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Advances in Applications of Burnup Credit to Enhance Spent Fuel Transportation, Storage, Reprocessing and Disposition

Proceedings of a Technical Meeting held in London, 29 August – 2 September 2005

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ADVANCES IN APPLICATIONS OF BURNUP CREDIT TO ENHANCE
SPENT FUEL TRANSPORTATION, STORAGE, REPROCESSING AND DISPOSITION

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FOREWORD

The most common assumption used in criticality safety analysis of spent nuclear fuel from power reactors is that spent fuel has the same reactivity as unburned fuel. This approach is typically known as the "fresh fuel" assumption and results in significant conservatism in the calculated value of the system reactivity. Current calculational methods have made possible taking credit for the reactivity reduction associated with fuel burnup, hence reducing the analysis conservatism while maintaining an adequate criticality safety margin. Spent fuel management is a common and costly activity for all operators of nuclear power plants. Implementing burnup credit offers the possibility to reduce fuel cycle costs, given the number of Member States dealing with increased spent fuel quantities and extended durations In fact, in many countries, burnup credit is already applied to transport systems, wet and dry storage facilities, and components of reprocessing plants. For disposal of spent fuel and reprocessing of some advanced fuel designs, burnup credit is considered to be important for viable schemes.

In 1997, the International Atomic Energy Agency (IAEA) initiated a task to monitor the implementation of burnup credit in spent fuel management systems, to provide a forum to exchange information, to discuss the matter and to gather and disseminate information on the status of national practices of burnup credit (BUC) implementation in the Member States. The IAEA started this active programme with an advisory meeting in 1997 (IAEA-TECDOC-1013, 1998) exploring worldwide interest in using BUC in spent fuel management systems. A second major meeting on BUC was held in Vienna in July 2000 and reported in IAEA-TECDOC-1241 (2001). The IAEA organized a third major BUC meeting in Madrid in April 2002 on requirements, practices, and developments in BUC applications (IAEA-TECDOC-1378, 2003). Following the recommendations of the Madrid meeting encouraging the IAEA to continue its activities on burnup credit including dissemination of related information, the IAEA planned and held a fourth technical meeting on burnup credit applications. This publiction reports on the results of the meeting held in London 29 August–2 September 2005, addressing advances in applications of burnup credit to reduce the number of transports, increase storage capacity, and enhance reprocessing and disposal capabilities.

The IAEA wishes to thank the UK Department for Transport and the UK Nuclear Installations Inspectorate (NII) for hosting the meeting, and in particular G. O'Connor (UK DfT) and D. Simister (NII) for their hard work. The IAEA also wishes to thank all participants of the meeting for their contributions and in particular J-C. Neuber (AREVA NP) for chairing the meeting and for preparing the meeting proceedings. The IAEA officer responsible for the organization of the meeting and overall coordination of this report was W. Danker of the Division of Nuclear Fuel Cycle and Waste Technology.

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MEETING OVERVIEW

This publication records the proceedings of a technical meeting organized by the IAEA and held in London 29 August–2 September 2005 with sixty participants from 18 countries. As indicated in the title, the objective of this meeting was to provide a forum for exchange of technical information on spent fuel burnup credit applications and thereby compile state-of-the-art information on technical advances related to spent fuel transportation, storage, reprocessing and disposition.

The term “burnup credit”(BUC) derives from a common assumption used in criticality safety analyses for spent power reactor fuel that spent fuel has the same reactivity as unburned fuel. This "fresh fuel" assumption results in significant conservatism in the calculated value of the system reactivity. Current calculational methods have made possible taking credit for the reactivity reduction associated with fuel burnup, hence reducing analytical conservatism while maintaining an adequate criticality safety margin.

The IAEA assigns a high priority to this topic since burnup credit applications offer significant efficiencies and attendant cost savings. In many countries, burnup credit is already applied to transport systems, wet and dry storage facilities, and components of reprocessing plants. Burnup credit is also considered important for disposal of spent fuel and reprocessing of some advanced fuel designs. The IAEA initiated its work to monitor implementation of burnup credit with a technical meeting in 1997, followed by major meetings in 2000, 2002, and 2005.

The following proceedings of the 2005 London meeting are organized in a manner consistent with the proceedings from the 2002 Madrid meeting (IAEA-TECDOC-1378, 2003). The majority of the proceedings (~400 pages) consist of the papers presented in the topical sessions held on 29–31 August 2005. This short meeting overview is followed by an extended summary (~70 pages) of the meeting, including (1) an introduction; (2) an overview of the topical presentations; (3) an overview of the burnup credit efforts by country; and (4) reports of group discussions analyzing four key areas. These group discussions were held in four parallel sessions on 31 August and 1 September and are summarized in section 4. Example conclusions derived from these working group summaries follow:

- Working group one (calculation methodology) identified three areas where insufficient guidance is readily available, for example the need to develop guidance as to what constitutes a complete set of documentation for burnup credit implementation.

- Working group two (validation and criticality safety criteria) provided thirteen specific conclusions, for example that experiments should be amenable to calculation without significant modeling approximations or assumptions and should include a thorough assessment of experimental uncertainty.

- Working group three (procedural compliance) provided six specific observations, for example that significant variation exists between standards with respect to whether measurement of burnup is a firm requirement or not.

- Working group four (regulatory aspects) identified six conclusions and four recommendations related to regulatory considerations, for example internationally accepted regulatory guidance for the implementation of burnup credit should be developed.

The leaders of these working group discussions presented their results at the closing plenary session on 2 September and then participated in an integrated panel discussion. Following the working group presentations and the panel discussion, the technical meeting chair presented his summary, concluding that the meeting represented an encouraging step forward in the application of burnup credit. The following conclusions and recommendations derived from deliberations during the 2005 meeting:

- Since BUC methodology is still developing, these international meetings play an important role in developing and maintaining technical capability as well as establishing good practice in BUC. Participants in this technical meeting in London therefore urged continuation of international activities in BUC including organization of these BUC technical meetings in the future.

- Since the lack of publicly available chemical assay data, particularly for VVER fuel, is a serious obstacle to BUC usage, the international community is urged to assist VVER validation.

- In addition, the international community is urged to support cooperation in performing new radiochemical assays and critical experiments appropriate to enhance application of BUC.

- Participants urged the development of international standards or guidelines for implementation of BUC in wet and dry storage systems, transport casks, reprocessing facilities, and for final disposal.

- The participants also recommended that the international community study the application of risk informed methods to BUC criticality safety assessments. It would be beneficial to develop methods of quantifying the risk factors due to the individual steps of BUC implementation and estimating the integral risk due to the use of BUC inclusive of its benefits.

MEETING AND WORKING GROUP SUMMARIES*

1. INTRODUCTION

In 1997, the International Atomic Energy Agency (IAEA) initiated a task to monitor the implementation of Burnup Credit (BUC) in criticality safety analysis and control of spent fuel management systems (wet and dry storage installations, transport casks, reprocessing facilities, final disposal systems), to provide a forum to exchange information, i.e.

- to discuss the matter
- to gather and disseminate detailed information on the status of national practices of BUC implementation in the Member States, and thus
- to contribute to improving the knowledge of:
  - BUC calculation methodologies
  - validation needs being imperative for BUC applications, and
  - implementation issues.

The IAEA started this active program with an Advisory Group Meeting in 1997 (resulting in IAEA-TECDOC-1013, 1998) exploring worldwide interest in using BUC in spent fuel management systems for

- UO2 (UOX) and MOX PWR and BWR fuels
- RBMK fuels, as well as
- VVER fuels.

As noted in IAEA-TECDOC-1013, even though economics is generally the primary factor in deciding to use BUC, other benefits contributing to public health and safety as well as resource conservation and environmental quality; in addition, cooperation of countries and organizations in developing and implementing BUC would mitigate resource requirements.

A second major meeting on BUC (resulting in IAEA-TECDOC-1241, 2001) was held in Vienna in July 2000. It concluded that use of BUC and understanding of related technical and regulatory issues continued to progress. It also reiterated recommendations that BUC information and data should be cooperatively developed and shared. In this context it was recommended that a BUC Training Course should be organized by the IAEA to transfer knowledge and expertise from Member States already applying BUC to Member States that are going to consider use of BUC in the near future.

The IAEA complied with this recommendation and contributed to a well received two week BUC training course held in the United States of America in October 2001. Most of the topics of this course were repeated in a second Training Course promoted by the IAEA at the China Institute of Atomic Energy (CIAE), Beijing in July 2002.

A third major BUC meeting was organized by the IAEA in Madrid in April 2002. As appears from IAEA-TECDOC-1378 (published in 2003), validation of BUC calculation codes and methods, key issues of safety assessment and implementation, and future applications were

* The views and recommendations expressed in this section are those of the meeting participants and do not necessarily reflect those of the IAEA.

addressed at the Madrid meeting. The outcomes of this meeting encouraged the IAEA to continue its activities on BUC.

Results from the Madrid meeting as well as the fact that several Member States have to deal with increasing spent fuel quantities led to a new task addressing advances in applications of BUC to reduce the number of transports, increase storage capacities, and enhance reprocessing and disposal capabilities. Therefore, a technical meeting (TM) was organized by the IAEA in London, United Kingdom, in August/September 2005. The agenda of this meeting included the recent developments and improvements in all key issues of BUC applications to spent fuel management systems.

2. OVERVIEW OF TOPICAL PRESENTATIONS

Implementation of BUC in a spent fuel management system requires application of a system specific loading criterion which indicates the requirements the spent fuel has to meet in order to be allowed to be loaded in the system. Any loading criterion used in BUC correlates the safety limit of a parameter appropriate to characterize the spent fuel with the initial enrichment of the fuel. The value of the safety limit of such a safety parameter at given initial enrichment depends on

- the used level of BUC (net fissile content, actinide-only, actinide-plus-fission-product level etc, cf. IAEA-TECDOC-1013, p.1) and
- the design of the spent fuel management system.

The safety parameter usually chosen to present the loading criterion is the average burnup of the spent fuel, i.e. the loading criterion is usually presented in form of a loading curve indicating the minimum average burnup necessary for fuel with a specific initial enrichment to be loaded in the spent fuel management system. For some systems, however, it is more convenient to present the loading criterion in a different form: For dissolver facilities in reprocessing plants, for instance, either the fissile content or the residual U235 enrichment of the spent fuel is chosen as safety parameter, i.e. the loading criterion indicates the maximum allowable fissile content or residual enrichment as a function of the initial enrichment.

The key steps in application of BUC to a spent fuel management system are the following:

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

Prediction of the isotopic inventory of the spent fuel by means of depletion calculations requires:

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

A loading criterion of a spent fuel management system designed for BUC usually applies to any fuel positions of the system. In fact a loading criterion can only be evaluated if no credit is taken for the real loading scheme. Therefore, evaluation of a loading criterion makes it necessary to look for a bounding irradiation history given by those depletion conditions (reactor operation conditions) which lead to the highest reactivity of the spent fuel. The depletion conditions are characterized by the following parameters or conditions:

- Fuel temperature
- Moderator temperature or density; void history (BWR reactor types)
- Presence of soluble boron (PWR reactor types)
- Use of fixed neutron absorbers in form of control rods or blades, control assemblies (VVER), burnable poison rods, axial shaping rods etc
- Operational strategies, reload patterns and core environment: specific power history, extended low power operation period at end of cycle, in-out/out-in strategies, presence of MOX fuel etc.

The depletion conditions are significantly impacted by fuel characteristics such as:

- the presence of integral burnable absorbers (in form of Gd or Er bearing fuel rods or so-called “IFBA rods” containing pellets with boron coating)
- axial zoning of initial enrichment and/or burnable absorbers, presence of axial blankets
- horizontally heterogeneous initial enrichment distributions (BWR, MOX fuel assemblies, VVER)
- presence of partial length fuel rods (BWR designs).

Criticality calculation and evaluation of the fuel loading criterion require:

- isotopic selection and validation
- validation of the criticality calculation code to be used,
- evaluation of the reactivity effect of axial and horizontal burnup profiles
- sensitivity studies on the reactivity effects of variations and tolerances in the parameters describing the characteristics of the spent fuel management systems.

By definition, for a specific initial enrichment the loading criterion provides a specific numerical value for the safety parameter chosen to present the loading criterion. This criterion must therefore cover the variety of reactivity effects due to the variety of axial and horizontal burnup profiles. So therefore, evaluation of a fuel loading criterion makes it necessary to look for a bounding axial burnup profile as well as for a bounding horizontal burnup profile. Since the shape of axial and horizontal burnup profiles and the reactivity effects due to these profiles change with the average burnup of these profiles the bounding axial burnup profile and the bounding horizontal burnup profile vary with the average burnup.

The complexity of fuel designs (modern BWR and VVER designs in particular), the complexity of the depletion conditions (use of fixed neutron absorbers, operational strategies, reload patterns and core environment), the complexity due to non-uniform burnup distributions and the complexity of the design of spent fuel management systems require careful choice of the BUC calculation methodology to be applied to assure sufficiently accurate presentation of the physics of the problem to be solved. Accordingly, the first technical topic on the agenda of the London TM, 2005, was:

- **Technical topic 1**: Principles of choosing the calculation methodology with respect to the fuel design and the spent fuel management system.

  The choice of the calculation methodology is inseparably linked with the isotopic selection and validation and the validation of the criticality calculation code chosen. Isotopic validation and validation of the reactivity calculations are necessary conditions for demonstrating the adequacy of the chosen calculation methodology. Therefore, the second technical topic on the agenda of the London TM was:

- **Technical topic 2**: Nuclear data and validation of depletion and reactivity calculations.

  Several papers were presented under topics 1 and 2. These papers addressed the following items:

  - Establishment of a database of spent BWR fuel data in the USA including physical fuel data and reactor operating histories to support BUC analysis for BWR fuel; application of multivariate data analysis methods to identify data clusters and bounding conditions.
  - BUC calculation methodologies for transport and storage casks:
    - A comprehensive survey of the US program to introduce actinide-plus-fission-product BUC in transport and storage casks was given. The activities relating to this program mainly address (a) the availability and applicability of existent isotopic assay data for validation of depletion calculations and existent critical experiments for validation of reactivity calculations, (b) the possibilities of increasing the size of isotopic assay data samples for the fission products relevant to BUC in particular, (c) the performance of new critical experiments with the relevant fission products, (d) the application of Sensitivity/Uncertainty tools for evaluating existing data and designing new experiments.
    - In a different paper the challenging idea is broadly outlined to apply BUC to actual cask loading schemes instead of generating a loading curve. This includes that the isotopic inventory and burnup distribution are calculated for each individual fuel assembly. This requires on-line core-following depletion calculation which is quasi-continuously recalibrated by means of in-core measurement. The paper gives no example, but it is obvious that the outlined procedure multiplies the complexity of BUC application with respect to burnup quantification and verification, depletion and reactivity calculation validation, implementation of cask loading.
  - Studies of BUC methodology for spent fuel storage in the People's Republic of China: A comprehensive program to introduce BUC in spent fuel storage has been initiated. This program includes validation of the calculational tools by means of experimental data and OECD/NEA BUC benchmarks as well as new critical experiments with spent fuel (from Qinshan nuclear power plant, for instance) planned to be performed at the China Institute of Atomic Energy.
  - BUC application to the post-closure phase of a spent fuel repository: Parametric criticality studies were presented for different fuel types, initial enrichments, burnup values and scenarios to demonstrate the need for BUC to minimize the probability of occurrence of the formation of critical and supercritical configurations. The BUC calculation procedure used for these

studies has the capability of two and three dimensional calculations of pin power distributions, isotopic inventories and reactivity.

- Improvements, developments and validations of two and three dimensional depletion calculation codes:
  - The improvements in the TRITON depletion sequences of the SCALE system were presented: TRITON has been enhanced by the addition of depletion sequences which use KENO V.a/VI for three dimensional (3D) transport solutions. This enables the performance of direct 3D depletion calculations. The results of isotopic assay validation calculations performed by means of the one, two and three dimensional depletion sequences of the SCALE system were presented.
  - The development and validation of the integrated depletion code MVP-ORBURN were described. This code combines the continuous energy Monte Carlo code MVP with the point depletion code ORIGEN 2. Results of BUC applications of MVP-ORBURN to PWR and BWR fuel were presented.
- Isotopic validation of BUC applications to VVER-440 spent fuel:
  - VVER-440 spent fuel isotopic assay data from RIAR Dimitrovgrad (ISTC project no. 2670): Radiochemical assay data for eight samples of spent VVER-440 fuel from Novovoronezh nuclear power plant unit 4 were presented. These data are the only spent VVER-440 fuel isotopic composition data publicly available outside of the Russian Federation which include fission products relevant to the actinide-plus-fission-product BUC level.
  - In a different paper these RIAR data served for validation of depletion calculations performed by means of the SAS2H sequence (employing ORIGEN-S) and the TRITON control module (using NEWT) of the SCALE system. These validation calculations included comparisons with 12 samples of actinide assay data taken at the Kurchatov Institute, Moscow, from irradiated fuel from Novovoronezh nuclear power plant unit 4. The calculations also included comparisons with the Takahama-3 assay data and the OECD/NEA VVER CB2-benchmark data.
- Evaluation of the REBUS experiments on PWR fuels: A feature of paramount importance in the REBUS program is that this program was aimed at providing an experimental database jointly usable for validation of depletion and reactivity calculation in such a way that a direct validation of the calculational tools commonly used in BUC criticality safety analysis is enabled (i.e. estimation of keff rather than reactivity perturbation calculations). The REBUS program therefore included integral reactivity worth measurements using fuel bundles from commercial PWR samples and, afterwards, radiochemical assay of the irradiated fuel. Two papers were presented providing preliminary analysis results obtained with a couple of different calculation codes.
- Sensitivity and uncertainty studies of the applicability of critical experiments to BUC criticality calculations: Two papers addressing this item were presented:
  - In one paper the applicability of critical MOX experiments selected from the International Handbook of Evaluated Criticality Safety Benchmark Experiments (IHECSBE) to the use of BUC for the compact storage installation at the VVER-440 Paks nuclear power plant, Hungary, and the VVER-440 fuel transport casks C-30 and TK-6 was studied. Fission rates, capture rates and neutron fluxes were

calculated in five broad energy groups, and the relative importance of the uranium and plutonium isotopes was analyzed. From the comparisons of the results obtained for the experiments and the application cases it was concluded that a "stand-alone" use of the selected MOX experiments is not sufficient for validating the use of BUC for the application cases.The other paper presented outcomes of a study of the applicability of more than 1000 critical benchmark configurations primarily taken from the IHECSBE to the Generic Burnup Credit GBC-32 prototypical high capacity US rail cask assumed to be loaded with 32 fuel assemblies of the 17x17-25 type with an initial enrichment of 4.0 wt.-% U235, an average burnup of 40 MWd/kg U and a cooling time of 5 years. This study was performed with the aid of the Sensitivity/Uncertainty (S/U) analysis sequences TSUNAMI of the SCALE system at Oak Ridge National Laboratory (ORNL), USA. It was found that only MOX configurations from the French Haut Taux de Combustion (HTC) experimental series are applicable: 156 HTC MOX configurations were considered in the study, 143 were found to be applicable.+) None of the 978 analyzed non-HTC experiment configurations including high-, intermediate and low enriched uranium as well as plutonium and non-HTC MOX configurations was identified as applicable. Only 45 of the non-HTC MOX configurations were classified as marginally applicable. However, an important aspect of assessing these outcomes is to consider whether all the BUC nuclides used in the application case (the GBC-32 cask) are really represented in the experiments. If some nuclides are missing in a critical configuration and if these missing nuclides have a significant reactivity worth in the application case, then this critical configuration is obviously tending to be rejected. The TSUNAMI procedure is therefore capable of calculating sensitivity profiles for any desired nuclide as a function of neutron energy. The degree of agreement between the nuclide-specific sensitivity profiles for experiments and the application case is described as "coverage". Coverage is provided by an experiment wherever the sensitivity profile of the experiment covers the sensitivity profile of the application case. Experiments which were identified as "not applicable" in total can have significant degrees of coverage for specific nuclides. Examples for such cases are given in the presented paper. Therefore, methods are under development at ORNL which utilize the relevant information from such experiments.

- Whilst in the study of the applicability of MOX experiments to VVER-440 spent fuel systems the experiments were selected in the traditional way (expert's judgment: comparisons of materials, geometries, gross integral parameters such as moderation ratio, lethargy of average neutron energy causing fission), S/U evaluation methods like TSUNAMI allow detailed quantitative comparisons of the similarity of nuclear system with respect to the underlying nuclear data characterizing the isotopic compositions in all the material zones of the

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+) Different from the recommendation reported in section 3.2.4.4 of this summary a criterion of $c_k \geq 0.9$ was used in the presented paper. Experiments with $0.8 < c_k < 0.9$ are identified as "marginally acceptable".

systems and hence their impact on the neutron spectra and the reactivity. Even though covariance matrices containing all the information about the "uncertainties" (variances and correlations) of the nuclear data (cf. section 3.2.4.4 of this summary) are still under development (for BUC analyses in particular), and even though no theory based rationale has been found until now for the decision criterion whether an experiment can be regarded as acceptable or not, S/U evaluation methods such as TSUNAMI provide a powerful tool for the selection of experiments. In addition, because of the capability of calculating nuclear specific sensitivity profiles, S/U methods can be used to design new experiments such that coverage with application cases is achieved.

Once an adequate BUC calculation methodology is chosen and validated, the task is to determine a criticality safety acceptance criterion from which the fuel loading criterion for the application case, i.e. the spent fuel management system to be analyzed, can be derived. Determination of a criticality safety acceptance criterion requires a consistent calculation route

- to evaluate the experimental information from chemical assay data in order to consider the isotopic bias in the keff value of the spent fuel management system,
- to evaluate the experimental information from reactivity worth measurements and critical benchmark experiments to take account of the calculated bias in the keff value of the spent fuel management system due to the criticality calculation procedure applied,
- to determine bounding irradiation histories including specific depletion conditions (e.g. use of control rods).
- to evaluate the reactivity effects of axial and horizontal burnup profiles under the conditions of the spent fuel management system,
- to cover the variability in keff value of the system due to variations and tolerances in the parameters describing the characteristics of the system.

There is a wealth of ways to come to a consistent calculation route. For instance, the isotopic bias can:

- either be covered by applying nuclide specific number density correction factors derived from comparisons of predicted isotopic concentrations to chemical assay data;
- or explicitly calculated by means of sensitivity analyses of the impact of the bias of the concentrations of the individual isotopes on the keff value of the spent fuel management system.

The bias in the prediction of the BUC nuclide reactivity worths can:

- either be conservatively covered by applying penalty factors to the isotopic number densities used in the calculation of the keff value of the spent fuel management system;
- or evaluated by means of sensitivity based criticality validation techniques.

Obviously, different ways of establishing a consistent calculation route result in different degrees of conservatism maintained in the k_{eff} value calculated for the spent fuel management system and hence in different safety margins. The third technical topic on the agenda of the London TM was accordingly described as "criticality safety criteria":

- **Technical topic 3**: Criticality safety criteria

Several presentations addressing this topic were given. All these presentations together reflected the wealth of methods of establishing a consistent calculation route. All aspects of determining a criticality safety acceptance criterion were covered for several fuel types and designs (PWR, BWR, VVER-440 and VVER-1000) and spent fuel managements systems (wet storage installations, transport and storage casks, dissolver facilities in reprocessing plants):

- Determination of bounding depletion conditions;
- Consideration of isotopic bias;
- Consideration of the reactivity bias in the reactivity calculations;
- Determination of bounding axial and horizontal burnup profiles and determination of the reactivity effects related to axial and horizontal burnup profiles and isotopic number density distributions;
- Evaluation of manufacturing tolerances and variations in the parameters describing fuel designs and spent fuel management systems.

In addition, the impact of the degree of conservatism due to the chosen depletion conditions and possible enhanced by applying conservatively derived isotopic correction factors to the number densities obtained from the depletion calculations on the estimates of reactivity effects due to non-uniform burnup distributions and manufacturing tolerances was described. This impact can be significant as exemplified for the reactivity effect due to axial burnup distribution.

The objective of a BUC criticality safety analysis of a spent fuel management system is to express the criticality safety acceptance criterion, derived for the system, in terms of a BUC loading criterion. Application of the loading criterion requires implementation of control and fuel handling procedures which assure compliance with the loading criterion. The fourth technical topic on the agenda of the London TM therefore was:

- **Technical topic 4**: Procedural compliance with the safety criteria

To assure compliance with the loading criterion the control and fuel handling procedures shall be aimed to prevent a "misloading error". By definition, a misloading error occurs when fuel that does not comply with the loading criterion of a spent fuel management system is anyway loaded in the system. The root cause for such an error is either an error in the information about the numerical value which the fuel has for the safety parameter chosen to present the loading criterion (e.g. average burnup) or an operational error. As with any other criticality safety scenario, the double contingency principle applies to the misloading event. Usually, this principle is applied in such a way that the misloading error is considered as one incident and a second concurrent event does not need to be considered. However, there is one problem which is inherent to the misloading error and distinguishes this error from most of the other accidental events to be considered in criticality safety analysis: If a misloading error does really

occur then there is a high probability that the error remains undetected. Then any other design basis accidental event that takes place at a later time cannot be regarded as a "concurrent" event; and the double contingency principle thus requires that the misloading event plus the additional accidental event have to be considered in the criticality safety analysis of the spent fuel management system. The consequences are system dependent but usually lead to the result that BUC cannot be applied to the system as shown in one of the papers presented under topic 4. This paper therefore comes to the conclusion that the misloading error has to be excluded as a design basis event; and a fuel handling procedure which meets this requirement is described in the paper. This fuel handling procedure has been developed under the responsibility of the German nuclear power plant Neckarwestheim II and is hence described as the "Neckarwestheim fuel handling procedure".

Once procedural compliance with the loading criterion is assured, one can enjoy the benefits of BUC. So, the fifth technical topic on the agenda of the London TM was:

- **Technical topic 5**: Benefits of BUC applications

  Several papers addressing this topic were presented:

  - The cost savings that can be achieved in the USA by extending BUC in transporting PWR spent nuclear fuel from the actinide-only to the actinide-plus-fission-product level were estimated.
  - The benefits that can be obtained by extending BUC for the receipt and storage of UOX PWR fuels in COGEMA/La Hague Pools from the actinide-only to the actinide-plus-fission-product level were discussed.
  - The Swedish Nuclear Fuel and Waste Management, SKB, regards the use of BUC as an option to increase the maximum allowable enrichment in the interim storage for spent nuclear fuel, CLAB, from 4.2 wt.-% to 5 wt.-% U235. Introduction of BUC could also reduce the rigorous control of fuel design parameters presently required before storage at CLAB. Preliminary studies of BUC applications to final disposal were also performed in Sweden.
  - The Nuclear Regulatory Authority of the Slovak Republic started a program to verify and validate application of BUC to VVER-440 spent fuel management systems. BUC is regarded as an important means to solve future storage and transport casks.
  - Studies of benefits from applying BUC to the spent fuel interim storage facility at Zaporizhya nuclear power plant were performed in the Ukraine.
  - The benefits from applying BUC to a future NUHOMS interim storage facility instead of using storage casks equipped with borated stainless steel channels were discussed in the Republic of Armenia.

  As regards application of BUC to VVER fuels there is a shortage of publicly available experimental data for validation of depletion calculations. This lack of data is commonly regarded as a serious obstacle for applying BUC. Instead of trying to introduce BUC without sufficient validation it seems to be more appropriate to look for a different solution serving as an interim solution at least. In the Czech Republic for instance, as was also reported under topic 5, application of partial boron credit to the wet storage facility at Dukovany nuclear power plant was approved in order to enable an increase of the initial enrichment of the fuel assemblies.

Due to the complexity of BUC cooperation of research institutes, industry, utilities and regulators is a must. It is important to set standards to guarantee consistency and dissemination of "good practice" and to give regulatory guidance, for applicants and regulators as well. The sixth technical topic on the agenda of the London TM was:

- **Technical topic 6**: Regulatory aspects in BUC

Besides a presentation which primarily reflected the state of the discussion in Sweden about the option of using BUC in the CLAB facility, reports were given on:

- the US NRC regulatory recommendations for actinide-only BUC in transport and storage casks; and
- the German BUC regulatory standards for wet storage, transport and dry storage of PWR/BWR UOX/MOX fuel.

The US NRC regulatory recommendations as well as the German BUC regulatory standards address all the steps which have to be taken in application of BUC, from the depletion calculations to the procedural compliance with the loading criterion.

Almost all the papers presented in the technical sessions of the London TM reflected the state of the art in applying BUC in the different Member States. A survey of the national practices, ongoing activities and regulatory status of using BUC in the different countries is given in the following section.

To encourage the discussion about all the topics of the technical sessions and foster the exchange of information about BUC practices four working groups were convened in parallel sessions after the technical sessions described above. The following topics were discussed in these groups:

- Calculation methodology
- Validation and criticality safety criteria
- Procedural compliance with the safety criteria
- Regulatory aspects in BUC.

Each working group produced a report summarizing the discussions conducted and including the recommendations and conclusions reached. These reports are provided in section 4.

Topical presentations at this meeting included opening overviews of international activities coordinated by the IAEA and the NEA:

- The first presentation summarized the activities the IAEA has initiated in this area. Because of the encouraging results of BUC-related activities and because of the observed trends towards more spent fuel storage capacity, longer storage durations, and higher initial enrichment and higher fuel burnup, the conclusion drawn in this presentation was that IAEA will continue to assign a high priority to activities associated with the implementation of BUC.
- The second presentation summarized the activities within the OECD/NEA. The attention was mainly focused on the work of the Expert Group on BUC Criticality, a subordinate group to the Working Party on Nuclear Criticality Safety (WPNCS) working under the Nuclear Science Committee (NCS). The main objective of the Expert group on BUC Criticality is to demonstrate that the available calculational tools are appropriate for BUC applications and that

a reasonable safety margin can be established. For this purpose the group established a suite of BUC criticality benchmarks that assesses the capability to calculate both spent fuel isotopic composition and reactivity of the spent fuel. The work of this group is ongoing.

The London TM was closed with presentations by the working group chairs on the discussions conducted, the conclusions drawn and the recommendations made by the groups and with a panel discussion. In the panel discussion, the attention was mainly focused on the benefits of using S/U-based criticality validation techniques and the Bayesian inference approach to the evaluation of experimental data (e.g. outcomes from differential reactivity worth measurements can be used to build an a priori knowledge distribution, results from integral worth measurements can then be employed to gain an a posteriori knowledge distribution with a reduced variance). As follows from the presentations given in the technical sessions, the discussion during these sessions and the discussion conducted in the working groups, even if significant improvements in BUC methodology and practice have been achieved since the Madrid TCM, 2002, there are still a lot of challenges; and there is a demand for harmonization of the BUC calculation and implementation methodologies and for guidance to achieve good practice. This demand results in the following conclusions and recommendations:

- BUC is a still developing methodology. Therefore, the international meetings organized by the IAEA since 1997 play an important role in developing and maintaining technical capability as well as establishing good practice in BUC. The IAEA is therefore urged by the London TM participants to continue its activities in BUC and to organize BUC TMs in the future.
- A serious obstacle to the use of BUC is the lack of publicly available chemical assay data. This goes for VVER fuel in particular. The IAEA is therefore urged to assist VVER validation.
- In addition, the IAEA is urged to support international cooperation in performing new radiochemical assays and critical experiments appropriate to enhance application of BUC.
- The IAEA is urged to develop or assist the development of international standards or guidelines for implementation of BUC in wet and dry storage systems, transport casks, reprocessing facilities, and final disposals.
- As already recommended at the Madrid TCM, 2002, the IAEA should take an action to study or assist a study of application of risk informed methods to BUC criticality safety assessments. It would be beneficial to develop methods of quantifying the risk factors due to the individual steps of BUC implementation and estimating the integral risk due to the use of BUC inclusive of its benefits.

The UK Department for Transport and the UK Nuclear Installations Inspectorate were thanked by the participants for hosting the meeting, in particular G. O'Connor (UK DfT) and D. Simister (NII). Participants also expressed appreciation to William Danker (IAEA) for organizing the meeting and to Jens-Christian Neuber (AREVA NP) for serving as Meeting Chair.

3. OVERVIEW ON THE BURNUP CREDIT EFFORTS BY COUNTRY

This chapter provides an overview on the national practices, ongoing activities and regulatory status of using burnup credit in different countries. The information was mainly gathered from

the countries participating in this TM and is divided according to the different areas of burnup credit application, including:

- Storage of spent fuel:
  - Wet storage (at reactor or away from reactor)
  - Dry storage (on site or off site)
- Wet and dry transport systems
- Reprocessing
- Disposal.

3.1. Wet storage of spent fuel

3.1.1. Wet storage at reactor

Information on the status of burnup credit applications to wet storage at reactor is presented in Table 3.1. This table also provides information on activities ongoing in different countries to get approval for implementing burnup credit.

As can be seen from this table, in several countries the *actinide plus fission product burnup credit level* is approved and implemented for wet storage of PWR UOX fuel at reactor. Use of the actinide plus fission product level means that credit is taken for the net fissile content of the fuel (taking into account both burnup and buildup of the different fissile nuclides), the absorption effect of the actinides and the neutron absorption in the major fission products (in the USA all fission products available except for Xe-135).

In some countries the *actinide-only burnup credit level* is applied to wet storage of PWR UOX fuel as well as RBMK fuel. In this case credit is taken only for the net fissile content of the fuel and the absorption effect of the actinides.

For the wet storage of BWR fuel (UOX as well as MOX) the *integral burnable absorber burnup credit level* is usually used. Credit is taken for the initial presence of integral burnable absorbers (e.g. gadolinium) in the fuel design, and the maximum reactivity of the fuel under the storage conditions of interest is used, which is often not the initial reactivity.

3.1.2. Wet storage away from reactor

Several countries have wet storage facilities that are away from reactor. In most cases, these pools are not borated. In PWR pools, criticality control may rely on a combination of burnup credit and soluble boron. Therefore, burnup credit approval may be different for PWR fuel in an unborated away from reactor wet storage system than that used at the plant.

The only away from reactor wet storage facility that utilizes burnup credit is in France. Prior to reprocessing, the spent fuel received at La Hague is put in a wet storage facility. This facility has approval for the actinide-only burnup credit level for PWR fuel. PWR burnup credit that covers selected fission products is under development. There is no burnup credit for any other fuel type.

Wet storage facilities away from reactor, which do not take credit for burnup, exist in Bulgaria, Germany, Japan, Russia, Slovakia, Sweden, Ukraine, and the United States of America. A new facility is build for Switzerland, which currently is not planned to take burnup credit.

TABLE 3.1: USE OF BURNUP CREDIT FOR WET STORAGE AT REACTOR (AR)

Country	PWR	BWR	MOX (PWR)	VVER	RBMK	Reactor Types[1]
Armenia	na	na	na	INT	na	VVER
Belgium	APU-1	na	na	na	na	PWR
Brazil	APU-2	na	na	na	na	PWR
Bulgaria	na	na	na	INT	na	VVER
China	INT	na	na	na	na	PWR
Czech Republic	na	na	na	INT	na	VVER
Finland	na	Gd	na	INT	na	VVER, BWR
France	Nc	na	Nc	na	na	PWR
Germany	APU-2	Gd	APC-2	Nc	na	PWR, BWR, VVER
Hungary	na	na	na	INT	na	VVER
Japan	INT	INT	INT	na	na	PWR, BWR
Korea	APU-2	na	na	na	na	PWR
Lithuania	na	na	na	na	Nc	RBMK
Mexico	na	Gd	na	na	na	BWR
Netherlands	APU-2	na	na	na	na	PWR
Russia	na	na	na	INT	APU-1	VVER, RBMK
Slovakia	na	na	na	UD-2	na	VVER
Slovenia	APU-2	na	na	na	na	PWR
South Africa	APU-2	na	na	na	na	PWR
Spain	APU-2	Gd	na	na	na	PWR, BWR
Sweden	Nc	Gd	na	na	na	PWR, BWR
Switzerland	APU-2	Gd	Nc	na	na	PWR, BWR
Ukraine	na	na	na	INT[2]	No	VVER, RBMK
UK	UD-1	na	na	na	na	PWR
USA	APU-2	Gd	UD-2	na	na	PWR, BWR

[1]Burnup credit is not currently envisioned for heavy water or gas cooled reactors so they are not listed
[2]Burnup credit is allowed by the regulations but actions to implement have not started.

Abbreviations:
APU-1: Approved and implemented burnup credit using the actinide-only level.
APU-2: Approved and implemented burnup credit using the actinide plus fission product level.
APC-2: Approved in concept burnup credit using the actinide plus fission product level.
UD-1: Preparing documentation for taking credit using the actinide-only level.
UD-2: Preparing documentation for taking credit using the actinide plus fission product level.
Gd: Use of the integral burnable absorber level.
INT: Interested, including some early analysis.
na: Not applicable.
Nc: Not being considered but potentially applicable.
No: No interest since the reactor is shutdown.

3.2. Dry storage of spent fuel

Information on the status of burnup credit applications to dry storage is presented in Table 3.2.

At present only a few countries are using burnup credit for dry storage. In Armenia the approval for burnup credit is limited to the use of the *net fissile content burnup credit level*. In this case credit is taken only for the net fissile content of the fuel. In contrast to this case the *actinide-only burnup credit level* is:

- approved in concept in the USA and
- already used in Germany and Ukraine.

Application of the actinide plus fission product level is approved in concept in Germany.

3.3. Transport of spent fuel

Information on the status of burnup credit applications to transport casks is presented in Table 3.3.

As in case of dry storage in Armenia the approval for burnup credit is limited to the use of the *net fissile content burnup credit level*. The *actinide-only burnup credit level* is allowed in USA and already used in France, Germany, Netherlands, Russia, and Switzerland.

A lot of activities are ongoing in several countries to get approval for application of the actinide plus fission product burnup credit level to transport casks. In Germany application of this level is already approved in concept.

3.4. Reprocessing

France: At La Hague actinide-only burnup credit is used for 10 years for storage in the pond and reprocessing. For liquids in tanks, some specific authorizations with fission product have been obtained. Activities to get approval for application of a burnup credit level which utilizes actinides and between 6 and 15 fission products are ongoing.

TABLE 3.2: USE OF BURNUP CREDIT FOR DRY STORAGE

Country	PWR	BWR	MOX (PWR)	VVER	RBMK	Reactor Types[1]
Armenia	na	na	na	APU-0	na	VVER
Belgium	Nc	na	na	na	na	PWR
Brazil	Nc	na	na	na	na	PWR
Bulgaria	na	na	na	INT	na	VVER
China	INT	na	na	na	na	PWR
Czech Republic	na	na	na	RR-2	na	VVER
Finland	na	nc	na	nc	na	VVER, BWR
France	Nc	na	Nc	na	na	PWR
Germany	APU-1 APC-2	Gd	APC-2	Nc	na	PWR, BWR, VVER
Hungary	na	na	na	INT	na	VVER
Japan	Nc	Nc	Nc	na	na	PWR, BWR
Korea	INT	na	na	na	na	PWR
Lithuania	na	na	na	na	INT	RBMK
Mexico	na	Nc	na	na	na	BWR
Netherlands	Nc	na	na	na	na	PWR
Russia	na	na	na	Nc	INT	VVER, RBMK
Slovakia	na	na	na	UD-2	na	VVER
Slovenia	Nc	na	na	na	na	PWR
South Africa	Nc	na	na	na	na	PWR
Spain	INT	INT	na	na	na	PWR, BWR
Sweden	na	na	na	na	na	PWR, BWR
Switzerland	INT	INT	INT	na	na	PWR, BWR
Ukraine	na	na	na	APU-1	INT[2]	VVER, RBMK
UK	na	na	na	na	na	PWR
USA	APC-1	INT	INT	na	na	PWR, BWR

[1]Burnup credit is not currently envisioned for heavy water or gas cooled reactors so they are not listed
[2]Burnup credit is allowed by the regulatory law but actions to implement have not beyond.

Abbreviations:
APU-0: Approved and implemented burnup credit using the net fissile content level.
APU-1: Approved and implemented burnup credit using the actinide-only level.
APC-1: Approved in concept burnup credit using the actinide-only level.
APC-2: Approved in concept burnup credit using the actinide plus fission product level.
RR-2: Under regulatory review for taking burnup credit using the actinide plus fission product level.
UD-2: Preparing documentation for taking credit using the actinide plus fission product level.
Gd: Use of the integral burnable absorber level.
INT: Interested, including some early analysis.
na: Not applicable.
Nc: Not being considered but potentially applicable.

TABLE 3.3: USE OF BURNUP CREDIT FOR TRANSPORT (TRANSPORT CASKS)

Country	PWR	BWR	MOX (PWR)	VVER	RBMK	Reactor Types
Armenia	na	na	na	APU-0	na	VVER
Belgium	INT	na	na	na	na	PWR
Brazil	Nc	na	na	na	na	PWR
Bulgaria	na	na	na	INT	na	VVER
China	INT	na	na	na	na	PWR
Czech Republic	na	na	na	RR-2	na	VVER
Finland	na	INT	na	INT	na	VVER, BWR
France	APU-1 UD-2	Nc	UD-1,2	na	na	PWR
Germany	APU-1 APC-2	Gd	APC-2	Nc	na	PWR, BWR, VVER
Hungary	na	na	na	INT	na	VVER
Japan	INT	INT	INT	na	na	PWR, BWR
Korea	INT	na	na	na	na	PWR
Lithuania	Na	na	na	na	INT	RBMK
Mexico	Na	Nc	na	na	na	BWR
Netherlands	APU-1	na	na	na	na	PWR
Russia	Na	na	na	APU-1	INT	VVER, RBMK
Slovakia	Na	na	na	UD-2	na	VVER
Slovenia	Nc	na	na	na	na	PWR
South Africa	Nc	na	na	na	na	PWR
Spain	INT	INT	na	na	na	PWR, BWR
Sweden	Nc	Nc	na	na	na	PWR, BWR
Switzerland	APU-1	INT	INT	na	na	PWR, BWR
Ukraine	Na	na	na	RR-1 /INT[2]	INT[2]	VVER, RBMK
UK	INT	Nc	Nc	na	na	PWR
USA	APC-1,UD-2	INT	INT	na	na	PWR, BWR

[1]Burnup credit is not currently envisioned for heavy water or gas cooled reactors so they are not listed
[2]Burnup credit is allowed by the regulatory law but actions to implement have not beyond.

Abbreviations:

APU-0 Approved and implemented burnup credit using the net fissile content level.
APU-1: Approved and implemented burnup credit using the actinide-only level.
APC-1: Approved in concept burnup credit using the actinide-only level.
APC-2: Approved in concept burnup credit using the actinide plus fission product level.
RR-1: Under regulatory review for taking burnup credit using the actinide-only level.
RR-2: Under regulatory review for taking burnup credit using the actinide plus fission product level.
UD-1: Preparing documentation for taking credit using the actinide-only level.
UD-2: Preparing documentation for taking credit using the actinide plus fission product level.
Gd: Use of the integral burnable absorber level.
INT: Interested, including some early analysis.
na: Not applicable.
Nc: Not being considered but potentially applicable.

Japan: Burnup credit is used in the spent fuel pool, which is part of the reprocessing facility. Burnup credit is also used for the dissolver.

Russian Federation: Burnup credit is currently used at the reprocessing facility.

United Kingdom: Actinide-only burnup credit is used in the dissolver.

3.5. Disposal

Reprocessing of fuel eliminates burnup credit for disposal. Also, fuel consolidation eliminates the need for consideration for burnup credit while the container is intact. Belgium, the Czech Republic, Germany and Korea, have performed some analysis of burnup credit in disposal. In Germany use of a risk-informed burnup credit methodology is presently under regulatory review. The USA and Sweden have actively pursued burnup credit for disposal to cover failed containers, which contain moderated fuel assemblies. The USA has submitted a risk informed Topical Report, which includes actinides and fission products to its regulatory body describing a burnup credit methodology, and has received approval of the Topical Report.

4. GROUP DISCUSSIONS

4.1. Calculation methodology

Leaders: **M. Brady Raap** (United States of America)
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4.1.1. Introduction

International efforts to develop an acceptable approach for the calculation of criticality safety limits for burnup credit have spanned almost two decades. Burnup credit has been studied for storage, transportation, and disposal applications, each of which may be characterized by different environments, configurations, and regulatory requirements. Hence, calculational approaches considered for prediction of and credit for the reduced reactivity worth of spent reactor fuel involve a diverse range of techniques, philosophies, and perspectives.

The working group on Calculation Methodology decided to try to provide common and consistent recommendations for the performance of burnup credit calculations, drawing from

[*] only temporarily present

collective experiences and understanding of key considerations in such an endeavor. However, in recognition of the broad scope of potential applications of burnup credit, no attempt is made to stipulate a detailed recipe that would proscribe any specific step-by-step procedure for performing calculations. Instead, this summary of the working group discussion attempts to provide a higher level overview of a general procedure that should be considered in the development of burnup credit calculation methods. It is hoped that the development of burnup credit methodologies by future parties will be able to benefit from the concepts presented here.

The calculation of criticality safety limits that take some degree of credit for the reduced reactivity in spent nuclear fuel discharged from a reactor is commonly referred to as burnup credit. *Burnup credit is an engineered approach to criticality safety that considers and applies biases, uncertainties, and physics in the solution of a complex problem, such that a realistic, yet conservative representation of irradiated fuel is used in the problem resolution.* As such, this method often requires an iterative approach in which improved solutions are obtained from knowledge gained in previous passes. The process described here is based on several years of experience and represents several iterations in advancing the state of the art of burnup credit. Nevertheless, the burnup credit practitioner who applies this experience is likely to require additional iterations in the development of a burnup credit method for a specific application. Hence, this publication describes a general phased approach for the development and implementation of burnup credit concepts, but allows for further refinement and adjustment according to specific applications, codes/methods used, and available data.

The burnup credit calculation process can be described in terms of four distinct phases:

- Preparation: involves definition of the problem and tools available.
- Depletion: considers the prediction of isotopic concentrations as a function of time.
- Criticality: applies the predicted isotopic concentrations in a criticality safety analysis of the spent fuel management system of interest.
- Implementation: is the follow-up to the calculation than ties the calculation and its assumptions to the actual physical process for treating the spent fuel in the system of interest.

The following sections describe in more detail the issues that should be considered in each phase of the process.

4.1.2. Preparation

The preparation phase of a burnup credit calculation begins with definition of the problem and assumptions made on criticality scenario(s). The level of burnup credit needed will depend on the particular application. One may seek to take credit for only fuel depletion, or for depletion plus production of actinide poisons (typically referred to as “actinide-only” burnup credit), or for actinide and fission-product credit. Burnup credit based on both actinides and fission products is often described as “full” burnup credit, although “actinide plus fission-product” burnup credit usually includes a limited set of nuclides.

The solution will be driven by the specifics of the environment in which fuel is burned. The range of operating histories of the fuel should be used to derive a set of bounding limits that bound the fuel depletion conditions with respect to the reactivity of the spent fuel; this includes not only fuel designs, fuel types, initial enrichments and burnup but also fuel

temperatures, moderator temperatures and densities, use of burnable poisons (if any) and typical control rod/blade insertion. This information drives the depletion phase of the burnup credit process. The criticality phase of the calculation is driven by the post-discharge scenario/configuration for which burnup credit is sought, i.e. the configuration of fuel assemblies, assumptions on moderator (presence, temperature/density, soluble boron), available poisons, etc. Many of these parameters are dictated not by real spent fuel conditions but by the design basis or limiting condition for which burnup credit is sought (e.g. a breached transportation cask fully flooded with water).

The preparation phase is also the time in which the designer begins to identify data that can be used in the validation of calculation methods applied in later phases. Burnup credit calculations will require both depletion/decay (i.e. isotopic composition) and criticality calculations, each that have unique requirements for validation. *The validation process is required to compare calculated results with experimental or theoretical benchmark data*[1)] *in order to demonstrate the applicability of the calculation methods that will be used and to derive the bias and uncertainty associated with each step of the calculation process.* Data used should be applicable to the scenario for which burnup credit is applied; i.e. the range of the data should be representative of the range of conditions of the spent fuel of interest. If existing data over the range of application turns out to be inadequate or insufficient, programs must be initiated at this time. These programs may require the performance of additional measurements or acquisition of appropriate proprietary data. Such efforts will potentially require considerable time and resources, and should be initiated early in this process. In addition, the use of advanced analysis methods may be required which are capable of revealing the applicability of benchmark data to the intended application by quantifying the degree of similarity of the application case to the benchmark data on the basis of the underlying nuclear data characterizing the isotopic compositions and their impact on the reactivity. However, these methods usually include analysis of the application case, and the validation process thus becomes part of the calculation process.

Consider first the validation of the depletion phase (i.e. fuel isotopic composition) of a burnup credit approach. A conservative use of burnup credit requires consideration of all fissile nuclides and hence validation of the calculated concentrations of these nuclides, and allows consideration of any neutron absorbing nuclides for which properties and quantities are sufficiently validated and hence known with sufficient certainty. The data used for validation should be representative of the fuel design and type of interest as well as the operating conditions representative of the fuel to which burnup credit is applied.

In addition to validation of the depletion process, the validation phase of the process will seek to validate methods applied for criticality calculations. Benchmark data are also required for this purpose; however, validation now relates primarily to testing of cross section data for nuclides used in the criticality calculation, and to the effect of environments similar to the application. As with depletion, one should be able to demonstrate the adequacy of data associated with each nuclide for which credit is sought. This may be accomplished in a nuclide-by-nuclide assessment, or by evaluation of integral experiments that assess the global ability to predict the criticality behavior of a set of relevant nuclides.

[1)] Experimental benchmark data = measured data.
Theoretical benchmark data:
(1) Benchmark data derived from measured data (e.g. benchmark configuration derived from a critical experiment).
(2) Results obtained by means of a validated calculation method (i.e. results with known bias and variance).

Data collected for use in burnup credit validation approaches for both depletion and criticality safety methods are available in open literature (more so for validation of criticality safety methods than for isotopic inventory calculations), and this knowledge base continues to grow. Collection of relevant existing data for a given application will also help to identify data deficiencies early in the process, so that avenues may be pursued to remove the deficiencies. Such data, when assembled, provides a mechanism to relate the predictive capabilities of a set of computational tools and associated data to physical measurements. Thus, validation data only have meaning with respect to the calculational tools. Hence, integral in the preparation phase of a burnup credit calculation is the selection of the computational tools to be used in the analysis. The tools selected should be consistent with the particular application. For example, reactor codes methods may appropriate for prediction of isotopic concentrations for relatively short times after discharge, but may be less accurate than other options when decay periods of thousands of years are considered, especially if fission product credit is sought. Open literature again is a valuable guide in selection of tools available for burnup credit analysis.

Once calculational tools have been selected and data for validation of those codes is available, phase one is completed and the burnup credit procedure moves into the next phase — depletion. The following section addresses the depletion phase of the burnup credit calculation approach. This will be followed by a discussion of phase three — criticality.

4.1.3. Depletion

Although depletion and criticality calculations are distinct phases of a burnup credit implementation, the two are intimately coupled in the sense that nuclides used in the criticality phase have their concentrations estimated during the depletion phase. The preparation phase of the calculation is used to determine those nuclides that will be used in the criticality phase. The depletion phase of the calculational process initiates the actual validation process, in which predicted concentrations of nuclides are compared to measured post-irradiation experiment (PIE) data. One method of accounting for the bias and uncertainty in the calculation of the isotopic inventory of the exposed fuel in a burnup credit calculation is to directly compare predicted and measured masses for specific isotopes. Ratios of predicted-to-measured values may be set for each nuclide for which measured data are available, and may be used to determine isotopic correction factors (ICFs). An ICF is a multiplier that may be used to correct a calculated isotopic prediction to be more representative of that which would be expected if another measurement were performed. International literature sources identify a number of approaches for determination of ICFs — both best estimate and conservative estimators of isotopic concentration corrections are available. The intent of the use of ICFs is to correct in a conservative manner for differences between measured and computed values for each nuclide.

Other methods of addressing the uncertainty and bias in the isotopic composition calculation/data may involve integral experiments or differential worth measurements. It is incumbent on the user of these data to provide a clear discussion characterizing the quality of the data utilized and a comprehensive analysis of how the data were used and the adequacy of the final evaluation/representation of bias and uncertainty in the depletion/criticality calculation. The discussion should clearly identify the degree of accuracy or conservatism in the analysis.

Although the set of nuclides for which validation is to be performed was identified in the preparation phase, it is possible that during the depletion validation process it may be found

that insufficient data exists for some nuclides, resulting in an excessive penalty in reactivity due to the bias and/or uncertainty obtained in the validation process. At this point it will be necessary to make a decision to pursue additional data or to omit the problematic nuclide(s). This is also an opportunity to evaluate the computational tools used in the depletion calculation – in general, biases arise not only from PIE measurements but also from the calculational method and associated data (cross sections, branching fractions, etc.) used directly in the calculational approach.

Also necessary for the depletion phase is a determination of the reference state for depletion calculations. Discharged fuel for which burnup credit is to be applied have been exposed over a range of depletion conditions — fuel temperatures, moderator temperatures and densities, soluble boron concentrations (if applicable), assembly designs, control rod histories, use and type of burnable poisons, etc. The range of operations will be driven by the range of fuel designs and types to which burnup credit is applied. Typically bounding depletion parameters are selected that represent a conservative limit for each parameter for the population of fuel for which burnup credit is sought. Care must be taken to remain conservative without accepting an excessively conservative limit. Excessive conservatisms may potentially offset any benefit provided by burnup credit. In addition, it has to be considered that the reference state used for the depletion calculations impacts the applicability of the validation data.

Once the validation process has been completed and bias and uncertainty incorporated for the nuclides of interest, it is possible to begin the depletion analysis process. Using reference state conditions, a broad database may be developed that contains sets of calculated isotopic concentrations computed over a representative range of burnups, enrichments, and assembly designs (including control rod states and burnable poison implementations for each design). Because the selection of final nuclides to be used are independent of the depletion process, depletion calculations can be initiated once the reference state is defined, and can be expanded as the range of fuel is adjusted. Nuclide number densities should be saved for all computed nuclides, to be potentially used at a future time if the list of burnup credit nuclides is expanded. The set of nuclides used in the criticality calculation, and bias and uncertainty corrections applied to these nuclides, is independent of the depletion process itself – these need only be defined prior to beginning the criticality phase of the calculation, and applied to the subset of the depletion database needed for the criticality analysis.

4.1.4. Criticality

While the depletion process is based on the conditions under which fuel is burned, i.e. the reactor environment, the criticality calculation is performed in the actual application environment — spent fuel pool, storage or transportation cask, disposal, etc. The criticality calculation is based on the predicted spent fuel isotopic concentration obtained from the depletion model, but is in an away-from-reactor environment defined by the specific application. In proceeding into the criticality phase, one must now validate the ability to perform criticality calculations, in which the goal is to determine biases and uncertainties in the estimation of the neutron multiplication factor in the application environment.

Typically, the approach to the criticality phase of the calculation is identical to that used in a non-burnup credit approach, with the exception of the fuel composition itself. In general, for any criticality safety calculation, one must begin by defining the base model, varying manufacturing tolerances, fuel geometries, moderator conditions, storage environment, etc, to obtain the most reactive base case. However, for the burnup credit implementation, one must include spent fuel concentrations and evaluate their effect on the system reactivity. The limiting state for assumed fresh fuel may be different than that for spent fuel because of

spectral differences due to the presence of higher actinides and fission products (if included). The accepted approach to this aspect of criticality calculations is to perform sensitivity calculations, in which system eigenvalues are calculated for nominal and perturbed states to determine the conditions that maximize k_{eff}. Calculations should be performed using only the set of fuel nuclides for which burnup credit is sought. Calculations should be performed using both best estimate and conservative (e.g. ICF adjusted) sets of nuclide concentrations, in order to

(1) assess the reactivity margin associated with the approach used to address the bias and uncertainty for fuel composition, and
(2) to determine any changes in the limiting case due to the inclusion of isotopic bias and uncertainty.

The latter concern results from the fact that the use of conservative bias and uncertainty for fuel isotopics may result in significantly non-physical system behavior. For example, use of highly conservative plutonium concentrations could potentially harden the spectrum to a point which could not be obtained under any real operating scenario, and for which relevant critical experiments are not available. Knowledge and judgment must be applied in balancing reality with conservatism. Conservatism is an important aspect of criticality safety, but conservatisms that alter the physical behavior of a system too far from reality may obscure a potentially unsafe configuration.

Direct validation of criticality methods for burnup credit approaches becomes difficult due to the lack of critical experiments with spent fuel. Hence, development of a direct calculational bias for a configuration of interest is unlikely. Validation of criticality methods must therefore be addressed in parts. One should understand the ability of a calculational tool to predict the eigenvalue of representative lattices under storage/transportation configurations — biases in the tool can be calculated for many prototypic conditions but with fresh fuel. Experiments have been performed in which one or more fission product nuclides are included; such experiments continue under various ongoing programs around the world. These will help to provide confidence in the cross section data for each nuclide evaluated, and may indicate deficiencies in the calculational approach or in cross-section data. Other reactivity worth experiments add confidence in cross section data, but are generally not prototypic of the lattice conditions for most burnup credit applications. Reactor critical calculations, in which the critical state is predicted using combined depletion and criticality calculations, provide an integral check of both methods, but are generally not prototypic of away-from-reactor conditions (elevated temperatures, presence of fresh or low burnup fuel, poisons, etc.) and may contain offsetting error components. Numerous approaches have been proposed for determining calculational biases and combining those biases in a criticality safety evaluation, although no consensus has been reached at this time.

Because of the lack of detailed experimental data for calculating biases, the general approach has been to use experimental data “representative” of the neutron spectrum anticipated to estimate bias and uncertainty in as best a fashion as can be done, and use these values to establish an upper sub critical limit. Typically this bias is used in conjunction with a fixed administrative margin. Additional uncertainties are addressed by assuming bounding conditions for other “unknown” parameters of the criticality model to ensure sub criticality. Similar to the conservative assumptions used in the depletion analysis to provide bounding estimates of nuclide concentrations, additional conservatisms are taken in the criticality analysis. These conservatisms may include spatial burnup distribution modeling, use of a bounding/most reactive fuel assembly design, and the omission of significant numbers (if not

all) of fission products and other neutron absorbers, which contribute a large fraction the reduced reactivity of spent fuel. The specific application to which burnup credit is being applied and the requirements of the user are often the key determinants in the degree of conservatism utilized.

In addition to the presence of additional nuclides introduced in a burnup calculation relative to fresh fuel isotopes, reactor operation also introduces axial and horizontal burnup gradients in a fuel assembly. Typically fuel assemblies are more burned near the axial center of the assembly, with significantly lower burnup toward the top and bottom ends of the fuel. Horizontally, burnup will vary across an assembly, depending on operational parameters and core locations in which the fuel is located. Conservatism is obtained by the selection of a most reactive axial profile that will bound all fuel assembly gradients in terms of its reactivity. Similarly, a bounding horizontal gradient may also be employed. The horizontal gradient also will have an axial component, so that simultaneous application of conservative axial and horizontal gradient effects will impart additional conservatism in the spatial model.

If burnup credit is sought for a number of assembly designs, or even for just one design but for a bounding state (e.g. assuming that burnable poison rods, even though only used in the first irradiation cycle in reality, are present in the fuel assembly for its entire exposure history — it should be however checked that this assumption is not overly conservative), then a burnup credit design that can be shown to remain sub critical for a full loading of limiting assemblies will have additional margin due to the fact that no real assembly will be operated at the limiting state. Additionally, if a burnup credit design is based on a given level of burnup, then any assembly actually loaded into the system with a burnup greater than the design burnup will provide additional margin relative to the criticality safety basis for which calculations are performed.

In general, a substantial criticality safety margin is provided by the combined set of conservative assumptions applied in developing the base criticality model. The exact magnitude of this margin cannot be precisely quantified due to the large number of possible variations in a broad range of parameters. However, there is value in attempting to quantify a best estimate reactivity margin relative to conservative assumptions, as a function of enrichment and burnup, to demonstrate the expected criticality safety margin present. This will provide the regulator with a degree of excess margin that can be used to offset uncertainties present and not directly addressed.

Once one has developed a base criticality model or set of models (perhaps as a function of burnup, assembly design, or poison loadings), the entire process should be reassessed. Assumptions made in the early preparation or in the depletion phases of the development may not be consistent with the final form of the model. For example, sensitivity calculations performed may indicate that a certain depletion parameter may not be conservative, or may be excessively conservative. Or nuclides for which burnup credit was initially desired may be found to be of little additional worth in the criticality evaluation, or have insufficient cross section validation data, and therefore may need to be deleted from the criticality model. A consistent depletion/criticality approach must be developed and applied before one is ready to move to the final implementation phase of the process.

4.1.5. Implementation

The preparation, depletion, and criticality phases of the burnup credit must be completed prior to initiating the implementation phase. Iteration may be required in complex cases (e.g. BWR spent fuel systems) in stepping through each of these three phases in order to define the base

depletion and criticality models, nuclides to be used, isotopic correction factors for each nuclide (based on depletion validation), bounding axial and horizontal burnup profiles, and criticality safety margins.

Implementation generally requires the generation of a loading criterion the spent fuel has to meet to be acceptable for loading in the spent fuel system of interest. This criterion is usually presented in form of a curve named as "loading curve" usually indicating the minimum burnup necessary to accept the spent fuel for loading as a function of the initial enrichment of the fuel. For some systems (e.g. the dissolver facility in a reprocessing plant) it is more convenient to present the loading criterion in a different form: Instead of indicating the minimum required burnup the limiting value of a related observable (e.g. the maximum allowable fissile mass) is determined as a function of the initial enrichment.

In whatever form a loading criterion is presented the related loading curve is defined by the maximum neutron multiplication factor allowable for the spent fuel system of interest, including all mechanical tolerances and calculational biases and uncertainties. In other words, a loading curve is generated by a reactivity equivalence relation. The validity of this relation and hence *the validity of the loading curve is restricted to the set of defining criteria (assumptions made in the burnup credit process to this point)*:

- the fuel design(s) and type(s) considered,
- the depletion parameters and conditions assumed (specific power and power history; fuel temperature; moderator temperature and density; presence of soluble boron in the moderator; presence and design of BPRs; control rod insertion history and control rod design; core loading strategies, e.g. out-in or in-out strategies, usage of MOX fuel; cooling time),
- the bounding axial and horizontal burnup profiles used,
- the design of the spent fuel system of interest and the design basis for which burnup credit is taken.

This set of defining criteria is characterized by a set of parameters which have to be controlled in the implementation phase additionally to the parameters specified by the loading curve (initial enrichment, minimum required burnup or the limiting value of a related parameter).

So, each loading curve generated will have associated with it a range of applicability based on the assumptions made in the procedure used to generate the loading curve. Therefore, each loading curve shall have associated with it a detailed description of all processes, data, and assumptions used in generating the loading curve. Constraints placed on a loading curve (e.g. exclusion of usage of BPRs) shall be clearly identified. This will allow independent verification of the process used for generating the curve, and will permit the inclusion of additional future fuel types that were not explicitly analyzed but that can be demonstrated to fall within the envelope for which the loading curve was derived.

4.1.6. Summary

Burnup credit is a concept that takes credit for the reduced reactivity of fuel discharged from the reactor to improve loading density of irradiated fuel assemblies in storage, transportation, and disposal applications, relative to the assumption of fresh fuel nuclide inventories in loading calculations. The penalty of the fresh fuel assumption, especially for newer, higher-enrichment fuel designs, can be especially costly and in some cases even non-conservative (i.e. low burnup, Gd-loaded BWR fuel). Additionally, existing storage facilities that are near

or at capacity based on fresh fuel assumptions can have their licensed capacity increased by utilizing burnup credit, also with significant cost savings. However, the burnup credit concept, while more credible and more closely representative of the actual state of discharged fuel, is much more complicated than the straightforward fresh fuel assumption. Burnup credit has been pursued largely for reasons of economics, because increased cask loadings will reduce the number of casks required for storing fuel, and because it will allow increased utilization of spent fuel cooling ponds. Furthermore, the number of shipments required to move a given number of fuel assemblies from one point to another can be significantly reduces. For storage, the size and capacity of a facility can be significantly reduced due to higher density loadings possible based on burnup credit calculations. But beyond economic, and certainly more importantly, proper application of burnup credit can reduce the risk of accidents and potential exposure by reducing the number of shipments needed in spent fuel transportation outside of a reactor facility. Because of these benefits, numerous organizations worldwide have studied calculational methods for accurate quantification of burnup effects, sought data to assess the validity of such calculations, and considered methods to combine measured data in many forms together with calculational models to make a solid case for criticality safety while at the same time taking credit, in part, for the reduced reactivity of spent nuclear fuel. Two decades of research, development, and in some cases implementation, have led to a general understanding of the best general process for applying burnup credit for particular applications. The intent of this report is not to spell out in detail the specific approach to be applied in a burnup credit application. Rather, it is simply informed guidance, developed by a number of experienced burnup credit analysts and practitioners from a range of countries and backgrounds.

In the development of this guidance, three issues were considered for which it was felt insufficient guidance was readily available. There was no solid position available for defining how to adequately address criticality validation; however, this issue was being addressed concurrently in an independent working group, so no recommendations were made within this working group. The second issue that was felt to be insufficiently resolved for practical application was the most appropriate method of treatment of isotopic uncertainties. Isotopic correction factors were discussed, but no details were provided as to how best to treat isotopic uncertainties in criticality evaluations. The ability to calculate correction factors and assess uncertainties is well developed, but methods to combine these effects in a meaningful manner in terms of their net effect of criticality are not as well defined. The concept of taking a maximum penalty for each nuclide used has been considered in a number of studies, but is generally considered to be unphysical and excessively conservative. Improved recommendations for combining isotopic corrections in a conservative yet more statistically meaningful manner are desired. Finally, with respect to implementation, it was felt that there is a need to develop guidance as to what constitutes a complete set of documentation for a burnup credit implementation.

This report has described a general four phase approach to be considered in burnup credit implementation. Much if not all of the background research and data acquisition necessary for successful burnup credit development in preparation for licensing has been completed. Many fuel types, facilities, and analysis methods are encompassed in the public knowledge base, such that in many cases this guidance will provide a means for rapid development of a burnup credit program. For newer assembly designs, higher enrichment fuels, and more extensive nuclide credit, additional research and development may be necessary, but even this work can build on the foundation that has been established to date. Those, it is hoped that this report will serve as a starting point with sufficient reference to existing knowledge and experience to be able to expedite future burnup credit program development.

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4.2. Validation and Criticality Safety Criteria

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4.2.1. Introduction

This group was assigned the task of reviewing the codes validation and the criticality safety criteria used in the application of Burnup Credit (BUC). The review has been made and is presented in the following main stages:

- Codes and nuclear data libraries
  - Codes and libraries used by participants
  - Process for verification, validation and qualification (experimental validation)
  - Experiments used for BUC qualification
  - Recent improvements in international data files (cross sections and fission product yields)
- Status of integral experiments for BUC qualification
  - Post Irradiation Experiments (PIE) for fuel inventory qualification (actinides and fission products)
  - High burnup challenge
  - Worth measurement of BUC nuclides
  - Global burnup worth experiments: Spent fuel measurements and LWR follow-up
  - Experimental needs and future programs
- Criticality safety criteria
  - Bounding conditions in assembly depletion calculation
  - Isotopic biases and penalty factors

[*] only temporarily present

- Errors in BUC nuclide worth prediction and bounding reactivity bias
- Sensitivity analyses: Representativeness of an experiment, propagation of nuclear data uncertainty, integral measurements and a-posteriori uncertainty
- Axial burnup databases and generation of bounding profiles.

The main issues of the group's discussion as well as the conclusions drawn and the recommendations made by the group are summarized in the following sections.

4.2.2. Codes and Nuclear Data Libraries

4.2.2.1. Summary of codes and nuclear data used by participants

A summary of information on codes used by the participants is presented in Table 4.1. It is seen that there is a wide range of techniques including deterministic and Monte Carlo methods and covering a range of energy group schemes.

4.2.2.2. Verification – Validation – Qualification Process

The first step in establishing the accuracy of a code/data package is usually based on numerical testing of the code (reliable algorithms, no bugs and regression in new code version) and its nuclear data library. This "Verification" phase normally includes checks to ensure that the processed cross-sections accurately represent the information contained in the basic nuclear data file.

"Validation" of the code may then be made by comparison with reference methods, such as continuous energy Monte Carlo. This validation process which aims for calibrating calculation biases (multigroup assumption, resonance self-shielding model, anisotropic scattering, etc) is generally carried out on representative simplified geometries (numerical benchmarks). Both standard route calculation and reference calculation must use the same nuclear data file.

Once this validation process has been performed, the global accuracy of the code/nuclear-data package is validated through benchmark experiments. This third phase is called "Experimental Validation" or "Qualification". In the context of BUC the qualification process generally consists of analyzing PIE data (to validate depletion calculations) as well as critical benchmark experiments and Spent Nuclear Fuel (SNF) worth measurements (to validate criticality, i.e. k_{eff} calculations). Some participants pointed out that, in the specific case of criticality safety, numerical and experimental validation are often gathered in a unique phase, as the general goal remains to compare Calculated (C) to Experimental (E) k_{eff} values and to acquire thus knowledge of the C/E in k_{eff} of the code scheme, according to the considered application.

Table 4.1: CODES USED FOR BURNUP CREDIT CALCULATIONS

Country/Organisation	**Depletion**		**Criticality**	
	Code	Data	Code	Data
Belgium/Tractebel	LWRWIMS	UKNDL	MCNP	ENDF-BV CE*
Belgium/BN	WIMS8	JEF2.2	KENO5A DANTSYS	JEF2.2 172g
Bulgaria/VVER-440	SCALE4.4	ENDF-BV	SCALE4.4	ENDF-BV 44g
Bulgaria/VVER-1000	NESSEL/NUKO	ENDF-BIV	SCALE4.4	ENDF-BV 44g
Czech Republic	SCALE4.4a/5	ENDF-BV	SCALE4.4a/5	ENDF-BV 44g
France	DARWIN- APOLLO2 CESAR	JEF2.2	CRISTAL TRIPOLI4	JEF2.2 172g JEF2.2 CE
Germany/AREVA	CASMO3/4 SCALE 5 KORIGEN	ENDF-BV/VI / JEF2.2 / PWR specific	SCALE-5 MCNP-5	ENDF-BV 44g ENDF-BV 238g ENDF-BVI CE
Germany/GRS	KENOREST-98	JEF2.2	KENOREST-98	JEF2.2
Germany/WTI (PWR)	SCALE4.4, HELIOS/SNF	ENDF-BV	SCALE4.4, MCNP	ENDF-BV/VI
Hungary	SCALE4/5 MULTICEL	ENDF-BVI	SCALE4/5 MCNP4c/5	ENDF / JEF / JENDL
Japan/JAERI	SRAC	JENDL3..2	VIM	JENDL3..2 CE
Japan/JNES	SRAC SCALE4	JENDL3..2 ENDF-BV	MVP SCALE4.4	JENDL3..2 CE ENDF-BV 44g
Russia/Kurchatov	KASSETA	BROND	MCU	BROND/ENDF
Russia/IPPE	CORE, ORIGEN	FOND-2.2 file	MMK, KENO-5a	ABBN 299g (FOND-2.2 file)
Slovakia	SCALE4.4/5	ENDF-BV	SCALE4.4/5	ENDF-BV 44g
Spain/ENUSA (PWR)	PHOENIX	ENDF-BV	SCALE4.3	ENDF-BV 44g
Spain/ENUSA (BWR)	TGBLA	ENDF-BV	SCALE4.4	ENDF-BV 44g
Spain/CSN	CASMO4	ENDF-BV	SCALE5	ENDF-BV 238g
Spain/SEA	MONTEBURNS	ENDF-BVI	MCNP	ENDF-BVI
Sweden	SCALE5	ENDF-BV/VI	KENO5a	ENDF-BV
Switzerland/PSI	BOXER	JEF1	BOXER	JEF1 20g
UK/BNFL	WIMS8 FISPIN	JEF2.2	WIMS8 MONK8	JEF2.2 CE
USA/Westinghouse	PHOENIX	ENDF-BV	KENO	ENDF-BV
USA	SCALE4.4 SCALE 5/5.1 CASMO4	ENDF-BV ENDF-BV ENDF-BV	SCALE4.4 SCALE 5/5.1 MCNP	ENDF-BV 44g ENDF-V 238g ENDF-BVI CE

* CE – Continuous Energy

Verification - Validation of the codes is usually carried out by the code developer, often in the context of a formal Quality Assurance (QA) framework. Typically the QA programme will include recommended procedures for optimal code utilization and the methodology for identifying, correcting faults, notifying the user community of such faults and any resulting corrections/improvements.

Experimental validation of the code is more commonly carried out by the users, although some differences in approach were noted. In particular, the French code scheme for BUC is developed, verified and validated by a team of specialists. Following this process a closely defined code scheme is released to the user community. No ad hoc changes to the code package are permitted; therefore, bias and associated uncertainty can be generated in a "semi-automatic" way from the validation dataset provided to the user. In other countries the responsibility for validation is entirely placed on the code user; and it is worthy of note that some regulatory bodies require the carrying out of the code validation of the user as a part of demonstrating the user's competence.

The following conclusions were also drawn:

- As a result of the increased validation requirement for BUC, there are significant advantages in the use of modern codes and data packages which have been verified and validated through coordinated programmes.
- For some BUC application areas validation data is limited, so that the use of approximate methods, which may contain compensating errors can lead to significant extrapolation uncertainties.
- It could be dangerous to use automated calculations of numerous ICSBEP benchmarks2), based on code data proposed in the Appendix of the ICSBEP evaluation, because on the one hand these code data could be wrong and on the other hand the physical analysis of the experimental information is missing.
- Only few validation guides exist: one ANS standard (USA), the German safety code KTA 3101.2 for validation of depletion calculation codes, and the German standard DIN 25478 for validation of criticality calculation codes.

4.2.2.3. BUC validation experiments

In general PIE data is used to validate depletion calculations, and critical experiments including SNF and BUC nuclide worth measurements are used to validate the criticality calculation code. One observes that, on the contrary to criticality experiments involving fresh fuel, few PIE and criticality experiments with spent fuel exist and are available for BUC validation. Participants also noted that evidence of code accuracy may be deduced from comparisons with LWR BOC measurements (CRC – Commercial Reactor Criticals) and boron let down during the fuel cycle in PWRs, although it is recognized that there are difficulties in applying reactivity effects in a hot reactor core to an accurate derivation of code bias for spent fuel environment.

In most countries depletion calculations have been validated against PIE data from public domain (OECD SFCOMPO database) or proprietary programmes. An important exception arises in Eastern Europe where there is an acute shortage of PIE data for validation of VVER depletion calculations; however, recent Russian results from the ISTC programme just became available. The participants noticed that the useful SFCOMPO database addresses only PWR and BWR fuels; furthermore this database should be enlarged to increased enrichments and larger burnup range. These new experiments should be more documented, with a detailed irradiation history.

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2) Benchmarks from the "International Handbook of Evaluated Criticality Safety Benchmark Experiments", NEA/NSC/DOC(95)03/I through VIIa, Nuclear Energy Agency, OECD Paris

Participants consider that the ICSBEP is a useful experiment database but not completely satisfying for BUC validation, due to the lack of BUC nuclide worth measurements as well as SNF experiments.

BUC nuclide worth measurements are very valuable because they allow direct validation of each BUC component. Furthermore these experiments enable actinide and Fission Product (FP) cross section improvements. Unfortunately, these BUC nuclide measurements are only performed in MINERVE reactor by oscillation of fuel samples containing one separated isotope. Some of the separated FP samples were also measured in the DIMPLE reactor in the frame of the CERES CEA-UKAEA collaborative programme. Six natural elements (Sm, Cs, Rh, Gd, Nd, and Mo containing the major BUC FP isotopes) were investigated in the Valduc Appareillage-B.

SNF worth measurements are required for the validation of BUC calculations because they supply the total burnup credit that allows the demonstration that a proposed BUC methodology is conservative. Two techniques are used in these experiments:

- Introduction of a SNF bundle at the center of the driver core of a zero power reactor: REBUS experiment in VENUS reactor, PWR/BWR assembly insertion at the center of PROTEUS.
- Oscillation of SNF rod sample at the center of the MINERVE reactor: This technique allows investigating the effect of different initial enrichments and spanning the total burnup range. Moreover the SNF samples are oscillated in various spectra (such as UO2-LWR, MOX-LWR and High Conversion LWR).

4.2.2.4. Recent Improvements in International Data Files

BUC experimental programmes have heavily contributed to the improvement of international datafiles ENDF-B, JEFF and JENDL.

In the European library particularly, the nuclear data evaluation of main actinides and FPs were modified from the previous JEF2.2 to the current JEFF3.1 file:

- U235 and Pu241 evaluations were modified with a significant +6% increase of their resonant (n,γ) cross section. Am241 evaluation was strongly modified with a +15% augmentation of the thermal-epithermal capture. A huge work on U238 resonance range reevaluation was performed in the framework of the OECD-WPEC-sg22 Group; a new JEFF3.1 evaluation was adopted with a 0.7% reduction of the U238 resonance integral. All these new evaluations modify the reactivity of the main BUC actinide nuclides.
- Concerning fission products, Nd143 evaluation was modified in order to fit the $\sigma 2200 = 338$ b cross section derived from Nd143 sample worth in MINERVE. Cs133 capture was reduced, that is consistent with recent Nakajima differential measurement and MINERVE results. Sm149 capture in the large first resonance was increased (+3% on Γn value). Rh103 cross sections were reevaluated on the basis of the recent measurement at the European GELINA LINAC. Thanks to the FP PIE results, the wrong JEF2.2 evaluations for Europium isotopes were corrected: ENDF/BVI.7 evaluations were adopted for Eu154 and Eu155.

US participants stated that FP evaluations in the future ENDF/BVII file will not be updated from the current ENDF/BVI.8.

4.2.3. Status of integral experiments for BUC qualification

Three kinds of integral experiments may be distinguished and are described in more details in the hereunder paragraphs. Tables 4.2 and 4.3 give an overview of the programmes available or in progress.

4.2.3.1. Post irradiation experiments for fuel inventory qualification

Several countries have their own PIE programmes, based upon domestic experiments or participation to international collaboration (e.g. ARIANE). The results obtained through those are often of a proprietary nature and the data are basically restricted to the participants of those programmes.

France undertook a large PIE programme, with PWR UO2 assemblies in the 2% - 4.5% enrichment range irradiated in Fessenheim2, Bugey3, Gravelines3 (from 10 to 60 Gwd/t), Cruas2 (up to 70 Gwd/t) as well as PWR MOX assemblies coming from St. Laurent des Eaux, Dampierre2 (from 10 to 60 Gwd/t). Radiochemical assays were also performed on a BWR assembly irradiated in Gundremmingen. Both major-minor actinides and BUC-FPs were analysed. The challenging problem of the total dissolution of metallic FPs for accurate chemical assays was discussed. Moreover CEA is presently participating to the MALIBU international programme.

Japan is involved in many PIE international programmes but also took the initiative to establish a database of PIE measurements for PWR and BWR fuel samples. The database, SFCOMPO, is available to members of the OECD through the Nuclear Energy Agency.

In the US, some TMI samples are openly available but they are however of poorer quality. US therefore participated to the ARIANE and recently engaged effort in other PIE programmes.

Among the international collaborations, one may quote the R&D programmes promoted by the Belgian company BELGONUCLEAIRE, sometimes associated to SCK•CEN. Such programmes are of proprietary nature and find customers in Western Europe, Japan and in the US. Three of them should be mentioned:

- GERONIMO / TOPGUN (TOPGUN being an extension of GERONIMO): these programmes primarily address BWR MOX fuel rod behaviour under irradiation up to high burnup (about 80 GWd/t) but offer PIE analysis for the major actinides, such that the PIE information can reveal valuable for the BUC community. These programmes are nearly completed.
- MALIBU: it is the extension of the previous ARIANE programme, but addressing higher burnup (also about 80 GWd/t) and providing scope improvements as compared to ARIANE, e.g. samples chosen from standard fuel coming from standard Nuclear Power Plants (NPPs), reduction of the decay time between EOL state and radiochemical analysis, radiochemical assay performed by at least 2 independent laboratories, etc. The scope of MALIBU comprises samples of different kinds of fuels irradiated in German and Swiss NPPs: PWR UO2, PWR MOX and BWR MOX. Discussions are still going on for a possible extension of the scope for BWR UO2 fuel. The nuclides list (actinides + fission products) addresses the needs of BUC community as well as those relevant for waste management (source term). The MALIBU programme is in progress and should come to an end around 2006.
- REBUS-PWR: is a R&D project that combines both aspects of criticals and PIE measurements. The scope includes the study of five critical configurations that are

loaded in the VENUS facility. They consist of a driver zone surrounding a central test bundle, which is successively composed of fresh and irradiated MOX and UO2 fuels. The rods are both being characterized by non-destructive and destructive examinations, for their criticality relevant composition (actinides + 19 fission products). REBUS-PWR brings 2 radiochemical assays, one PWR UO2 3.8 w% at a burnup of 54 GWd/t and one PWR MOX Pu-fiss 6.75 w% at only 22 GWd/t but characterized by a large decay time of 15 years. REBUS-PWR was completed during summer 2005. It is worth noting that it is followed by a REBUS-BWR-MOX programme using the BWR MOX fuel that was considered for MALIBU.

As far as the VVER application area is concerned, the ISTC project No.2670 on VVER PIEs in Research Institute of Atomic Reactors (RIAR), Dimitrovgrad, Russia, funded by US and led by pair of the managers of Lawrence Livermore National Laboratory (LLNL), California, and RIAR, Dimitrovgrad, was completed in early 2005 (see Chetverikov et al., this meeting). The final report entitled as 'Radiochemical Assays of Irradiated VVER-440 Fuel for Use in Spent Fuel Burnup Credit Activities' was issued in LLNL (UCRL-TR-212202) being released without any restrictions. The work was performed under the auspices of the U.S. Department of Energy by University of California, LLNL.

TABLE 4.2: OVERVIEW OF THE POST IRRADIATION EXPERIMENTS

Project name	**Fuel type**	**Organizing country / organization**	**Status**	**Access**
ARIANE	PWR UO2, PWR MOX BWR UO2 BWR MOX	Belgium	Completed	Restricted
French Programme	PWR UO2, PWR MOX BWR UO2	France	Ongoing Completed Completed	Restricted
MALIBU	PWR UO2, PWR MOX BWR MOX (BWR UO2)	Belgium	Ongoing	Restricted
REBUS-P REBUS -B	PWR UO2, PWR MOX BWR MOX	Belgium	Completed Ongoing	Restricted Restricted
ISTC N°2670	VVER-440	Russia (RIAR)	Completed	Open
SFCOMPO	PWR UO2, BWR UO2	Japan / OECD	Ongoing	Open (OECD members)
TMI	PWR UO2	USA	Completed	Open

The data for eight spent fuel samples resulting from the project became the first VVER PIE data, which can be used for comparison with predicted concentrations of the "OECD BUC isotopes" (actinides and major fission products). The participants recommend the inclusion of the data as well as any further VVER PIE data in the SFCOMPO database.

Based on the measured sample data and operation history a new international benchmark focused on the VVER 440 inventory prediction is intended to be specified by the analysts working in the VVER environment.

As in the VVER application area there has been a lack of released well documented PIE data for many years, the data of the ISTC project No.2670 are appreciated much by the VVER criticality safety analysts. However, the obtained data of only eight samples still represent a too small statistics and moreover do not cover sufficiently the current application range of depletion conditions, as well as designs of the VVER fuel assemblies currently used in the VVER 440 and 1000 reactor units. That is why a further VVER PIE project is being proposed and first negotiations are being made as for its funding. The Working Group concurs indeed that 8 samples is not sufficient.

The crosschecking principle (applied in MALIBU for instance) has been mentioned for such VVER fuel. Unfortunately shipment of spent fuel across the Russian borders is forbidden.

Table 3.2 summarizes the PIE experimental programmes for BUC qualification.

4.2.3.2. Worth measurement of BUC nuclides

The second kind of experiment for BUC qualification consists in reactivity worth measurements for individual nuclides. Although some work was performed in the US (Rh103 detectors at SANDIA lab), most data come from the French programme. In particular it is noted that the French programme for fission products is presently completed.

The poisoning worth of the 14 main BUC-FPs was measured in the MINERVE reactor at Cadarache. The measurement was carried out by oscillation of separated fission product samples: Mo95, Tc99, Rh103, Ag109, Cs133, Nd143, Nd145, Sm147, Sm149, Sm152, Eu153, Gd155, and Ru, Mo, Ag, Nd, Sm element samples.

Four experiments were performed in the framework of the Burnup Credit programme:

- R1-UO2 devoted to BUC investigation in storage pool and PWR-assembly transportation,
- R2-UO2 with a softer spectrum corresponding to the optimum moderation ratio in a fuel dissolver and
- R1-MOX that corresponds to oscillations at the center of a MOX Pu4.0w% lattice.

The VALDUC fission product experiments were performed in three gradual steps at VALDUC in ‘Apparatus B’ by using the sub-critical approach technique based on the rising of moderating and reflecting water of a driver array. In the centre of the driver, FPs are in solution in a Zr tank, alone or mixed, with or without interactions with U, Pu and Am241. The following isotopes have been studied Rh103, Cs133, Sm149, as well as Gd element (Gd155 + Gd157 capture) and Nd element (Nd143 + Nd145 capture). The series of experiments was aimed at testing the capacity of the codes to calculate some critical situations representative of storage, transport and dissolution:

- The first series of experiments, called ‘Physical’ type experiments, is representative of storage and transportation conditions: the square pitch of the driver array (1.3 cm) leads to a thermal neutron spectrum, representative of the nominal square pitch (1.27 cm) of storage and transportation. The 1.3 cm square pitch accounts for assembly water holes.
- A second series of experiments, named ‘Elementary Dissolution’ type, has been performed to improve the ‘dissolution’ qualification (whose neutron spectrum is more thermal than the previous ones). FPs are then in close interaction with the U, Pu & Am isotopes of inner array of UO2 rods or “Haut Taux de Combustion” (HTC) rods (with

a square pitch of 1.272 cm) in the Zr tank. The UO2 rods have an initial fuel enrichment of 4.738 wt% U235, and the HTC rods simulates U and Pu composition for a UO2 fuel with initial enrichment of 4.5 wt% U235 irradiated at 37 GWd/t, without FP. This second series is itself divided in two cases: FPs in acid solutions (HNO3 - 1N) or FPs in Depleted Uranyl Nitrate Solution (DUN). All these cases are completed. Other experiments were performed on natural Gd solution, on 95Mo in thin slices of CH2/natural metallic Mo, and on F as polytetrafluorethylene (PTFE) solid block. In all, 156 experiments have been performed.

- A third series of experiments, named 'Global or Advanced Dissolution' type, is planned. It consists of a large SS tank (70.4 x 70.4 cm2) containing a 44 x 44 HTC rod array steeping in a DUN solution poisoned with 6 FPs. The 1.6 cm pitch leads to an even more thermal neutron spectrum, more convenient to 'dissolution' qualification if necessary.

France is now considering the OSMOSE programme addressing the actinides, including Am241, Am243 and Cm isotopes. Each separated actinide isotope (from Th232 to Cm245) is mixed to a UO2 support in oscillation samples. The oscillation of these actinide samples has just started in the PWR-UO2 experimental lattice of the MINERVE reactor. This programme, also motivated by other purposes (transmutation, high burnup, nuclear cross section assessment), should allow deriving the actual penalty factors associated with JEF2.2 and JEFF3.1 libraries, to be accounted for in the BUC criticality studies.

TABLE 4.3: OVERVIEW OF THE REACTIVITY EXPERIMENTS

Experiment	**Nuclides**	**Technique**	**Organizing Country**	**Status**	**Access**
AppareillageB	FPs	Criticals	France	Completed	Restricted
OSMOSE	Actinides	Oscillations	France	Ongoing	Restricted
CERES	FPs	Oscillations	France/UK	Completed	Restricted
Minerve LWR UO2 & MOX	Spent fuel	Oscillations	France	Completed	Restricted
REBUS-P -B	Spent fuel	Criticals	Belgium	Completed Ongoing	Restricted
PROTEUS	Spent fuel	Criticals	Switzerland	Completed	Restricted
SANDIA	Rh103	Foil	US	Completed	Open

4.2.3.3. Global burnup worth experiments

"Commercial Reactor Criticals" (CRC) could be included in this category. The use of such data for BUC qualification is generally quoted as difficult to analyze. Global worth experiments performed in an experimental facility should be preferred. Among those experiments, one distinguishes between oscillations and direct criticality techniques.

The French programme includes oscillations of spent fuel rod samples in the MINERVE facility (fuel inventory is known through chemical analyses performed on contiguous pellets of the oscillation sample). The burnup of the samples are varied from 0 to 60 GWd/t for PWR-UO2 samples, from 0 to 45 GWd/t for MOX, and from 30 to 45 GWd/t for BWR samples. PWR and BWR rod cuts were oscillated at the center of the LWR 1.26cm pitch regular lattice, meanwhile the MOX spent rods were oscillated at the center of the R1-MOX lattice.

For the second technique, one finds the REBUS-PWR project that associates both aspects of PIE and criticality measurements, performed in the VENUS critical facility (SCK•CEN). The fuel is either at fresh or burnt state. In the latter case it is irradiated in a standard way, up to a high burnup. Such a global reactivity worth approach makes the burnup effect very demonstrative and provides a kind of "real" case test for BUC validation and implementation. The reactivity loss induced by the burnup, assessed through a critical water level difference (20 - 30 cm) between the couple of configurations, is large enough to reduce the relative importance of all uncertainty sources. Such a reactivity effect is about -1900 pcm and -2300 pcm for the MOX and the UO2 fuel, respectively.

Participants finally cited the PROTEUS (PIE + spent fuel reactivity + individual fission product cross-section) programme, at PSI, Switzerland. It is worthy of note that the PROTEUS programme include burnup values up to about 80 GWd/t.

Table 3.3 summarizes the reactivity experimental programmes for BUC qualification.

4.2.3.4. High burnup challenge

The trend towards higher burnups, and thus higher enrichments, is more challenging for the rod mechanical stability, corrosion and fission product release, rather than for the BUC itself. However it is noted that high burnup of MOX fuel, for instance, is characterized by a non linear increase of some minor actinides, having sometimes a large positive reactivity contribuion (e.g. Cm245).

In some cases PIE data is a little limited for higher initial enrichment (IE), but it is noted that no trend with IE is seen for many code/data schemes. Generally it is concluded that coverage of burnup is the more important parameter.

4.2.3.5. Experimental needs and future programmes

As stated during the 2002 TCM on BUC in Madrid (cf. IAEA-TECDOC-1378), the main future developments in BUC could be anticipated to arise in the following areas:

- Increase in initial enrichment and burnup for PWRs.
- Development of BUC for BWRs.
- Development of BUC for MOX fuels; due to the required MOX and UOX assembly equivalence, Pu load up to 12w% and burnup up to 60Gwd/t must be considered.
- Move to "full" BUC credit, including fission products, for applications different from PWR wet storage systems.
- Such developments could motivate new experiments, like those recently performed within the French programme addressing higher burnups (70GWd/t for UO2 and 60GWd/t for MOX), REBUS-like experiments for various spectra and high burnup MOX fuel, etc. However more immediate needs are quoted: First it is recalled the Eastern European countries running VVER kind of NPP do have today the opportunity to implement a rigorous validation on VVER fuel chemical assays, and that this brand new database is expected to be enlarged in the future.
- Second the proprietary nature of some experimental programmes is of concern by the same Eastern European countries. The acquisition of some data from such proprietary programmes requires the approval of the fuel manufacturer and of the fuel owner.

4.2.4. Criticality safety criteria

4.2.4.1. Bounding conditions in assembly depletion calculation

The reactivity of spent fuel is affected by the degree of spectrum hardening caused by the depletion conditions. Keeping all the depletion parameters constant (specific power, fuel temperature, moderator temperature and density, presence of soluble boron in the moderator, presence of burnable poison rods, control rod insertion - cf. working group report given in section 3.1), increasing of the fuel's burnup results in an increasing hardening of the neutron spectrum during irradiation of the fuel. Additional spectrum hardening due to a change in any depletion parameter (e.g. decrease in moderator density) or any depletion condition (e.g. change in the core environment due to usage of MOX fuels) results, compared to the case of unchanged depletion conditions, in an increase of the reactivity of the spent fuel. Possible changes in depletion parameters and conditions have therefore to be covered by choosing bounding depletion parameters and conditions.

- Specific power:

  It was agreed that a change in the specific power has a negligible effect on the reactivity of the spent fuel. However, participants pointed out that the value used for the specific power during depletion calculations has a complex, but slight effect on the reactivity depending on the burnup credit model (actinide-only or actinide + FP BUC) and the cooling time.

- Fuel (pellet) temperature:

  It is conservative to use a high value for the fuel temperature as it leads to more resonant captures on U238 and hence to more production of Pu239. After the first irradiation cycle, the pellet average temperature usually decreases below 600°C. Therefore, usage of a pellet average temperature of Tf = 600°C were often recommended in the past, but meanwhile it was also proposed (in USA in particular) to use Tf = 1000K as conservative fuel temperature.

- Moderator temperature and density (PWR):

  Participants agreed on the use of the water outlet temperature, which corresponds to the lower moderator density and consequently to the higher conversion factor.

- Moderator temperature and density (BWR):

  Due to the physics of an operating BWR the moderator temperature changes very little axially once the height were boiling begins is reached, but the moderator density significantly changes axially since the void fraction increases with increasing height. Due to the variations in the axial power peaking in an operating BWR the void fraction can change significantly both axially and as a function of time. It is obvious, therefore, that depletion effects have to be studied as a function of moderator density or void fraction instead of moderator temperature.
  Variations in moderator temperature are in fact of no interest. In an operating BWR the outlet temperature usually is about 560 K. A variation of this temperature by 5 K results in a variation of the pressure by more than 5 bar (= 5·105 Pa = 72.52 psi). In reality variations of the mean core pressure are less than 2 bar (under stretch-out operating conditions less than 2.5 bar). So, actual variations in the core moderator temperature are irrelevant to burnup credit criticality safety analysis of BWR spent fuel.

- Soluble boron concentration of the moderator (PWR):

An increase in the B10 concentration of the moderator results in spectral hardening due to stronger absorption of thermal neutrons. Consensus was reached that use of a cycle-averaged boron concentration represents a bounding condition. (In many cases a cycle-averaged boron concentration of CB = 600 ppm may be bounding, but the actual bounding cycle-averaged boron concentration should be derived from the boron let-down curves of all the cycles to be considered.)

- Presence of integral burnable absorbers in the fuel design:

  The effect of integral burnable absorbers initially present in a fuel design of interest should be studied in a sensitivity analysis on the spent fuel reactivity. The presence of so called "IFBA" rods (fuel rods with boron-coated pellets) in particular can result — compared to the same but unpoisoned fuel design — in an increase of the reactivity of the irradiated fuel after burnout of the absorber.
  Since integral burnable absorbers are usually used in the center region of the fuel zone only the initial presence of the absorber can impact the axial end effect (i.e. the reactivity effect due to the axial distribution of the burnup) even after burnout of the absorber.

- Usage of removable Burnable Poison Rods (BPRs):

  BPRs, inserted in guide thimbles of fuel assemblies during irradiation, are usually removed at the end of the first irradiation cycle of the fuel assemblies. As noticed by US representatives, the increase Δk in reactivity of pool storage or transport cask due to the usage of BPRs in the first cycle is less 0.01. The usage of BPRs may impact the axial end effect.
  In contrast to many other countries BPRs are not used in Germany (with the exception of the first cycle, where the initial enrichments were however low so that a burnup credit, if needed at all, is of very small amount).

- Control Rod (CR) insertion (PWR):

  In France the depletion calculations for UOX fuel assemblies is carried out with CRs fully inserted throughout all the irradiation. This procedure results in a significant increase of the reactivity of the fuel at end of life and thereafter: As noticed by French representatives, reactivity increases in the range of $\Delta k = 0.03$ to $\Delta k = 0.04$ have been observed for pool storage racks or transport casks loaded with 40 GWd/t 17x17 assemblies. In the ongoing Phase II-E Burnup Credit Benchmark conducted by the Expert Group on Burnup Credit Criticality Safety under auspices of the NEA/OECD it has been found that the assumption of full CR insertion during the entire irradiation time increases the reactivity of a conceptual transport/storage cask by about $\Delta k = 0.035$ for loadings with 30 GWd/t 17x17 fuel assemblies and by about $\Delta k = 0.06$ for loadings with 50 GWd/t 17x17 fuel assemblies. In addition it is observed that it may happen that a "bounding" axial burnup profile does not remain bounding (i.e. results in a negative end effect, so that the uniform distribution of the average burnup then represents the bounding profile) when the CR insertion depth becomes greater than about the half of the active fuel length (see J.C. Neuber, this meeting, Figures 8 and 9 in the paper entitled "Calculation Routes to Determine Burnup Credit Loading Curves").

  In USA PWRs typically do not operate with CRs inserted. The tips of the rods may however rest at the fuel ends, which results in an insignificant reactivity effect ($\Delta k < 0.002$) on a burnup credit cask. Studies performed by Oak Ridge National Laboratory (ORNL) under auspices of the U.S. NRC show that full CR insertion for burnups around 5 GWd/t leads to an increase in cask keff values of the same order as observed for BPRs. Therefore, since BPRs and CRs cannot be inserted in an assembly

at the same time, it follows that inclusion of BPRs in the assembly irradiation model (up to burnup values that encompass realistic operating conditions) adequately account for potential reactivity increase that may occur for spent fuel exposed to CRs during irradiation (for more details see C.V. Parks et al., this meeting, paper entitled "U.S. regulatory recommendations for actinide-only burnup credit in transport and storage casks").

As in USA, to avoid burnup delays, PWRs in Germany typically do not operate with CRs inserted, although the tips of the CRs may rest at the fuel ends. Since BPRs are not used in Germany it may be demonstrated by means of sensitivity studies on plant-specific bounding CR insertion histories derived from CR insertion statistics of the plant of interest that the reactivity effect due to some CR usage is typically insignificant and covered by far by assuming for instance a soluble boron concentration somewhat higher than the cycle-averaged boron concentration. Since the determination of the axial end effect has to be performed in Germany on the basis of a sufficient number of plant-specific axial burnup profiles, any distortion of a burnup profile due to CR insertion is covered by deriving an average-burnup-dependent bounding axial burnup profile from all the EOC burnup profiles available from the plant of interest.

- Control blade insertion (BWR):

  Siemens Power Generation Group (KWU) demonstrated within the framework of a reracking project for the storage pool of the Spanish BWR plant Santa Maria de Garoña in 1996 that full insertion of control blades does not lead to a change in the bounding reactivity level at the maximum reactivity point of the BWR fuel. However, the burnup value where the maximum reactivity point is situated was slightly changed, so that the curve showing keff as a function of the burnup was slightly changed. This change however is probably mainly due to the fact that it were taken into account that insertion of control blades results in a reduction of power and hence a decrease in the void fraction, which counteracts spectral hardening.
  It should be noted that the insertion depth of the control blades has a significant effect on the axial power shape and hence on the resulting axial burnup profile.

- MOX environment effects:

  Participants agreed on the need to account for MOX environment effects in UOX assembly depletion calculation in PWRs.
  French representatives told that in France for 900MWe reactors recycling the plutonium from La Hague reprocessing plant conservative depletion calculation of UOX assemblies is performed with MOX completely surrounding an UOX assembly.
  A bounding, but more realistic approach is usually taken in Germany, since in reality an UOX assembly is not completely encircled by MOX assemblies.

4.2.4.2. Isotopic biases and correction factors

Isotopic correction factors are derived from comparisons of Calculated (C) (predicted) isotopic masses/concentrations to Experimental (E) isotopic masses/concentrations measured in Post-Irradiation Experiments (PIE). It was agreed that isotopic correction factors, given for instance in form of C/E values for each BUC nuclide, make up the basis for estimating the isotopic bias of the neutron multiplication factor of a spent fuel configuration of interest.

US representatives however emphasized the lack of *unrestricted* radiochemical assays that jeopardizes the definition of reliable isotopic correction factors on BUC nuclide concentrations.

With respect to the application of isotopic correction factors French representatives proposed to apply these factors as fixed "penalty factors" such that a conservative estimate of the spent fuel's reactivity is a priori guaranteed.

The standpoint of German representatives was as follows: First of all, C/E values are not fixed numbers but statistics (random numbers) since experimental results are statistics. Second, the C/E values of groups of the BUC isotopes are correlated, first due to the production processes of these isotopes during irradiation and thereafter and second due to the chemical separation processes required and measurement methods applied. Therefore, a correct statistical analysis of the C/E values is required in order to be able to derive the isotopic bias *with sufficient confidence*. In addition it has to be considered that significant systematic deviations in experimental results have been observed (this was for instance the case in the ARIANE experiments, where experimental results delivered from different laboratories for one and the same sample differed by a factor of 2). So, it is necessary to try to asses the quality of the experimental data by checking the consistency of the data with respect to the physics of production, depletion and decay of the BUC isotopes. This may be very difficult in many cases or even impossible, so that the only solution is to analyze for each BUC isotope as many C/E values as available. With respect to this requirement there is still a need for more experimental results for some of the BUC isotopes.

4.2.4.3. Bias in prediction of BUC nuclide reactivity worth and correction factors

In France, the reactivity worth of each BUC fission product was measured, and the current OSMOSE oscillation experiment in MINERVE will supply the reactivity worth of each actinide. For each BUC nuclide a conservative penalty factor is or will be derived from the measurement results. This factor is combined with the penalty factor linked to fuel inventory bias. This conservative method is only used in France at present, even though reactivity worth correction factors (named as "confidence factors") are listed in Appendix D of the June 2005 draft version of ANSI 8.27.

4.2.4.4. Sensitivity-based criticality validation techniques, representativeness of experiments, and a posteriori uncertainty

With the SCALE-5 package the Sensitivity/Uncertainty (S/U) module TSUNAMI became available. This module computes sensitivity coefficients Sk = δkeff/δα of the calculated keff to parameters α (e.g. cross-sections) within an energy structure. Sk is an M × N matrix where M is the number of systems being considered and N is the number of nuclear data parameters being involved. (Typically, N is given by the number of nuclide-specific reaction channels times the number of energy groups used.). Due to the linearity of the changes δ_{keff} to the perturbations δα the covariance matrix (also named as "uncertainty matrix") of the changes δ_{keff} is given by $C_{kk} = S_k C_{\alpha\alpha} S_k^T$. The covariance matrix $C_{\alpha\alpha}$ (which obviously is an N × N matrix) contains all the information about the uncertainties in the α parameters. The resulting C_{kk} matrix is an M × M matrix. Its diagonal elements give the variances σ_i^2 for each of the systems considered, and its off-diagonal elements represent the covariance cov_{ij} between the systems. $cov_{ij} \neq 0$ means that the systems i and j are correlated. The degree of correlation is expressed by the *correlation coefficient*

$$c_k(i,j) = cov_{ij} \Big/ \sqrt{\sigma_i^2 \cdot \sigma_j^2} \; .$$

So, if one of the systems represents an application case and all the other (M – 1) systems are experiments, the correlation coefficients c_k express the representativeness of each of the experiments with respect to the application case.

At the present the ORNL team recommends to use the criterion $c_k > 0.8$ for accepting an experiment as representative of an application of interest. No theory-based rationale has been found for this criterion up to the present. However, as in mathematical statistics, a correlation coefficient of 0.8 represents a significant degree of correlation and hence a significant degree of similarity of two systems.

TSUNAMI has already been used extensively in USA (see D.E. Mueller et al, this meeting, paper entitled "Application of Sensitivity/Uncertainty Methods to Burnup Credit Validation"; in addition the ORNL team reported that c_k coefficients have also been used to assess the representativeness of Pu experiments for MOX powder validation). TSUNAMI is also used in Germany and will be probably used in other Central European countries.

S/U tools also exist in Russia but they are not used for BUC studies. In the UK, the code MONK also has some S/U capabilities.

In the French CRISTAL package, adjoint flux and sensitivity profiles can be calculated. A "Characterization System" allows the location of the calculated application amongst the available experimental benchmarks (through neutron balance comparison). A new approach based on representativeness factors will soon enable the automated calculation of the a priori k_{eff} uncertainty (due to nuclear data covariances) and the a posteriori uncertainty associated with the corrected k_{eff} value accounting for experimental information; this method has already been implemented for MOX powders and MOX fuel pin storage at MELOX plant.

As noted by participants, the utilization of MCNP for sensitivity calculations is not obvious.

Finally, the working group emphasized the need for multigroup covariances of (n,γ) cross section of BUC nuclides, as well as covariances for fission and multiplicity of fissile isotopes. It was therefore asked to the Working Party of Nuclear Criticality Safety (WPNCS) of the OECD/NEA to establish in the JEF 15-macrogroup structure the standard deviations/correlations for these BUC nuclide nuclear data (associated with the best evaluations involved in the recent international files).

4.2.4.5. Axial burnup databases and generation of bounding profiles

As a preamble, the issue of axial burnup profiles has to integrate the uncertainties within which a burnup is known. In general, it seems that the FA average burnup is conservatively known at ± 5 %, with ± 2 % around the mid-plan and ± 5 % around the axial ends. The burnup data are obtained either through in core measurements or using spectrometric methods after unloading.

US representatives stated that the available Yankee Atomic Corporation (YAC) axial burnup database, analyzed by Paris and Chen, has not been enlarged. Concerning bounding profiles, ORNL has revised the analysis from YAC database. New bounding profiles more realistic and meaningful are proposed (only 3 profiles are defined to span the whole burnup range).

The French database actually is a burnup profile information base. Different from US or German databases which are usually derived with the aid of core calculations from in core axial flux measurements, the French database consists of axial burnup profiles which were

measured by means of gamma-spectroscopy on unloaded PWR assemblies at La Hague facility. The database contains for instance 600 profiles corresponding to PWR assemblies from 1300MWe reactors, mainly in the 30-40 MWd/t average burnup range and often located under CR clusters. Evaluation of the database has shown that the measured axial profiles are very similar irrespective of their burnup and have no asymmetry.

EDF and the French BUC Group should derive bounding profiles from the La Hague burnup database before end of 2005. The French Safety Institute IRSN proposes the generation of more conservative profiles, which does not preserve the assembly average burnup: for each axial location the bounding burnup corresponds to the minimum measured value.[3)]

In Germany, the axial end effect has to be determined from a sufficient number of *plant-specific* axial burnup profiles. Up to the present more than 20000 EOC profiles from eight different plants were analyzed; and bounding profiles were derived, for each plant separately. The analyzed profiles are very similar irrespective of their average burnup, but they show some plant-specific components; and, in contrast to the French profiles, quite a lot of them have some asymmetry which decreases with increasing average burnup. The resulting plant-specific bounding profiles are therefore given as continuous functions of the average burnup.

The calculation procedure usually used in Germany for deriving axial burnup profiles from in-core flux measurements has been validated several times against samples of profiles measured by means of gamma spectroscopy. These samples include profiles from fuel assemblies exposed to partial CR insertion as well as profiles from fuel designs with integral burnable absorbers (Gd in particular).

The spent fuel rods provided by the German NPP Neckarwestheim 2 for one of the REBUS experiments were chosen since the calculated axial burnup profile of the fuel assembly from which the rods were taken showed no asymmetry. This was confirmed by rod gamma scans performed by SCK•CEN: Calculated and measured profiles were in excellent agreement.

Finally, German participants mentioned that about 2000 axial profiles from the NPPs Neckarwestheim 1 and 2 are available at OECD NEA. In addition to these profiles 850 profiles were provided by NPP Neckarwestheim 2 for the Phase II-C Burnup Credit Benchmark conducted by the Expert Group on Burnup Credit Criticality Safety under auspices of the OECD NEA.

In other Central European countries axial burnup profiles are derived from calculations. Studies performed in Hungary and Czech Republic showed very small positive or even negative end effects for VVER fuel.

Finally, the working group recommended the use of bounding burnup profiles in criticality-safety calculations, if at all possible.

[3)] Comment (J.C. Neuber): The end effect, usually first negative for low average burnup values, increases - after having reached its minimum at a certain average burnup - with increasing average burnup. So, if the assembly average burnup is not preserved and no parameter is introduced to relate the "conservative profile" to the average burnup values of the profiles from which the "conservative profile" is derived how it is possible to demonstrate that the "conservative profile" is really bounding for any given average burnup in the range of positive end effects? (A bounding non-uniform burnup profile is always related to positive end effects; for negative end effects the uniform distribution of the average burnup is the bounding profile.)

4.2.5. Conclusions and recommendations

The main conclusions of the group discussion are as follows:

- As a result of the increased validation requirement for BUC, there are significant advantages in the use of modern codes and data packages which have been verified and validated through coordinated programmes.
- The OECD database SFCOMPO for radiochemical assays on SNF and the ICSBEP handbook for critical experiments, are very useful.
- However, participants noticed that SFCOMPO is only partially well suited for BUC validation at present. PIE results for VVER fuel designs as well as modern western PWR and BWR designs should be introduced with a more detailed description of irradiation histories.
- The advantage of well defined benchmarks is noted. In particular, experiments should be amenable to calculation without significant modeling approximations or assumptions and should include a thorough assessment of experimental uncertainty. The benchmark evaluation process used in the production of data for the ICSBEP handbook is felt to be an example of good practice in this area.
- The currently available PIE VVER database (8 samples) is not sufficient. The achievement of a complementary programme is recommended.
- Thanks to BUC experimental programmes (PIE and BUC nuclide worth measurements), nuclear data evaluations were improved in the international nuclear data files. In the new JEFF3.1 European file, for instance, the U235, U238, Pu241 and Am241 evaluations were improved; the capture cross sections of the FP BUC isotopes were reevaluated, and reliable Sm149 - Nd143- Cs133 – Rh103 – Eu155 (n,γ) cross sections are available. Furthermore some FP yields from U235 and Pu239 were modified. In comparison to JEF2.2 these improvements will result in less isotopic bias in BUC calculations.
- The working group agreed on the necessity of a bounding approach in the fuel depletion calculations. Which approach is in fact the bounding one this depends on the specific fuel design, the specific reactor operation conditions, the specific fuel management system of interest and the requirements one makes on this system.
- The working group agreed on using isotopic correction factors as a basis for estimating the isotopic bias of the neutron multiplication factor of a spent fuel management system of interest. It is desirable to take account of the variances and covariances of the isotopic correction factors. Measurements of isotopic masses/concentrations should be performed in such a way that systematic errors in the applied measurement methods can be revealed.
- The working group concurs that reactivity worth measurements of individual BUC nuclides provide essential information for criticality calculation validation. Integral burnup reactivity worth measurements, such as the LWR spent fuel worth measurements performed in the reactors VENUS and MINERVE, are essential to demonstrate the applicability of depletion calculation and criticality calculation codes. Analysis of “Commercial Reactor Criticals” can provide an integral check of isotopic compositions, cross sections and fuel lattice geometry.
- Sensitivity/Uncertainty (S/U) technique was judged as a powerful tool for validation issues. Participants suggested that the efficient utilization of S/U tools, such as TSUNAMI in the SCALE package, should be established and recommended to users.
- S/U studies require the knowledge of covariance matrices. Thus, the working group proposed that the OECD-WPNCS Criticality Working Party addresses this issue and

define the standard deviations & correlations in the JEF 15-group structure for every BUC nuclide.

- Axial burnup databases are useful information to define bounding profiles. Different methods of generating bounding profiles were developed in different countries. Since axial burnup profiles are fuel-design- and operational-dependent and since the method of generating bounding profiles can be impacted by the requirements the spent fuel management system of interest has to meet, it is not to be expected at present that consensus on a standardized method of generating bounding profiles will be achieved. The working group however proposed that this topic should be further pursued, particularly in international expert groups such as the OECD-WPNCS Expert Group on Burnup Credit Criticality Safety Analysis.
- BUC is a still developing methodology. Therefore, access to international activities on nuclear data, methods development and code benchmarking as well as participation at the international technical meetings organized by IAEA since 1997 play an important role in developing and maintaining technical capability as well as establishing good practice in BUC. IAEA is therefore asked by the working group to continue its activities in BUC.

4.3. Procedural Compliance with Safety Criteria

Leaders: **J. Gulliford** (United Kingdom)
J.C. Wagner (United States of America)

Members: **G. Caplin** (France)
A. Chesterman (United Kingdom)
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A. Marc (France)
L. Milet (France)
D. Simister (United Kingdom)
P. Wilson (United Kingdom)
J.C. Neuber[*] (Germany)

4.3.1. Introduction

4.3.1.1. General

This working group was assigned the task of reviewing the status of methods used to demonstrate compliance with safety criteria among nations currently applying BUC. In particular the discussions of this workshop focused on methods of verifying assembly burnup and reviewed examples of how this is put into practice as part of spent fuel operations in France, Germany, the United Kingdom and in the USA. These examples were used to highlight both similarities and differences in practice. Where differences were found the group sought to identify the causes both in terms of variations in operational design/requirements and in terms of any differences in underlying safety philosophy. In general it was found that the safety philosophy and associated methods are very similar and differences in outcome with respect to compliance issues arise mainly from differences in the operational environments. Examples of operational practice used to inform the review included:

[*] only temporarily present

- Fuel Dissolution (Cap de La Hague, France and Thorp, UK)
- Fuel Transport (France)
- Reactor Pond Storage (Germany & USA)

A review was also made of measurement methods currently employed in the verification of burnup and potential development of the technology was also discussed.
The main observations of the group discussions are summarized and recommendations arising from the discussions are recorded. The group recommended that the IAEA Standard on transport should be reviewed with respect to the current "requirement" for a measurement of burnup.

It should be noted that this report aims to reflect the views of the individuals involved in the workshop and is not necessarily a statement of any corporate or regulatory position on BUC.

4.3.1.2. Additional material used to help inform the workshop discussions

The following material was also reviewed as part of workshop discussions:

- Report from the Working Group (Workshop) on Safety Assessment and Implementation at Madrid 2002 IAEA TCM [1]
- Extract from ANS Standard on 8.17 – paragraph 4.10 [2]
- Extract from ANS Standard on Burnup Credit - paragraph 8 [3]
- Extract from IAEA Standard on transport - paragraph 674 (need for burnup measurement) [4]

The following observations were drawn:

- There is significant variation between standards with respect to whether a measurement of burnup is a firm requirement or not.4)
- The Madrid workshop tended to concentrate on general principles of implementation, so it would be appropriate for this workshop to focus on practical examples and highlight relevant operational experience.

4.3.1.3. Outline of working group (Workshop) agenda and report

Following introduction of the participants and preliminary discussions on what topics were of particular interest the following outline was identified to serve both as a rough agenda and as a suitable structure for this report:

- Differences from Fresh Fuel Assumption
- Examples of Current Practices and Regulatory Requirements
- Types of Safety Measure Applied & Measurement Methods
- Summary of Recommendations and Observations

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4) During presentation of the workshop discussions it was noted that while the IAEA Standard requires that 'a measurement shall be performed', (paragragh 674 of [4]), the advisory material explains that the exact nature of the measurement is very flexible and might be taken to mean some visual method to confirm the identity of the assembly, for example.

4.3.2. Methodology differences from fresh fuel assumption

The group concluded that there is no fundamental difference in safety philosophy between criticality assessments using fresh fuel assumptions and assessments applying BUC. In both cases the general approach is to identify a safe envelope against appropriate safety criteria (which incorporate suitable margins of safety) and show that all normal conditions fall within that envelope. For those accident conditions where a chain of events (a Fault Sequence) might credibly lead to a criticality, safety measures are identified to provide protection such that the abnormal condition is recognized and appropriate action taken before the safe envelope is breached. The adequacy of the safety measures is usually judged against principles of "defense-in-depth" and sometimes against risk criteria. A key part of implementation of the safety case is that checks are made to demonstrate that the operation is being carried out in compliance with operational limits associated with the safe envelope and that any safety measures provide the level (reliability & sensitivity) of protection foreseen in the assessment.

This approach appears to be commonly applied around the world and in the same way to "fresh fuel" assessments or assessments applying BUC. In discussing examples of how BUC is implemented the group concluded that the main difference between traditional fresh fuel and BUC-based assessment is essentially a question of degree of complexity. This arises both in the methodology (e.g. identification of bounding irradiation parameters, effects of spatial variation, selection of BUC nuclides, validation requirement) and in implementation (e.g. compliance with both Initial Enrichment (IE) and Burnup (BU), interpretation/calibration of measurements, allowance for uncertainty in BU).

4.3.3. Examples of current practice and regulatory requirements

4.3.3.1. Summary of current practices

The following examples of current BUC practice were presented and discussed by the group:

- Dissolution in head-end of Thorp reprocessing plant (no BUC is taken in the reception pond)
- Transport to/Reception of Spent LWR fuel at Cap de La Hague and dissolution in head-end of reprocessing plant
- Reactor pond storage in at PWR & BWR NPP in the USA
- Reactor pond storage in Germany

In addition some details of historic BUC application to transport assessment in Germany and planned application of BUC to dry transport in the USA was presented and discussed. A summary of each example is given below.

Thorp

Fuel is shipped to Thorp under criticality assessments based on the fresh fuel assumption. BUC is applied to the head-end plant, particularly in the Dissolvers where large batches of fuel comprising several fuel assemblies are sheared into small lengths and dropped into hot nitric acid doped with gadolinium. The original assessment was based on fresh fuel assumptions which resulted in a requirement for significant concentrations of Gd to comply with criticality safety criteria. To reduce the Gd loading (and hence reduce waste volumes) Actinide-only BUC has been applied which reduces the Gd requirement by nearly a factor of 2.

Safety criteria are based on a margin of 5% in k_{eff} for normal conditions, with additional allowance for code bias and uncertainty (including uncertainty associated with BUC). For some low probability accident conditions the fixed margin is reduced to 2% in k_{eff}. Fuel packing fractions in the dissolver are optimized in calculations made to derive the loading curve, which represents an additional margin of about 14% in k_{eff} relative to typical packing fractions. Other conservatisms (in the depletion calculations) include:

- high soluble boron
- high fuel/moderator temperature
- low cooling time

Some analysis of the effect of axial burnup profile was made, but due to the fact that multiple assemblies are sheared and dropped into each batch the sensitivity to this parameter is low.

Defense-in-depth assessment of safety measures is made against double contingency and Design Basis Accident Analysis (DBAA) criteria.

Compliance with the safe loading curve is made through a combination of checks against supplier's data and through measurements made on each assembly by the Thorp Fuel Pond Feed Monitors (FPFM). The measurement is based on γ spectroscopy and neutron counting which provide information on cooling time, burnup (BU), initial enrichment (IE) and residual enrichment (RE). A go/no-go trip is set against the measured RE value which prevents any assembly above the limiting RE being fed forward to the shearing machine. The assessment demonstrates that the RE at the zero burnup end of the loading curve is bounding (i.e. a minimum) for the rest of the curve, so this value, with allowance for measurement/calibration uncertainties is used as a test of compliance for all assemblies. Calibration is made through measurements on selected fuel assemblies prior to each campaign. The measurements are calibrated against suppliers' data. Checks are made during and after each campaign to confirm calibration constants and uncertainties. The measured RE is based on neutron counts taken at a single axial location near the centre of the element.

Additional operating requirements identified during regulatory review of the BUC criticality assessment include:

- Pre and post campaign reviews of FPFM results to confirm overall performance
- Additional margin in keff to allow for "prudence and caution"

The UK regulator (NII - Nuclear Installation Inspectorate) has also taken the position that regulatory review in support of a movement away from a fresh fuel based assessment to a safety case based on BUC would not be given high priority unless it were clear that some safety (as opposed to purely economic) benefit would result. In the case of Thorp this is provided by the reduction in waste volumes associated with reduction in Gd concentrations.

Cap de La Hague

In the La Hague reprocessing plant, BUC is applied to the spent PWR fuel storage in ponds and to the PWR dissolution (thus especially in the rotary dissolvers where sheared rods are dropped into one of the dissolver's bucket soaking in hot nitric acid –non-poisoned). For these two applications, (where the required level of credited burnup is different), the BUC methodology approach is also noticeably different.

Fuel is shipped to La Hague under criticality assessments based on BUC. The principles of the safety assessment for shipping fuels are very similar to those applied for the fuel storage in ponds (discussed hereafter).

- Main principles shared by the different applications of BUC
- The Defense-in-depth assessment of safety measures is made against the Criticality Fundamental Safety Rule (Règle Fondamentale de Sûreté – RFS): a single event must not lead to a criticality accident and if a criticality can occur from two events they must be proved to be strictly independent, unlikely and each event must be reliably detected within an acceptable delay with regards to the corrective operation.
- The safety criteria are based on a margin of 5% in keff for normal conditions and some accident conditions with additional allowance for code uncertainty. For some unlikely accident conditions, the fixed margin is reduced to 3% or 2% in keff. Uncertainties associated with BUC are taken into account especially by additional conservatisms in the depletion calculations.
- Distinctive features of BUC for PWR fuel storage in pondsThe original assessment was based on fresh fuel assumptions which were enough to comply with criticality safety criteria for low enriched fuels. At present, for higher enrichment (depending on the fuel assembly design), the assessment is based on Actinide-Only BUC. And then, two cases appear:
    - If the burnup requirement is lower than or equal to 3.2 GWd/t, a simple gross γ measurement (or a validated by regulators equivalent method to confirm the irradiation) is sufficient.
    - If the burnup requirement is higher than 3.2 GWd/t, a validated (by regulators) burnup measurement must be performed. This measurement must prove that the irradiation in the 50-least-irradiated-centimetres (axially) of the Fuel Assembly is higher than the burnup required.

  These burnup verifications are performed before loading the fuels in their transport cask and are completed by the supplier. The type of measurement, made in supplier's plant, depends on the supplier (French, German, Swiss, etc.) but must be approved by French regulators.

  The safety assessment for shipping fuels is based on the same principles.

  Some studies are in progress in order to use Actinides plus Fission Products BUC and to take into account the axial profile for transport and for storage in ponds. These studies are mainly based on the work of the French Working Group on BUC and should be submitted to regulators shortly. This approach will allow extension of the enrichment field in which a quantitative burnup measurement is not necessary for safety reasons (gross γ measurement only).

- Distinctive features of BUC for dissolution in rotary dissolvers
  The safety assessment is based on Actinide-only BUC which leads to safe loading curves: maximal permissible mass per bucket against burnup (one safe curve per initial enrichment). Each of these curves presents a burnup limit over which the criticality safety is ensured by the bucket geometry (without restrictions on loaded mass). A PWR fuel assembly is typically loaded into 3 or 4 buckets (about 110 liters volume each).

The compliance with the safe loading curves and the determination of the number of buckets needed to load the fuel assembly is made through a combination of checks against supplier's data and through measurements made on each assembly. The measurements must provide information on the initial enrichment, the average burnup and the axial profile. These calibrated and validated measurements are implemented between the storage ponds and the dissolver's workshop, and consist in a double gamma scanning (on two opposite faces of the fuel assembly) and passive neutron measurements (on the two other opposite faces). These axial scannings are interpreted by an on-line evaluation code. A go/no-go trip is set against the comparison between the average burnup measured and the supplier's data.

Spent fuel pond storage in the United States of America

Storage of spent fuel in underwater racks at reactors has been standard practice in the United States since the start of the nuclear industry. Spent fuel ponds at reactors are licensed under the regulations governing reactors and represent a controlled facility operated in conjunction with the reactor operation. In contrast, transportation casks are licensed under a different set of regulations (packaging and transportation) because they may be used in any facility and transported over public roads, where the environment is more unpredictable and the controls less reliable (as compared to spent fuel ponds). Hence, for these two applications, the BUC methodology approaches are notably different.

The main principals shared by the two applications of BUC are consistent with those used throughout the world. The safety criteria are based on a margin of 5% in k_{eff} for normal conditions, with additional allowance for code bias and uncertainty (including uncertainty associated with BUC), and safety measures to provide defense-in depth are employed.

- Distinctive features of BUC in spent fuel ponds
  In lieu of credit for the soluble boron present in the spent fuel pond water, the US regulatory authority (NRC) has licensed the use of full (actinides and fission products) burnup credit in borated spent fuel ponds at PWR plants. Limited credit for the soluble boron during normal and off-normal conditions is permitted. Hence, the uncredited soluble boron in the pond water provides defense-in-depth. Fuel assembly burnup from plant records (including an adjustment for burnup uncertainty) is compared against a loading curve from the safety analysis for determination of compliance. No measurement of burnup is required. True "burnup credit" for BWR storage pools (where there is no soluble boron present) has not been licensed in the USA; instead, the approach is based on the peak reactivity anticipated for the BWR fuel during the depletion process (reactivity initially increases early in life due to depletion of the gadolinium absorber in the assembly). This approach is sometimes referred to as "gadolinium credit".

- Distinctive features of BUC in transportation casks
  For transportation on public roads, the regulations require consideration of flooding by unborated water in the safety analysis. Allowance for BUC is a relatively recent development, and hence no transportation casks are yet licensed with BUC. The current regulatory guidance permits credit for actinides only and verification of fuel burnup via measurement is required. Flooding of a cask during transportation is considered by many to be highly unlikely, and hence the regulatory requirement for considering the condition is often cited as providing defense-in-depth.

Spent fuel pond storage in Germany

In Germany wet storage facilities for LWR fuel assemblies have to meet the requirements laid down in the German safety codes KTA 3602 and DIN 25471. Transportation and dry storage casks have to meet the requirements laid down in the German safety code DIN 25712. Even though licensed under different regulations, the requirements laid down in these regulations for BUC applications to wet storage pools and transport/storage casks are completely consistent with respect to implementation and validation of the depletion calculations, isotopic selection, implementation and validation of the criticality calculations, evaluation of axial and horizontal burnup profiles, determination of criticality safety acceptance criteria and loading criteria (loading curves), as well as prevention of misloading events.

Nuclides with negative reactivity worth may be used if their contribution to the isotopic bias of the k_{eff} value of the spent fuel pool or spent fuel cask can be validated and if they are non-volatile under the conditions to be considered (normal as well as accidental conditions).

The evaluated maximum neutron multiplication factor k_{eff} shall not be greater than $(1 - \Delta k_S)$ and shall include all calculational and mechanical uncertainties with a 95% probability at a 95% confidence level. For the safety margin Δk_S the value of 0.05 has to be chosen for normal operation conditions, $\Delta k_S = 0.05$. This also goes for abnormal and accidental conditions with the following exceptions:

- Borated wet storage ponds: If no BUC is taken Partial Boron Credit (PBC) may be taken. For the case of a hypothetical boron dilution transient down to 0 ppm in the pool water $\Delta kS = 0.02$ may be used.
- Wet storage ponds: For events with very low frequency of occurrence and very low consequence in case of a hypothetical criticality a lower value than 0.05 may be used for ΔkS provided that the fuel configurations related to these events are modeled as exactly as reasonable possible and that the calculation models used are well validated. In no case a lower value than $\Delta kS = 0.02$ may be used.
- Transport/storage cask: If no BUC is taken a lower value than 0.05 may be used for ΔkS in compliance with regulations endorsed by DIN 25712 (cf. for instance IAEA TS-G-1.1).

Both the standard DIN 25471 and the standard DIN 25712 require that the misloading event has to be excluded as a design basis event by applying the double contingency principle directly to the misloading event: At least two independent, unlikely and concurrent incidents have to happen before a misloading event can occur. This application of the double contingency principle and hence the exclusion of the misloading event as a design basis event from the criticality safety analysis is achieved by applying independent layers of hardware and software measures ensuring the reliability of the reactor record data and the fuel handling procedures applied to the pond and cask loading operations (see J.C. Neuber et al, this meeting, paper entitled “Double Contingency Principle and Prevention of Misloading Events”). Therefore, no burnup measurement is required. However, since stipulated in paragraph 674 of IAEA TS-R-1 a burnup measurement is prescribed by DIN 25712 for cask loading.

- Distinct features of BUC in spent fuel ponds
  The codes KTA 3602 and DIN 25471 allow application of actinide-plus-fission-product BUC to LWR UOX and MOX fuel irrespective of whether the spent fuel pool is normally borated or not.

If the pond is normally borated it is allowed to take account of the presence of the boron in the criticality analysis of abnormal and accidental events in compliance with the double contingency principle. But it is not allowed to apply PBC for the normal operation geometry if BUC is used. So, PBC can only be applied to region I of a storage pond (region I = region designed to accommodate fuel with maximum allowable enrichment at the maximum reactivity point in its lifetime).

- At present actinide-plus-fission-product BUC is applied in the storage ponds of two PWR NPPs, the Convoy Series Plants Neckarwestheim II (GKN II) and Emsland (KKE). A relatively low burnup of 10 MWd/kg U is required for UOX fuel assemblies with initial enrichments between 4.0 wt.-% U-235 and the maximum allowable enrichment of 4.4 wt.-%. This minimum required burnup covers also storage of Enriched Reprocessed Uranium (ERU) UOX fuels which are reactivity equivalent to fresh fuel initial enrichments between 4.0 wt.-% and 4.4 wt.-% U-235 under reactor operation conditions.5) There is no need for higher burnups or BUC for MOX fuel assemblies, since there is no strong demand for increasing the presently available storage capacities because of the following reasons:
- Since the initial U-235 enrichment already used is relatively high6) and the cycle lengths used in Germany are usually not greater than 12 months the number of fuel assemblies discharged at EOCs is relatively small.
- Spent UOX fuel is loaded into casks after 5 years cooling time at the latest; and spent MOX fuel is loaded into casks after 10 years cooling time at the latest.
- MOX fuel which is reactivity equivalent to 4.4 wt.-% U-235 enriched UOX fuel under reactor operation conditions can be stored in region II without using BUC.The integral burnable absorber BUC using actinides plus fission products is applied to wet storage of UOX and MOX fuel assemblies for all BWR plants in such a way that compliance with the criticality safety acceptance criterion is ensured for the maximum reactivity points of the fuel types. This type of BUC is also used for some fuel types in PWR plants.
- Distinct features of BUC in transportation/storage casks

  The code DIN 25712 allows application of actinide-plus-fission-product BUC to LWR UOX and MOX fuel. Flooding of the transport casks as well as of the storage casks has to be considered. In contrast to wet storage, a burnup verification measurement for fuel to be loaded in a cask is stipulated. Optimized loading schemes with storage-position-specific minimum required burnups are allowed (see J.C. Neuber, this meeting, paper entitled “The German Burnup Credit Regulatory Standards”).

  At present a limited actinide-only BUC is used. But a new generation of casks under development will apply higher margins of BUC in compliance with the safety code DIN 25712.

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5) Reactivity equivalence of two different fuel types under reactor operation conditions means that the reactivity integral $\int k_{inf}(B)\, dB$ is virtually conserved for the range $B \in [0, 2 \cdot B_e]$ where B_e denotes the target burnup of the reference fuel type.

6) An isolated fully assembled Convoy Series fuel assembly with 4.4 wt.-% U-235 fresh fuel initial enrichment has a k_{eff} value of 0.95 when completely immersed in pure water

4.3.3.2. Discussion: Differences/Similarities

A number of differences between the examples of current practices are observed. Some of the differences include:

- Measurement practices (and hence compliance procedures) vary significantly depending on how much BU is needed to satisfy the safety criteria (notably at Cap de La Hague, for example).
- Different facilities and operations show varying levels of sensitivity to spent fuel characteristics — e.g. low sensitivity to axial profile in the Thorp dissolver (where multiple assemblies are processed in a single batch) compared to higher sensitivity in the COGEMA dissolver system where individual sections of the assembly are passed through the dissolution stage.
- Different facilities and operations show varying levels of tolerance to a misloading error (i.e. fuel which IE is too high or which BU is too low relative to the loading curve is anyway loaded in the facility). Examples of high tolerance include operations in some spent PWR fuel ponds where the presence of soluble boron provides a significant additional margin and dry storage/transport where a misloading would need to occur simultaneously with a flooding event.7)

On the other hand we see many important similarities in the approach taken:

- Very similar keff criteria, including relaxation on margin for low probability accident conditions
- Similar defense-in-depth criteria (double contingency, DBAA)
- Same hierarchy in preferred type of safety measure (Passive Feature-Active Engineered-Operator)
- All BUC practitioners in the group noted the importance of quantifying the sensitivity to controlled parameters, particularly as the limit of the safe envelope is approached.

In discussing the examples it became apparent that consideration of compliance issues may fundamentally affect the selection of which parameters are used to define the safe envelope. For example in Thorp significant "credit" might be available by consideration of "real" packing fraction but this would be a difficult parameter to demonstrate procedural compliance (due to local variations).

In summary, although the approach to BUC criticality assessment is very similar, the outcomes with respect to compliance procedures, particularly with regard to requirements for BU verification measurements, vary considerably. In general the level of reliance placed on BU verification depends on:

- the amount of burnup being credited
- the level of confidence in other sources of information (e.g. reactor records)

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7) Comment (J.C. Neuber): It is important to pay attention to the fact that the word "*simultaneously*" is used. This shows that it is presupposed that the *double contingency principle* can be applied, so that the misloading error is regarded as one unlikely incident and a second *concurrent* (independent and unlikely) incident as for instance flooding need not be assumed. However, as experience shows (e.g. , the Dampierre misloading error was discussed by the working group), when a misloading error really happens then there is a non-negligible probability that this error remains undetected. Then the double contingency principle cannot be effective when a non-concurrent incident (e.g. flooding) takes place at any time later on.

- the presence of other contingencies (e.g. loss of control over boron concentration in a storage pond, flooding in a dry transport cask)
- the presence of other margins of safety not explicitly credited in the assessment (e.g. packing fraction in dissolvers, soluble boron in spent fuel ponds).

4.3.4. Summary of measurement/verification techniques

There is a well established range of spent fuel measurement techniques available to support BUC verification. A useful set of these, along with application examples is presented in a burnup credit training course that was run in Beijing [5].

The principal types of these non-destructive measurement techniques may be summarized as:

- Gamma ray counting
  - Gross gamma ray counting:

    This technique may be used to provide a quick verification that an assembly has been irradiated. With information on cooling time a crude measure of irradiation can be achieved.

  - Gamma spectrometry:

    This powerful technique measures the intensity of emissions from individual fission and activation products and may be used to measure burnup, cooling time and, to a limited extent, enrichment under certain conditions. The technique may be applied with high, medium and low resolution detectors and their associated pulse processing electronic systems. The choice of detector type will depend on a number of parameters linked to the desired measurement performance, i.e. accuracy and precision, as well as local dose rate conditions and budgetary constraints.

    Dependent on the cooling time the decay or the activity ratio of the following isotopes can be used for determining the burnup of spent fuel: Cs-134, Cs-137, Eu-154, and Cs-134/Cs-137, Eu-154/Cs-137, (Ru-106 · Cs-137)/(Cs-134)2 respectively, cf. References [5] and [6].

    Since gamma rays can be collimated, gamma ray counting is also used for determination of axial burnup profiles [6].

    Measurement of isotopic ratios by means of high resolution gamma spectrometry does not require knowledge of the detector yield, but the measurement result significantly depends on the irradiation history of the fuel assembly of interest.

    In addition, due to shielding of gamma rays in the fuel pins only a peripheral assay of the fuel assembly can be achieved. This is different from neutron counting. Accordingly, placing two neutron detectors (e.g. fission chambers) at opposite lateral faces of the fuel assembly and averaging the results from the two detectors leads to a very low sensitivity to horizontal burnup gradients in the fuel assembly. There are in fact detector designs with at least two measurement arms embracing the fuel assembly; each arm is equipped with one neutron detector at least (see below).

- Neutron measurements
  - Passive neutron counting:

    This is a very sensitive technique that measures spontaneous fission neutrons from higher actinides, mainly Cm-244. The technique has a strong count rate dependency on burnup, cooling time and initial enrichment so that, if the enrichment and cooling time are known and there is good geometrical control during measurement, an accurate measure of burnup can be made.

- Active neutron counting:

  This technique measures the enhanced neutron flux from induced fission events that occur in the residual fissile material. The induced fissions are induced or "activated" by an external source, normally Cf-252, though a development of the technique has been explored that uses the assembly's inherent passive neutron emission to act as the activation or interrogating source [7]. This development is called "self-interrogation".

  Usually an axial section of the fuel zone of several centimeters in length contributes to the signals recorded in a neutron detector. Consequently, neutron signals are usually used to evaluate the average burnup, whereas burnup profiles are measured with collimated gamma rays.

Since the first derivate of the correlation between count rate and burnup is smaller for the gamma emitters than for neutron emission, application of passive neutron counting results in a more accurate estimate of the burnup than use of gamma measurement methods. For example, the following accuracies have been reported:

- For the BNFL Spent Fuel Monitor, which takes measurements on Cs-137 (i.e. on the 662 keV gamma ray following the decay of Cs-137) and the activity ratio Cs-134/Cs-137, accuracies of 10% at 15 MWd/kg and 5% at 33 MWd/kg [5]
- For FORK and FORK+ detectors, having two measurement arms each quipped with on thermal neutron detector, one epithermal detector, and one gross gamma detector, accuracies of about 2% [5] (the FORK+ design is equipped with one detector more, a CZT – Cadmium – Zinc – Telluride gamma detector, usually adjusted to the 662 keV ray from the Cs-137 decay)
- For the PYTHON device applied to PWR (using passive neutron and collimated gross gamma counting) 2% in prototype testing and up to 5% under operation conditions [6]
- For the NAJA device (developed to segregate UOX fuel from MOX fuel and to estimate initial enrichment, burnup and reactivity, and using passive and active neutron measurements) up to 2.5% for average burnup, 0.5% for initial U-235 enrichment for fresh UOX fuel between 3 and 4 wt.-% initial enrichment, and about 0.3% in keff [8].

As regards the "assayability" on U-235, U-238, fissile Pu isotopes and total Pu content accuracies between 1% and 5% are listed in Ref. [9] for applications of active and passive neutron measurements to fuel rods (including neutron coincidence collar and rod scanning techniques).

The use of active neutron counting, using a source or self interrogation, has had only limited development as a procedure for the measurement of k_{eff}. Depending on the industry's need development in this area may be useful.

Developments in small solid-state gamma spectrometry detectors, e.g. CZT and combined gamma and neutron counter, e.g. silicon carbide, could potentially find applications in "in built" cask monitoring devices.

Other measurement data also available from reactor monitoring systems include:

- In-core detectors
- Ex-core detectors (limited use, particularly for assemblies away from edge of core)
- Core following data.

One important feature of the techniques mentioned above is that they all involve some level of interpretation or calibration. For example, due to the multiplication effect in a fuel assembly the measured count rate in a passive neutron measurement is not only dependent on the neutron emission and the detector yield but is also impacted by the neutron multiplication factor k_{eff} of the fuel assembly. Since k_{eff} depends on the initial enrichment, the burnup, the irradiation history, and the measurement environment (measurement in pure or borated water, air, etc.) determination of the fuel assembly's burnup from the measured count rate requires to make use of the information about the initial enrichment and the irradiation history from the reactor records to obtain a priori known relations between k_{eff} and burnup or count rate and burnup at given initial enrichment, irradiation history, and given measurement environment. Such relations can be for instance obtained and used as follows:

- Use of a measured calibration curve: For a set of fuel assemblies with very similar irradiation histories, known initial enrichments and known burnups the count rates are measured under the given measurement environment conditions. From the results calibration curves are derived for the correlation between count rate and burnup at given initial enrichment and irradiation history. The irradiation histories of the fuel assemblies for which the burnup has to be determined with the aid of the calibration curves have to be very similar to the irradiation histories on which the calibration curves are based.
- Use of a calculated correlation curve: By means of a validated online depletion calculation code the isotopic content is calculated as a function of burnup for the initial enrichment and the irradiation history of the fuel assembly to be measured. Then, considering spontaneous fission and (alpha, neutron) reactions the online code calculates the correlation between burnup and actual neutron emission (from Cm-244 for instance) for the given initial enrichment and irradiation history. In addition, by means of a validated reactivity calculation code the neutron multiplication factor keff of the fuel assembly under examination has to be calculated as a function if the burnup (B) for the given initial enrichment (e), irradiation history (IH) and measurement environment conditions (MEC), keff = keff (B|e,IH,MEC). Using this function and the correlation between burnup and actual neutron emission the measured count rate of the fuel assembly can be evaluated. This is usually done in an iterative way: Using the measured count rate and a first guess of keff a first estimate of the burnup is obtained. With this first estimate a new estimate of keff is obtained from keff = keff (B|e,IH,MEC). With this new estimate and the measured count rate a further estimate for burnup is obtained. The procedure is continued till convergence in keff is achieved.

In any case it is required to use information from the reactor records for the evaluation of the measurement results. (The more complex the applied measurement technique is the smaller is usually the amount of information needed from the reactor records.) The evaluation of the measurement results cannot be performed independently from reactor record data, but the measurement results are of course completely independent from the reactor record data. So therefore, the evaluation of the measurements results is an *independent consistency check of reactor record data*. Sensitivities of measurement techniques and evaluation procedures to variations in reactor record data are discussed in detail in Ref. [10].

4.3.5. Observations and recommendations

The group made the following observations:

- There appears to be significant variation between standards on the requirement for a measurement to verify compliance with BUC criteria.
- There has been considerable further accumulation of practical experience in operating facilities where BUC is applied since the Madrid TCM.
- Although the approach to BUC criticality assessment is very similar, the outcomes with respect to compliance procedures, particularly with regard to requirements for burnup verification measurements, vary considerably.
- There is significant variation between standards with respect to whether a measurement of burnup is a firm requirement or not.
- Measurement techniques are well established and sufficient accuracy in average assembly burnup is readily achievable.
- Further development of measurement techniques is expected but it is anticipated that there will always be some component of operator action (e.g. interpretation/calibration) in this type of safety measure.

Based on the discussions reported above, the group recommends that the paragraph 674 of the IAEA Standard on Transport [4] should be reviewed. The review should take account of the significant variation seen in level of reliance placed on this type of measurement and on the extensive experience now accumulated in safe implementation of BUC based criticality safety cases.

4.3.6. References

[1] IAEA TCM on Requirements, Practices, and Developments in BUC Applications, Madrid April 2003, IAEA-TECDOC-1378.

[2] Criticality Safety Criteria for Handling, Storage and Transportation of LWR Fuel Outside Reactors, ANS 8.17.

[3] Criteria For Taking Credit For Irradiation Of LWR Fuel in Criticality Safety Analysis For Applications Outside of Reactor Cores. ANS 8.27.

[4] Regulations for the Safe Transport of Radioactive Material, 1996 Edition (Revised) Safety Requirements. Safety Standards Series No. TS-R-1.

[5] Training Course on Burnup Credit in Spent Fuel Management Systems, China Institute of Atomic Energy, Beijing, 1–12 July 2002.

[6] A. Lebrun and G. Bignan, "Non Destructive Assay of Nuclear LEU Spent Fuels for Burnup Credit Applications", IAEA-TECDOC-1241, pp. 251–268

[7] http:/www.bnl.gov/ispo/SP-1s/05TNS-011.htm

[8] G. Bignan and D. Janvier, "NAJA: A New Non-Destructive Automatic On-Line Device for Fuel Assembly Characterisation and Core Loading Conformity Control", http:/www.nea.fr./html/science/rsd/ic96/4-6.pdf

[9] N. Ensslin, D. Reilly, and H. Smith, Jr., "Application Guide", http:/www.lanl.gov/orgs/n/n1/panda/00326418.pdf.

[10] A. Lebrun, C. Riffard and H. Toubon, "Cross-checking of the operator data used for burnup measurements", IAEA-TECDOC-1378, pp. 402–412.

4.4. Regulatory Aspects in Burnup Credit

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4.4.1. Introduction

This group discussed various regulatory aspects and issues related to the implementation of burnup credit. The group did not involve itself in the technical aspects nor did it seek to establish acceptance values for approving burnup credit analyses. Acceptance values were left to each member country to establish based on the regulatory practices and goals in that country.

The group began by assembling an update of the upper limits on k_{eff} of the neutron multiplication factor set by the various countries when burnup credit is used in the criticality analysis of a spent fuel cask or activity. This effort updated a table that was prepared during the previous technical meeting in 2002.

Concurrent with the updating process, the group identified general conclusions regarding burnup credit that had the general agreement and consensus of the group.

Finally, the group developed specific recommendations that followed from the general conclusions.

4.4.2. Upper Limits on k_{eff} for Burnup Credit Applications

The group prepared a table (cf. Table 3.4) of the regulatory upper limits on k_{eff} when burnup credit is used to analyse the criticality safety of activities involving spent fuel. All member countries polled, established a 5% safety margin for criticality safety under the normal conditions of operations in wet pools when taking burnup credit. A couple of countries varied from a 5% margin for special cases of abnormal conditions involving spent fuel. When credit is taken for the burnup of fuel, some countries increase the safety margin for some operations versus an analysis without credit for burnup.

For accident conditions, some countries have a 5% safety margin while others reduce the safety margin to 2% (0.98) (including uncertainties and tolerances). If the margin is reduced,

* only temporarily present

it usually reflects a conclusion that the probability of such accidents is very low. In the UK a limit on k_{eff} in wet storage under normal conditions is raised to 0.98 for a dropped fuel accident on a case-by-case basis. In addition, some countries reduce their safety margin for very extreme accident scenarios such as optimum moderation conditions and for package arrays.

Some limits have not been finalized and are indicated as proposed limits. Entry where no limit is given means that the country does not have any spent fuel operations in that category.

4.4.3. Conclusions about burnup credit implementation

4.4.3.1. Burnup credit is useful and viable to develop and implement

There has been much interest by the nuclear industry in including credit for the burnup of irradiated fuel in the criticality safety analysis of spent fuel operations. It is recognized that significant benefit can be derived from the implementation of burnup credit in the storage, transport, disposal, and reprocessing of spent fuel. The benefits include economic savings, reduced risk, and the passive presence of the effect of burnup.

Under burnup credit, the capacity of spent fuel casks can be increased resulting in the need for fewer numbers of casks for dry storage and transport. Burnup credit can be used as an alternative to poison plates or other neutron absorbing components. Reduction in the loading and/or number of neutron absorbing components results in reduced cost for manufacturing spent fuel casks.

Cost savings in pool storage can occur by allowing closer spacing of fuel which increases the storage capacity of the pool. Utilizing burnup credit in the design of a disposal site can reduce the size and cost of the disposal field as well as reducing the cost of the disposal unit. Employing burnup credit in the design of reprocessing dissolvers can result in cost savings from increased throughput, simplified design and operations, and reduced waste streams.

In addition to the above economic benefits, a reduction in the number of spent fuel casks results in fewer loading operations for the same number of spent fuel assemblies. This reduction in the number of loading operations results in reduced operational costs as well as a reduction in operational exposure. Fewer casks also mean fewer spent fuel shipments. This reduction in exposures and number of shipments, results in a general reduction in the overall risk from activities involving spent fuel.

Table 4.4. Upper Limits k_{limit} of the Neutron Multiplication Factor in Burnup Credit Applications in Different Countries (Figures in italics are non-administrative values)

		Armenia	Bulgaria	China	Czech Republic	France	Germany	Hungary	Lithuania	Slovakia	Spain	Sweden	UK	Ukraine	USA
Normal Conditions	Wet Storage	0.95	0.95	0.95	0.95	0.95	0.95	0.95	0.95	0.95	0.95	0.95	0.95	0.95	0.95
	Dry Storage / Transport	0.95	0.95	0.95	0.95	SP: 0.95** PA: 0.98**	0.95	0.95	0.95	0.95	0.95	0.95**	0.95**	0.95	0.95
	Disposal	-	-	-	-	-	0.95*	-	-	-	-	***	-	-	<1.0*
	Dissolver (Reprocessing)	-	-	-	-	0.97 – 0.98 (case dependant)	-	-	-	-	-	-	0.92 – 0.98 (case dependant)	-	-
Accident Conditions	Wet Storage	0.95	0.95	0.98	0.95 (OM: 0.98)	0.95 – 0.97 (case dependant)	0.95 – 0.98 (case dependant)	0.98	0.95	0.98 OM	0.95	0.98	0.95 – 0.98 (case dependant)	0.95	<1.0
	Dry Storage / Transport	0.95	0.95	0.95 (OM: 0.98)	0.95 (OM: 0.98)	SP: 0.95** PA: 0.98**	0.95	0.98	0.95	0.98 OM	0.98 OM	0.98**	0.95**	0.95	0.95
	Disposal	-	-	-	-	-	<1.0*	-	-	-	-	***		-	<1.0*
	Dissolver (Reprocessing)	-	-	-	-	0.97 – 0.98 (case dependant)	-	-	-	-	-	-	0.92 – 0.98 (case dependant)	-	-

* = proposed limits, ** = no dry storage, *** = not yet decided, OM = optimal moderation, SP = single package, PA = package array

A third benefit of burnup credit to criticality safety is the fact that burnup effects are a passive feature. The effect of fuel burnup is an ever present property of the fuel and does not need any special operating procedures or administrative controls for it to be activated.

Because of these features, burnup credit is being actively pursued by many groups in a number of the member countries. Their efforts have resulted in a number of reports and documentation in the open literature. The development efforts in burnup credit include technical studies of its effects, experimental measurements, and assessments of the benefits. As a result, the general phenomena and benefits of burnup credit are fairly well known and understood.

Based on the available information, the group concluded that the development of burnup credit has matured to the point where it is worthwhile to pursue and has great promise for implementation. Although additional work needs to be done in the development of burnup credit before its full potential can be realized, it is felt that the benefits can be significant and such development is worthwhile. Thus, continued effort in the development of burnup credit is encouraged and it is recognized that when a proper technical foundation is presented, burnup credit can be an acceptable safety measure.

4.4.3.2. Important to set standards

There is a general consensus that the application of burnup credit is a very complex issue, which requires highly sophisticated methodologies for calculating burnup and depletion values as well as challenging criticality calculations for spent fuel assemblies under a variety of conditions. Codes have to be validated, and safety margins and uncertainties of every step of an analysis must be determined. Compliance with existing safety criteria must be ensured. Therefore it seems very advantageous, from the regulator's point of view as well as from the viewpoint of applicants, to have standards and/or guidelines for the application of burnup credit.

The degree of detail given in a standard depends on the safety approach, the method of application, and safety philosophy. In a standard, the basic steps of a criticality analysis including burnup credit considerations should be described and safety limits should be documented. The following topics should be covered:

- Required conditions for inventory calculations for the spent fuel
- Specification of isotopes (Actinides and Fission Products (FPs)) allowed for consideration
- Demonstration that the assumptions and approach are appropriately conservative
- Specification of safety margins, which need to be determined
- Validation of the applied codes for calculating isotopic inventory and criticality
- Requirements concerning uncertainty analyses
- Requirements for determining the actual burnup of the spent fuel
- Requirements to ensure procedural compliance with the safety criteria
- Requirements on risk informed methods, if in compliance with the safety approach of the respective country.

Several countries have already developed regulatory and/or guidance documents on burnup credit implementation for wet and dry storage and transportation of spent nuclear fuel. Examples are the interim staff guidance (ISG)-8 of the USA and the DIN standards 25471 and 25712 of Germany.

The group agreed that the development of *international* standard, guidelines or recommendations would be very helpful for those countries, which plan to allow burnup credit application in the future. Further, *international* guidelines could be useful for countries which have already developed regulatory documents so they can review these recommendations and, if desired, make their regulations consistent with the international guidelines.

4.4.3.3. Industry and public involvement in standards development

Regulatory guidance is often created by the regulators and supporting organizations. Guidance development has benefited from discussions with the industry during the draft phase. However, the regulators give final approval to the guidance.

Industrial standards (such as the DIN standards) are normally created by a group of specialists from industry, the regulators, and expert organizations. Such standards are the result of discussions within this group and may not necessarily result in full agreement by every member of the group.[8)] The draft of the standard is often published for comment by the public.

The existing standards and regulatory guidance establish conditions for the application of burnup credit in the country of origin. The legal state of the document will determine how closely the applicants and regulators must follow it. The standards and guidance help applicants by specifying requirements that shall be met for a successful application. On the other hand, during the preparation of the standard/guidance the knowledge of the regulators and the applicants about the process of implementing burnup credit can be significantly improved (by discussions and substantial contributions from the participants).

The discussion group believes it is important that the nuclear industry and any other interested parties are given the opportunity to comment on the proposed burnup credit standard.

It is felt that a wide consultation will lead to a greater understanding and appreciation of the issues that are important to each of the stakeholders. In turn, this will lead to the production of a burnup credit standard that meets the needs of industry as well as being acceptable to other interested parties. However, it must be recognized that some countries may have restrictions in place that would limit or prohibit public consultation.

It is also considered appropriate that once a standard is issued, it must be reviewed on a regular basis (for example, every five years may be appropriate) to ensure that its provisions continue to meet the needs of the industry, the regulators and other stakeholders. It is appropriate to leave the form of the consultation up to each individual country (a possible method could be to post a notice on a designated website).

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8) Comment (J.C. Neuber): This is also true very often for non-industrial regulatory standards (as for instance for the German KTA rules).

4.4.3.4. Need for more assay measurement data

Knowledge of the nuclide composition of irradiated fuel is necessary when performing a burnup credit analysis. The group agreed that there was an important need for more measurement data that are publicly available. The areas of greatest need exist for data on VVER-440/1000 fuel, high burnup fuel and MOX fuel. These data are needed to validate and benchmark the depletion codes used to provide a calculated estimate of the isotopic inventory in the spent fuel based on its irradiation history in the reactor core. Although data needs exist for validating both criticality and depletion calculations, the group recognized a particular lack of PIE assay data.

To deal with the task, the following has to be "well documented":

- full description of initial fuel composition (not only U235 and U238, but some of the typical impurities in fuel as U236), as well as, cladding composition and exact dimensions of pins, cladding, shroud, etc.

- full description of irradiation history including exact power (either relative or absolutely), time length of operations and outage, and position in assembly (pin and high), for each measured sample

- full description of measured parameters (burnup, error of measurement, cooling time, isotopic composition) for each measured sample.

4.4.3.5. Performing code validation and setting biases is necessary

While it is important to take care in performing all parts of a burnup credit analysis, the group believes that the area of code validation needs to be emphasized as requiring particular attention. Validation and a conservative assessment of uncertainties (biases, tolerances, operational data, etc.) for computer codes and their corresponding libraries is a strict requirement which is necessary for burnup credit applications.

Validation of the criticality codes as well as the depletion methodology must be performed using experimental data. Experimental data need to be of a similar fuel type to the fuel type on which the burnup credit analysis is performed. Validation of the depletion calculation methodology should be carried out against PIE assay data (isotopic composition) for the specified fuel.

Also, the validation process should determine the range of parameters for which the code is valid and can be applied for burnup credit analyses.

4.4.3.6. Use of risk informed considerations should be investigated

The prime objective of any criticality assessment based on burnup credit is to determine that the probability of a criticality event (e.g. a latent misloading event with a subsequent increase in moderation) is "sufficiently" unlikely that it can be discounted as being outside of the design basis. This is traditionally demonstrated via a deterministic approach to show defense in depth.

There are significant human factors involved in a number of the steps required for the application of burnup credit — both in the formulation of the criteria and the handling of the fuel assemblies. Operational experience suggests that people will make mistakes. In addition,

the steps involved in a burnup credit analysis should be reviewed to identify and target the steps with the highest safety significance.

If a criticality occurs, the consequences could vary greatly depending on the type of application (e.g. underground repository, transport). Without any knowledge of the consequences, it is difficult to define what would be an acceptably small frequency for a criticality event. If the risk is known, then its acceptability can be judged in the context of the overall safety analysis for the plant. Work is therefore needed to establish the likely consequences for the various applications.

The adoption of a risk informed approach could help to determine the need, if any, for additional engineered safeguards/protection (e.g. criticality detectors, boron monitors). It would also allow the utility and the regulator to focus their efforts toward the risk significant issues.

The benefits of a risk informed approach to burnup credit would include a more rigorous consideration of the fuel properties to better optimize spent fuel storage, and thus, could lead to a reduction in the number of overall spent fuel movements.

4.4.4. Recommendations

4.4.4.1. The IAEA is urged to assist VVER validation

The group recognized that there is a special need for measured assay data that can be used to validate the depletion code calculations when applying burnup credit to spent VVER fuel. Assay data for this fuel type are very limited and the countries which are seeking to apply burnup credit to VVER fuel have limited financial resources.

A group of VVER users is currently engaged in a series of meetings to establish a basis for sharing the existing data and the costs of a follow-on project to obtain additional assay data. To support this initiative, it would be beneficial to organize a workshop focused on defining the scope necessary for obtaining new PIE data.

The group recommends that the IAEA become involved in these efforts to help facilitate international coordination and assist in establishing a program to obtain additional assay data.

4.4.4.2. There is a need for analysis of alternatives to burnup credit

Burnup credit analyses require a complex set of calculations which can be costly to develop. This is particularly true when considering the need for experimental data which is sufficient to validate and benchmark the calculation codes. The group recognizes that there may be some applications where measures other than burnup credit could be relied upon to provide criticality control. These measures include partial credit for the boron content in a wet storage pool, credit for dry storage casks to maintain its integrity against water intrusion during storage, and arrangements which allow the use of a sufficient amount of solid neutron absorbers in lieu of burnup credit.

The group concluded that analyses of these alternative measures, which compared their usefulness as criticality controls in place of burnup credit, would be beneficial to countries which find the cost of implementing burnup credit burdensome and may wish to evaluate less costly alternatives. Thus, efforts to organize analysis activities and potentially a working group in this area are encouraged by the discussion group.

4.4.4.3. The IAEA is urged to assist the development of a standard or guidelines for implementing burnup credit

The group agreed it would be advantageous for an international organization to promote the development of standards or recommended guidelines on burnup credit. Guidance developed under the sponsorship of an international agency such as the IAEA would have sufficient stature to gain general acceptance in countries seeking to implement burnup credit. Such guidance would be very helpful for those countries which plan to allow the implementation of burnup credit in the future and which do not have the financial resources to develop a comprehensive set of guidance on their own. This guidance could serve as a template for structuring a national program for implementing burnup credit. Therefore, the group recommends that the IAEA assist in efforts to develop internationally accepted guidance for the implementation of burnup credit.

4.4.4.4. Study into the applicability of risk informed methods to burnup credit criticality safety assessment is recommended

The group believes that consideration of risk factors during the implementation of burnup credit would be beneficial in concentrating efforts and resources into those areas of greatest safety significance. Due to the complexity and difficulty of a valid burnup credit analysis, efficient allocation of the resources needed is beneficial. When implementing burnup credit, decisions must be made on the acceptability of simplifying assumptions, approximations, extrapolations beyond the data range, and levels of accuracy of any proposed analysis methodology. Thus, efforts to organize analytical studies and a possible working group in this area are encouraged by the discussion group. The results of such efforts should be made publicly available to the interested community.

INTERNATIONAL ACTIVITIES

(Session 1)

Overview of IAEA spent fuel management activities

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Abstract.. This paper summarizes activities of the International Atomic Energy Agency (IAEA) relevant to the application of burnup credit in the backend of the nuclear fuel cycle. It highlights related work in spent fuel management that the IAEA has carried out for the past twenty-five years. It summarizes efforts initiated in 1997 to establish a forum for exchange of technical information and thereby compile state-of-the-art information on advances in burnup credit applications. It summarizes global trends that motivate Member State interest in pursuing the efficiencies and attendant cost savings that burnup credit applications offer. It clarifies that the IAEA is committed to maintaining a high priority for these burnup credit activities.

1. Introduction

As delays are incurred in implementing plans for geologic repositories and for reprocessing, storage of power reactor spent fuel for extended durations is becoming a progressive reality. This trend of more storage for longer durations is expected to continue. The situation is complicated by trends toward higher initial enrichment, higher fuel burnup, as well as other considerations including the use of evolving fuel designs and mixed oxide (MOX) fuel. Since over 10,000 metric tons of heavy metal (tHM) are unloaded from the world's ~440 reactors each year and less than one third is reprocessed, about 8 000 t HM/year on average will need to be placed into interim storage facilities. While most spent fuel is in wet storage, use of dry storage is becoming more important as durations extend. And as some nations consider storage periods of 100 years and even beyond, more attention is being directed toward securing and maintaining related prerequisites including preservation of technical knowledge, records, and stability in funding, and infrastructure. Given the importance of effective spent fuel management to sustainable utilization of nuclear energy, Member States of the International Atomic Energy Agency (IAEA) maintain an active interest in related work, as evidenced in part by participation in IAEA-sponsored meetings

2. Spent fuel management programme

For the last twenty-five years, the IAEA has been proactively involved in spent fuel management activities. The Nuclear Fuel Cycle and Materials Section within the Department of Nuclear Energy organizes various meetings, often focused on producing technical documentation available to all Member States on a topic of interest. IAEA technical documents can be downloaded free of charge at http://www-pub.iaea.org/MTCD/publications/tecdocs.asp. As a result of the trends noted above, IAEA activities on spent fuel management have enhanced scrutiny of issues associated with long term spent fuel storage, with following examples. As storage durations extend, obtaining and extrapolating information on the behavior/performance of fuels and materials in storage is an important ingredient in continued confidence of both implementers and regulators. The IAEA coordinated research project on spent fuel performance assessment and research (SPAR-II) initiated last year is focused on specific research objectives involve surveillance and monitoring programmes for spent fuel storage facilities, fuel/materials performance evaluation for wet/dry storage, and collection and exchange of spent fuel storage experience. In addition to documentation of this on-going SPAR-II work and the burnup credit activities described below, a range of technical documents will be published on topics including data requirements and records maintenance, economics of spent fuel storage, operations/maintenance of

casks and containers, regional spent fuel storage aspects, and cask/container loading optimization. The IAEA will also continue plans for periodic large conferences on spent fuel management to foster a wide exchange of current information and to stimulate creative dialogue on emerging trends. One hundred twenty-five representatives from thirty-five Member States and three international organizations participated in the 2003 IAEA spent fuel conference held in Vienna (proceedings can be accessed at http://www-pub.iaea.org/MTCD/publications/PDF/csp_020c/Start.pdf). The next IAEA conference on spent fuel management is planned for 19–23 June 2006 in Vienna. Further information regarding IAEA spent fuel management activities can be accessed at http://www.iaea.org/OurWork/ST/NE/NEFW/nfcms_b3.html.

3. Activities focused on burnup credit applications

For almost a decade, the IAEA has taken an active role in disseminating information related to applications of burnup credit for spent fuel management. The IAEA monitors the status of burnup credit application and provides a forum to exchange related information, for example regarding the status of national practices of burnup credit implementation in the Member States.

In October 1997, the IAEA organized a technical meeting in Vienna to examine and report on the status of burnup credit for storage, transport, reprocessing, and disposal of PWR, BWR, VVER, RBMK and MOX spent fuel. The proceedings of that meeting were published in April 1998 as IAEA-TECDOC-1013, entitled "Implementation of burnup credit in spent fuel management systems."

In July 2000, the IAEA organized a second technical meeting on this topic in Vienna. 35 experts from 17 countries and 2 international organizations surveyed the progress and status of international activities related to the use of burnup credit for spent fuel applications. Participants recognized the value of international cooperation on this topic and recommended further studies of axial effects, and verification methods for fuel burnup values, including cooperation in future experimental programmes and sharing of available data. Participants also recommended holding a training course for potential users of burnup credit and their respective regulators. The proceedings of the technical meeting in 2000 were published in August 2001 as IAEA-TECDOC-1241, also entitled "Implementation of burnup credit in spent fuel management systems."

As recommended in the latter technical meeting, a training course on the implementation of burnup credit in spent fuel management systems was held in the United States of America at the Argonne National Laboratory 15–26 October 2001 with 25 course participants from 12 different countries.

In April 2002, the IAEA held its third technical meeting on burnup credit applications in Madrid with participation from 54 experts from 18 countries. Building on the results of preceding meetings and related developments, participants presented 40 reports in eight sessions. Thereafter, four parallel working groups focused on code validation, key issues, safety assessments, and future applications. The meeting concluded with a recommendation that the IAEA continue its activities on burnup credit due to its increasing importance for Member States having to deal with increasing spent fuel storage quantities and durations. The proceedings of this meeting were published in 2004 as IAEA-TECDOC-1378, entitled "Practices and developments in spent fuel burnup credit applications."

In the area of technical cooperation, the IAEA has coordinated a project with China focused on technology transfer related to burnup credit. Project activities from 2001 to 2005 included scientific visits, expert missions, equipment procurement, and fellowships.

In addition to the larger technical meetings (TMs) described above, small consultancies have been held over the years both to handle pre- and post-TM actions and to monitor interim progress in burnup credit implementation. For example, a consultancy meeting was held June 2004 in Vienna to review progress and to prepare for the 2005 technical meeting held in London.

Prior to the 2005 technical meeting, participants were requested to review and update the tables in IAEA-TECDOC-1378 summarizing current BUC applications status and the BUC level implemented

in each application by country. Once all updates are received and incorporated, the revised tables will be included in the TECDOC proceedings of the meeting.

IAEA technical meetings held to date have concluded that the use of burnup credit for spent fuel management continues to progress and have recommended continued acquisition of data to support burnup credit. As one example of current interest in burnup credit applications, a June 2005 letter to the USNRC from their Advisory Committee on Nuclear Waste advised they consider allowing realistic burnup credit in the certification of spent fuel transportation casks. IAEA meeting proceedings to date have documented significant developments serving to advance the use of burnup credit in Member States. Participating experts continue to make important contributions in this regard. In the future, the IAEA plans to continue to assign a high priority to activities related to burnup credit applications. For example, the budget cycle for 2006/2007 contains a sequel task on burnup credit applications calling for a consultants meeting to prepare for a subsequent fifth technical meeting. Also, an earlier proposal for a task related to chemical assay data of WWER fuel was shelved pending availability of these data. Results of the 2005 TM will be useful in determining if the IAEA should pursue this activity in the pending budget cycle.

4. Conclusions

Spent fuel storage has been carried out safely and effectively for decades, and there is high confidence that this will continue to be the case. Yet as storage inventories and durations increase, issues associated with long term storage compel more attention, as witnessed by participation of IAEA Member States in IAEA meetings such as the Agency's 2003 spent fuel storage conference and the 2004 Scientific Forum on nuclear fuel cycle issues. Trends toward more storage capacity for longer durations are complicated by trends toward higher initial enrichment, higher fuel burnup, as well as evolving fuel designs. Motivated by these trends, the IAEA has enhanced scrutiny of issues associated with extended spent fuel storage durations and quantities. Recent activities have examined issues associated with materials aging, performance monitoring, economics, maintenance, data requirements, cask loading, spent fuel treatment, regional facilities, and facility selection criteria. The IAEA continues to assign a high priority to activities associated with implementation of burnup credit, given the potential for increased storage capacity and resultant reduced costs and operational exposure.

Status of burnup credit activities at OECD/NSC

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Abstract. This paper summarizes activities within the Organisation for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA). The focus is on the work of the Expert Group on Burnup Credit* Criticality, a subordinate group to the Working Party on Nuclear Criticality Safety (WPNCS) working under the Nuclear Science Committee (NSC). The Working Group on Operating Experience/Fuel Cycle Safety (WGOE/FCS), a subgroup of the Committee on the Safety of Nuclear Installations (CSNI), has also expressed interest and is monitoring the licensing and use of burnup credit in member countries. Indirectly through the benchmarking and validation efforts associated with burnup credit, there is a relationship with the nuclear data development activities of the Working Party on International Nuclear Data Evaluation Cooperation (WPEC).

The WPNCS of the OECD/NEA coordinates and carries out work in the domain of criticality safety at the international level. Particular attention is devoted to establishing sound databases required in this area and to addressing issues of high relevance such as burnup credit. The activities of the expert group are aimed toward improving safety and identifying economic solutions to issues concerning the back-end of the fuel cycle. The group has established and evaluated a number of calculational benchmarks addressing the physics and modeling needs for performing burnup credit analyses for light water reactor fuels.

The discussion in this paper is to review the results of 14 years of cooperative investigation into the development and validation of burnup credit computational methodologies based on the work of the Expert Group on Burnup Credit Criticality (1991–present).

1. Introduction

The importance of the safe handling of fissile materials was recognized at an early stage both by the scientific community and the responsible authorities. In fact, nuclear criticality safety was established as a discipline more than 50 years ago in response to several accidents that occurred in nuclear weapons programs. At the beginning, intensive experimentation with a large variety of configurations and materials took place in order to establish a basis of knowledge for such systems. Over the years, substantial progress has been made in developing nuclear data and computer codes to evaluate criticality safety for nuclear fuel handling. The accuracy and reliability of computer code calculations has been extensively benchmarked using the experimental data that had formed the foundation for criticality safety. These validated criticality calculational tools can be utilized to evaluate proposed fissile equipment designs and operational activities and establish limits and controls to assure safety. This application of state-of-the-art calculational tools for criticality safety evaluations has led to reduction of the uncertainties in safety margins and has allowed rational and more economical designs for manipulation, storage and transportation of fissile materials.

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* *Burn-up credit is a term that applies to the reduction in reactivity of burned nuclear fuel due to the change in composition during irradiation.*

2. Working party for nuclear criticality safety (WPNCS)

OECD/NEA has coordinated the activities of this criticality safety benchmark group for more than two decades. The Working Party for Nuclear Criticality Safety (WPNCS) was officially chartered in 1997 to review and coordinate the activities of the existing expert groups operating under the auspices of OECD/NEA and to propose establishing task forces (expert groups) corresponding to new demands on methods development, experimental needs and international handbook data in the field of nuclear criticality safety. Groups of criticality safety experts have been working under the auspices of OECD/NEA under different names since 1980 to address topics of common interest and concern such as nuclear fuel transportation and storage and fuel dissolution issues.

The scope of the WPNCS covers technical criticality safety issues relevant to fabrication, transportation, storage and other operations related to the fuel cycle of nuclear materials. Figure 1 illustrates the current scope of activities being addressed by the WPNCS. The working party primarily provides guidance to promote and coordinate the identification and investigation of high priority issues of common interest to the international criticality safety community. In doing this, the WPNCS maintains a priority list of the needs of the nuclear criticality safety community and submits proposals to the OECD/Nuclear Science Committee (NSC) on the establishment of specific expert groups to address these issues as deemed appropriate.

Expert groups have been established for:

- Developing an experiments database for critical and sub-critical experiments — International Criticality Safety Benchmark Evaluation Project (ICSBEP)
- Identifying needs for critical, subcritical and supercritical experiments — Experimental Needs
- Establishing/updating basic criticality condition data —Minimum Critical Values
- Verifying the adequacy of existing codes and data for application with burned fuel - Burnup Credit Studies.
- Analysing convergence problems associated with criticality calculations of loosely coupled fissile units
- Studying the phenomenology of criticality excursion.

Several of the issues currently being addressed at the level of the WPNCS were initially identified within the EGBUC and then found to have impact/interest beyond burnup credit. Examples of these issues include:

- Numerical convergence in computing criticality of decoupled fissile systems such as spent fuel assemblies. This problem needs to be addressed for both deterministic and stochastic methods (a specific benchmark has been proposed for Monte Carlo methods).
- Effects of geometrical approximations in pin cells, e.g. square versus cylindrical.
- Mixed configurations of different units with fissionable material.

Information about the current activities and links to publications of the OECD/NEA Working Party on Nuclear Criticality Safety may be found at http://www.nea.fr/html/science/wpncs/ The OECD/NEA Secretariat for WPNCS is Y. Rugama (Yolanda.RUGAMA@oecd.org) who may be contacted for additional information.

Figure 1 also illustrates the different levels of coordination within OECD/NEA. Validation of codes and data, benchmarking, criticality safety handbooks and standards are common themes among the different expert groups within the WPNCS. These activities often require integration and coordination with the Working Party on International Nuclear Data Evaluation Cooperation (WPEC). The NEA's nuclear data evaluation co-operation activities involve evaluation projects in the following regions: Japan (JENDL), United States (ENDF), Western Europe (JEFF), and non-OECD member countries (BROND, CENDL, and FENDL). The participation of non-OECD member countries in these evaluation projects is channeled through the Nuclear Data Section of the International Atomic Energy Agency (IAEA). Information about the current activities and links to publications of the OECD/NEA Working Party on Nuclear Criticality Safety may be found at http://www.nea.fr/html/science/wpec/ The OECD/NEA Secretariat for WPEC is C. Nordborg (Claes.NORDBORG@oecd.org) who may be contacted for additional information.

A similar overlap exists between the EGBUC and the Subgroup on Fuel Cycle Safety (FCS), which is under the Committee on the Safety of Nuclear Installations (CSNI). The subgroup is a joint regulator and industry group devoted to nuclear fuel cycle safety in the international community. The group has broad wide ranging interests covering topics which encompass the full scope of fuel cycle activities, including but not restricted to; safety assessments, nuclear criticality safety, probabilistic safety assessment, safety management, decommissioning and site remediation, fire protection and human factors as well as other wide ranging topics. The Fuel Cycle Safety subgroup has developed a burnup credit questionnaire to establish a mutual understanding of burnup credit from the regulatory view-point and to evaluate the differences from country to country and establish the possibility of standardized burnup credit in regulation for the future. Additionally, the group has just published the 3rd edition of The Safety of the Nuclear Fuel Cycle which contains general safety information on criticality. Information about the current activities and links to publications of the OECD/NEA WGOE/FCS may be found at http://www.nea.fr/html/nsd/csni/fcs.html The OECD/NEA Secretariat for this work is B. Kaufer (Barry.KAUFER@oecd.org) who may be contacted for additional information.

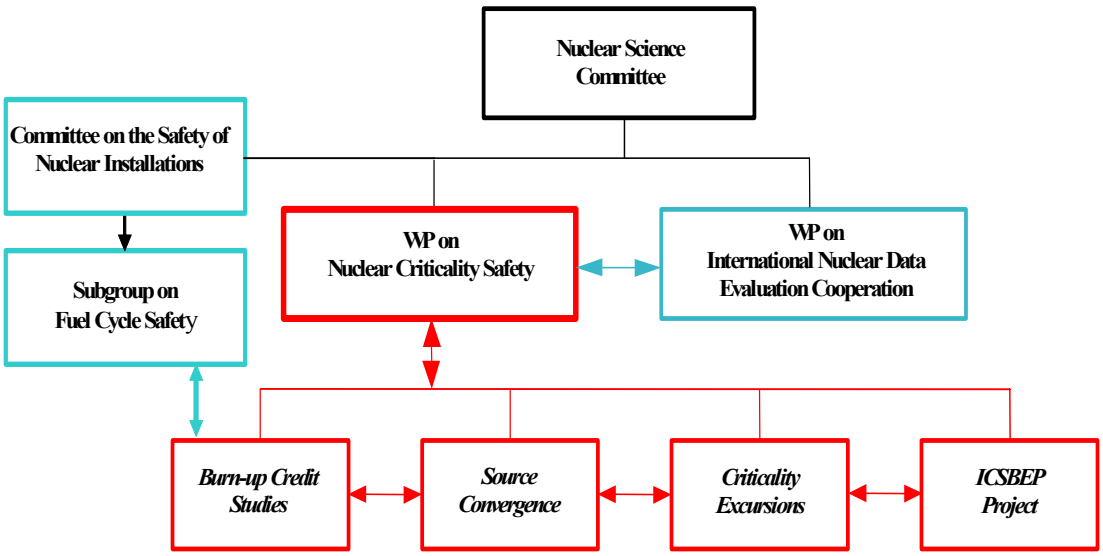

Figure 1. Existing relationship between working parties reporting to the OECD/NEA Nuclear Science Committee and the criticality safety expert groups.

3. Expert group on burnup credit criticality

The scope of the Expert Group is to study Burnup Credit as applied to criticality safety in the transportation, storage, and treatment of spent fuel for a wide range of fuel types, including UOX and MOX fuels for PWR, BWR and VVER.

Under the guidance of the Working Party on Nuclear Criticality Safety, the major objectives of the Expert Group include:

- carrying out international comparison exercises and benchmarks and to assess the ability of code systems to predict the reactivity of spent nuclear fuel systems, including comparison with experimental data as available;
- investigation of the physics and predictability of burn-up credit based on the specification and comparison of calculational benchmark problems;
- publication of the results for the benefit of criticality safety community, so that the work may be used to help establish suitable safety margins.

Official information about the Expert Group is available at http://www.nea.fr/html/science/wpncs/buc. The OECD/NEA Secretariat for the EGBUC is Y. Rugama (Yolanda.RUGAMA@oecd.org) who may be contacted for additional information.

The main goal of the activities of the OECD/NEA Expert Group on Burnup Credit Criticality is to demonstrate that the available criticality safety calculational tools are appropriate for application to burned fuel systems and that a reasonable safety margin can be established. For this purpose the Expert Group established a suite of burnup credit criticality benchmarks that assess the capability to calculate both spent fuel composition and reactivity of spent fuel. The benchmarks were carefully specified to allow a comparison of results using a wide variety of calculational tools and nuclear data sets. Throughout the tenure of the activities of the Expert Group on Burnup Credit Criticality, experts from 17 countries (Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Hungary, Italy, Japan, South Korea, the Netherlands, Spain, Sweden, Switzerland, United Kingdom and the United States) have participated in various phases of the benchmark exercises. Participants used a wide variety of codes and methods based on transport theory, using SN, nodal and Monte Carlo techniques. Nuclear data (both cross-section and decay data) were taken from a variety of sources: multiple versions of the Evaluated Nuclear Data Files (ENDF/B), the Japan Evaluated Nuclear Data Libraries (JENDL) and the Joint Evaluated Fission and Fusion (JEFF) Libraries. Both multi-group and continuous energy cross-section data were used in the study.

Table I is a summary of the benchmark problems addressed noting both the primary objective and current status of each.

Phase I and Phase II included both criticality and depletion benchmarks for pressurized water reactors (PWRs). A set of selected nuclides including 7 major actinides (U-234, 235, 236, and 238; Pu-239, 240 and 241), 5 minor actinides (Pu-238 and 242; Am-241 and 243; Np-237) and 15 fission products (Mo-95; Tc-99; Ru-101; Rh-103; Ag-109; Cs-133; Sm-147, 149, 150, 151 and 152; Nd-143 and 145; Eu-153; and Gd-155) were used in these studies. The results showed no trends in standard deviation among participants with burnup or cooling time in the criticality analyses. Consistently the largest deviations among participants were for the fresh fuel cases. In the depletion analyses, there was evidence of a significant trend in the standard deviation among participants for the residual U-235 (the trend was small for most other isotopes). A number of nuclides have been identified for additional study based on the sensitivity of k to the observed standard deviations: Pu-239, Gd-155, U235, Pu-241, Pu-240 and Sm-151. Much of the differences are assumed to be in the basic nuclear data. Both 2-D and 3-D models have been used to evaluate the impact of axially distributed burnup. It was determined that 70% of the total fissions occur in the upper 40cm of fuel that illustrates the potential

importance of this parameter. Good agreement was seen among the participants relative to the calculated "end effect". It has been noted by the group that the effect on k is strongly a function of the system being evaluated and may be even more important under postulated accident conditions that result in axial heterogeneity. Two remaining issues associated with the axial effect continue to be investigated in the expert group: (1) limited availability of measured axial profile data and detailed power history data in the open literature, and (2) defining/performing analyses to determine the sensitivities due to different axial burnup profiles across the full range of burnups.

TABLE I. SUMMARY OF BENCHMARK PROBLEMS ADDRESSEDBY THE OECD/NEA EXPERT GROUP ON BURNUP CREDIT CRITICALITY

Benchmark	Primary Objective	Status
Phase I-A	Examine effects of seven major actinides and 15 major fission products for an infinite array of PWR rods. Isotopic composition specified at 3.6 wt.% ^{235}U at 0, 30 and 40 GWd/MTU and at one- and five-year cooled.	Completed (Ref. 1)
Phase I-B	Compare computed nuclide concentrations for depletion in a simple PWR pin-cell model, comparison to actual measurements at three burnups (27.34, 37.12 and 44.34 GWd/MTU).	Completed (Ref. 2)
Phase II-A	Examine effect of axially distributed burnup in an array of PWR pins as a function of initial enrichment, burnup and cooling time. Effects of fission products independently examined.	Completed (Ref. 3)
Phase II-B	Repeat study of Phase II-A in 3-D geometry representative of a conceptual burnup credit transportation container. Isotopic compositions specified.	Completed (Ref. 4)
Phase II-C	Key sensitivities in criticality safety to burnup profiles.	Report in Draft
Phase II-D	Effect of absorbers/control rods.	Final Draft Approved 2005
Phase II-E	Combination of Phase II-C and Phase II-D and benchmark the asymmetry effect on the end effect with the CR insertion effect on the isotopic inventory	Proposed
Phase III-A	Investigate the effects of moderator void distribution in addition to burnup profile, initial enrichment, burnup and cooling time sensitivities for an array of BWR pins.	Completed (Ref. 5)
Phase III-B	Compare computed nuclide concentrations for depletion in a BWR pin-cell model.	Completed (Ref. 6)
Phase IV-A	Investigate burnup credit for MOX spent fuel pin-cell for three plutonium vectors (first recycle, fifth recycle, weapons-grade)	Completed (Ref. 7)
Phase IV-B	Compare computed nuclide concentrations for depletion in a MOX super-cell.	Completed (Ref. 8)
Phase V	VVER burnup credit. Similar to Phases I and II for PWRs but with hexagonal geometry and WWER fuel specification	Independent/ Parallel Study

Phase III included both criticality and depletion benchmarks for boiling water reactors (BWRs). For the most part the results are consistent with those for PWRs: the largest deviations among participants are for the fresh fuel cases, and deviations are higher for distributed burnups versus modeling the average burnup. Larger void fractions (i.e. use of a 70% uniform void distribution) tended to increase the deviation among participants. The complex geometry of the BWR fuel assemblies added

complexity to the depletion calculation. These results are in final review and should be published shortly.

Mixed oxide (MOX) fuels in PWRs were investigated in Phase IV. The problems included a MOX pincell calculation to identify sensitivities specific to MOX. The primary result of this early benchmark was to identify the need to include curium isotopes in both the criticality and depletion calculations, as Cm contributes up to 1.5% in k.

Phase V is a completely parallel study being led by L. Markova under the AER Subgroup E and is addressed in another paper at this meeting.

Since the objective of the Expert Group on Burnup Credit Criticality thus far has been to assess code capabilities, the results are most often presented as the standard deviation among participants. There has been no attempt to make a safety case for licensing or to provide bounding values on the observed trends or physical phenomena (e.g. the effect of axially distributed burnup). However, the group does discuss specific or suspected sources of discrepancies, leading to the identification of further studies. [9, 10]

The EGBUC currently has three milestones to the Nuclear Science Committee, (1) publish the Phase II-C (Evaluation of asymmetry of burnup distribution on the end effect) report in 2004; (2) publish the Phase II-D (evaluation of the effect of control rods) report in 2005; and (3) publication of the Summary Report of the activities of the Expert Group in 2005.

The Phase II-C report has been delayed. The report is being modified per agreements made at the August meeting and should be submitted for publication by mid-2006.

The Phase II-D report has been approved and submitted for publication as scheduled.

Responsibilities for coordinating contributions for the summary report have been assigned for the major chapters:

- Discussion of validation issues
- Application to PWR - UOX fuel (square-pitch design)
- Application to PWR MOX Fuel
- Application to PWR VVER-design fuel
- Application to BWR fuel.

The plan remains to consolidate these articles into an OECD/NEA report.

The group continues to discuss a proposal for Phase II-E which will look at the effect of the partial insertion of control rods during irradiation.

Highlights from the country reports at the 14th meeting of the EGBUC included the availability of REBUS results to it's investors; advancements in the development of an American National Standard on burnup credit; review of burnup credit methodology in the Czech Republic is pending the availability of new PIE data; new approval of actinide-only burnup credit in Slovakia; US-DOE efforts to expand PIE data for western LWRs; revisions to Interim Staff Guidance 8 (ISG-8) issued by the US/NRC and revisions/release of JEF 3.1 with corrections for nuclides important for BUC [e.g. major actinides (235, 238 U and 239, 241 Pu) and fission products (103Rh, 149Sm, 154Eu, etc)]. Japan (JNES) has been developing an integrated depletion code named MVP-ORBURN, by means of combining the continuous energy Monte Carlo code MVP and point depletion code ORIGEN2.

The need for Post Irradiation Examination (PIE) data for VVER fuels, specifically, as well as other LWR fuel types continues to be an issue of concern to the Expert Group.

The work of the EGBUC will continue as the interest from member countries continues to grow and the economic and risk benefits of burnup credit continue to emerge.

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TECHNICAL TOPICS

(Session 2)

PRINCIPLES OF CHOOSING THE CALCULATION METHODOLOGY WITH RESPECT TO THE FUEL DESIGN AND THE FUEL MANAGEMENT SYSTEM

(Session 2.1)

A coordinated U.S. program to address full burnup credit in transport and storage casks

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Abstract. The benefits of burnup credit and the technical issues associated with utilizing burnup credit in spent nuclear fuel (SNF) casks have been studied in the United States for almost two decades. The issuance of the U.S. Nuclear Regulatory Commission (NRC) staff guidance for actinide-only burnup credit in 2002 was a significant step toward providing a regulatory framework for using burnup credit in transport casks. However, adherence to the current regulatory guidance (e.g. limit credit to actinides) enables only about 30% of the existing pressurized-water-reactor (PWR) SNF inventory to be transported in high-capacity (e.g., 32-assembly) casks. Work has been done to demonstrate that the allowable inventory percentage could potentially increase to nearly 90% if credit for fission products were allowed. Thus, Oak Ridge National Laboratory has worked with the U.S. Department of Energy Office of National Transportation (DOE/ONT), the NRC, and the Electric Power Research Institute (EPRI) to coordinate a research program that will (a) obtain and evaluate experiment data to support the safety basis for fission product credit validation, (b) investigate unresolved technical issues associated with PWR full burnup credit, and (c) recommend approaches for boiling-water reactor (BWR) burnup credit in transport and storage casks. This paper will review the program of research and discuss the progress to date.

1. Introduction

Safe, efficient, and effective management of spent nuclear fuel (SNF) from U.S. commercial nuclear power plants will demand increasing attention to transport and storage in casks. Historically, spent fuel cask designs have had to demonstrate criticality safety and structural integrity while meeting limits on weight, thermal loading, external dose, and containment. With the reduced thermal load and dose provided by a minimum 5-year cooling time for transport of domestic SNF, it became apparent in the late 1980s that SNF cask capacity would often be limited by the conservative, yet simple fuel assumption of unirradiated fuel (i.e. no credit for the fuel burnup) used in criticality safety evaluations. For pressurized-water reactor (PWR) SNF, burnup credit eliminates the need for the gapped basket structures (i.e. flux traps) used for separation and criticality control — thus providing an important degree of flexibility to cask designers. Elimination of the flux traps increases the capacity of PWR rail casks by at least 30%.

The use of high-capacity casks leads to reduced risk and reduced cost relative to storage and transport operations. Although crediting the reactivity reduction from burnup (i.e. burnup credit) is an important component of enabling SNF casks to have high capacity, the current regulatory guidance recommends credit only for the reactivity change due to major actinides (a reduction in actinides that fission and an increase in actinides that absorb neutrons). The current regulatory position [1] for transport and storage is provided in the U.S. Nuclear Regulatory Commission's (NRC's) Interim Staff Guidance 8, Revision 2 (ISG-8R2). This guidance will enable no more than ~30% of the domestic SNF inventory from PWRs to be loaded in high-capacity (~32-PWR-assembly) casks. Additional burnup credit provided by fission products (nuclides produced during burnup with neutron-absorbing properties) is necessary to enable high-capacity casks to handle the majority (up to 90%) of the domestic PWR SNF inventory [2].

In 2004, Oak Ridge National Laboratory (ORNL) prepared a roadmap for a project whose goal is to develop and/or obtain the scientific and technical information necessary to support preparation and review of a safety evaluation for cask designs that use full (actinide and fission product) burnup credit to transport PWR SNF. Subsequently ORNL has worked cooperatively with the NRC, the Electric Power Research Institute (EPRI), and the U.S. Department of Energy (DOE) Office of National Transportation (ONT) to execute the project plan. Existing critical experiments and assay measurement data will be obtained and assessed for technical value in developing an adequate safety evaluation that includes both actinide and fission product credit. In addition, the use of burnup credit in boiling-water reactor (BWR) SNF casks will be investigated, with the goal of recommending the technical approach and associated data needs for BWR fuel with enrichments up to 5 wt % to be transported in high-capacity casks.

2. Data base of critical experiments for full burnup credit

2.1. Background and approach

The potential benefits of burnup credit relative to the increased inventory of PWR SNF that could be transported in high-density casks have been demonstrated in Ref. [2] The cost savings from this inventory increase varies from a minimum of $156M to $400M depending on the assumptions relative to cask sizes. The project being discussed in this paper is seeking to obtain the data needed to enable straightforward and effective preparation and review of a criticality safety evaluation with full burnup credit. The rationale for restricting the ISG-8R2 to actinide-only is based largely on the lack of definitive experiments that can be used to estimate the bias and uncertainty associated with best-estimate analyses needed to obtain full burnup credit. Applicants and regulatory reviewers are constrained by both a scarcity of data and a lack of clear technical bases (e.g. criteria) for demonstrating applicability of the data.

Under this project, ORNL is working to obtain, and make available to industry, a well-qualified experimental data base that can ensure reliable and accurate estimation of any bias and uncertainty resulting from the codes and data used to predict the system neutron multiplication factor, k_{eff}. Rather than an *a priori* decision on suitability of candidate experiments, ORNL is seeking to obtain and assess critical experiment data from the following sources:

(a) critical experiments within the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (*IHECSBE*) [3];

(b) proprietary critical experiment data;

(c) commercial reactor criticals (CRCs); critical state points from operating reactors; and

(d) proposed new critical experiments.

The applicability and value of this data base of critical experiments are being assessed using sensitivity and uncertainty (S/U) analysis tools developed at ORNL [4] and incorporated within Version 5 of the SCALE code system [5]. The TSUNAMI-3D sequence within SCALE uses first-order linear perturbation theory [6] to calculate the sensitivity of k_{eff} for systems (e.g. SNF casks) and/or critical experiments to variations in nuclear data. Energy-, nuclide-, reaction-, and position-dependent sensitivity profiles are generated and saved in sensitivity data files. TSUNAMI-IP uses the sensitivity data file information and cross-section uncertainty data to evaluate the similarity of different systems. One of the products of this comparison is an integral index, referred to as c_k, that is a single-valued quantity used to assess similarity of uncertainty-weighted sensitivity profiles between a modeled system and a criticality experiment for all nuclide reactions. A c_k index is similar to a correlation coefficient, and a value of 1 indicates that the compared systems have identical uncertainty-weighted sensitivities. A value of 0 indicates that the systems are completely dissimilar. The current guidance [4] is that critical experiments with a c_k value of at least 0.9 are applicable for validation purposes and that c_k values between 0.8 and 0.9 indicate marginal applicability.

The SCALE S/U tools were used to analyze the GBC-32 prototypical high-capacity rail cask [7] loaded with Westinghouse 17×17 fuel (see Fig. 1) having accumulated burnups of 10 to 60 GWd/MTU. The results from this cask model serve as the reference for applicability comparisons with the sets of critical experiments under consideration.

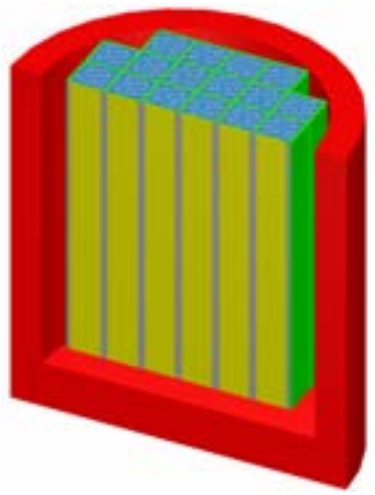

FIG. 1. GBC-32 cask model.

2.2. Assessment of IHECSBE and French proprietary experiments

As part of this project, ORNL was able to negotiate a multioption contract with Cogema to gain access to proprietary critical experiments performed at the Valduc research facility in France. These experiments are part of a larger French program [8] to develop a technical basis for burnup credit. Subsequent to assessment and evaluation, data obtained by ORNL under the contract will be made available to industry for use in cask design and licensing activities.

In late July 2005, ORNL received the first set of critical experiment data documented using the format of the IHECSBE. These experiments were performed with rods having uranium and plutonium isotopic compositions similar to $U(4.5\%)O_2$ fuel with a burnup of 37,500 MWd/MTU. The experimental series, referred to as the HTC experiments, investigated 156 configurations divided into 4 groups, as illustrated in Fig. 2. The first group is a single clean-water-moderated and water-reflected array of HTC rods with the pin pitch varied from 1.3 to 2.3 cm. The second group is similar to the first, except that boron or gadolinium is dissolved in the water at varying concentrations. The third group has four separate assemblies of HTC rods, separated by varying distances, and with borated steel, Boral™, or cadmium plates on the outsides of the assemblies in 11 of the critical configurations. The fourth group is similar to the third group, except that a thick lead or steel shield is placed around the outside of the four assemblies to simulate the type reflector representative of a cask.

These 156 HTC critical experiments, together with nearly 1000 critical configurations from the IHECSBE, have been analyzed with the TSUNAMI-IP sequence, and the sensitivity data obtained have been compared with sensitivity data for the reference cask model loaded with assemblies burned to 40 GWd/MTU. (Actinides and fission products are included in the reference model.) Figure 3 shows the distribution of the c_k values for the 1134 critical configurations when compared with the reference burnup credit cask model. As shown in the figure, the 170 ^{233}U experiments, the 150 high-enrichment uranium experiments, the 4 intermediate-enrichment uranium experiments, the 197 plutonium-only configurations, and the 256 low-enrichment-uranium experiments, all have c_k values of < 0.8. Only 45 of the 201 non-HTC mixed-oxide (MOX) configurations have c_k values ≥ 0.8, with none having c_k values ≥ 0.9. (Additional non-HTC MOX experiments continue to be assessed.) However, the strong applicability of the HTC MOX experiments is demonstrated by the fact

that 152 of the 156 configurations have c_k values ≥ 0.8, with 143 c_k values ≥ 0.9. The few experiments with c_k < 0.9 all had high soluble gadolinium concentrations to simulate systems in fuel reprocessing. The results of these studies confirm the significant value of the HTC experiments for criticality validation of the primary actinides and the weaker validation basis that exists without the HTC experiments.

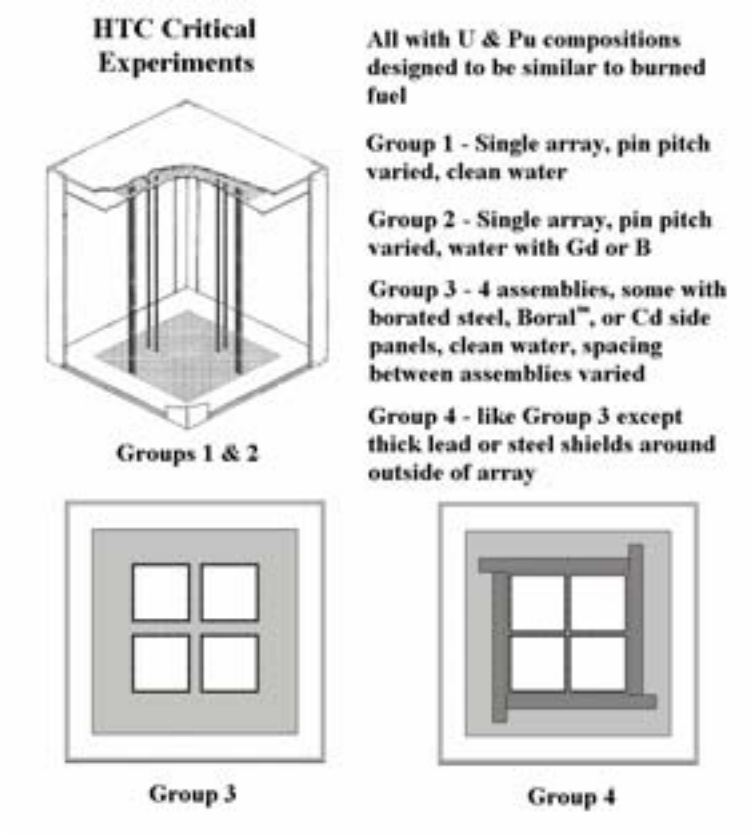

FIG. 2. French HTC critical experiments.

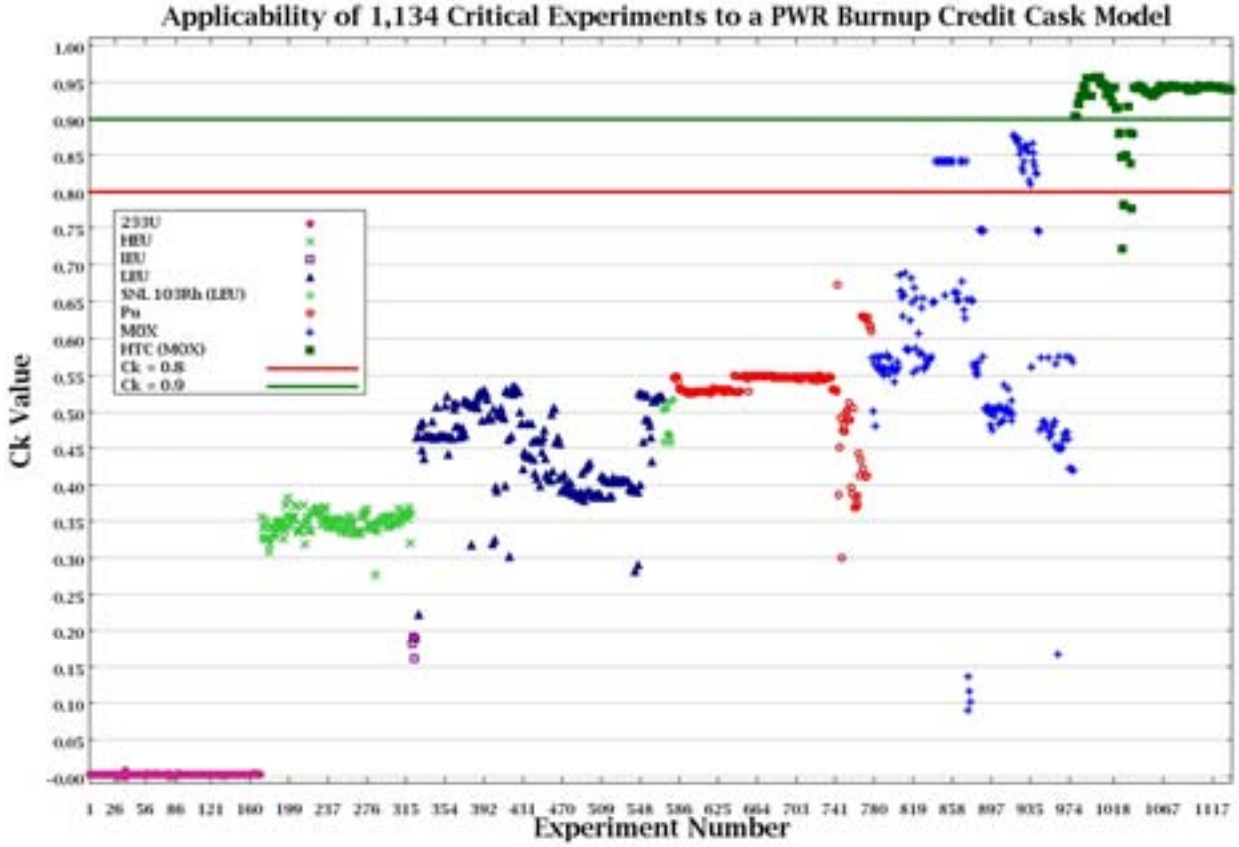

FIG. 3. Critical experiment applicability to burnup credit.

However, the HTC experiments do not provide validation for the fission product compositions in the SNF, and work has been initiated to assess critical experiments that will address this validation need. In 2005, work was performed to assess two sets of critical experiments involving fission products. The first set of experiments was performed in 2003 at Sandia National Laboratories (SNL) as part of a DOE Nuclear Energy Research Initiative (NERI). The set of experiments included thin ^{103}Rh foils stacked between fuel pellets in UO_2 rods placed in a hexagonal array. Under this current project, the final documentation and review of these experiments were completed and published as part of the 2005 release of the IHECSBE data base.

The S/U analyses have been performed for the SNL ^{103}Rh critical experiments, and the results have been compared with S/U analyses results for the GBC-32 cask model. A comparison of the energy-dependent sensitivity profiles shows reasonably good agreement except in the 1- to 2-eV neutron energy range. Studies have been performed to show how a modified experiment design (use of thinner foils) could improve the applicability of the experiments. The S/U tools will be employed in the design process of planned SNL experiments (see Sect. 2.4) to ensure maximum applicability [9].

The second series of experiments being assessed for their value in validation of the fission product burnup credit are the second set of critical experiments that ORNL is seeking to obtain from COGEMA via the contract noted above. ORNL has received preliminary reports that describe 147 critical configurations (referred to as the "PF" experiments), 74 of which contain fission products. The HTC critical experiment MOX rods were used in 29 of the critical configurations, and 14 of these contained fission products. The fission products were present in solution either individually or as mixtures. The first group of experiments uses a central tank filled with water, borated water, or fission product solution. The central tank is surrounded by $U(4.7)O_2$ fuel rods in water. The second group of experiments uses a central tank containing an 11 × 11 array of either $U(4.7)O_2$ or HTC MOX rods in uranyl nitrate solutions with dissolved fission products. The central tank is surrounded by $U(4.7)O_2$ fuel rods in water. The third group of experiments uses a large tank containing an array of either

$U(4.7)O_2$ or HTC MOX rods in depleted uranyl nitrate solutions. Four of the Group 3 experiments with HTC MOX rods also contain fission products. In Group 3, the tank is surrounded by water. Preliminary sensitivity analyses of these French fission product experiments using TSUNAMI-3D and TSUNAMI-IP indicate that only 4 of the 147 critical configurations are sufficiently similar to the GBC-32 cask model to yield c_k values greater than 0.8. These four configurations are nearly identical and yield c_k values of about 0.97. Preliminary observations indicate that the HTC MOX rods dominate the overall c_k comparison between these experiments and the GBC-32 model. Work in progress involves investigation of the sensitivity profiles by nuclide. Using TSUNAMI-IP, the goal of the project is to quantify an uncertainty allowance for the fission products by using the sensitivity profile information for all the criticals and the limited number of applicable critical configurations that have high c_k values.

2.3. Assessment of commercial reactor critical (CRC) configurations

Work currently in progress includes modeling and S/U analyses for more than 60 CRC state points. The initial focus has been on the reactor core configurations and material compositions for 33 Crystal River Unit 3 state points that are documented in great detail in the Yucca Mountain Project (YMP) reports [10–11]. The CRC state points require very large, complex computational models with the following information needed for completeness: fuel assembly locations during reactor cycles and 18-node fuel rod compositions; burnable poison rod assembly (BPRA) core locations and 17-node compositions; rod cluster control assembly (RCCA) and axial power shaping rod assembly (APSRA) core locations, compositions, and insertion heights; and a description of assembly hardware. Figure 4 shows an overhead view of the Crystal River Unit 3 model as generated by the SCALE graphical display package.

Preliminary results for three of the Crystal River CRC state points show $c_k > 0.85$ for CRC cases with effective full-power days ranging from 0 to 515. In addition, comparison of the sensitivity files show reasonable similarity for many of the key fission products. Work is continuing to analyze all of the available CRC state points and assess their utilization in burnup credit criticality evaluations.

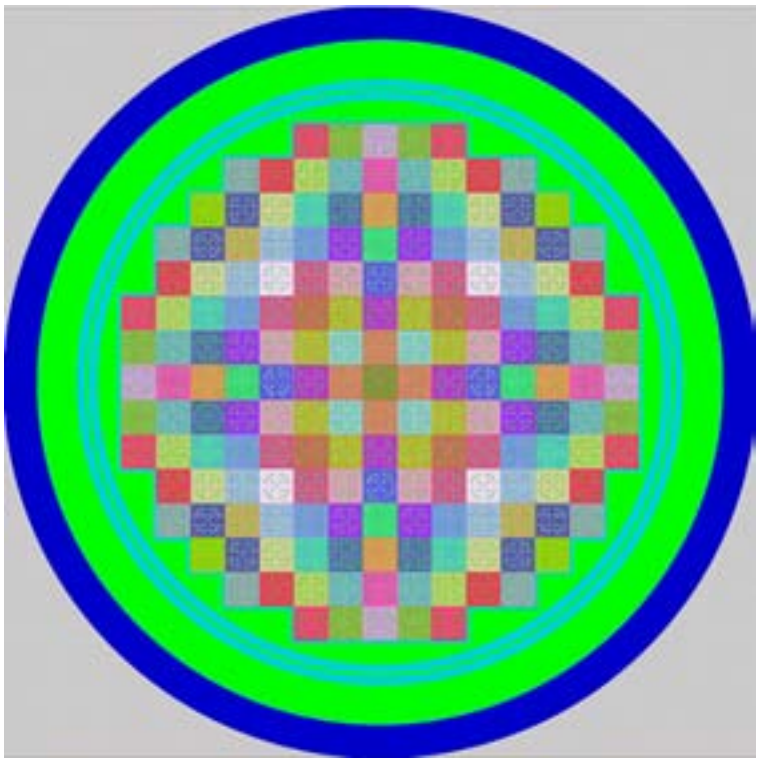

FIG. 4. Commercial Reactor Critical (CRC) model.

2.4. Proposed new critical experiments

This coordinated project is seeking to pursue all existing options to help bring closure to the current technical issues related to burnup credit. To this end, the project is pursuing planning activities to perform additional experiments with the principal fission products. The experiments are to be performed at SNL and would be a follow-on to the critical experiment with ^{103}Rh performed under the DOE/NERI project. The S/U analysis tools, which were not available when the ^{103}Rh critical experiments were designed, will be used in the design of the critical configurations. The goal will be to address any technical needs that may not be adequately addressed with the data obtained from Cogema (e.g. data that might be needed to address burnup credit for BWR SNF). Planning activities were initiated in 2005.

Through an NRC-supported agreement with Belgonucleaire, ORNL will also be able to assess critical experiments performed as part of the REBUS international program using the VENUS critical facility. These experiments involve critical UO_2 pin lattice configurations with portions of commercial BWR and PWR SNF assemblies inserted in the middle of the configuration. Final documentation of the critical experiment should be received by the end of 2005, and ORNL will initiate an evaluation of the experiment in 2006.

3. Data base of isotopic assay data for PWR full burnup credit

3.1. Evaluated assay data for fission products

Just as there are limited benchmark critical experiments that can be used to estimate the bias and uncertainty due to the presence of fission products in SNF cask systems, the existing regulatory guidance of ISG-8R2 indicates there is a definitive lack of measurements that can be applied to estimate the bias and uncertainty in the prediction of the fission product compositions in SNF. Figure 5 illustrates the individual reactivity worth or importance of the major fission products for Westinghouse 17 × 17 SNF loaded in the GBC-32. Regardless of the burnup or decay time, the top six fission products accounting for approximately 75% of the total worth of all fission products are ^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{151}Sm, and ^{155}Gd. These six fission products are the focus of this project's efforts to obtain and assess potential sources of data that can support a strengthened technical basis for fission product credit.

Although radiochemical assay measurements have been reported for a large number of spent fuel samples, most measurements include only the major actinides. Relatively few measurements include the largely stable fission products important to burnup credit (i.e. ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Rh, ^{109}Ag, ^{133}Cs, ^{143}Nd, ^{145}Nd, ^{147}Sm, ^{149}Sm, ^{151}Sm, ^{152}Sm, ^{155}Gd, and ^{153}Eu) [12]. Of the 56 PWR spent fuel samples that had been evaluated by ORNL prior to 2005 [13], only 19 included any of these fission products, and many samples have measurements for only a small number of fission products. No measurements are available for three fission products (^{95}Mo, ^{101}Ru, and ^{109}Ag), and ^{103}Rh had just one measurement [14]. Table I provides a summary of the total number of measurements assessed and accepted by ORNL for each fission product in general order of descending importance. The fission product assay measurements shown in Table I are from just two reactors: the Calvert Cliffs fuels [designated as Approved Testing Materials (ATM)-103, ATM-104, and ATM-106 fuels] measured by Pacific Northwest National Laboratory (PNNL) and the V. G. Klopin Radium Institute (St. Petersburg, Russia) [15] and the Japanese Takahama Unit 3 PWR fuel measurements performed by the Japan Atomic Energy Research Institute [16].

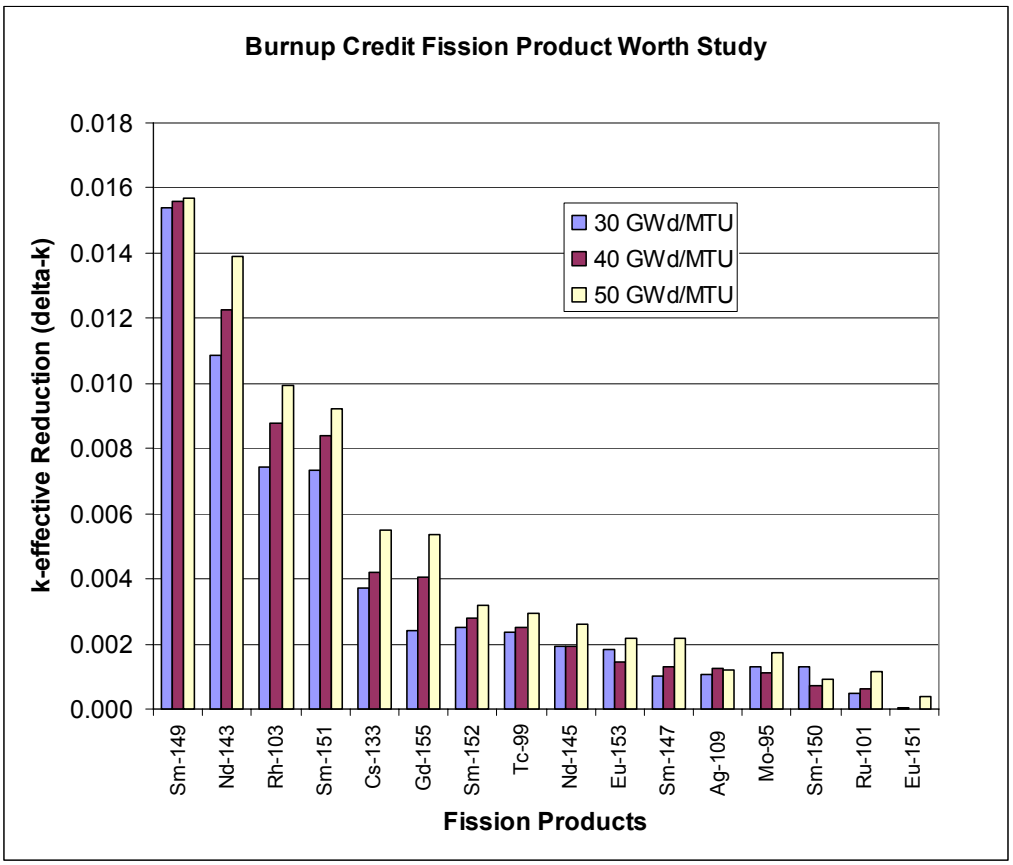

FIG. 5. Fission product worth calculated for WE 17 × 17 SNF assemblies with 4 wt % initial enrichment and loaded in the GBC-32 after a 5-year cooling time.

TABLE I. NUMBER OF MEASUREMENTS AND RELATIVE IMPORTANCE OF FISSION PRODUCTS TO BURNUP CREDIT

(Highest Importance)														(Lower Importance)
^{149}Sm	^{143}Nd	^{103}Rh	^{151}Sm	^{133}Cs	^{155}Gd	^{152}Sm	^{99}Tc	^{145}Nd	^{153}Eu	^{147}Sm	^{109}Ag	^{95}Mo	^{150}Sm	^{101}Ru
9	14	1	9	3	4	9	9	14	4	9	0	0	9	0

In 2005, ORNL performed a thorough review of existing information on measured assay data with the goals of (a) collecting all of the relevant data into a single data base and (b) identifying measurement data that are not currently being utilized. The calculated-to-experiment (C/E) ratio obtained for the measurements noted in Table I was used to investigate the potential improvement (additional negative reactivity that could be credited) that would be obtained with availability of similar quality measurements. Statistically, the uncertainty is best estimated if at least 15 to 20 measured samples are available; the project goal is thus to have this minimum number of measurements available for the validation of the principal fission product nuclides.

3.2. Sources of additional assay data—proprietary

This section describes potential foreign sources of isotopic assay data that ORNL has explored as a means to support code validation for burnup credit using fission products. The sources include existing proprietary programs, currently active programs, and opportunities to perform new measurements.

The Commissariat à l'Energie Atomique (CEA) of France has established experimental programs to provide data for the validation of French computer codes. The programs include spent fuel assay measurements in support of fuel inventory and fuel cycle studies, including burnup credit [8]. e data from these programs are proprietary, but through the contract with Cogema (one of the optional purchases under the contract discussed in Sect. 2), ORNL can obtain and distribute the data for use with burnup credit design and review activities. The available Bugey assay measurements include only two SNF samples of 2.1 wt % and 3.1 wt % enrichment, with burnup less than 38 GWd/MTU. The available Gravelines assay measurements include three SNF samples with initial enrichments of 4.5 wt % and burnup values of 39.1, 51.6, and 61.2 GWd/MTU. All of these samples include measurements for the fission products of interest. If the CEA data are acquired, assay measurements for three BWR SNF samples from the German Gundremmingen reactor would also be provided.

The CEA fission product data are viewed as highly beneficial to strengthening the technical basis to support quantifying fission product uncertainty because of (a) the high-accuracy radiochemical analysis methods employed, (b) the wide range of enrichments and burnups (covering most commercial U.S. fuels), (c) the use of standard commercial fuel assemblies (nonreconstituted), and (d) the fact that the fuel is likely well characterized (because it was selected specifically to support code validation in France). Although not thought to be a significant issue, any differences between the operations of French plants as compared with domestic plants may introduce subtle biases in the measurements that may not be applicable to domestic plants. However, the quantity of CEA fission product assay data is limited to 5 PWR samples, thus leaving the total number of measurements available for many nuclides well below the target value of about 20.

Belgonucleaire is coordinating the international REBUS program to obtain worth measurements for SNF and the MALIBU program to obtain isotopic assay data for high-burnup spent fuel. Through support from NRC and DOE, ORNL is participating in both of these programs, which will provide fission product assay data measured by several independent laboratories using state-of-the-art methods. The REBUS program will provide fission product assay data for one PWR SNF sample, while the MALIBU program will provide fission product assay data for two PWR SNF samples. However, the number of assay samples that are being evaluated is small, and the burnup range is high (> 50 GWd/MTU). The data will be commercial proprietary for a period of 3 years after the final report is issued, expected late in 2005.

3.3. Sources of additional assay data—nonproprietary

In 2005, ORNL contracted with PNNL to investigate and assess whether there are existing, U.S.-origin spent nuclear fuel samples that can be retrieved and made available for expanding the data base of radiochemical assay data for validation of fission product burnup credit. A large percentage of the existing usable fission product assay data was generated by the Material Characterization Center (MCC) at PNNL as part of the ATM program in the late 1980s and early 1990s. ORNL has received a draft report from PNNL identifying available samples. ORNL plans to evaluate the need for performing measurements on some or all of these samples.

A major activity in the last half of 2005 has been work to reassess reported measurements of Three Mile Island Unit 1 (TMI-1) SNF that were performed circa 1999 to support the YMP [17] earlier assessment of the TMI-1 data by ORNL deemed the TMI-1 data were not suitable for use in obtaining the bias and uncertainties for prediction of fission product nuclides. The basic reason for this conclusion was that analyses performed by both ORNL and staff at the YMP [18]ed the C/E results to be highly discrepant compared with the results from the other 56 samples analyzed by ORNL and those reported by the CEA and Belgonucleaire programs. For example, Ref. [19] reports differences of

30–40% between measured and calculated predictions for ^{239}Pu. Reanalysis performed by ORNL in 2005 using state-of-the-art multidimensional reactor physics codes (both SCALE and HELIOS) show discrepancies of 10–20%. This compares with typical calculated-to-measured differences of ±5% for ^{239}Pu. The TMI-1 fuel was originally selected for postirradiation examination because it had experienced extreme crud buildup during irradiation and possible fuel cladding failure of the assembly [19]. The reactor conditions experienced by these fuel samples are not well known. Several suspected local conditions [**Error! Bookmark not defined.**] that could significantly impact the predictions are potentially the reason for the large C/E discrepancies.

Nevertheless, the difficulty in obtaining the quantity and quality of measured assay data for fission product nuclides has led ORNL to revisit the potential usefulness of the TMI-1 data. There are 19 TMI-1 measured samples having a desirable range of initial enrichments (4.0–4.65 wt %) and burnup values (23–55 GWd/MTU). Thus, the TMI-1 samples provide the number of additional measurements recommended for adequate statistical estimation of the uncertainties. The supposition is that a number of samples of "poor" quality (high bias and uncertainty caused by unknown reasons) might be similar to a small number of samples deemed to be of high quality (accurate radiochemical measurements with well-known reactor conditions). Thus, ORNL has recently investigated the distribution of the TMI-1 C/E values and carefully studied the available information on the TMI-1 reactor conditions for this fuel.

The initial recommendation from this reinvestigation, pending further work in 2006, is that the TMI-1 samples are not considered sufficiently qualified for code benchmark purposes (demonstrating that the code and its input data are accurately predicting reality). However, the samples may be useful in supporting a safety basis, provided that the uncertainties are adequately addressed and that use of the data can be demonstrated to yield conservative results. To demonstrate that use of the TMI-1 data provides conservative results requires, at a minimum, a few high-quality measurements from other sources. For fission product nuclides having no previous measurements (e.g. ^{95}Mo, ^{101}Ru), it will be difficult to establish that the TMI-1 results are representative or conservative without having independent data. Also, with any use of the TMI-1 data, it must be recognized that the uncertainties derived from the data may not be representative of modern high-burnup fuel. Ultimately, it should be demonstrated that use of the data does not reduce the margin because of the addition of data that may exhibit abnormal biases. Some additional work in this area is expected prior to final recommendations. The outcome of this work may also influence the effort expended under this project to obtain proprietary data or additional domestic assay data.

4. Nuclear data assessment, measurement, and evaluation

The technical rigor (physics measurements and evaluations to smoothly fit data over the entire energy range) utilized in acquiring current fission product cross-section data is deficient relative to that for major actinides and can impact the uncertainty and credibility of the validation process. This discrepancy in technical rigor has long been a concern (albeit, a secondary concern, if sufficient integral assay and critical measurements with fission products are available) of NRC staff in its consideration of allowing fission product credit. Under this project, ORNL is working to assess the quality of cross-section data (from domestic and international sources) for the key fission product nuclides (i.e. ^{103}Rh, ^{143}Nd, ^{149}Sm, ^{151}Sm, ^{133}Cs, and ^{155}Gd). As needed and justified, new measurements will be performed under a cooperative DOE–Euratom agreement. Work has already been initiated on new measurements and evaluation for ^{103}Rh. Production cross-section libraries will be prepared that are consistent with the quality and rigor now provided in the actinide data.

5. Other activities

5.1. Data for improved safety analyses

ORNL utilized a summer intern to gather and organize operational parameter data from PWR and BWR CRC information to support establishment of more realistic bounding assumptions for use in the safety analyses. Soluble boron concentrations, maximum fuel temperature, and minimum moderator

densities were the initial parameters investigated. Using the range of data values obtained and investigating the mean standard deviations, ORNL is working to provide a technical basis for recommending bounding assumption values that can be used in the safety analysis. A reduction in conservative values recommended in earlier reports is anticipated, and the reduction should allow a larger fraction of spent PWR fuel to be considered as acceptable for transport in fully loaded high-capacity casks. This activity is a continuing effort.

5.2. BWR burnup credit

ORNL has performed analyses that confirm the need for relatively little burnup credit in a high-capacity BWR SNF rail transport cask. In addition, analyses were performed to determine to what extent current high-capacity rail casks, which have a maximum initial enrichment limit of ~4.0 wt %, would need to be de-rated (capacity reduced) to accommodate maximum enrichment (5.0 wt %) BWR assemblies without burnup credit. The analyses suggest that a reduction in capacity of a 68-assembly cask to 64 assemblies will enable loading of 5.0 wt % BWR assemblies without credit for fuel burnup. A simplistic cost savings analysis, based on reduction in the number of shipments, for BWR burnup credit was performed. This cost savings analysis and the work to date on BWR burnup credit will be documented in 2006. Approaches that are simple, but reliable, for using burnup credit to assure full cask loadings of all inventory up to 5 wt % will also be explored.

6. Summary

This report has summarized the current U.S. project on burnup credit and the activities performed to date. The highest-priority data have been obtained (HTC critical experiment set in final form and the PF or fission product critical experiment set in draft form) and are currently being evaluated for applicability to SNF transport and storage casks. The initial results indicate that the HTC data set will provide a strong technical foundation for the actinide portion of burnup credit and enable more flexibility in the criteria by which credit for fission products is considered.

Radiochemical assay data needed for estimating bias and uncertainties in predicted fission product nuclides continue to be a challenge. ORNL has investigated all known sources of assay data and initiated a new effort to reassess and provide guidelines on utilizing the TMI-1 measured data that provide large and atypical C/E values relative to all other known sources of data.

ORNL also has continued to seek a diverse path in assuring that all technical approaches are studied and understood to (a) provide flexibility in future safety analyses and (b) ensure that a solid technical basis consistent with cost and benefit is established. Thus, the CRC data continue to be assessed for applicability to cask systems, efforts to improve the cross-section data for fission product nuclides have been initiated, and activities are ongoing to increase the data base via domestic (e.g. new critical experiments at SNL and assay data measurements at PNNL) or international participation in research programs. By the end of 2006, ORNL is seeking to provide NRC with draft recommendations on implementing fission product credit using the data that have been obtained and to demonstrate where future work (e.g. planned experimental data or an improved reactor operating history data base) might improve implementation of full burnup credit.

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Improved radiochemical assay analyses using TRITON depletion sequences in SCALE

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Abstract. With the release of TRITON in SCALE 5.0, Oak Ridge National Laboratory has made available a rigorous two-dimensional (2D) depletion sequence based on the arbitrary-geometry 2D discrete ordinates transport solver NEWT. TRITON has recently been further enhanced by the addition of depletion sequences that use KENO V.a and KENO-VI for three-dimensional (3D) transport solutions. The Monte Carlo–based depletion sequences add stochastic uncertainty issues to the solution, but also provide a means to perform direct 3D depletion that can capture the effect of leakage near the ends of fuel assemblies. Additionally, improved resonance processing capabilities are available to TRITON using CENTRM. CENTRM provides lattice-weighted cross sections using a continuous energy solution that directly treats the resonance overlap effects that become more important in high-burnup fuel. And beginning with the release of SCALE 5.1 in the summer of 2006, point data and fine-structure multigroup libraries derived from ENDF/B-VI evaluations will be available. The combination of rigorous 2D and 3D capabilities with improved cross section processing capabilities and data will provide a powerful and accurate means for the characterization of spent fuel, making it possible to analyze a broad range of assembly designs and assay data. This in turn will reduce biases and uncertainties associated with the preduction of spent fuel isotopic compositions. This paper describes advanced capabilities of the TRITON sequence for depletion calculations and the results of analyses performed to date for radiochemical assay data.

1. Introduction

Historically, the one-dimensional (1D) SAS2H depletion sequence within the SCALE nuclear analysis suite [1] has provided a simple and rapid approach for spent fuel characterization. However, with the evolution of modern fuel assembly designs, SAS2H is often no longer an appropriate choice for spent fuel characterization. Accurate calculation of the depletion of nuclear materials requires careful determination of the neutron flux density and spectrum in the region(s) of interest. Increasing complexity in reactor designs, evolutionary concepts, and nonreactor applications such as safeguards, security, and nonproliferation require more robust geometrical modeling capabilities than those available in SAS2H in order to properly characterize neutron transport in such complex configurations.

With the release of the TRITON control module in SCALE 5.0, ORNL has made available a rigorous two-dimensional (2D) depletion sequence based on the arbitrary-geometry 2D discrete ordinates transport solver NEWT [2, 3]. NEWT has continued to evolve since this initial release and will be significantly updated with the release of version 5.1 of SCALE in 2006. The update to NEWT will include a completely rewritten geometry-processing package based on the SCALE Generalized Geometry Package (SGGP) used by KENO-VI. Figure 1 illustrates the detailed modeling capabilities available within NEWT to capture the geometric detail of a boiling water reactor (BWR) fuel assembly with a control blade inserted. Also introduced with the 5.1 version of NEWT are a coarse-mesh finite-difference accelerator, pin power calculation capabilities, and an expanded set of lattice physics parameters. Minor code changes have been made to improve the accuracy of the solution and to further enhance computational performance.

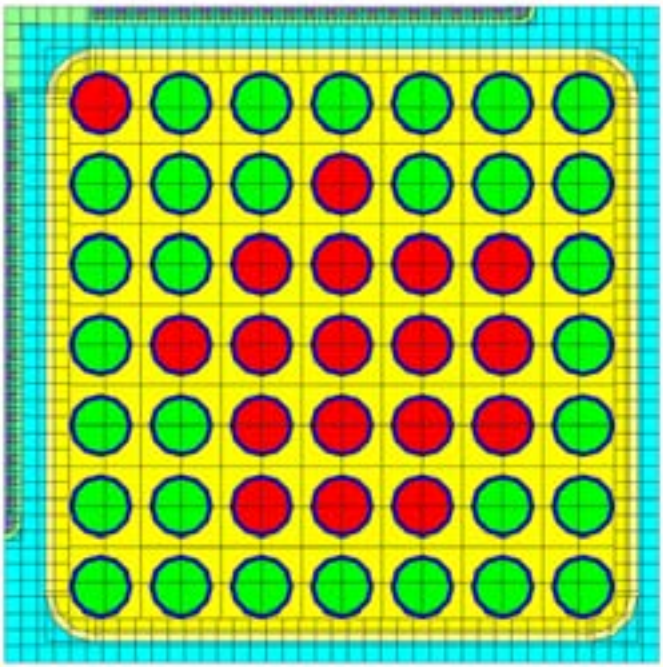

FIG. 1. NEWT model for a BWR design with control blade insertion.

Within TRITON, the T-DEPL sequence is used to perform 2D depletion analysis. This analysis sequence combines cross-section processing via BONAMI/CENTRM (or, optionally, BONAMI/NITAWL), the NEWT transport solution, and COUPLE and ORIGEN-S depletion calculations. In a calculation invoking the T-DEPL sequence, NEWT is used to create a three-group weighted cross-section library based on calculated and volume-averaged fluxes for each mixture. COUPLE is used to update the ORIGEN-S library with cross-section data read from the weighted library. Three-group fluxes calculated by NEWT are supplied to ORIGEN-S for depletion calculations. ORIGEN-S calculations are repeated for each mixture being depleted, as specified in input, using mixture-specific cross-section data and fluxes.

Because spatial fluxes are burnup-dependent, changing with nuclide inventories, and because mixture cross sections will also change with burnup, the T-DEPL sequence uses a predictor-corrector approach to update both fluxes and cross sections as a function of burnup. T-DEPL calculations can be considered to consist of two components during this iterative phase: (1) transport calculations (cross-section processing and the transport solution) and (2) depletion calculations. Transport calculations are used to calculate fluxes and prepare weighted cross sections based on a given set of nuclide concentrations; depletion calculations are used to update nuclide concentrations, which can be used in the following transport calculation.

Other SCALE 5.1 features that are available to TRITON provide additional capabilities. The use of the 1D continuous-energy discrete-ordinates transport module CENTRM within TRITON allows for the preparation of multigroup cross sections weighted with a continuous energy treatment for increased accuracy. TRITON uses ORIGEN-S to perform the depletion/decay calculations; ORIGEN-S underwent significant upgrades in the SCALE 5.0 release, including completely updated nuclear data from ENDF/B-VI, FENDL-2, and EAF-99. Nuclear data were added for hundreds of nuclides that previously were not modeled in any version of ORIGEN. The fission product yield data were increased from 5 fissile nuclides to 30. The methods in ORIGEN-S have also been upgraded to support nontraditional systems. (The more widely recognized ORIGEN2 code has not been updated in more than 10 years and is no longer supported at ORNL.)

Despite the broad applicability of the 2D fuel depletion analysis capability of TRITON, there are some domains in which accurate three-dimensional (3D) depletion capabilities are necessary. For example, criticality analysis for commercial spent fuel in transportation and storage is concerned with the positive reactivity effects of low-burnup fuel near the ends of a fuel assembly where axial leakage effects (not captured by 2D methods), may be important. Deterministic transport methods are also unable to perform full-core analysis in a practical sense because of the computational overhead of such large-scale discretization. Additionally, conceptual advanced reactor designs, such as designs for space reactors, Generation IV commercial power reactors or research reactors and other small cores, depart from traditional design attributes so that more robust 3D methods may be required to track fuel depletion or provide reference solutions for 2D methods. For these reasons, among others, a 3D depletion capability has been integrated into TRITON, using the 3D Monte Carlo–based KENO V.a and KENO-VI functional modules of SCALE [4,5]. These options are available within the T5-DEPL (KENO V.a) and T6-DEPL (KENO-VI) sequences of TRITON.

Because of the modular nature of SCALE, the process for replacing the deterministic 2D NEWT transport solution with 3D KENO solutions was relatively straightforward. However, certain functions available within NEWT (e.g., calculation of averaged three-group fluxes and fission/capture power calculations) were not readily available within either KENO module. Rather than modify KENO, the KENO postprocessing codes KMART and KMART6 (for KENO V.a and KENO-VI, respectively) have been adapted to provide collapsed cross sections and fluxes required by TRITON for setting up ORIGEN-S depletion calculations. Additionally, the restart capabilities of the KENO codes have been used to provide an improved starting source for each depletion step, further improving calculation times by reducing the number of calculations required to obtain source convergence.

Beyond these changes, however, the logical flow through TRITON in the KENO-based sequences mirrors that of the NEWT-based T-DEPL sequence. Figure 2 illustrates the computational flow through TRITON for both of the 3D depletion sequences. Because all cross-section processing and depletion processes are identical between each of the three depletion sequences, a direct comparison of results is possible, with differences attributable solely to differences in the transport solution.

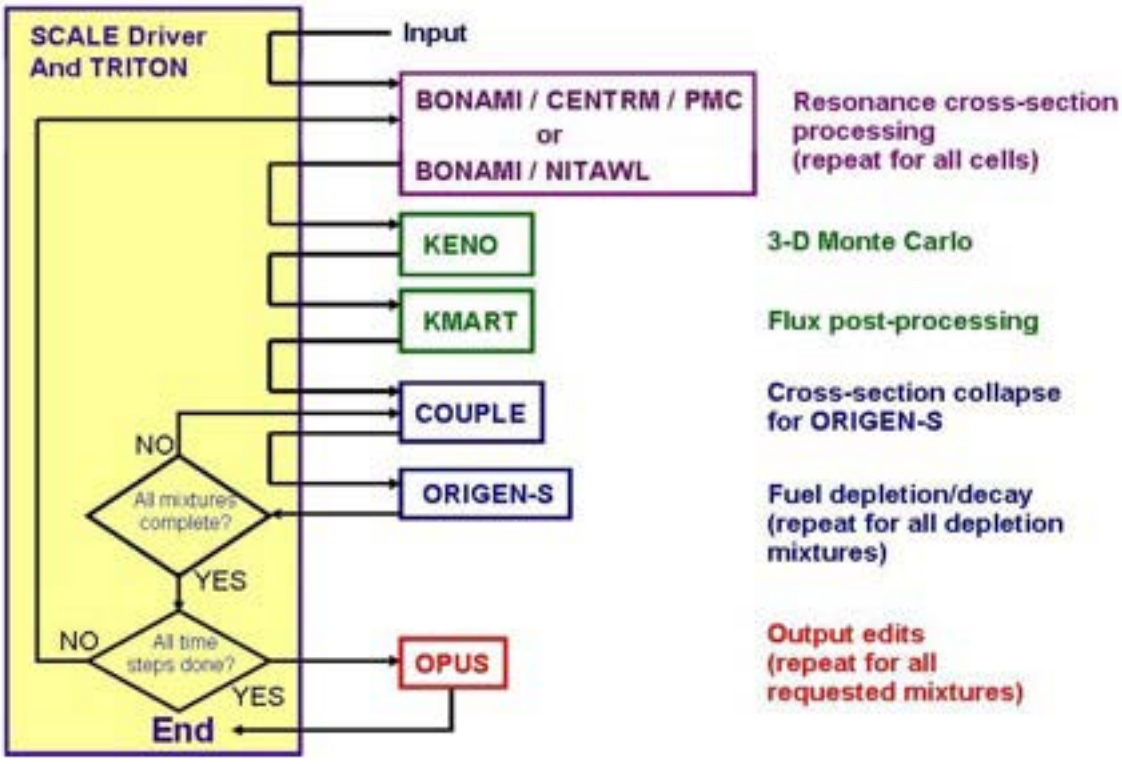

FIG. 2. TRITON sequence for KENO-based depletion.

2. Issues in Monte Carlo depletion

Use of Monte Carlo methods for depletion analyses introduces new challenges that should be addressed [6]. The Monte Carlo transport solution introduces stochastic uncertainty in fluxes. Because these fluxes are used to collapse cross sections, to estimate power distributions, and to deplete the fuel within ORIGEN-S, the predicted number densities contain random uncertainties due to the Monte Carlo solution. Depletion and decay calculations are by their nature extrapolations, so errors can be compounded with time.

Flux errors may be minimized by using very large numbers of neutron histories. Flux errors will be smallest in most reactive regions, where the greatest sampling occurs, but larger in the lower flux regions. Variance reduction will be important to force significant neutrons out to all regions of interest. Propagation of uncertainties from cross sections to isotopic concentrations will help in assessing the effect of potentially large flux variances. Both these issues will be addressed in future research at ORNL.

Nevertheless, stochastic limitations are offset by the ability to apply the power of Monte Carlo methods for complex 3D geometries. Furthermore, because of the particle tracking method of KENO V.a, it can perform extremely fast transport calculations relative to other Monte Carlo codes such as MCNP or KENO-VI. KENO-VI, on the other hand, provides complete flexibility in model development due to its combinatorial geometry input specification. Finally, because this methodology is built on the existing T-DEPL methodology in SCALE, direct benchmark comparisons can be made between the NEWT and KENO versions of TRITON for validation. The following section describes the results of such analyses.

3. Validation

Benchmark calculations have been performed using pressurized water reactor (PWR) fuel assembly data provided in validation reports of the 1D SAS2H depletion sequence in SCALE [7-9]. Benchmark models of the spent fuel assemblies have been developed with both T5-DEPL and T6-DEPL. Calculated results have been compared with the measured radiochemical spent fuel assay data given in the reports and with previously calculated SAS2H and T-DEPL results. Benchmark calculations have been performed for a wide variety of fuel assemblies; additional validation work is ongoing at ORNL. This paper provides results obtained from four PWRs:

- Calvert Cliffs
- Obrigheim
- San Onofre
- Trino Vercelles

3.1. Calvert Cliffs 14 × 14 fuel

The Calvert Cliffs fuel assembly is a Combustion Engineering (CE) 14 × 14 fuel assembly design. The fuel assembly modeled was D047. The specific location in the assembly of the measured sample was rod MKP109 at an elevation of 165.22 cm with a burnup of 44.34 GWd/MTU [7]. Measured data were obtained for the major actinides, cesium isotopes, and other fission products of importance to burnup credit (i.e., strong neutron absorbers). A comparison of the calculated results from SAS2H, T-DEPL, T5-DEPL, and T6-DEPL with measured data are presented in Table I and Figs. 3 (actinides) and 4 (fission products).

These results demonstrate consistency between the 1D, 2D, and 3D SCALE depletion sequences. The comparisons with the measured data show errors of generally 10% or less for the actinides and most fission products. Results for six of the fission products deviate from the measured data by approximately 20%. More importantly, results are generally consistent among the various codes.

Differences are seen between the 1D and multidimensional results for those nuclides that are most sensitive to the thermal spectrum (i.e. ^{235}U and Pu isotopes), indicating the possibility of inadequate characterization of the thermal spectrum in the 1D model.

TABLE I. CALVERT CLIFFS Fuel Assembly D047, Rod MKP109 (44.34 GWd/MTU)

Nuclide	Measured (g/gUO2)	SAS2H %Diff.	T-DEPL (NEWT) %Diff.	T5-DEPL (KENO V.a) %Diff.	T6-DEPL (KENO-VI) %Diff.
U-234	1.20E–04	1.40	1.14	1.16	1.14
U-235	3.54E–03	–8.70	–5.05	–5.44	–5.40
U-236	3.69E–03	1.90	–1.81	1.81	1.80
U-238	8.25E–01	–0.10	–0.16	–0.16	–0.18
Pu-238	2.69E–04	–5.00	–6.63	–6.56	–6.59
Pu-239	4.36E–03	–1.50	6.26	4.96	5.00
Pu-240	2.54E–03	–3.90	–0.17	–0.70	–0.96
Pu-241	1.02E–03	–2.40	–0.71	–1.30	–1.14
Pu-242	8.40E–04	4.10	–0.90	–0.65	–0.49
Np-237	4.68E–04	7.20	7.25	7.13	7.25
Cs-133	1.24E–03	3.40	3.47	3.47	3.46
Cs-134	3.00E–05	–18.60	–19.45	–19.45	–19.43
Cs-135	4.30E–04	1.70	3.42	3.20	3.22
Cs-137	1.25E–03	1.20	–0.40	–0.40	–0.41
Nd-143	7.63E–04	0.50	1.63	1.47	1.48
Nd-144	1.64E–03	0.20	–0.07	0.03	0.02
Nd-145	7.44E–04	–0.60	–0.39	–0.34	–0.31
Nd-146	8.30E–04	1.30	1.74	1.74	1.72
Nd-148	4.28E–04	0.30	0.60	0.64	0.61
Nd-150	2.08E–04	4.20	4.71	4.71	4.68
Pm-147 + Sm-147	2.68E–04	–4.80	–5.88	–5.80	–5.76
Sm-148	2.22E–04	–18.20	–17.98	–17.98	–17.99
Sm-149	4.70E–06	–49.10	–51.18	–51.55	–51.39
Sm-150	3.61E–04	–5.60	–6.03	–6.03	–5.99
Sm-151 + Eu-151	9.78E–06	38.50	N/A	35.18	34.62
Sm-152	1.21E–04	22.00	20.99	20.68	20.89
Eu-153	1.48E–04	2.50	0.54	0.79	0.75
Sm-154 + Eu-154 + Gd-154	8.42E–05	–3.40	–3.73	–3.94	–4.01
Eu-155 + Gd-155	9.82E–06	–25.30	–23.96	–24.16	–24.17

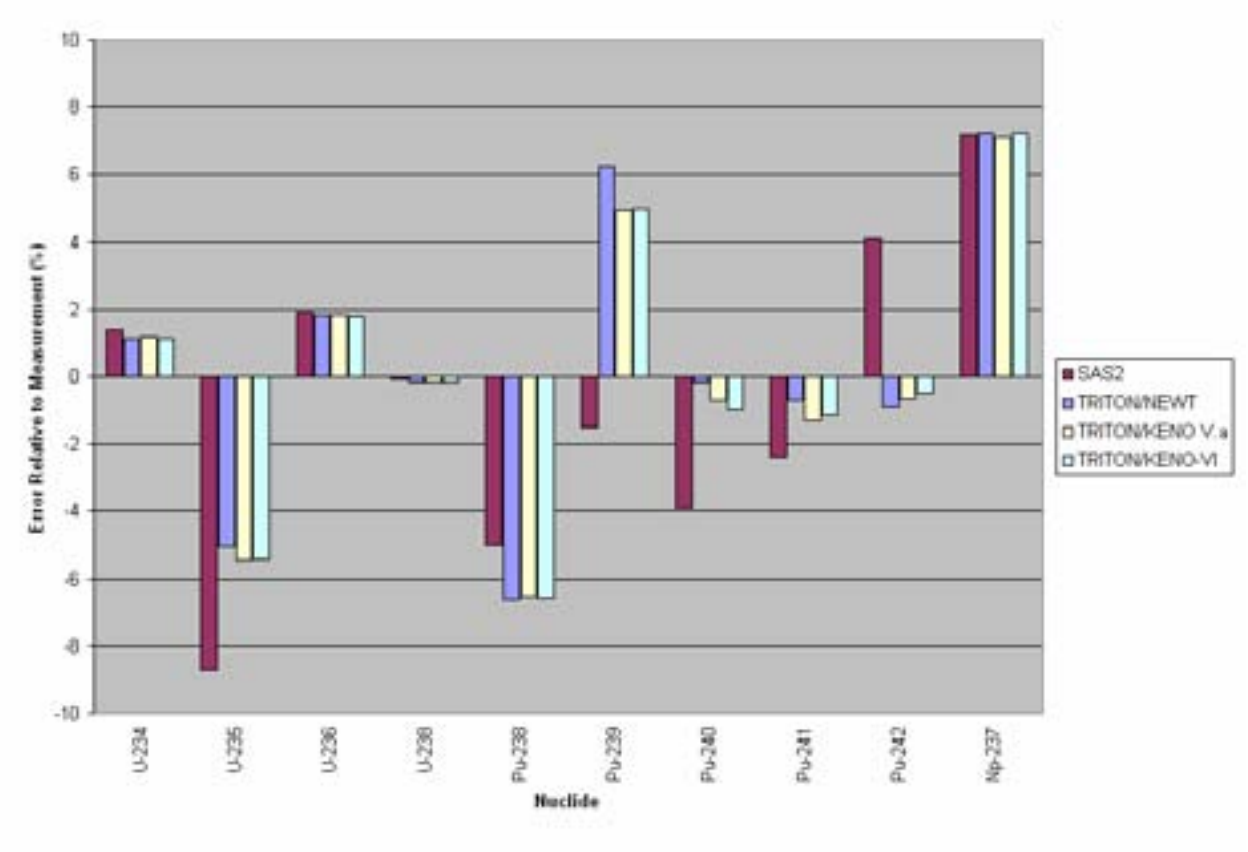

FIG. 3. Calvert Cliffs calculated results vs measured data for actinides.

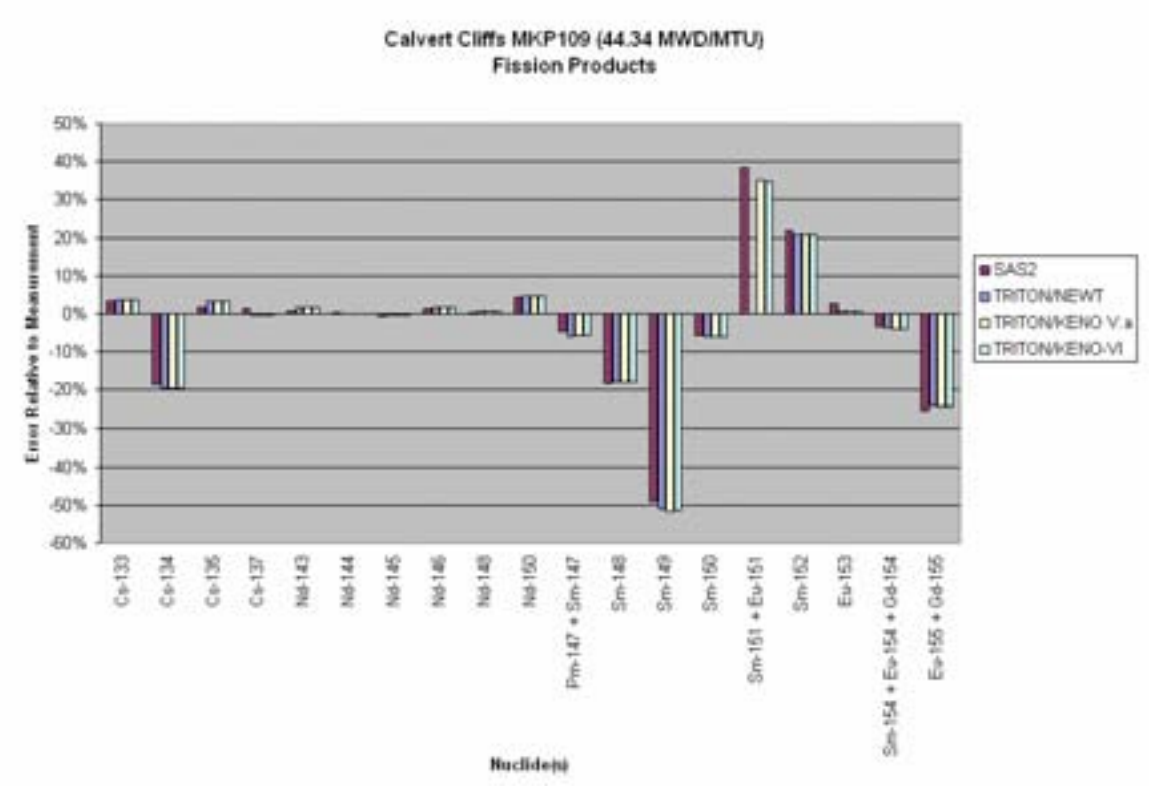

FIG. 4. Calvert Cliffs calculated results vs measured data for fission products.

3.2. Obrigheim

Isotopic measurements of the Obrigheim German PWR 14 × 14 assemblies were performed in Europe. For these measurements, each assembly was cut in half lengthwise and dissolved. The radiochemical analysis for a number of actinide and fission products was subsequently carried out by four independent institutes. The Obrigheim measurements thus provide "assembly average" isotopic values that, in comparison with individual pellet measurements, are more consistent with the spatially independent (i.e. assembly average) point-depletion techniques typically used to characterize spent fuel for away-from-reactor applications.

The assembly modeled in this study was assembly 176, batch 90, with an enrichment of 3.1 wt % and a burnup of 29.52 GWd/MTU [7]. The comparison of results in Table II shows good agreement between measurements and calculations, except for ^{242}Cm, one of the lesser actinides for burnup credit applications. Because of the isotopic homogenization of this assembly, the homogenization approximation applied by SAS2H yields exceptionally good results relative to multidimensional methods.

TABLE II. OBRIGHEIM FUEL ASSEMBLY 176 (29.52 GWd/MTU)

Nuclide	Measured (mg/gU)	SAS2H %Diff.	T-DEPL (NEWT) %Diff.	T5-DEPL (KENO V.a) %Diff.	T6-DEPL (KENO-VI) %Diff.
U-235	9,180.00	-2.0	-2.0	-0.2	-0.3
U-236	3,810.00	1.2	0.7	0.2	0.3
Pu-238	107.1	3.0	-2.2	-2.6	-2.5
Pu-239	4,943.00	< 0.1	-0.2	1.1	1.1
Pu-240	2,040.00	-0.1	1.1	1.0	0.9
Pu-241	1,128.00	0.5	-0.9	-0.4	-0.4
Pu-242	438	-4.7	-8.1	-9.3	-9.2
Cm-242	21.8	-23.1	-27.1	-27.3	-27.2
Cm-244	19.2	-9.1	-9.9	-11.6	-11.2

3.3. San Onofre mixed oxide (MOX) fuel

The EEI-Westinghouse Plutonium Recycle Demonstration Program—sponsored by Edison Electric Institute, Westinghouse Electric Corporation, and the Atomic Energy Commission—was conducted between 1968 and 1974. A significant part of the program involved the measurement of isotopic compositions of uranium, plutonium, and a few other actinides in irradiated MOX fuel from the San Onofre PWR Unit 1, a reactor with a Westinghouse design and operated by Southern California Edison and San Diego Gas & Electric companies. Four MOX fuel assemblies were loaded at the start of Cycle 2 at the San Onofre Nuclear Generation Station Unit 1 and irradiated during both Cycles 2 and 3. Isotopic composition analyses were conducted by Westinghouse Electric Corporation on six sample pellets from four fuel rods of the MOX test assembly D51X. The measured actinide inventories have been used to benchmark the use of SAS2H depletion calculations for MOX fuel [8].

As part of the current validation, the sample pellet from pin 079 at an elevation of 49 in. with a burnup of 20.89 GWd/MTU was modeled. Comparisons of the calculated results from SAS2, T-DEPL, T5-DEPL, and T6-DEPL with the measured data are presented in Table III. Once again, the calculated results are consistent and generally agree well with the measured data. The two nuclides with poor results, ^{234}U and ^{238}Pu, have relatively low concentrations and importance.

TABLE III. SAN ONOFRE MOX FUEL ASSEMBLY DX51, PIN 079 (20.89 GWd/MTU)

Nuclide	Measured	SAS2H %Diff.	T-DEPL (NEWT) %Diff.	T5-DEPL (KENO V.a) %Diff.	T6-DEPL (KENO-VI) %Diff.
U-234	4.66E-02	-13.1	-13.4	-13.7	-13.5
U-235	4.40E+00	-2.0	0.8	1.1	1.0
U-236	4.89E-01	6.6	2.6	2.6	2.4
U-238	9.43E+02	<0.01%	<0.01%	<0.01%	<0.01%
Pu-238	2.82E-01	-36.3	-35.4	-34.8	-35.1
Pu-239	1.65E+01	5.2	3.0	4.0	3.8
Pu-240	7.68E+00	-3.3	2.6	1.9	2.3
Pu-241	3.66E+00	1.5	0.1	1.6	1.1
Pu-242	8.97E-01	5.9	3.6	3.2	3.4
Nd-148	2.27E-01	0.1	0.6	0.6	0.6

3.4. Trino Vercelles

Trino Vercelles is a 825-MW Westinghouse PWR in Italy. The reactor is based on one of the earlier Westinghouse designs and is unlike most PWR designs in the United States, but similar to that of the Yankee Rowe PWR. Use of this uncommon design will serve to demonstrate the modeling capabilities of KENO for non-uniform fuel assembly designs. The fuel assembly design is based on a 15 × 15 lattice of fuel pins with 16 of the outer pins excluded to accommodate cruciform positions, as illustrated in the 2D plot of the KENO V.a model in Fig. 5.

Radiochemical assay data obtained from assembly 509-069, irradiated during both the first and second fuel cycles, were used for benchmarking in this validation [9]. Comparisons of the calculated results from the four depletion options of SCALE with the measured data are presented in Table IV. Once again, the calculated results are consistent and, except for ^{134}Cs and ^{154}Eu, generally agree well with the measured data. The Cs-134 error is known to derive from weaknesses in ENDF/B data.

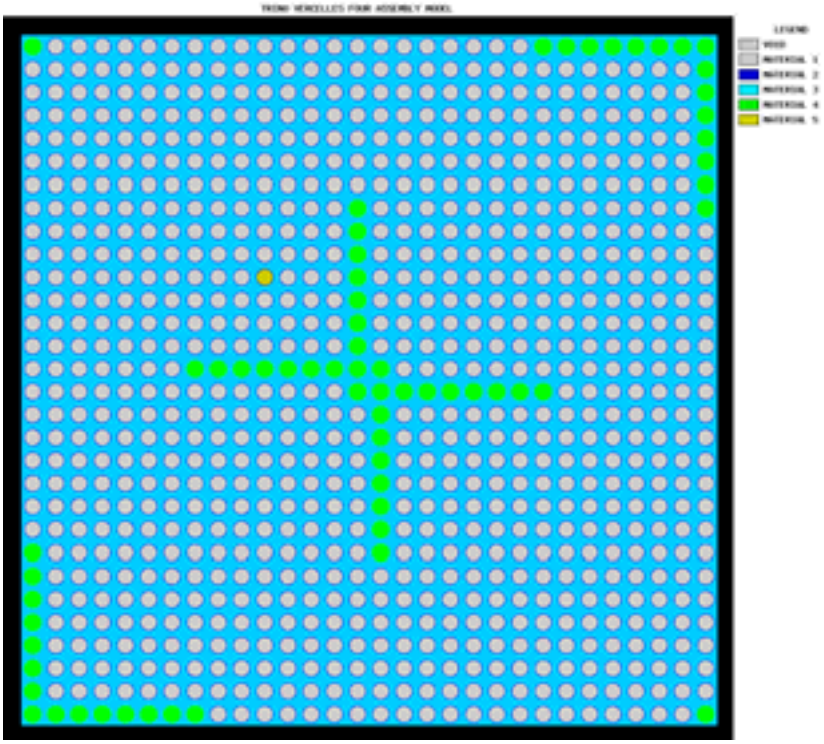

FIG. 5. 2-D plot of KENO V.a model for Trino Vercelles assembly 509-069.

TABLE IV. TRINO VERCELLES FUEL ASSEMBLY 509-069, ROD E11 (12.859 GWd/MTU)

Nuclide	Measured (mg/g U)	SAS2H %Diff.	T-DEPL (NEWT) %Diff.	T5-DEPL (KENO V.a) %Diff.	T6-DEPL (KENO-VI) %Diff.
U-235	1.95E+01	0.51	0.46	0.49	0.53
U-236	2.45E+00	-5.75	-6.77	-6.85	-6.87
U-238	9.59E+02	-0.05	-0.01	-0.01	-0.05
Pu-239	4.58E+00	-1.51	-1.60	-1.69	-1.85
Pu-240	8.40E-01	8.20	8.90	8.28	8.97
Pu-241	4.00E-01	3.62	-3.14	-2.84	-3.51
Pu-242	4.60E-02	10.26	1.53	1.62	1.45
	(Curies/g U)				
Cs-134	2.49E-02	-25.94	-27.87	-27.90	-27.97
Cs-137	3.94E-02	0.71	-0.88	-0.88	-0.86
Eu-154	1.37E-03	-25.62	-23.33	-23.63	-23.48

4. Conclusions

The updated TRITON depletion sequences using NEWT and the 3D Monte Carlo codes KENO V.a and KENO-VI show tremendous potential for application in 3D configurations. Performance of both Monte Carlo depletion sequences has been assessed by comparison with 1D and 2D results obtained using deterministic transport methods and by direct comparison to measured spent fuel data. Results show excellent agreement with other codes and data. These calculation sequences provide simple and straightforward analysis capabilities for a wide variety of applications. Planned future work includes implementation of variance reduction techniques to improve computational efficiency and statistical uncertainty propagation from the Monte Carlo calculations to the predicted isotopic concentrations.

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Integrated depletion code MVP-ORBURN
Development, validation and application study to the burnup credit evaluation

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Abstract. JNES has been developing an integrated depletion code, named MVP-ORBURN, by means of combining the continuous energy Monte Carlo code MVP[(1)] and point depletion code ORIGEN2[(2)]. The present effort focuses the improvement of prediction accuracies of the nuclide composition of a spent LWR fuel and the evaluation of the burnup credit on criticality safety evaluation for the safety regulation. A detailed modeling of heterogeneous structure of the fuel assemblies, including control rods, gadolinium fuels etc. is achieved, and thereby stable and reliable techniques and latest nuclear data libraries for the depletion analysis are established.

The accuracy of MVP-ORBURN for the nuclide composition analysis of spent fuel was validated against typical post irradiation data. For the PWR fuel, the MVP-ORBURN analysis showed sufficient accuracy, i.e. the differences of the calculated values to the experimentally measured values (C/E ratios) were less than 10% for the major actinide concentrations. On the other hand, for the BWR fuel it was suggested that further developments were required, because differences of the C/Es for the few actinides exceeded over 15%. Therefore new calculation schemes to include the axially distributed void fractions and the sophisticated allocations of burnable poison gadolinium were introduced, and the C/E differences were reduced to nearly 10%.

Application studies were also made for the analysis of spent fuel composition and the evaluation of burnup credit with a transport cask containing spent fuels, mainly for the uncertainties and their causes. It revealed that the PWR fuel composition predicted by ORIGEN-2 was conservative especially for low burnup conditions compared with the realistic composition predicted by MVP-ORBURN. It is also confirmed that the uncertainty associated with defining acceptable ranges of burnup that must be specified an operation condition for the facility were another important causes of uncertainties.

1. Outline

To establish an accurate and reliable analytical tool for predicting the composition of spent LWR fuel for the criticality safety evaluation incorporating burnup credit (hereinafter called BUC), JNES has been developing the MVP-ORBURN code (hereinafter called MVP-ORBURN) by combining the MVP and the ORIGEN2. Using these reliable codes and the latest nuclear data, the depletion analyses with a detailed modeling of the heterogeneous structure of fuel assembly, including control rods and gadolinium fuels, have been made.

The accuracy of the fuel composition evaluation with MVP-ORBURN was studied through the validation effort against post irradiation examination (PIE) data, and the results showed that the PWR fuels have the C/E ratios of nearly 10% or lower for major nuclides, while the BWR fuel case may exceed uncertainties over 15% for some major nuclides. The BWR fuel case suggests that improvement is required in the calculation accuracy. We have improved mainly the axial calculation function to take the void distribution and the gadolinium distribution into account, and have achieved the accuracy level of about 10% or lower, even for the BWR fuels.

Taking spent fuel transportation casks as examples, a study on the nuclide composition evaluation using MVP-ORBURN and ORIGEN was also made for examining the influence of the differences in nuclide composition and various kinds of uncertainties caused by application of the burnup credit. As

a result, it was confirmed that the evaluation of nuclide composition with ORIGEN produced conservative results, particularly under the low burnup conditions. The setting of acceptable burnup specified for facility and equipment had relatively great influence on the margin obtained by applying burnup credit.

2. Development of MVP-ORBURN

2.1.Outline of MVP-ORBURN

MVP-ORBURN is composed of the MVP module, the ORIGEN module, the ART and other preprocessors for temperature interpolation of the cross section for MVP calculation, and the driver modules. All of these modules are installed on a PC (OS: Windows XP).

MVP-ORBURN calculations consist of the repetition of a single depletion calculation (in the three steps, (1) to (3), described below) for each of the fuel assembly shapes and the initial preparation to receive input from the LATTICE preprocessor, as shown in Figure 1.

(1) The MVP module calculates the neutron flux distribution for a fuel assembly of three-dimensional XYZ, and delivers the resultant neutron reaction cross section and neutron flux of each fuel rod to the driver modules.

(2) The driver modules determine the burnup for the fuel assembly, normalize it using the input burnup and depletion period, modify the neutron reaction cross section library to determine the specific power of each fuel rod, and deliver it to the ORIGEN module.

(3) The ORIGEN module carries out the depletion calculation for each fuel rod, and delivers the resultant nuclide composition to the MVP module via the driver modules.

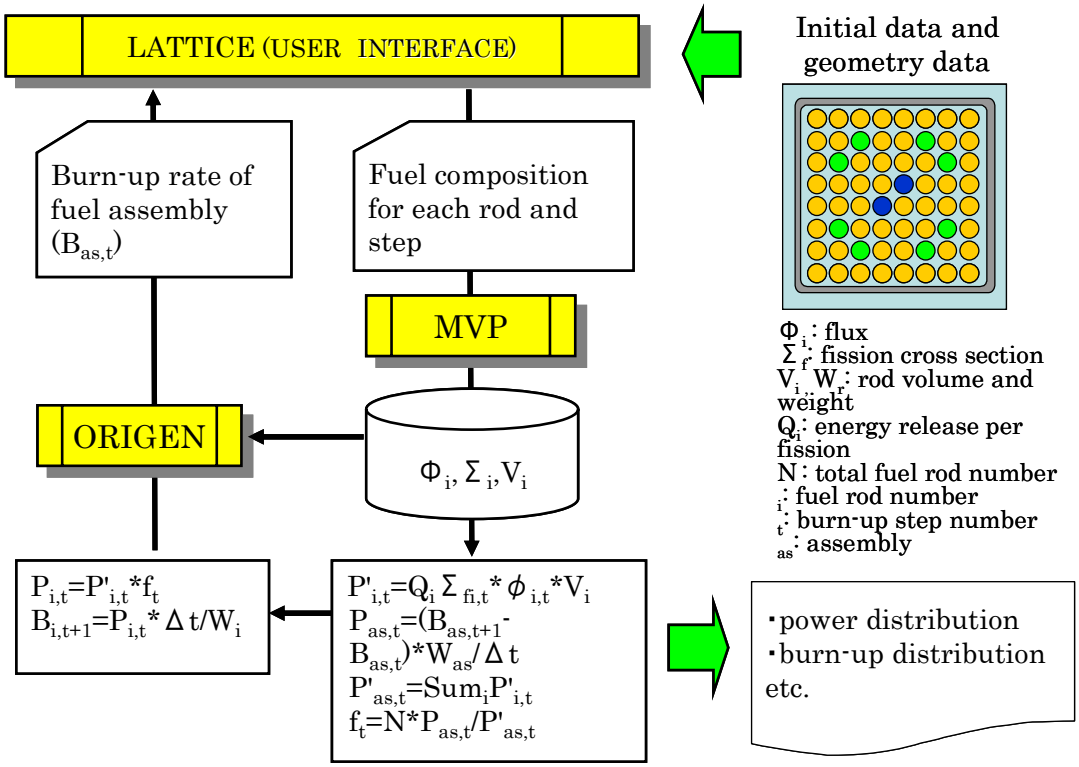

FIG.1. Calculation flow of MVP-ORBURN.

2.2.Improvement of MVP-ORBURN

2.2.1. Sensitivity analysis

The previous validation analysis against PIE data by MVP-ORBURN revealed that the PWR fuels had the C/E ratios of nearly 10% or lower for main nuclides, while the BWR fuels showed uncertainties of 15% or higher for some major nuclides, such as plutonium. In addition, MVP-ORBURN had a disadvantage of requiring a long time to perform a series of depletion calculations due to use of the Monte Carlo calculation. To solve this problem, we tried to improve MVP-ORBURN through various kinds of examinations. Main topics in the improvement process are described below.

(1) Analysis of forced increase and decrease in depletion neutron flux

To examine the sensitivity of MVP-ORBURN in the nuclide composition calculations, depletion analysis for PWR fuels that have representative burnup was made with different depletion neutron fluxes, changed by ±5% and ±20%, and the calculated nuclide compositions were compared with the measured PIE compositions.

The residual weight of the major nuclides at depletion of nearly 37 GWd/t changed almost monotonously with regard to the depletion neutron flux level. The resultant accuracy showed minimal values on the whole when the depletion neutron was increased by about 5%, as shown in Figure 2.

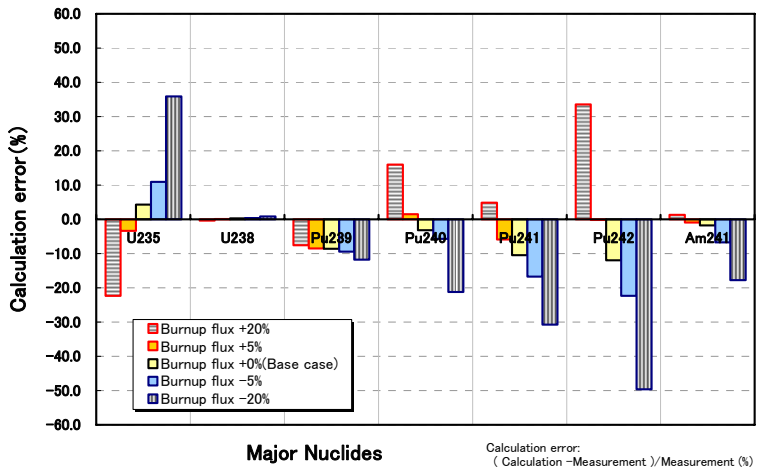

FIG.2. Influence of deploetaion flux level (PWR Fuel SF95-4).

(2) Analysis of changing the power conversion factor of MVP

There are two kinds of energies generated in a reactor. One is the energy directly generated by nuclear fission, such as prompt gamma rays. The other is delayed energy generated when neutrons are captured by structural materials, such as delayed gamma rays. Usual depletion calculation code uses the power conversion factor (hereinafter called the Q value), which is a sum of the effective nuclear fission energy (Q_{eff}) given in nuclear data libraries and the energy generated when neutrons are captured by structural materials (Q_c).

The burnup of depleted fuels is based on reactor core management data and these data are evaluated by the all thermal energy generated in a reactor. Therefore it is reasonable to use the Q value defined by $Q_{eff} + Q_c$ in usual depletion calculations. However, the measured results of mass spectrometric analysis, indicated in the PIE data, are given by the depletion rate (% FIMA), and a conversion from the depletion rate to the burnup (GWd/t) is required. The data shown as measured values contain several percentage of uncertainties as described in the ANSI standard [3]. This is because: (i) the fixed value specified in the ANSI standard is used for the conversion, but there are differences in Q_{eff} values for each nuclide between ENDF/B-VI in the United States and JENDL-3.3 in Japan (as shown in

Table 1), (ii) there is some variation due to differences in nuclide composition, and (iii) the Q_c value varies depending on the positions in fuel assembly.

Table 1. Power conversion factors (Q value) for each nuclide in MVP-ORBURN

Nuclide	MVP-ORBURN Old value	Q value on JENDL-3.3	Q value on ENDF/B-VI
U-233	-	190.850	
U-234	-	189.999	
U-235	202.18	193.580	193.720
U-236	-	189.999	
U-238	205.87	195.160	198.060
Np-237	-	200.000	
Np-239	-	189.999	
Pu-249	210.91	199.620	199.920
Pu-240	210.96	198.000	199.470
Pu-241	213.22	201.650	201.980
Pu-242	241.62	201.360	201.580
Am-241	-	200.000	
Am-242	-	200.000	
Am-242m	-	200.000	
Am-243	-	200.000	
Am-244	-	200.000	
Am-244m	-	199.999	
Cm-242	-	200.000	
Cm-243	-	200.000	
Cm-244	-	200.000	

MVP-ORBURN determines the depletion neutron flux level using these Q values while adjusting the average burnup of fuel assembly to the reactor core management data. However, as described above, the reactor core management data is not always consistent with the PIE data for control rod allocations. Consequently, there is a possibility that the neutron flux level in MVP-ORBURN is underestimated.

Based on the sensitivity analysis using the Q value as a parameter, it has been indicated that there is a tendency of increasing the neutron flux and accelerating the depletion when only the Q_{eff} value given in the JENDL-3.3 library is used. And, this tendency agrees with that the increase of around +5% in the depletion neutron flux level as described above in item (1) improves the accuracies for all the major nuclide compositions.

2.2.2. Improvement

The basic functions of MVP-ORBURN were improved as follows.

(a) The input module was modified so that the power conversion factor, Q value, can be changed in any of the three ways, setting the Q value on the basis of JENDL-3.3, the Q value with the capture gamma ray energy, and any Q value for each nuclide.

(b) An automatic determination procedure whether or not to perform the MVP calculation was provided, in addition to the ordinary calculation with specified depletion steps. A simplified calculation for a single fuel pin is executed, and the MVP calculation is executed only when the sum of macroscopic fission cross section is larger than the value in the preceding MVP calculation step by a specified factor.

(c) Flexibility was given so that the MVP calculation and the cross section changes are executed only for specified nuclides.

(d) The axial direction was divided into 3 to 5 areas to the specified dimensions to allow depletion calculations for the low burnup, including the parts near the end of fuel. It was also enabled to adjust the axial void fraction, the fuel enrichment, the BP distribution and the ORIGEN cross section.

2.3.Analysis of PIE data using MVP-ORBURN

2.3.1. Calculation conditions

The PIE data was obtained in the positions for the fuel arrangement as shown below. And these analysis data for PWR and BWR fuels are shown in Table 2.

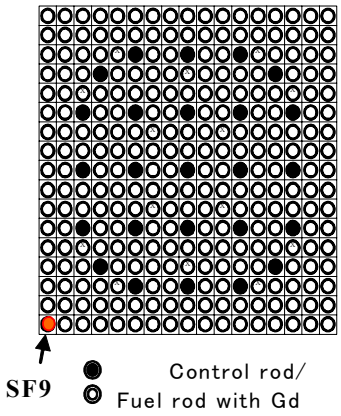

FIG. 3. PWR fuel assembly.

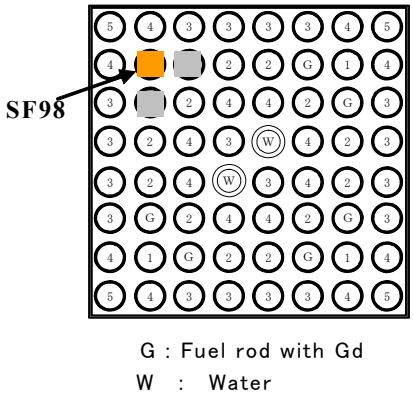

FIG. 4. BWR fuel assembly.

Table 2 Fuel Specifications in the JAERI Post Irradiation Examination

Takahama Nuclear Power plant-3 (PWR : 17 X 17)

Parameter	Value
Fuel rod pitch (mm)	about 12.6
Outer diameter of pelet (mm)	about 8.05
Outer diameter of fuel rod (mm)	about 9.5
Clad thickness (mm)	about 0.64
Fuel temperature (k)	900
Clad temperature (k)	600
Moderator temperature (k)	600
Assembly pitch (mm)	about 214

SF95	Measured point				
	SF95-1	SF95-2	SF95-3	SF95-4	SF95-5
Burnup (GWd/t)	14.30	24.35	35.42	36.69	30.40
Axial position (From bottom of active fuel, mm)	3606	3446	2926	1646	246

Measured after 3.6y

Fukusima Second Nuclear Power plant-2 (BWR : 8 X 8)

Parameter	Value
Fuel rod pitch (mm)	about 1.63
Outer diameter of pelet (mm)	about 1.03
Outer diameter of fuel rod (mm)	about 1.23
Clad thickness (mm)	about 0.86
Fuel temperature (k)	900
Clad temperature (k)	600
Moderator temperature (k)	600
Inner diameter of channel box (mm)	134
Outer diameter of channel box (mm)	2.5

SF98	Measured point				
	SF98-3	SF98-5	SF98-6	SF98-7	SF98-8
Void ratio (%)	3.0	32.0	54.5	68.0	73.0
Burnup (GWd/t)	36.94	43.99	39.92	39.41	27.18
Axial position (From bottom of active fuel, mm)	423	1214	2050	2757	3397

Measured after 5.5y for SF98-1,2,3,4

Measured after 6.2y for SF98-6

Measured after 5.9y for SF98-5,7,8

2.3.2. Calculation results

The results of comparison between the nuclide weight obtained using improved MVP-ORBURN and the measured values are listed in Table 3 and shown in Figures 5 and 6. For the PWR fuels, there is a tendency that the fissile nuclides are underestimated, while the other nuclides are overestimated. However, the C/E ratios were almost within 10%. Because the specimens were positioned at the corners of the assembly, there is a possibility that the water gap and surrounding fuels affected the results.

For the BWR fuels, significant improvement could not be achieved by consideration of the axial void fraction only. However, we obtained predictions that accuracies can be achieved almost within 10% in most cases. The position of samples next to the fuel rods containing gadolinium is one of the factors in making it difficult to improve the accuracy.

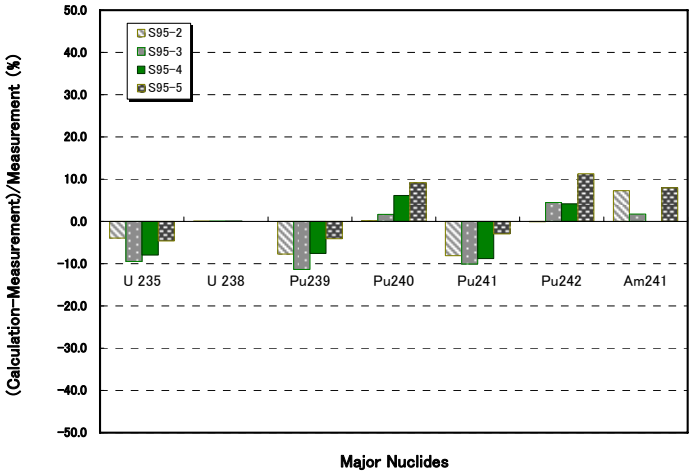

FIG. 5. Benchmark calculation results by MVP-ORBURN for JAERI PIE data (PWR spent fuel: SF-95).

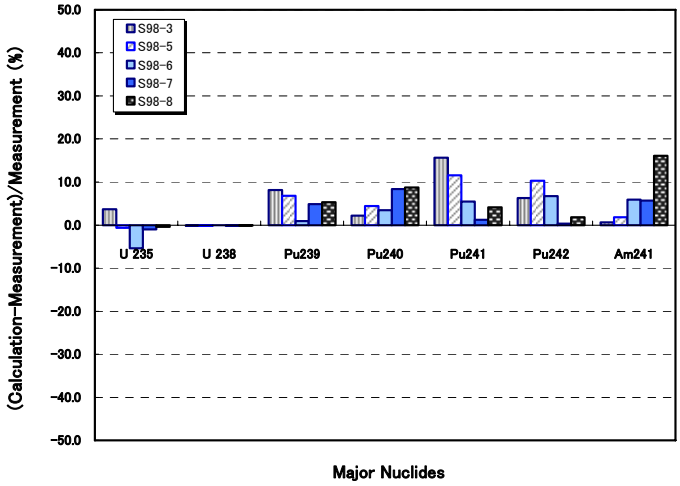

FIG. 6. Benchmark calcualtion results by MVP-ORBURN for JAERI PIE data (BWR spent fuel: SF-98)

Table 3 Results of Benchmark Analysis on PIE Data with MVP-ORBURN

Takahama Nuclear Power plant-3 (PWR : 17 X 17)

Nuclide	SF95-2			SF95-3			SF95-4		
	Calculation	Measurement	(C-M)/M (%)	Calculation	Measurement	(C-M)/M (%)	Calculation	Measurement	(C-M)/M (%)
U 235	1.851E+04	1.927E+04	-3.9	1.200E+04	1.326E+04	-9.5	1.132E+04	1.230E+04	-8.0
U 238	9.431E+05	9.424E+05	0.1	9.348E+05	9.338E+05	0.1	9.338E+05	9.335E+05	0.0
Pu239	5.218E+03	5.655E+03	-7.7	5.490E+03	6.194E+03	-11.4	5.547E+03	6.005E+03	-7.6
Pu240	1.541E+03	1.539E+03	0.1	2.222E+03	2.186E+03	1.6	2.342E+03	2.207E+03	6.1
Pu241	8.800E+02	9.578E+02	-8.1	1.336E+03	1.486E+03	-10.1	1.337E+03	1.466E+03	-8.8
Pu242	1.842E+02	1.844E+02	-0.1	4.717E+02	4.516E+02	4.5	5.002E+02	4.803E+02	4.1
Am241	2.514E+01	2.344E+01	7.3	3.367E+01	3.310E+01	1.7	3.255E+01	-	-

Nuclide	SF95-5		
	Calculation	Measurement	(C-M)/M (%)
U 235	1.473E+04	1.544E+04	-4.6
U 238	9.387E+05	9.388E+05	0.0
Pu239	5.402E+03	5.635E+03	-4.1
Pu240	1.988E+03	1.821E+03	9.2
Pu241	1.119E+03	1.153E+03	-2.9
Pu242	3.311E+02	2.976E+02	11.3
Am241	3.067E+01	2.840E+01	8.0

Fukusima Second Nuclear Power plant-2 (BWR : 8 X 8)

Nuclide	SF98-3			SF98-5			SF98-6		
	Calculation	Measurement	(C-M)/M (%)	Calculation	Measurement	(C-M)/M (%)	Calculation	Measurement	(C-M)/M (%)
U 235	8.443E+03	8.142E+03	3.7	6.275E+03	6.315E+03	-0.6	8.573E+03	9.062E+03	-5.4
U 238	9.388E+05	9.406E+05	-0.2	9.311E+05	9.328E+05	-0.2	9.328E+05	9.334E+05	-0.1
Pu239	3.995E+03	3.694E+03	8.2	4.555E+03	4.265E+03	6.8	5.355E+03	5.305E+03	0.9
Pu240	2.181E+03	2.135E+03	2.2	2.728E+03	2.613E+03	4.4	2.721E+03	2.630E+03	3.5
Pu241	1.035E+03	8.949E+02	15.7	1.307E+03	1.172E+03	11.5	1.362E+03	1.292E+03	5.4
Pu242	4.913E+02	4.623E+02	6.3	7.651E+02	6.939E+02	10.3	5.796E+02	5.431E+02	6.7
Am241	3.292E+01	3.271E+01	0.6	3.802E+01	3.734E+01	1.8	4.333E+01	4.091E+01	5.9

Nuclide	SF98-7			SF98-8		
	Calculation	Measurement	(C-M)/M (%)	Calculation	Measurement	(C-M)/M (%)
U 235	9.261E+03	9.357E+03	-1.0	1.539E+04	1.545E+04	-0.4
U 238	9.315E+05	9.332E+05	-0.2	9.420E+05	9.431E+05	-0.1
Pu239	5.901E+03	5.628E+03	4.9	5.626E+03	5.341E+03	5.3
Pu240	2.890E+03	2.668E+03	8.3	1.975E+03	1.816E+03	8.8
Pu241	1.371E+03	1.355E+03	1.2	9.453E+02	9.079E+02	4.1
Pu242	5.456E+02	5.439E+02	0.3	2.260E+02	2.220E+02	1.8
Am241	4.636E+01	4.388E+01	5.7	3.824E+01	3.295E+01	16.1

2.4.Analysis of applicability of burnup credit using MVP-ORBURN

2.4.1. Calculation conditions

(1) Calculation of nuclide composition

Using improved MVP-ORBURN, were obtained the nuclide compositions of typical PWR and BWR fuels by changing the following parameters.

(a) Parameters changed for PWR fuels: Burnup credit level, initial enrichment, fuel burnup and cooling period for depleted fuel

(b) Parameters changed for BWR fuels: Burnup credit level, initial enrichment, fuel burnup, average void fraction and cooling period for depleted fuel

Nine actinides, ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu and ^{241}Am were considered for burnup credit level 1, and twelve fission product nuclides, ^{149}Sm, ^{103}Rh, ^{143}Nd, ^{133}Cs, ^{99}Tc, ^{152}Sm, ^{155}Gd, ^{145}Nd, ^{147}Sm, ^{95}Mo, ^{153}Eu and ^{150}Sm in addition to the nine actinides for burnup credit level 2.

MVP-ORBURN was based on the JENDL-3.3 based MVP library as previously mentioned, and the JENDL-3.2 based ORIGEN libraries for LWR were developed by JAERI [4] (refer to Table 4).Acknowledgements

(2) Criticality calculation when the burnup credit is applied

Based on the nuclide compositions obtained above, the cross sectional views of the typical PWR and BWR fuel transportation casks were prepared, as shown in Figure 6, for the CSAS25 sequence of the SCALE-4.4a system [5]. And the criticality calculations with the multigroup energy Monte Carlo code, KENO-Va, were performed to obtain the effective multiplication factors. The 238 Group nuclear library, based on the ENDF/B-V cross section files, was used.

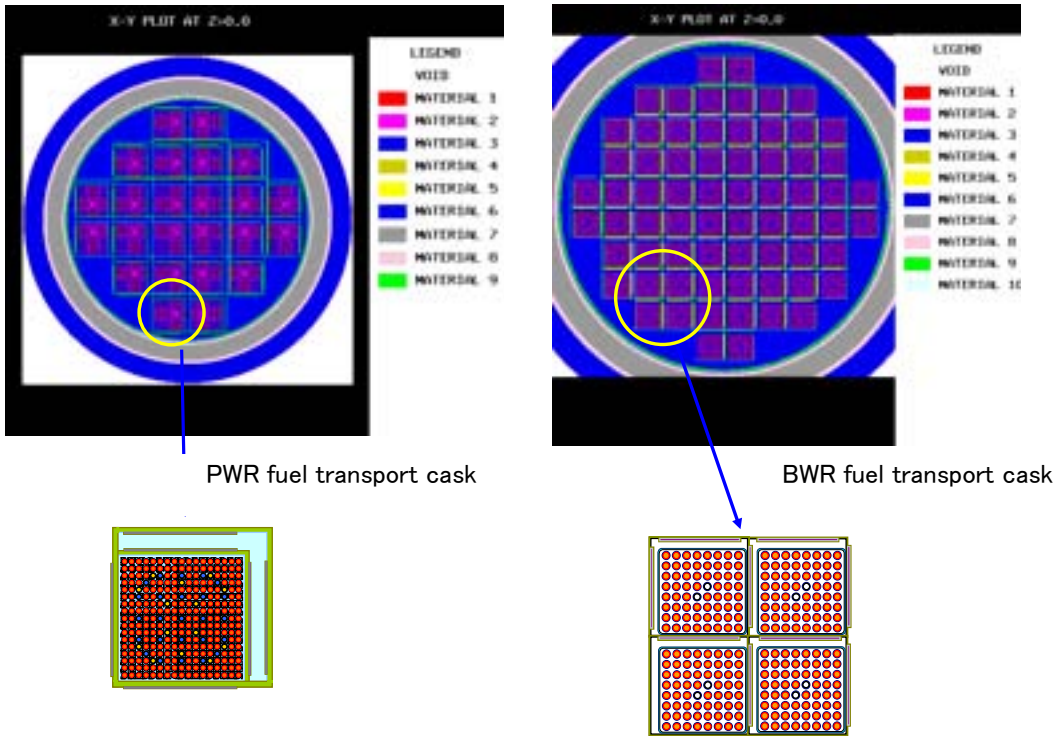

FIG. 7. Cross spectional view of criticality calculation model of tranport cask for PWR and BWR fuels.

2.4.2. Calculation results

(1) Evaluation of influence of nuclide composition

Before performing the criticality calculations, we analyzed and compared the nuclide compositions of MVP-ORBURN and ORIGEN for the PWR fuels that have the initial enrichment of 3% and 5% and are depleted to the burnup of 30 GWd/t, and examined differences in the nuclide compositions and general tendencies in the criticality calculations. Most of the PWR fuels commonly used in Japan have the enrichment of 3.4 to 4.1%, and are included in the range of this analysis.

As shown in Figures 8 and 9, the depletion characteristics greatly vary depending on the initial enrichment. It was recognized that ORIGEN overestimated the generation of ^{239}Pu and underestimated the consumption of ^{235}U. This is probably because ORIGEN cannot reflect the fluctuation of effective cross section due to depletion.

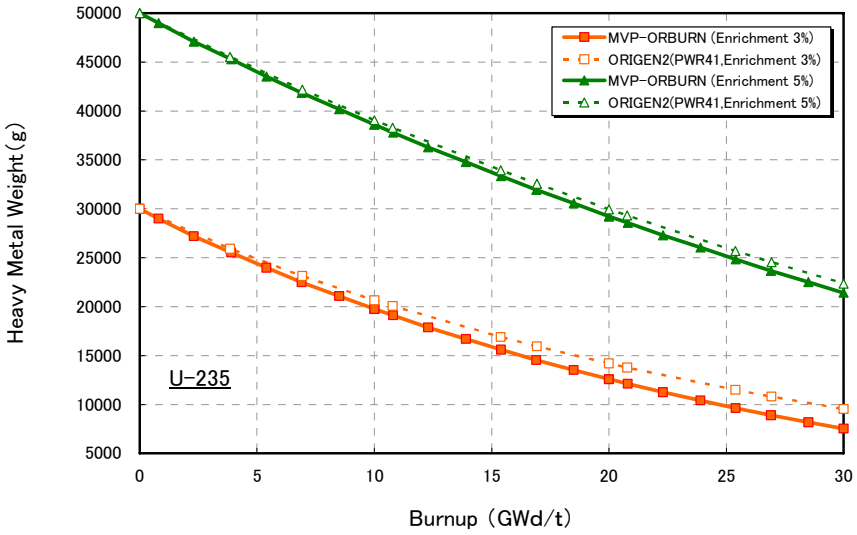

FIG. 8. Comparison of MVP-ORBURN and ORIGEN (depletion characteristics for U-235).

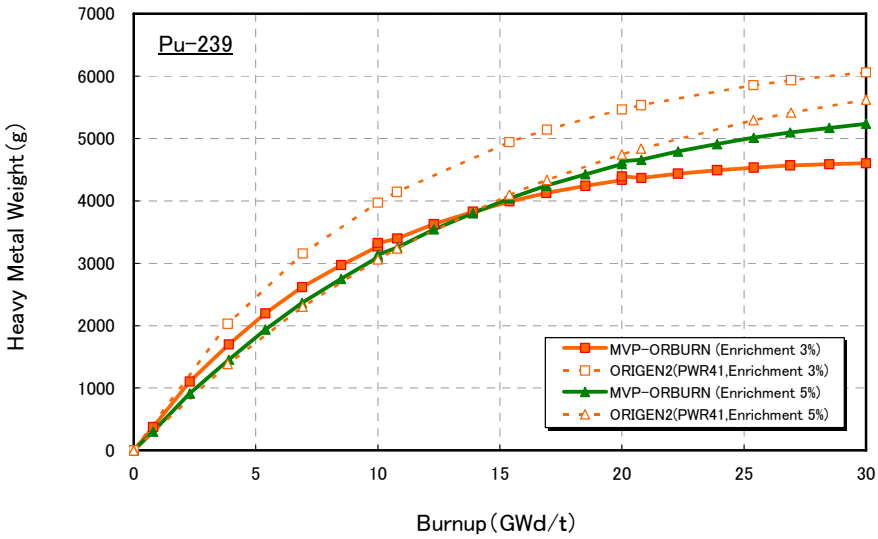

FIG.9 Comparison of MVP-ORBURN and ORIGEN (depletion characteristics for Pu-239).

The results of the criticality calculation using these compositions are shown in Figures 10 to 13. Figure 10 shows the results of applying burnup credit level 1 to the PWR fuels, and Figure 11 shows

the results of applying burnup credit level 2. Figure 12 shows the results of applying burnup credit level 1 to the BWR fuels, and Figure 13 shows the results of applying burnup credit level 2.

For any type of fuel, it was revealed that detailed depletion calculation by MVP-ORBURN overestimated the effect of burnup credit, while the evaluation by ORIGEN produced conservative results.

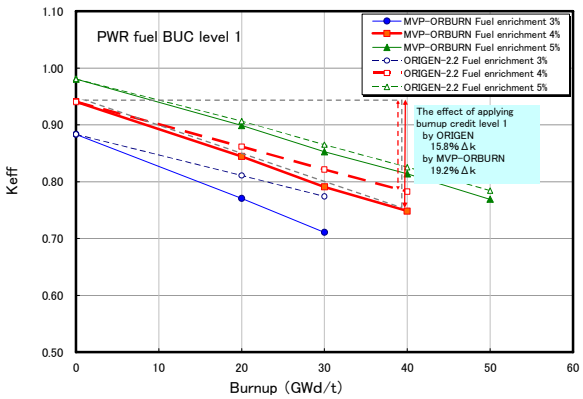

FIG. 10. Difference between the initial enrichment and the burnup credit level 1 calculations for PWR fuel.

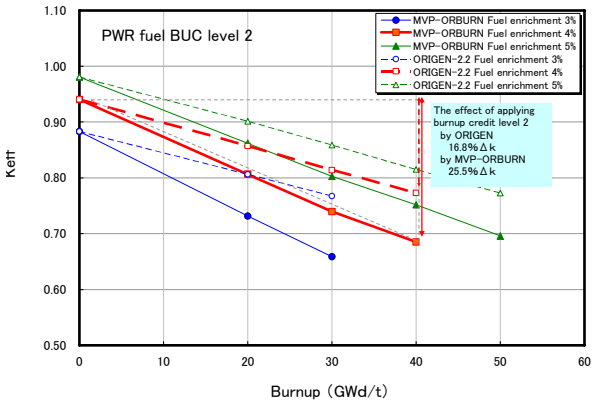

FIG. 11. Differnce between the initial enrichment and the burnup credit level 2 calculations for PWR fuel.

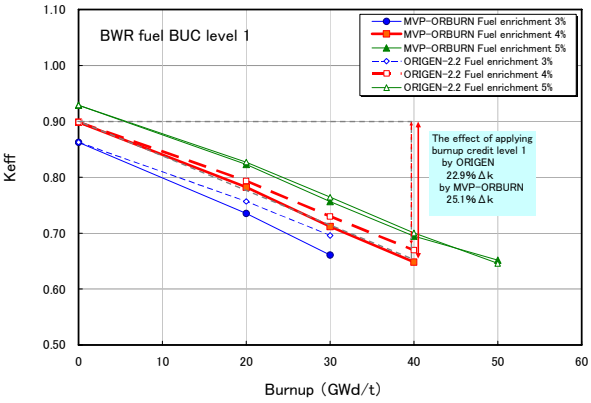

FIG. 12. Differnces between the initial enrichment and the burnup credit level 1 calculation for BWR fuel.

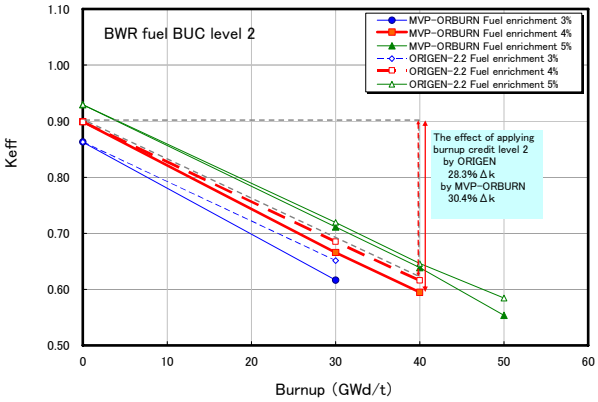

FIG. 13. Differnces between the initial enrichment and the burnup credit level 2 calculation for BWR fuel.

(2) Criticality calculation when burnup credit is applied

From the changes in the effective multiplication factor due to different parameters applied to the PWR and BWR fuel transportation casks, we examined what degree of effective margin can be expected when the burnup credit is applied. For this purpose, we classified the basic uncertainties into four types: (i) differences due to presence or absence of the burnup credit and its level, (ii) influences of differences of the method for evaluating the number density in the depletion and storage conditions, (iii) influences of the range of acceptable burnup and the burnup management and measurement, and (iv) influences of uncertainties in the criticality calculation model and boundary conditions.

When the spent PWR fuel with the initial enrichment of 4% and the burnup of nearly 40 GWd/t is evaluated by applying the ORIGEN based burnup credit level 1, the effect of the effective multiplication factor due to presence or absence of the burnup credit is nearly 15.8%Δk. When the uncertainties due to the depletion condition, the burnup management and other reasons, about 12.4%Δk in total, are taken into account, the net effect becomes nearly 3.4%Δk, as shown in Figure 16. When MVP-ORBURN is used, the net effect can be increased by nearly 3%Δk.

When the spent BWR fuel with the initial enrichment of 3.5% and the burnup of nearly 40 GWd/t is evaluated by applying the ORIGEN based burnup credit level 1, the effect of effective multiplication factor due to the presence or absence of the burnup credit is nearly 22.9%Δk. When the uncertainties due to the depletion condition, the burnup management and other reasons, 21.0%Δk in total, are taken into account, the net effect becomes nearly 1.9%Δk, as shown in Figure 17. When MVP-ORBURN is used, the net effect can be increased by nearly 2%Δk.

These results showed that introduction of the burnup credit evaluation does not have a great effect for the current fuels with relatively low average burnup and that the uncertainties in operation, such as the setting of acceptable burnup specified for facility and equipment which determines the tolerable limit of including low burnup fuels, have a relatively great influence on the margin obtained by applying burnup credit.

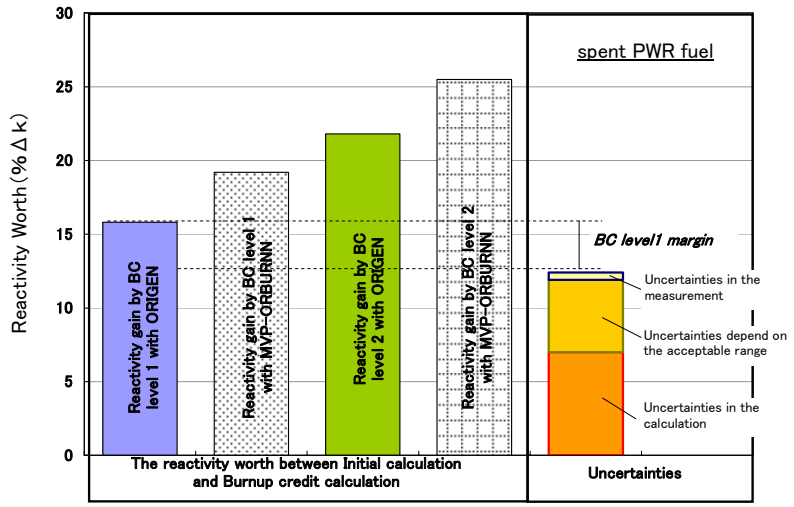

FIG. 14. Realistic evaluation of the burnup credit margin for PWR spent fuel tranport cask.

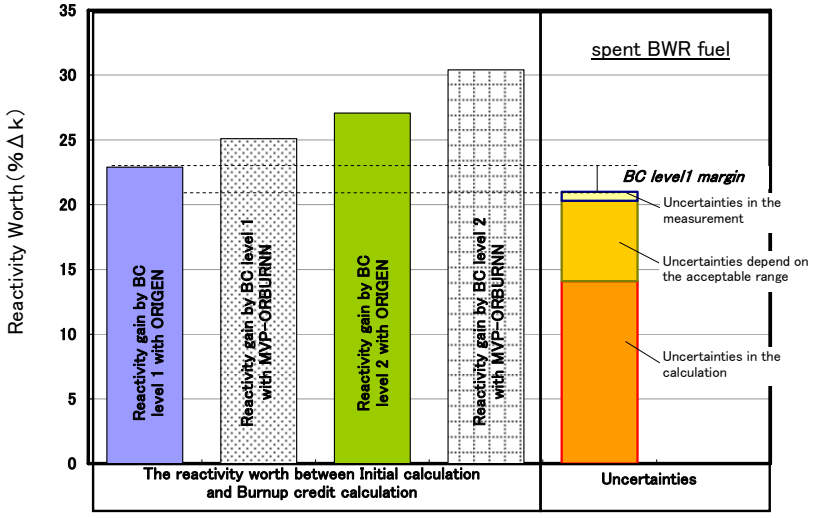

FIG. 15. Realistic evaluation of the burnup credit margin for PWR spent fuel tranport cask.

3. Summary

The accuracy of MVP-ORBURN was evaluated in the validation analysis against PIE data. The results showed that the PWR fuels had the C/E ratios of nearly 10% or less for major nuclides. For the BWR fuels, however, the accuracy level of 10% or lower was achieved mainly by the improvement made to

the calculation in axial direction. For these reasons, we consider that MVP-ORBURN code has nearly reached the level of practical use.

In addition, taking spent fuel transportation casks as an example, we examined the influence of differences in nuclide composition on MVP-ORBURN and ORIGEN and the various kinds of uncertainties due to application of the burnup credit. We obtained the conclusions that the evaluation of nuclide composition using ORIGEN produced conservative results, particularly when the burnup level was low, and that the setting of acceptable burnup specified for facility and equipment had relatively great influence on the margin obtained by applying burnup credit.

Improvements efforts are now proceeding on MVP-ORBURN for establishing it more accurate and convenient. And we are also planning to evaluate the margin obtained by applying burnup credit, and the influence of uncertainties due to applying burnup credit for the dissolver of the reprocessing facility.

REFERENCES

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A burnup credit concept for CASTOR® transport and storage casks with PWR spent fuel

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Abstract. The trend towards higher initial enrichments, extended burnup of the fuel and an economic loading strategy for transport and storage casks requires the use of burnup credit (BUC). The application of BUC markedly complicates the criticality safety analysis. The methodology requires both depletion calculations and subsequent criticality calculations containing the predicted spent fuel composition. In addition, it combines design data with irradiation conditions during reactor operation. This paper describes the methodology of applying BUC in future criticality safety analyses for PWR spent fuel casks.

1. Situation in Germany when applying BUC

Fig. 1 shows the NPP locations in Germany as well as their operating companies.

If you are applying BUC in calculations for a NPP wet storage there is only one supervisor checking your calculations and giving the approval. Furthermore, this supervisor is involved in the supervision of the NPP and therefore has access to detailed information about the fuel assemblies, e.g. axial burnup profile, irradiation time.

In the case of transport and storage casks there are several authorities and supervisors involved. First you are dealing with the authority for licensing of the transport cask and the supervisor for the approval of transportation. After having received the approval for the cask, a second authority, the licensing authority for the storage installation, and the supervisor for the NPP will become involved for loading the cask.

In order to minimize the investment for the supervision there are two possibilities. The first one is the use of very conservative boundary conditions and maximal safety factors in the calculations. Another possibility is to establish a calculation methodology and a calculation chain which considers all uncertainties and calculational variations.

2. Procedure for applying BUC

Fig. 2 illustrates the procedure for applying BUC in criticality analysis for PWR spent fuel casks. Based on the reactor operation data, the spent fuel geometry and material as well as on burnup profiles, the nuclide inventory is determined by taking into account the results of burnup benchmarks for the nuclides of interest. In the next step, these nuclide inventory data are considered in the determination of k_{eff}.

3. Standard procedure — Loading curve

A loading curve is a simple graph for demonstrating the permissible cask loading since it connects the initial uranium enrichment with the burnup of the loaded fuel assemblies. To determine such a loading curve for cask loadings, many k_{eff} calculation runs have to be done in order to find the threshold, where the minimal burnup meets the max. k_{eff} of several initial enrichments (refer to Fig. 3). The scope of the loading curve is directly connected with the assumed (conservative) reactor operation data including the irradiation history, axial burnup profiles, the position of the fuel assembly, the position of absorber elements and control rods, burnable absorbers, adjacent fuel assemblies as well as axial temperature profile, etc. These elements already had been taken into account while determining the nuclide inventory of the fuel assemblies and its distribution within the fuel assemblies.

The benefits of a loading curve are having a simple criterion for permissible cask loadings and that illustrates out the potential capacity of the cask.

A drawback of a loading curve is, that the accumulation of several excessive conservative boundary conditions leads to a high minimal burnup of the fuel assembly and thus, to a restricted use of casks in some cases. Furthermore, it has to be verified thoroughly that the chosen boundary conditions cover the real data for the specific fuel assemblies of the NPP. Therefore, the chosen conservative boundary conditions need to be appropriate and all assumptions need to ensure that future changes to NPP operations are considered.

Since the casks should be used for all German PWR NPPs, a sufficient amount of data has to be investigated and a lot of calculations have to be done during the approval phase of a cask.

4. Alternative approach

As mentioned, it is possible to get a large or a small scope of the loading curve depending on the chosen (conservative) assumptions. Therefore, it may happen that a cask loading requires a single approval, if one or more fuel assemblies do not meet these requirements. From this point of view, an alternative approach is derived.

Applying the alternative approach, the nuclide inventory is calculated for each individual fuel assembly in a cask loading and a criticality analysis for this nuclide inventory with the actual cask loading plan for the specific cask loading is performed.

Following this line, step 1 establishes the method and gets the approval for the cask loading. In step 2, the approved method is applied.

There are two possibilities to determine the nuclide inventory for each fuel assembly. According to the standard procedure it can be determined based on reactor operation data (derived from /1/) or the nuclide inventory can be derived directly from reactor operations records. The second possibility offers the benefit that the transmitted nuclide inventory is based on continuous recalibration from the coupling of incore measurements and calculations. Therefore, it provides a more realistic representation of the nuclide inventory and its distribution within each fuel assembly after operation. It minimizes the conservatism of the assumptions (no excessive conservatism). It is an even more valuable benefit that this method is approved by the licensing authorities for the NPPs.

The next step is the establishment of the calculation model for the criticality analysis. This calculation model needs to take into account conservative boundary conditions concerning cask material and geometry, a conservative nuclide inventory (incl. safety factors), the number of examined cases (normal, off-normal and accident conditions) and special features of the loading (e.g. misloading).

Finally, the authorities for licensing of transport and storage installation need to certify the procedure / method.

By an upcoming cask loading campaign the examined and approved method will be applied for the specific cask loadings. First, the nuclide inventory for each fuel assembly will be determined based on NPP specific boundary conditions or it will be taken from reactor records. In the following, a criticality analysis for the specific cask will be performed.

This approach is beneficial since it gives a more realistic representation of the nuclide inventory and its distribution within each fuel assembly after operation and since it minimizes the conservatism of the criticality safety analysis. The result of this approach leads to a lower maximal burnup of the fuel assemblies.

Drawbacks of this alternative approach are that a separate set of calculations has to be done for each cask loading and that there exists a strong relationship between reactor physics calculations and the cask analysis.

It has to be mentioned that not all nuclides of interest in burnup calculations are considered in the reactor physics calculations, e.g. all generated fission products. Therefore, some effort has to be done for including these nuclides in reactor physics calculations.

5. **Conclusion**

To sum up, there are the following benefits in applying the alternative approach:

- Reduction of excessive conservative assumptions and
- a safe long term usage of the transport and storage casks.

The application of this alternative approach as well as the application of the loading curve does not remove the responsibility to ensure that the fuel assemblies possess the demanded burnup. This can be done e.g. by measuring the burnup of each fuel assembly before loading it into the cask.
It is likely that a combination of a loading curve with a limited scope and the alternative approach may be used in the approval phase for future cask loadings.

Fig. 4 shows a CASTOR® cask of the new generation where the burnup credit concept will be implemented.

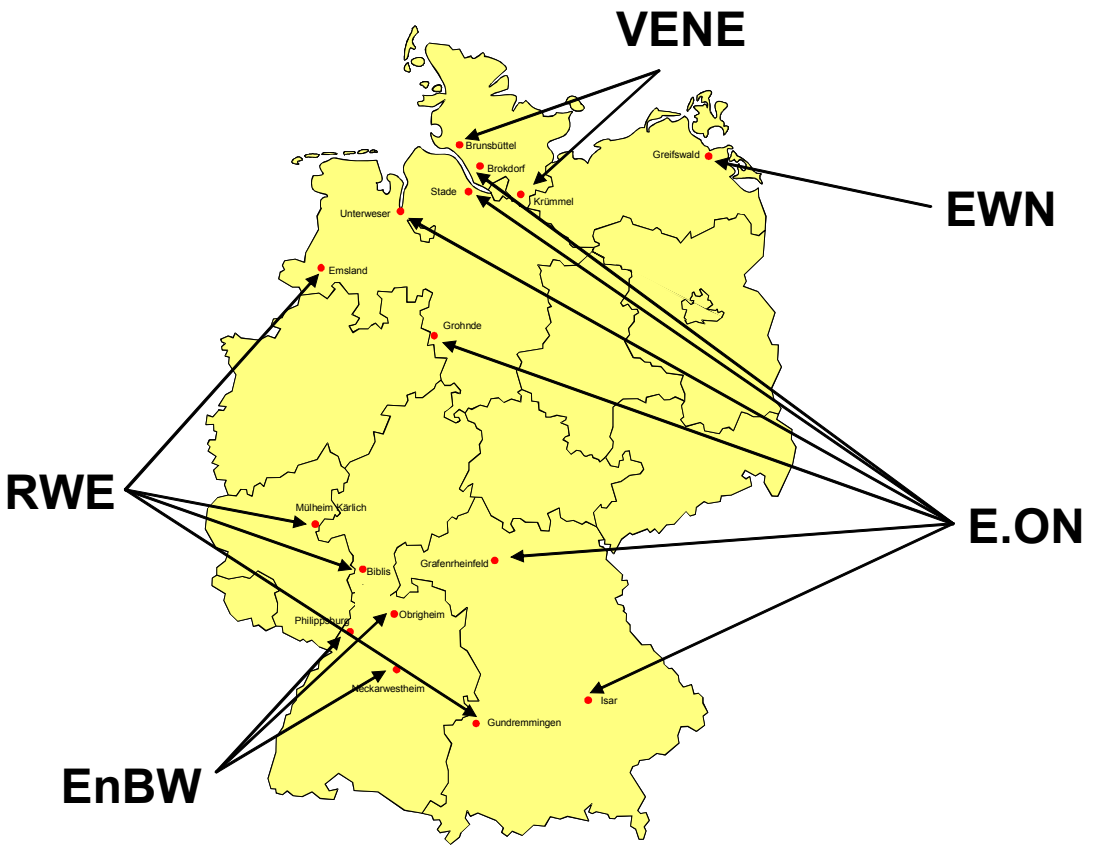

Fig. 1 NPP Locations in Germany and their operating companies.

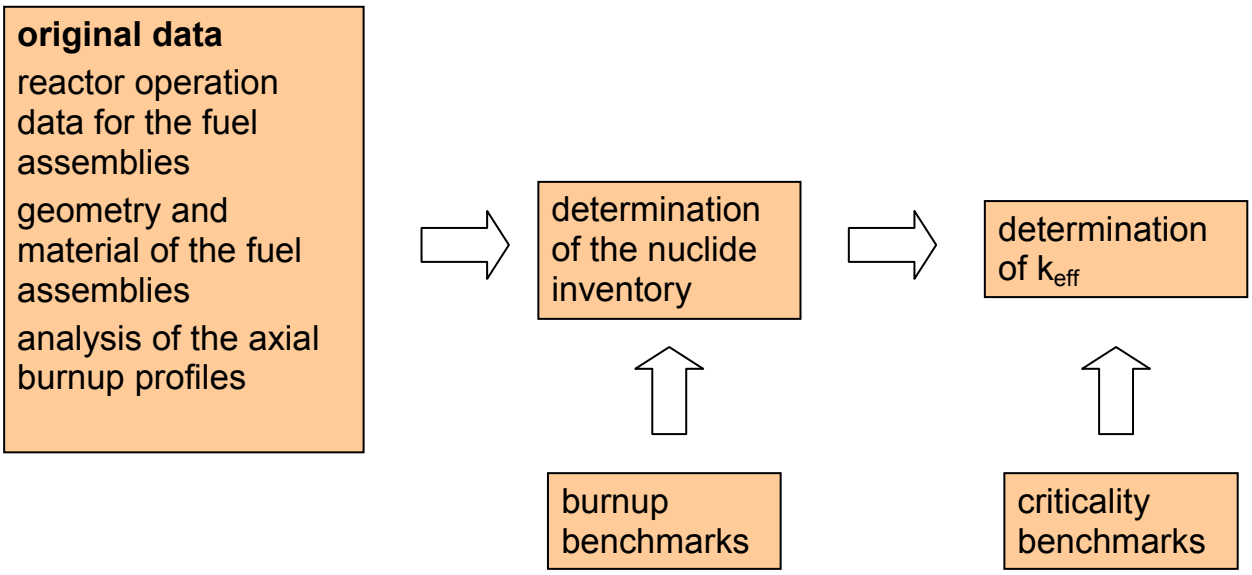

Fig. 2 Methodology for applying BUC in criticality analysis for PWR spent fuel casks.

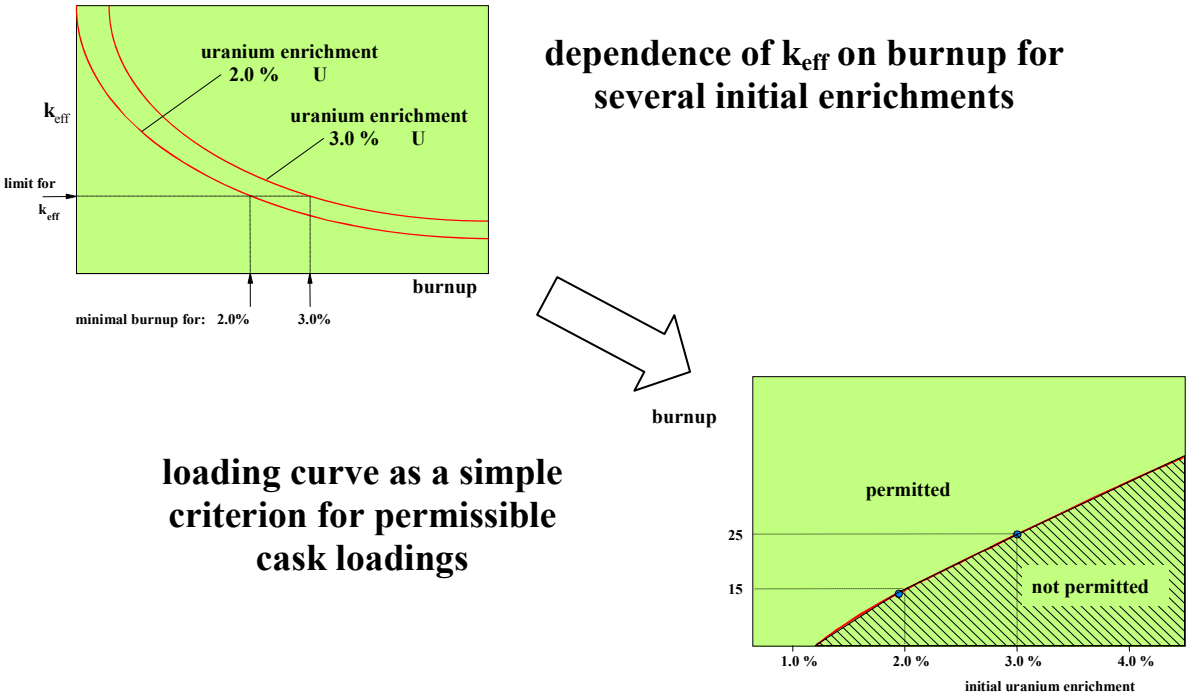

Fig. 3 Standard procedure – loading curve.

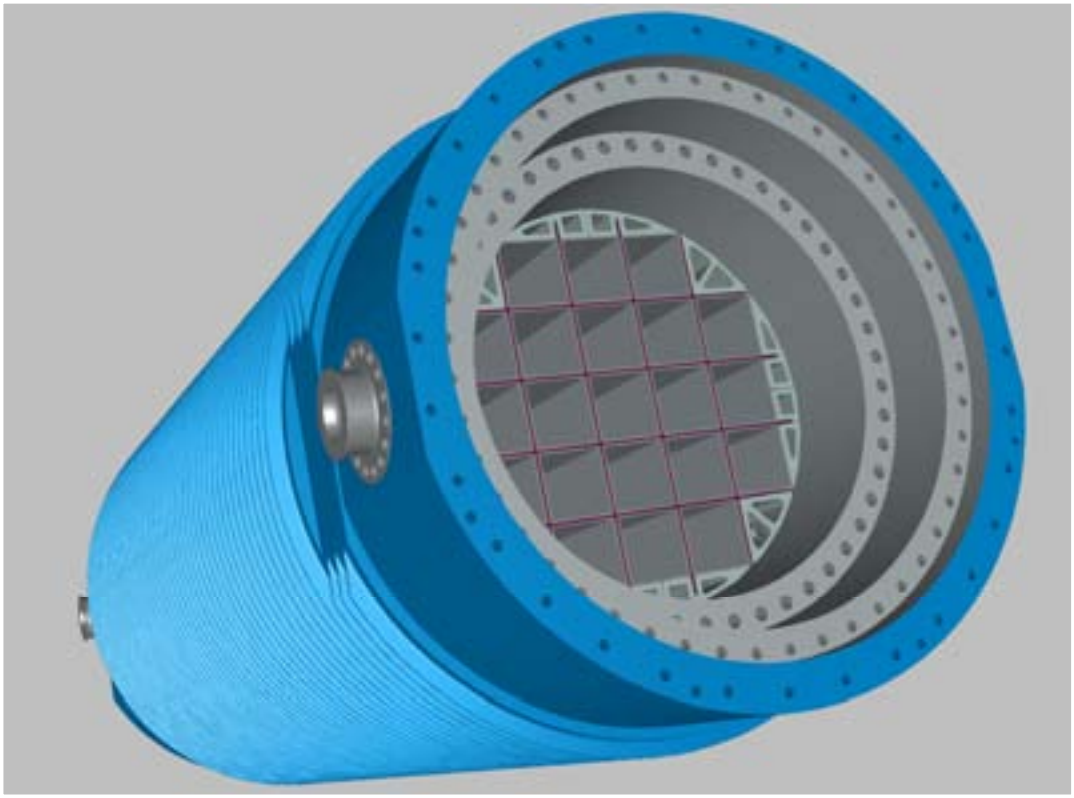

Fig. 4 Transport and storage cask CASTOR®.

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Issues and future plans of burnup credit application for final disposal

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Abstract. In Germany the major current work for final disposal of spent nuclear fuel is directed towards comparison of the different potential host rock materials salt, clay and crystalline rock for their suitability under the viewpoint of long term safety. Recent criticality analyses were performed for generic disposal facilities in clay and granite rock respectively, to compare the results with those from earlier analyses for salt rock. Average burnup values between 0 and 55 GWd/t HM were taken into account for the spent fuel. Due to the high package density of consolidated rods and the use of borated steel for structure material in the disposal cask POLLUX no burnup credit (BUC) is needed for normal and flooded conditions of the cask. The calculated k_{eff} is ≤ 0.9 for fresh fuel up to 4.5 % ^{235}U or 5.3 % Pu_{fiss}. But for certain scenarios, which characterize the possible development of a waste package after assuming water intrusion and corrosion of the fuel matrix, burnp credit is needed to demonstrate sub-criticality. An example of such a scenario is the creation of a pure solid mineral phase from dissolved fuel material e. g. Meta-Schoepite $UO_3 \cdot 2\ H_2O$. Some details and results of the recent criticality calculations for spent fuel are presented in the paper. Additional information is given on the calculation methods used. Besides criticality calculation the proper determination of the nuclide inventory is an important issue in BUC application. Considerable efforts have been preformed at GRS to improve the 2D/3D burnup code system KENOREST which is capable of two and three dimensional inventory calculations. Results of benchmark calculations are presented.

1. Background

In Germany direct disposal of spent nuclear fuel is being pursued as the second path of spent fuel management in parallel to reprocessing since the beginning of the 1980s. Thereby a pre-selection for disposal in a deep geologic salt rock formation was met and salt rock has been investigated since 20 years for its suitability. According to the nuclear phase out agreement of 2001 between the federal government and the nuclear power utilities the transport of SNF to the reprocessing facilities in France and the UK was terminated by end of June 2005 and spent fuel will further-on be stored at the NPP sites until a disposal facility is available. The disposal facility operation start-up is expected not before 2030. Furthermore it was agreed upon to broaden the scope of the site selection process taking into account also clay and crystalline rock formations. Concomitantly special safety related questions are to be investigated by generic analyses, one of which is long-term criticality safety. The aim hereby is to provide information for comparing different host formations with regard to long term safety.

2. Criticality studies for final disposal

There are two aspects of criticality safety of the disposal facility, were burnup consideration may be of interest. The first one is the operational phase of the facility covering interim storage, transport and placement of the disposal canisters, which is not subject of this paper. The second one is the post-closure phase, which begins after backfilling and sealing of the repository. In this phase the disposed containers are no longer under direct control and underlie the influence of geological and geochemical processes. The time frame for the period of investigation was set to 1 Million years.

Our recent criticality studies for disposal of spent nuclear fuel on behalf of the Federal Office for Radiation Protection were directed to long-term analyses for the host rock formations clay and granite on the background of providing data for comparison with a disposal site in salt rock [1, 2]. Among others criticality calculations for disposed containers and distributions of fissile material in the near field of a repository were performed on the base of geochemically justified long-term scenarios (see Table III). For these calculations spent fuel from commercial light water reactors (PWR type) with mean burnup values from 20 up to 55 GWd/t HM was assumed (Table I). For the nuclide inventory actinides only were taken into account as specified in Table II. The reason for neglecting the fission products was that so far there is no justification that fission products would show the same geochemical behaviour as the actinides, or in other words the occurrence of a geochemical process, which may effect separation of actinides and fission products can not be excluded.

Another aspect of burnup consideration in disposal criticality analyses is the variation of the concentration of nuclides of the inventory with time due to the radioactive decay. This is of special importance for the Plutonium isotopes ^{239}Pu and ^{240}Pu. The decay of the latter ($t_{1/2}$ = 6563 y) which is a strong neutron absorber may result an increase of reactivity dependent on its fractional presence in the fuel composition. The decay of ^{239}Pu into ^{235}U can also be seen as a competitive process of a presumed geochemical separation of Plutonium. To evaluate these effects the nuclide inventories according to different cooling times have been determined for subsequent criticality calculations. As an example Figure 1 shows the calculated nuclide densities of the actinides considered in dependence of cooling time for spent MOX fuel with an average burnup of 40 GWd/t HM.

TABLE I: STANDARD FUEL TYPES AND BURNUP VALUES ASSUMED FOR DISPOSAL CRITICALITY CALCULATIONS

Fuel type	**Initial enrichment [wt-%]**	**Burnup [GWd/tHM]**	**Cooling time [years]**
UO_2 PWR	3.6 % ^{235}U	20, 40	1, 10^2, 10^3, 5x10^4, 10^6
UO_2 PWR	4.4 % ^{235}U	55	1,10^2, 5x10^4, 10^6
Mixed Oxide PWR	3.7 % Pu_{fiss}	20, 40	10^3, 5x10^4, 10^6
Mixed Oxide PWR	4.6 % Pu_{fiss}	55	1,10^2, 5x10^4, 10^6
MOX disposal rods	3.5 % Pu_{fiss}	No burnup	10^3, 5x10^4, 10^6
	5.0 % Pu_{fiss}	No burnup	10^3, 5x10^4, 10^6

Table II: Actinides taken into account for criticality calculations

^{230}Th	^{233}U	^{238}Pu	^{241}Am	^{242}Cm
^{232}Th	^{234}U	^{239}Pu	^{242m}Am	^{243}Cm
^{231}Pa	^{235}U	^{240}Pu	^{243}Am	^{244}Cm
^{237}Np	^{236}U	^{241}Pu		^{245}Cm
	^{238}U	^{242}Pu		

The burnup values were assumed to be representative and sufficient conservative to cover most of the spent fuel designated for final disposal. The influence of burnup and cooling time on the neutron multiplication factor k is demonstrated by the Figures 2, 3 and 4. Calculated k_{inf} of homogeneous fuel-water-mixtures for three different burnup values and cooling time 1 and 50 000 years respectively are represented in Figures 2 and 3. Apart from the expected reactivity decrease due to burnup there is a significant increase of k for long cooling time of MOX fuel, caused by the decay of ^{240}Pu. For UO_2 the effect of the decay of Plutonium on the reactivity is obviously weaker. Generally there is a decrease of k during the first 100 years after discharge, which can be explained by the decay of ^{241}Pu ($t_{1/2}$ = 14.4 y). This effect is equalized later on by the decay of ^{240}Pu. Figure 4 is showing k_{inf} of a fuel rod lattice filled with $Fe(OH)_3$ (representing corrosion products of the cask) for three different fuel types, UO_2 3.6 % 40 GWd/THM, MOX 3.7 % Pu_{fiss} 40 GWd/t HM, MOX 5 % Pu_{fiss} no burnup, for 1000 years and 50 000 years cooling time respectively. Again considerable variation in k can be observed for MOX fuel, changing its behaviour from a Pu fissile system to a U system. The increase of the maximum k-value is due to the decay of ^{240}Pu.

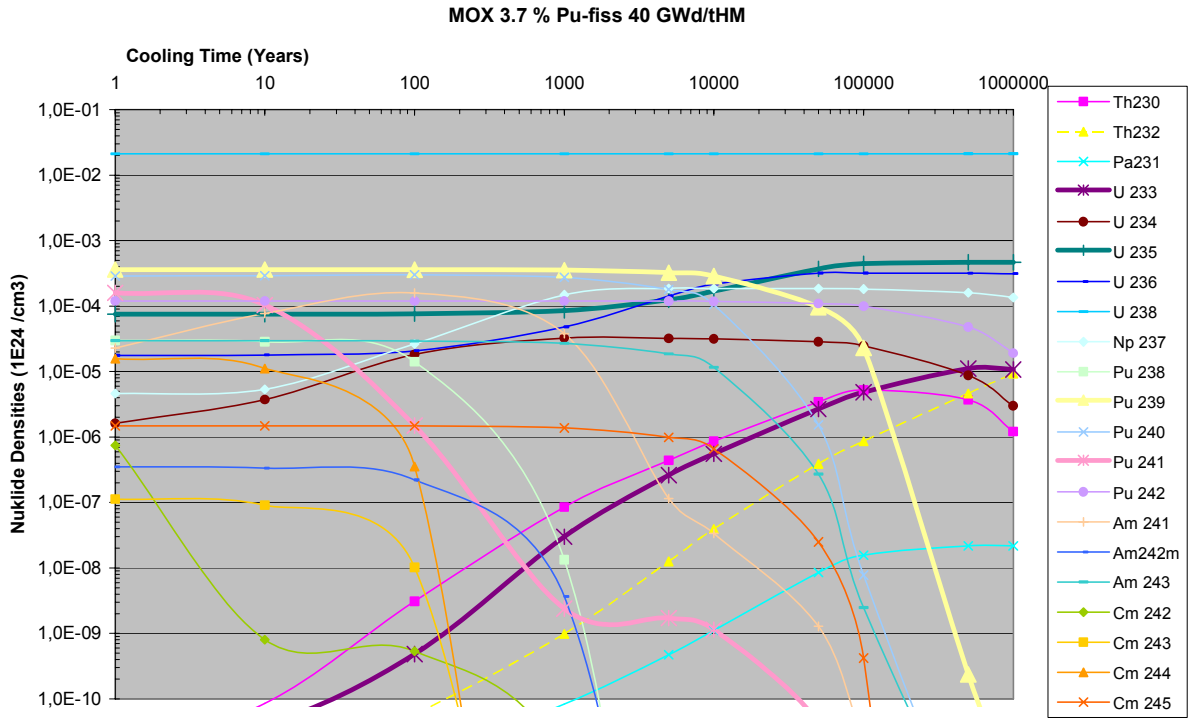

Fig. 1: Variation of Nuclide densities with cooling time.

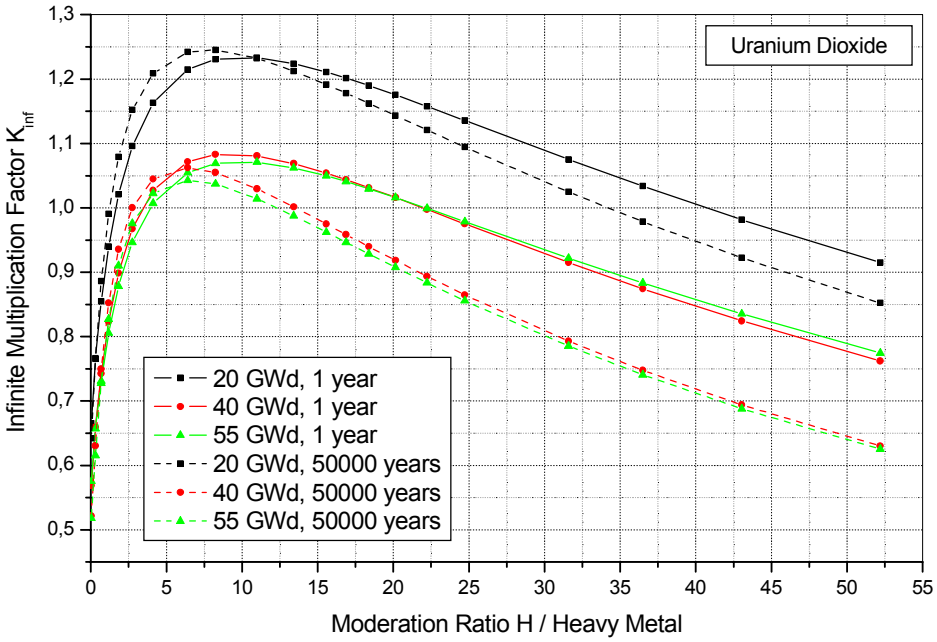

Fig. 2: Calculated k_{inf} of homogeneous spent UO_2 fuel-water-mixtures for various values of burnup and cooling time.

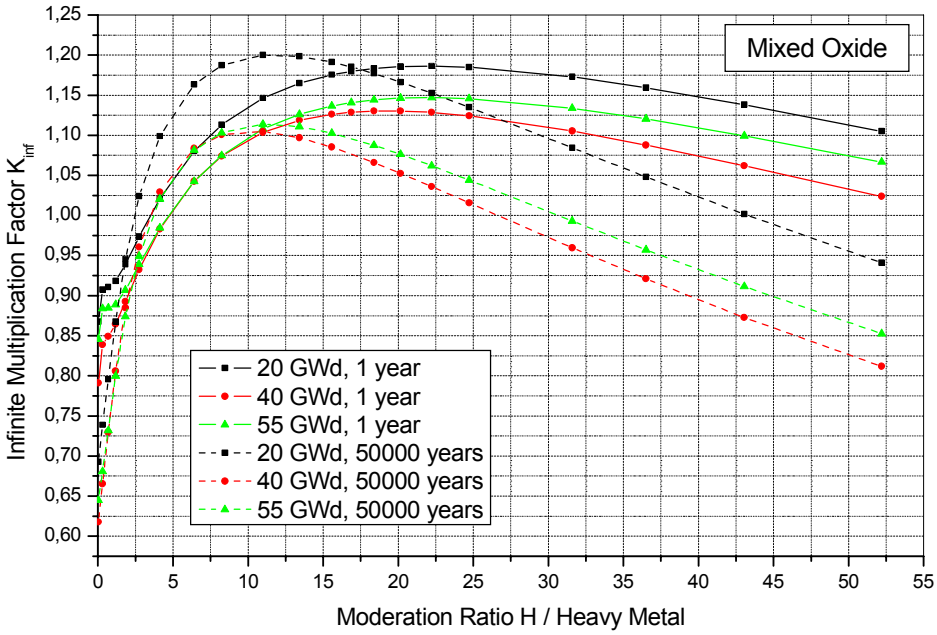

Fig. 3: Calculated k_{inf} of homogeneous spent MOX fuel-water-mixtures for various values of burnup and cooling time.

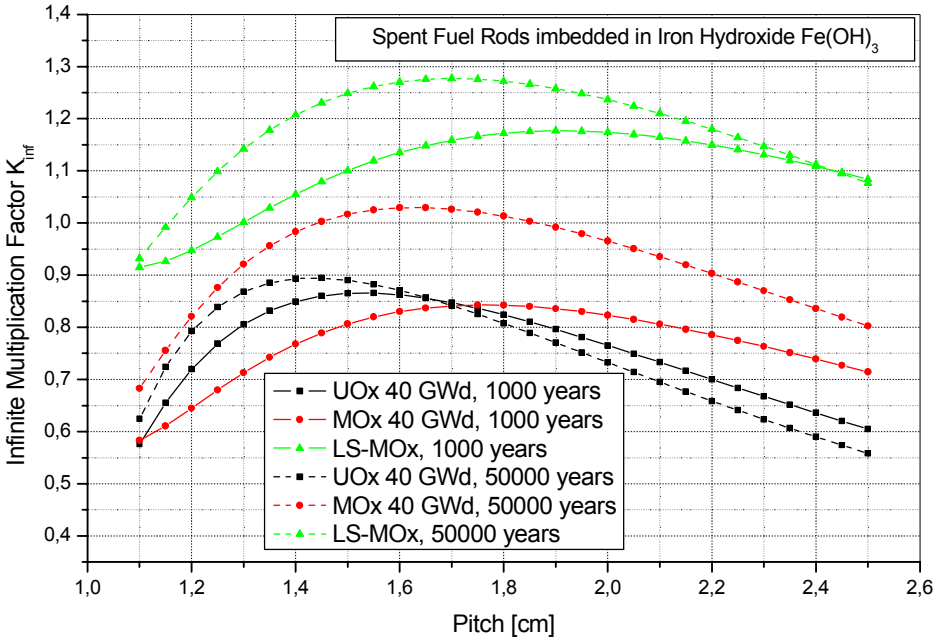

Fig. 4: Calculated k_{inf} for spent fuel rod lattice filled with Iron Hydroxide for various values of burnup and cooling time.

Two types of disposal canisters are currently being discussed for use in a future repository. The multi-purpose cask POLLUX for transport, storage and disposal is designed to accommodate fuel rods of up to 10 dismantled PWR or 30 BWR assemblies. The POLLUX cask consists of an outer shielding cask made of cast iron of 27 cm wall thickness and an inner fuel container made of steel with a wall thickness of 16 cm and with two lids. The outer dimensions are 156 cm diameter and 552 cm length. An alternative concept is the unshielded fuel rod canister BSK 3 which is designed for accommodating 3 dismantled PWR or 9 BWR assemblies respectively. The BSK 3 is a stainless steel canister of 5 cm wall thickness and 490 cm in length. The outer diameter of 43 cm corresponds to the HLW canister of the COGEMA type. The BSK 3 is designed for disposal in bore holes. For the POLLUX cask the placement and backfilling in horizontal galleries is planned. Descriptions of the disposal containers is given in references [3] and [4].

It should be mentioned that for the POLLUX cask no burnup credit is required to demonstrate criticality safety under transport conditions for UO_2 fuel up to 4.5 % ^{235}U enrichment and for MOX up to 5.3 % Pu_{fiss} respectively. This is achieved by consolidation of the fuel rods and the use of borated steel for the inner basket. In spite of the larger inventory the calculated k_{eff} for the POLLUX cask in the flooded case is not considerably higher than for the BSK 3 due to a slightly higher package density and the presence of neutron absorber. For corrosion scenarios the fraction of corrosion products of the container, which cause a decrease of neutron multiplication factor k is also higher for the larger POLLUX cask.

The criticality calculations performed for the post-closure phase were based on the geochemically justified long-term performance of the disposed container surrounded by clay-based backfilling. An important condition for criticality analyses is that the containers are being assumed to remain separated from each other and no accumulation of fissile material from more than one container is presumed. This is caused by the hydrologic properties of clay. Most of the criticality calculations were performed for assuming spent UO_2 and MOX fuel at 40 GWd/t HM average burnup and for un-irradiated MOX fuel (see Table I).

As can be seen from Table III the analyses of the investigated scenarios resulted in critical configurations only in the cases where low burnup or un-irradiated fuel has been assumed, and for the very unlikely scenarios, where selective migration and deposition of Plutonium has been presumed. According to the geochemical evaluation, the formation of a new solid mineral phase of Uranium after dissolution of fuel und precipitation seems possible. Such a geochemical 'conversion process' could take place within a container, following leakage and water intrusion. Criticality calculations for different mineral compounds of Uranium yielded the highest k-value for the hydated mineral Metashoepit $UO_3 \cdot 2H_2O$. Further on criticality analysis for a long-term scenario resulting in formation of this mineral showed the theoretical possibility of a critical configuration for unirradiated or low burnup (≤20 GWd/t HM) fuel.

TABLE III: SCENARIOS ASSUMED FOR CRITICALITY CALCULATIONS FOR THE POST-CLOSURE PHASE OF A REPOSITORY IN A CLAY OR GRANITE FORMATION BACKFILLED WITH CLAY

Nr.	Scenario	Time Frame	Criticaliy Calculations perfomed	Results [k]
1	Disposal canister intact flooding by water assumed (design requirement)	Up to 1000 years	UO_2 3,6 % 40 GWd, 1000 y POLLUX, BSK-3 MOX 3,7 % Pu_{fiss}, 40 GWd 1000 y MOX rods 5 % Pu_{fiss} no burnup 1000 y cooling time	< 0.48 < 0.5 < 0.75
2	Deformation of canister by ext. pressure, only compaction	> 1000 years	Covered by scenario no 1	
3	Container in the repository, backfilled with clay	> 1000 years	UO_2 3,6 % 40 GWd, 1000 y POLLUX, BSK-3 MOX 3,7 % Pu_{fiss}, 40 GWd 1000 y cooling time MOX rods 5 % Pu_{fiss} no burnup 1000 y cooling time	≤ 0.50 ≤ 0.45 < 0.66
4	Corrosion of container inside and outside	10^3 to 10^5 years		
4.1	$Fe(OH)_3$ fills free space in the cask, fruel rod remain intact		MOX rods 5 % Pu_{fiss} no burnup, 10^3, $5x10^4$, 10^6 y c.t.	< 0.66
4.2	Expansion of rod lattice by corrosion product $Fe(OH)_3$		MOX rods 5 % Pu_{fiss} no burnup BSK-3 MOX rods 5 % Pu_{fiss} no burnup POLLUX Mixed loading regime MOX no burnup, spent UO_2	< 0.92 ≈ 0.7–1.12 < 1.0
4.3	Degradation of fuel rods, mixing with $Fe(OH)_3$		Covered by scenario no 4.1	< 0.5
4.4	Collapsing of container after wall corrosion, disruption, degradation of fuel rods		Mixtures of fuel particles, Zircalloy and water reflected by $Fe(OH)_3$ or Fe_3O_4	< 0.8
5.1	Formation of a solid pure U-mineral through diffusion process and precipitation from dissolved fuel	Appox. 10^6 years	k_{inf} for different U-minerals, U composition based on spent UO_2 fuel 40 GWd/t HM	≈ 0.8-1.07
5.2	Formation of the mineral $UO_3 \cdot 2H_2O$ within one container	Up to 10^6 years	$UO_3 \cdot 2H_2O$ reflected by $Fe(OH)_3$ or Fe_3O_4 , critical mass from one cask load only for low (20 GWd) and 0 burnup fuel	230- 2100 kg U
6	Mixture of U mineral phase and corrosion products	Up to 10^6 years	k-inf for homogeneous mixtures of $UO_3 \cdot 2H_2O$ + $Fe(OH)_3$	< 1.0
	Scenarios for selective accumulation of fissile nuclides in the near field			
7.1	Sorption of Pu on corrosion products, up to 4,3 g Pu per kg of Fe_3O_4	10^5 – 10^6 years	k_{inf} for homogeneous mixtures of Pu and Fe_3O_4	< 1.0
7.2	Deposition of Pu by suface precipitation with $Fe(OH)_3$ up to 62 g Pu/l	10^5 – 10^6 years	k_{inf} for homogeneous mixtures of Pu and $Fe(OH)_3$ Minimum critical mass for LWR-Pu at 50 000 y cooling time Pu decays before the required mass could accumulate	> 1.0 12.5 kg

3. Plans for future work

The criticality analyses for final disposal performed so far were of generic type. The next step is to develop a "practically applicable" approach for considering the fuel burnup in the criticality analysis for the safety case of a decided repository. This means the burnup characteristics of the spent fuel assemblies, which are intended for direct disposal, must be in compliance with, or conservatively covered by the assumptions used for the long-term criticality analyses for the repository. Therefore a sufficiently conservative approach for deriving a loading curve and bounding profiles [5-7] will be required. This includes a systematic screening of burnup influence on long term sub-criticality and the development of bounding burnup criteria for disposal canisters. Thereby has to be taken into account, that no fuel assembly may be rejected from final disposal due to insufficient burnup. A possible approach with regard to long term scenarios with degradation of the fuel structure could be to determine average and minimum required burnup values for the total fuel charge of a container, in dependence of the fuel enrichment. Preliminary criticality calculations have been performed for assuming a mixed loading regime of unburned disposal rods and spent fuel rods for the POLLUX cask. This issue will be subject of a separate study which is planned to start later this year.

4. Calculation methods

For k_{eff} calculations the generally established and proven Monte Carlo codes KENO Va and MCNP are being used. For neutron transport calculations using S_N method a 2d system DORTABL is under development, which includes the 1d burnup system OREST [8] and the 2d deterministic criticality code DORT from the DOORS package [9]. The system calculates k_{eff} for finite cylinder models in r-z-geometry and is a useful tool for fast calculation on simple geometric models.

Besides criticality calculation the appropriate determination of the nuclide inventory is an important issue in BUC application. For burnup and inventory calculations the GRS-developed coupled code systems OREST for 1d problems and KENOREST [10] for 2d/3d problems are being applied. Some recently performed improvements of the KENOREST system and validations are described briefly in the following. A more detailed description can be found in reference [11].

KENOREST is a coupled inventory and reactivity calculation system and provides a coupling of the 3d Monte Carlo code KENO Va and the 1d deterministic pin cell depletion calculation system OREST based on the codes HAMMER and ORIGEN [10]. The system is capable of 2d and 3d calculations of reactivity, pin power distributions and inventories. Several substantial improvements have been implemented in the system during the last years.

- The KENO VI code has been included into the system to enable burnup calculations for hexagonal lattice geometry.

- A radial multi-zoning option for the fuel pellet region was implemented to enable better calculation of the burnout inside the pellet, which is indispensable for burnable poison rods.

- A convergence control routine was introduced to eliminate convergence problems, which occurred due to the coupling of the Monte Carlo code with deterministic codes. Thereby the global parameter k_{inf} and local results of power distributions are compared for subsequent KENO runs.

- A complete update of the cross section libraries based on JEF2.2, ENDF/B-6, JENDL3.2, EAF97 data, and an extension of the depletion code ORIGEN were carried out. The GRS version ORIGEN-X has been extended to handle and treat up to 25 fission yield sets and up to 900 irradiation steps (from previously 5 sets and 10 steps respectively).

A number of benchmark calculations were carried out for validating the improved code system. An example is given in Figure 5 from recalculation of spent fuel sample analyses data from the Takahama-3 reactor [12]. The figure shows calculated (c) over measured (e) nuclide concentrations of three samples in a mid burnup range. The calculations were performed in 2004 by using a test version

of the KENOREST system. There are still some outliers among the list of calculated nuclides for example ^{125}Sb and ^{237}Np, but most of the criticality relevant nuclides are met within 10%. The underestimation of Cm isotopes can be observed for most of the burnup codes [12]. The KENOREST system is currently used for benchmark calculations of the OECD Working Group on Brunup Credit (Phase II d and II e) and for the Depletion Calculation Benchmark of the Working Party on Scientific Issues of Reactor Systems (WPRS).

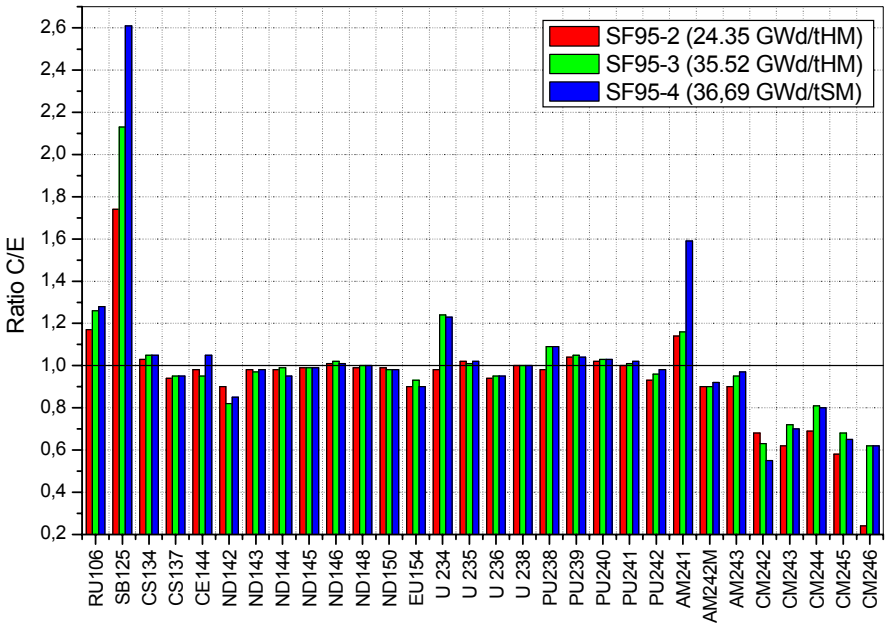

Fig. 4: Ratio of calculated (C) and measured (E) nuclide fractions for the SF95 sample series of spent fuel samples from Takahama-3 reactor.

5. Conclusion

From the analyses performed so far, some preliminary conclusions concerning burnup credit application in disposal criticality analyses can be drawn:

The decrease of reactivity of spent fuel due to the burn-out of fissile nuclides is beneficial also for long-term analyses. Fission products should only be taken into account, if their presence in the long-term range can be justified under the criteria of geochemistry. A possible long-term increase of reactivity must be considered for MOX fuel. For scenarios where degradation of the fuel rod structure must be assumed no axial burnup distribution can be applied. The determination of a conservative equivalent average burnup and a minimum burnup value for the inventories should be sufficient for long-term criticality analyses.

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An intelligent spent fuel database for BWR fuels

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Abstract. The present aim is to establish an intelligent database of Spent Fuel Data (including physical fuel data and reactor operating history information) to support burnup credit analyses for Boiling Water Reactor Fuel. At a later date, information of Pressurized Water Reactor Fuel and existing Post-Irradiation Examination (PIE) data for benchmarking fuel composition calculations may be integrated into the database.

1. INTRODUCTION

The technical approach involves identifying and collecting data to address a large region of fuel types for Boiling Water Reactors. The development of data tables is the first step in creating a useful database. The data tables will need to be comprehensive enough to support the calculation of burnup uncertainties, burnup distributions both axially and radially, different methods of performing depletion analyses and the potential for the use of reactor data in validation/verification calculations. Additionally, specific analyses will be performed within the database to identify and prioritize specific fuel types and/or reactors for which additional data should be collected from the existing sources.

Key elements of the data compilation activity are to identify data sources and obtain detailed data consistent with that used in and produced by 3-D core simulation calculations from utilities. The data will be incorporated into a database using a commercially available database platform (e.g. ACCESS ™, SQL ™, OPUS ™, etc).

This activity includes incorporating the spent fuel data into a database and using an analysis tool developed by Battelle to identify atypical fuel conditions and pattern detection among spent fuel data. This will permit a systematic evaluation of the range of fuel parameters that are important to be addressed in the criticality safety analyses to support the YMP. The result will aide in targeting specific fuel types and conditions (e.g. enrichment, burnup, presence of different absorbers) to establish the widest area of applicability with the fewest data points.

2. PROPOSED DATABASE ACTIVITY

The database tool was initially developed to provide aviation experts with information about flight patterns and operational conditions that they need to identify both potentially risky pilot behavior and mechanical problems by analyzing huge amounts of data previously unanalyzed. The technology was developed under contract with NASA and successfully used by commercial airlines to create the "Morning Report" analysis. The existing "Morning Report" technology can easily be adapted to the analysis of spent fuel data. It will permit the data to be organized in a standard format by which queries can be executed to automatically build input files for both depletion and criticality calculations. This technology was designed to pass sensitive data through a filter that would hide the identification to its source. The feature was built in to protect the airlines and their pilots from anyone being able to correlate safety problems with a particular company. For the proprietary fuels data this feature offers obvious advantages.

Database development/data mining will be used to establish standard data tables. An additional advantage of the "Morning Report" technology is that the tabular data can be sanitized such that the

data tables to be analyzed will only include unique assembly identifiers without traceability to utility. However, a QA table will be produced to provide traceability and data verification. Vendor identity will also be masked such that only the array size (e.g. 7x7, 8x8) is identified in the general database. The data will be reviewed and analyzed to identify data clusters, bounding conditions, etc. (i.e. to address the assumption that the newer fuel will identify more reactive fuel conditions). All data will have documentation showing transparent traceability back to a Qualified Source (e.g. a transmittal letter identifying the source of the data as a report developed under an NRC accepted QA program).

Data and analyses will be tailored to permit the study of three key information needs: uncertainty in reactor reported assembly burnup; effect of radial burnup gradients; and data for CRC evaluations.

Data need to establish the range of uncertainty in BWR spent fuel assembly average burnup will be incorporated into the database fields. The uncertainty data would be obtained from calculated and measured Transversing Incore Probe (TIP) data at various statepoints through the cycle. TIP measurements are taken at the plant between every 30 to 60 days. Available measured and calculated nodal TIP data will be used to determine the uncertainty in the fuel burnup. The availability of the data depends on the code package being used, and whether the utility calculates the TIP responses in the 3D simulator code used by the utility.

Once the BWR data have been made available in the database, they will be evaluated and uncertainty results reported consistent with the uncertainty data previously developed for PWRs. This may include using the Anderson-Darling method, if verified appropriate for these data, to test for normality at the 95% confidence level, calculating the mean and standard deviation for that data which passes the normality test, and determining the uncertainty of either “D” or “P” using the one-sided tolerance limit method at the 95% confidence and 95% probability levels. Data that does not pass the normality test will be further evaluated to determine if alternate tests can be used to allow its use in the uncertainty analysis.

Secondly, data needed to evaluate radial burnup gradients for BWR spent fuel assemblies will also be included. Pin-by-pin data to be assembled includes burnup (both “calculated” and “measured” if available) for various nodal heights, assembly-average burnups, and nodal burnup levels. The assembly type, number of cycles irradiated, and initial enrichment for each assembly will also be assembled as will the presence or absence of control blades and supplemental burnable poison rods. The assembled data will be compiled into a user-friendly, searchable database (e.g. ACCESS).

Initially, data for assemblies discharged from the Limerick 1, Susquehanna 1, Hope Creek, and Columbia Generating Station will be incorporated into the database. These reactors represent reactor classes BWR-4 and BWR-5 and BWR assembly types GE 8x8, GE 9x9, ANF 8x8, ANF 9x9, ANF 10x10, and ABB/Westinghouse 10x10. While data will be assembled for spent fuel assemblies having the variety of characteristics described previously, the primary effort will be focused on assembling data for those spent fuel assemblies believed to have the greatest radial burnup gradient (i.e. those assemblies that were positioned on two of four sides next to deeply inserted control blades for long periods during its life-cycle relative to other assemblies). Efforts will also be made to expand the database with data from additional BWR power plants having reactor types and fuel types different than the cited plants. The focus of these efforts will be to fill data gaps identified using data clustering/gap analysis tools developed by Battelle.

The pin by pin exposures for each node will be obtained from the utility 3D simulator code at the end of each cycle collected, if available. The availability would depend on whether the utility is using a modern nodal code that performs pin power reconstruction, or an older code that does not compute pin powers and only provides nodal data. The pin by pin exposures can also be obtained from the TRITON depletion calculations.

Once the data has been made available in the database, the data will be evaluated and summarized in a report in a similar manner to that performed for horizontal gradient in PWR spent fuel assemblies (i.e. deviation of burnup at various pin locations relative to assembly average burnup).

Thirdly, a database of BWR commercial reactor critical (CRC) data for performing CRC depletion and reactivity calculations will be assembled. The data should be sufficient to perform simple depletion calculations such as those using the SCALE 5 package, specifically the SAS2H and TRITON modules. The various fuel assemblies will be depleted by these modules through their unique operating histories such that the modified fuel compositions will be available at the specific exposure times corresponding to the approved statepoints. The fuel assembly depletion calculations will be based on detailed core follow information for each assembly. Detailed information for each assembly including burnup, fuel temperature, moderator density, fuel enrichment, gadolinium loading, fuel density, and control blade position will be included as available. This data includes the fuel lattice fuel enrichment and gadolinia loading distributions. Detailed axial exposure and moderator density/void history profile information for all assemblies will be collected and used to support both the TRITON and SAS2H calculations. The assembled data will be compiled into a user-friendly, searchable database (e.g. ACCESS).

CRC data will be assembled for spent fuel assemblies discharged from among available data from Limerick 1, Susquehanna 1, Hope Creek, and Columbia Generating Station. These reactors represent reactor classes BWR-4 and BWR-5 and BWR assembly types GE 8x8, GE 9x9, GE 10x10, ANF 8x8, ANF 9x9, ANF 10x10, and ABB/Westinghouse 10x10. Data from two critical measurement techniques could be available from these plants; in-sequence and local. When a BWR is started, the control blades are pulled in a predefined sequence until it goes critical. This is defined as an in-sequence critical. This is the typical type of critical performed at BWRs. Very rarely a special test is done called a local critical. When a local critical is performed, only 2 or 3 control blades adjacent to each other are pulled (as opposed to 40 to 60 blades that are arranged symmetrically around the core for an in-sequence critical). CRC data will only be assembled for one of these reactors and only after Battelle receives concurrence from the CONTRACTOR that the specific reactor chosen for this effort is acceptable. CRC data assembled will meet the following four criteria:

- contains no xenon
- accounts for samarium
- minimizes temperature effects
- represents (as a set) a reasonable portion of the spent nuclear fuel inventory.

3. CONCLUSION

The need for an organized database of spent fuel information has been recognized. This proposed database and the ability to query and evaluate the data for outliers, correlations and gaps should be extremely beneficial in the further development of a regulatory basis for the acceptance of burnup credit in the storage and transportation of spent nuclear fuel.

NUCLEAR DATA
AND VALIDATION OF
DEPLETION AND REACTIVITY CALCULATIONS

(Session 2.2)

Investigation of burnup and nuclide composition of spent nuclear fuel for use when solving "burnup credit" tasks — RIAR's experience

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Abstract. Accumulation of considerable quantity of spent nuclear fuel creates problems with its handling. There is a need for rational calculation of storing, transport and processing processes taking into account safety and economy issues by means of fuel burnup for system safety control. In order to estimate rationally spent fuel reactivity it is necessary to use accurate data on burnup, nuclide composition including heavy isotopes and fission products. Radiochemical method is applied as a basic one for obtaining this data. This method has been used by RIAR, who has many years' experience in spent nuclear fuel investigation, for the spent fuel investigation for Lawrence Livermore National Laboratory, USA, ISTC project No. 2670p. The investigation was performed in order to obtain experimental data for verification of calculation programs and codes for spent VVER-440 fuel handling systems, and included solving of the following tasks:

- methodology of selection of fuel assembly (FA), fuel rods and specimens for investigation;
- neutron-physical calculation of FA irradiation parameters;
- fuel rods gamma-spectrometry and selection of specimens for radiochemical investigation;
- radiochemical investigation of nuclide composition;
- burnup depth evaluation;
- analysis of the results.

1. Methodology of selection of FA, fuel rods and specimens for investigation.

A major share of electric power is produced at the nuclear power plants (NPPs) operating the VVER-440 reactors in the Eastern Europe, FSU countries and Russia. A lot of spent FAs appeared in the course of their operation. Their handling results in considerable expenditures for the nuclear power industry in the countries operating the VVER-440 power units. FAs at 3.6 % enrichment ^{235}U, which had been operated up to an average burnup of ~40 MW·day/kg$U_{initial}$, were used at the first stage of the VVER-440 fuel cycle. Spent FAs with the above-given characteristics represent a massive accumulation of nuclear fuel that is stored in the NPP storage pools. The FA, fuel of which is subjected to investigation, has to have uniform burnup, not to be located close to the fuel assemblies of control group of protection control means during all fuel cycles and not to contain unsealed fuel rods. From the above reasoning, FA No.13626135 (hereinafter referred as FA No. 135) operated up to an average burnup of 38.5 MW·day/kgU in the 4th power unit of Novovoronezh NPP was selected for investigation. The FA had regular design, regular fuel and was manufactured at Joint-Stock Company Machinery Building Plant.

In order to obtain data on nuclides distribution and fuel burnup depth throughout the FA axis and radius for radiochemical investigation not less than four fuel rods were selected. They met the following requirements:

- one fuel rod at a maximum fuel burnup level;
- one fuel rod at a minimal fuel burnup level;
- the other two should be at an intermediated burnup level;
- all the fuel rods should be located on the same diagonal of FA.

Three specimens are cutout throughout the height of fuel rods at a maximum and minimal burnups. One specimen is cutout from the central part of fuel column of two other fuel rods. A total of eight fuel specimens those burnup varies significantly is selected for radiochemical investigation. Fuel rods Nos. 65, 67, 68, 69 meeting the above requirements were selected from FA No. 135 for radiochemical investigation.

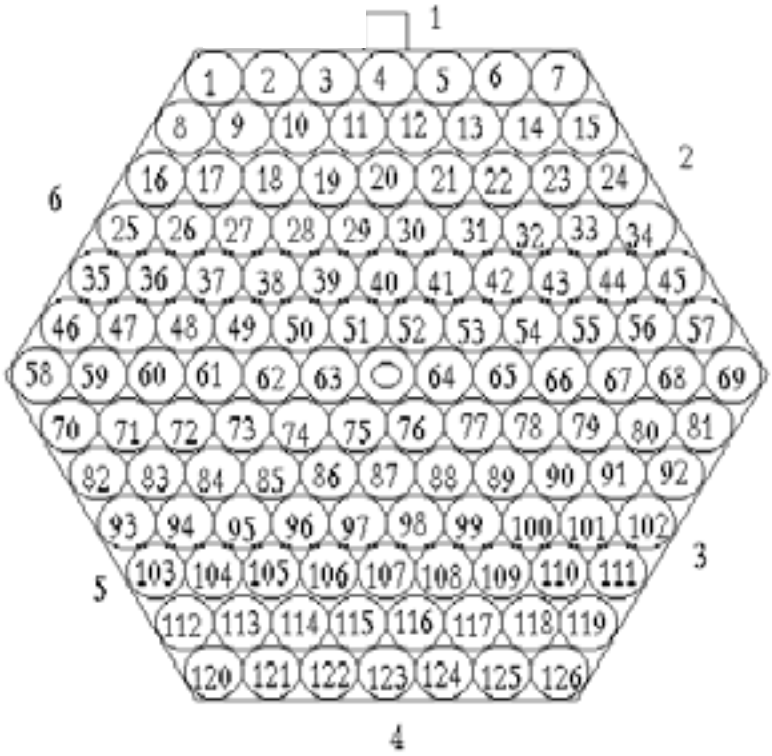

Fig. 1.1 Schematic arrangement of fuel rods in FA No. 135.

FA No. 135 was operated as a member of 15–18 core loads in the forth power unit of Novovoronezh NPP.

Table 1.1 and Figure1.2 show fuel cycle data and position of the FA in the reactor core.

The analysis of data given in Table 1.1 and Fig. 1.2 reveals that FA No. 135 was operated at the same burnup rate during all the fuel cycles. It was not located close to the fuel assemblies of control group of protection control means during all the cycles. According to the results of cladding integrity control in the 4th power unit of Novovoronezh NPP and post-irradiation experiment data, fuel assembly didn't include unsealed fuel rods.

TABLE 1.1: DURATION OF FUEL CYCLES AND FA NO. 135 MOVEMENT DIAGRAM

Fuel cycle No.	**15(1)**	**16(2)**	**17(3)**	**18(4)**
Beginning End	10.87 08.88	10.88 12.89	01.90 05.90	07.90 7.91
Effective days	296	389.8	98.2	325.1
Average burnup in a fuel cycle, MWd/kgU	11.0	14.5	3.5	9.5
Average burnup at the end of fuel cycle, MWd/kgU	11.0	25.5	29.0	38.5
Burnup rate, $\frac{\text{MWd/kgU}}{\text{effective days}}$	0.037	0.037	0.036	0.029
FA coordinate in the reactor core	21–38	19–40	19–40	16–45

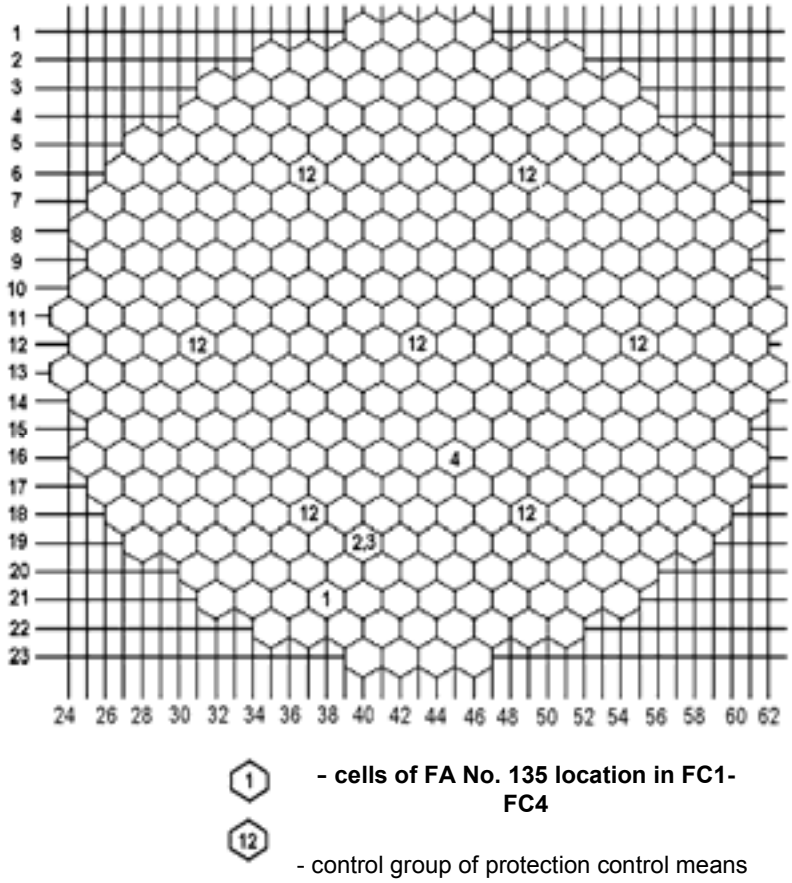

Fig. 1.2 Location of FA No. 135 in the core of the 4th power unit Novovoronezh NPP.

2. Neutron-physical calculation of FA No. 135 irradiation parameters.

Neutron and physical calculations were made for each fuel rod of the FA using program system BIPR-7A – PERMAK-A in RSC "KI". The following results were found:

- distribution of fuel burnup depth throughout height;
- distribution of fast neutron fluence;
- linear power distribution;
- cladding temperature values,
- linear power values.

Table 1.2 shows calculation values of burnup for six fuel rods at the end of irradiation.

A satisfactory agreement between the calculated and experimental data on burnup depth determination was revealed. The experimental data on burnup depth determination were obtained using nondestructive measurement method presented below.

TABLE 1.2: CALCULATED VALUES OF FUEL BURNUP THROUGHOUT THE HEIGHT OF FUEL RODS AT THE END OF IRRADIATION

Core Node No.	Burnup, MWd/kgU					
	fuel rod 6	fuel rod 16	fuel rod 32	fuel rod 54	fuel rod 82	fuel rod 116
25	19.0	18.7	18.9	19.6	20.6	22.2
24	25.2	24.8	25.1	25.9	27.2	29.1
23	30.3	29.8	30.1	31.1	32.5	34.6
22	33.6	33.3	33.6	34.5	36.1	38.3
21	35.9	35.6	35.9	36.9	38.4	40.7
20	37.8	37.3	37.6	38.7	40.2	42.6
19	39.2	38.7	39.0	40.1	41.9	44.2
18	40.5	40.1	40.5	41.8	43.5	45.8
17	41.5	41.2	41.5	42.7	44.6	47.0
16	42.3	41.9	42.3	43.5	45.4	47.9
15	48.8	42.2	42.8	44.0	45.9	48.2
14	43.1	42.5	43.1	44.3	46.2	48.6
13	43.3	42.8	43.3	44.5	46.5	48.8
12	43.5	43.1	43.5	44.7	46.7	49.0
11	43.7	43.3	43.7	44.8	46.8	49.2
10	43.8	43.4	43.8	45.0	46.9	49.3
9	43.8	43.4	43.8	45.0	47.0	49.6
8	43.8	43.4	43.8	45.0	47.0	49.6
7	43.7	43.3	43.7	44.9	46.8	49.4
6	43.3	42.7	43.3	44.4	46.4	48.9
5	42.1	41.7	42.1	43.4	45.2	47.8
4	40.1	39.7	40.1	41.4	43.2	45.8
3	36.7	36.2	36.7	37.9	39.6	42.1
2	30.7	30.3	30.7	31.7	33.4	35.7
1	22.0	21.8	22.1	22.8	24.2	26.0

3. Gamma-spectrometry of fuel rods. Selection of specimens.

Gamma-spectrometry of fuel rods was performed to obtain data on axial and radial distribution of fission products and fuel burnup, selection of fuel rods and coordinates.

TABLE 1.3: MAXIMUM AND AVERAGE FUEL BURNUP AND PEAKING FACTORS (KZ) OF FISSION PRODUCT DISTRIBUTION ALONG FUEL RODS OF FA NO. 135

Fuel rod No.	**Maximum burnup, MWd/kgU**	**Kz (^{137}Cs)**	**Average burnup, MWd/kgU**
1	48.4	1.16	41.6
7	48.2	1.16	41.5
9	45.1	1.17	38.6
14	45.0	1.16	38.7
18	43.1	1.16	37.0
22	43.2	1.16	37.1
28	42.2	1.17	36.1
31	42.5	1.16	36.5
39	42.2	1.16	36.2
41	42.7	1.16	36.8
51	43.2	1.17	36.9
52	43.2	1.16	37.1
58	48.7	1.15	42.3
59	45.8	1.16	39.3
60	43.9	1.16	37.7
61	42.6	1.16	36.6
62	42.8	1.16	36.8
63	43.8	1.16	37.6
64	43.5	1.16	37.4
65	**42.3**	**1.16**	**36.4**
66	42.5	1.16	36.5
67	**43.8**	**1.16**	**37.6**
68	**45.7**	**1.16**	**39.3**
69	**49.3**	**1.17**	**42.2**
75	43.3	1.16	37.3
76	43.6	1.17	37.4
86	42.6	1.16	36.6
88	42.4	1.17	36.1
96	42.8	1.16	36.8
99	42.6	1.17	36.6
105	44.4	1.16	38.2
109	43.3	1.17	37.2
113	45.7	1.16	39.3
118	44.8	1.16	38.5
120	49.1	1.15	42.8
126	47.9	1.16	41.4
Minimum	42.2	1.15	36.1
Maximum	49.3	1.17	42.8
Average	44.3	1.16	38.1
Mean-square deviation	2.18	0.005	1.97

Results of measurement of the selected specimens are marked with dark color.

of specimen selection for radiochemical investigation. Table 1.3 gives the experimental values for axial power peaking factor (Kz) of the ^{137}Cs-fission product accumulation as well as maximum and average burnup values in fuel rods of the FA.

Figure 1.3 shows distributions of average fuel burnup in diagonal fuel rods of the FA.

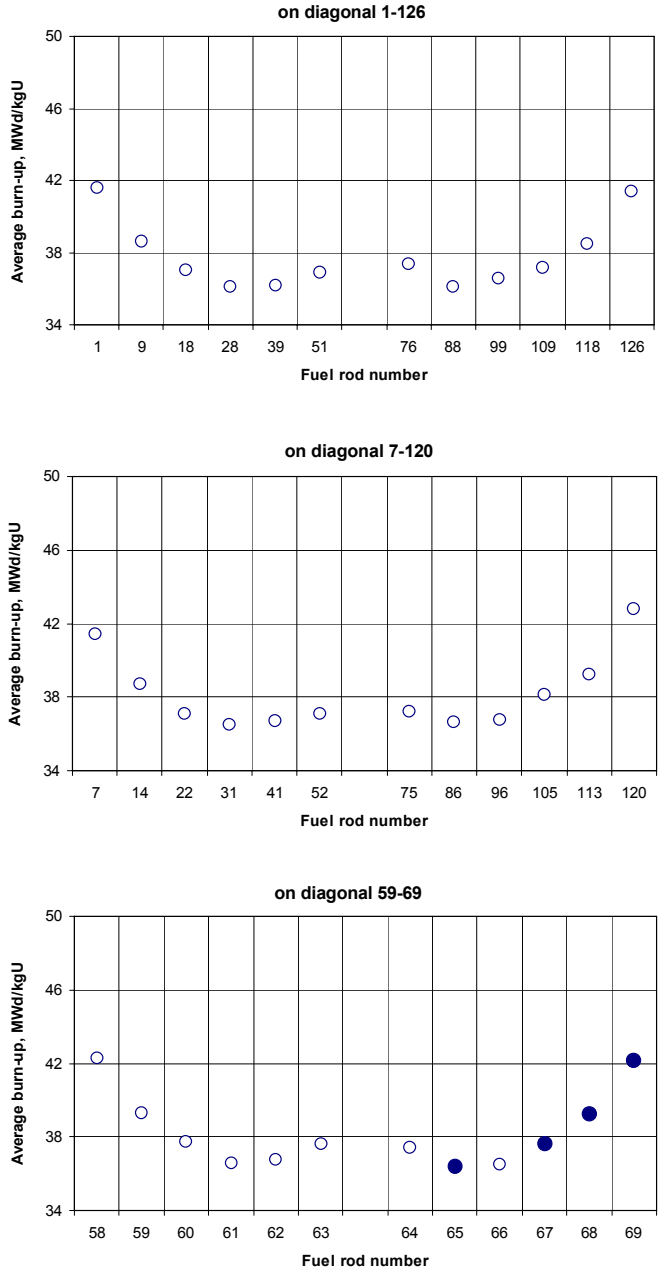

Fig. 1.3. Distribution of the average burnup in diagonal directions of FA No.13626135 fuel rods.

A radial burnup peaking factor (K_r) equals to 1.15. Fuel rod No. 69 achieved a maximum burnup. Fuel rod No. 65 achieved a minimal burnup. An intermediate fuel burnup level was achieved in fuel rods No. 67 and No. 68.

Figures 1.4 and 1.5 demonstrate the dismantling procedures of fuel rods for cutting out specimens and a location of specimens selected along fuel rods relative to the profile of fission product distribution throughout the height.

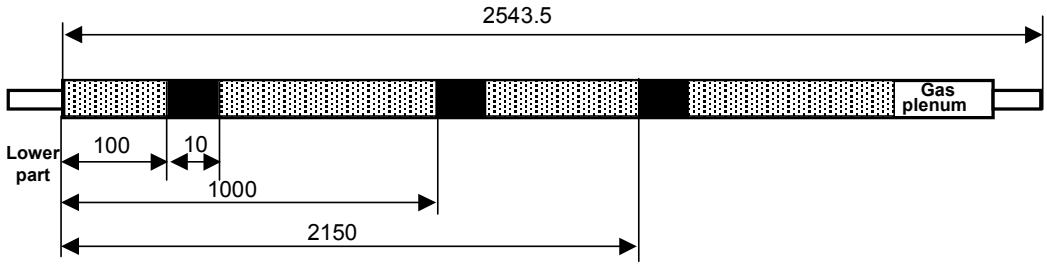

Fig. 1.4 Cutting procedure of specimens from fuel rods No. 69 in FA No. 135.

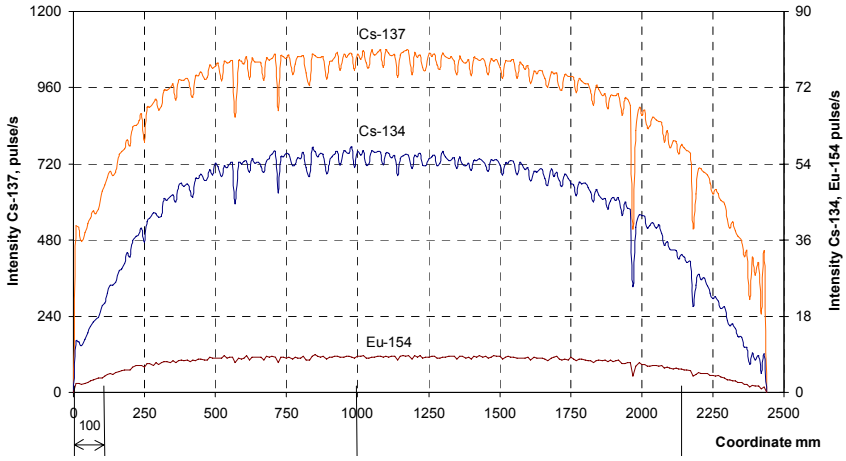

Fig. 1.5 Distribution of fission product throughout the height of fuel rod No. 69, FA No. 135 (Internal collimator– 3 mm, outer one – 5 mm, pitch – 10 mm).

The cutout fuel specimens were placed in Al-containers whose covers had numbers. Table 1.4 shows marking of the Al-containers with the corresponding fuel specimens. The Al-containers loaded with the fuel specimens were sent to radiochemical laboratory for radiochemical investigation.

TABLE 1.4: MARKING OF FUEL SPECIMENS CUTOUT FROM FUEL RODS OF FA NO. 135

Fuel rod No.	Coordinate from the lower part of fuel rod, mm	Number of Al-container with a fuel specimen
65	100	182
	1000	21
	2150	69
67	1000	149
68	1000	162
69	100	135
	1000	79
	2150	57

4. Radiochemical investigation of nuclide composition.

We have developed a general scheme of irradiated fuel analysis (Fig. 1.6) to evaluate nuclide composition of heavy atoms and fission products.

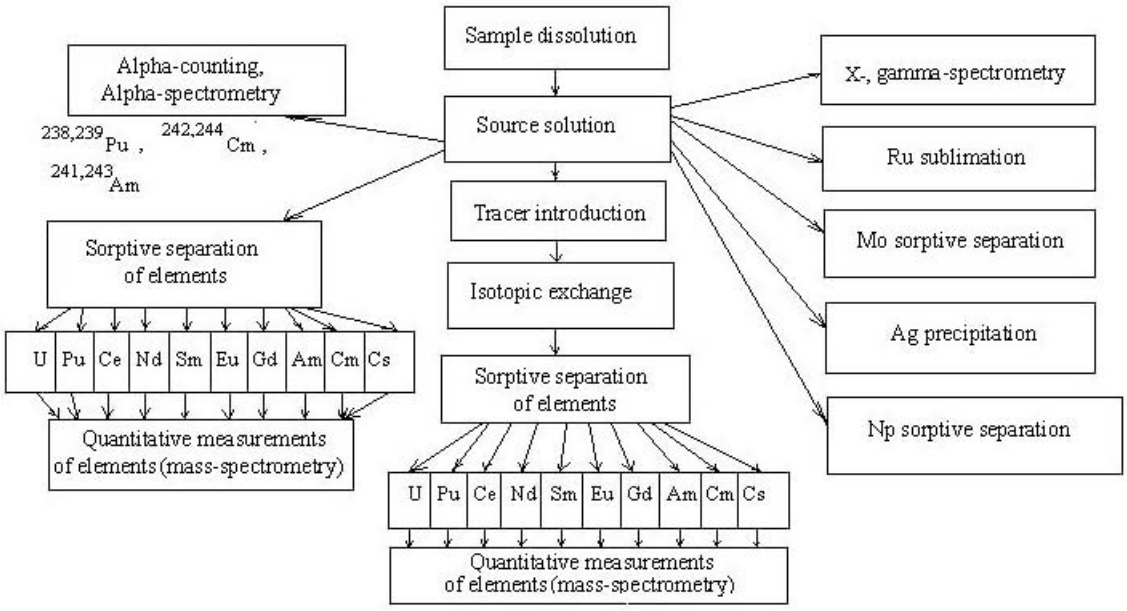

Fig. 1.6 General scheme of irradiated fuel analysis.

This scheme includes the following procedures:

- dissolution of the sample to be analyzed;
- calculation of solution volume required for analysis;
- radiochemical extraction of U, Pu, Ce, Nd, Sm, Eu, Gd, Am, Cm and Cs for mass-spectrometer measurements of isotope composition;
- repeated radiochemical extraction of these elements in the presence of the complex tracer for quantitative measurements of elements;
- individual extraction of Np, Tc, Ag, Mo, Ru and Pd in accordance with individual techniques.

4.1. Specimen dissolution

The irradiated fuel specimen weighing as much as 2 up to 10 g is placed into a beaker flask with a volume of 150-200 cm^2. HNO_3 solution at 8 mole/l concentration with an addition of hydrofluoric acid (0.1 mole/l) is used for dissolution in poor boiling of the solution.

4.2. Radiometric measurements

A sample is taken from the initial solution after dissolution of fuel specimen to evaluate the total α-activity and measure α-, X-, γ-spectra. The results of measurements are used as the basis for estimation of the solution quantity necessary for analysis.

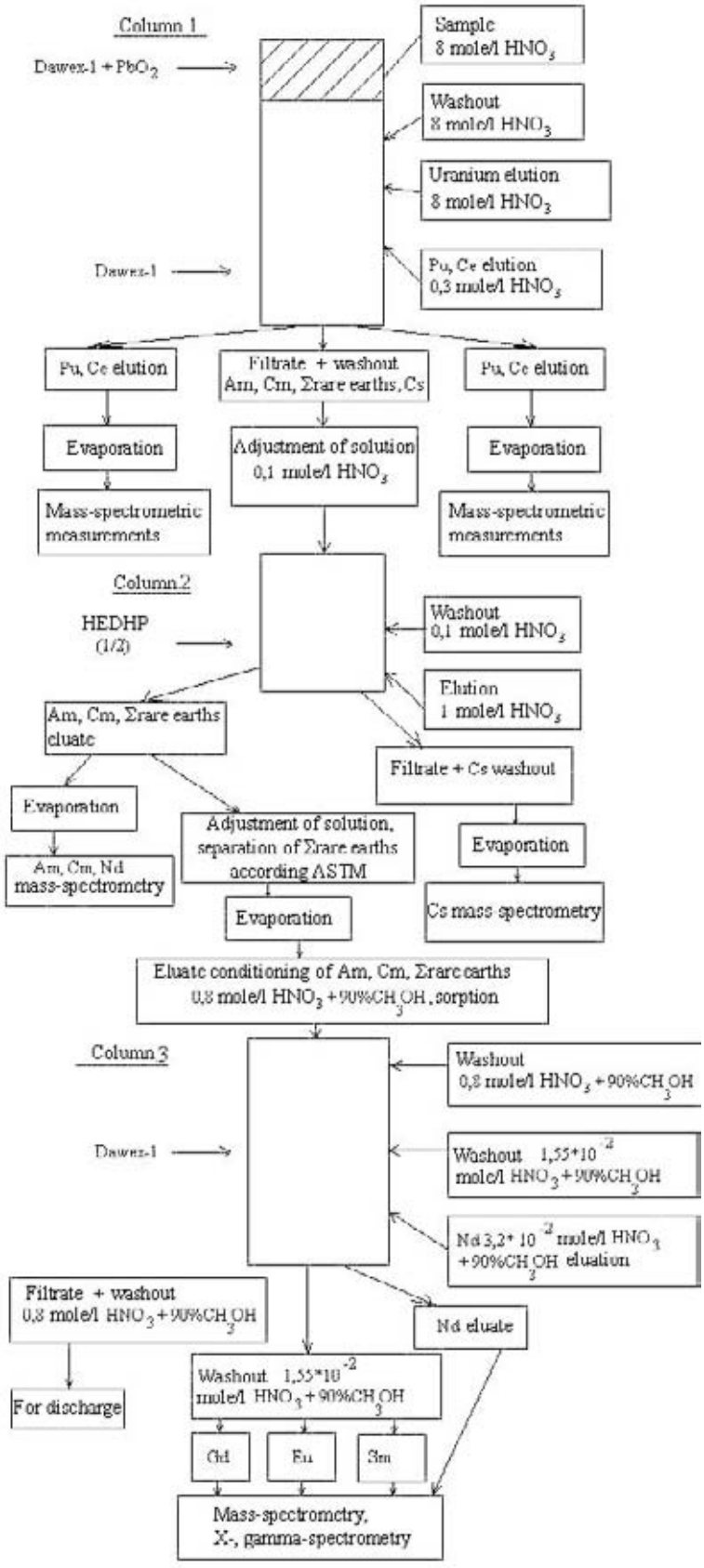

Fig. 1.7 Scheme of radiochemical extraction.

4.3. Radiochemical analysis, element mass determination

Figure 1.7 represents a radiochemical extraction scheme for U, Pu, Am, Cm, Cs, Ce, Nd, Sm, Eu, Gd for mass-spectrometer measurement of isotope composition.

Individual extraction of Np, Tc, Ag, Mo, Ru, Pd for isotope composition measurement was carried out by means of special complex methods.

Method of isotope dilution with mass-spectrometer ending of analysis was used as a main method of measurement of nuclide content in fuel of the specimens. The method error didn't exceed 1.5 – 3 % and was mainly determined by error of standard samples using as tracers during isotope dilution. When measuring content of U, Pu, Am, Cm, Nd, Ce the standard solutions of these elements prepared in Radium Institute (Saint Petersburg) were used as tracers. When measuring content of Gd, Eu, Sm, Cs, Mo, Ru, Ag, Pd the standard solutions of these elements prepared from the specimens of natural isotope composition were applied as tracers. When the ^{99}Tc content was measured, a standard solution was made of technetium specimen where the ^{98}Tc portion was not less than 12 %. Such technetium specimen was produced according to the reaction (n, 2n) during ^{99}Tc irradiation in the nuclear reactor.

Although the analytical capabilities of this procedure are good, the data on ^{101}Ru content are of evaluation character only. The reason is that some ruthenium can be lost in dissolution of fuel specimen due to a very high volatility of this element. It was difficult to process the measurement results because of the complex chemical composition that results in difficulties associated with chemical separation of elements and presence of a great number of isobar lines in the mass region. That's why we consider the obtained mass values of ^{95}Mo, ^{99}Tc, ^{109}Ag and ^{105,108}Pd to be evaluation data. The ^{103}Rh mass fraction wasn't measured because of the fact that its portion is 100% in the natural isotope composition. ^{242}Cm, ^{243}Cm and ^{244}Pu were not also analyzed. ^{242}Cm was nearly subjected to a total decay because there were 28 half lives of ^{242}Cm from the date of the FA discharge from the reactor and radiochemical analysis. ^{243}Cm and ^{244}Pu contents in fuel of specimens were lower than the detection limits of the techniques used for their measurements.

Results of measurements given for the end of the fuel irradiation period are included in Tables 1.5–1.12.

TABLE 1.5: MASS FRACTION OF URANIUM ISOTOPES (kg FP/тU$_{initial}$)

Specimen No.	^{234}U	^{235}U	^{236}U	^{238}U
21	0,189(47)	7,65(28)	5,19(28)	932,05(37)
149	0,189(47)	6,989(94)	5,289(94)	931,99(38)
162	0,188(46)	6,12(19)	5,37(19)	930,45(35)
79	0,094(68)	4,13(19)	5,26(19)	930,36(35)
57	0,105(47)	9,52(47)	4,85(19)	937,40(40)
135	0,240(38)	12,87(19)	4,419(96)	943,05(42)
182	0,135(36)	16,06(19)	4,06(19)	945,59(42)
69	0,172(46)	11,77(28)	4,692(95)	940,86(42)

TABLE 1.6: MASS FRACTION OF PLUTONIUM ISOTOPES (kg FP/тU$_{initial}$)

Specimen No.	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
21	0,227(24)	6,308(14)	2,686(10)	1,708(12)	0,728(5)
149	0,228(23)	6,246(9)	2,857(9)	1,768(20)	0,830(9)
162	0,280(28)	6,019(8)	2,915(12)	1,775(20)	0,928(12)
79	0,329(33)	5,785(7)	3,025(13)	1,566(12)	0,965(14)
57	0,164(16)	6,019(8)	2,361(15)	1,489(12)	0,557(9)
135	0,0884(88)	5,315(10)	1,883(16)	1,070(5)	0,314(4)
182	0,0733(73)	5,769(9)	1,597(8)	0,903(6)	0,1876(85)
69	0,143(14)	6,408(11)	2,255(13)	1,382(12)	0,413(11)

TABLE 1.7: MASS FRACTION OF TRANSPLUTONIUM ELEMENTS ISOTOPES (kg FP/тU$_{initial}$)

Specimen No.	^{237}Np	^{241}Am	^{242m}Am $\cdot 10^3$	^{243}Am	^{244}Cm	^{245}Cm $\cdot 10^2$	^{246}Cm $\cdot 10^4$
21	1,43 (10)	0,0650 (17)	1,45 (10)	0,1435 (87)	0,0718 (36)	0,473 (9)	4,2 (4)
149	1,46 (10)	0,0672 (14)	1,09 (10)	0,1418 (90)	0,0877 (44)	0,378 (7)	4,5 (4)
162	1,49 (10)	0,0918 (61)	0,50 (6)	0,1576 (63)	0,1222 (61)	0,722 (14)	5,8 (6)
79	1,59 (11)	0,1627 (81)	1,03 (10)	0,266 (13)	0,1888 (94)	1,056 (21)	6,4 (6)
57	1,110 (78)	0,0557 (28)	1,05 (8)	0,1032 (52)	0,0486 (24)	0,135 (3)	-
135	0,780 (56)	0,0380 (19)	0,85 (5)	0,0412 (20)	0,01348 (67)	0,052 (1)	0,68 (7)
182	0,710 (50)	0,0446 (27)	1,04 (10)	0,0243 (12)	0,00675 (33)	0,0250 (5)	0,21 (2)
69	1,030 (71)	0,0486 (24)	1,16 (10)	0,0702 (42)	0,0224 (11)	0,138 (3)	1,32 (13)

TABLE 1.8: MASS FRACTION OF NEODYMIUM (kg FP/тU$_{initial}$)

Specimen No.	^{142}Nd	^{143}Nd	^{144}Nd	^{145}Nd	^{146}Nd	^{148}Nd	^{150}Nd
21	0.0294 (9)	0.965 (9)	1.043 (10)	0.898 (9)	0.821 (8)	0.473 (5)	0.228 (2)
149	0.0318 (9)	0.969 (9)	1.080 (11)	0.912 (9)	0.840 (8)	0.482 (5)	0.232 (2)
162	0.0339 (9)	0.981 (9)	1.137 (11)	0.937 (9)	0.874 (8)	0.499 (5)	0.241 (2)
79	0.0386 (9)	0.992 (10)	1.284 (13)	0.986 (9)	0.947 (9)	0.526 (5)	0.246 (2)
57	0.0206 (6)	0.864 (8)	0.869 (8)	0.777 (8)	0.695 (7)	0.400 (4)	0.187 (2)
135	0.0155 (5)	0.779 (8)	0.690 (7)	0.665 (6)	0.579 (6)	0.336 (3)	0.1570 (15)
182	0.00775 (23)	0.618 (6)	0.490 (5)	0.506 (5)	0.430 (4)	0.252 (2)	0.116 (1)
69	0.0131 (5)	0.797 (8)	0.733 (7)	0.692 (7)	0.608 (6)	0.354 (3)	0.166 (2)

TABLE 1.9: MASS FRACTIONS OF CESIUM AND CERIUM ISOTOPES (kg FP/тU$_{initial}$)

Specimen No.	^{133}Cs	^{134}Cs	^{135}Cs	^{137}Cs	^{140}Ce	^{142}Ce	^{144}Ce
21	1,328 (46)	0,117 (6)	0,476 (19)	1,490 (50)	1,540 (50)	1,407 (50)	0,634 (51)
149	1,416 (50)	0,155 (7)	0,488 (19)	1,596 (50)	1,558 (50)	1,423 (50)	0,642 (52)
162	1,471 (51)	0,128 (6)	0,480 (18)	1,675 (50)	1,640 (51)	1,498 (60)	0,674 (55)
79	1,462 (50)	0,127 (6)	0,445 (16)	1,680 (67)	1,729 (60)	1,585 (61)	0,718 (56)
57	1,253 (43)	0,135 (6)	0,449 (16)	1,323 (53)	1,285 (53)	1,178 (50)	0,533 (50)
135	1,078 (38)	0,142 (7)	0,415 (17)	1,100 (44)	1,095 (44)	1,004 (50)	0,458 (37)
182	0,848 (34)	0,109 (5)	0,389 (16)	0,850 (40)	0,832 (40)	0,764 (38)	0,350 (35)
69	1,110 (36)	0,174 (8)	0,460 (17)	1,178 (47)	1,149 (46)	1,053 (53)	0,477 (38)

TABLE 1.10: MASS FRACTION OF SAMARIUM ISOTOPES (kg FP/$tU_{initial}$)

Specimen No.	147	148	149	150	151	152	154
182	0.2543 (46)	0.0896 (16)	0.00309 (11)	0.2053 (37)	0.01122 (20)	0.0852 (15)	0.0233 (12)
135	0.2565 (47)	0.1178 (25)	0.00299 (11)	0.2405 (40)	0.01110 (20)	0.0922 (16)	0.0287 (15)
69	0.2630 (47)	0.1272 (23)	0.00306 (10)	0.2544 (46)	0.01131 (21)	0.0960 (17)	0.0306 (17)
57	0.2692 (50)	0.1551 (28)	0.00304 (10)	0.2907 (50)	0.01139 (22)	0.1039 (18)	0.0362 (20)
21	0.2776 (50)	0.1932 (33)	0.00305 (10)	0.3399 (55)	0.01148 (25)	0.1148 (20)	0.0435 (22)
149	0.2980 (52)	0.2103 (38)	0.00319 (13)	0.3686 (66)	0.01223 (22)	0.1240 (22)	0.0472 (24)
162	0.2882 (52)	0.2208 (41)	0.00312 (10)	0.3777 (60)	0.01171 (22)	0.1237 (22)	0.0490 (26)
79	0.2804 (51)	0.2315 (42)	0.00294 (10)	0.3878 (70)	0.01122 (20)	0.1243 (22)	0.0507 (25)

TABLE 1.11: MASS FRACTION OF EUROPIUM ISOTOPES (kg FP/$tU_{initial}$)

Specimen No.	151	153	154	155
182	0.000460 (18)	0.0199 (10)	0.00289 (14)	0.00125 (11)
135	0.000429 (16)	0.0224 (11)	0.00366 (18)	0.00135 (12)
69	0.000409 (16)	0.0222 (11)	0.00370 (18)	0.00133 (12)
57	0.000338 (14)	0.0217 (11)	0.00390 (19)	0.00127 (11)
21	0.000283 (12)	0.0222 (11)	0.00431 (21)	0.00127 (11)
149	0.000268 (12)	0.0214 (11)	0.00416 (21)	0.00123 (11)
162	0.000228 (10)	0.0205 (10)	0.00412 (21)	0.00115 (10)
79	0.000153 (6)	0.0152 (7)	0.00314 (16)	0.00085 (7)

TABLE 1.12: MASS FRACTION OF NUCLIDES-FISSION PRODUCTS (kg FP/$tU_{initial}$)

Specimen No.	^{95}Mo	^{99}Tc	^{101}Ru	^{105}Pd	^{108}Pd	^{109}Ag	^{155}Gd
182	0.533 (96)	0.55 (10)	0.531 (96)	0.261 (47)	0.064 (12)	0.0299 (84)	0.000746 (63)
135	0.69 (12)	0.72 (13)	0.70 (13)	0.348 (63)	0.091 (16)	0.038 (11)	0.000960 (82)
69	0.72 (13)	0.75 (14)	0.73 (13)	0.364 (66)	0.099 (17)	0.041 (11)	0.001000 (85)
57	0.78 (14)	0.82 (15)	0.82 (15)	0.439 (79)	0.115 (21)	0.046 (13)	0.001140 (97)
21	0.92 (17)	0.96 (17)	0.98 (18)	0.529 (96)	0.147 (26)	0.054 (15)	0.00134 (12)
149	0.93 (17)	0.96 (17)	0.99 (18)	0.552 (99)	0.155 (28)	0.054 (15)	0.00140 (13)
162	0.97 (17)	1.00 (18)	1.04 (19)	0.59 (11)	0.171 (31)	0.055 (16)	0.00143 (13)
79	1.01 (18)	1.02 (18)	1.08 (19)	0.58 (11)	0.166 (30)	0.051 (14)	0.00142 (13)

5. Burnup depth evaluation.

Burnup depth that is defined as a ratio of heavy nuclei fissioned during irradiation to their total initial amount in fuel is calculated in two ways:

- based on the isotopic composition of fuel that is measured before and after irradiation by means of expressions (heavy atoms method);
- based on fission product accumulation.

5.1. Heavy atoms method (HAM).

To evaluate burnup we used algorithm of international standard ASTM E244-80 [1,2]. According to this algorithm total burnup is as follows:

$$F_T = F_5 + F_8 + F_9 + F_1,$$

where F_5, F_8, F_9, F_1 are burnup ^{235}U, ^{238}U, ^{239}Pu, ^{241}Pu accordingly. F_5, F_8, F_9, F_1 were found by means of calculation expressions of the given analysis. Table 1.13 represents values of nuclear constants used during the calculation.

TABLE 1.13: USED NUCLEAR CONSTANTS

Para-meter	Neutrons per fission, ν_i [4]	Capture cross section - fission cross section ratio, α_i [4]	$Y_{145+146Nd}$, % [3]	Y_{137Cs}, % [3]	σ_i, barn [1]
^{235}U	2.4251(17)	0.1687(7)	6.82±0.15	6.27±0.12	556
^{238}U	1.980(15)	0.089(1)	6.70±0.16	6.33±0.12	
^{239}Pu	2.877(3)	0.5870(16)	5.66±0.55	6.48±0.77	
^{241}Pu	2.937(3)	0.4200(28)	5.79±0.17	6.52±0.39	1550.0
^{236}U	-	-	-	-	85
ε	1.046 [1]				
λ_1,c	$1.53 \cdot 10^{-9}$[1]				

Values of total burnup F_T, fissile nuclide burnup F_i and their fission fractions q_i calculated according to HAM are given in Table 1.14.

TABLE 1.14: BURNUP of ^{235}U, ^{238}U, ^{239}Pu and ^{241}Pu ISOTOPES AND THEIR FISSION FRACTIONS IN TOTAL BURNUp

Parameter	Fuel rod No.65 Specimen No. 21		Fuel rod No.67 Specimen No.149		Fuel rod No.68 Specimen No. 62		Fuel rod No.69 Specimen No. 79	
	Burnup kg/t U	Fission fraction q_I, %	Burnup kg/t U	Fission fraction q_I, %	Burnup kg/t U	Fission fraction q_I, %	Burnup kg/t U	Fission fraction q_I, %
F_5	22,07	53,66	22,55	53,71	23,22	53,03	25,05	54,51
F_8	4,52	11,00	4,62	11,00	4,82	11,02	5,05	11,00
F_9	12,67	30,82	12,83	30,56	13,52	30,89	13,55	29,48
F_1	1,86	4,52	1,98	4,71	2,21	5,06	2,30	5,01
F_T	41,12		41,98		43,77		45,95	
Fluence, cm^{-2}, 10^{21}	2,70		2,86		3,09		3,995	
Flux, 10^{13} $cm^{-2} s^{-1}$	2,819		2,986		3,229		3,828	
$F_{weighted}$ FP, kg/t U	41,5		41,9		44,2		46,3	

Values of total burnup measured per fission products are shown in the last line of the table. We can see a good conformity of the burnup evaluation results obtained by two methods in the table.

5.2. Method of evaluation per fission products.

Stable products of neodymium isotope fission (^{148}Nd and $^{145+146}$Nd) and long-lived isotope ^{137}Cs were used as burnup monitors. These isotopes aren't included into the chemical composition of unirradiated fuel, and their precursors in the radioactive-decay chain are short-lived nuclides, have small neutron radiative capture cross-sections.

Burnup F_{FP} evaluation was carried out according to the formula:

$$F_{FP} = \frac{1000 \cdot N_M / (Y_{effec} \cdot N_U)}{1 + N_{Pu} / N_U + N_{Np} / N_U + N_{TPE} / N_U + N_M / (Yeffec \cdot N_U)} \cdot \frac{M_C}{M_O},$$

where N_U, N_{Pu}, N_{Np}, N_{TPE}, N_M – experimentally measured numbers of atoms of uranium, plutonium, neptunium, transplutonium elements and burnup monitor per unit mass of analyzed solution of irradiated fuel.

M_C, M_O – average masses of fissionable nucleuses and nucleuses of initial uranium.

Y_{effec} - effective yield of burnup monitor calculated according to the formula [2]:

$$Y_{effec} = \sum_{i=1}^{n} q_i \cdot Y_i$$

where Y_i – cumulative yield of burnup monitor during *i*-nucleus fission,

q_i – contribution (fraction) of *i*-fission nucleus to total number of fission (Table 1.14),

n – number of fission nuclides, usually *n* = 4 (^{235}U, ^{238}U, ^{239}Pu, ^{241}Pu) for fuel of thermal reactors.

The measured burnup values are given in Table 1.15.

TABLE 1.15: FUEL BURNUP (kg FP/т $U_{initial}$)

Specimen No.	$^{145+146}$Nd	^{148}Nd	^{137}Cs	Weighted average
21	41,5 (12)	41,55 (83)	41,6 (15)	41,5 (15)
149	41,6 (14)	42,02 (84)	42,1 (16)	41,9 (15)
162	43,8 (15)	44,27 (88)	44,2 (16)	44,2 (14)
79	46,2 (16)	46,24 (94)	46,6 (17)	46,3 (14)
57	36,2 (12)	36,08 (72)	36,4 (14)	36,2 (12)
135	30,25 (91)	30,06 (60)	29,20 (89)	29,90 (91)
182	23,12 (76)	22,65 (45)	23,23 (77)	22,86 (72)
69	31,4 (11)	31,11 (62)	31,9 (11)	31,32 (98)

As it is shown in the Table, range of measured burnup values is 22.86 – 46.30 kg FP/т $U_{initial}$.

6. Analysis of the results.

Dependencies of fission nuclide contribution to fuel burnup on burnup are given in Fig. 1.8.

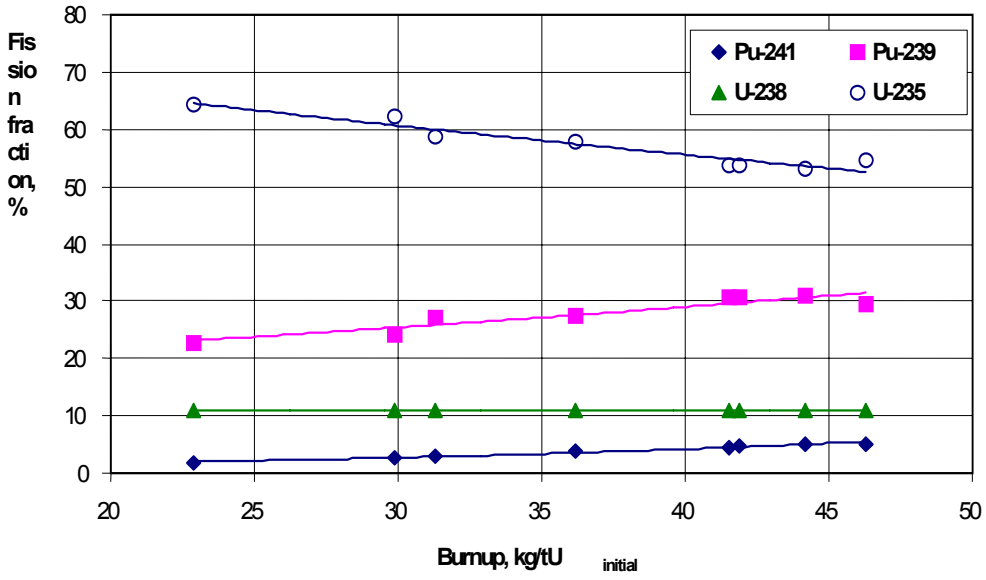

Fig.1.8 Fissile nuclide contribution to fuel burnup

^{238}U contribution to fuel burnup is insignificant and it makes up 10-11% within the examined burnup range. As the burnup level increases, the ^{235}U contribution to total burnup essentially decreases. ^{235}U contribution makes up 65% at the beginning of the range and the one (46.3 kg FP/т $U_{initial}$) is about 54% at the end of the range. ^{239}Pu contribution is about 30%, ^{241}Pu one is about 5%.

Transplutonium elements accumulation as a function of fuel burnup is shown in Fig. 1.9.

^{145}Nd/^{146}Nd ratio change as a function of fuel burnup is given in Fig. 1.10.

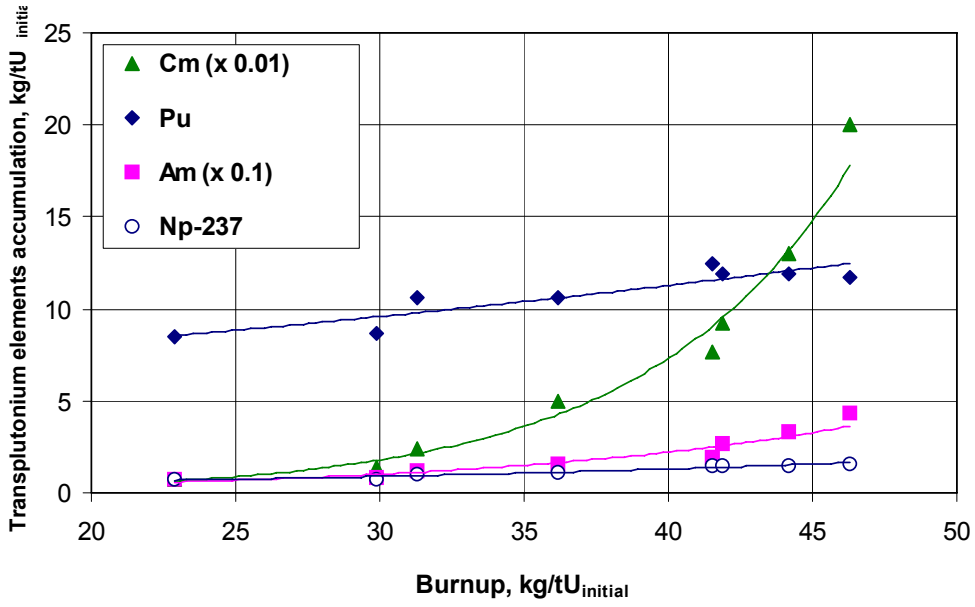

Fig. 1.9 Transplutonium elements accumulation as a function of fuel burnup.

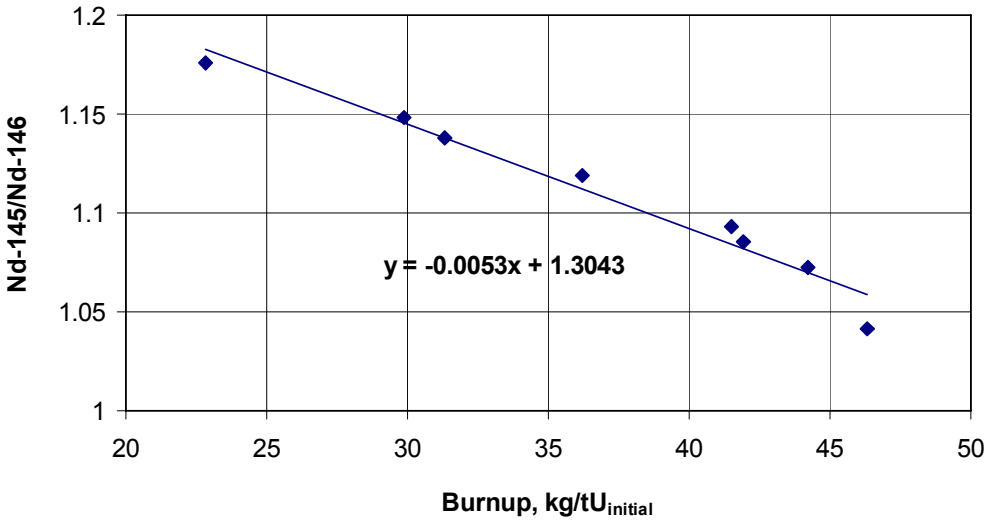

Fig. 1.10 Nd-145/Nd-146 ratio change as a function of fuel burnup.

It is possible to use the $^{145}Nd/^{146}Nd$ ratio for the burnup evaluation due to the linear character of the relationship shown in Fig. 1.10.

7. Conclusions

Isotope compositions of U, Pu, Am, Cm, Nd, Eu, Cs, Ce, Sm, masses of these elements and ^{99}Tc, ^{95}Mo, ^{101}Ru, ^{105}Pd, ^{108}Pd , ^{109}Ag, ^{155}Gd (by isotope dilution method) and fuel burnup depth were measured by radiochemical and mass-spectrometer methods in fuel of the specimens. Range of the measured burnup values, taking into account axial and radial distributions in FA, is 22.86 – 46.30 kg FP/т $U_{initial}$. The obtained experimental data, taking into account information on initial fuel and its irradiation history, can be used for reactivity calculation, evaluation of degree of spent fuel handling system criticality and for verification of calculation codes. But in order to obtain statistically valid results, due to considerable variation of data on fuel irradiation history, it is required to carry out additional experiments on investigation of nuclear fuel of other FAs.

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Preliminary analysis of the REBUS-PWR results

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Abstract.

REBUS is an International Programme launched jointly by BELGONUCLEAIRE (BN) and SCK•CEN in the beginning of the year 2000. The PWR phase of this R&D project is now completed. As results, it provides a unique database for criticality benchmarking with implementation of Burnup Credit. Such an analysis is here presented, comparing the experimental measurements to the BN calculations.

KEYWORDS: REBUS, WIMS, burnup, criticality, validation

1. Introduction

Taking into account the loss of reactivity with burnup in criticality studies may result in significative economical benefits as well as in a decrease of radioactivity burden. This concept, known as "burnup credit" (BUC), has motivated nuclear industry and national research institutions to carry out various R&D programmes in that field.

To this end, the International Programme REBUS-PWR (Reactivity tests for a direct Evaluation of the Burn-Up credit on Selected irradiated LWR fuel bundles) has been initiated jointly by the Belgian Nuclear Research Center SCK•CEN and Belgonucleaire. Additionally sponsored by EdF (France), VGB (German nuclear utilities), JNES (Japan) and more recently ORNL (US), REBUS aims at providing a unique experimental database for validation of criticality and reactor physics codes devoted to burnup credit.

Four major features make REBUS an outstanding programme :

(1) The use of a commercial fuel irradiated in a standard way, up to a high burnup. The "integral" approach (no isotope-wise measurement) makes the burnup effect very demonstrative.

(2) The irradiated fuel is chemically characterized, in a similar way to the ARIANE [1] programme, but with the difference that the analyses focus on a slightly restricted list of nuclides (actinides, burnup indicators and the most important fission products (FPs) with regard to neutron absorption) to be measured only by the SCK•CEN.

(3) The antireactivity induced by the burnup, assessed through critical configurations, is foreseen to be large enough to reduce the relative importance of all uncertainty sources.

(4) The critical configurations are made as simple as possible so that they provide a valuable database for code benchmarking.

First the paper recalls briefly the scope of the programme. Then it describes the BN calculation codes, as used in the design and follow-up of the programme. Finally, the main part of the paper is devoted to calculation-to-measurement comparisons, based mostly on the design calculations.

2. The REBUS-PWR scope

The scope of the REBUS-PWR International Programme was described into details elsewhere [2, 3], so that one only gives here a summary. The scope includes the study of five critical configurations, that are loaded in the VENUS facility. They consist of a driver zone surrounding a central test bundle, which is successively composed of fresh and irradiated MOX and UO2 fuels. The rods are both being characterised by non-destructive and destructive examinations, for their criticality relevant composition.

2.1.Hot cell work

Refabrication of the industrial 4 meter long GKN UO2 spent fuel rods into 1 meter rodlets was executed in the SCK•CEN hot cell laboratory. This was a challenging aspect of the project, covering several steps, among which are cutting, flattening, defuelling, cleaning, end cap fitting, welding, decontamination and leak test. For the BR3 MOX spent fuel rods, no refabrication was needed as the original length was already 1 meter. The rodlets were assembled in an 7×7 test bundle.

2.2.Non destructive fuel characterization

Two types of γ-activity based measurements were performed. Firstly, all spent fuel rods were scanned for the total γ–activity distribution all along their axis. Secondly, one selected rod per irradiated bundle was investigated all along its axis by a detailed γ–spectrometry, with respect to Cs137 activity. The combination of the total activity scans and the γ–spectrometry scan provided the axial burnup profile of all rodlets.

2.3.Radiochemical assay

A radiochemistry characterization [1] was performed for one sample per irradiated fuel (MOX and UO2). The sample was taken from the same rod measured by spectrometry. It aimed at determining both the actinides content, some burnup indicators and the 19 most important fission products with respect to neutron absorption.

2.4.Critical experiments at VENUS

Five critical configurations (Figure 1) associated to five different 7×7 fuel bundles were considered in the REBUS-PWR programme :

Congiguration 49/00 : reference 3.3 w/o enriched UO_2 fuel, calibration critical configuration,

Congiguration 50/00 : fresh PWR MOX fuel, manufactured by BN for the experimental Belgian PWR BR3,

Congiguration 51/00 : fresh commercial PWR UO_2 fuel, manufactured by Framatome ANP GmbH,

Congiguration 52/00 : irradiated PWR MOX fuel (20 GWd/tM + 15 years cooling), from BR3 reactor.

Congiguration 53/00 : irradiated commercial PWR UO2 fuel (54 GWd/tM + 7 years cooling), originating from Neckarwestheim NPP and belonging to GKN (Germany).

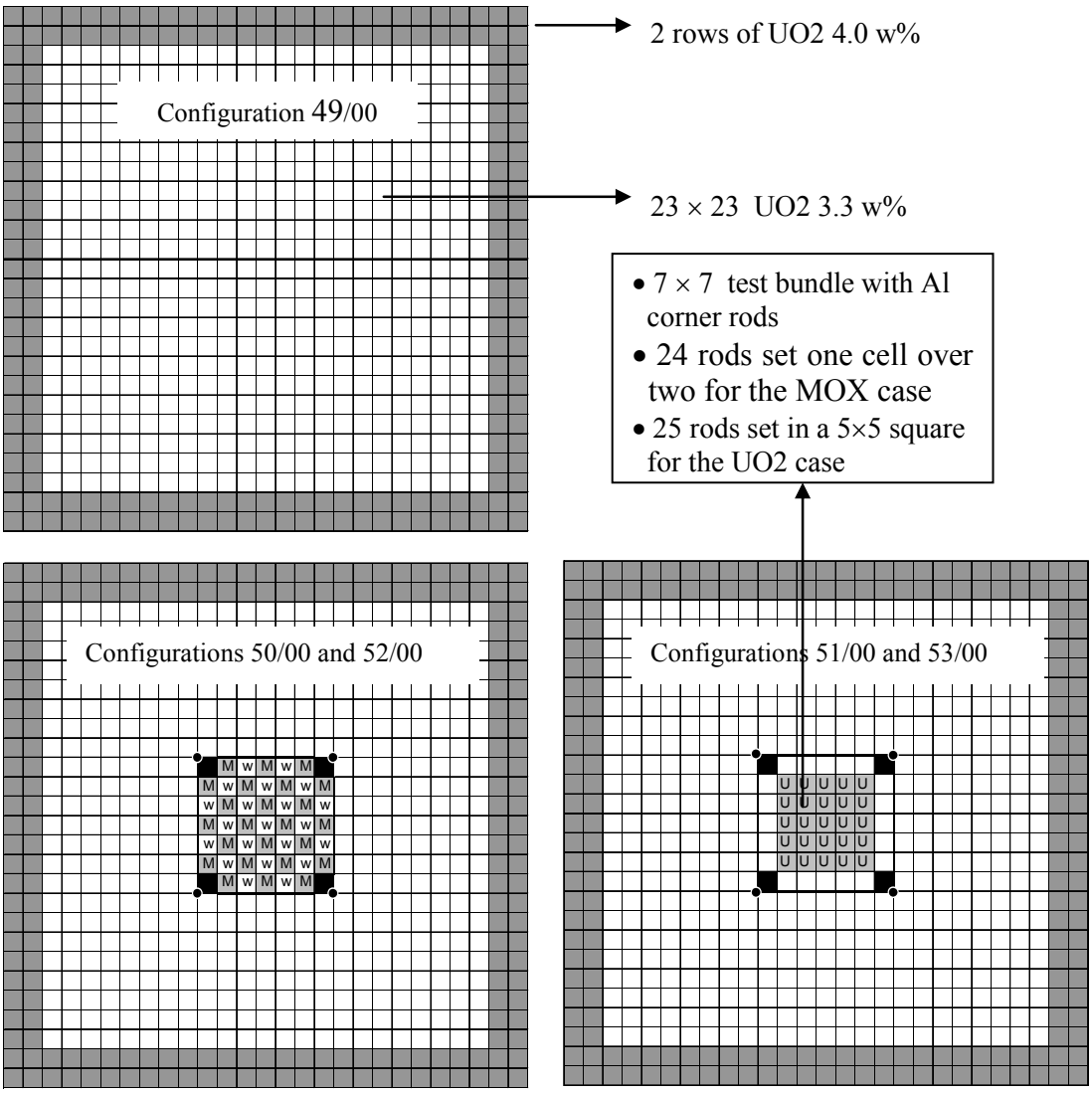

Figure 1. XY view of the critical configurations achieved in VENUS, in the frame of REBUS-PWR

The driver zone was made of 3.3 w/o enriched UO2 fuel rods (23×23 rods) and two additional rows of 4.0 w/o enriched UO2 fuel rods, leading to external dimensions of 27×27 rods. The same driver zone was used for the five critical configurations.

The reactivity effect induced by the burnup was measured by loading successively two different bundles (same fuel type but two different burnup states) in the centre of the driver zone and measuring each time the critical water level and the water level reactivity. Both parameters allow to get the reactivity effect induced by the burnup. Indeed, only a critical water level difference can be measured in the VENUS facility, that has to be converted in terms of reactivity effect through the use of the water level reactivity at both critical water levels (fresh and irradiated states) [4]. The burnup induced reactivity effect may be deduced by means of two approaches : (i) a linear one and, (ii) the proper integration method. The latter one is recommended as soon as large critical water level differences occur, as it is the case for REBUS.

Besides the critical water level and water level reactivity, radial and axial fission rate and flux distribution measurements were performed.

3. BN calculation tools

3.1. The computer codes

The calculations were perfomed with WIMS8a [5] modular codes for the main part. WIMS can be used « stand alone » or coupled to other codes, such as DANTSYS [6] and KENO-Va [7], with the help of a home-made interface software. WIMS is fed by its own WIMS'97 multigroup library (172 groups) based on JEF2.2. A summary of the calculation codes and the parameters they address is given in Table 1. More details about this item can be found in [8].

TABLE I. SUMMARY OF THE CALCULATION CODES AND THE ASSOCIATED PARAMETERS

Parameter	Code	Method
Spent fuel isotopic inventory	WIMS8a (WIMS'97 fed, 172 gr.)	Transport + burnup equations
Critical axial buckling Error! Objects cannot be created from editing field codes.	WIMS8a (WIMS'97 fed, 172 gr.)	Transport characteristics
Delayed neutron fraction β and importance factor I		Delayed neutron data [9] processing
Delayed neutrons parameters a_i and λ_i		
Prompt neutron lifetime Error! Objects cannot be created from editing field codes.		
Cross sections preparation	WIMS8a	16 energy groups, P_1 Legendre expansion
Critical water level H_c	KENO-Va	Monte Carlo 2000 × 2000
Reactivity of the water level $\partial\rho/\partial H$	THREEDANT	Transport S_{16}
Total extrapolated length λ	/	Buckling formula

3.2. Depletion calculations

Depletion calculations were performed in order to get the isotopic inventory of the spent fuel to be loaded in VENUS (Configurations 52/00 and 53/00) and consequently to determine the related reactivity effect compared with the fresh fuel bundle.

So far, only one representative spent fuel rod per bundle (the bundle contents 24 test rodlets for the MOX and 25 for the UO2) was calculated, taking into account an average irradiation history.

For the MOX case, in account of the complex BR3 geometry, a special macrocell representing a BR3 assembly comprised between two moderator tubes was developed.

For the UO2 case, a standard fuel assembly calculation was performed, representing the original GKN 18×18 PWR assembly.

3.3. Venus 2D calculations

2D calculations of the VENUS configurations were performed with WIMS to get several physical parameters :

- the critical axial buckling B_z^2 (cm^{-2}),
- the radial pin power distribution,

- the effective delayed neutron fraction β_{eff},
- the relative fraction of delayed neutrons and decay constants,
- the prompt neutron lifetime ℓ .

The calculation scheme implemented in WIMS is nearly the same as for depletion calculations, but the geometrical model represents the flooded fuel region of the REBUS-PWR configuration.

3.4. Cross sections preparation

After the VENUS 2D calculation, the cross sections of the fuel, cladding and moderator are smeared together to get equivalent cell cross sections. This permits to build the 3D geometrical model by means of processed elementary cells.

3.5. Venus 3D calculations

Both the deterministic THREEDANT and statistical KENO-Va codes are used for 3D calculations. THREEDANT is the 3D module of the DANTSYS package (Sn transport). KENO-Va is a Monte Carlo code and the calculations were performed with a statistics of 2000 neutrons per generation × 2000 generations.

The critical height H_c of the flooded fuel region was obtained by means of few calculations, from which we also deduced the reactivity of the water level $\partial\rho/\partial H$.

4. Analysis of the results

4.1. Spent fuel inventory

For the MOX fuel, two depletion calculations were performed : one for the representative average fuel composition of the bundle and the other one dedicated to the fuel sample analysed by radiochemistry, which has a slightly higher burnup of 22 GWd/tM. The calculation to measurement ratios C/E are however considered relevant for the entire bundle and will be thus used in the reactivity analysis (§ 4.4).

For the UO2 fuel, only one depletion calculation is enough, due to the axial flat burnup profile inside the bundle.

The C/E comparisons are reported in the Tables II.a and b, respectively for the actinides and the fission products. Note that decay correction were so far only performed for the actinides, so that the results are provided for both the date of criticality in VENUS (relevant for the criticality analysis) and the EOL state (relevant for the reactor physics and cross sections data).

A comparison against ARIANE samples is finally given. Although burnup and decay time are not the same (BM-1 irradiated up to 45.5 GWd/tM ; GU-3 irradiated up to 51.9 GWd/tM) as for REBUS-PWR, this provides evidence that the C/E observed in REBUS are globally compatible with those observed previously in the frame of the ARIANE project [1, 10, 11].

TABLE II.a. SPENT FUEL INVENTORY CALCULATION TO MEASUREMENT COMPARISON (C/E) FOR THE ACTINIDES

Nuclide	MOX fuel			UO2 fuel		
	EOL	VENUS	ARIANE BM-1	EOL	VENUS	ARIANE GU-3
U-234	0.896	0.931	0.94 - 0.98	1.293	1.212	1.46 – 1.51
U-235	0.993	1.001	1.030	1.094	1.094	1.18
U-236	0.492	0.687	0.884	0.966	0.967	0.98
U-238	1.000	1.000	1.000	1.000	1.000	1.00
Pu-238	0.891	0.971	0.93 - 0.94	0.785	0.829	0.89
Pu-239	1.022	1.023	1.078	1.104	1.110	1.11
Pu-240	1.000	0.998	1.005	1.017	1.013	1.05
Pu-241	1.003	1.014	1.024	1.015	1.021	1.07
Pu-242	0.998	0.998	0.982	0.891	0.891	0.93
Am-241	nc	1.240	1.08 - 1.24	nc	1.315	1.24 – 1.94
Am-242m	0.731	0.731	0.708	0.830	0.830	0.88
Am-243	1.094	1.094	0.958	1.041	1.041	0.99
Cm-242	nc	nc	0.966	nc	nc	0.94
Cm-243	1.241	0.860	0.779	1.027	0.878	1.26
Cm-244	0.779	0.786	0.856	0.733	0.770	0.67 – 0.80
Cm-245	0.756	0.756	0.888	0.704	0.704	0.70 – 0.84
Np-237	0.241	0.250	0.689	1.063	1.069	0.77

For the MOX fuel, a very good agreement is obtained for U235 and the Pu isotopes. One may however quote a strong overestimation of Am241, strong underestimation of Np237 and a global overestimation of the metallics FP. Such trends were already observed in [1, 10] but they are here still enhanced.

For the UO2 fuel, U235, Pu239 and Am241 suffer a strong overprediction, once again already observed for the GU-3 ARIANE sample. Np237 and the Cm isotopes are underestimated and the metallics FP are overestimated, although to a less extent than for the MOX case. Finally Nd143 is overestimated by 4 % whereas Nd144 is underestimated by 3-4 %, such that the prediction for the sum of both nuclides is close to 1.

Such trends were studied in details for the MOX fuel in the so-called european project VALMOX [10, 11]. It is therefore known that part of the discrepancy arises from the calculation method, whereas the other part can arise form the cross section data. The case of Am241 was especially pointed out as a nuclide for which improvement of the cross sections data is desirable, for instance to be implemented in the forthcoming JEFF-3 database.

TABLE II.b. SPENT FUEL INVENTORY CALCULATION TO MEASUREMENT COMPARISON (C/E) FOR THE FISSION PRODUCTS

Nuclide	MOX fuel		UO2 fuel	
	VENUS	ARIANE BM-1	VENUS	ARIANE GU-3
Nd-143	0.996	0.992	1.045	1.04
Nd-144	0.979	0.969	0.973	0.96
Nd-143+144	0.987		0.995	
Nd145	0.999	0.998	1.014	1.01
Nd-146	0.996	0.999	1.005	0.99
Nd145+146	0.998		1.009	
Nd-148	1.013	1.014	1.027	1.01
Nd-150	0.975	0.999	1.017	0.99
Sm-147	1.016	0.999	0.974	1.04
Sm-148	0.843	0.922	0.907	0.91
Sm-149	0.956	0.851	1.082	0.82
Sm-150	0.989	0.967	0.996	1.06
Sm-151	0.985	1.078	1.170	1.26
Sm-152	1.015	1.037	1.101	1.15
Sm-154	1.038	1.003	1.019	1.11
Eu-153	0.933	1.029	1.112	1.10
Eu-154	1.136	1.115	1.917	1.74
Eu-155	1.725	1.042	1.016	1.08
Gd-155	1.483	1.300	0.997	1.12 – 1.28
Cs-133	1.010	0.922	1.037	1.03
Cs-134	1.185	0.878	1.078	1.00
Cs-135	0.969	nc	0.996	nc
Cs-137	nc	0.845	nc	1.01
Mo-95	1.125	1.083	1.116	0.87
Tc-99	0.987	1.606	0.974	1.07
Ru-101	2.123	1.763	1.293	1.09
Rh-103	2.247	1.988	1.173	1.12
Pd-105	1.554	nc	1.599	nc
Pd-108	1.588	nc	1.576	nc
Ag-109	1.340	2.482	1.223	1.01

4.2.Critical water level and water level reactivity

The calculation to measurement discrepancies with respect to the critical water level and water level reactivity are reported in the Table III, in terms of percentage for H_c and in pcm. The BN calculations are characterized by a mean conservative bias of ~ 300 pcm, that is essentially determined by the driver zone. The worst case is obtained for the irradiated MOX fuel bundle, due to the axial burnup profile that is so far neglected. The agreement between the 3D deterministic and the stochastic approaches is satisfactory, as it should be since the geometrical modelling and the cross sections are the same.

TABLE III. ABSOLUTE CRITICAL WATER LEVEL COMPARISON BETWEEN CALCULATION AND MEASUREMENT

VENUS core	49/00	50/00	51/00	52/00	53/00
Comment	Reference	Fresh MOX	Fresh UO2	Burnt MOX	Burnt UO2
Hc – DANT (%)	-1.4	-4.1	-3.3	-10.3	-1.6
Hc – KENO (%)	-0.9	-3.8	-3.3	nc	-3.2
Keff – DANT (pcm)	-123	-370	-298	-640	-95
Keff – KENO (pcm)	-78	-340	-298	nc	-192

The water level reactivity was calculated at both water levels (Table IV) : the critical state and the level considered in the period measurement (~ Hc + 1 cm). Such a correction is worthful for the 2D calculations (WIMS - only), whereas no systematic trend is observed for the 3D calculations. Globally the dρ/dH are foreseen with a standard deviation of ~ 10 %, with the worst case for the irradiated MOX fuel.

TABLE IV. WATER LEVEL REACTIVITY COMPARISON BETWEEN CALCULATION AND MEASUREMENT

VENUS core		49/00	50/00	51/00	52/00	53/00
Comment		Reference	Fresh MOX	Fresh UO2	Burnt MOX	Burnt UO2
(dρ/dH) (%) at Hc	DANT	-12	-3	-1	19	-14
	WIMS	10	12	16	35	22
(dρ/dH) (%) at Hc + ΔH	DANT	-15	-8	-4	10	-19
	WIMS	5	6	12	26	16

4.3. Flux and fission rate distributions

The radial fission rate distributions across the REBUS configurations are plotted in Figures 2. Such distributions are obtained choosing one UO2 3.3 w% driver zone rod as a monitor, which assumes no discrepancy for this particular rod and has a potential influence on the other ones. Note that there is no fission rate measurement inside the irradiated test bundle, so that the large differences observed for the cells 0 to 3 between the fresh and the irradiated states are only calculated ones. One may however quote that the calculation underestimates quite strongly the fission rate of the fresh UO2 fuel rods, by 4 to 6 %. The Table V gives a numerical summary of the pinwise results through the use of the standard deviation of the individual discrepancies. The configuration 51/00 incorporating the fresh UO2 bundle presents indeed the largest deviation.

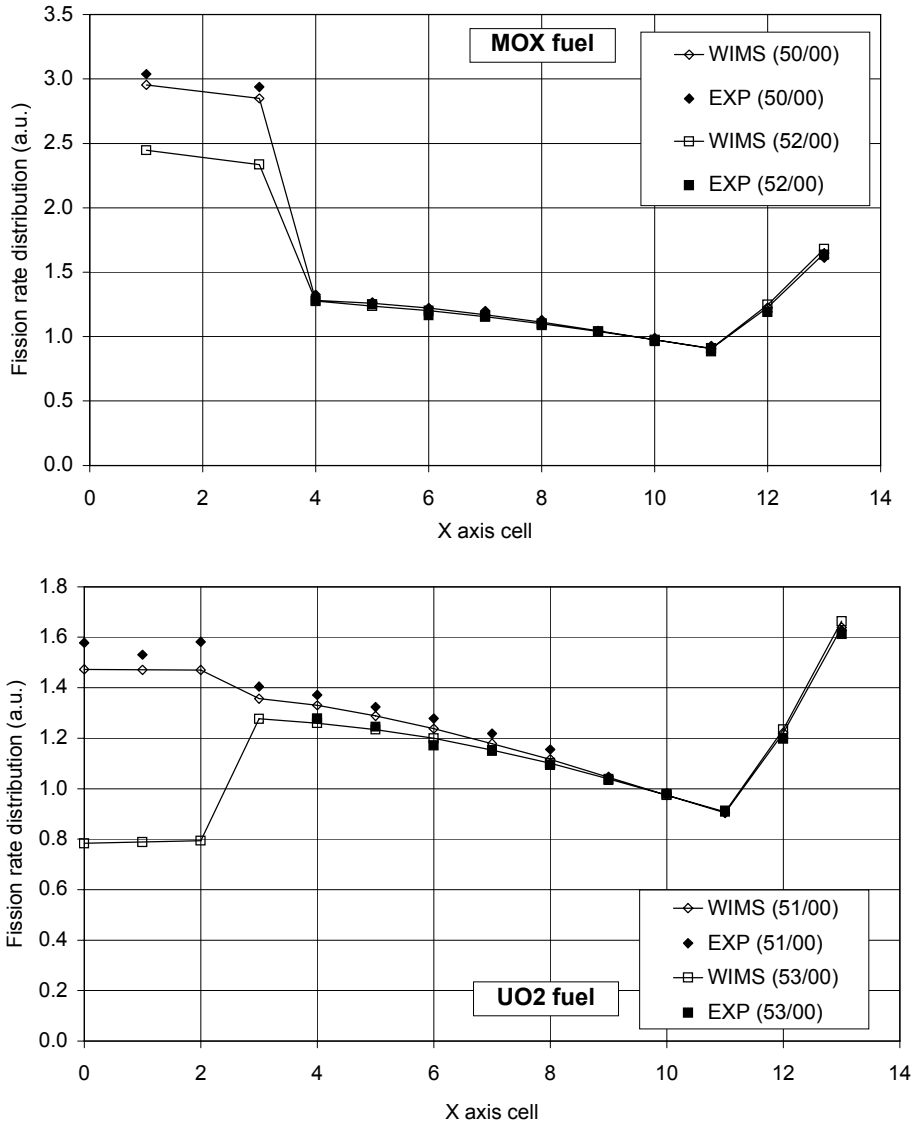

Figure 2. Comparison between the calculated and measured fission rate distributions.

TABLE V. PINWISE FISSION RATE COMPARISON BETWEEN CALCULATION AND MEASUREMENT

1. VENUS core	**50/00**	**51/00**	**52/00**	**53/00**
Comment	**Fresh MOX**	**Fresh UO2**	**Burnt MOX**	**Burnt UO2**
σ_X (%)	2.05	2.40	1.89	1.65
σ_{XY} (%)	2.79	3.80	2.48	1.73

The analysis of the radial flux distribution is complicated by off-centered position of the UO2 3.3 w% monitor rod and by the intermediate XY position of the activation wires with which the measurements were made. Therefore the analysis is so far restricted to the X axis and is reported as standard deviations of the point-wise discrepancies, in the Table VI. Once again the configuration 51/00 presents the worst result. Moreover, one observes a local discrepancy in the shape of the radial flux distribution in the bundle region (convex / concave shapes).

TABLE VI. RADIAL FLUX DISTRIBUTION COMPARISON BETWEEN CALCULATION AND MEASUREMENT

2. VENUS core	50/00	51/00	52/00	53/00
Comment	Fresh MOX	Fresh UO2	Burnt MOX	Burnt UO2
σ_X (%)	3.22	5.35	4.71	4.42

4.4.Burnup induced reactivity effect

The final burnup induced reactivity effects were about –1900 and – 2300 pcm, respectively for the MOX and the UO2 fuels. The comparison between calculation and the measurement reveals that such a reactivity effect can be foreseen at ± 10 % without too much calculation modelling effort, as far as the axial burnup profile is flat. In such a case also, a 2D calculation already provides a satisfactory agreement, whereas this is no longer the case when there is a significant axial burnup gradient, as for the MOX fuel. The line quoted as "Bu corr." stands for "burnup corrected", for which we took into account, only by a rule of thumb, that the average immerged burnup of the MOX fuel test bundle differs from the average burnup over the bundle total length (H_c < 1 m active length). This indicates that an explicit 3D VENUS modelling, accounting for the burnup axial profile, should provide a better agreement.

TABLE VII. COMPARISON BETWEEN THE CALCULATED AND MEASURED REACTIVITY EFFECTS

(C-E)/E in %	2D (WIMS)	3D calculations		
Case	$\Delta\rho$	ΔHc	$\Delta\rho_{lin}$	$\Delta\rho_{int}$
MOX fuel	-20	-32	-18	-13
MOX fuel Bu corr.		-14	-10	-4
UO2 fuel	-6.9	0.3	19	-5.4

TABLE VIII. VENUS REACTIVITY EFFECT BREAKDOWN (%) AMONG THE MAIN NUCLIDES — BASE AND C/E CORRECTED VALUES ARE GIVEN — PRELIMINARY RESULTS

Nuclide	MOX fuel		UO2 fuel	
	Base	C/E corrected	Base	C/E
U234	0.16	0.17	-0.20	-0.24
U235	0.74	0.74	**113.95**	**115.64**
U236	0.05	0.08	1.11	1.15
U238	-0.15	-0.15	-1.18	-1.18
Pu238	-0.56	-0.49	0.94	1.14
Pu239	**34.17**	**35.79**	**-48.19**	**-43.41**
Pu240	4.25	4.37	13.32	13.15
Pu241	15.72	16.01	-12.36	-12.11
Pu242	0.41	0.42	1.28	1.43
Am241	**24.98**	**24.68**	**4.20**	**3.19**
Am242m	-0.11	-0.11	-0.09	-0.11
Am243	0.90	0.90	0.65	0.62
Cm242	0.00	0.00	0.00	0.00
Cm243	0.00	0.00	-0.01	-0.01

Cm244	0.02	0.03	0.06	0.08
Cm245	-0.02	-0.02	-0.22	-0.31
Np237	0.05	0.19	1.41	1.32
ACTINIDES	80.63	82.61	74.68	80.36
Mo95	0.19	0.17	0.66	0.59
Tc99	0.50	0.51	1.12	1.15
Ru101	0.20	0.09	0.50	0.39
Rh103	**1.70**	**0.76**	**3.12**	**2.66**
Pd105	0.25	0.16	0.42	0.27
Pd108	0.13	0.08	0.17	0.11
Ag109	0.66	0.49	0.61	0.50
Cs133	0.82	0.82	1.66	1.60
Cs135	0.08	0.08	0.14	0.14
Nd143	1.37	1.37	3.76	3.60
Nd145	0.30	0.30	0.91	0.89
Sm147	0.49	0.48	0.67	0.69
Sm149	5.00	5.24	1.91	1.76
Sm150	0.19	0.19	0.66	0.66
Sm151	2.81	2.85	1.91	1.63
Sm152	0.82	0.81	1.14	1.03
Eu153	0.33	0.35	1.16	1.05
Eu154	0.00	0.00	**0.63**	**0.33**
3. Gd155	**3.48**	**2.35**	3.99	4.00
PseudoFP	0.04	0.04	0.19	0.19
Fission Products	19.37	17.15	25.32	23.23
TOTAL	100.00	99.77	100.00	103.60

Further analysis is provided in the Table VIII. It deals with the reactivity effect breakdown between the major nuclides. Two columns are given for each kind of fuel. They correspond to the base calculation and to what would be expected when the isotopic content is corrected by the C/E ratios reported previously.

For the MOX fuel, 80% of the reactivity loss is attributed to the actinides, among which Pu239 and Pu241 depletions, and Am241 build-up (large decay time) are the major contributors. When correcting by the inventory C/E ratios, the loss of reactivity effect should remain about the same, but giving still more weight to the actinides contribution (Pu239 and Am241) and decreasing the FP contribution mainly through the Rh103 and Gd155 isotopes.

For the UO2 fuel, about 75% of the reactivity loss is attributed to the actinides and one can observe that the overestimation of Pu239 (build-up) is counter-balanced by the overestimation of U235 (depletion). Once corrected by the inventory C/E ratios, the total reactivity loss is increased by 3 %, providing a better agreement against the experimental value. This is to be attributed to the corrections on U235, Pu239, Am241, for the actinides, and to Rh103 and Eu154 for the FP.

5. Uncertainties

So far an uncertainty analysis with respect to the burnup induced reactivity effect has not been performed. Both calculation and experiment uncertainties have to be considered in order to

assess whether the discrepancy between the calculated and the measured Δk_{eff} is significant or not.

Most of the experimental uncertainty sources affect each one of the couple of critical configurations:

- uncertainty in the initial enrichment,
- uncertainty in the oxide density,
- uncertainty in the cladding thickness,
- uncertainty in the pitch size,
- uncertainty in the critical water level,
- uncertainty in the water level reactivity.

The latter uncertainty may comprise a significant contribution from calculation, since the water level reactivity is deduced from period measurements and combined with the calculated delayed neutron parameters.

A particular experimental uncertainty arises in the determination of the burnup induced reactivity effect [4]. This is obtained through the analytical integration of the H^{-3} behaviour of the water level reactivity, between the two critical water levels (fresh / burnt). However the proportionality constant between $d\rho/dH$ and H^{-3} behaviour may differ significantly from the fresh to the burnt configuration and an average value has to be chosen.

For what concern the calculation uncertainties, we have to take into account the both steps of depletion simulation and 3D criticality calculation. Modeling approximations and cross sections data remain the major contributors to the calculation uncertainty.

Although the uncertainties analysis is not yet performed, it is believed that the presently observed trend of 5 % reactivity effect underestimation is a significant signature of the calculation scheme, thus above the cumulated uncertainties.

6. Conclusions

The paper provides a preliminary overall study of the REBUS-PWR experimental results, using mainly the design calculations. It is shown that the REBUS programme provides a clear evidence of the reactivity effect induced by burnup, that can be simulated by calculation successfully, without too much sophisticated experiment modelling.

The salient preliminary observations are the following:

- the burnup induced reactivity effects are presently foreseen at ± 10% ;
- a 3D model is needed for the analysis as soon as a burnup axially non homogeneous profile is considered (MOX case) ;
- given a sufficiently accurate geometrical model for the VENUS modelling, the major part of the discrepancy for the burnup induced reactivity effect is explained by U235, Pu239 and Am241 nuclides (actinides contribution always underestimated) ;
- moreover, some fission products devote further attention, due to the combined effect of their irradiated fuel concentration and their reactivity weight, namely Rh103, Eu154 and Gd155.

Although the preliminary analysis is encouraging, a rigorous validation work should be performed, which should be very fruitful for the Burnup Credit implementation in criticality studies and for the safety in general. From the BN side, one contemplates the following steps:

- refinement of the VENUS modelling through a more detailed fuel rod and reflector definition;
- recheck and possible refinement of the depletion calculations ;
- further analysis of the depletion calculation for the radiochemical samples and explicit decay time correction for all isotopes ;
- further analysis of the burnup induced reactivity breakdown among the nuclides ;
- uncertainties analysis (modelling, initial U235 and Pu content of the fuel, critical water level measurement, water level reactivity determination, reactivity effect determination throughout the integral formula method, etc.) ;
- test of WIMS9 and of JEFF3.

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Monte Carlo calculations of the REBUS critical experiment for validation of burnup credit

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Abstract. The application of burnup credit (BUC) to criticality safety analysis for Spent Nuclear Fuel (SNF) configurations requires the implementation of both estimation of the SNF composition with the aid of depletion calculation tools and estimation of the SNF reactivity with the aid of criticality calculation tools.

Amongst the several experimental programs dedicated to the validation of both calculation tools, REBUS is distinguished by a combination of chemical analysis and critical experiment. In addition to detailed assays of irradiated fuel, the reactivity worth of the fuel rods under investigation is measured both before and after irradiation. Since a whole bundle of fuel rods is used in the experiment, the change in reactivity is significant enough to be observable by Monte Carlo calculations. Thus, the calculation tools which see the most widespread use in SNF critical safety applications can be validated directly.

Apart from the effective neutron multiplication factor k_{eff}, REBUS also provides measurements of the flux and fission rate distributions. While the program comprises investigation of commercial UO_2 fuel rods and mixed oxide (MOX) fuel from a research reactor, the presentation will focus on the commercial UO_2 fuel with an overview of the experimental setup and first results from the analysis.

1. Introduction

Analysis of the criticality safety for Spent Nuclear Fuel (SNF) is usually based on the assumption of unburned fuel. The application of burnup credit takes the reactivity loss of the irradiated fuel into account and requires well-validated calculation tools for predicting the nuclide inventory of depleted fuel (reactor physics) and for estimating the SNF reactivity (criticality methodologies).

While the calculation tools in the reactor physics domain allow accurate and thoroughly validated predictions of reactivity loss with burnup in the reactor core, predictions of the SNF reactivity outside the core in storage, transport or disposal conditions are less well validated. In addition, assuring the criticality safety of SNF operations outside the reactor usually relies more on the accuracy of the calculated reactivity than on monitoring equipment. For these reasons, there is a strong requirement for validating the calculation methods used in burnup credit for SNF. This led to the establishment and implementation of several experimental programs in Europe aimed at providing data for these validation efforts.

The REBUS experimental program is distinguished by a combination of chemical analysis and critical experiment. In addition to detailed assays of irradiated fuel, the reactivity worth of the fuel rods under investigation is measured both before and after irradiation. Since a whole bundle of fuel rods is used in the experiment, the change in reactivity is significant enough to be observable by Monte Carlo calculations. Thus, the calculation tools which see the most widespread use in SNF critical safety applications can be validated directly.

In comparison with earlier experimental programs, REBUS extends the fuel type, the enrichment range and the number of fission products analyzed. One of the conclusions of the CERES program [1]

was that the most significant uncertainty in fission product contribution to BUC arises from the uncertainty in the inventory predictions, which shows the relevance of the REBUS chemical assay data.

Apart from the effective neutron multiplication factor k_{eff}, REBUS also provides measurements of the flux and fission rate distributions. The program entails investigation of commercial UO_2 fuel rods and mixed oxide (MOX) fuel from a research reactor.

2. The REBUS PWR critical experiment

A feature of paramount importance in the REBUS program is the use of fuel bundles from commercial PWR UOX samples for the reactivity measurements, since the use of fuel bundles controlling significant amounts of reactivity permits analysis of the measurements by direct Monte Carlo methods. This allows a more direct validation of the calculation code schemes commonly used in criticality safety analysis (i.e., estimation of k_{eff} rather than reactivity perturbation calculations).

The REBUS PWR bundle contains 25 fuel rodlets of 100 cm length fabricated from Neckarwestheim II fuel rods. They were cut from the center of the active zone to suppress uncertainties arising from the axial burnup distribution. This bundle is placed in a 27x27 driver zone of VENUS fuel rods surrounded by a water reflector. The water level of the reflector is used to control the neutron multiplication of the setup.

Fig. 1 shows a cutaway view of the three-dimensional calculation model of the VENUS/REBUS setup, whereas Fig. 2 and Fig. 3 show horizontal and vertical cuts of the calculation model.

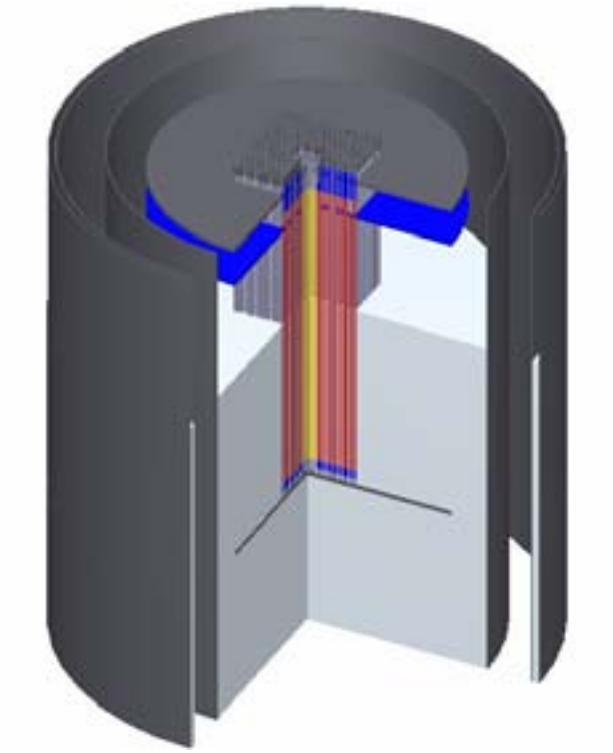


Fig. 1: 3D-View of the VENUS/REBUS setup.

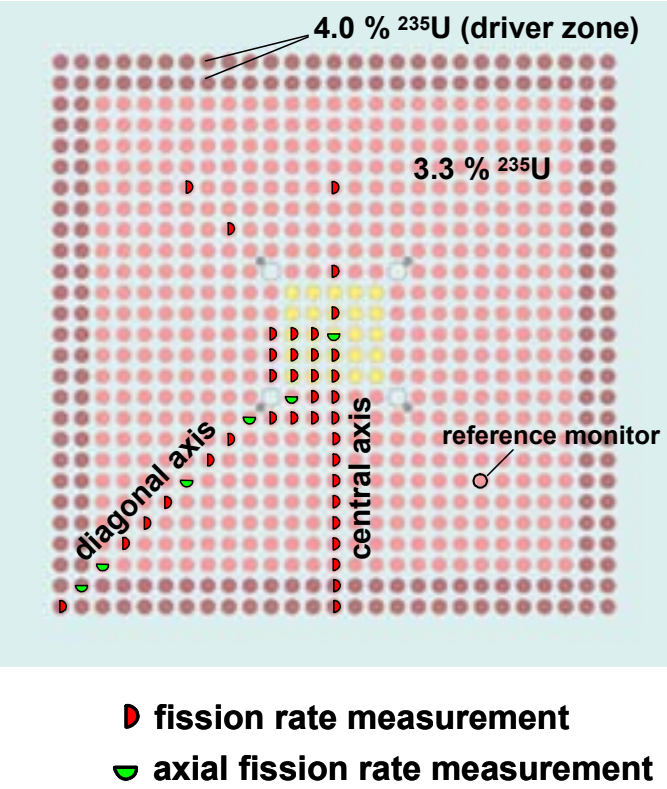


Fig. 2: Horizontal cut of the REBUS PWR core with fission rate measurement positions.

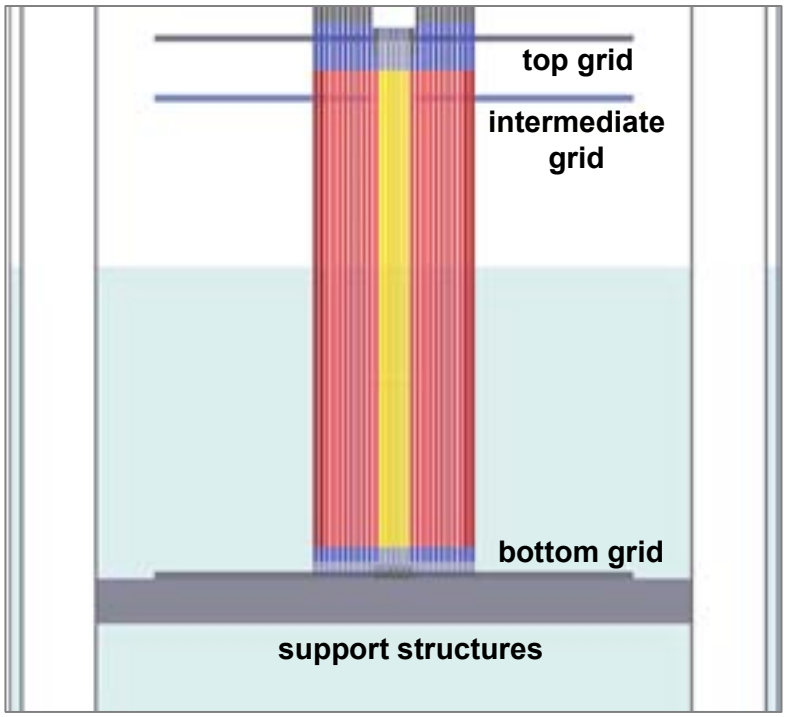


Fig. 3: Vertical cut of the REBUS PWR core.

In addition to the measurements of the reactivity effects and critical heights of the water level for the fresh and the irradiated Neckarwestheim II bundle, axial and horizontal fission rate distributions are determined by gamma-scans and horizontal neutron flux distributions are monitored with the aid of activation sensors.

The positions of the fission rate measurements within the VENUS/REBUS core are also shown in Fig. 2. Fig. 4 shows the positions of the scandium wires used to determine the radial neutron flux distribution.

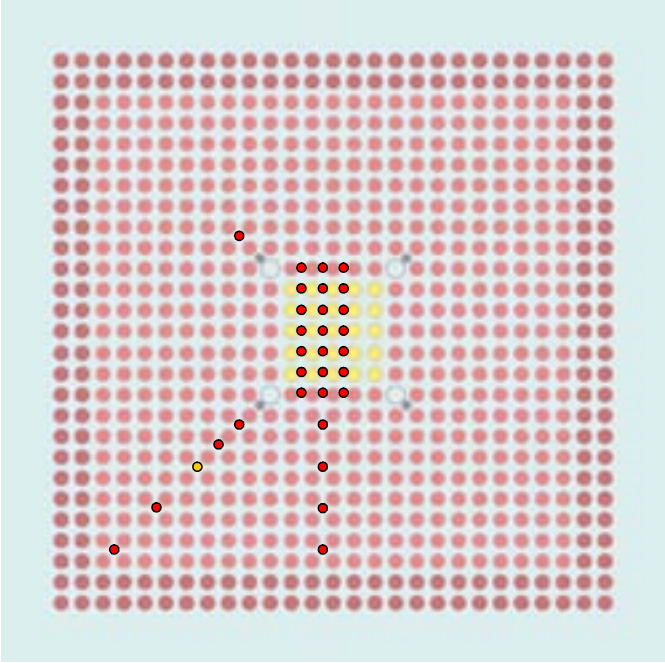


Fig. 4: Horizontal cut of the REBUS PWR core with Sc wire positions.

3. REBUS PWR Analysis at WTI

Since the emphasis of the REBUS experiment lies on the validation of a workable burnup credit methodology for SNF casks, the analysis is performed with the tools also used in our day-to-day nuclear safety work. These programs include the depletion codes ORIGEN and TRITON of the SCALE 5 system and the criticality calculation Monte Carlo codes KENO V and KENO VI of this system as well as the Monte Carlo code MCNP5.

The aim of benchmarking day-to-day tools against the REBUS PWR measurements extends to the neutron cross section libraries: instead of data sets geared specifically towards this experiment, the default libraries of the computation codes are applied. The general purpose Monte Carlo code MCNP uses point-wise continuous cross-section data based mainly on ENDF/B-VI and, for some nuclides, ENDF/B-V data. The codes of the SCALE system include 238-group and 44-group cross section libraries derived from ENDF/B-V data. The 44-group library, collapsed from the 238-group library, has been especially developed for the analysis of well moderated light water reactor fuel packages and is used in the calculations presented here.

Although the effective neutron multiplication factor k_{eff} and its dependence on the water level are the principal results of the Monte Carlo REBUS simulations, the validity of these results varies with the quality of the calculation model. To ensure that the level of detail of the model is appropriate, all experimental results, especially the radial distributions of flux and fission rates, were compared with the results from calculations — if the neutron distribution within the reactor is not well represented by the calculation, one cannot expect to gain good results for k_{eff}.

The results of these calculations of the three-dimensional core characteristics are described in this paper, together with the critical water level and the reactivity effect of the water level for the REBUS PWR setup. To back up the criticality analysis conducted so far, sensitivity studies will be performed to gauge the influence of material properties, geometrical uncertainties and modelling assumptions on the calculated results.

Furthermore, the results of the chemical assay data will be used to benchmark the depletion codes of the SCALE system by comparing measured and calculated isotopic contents. The ability of the Monte Carlo criticality codes to predict the SNF reactivity worth in the VENUS configuration will be tested with both the chemical assay data and the isotopic concentrations derived from depletion calculations. The latter constitutes a test of the complete calculation chain used for burnup credit.

4. Calculation Results

The radial symmetry of the VENUS core is used for both fission rate and flux calculations to increase the statistics for a detector position. If the experiment provided measurements at equivalent positions (e.g. fission rates are given for three of the four rods directly adjacent to the center rod), the mean of the measurement results is used to benchmark the calculations.

For the fission rate calculations, fissions over the whole length of the fuel rods are considered in MCNP, whereas only the fuel below water level is accounted for in SCALE/KENO. As can be seen in Fig. 5, the radial fission rate distribution is very well represented by calculations, with only a few percent deviation between measured and calculated results. It is especially noteworthy that the agreement between the two different Monte Carlo codes (SCALE/KENO and MCNP) is even better, in spite of a very different treatment of the neutron cross sections. All results are normalized to the measured result of the reference monitor rod (see Fig. 2). The rising fission rates at the edge of the REBUS core are due to the higher enrichment of the outer two rows (the driver zone).

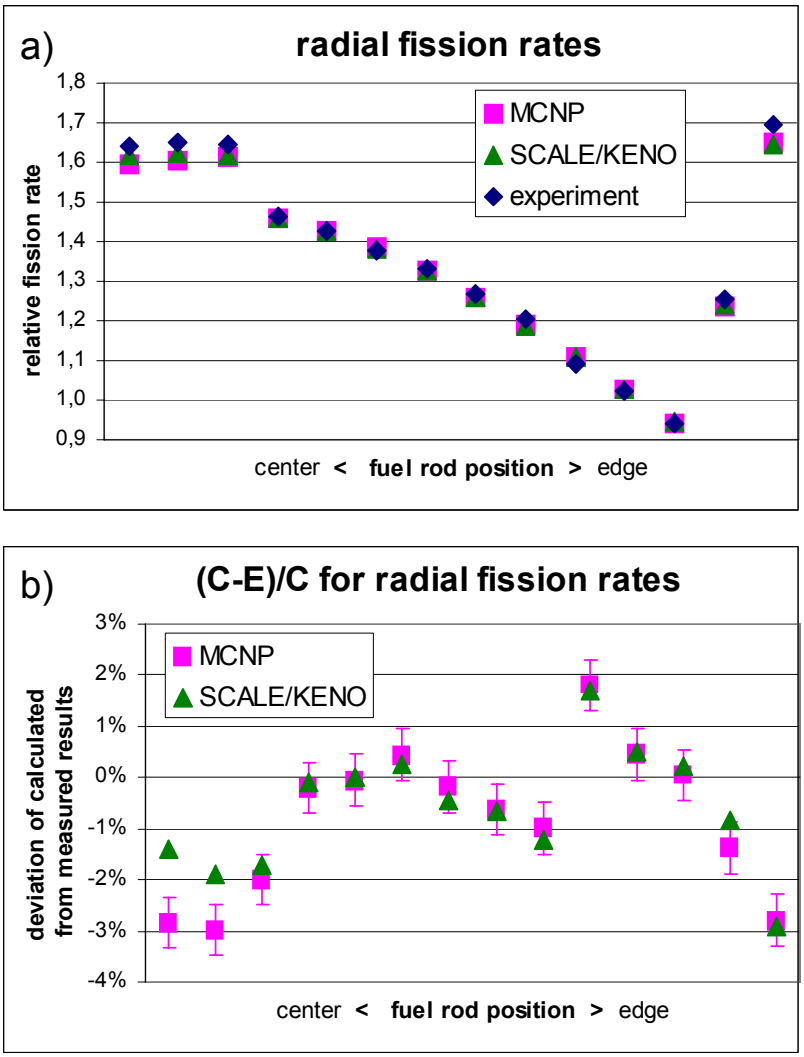


Fig. 5: Fission rates along the x=0 axis: a) relative fission rates, b) comparison between measured and calculated values.

The flux distribution measurement of the experiment is based on relative activation rates of scandium wires placed between the fuel rods, roughly in the axial center of the active zone. While both MCNP and SCALE/KENO permit flux calculations, only MCNP offers a simple method to derive Sc activation rates.

Comparison of the measured and calculated values for the Sc detectors along the x=0 axis (see Fig. 6) shows very good agreement between MCNP and SCALE/KENO for the flux calculation, but pronounced differences between calculations and measurement. Within the REBUS bundle, the radial flux increases for measured values and decreases for calculated values; outside the bundle, measured values are underestimated by a margin of roughly 5 %.

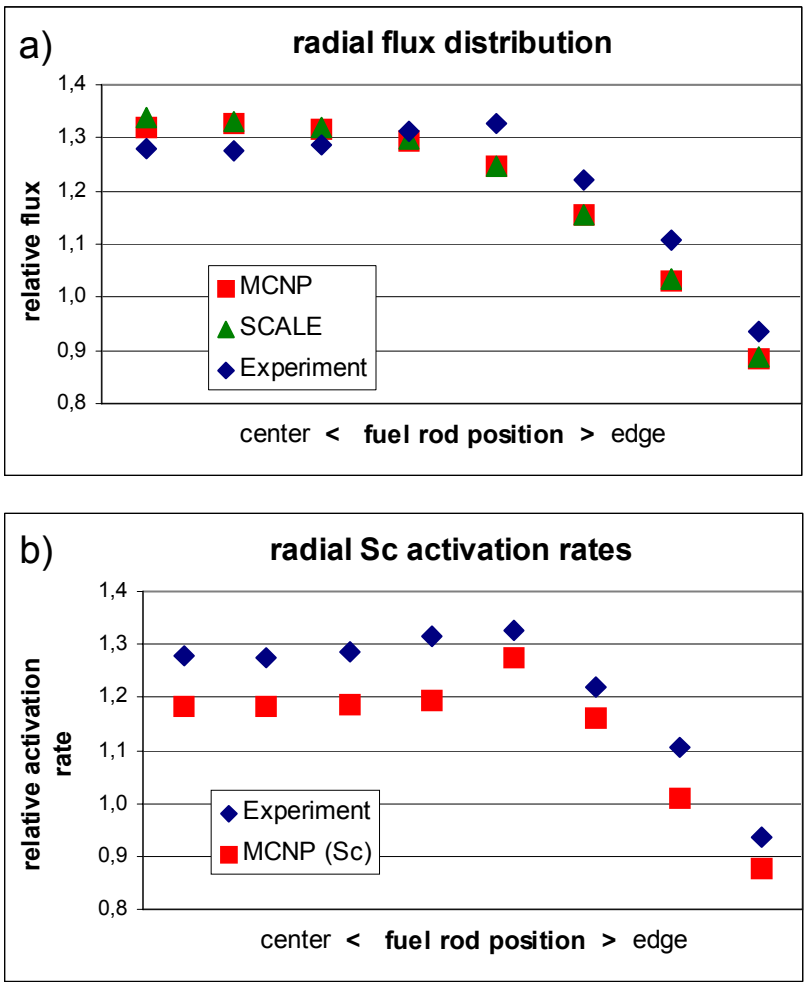
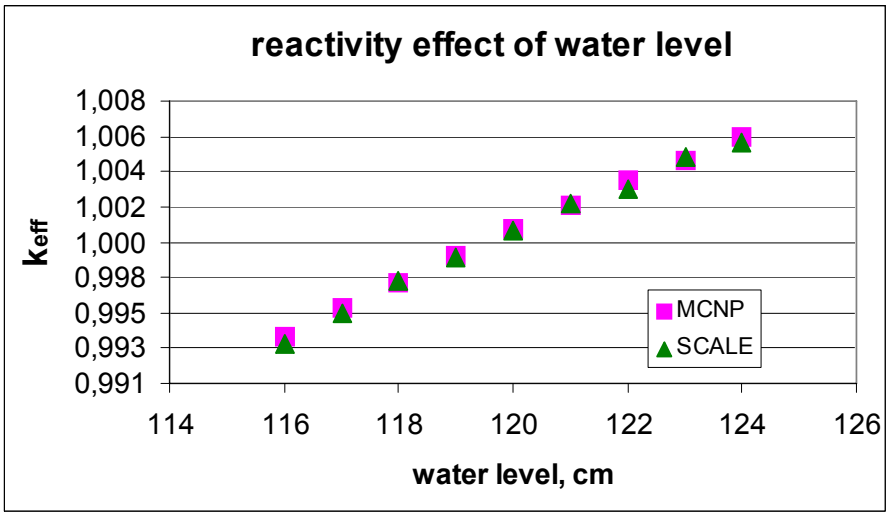


Fig. 6: Flux along the x=0 axis: (a) calculation values for relative flux, (b) calculation values for relative Sc activation rates.

When comparing the measured values to the calculations for Sc activation rates, the trend within the REBUS bundle is better represented, but now all calculated values underestimate the measurements. A possible explanation for this discrepancy might be the influence of the aluminium housings of the Sc wires, which are not accounted for in the calculation models.

Results for the critical water level, shown in Fig. 7, display a linear dependance on the water level. The critical water level and its reactivity effect are derived from linear extrapolation of the calculated values. While the measured point of criticality is overestimated by 3.5 cm (which translates into an underestimation of the effective neutron multiplication factor k_{eff} by ~500 pcm), there is excellent agreement for the reactivity effect. Criticality results for the reference setup - essentially the VENUS core with standard rods replacing the REBUS bundle - also underestimate k_{eff} by ~500 pcm and represent the measured reactivity effect of the water level very well.



	critical water level	uncertainty	reactivity effect	uncertainty
experiment	116.39 cm	0.020 cm	0.153 %/cm	0.002 %/cm
MCNP	119.89 cm	0.002 cm	0.148 %/cm	0.002 %/cm
SCALE	119.99 cm	0.002 cm	0.150 %/cm	0.002 %/cm

Fig. 7: Criticality results for the REBUS PWR core.

REFERENCE

[1] J. GULLIFORD et al., "Measurements of the Reactivity Loss with Burnup in PWR, BWR and MOX Fuel made in the CERES Collaborative Programme", Vol. 2, Proc. PHYSOR'96, September 1996, Mito, Japan

Application of sensitivity/uncertainty methods to burnup credit validation

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Abstract. The responsible use of calculational methods in nuclear criticality safety includes a determination of bias and bias uncertainty that may exist between the calculated results and reality. Such biases exist due to approximations used to model the real world, uncertainties in nuclear data, and approximations associated with the calculational method (e.g. Monte Carlo method). The bias and bias uncertainty are typically determined by using the modeling approximations, nuclear data, and calculational method to model well-known, usually critical, systems. Unfortunately, the bias and bias uncertainty determined in this manner can vary significantly depending on the characteristics of the known "benchmark" systems. The most accurate determination of bias and bias uncertainty is obtained by using benchmark systems that are very similar to the real-world operational configuration, the subcriticality of which must be safely ensured. Historically, similarity has typically been determined using comparisons of gross integral parameters (e.g. lethargy of average energy of neutrons causing fission and hydrogen to fissile nuclide ratio) and on qualitative comparisons of the geometry and materials present in the benchmark and application systems. The development of sensitivity/uncertainty methods permits detailed quantitative comparison of these systems. The work presented in this paper is a sensitivity/uncertainty-based study of the similarity or applicability of many critical experiment models to a model of a high-capacity transportation cask that is loaded with spent commercial nuclear fuel and flooded with water. This paper includes descriptions of the sensitivity/uncertainty methods used, the operational configuration of interest, benchmark critical configurations used for comparisons, and discussion of the results from the sensitivity/uncertainty analyses.

1. Introduction

Historically, criticality safety analyses for commercial light-water reactor spent nuclear fuel (SNF) storage and transportation casks have assumed the SNF to be fresh (unirradiated) with uniform isotopic compositions corresponding to the maximum allowable enrichment. This fresh-fuel assumption provides a simple bounding approach to the criticality analysis and eliminates concerns related to modeling the fuel operating history. However, because this assumption ignores the decrease in reactivity as a result of irradiation, it is very conservative and can result in a significant reduction in SNF capacity for a given storage or cask volume. Numerous publications, an extensive set of which is listed in the reference section of NUREG/CR-6800 [1], have demonstrated that increases in SNF cask capacities from the use of burnup credit can enable a reduction in the number of casks and shipments and thus have notable financial benefits while providing a risk-based approach to improving overall safety. The concept of taking credit for the reduction in reactivity due to irradiation of nuclear fuel (i.e. fuel burnup) is commonly referred to as burnup credit. The reduction in reactivity that occurs with fuel burnup is due to the change in concentration (net reduction) of fissile nuclides and the production of parasitic neutron-absorbing nuclides (nonfissile actinides and fission products). The work presented in this paper used sensitivity/uncertainty (S/U) analysis to explore the potential applicability of critical experiments to validation of burnup credit calculations. The S/U analysis was performed using TSUNAMI-3D [2] sequence and TSUNAMI-IP [3] module from SCALE 5 [4].

TSUNAMI-3D is a Monte Carlo–based eigenvalue sensitivity analysis sequence that was released with SCALE 5. This software tool permits energy-, mixture-, nuclide- and region-dependent examination of the sensitivity of the system k_{eff} to variations in nuclear data of modeled materials. TSUNAMI-3D uses first-order linear-perturbation theory to produce sensitivity coefficients. As such, the sensitivity coefficients are valid only for small perturbations.

TSUNAMI-IP uses sensitivity data generated by TSUNAMI-1D and/or TSUNAMI-3D and cross-section uncertainty data to generate several relational parameters and indices that can be used to determine the degree of similarity between two systems. The sensitivity profiles generated for a particular system of interest may be compared with the sensitivity profiles for critical experiments used to generate subcritical limits in validation studies. Such comparisons enable the analyst to reach conclusions regarding the adequacy or applicability of critical experiments used in the validation study.

Work is being performed at Oak Ridge National Laboratory (ORNL) to generate recommendations and develop computational methods related to taking credit for the in-reactor burnup of commercial nuclear fuel during out-of-reactor storage and transport. The work reported in this paper involved an evaluation of the applicability of more than 1000 critical configurations to the validation of criticality calculations for a high-capacity rail cask loaded with 32 spent pressurized-water reactor (PWR) fuel assemblies.

2. Applicability of benchmark critical experiments

The complexity of many systems having criticality accident potential necessitates heavy reliance on computer calculations to establish the safety of the system under normal and upset conditions. One of the responsibilities of the safety analyst using computer calculations is to use validated computational methods to determine the system multiplication factor (k_{eff}) and the maximum k_{eff} considered safely subcritical. Typically, validation is performed by using a computational method to calculate the k_{eff} for a set of applicable critical experiments. A computational method is defined by the modeling approximations, the computer code, computer code input options, and the nuclear data used. The analyst performing the validation then uses the calculated k_{eff} values and associated uncertainties and the critical experiment measured k_{eff} and uncertainties in a statistical analysis to determine the bias and bias uncertainty for the computational method. This bias quantifies the relationship between the calculated and actual k_{eff} values for a modeled system.

For any given computational method, the values of the bias and bias uncertainty can vary significantly depending on geometry and materials present in the modeled critical experiments. For example, the computational method bias for an array of fuel rods may be significantly different if the system is dry as compared to the bias for a similar array flooded with water. Consequently, the most accurate values for bias and bias uncertainty for use in a criticality analysis are determined using critical experiments that are similar to the calculations used in the criticality analysis. It would not be appropriate to use dry, high-enriched uranium critical experiments to determine a bias and bias uncertainty for a safety analysis of an optimally moderated low-enrichment uranium system. The bias developed in this way is a total bias that combines the biases resulting from individual bias sources. For example, the total bias may include, among many others, a bias resulting from the modeled presence of fission product ^{103}Rh and a bias related to hydrogen scattering. The size and sign of such biases vary depending upon the neutronic environment of the modeled systems. These biases may be of opposite sign, thereby partially compensating for each other. Critical experiments may also contain extra features or materials not present in the criticality analysis model, also referred to in this paper as the application. The extra biases associated with the features present only in the experiments may cancel out other biases that are present in both the experiment and application, thus hiding a real bias that should be applied to the application. The calculated partial biases vary with conditions. A bias calculated for a fission product using experiments that include only that fission product or include it in a system that is rather different than the application may not be correct. Neutron energy spectrum shifts, associated with the presence of other fission products or other materials, may significantly affect the bias associated with the fission product of interest.

A conclusion drawn from this discussion is that it is important to include only critical experiments that are similar to the evaluation case when determining bias and bias uncertainty that are appropriate for the evaluation case. Sensitivity/Uncertainty tools developed at ORNL and distributed as part of the SCALE 5 package permit a detailed, quantitative comparison of modeled systems. This comparison can be used to determine how similar a critical experiment model is to an application.

3. Sensitivity/uncertainty analysis tools and concepts

TSUNAMI-3D calculates total and partial sensitivity coefficients for various neutron interactions with each nuclide in each region. A sensitivity coefficient is the relative impact of a change in some nuclear data (e.g. Σ_a) on the system k_{eff} and is defined as $S_\alpha = (dk/k)/(d\alpha/\alpha)$, where α is the nuclear data of interest.

In some cases, these sensitivity coefficients may be further broken down into explicit and implicit components. The explicit component results from the sensitivity of k_{eff} to variation of the resonance self-shielded macroscopic cross section. The implicit component results from cross-section adjustments in the resonance self-shielding calculations. For example, the explicit sensitivity of hydrogen in the moderator around a fuel pin results directly from the sensitivity of k_{eff} to changes in the hydrogen cross section. The implicit sensitivity includes the effects of the sensitivity of the fuel macroscopic cross sections to changes in the moderator cross sections. The implicit component is calculated using derivatives produced during problem-dependent cross-section processing.

The TSUNAMI-3D sequence uses KENO V.a to perform forward and adjoint calculations. Then the SAMS program uses the forward and adjoint solutions in a standard linear perturbation theory method to produce neutron energy-dependent sensitivity profiles. The profiles for each modeled system are saved into a sensitivity data file (SDF).

TSUNAMI-IP, also a part of the SCALE 5 computer software package, is used to compare the sensitivity data for two systems. It generates a variety of total and partial relational parameters that quantify the similarities between the two systems. The work reported in this paper utilizes the c_k parameter. The c_k parameter is a single-valued parameter used to assess similarity of uncertainty-weighted sensitivity profiles for all nuclide-reactions between a design system and a criticality experiment. The value of c_k varies between zero, for two completely dissimilar systems, and 1.0, for two identical systems. The premise behind the c_k parameter is that biases are primarily due to cross-section data with larger uncertainties. Systems that demonstrate similarly high sensitivities to highly uncertain cross section data will have similar computational biases. The current guidance based on experience at ORNL is that a critical configuration is applicable to an evaluation case if the c_k value is ≥ 0.9, a critical configuration is considered marginally applicable if c_k is ≥ 0.8 and <0.9, and a critical configuration is not applicable if $c_k < 0.8$.

Another use of sensitivity data is to evaluate "coverage". Coverage for a specific nuclide and cross section is a measure of whether a critical configuration is at least as sensitive to change in the cross section as is the application. Figure 1 below shows that the ^{1}H total sensitivity for critical configuration LEU-COMP-THERM-050, case 18, from the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* [5] (IHECSBE); it covers the sensitivity for the GBC-32 for most of the energy range. It also shows that there is significant noncoverage in the 0.1- to 10-eV range. For a fully covered sensitivity profile, the blue curve in Fig. 1 would completely cover the red curve.

4. Burnup credit cask model and methods

A generic cask model with a 32-PWR assembly capacity was previously developed and is described in NUREG/CR-6747 [6]. This model, referred to as the GBC-32, was created to serve as a computational benchmark. The features of the GBC-32 include 32 cells with 365.76-cm-tall and 19.05-cm-wide Boral (0.0225 g $^{10}B/cm^2$) panels between and on the external faces of each cell. The cell walls are constructed of stainless steel having inner dimensions of 22 by 22 cm and are spaced on 23.76 cm centers. The cells sit 15 cm above the bottom of a stainless steel cask having an inner radius of

87.5 cm and internal height of 410.76 cm. The radial thickness of the side walls is 20 cm, and the cask bottom and lid are 30 cm thick. Figure 2 shows a half-cask model with the top removed.

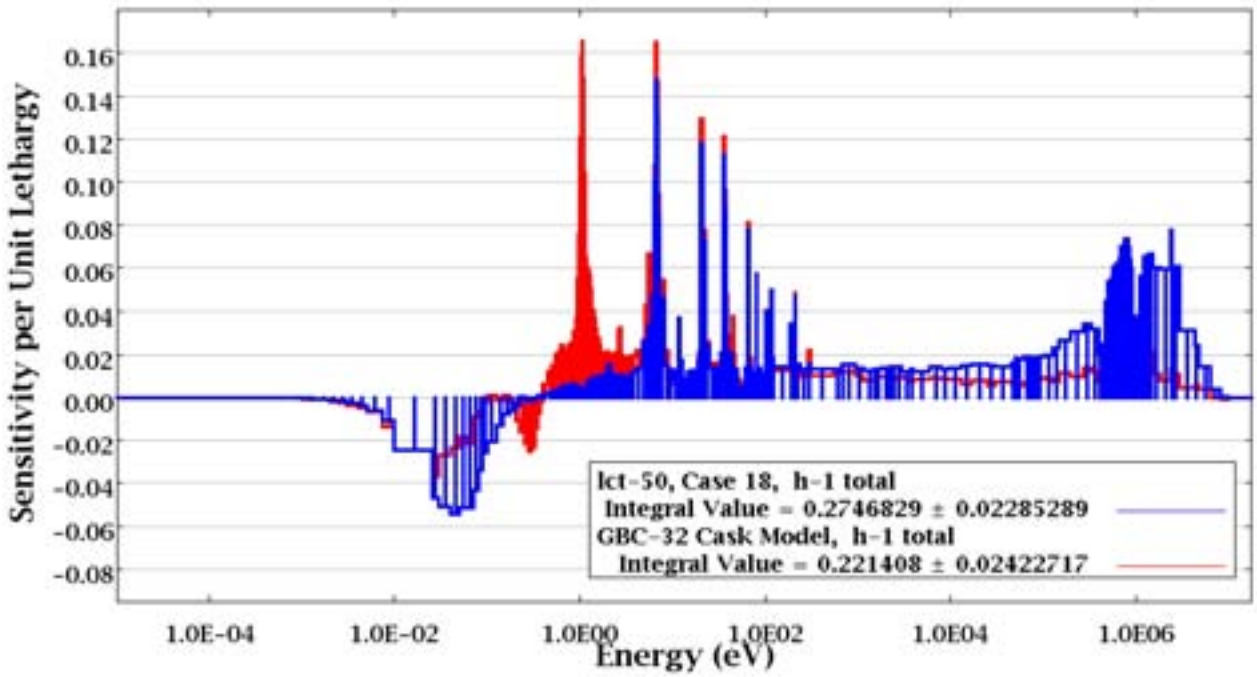

FIG. 1. The two curves show the sensitivity of k_{eff} to changes in the total hydrogen macroscopic cross sections for a burnup credit cask model (red curve) and for a critical experiment model (blue curve). "Coverage" is provided by the experiment wherever the blue curve bounds the red curve.

For purposes of the analyses documented in this paper, the cask was modeled as loaded with Westinghouse 17 by 17 Optimized Fuel Assemblies (W17×17OFA). The dimensions for the W17×17OFA were taken from Table 3 of Ref. [6]. The interior of the cask was modeled as filled with water.

The fuel had an initial enrichment of 4 wt % ^{235}U and was burned to 40 GWd/MTU. The STARBUCS [7] sequence in SCALE 5 was used to generate 18 axial location-dependent burned fuel compositions. The STARBUCS sequence and available input parameters are discussed in Ref. [7]. The normalized burnup profile from Table 5 of Ref. [6] was used. The fuel burnup was modeled at a power density of 40 MW/MTU for 1000 d, with a postshutdown cooling period of 5 years. The fuel burnup calculations model the depletion of the ^{235}U and the in-growth of actinide and fission product nuclides. From the depletion calculations, fuel compositions for the following nuclides were retained for the criticality calculations: ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Rh, ^{109}Ag, ^{133}Cs, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{151}Sm, ^{152}Sm, ^{143}Nd, ^{145}Nd, ^{151}Eu, ^{153}Eu, and ^{155}Gd.

Sensitivity analysis for the GBC-32 cask was performed with TSUNAMI-3D and was checked using direct perturbation calculations. As is noted in the SCALE documentation, the TSUNAMI-3D version distributed with SCALE 5 will not work if the model has more than 50 nuclides that have resonance information. Each occurrence of the same resonance nuclide in multiple mixtures is counted toward this limit. The GBC-32 cask model included different mixtures of uranium, plutonium, other actinides, and 15 fission products in each of 18 axial zones. There were in excess of 500 resonance nuclides in all mixtures. Consequently, special development versions of some of the TSUNAMI-3D sequence programs were created by the SCALE development staff to support this work. The 50-resonance-nuclide limitation will be removed in SCALE 5.1. However, users will need to keep this 50-nuclide limit in mind during model development with SCALE 5.

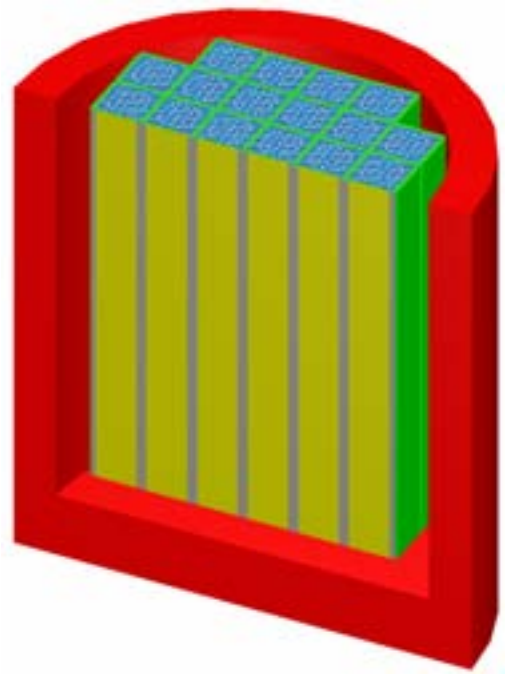

FIG. 2. GBC-32 cask model.

5. Comparison with critical experiments

Sensitivity analyses have been performed by ORNL staff for more than 1,000 critical configurations. The sensitivity data files have been accumulated as a resource for identifying critical configurations that may be useful for validation studies. The critical configurations are primarily from the IHECSBE [5]. At the time this work was presented at the International Atomic Energy Agency (IAEA) meeting in London, the collection of sensitivity data files included 170 ^{233}U, 150 high-enrichment uranium (HEU), 4 intermediate-enrichment uranium (IEU), 256 low-enrichment uranium (LEU), 197 Pu, 201 mixed-oxide (MOX), and 156 MOX configurations from the French Haut Taux de Combustion (HTC) experiment series.

The French HTC experiment data were purchased from the French under an agreement that limits release of the information. This series was designed to support validation of actinide-only burnup credit. The uranium and plutonium compositions of the rods were selected to be consistent with PWR U(4.5 wt %)O_2 rods with 37,500 MWd/MTU burnup. The French categorized these experiments into four groups. The first group was 18 configurations involving a single square-pitched array of rods with rod pitch varying from 1.3 to 2.3 cm. The arrays were flooded and reflected with clean water. The second group was 41 configurations that were similar to the first group except that the water used as moderator and reflector included either boron or gadolinium in solution. The third group was 26 configurations with the rods arranged into four assemblies that were arranged in a 2×2 array. The spacing between assemblies was varied, and some of the assemblies had borated steel, Boral®, or cadmium plates attached to the sides of the four assemblies. The fourth group was 71 configurations similar to the group 3 configurations except thick steel or lead shields were placed around the outside of the 2×2 array of fuel assemblies.

One of the primary objectives of the work reported in this paper was to evaluate the applicability of the HTC experiments to burnup credit calculations.

The TSUNAMI-IP code from SCALE 5 was used to compare the sensitivity data file for the GBC-32 cask with the sensitivity data files for 1134 critical configurations. The results from this comparison are presented in Fig. 3.

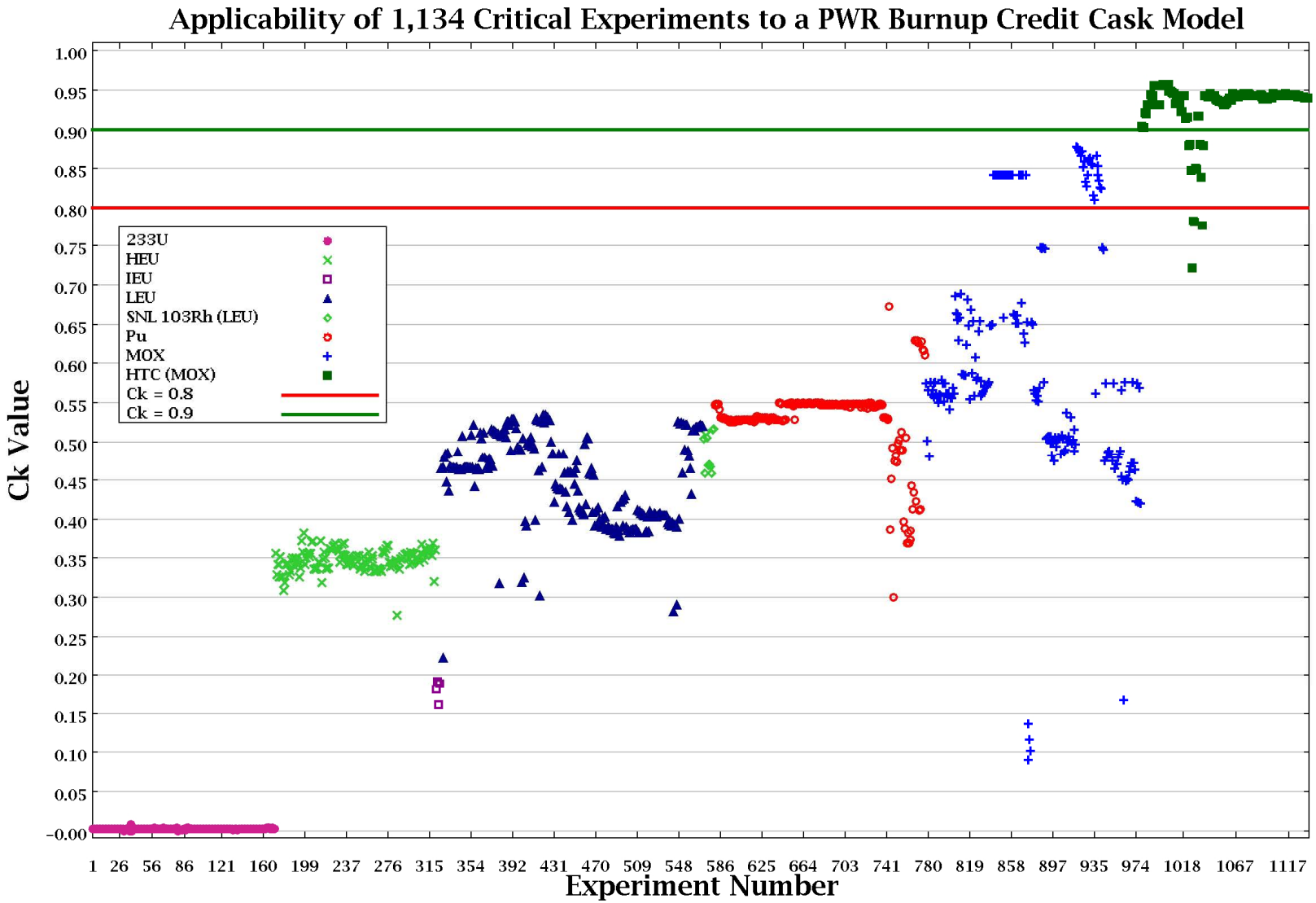

FIG. 3. Sensitivity/Uncertainty analysis results.

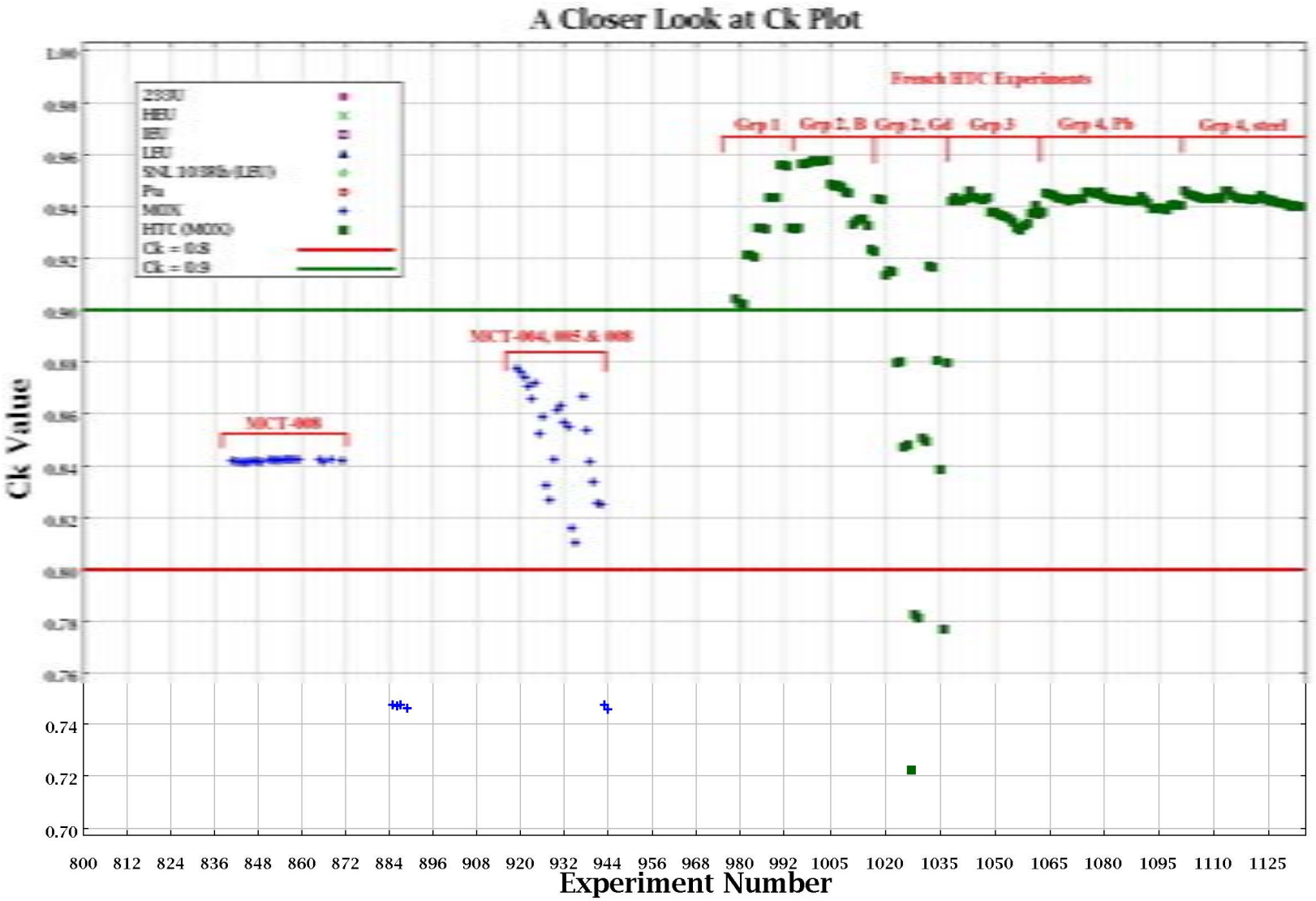

FIG. 4. Closer look at S/U analysis results.

The critical configurations are grouped by type of fissionable material. The trend of c_k values with fissionable material composition appears reasonable. The GBC-32 model has LEU mixed with Pu. The nearly zero c_k values for ^{233}U configurations show that these configurations have very little in common with the GBC-32. The HEU, IEU, and LEU have ^{235}U, but no Pu. Consequently, their c_k values are higher than for the ^{233}U configurations but are still significantly below the 0.8 cut-off value for being considered marginally applicable. The MOX critical configurations are the most similar to the GBC-32 model. Figure 4 shows a closer look at the MOX data from Fig. 3 and includes some annotations identifying specific sets of critical configurations.

The results for the HTC MOX configurations show that these configurations are generally very similar to the burned fuel in the GBC-32 cask model. The HTC configurations with lower c_k values are from HTC group 2, and all have a significant quantity of gadolinium dissolved in the moderator/reflector. The amount of gadolinium present far exceeds that present in the burned fuel as a fission product. The results from the S/U analyses confirm the value of the HTC MOX critical configurations for validation of burnup credit calculations. The set of MOX critical configurations considered in this study is not complete. There are additional configurations in the IHECSBE [5] not considered in this study that are expected to be at least marginally applicable.

Of the 1134 critical configurations considered, 937 had c_k values lower than 0.8 and, by the guidelines presented earlier in this paper, would not be considered applicable to the validation of burnup credit cask calculations. Table I below presents the S/U analysis results for the 1134 critical configurations considered. Of the 978 non-HTC experiment configurations, none were identified as applicable ($c_k \geq 0.9$), 45 configurations were marginally applicable ($0.9 > c_k \geq 0.8$), and 933 were not applicable. Of the 156 HTC configurations, 143 were applicable, 9 were marginally applicable, and 4 were not applicable.

TABLE I. SENSITIVITY/UNCERTAINTY ANALYSIS RESULTS FOR 1134 CRITICAL CONFIGURATIONS

	^{233}U	HEU	IEU	LEU	Pu	MOX	HTC	All
$c_k \geq 0.95$	0	0	0	0	0	0	11	11
$c_k \geq 0.9$	0	0	0	0	0	0	143	143
$c_k \geq 0.8$	0	0	0	0	0	45	152	197
$c_k < 0.8$	170	150	4	256	197	156	4	937
Total	170	150	4	256	197	201	156	1134

5.1. Coverage

An additional consideration is whether all of the nuclides in an application are represented in the critical configurations. If they are not present in the experiments, any bias associated with the missing nuclides will not be present in the bias calculated using the critical experiments. If the missing nuclides are a significant contributor to the k_{eff} of the application, the c_k values for the experiments should as a result be lower. If they do not contribute significantly to the application multiplication factor, then any associated bias should be insignificant, too. The degree to which the experiments cover the sensitivities for a nuclide is referred to as "coverage." If the experiment or group of experiments have sensitivity profiles that are at least as large as those for the application at all neutron energies, the application is considered covered by the experiments.

The most important fission product for burnup credit is ^{149}Sm. With current S/U methods, critical experiments should have a c_k value of at least 0.9 and provide coverage for all significant nuclides. In this context, significant means that the presence of the nuclide has a statistically significant effect on the application k_{eff} value. Of the 1134 critical configurations evaluated for applicability, none containing ^{149}Sm had a c_k value greater than or equal to 0.8. However, the critical configurations did include cases 8 through 18 of LEU-COMP-THERM-050 from the 2004 IHECSBE [5], which include

^{149}Sm in solution in a tank in the middle of an array of LEU fuel rods. These critical configurations are experiments 560 through 570 in Fig. 3. Figure 5 shows the ^{149}Sm sensitivity profiles for 11 cases from LCT-050 and for the GBC-32 cask loaded with fuel burned to 40 GWd/MTU. The thick black curve shows the total sensitivity curve for ^{149}Sm in the GBC-32 cask. The other curves in Fig. 5 show the ^{149}Sm sensitivity profiles for cases 8 through 18 of LCT-050. Note that the LCT-050 ^{149}Sm curves show that the experiments are at least as sensitive to the presence of ^{149}Sm as is the GBC-32 cask. Also note that the magnitude and shape of the LCT-050 curves are similar to those from the GBC-32. If the c_k value had been high enough, the user could have some confidence that any bias caused by the presence of ^{149}Sm would be adequately included in the overall bias. However, since the c_k values of the LCT-050 configurations are around 0.45, a ^{149}Sm bias calculated using the LCT-050 configurations may not be appropriate for the GBC-32. If the LCT-050 configurations were modified to contain fissionable materials, moderators, and absorbers more similar to the GBC-32, thus yielding a higher c_k value, the ^{149}Sm bias calculated from the modified experiments might be significantly different.

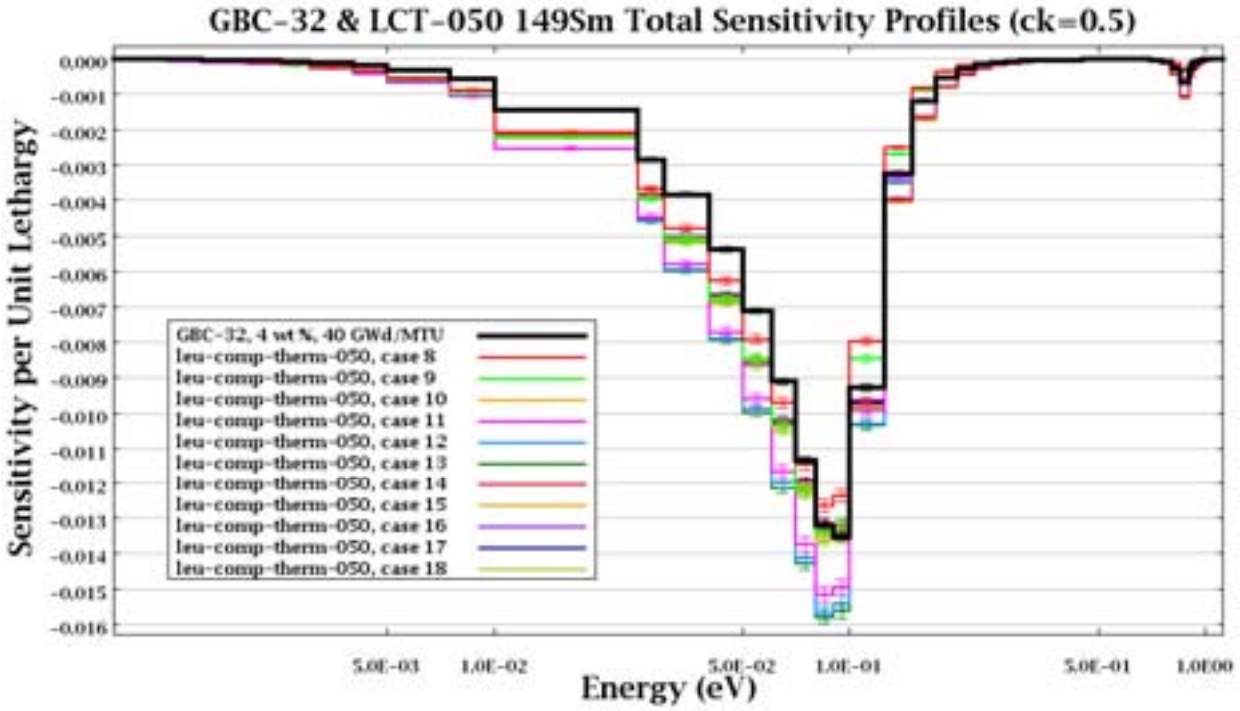

FIG. 5. ^{149}Sm sensitivity profile coverage.

The next coverage example examines the ^{155}Gd present in the GBC-32 and in some of the French HTC experiments. Note from Fig. 4 that the c_k values for the group 2 Gd experiments vary significantly from the non-Gd HTC experiments. From the S/U analysis, the c_k values decreased as the Gd concentration increased. Figure 6 shows the ^{155}Gd sensitivity curves from the GBC-32 and from a subset of the HTC group 2 Gd solution experiments. The figure shows that the sensitivity profiles from the HTC group 2 Gd configurations completely cover the ^{155}Gd sensitivity profile from the GBC-32 model. The peak sensitivity for the HTC group 2, Gd case 8 configuration is nearly 4 times as great as the peak sensitivity for the GBC-32. Consequently, the ^{155}Gd bias calculated using this experiment could be significantly different than a bias calculated using experiments having more similar ^{155}Gd sensitivity profiles.

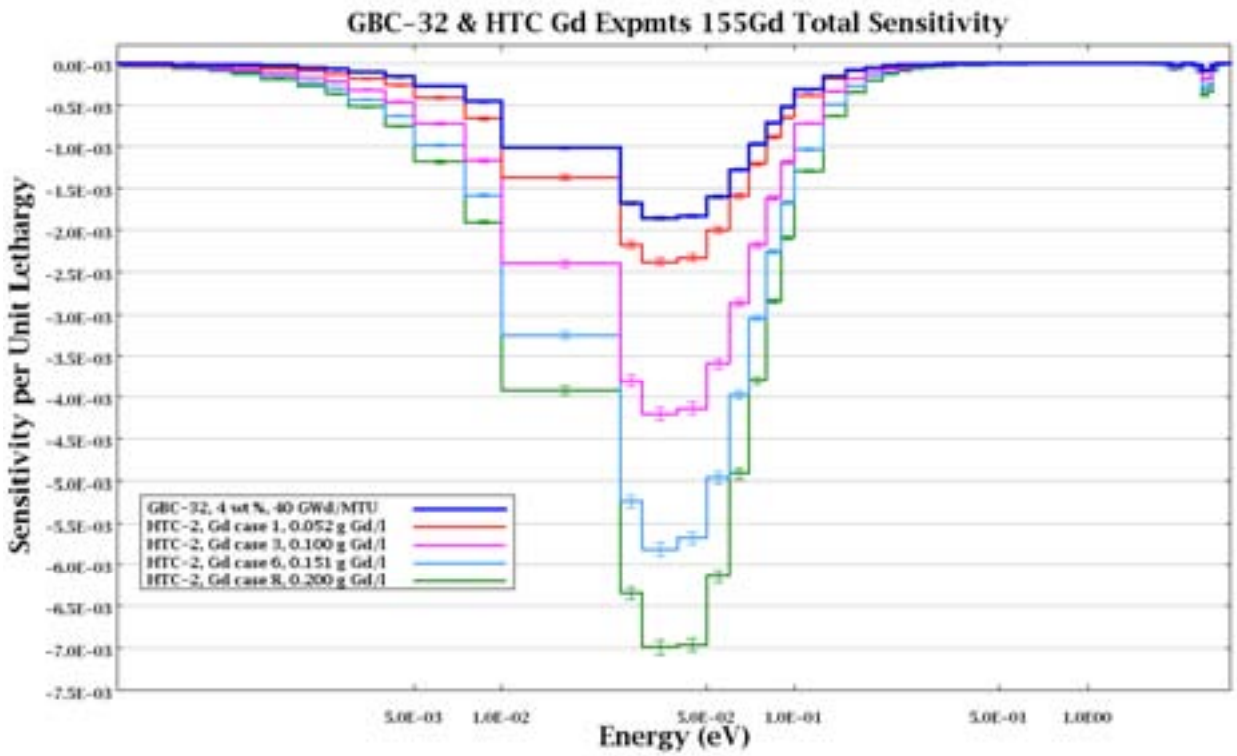

FIG. 6. ^{155}Gd sensitivity profile coverage.

As a final example of how coverage may be used, let us examine the ^{103}Rh sensitivity profiles from the GBC-32 model and a series critical experiments performed at Sandia National Laboratories (SNL). These experiments, shown as a separate group in Fig. 3, are documented in the 2005 version of the IHECSBE [5] as evaluation LEU-COMP-THERM-079. These experiments were designed to support validation of ^{103}Rh in burnup credit applications. The experiments have thin (25-, 50- or 100-microns thick) ^{103}Rh foils stacked between the U(4.32 wt %)O_2 pellets in some of the rods in a water moderated and reflected triangular pitched (2.0- or 2.8-cm) array. Figure 7 shows the ^{103}Rh sensitivity profiles for the GBC-32 and four of the SNL ^{103}Rh experiments. Generally, the ^{103}Rh coverage provided by the SNL experiments is good, except in the 1- to 2-eV range. In this neutron energy range it appears that the GBC-32 model is significantly more sensitive than the experiments. Consequently, it is not clear that a bias calculated using these experiments would be correct. Note that for the SNL experiments, the sensitivity around the 1- to 2-eV range increases with decreasing foil thickness. This indicates that even thinner or more lightly loaded ^{103}Rh foils might produce a more similar sensitivity profile. Based on these observations, conceptual critical experiment models were developed using thinner foils. The resulting sensitivity profiles are shown in Fig. 8. This figure shows that 5-micron thick foils in the modified experiment model produced a ^{103}Rh sensitivity profile that is very similar to the GBC-32 cask model ^{103}Rh sensitivity profile.

This exercise demonstrated how the S/U tools could be used to design critical experiments that are intended to support a specific application, such as a high-capacity burnup credit cask model. The S/U tools were not available when the SNL ^{103}Rh experiments were designed and conducted. The S/U tools will be used to evaluate and, where appropriate, modify future SNL fission product experiment designs. This will maximize the usefulness of the experiments and maximize the return on the experiment sponsor's investment.

Due primarily to the absence of Pu in the SNL ^{103}Rh experiments, the c_k values generated by comparing the experiments to the GBC-32 cask model were around 0.5 and would not be considered "applicable" for a traditional calculation of bias and bias uncertainty. Methods based upon Generalized Linear Least-Squares Methods [8] are being developed at ORNL to utilize the relevant information from experiments with relatively low c_k values for validation.

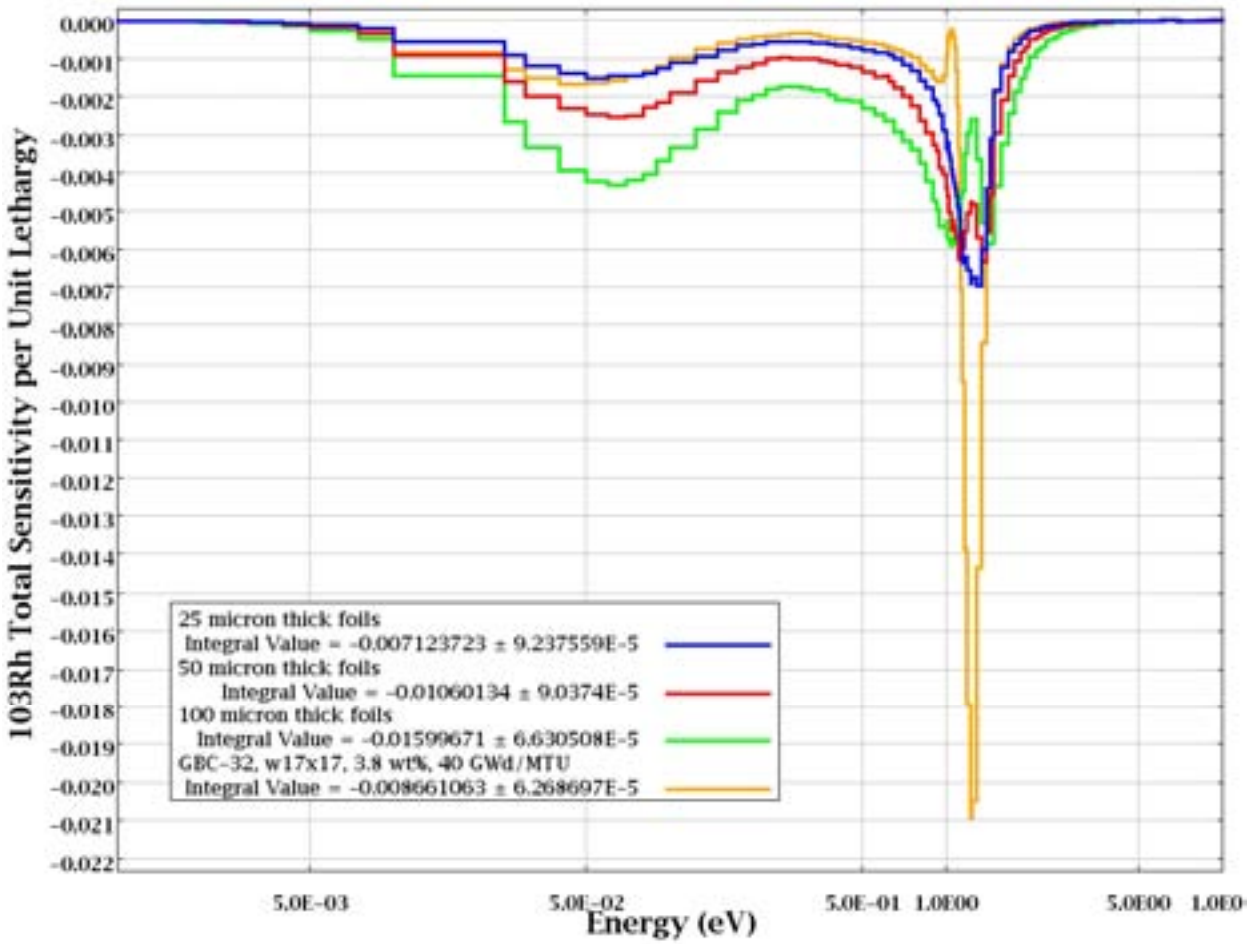

FIG. 7. GBC-32 and SNL experiment ^{103}Rh sensitivity profiles.

6. Conclusions

The objective of the work reported in this paper was to use S/U analysis to identify critical configurations that could be used to validate burnup credit calculations. Sensitivity calculations have been performed on a generic burnup credit cask and on 1134 critical configurations, 156 of which are from the French HTC series. The S/U analysis shows that 45 of the non-HTC critical configurations are marginally applicable for validation. The analyses also show that 143 of the 156 HTC experiments are applicable for validation, and an additional 9 HTC critical configurations are marginally applicable. The HTC experiments provide an excellent source of data for validating burnup credit calculations.

Discussion of the concept of coverage was included in this paper and an example of how S/U tools could be used in the review and design of critical experiments.

Future S/U analysis work will include evaluation of French critical experiments with fission products, evaluation of casks loaded with spent boiling water reactor fuel, design and evaluation of future SNL fission product critical experiments, and application of S/U tools to commercial reactor criticals and to the REBUS experiment [9].

Acknowledgment

This work was sponsored by the Office of National Transportation in the U.S. Department of Energy Office of Civilian Radioactive Waste Management.

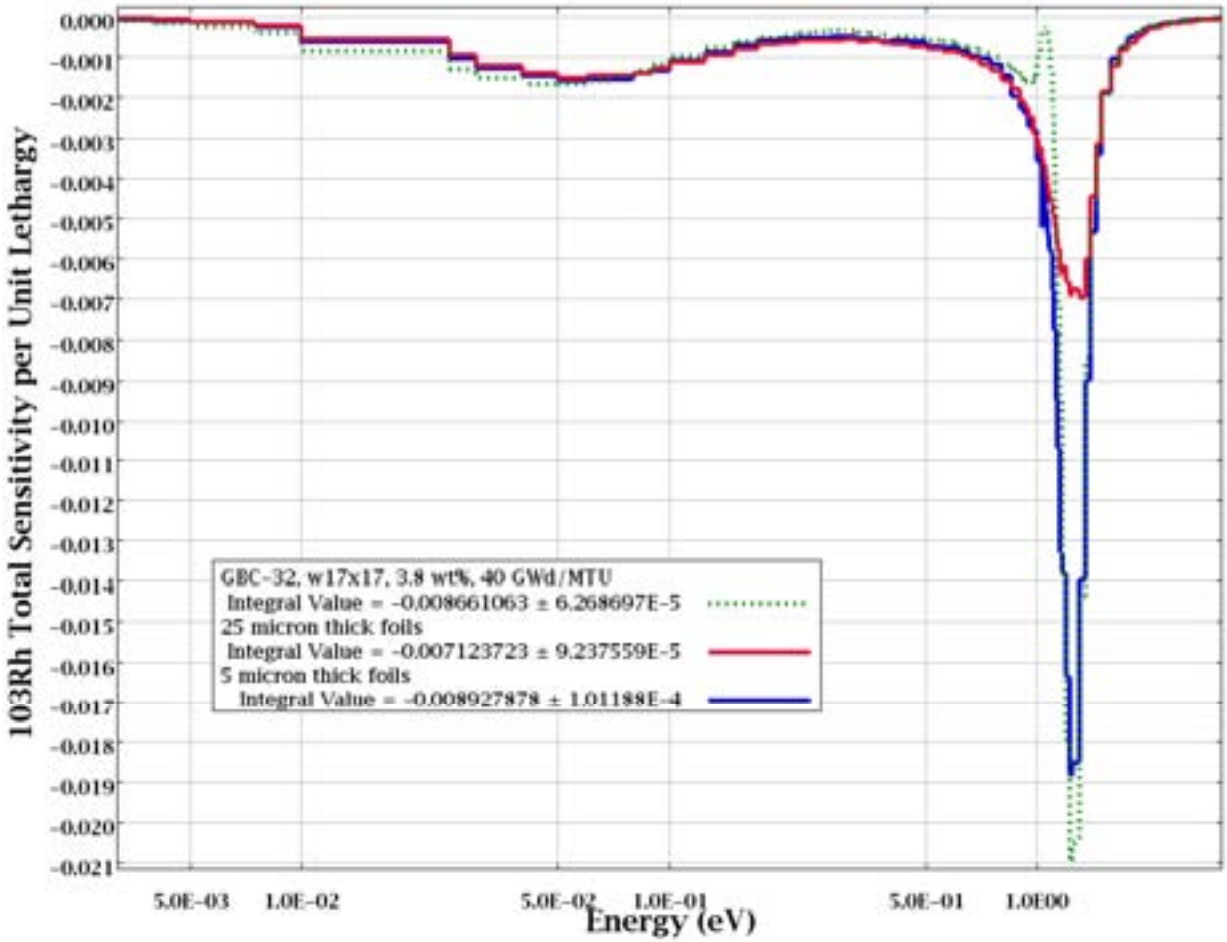

FIG. 8. ^{103}Rh sensitivity profiles with modified SNL ^{103}Rh experiment.

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Investigation of the influence of the plutonium and uranium cross section uncertainties in burnup credit application

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Abstract. The uncertainty of the criticality calculations arising from the error of the nuclear cross section libraries is studied based on a number of calculations for critical facilities. Several UO_2 and MOX critical benchmark experiments selected from the ICSBE Handbook are investigated using the major data libraries. Using trending analysis, a subcriticality safety limit is derived for fresh VVER type UO_2 fuel. The possibility of using the available MOX critical experiments for deriving similar subcriticality safety limit for burnup credit application is investigated. Qualitative arguments suggest that stand-alone use of these MOX experiments is insufficient for this purpose. Use of sensitivity/uncertainty method and additional experiments is planned in the future.

1. Background

According the Hungarian regulation, the criticality safety analysis of a transport/storage device should be based on the fresh fuel assumption, and the subcriticality limit should be ensured by a conservative safety margin, covering all kind of uncertainties. A major source of the uncertainties is the error in the nuclear cross section libraries. Recently, a project was continued to investigate the influence of this error in the case of the ENDF/B-VI.2, ENDF/B-VI.8, JENDL 3.2, JENDL 3.3 and JEF 2.2 libraries. The project was supported by the Hungarian Atomic Energy Authority. The primary purpose of the work was to derive a subcriticality limit for analysis with fresh fuel, however, efforts were made to investigate the possibility of deriving a safety margin for actinides only burnup credit safety analysis. This effort was motivated by the realization, that using the fresh fuel assumption the presently used transport/storage facilities can't be used with full capacity in some cases, if the planned new, advanced fuel types will be introduced. (These cases will be specified in the next point.) For the criticality, the most important elements are the uranium and plutonium, so investigating the possibility of using MOX critical experiments for this purpose was quite evident. This approach can not replace the critical experiments with more realistic fuel composition by any mean; however, it may give some information about the influence of the error in the cross sections of these isotopes in the different nuclear data libraries.
For this purpose, calculations for MOX experiments selected from the ICSBE Handbook [4] were performed and qualitative considerations were made regarding their applicability.

2. Sample applications where burnup credit usage would be beneficial

Paks NPP, Hungary has four VVER-440 units. These units have hexagonal cores with pitch p=14.7 cm. The outer diameter of a fuel assembly is 14.4 cm. One assembly contains 126 fuel pins. The pins in the original assemblies have 1.22 cm pitch and 3.6 % maximal enrichment. The enrichment was increased recently up to 3.82 % with radial zoning, and the change of lattice pitch from 1.22 cm to 1.23 cm is now considering.

After the removal from the core the fuel is loaded into the wet storage pool at the reactor. The pool consists from two parts. One of them has 22.5 cm spacing between the assemblies, the other is a compact storage part containing boron steel plates and so the spacing is 16 cm. After some years of cooling the spent assemblies are transported either to the dry interim storage module or into the Russian Federation. Two types of cask used for these transport actions: C-30 and TK-6. The maximal storage capacity of both casks is 30 assemblies.

If the criticality analysis is based on fresh fuel assumption, there are two cases, when the $k_{eff} < 0.95$ subcriticality requirements can be insured only with technical measures: the compact storage pool with normal water density and the TK-6 cask in the case of optimal moderation. The compact storage pool has its maximal multiplication properties with water density 1 g/cm^3. The planned new assemblies (3.82 enrichment, 1.23 cm pitch) can be loaded into the compact storage only, if absorber assemblies are loaded into every 12th positions. (This number was derived by the assumption, that all technological parameters have their "worst" values.)

In the TK-6 cask the spent fuel assemblies are stored under water, and there is a water-drain cock at the bottom of the cask. In principal, the accident that the cock is broken can not be excluded. In this case the water would leak out at the bottom of the cask, the water level and the pressure above the water level would decrease, so a mixture of steam and boiling water could occur. This makes necessary the investigation of the optimal moderation conditions. The results of the analysis show that the cask containing 30 fuel assemblies would became supercritical. The technical measures currently applied are:

(a) loading only 24 assemblies into the cask, or

(b) loading 27 assemblies and 3 absorber assemblies. The criticality analysis shows, that with this loading strategy the cask would be subcritical in the optimal moderation conditions.

However, these technical measures decrease the transport/storage capacity of these facilities. Simplified calculations have shown that the meeting of subcriticality requirements for the compact storage pool and the TK-6 cask could be insured taking into account the real uranium and plutonium content corresponding approximately to 15 MWd/kgU and 30 MWd/kgU burnup.

3. Derivation of subcriticality limit

The subcriticality of a transport/storage device should be insured by sufficient margin accounting for all uncertainties of the criticality calculation process, including the technical uncertainties and the uncertainties of the computational model. The uncertainties of the computational model arise from the approximations introduced while modeling the application, and from the error of the selected code and the selected nuclear data library. In this paper the second source of error will be studied. For insuring the proper account of this kind of uncertainties, validation of the used computational tool (code + nuclear data) should be performed. In the validation process, calculated results should be correlated with experiments to establish bias and uncertainties and establish a reactivity margin.

A traditionally used approach is performing a series of criticality calculations for a set of selected critical benchmark configurations, which has neutronic characteristics close to the application(s) in question. Based on the statistical investigation of deviations between the calculated and measured results an appropriate margin can be derived for the used nuclear data library and criticality code. The selected benchmarks should be similar to the application under consideration, and the possible dependence of the calculated margin on the relevant neutronic quantities (enrichment, lattice pitch, hydrogen/uranium ratio etc.) should be examined. Based on this approach two methods for derivation an upper subcriticality limit (USL) were developed in ORNL. The USLSTAT code implementing these methods was also developed in Oak Ridge. They are described in details in [1]. Since a few years, one of these methods, the confidence band with administrative margin approach has been used

for criticality safety analysis in KFKI AERI. In its general form, the derived upper safety limit can be written as

$USL(x) = 1- \Delta k_m -W - \beta(x)$,

where x is the physical parameter which gives the most conservative USL in the trending analysis, Δk_m is the administrative safety margin (usually 0.05), $\beta(x)$ is the bias, i.e. the difference between the linear fit and the measured k_{eff} values. W is the confidence band width for the lower confidence limit. On a specified confidence level it is determined by the deviation of calculated and measured k_{eff} values.

The key point of making this type of analysis is that a sufficient number of experiments should be available which are judged to be similar to the application. This requirement contains a degree of subjectivity, because the analyst should be decide, based on his experience, that the benchmarks are similar enough or not to use in the analysis. Also, this method relies upon to the analyst's judgment about the range of applicability of the derived safety limits.

This problem can be handled relatively reliably in cases, when there is a plenty of appropriate benchmark experiment with similar or identical primary physical parameters (enrichment, material compositions, lattice pitch, cladding material etc.) to the application. However, in those cases when the physical properties of the application(s) are covered only partly with the benchmark experiments, it may be difficult to decide based on intuitive judgment whether the USL derived from the particular benchmark set is applicable or not. In the last years, new methods were developed in Oak Ridge National Laboratory to quantify the similarity and help to decide that a combination of experiments is adequate for benchmarking an application or not [2] [3]. This was included into the last release of SCALE program package.

The usage of these methods is planned in the near future in KFKI AERI, but they are still not in use. Because the MOX experiments described in the ICSBE Handbook are somewhat different from a burnup credit application, in this paper some simply physical considerations will be used for the investigation, whether an USL derived by the confidence band with administrative margin from these experiments is applicable to burnup credit applications or not.

4. Selected libraries and the investigated criticality benchmark experiments

The criticality calculations were performed by the MCNP4C code using different libraries. In the following the used libraries are given together with the institute, where the "basic" library was transformed into MCNP format.

- ENDF/B-VI.2 Nuclear Data Team, Los Alamos National Laboratory, USA
- ENDF/B-VI.8 Royal Institute of Technology, Stockholm, Sweden
- JENDL 3.2 Royal Institute of Technology, Stockholm, Sweden
- JENDL 3.3 Japanese Atomic Energy Research Institute (JAERI), Japan
- JEF 2.2 ENEA, Bologna, Italy

Four groups of critical experiments were selected from the ICSBE handbook:

- thermal, low enriched UO_2 in homogeneous lattice (86 experiments)
- thermal, low enriched UO2, boron steel plates in the lattice (35 experiments)
- thermal, MOX fuel in homogeneous lattice (44 experiments)
- thermal, MOX fuel, absorber rod in the center of lattice (88 experiments)

While there is a plenty of appropriate benchmark experiment, if the fresh fuel assumption is used (i.e. there are a lot of critical experiment with fresh UO_2 fuel), practically there is no experiment for burned fuel in the public domain. Because for the criticality the most important elements are the uranium and

plutonium, a possibility is the using MOX critical experiments for this purpose. This approach can not replace the critical experiments with more realistic fuel composition by any mean; however, it may give some information about the influence of the error in the cross sections of these isotopes in the different nuclear data libraries.

The behaviour of the k_{eff} values calculated by different libraries for the different configurations are overviewed in Table 1.

TABLE 1. OVERVIEW OF THE CALCULATED k_{eff} VALUES. σ_k IS THE SPREAD OF THE k_{eff} VALUES CALCULATED.

UO_2 fuel in homogeneous lattice					
	ENDF/B-VI.2	ENDF/B-VI.8	JENDL 3.2	JENDL 3.3	JEF 2.2
$k_{average}$	0.9957	0.9961	1.0037	0.9994	1.0004
σ_k	0.0056	0.0056	0.0051	0.0053	0.0050
k_{min}	0.9835	0.9820	0.9915	0.9847	0.9866
k_{max}	1.0101	1.0110	1.0161	1.0119	1.0116
UO_2 fuel with boon steel plate absorbers					
	ENDF/B-VI.2	ENDF/B-VI.8	JENDL 3.2	JENDL 3.3	JEF 2.2
$k_{average}$	0.9952	0.9964	1.0038	1.0003	1.0003
σ_k	0.0011	0.0013	0.0014	0.0014	0.0018
k_{min}	0.9929	0.9929	1.0008	0.9972	0.9960
k_{max}	0.9975	0.9995	1.0059	1.0030	1.0035
MOX fuel in homogeneous lattice					
	ENDF/B-VI.2	ENDF/B-VI.8	JENDL 3.2	JENDL 3.3	JEF 2.2
$k_{average}$	0.9945	0.9954	0.9999	0.9999	0.9988
σ_k	0.0035	0.0033	0.0029	0.0029	0.0022
k_{min}	0.9872	0.9882	0.9946	0.9945	0.9946
k_{max}	1.0014	1.0024	1.0062	1.0054	1.0055
MOX fuel in lattice, absorber rod at the centre					
	ENDF/B-VI.2	ENDF/B-VI.8	JENDL 3.2	JENDL 3.3	JEF 2.2
$k_{average}$	0.9888	0.9898	0.9943	0.9942	0.9927
σ_k	0.0028	0.0029	0.0026	0.0027	0.0027
k_{min}	0.9833	0.9843	0.9895	0.9894	0.9876
k_{max}	1.0010	1.0033	1.0075	1.0076	1.0056

Note: The standard deviation of the individual multiplication factor is generally 5×10^{-4} or less.

Some general conclusions can be drawn from this Table:

- using ENDF/B-VI.2 or ENDF/B-VI.8 gives approximately 0.001 difference
- using JENDL 3.2 or JENDL 3.3 gives approximately 0.003-0.004 difference for UO_2, and negligible difference for MOX

- using ENDF/B-VI or JENDL libraries results difference about 0.005, the larger results is given by JENDL
- generally the JEF 2.2 gives results close to those calculated by JENDL 3.3, except the case of MOX with absorber rod, when the difference is about 0.0025
- for homogeneous MOX lattices all but JENDL3.3 libraries give lower result than for UO_2 lattices
- for cases of MOX fuel with absorber rod, all libraries underestimate the experimental results, the ENDF/B-VI libraries with more than one percent, the JENDL libraries with about half percent

From these observations we concluded, that the best overall agreement with the experiments is given by the JENDL3.3 and the subsequent analysis was performed by this library.

In cases, when it was applicable, the study of the results was performed by the USLSTAT code. Trending analysis was made according several variables. These variables were the uranium enrichment (ratio of the ^{235}U atoms to total number of uranium atoms), plutonium enrichment (ratio of fissionable plutonium atoms to the total number of plutonium atoms), H/X (ratio of number of hydrogen atoms and fissionable atoms), lattice pitch, outer clad diameter, and the energy of average lethargy causing fission (EALF). For these parameters the correlation coefficient for the calculated k_{eff} values were evaluated, and if the correlation was statistically significant, the upper safety limit due to this parameter was calculated. Finally, the most restrictive USL was determined.

UO_2 fuel in homogeneous lattice

In this case it was easy to select a suitable set of benchmarks, because there were a number of experiments with enrichment, lattice pitch, material composition, dimensions etc. close to the VVER-440 fuel. In an essential part of the examined experiments, the fuel was completely identical with those used in VVER-440 reactor. Other, similar experiments gave a sufficient range for trending analysis. The enrichment, lattice pitch and the $H/^{235}U$ ratio varied from 2.6 % to 5.12 %, from 0.8 cm to 2.54 cm and from 30 to 450. (The corresponding values for fresh fuel applications are 3.6-3.8 %, 1.22 or 1.23 cm and approximately 120 for normal water density). According our judgment this benchmark set is clearly suitable for derivation of a criticality safety limit.

Applying the above described procedure for this benchmark set it was found, that the USL is determined by the variation of $H/^{235}U$. The correlation coefficient is about 0.44, and the derived USL can be written as:

$U(x) = 0.9370 + 2.0685E\text{-}05 \times x$ ha $x < 229.71$ or

$U = 0.9417$ if $x > 229.71$

The value of $x=H/^{235}U$ for normal water density is about 120 for VVER-440 fuel, so the applicable limit is USL=0.9395.

UO_2 fuel with boon steel plate absorbers

Apparently, we have a number of experiments is this case. However, all the 36 experiments were performed on two facilities (six and twenty nine experiments); both of them having fixed enrichment and lattice pitch (4.31 % with 2.54 cm and 2.54 % with 1.96 cm). This might imply that the systematic error (arising almost necessarily from the description of the experiments) is the same for the majority of the experiments. The varying condition was the strength of the absorber, which was compensated by amount of fuel rods. Trending analysis against the absorber atoms is planned, but not yet has done. Now we think it is better not to use this set for USL derivation before further analysis planned in the future.

MOX fuel in homogeneous lattice

In this case there are also a number of experiments, but their characteristics covers only partly the neutronic features of the applications of interest (compact storage and TK-6 transport cask). The PuO_2 content of the MOX fuel varies from 1.5 % up to 20 % in these experiments. The lower bound of this interval is close to the plutonium content of fuel having burnup interesting for us, which is approximately 1 %. The "plutonium enrichment", i.e. the ratio of fissionable plutonium atoms to the total number of plutonium atoms is ranging approximately from 70 % to 90 %, which is applicable for our cases, where this quantity is about 75-77 %. The lattice pitch is larger in all experiments compared those which is real in VVER-440 fuel, but this could be extrapolated. However, in all but one experiment natural uranium dioxide was used, i.e. the ^{235}U content is approximately 0.7 percent only. In the single exceptional case depleted uranium was used with 0.22 percent ^{235}U content. The influence of this difference may be significant and should be examined more closely.

To investigate this phenomenon some quantities related to multiplication factor were calculated for the benchmark experiments and for the two investigated burnup credit applications. These were the fission, capture and neutron flux calculated in 5 energy groups. The following energy group boundaries were used: 0.1 eV, 10 eV, 10 keV, 100 keV and 20 MeV. The relative importance of different isotopes in fission and capture processes was also investigated. The graphical comparisons of the five group quantities are shown on Figures 1–3. This comparison suggests that these broad group characteristics of the applications are covered fairly well by the experiments.

We also tried to quantify the deviation of these five-group quantities. The relative distance between a benchmark spectra and an application can be defined as:

$$D^2 = \frac{\sum_{i=1}^{5}\left(x_i^2 - A_i^2\right)}{\sum_{i=1}^{5} A_i^{\,2}}$$

where x_i and A_i is the five group quantities for the benchmark and for one of the two application, respectively. On Figs. 4–6 this relative distance is shown as a function of benchmark number for fission, capture and neutron flux. Generally it can be said, that the five group quantities calculated for the benchmarks and for the applications are not very different, i.e. the there are some degree of similarity in these global course-group quantity. (It is worthy to note, that the relative distance shows a quite strong correlation with the H/X ratio. For example the correlation coefficient for the "fission distance" from compact storage is 0.77.)

Unfortunately, this is not the case with the relative importance of the uranium and plutonium in fission and capture. The ratio of number of fissions on plutonium and number of fissions on uranium, as well as the similar ratio for capture was evaluated for the benchmarks and for the two burnup credit applications. These ratios are shown on Figs. 7-8. As it can be seen from the Figure, this ratio is higher by a factor of 5-20 for fission and by a factor of 3-10 in the benchmark experiments, then in the burnup credit applications. This suggests that the uranium has a much less role in the selected MOX experiments, than in the investigated burnup credit application. Consequently, if we derive an upper safety limit by the traditional method from these experiments, the uncertainties of the plutonium would be overemphasized and the uncertainties of the uranium would be underemphasized. Combination of these benchmark set with other experiments may be useful, but it need further considerations.

MOX fuel with absorber rod at the centre

In the case of this set of experiments, practically the same holds for the five group quantities and for the reaction ratios, and consequently this benchmark group is also not suitable for derivation an USL

applicable for burnup credit application. Furthermore, all these 88 experiments are very similar to each other, as it was in the case of UO_2 fuel with absorber plates. They are all characterized with a few values of composition and lattice pitch, and the strong variation of the absorber rod is compensated by variation of fuel amount. This strong similarity is clearly illustrated on Fig.9, where the 5 group fission is shown for the 88 cases, but most curves are practically indistinguishable.

VENUS-2 experiment

The above described discrepancies obviously follow from the fact, that the uranium enrichment was much lower, than in the burnup credit application. A few publicly available MOX experiments exist with low enriched uranium; one of them is the VENUS-2 MOX core experiment. Its details were published as OECD/NEA benchmark. The arrangement of this experiment is quite different from the burnup credit application under consideration, because the core consists from three parts, each filled with different fuels: 3.3 % enriched UO_2 fuel, 4.0 % enriched UO_2 fuel and with MOX fuel containing 2.0 % enriched uranium and 2.7 wt % plutonium oxide. The plutonium on the MOX fuel is reactor grade quality, so it fissile isotope content corresponds to the desired applications. In spite of the essential difference in the geometrical arrangement, it can be demonstrated, that averaging over the whole core, the particular isotopes play a similar role, than in the two investigated application. The balance of fission and capture on different isotopes is shown on Fig. 10, averaged over all fuel pins (fission+capture normalised to 100 for all fuel). It can be seen, that the reaction rates on different isotopes are not far from the values found in the applications.

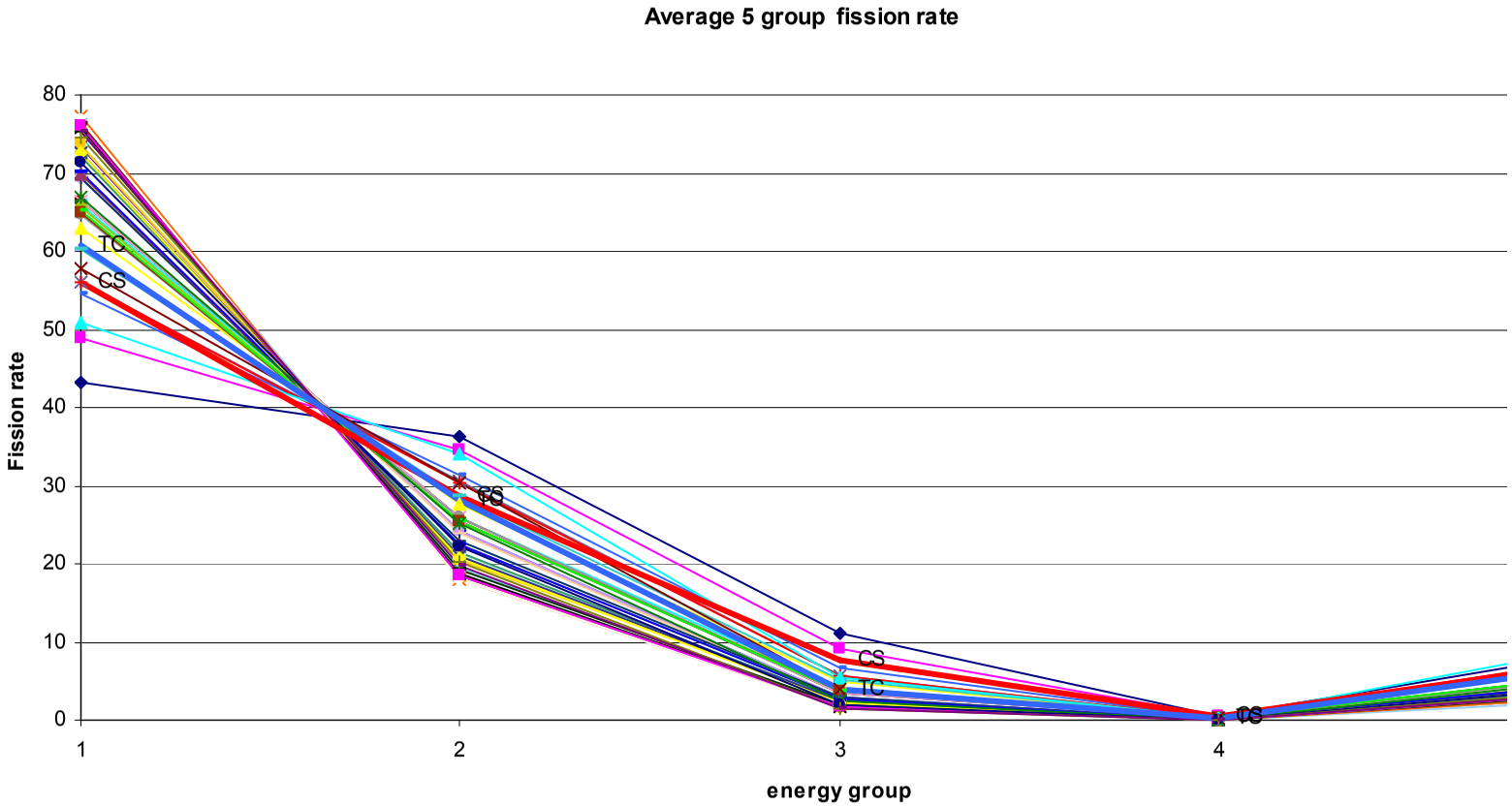

Fig. 1. Five group fission rate calculated for the homogeneous MOX benchmark experiments and for the two BC applications. The thick red line is the compact storage; the thick blue line is the TK-6 container.

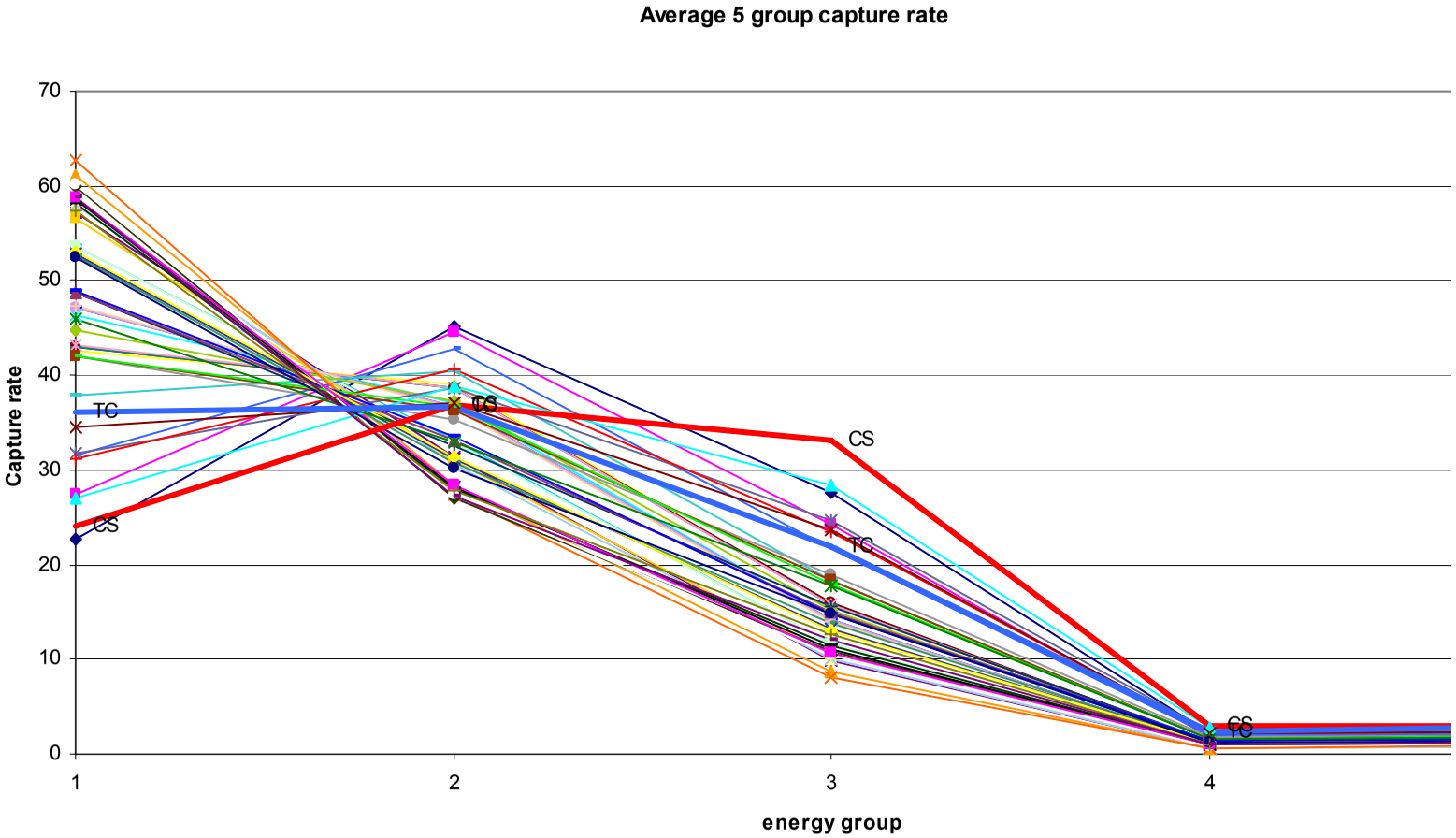

Fig. 2. Five group capture rate calculated for the homogeneous MOX benchmark ex periments and for the two BC applications. The thick red line is the compact storage, the thick blue line is the TK-6 container.

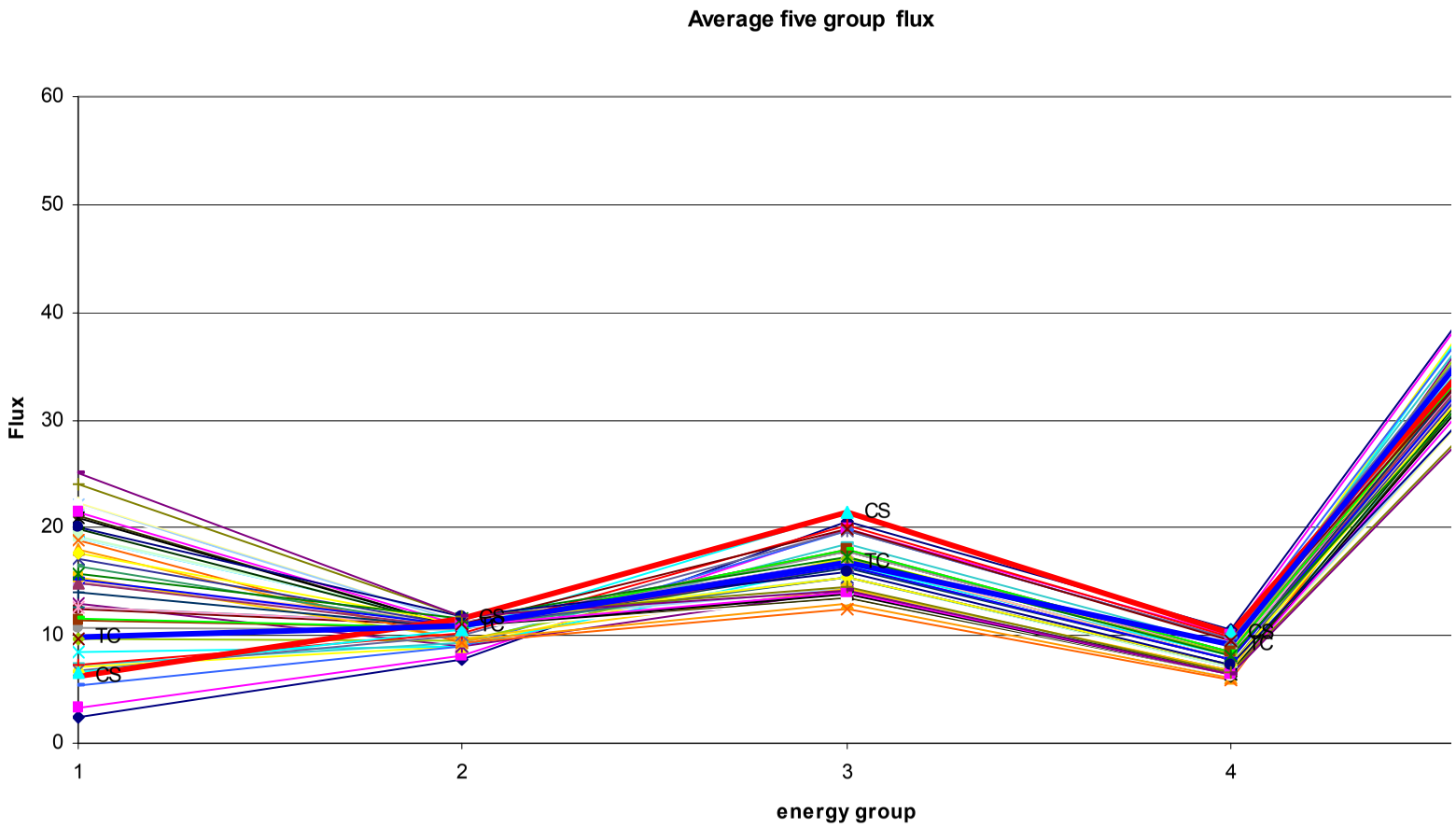

Fig. 3. Five group flux calculated for the homogeneous MOX benchmark experiments and for the two BC applications. The thick red line is the compact storage, the thick blue line is the TK-6 container.

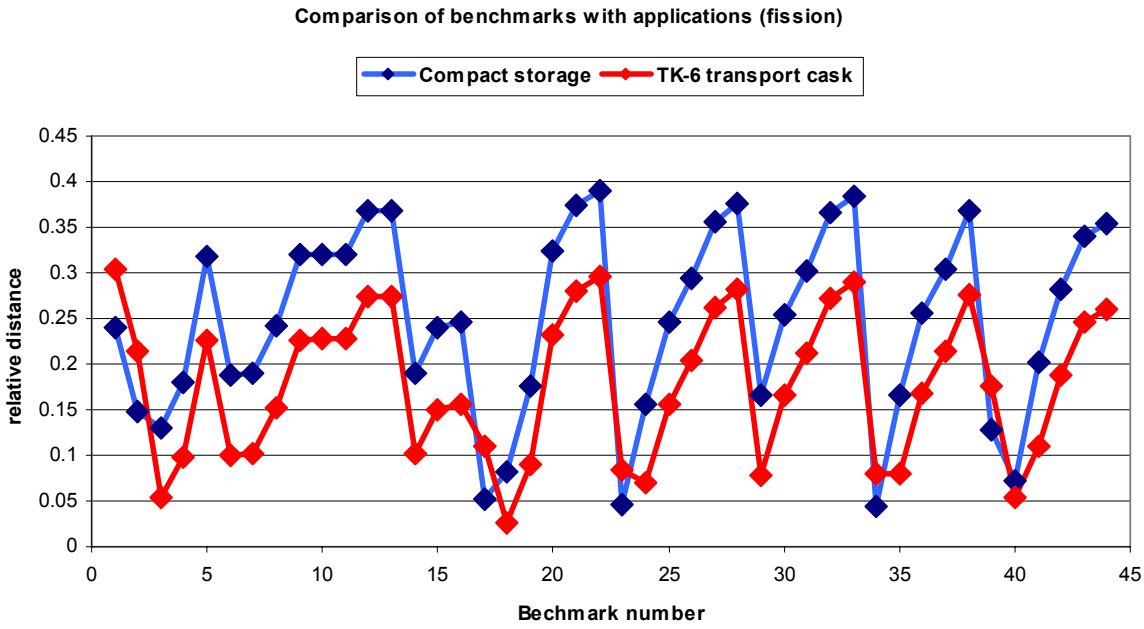

Fig. 4. Relative distance of homogeneous MOX benchmarks from applications for 5 group fission.

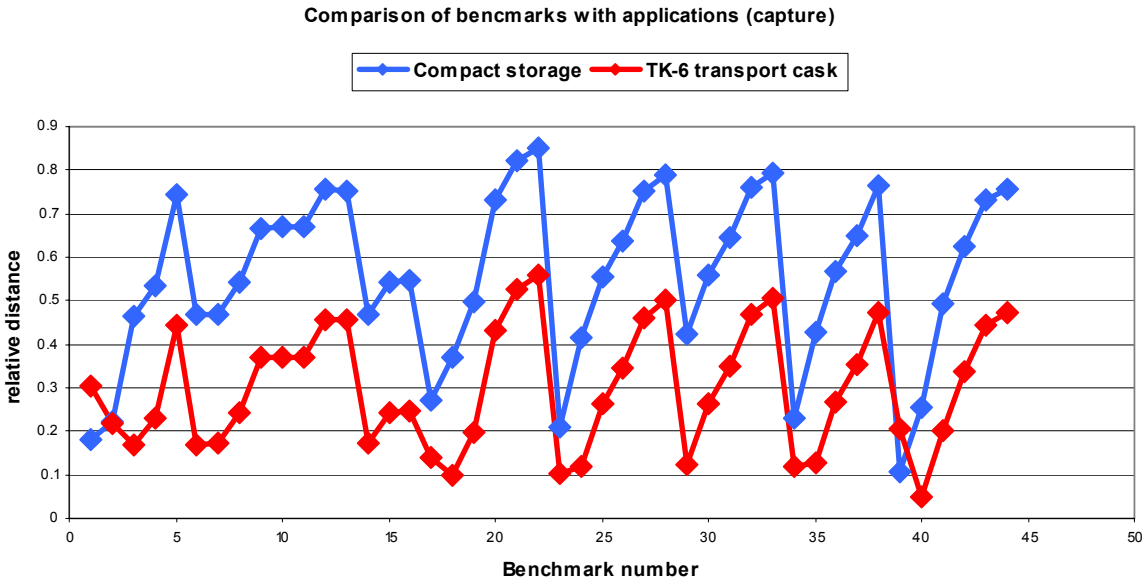

Fig. 5. Relative distance of homogeneous MOX benchmarks from applications for 5 group capture.

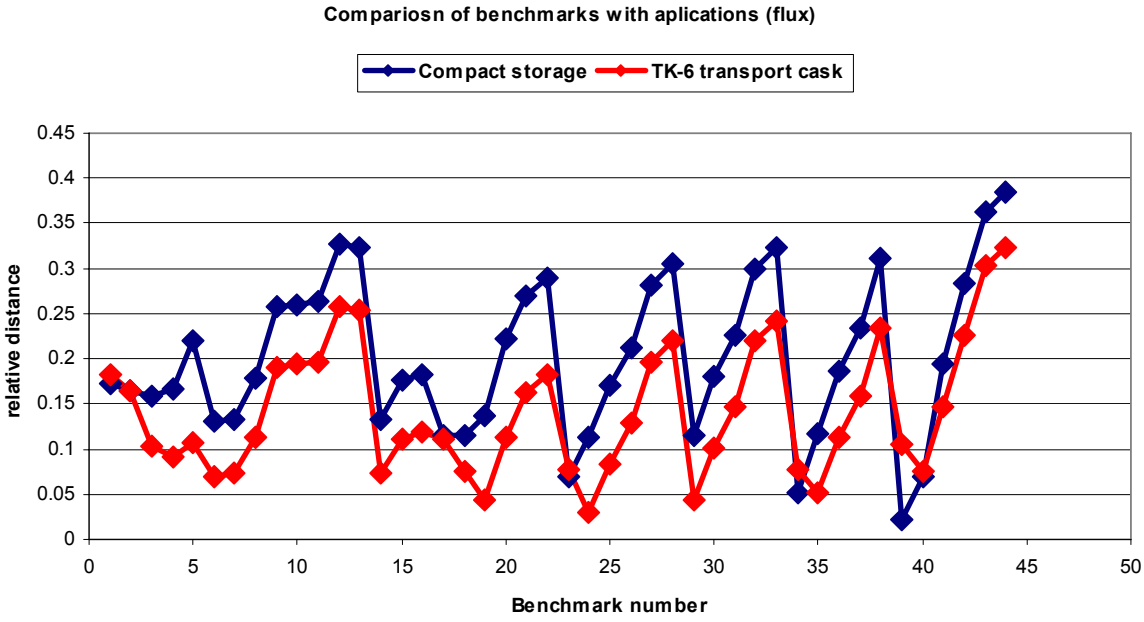

Fig. 6. Relative distance of benchmarks from applications for 5 group flux.

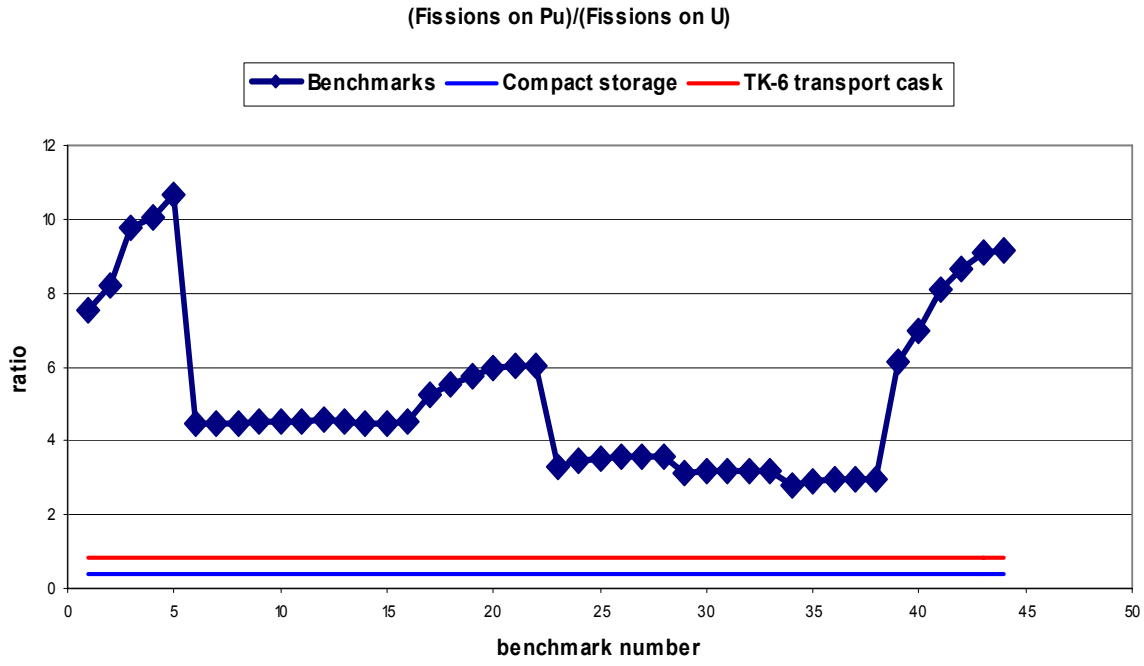

Fig. 7 Ratio of fissions happening on plutonium to the fission happening on uranium for homogeneous MOX benchmarks.

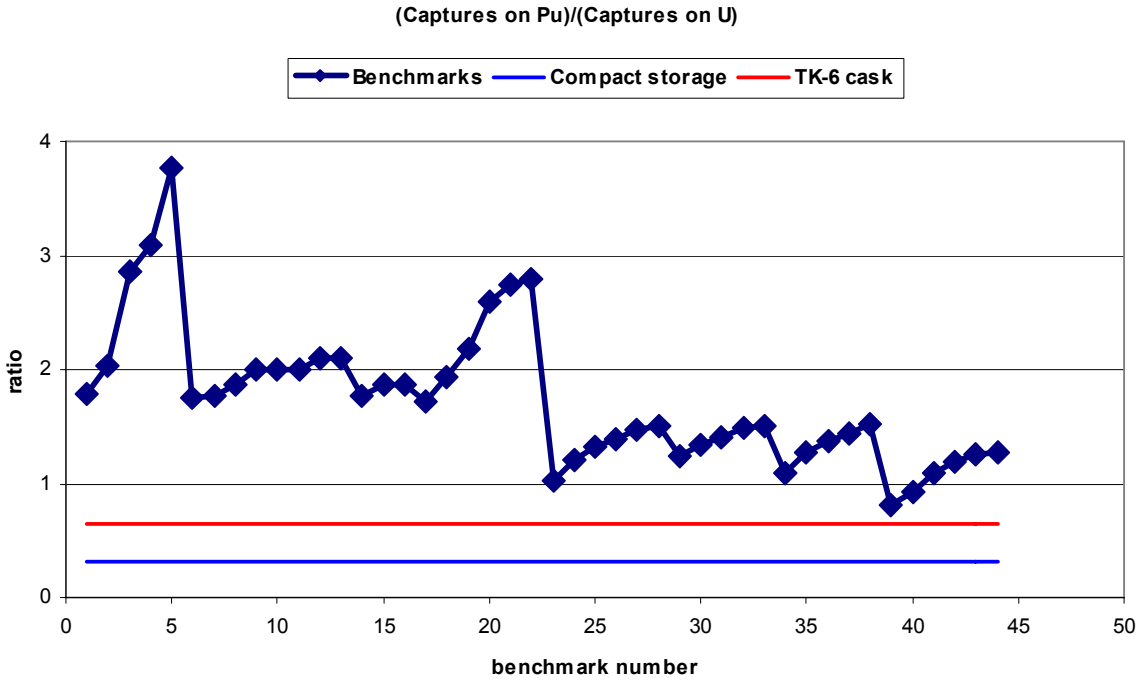

Fig. 8. Ratio of captures happening on plutonium to the fission happening on uranium homogeneous MOX benchmarks.

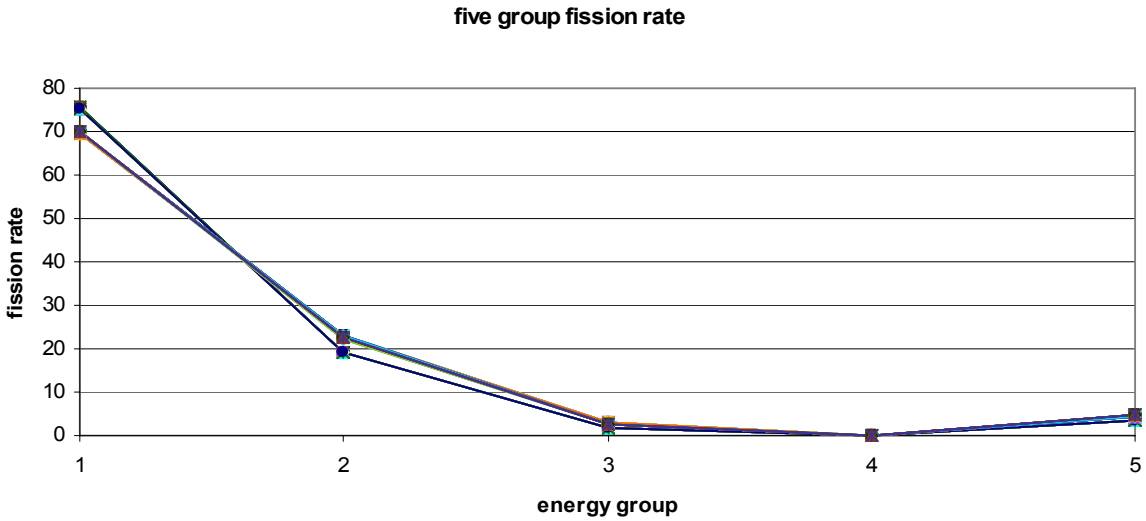

Fig. 9. Five group fission rates calculated for the MOX benchmark experiments with central absorber rod. In fact 88 curves are shown on this figure.

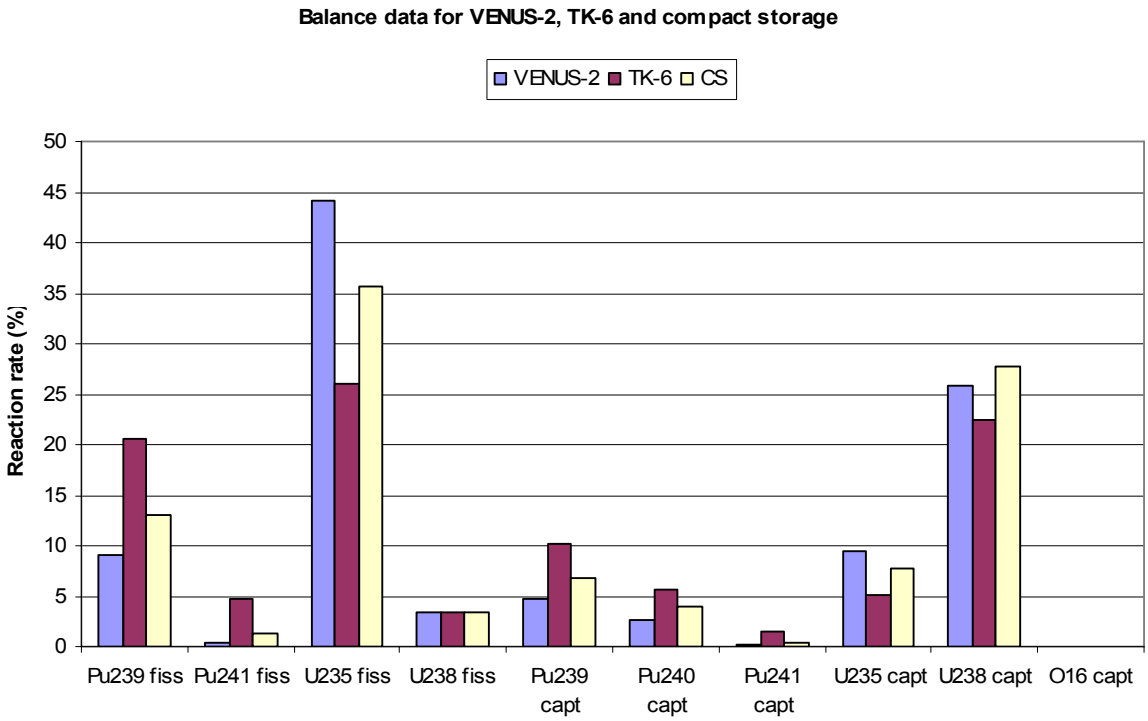

Fig.10. Comparison of reaction rate balance of compact storage pool and TK-6 cask with burnup credit and of VENUS-2 benchmark.

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Calculations of criticality and nuclide compositions for VVER-440 fuel by new vesion of the scale 5 code

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Abstract. In this article are compared theoretical results by new version of the SCALE 5 code with experiments or other theoretical calculation for:

(1) criticality	-measurement on ZR-6 and LR-0
	-numerical benchmark No.1, 3 and 4 (CB1, CB3, CB4)
(2) nuclide compositions	-measurement in Kurchatov institute for 3.6%
	-measurement in JAERI (PWR 17x17)
	-numerical benchmark No.2 (CB2)

The focus is on modules KENO VI, TRITON and ORIGEN-S.

1. Introduction

In russian reactor VVER-440 are used hexagonal assemblies with triangular lattice pitch. In western PWR are used assemblies with square geometry. The change from square geometry to triangular is sometimes big problem. It is necessary to verify code used by calculations of spent fuel (VVER-440) storage. The best way is to compare theoretical results with experiments. If experiments don't exist with other calculations. In this paper are results of the SCALE 5 code (distributed in 2004) [1] for calculations of criticality and nuclide compositions. The computing system SCALE 5 includes several modules: KENO V.a, KENO VI, SAS2, SAS4, ORIGEN-S, TRITON etc. The library 44GROUPNDF5 was used.

2. Criticality

For verification of criticality calculations exist a lot of critical experiments and several sets of numerical benchmarks.

The criticality experiments were selected from the International Handbook of Evaluated Criticality Safety Benchmark Experiments (September 2004 Edition) [2]. The basis of the selection was that the material and geometrical characteristics of the investigated systems should be similar to the characteristics of the VVER-440 system.

The total number of experiments is 297. 271 experiments were done in KFKI Budapest on reactor ZR-6 (LEU-COMP-THERM 015 and 36), 9 in VALDUC (LEU-COMP-THERM 052 and 007), one in PNL (LEU-COMP-THERM 005) and 12 in NRI Rez [3]. All selected experiments are with regular hexagonal lattice of low enriched uranium (UO2) rods in water (some times in borated water). The triangular pitch is between 1.1 and 2.598 cm, the most (202 experiments) is 1.27 cm. The enrichment of U235 is between 2.0 and 4.742%, the most (254 experiments) is 3.6%. The boron concentration is between 0 and 7.2 g_{H3BO3}/g_{H2O}, the most (251 experiments) is only water.

All calculations were done with the KENO VI module (sequence CSAS26) and library 44GROUPNDF5. The total number of neutrons was 500 000, 2 000 per generation, 10 skipped generation.

In Fig.1.1 is histogram of deviation between calculated and measured k_{ef} shown. The square root deviation (σ) is 0.45%, interval of deviations is between -1% and 2% (only three cases have deviation more then 1%). Upper tolerance limit $2\sigma = 0.0090$.

The numerical benchmarks CB1, CB3 and CB4 [4,5,6] are second set for comparison (total 37 cases). In Fig.1.2 is histogram of deviation between calculated and evaluated k_{ef} shown. The square root deviation (σ) is 0.31%. The agreement is very good, interval of deviations is between -0.3% and 0.7% (only two cases have deviation more then 0.5%).

In Fig.1.3 is deviation (minimal, average and maximal) for both (experimental and numerical) cases shown.

3. Nuclide compositions

In ORIGEN-S calculations were used libraries prepared by the module SAS2. Basic library was used library 44GROUPNDF5 and 1 library per cycle. Time step by irradiation was maximal 100 days. The new 2D modules (NEWT and TRITON) for inventory calculations were tested. By TRITON calculation was used 1 step per cycle.

The nuclide composition measurement (only some actinides) of spent fuel assembly VVER-440 was made in Kurchatov Institute in Moscow [7–10]. 12 samples (11 actinides) are from one assembly 3.6% from NPP Novovoronezh Unit 4 (In text Novovoronezh 1). New nuclide composition measurement for burnup credit application for VVER-440 reactor was made in RIAR in Dimitrovgrad (ISTC project) [11]. 8 samples (16 actinides and 32 fission products) are from one assembly 3.6% from NPP Novovoronezh Unit 4 (In text Novovoronezh 2). We used module ORIGEN-S and library prepared for fuel VVER-440 (by module SAS2) and the new 2D module TRITON (April 2005 release). The results are in Tab.2.1 and Tab.2.2 and in Fig.2.1 and Fig.2.2 (maximal, minimal and average deviation) shown. For U and Pu is agreement good - deviations are less then 15 % (except Pu238), for Am and Cm may be large differences - up to 60 %. The deviations according TRITON are smaller (except U235 and Am243) then deviation according ORIGEN-S. In the second experiment are results for actinides similar to the first experiment. Very high deviation is in one sample for U235 and in one sample for Am242m. The initial concentration of U234 is not known. For fission products are small deviation (less than 10%) for Nd143, Nd146, Nd148, Nd150, Cs133, Cs137, Ce140, Ce142, Sm150 and Sm154. Very high deviation (more than 50%) are for Sm147, Sm151, Eu151, Ag109 and Gd155. Extreme high deviation (more than 150%) are for Eu153, Eu154 and Eu155.

Because some nuclides have very high deviation, the comparison was done for one PWR assembly. The measured data are from SFCOMPO data bank operated by OECD/NEA [12]. It was used 5 samples (SF97-2 up to 6, 19 actinides and 20 fission products) from TAKAHAMA-3 (PWR, 17x17). In ORIGEN-S calculation was used ARP library "17x17" and operational history with small time step. Additional data are in [13]. The results are in Tab.2.3 and Fig.2.3 shown. For U and Pu is agreement good - deviations are less then 10 % (except Pu238), for Am and Cm may be large differences - up to 50 %. Generally results from TRITON are not better then from ORIGEN-S. For Cm246 and Cm247 (very low concentration) are very high difference between ORIGEN-S and TRITON. For all isotopes of Nd are deviation less than 5%, for other fission products are deviation up to 20%, except Sb125 (up to 80%), and Sm151 and Sm152 (up to 35%).

Calculational Burnup Credit Benchmark No.2 was defined and evaluated in [14, 5] (only 11 actinides and 15 fission products important for burnup credit calculations). The library was prepared by the module SAS2, nuclide composition was calculated by the module ORIGEN-S. The results are in Tab.2.4 and in Fig.2.4 shown. Deviations are less then 10 % (except Am243, Sm151 and Gd155). Big differences between concentration according SAS2 and TRITON are for U235 and isotopes of Pu.

For nuclides compared in all cases (except fission products, not exist in measurement Voronezh 1, 10 actinides and 7 fission products) the maximal deviation is usually in measurement Voronezh 2 and minimal deviation is usually for numerical benchmark CB2. Usually deviation for measurement Takahama is lower then for Voronezh 1 and 2.

Pins in assembly VVER-440 have different ratio uranium/water (by central tube, at corner, by side, surrounding by fuel pins) and therefore the concentration depends on pin position. In [15] is detailed analyse of nuclide concentration in pins. The maximal deviation is usually in corner's pin and for the most of nuclides is less 3%, maximal is 6%.

4. Conclusion

The criticality calculation of fresh assembly with triangular lattice has very high accuracy - deviation of k_{ef} is less 1%.

The nuclide composition calculation of spent fuel assembly has different deviation for different nuclides. 17 nuclides were compared in 3 (2) measurements and one numerical benchmark. Very good agreement (less 5%) is for U236, U238, Nd143, Nd145. Good agreement (less 15%) is for U235, Pu and some Sm isotopes. For Am243, Cm244 and some Sm isotopes is deviation higher (more than 25%). Some nuclides (Eu isotopes) were measured only in Voronezh 2 experiment and have extreme high deviation (more than 150%). The high deviation is usually for nuclides with very low concentration.

The system SCALE 5 is very good tool for calculation of spent fuel VVER-440. The accuracy of criticality is very high. By nuclide composition the accuracy depends on nuclide, it is necessary to continue in investigation.

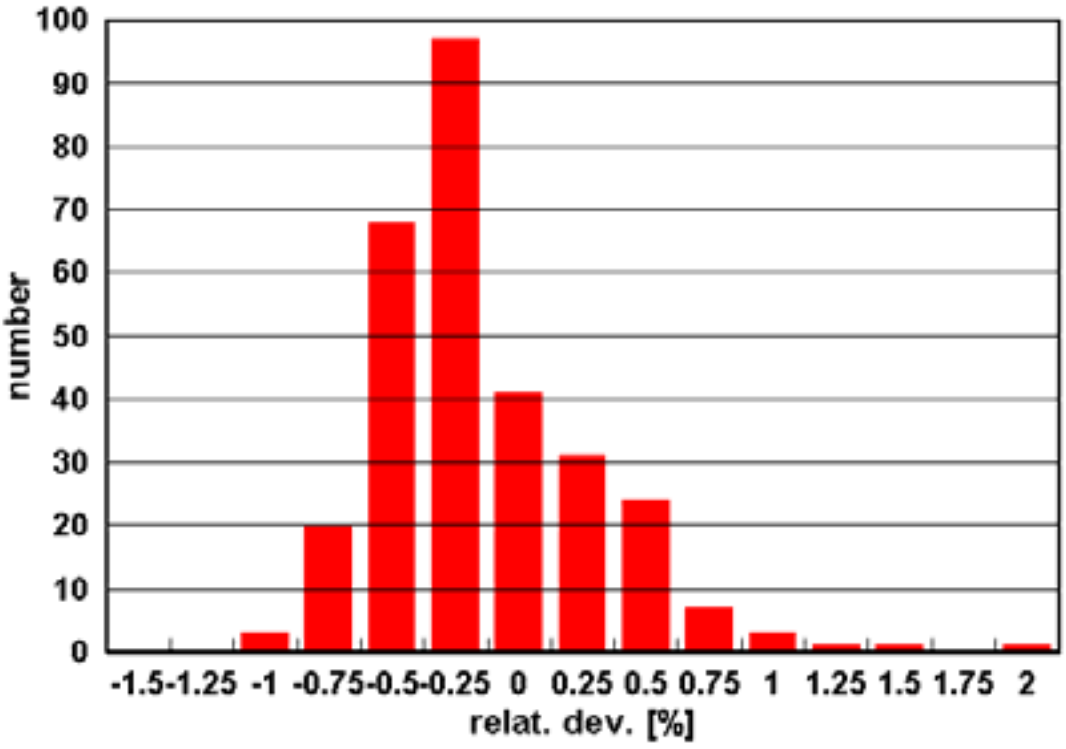

Fig.1.1 Histogram of Kef deviation for KENO VI, 297 experiments. $2\sigma = 0.0090$.

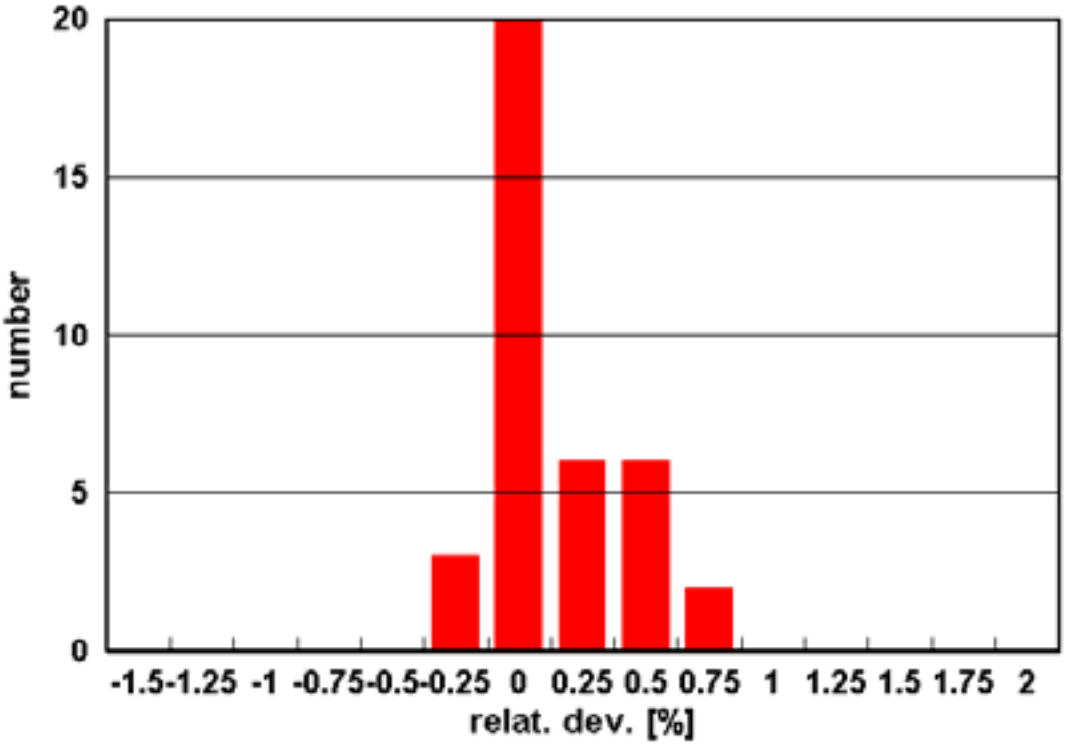

Fig.1.2 Histogram of Kef deviation for KENO VI, 37 cases CB1,3,4. $2\sigma = 0.0062$.

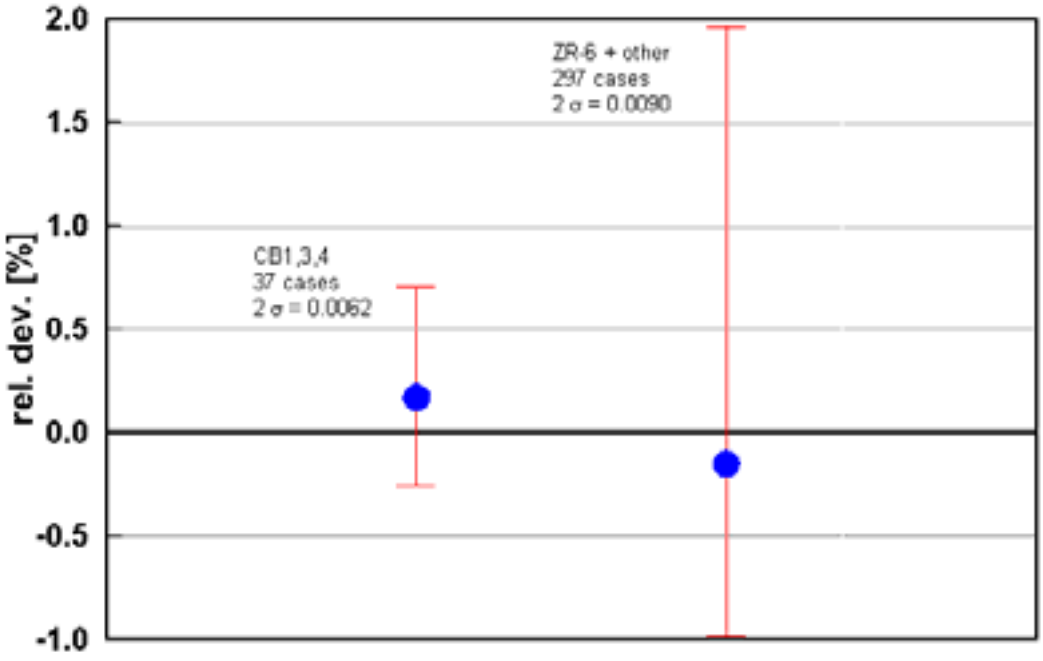

Fig.1.3 Kef deviation for KENO VI (minimal, average and maximal).

TABLE.2.1. DEVIATION (%) OF NUCLIDE CONCENTRATION, EXPERIMENT NOVOVORONEZH 1 (12 SAMPLES)

	ORIGEN-S				TRITON			
nuclide	min	average	max	σ	min	average	max	σ
U235	2.75	7.33	12.69	7.96	1.24	8.00	16.63	9.24
U236	-7.94	-6.17	-4.00	6.25	-8.05	-6.03	-3.97	6.15
U238	-0.11	0.05	0.35	0.13	-0.11	0.01	0.22	0.08
Pu238	-28.01	-18.63	-11.19	19.30	-23.03	-16.11	-8.97	16.75
Pu239	-3.72	0.50	8.85	3.59	-7.10	0.57	6.07	3.87
Pu240	-10.21	-3.01	9.02	7.42	-7.58	-4.08	0.78	4.83
Pu241	-12.95	-4.16	3.52	6.03	-6.52	0.51	7.04	3.14
Pu242	-13.73	-7.03	0.60	8.29	-10.79	-3.03	2.42	5.14
Am243	-15.40	16.49	38.28	23.42	-10.29	26.08	49.26	31.17
Cm242	-52.17	-35.84	-9.80	38.14	-45.96	-27.78	-5.87	29.90
Cm244	-32.70	-17.55	13.32	22.38	-23.85	-8.07	11.59	12.99

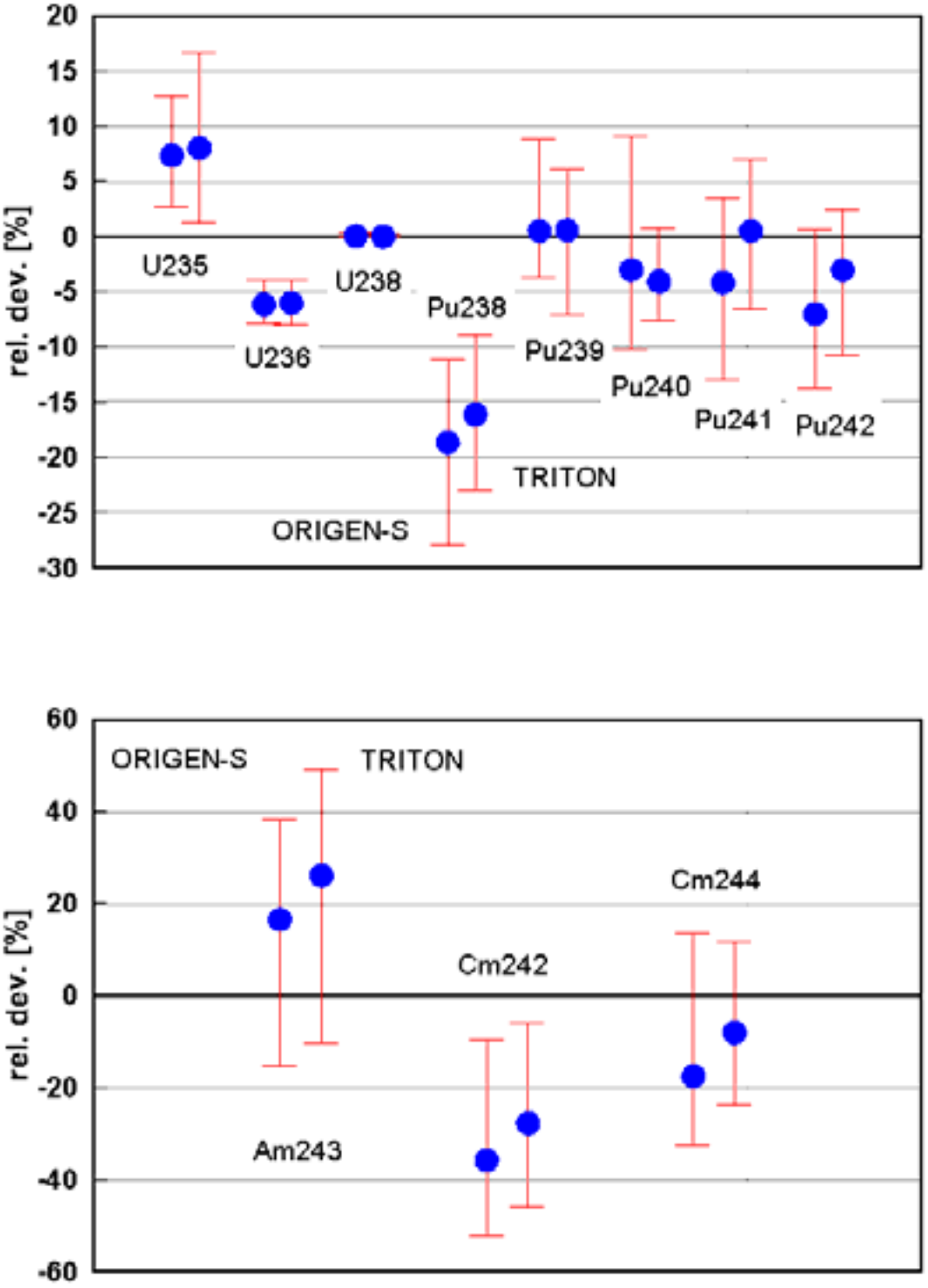

Fig.2.1. Deviation of nuclide concentration, experiment Novovoronezh 1 (12 samples).

TABLE.2.2. DEVIATION (%) OF NUCLIDE CONCENTRATION, EXPERIMENT NOVOVORONEZH 2 (8 SAMPLES)

	ORIGEN-S				TRITON			
nuclide	min	average	max	σ	min	average	max	σ
u234	-99.28	-98.44	-96.72	98.44	-99.3	-98.34	-96.7	98.34
u235	-0.39	13.79	60.61	23.15	-4.04	13.05	46.92	19.68
u236	-13.1	-9.81	-7.11	9.99	-4.25	-2.25	1.56	3.05
u238	-0.13	0.06	0.25	0.12	-0.1	-0.01	0.14	0.08
pu238	-24.05	-4.88	20.59	13.8	-10.43	7.94	30.88	14.31
pu239	-3.73	4.28	15.18	7.14	-3.37	5.1	11.76	6.54
pu240	-12.93	-3.93	10.67	8.7	-10.35	-5.27	2.5	6.55
pu241	-15.23	-2.27	15.01	9.36	-7.79	0.13	7.15	4.7
pu242	-18.5	-10.65	1.56	12.25	-16.29	-8.58	5.54	10.7
np237	-64.28	-59.76	-52.74	59.85	-58.36	-55.1	-48.71	55.16
am241	-58.63	-5.54	36.95	27.62	-61.28	-1.3	34.84	28.49
am242m	-31.34	38.43	200.6	76.03	-13.98	40.32	194.4	72.45
am243	-18.4	12.08	35.53	23	-13.38	16.07	40.46	23.03
cm244	-52.89	-30.76	-7.12	33.84	-51.21	-26.88	-5.19	30.57
cm245	-64.03	-36.02	0.37	41.85	-55.8	-32.29	-6.3	36.71
cm246	-81.62	-46.43	-11.72	51.95	-74.18	-42.42	-12.25	47.48
nd142	-14.28	-6.48	12.6	10.69	-14.37	-5.92	12.9	10.32
nd143	-3.61	-1.28	2.69	2.29	-3.34	-1.35	2.86	2.28
nd144	12.38	21.81	29.27	22.31	15.73	21.93	27.65	22.37
nd145	-12.46	-9.2	-3.42	9.64	-12.48	-9.43	-4.05	9.84
nd146	1.68	3.88	5.88	4.08	1.85	3.87	5.42	4.03
nd148	-6.93	-4.83	-2.22	5.06	-6.99	-4.95	-2.26	5.15
nd150	-5	-2.5	0.43	3.08	-4.83	-2.6	-0.7	3
cs133	-4.92	-2.9	1.43	3.67	-5.42	-3.46	0.6	4.11
cs134	-58.17	-9.39	43.07	38.34	-55.99	-9.7	40.16	37.34
cs135	-6.89	-0.58	8.83	4.33	-5.14	-0.35	4.37	2.97
cs137	-7.14	-4.45	-1.54	4.83	-7.27	-4.56	-1.68	4.93
ce140	-4.22	-1.68	1.79	2.65	-2.43	-0.27	3.19	2.05
ce142	-5.8	-2.45	0.93	3.41	-5.43	-2.54	0.93	3.44
ce144	-50.17	-47.45	-44.77	47.48	-51.46	-48.03	-45	48.07
sm147	-69.85	-66.84	-65.62	66.86	-70.52	-66.88	-64.9	66.9
sm148	-22.39	-15.07	-10.38	15.62	-19.78	-15.05	-9.59	15.49
sm149	-21.49	-12.04	-4.76	12.82	-23.71	-15.4	-3.77	16.7
sm150	-11.06	2.84	7.71	6.39	-10.96	2.69	7.89	6.29
sm151	23.71	53.51	79.23	55.89	31.08	54	79.62	56.27
sm152	4.18	22.77	29.69	24.03	2.11	21.47	30.09	22.96
sm154	-12.92	-0.6	3.54	5.15	-11.24	-0.53	4.78	4.59
eu151	-92.52	-90.52	-84.75	90.55	-93.36	-90.48	-86.54	90.5
eu153	221.16	524.94	1008.55	573.91	226.73	523.73	1009.21	572.14
eu154	278.2	623.02	1147.13	672.43	303.46	626.64	1092.36	670.52
eu155	117.28	413.81	878.71	472.03	116.4	413.61	879.88	472.57
mo95	-12.64	-11.05	-7.4	11.19	-12.68	-10.88	-6.63	11.04
tc99	-2.7	-0.91	1.7	1.74	-3.14	-1.17	1.41	1.89
ru101	-3.51	-1.94	1.15	2.49	-3.83	-2.08	0.88	2.6
pd105	-11.72	-6.71	-1.93	7.29	-9.73	-6.64	-2.86	7.1
pd108	8.2	21.19	31.93	22.27	12.31	21.66	30.9	22.37
ag109	32.94	71.7	115.29	75.27	36.92	71.95	115.29	75.21
gd155	-94.55	-93.72	-91.5	93.72	-94.42	-93.52	-92.15	93.53

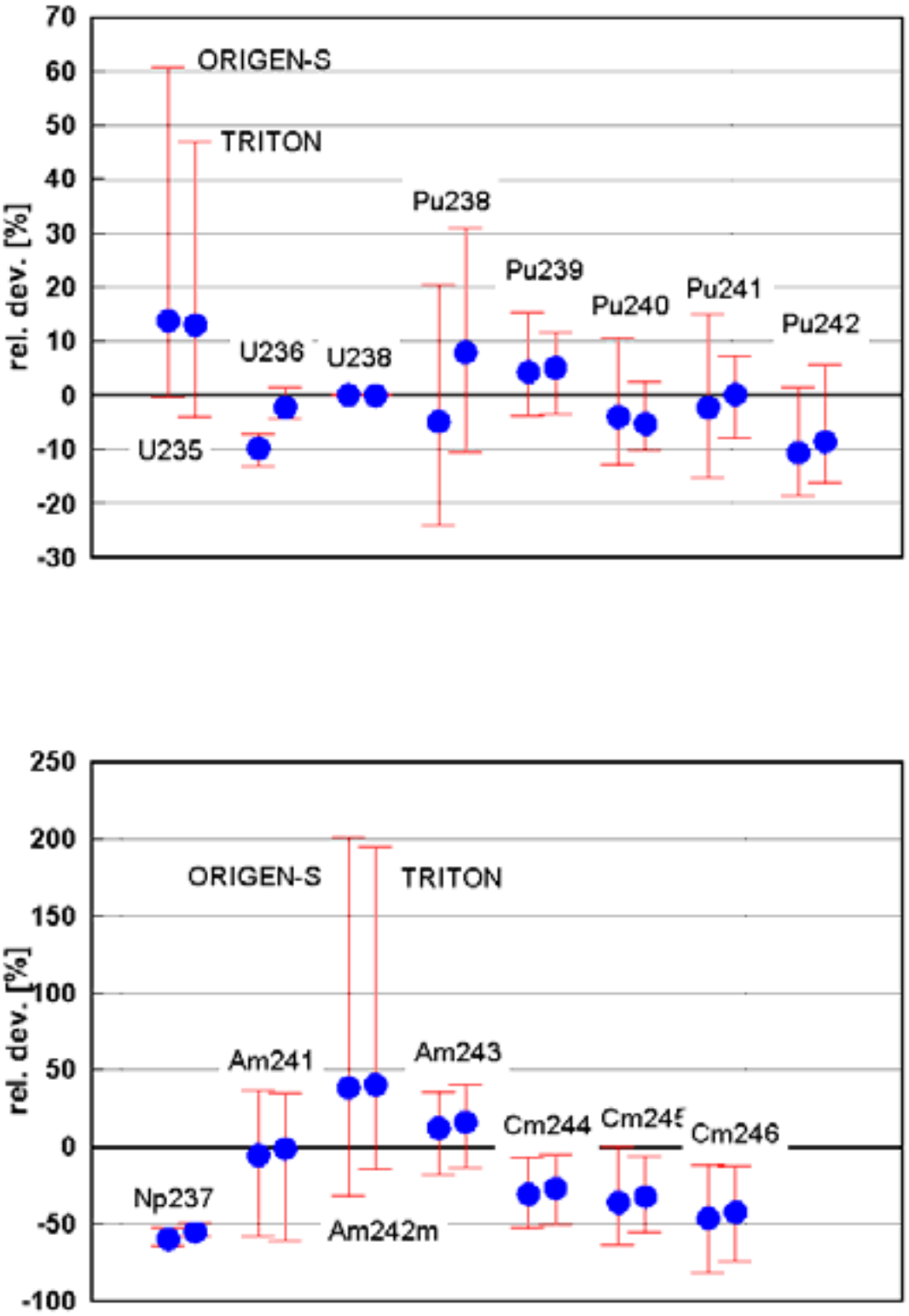

Fig.2.2a Deviation of nuclide concentration, experiment Novovoronezh 2 (8 samples).

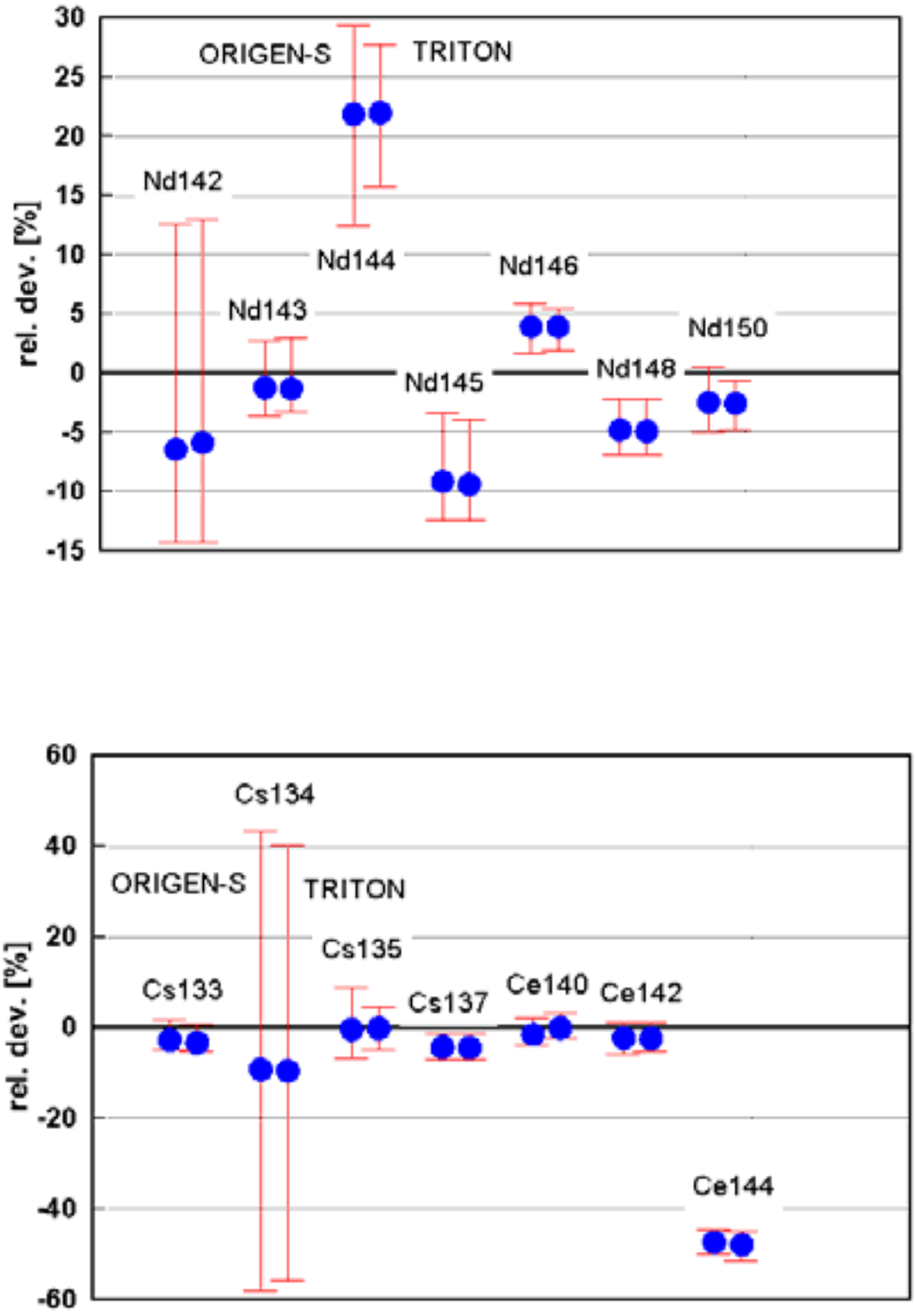

Fig.2.2b Deviation of nuclide concentration, experiment Novovoronezh 2 (8 samples).

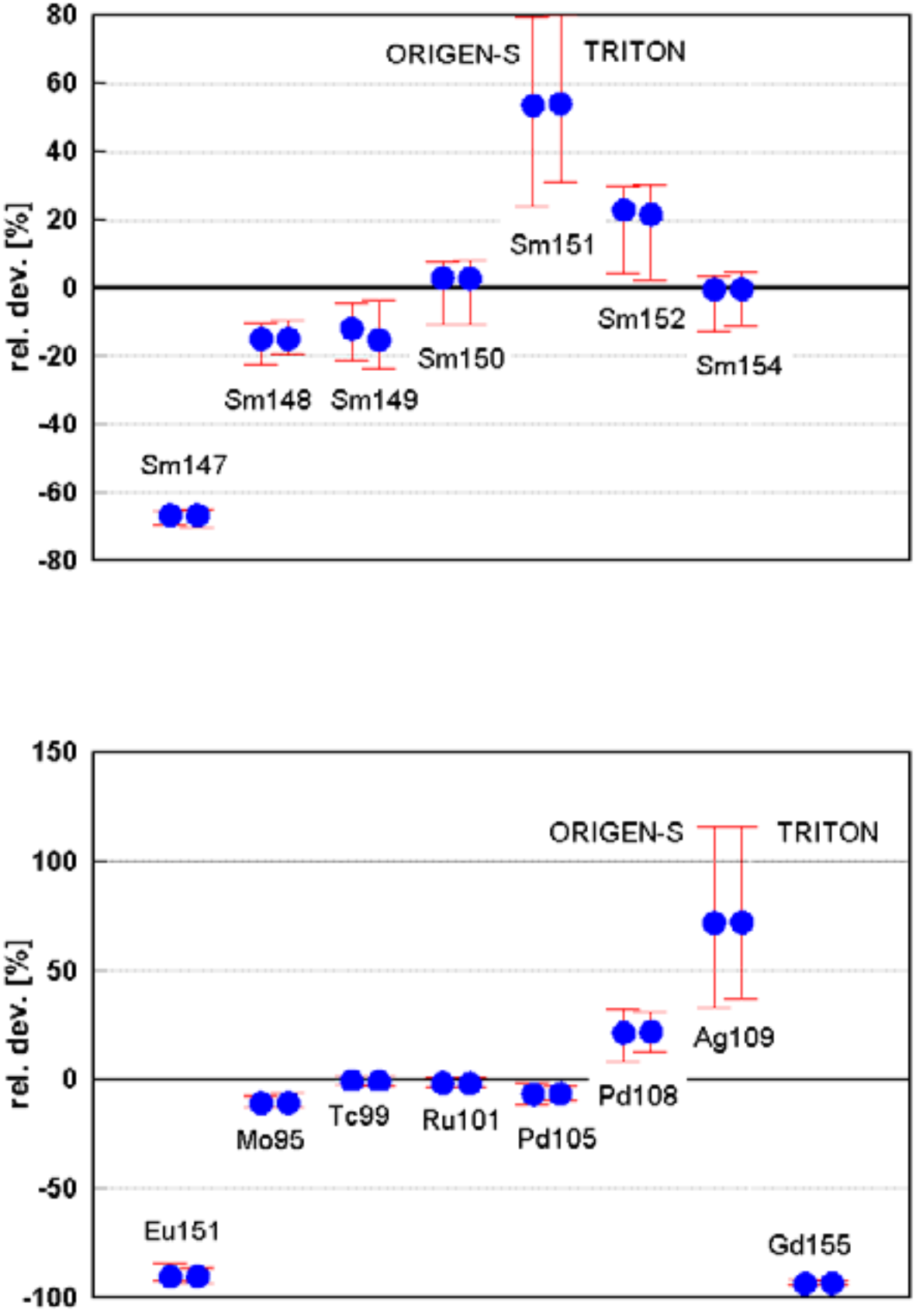

Fig.2.2c Deviation of nuclide concentration, experiment Novovoronezh 2 (8 samples).

TABLE.2.3 DEVIATION (%) OF NUCLIDE CONCENTRATION, EXPERIMENT TAKAHAMA-3 (5 SAMPLES)

	ORIGEN-S				TRITON			
nuclide	min	average	max	σ	min	average	max	σ
u234	5.47	7.00	10.26	7.22	7.48	8.83	11.75	8.97
u235	-5.09	-1.91	1.77	3.00	-6.00	-2.91	0.79	3.68
u236	-0.49	-0.15	0.13	0.27	-0.34	-0.04	0.17	0.18
u238	-0.11	0.00	0.08	0.06	0.31	0.42	0.50	0.43
np237	-2.01	0.71	2.14	1.61	-4.39	-2.04	-0.54	2.42
pu238	-11.93	-9.16	-2.80	9.75	-15.61	-12.89	-6.62	13.29
pu239	-2.67	0.75	6.39	3.31	-6.10	-2.74	2.73	4.08
pu240	3.10	4.70	7.27	4.92	-2.47	-1.04	1.59	1.79
pu241	-7.37	-3.97	1.87	5.24	-9.06	-5.79	0.20	6.71
pu242	-1.79	-0.94	0.54	1.25	-1.43	-0.33	1.39	1.00
am241	8.77	16.91	31.67	18.86	13.03	21.55	37.12	23.27
am242m	2.84	10.75	26.08	13.67	1.78	9.71	25.15	12.89
am243	7.81	11.84	19.40	12.49	3.82	10.13	17.69	11.10
cm242	-8.44	3.83	15.71	8.98	-11.74	1.13	13.08	8.41
cm243	-25.24	-19.15	-12.64	19.65	-30.11	-22.88	-16.52	23.37
cm244	-11.63	-5.28	5.69	7.99	-18.14	-9.98	0.59	11.79
cm245	-45.73	-40.03	-27.38	40.57	-45.79	-37.95	-24.93	38.65
cm246	-19.74	-16.37	-5.16	17.31	-46.28	-41.33	-33.45	41.55
cm247	13.73	22.58	39.76	24.44	-52.04	-44.17	-36.57	44.51
nd142	-4.04	1.08	3.15	2.84	-4.86	0.64	2.92	2.90
nd143	-1.59	-0.60	0.23	0.84	-1.59	-0.60	0.33	0.88
nd144	-4.57	-1.72	2.52	3.03	-2.25	0.60	4.76	2.54
nd145	0.23	0.88	1.60	1.00	1.20	1.80	2.48	1.85
nd146	0.49	0.65	0.98	0.68	0.89	1.15	1.55	1.18
nd148	-0.27	0.03	0.59	0.31	0.21	0.52	1.06	0.62
nd150	0.49	0.87	1.64	0.97	0.94	1.35	2.21	1.43
cs134	-20.35	-15.29	-12.50	15.59	-23.36	-18.16	-15.35	18.43
cs137	-1.01	-0.63	-0.11	0.72	-2.47	-2.12	-1.66	2.14
eu154	-5.27	2.10	7.97	5.16	-10.29	-3.23	2.45	5.55
ce144	-9.44	-0.35	7.28	6.49	-12.32	-3.70	3.55	7.17
ru106	-14.34	-3.81	8.11	8.21	-17.10	-6.99	4.60	9.92
sb125	14.93	54.05	83.57	59.94	19.27	58.59	89.22	64.29
sm147	-12.51	-10.44	-6.88	10.63	-2.18	-0.10	3.37	2.01
sm148	-17.18	-14.48	-11.22	14.71	-13.07	-10.22	-6.87	10.58
sm149	-4.88	8.89	20.51	12.83	-17.43	-6.27	3.47	9.92
sm150	7.24	8.31	9.68	8.35	4.57	5.13	6.63	5.19
sm151	23.32	33.68	45.44	34.58	15.17	23.83	34.85	24.84
sm152	24.66	28.76	31.82	28.91	24.87	28.70	31.59	28.83
sm154	0.20	1.75	3.66	2.08	-2.08	-1.00	0.97	1.45

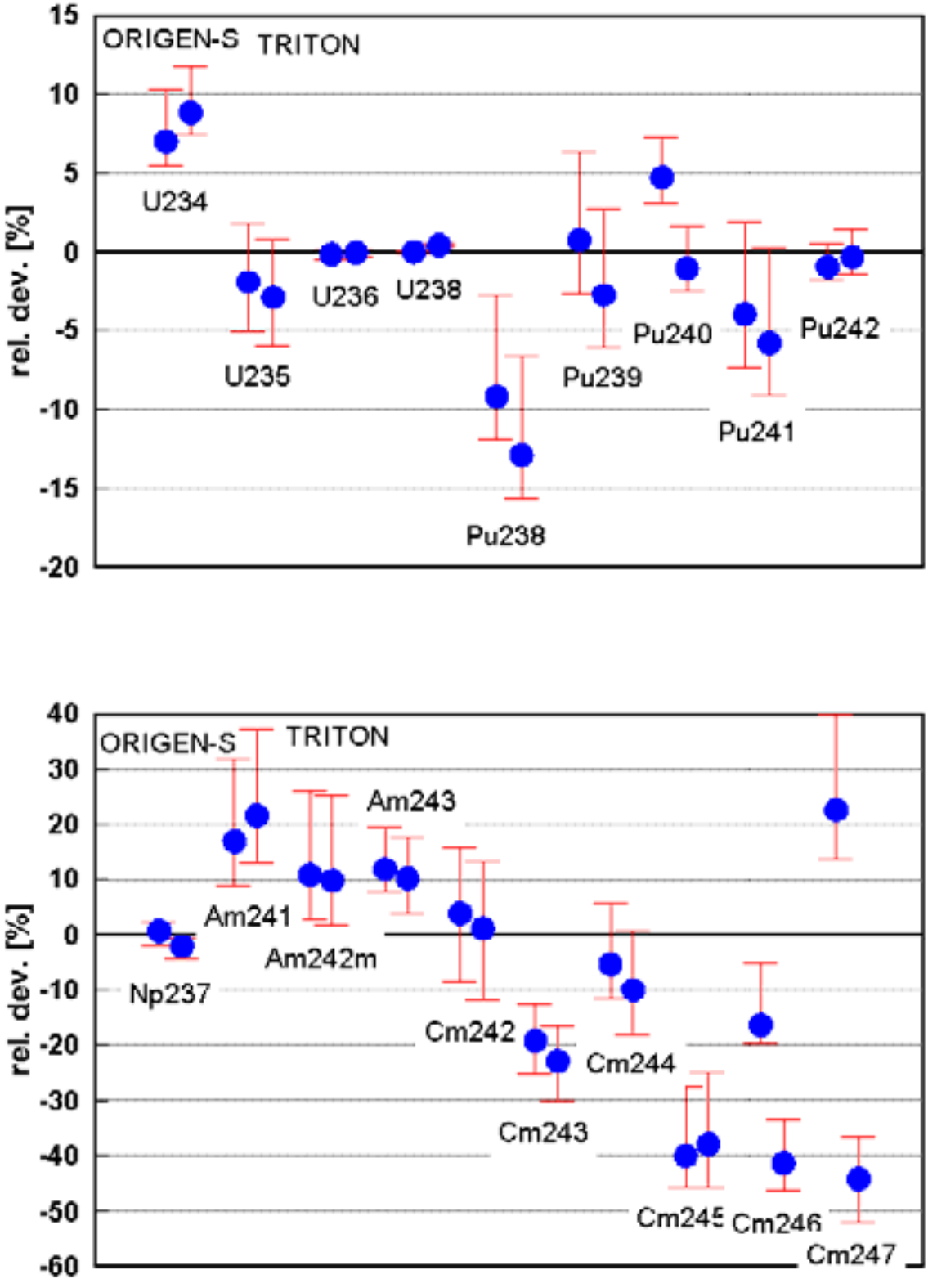

Fig.2.3a Deviation of nuclide concentration, experiment Takahama (5 samples).

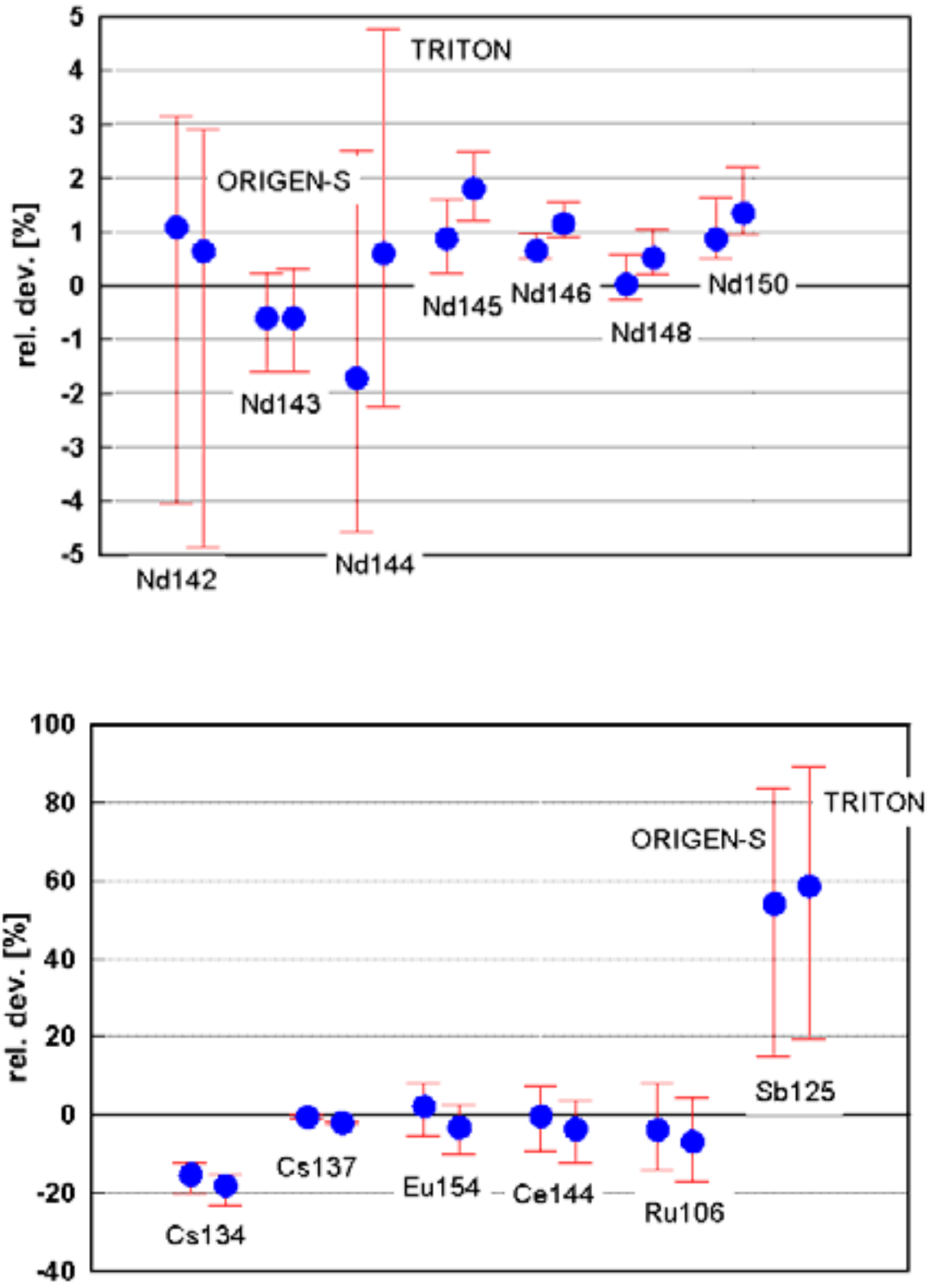

Fig.2.3b Deviation of nuclide concentration, experiment Takahama (5 samples).

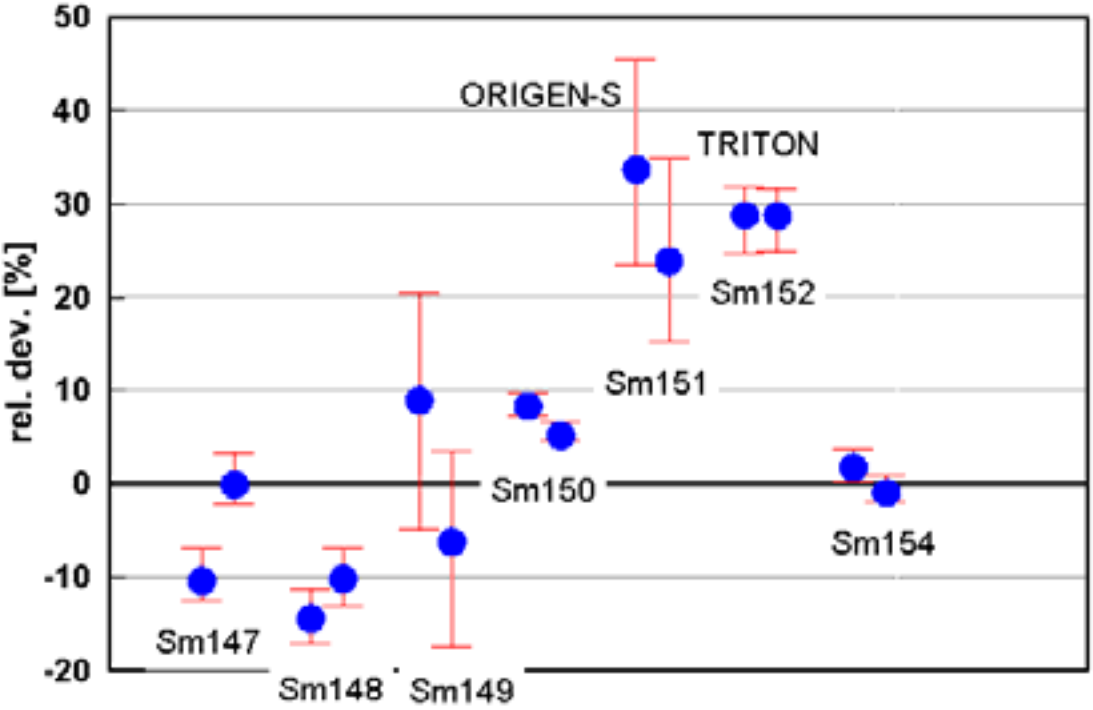

Fig.2.3c Deviation of nuclide concentration, experiment Takahama (5 samples).

TAB.2.4 DEVIATION (%) OF NUCLIDE CONCENTRATION, NUMERICAL BENCHMARK CB2 (6 SAMPLES)

	SAS2				TRITON			
nuclide	min	average	max	sigma	min	average	max	sigma
u235	-1.29	-0.79	-0.51	0.83	-5.82	-4.53	-3.46	4.64
u236	0.69	1.03	1.26	1.05	1.36	1.56	1.94	1.57
u238	-0.13	0.10	0.33	0.25	-0.14	0.09	0.33	0.25
np237	0.26	0.63	0.98	0.68	-0.89	-0.50	-0.08	0.57
pu238	-0.87	-0.02	1.36	0.86	-3.17	-2.40	-1.16	2.50
pu239	-1.64	-0.49	0.50	0.97	-5.23	-3.90	-2.48	4.04
pu240	-1.09	0.21	1.67	1.12	-1.31	0.10	1.88	1.29
pu241	1.50	1.98	2.78	2.03	-0.66	0.34	1.63	0.96
pu242	3.74	4.19	5.26	4.22	7.31	7.75	8.83	7.77
am241	-0.37	1.63	2.78	2.00	-2.48	0.02	1.90	1.43
am243	10.72	12.04	13.97	12.11	13.22	14.45	16.29	14.50
mo 95	-1.75	-0.60	0.09	0.90	-0.17	0.69	1.40	0.89
tc 99	-0.24	-0.03	0.08	0.12	0.83	1.03	1.18	1.04
ru101	0.90	1.10	1.20	1.11	2.09	2.22	2.39	2.23
rh103	2.04	3.74	4.77	3.85	3.14	4.52	5.33	4.59
ag109	-5.28	-3.44	-2.09	3.60	-4.91	-3.59	-2.76	3.67
cs133	1.44	1.92	2.38	1.95	2.33	2.63	2.95	2.64
nd143	-0.17	0.01	0.33	0.17	-0.47	0.08	0.73	0.46
nd145	0.27	0.46	0.96	0.52	1.29	1.58	1.99	1.59
sm147	-0.37	0.36	0.90	0.61	0.69	1.69	2.60	1.85
sm149	1.15	2.78	4.11	2.95	-9.02	-6.28	-4.11	6.46
sm150	1.64	2.58	3.39	2.67	1.49	2.38	3.15	2.48
sm151	8.50	10.31	11.70	10.39	3.38	5.31	7.16	5.48
sm152	3.55	4.83	5.91	4.94	5.00	6.24	7.27	6.32
eu153	-3.12	-2.37	-0.89	2.50	-2.30	-1.49	-0.08	1.69
gd155	-42.93	-32.11	-6.46	34.59	-42.60	-32.81	-6.22	35.16

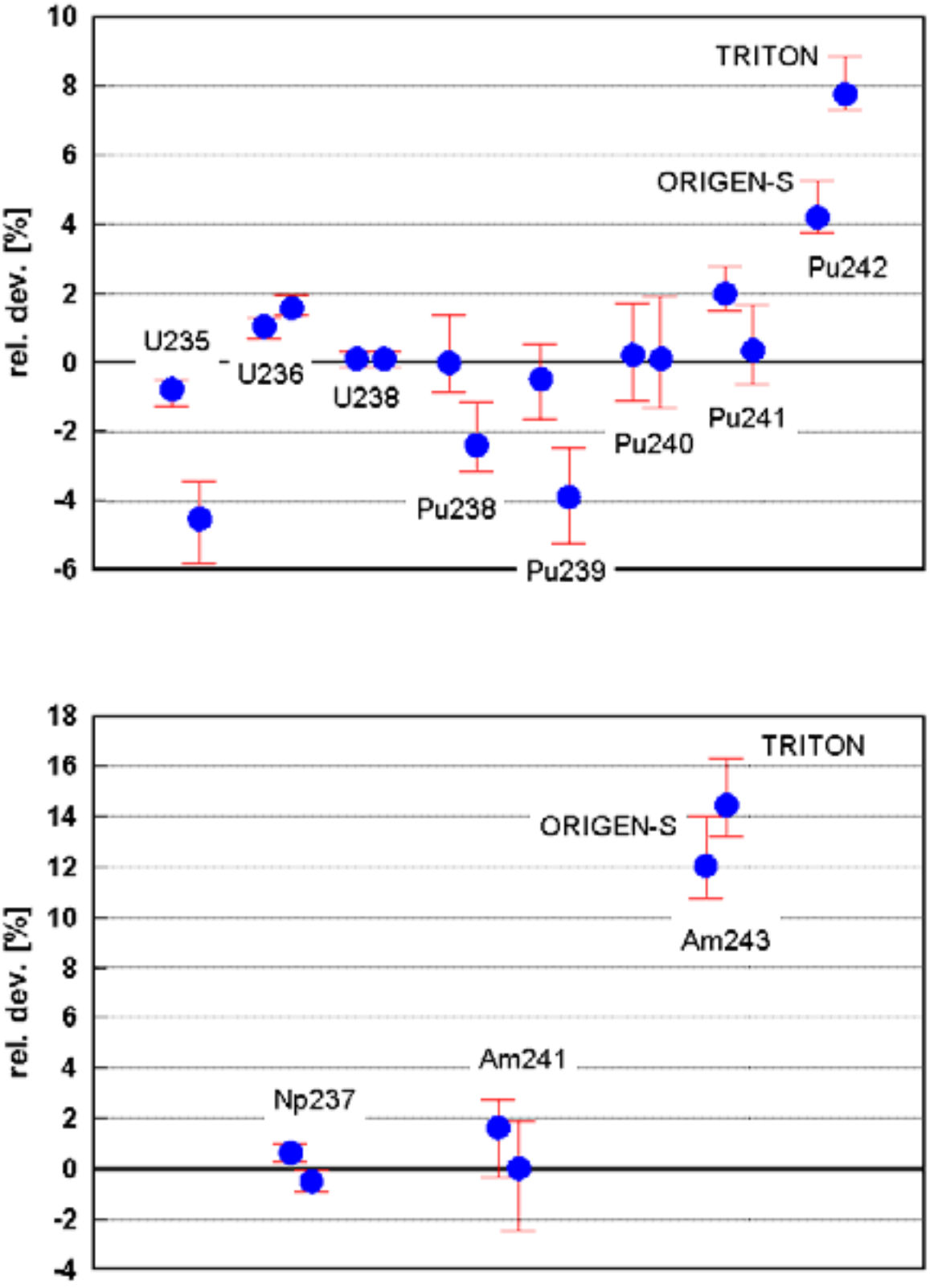

Fig.2.4a Deviation of nuclide concentration, numerical benchmark CB2 (6 samples).

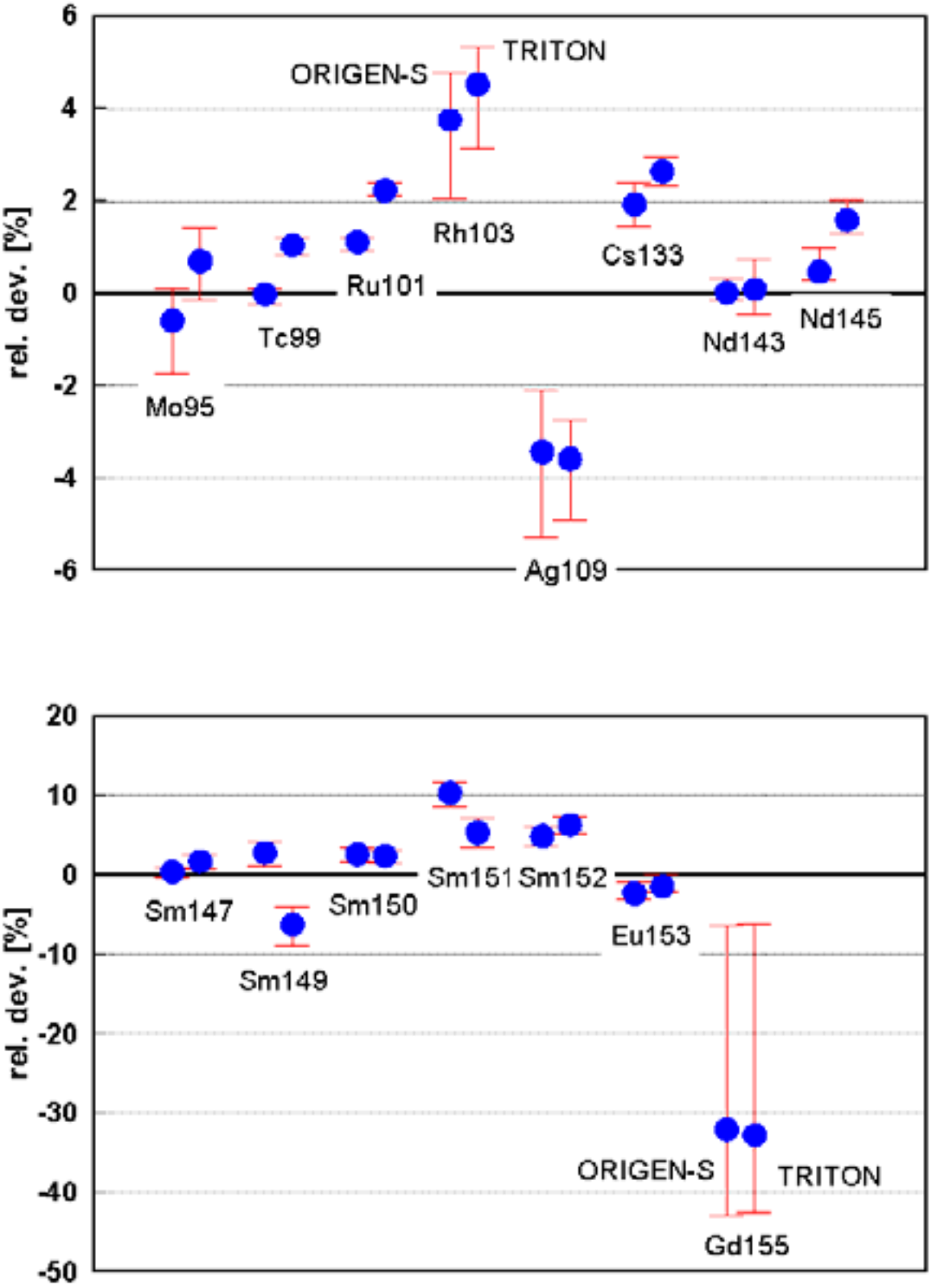

Fig.2.4b Deviation of nuclide concentration, numerical benchmark CB2 (6 samples).

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The study of burnup credit technology for spent fuel storage in China

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Abstract. This article briefly presents the status and trends of spent fuel management in China, the activities carried out and problem faced with. Research programmes on the application of burn-up credit technology to storage of spent fuel, conducted at China Institute of Atomic Energy (CIAE), is introduced in this paper. A 5-year project on criticality experiment is planned, which will begin in the next year. The criticality experiment to simulate the spent fuel storage will also be included in this project.

1. Introduction

Nuclear power has been developed with a great success in China. The first two nuclear power plants, i.e. Qinshan-I and Daya Bay NPP, have been operated for more than 10 years. Recently, six new units in China have put in operation. Other two units will be connected in grid by 2005 (see Table 1). The government of China is developing a plan up to 2020 to build more nuclear power plants in China. The total nuclear capacity will reach to about 40 GW by 2020.

Spent fuel from NPPs in China is in wet-storage on site. Since the capacity of storage facility in Daya Bay NPP is near fully occupied, transportation of spent fuel from Daya Bay to reprocessing plant is in process. The capacity of the storage pool in Qinshan NPP is going to be full soon. At present, traditional criticality analysis for spent fuel storage and transport is applied in China. This approach provides considerable criticality safety margin. In fact, the international practice of burn-up credit technology used in spent fuel storage and transport has demonstrated the benefits of enhanced capacities. As the initial fuel enrichment of commercial nuclear reactor increases, and as higher discharge burn-up is achieved, there are considerable benefits to use the burn-up credit technology. For example, the initial fuel enrichment of Daya Bay NPP has increased from 1.8% to 4.45%, and the burn-up has increased from 20,000 MWd/MTU to 45,000 MWd/MTU.

TABLE 1 THE STATUS OF NUCLEAR POWER PLANTS IN CHINA

NPP unit name	Capacity (MWe)	Year commission and operation
Qinshan-I	300 (Chinese design)	1991.12
Daya Bay	2×944 (Framatome)	1993, 1994
Qinshan-II	2×600 (Chinese design)	2002, 2004
Qinshan-III	2×728 (CANDU-6)	2002, 2003
Lingao	2×984 (Framatome)	2002, 2003
Tianwan	2×1000 (VVER)	2005
SUM	~8800	2005
Projected capacity	40000	by 2020

In China, burn-up credit concept has been investigated. the technology study in this area is just at its beginning. We have lack of experience in operable methods for concrete projects, establishment of

regulations and rules, and so on. Thus we are in urgent need of technology assistance and more international academic exchange.

2. Study on burn-up credit technology

Since 1990s, the Reactor Physics Division of China Institute of Atomic Energy (CIAE) has conducted a number of works on the burn-up credit technology. At that time, a preliminary criticality analysis was performed by taking credit for fuel burnup for three improved schemes of Daya Bay Nuclear Power Station spent-fuel storage pool. It is shown that use of burnup credit in the criticality analysis and design of spent-fuel storage pool could result in considerable benefits. The k_{eff} of the storage pool could be decreased about 20% if we only consider the affect of main actinide, and decreased about 30% if we consider the combinative affect of actinide and the major fission product poisons. [(1)] Such benefits interest our government and owner of power station plant on using BUC technology.

From the year of 2002, we are carrying out a technical co-operation project on burn-up credit technology with the IAEA. Six young engineers were trained on theoretical analysis and criticality experiments in Germany, France and Japan. Three senior scientists conducted scientific visits to Germany, Belgium and Czech Republic. With assistance of IAEA experts a national training course on "Implementation of Burn-up Credit in Spent Fuel Management Systems" was held in July 2002 at CIAE.

The calculation adopting burn-up credit technology consists of two steps. At first, the isotopic concentrations of spent fuel should be calculated. Secondly, the criticality safety analysis will be performed using the concentrations resulted from the first step. The main issues, which are entirely due to the application of burn-up credit, are as set below:

- Validation of depletion codes applied;
- Establishing appropriate isotope sets;
- Validation of the criticality codes for burn-up credit;
- Determination of enveloping irradiation histories in conjunction with the evaluation of the reactivity effects of axial and horizontal burn-up distributions;
- Determination of the minimum required burn-up (usually as a function of initial enrichment);
- Verification of the fuel assembly burn-up before loading it in the spent fuel management system of interest.

In our working group, the calculation of depletion and isotopic concentrations of fuel assemblies is performed with standardized computer code: ORIGEN-2 and the criticality safety analysis is performed with a Monte Carlo n-particle transport code MCFR, which was developed by a research group on Monte Carlo method at China Institute of Atomic Energy, and its function is similar to the well-known criticality calculation code MCNP. The point cross-section data is taken from ENDF/B-V cross-section library.

In order to verify that the available criticality safety analysis tools and depletion codes are suitable for application to burn-up credit technology, we have performed some calculations according to the OECD/NEA burn-up benchmark problems.

The OECD/NEA burn-up benchmark Phase I-A problem is an infinite array of a simple PWR spent fuel rod. The objective is to examine effects of seven major actinides and 15 major fission products for an infinite array of PWR rods. Isotopic composition specified at 3.6 wt% U-235 at 0, 30, 40 GWd/MTU and at one- and five-year cooling time. The Benchmark consists of 13 cases. We calculated the first nine cases as shown in Table 2, which consider all actinides.

TABLE 2 BENCHMARK REFERENCE CASE NUMBERS

Cooling Time	Considered F.P.s	Burnup（GWd/MTU）		
		Fresh	30	40
1 year	Selected	Case 1	Case 2	Case 3
	No F.P.s		Case 4	Case 5
5 years	Selected	(Case 1)	Case 6	Case 7
	No F.P.s		Case 8	Case 9

We used MCFR to calculate the nine problems to get multiplication factors (k). Trends in the multiplication factors with burn-up and cooling time were as expected: k decreases as both burn-up and cooling time increase. The MCFR code we used, has cross section data for all the major fission products and actinides. The results from MCFR calculation, as shown in Table 3, have good agreement with results of the participants of the benchmark exercises, so we could proceed to next steps of our study with this code.

TABLE 3 RESULT COMPARISON OF BENCHMARK PHASE I-A

Case no.	17 Participants		MCNP-4 by JAERI	MCFR by CIAE
	Average	2*sigma		
1	1.4378	0.0175	1.4427	1.4419/0.0004
2	1.1402	0.0169	1.1382	1.1391/0.0006
3	1.0638	0.0170	1.0591	1.0621/0.0006
4	1.2456	0.0107	1.2426	1.2460/0.0005
5	1.1885	0.0110	1.1930	1.1883/0.0005
6	1.1123	0.0164	1.1042	1.1111/0.0006
7	1.0240	0.0156	1.0216	1.0236/0.0006
8	1.2284	0.0109	1.2231	1.2302/0.0005
9	1.1657	0.0099	1.1668	1.1657/0.0005

The purpose of the OECD/NEA burn-up benchmark Phase I-B was to compare computed nuclide concentrations for depletion in a simple pin-cell model. This benchmark consists of three cases, each with a different burn-up (27.35, 37.12 and 44.34 GWd/MTU). The specific power and boron concentrations for each cycle and cumulative burn-up were given in the problem description. Initial isotopic compositions for both the fuel and the moderator were given. The concentration for the 12 actinides and 15 fission products were requested to be calculated and compared to actual measurements.

The code we used is ORIGEN-2. The results given by the Table 4 is for the case with 27.35GWd/MTU. Only the concentrations of the actinides are shown here.

TABLE 4 RESULTS FOR THE CASE OF 27.35GWD/MTU ACTINIDES (MG/G UO_2)

Nuclide	Measurement	Average/Std.Dev	ORIGEN-2
^{234}U	1.600E-1	1.590E-1 /5.19%	1.646E-1
^{235}U	8.470E+0	8.190E+0 /2.98%	8.473E+0
^{236}U	3.140E+0	3.224E+0 /2.91%	3.189E+0
^{238}U	8.425E+2	8.375E+2 /0.12%	8.371E+2
^{238}Pu	1.012E-1	9.027E-2 /15.68%	9.926E-2
^{239}Pu	4.264E+0	4.230E+0 /5.16%	4.313E+0
^{240}Pu	1.719E+0	1.710E+0 /3.95%	1.753E+0
^{241}Pu	6.812E-1	6.697E-1 /6.45%	7.096E-1
^{242}Pu	2.886E-1	2.761E-1 /8.69%	2.765E-1
^{241}Am	Lack	2.426E-1 /4.22%	2.491E-1
^{243}Am	Lack	4.101E-2 /11.31%	4.347E-2
^{237}Np	2.680E-1	2.912E-1 /8.61%	3.157E-1

We have also performed some sensitivity analyses to determine the input parameters that can have a significant effect on the depletion and criticality analyses. For depletion analysis, these input parameters are specific power, dissolved boron, moderator temperature, fuel pellet temperature, and control rod insertion history. For the criticality analysis, these input parameters are moderator density, fuel temperature, axial burn-up distribution, and horizontal burn-up distribution. Each of these parameters was studied over a range of enrichments, burn-up, and cooling times. We would work to find the bounding values for the input parameters, which can make the safety analysis conservative.

3. The primary critical calculation for a proposed critical experiment facility

Using the fuel rods from Qinshan nuclear power plant, a critical experiment facility including spent fuel rods is proposed to build in our laboratory. The primary critical calculation for the core of this facility will be introduced in this paper.

The lattice pitch of fresh fuel rods is determined by the change of k_{eff} with the changing the lattice pitch. The relation between k_{eff} and lattice pitch is as Figure 1:

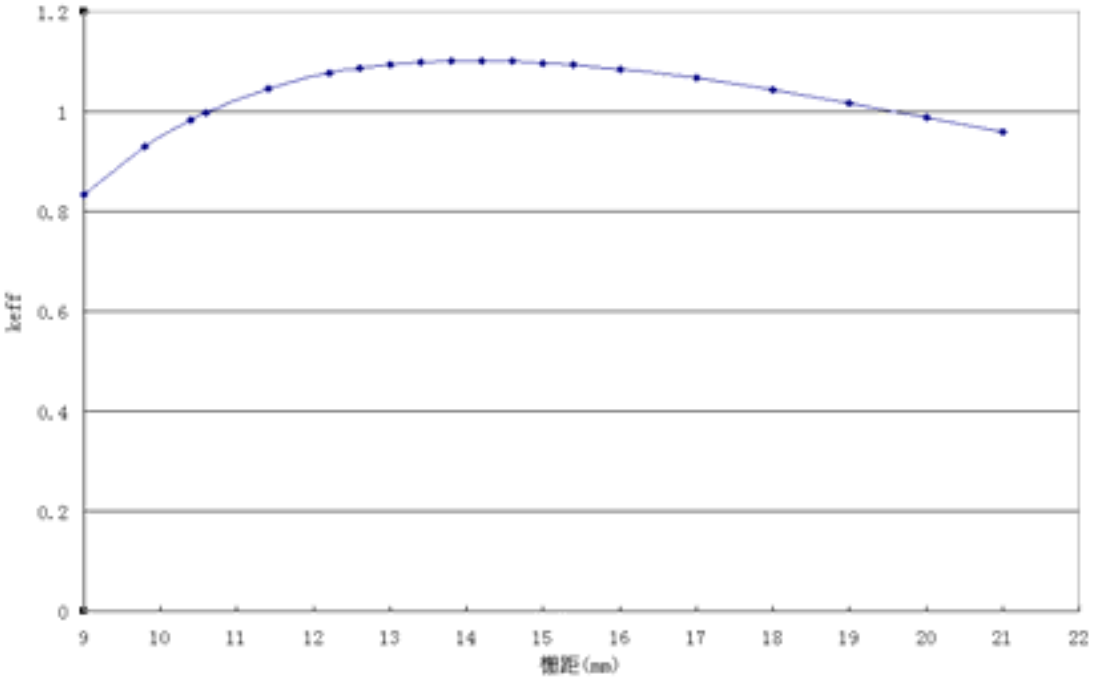

Figure 1. The k_{eff} values with changing lattice pitch.

Through a thorough analysis, the pattern of the core is basically confirmed as below (see Figure 2):

- The lattice pitch is 13.2mm;
- In the center of core, one spent fuel assembly(3×3)is loaded;
- Two control-rod assembly (each has one control rod in the center);
- Two safety-rod assembly (each has four safety rods symmetrically).

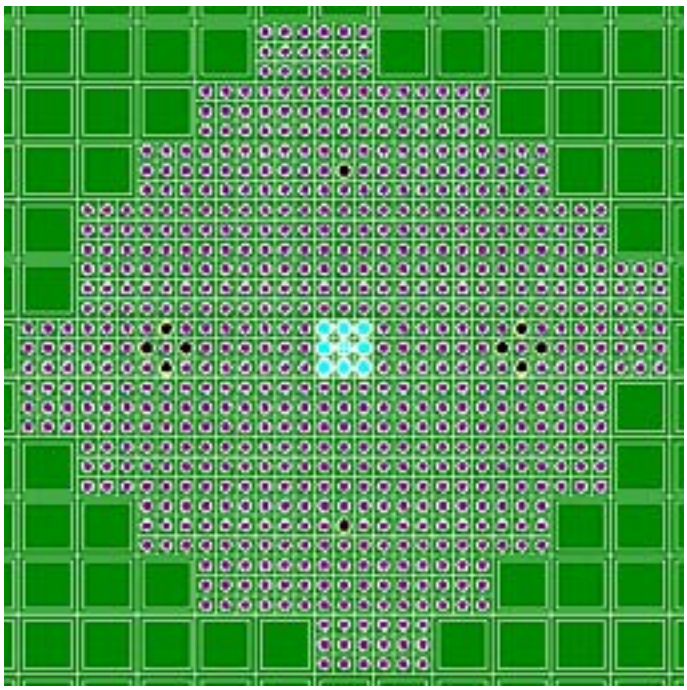

Figure 2. The conceptual design of the core.

Using ORIGEN-2 code, the isotopic inventory of the spent fuel rods is calculated. The calculated results are used by M-C code to calculate the k_{eff} value. Both the values leading a maximum k_{eff} value and a minimum k_{eff} value are used to get the two-side boundary k_{eff} value. The standard k_{eff} which the used isotopic inventory is calculated under the normal operating history is also calculated. Calculated results are shown in table 5.

TABLE 5 CRITICAL CALCULATION RESULTS

	One spent fuel assembly loaded in the center						No spent fuel assembly	
	minimum k_{eff}		standard k_{eff}		maximum k_{eff}			
	k_{eff}	σ	k_{eff}	□σ	k_{eff}	□σ	k_{eff}	□σ
No CR No SR	1.00156	0.00084	1.00266	0.00084	1.00437	0.00088	1.00752	0.00085
One CR No SR	0.99660	0.00078	0.99651	0.00083	0.99868	0.00087	1.00354	0.00074
Two CR No SR	0.99029	0.00083	0.99052	0.00075	0.99382	0.00085	0.99750	0.00083
One SR No CR	0.98809	0.00077	0.98839	0.00077	0.99126	0.00077	0.99642	0.00078
Two SR No CR	0.97373	0.00075	0.97478	0.00076	0.97724	0.00081	0.98253	0.00087

σ is the standard deviation of the M-C code.

4. Next five-year project

We have made a five-year project proposal on BUC technology. In this project, four issues will be carried out to investigate on BUC:

Firstly, the code will be improved, both the code for estimation of the spent fuel composition and the criticality calculation code.

Secondly, chemical assay for spent fuel to get the isotopic inventory will be done. The date will be used for validation of the prediction of the isotopic inventory of spent fuel.

Thirdly, we will also investigate burn-up measurement technology and develop spent fuel assembly burnup measuring system.

Finally, it is very important to carry out critical experiment on spent fuel for validating calculation codes using burnup credit technology. In this proposal, a critical experiment facility using spent fuel will be built. As described in the section 3, the critical calculation on the reactor core has been basically completed by now. Furthermore, there is a lot of work to do for the design of reactor core.

5. Conclusion

With the assistance of IAEA and our study, we have known a whole analysis method on BUC technology. A code package including ORIGEN-2 and M-C code is established in our laboratory. Some validating work according the OECD/NEA BUC benchmark problems have been performed. A five-year project on BUC technology is proposed. For this project, the primary critical calculation for a conceptual design core using spent fuel rods from Qinshan Power Plant is done.

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CRITICALITY SAFETY CRITERIA

(Session 2.3)

Recent advances in French validation program and derivation of the acceptance criteria for UOX fuel

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Abstract. As part of the implementation of burnup credit (BUC) in French spent fuel management systems, the CEA and the IRSN are developing, in collaboration with French nuclear industry, a "BUC" calculation route for UOx fuel, based on the connection of the French depletion codes DARWIN or CESAR and the French Criticality-Safety Package, CRISTAL. The purpose of this route is to take into account the depletion of fissile content, the production of absorbent isotopes and an axial burnup profile in the criticality calculations. Furthermore, a conservative depletion scheme has been defined in order to guarantee the conservatism of the spent fuel inventory for criticality calculations. Finally, conservative correction factors are applied to calculated isotopic concentrations before being integrated into the criticality calculations, for isotopes whose total calculated reactivity worth is not conservative. These correction factors have been determined from the experimental validation of both spent fuel inventory and reactivity worth. This paper details the recent advances in French "BUC" calculation route for UOx fuel and presents a comparison of its possible application with the earlier methodology termed, "actinide-only".

1. Introduction

Up to the 1980's, nuclear facilities dealing with spent fuel were designed with the assumption of fresh fuel. This assumption led to considerable safety margins.

In the early 1980's, in order to use the existing devices at the La Hague reprocessing plant for some irradiated UO_2 fuel initially enriched at 4.4% (an enrichment higher than the highest design stage enrichment of 3.5%), a method, termed "actinide-only", was proposed by the COGEMA enabling them to consider a certain amount of burnup in criticality studies. But in order to guarantee safety margins, only the depletion of ^{235}U and the formation of the following actinides ^{236}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu were taken into account. The axial burnup profile was considered flat and the burnup equal to the mean value of the end 50 least-irradiated centimetres.

This "actinide-only" method was accepted by the French safety authorities and afterwards was used for the transport of irradiated fuel and also at the design stage of the UP3 and UP2-800 La Hague reprocessing plants.

However, given the increasing initial enrichment and the growing needs of interim irradiated fuel storage, it is becoming necessary for the nuclear industry to reduce the conservatisms due to the very pessimistic assumptions of this "actinide-only" method.

As part of this work, the CEA and the IRSN are developing, in collaboration with French nuclear industry, a "burnup credit" (BUC) calculation route [1] for PWR-UOx assemblies, based on the connection of the French depletion codes DARWIN [2] or CESAR [3] and the French Criticality-

Safety Package, CRISTAL [4]. The methodology implemented in this "BUC" calculation route deals with :

- the reduction of the fissile content, the build-up of higher actinides and the increase of the concentration of fission products (FPs) with fuel burnup,
- the definition of conservative conditions of irradiation, in order to guarantee the conservatism of the spent fuel inventory for criticality calculations,
- the experimental validation of the spent fuel inventory and the reactivity worth for actinides and FPs taken into account for their participation in the reduction of the reactivity. This experimental validation will be used to elaborate conservative isotopic correction factors, to guarantee the conservatism of the spent fuel reactivity in criticality calculations,
- the description of an axial burnup profile.

2. Presentation of the French "BUC" calculation route

The French "BUC" calculation route is based on the connection of the French depletion codes DARWIN [2] (reference code) or CESAR [3] (industrial code) and the French Criticality-Safety Package, CRISTAL [4].

The DARWIN and CESAR depletion codes calculate the concentrations of isotopes at the end of irradiation or after cooling time. Then, these concentrations, to which a correction factor can be applied, are used as input data in the Criticality-Safety Package, CRISTAL, which provides the neutron multiplication factor associated to the calculated situation. The sequence of the codes is presented below.

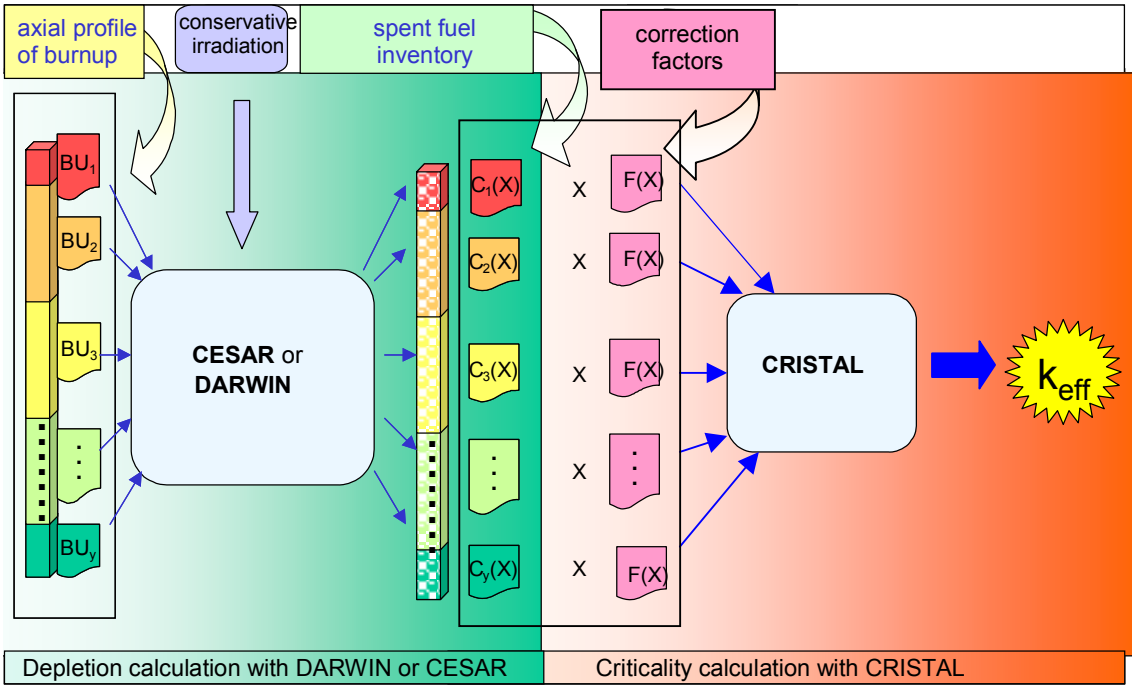

Figure 1: Presentation of the French "BUC" calculation route

3. List BUC isotopes

In order to take into account the burnup of the UOx fuel, through the depletion of fissile content and the build-up of absorbent isotopes, a study has been carried out to determine which nuclides contribute significantly to the negative burnup reactivity worth [5]. In addition to major actinides, some minor actinides and 15 fission products (FPs) have been selected for the BUC application. The 15 FPs

correspond to the most absorbing nuclides, stable and non-volatile ; they represent 79% of the total FP reactivity worth. This selection of nuclides, presented below in Table 1, has been approved by the OECD/Expert Group on BUC Criticality Safety. Table 1 presents the negative reactivity worth of the capturing isotopes for a UOx assembly with an initial enrichment of 3.3%, a burnup of 40 GWd/t and a cooling time of 5 years [5].

Table 1: BUC nuclides and negative reactivity worth of the capturing isotopes for UOx fuel

Actinides	Reactivity worth (in pcm)	Fission Products	Reactivity worth (in pcm)
^{234}U	-120	^{143}Nd	-900
^{235}U		^{145}Nd	-410
^{236}U	-910	^{147}Sm	-230
^{238}U		^{149}Sm	-1030
^{238}Pu	-310	^{150}Sm	-270
^{239}Pu		^{151}Sm	-500
^{240}Pu	-8370	^{152}Sm	-490
^{241}Pu		^{95}Mo	-290
^{242}Pu	-710	^{103}Rh	-1360
BUC Major actinides	**-14430**	^{155}Gd	-1500
^{237}Np	-620	^{153}Eu	-390
^{241}Am	-1290	^{109}Ag	-250
^{243}Am	-280	^{99}Tc	-440
BUC Minor actinides	**-2190**	^{101}Ru	-220
BUC Major + Minor actinides	**-16620**	^{133}Cs	-750
BUC actinides + 15 FPs	**-25650** (90% of the total BUC)	**BUC 15 FPs**	**-9030** (79% of the total FP BUC)

4. Conservative conditions of irradiation

The reactivity of burned fuel increases when the irradiation conditions lead to hardening neutron spectrum, this hardening being dependent on control rod insertion, MOx environment and irradiation parameters (increase of moderator and fuel temperature, increase of boron concentration).

Studies on UOx assemblies have been carried out to evaluate the effect of these irradiation conditions on criticality calculations (while considering a flat axial burnup profile) [1] [6]. Based on these results, recommendations for the "BUC" calculation route have been elaborated.

4.1. Environment of the UOx assembly

In France, plutonium from the La Hague reprocessing plant is recycled in PWRs. Twenty 900 MWe PWRs are currently devoted to recycling Pu in 30% mixed core loading. The effect of MOx fuel around a UOx assembly has been quantified : the reactivity of a pool storage of UOx assemblies, irradiated at 40 GWd/t in a complete MOx environment, is 1200 pcm higher than the reactivity of a pool storage of UOx assemblies irradiated in a complete UOx environment [6].

In order to be conservative, the irradiation of the UOx assembly is modelled, in the depletion part of the "BUC" calculation route, with a MOx environment, composed of either 8 irradiated MOx assemblies, or 4 irradiated MOx assemblies and 4 irradiated UOx assemblies. The complete MOx environment is applied by the French fuel cycle industrial COGEMA. The choice of 8 MOx assemblies is justified by the fact that COGEMA stores and transports assemblies coming from not only French reactors but also assemblies coming from foreign reactors which can contain more than 30% of MOx ass

4.2. Control rod insertion

French PWR operations can involve periods of partial control rod (CR) insertion. In order to maximise the reactivity effect due to this insertion, the effect of a full axial control rod insertion (Ag-In-Cd or B_4C) during the entire burnup has been quantified : the reactivity of a pool storage of UOx assemblies irradiated with CRs fully axially inserted from 0 to 40 GWd/t is about 4000 pcm higher than the reactivity of a pool storage of UOx assemblies irradiated without CRs [6].

These 4000 pcm include two large conservatisms : the time of CR insertion (all the irradiation) and the level of axial insertion of CRs (full axial insertion). But although the probability is low that an assembly is irradiated with CRs inserted during all its irradiation, it is difficult to exclude that possibility. Furthermore, it is assumed that an operator is not able to guarantee a low axial CR insertion.

In order to be conservative, the irradiation of the UOx assembly is modelled, in the depletion part of the "BUC" calculation route, with CRs inserted throughout all the irradiation. The criticality calculation uses the fuel inventory calculated with this depletion scheme on the full length of the assembly, in order to simulate an irradiation with a full axial CR insertion.

4.3. Fuel temperature

It is conservative to consider a high value for fuel temperature as it leads to more resonant captures on ^{238}U, and then to further production of ^{239}Pu. However, the fuel temperature used in the depletion calculation has a small effect on the reactivity of a pool storage, about +5 pcm/°C at 50 GWd/t [8].

Thermo-mechanical calculations have shown that the fuel average temperature in PWR reactors is always below 700°C in the plateau at fuel mid-height (on the lower and upper fuel column, fuel temperature is lower). However, after the first irradiation cycle, the average temperature decreases below 600°C [7]. The conservative fuel temperature value used in the depletion calculation is therefore 600°C, furthermore this value is applied to the full length of the assembly.

4.4. Moderator temperature

In a PWR, as the moderator temperature increases, the moderator density decreases, which results in reduced moderation and therefore in spectral hardening. So it is conservative to consider a high value of the moderator temperature as it leads to an increase in the reactivity (about +80 pcm/°C at 50 GWd/t [8].

As the moderator temperature increases with the fuel length (i.e. the moderator temperature is higher at the top of the fuel than at the bottom, with a mean value of 304°C), the value of the conservative moderator temperature used in the depletion calculation is the mean temperature of the outlet water. This value, 325°C in nominal conditions, is applied to the full length of the assembly.

4.5. Boron concentration

Increasing boron concentration results in spectral hardening due to the absorption of thermal neutrons in the moderator by boron. So, it is conservative to consider a high value of the boron concentration as it leads to an increase in the reactivity (can reach 3 pcm/ppm at 50 GWd/t [8]).

A soluble boron concentration of 1200 ppm is typical of Beginning-Of-Cycle (BOC). This value decreases down to 900 ppm during the first days due to Xenon and Samarium poisoning, and afterwards decreases linearly down to 0 ppm at the End-Of-Cycle (mean value during an irradiation cycle : C_B = 456 ppm). Sensitivity calculations have shown that a C_B = 600 ppm constant value is fully bounding the actual boron decrease during irradiation cycle.

4.6. Fuel specific power

The specific power used during depletion calculations has a slight effect on the reactivity in the 30-48 W/g PWR operating range. The value of the conservative specific power used in the depletion calculation is thus 40 W/g, with no inter-cycle downtime in order to minimize ^{241}Pu decay.

5. Determination of the isotopic correction factors

The fuel inventory, calculated with the DARWIN or CESAR depletion codes, is used as input data in the Criticality-Safety Package CRISTAL, which provides the neutron multiplication factor associated to the calculated situation. In order to guarantee the conservatism of the fuel reactivity in criticality calculations, correction factors can be applied to the concentrations of isotopes.

Determination of the correction factors for UOx fuel is based on the experimental validation [9] of the spent fuel inventory carried out using the DARWIN package, and on the experimental validation of the reactivity worth carried out with the CRISTAL package.

5.1. Experimental validation of the spent fuel inventory

The depletion calculations are performed using the French depletion codes DARWIN [2] or CESAR [3], the latter being validated by a code to code comparison with DARWIN. The current version of these two codes uses the same library JEF2.2. The determination of the isotopic correction factors concerning the experimental validation of the spent fuel inventory is therefore based on the experimental validation of the DARWIN depletion code [10]. This experimental validation consists in comparing calculated spent fuel inventories with spent fuel chemical assays. The experimental data is 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 plant. This enables us to cover a large range of UOx fuels with various enrichments in ^{235}U, from 3.1% to 4.5%, and burnups from 10 GWd/t to 60 GWd/t.

In order to have an unique correction factor per isotope for any burnup, the Calculation to Experiment ratio (C/E) on fuel inventory used to elaborate isotopic correction factors for each isotope is :

- either the mean value obtained from all the results when there is no shift of the (C/E) with the burnup,
- or the value which minimises the neutron multiplication factor if there is a shift of the (C/E) with the burnup, or if any trend from all the results is observed. In the case of a fissile nuclide, it corresponds to the lower (C/E), and in case of an absorbent nuclide, it corresponds to the higher (C/E).

5.2. Experimental validation of the reactivity worth

5.2.1. Fission products

The experimental validation of the FP reactivity effect, carried out with the CRISTAL package [4], consists in comparing the calculated reactivity worth with the experimental reactivity worth for each BUC FP [11]. The measurements have been carried out using the oscillation technique in the French experimental reactor MINERVE [9].

This experimental validation enables us to determinate the Calculation to Experiment ratio (C/E) on each FP reactivity worth. These (C/E) are used to elaborate isotopic correction factors.

5.2.2. Actinides

Currently, there is no available experimental programme devoted to the validation of the actinide reactivity worth, excepted ^{235}U. In substitution, we have used the uncertainties deduced from French

work on JEF2.2 nuclear data uncertainty [12] : for fissile isotopes, we took uncertainties in neutron multiplicity, fission and absorption cross sections into account ; for other isotopes, we have used the uncertainties linked to thermal capture cross section and resonance integral.

These uncertainties on the actinide reactivity worth are translated into isotopic correction factors.

5.3. Determination of the isotopic correction factors

The correction factor applied to a BUC nuclide concentration is the product of the (E/C) deduced from the experimental validation of the spent fuel inventory with the (E/C) deduced from the experimental validation of the reactivity worth (or JEF2 data uncertainty), if and only if the product of this correction factor with the nuclide concentration calculated with the DARWIN package results in a new concentration which leads to an increase in the K_{eff}. If it is not the case, the correction factor is equal to 1 (the calculated concentration is not modified) in order to maintain the conservatism of the concentration.

Table 2 presents the correction factors associated to DARWIN-JEF2 tool for most of the BUC nuclides. The application of these correction factors enables us to correct the underestimation of some fissile isotopes and the overestimation of some absorbent isotopes for the total calculated reactivity worth.

Table 2: Correction factors for PWR-UOx assemblies

Actinides	Correction factor	Fission product	Correction factor
^{234}U	0.96	^{143}Nd	1
		^{145}Nd	1
^{236}U	1	^{147}Sm	1
^{238}Pu	1	^{149}Sm	1
		^{150}Sm	1
^{240}Pu	1	^{151}Sm	0.86
^{241}Pu	1.06	^{152}Sm	0.94
^{242}Pu	1	^{95}Mo	1
^{237}Np	1	^{103}Rh	0.85
^{241}Am	0.96	^{155}Gd	0.91
^{243}Am	1	^{153}Eu	0.88
		^{109}Ag	1
		^{99}Tc	0.92
		^{101}Ru	0.72
		^{133}Cs	1

6. Axial burnup profile

The dynamics of reactor operations result in non-uniform axial burnup profiles in fuel with any burnup.

The effect of the assembly axial burnup profile in burnup credit calculations was investigated in the OECD/NEA benchmarks Phase IIA [13], IIB [14] and IIC [15]. Due to a strong shift of the flux towards the fuel pin extremities with increasing burnup, the simplified flat burnup model is not conservative for burnup credit calculations of PWR casks and storage for highly irradiated assemblies, i.e. mean BU > 30 GWd/t.

French axial profiles of burnup measured at the La Hague reprocessing plant are being examined in order to determine a realistic bounding burnup profile.

7. Applications

The burnup credit reactivity versus burnup for a cooling time of 1 year has been evaluated using the "BUC" calculation route (the CRISTAL calculations were performed with the design route, APOLLO2 Pij and Sn [16]) for a pool storage of PWR-UOx assemblies, and compared to the earlier methodology termed, "actinide-only", and a best-estimate burnup credit.

The assumptions associated with each methodology are the following:

- No BUC
  - no depletion calculation,
  - fuel inventory in criticality calculation = fresh fuel,
- BUC "actinide-only"
  - depletion calculation with mean conditions of irradiation (see table 3),
  - fuel inventory in criticality calculation composed of the following isotopes ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu,
  - flat burnup profile, the burnup used in the criticality study is equal to the mean value in the end 50 least-irradiated centimetres,
- BUC "actinides + FPs"
  - depletion calculation with conservative conditions of irradiation,
  - fuel inventory in criticality calculation composed of the following actinides ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{237}Np, ^{241}Am and ^{243}Am, and the following 15 FPs ^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{152}Sm, ^{155}Gd, ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{109}Ag, ^{145}Nd, ^{147}Sm, ^{150}Sm, ^{151}Sm and ^{153}Eu,
  - application of correction factors to the concentrations of concerned isotopes,
  - axial burnup profile, equal to the mean German profile of the OECD/phase IIC benchmark (at the time of the study, the French axial burnup profiles had not yet been examined).
- Best-estimate BUC
  - depletion calculation with representative conditions of irradiation,
  - fuel inventory in criticality calculation composed of irradiated fuel (all actinides and 74 main FPs),
  - axial burnup profile, equal to the mean German profile of the OECD/phase IIC benchmark.

Table 3 presents the representative, the conservative and the mean conditions of irradiation.

Table 3 : Representative, conservative and mean conditions of irradiation

Condition of irradiation	Mean	Conservative	Representative
Fuel temperature	550°C	600°C	Axial profile
Moderator temperature	304°C	325°C	Axial profile
Mean boron concentration	456 ppm	600 ppm	456 ppm
Specific power	36 W/g	40 W/g	36 W/g
Environment	no	4 MOx assemblies and 4 UOx assemblies (can be 8 MOx assemblies)	no
Control rods	no	24 AIC	no
Insertion of CRs	-	Throughout all the irradiation	-

Table 4 presents the BUC associated to each application for a pool storage of UOx assemblies (^{235}U 4.2 wt% enrichment).

Table 4 : Comparison of "Actinide-only" and "Actinides + FPs" BUC

	"Actinide-Only" BUC	"Actinides + FPs" BUC	Best-estimate BUC
20 GWd/t, Ctime 1 year	4400 pcm	9000 pcm	12000 pcm
30 GWd/t, Ctime 1 year	6900 pcm	12600 pcm	17500 pcm
40 GWd/t, Ctime 1 year	9500 pcm	16000 pcm	23300 pcm
50 GWd/t, Ctime 1 year	12400 pcm	19000 pcm	29300 pcm
% of best-estimate BUC	**~ 40%**	**~ 70%**	**100%**

The new “Actinide+FP” calculation route in French tools enables us to handle 70 % of the best-estimate BUC, compared to 40% in the earlier "Actinide-Only" BUC methodology (pool storage application).

8. Conclusion

This paper has presented the recent advances in French "BUC" calculation route for UOx fuel, based on the connection of the French depletion code DARWIN (reference code) or CESAR (industrial code) and the French Criticality-Safety Package CRISTAL.

The various assumptions linked to the depletion calculation or the criticality calculation are summarised below :

(1) Depletion calculation

- Description of an environment of 4 irradiated MOx assemblies and 4 irradiated UOx assemblies, or 8 irradiated MOx assemblies (depending on the choice of the French nuclear industry),
- Full insertion of 24 Ag-In-Cd or B_4C control cluster in the central UOx assembly, throughout all the irradiation,
- Fuel temperature = 600°C,

- Moderator temperature = 325°C,
- Boron concentration = 600 ppm,
- Specific power = 40 W/g.

(2) Criticality calculation

- Fuel inventory composed of the following BUC actinides 234U 235U, 236U, 238U, 238Pu, 239Pu, 240Pu, 241Pu, 242Pu, 237Np, 241Am and 243Am, and the following 15 BUC-FPs 103Rh, 133Cs, 143Nd, 149Sm, 152Sm, 155Gd, 95Mo, 99Tc, 101Ru, 109Ag, 145Nd, 147Sm, 150Sm, 151Sm and 153Eu,
- Some nuclide concentrations are multiplied by a conservative correction factor,
- Description of an axial profile of burnup.

Table 5 presents the conservatisms linked to these assumptions for a mean burnup of 40 GWd/t (considering an axial burnup profile) and a cooling time of 1 year.

Table 5 : Conservatisms introduced in the "BUC" calculation route (at 40 GWd/t)

Irradiation & Criticality model	**Conservatisms**
Fuel temperature	+ 150 pcm
Moderator temperature	+ 300 pcm
Boron concentration	+ 300 pcm
Specific power	+ 30 pcm
Environment	+ 900 pcm
CRs inserted on the full length of the fuel and throughout all the irradiation	+ 3400 pcm
Limitation to BUC isotopes	+ 2000 pcm
Correction factors	+ 600 pcm
TOTAL	**+ 7680 pcm**

The French "BUC" calculation route has been evaluated for a pool storage of PWR-UOx assemblies. This BUC amounts to -16000 pcm for a 4.2% UOx fuel irradiated at 40 GWd/t, which represents about 70 % of the best-estimate BUC.

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Development of burnup credit loading criterion for the Sizewell B spent fuel storage ponds

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Abstract. In 1996, prior to discharge of any irradiated fuel, the Sizewell B PWR spent fuel storage pond was re–racked. The new racks were partitioned into two regions. Region 1, a low density region, consists of 308 cells each a pitch of 280mm, in which each cell is separated by two boral absorbers plates. This was designed for the storage of fresh fuel of enrichment up to 5.0w/o. Region 2, a relatively high density region, consists of about 1600 cells with a pitch of 230mm, in which each cell is separated by one boral plate. This was designed for the storage of irradiated fuel with credit taken for burn-up. Note that Sizewell B uses 17 by 17 fuel which has been supplied by either BNFL-Westinghouse or FANP. A minimum irradiation versus enrichment loading criterion was developed for Region 2 and all fuel has been loaded into Region 2 only if it meets this criterion. This paper describes the development of this loading criterion. The burn-up credit case has not yet been presented to the regulator. This awaits a satisfactory case to cover misloading in Region 2 of the pond. This paper doesn't address misloading. The current safety case for Region 2 relies on checkerboard loading of fuel into Region 2 with loading of alternate cells prevented by inserts. This case has been made for fresh fuel up to 5w/o although all fuel loaded to date has, as noted above, met the derived criterion in anticipation of a burn-up credit case.

1. Overall basis for the case

The burn-up criterion was developed on the basis of credit for actinide and fission product absorption. The Xe-135 number density was set to zero and no credit was taken for any other fission product or actinide decays after discharge. Over time these considerably reduce reactivity.

The criticality case was made on the standard UK basis of:

$$\text{k-eff} + S + 3\sigma < 0.95$$

where the systematic term, S, includes the systematic bias of the code used, as derived from appropriate benchmark critical experiments, and manufacturing tolerances (enrichment, fuel density, fuel dimensions, rack dimension, B-10 concentration in the boral). The random uncertainty, σ, included the Monte Carlo sampling uncertainty and the accuracy of the code employed when compared against a relevant range of critical benchmark experiments. Note this is the sample standard deviation from experimental comparisons rather than the standard deviation of the mean as is used in some analyses. Further systematic and uncertainty components for irradiated fuel were added to those for fresh fuel as described below. For 5w/o fuel at the minimum burn-up required for loading in Region 2 46.5GWd/tU, the total systematic allowance added excluding the code bias was 0.049 and the total, 3 sigma random uncertainty was 0.020 i.e. the best estimate k-eff was < 0.88.

The analysis took no credit at all for soluble boron in the fuel pond even though the required minimum level is 2000ppm during fuel movement and in practice it is maintained above 2500ppm at all times. This is a very significant conservatism. At 2500ppm the criticality criterion can be met for the Region 2 racks full of fresh 5.0w/o fuel. It was also assumed that the pond water was at 4^{o}C to maximise reactivity.

The calculations for the rack cell geometry were performed in 3-D but generally for a single storage cell, assuming an infinite array of similar cells.

2. Codes used

The main analysis was made by a contractor, Holtec, using the KENO-5a, Monte Carlo, multigroup neutron transport code plus the lattice code CASMO-3 for depletion analysis. Note that KENO required a subsidiary U-238 resonance shielding calculation – NITAWL. This analysis was performed on the basis of equivalent fresh fuel reactivity i.e. CASMO was used to derive a fresh fuel enrichment which gave the same reactivity as the depleted fuel at cold conditions with zero soluble boron. The KENO calculations were then performed with this equivalent fresh fuel. Note however that this approach was found to give quite poor accuracy for conditions which depart significantly from those for which the equivalence has been derived. In particular at high soluble boron a new fresh fuel equivalence should be derived. Most of the KENO calculations were run with 20 neutron groups but there were some confirmatory calculations with 218 groups and with the continuous energy Monte Carlo transport code MCNP.

The Holtec analysis was verified and extended by British Energy using the UK codes, MONK (version 6) a very fine mesh Monte Carlo transport code, and LWRWIMS a reactor lattice code that at that time ran in up to 69 neutron groups. Both codes then had nuclear cross-section libraries derived in 1986 based on the UK nuclear data library and LWRWIMS also had a later 1996 library based on JEF 2.2. Note two differences here. As a very fine mesh code MONK doesn't require a special treatment of resonances. Secondly the MONK calculations were not performed on the basis of equivalent fresh fuel reactivity but by transferring the actinide and fission product number densities calculated by LWRWIMS to MONK.

The LWRWIMS code with either the JEF2.2 cross-sections or the 1986 UKNDL cross-sections predicted essentially the same, somewhat higher reduction of reactivity with irradiation than CASMO. At the highest irradiation LWRWIMS predicted a slightly smaller reduction in reactivity by 1.3% after 43GWd/tU (a $\otimes$k of -0.2970 compared to -0.3008) than CASMO. This difference would have increased a little to 2.5% if consistent fuel temperatures had been used in the two depletions. However this is well inside the claimed uncertainty for CASMO of 5% for prediction of this reactivity reduction.

MONK and KENO were compared for an infinite array of region 2 cells filled with 5w/o fuel depleted to 43GWd/tU. After application of the respective code biases (+0.0103 for KENO and –0.0053 for MONK) the MONK prediction was higher by 0.0059. If the LWRWIMS-CASMO depletion difference noted above is subtracted this difference reduces to only 0.0021. Again this is smaller than the calculational one sigma random uncertainty claimed for either code, 0.0028 for KENO and 0.0027 for MONK.

3. Assembly irradiation distribution and depletion effects

There are two main difficulties with a loading criterion based on assembly average irradiation.

Firstly both the axial and radial irradiation distribution vary within an assembly. In circumstances discussed below these can both lead to higher reactivity in the fuel pond than uniform distributions.

Secondly the conditions under which the fuel is irradiated can affect the isotopic mix and hence reactivity for a given irradiation. Most important of such conditions are water density, the presence of absorbers during core depletion particularly those which are not then present in the fuel pond (either soluble boron, burnable poisons or control rods) and fuel temperature. Low water density or absorber presence harden the neutron spectrum and increase the Pu-production rate. High fuel temperature leads directly to a higher conversion ratio. Low water density and high fuel temperature can arise from high temperature inlet conditions and from high channel and overall power. High power also gives rise to

high xenon levels but this is a relatively small effect because Xe-135 is nearly saturated at Sizewell B average rating levels.

A further complication is that these two effects are intimately linked. Most importantly the impact of axial irradiation distribution has to be combined with the effect of the axial water density profile during irradiation. This water density profile significantly increases the axial re-distribution effect. Some past analyses in the literature have neglected this effect.

It is most unlikely that any single fuel assembly experiences the maximum of all these effects over a long enough time for these to affect depletion. The problem for the analyst is to cover these in a sufficiently conservative manner.

The Sizewell B analysis covered them in the following way:

3.1. Axial irradiation distribution effect

The CASMO calculations were performed for a number of axial layers of the core with an assumed water density profile. Irradiation distributions were taken from US PWR data and an equivalent fresh fuel reactivity derived for each layer. These were then input to the KENO model for a single storage cell. The following results were found.

Enrichment/Burnup	**Reactivity Effect of Axial Irradiation**
Up to 30 GWd/tU	Negative effect but taken to be zero
4w/o, 32.3GWd/tU	0.0050
4.5w/o, 39.5GWd/tU	0.0145
5w/o, 46.5GWd/tU	0.0245

These results are in reasonable agreement with those reported in Reference 1. Note that these calculations produce an extremely skewed axial distribution with a flux peak right at the top of the core. As such these calculations are extremely sensitive to the assumed properties of the materials in the pond above the active fuel.

Subsequently the irradiation distributions used in the analysis were compared with those of fuel discharged from Sizewell B. It was judged that the US histories used adequately bounded those from Sizewell B apart from fuel that had spent part of its irradiation in an operational control rod (Z-bank) position. Such histories can have a significantly bigger axial effect. This has been allowed for in the application of the criterion at Sizewell by imposing a requirement for such assemblies to have an additional margin to the loading criterion. This additional margin is 3GWd/tU if the last cycle dwell was not in a Z-bank position and an additional 8GWD/Te for fuel which was irradiated in its last cycle of operation in a Z-bank position. (Note that the allowances cover increased Z-bank insertion through periods at part power).

3.2. Radial irradiation distribution effect

Assemblies irradiated for one or more dwells on the core boundary can have a significant variation of burn-up across the fuel array. The maximum quadrant irradiation mismatch seen in Sizewell B discharge irradiations was found to be +/- 12%. This was for fresh fuel placed on the core edge in its first dwell. The potential impact of this on reactivity in Region 2 was bounded by doing a supercell calculation which represented four such tilted irradiation assemblies in the racks with the four low irradiation quadrants adjacent. The increase in reactivity was found to 0.0007.

This effect has not been included in the loading criterion for the following reasons:

- the value above is calculated very conservatively
- the effect is small and much smaller than the allowance made for the presence of discrete burnable absorbers (see below) and it is very unlikely that such low burn-up quadrants would contain burnable absorbers
- axial flux redistribution skews the flux right to the top of the fuel where irradiations are much smaller and the radial effect would be much smaller.

3.3. Allowances for core average depletion conditions

- Soluble boron concentration

The analysis was performed with a cycle average soluble poison concentration of 700pm which at the time bounded operation at Sizewell B. However there were some proposed cycle designs with higher cycle averages, of up to 850ppm. As a result allowance was made for a fuel lifetime averaged soluble poison concentration of 900ppm. Compared to depletion of 700ppm this added 0.0054 to the reactivity of fuel of 5w/o @ 46.5GWd/Te. However, today, after 8 cycles of operation the highest cycle average value has still been less than 700ppm.

- Burnable absorbers

An allowance was made for the presence during first dwell of discrete burnable poison rods (BPRA - pyrex) which are then removed. This effect was calculated assuming 20 BPRA rods were inserted to 23GWd/tU. The effect was found to be an increase in reactivity of fuel of 5.0w/0 at 46.5GWd/tU of 0.004. Note that BPRAs were only present in Sizewell cycle 1. A different burnable absorber design, WABAs, were used in cycle 2 with up to 24 BP rods. These have less of a depletion effect because they displace less water even though there were an increased number of rods. Subsequent cycles have only used integral (Gd) burnable poisons. Whilst they also have the effect of hardening the neutron spectrum and thereby increase reactivity for a given irradiation, the fuel is never more reactive for a given mean irradiation than unpoisoned fuel.

- Fuel temperature

The CASMO depletion calculations were carried out for a conservatively high through- life average fuel temperature of 1016K. The value calculated by the ENIGMA code at zero irradiation and core average rating is 910K. This effect increases reactivity at 46.5GWd/tU for 5w/o by about 0.004. Moreover the time averaged fuel temperature is predicted to reduce with irradiation as the fuel-can gap closes despite reduction of fuel conductivity. The assumed value of 1016K allows for a sustained rating of 30% higher than average. This is very conservative even more so because the axial irradiation effect leads to flux peaks at the top of the fuel where the rating and fuel temperature during depletion would be considerably lower.

- Water temperature and density

An average water temperature of 583.9K, water density of 0.703gm/cc was assumed in the CASMO depletion calculations. This is slightly lower than the design core average value of 0.704gm/cc and therefore slightly conservative on average. The greatest sustained variation in assembly power at discharge for Sizewell B is < +/-18%. At this maximum such channels would be irradiated at a mean density of ~ 0.690gm/cc. Lattice calculations show that this would increase reactivity by about 0.0030 for 4.0w/o fuel at around 40GWd/tU. This variation is smaller than the other effects allowed for as noted above and has not been allowed for explicitly because such assemblies with a higher than average irradiation than the batch mean meet the loading criterion with some margin. Furthermore periods of high assembly power are likely to occur early in fuel life and their effects will tend to burn-out towards end of life when the assembly power will be lower. Note that the axial variation in water temperature and density has been included in the axial burn-up variation allowance.

- Assembly power

As noted above the greatest sustained variation in assembly power at discharge for Sizewell B is < +/- 18% for fuel irradiated to about 40GWd/tU. The possible effect of sustained variations in assembly power on isotopic depletion are mainly through high fuel temperature and low water density and these have been covered above. There is a further effect of high assembly power on depletion, the Xe-135 concentration, which has not been explicitly allowed for. However the xenon concentration is close to saturation at core average rating and this is judged to be a small effect.

3.4. Combined effects

The irradiation and depletion distribution allowances identified in the above sections have been added as systematics. They are systematic in the sense of being predictable but their occurrence, particularly their coincidence in a single or adjacent assemblies within the storage pond, would have a random character.

All the depletion condition allowances noted above in section 4.3 were derived for assembly average irradiations. As pointed out above the axial redistribution effect pushes the peak flux right to the top few percent of the fuel assembly where irradiations are much smaller. It will therefore be conservative to compute the depletion allowances at the assembly average irradiation. Furthermore it is conservative to assume a high fuel temperature for the top of the core where in reality the fuel temperatures would be considerably lower than average.

It is judged that the overall combination of depletion conditions and irradiation distribution factors considered bound the reactivity effects that could occur. It is judged that further work on evaluation of the axial irradiation distribution and depletion effects should be done in which the effects are all combined in single calculation representing actual fuel histories. This could be done by employing a best estimate microscopic depletion calculation in the PANTHER core follow route and transferring nuclide number densities for discharged fuel to a Monte Carlo code representing the storage rack. This would then be compared directly with the keff < 0.95 criterion. This further work also needs to look at the assumed top axial reflector boundary condition to which the calculations have been found to be extremely sensitive.

4. Irradiated fuel uncertainties

It was noted above that two additional random uncertainties for irradiated fuel were added to the standard (KENO) calculational random uncertainty for fresh fuel. The values for these were derived and justified as follows.

4.1. Depletion uncertainty

The quoted uncertainty on CASMO prediction of reduction of reactivity with burn-up was claimed to 5%. This was only based upon the accuracy with which cycle lengths are predicted. However the LWRWIMS code had been compared against the CERES 1 reactivity measurements of 4 spent PWR fuel samples made in the DIMPLE reactor (Ref 2). These had irradiations ranging from 20Gwd/tU to 60GWd/tU. LWRWIMS was found to slightly underpredict the reactivity reduction by 1.1% with a standard deviation of 2%. The 5% uncertainty applied to the CASMO core was therefore consistent with a 3sigma value for LWRWIMS. Note there were further reactivity measurements in the CERES programme made in the MINVERVE reactor (Ref 3) which were not available to the author but are said to be consistent with the DIMPLE results.

4.2. Irradiation uncertainty

The assembly irradiations used at Sizewell B to compare against the loading criterion are derived using the PANTHER code that follows the core history. An assessment was carried out of the accuracy of such calculations. This accuracy was based primarily on start-up assembly power

measurements and how such assembly power differences are sustained through irradiation, the accuracy of overall power calibration and the accuracy with which the core follow history can be followed, particularly through changes in load. This assessment concluded that an appropriate 3sigma uncertainty was 5.4%. This assessment was associated with recommendations about how to carry out the core follow calculations conservatively, This included the adequacy of time steps, particularly through load changes, and what to do about any missing data.

5. Summary

A minimum assembly irradiation versus enrichment criterion has been developed for Region 2 of the Sizewell B fuel storage racks which takes credit for actinide and fission product burn-up. The criterion incorporates conservative allowances for the effects of within assembly axial irradiation distribution and the effects of conditions that can occur in operation which affect depletion. These include absorber presence, low water density and high fuel temperature. Additional irradiation uncertainties for the prediction of reactivity reduction with irradiation and for the derivation of assembly irradiation have been derived and justified. Comparisons of two different code sets used for these analyses CASMO/KENO and LWRWIMS/MONK have given good agreement within the uncertainties allowed for.

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Analysis of axial burnup profile and burnable poison loading on spent BWR fuel reactivity in the THORP dissolvers

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Abstract. The criticality safety case for the THORP Dissolvers applies 'Actinide-Only' Burnup Credit to show that the plant is safe from criticality for a range of normal and accident conditions. The original assessment was based primarily on calculations for PWR fuel, with design and irradiation parameters set to give conservative estimates of spent fuel reactivity. In particular those parameters affecting neutron spectrum were set to harden the spectrum and so enhance Pu build-up. At that time anticipated BWR enrichment was low enough so that criticality criteria would not be challenged and the PWR fuel represented the most limiting case.

To assess the impact of potential increases in BWR enrichment a series of calculations was made to establish that the original PWR-based assessment would continue to bound likely BWR fuels. In particular the effect of the axial void profile, axial blankets, and burnable poisons on burnup and Pu build-up were studied in detail.

3D MONK calculations to model depletion of BWR fuel were made for a range of burnups and initial enrichment. These calculations provided the spent fuel composition for WIMS calculations of k-eff in the dissolver. The results confirmed that the original PWR-based assessment remains conservative relative to any forseen BWR enrichment. Comparisons are made between the axial burn-up profiles generated by the 3D MONK calculation and typical BWR burnup profiles. It is seen that, above about 30GWd/t, the profiles generated by the MONK calculation are consistent with the typical BWR axial burnup profile. Depletion and k-eff calculations were also made for a range of burnable poison loadings in BWR fuel. These also demonstrated that the original analysis for PWR fuel remains bounding.

A summary of the calculation route and discussion of the results is presented in this paper.

1. Introduction

The original THORP Burnup Credit (BUC) case was based on analysis of 'worst-case' PWR fuel. At that time Initial Enrichment (IE) of BWR fuel programmed for dissolution was sufficiently low that very limited credit for burnup (BU) was needed to comply with criticality criteria. In anticipation of higher IE for BWR fuels, sensitivity calculations were made to assess impact of BWR specific design/irradiation parameters.

This paper summarises the results of those calculations, which were made to assess the effects of:

- axial BU Profile (including axial blanket)in BWR Fuel
- presence of fixed burnable poisons

A brief recap of the Burnp Credit methodology applied to THORP is also given.

2. Recap of BUC methodology as applied to THORP Head-End Plant

Details of the THORP Head-End Plant and the implementation of a BUC criticality assessment are given in Reference 1. In summary, spent fuel is taken from the THORP Feed Ponds via the Fuel Pond Feed Monitor (FPFM) and sheared into short lengths before being passed into a perforated steel basket in dissolver vessel. The Dissolver contains hot nitric acid and Gd nitrate. Once the requisite amount of fuel has been fed, the Dissolver lid is closed and the contents heated to leach temperature.

BUC was seen as way of reducing Gd poison concentration and hence reduction in waste volumes. Prior to applying BUC consideration was also given to other alternatives, e.g.

- credit for fuel dissolution and/or 'real' packing fractions
- batching the fuel by initial enrichment band

Optioneering studies concluded that Actinide-only credit offered the most practicable way forward for a number of reasons:

- burnup is an intrinsic feature of the fuel
- other candidates (e.g. fuel dissolution) might be undermined by accident conditions (dilution)
- global packing fraction in the dissolver is reasonably well known, but local variations are difficult to quantify
- batching by initial enrichment would involve variable safe limits on gadolinium which would be complex to implement.

On that basis an Actinide-Only Credit methodology was developed with the following key features and simplifying assumptions:

- credit for changes to U235, U238, Pu238, Pu239, Pu240, Pu241, Pu242 content in fuel
- packing fraction in the Dissolver to be optimised (i.e. no credit for 'real' packing fraction)
- worst-case fuel & irradiation parameters
- no credit for Fission Products.

A large number of depletion calculations for fuel in the reactor and k-eff calculations for spent fuel conditions (normal and accident) in the Dissolver were then made to establish safe loading curves with respect to combinations of Initial Enrichment (IE) and Burnup (BU), as illustrated below in Figure 1.

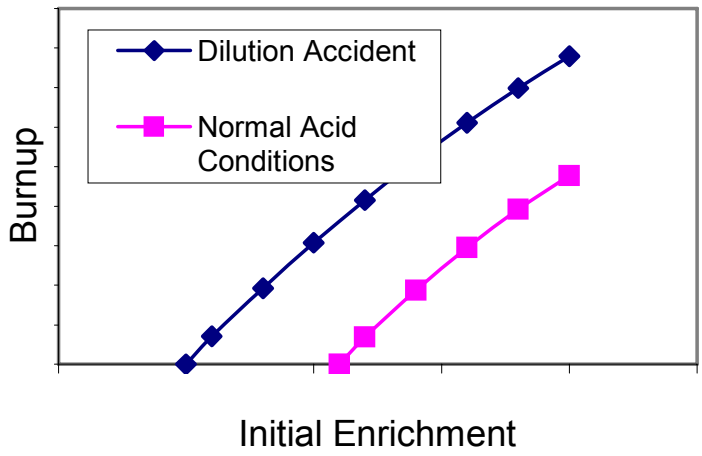

Fig. 1. Safe loading curve for normal and accident conditions.

Compliance with the loading curve is demonstrated through a combination of checks on supplier's data and measurement of Residual Enrichment (RE) by the FPFM.

Most of the original analysis was based on calculations for a 'worst-case' PWR fuel design:

- high fuel-to-moderator (maximum rod diameter, minimum pitch)
- high soluble boron concentration
- low cooling time
- high power rating.

Some additional calculations were made for sensitivity to fuel type (AGR, BWR), but at that time (circa 2002) the maximum IE for these fuels did not challenge the loading curve for normal conditions. Also the PWR spent fuel isotopics were found to be bounding.

Following implementation of the new BUC criticality safety case further analyses have been made to confirm that the original assessment was not being challenged by the addition of other fuels to the THORP programme. The remainder of this paper summarises such calculations made for BWR fuels at slightly higher IE than those considered under the original analysis.

3. Additional analyses for BWR fuel

3.1. Effect of axial BU profile and blankets

BWR fuel designs often include axial blankets at top and bottom of fuel elements and it was argued that it is conservative to ignore the blanket and consider only the enriched central axial zone when assessing compliance with loading curves. A second important feature of BWR fuel irradiation is the presence of voids in the coolant, particularly at top of the elements. This results in higher production of Pu in the harder spectrum at these locations and a shift in the peak in axial BU shifts towards the top of fuel element.

In the original assessment a very simple 2-zone model of concentric cylinders of BWR fuel fragments in the dissolver, illustrated below, was considered sufficient to assess the impact of these effects.

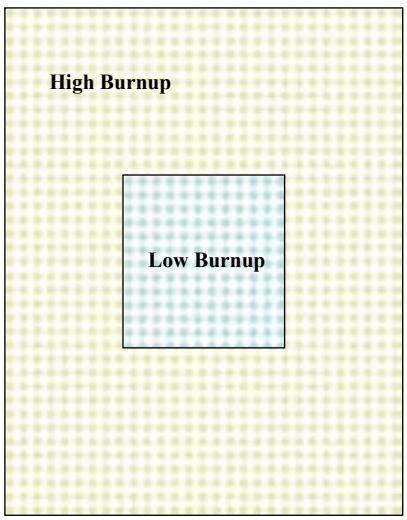

Fig. 2. Schematic of original 2-zone model for BWR fuel in dissolver (side view).

In this model the inner cylinder was filled with BWR fuel fragments, (represented as spheres) with low BU representative of fuel irradiated at the upper part of the central zone. The outer zone is filled with fuel representative of the middle region of the central axial zone. This was considered to be conservative because:

- no credible mechanism exists for sorting of cut fuel pieces in this way (the full dissolver batch is more than one fuel element)
- no credible mechanism for suspension of fuel pieces can be identified
- real packing fractions in the Dissolver are much less reactive than the optimised model.

This representation gave slightly higher k-eff than the base model (element average BU over the whole Dissolver), but the difference was small compared to other margins.

In later campaigns the IE of BWR fuels programmed for dissolution increased and it was decided that more refined modelling of the BU profile was required to ensure that margins were not challenged. This work was carried out on behalf of BNFl by Serco Assurance [2,3] These calculations took advantage of the development of a 3D depletion capability in the MONK code [4] to derive a more detailed representation of axial BU variation (taking into account both absolute BU level and the effect of changes in neutron spectrum due to moderator voids). The fuel element design used in theses studies was based on the OECD-NEA BUC Benchmark Phase III [5], with nominal fresh fuel compositions.

A schematic of the fuel is shown below.

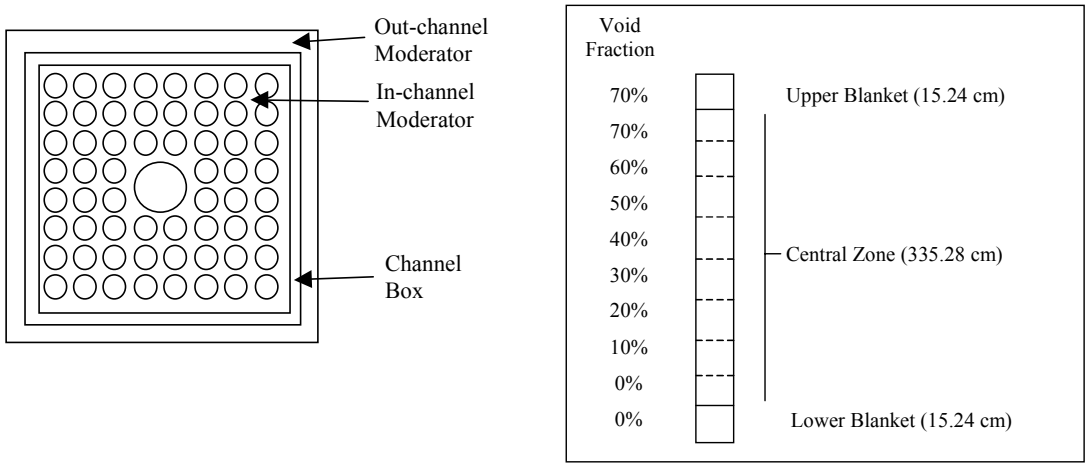

Fig. 3. Schematic of nominal BWR fuel element based on OECD-NEA BUC Phase III benchmark.

The central zone was modelled at an IE of ~3.75w/o with natural uranium in the blankets. These were chosen to represent reasonable upper bound on expected fuel design and to give axially averaged IE of 3.5w/o which provided a useful additional point of comparison with the original PWR calculations. Other features of the model include:

- Central Zone divided into 8 sub-zones
- axial blankets modelled as single zone
- radial pin map modelled explicitly
- infinite array of identical elements in the X/Y plane
- power fixed at 30MW/teHM
- no control rods
- MONK with 172 group JEF2.2 nuclear data library used for depletion calculation.

Following the depletion calculation actinide-only fuel compositions for each sub-zone were passed to a WIMS model of fuel in the THORP Dissolver. In this model the fuel is represented as a series of layers of fuel fragments corresponding to a 'stacking' process as the sheared fuel is fed into the Dissolver. A schematic is shown below.

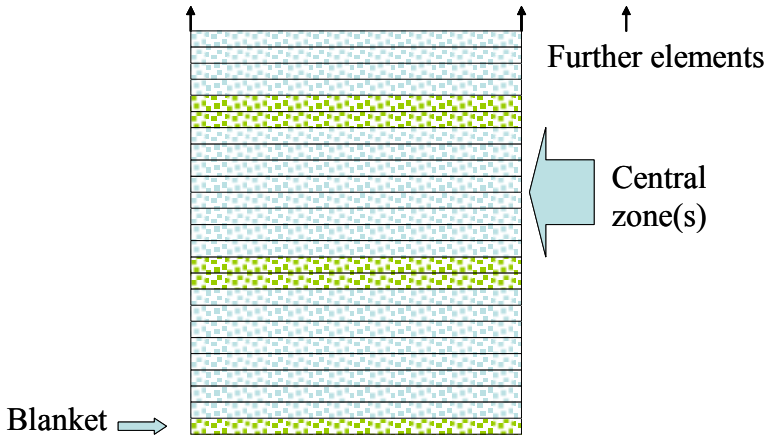

Fig. 4. Schematic of Refined Model for BWR Fuel in Dissolver (side view).

Figure 5 shows a plot of calculated Nd148 inventory as a function of axial zone and for various levels of average element burnup. This provides a useful indication of the level of burnup as a function of axial position.

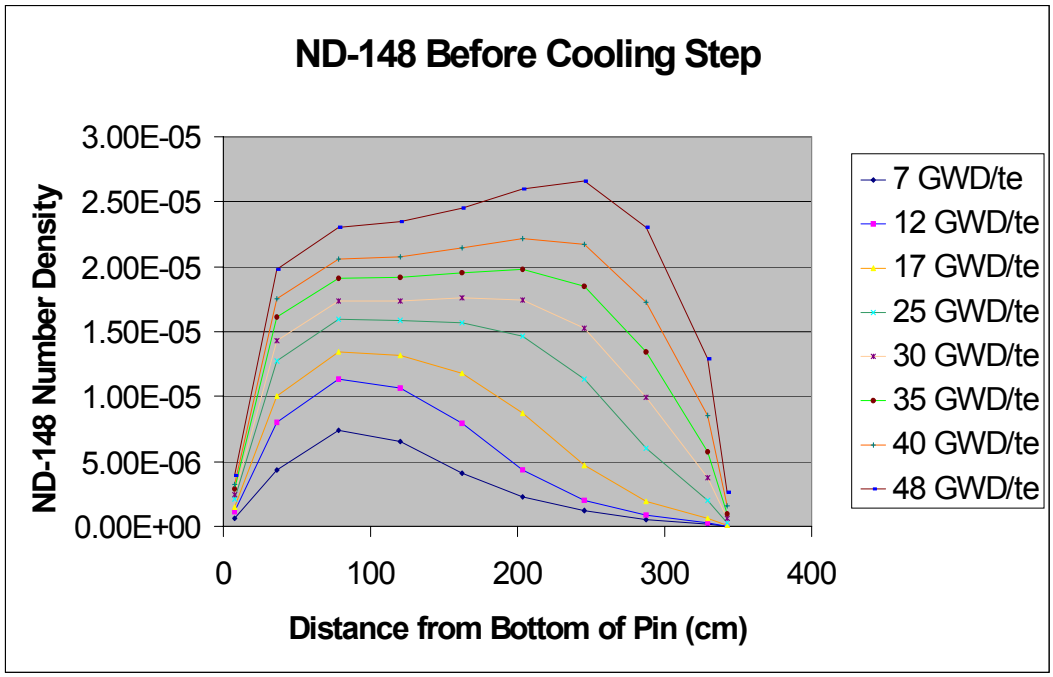

Figure 5. MONK prediction of Nd148 inventory as a function of axial position and for various levels of average element burnup.

Generally it is seen that the axial distribution of BU is in reasonable agreement with expectation, except at the early part of the calculation where BU is much lower than the average BU in an operating BWR. At this level the profile is artificially weighted towards the bottom of the fuel,

because the calculation is taking no account of neighbouring elements having higher BU, (as would be the case in a real reactor). As BU increases to a level more akin to 'typical' core average values the profile begins to adopt a more reasonable form.

Comparison of the WIMS k-eff results for the Dissolver with the original analysis for worst-case PWR fuel showed that the original loading curve still remained bounding (□k ~1-2%).

3.2 Effect of burnable poisons in BWR fuel

Many BWR fuel designs with higher IE include Gd as a fixed burnable poison to reduce reactivity loss during irradiation and thereby extend the overall BU which may be achieved before discharge. The presence of the poisons tends to harden spectrum during irradiation and can lead to increased Pu production (relative to un-poisoned elements at the same BU level).

The original PWR calculations assumed fixed/high soluble boron poison designed to give conservative discharge levels of Pu. Increases in Gd poisons in later BWR fuel programmed for THORP prompted additional analysis to assess whether the spectrum hardening effect was significant.

In this analysis CASMO was used to calculate Actinide-only composition without Gd, for two Gd loadings representative of real fuel BWR design in the THORP programme. The reactivity effect of changes to the Pu composition in spent fuel was determined by combining the CASMO depletion results with WIMS reactivity sensitivity coefficients for the Dissolver, (these were already available from original analysis).

The results are shown in the Table below.

	Change in k-eff in Dissolver		
Nuclide	20GWd/t	30GWd/t	40GWd/t
U235	-0.001382	-0.001662	-0.001773
U238	-0.000025	-0.000023	-0.000019
Pu239	-0.001454	-0.001030	-0.000924
Pu240	0.000801	0.000499	0.000298
Pu241	-0.000379	-0.000500	-0.000497
Pu242	0.000025	0.000029	0.000025
Total	-0.002414	-0.002688	-0.002890

These correspond to an element containing 12 poisoned pins at 6w/o Gd. From this analysis it was seen that the effect was not large for BWR fuel designs programmed in THORP and none of programmed elements challenged the original loading curve, i.e. the original PWR analysis remained bounding.

4. Conclusion

Following implementation of the original BUC criticality safety case in THORP additional analyses have been made for various fuel campaigns not covered in the original assessment. In particular modelling of BWR fuel has been refined to take better account of the effects of axial variations (blankets, moderator voids) and fixed burnable poisons.

The additional calculations confirmed that the original assessment, which was based on analysis of a 'worst-case' PWR fuel design remains bounding.

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Criticality safety analysis of WWER spent fuel casks with radial burnup profile implementation

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Abstract. The aim of this paper is to evaluate the impact of radial burnup profile on the criticality of WWER spent fuel casks. The radial zone wise dependent spent fuel inventories of two WWER assemblies, discharged from Unit 1 and Unit 5 of Kozloduy NPP respectively, have been calculated by the NESSEL-NUKO code system. For criticality calculations the modular code system SCALE4.4 has been applied. Calculations have been performed for both: cask with 30 WWER-440 fuel assemblies with initial enrichment 3.6% of ^{235}U and burnup up to 40MWd/kgU and cask with 12 WWER-1000 fuel assemblies with initial enrichment 3.3% of ^{235} U and burnup up to 40MWd/kgU. The results obtained show that the influence of radial burnup credit on Keff of the considered transport casks is very small, which is in good agreement with the published results for WWER-440 and for PWR fuel.

1. Introduction

In this paper the results of criticality analysis of two types transport casks: for WWER-440 and for WWER-1000 assemblies with radial burnup profiles are presented.

The real geometry of assembly (e.g. presence of absorber rods, periphery areas, asymptotic areas etc.) is accounted for accordingly in the neutron spectrum and cross section calculations. That results in determination of the isotope inventory in different radial zones of WWER-440 and WWER-1000 spent fuel assemblies. The calculated by zones isotope inventory has been applied in criticality safety analysis of the spent fuel transport casks.

2. Calculation methodology

The calculation methodology [1] is based on the two worldwide well known and used code systems for depletion and criticality calculations:

- NESSEL-NUKO — for depletion calculations with spatial dependence of isotope inventory [2,3];
- SCALE-4.4 — for criticality calculations [4].

2.1. NESSEL-NUKO code system

The spectral and burnup calculations have been performed by the NESSEL code [2] especially designed for neutronics calculations of WWER type of reactors. It uses data library containing microscopic cross sections for more than 200 isotopes based on the evaluated nuclear data files ENDF/B-IV/V. NESSEL calculates effective few group diffusion parameters and depletion for given radial zone accounting for the neutron spectrum in this zone.

The spatial dependence is considered by the so called method of “step by step” homogenization. In this method the real assembly is represented by a set of cylindrical cells of Wigner-Zeitz, unified with

respect to their properties in “homogeneous area”. Each zone contains either “elementary zones” (primary material zones) or already homogenized with respect to flux and volume zones of lower rank.

The NESSEL code is accounting for the real geometry of the fuel assembly (e.g. presence of absorber rods, edge fuel pin cells, asymptotic areas etc.) in the neutron spectrum and isotope concentration calculation. NESSEL calculates all nuclides with considerable influence on the neutron spectrum explicitly (U and Pu isotopes and saturation fission predicts Xe and Sm) in both different radial zones and entire WWER-1000 and WWER-440 assembly.

The NUKO code [3] calculates the concentrations of actinides and fission products important for practice with medium decay period.

2.2. SCALE- 4.4 modular code system

The SCALE-4.4 modular code system [4] is verified and world-widely used for criticality safety analyses of PWR spent fuel storage facilities. The system has been recently in process of international testing for WWER applications [5, 6, 7]. It was verified also at the INRNE, BAS for analyses of WWER spent fuel storage and transportation facilities [8]. The analytical sequence CSAS6 has been applied for the criticality calculations. It includes the modules BONAMI, NITAWL-II and XSDRNPM for neutron data preparation, as well as the 3D multi-group Monte Carlo criticality code KENO-VI. The 44-group neutron data library 44GROUPNDFB5 based on evaluated data file ENDF/B-V has been used.

3. Results

3.1. Depletion calculations

The considered WWER-440 fuel assembly with initial enrichment 3.6% was from Unit 1 of Kozloduy NPP, operated 4 years from cycle 17, when it has been inserted fresh, to cycle 20, when it has been discharged from the core with burnup reached 40.79 MWd/kgU [9].

The analyzed WWER-1000 fuel assembly with initial enrichment 3.3% has been operated at the Kozloduy NPP Unit 5 for 3 years from cycle 4, when it has been inserted fresh, to cycle 6, when it was discharged. The burnup level reached after its discharging from the core was 36.26 MWd/kgU [9,10].

The isotopes included in NESSEL-NUKO depletion calculations are as in Ref. [5]: 12 major and minor actinides (U-235, U-234, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-243, Np-237) and 15 fission products (Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Nd-143, Nd-145, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Eu-153 and Gd-155).

The geometry data and material content of each assembly have been modeled corresponding to the homogenization procedure of the NESSEL code, generating the isotope concentrations, the fission and capture microscopic cross sections and the 34-group neutron spectrum in dependence on fuel burnup. The real operational power history and the outages for reloading of the given assemblies are described in Ref [9, 10].

The isotope concentrations have been determined by the NESSEL-NUKO code system in two radial zones according to the pin cell location inside WWER-440 fuel assembly and in three radial zones, in accordance with the pin cell location inside WWER-1000 assembly. The WWER-440 and WWER-1000 assembly with different fuel pin cell types located in different radial zones are shown in Figures 1 and 2.

The WWER-440 spent fuel assembly inventory has been calculated up to average burnup of 40.79 MWd/kgU by step 5 MWd/kgU. The inventory of the WWER-1000 spent fuel assembly has been calculated up to average burnup of 36.26 MWd/kgU by step 3 MWd/kgU.

Detailed results for deviations between zone wise and assembly average burnup and isotope concentrations for WWER-1000 spent fuel assembly are given in Ref. [10]. Maximal deviation in burnup is in the periphery zone (about 20%).

The spatial change of isotope concentrations is illustrated graphically in Figures 3–6. In Figures 3–4 the concentrations of ^{235}U and ^{239}Pu for two radial zones (asymptotic and periphery) and for assembly average burnup for WWER-440 spent fuel assembly are presented. In Figures 5–6 the concentrations of ^{235}U and ^{239}Pu for three radial zones (cluster, asymptotic and periphery) and for assembly average burnup for WWER-1000 spent fuel assembly are given.

3.2. Criticality calculations

The effective multiplication factor Keff of two transport casks has been calculated by the SCALE4.4 (control module CSAS6 with KENOVI) in two cases: radial burnup profile ($Keff_{prof}$) and flat burnup ($Keff_{flat}$). One of the casks contains 30 WWER-440 spent fuel assemblies (see Fig.7), irradiated up to burnup level of 40.76 MWd/kgU. The other one is with 12 WWER-1000 spent fuel assemblies (see Fig.8), irradiated up to burnup level of 36.26 MWd/kgU. The both assemblies have been described above.

The calculation results for

$\Delta Keff = Keff_{prof} - Keff_{flat}$

for WWER-440 and WWER-1000 spent fuel casks are given in Tables 1 and 2, respectively. They show that the influence of the radial burnup profile on the criticality is very small. The difference is within the statistical error for all of the cases. For WWER-440 and WWER-1000 spent fuel cask $\Delta Keff$ is in interval 1-4σ, σ - Monte Carlo standard deviation.

These results are in good agreement with the published results for WWER-440 and for western PWR [12, 13].

4. Conclusions

On the basis of the obtained results it could be concluded:

- The influence of the radial burnup profile on the multiplication factor of WWER-440 and WWER-1000 spent fuel casks is very small. The difference between Keff for radial burnup profile and Keff for flat burnup is within the statistical error.

- The calculated results are in good agreement with the results published for WWER-440 and PWR spent fuel facilities.

- Further investigations should be carried out for new advanced WWER-1000 fuel with Gd absorbers and for 4.4% profiled WWER-1000 assembly. It could be expected the analogous results but the influence of radial burnup credit should be evaluated also in these cases, important for future WWER-1000 spent fuel management.

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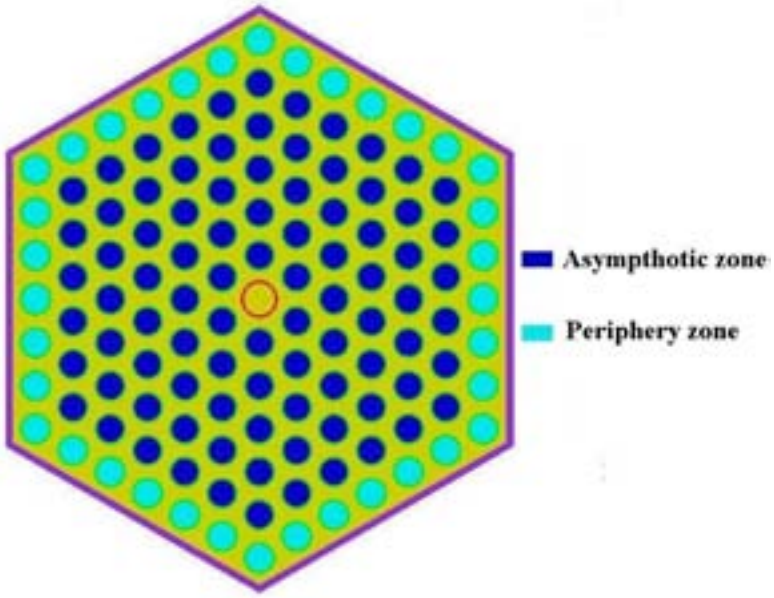

FIG. 1. WWER-440 spent fuel assembly (x-y plane). Visualization by KENO VI.

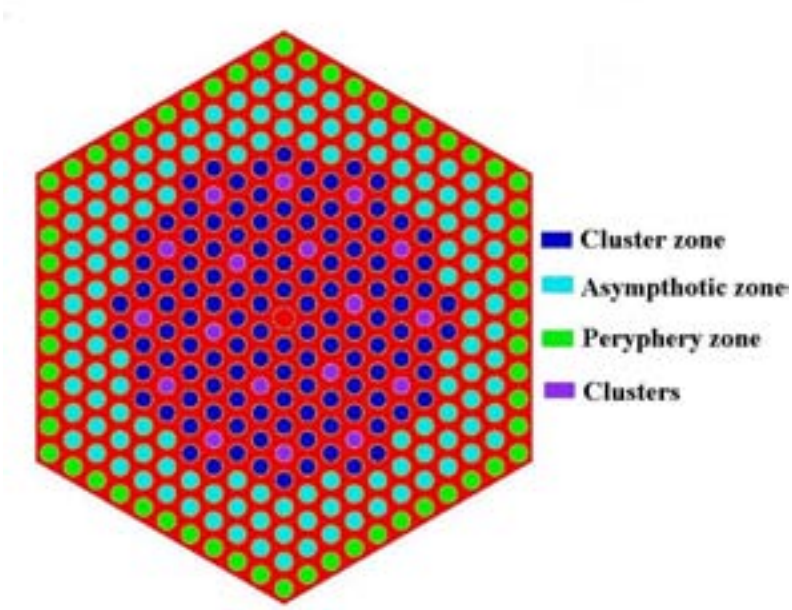

FIG. 2. WWER-1000 spent fuel assembly (x-y plane). Visualization by KENO VI.

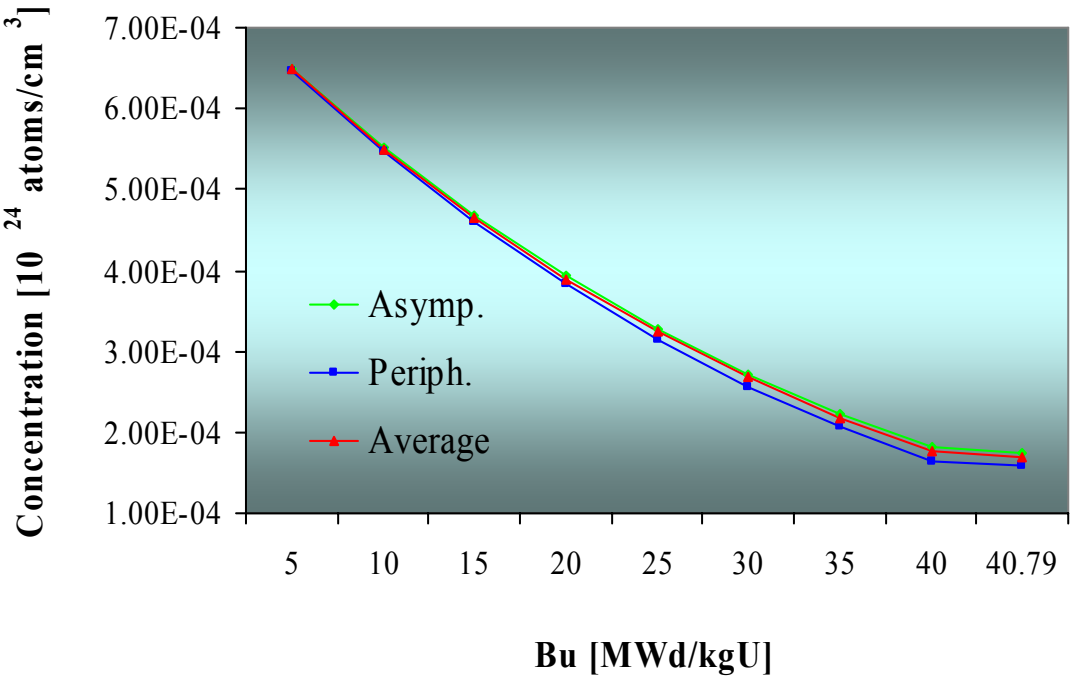

FIG. 3. Concentration of ^{235}U, WWER-440 spent fuel assembly.

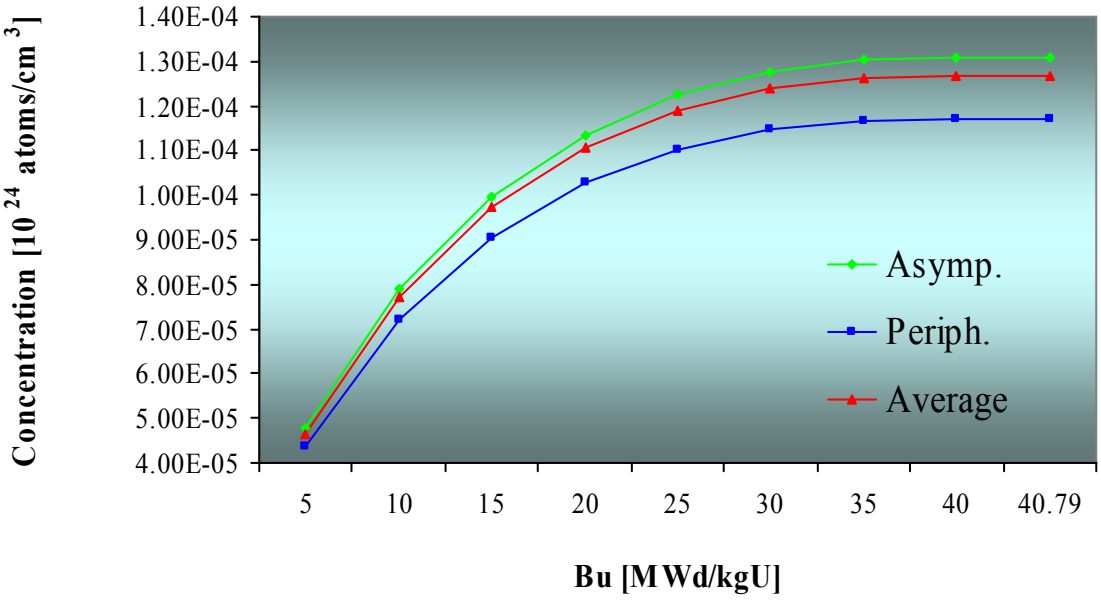

FIG. 4. Concentration of ^{239}Pu, WWER-440 spent fuel assembly.

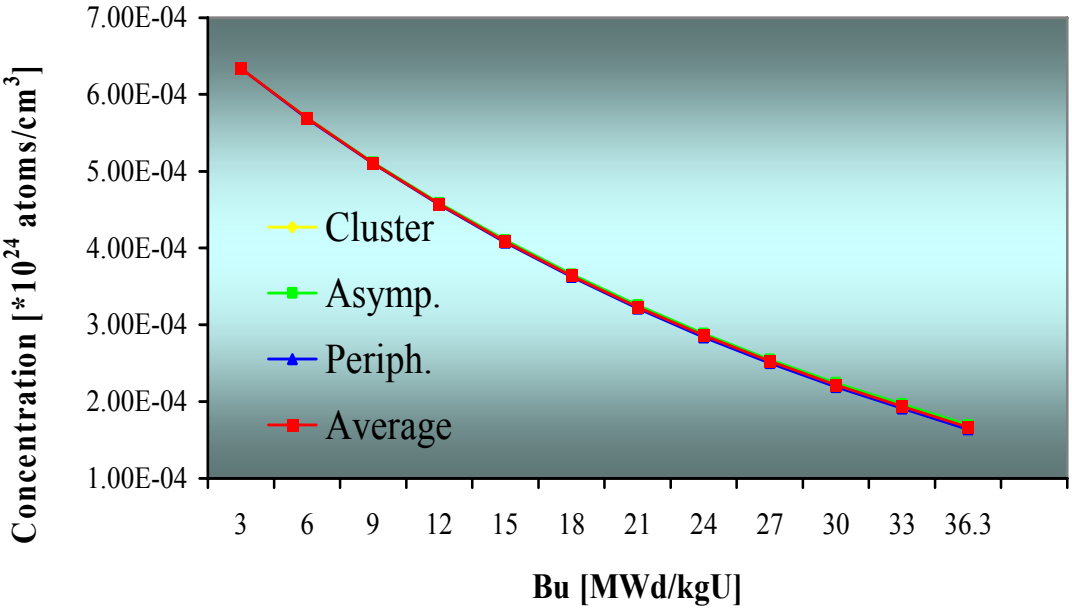

FIG. 5. Concentration of ^{235}U, WWER -1000 spent fuel assembly.

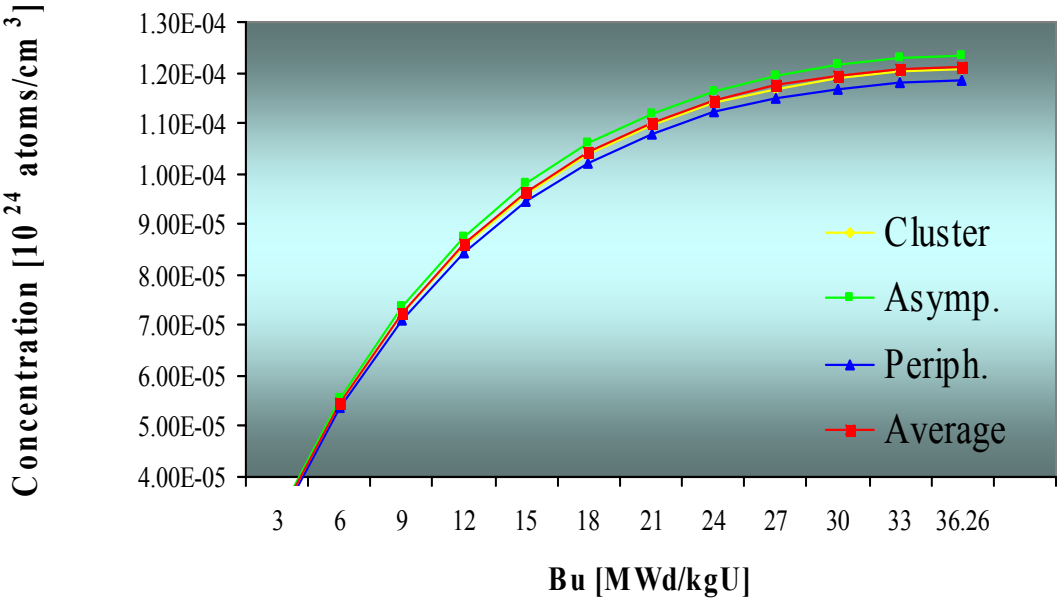

FIG. 6. Concentration of ^{239}Pu, WWER-1000 spent fuel assembly.

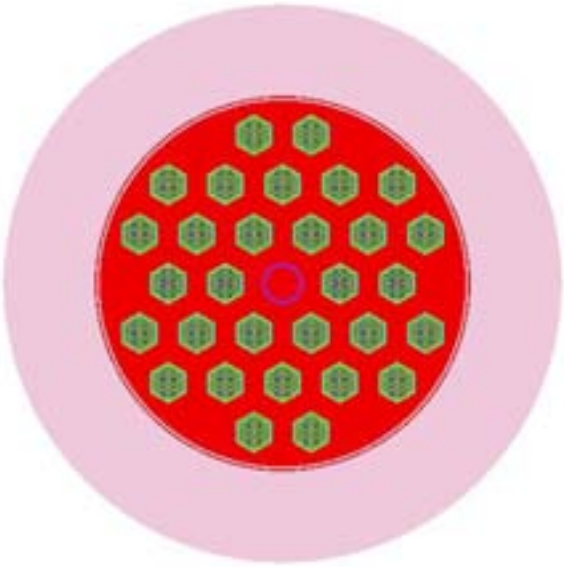

FIG. 7. WWER-440 spent fuel cask with 30 assemblies.Visualization by KENO VI (x-y plane).

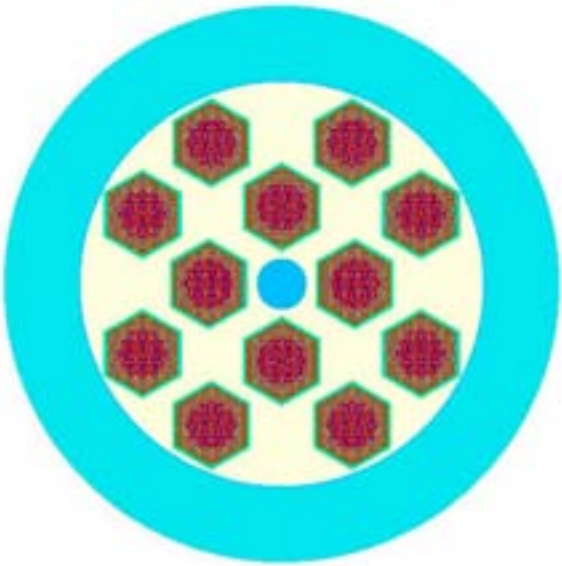

FIG. 8. WWER-1000 spent fuel cask with 12 assemblies. Visualization by KENO VI (x-y plane).

TABLE 1. CALCULATED $K_{eff\ values}$ FOR TRANSPORT CASK WITH 30 WWER-440 SPENT FUEL ASSEMBLIES, IRRADIATED UP TO BURNUP 40.79 MWd/kgU

Average Burnup [MWd/kgU]	profile		flat		ΔK_{eff}=Keff $_{prof}$-Keff $_{flat}$
	Keff$_{prof}$	$\pm\sigma_{MonteCarlo}$	Keff $_{flat}$	$\pm\sigma_{MonteCarlo}$	
0	0.83889	0.00064	0.83889	0.00064	0.
10	0.78693	0.00065	0.78759	0.00078	-0.00066
20	0.74114	0.00071	0.74327	0.00057	-0.00213
30	0.69493	0.00053	0.69704	0.00062	-0.00211
40	0.64956	0.00068	0.65241	0.00066	-0.00285
40.79	0.64543	0.00057	0.64838	0.00071	-0.00295

TABLE 2. CALCULATED K_{eff} VALUES FOR TRANSPORT CASK WITH 12 WWER-1000 SPENT FUEL ASSEMBLIES, IRRADIATED UP TO BURNUP 36.26 MWd/kgU

Average Burnup [MWd/kgU]	profile		flat		ΔK_{eff}=Keff$_{prof}$-Keff$_{flat}$
	Keff$_{prof}$	$\pm\sigma_{MonteCarlo}$	Keff$_{flat}$	$\pm\sigma_{MonteCarlo}$	
0	0.85464	0.00058	0.85464	0.00058	0.0
9	0.78965	0.00062	0.79017	0.00059	-0.00052
21	0.72905	0.00049	0.73135	0.00053	-0.00230
30	0.68657	0.00045	0.68832	0.00049	-0.00175
36.26	0.65877	0.00047	0.66087	0.00051	-0.00210

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Calculation routes to determine burnup credit loading curves

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Abstract. The objective of the paper on hand is to describe the key steps of the calculation routes used for evaluating burnup credit loading curves and to discuss procedures which are adequate to estimate the biases and variances in the calculation routes. In addition, impacts of the formulation of bounding or conservative approaches on the estimates of these biases and variances as well as on the reactivity effects due to the non-uniformity of the burnup distribution within the fuel are discussed.

1. Introduction

The objective of a burnup credit criticality safety analysis of a spent nuclear fuel system (e.g. wet storage system, transport or storage cask, dissolver in a reprocessing plant) usually is to determine a loading criterion the fuel has to meet to be acceptable for loading in the system. This criterion is usually given in form of a curve named as "loading curve" which indicates the minimum burnup necessary (or a related parameter's minimum value necessary or, as the case may be, maximum value allowable) for fuel with a specific initial enrichment to be loaded in the spent fuel system (Figure 1).

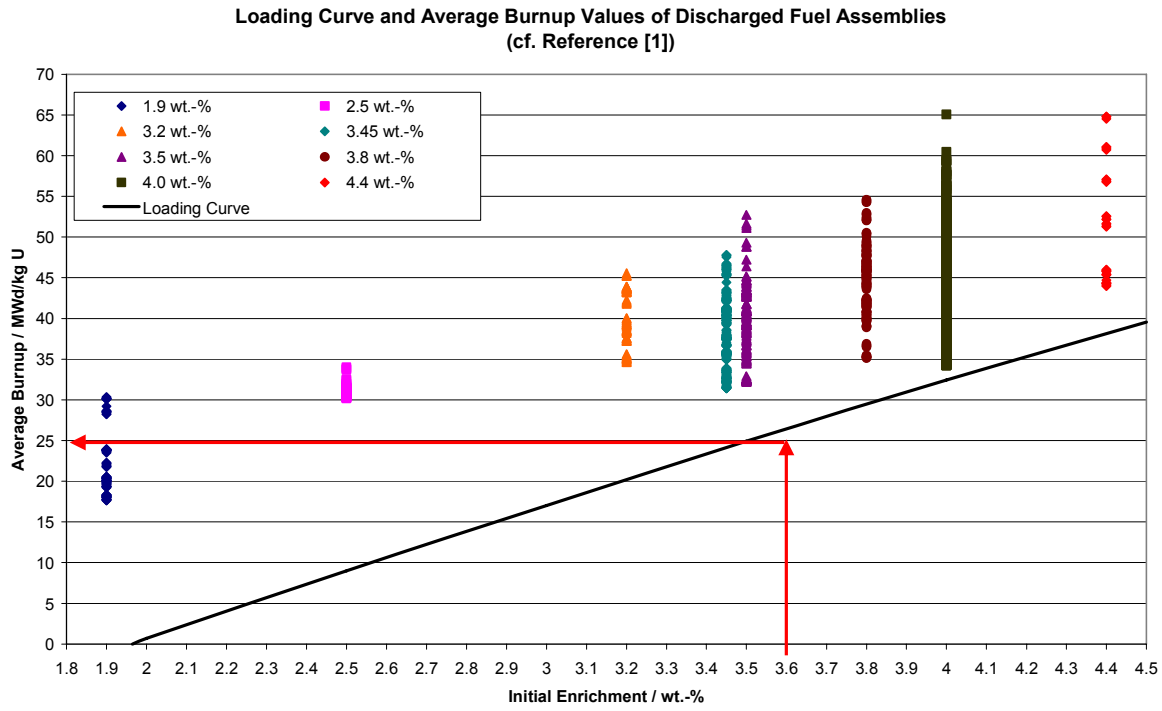

FIG. 1. Example for a Loading Curve (LC) and for average burnup values of spent fuel assemblies to be loaded in the spent fuel system for which the LC is determined.
(LC indicates the minimum burnup required for a fuel assembly with a specific initial enrichment).

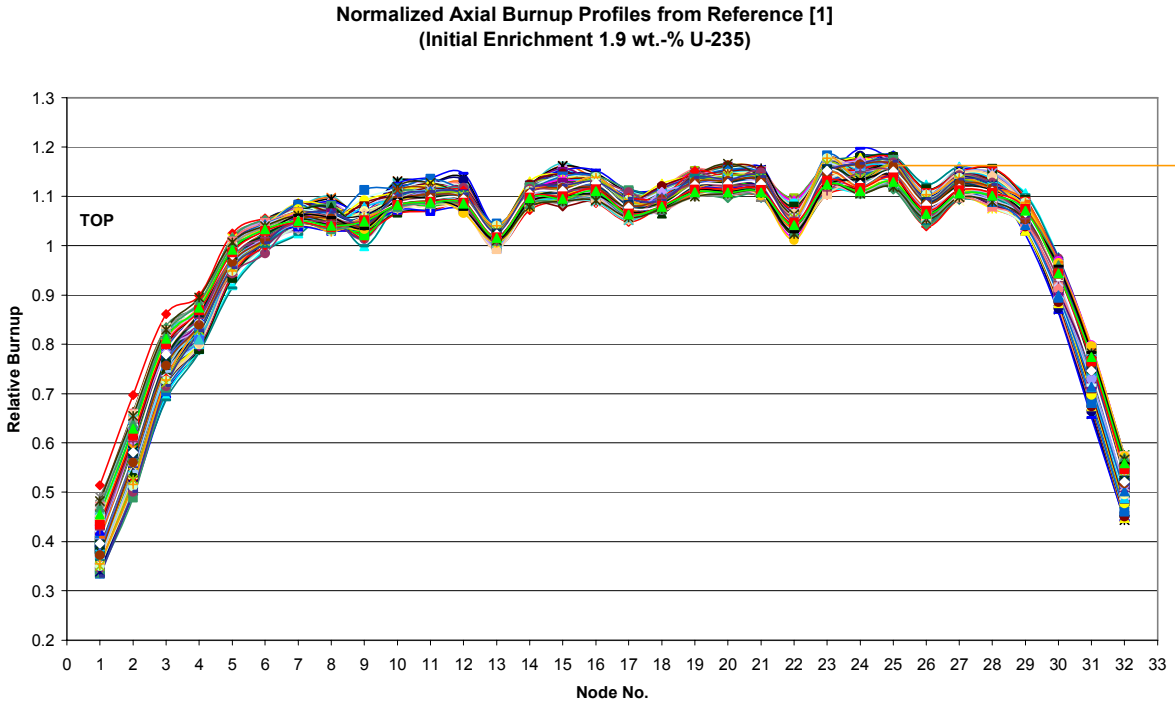

FIG. 2: Examples for axial burnup profiles obtained for fuel assemblies with 1.9 wt.-% initial enrichment.

By definition, a loading curve specifies a unique *average* burnup value (or a corresponding value of a related parameter) for a given initial enrichment. A loading curve accordingly applies to any fuel position of the spent nuclear fuel (SNF) system of interest and does not take credit for any real loading scheme of the system. A loading curve must therefore cover the variety of the irradiation histories to be taken into account as well as the variety of axial and horizontal burnup profiles to be considered (Figures 2 and 3). The task to determine a loading curve thus implies the need for

- looking for a *bounding* irradiation history given by those reactor operation conditions leading, at given initial enrichment and given burnup, to the highest reactivity of the SNF under the conditions of the SNF system of interest,

- generating a *bounding* axial burnup profile, i.e. a model profile which covers, under the conditions of the SNF system of interest, the end effects (i.e. the reactivity effects due to the non-uniformity of the axial burnup distribution) of all the real axial burnup profiles to be taken into account, and

- generating a *bounding* horizontal burnup profile, i.e. a model profile which covers the reactivity effects of all the horizontal burnup profiles to be considered.

Both the shape of the bounding axial burnup profile and the shape of the bounding horizontal profile have to be described as continuous functions of the average burnup in order to get an unambiguous loading curve [2][3][4].

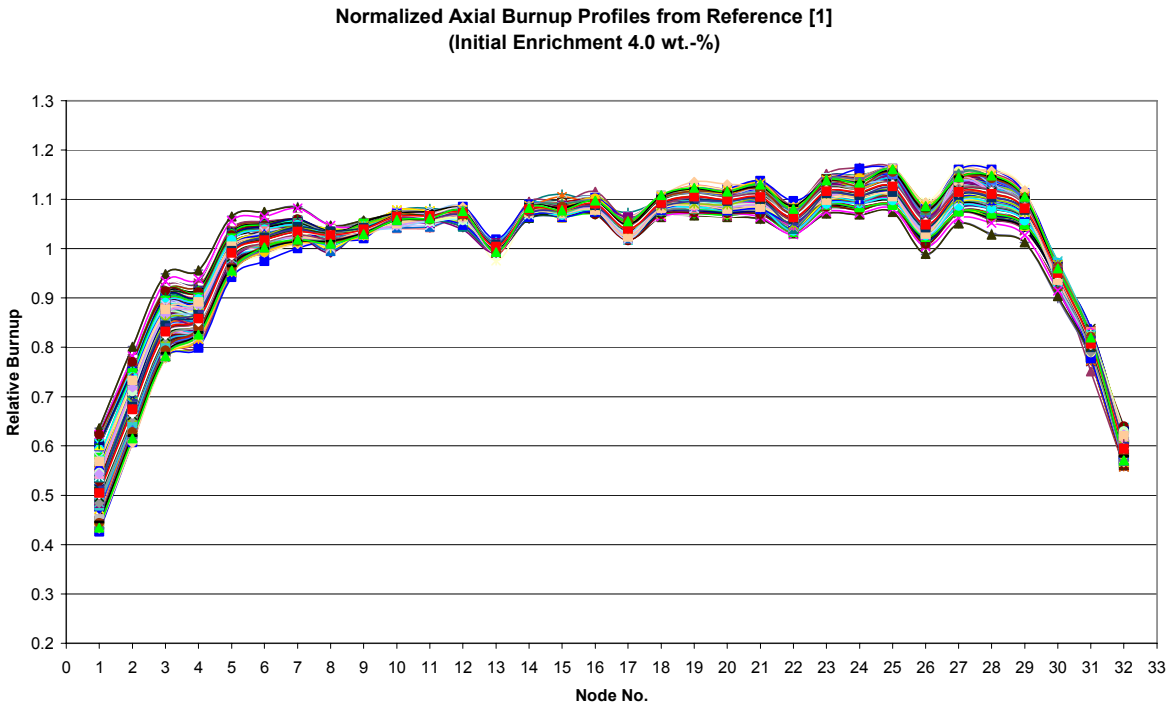

FIG. 3. Examples for axial burnup profiles obtained for fuel assemblies with 4.0 wt.-% initial enrichment.

2. Reactivity equivalence relation

Using a bounding axial burnup profile the shape of which is a continuous function of the average burnup $\overline{B}$ and using a bounding horizontal burnup profile the shape of which is a continuous function of $\overline{B}$ as well the loading curve of the SNF system is given by the reactivity equivalence condition

$$\hat{k}(e,\overline{B}) + \lambda\hat{\sigma}(e,\overline{B}) + \Delta k_U(e,\overline{B}) = (1 - \Delta k_m) \quad (1)$$

where $(1-\Delta k_m)$ represents an adequate upper bound of subcriticality.

$\hat{k}(e,\overline{B})$ is the effective neutron multiplication factor k_{eff} of the SNF system of interest calculated at initial enrichment e and average burnup $\overline{B}$ using bounding shapes for the axial and horizontal distribution of the burnup. $\lambda\hat{\sigma}(e,\overline{B})$ represents either the statistical tolerance (if a statistical calculation code is used for calculating $\hat{k}$) or the numerical error (if a non-statistical calculation procedure is applied) of the calculated value $\hat{k}$ of the neutron multiplication factor k_{eff}.

$\Delta k_U(e,\overline{B})$ in equation (1) is the tolerance of $\hat{k}$ due to the biases and uncertainties in the applied burnup credit calculation route. All burnup credit calculation routes consist in implementation of the following key steps:

- Prediction of the isotopic inventory of the fuel as a function of initial enrichment and burnup, using a bounding irradiation history (i.e. using bounding depletion conditions)
- Determination of the loading criterion (loading curve) based on the estimation of the SNF system's neutron multiplication factor (Figure 4).

- Quantification and verification of the average burnup of the fuel destined to be loaded in the SNF system.

Accordingly, $\Delta k_U\left(e, \overline{B}\right)$ in equation (1) is determined by the following three tolerances:

- $$\Delta k_{DC} = f\left(\left(\Delta k_B\right)_{DC}, \left(\Delta k_{FD}\right)_{DC}\right) \quad (2)$$

  where $(\Delta k_B)_{DC}$ denotes the tolerance in the SNF system's k_{eff} value due to the biases and uncertainties in the predicted isotopic number densities (DC:= depletion calculation),

  $$\left(\Delta k_B\right)_{DC} = g\left(e, \overline{B}\right) \; ; \quad (3)$$

  $(\Delta k_{FD})_{DC}$ reflects the variance of the system's k_{eff} value due to the impact of manufacturing tolerances in the fuel design on the predicted isotopic inventory.

- $$\Delta k_{CC} = f\left(\left(\Delta k_B\right)_{CC}, \left(\Delta k_{FD}\right)_{CC}, \Delta k_S\right) \quad (4)$$

  where $(\Delta k_B)_{CC}$ denotes the tolerance in the SNF system's k_{eff} value arising from the bias in the k_{eff} value due to the criticality calculation code applied (CC:= criticality calculation); $(\Delta k_{FD})_{CC}$ and Δk_S reflect the variances of the SNF system's k_{eff} value due to manufacturing tolerances in the fuel design and the design of the SNF system, respectively.

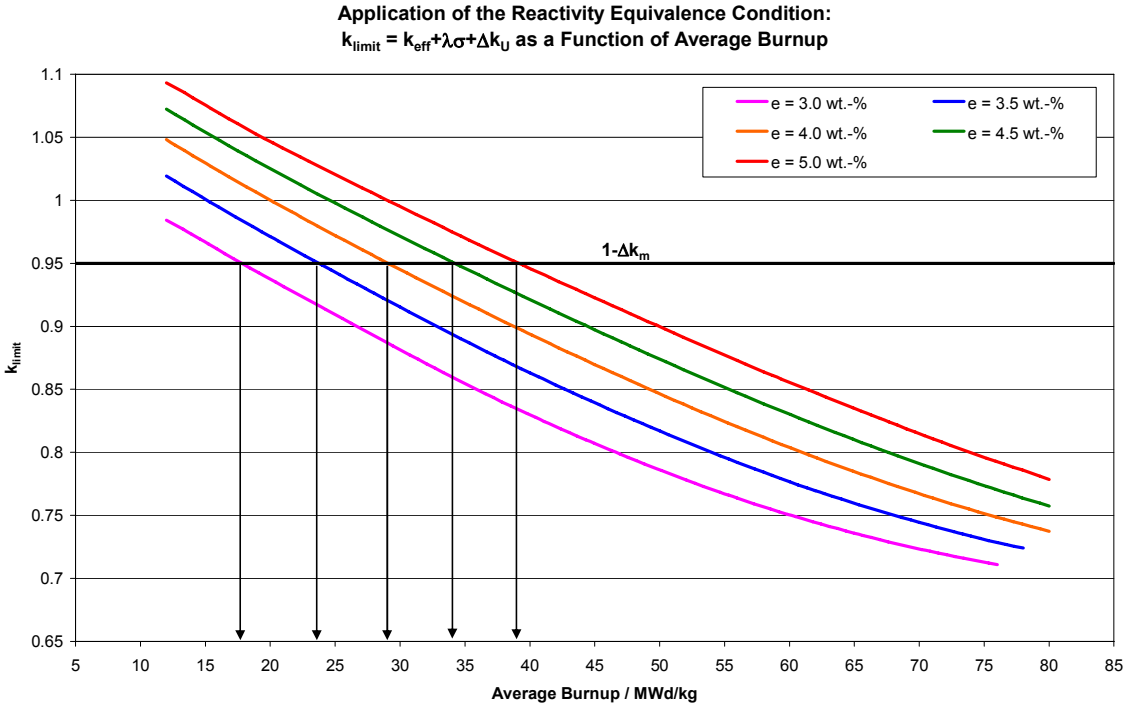

FIG 4. Illustration of the determination of a loading curve.

- $$\Delta k_V = f\left(\Delta B\right) \quad (5)$$

  reflecting the variance of the SNF system's k_{eff} value due to the bias and variance in the quantification of the average burnup of the fuel.

Usually the term Δk_V is not included in the term $\Delta k_U\left(e,\overline{B}\right)$ of eq.(1) since it is quite impractical to make the outcome of the criticality calculation dependent on a nuclear-power-plant-specific procedure used to quantify the average burnup. Therefore, the expression

$$\Delta k_U\left(e,\overline{B}\right) = \Delta k_{DC} + \Delta k_{CC} \tag{6}$$

is usually used for the term $\Delta k_U\left(e,\overline{B}\right)$ in eq.(1).

The term Δk_V got a mention here because the bias and the variance has to be considered in form of an increment $\Delta B = \Delta B\left(e,\overline{B}\right)$ to the average burnup value specified by the loading curve obtained by applying eq.(6) in eq.(1). Such an increment can however be added later on.

3. Effects of the depletion conditions on results and validation of the calculation routes

The isotopic composition of spent fuel, required as input to burnup credit criticality safety analysis, is predicted by means of depletion calculations. To perform a depletion analysis

- the fuel design characteristics,
- the fuel depletion conditions (reactor operation conditions), and
- the cooling time

have to be specified.

The depletion conditions for PWR or BWR UOX fuel are characterized by the following parameters:

- Specific power and operating history
- Fuel temperature
- Moderator temperature and density
- Presence of soluble boron in the core (PWR)
- Core environment (e.g. presence of MOX fuel in the core)
- Use of fixed neutron absorbers (control rods, burnable poison rods, axial power shaping rods)
- Use of integral burnable absorbers in form of gadolinium (Gd) or erbium (Er) bearing fuel rods or IFBA rods (rods containing pellets with burnable absorber coating, e.g. B-10 coating).

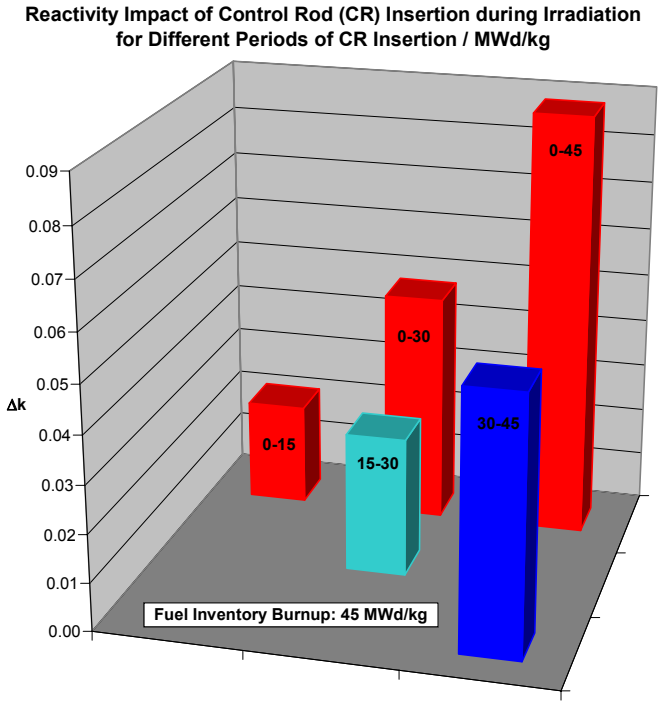

FIG. 5. Example for the Increase Δk in the neutron multiplication factor due to the use of fixed neutron absorbers.

It has been already shown at different opportunities ([5], for instance) how the depletion parameters have to be chosen to assure application of a bounding history leading, at given initial enrichment and given burnup, to the highest reactivity of the SNF under the conditions of the SNF system of interest. Apart from specific power and operating history effects all the other parameters are directly related to *neutron spectrum hardening*. Spectrum hardening results in an increased build-up rate of plutonium due to increased neutron capture in U-238 and has therefore the effect of increasing the reactivity of the fuel at shut-down and after that. So therefore, bounding values can be found for the fuel temperature, moderator temperature and density as well as the soluble boron content and the presence of MOX fuel in the core. The presence of fixed neutron absorbers used for reactivity control as well as enhanced fuel utilization has a significant effect on the reactivity of the SNF (Figure 5). Burnable poison rods are usually used in fresh fuel assemblies and are removed after one cycle. Thus, bounding models for the usage of burnable poison rods can be determined. For full and partial insertion of control rods (CR), bounding models can be derived from the reactor operation strategies used, as can be demonstrated by means of sensitivity studies on the impact of the use of control rods on the SNF reactivity.

The effect of integral burnable absorbers on the SNF reactivity depends on their initial poison loading, their position in the fuel assembly, the distribution of the poison within the pellets, and the burnup (Fig. 6). The presence of integral burnable absorbers results in spectrum hardening. However, spectrum hardening does not only lead to an increase in the plutonium build-up, but also to a delay in the burn-out of the integral burnable absorbers, and this delay tends toward a decrease in reactivity. All these effects are impacted by the fuel characteristics.

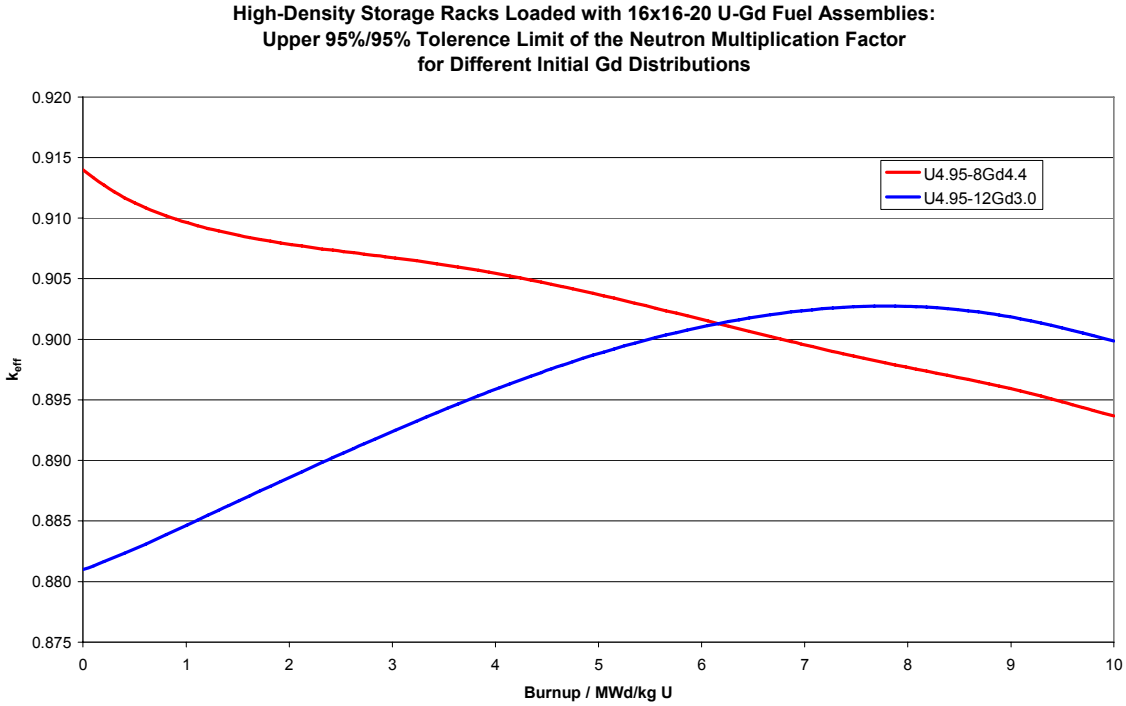

FIG. 6. Examples for the reactivity impact due to the usage of integral burnable absorbers.

Apart from the case of using integral burnable absorbers, which results in a more complex reactivity behavior, it seems to be clear that a bounding or rather conservative irradiation history [6] can be generated by choosing the depletion parameters in such a way that neutron spectrum hardening is maximized. And, in fact, tendencies to proceed in this way are observed in practice: For PWR UOX fuel burnup credit cases, for instance, it has been already observed that the highest fuel temperature and the highest moderator temperature and hence the lowest moderator density are combined with the highest (or at least a very conservatively estimated) soluble boron concentration of the moderator and the assumption that control rods are completely inserted in the fuel assemblies during all the operation cycles. And that's not all; there is additionally assumed that each of the UOX fuel assemblies, for which burnup credit is intended to be taken, is completely surrounded by MOX fuel assemblies inside the core during all the operation cycles. Maximizing spectrum hardening in such a way amounts to buying a ticket for departure from reality. First of all, the combination of the highest fuel and moderator temperature with the highest (or a very high) soluble boron concentration and the case of fully inserted control rods is contradictory to physics. To get a bounding irradiation history it is not necessary to choose depletion parameter combinations which are contradictory to physics. It makes sense

- first, to spend an adequate amount of effort on the study of the specific reactor operation strategies used in the nuclear power plant of interest and
- then, to figure out by means of sensitivity studies a depletion parameter combination suitable for the plant of interest.

But even if such studies are not carried out, it is a priori known that control rods are not completely inserted in a fuel assembly during all the operation cycles. Even if there is not enough information about the CR usage, the assumption, that the control rods are inserted during the last cycle is, as follows from Figure 5, still a conservative assumption, under usual operation conditions at least. In addition, it is questionable whether or not it really makes sense to combine the assumption that control rods are fully inserted in the PWR UOX fuel assemblies of interest with the assumption that each of these assemblies is fully surrounded by MOX fuel assemblies. Insertion of control rods results, as follows from Figure 5, in a significant spectrum hardening. Surrounding the UOX fuel with MOX fuel

results in a significant spectrum hardening as well and leads to a decrease of the control rod efficiency, therefore. So,

- additional sensitivity studies are obviously required for the combination of CR insertion and MOX presence.

And, in addition, with respect to CR usage it has to be taken into account that, at beginning of the fuels' lifetime, MOX fuel is usually less reactive than UOX fuel, whereas the contrary is usually true at the end of the fuels' lifetime.

It should be kept in mind that *the chosen depletion parameter combination has a significant impact on the economical benefit of burnup credit*. For instance, as follows from Figure 5, going on the assumption that control rods are fully inserted during all the operation cycles leads to a significant decrease in the economical benefit of burnup credit, since the significant reactivity increase due to CR usage makes it necessary to use more neutron absorbing material and/or greater distances between the fuel positions inside the SNF system of interest. Both increase in the required amount of neutron absorbing material and decrease in the number of fuel cells per unit volume of the SNF system result in a significant increase in the costs of manufacturing and operating the SNF system.

It has to be noted that *increase in spectrum hardening*, due to CR insertion for instance, *leads to a significant change in the estimated end effect*, i.e., the reactivity effect due to the non-uniformity of the axial burnup distribution. This is due to the fact that, as appears from Figure 7 as well as Reference [5], the increase in the Pu-239 number due to spectrum hardening and hence the correlated increase in the U-235 number density is greater in the center of the fuel assembly, where the burnup is high (Figures 2 and 3), than at the ends of the active zone of the fuel assembly, where the burnup is low (cf. ibid.). In other words, the reactivity importance of the center zone of the fuel assembly increases faster with increasing spectrum hardening than the reactivity importance of the end zones of the fuel assembly. From that it follows immediately, that, if the spectrum hardening is caused by CR insertion, the change in the end effect is significantly dependent on the CR insertion depth assumed in the depletion calculation. As appears from Figures 8 and 9, the end effect first increases with increasing CR insertion depth (since the center zone of the fuel is still assumed to be not exposed to CR insertion), reaches then a maximum (which is dependent on the average burnup), decreases then significantly (since an increasing part of the center zone of the fuel is now assumed to be exposed to CR insertion), and may become, even if the average burnup is significantly higher than the average burnup gained in the first operation cycle, negative at full CR insertion (Figure 8). Therefore, unnecessary maximizing of spectrum hardening results in a significant bias of the estimated end effect.

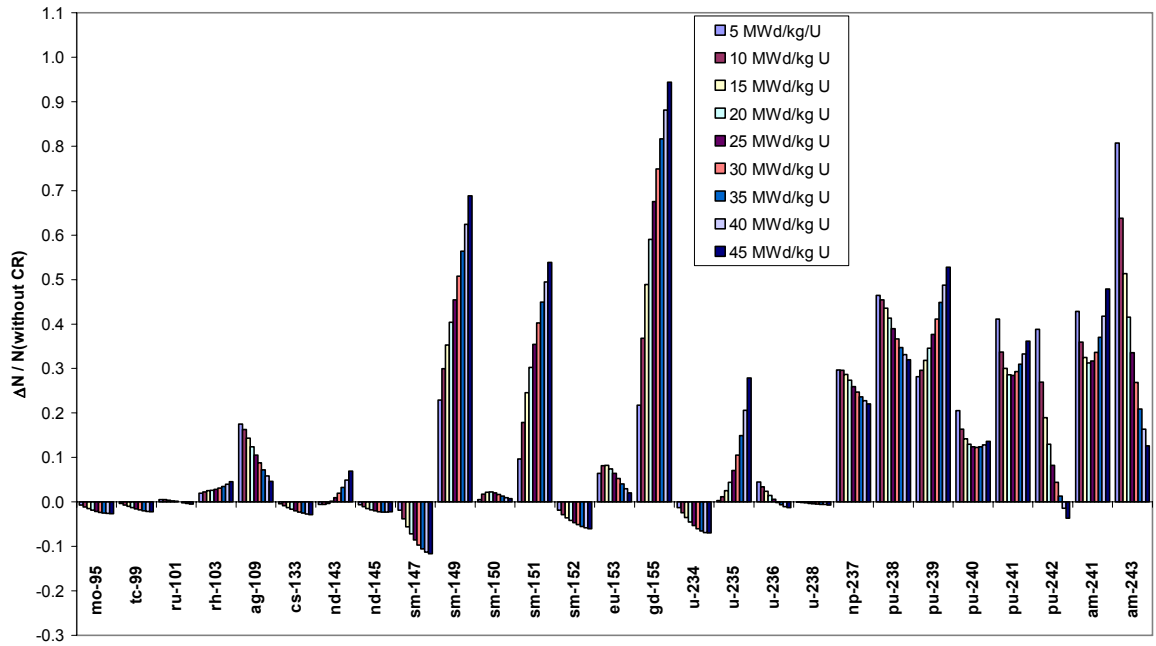

FIG. 7. Relative change of isotopic number densities due to CR insertion (Zero cooling time).

The chosen depletion parameter combination can have a significant impact on the outcome of the validation of the predicted isotopic inventory, i.e. on the term $(\Delta k_B)_{DC}$ of eq.(2). Validation of the predicted isotopic inventory is usually achieved by correcting the predicted isotopic concentrations with the aid of correction factors derived from comparisons with chemical assay data. The correction factors are usually applied to individual isotopes within the predicted isotopic inventory. This approach possibly has a tendency to lead to an overly conservative estimate of the isotopic inventory since the correlations between the isotopic concentrations due to cross sections of different reaction channels (fission, capture, elastic and inelastic scattering, (n, 2n)-reaction, (n, α)-reaction, etc.), fission yields, decay and branching ratios are disregarded (Figure 7 where the correlations between U-236 and Np-237 as well as U-238 and Np-237, between Np-237 and Pu-238, between Pu-241 and Am-241 as well as between Pu-242 and Am-243 are apparent, for instance). In fact in practice isotopic correction factors are often applied in such a way that number densities of fissile nuclides are corrected only then, if and only if underestimated, and number densities of absorbers are corrected only then, if and only if overestimated in comparison to chemical assay data. This way to perform the isotopic validation significantly increases the tendency to lead to overly conservative estimates of the isotopic concentrations. Since chemical assay data usually originate from commercial fuel, it is obvious that unnecessary maximizing of spectrum hardening in conjunction with the application of isotopic correction factors results in a departure from reality.

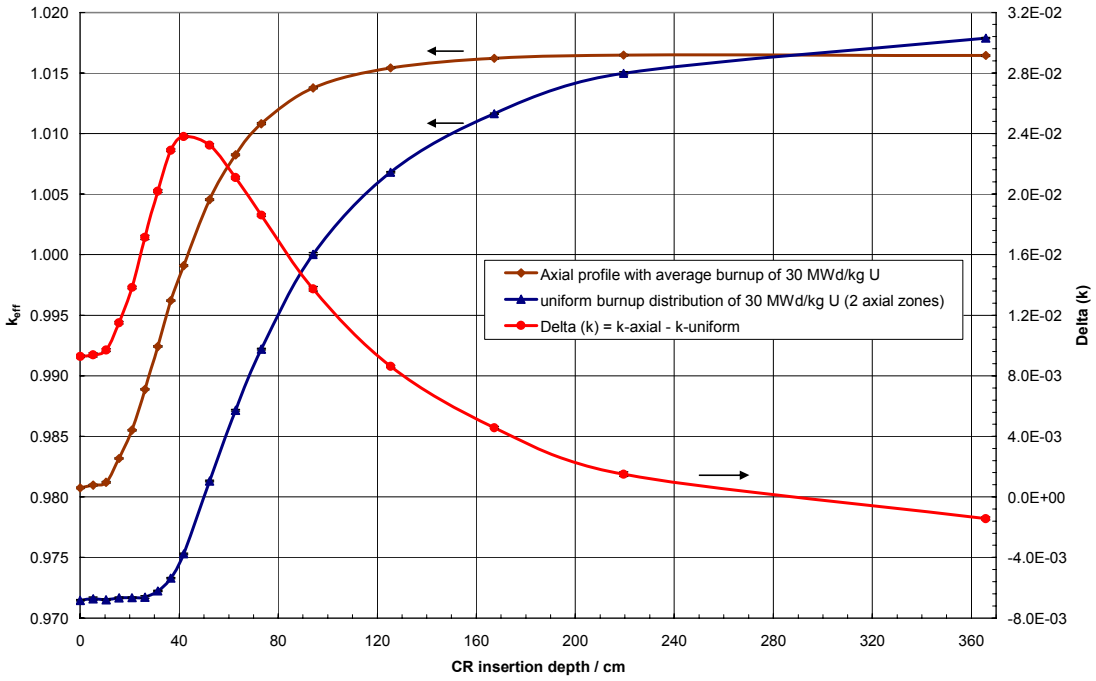

*FIG. 8. Cask loaded with 21 FA of type 17*17-(24+1) (OECD Phase II-E)*
End effect ΔK at 30 MWd/kg U average burnup
as a function of control rod insertion depth during depletion.

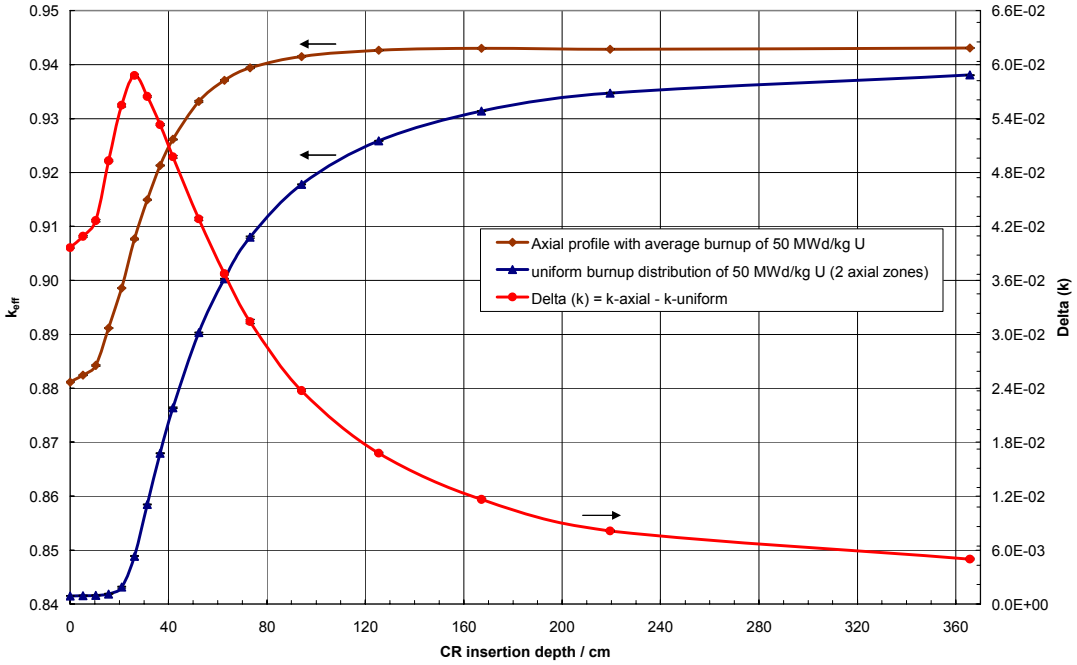

*FIG. 9. Cask loaded with 21 FA of type 17*17-(24+1) (OECD Phase II-E)*
End effect ΔK at 50 MWd/kg U average burnup
as a function of control rod insertion depth during depletion.

The usual application of correction factors can result in an underestimation of the axial end effect since the plutonium concentration and the fission product concentration is higher in the center zone of a fuel assembly than in the region of the ends of the active zone of the fuel assemblies. A bias in the estimated end effect can only be avoided when

- the mean values c_i of the isotopic correction factors are given as functions of initial enrichment e and burnup B

  $$c_i = c_i(e, B) \ ,$$

- the variances $\sigma^2(c_i)$ of the isotopic correction factors are given as function of initial enrichment and burnup,

  $$\sigma^2(c_i) = g_{ii}(e, B) \ ,$$

  and when

- the correlations between the isotopic concentrations are considered in terms of covariances $cov(c_i, c_j)$ of the isotopic correction factors c_i, c_j, given as functions of initial enrichment and burnup,

  $$cov(c_i, c_j) = g_{ij}(e, B) \ .$$

In that case, a confidence interval of the end effect as a function of initial enrichment and burnup can be estimated by means of Monte Carlo sampling from the confidence region in the space of the isotopic correction factors. However, this case is usually not given. Therefore, it is preferable to proceed as follows:

- The neutron multiplication factor of the SNF system of interest is estimated as a function of burnup at different initial enrichments, first by using the predicted isotopic concentrations and then by employing the corrected isotopic number densities obtained by applying the isotopic correction factors derived from comparison with chemical assay data (Fig. 10).

- The resulting differences

  $$\Delta k_{eff}(e, B_{max}) = k_{eff}\left((c_i)_{corrected} \middle| e, B_{max}\right) - k_{eff}\left((c_i)_{predicted} \middle| e, B_{max}\right) \qquad (7)$$

  obtained at the selected initial enrichments and that burnup B_{max}, which covers the highest burnup value of all the axial and horizontal burnup profiles to be considered in the criticality analysis of the SNF system of interest, are fitted with the aid of the linear-least-squares method (cf. Figure 11, where the Δk_{eff} values for the enveloping burnup value B_{max} = 60 MWd/kg U are evaluated).

- The resultant upper 95%/95% tolerance limit of the fit gives the term $(\Delta k_B)_{DC}$ in eq.(2).

Since this term covers the biases and uncertainties in the predicted isotopic number densities for all the initial enrichments and all the burnup values to be considered in the criticality calculations, all these criticality calculations can be performed by using the *predicted* (non-corrected) isotopic concentrations, provided of course that the term $(\Delta k_B)_{DC}$ is explicitly considered, via eq.(6), in the reactivity equivalence condition eq.(1).

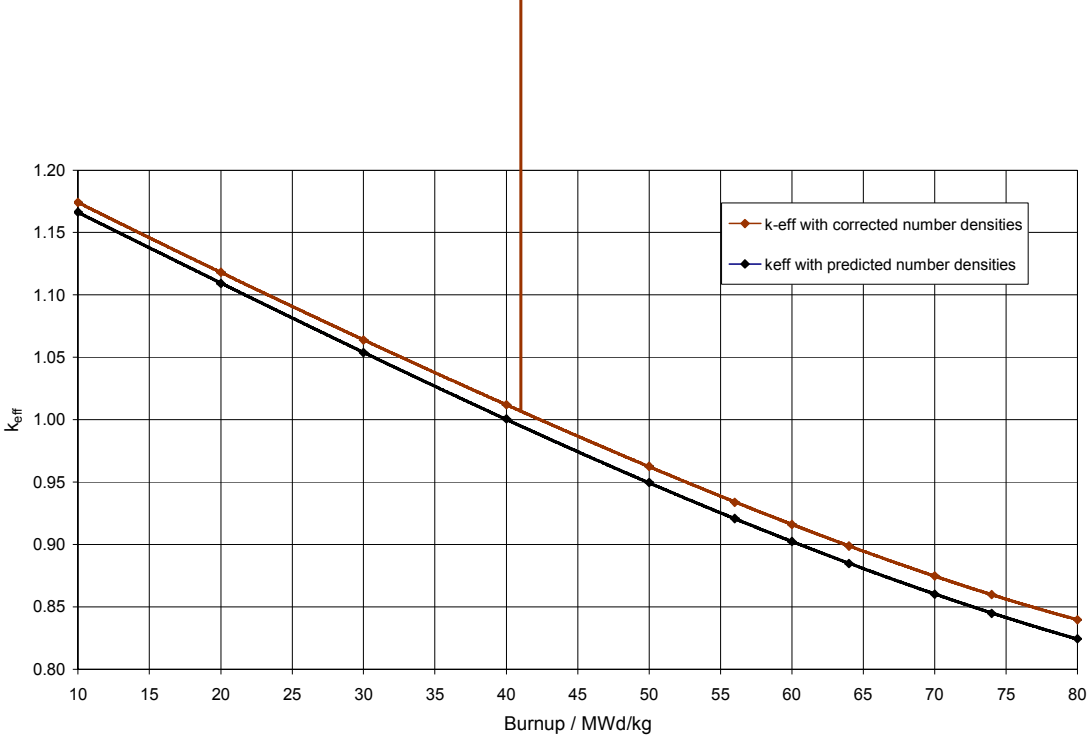

FIG. 10. Isotopic validation: $k_{eff} = f(B \mid e = 5$ wt.-%).

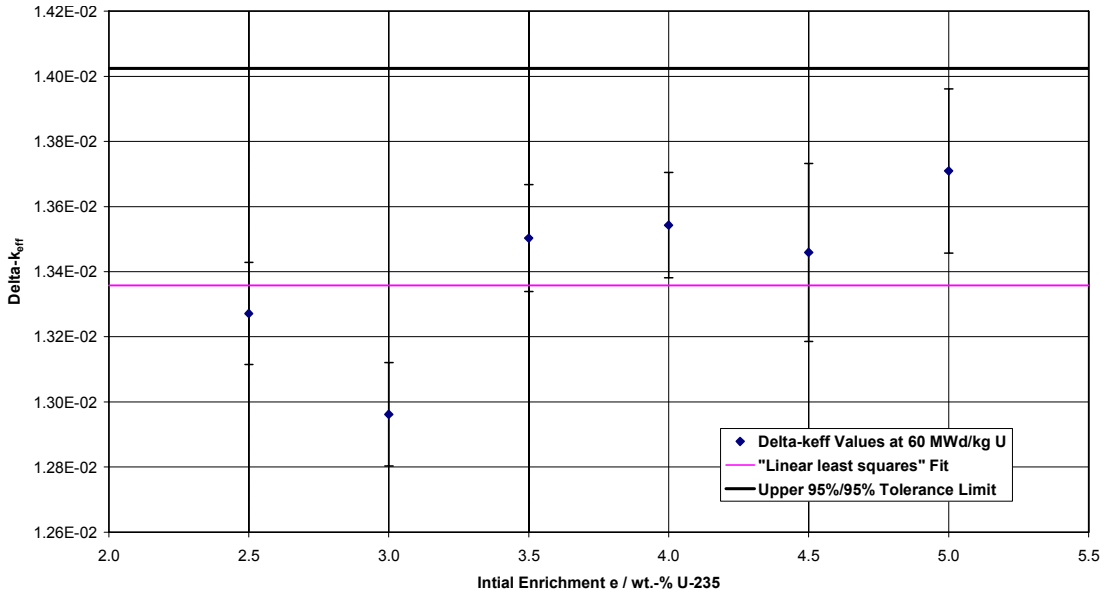

FIG. 11. Isotopic validation: $\Delta k_{eff}(e \mid B_{max}) = f(e)$.

The term $(\Delta k_{FD})_{DC}$ in eq.(2) which reflects the variance of the SNF system's k_{eff} value due to the impact of manufacturing tolerances in the fuel design on the predicted isotopic inventory is usually covered by the use of bounding depletion conditions. In addition, some manufacturing tolerances in the fuel design are of no interest or can be covered in a simple way. In fact, any tolerance in the initial enrichment is of no interest since the loading curve is a function of the initial enrichment. The tolerance in the pellet density can be covered by using the upper tolerance limit of the pellet density for calculating the initial fissile isotope number densities (and, if required, the lower tolerance limit of the pellet density for calculating the initial isotope number densities of integral burnable absorbers). So, in conclusion, by means of the procedure illustrated in Figs 10 and 11 a sufficiently bounding estimate for the term Δk_{DC} in eq.(6) can be obtained.

The chosen depletion parameter combination has a significant impact on the selection of experiments chosen to validate the criticality calculation code applied, i.e., to estimate the term $(\Delta k_B)_{CC}$ in eq.(4). Let's take the cores No.4 and No.5 of the REBUS experimental program [7] as an example. With these cores reactivity measurements were performed using fuel bundles from the German nuclear power plant Neckarwestheim II (GKN II) (core No. 4: fresh UOX fuel; core No. 5: spent UOX fuel). Since

the use of these bundles controls significant amounts of reactivity the evaluation of the measurement results with the aid of direct Monte Carlo methods has been made possible. In fact, besides critical reactor configurations, this experiment is the only one that permits *direct validation of the calculation routes commonly used* in criticality safety analysis (i.e. estimation of k_{eff} rather than reactivity perturbation calculations).

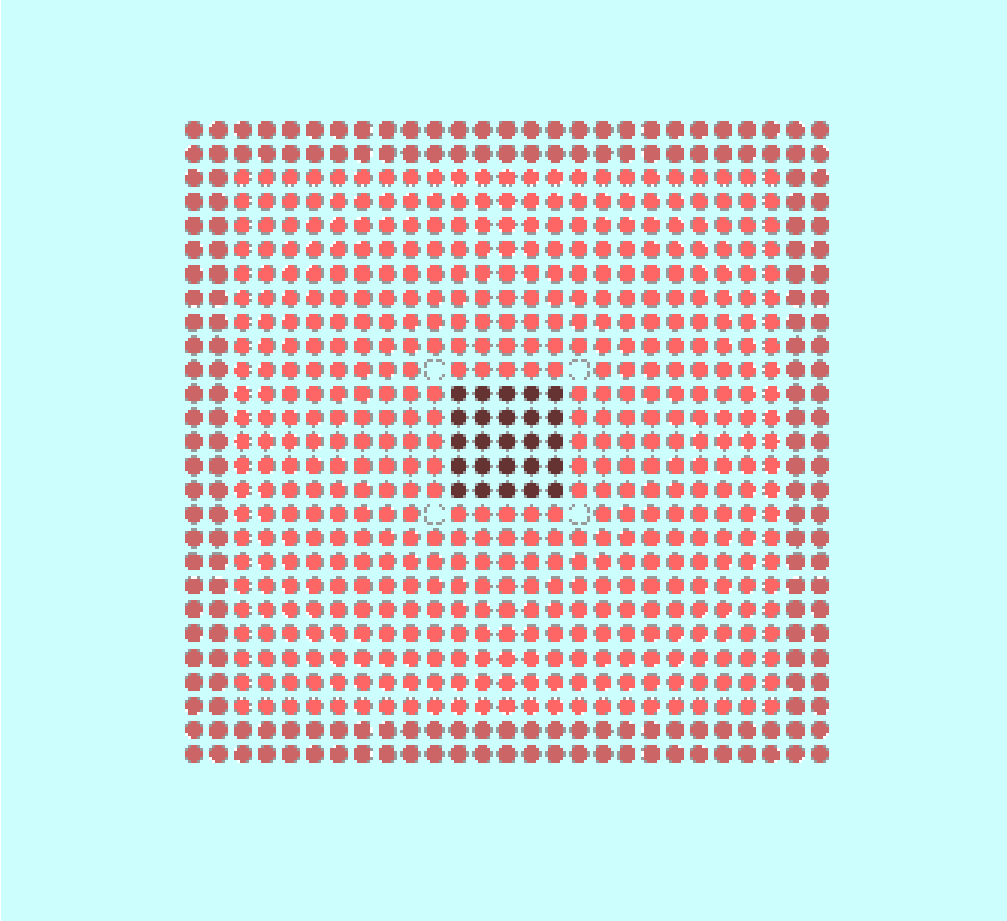

FIG. 12. REBUS Core No. 5 [7].

The REBUS test bundle made up of the GKN II fuel rods consists of 25 fuel rods (Fig. 12). This bundle is placed in a 27x27 driver zone of VENUS UOX fuel rods surrounded by a water reflector (cf. ibid.). Due to the dimensions of test bundle and driver zone it is obvious that the neutron spectrum of the REBUS core No.5 is relatively soft. So, any method used to check the applicability of experimental data which is based on the analysis of the similarity of sensitivity coefficients

$$s_x = \frac{dk_{eff}/k_{eff}}{d\Sigma_x/\Sigma_x}, \quad \Sigma_x := \text{macroscopic cross section for reaction x,} \tag{8}$$

suggests more and more to exclude the REBUS core No.5 from the validation of the criticality calculation procedure, the harder the neutron spectrum of the SNF system of interest becomes.

More general, the harder the neutron spectrum of the application case becomes, the more experimental data from experiments with UOX fuel will be rejected and the more experimental data from experiments with MOX fuel will be accepted. However, it is not exactly a brilliant situation to exclude just that experiment which is the only out-of-commercial-reactor experiment that permits direct validation of the calculation routes commonly used in criticality safety analysis. And unnecessary maximizing of spectrum hardening seems to significantly reduce the number of acceptable experiments.

Once a set of acceptable experiments is found, the term $(\Delta k_B)_{CC}$ in eq.(4) can be determined.

4. Evaluation of the effects of manufacturing tolerances on the neutron multiplication factor

The terms $(\Delta k_{FD})_{CC}$ and Δk_S in eq.(4) can be determined by studying the sensitivity of the k_{eff} value of the SNF system of interest to variations of the parameters $\vec{p} = (p_1,...,p_n)$ describing the criticality-relevant characteristics of the fuel design and the SNF system. The variance $V[k_{eff}(\vec{p})]$ of k_{eff} arising from the variances $V[p_i]$ and covariances $cov(p_i, p_j)$ of the set of parameters $\vec{p} = (p_1,...,p_n)$ is given by the expectation of the function $(k_{eff}(\vec{p}) - E[k_{eff}(\vec{p})])^2$,

$$V[k_{eff}(\vec{p})] = E\left[(k_{eff}(\vec{p}) - E[k_{eff}(\vec{p})])^2\right] \tag{9}$$

where $E[k_{eff}(\vec{p})]$ denotes the expectation value of $k_{eff}(\vec{p})$.

Expanding the function $k_{eff}(\vec{p})$ in a Taylor Series about the expectation values of $\vec{p}$, $E[\vec{p}] \equiv \vec{\mu} = (\mu_1,...,\mu_n)$, makes it possible to calculate the expression (9). Usually the first order Taylor expansion is sufficient,

$$k_{eff}(\vec{p}) \approx k_{eff}(\vec{\mu}) + \sum_{i=1}^{n} (p_i - \mu_i) \left.\frac{\partial k_{eff}(\vec{p})}{\partial p_i}\right|_{\vec{p}=\vec{\mu}} . \tag{10}$$

Only in the case that $k_{eff}(\vec{\mu})$ is in the range of an extreme value of k_{eff}, the second order expansion is required:

$$k_{eff}(\vec{p}) \approx k_{eff}(\vec{\mu}) + \sum_{i=1}^{n} (p_i - \mu_i) \left.\frac{\partial k_{eff}(\vec{p})}{\partial p_i}\right|_{\vec{p}=\vec{\mu}} + \frac{1}{2} \cdot \sum_{i,j=1}^{n} (p_i - \mu_i)(p_j - \mu_j) \left.\frac{\partial^2 k_{eff}(\vec{p})}{\partial p_i \partial p_j}\right|_{\vec{p}=\vec{\mu}} . \tag{11}$$

(In the case that $k_{eff}(\vec{\mu})$ is in the range of a saddle point the third order expansion is required, but such a case is hardly to be expected.)

5. Application of the reactivity equivalence relation and the need for bounding axial and horizontal burnup profiles

With the reflections, given in section 3, on the determination of the terms $(\Delta k_B)_{DC}$ and $(\Delta k_{FD})_{DC}$ in eq.(2) as well as the reflections, given in sections 3 and 4, on the determination of the terms $(\Delta k_B)_{CC}$, $(\Delta k_{FD})_{CC}$ and Δk_S in eq.(4) the contributions Δk_{DC} (cf. eq.(2)) and Δk_{CC} (cf. eq.(4)) to the term Δk_U in eq.(1) are completely known (cf. eq.(6)). The reactivity equivalence condition eq.(1) can therefore be applied now to determine the loading curve, provided that one has a bounding description of the reactivity effects due to the axial and horizontal distribution of the burnup.

A bounding description of the axial end effect can be obtained in two ways as set forth below:

- A sufficiently large set of power-plant-specific axial burnup profiles is selected, and the end effects are determined for each of the profiles separately. From the sample of profile-specific end effects thus obtained a bounding correlation of the end effect to the averaged burnup of the profiles is

generated; i.e. since the end effect due to a burnup profile varies with the shape and the average burnup of the profile [4] one looks for a correlation between the end effect and the average burnup which bounds all the individual profile-specific end effects of the set of selected axial burnup profiles [3].

- The set of the power-plant-specific axial burnup profiles is compiled; this set contains as many profiles as available. From this set a bounding axial burnup profile is generated which bounds, by definition and construction, all the end effects of the set of real axial burnup profiles evaluated [2]. Since the shape of an axial burnup profile is correlated to the average burnup of the profile [4] the shape of a bounding profile is a function of the average burnup [2]. A bounding correlation of the end effect to the average burnup can therefore be determined by means of calculating the end effect of the bounding profile.

The difference between the two procedures described is that in the latter procedure the generation of a bounding correlation between end effect and average burnup is based on an intermediate step, on the generation of a bounding profile in fact, whereas in the former procedure the determination of a bounding correlation is directly based on the end effects of the real shapes. Therefore, the advantage of the former procedure obviously is that it can be seen at a glance that the correlation obtained is really bounding [3], whereas in case of choosing the second method one is left with the problem to demonstrate that the method used for generating the bounding profile results in a profile which is really bounding. However, the disadvantage of the first procedure is that it is a huge amount of calculational work to determine the end effects of all the individual profiles contained in one set of profiles to be evaluated. And, in addition, one is left with the problem to demonstrate that the number of profiles of such a set is really sufficiently large to attain sufficient confidence that the highest end effect occurring due to the reactor operation strategies used is covered by the set of profiles analyzed and hence bounded by the correlation generated. In fact, a 100% confidence can only be achieved when the sets of profiles include all the profiles available.

Therefore, the second method developed in Ref. [2] is reconfirmed in a different paper presented in the Technical Meeting on the “Advances in applications of burnup credit to enhance spent fuel transportation, storage, reprocessing, and disposition”[8].

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Presentation of axial and horizontal burnup profiles

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Abstract. In the paper on hand the derivation of methods is presented which are capable to generate bounding axial burnup profiles as well as bounding horizontal burnup profiles as a function of the average burnup for different fuel designs and different reactor operation conditions. The profiles that can be generated by means of these methods are bounding by construction, irrespective of changes in the isotopic inventory due to changes in the depletion conditions.

1. Introduction

In a burnup credit criticality safety analysis usually the task is given to figure out a decision criterion that provides the average burnup which a fuel assembly must have reached to be acceptable for loading in any position of a spent fuel management system designed for burnup credit. This criterion is usually given in form of a loading curve indicating the minimum average burnup necessary for a fuel assembly with a specific initial enrichment to be loaded in the spent fuel management system. So, for a given initial enrichment a given loading curve specifies a unique average burnup value. Therefore, this value must cover the variety of irradiation histories and the variety of axial and horizontal burnup profiles. The task to determine a loading curve thus implies the need for

- looking for a bounding irradiation history given by those reactor operation conditions leading to the highest reactivity of the spent fuel,
- generating a bounding axial burnup profile the shape of which is a continuous function of the average burnup, and
- generating a bounding horizontal burnup profile the shape of which is a continuous function of the average burnup.

It is the objective of the paper on hand to present the derivation of methods to generate bounding axial burnup profiles as well as bounding horizontal burnup profiles as a function of the average burnup for different fuel designs and different reactor operation conditions. In addition the impact of a change in the isotopic inventory due to a change in the depletion conditions on the axial end effect is analyzed.

2. Description of the method of generating bounding axial burnup profiles as a continuous function of the average burnup

The method of generating a bounding axial burnup profile the shape of which is a continuous function of the average burnup was already described in Ref. [1]. This method is based

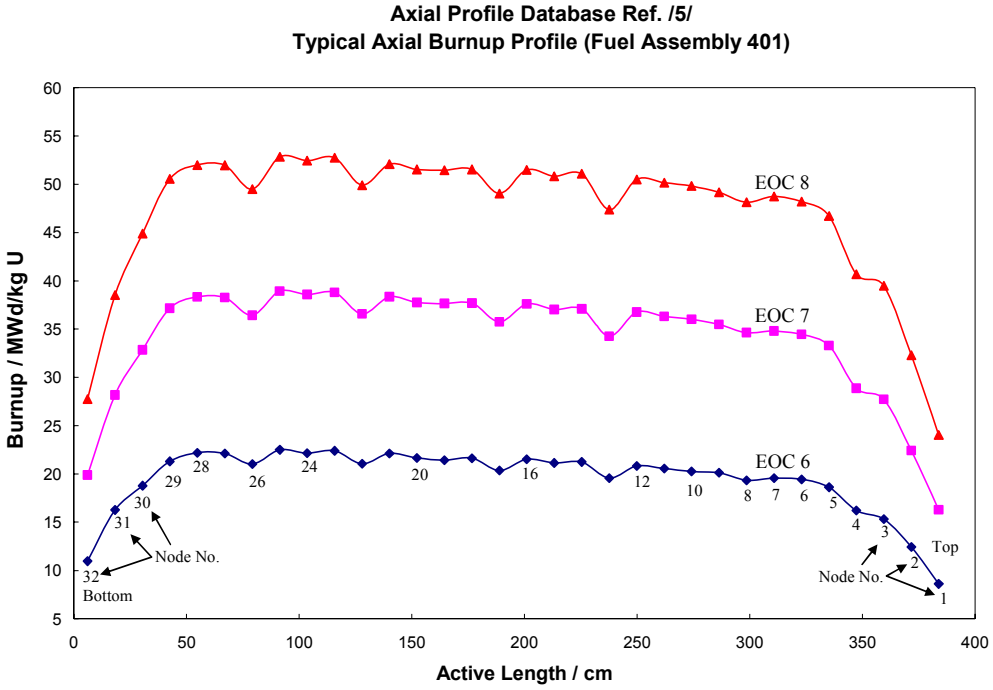

FIG. 1. Typical PWR axial burnup profiles.

- on an analysis of the characteristics of the shapes of axial burnup profiles received from nuclear power plants and

- on application of the results of the Phase II benchmarks conducted by the Expert Group on Burnup Credit Criticality Safety under the auspices of the Nuclear Energy Agency (NEA) of the Organization for Economic Co-operation and Development (OECD) [2][3][4].

As described in Ref. [1], axial burnup profiles are design and plant specific. Therefore, for the estimation of the axial end effect under the conditions of the spent fuel management system of interest profiles from different designs and plants have to be evaluated separately. The method of generating bounding axial burnup profiles described in Ref. [1] is sensitive to the design and plant specific characteristics of burnup profiles since it consists in an analysis of the characteristics of the shapes of real axial profiles.

Figure 1 shows typical EOC axial burnup profiles from an axial profile datafile received from Nuclear Power Plant (NPP) Neckarwestheim II (GKN II) [5]. At the present time this datafile includes 3238 EOC axial burnup profiles from UOX fuel assemblies.

The profiles were derived from in-core 3D power density distribution measurements based on flux measurements. At 28 fuel assembly positions the flux data are monitored at 32 equidistant axial nodes. These measurements are performed every fourteenth day.

To extract from these profiles a bounding profile the shape of which is a continuous function of the average burnup the following steps are taken:

First of all the shapes are normalized by forming the ratios

$$\alpha_{ij} = \frac{B_{ij}}{B_j}, \begin{cases} i = 1,..., n \ (n = 32 := \text{number of nodes, cf. Figure 1}) \\ j = 1,..., N \ (N = 3238 := \text{number of profiles}) \end{cases} \quad (1)$$

of the burnup B_{ij} at node i of the j-th axial profile to the average burnup B_j of this profile. For n equidistant nodes B_j is given by

$$B_j = \frac{1}{n}\sum_{i=1}^{n} B_{ij} \quad . \tag{2}$$

As indicated in Fig. 1, in compliance with the nomenclature used at NPP GKN II i = 1 denotes the node at the top of the fuel zone (top node), and i = n = 32 denotes the node at the bottom of the fuel zone (bottom node).

Let $\alpha(i) = f_i(\overline{B})$ be the function that describes (if exists) the normalized burnup at node i as a *continuous function of the average burnup* $\overline{B}$. However, to get the bounding profile sought-after as a continuous function of the average burnup, it is not only necessary that all the functions $\alpha(i) = f_i(\overline{B})$, $i = 1,...,n,$ exist, it is also necessary that all these functions are correlated with each other in such a way that all $\alpha(i)$ can be derived from each other. So, it must be possible in particular that all $\alpha(i) = f_i(\overline{B})$, $i = 2,...,n,$ can be derived from the function $\alpha(1) = f_1(\overline{B})$ describing the normalized burnup at node 1 as a continuous function of the average burnup $\overline{B}$,

$$\alpha(i) = f_i(\overline{B}) = f_i(f_1(\overline{B})) \quad , i = 2, \ldots, n. \tag{3}$$

Such functions in fact exist as already shown in Ref. [1] on the basis of the 850 evaluated axial burnup profiles used in Ref. [4] and as re-confirmed in the paper on hand on the basis of the 3238 axial burnup profiles from Ref. [5]. (The 850 profiles used in Ref. [6] are included in the datafile Ref. [5].) As can be seen from Figs 2 through 9, there are

- strong correlations between the α_{ij} and α_{i-1j} values of the top nodes i = 2 through i = 7 as well as between the bottom node values α_{n-1j} and α_{nj} (n = 32), and there is
- a significant correlation between the bottom node values α_{nj} and the top node values α_{1j}.

Therefore, by means of linear regression analysis [6] the following recurrence formulae can be extracted from the α_{ij} data:

$$\alpha(i) = r_i(\alpha(i-1)) \quad \text{for } i = 2, \ldots, 7, \tag{4}$$

$$\alpha(n-1) = r_{n-1}(\alpha(n)) \tag{5}$$

$$\alpha(n) = r_n(\alpha(1)) \tag{6}$$

where r_v denotes the respective regression functions resulting from the regression analysis.

The black line in Figs 2 through 9 represents the regression functions obtained for i = 2, … 6, i = n-1 and i = n respectively. The green line in these figures gives the respective one-sided lower 95%/95% tolerance limits of the regression functions obtained [6].

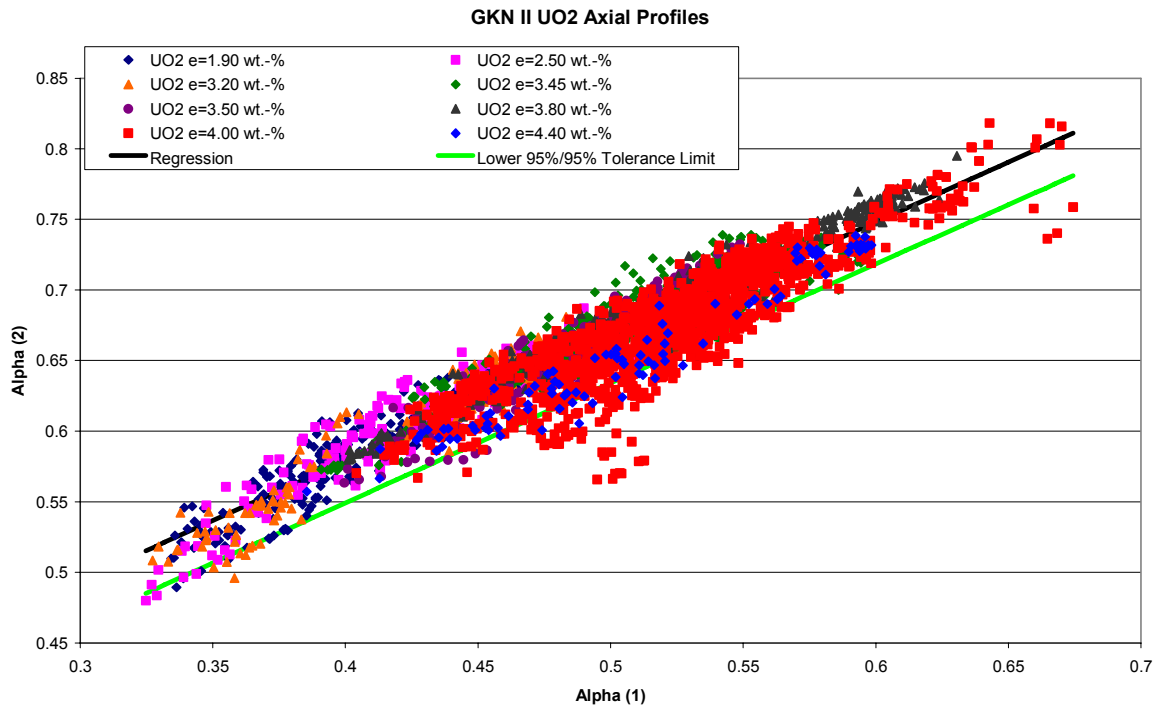

FIG. 2. GKN II UO_2 axial profiles.

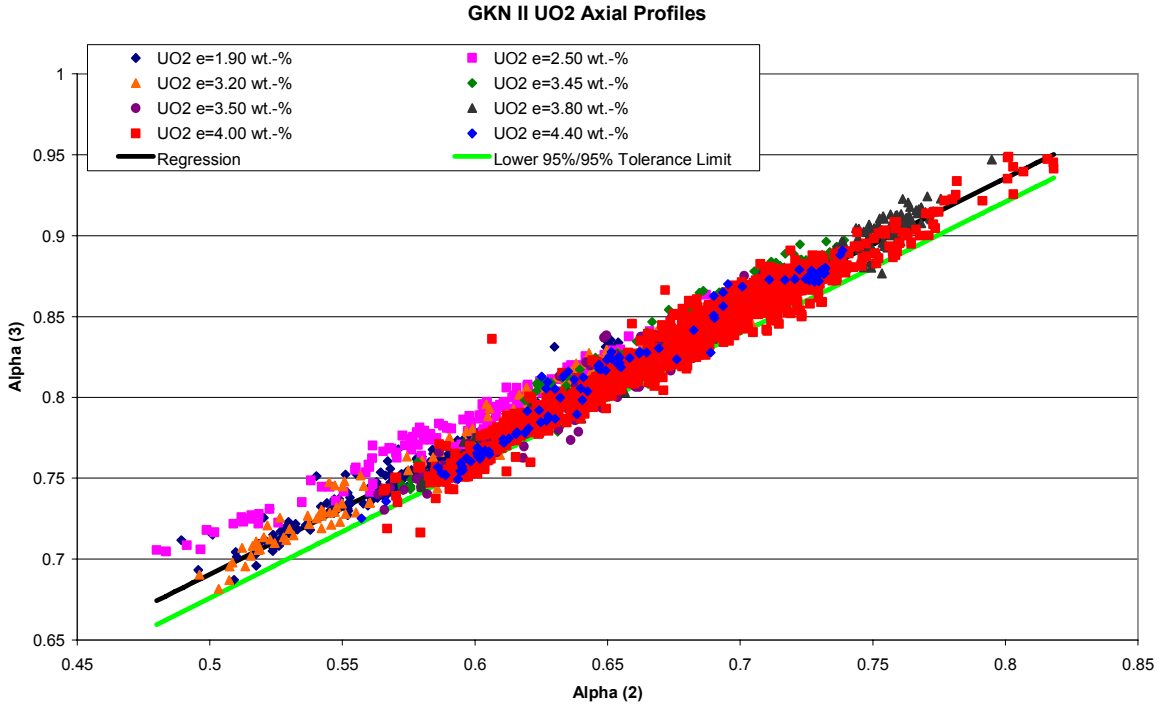

FIG. 3. GKN II UO_2 axial proflies.

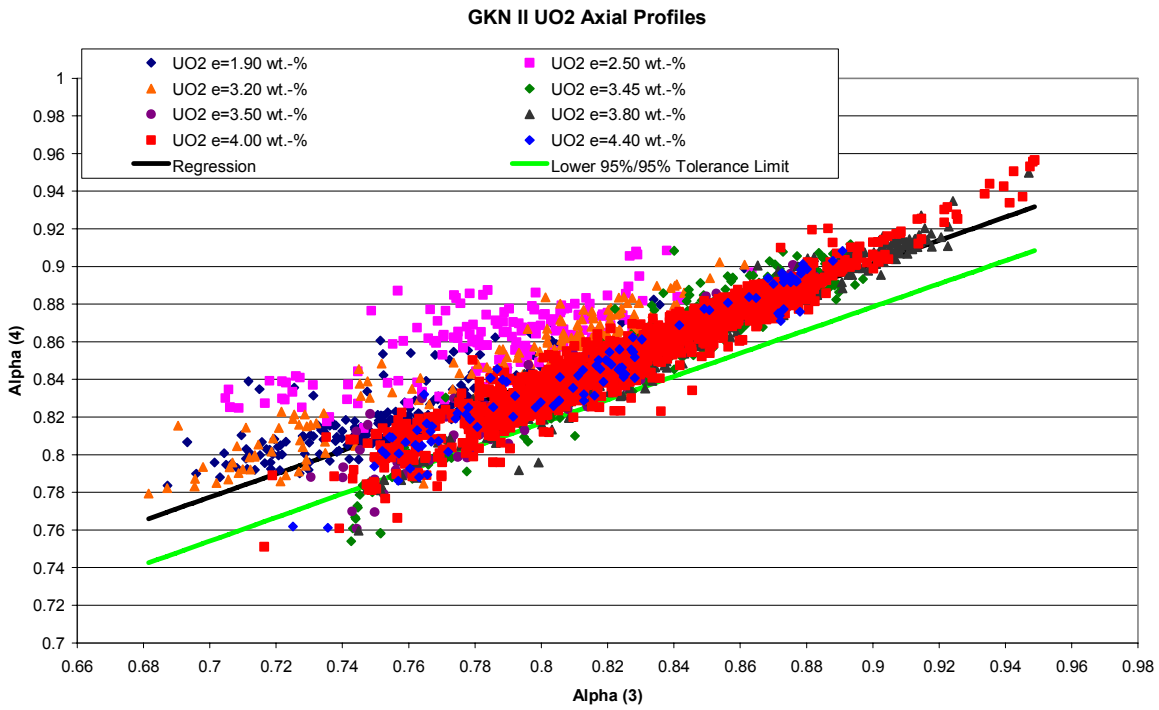

FIG. 4. GKN II UO_2 axial profiles.

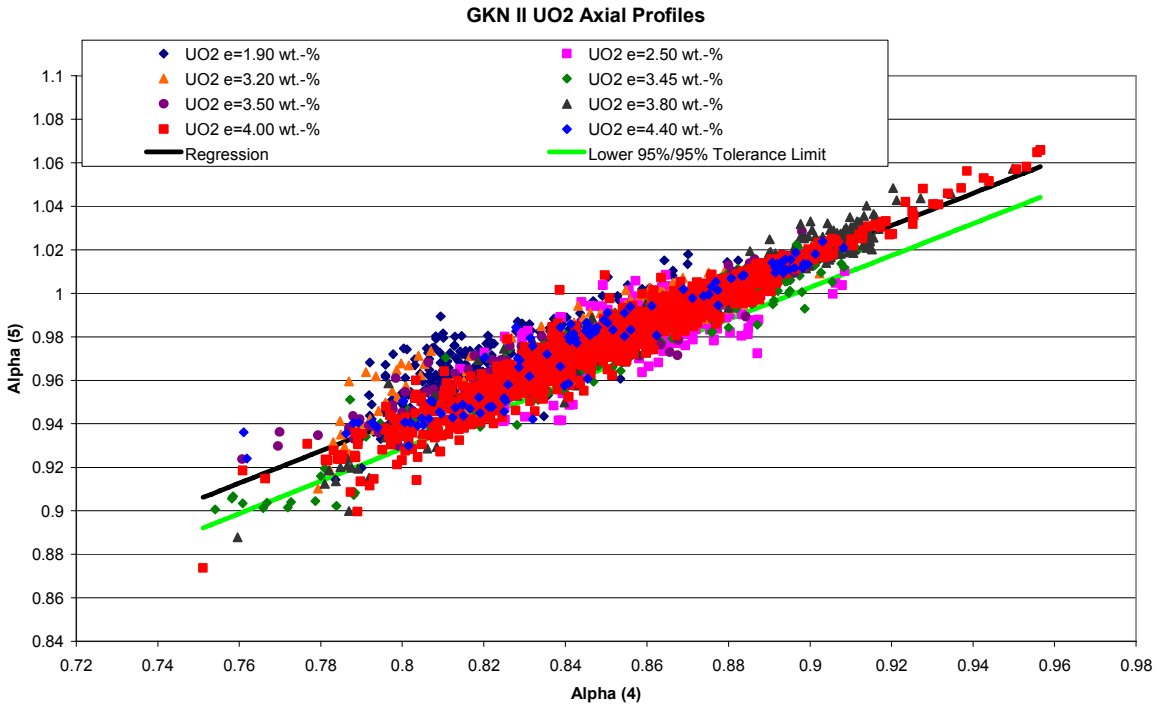

FIG. 5. GKN II UO_2 axial profiles.

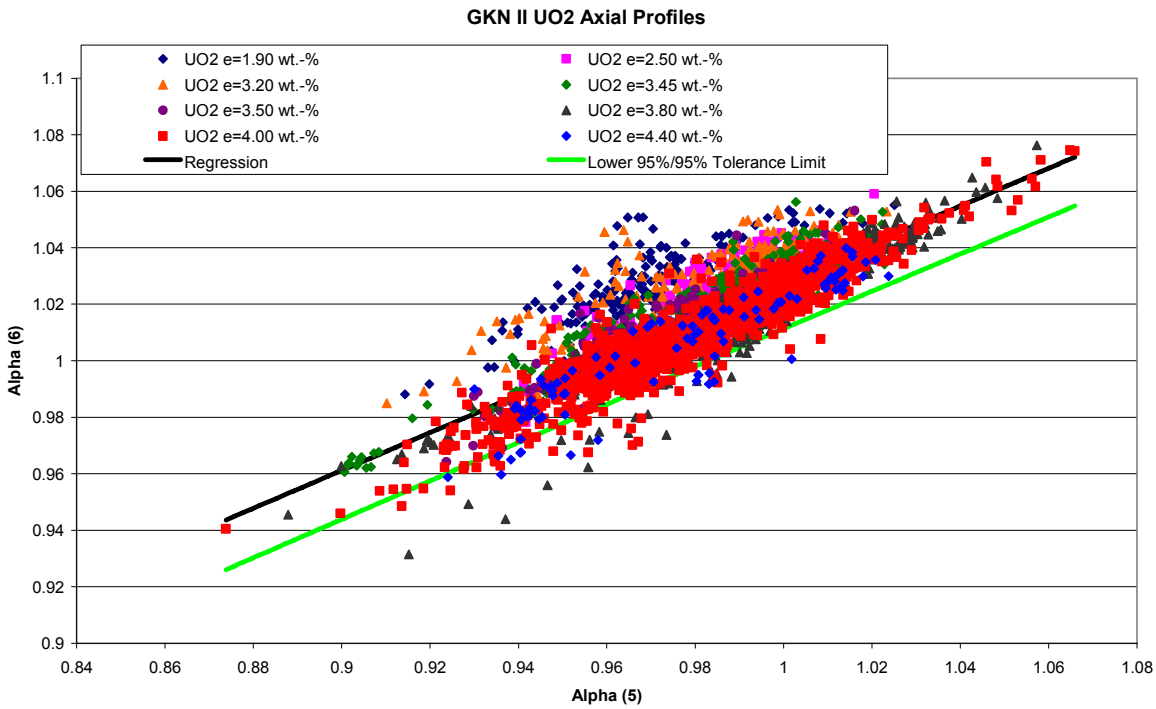

FIG. 6. GKN II UO_2 axial profiles.

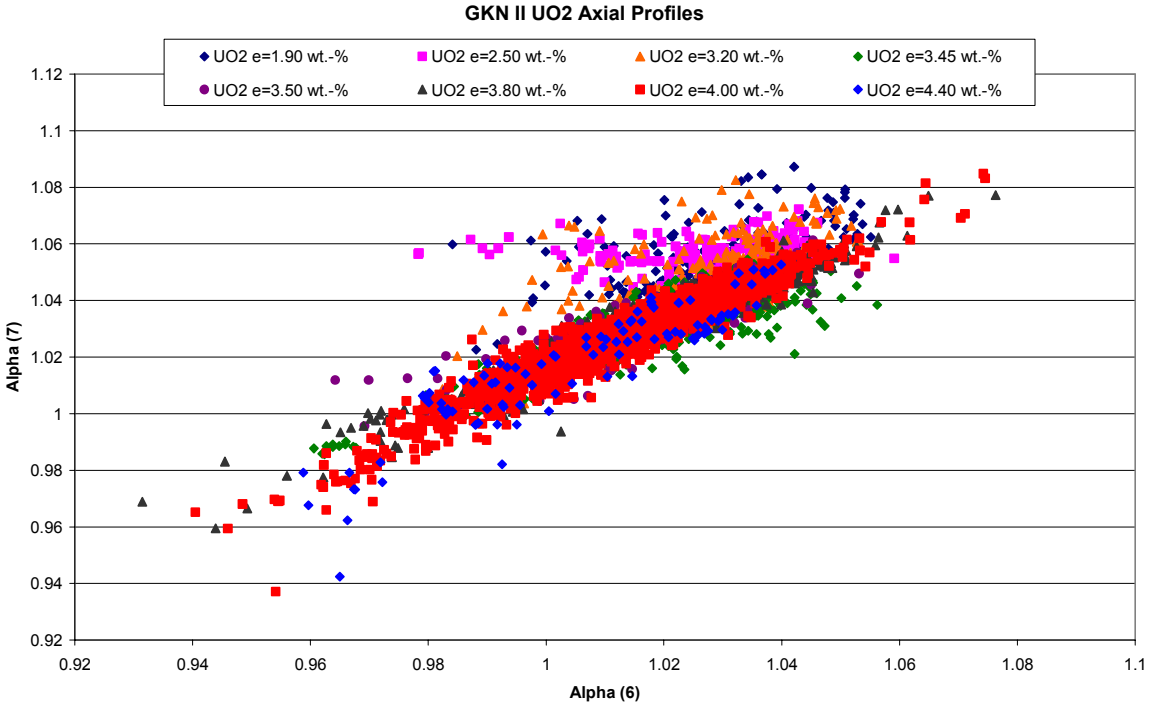

FIG. 7. GKN II UO_2 axial profiles.

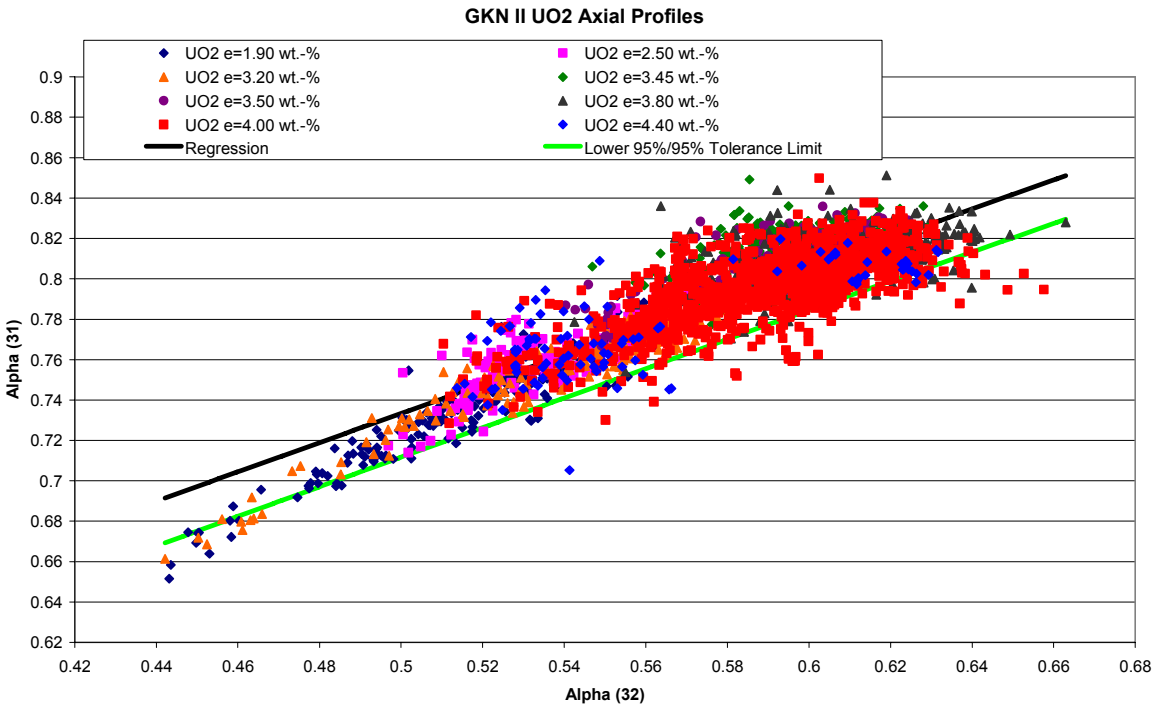

FIG. 8. GKN II UO_2 axial profiles.

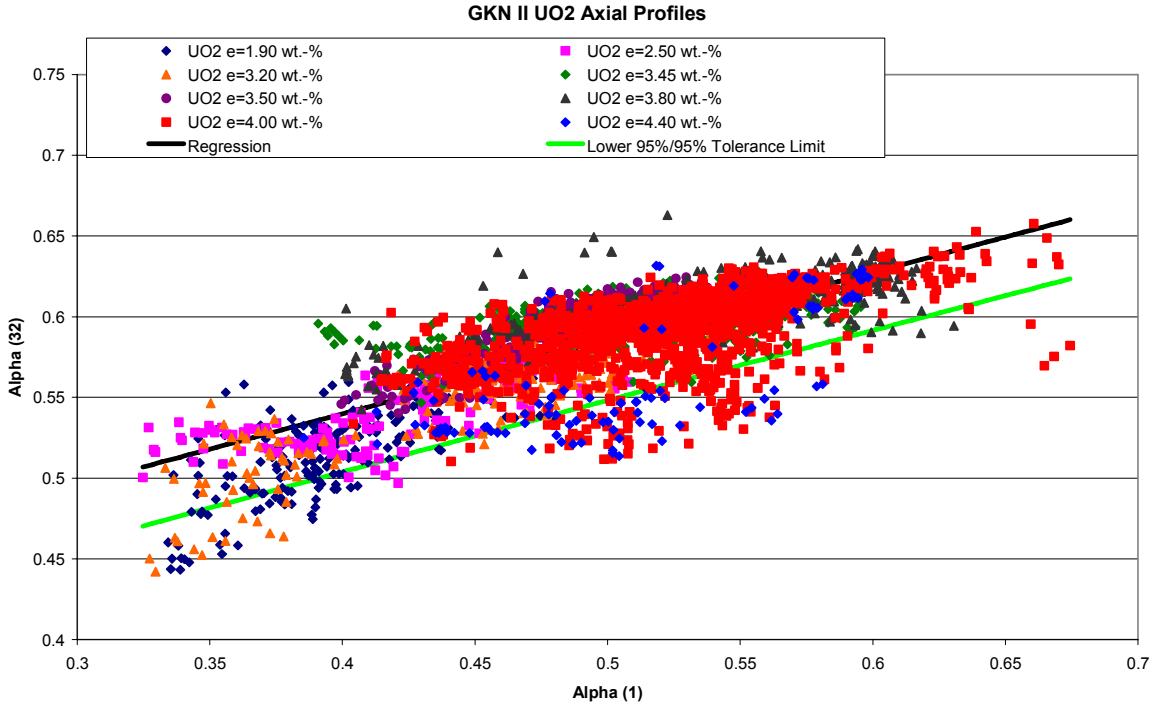

FIG. 9. GKN II UO_2 axial profiles.

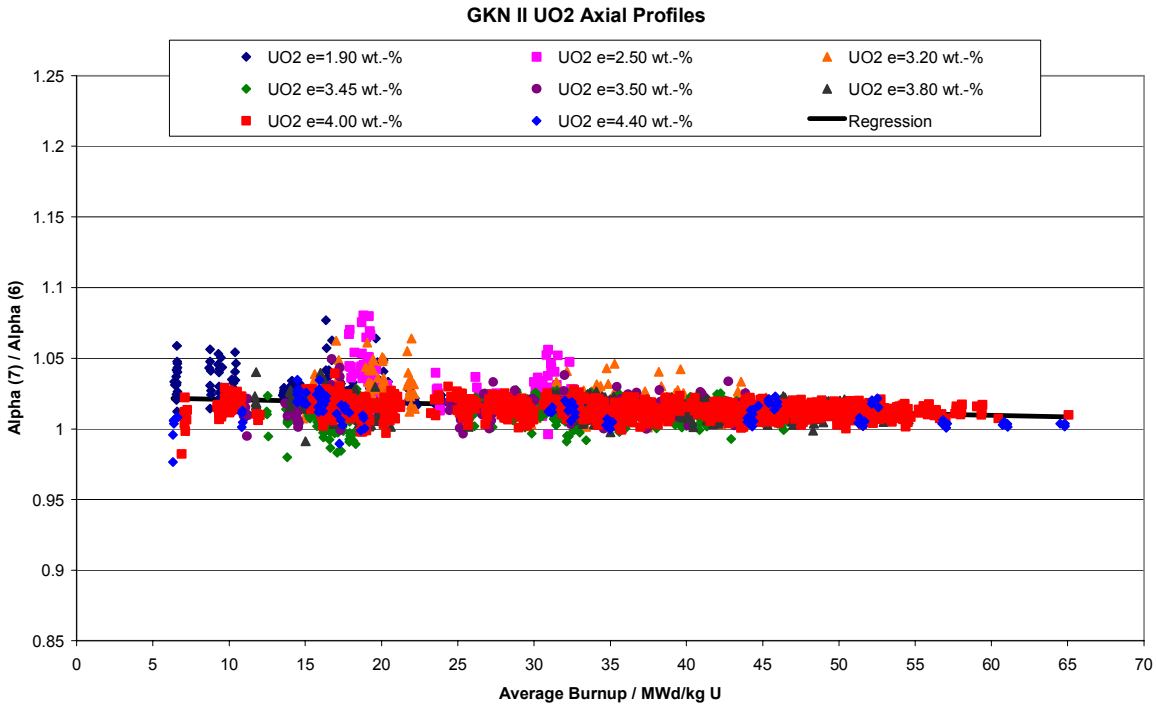

FIG. 10. GKN II UO_2 axial profiles.

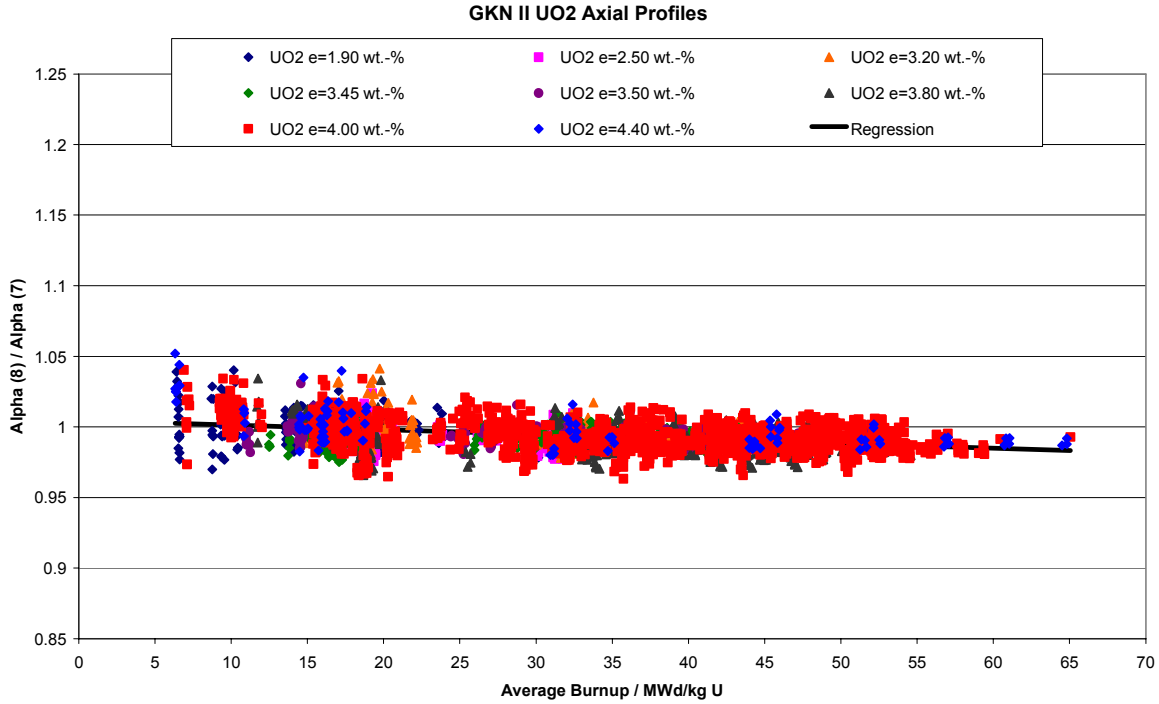

FIG. 11. GKN II UO_2 axial profiles.

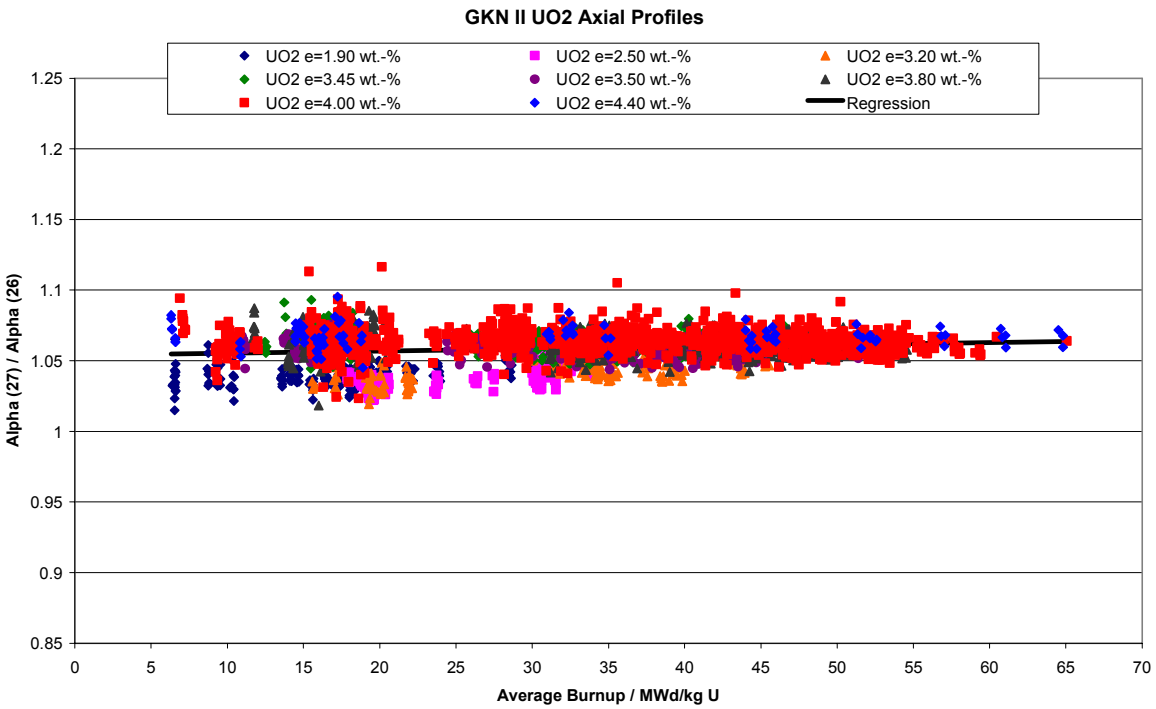

FIG. 12. GKN II UO_2 axial profiles.

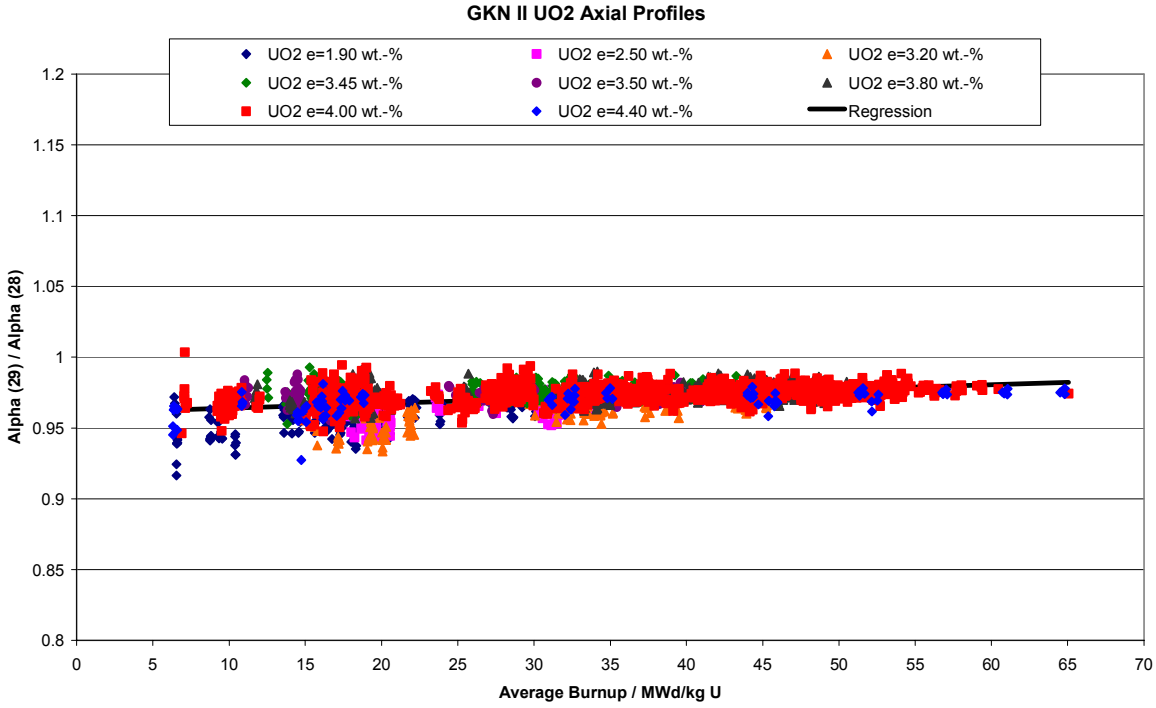

FIG. 13. GKN II UO_2 axial profiles.

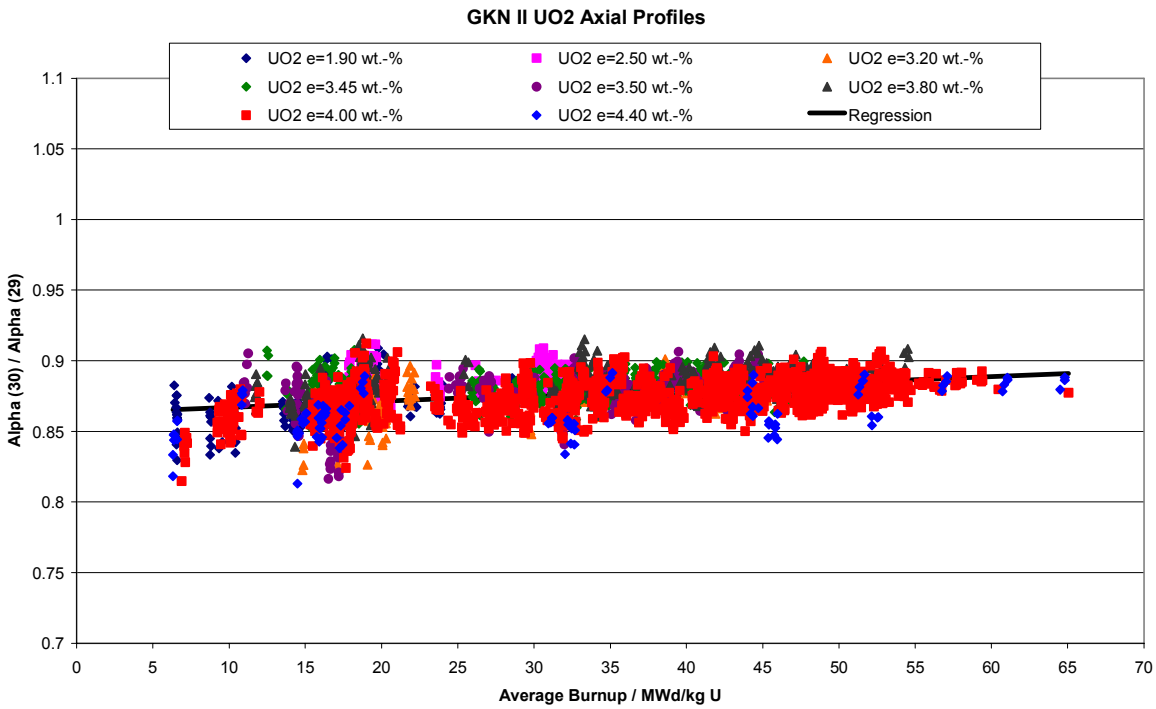

FIG 14. GKN II UO_2 axial profiles.

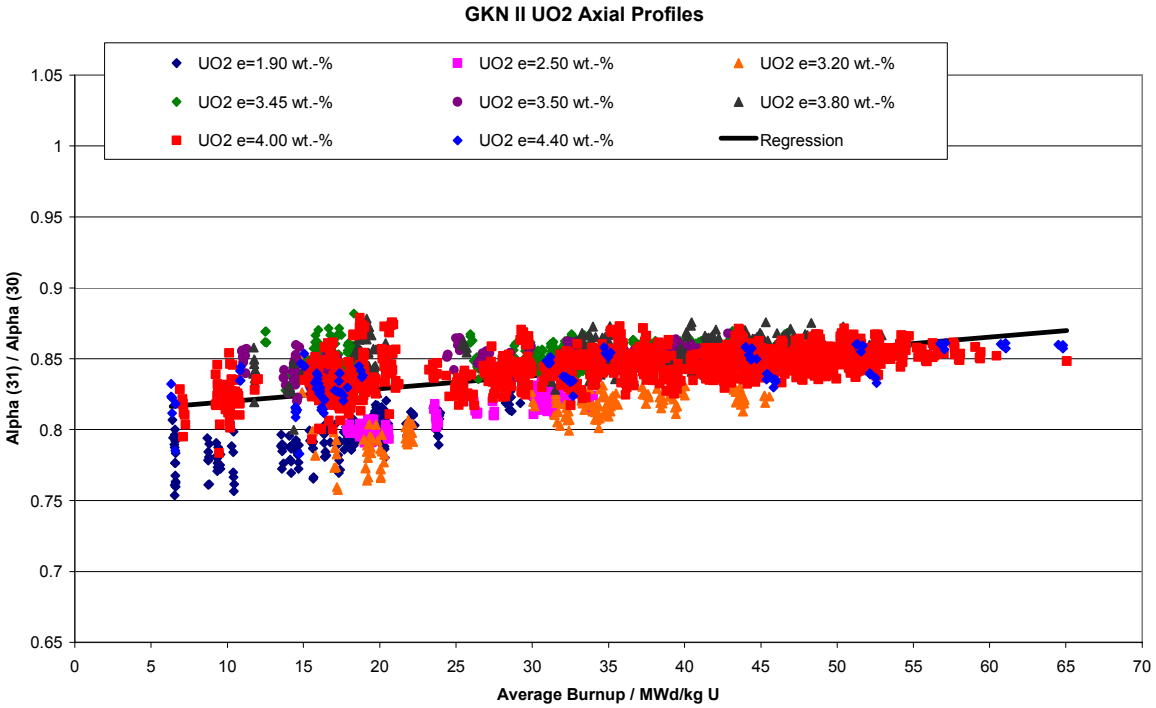

FIG 15. GKN II UO2 axial profiles.

As exemplified in Figs 10 through 15,

- the quotients $\alpha_{ij} / \alpha_{i-1j}$ are virtually independent from the average burnup for i = 7 through i = n – 1 = 31.

This was to be expected for i = 7 through i = 29 at least, since the plateau region of the axial profiles ranges from node 6 to node 29 (Fig. 1).

By means of linear regression analysis the following recurrence formulae can be extracted from the α_{ij} data for i = 7 through i = n-1:

$$\alpha(i) = \alpha(i-1) \cdot r_{i,i-1}(\overline{B}) \quad , i = 7, \ldots, n-1, \tag{7}$$

where $r_{i,i-1}$ denotes the respective regression functions resulting from the regression analysis of the quotients $\alpha_{ij} / \alpha_{i-1j}$ as a function of the average burnup.

The black line in Figs 10 through 15 represents the regression functions obtained.

By means of the set of recurrence formulae given by equations (4) through (7), all the functions $\alpha(i)$, i = 2, …, n can be transformed into functions of the form of eq.(3). In other words, with the set of recurrence equations (4) through (7) a set of model functions $\alpha(i)$ is given forming an axial burnup model shape which can be calculated if one has a model

$$\alpha(1) = f_1(\overline{B}) \tag{8}$$

for node 1 that describes $\alpha(1)$ as a continuous function of the average burnup $\overline{B}$.

The procedure of generating a model function for $\alpha(1)$ which results with the aid of the recurrence equations (4) through (7) in a bounding profile is mainly based on the observations of the Phase II benchmarks conducted by the Expert Group on Burnup Credit Criticality Safety under the auspices of the Nuclear Energy Agency (NEA) of the Organization for Economic Co-operation and Development (OECD) [2][3][4].

In Refs [2][3][4] different definitions of the end effect (:= reactivity effect due to the non-uniform axial distribution of the burnup in the active zone of fuel assemblies) are given. In the paper on hand the most common definition is used:

- The end effect due to an axial burnup profile is the difference Δk between the spent fuel configuration's neutron multiplication factor obtained with the burnup profile and the configuration's neutron multiplication factor obtained by assuming a uniform distribution of the averaged burnup of the profile.

As regards the impact of the shape of axial burnup profiles on the end effect it has been found in Refs [2][3][4] that

- the end effect is dictated by the shape of the profiles at the lesser burnt end of the fuel zone of the fuel assemblies.

The lesser burnt end of the fuel zone usually is, due to the lower moderator density in the upper half of an operating core, the top end region of the fuel zone. For the fuel assemblies under examination the top end region is given by the region which includes the nodes No.1 through No.6 (Fig. 1).

The main observations made in Refs [2][3][4] are: The end effect Δk due to an axial burnup profile

- is fundamentally determined by the degree of the asymmetry of the axial burnup profile and
- can be significantly affected by the "local asymmetry" of the shape of the profile at the top end of the profile.

As shown in Ref. [4], with respect to its impact on the end effect the degree of asymmetry of an axial burnup profile can be adequately described by the "top end sum"

$$S6 = \frac{1}{n} \sum_{i=1}^{6} \alpha(i) \quad , \tag{9}$$

with

$$\alpha(i) = \frac{B(i)}{\overline{B}} \quad , i = 1,\ldots, n. \tag{10}$$

By definition, the lower the value of the top end sum eq.(9) is, the higher is the asymmetry of the axial burnup profile. It has been observed and described in Ref. [4].

- that the asymmetry of axial burnup profiles tends to decrease with increasing average burnup (Fig. 16),
- that at given asymmetry the end effect is increasing with increasing average burnup (Fig. 17), and
- that at given average burnup the end effect is the higher, the higher the asymmetry of the profile is, cf. ibid.

Figure 17 shows the end effects Δk observed in Ref. [4] as a function of the top end sum S6, i.e., as a function of the degree of asymmetry of the axial profiles analyzed in Ref. [4] for average burnup values of 32 MWd/kg U and 50 MWd/kg U respectively. The solid lines in this figure represent the respective regression functions resulting from the Δk values observed. As appears from Fig. 17, the end effect is increasing with increasing asymmetry (decreasing S6) and increasing average burnup; and the sensitivity of the end effect on the asymmetry of an axial profile is increasing with increasing average burnup of the profile.

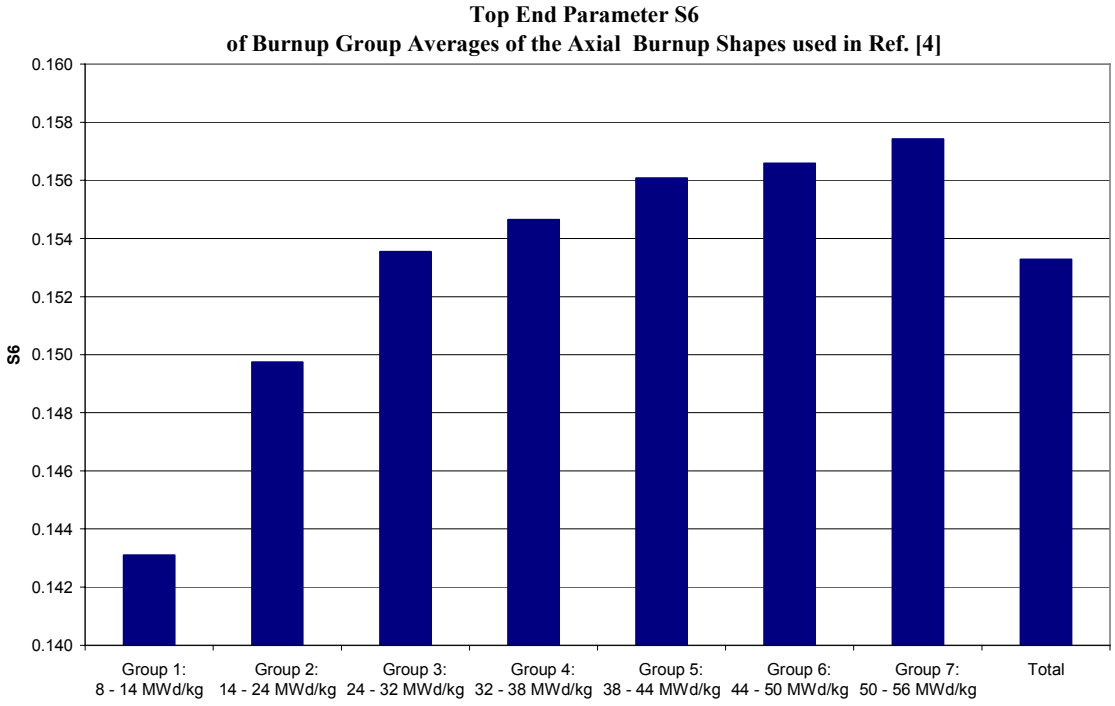

FIG. 16. Top end parameter S6 of burnup group averages of the axial burnup shapes used in Ref. [4].

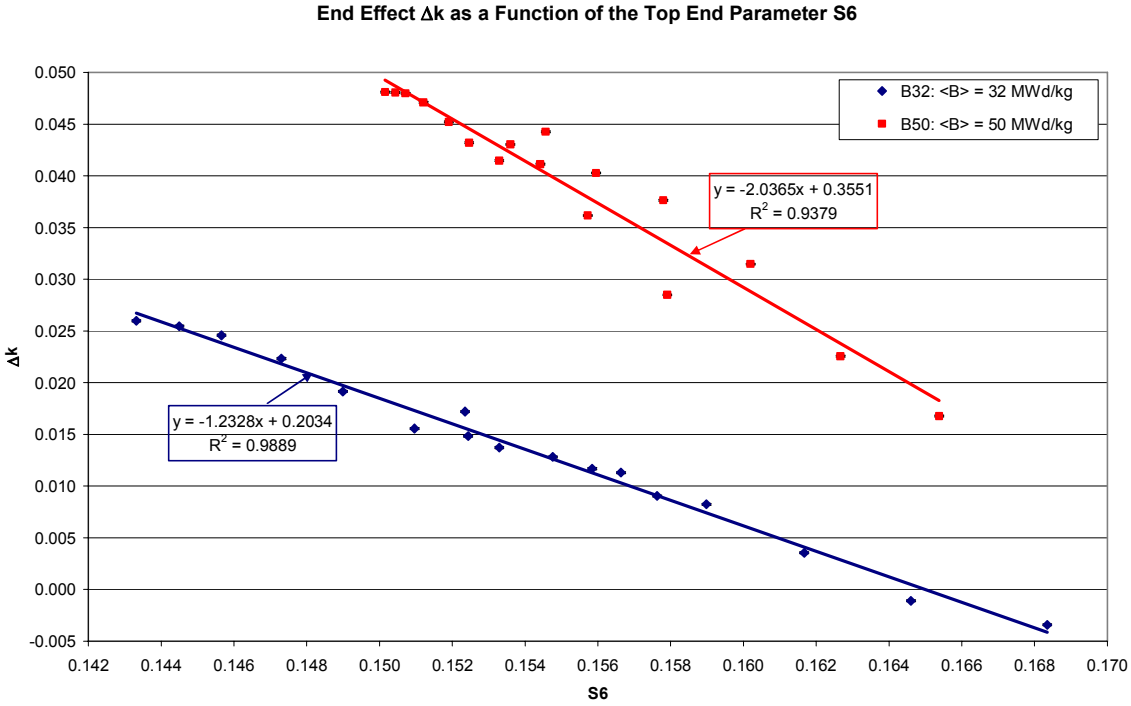

FIG. 17. End effect Δk as a function of the top end parameter S6 [4].

As shown in Ref. [4], with respect to its impact on the end effect the degree of "local asymmetry" can be described by the ratio S13/S6 of the "partial top end sum" S13 to the top end sum S6,

$$S13 = \frac{1}{n}\sum_{i=1}^{3}\alpha(i) = S6 - \frac{1}{n}\sum_{i=4}^{6}\alpha(i) \qquad (11)$$

By definition, the lower the value of the ratio S13/S6 is, the higher is the "local asymmetry". It has been observed in Ref. [4] that

- the impact of the "local asymmetry" of an axial burnup profile on the end effect Δk is the higher, the higher the average burnup of the profile is.

This can be seen from Fig. 18. In this figure the Δk values already presented in Fig. 17 are plotted versus the parameter

$$A \equiv S6 + g \cdot \frac{S13}{S6} \qquad (12)$$

By definition, the factor g in this expression couples the profile's asymmetry with the "local asymmetry" of the profile's top end shape. In the "uncoupled" case, i.e., with $g = 0$ one gets the regression curves presented in Fig. 17. In Fig. 18 these curves serve as starting

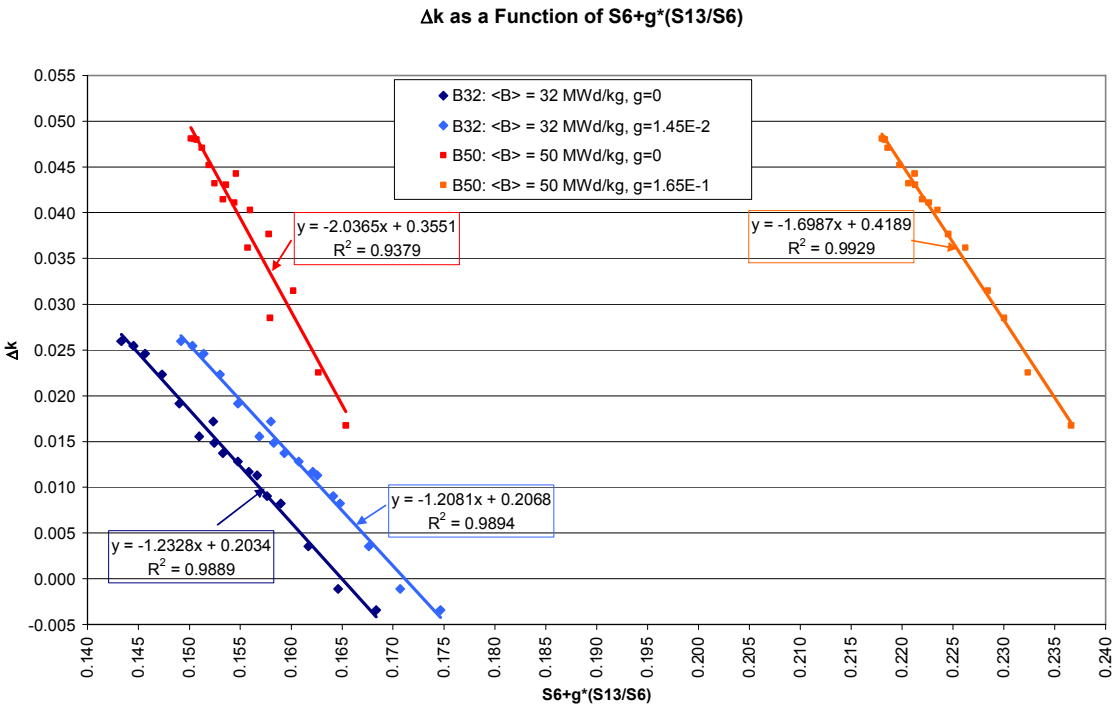

FIG 18. Δk as function of S6+(S13/S6) [4].*

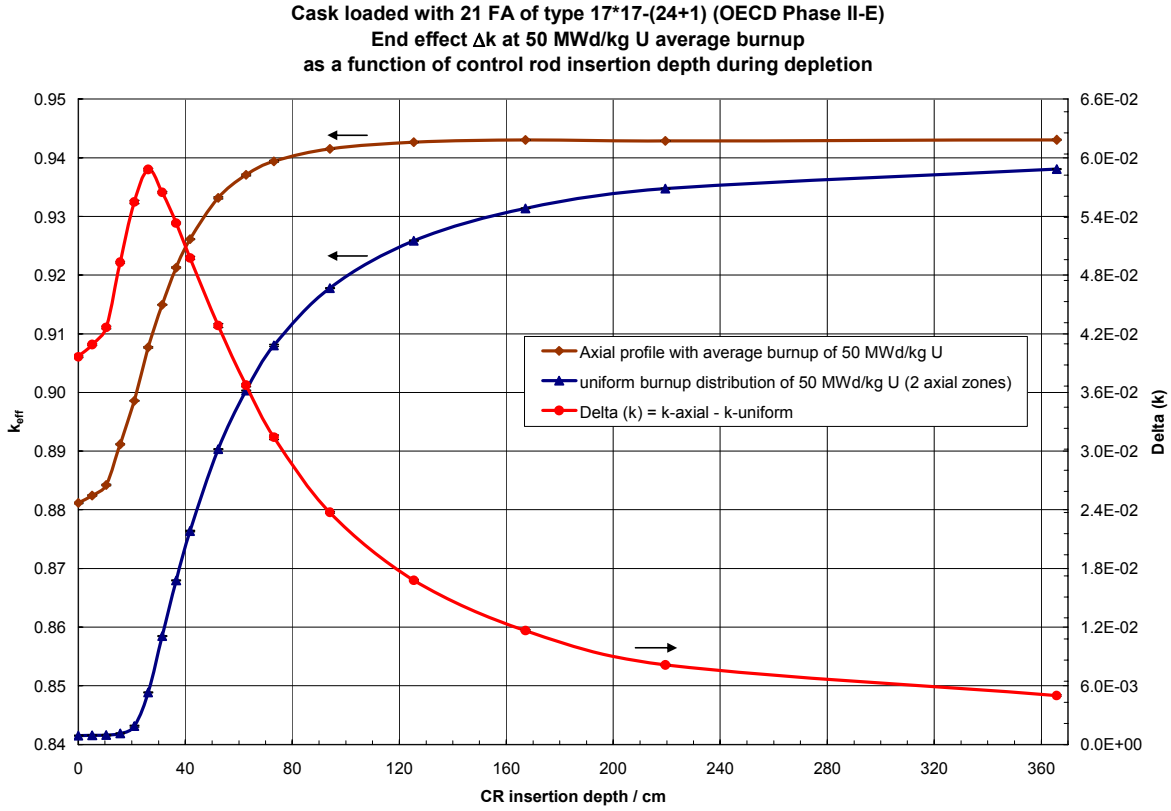

*FIG 19. Cask loaded with 21 FA of type 17*17-(24+1)(OECD Phase II-E) end effect Δk at 50 MW d/kg U average burnup as a fuction of control rod insertion depth during depletion.*

curves now. Starting with these curves the factor g is increased such that the square of the sample correlation coefficient R of the regression function is maximized. The final regression curves belonging to the maximum R^2 values achieved are presented in Fig. 18. As appears from this figure,

the increase in R^2 obtained for 32 MWd/kg U axial burnup profiles is insignificant, whereas the increase in R^2 obtained for 50 MWd/kg U burnup profiles is significant. This shows that the impact of "local asymmetries" on the end effect increases with increasing average burnup of axial burnup profiles.

The end effect due to an axial burnup profile is not only determined by the shape of the profile, but also by the axial distribution of the isotopic number densities. The end effect varies with varying axial distribution of the isotopic number densities, as is reflected for instance by the fact that the end effect is changing with the cooling time of the fuel [2][3][7].

The axial distribution of the isotopic number densities depends on the depletion conditions. Therefore, for a given axial burnup shape the impact of the "local asymmetry" of the profile on the end effect can be significantly increased, for instance, by partial control rod insertion during irradiation of the fuel since spent fuel exposed to control rod (CR) insertion during irradiation has, at given burnup, a higher reactivity than spent fuel which has not been exposed to control rod insertion (Fig. 19). This Figure shows the k_{eff} value of a spent fuel cask configuration as a function of the CR insertion depth during depletion for the following cases:

- A bounding axial burnup profile of 50 MWd/kg U average burnup derived from a profile database for 17x17-25 UOX fuel assemblies is taken for the 21 fuel assemblies assumed to be loaded in the cask.

- The average burnup of this profile is assumed to be uniformly distributed throughout the active zone of the fuel assemblies. The active zone is divided into two regions, that one which has been exposed to control rod insertion, and that one which has been not.

As appears from Fig. 19, the end effect reaches a maximum for insertion depths just in the range of the top end zone of the fuel assemblies. Therefore, it is obvious that impact of the "local asymmetries" of the profile on the end effect can be significantly increased by CR insertion during depletion.

To cover the impact of the asymmetry of axial burnup profiles on the end effect as well as the impact of "local asymmetries" at the top end region of these profiles on the end effect the model distribution eq.(8) for $\alpha(1)$ described in Ref.[1],

$$\alpha(1) = \frac{a}{b \cdot \left(\overline{B} + c\right)^2 + 1} + d \cdot \left[1 - \exp\left(-\lambda \cdot \left(\overline{B} + c\right)\right)\right], \quad a, b, c, d, \lambda = \text{const.} \in \Re\,, \tag{13}$$

is chosen such that the values of the bounding profile resulting for the top nodes i = 1 through i = 6 from the relations (13) and (4) remain below the α_{ij} values of all the 3238 profiles from the database Ref. [5] (Figs 20 through 25). If this cannot be achieved for i = 2 through i = 6 with the aid of the regression functions (4) then the lower 95%/95% tolerance limit of these functions is applied (Figs 2 through 6).

According to equations (1) and (2) the ratios a_{ij} are bounded by

$$\sum_{i=1}^{n} \alpha_{ij} = n \,. \tag{14}$$

Due to the fact that the model function eq. (13) represents a lower bound (Fig. 20) and that the regression functions in equations (4) through (7) do not necessarily conserve the normalization condition eq.(14), the resulting $\alpha(i)$ values have to be re-normalized. However, because of being defined as a lower bound $\alpha(1)$ given by eq. (13) cannot be included in the re-normalization. In addition, the regression functions in eq. (4) reflect the strong correlations between the α values of nodes 1 through 7. Therefore, these correlations cannot be included in the re-normalization, since $\alpha(1)$

has to be excluded from the re-normalization. The re-normalization is therefore restricted to a correction of the α values of nodes k = 10 through m = 28. The α values of these nodes are multiplied with a node-dependent factor g_i given by

$$g_i = s \cdot (i - (k-1)) + 1 \quad \text{for } i = k, \ldots, m, \tag{15}$$

where s is given by

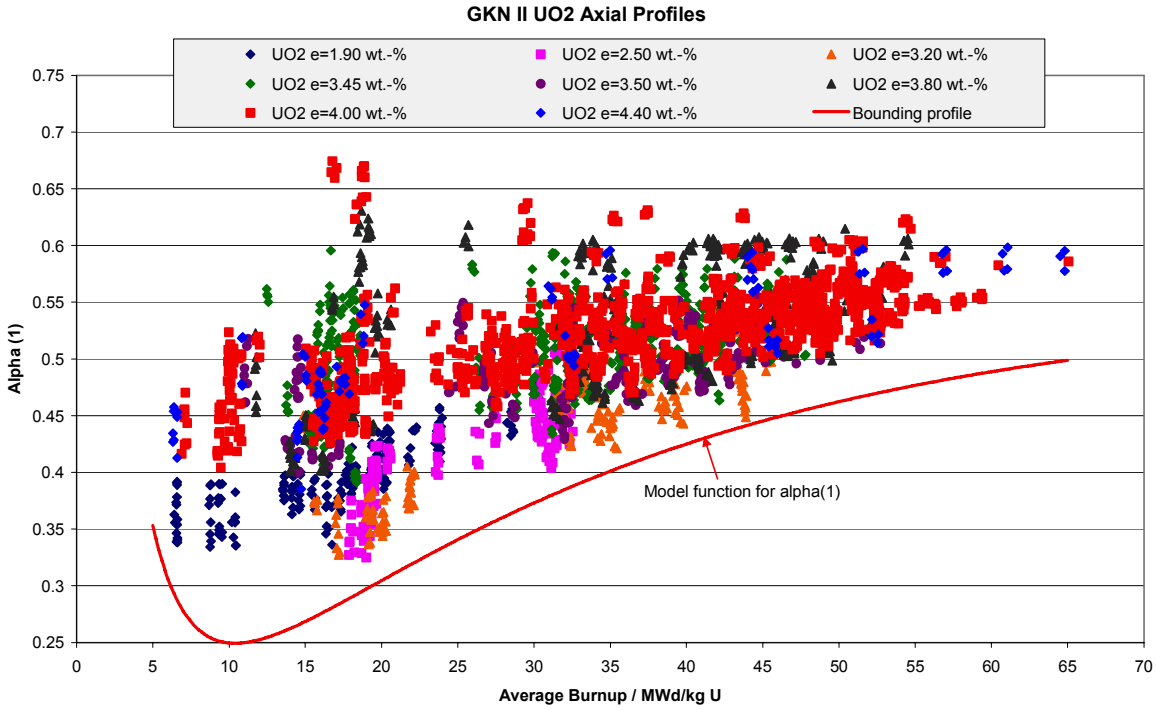

FIG. 20. GKN II UO_2 axial profiles.

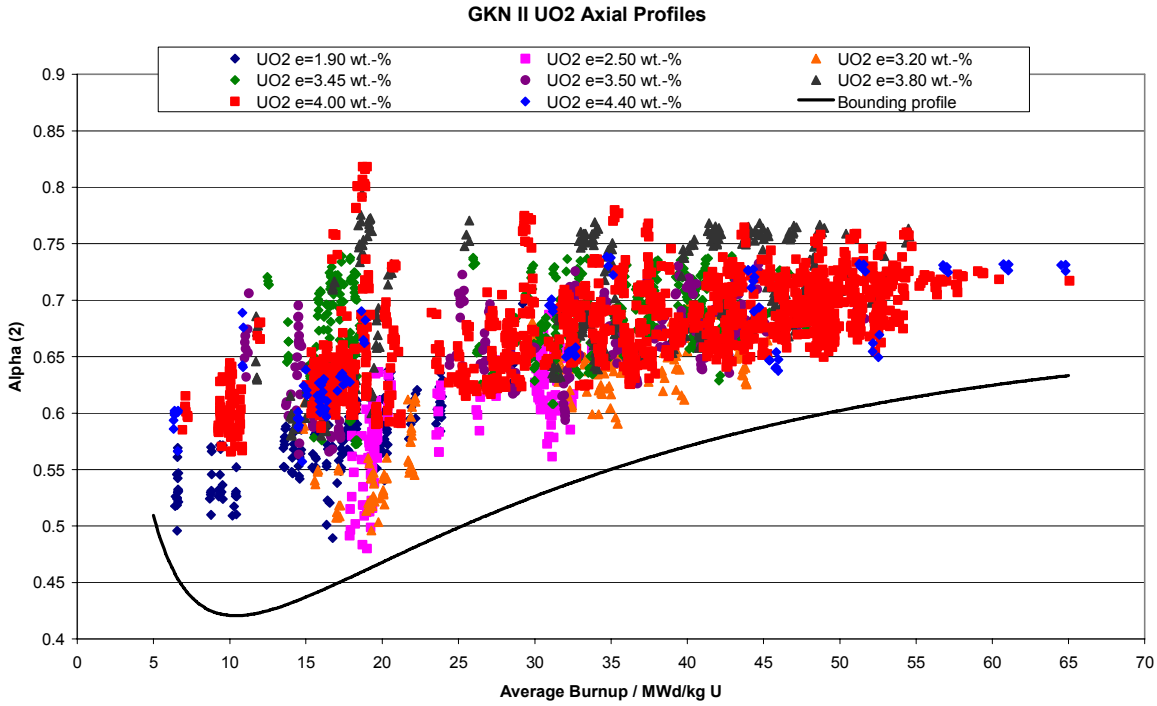

FIG 21. GKN II UO_2 axial profiles.

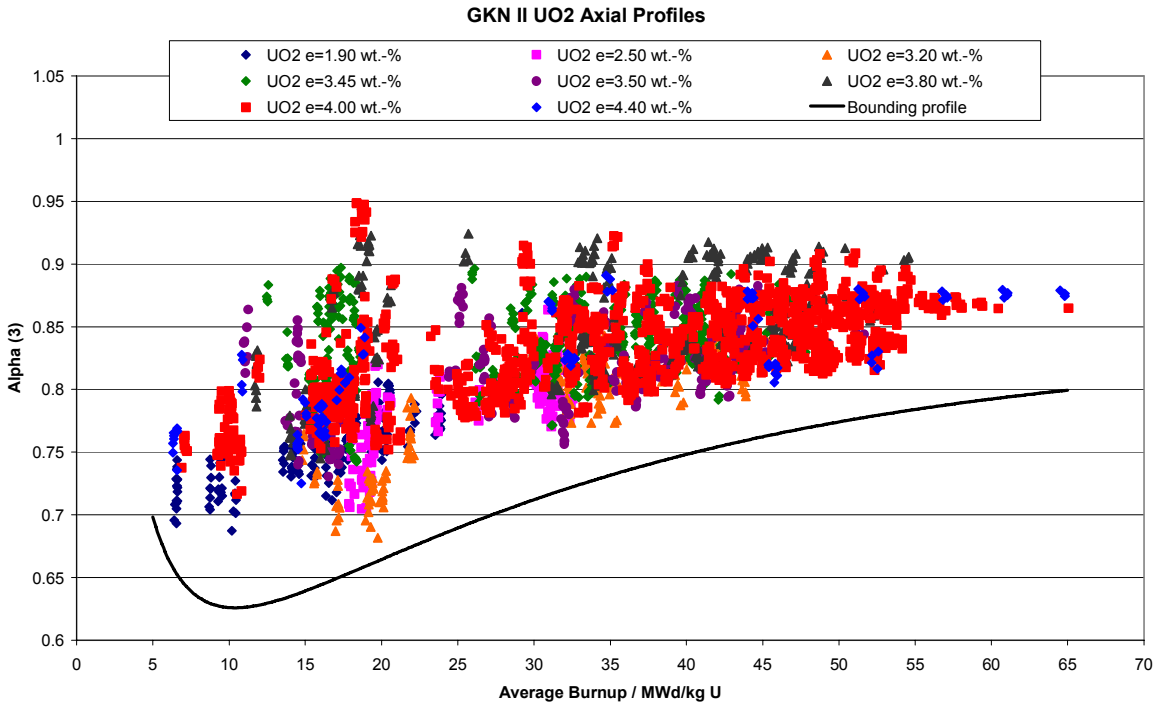

FIG 22. GKN II UO_2 axial profiles.

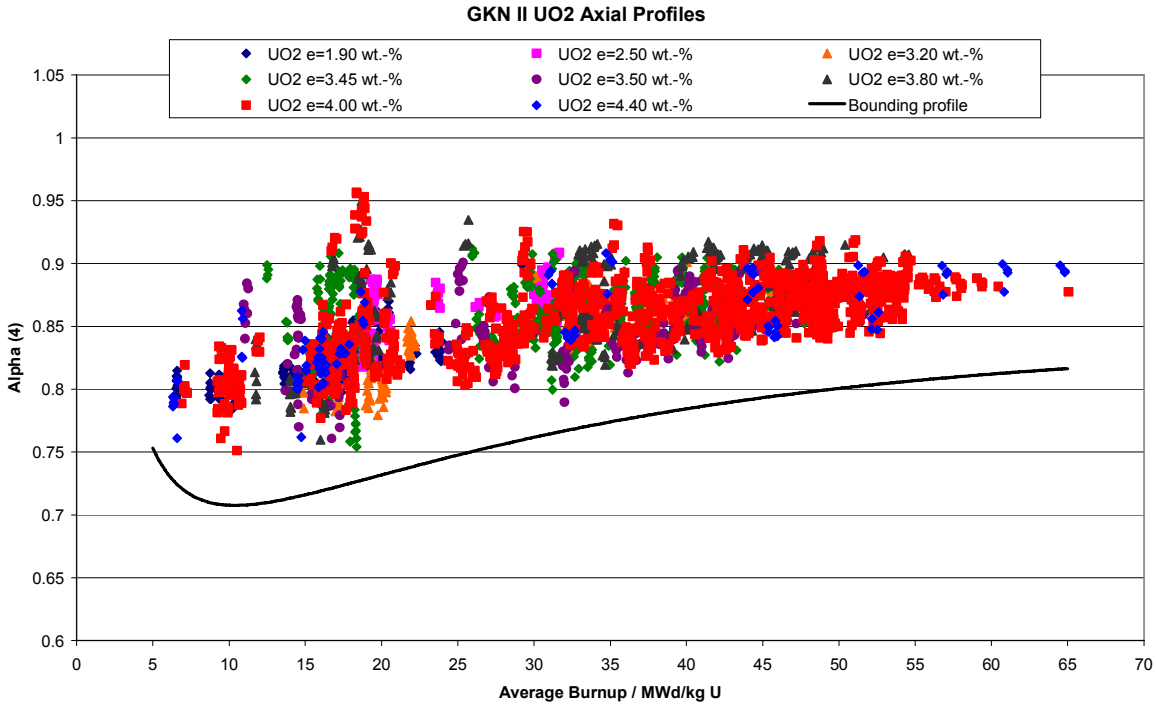

FIG 23. GKN II UO2 axial profiles.

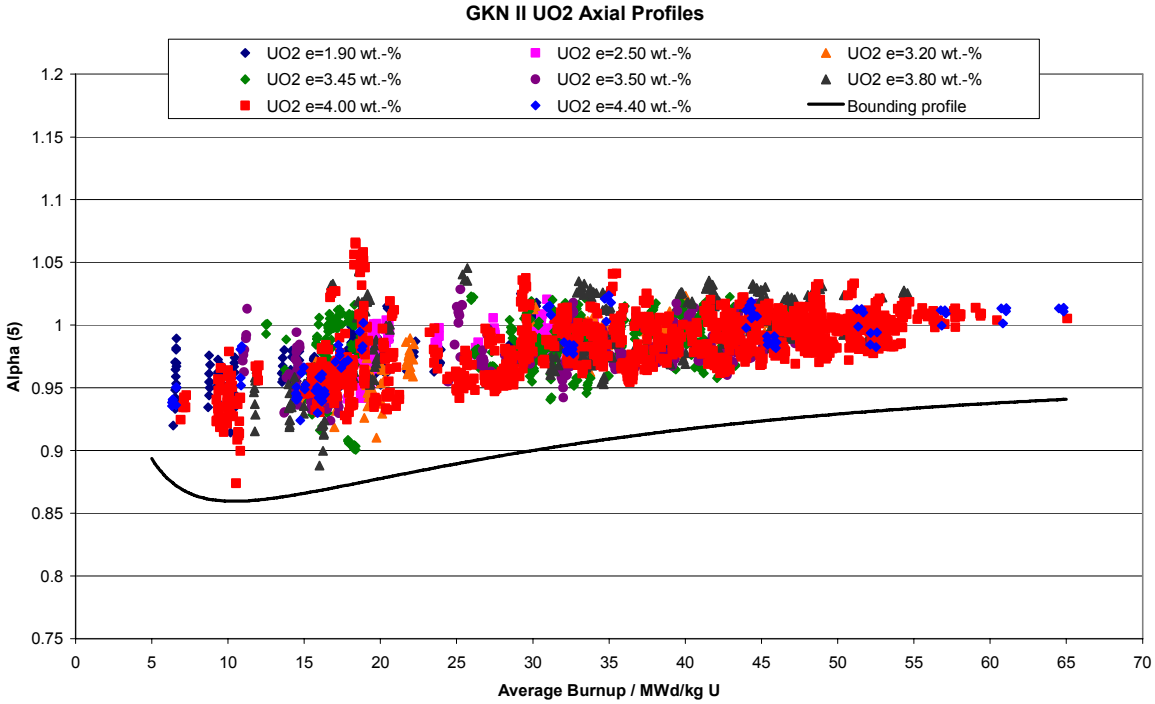

FIG. 24. GKN II UO_2 axial profiles.

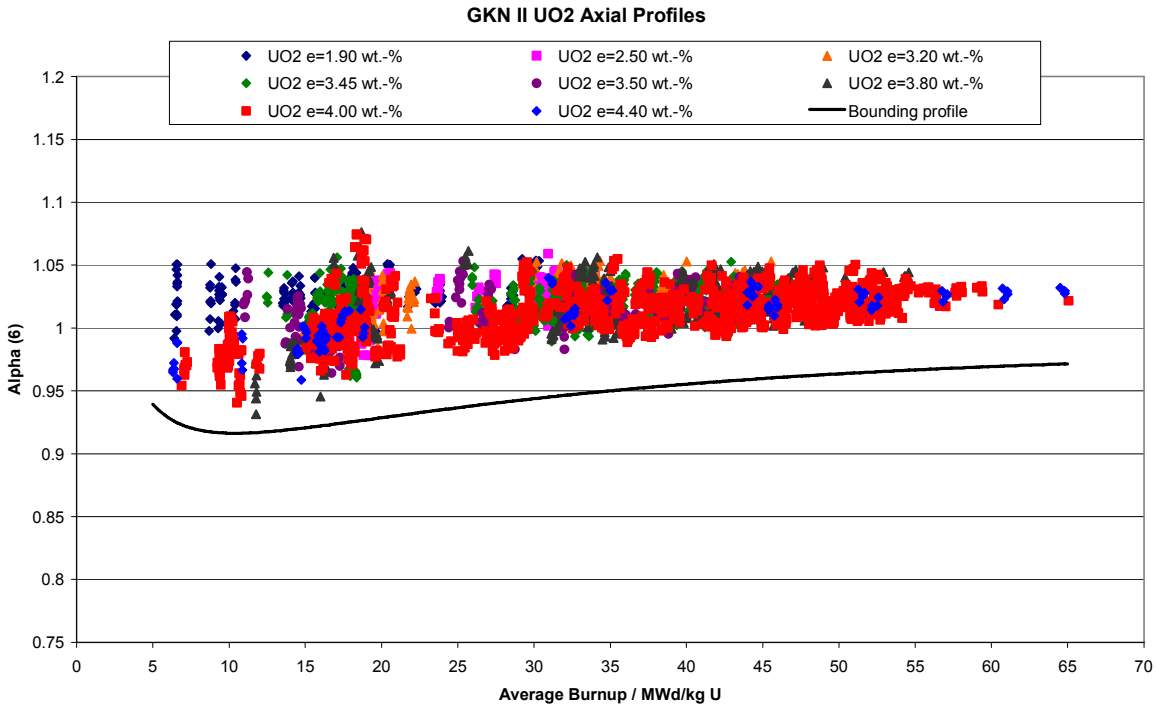

FIG 25. GKN II UO_2 axial profiles.

$$s = \frac{n - \sum_{j=1}^{n} \alpha(j)}{\sum_{j=k}^{m} \alpha(j) \cdot (j - (k-1))} \quad , \; k = 10 \;\; \text{and} \;\; m = 28 \;\; . \tag{16}$$

The bounding profile thus obtained covers the end effects of all the real profiles on which its generation is based. Since its values resulting for the top nodes i = 1 through i = 6 remain below the α_{ij} values of all these real profiles all impacts on the end effect due to "local asymmetries" are covered, even if a section of the top end region of the fuel zone juts out of the neutron absorbing channels of the storage cells of the spent fuel racks or casks of interest, or even if the axial isotopic inventory distribution is impacted by partial control rod insertion during irradiation of the fuel.

3. Description of the method of generating bounding horizontal burnup profiles as a continuous function of the average burnup

In Ref. [8] it was already shown that the loading curve might be affected by horizontal burnup profiles within fuel assemblies. In the case analyzed in Ref. [8] a horizontal burnup gradient parallel to two opposite lateral faces of each fuel assembly was assumed. In the paper on hand the attention is therefore focused on horizontal burnup gradients diagonal through the fuel assemblies.

Let us first take n x n fuel assemblies with *even* number n of lattice positions per row and column. As in Ref. [8] a linear model is used for the horizontal burnup tilt which conservatively covers the maximum assembly quadrant deviations from the fuel assembly average burnup determined in Ref. [9] from the horizontal database given in Ref. [10]. The assembly quadrant deviation is assumed to be given by

$$\frac{B_{UL} - \overline{B}}{\overline{B}} = 0.33 - \frac{0.08}{15} \cdot \left(\frac{\overline{B}}{MWD/kgU} - 10 \right) \tag{17}$$

where $\overline{B}$ is the fuel assembly's average burnup and B_{UL} denotes the average burnup of the "upper left" quadrant of the fuel assembly, see Fig. 26. Assuming first that each position of the n x n lattice describing the fuel assembly is filled with a fuel rod, and assuming

- that along the main diagonal from the upper left (UL) corner to the lower right (LR) corner of the fuel assembly (Fig. 26) the burnup decreases linearly, i.e. drops by a constant amount from fuel rod position to fuel rod position, and assuming

- that along each line perpendicular to the main diagonal the burnup remains constant

$$B_{j,i-(j-1)} = B(i), \begin{cases} j = 1,..., i \text{ for } i = 1,..., n \\ j = i - (n-1),..., n \text{ for } i = n+1,..., 2n-1 \end{cases} \tag{18}$$

with

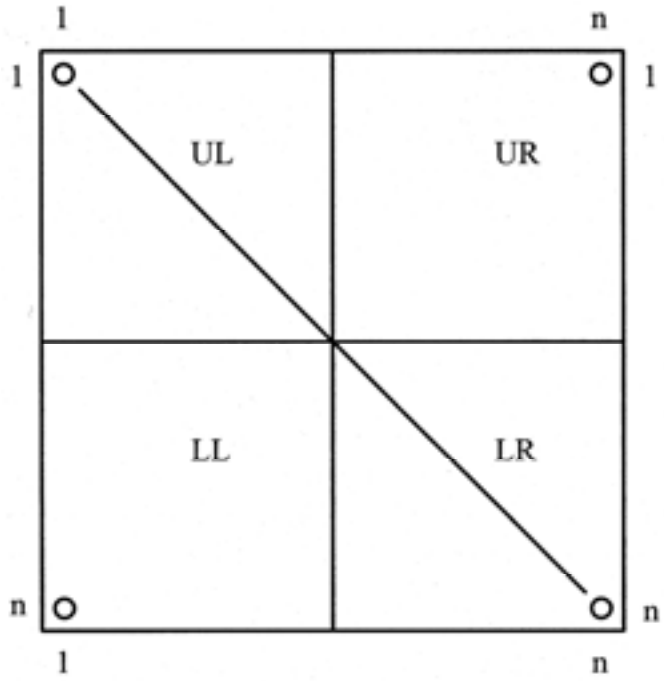

FIG. 26. Notation used for characterizing horizontal profiles (for even numbers n).

$$B(i) = 2 \cdot \left(B_{UL} - \overline{B}\right) \cdot \left(1 - \frac{i}{n}\right) + \overline{B}\ ,\ i = 1,\ldots,\ 2n-1 \tag{19}$$

Using then the real fuel rod configuration and hence re-normalizing the burnups eq.(18) in such a way that the average burnup of each assembly quadrant remains constant one gets the final horizontal profile: The final burnup $(B_{\nu\mu})_q$ at lattice position (ν,μ) in the assembly quadrant q is given by

$$\left(B_{\nu\mu}\right)_q = f_q \cdot B_{\nu\mu} \cdot \delta_{\nu\mu}(FR) \tag{20}$$

where $B_{\nu\mu}$ is given by equations (18) and (19), $\delta_{\nu\mu}$ (FR) is given by

$$\delta_{\nu\mu}(FR) = \begin{cases} 1\ , \text{if the position } (\nu,\mu) \text{ is filled with a fuel rod} \\ 0\ , \text{otherwise} \end{cases}, \tag{21}$$

and f_q is given by

$$f_q = \frac{\left(\frac{2}{n}\right)^2 \sum\limits_{(\nu,\mu)\in q} B_{\nu\mu}}{\left(\frac{1}{\sum\limits_{(\nu,\mu)\in q} \delta_{\nu\mu}(FR)}\right) \sum\limits_{(\nu,\mu)\in q} B_{\nu\mu} \cdot \delta_{\nu\mu}(FR)}\ ,\ q = \text{UL, LL, LR, UR (Fig. 26).} \tag{22}$$

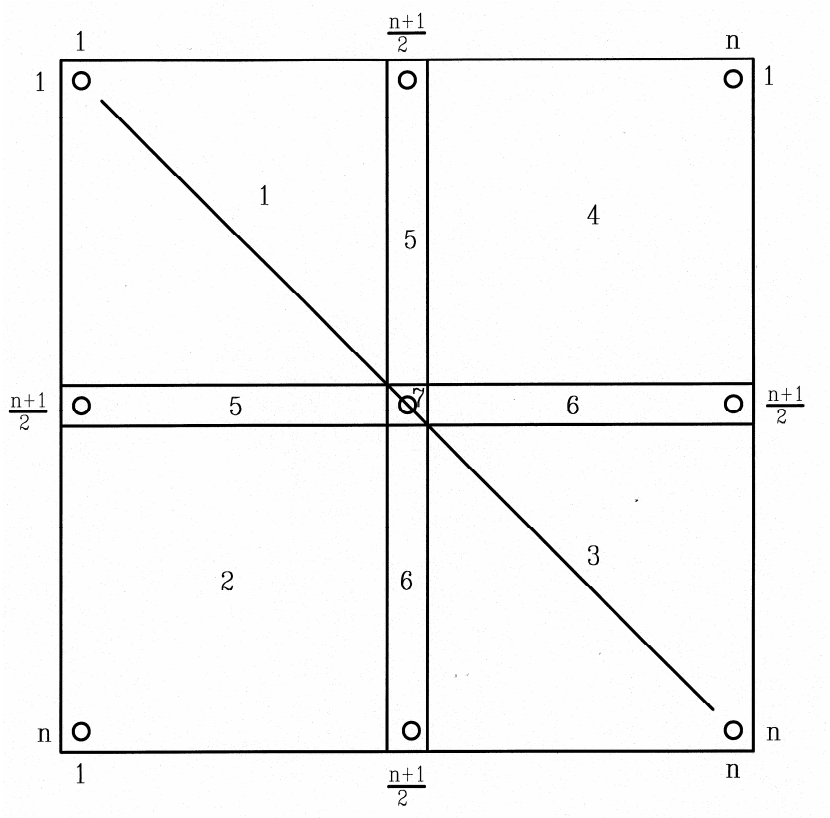

FIG. 27. Division of a n x n fuel rod lattice in 7 zones (for odd numbers n).

Instead of equations (20) through (22) one gets the following relations for n x n fuel assemblies with *odd* number n of lattice positions per row and column:

$$\left(B_{\nu\mu}\right)_Z = f_Z \cdot B_{\nu\mu} \cdot \delta_{\nu\mu}(FR)\ , \tag{23}$$

f_Z given by

$$f_Z = \frac{\left(\frac{2}{n-1}\right)^2 \sum\limits_{(\nu,\mu)\in Z} B_{\nu\mu}}{\left(\frac{1}{\sum\limits_{(\nu,\mu)\in Z}\delta_{\nu\mu}(FR)}\right)\sum\limits_{(\nu,\mu)\in Z} B_{\nu\mu}\cdot\delta_{\nu\mu}(FR)} \quad \text{for z = 1, 2, 3, 4 (Fig. 27),} \tag{24}$$

$$f_Z = \frac{\left[\sum\limits_{(\nu,\mu)\in Z} B_{\nu\mu} + \frac{1}{2}\cdot B_{\frac{n+1}{2},\frac{n+1}{2}}\cdot\left(1-\delta_{\frac{n+1}{2},\frac{n+1}{2}}(FR)\right)\right]\cdot\sum\limits_{(\nu,\mu)\in Z}\delta_{\nu\mu}(FR)}{\left[(n-1)+\frac{1}{2}\cdot\left(1-\delta_{\frac{n+1}{2},\frac{n+1}{2}}(FR)\right)\right]\cdot\sum\limits_{(\nu,\mu)\in Z} B_{\nu\mu}\cdot\delta_{\nu\mu}(FR)} \quad \text{for z = 5, 6 (cp. ibid.),} \tag{25}$$

$$f_Z = \delta_{\frac{n+1}{2},\frac{n+1}{2}}(FR) \quad \text{for z = 7 (cp. ibid.).} \tag{26}$$

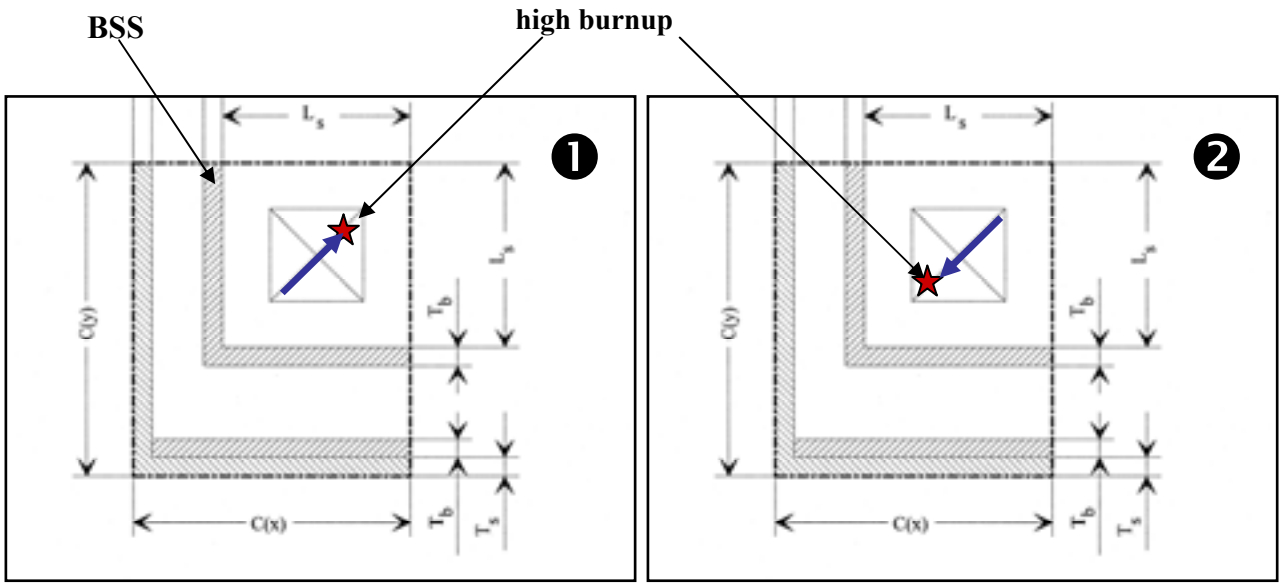

FIG. 28. Storage cell with non-isotropic arrangement of neutron absorbing panels. (Periodic boundary conditions are assumed for all lateral directions)

To study the impact of the assumed horizontal burnup gradients on the reactivity of a spent fuel configuration let us assume a wet storage configuration equipped with storage cells which have a non-isotropic arrangement of neutron absorbing materials. As appears from Fig. 28, each cell is equipped with an unborated stainless steel chevron, a borated stainless steel (BSS) chevron and an additional BSS panel. As indicated in Fig. 28, two cases have been studied:

- (1): It is assumed that the fuel assembly's corner, which has the maximum burnup, faces towards the stainless steel sides.

- (2): It is assumed that the fuel assembly's corner, that has the maximum burnup, faces towards the BSS chevron.

The neutron multiplication factors obtained for these cases were evaluated by means of the methods described in Ref. [8].

As can be seen from Fig. 29, when the maximum burnup corner of the fuel assembly faces towards the BSS sides of the storage cell (case (2)) one gets a slight increase in the reactivity with increasing burnup due to spectrum hardening. The increase in spectral hardening slows down with increasing burnup. Therefore, the reactivity difference remains positive but becomes more or less constant for higher burnups.

When the maximum burnup corner of the fuel assembly is facing towards the unpoisoned steel sides of the storage cell (case (1)) one gets a slightly negative reactivity effect (Fig. 30) because the BSS sides are facing now towards lower burnup, i.e., less spectrum hardening. The BSS panels are therefore more efficient as a neutron absorber than in case of the horizontally averaged burnup.

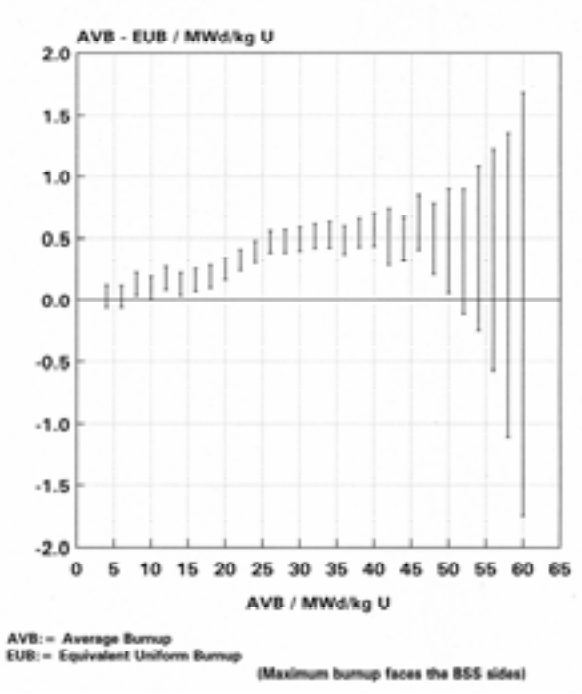

FIG. 29. Storage Region 2 with Standard FA
Impact of Horizontal Burnup Profiles
AVB-EUB = f(AVB/e = 3.0wt.-%).

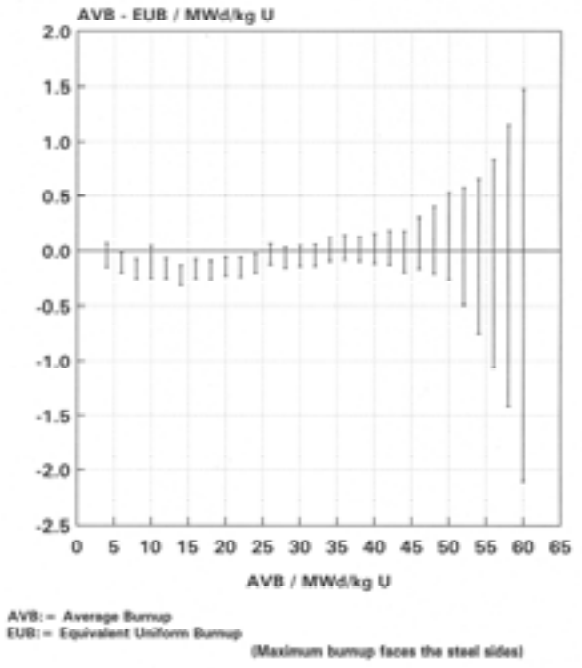

FIG. 30. Storage Region 2 with Standard FA
Impact of Horizontal Burnup Profiles
AVB-EUB = f(AVB/e = 3.0wt.-%).

So therefore, what is observed here is not the reactivity effect of horizontal burnup profiles in itself, but the change of the efficiency of the BSS panels as a neutron absorber due to spectral hardening which increases with increasing burnup. That this effect is observable here is due to the specific design of the storage cells assumed, and the horizontal burnup profiles serve as a means of making this effect observable.

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PROCEDURAL COMPLIANCE WITH THE SAFETY CRITERIA

(Session 2.4)

Double contingency principle and prevention of misloading events

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Abstract. Burnup credit implementation in wet storage, transport and dry storage of spent fuel requires use of fuel handling procedures which ensure prevention of misloading events. In some countries such as Germany misloading events have to be excluded by application of the double contingency principle, i.e. at least two independent, unlikely and concurrent incidents have to happen before a misloading event can occur. If this requirement is met, the misloading event is ruled out and needs not to be considered as a design basis event. For this purpose fuel handling procedures based on technical measures employing software and hardware controls have been set up. The paper on hand presents the basic procedure developed at the German nuclear power plant Neckarwestheim II.

1. Introduction

The outcome of a burnup credit criticality safety analysis of a spent fuel management system (wet storage pool, transport or storage cask, etc.) is a loading criterion the fuel has to meet to be acceptable for loading in the system. Usually this criterion is expressed in form of a curve named as “loading curve” which indicates the minimum required burnup as a function of the initial enrichment of the fuel (Fig. 1).

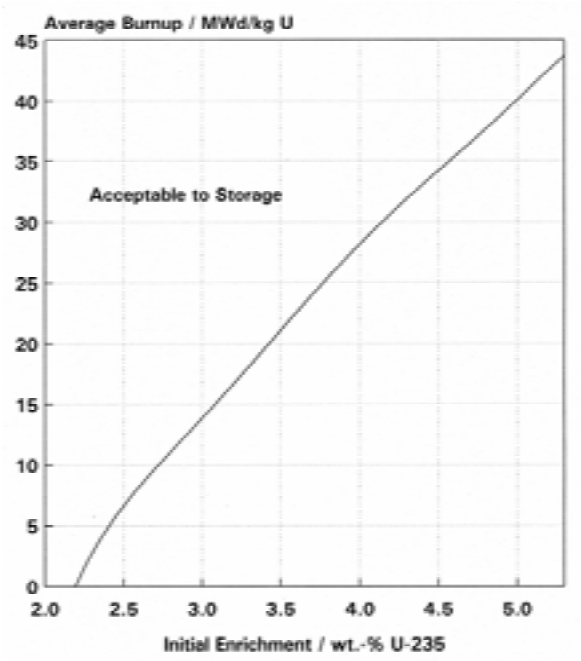

FIG. 1. Example for a loading curve of a spent fuel management system.

By definition, a *misloading error* (misloading event) occurs when fuel which does not comply with the loading criterion of the spent fuel management system of interest is anyhow loaded in the system. The probability and the consequences of such an error are strongly system-specific.

In some countries such as Germany *misloading events have to be excluded by application of the double contingency principle*, i.e. at least two independent, unlikely and concurrent incidents have to happen before a misloading event can occur. If this requirement is met [1)], the misloading event is ruled out and needs not to be considered as a design basis event in the analysis. In the paper on hand rationales for taking this approach are given, and the basic fuel handling procedure developed at the German nuclear power plant Neckarwestheim II for ensuring exclusion of misloading events in compliance with the double contingency principle is described.

2. Double contingency principle and prevention of misloading events

2.1. Basic principles

If a misloading error really occurs, the root cause of such an error is

- either an error in the *burnup information* from the reactor records
- or an *operational* error.

Accordingly, the probability of a misloading event can be reduced by imposing *different and independent* layers of administrative or technical verifications on the generation of the burnup information and the steps of the fuel handling procedure used to load the fuel into the spent fuel management system of interest. Burnup information control procedures and fuel handling procedures based on technical measures employing software controls, measurement devices or hardware controls are preferable to administrative verifications.

[1)] In Germany this requirement is laid down in the burnup credit criticality safety standards DIN 25471 and DIN 25712 [1][2].

2.1.1. Error in the burnup information

Verification of such fuel assembly's basic parameters as identification, initial enrichment and burnup is usually based on information from the reactor records. The information about the burnup of fuel assemblies is generated using measurements of the core flux distribution and reference core power distribution calculations. It has already occurred that a systematic error slipped in the process of generating this information. This type of error can affect the information about an entire core.

A different type of error occurs when a burnup information is inadvertently assigned to a wrong fuel assembly. This type of error in the burnup information is due to a mismanagement of the basic parameters of the fuel assemblies and is hence considered as an operational error in the paper on hand.

2.1.2. Problems with the usual application of the double contingency principle in burnup credit criticality safety analysis

As with any other criticality safety scenario, the *double contingency principle* applies to misloading events. Usually, this principle is applied to a misloading event in such a way that the misloading event is considered as one incident and a second concurrent event does not need to be considered [3] for example). Normally, a misloading error involving only a single fuel assembly is considered, unless there are circumstances which make multiple misloading errors credible (cf. ibid.). However, the administrative and/or technical verifications and controls included in the applied burnup information generation procedure and the employed fuel handling procedure shall be aimed to avoid the possibility of "common mode" errors, which have the possibility to lead to multiple misloading errors. By virtue of the double contingency principle two or more *concurrent* misloading errors do not need to be considered if they can be considered as independent events. But if the same administrative error can lead to a misloading of more than one fuel assembly, multiple misloading has to be considered as one event [2)].

However, even if multiple misloading errors can be excluded by virtue of administrative and/or technical measures incorporated into the procedures employed for burnup information generation and fuel handling there remains one problem which is inherent to the misloading event and distinguishes this event from most of the other accidental events to be considered in criticality safety analysis: If a misloading event does really occur then there is a high probability that the error remains *undetected*. Then any other design basis accidental event that takes place *at a later time* cannot be regarded as a "concurrent" event. In this case application of the double contingency principle *requires* that the misloading event *plus* the additional accidental event have to be considered in the criticality safety analysis. The consequences are system-dependent as briefly discussed in the following sections, but it is already obvious that one of the additional accidental events which have to be considered is a *second misloading event* because of the fact that a system, which cannot withstand the misloading of one fuel assembly, cannot withstand the misloading of more than one fuel assembly.

2.2.Borated wet storage pools

The following cases have to be discussed:

- First case: *One* misloading error occurs during a fuel handling campaign and *is detected* during or at the end of the campaign because of the control procedures implemented during and/or at the end of the campaign: Since the boron concentration of the pool coolant is sufficiently controlled during a fuel handling campaign the presence of the boron can be credited to the extent guaranteed by the plant technical specifications: The misloading event is one incident and, by virtue of the double contingency principle, a concurrent boron dilution event need not be assumed.

[2)] An example for such a case is the Dampierre misloading event that occurred during core loading some years ago: A single error led to a chain of multiple misloading errors.

However, the question may arise whether an additional accidental event due to improper fuel handling (e.g. drop of a fuel assembly) has to be postulated to occur *before* the misloading event is detected. The answer to this question is that the frequency of the controls has to be chosen such that the combination of the additional mishandling event and the misloading event can be excluded by virtue of the double contingency principle [3)].

- Second case: Multiple misloading errors occur during the handling campaign; all these errors are detected at the end of the campaign (at the latest) because of the control procedures implemented at the end of the campaign: There is in principle no difference between this case and the first case; one has only to take more misloaded fuel assemblies into account. In fact, in a criticality safety analysis the assumption of multiple misloading events results in assuming all the storage positions, which are designed for burnup credit, to be loaded with fresh fuel of the maximum allowable enrichment. However, this assumption can lead to the result that the minimum boron content required for maintaining the neutron multiplication factor of the storage facility below the maximum allowable value is greater than the boron content guaranteed by the plant technical specifications. Such a result would mean that the design of these storage positions must not be based on the use of burnup credit.

  However, since these storage positions are designed for burnup credit the fuel handling procedure shall be aimed to avoid the possibility of multiple misloading errors. It is therefore assumed in the following that common mode errors and hence multiple misloading events during a fuel handling campaign can be excluded. So therefore, it is assumed in the following that misloading errors can be considered as independent events.

- Third case: A misloading error occurs during a fuel handling campaign; the error remains undetected until over the end of the campaign: In this case any other incident taking place at a later time cannot be regarded as "concurrent" event. Therefore, by virtue of the double contingency principle, all the combinations of the misloading event and each of the other incidents have to be considered. In particular the following combinations have to be analyzed:

  - combination of the misloading event and a boron dilution event (taking place later on outside of a fuel handling campaign),

  - combination of the misloading event and a different misloading event occurring during a later fuel handling campaign.

  In case of the first combination one has to determine the minimum boron concentration of the pool-water which is required to maintain the neutron multiplication factor of the storage facility below the maximum allowable neutron multiplication factor; and one has to demonstrate for the worst boron dilution transient that there is sufficient time to implement all administrative and/or technical measures which are required to maintain the actual boron content above the minimum required content.

  The second combination brings back the occurrence of a multiple misloading event: The first and the second misloading event have to be considered as one incident which has to be combined again, if the second misloading event remains undetected, with each of the incidents taking place later on, in particular with

  - a boron dilution event taking place later on outside of a fuel handling campaign or

  - a third misloading event occurring during a later fuel handling campaign.

[3)] This does not exclude the case that a drop of a fuel assembly can result in a misloading event.

It is obvious that this finally results in a multiple misloading scenario with all the storage positions assumed to be loaded with fresh fuel assemblies having the maximum allowable enrichment. And a combination of this scenario and a boron dilution event obviously leads to the result that burnup credit cannot be used for the design of the storage facility of interest.

It follows therefore that *it is necessary to reduce the probability of occurrence of a misloading event to a negligible figure* so that the misloading event does not need to be considered as a design basis event. This is just achieved by applying the double contingency principle to the misloading event in such a way that at least *two independent, unlikely and concurrent incidents have to happen before a misloading event can occur*.

2.3. Unborated wet storage pools

Since a loading curve is generated by a reactivity equivalence relation with the maximum allowable neutron multiplication factor [4)] it is self-evident that any misloading event has to be excluded if the loading curve is evaluated for normal operation conditions. If a misloading event cannot be excluded then the loading curve has to be based on the assumption of a misloading event. However, to keep a loading curve based on this assumption practicable the storage facility has to be designed such that the economical benefit of using burnup credit usually turns out to be small since each of the storage cells of the fuel assembly storage racks has to be designed against the misloading event.

In addition, since a misloading event has a high probability of remaining undetected one is confronted with the same problem as was already discussed in the previous section: One has to consider the possibility of a chain of non-concurrent misloading events resulting, step by step, in a multiple misloading event. The consequence is that burnup credit cannot be used. So, if burnup credit shall be used any misloading event has to be excluded by applying the double contingency principle in such a way that at least two independent, unlikely and concurrent incidents have to happen before a misloading event can occur.

2.4. Transport casks (dry transport of spent fuel)

For dry transports of spent UO_2 fuel with an initial enrichment not greater than 5 wt.-% U-235 burnup credit, if used, is only needed if

- the transport casks are loaded in pure water and/or
- re-flooding of the casks with pure water under normal or accident conditions has to be taken into account.

In Germany as well as in some other countries re-flooding of a cask under accident conditions has to be considered as a *design basis* event. Therefore, even if multiple misloading events can be excluded for a transport cask the presumption of only one misloading error results in a considerable reduction in the economical benefit of using burnup credit, because each and every fuel storage cell inside the cask has to be designed against the misloading event. So, the use of burnup credit is of interest only then, if any misloading event can be excluded by applying the double contingency principle in such a way that at least two independent, unlikely and concurrent incidents have to happen before a misloading event can occur.

2.5. Interim storage in dry storage casks

The same conclusion as drawn at the end of the previous section goes for interim storage in dry storage casks if re-flooding of the cask with pure water under accident conditions has to be considered.

---

4) Cf. Reference [4], section 2.4.

As follows from Ref. [2], in Germany re-flooding of a dry storage cask under accident conditions has to be analyzed as a design basis event.

2.6. Résumé

As shown in sections 2.2 through 2.5, straightforward application of the double contingency principle leads in any case to the conclusion that the misloading event must not be a design basis event. The misloading event has to be excluded by application of the double contingency principle in such a way that at least *two independent, unlikely and concurrent incidents have to happen before a misloading event can occur*. The double contingency principle has therefore to be imposed on each step of the procedure employed for generating the burnup information and on each step of the procedure used for planning and implementing loading of the fuel into the spent fuel management system of interest. As already mentioned, these requirements are laid down in the German burnup credit regulatory safety codes DIN 25471 and DIN 25712 [1][2], respectively.

In the following section a basic fuel handling procedure which fulfills these requirements is presented. This procedure is characterized as "basic" not because it is the only possible procedure but because it is a *necessary and sufficient* procedure to meet the requirement to exclude the misloading event as a design basis event. This procedure based on technical measures combining software and hardware controls has been developed under the responsibility of the German nuclear power plant Neckarwestheim II and is therefore named as "the Neckarwestheim fuel handling procedure".

3. The Neckarwestheim fuel handling procedure

The Neckarwestheim fuel handling procedure has been developed to ensure prevention of errors in all the handling operations inside a nuclear power plant at the *planning and the operation stage*. Since the attention of this IAEA technical meeting is mainly focused on the use of burnup credit and hence, inter alia, on ensuring procedural compliance with a burnup credit loading criterion the handling procedure as applied, so to speak, "at home", i.e. in the German Convoy Series nuclear power plant Neckarwestheim II (GKN II), is described in the paper on hand.

3.1. Description of the procedure

To prevent errors at the *operation stage* an interlock logic protected against malfunction is installed in the control unit of the fuel loading machine hindering this machine from handling operations which are not laid down in a *handling sequence plan* established by an authorized person and checked by an empowered person according to the quality assurance requirements to be applied [1][2][4]. Hindering the loading machine means *blocking* the functions "lifting" and "lowering" of the main hoist of the loading machine.

The handling sequence plan includes the sequence of all the handling operations which are part of *one and the same* well defined fuel handling *action*. With respect to the application of burnup credit the following fuel handling actions are of particular interest:

- *Fresh fuel transfer*: Transfer of unirradiated fuel from the new fuel store to region I of the storage pond (Fig. 2),
- *Reshuffling*: Reshuffling of the core,
- *Transfer to region II*: Transfer of spent fuel from region I to region II of the storage pond (Fig. 2),

- *Cask loading*: loading of a spent fuel transport/storage cask [5)].

Fuel handling operations laid down in a handling sequence plan can be executed *then and only then*, if the handling sequence plan is installed in the control unit of the fuel loading machine by an authorized person.

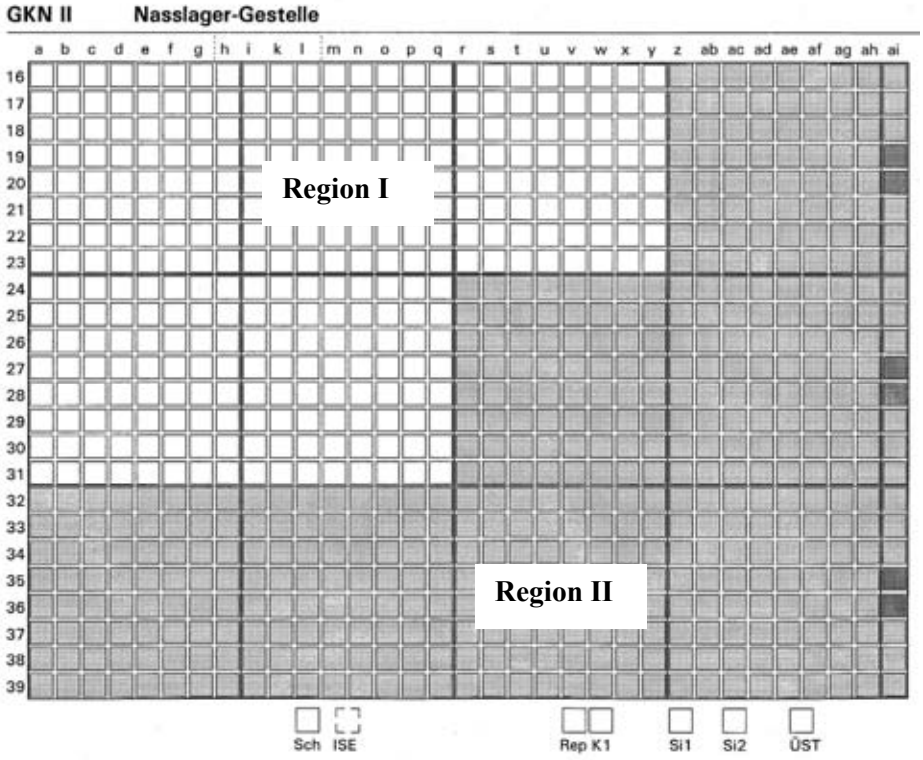

FIG. 2. Neckarwestheim wet storage pond: Application of burnup credit has made it necessary to divide the storage pool into two storage regions [1].

Region I with storage racks designed to accommodate fuel which is at the maximum reactivity point of its life, and region II with storage racks designed to accommodate fuel for which burnup credit is taken.

3.1.1. Principles of the procedure applied to prevent misloading events

To prevent misplacement of a fuel assembly that does not meet the region II loading criterion (loading curve) into a region II storage cell the Neckarwestheim fuel handling procedure is based on the following principles:

- Unirradiated fuel is stored in *only one* of the five region I storage racks (Fig. 3). It is impossible to include calls of the loading machine at storage positions of this particular rack in a handling sequence plan for an action "transfer to region II".

- During the action "reshuffling" (reshuffling of the core) only the storage region I is available to fuel handling operations (Fig. 4).

[5)] In the following the attention is mainly focused on the procedure employed to prevent misloading events in region II of the wet storage pond. However, the principles of this procedure apply to cask loading too; but because of some special requirements to the fuel handling procedure during cask loading some additional steps are required which assure a definite assignment of a fuel assembly to be loaded in the cask to the location inside the cask where the assembly is destined for. These additional steps are described in detail in Reference [4].

- Except for the action “transfer to region II” the *storage region II is always closed* to relocating a fuel assembly to a region II storage cell. To meet this requirement the following measures are taken:

  - In addition to the blocking of the loading machine against handling operations that are not included in a handling sequence plan the loading machine is *blocked with the aid of a key switch* hindering the machine from transferring fuel to a region II storage cell.

  - The blocking of the loading machine is ensured through *the operational and the fail-safe control* of the machine.

  - The key is maintained under *positive administrative control* in the safety control room of the nuclear power plant.

  - Issue of the key is only effected on presentation of that *work order* in written form which is required for an action “transfer to region II” and *approved by an authorized person*.

  - The blocking of the loading machine is raised *only for the duration* of an action “transfer to region II”. *Immediately* after completion of such an action the blocking of the loading machine is switched on again.

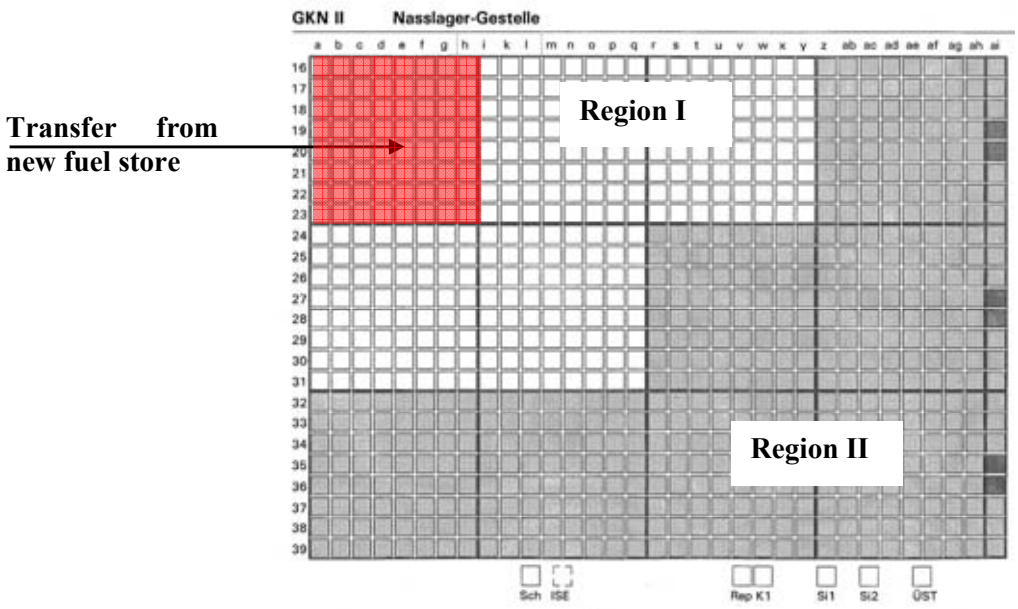

FIG. 3. Principles of the Neckarwestheim fuel handling procedure.

Unirradiated fuel is stored in only one of the five region I storage racks. Calls of the loading machine at storage positions of this particular rack cannot be included in a handling sequence plan for an action “transfer to region II”.

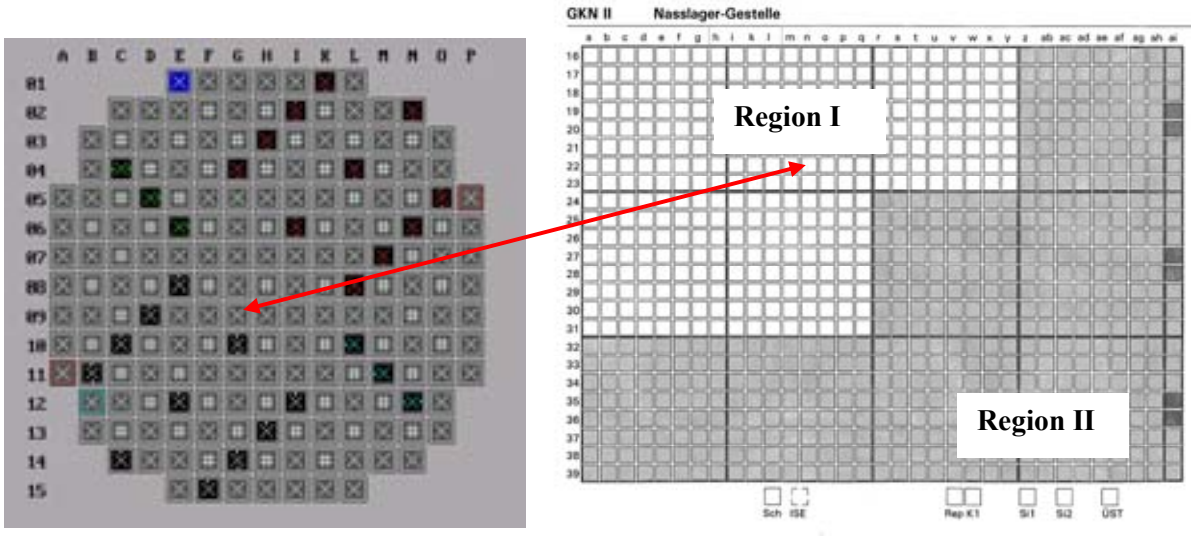

FIG. 4. Principles of the Neckarwestheim fuel handling procedure.

During reshuffling of the core only the storage region I is available to fuel handling operations.

- In a handling sequence plan for an action "transfer to region II" only fuel assemblies can be included which *comply with the region II loading criterion*. Generation of the handling sequence plan includes therefore, as described in more detail in the next section,

  - evaluation of the reactor records and

  - application of an appropriate interlock logic scheme that discriminates the fuel assemblies which do not comply with the loading criterion.

3.1.2. Generation of a fuel handling sequence plan

Each handling sequence plan is generated with the aid of the computer code system ALFA. This code system serves for proper *identification, management and documentation* of locations and relocations of fuel assemblies and internals within the plant (Fig. 5)[6]. ALFA has access to all pertinent data files such as the *reactor records* for instance to get all the data required such *as names, initial fuel enrichments and topical burnups* of the fuel assemblies. So, with ALFA any handling of fuel assemblies or internals can be simulated and hence planned in compliance with the quality assurance requirements to be applied [1][2][4].

To prevent errors at the *planning stage* ALFA distinguishes between the different fuel handling actions by applying appropriate *interlock logic schemes* as required by the principles listed in section 3.1.1. In particular, except for the action "transfer to region II"[7] the *storage region II is always closed to planning of any fuel handling operation*. Accordingly, when the action "fresh fuel transfer" is called in ALFA, for instance, *it is impossible* to call the action "transfer to region II" in ALFA. Therefore, no fuel handling operation belonging to an action "transfer to region II" can be included in a handling sequence plan for an action "fresh fuel transfer". Likewise, if the action "reshuffling" is called in ALFA *it is impossible* to call the action "transfer to region II". So therefore, no fuel handling operation belonging to an action "transfer to region II" can be included in a handling sequence plan for an action "reshuffling" (Fig. 6). Since the loading machine is blocked against handling operations not

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[6] ALFA has been generated and is maintained by Dr. Hans-Georg Johann, one of the authors of the paper on hand.

[7] The code ALFA is used in 11 European nuclear power plants. Most of these plants do not use burnup credit. Accordingly, the versions of ALFA applied in these plants do not include the action "transfer to region II".

included in a handling sequence plan and since the loading machine is blocked by means of a key switch against any access to region II fuel storage cells, transfer of any fuel to region II cannot take place during any fuel handling action different from the action “transfer to region II”.

When the action “transfer to region II” is called in ALFA the particular region I storage rack which is designed for accommodating unirradiated fuel (cp. section 3.1.1) is *closed* to the planning of fuel handling operations; and hence, no transfer of unirradiated fuel can be included in a handling sequence plan for an action “transfer to region II”. The storage positions of the particular region I rack are visually differentiated by ALFA from the remaining storage region I positions and the region II positions to indicate that any fuel transfer from the particular region I rack is *unacceptable* for ALFA (Fig. 7).

ALFA screen shots showing certain loadings of the core and the wet storage racks. (In the storage cells of that part of region II, which is used for fuel from GKN I at the present time, adapter channels are inserted because the GKN I fuel assemblies have a smaller cross section. The presence of these adapter channels makes it impossible to insert GKN II fuel into these storage positions. Due to the interlock logic schemes used in ALFA it is impossible to transfer GKN II fuel into a storage cell reserved for GKN I fuel or to transfer GKN I fuel into a storage cell reserved for GKN II fuel.)

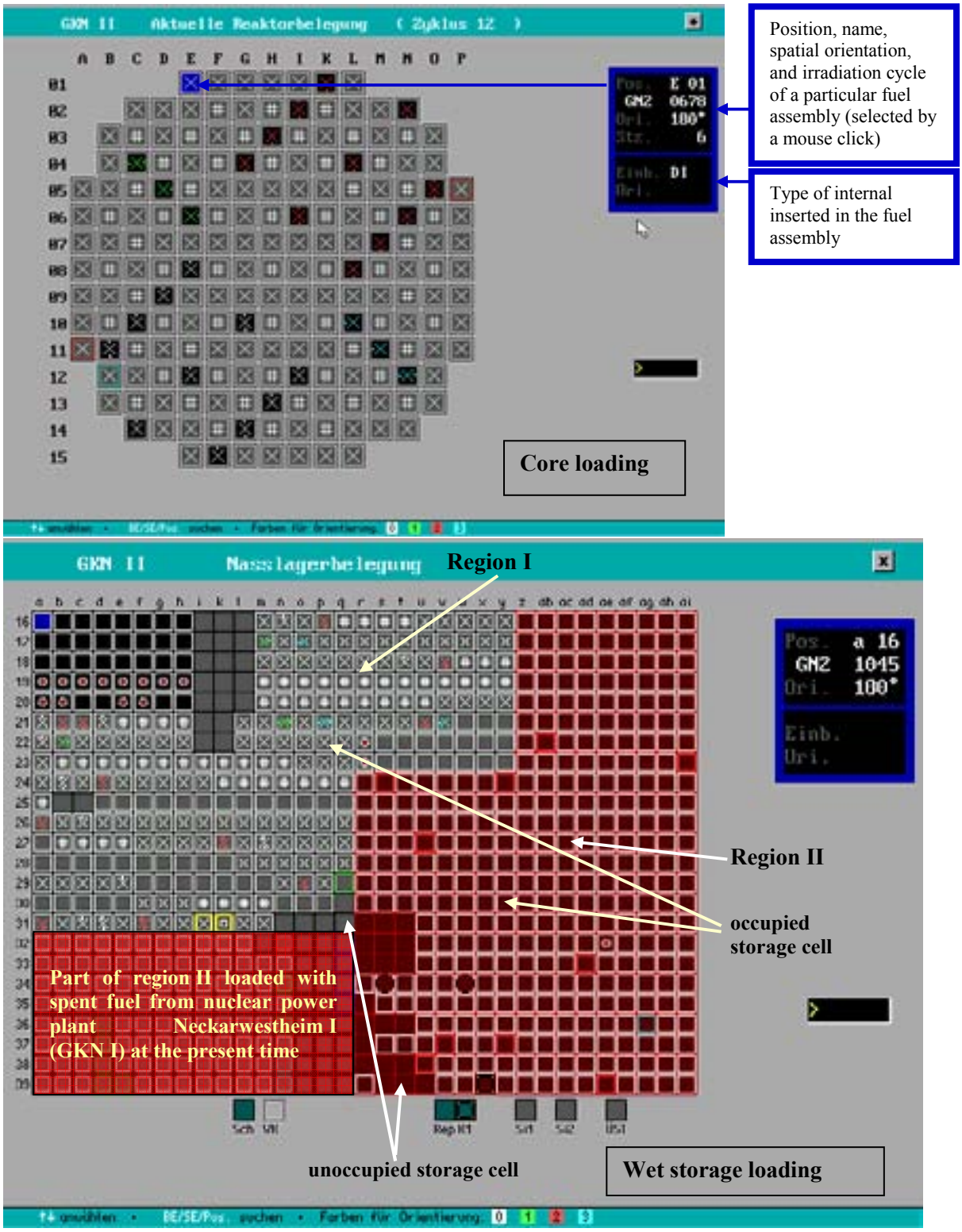

FIG. 5. Examples for the identification and documentation of the locations of fuel assemblies and internals within the nuclear power plant Neckarwestheim II (GKN II) with the aid of the code system ALFA.

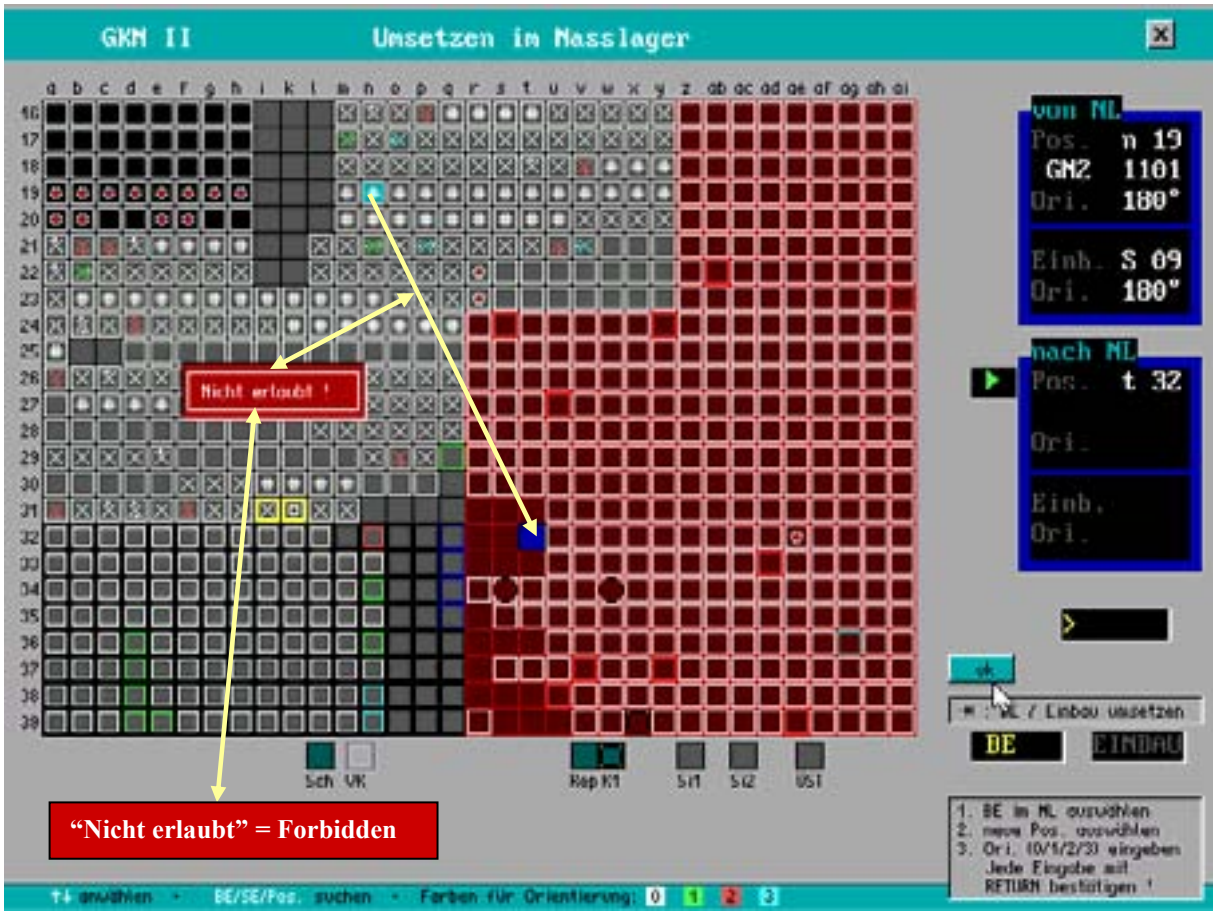

FIG 6. Example for the interlock logic schemes applied by ALFA to the planning of fuel handling operations.

The ALFA screen shown belongs to an action which is *different* from the action "transfer to region II". The region II storage cells (reserved for GKN II fuel, Fig. 5) are therefore presented in red color to indicate that any fuel transfer to region II is forbidden. If one still tries to include a transfer to a region II storage cell (e.g. as shown, from the region I position n 19 to the region II position t 32) then this transfer is *rejected* by ALFA ("Nicht erlaubt"=forbidden) and is therefore *not* included in the handling sequence plan generated by ALFA under the chosen action (e.g. the action "reshuffling").

To all the fuel assemblies which are located in the remaining region I storage cells ALFA applies, under the action "transfer to region II", the *region II loading criterion*[8]. Those fuel assemblies, which meet the loading criterion, are visually differentiated by ALFA from those ones, which do not comply with the loading criterion. As demonstrated in Fig. 8, it is *impossible* for ALFA to accept a transfer of a fuel assembly to region II which does not comply with the region II loading criterion.

[8] The region II loading criterion is often given in form of a loading curve (as illustrated in Fig. 1) but not always. Dependent on the region II storage rack design, it can be sufficient to divide the fuel assemblies into two groups: One group with lower initial enrichments for which no burnup credit is required, and a second group, starting with a certain initial enrichment (confirmed by criticality safety analysis) for which a fixed average burnup value (confirmed by criticality safety analysis) is credited. In other words, in this case the loading curve is expressed in form of a step function. If the fuel assemblies present in a nuclear power plant do not differ so much in their initial enrichment values it may be simpler to stipulate the specified average bunup value for all the fuel assemblies to be loaded into region II whatever their initial enrichment might be.

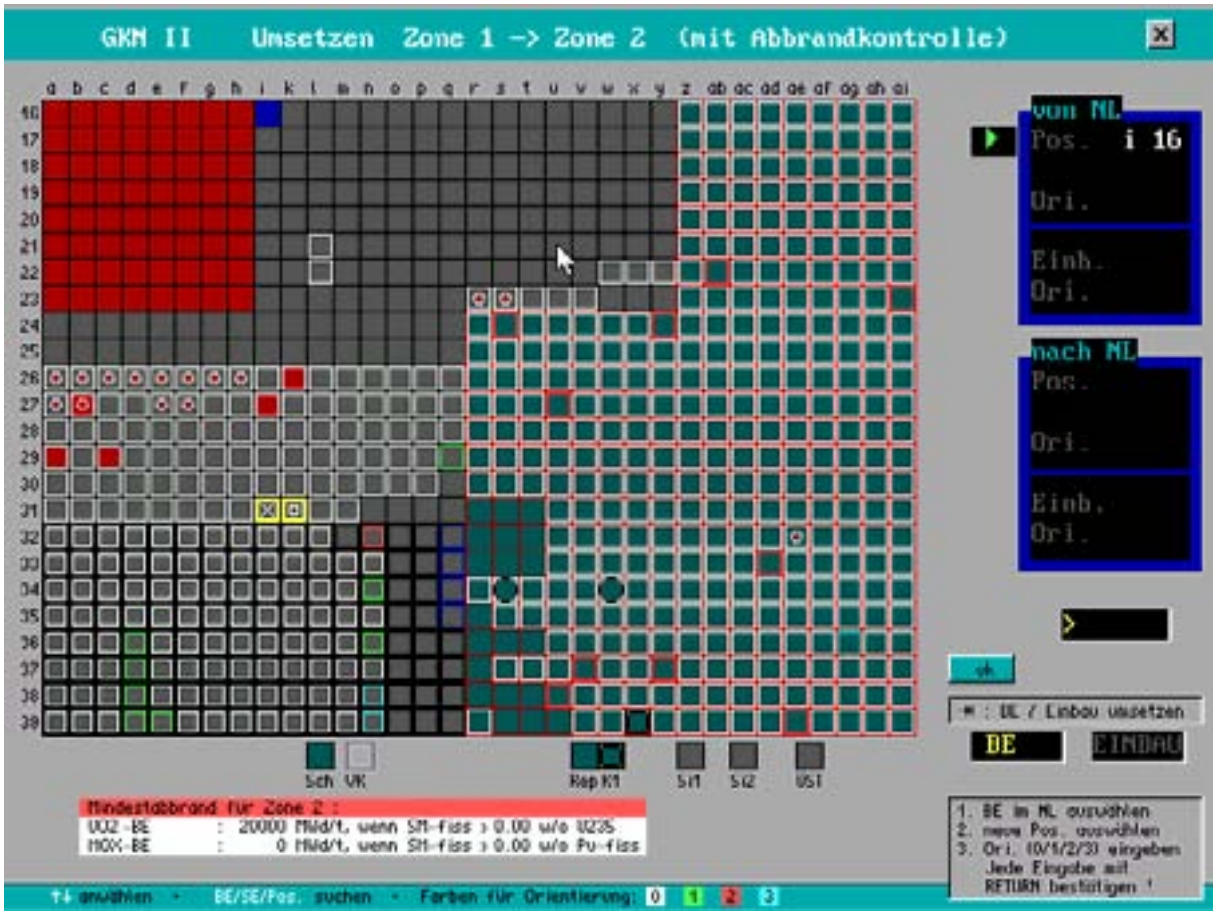

FIG. 7. Example for the interlock logic schemes applied by ALFA to the planning of fuel handling operations.

The ALFA screen shown belongs to the action “transfer to region II”. The region II storage cells (reserved for GKN II fuel, Fig. 5) are therefore presented in green color now, whereas the storage positions of the particular region I rack designed for accommodating unirradiated fuel (cf section 3.1.1) are shown in red color to indicate that any fuel transfer from this region I rack is *unacceptable* for ALFA under the action “transfer to region II”. ALFA applies to all fuel assemblies placed in the remaining region I storage cells the region II loading criterion. Fuel assemblies which do not comply with this criterion are shown in red color (cf. for instance region I storage position c 29) to indicate that transfer of these fuel assemblies to region II is *unacceptable* for ALFA. If one still tries to transfer such a fuel assembly to a region II storage cell then this trial is *rejected* by ALFA, as demonstrated in Fig. 8.

For each of the actions to be planned the code system ALFA has interlock logic schemes *appropriate to prevent handling errors*. A fuel handling operation which is rejected by ALFA is *not* included in the handling sequence plan for the action of interest. Fuel handling operations which are not included in the handling sequence plan *cannot be executed* because this plan is installed in the control unit of the fuel loading machine.

A handling sequence plan can be generated and authorized *by empowered persons only*. An *authorized* handling sequence plan can be installed in the control unit of the fuel loading machine *by an empowered person only*. The handling sequence plan *cannot be executed* until

- it is printed out and signed by the persons who generated and authorized this plan and
- all the other working orders and permits required are given.

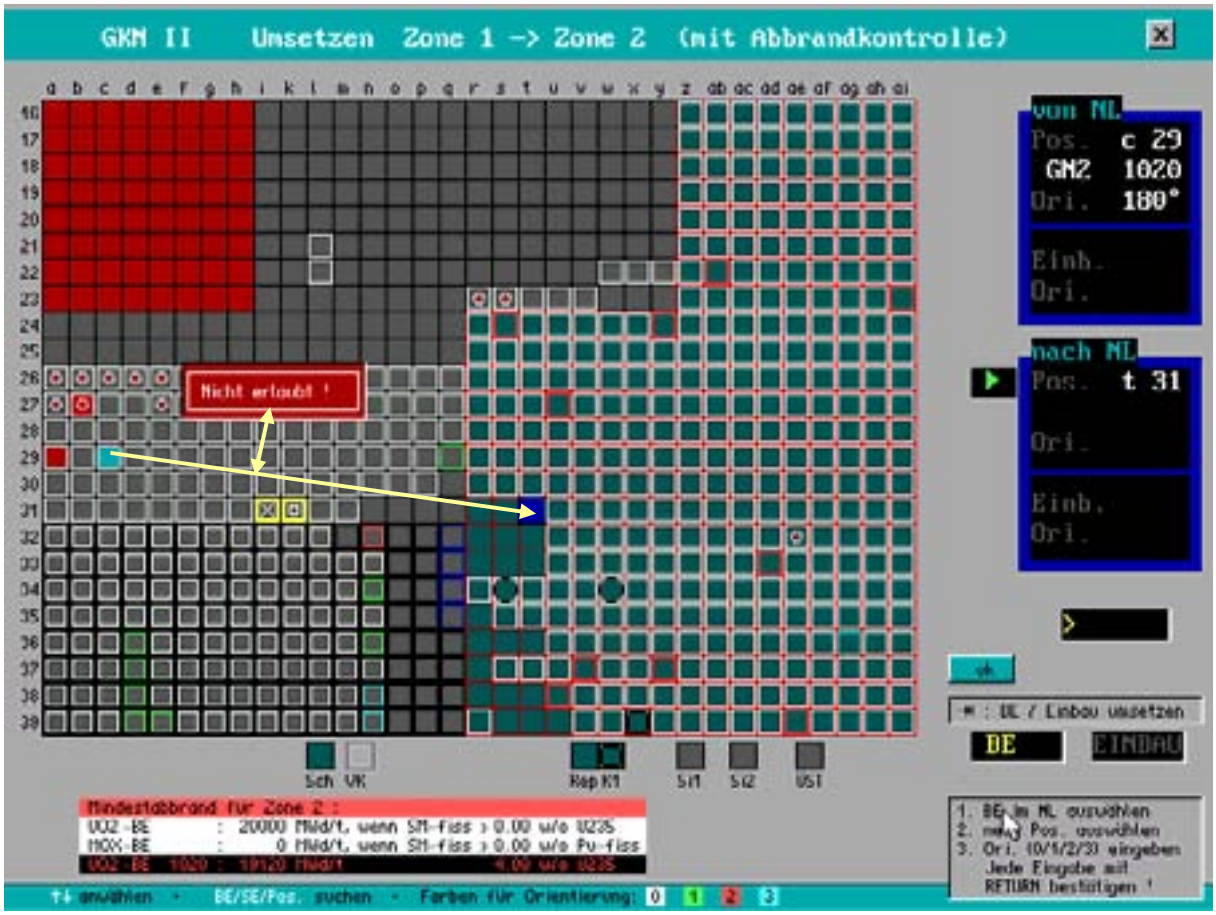

FIG. 8. Example for the interlock logic schemes applied by ALFA to the planning of fuel handling operations under the action "transfer to region II".

A transfer of the fuel assembly located at the region I position c 29 to the region II position t 31 (or to any other region II position) is *rejected* by ALFA ("Nicht erlaubt"=forbidden) and remains therefore excluded from the handling sequence plan generated by ALFA under the chosen action, because the fuel assembly at position c 29 does not meet the region II loading criterion (the fuel assembly has an average burnup of 19.12 MWd/kg U whereas the minimum required burnup is assumed to be 20 MWd/kg U).

3.2. Implementation of the procedure

In case of an action "*transfer to region II*" or an action "*cask loading*" (if burnup credit is used for the cask to be loaded[9]) first of all it has to be ensured in compliance with the quality assurance requirements to be applied [1][2][4].

- that the burnup data of the fuel assemblies stored in region I are updated and
- that the quality assurance measures which have to be applied to the updated data [4] are actually carried out[10].

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[9] As already mentioned, in case of the action "cask loading" some additional steps are required. These steps are described in detail in Ref. [4].

[10] By the way, if it is postulated that an update of the burnup data of the fuel assemblies stored in region I has been omitted, then the recorded burnup values are lower than the actual burnup values of these assemblies. Therefore, application of the loading criterion can only result in a rejection of some fuel assemblies which are actually acceptable already.

Then, for all actions, the following steps have to be taken:

- Generation of the handling sequence plan with the aid of the code system ALFA by an *authorized* person.
- Checking of the handling sequence plan by an *authorized person not involved* in the generation of the plan. *Without an authorized handling sequence plan any fuel handling operation is forbidden.*
- Installation of the *authorized* handling sequence plan in the control unit of the fuel loading machine by an *empowered* person. *Without installation in the control unit of the loading machine no fuel handling operation is possible.*
- Issue of *authorized* working orders and permits for performing the fuel handling operations. *Without these orders and permits no fuel handling operation is allowed.*
- In case of an action "transfer to region II": *Issue of the key* necessary to raise the blocking of the loading machine for region II.
- *Written* confirmation of the execution of the handling operations. In case of an action "transfer to region II": *Immediate return of the key and blocking of the loading machine for region II.*
- Update of the ALFA data files documenting the actual loading of the reactor core and/or the storage facilities.

Due to its principles, interlock logic and blocking schemes as well as its implementation the Neckarwestheim fuel handling procedure ensures prevention of a misloading event in compliance with the conclusion drawn in section 2.6. Except for the actions related to the use of burnup credit and the blocking of the loading machine this procedure has been already employed since approximately 20 years in many European nuclear power plants. The actions related to burnup credit and the blocking of the loading machine are in use since more than 5 years now in all German nuclear power plants applying burnup credit to their wet storage ponds.

The Neckarwestheim fuel handling procedure demonstrates that a combination of a considerable number of different software, hardware and administrative controls is required to be able to exclude the misloading event as a design basis event by virtue of the double contingency principle. But even if the number of controls is considerable the combination of these controls as realized in the Neckarwestheim fuel handling procedure is necessary and sufficient to constitute a solid and robust procedure capable of preventing misloading events in compliance with the conclusion drawn in section 2.6.

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A conservative approach to consider burnup credit in criticality studies

Keywords: burnup credit, axial profile, fission products, conservative approach

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Abstract. In order to avoid criticality risks, several facilities operating spent PWR fuels have been designed considering the fuel as fresh. This choice has obviously led to considerable safety margins.

In the early 1980's, a method allowing to consider the changes in the fuel composition during the depletion with some very pessimistic hypothesis (only actinides were considered and the amount of burnup used in the studies was equal to the mean burnup in the 50-least-irradiated centimeters) was accepted by the French Safety Authorities.

As many facilities still want to optimize their processes (e.g. transport, storage, fuel reprocessing), the main French nuclear companies, researchers and IRSN set up a working group in order to study the way to take into account Burnup Credit in the criticality calculations, considering some fission products and a more realistic axial profile of burnup.

The article presents the different questions that have been raised while studying the possibility to set an axial shape of the burnup and the absorption of the neutrons by some of the fission products. However the questions would also apply to a method with actinides only.

They are sorted in the following topics: (i) which axial shape of the burnup should be considered, (ii) which composition of the irradiated fuel should be considered for every axial sub-part of the assembly, and finally, (iii) which confidence can we have in the criticality codes used to determine the k-effective. The paper highlights the questions raised during the work of a French working group on Burnup Credit but doesn't present a recommended approach. This approach is to be issued by the end of the year.

1. Introduction

Up to the 1980's, the nuclear facilities dealing with spent fuel were designed, regarding criticality aspects, with the assumption of fresh fuel. This assumption led to considerable safety margins.

In the early 1980's, in order to use the existing devices at La Hague reprocessing plant for some irradiated UO_2 fuel initially enriched at 4.4% (this enrichment was higher than the highest enrichment of 3.1% considered at the designing stage), a method was proposed by COGEMA to enable them to consider a certain amount of burnup in the criticality studies. However some pessimistic assumptions were made to guarantee some safety margins:

- only uranium and plutonium were considered after the depletion of ^{235}U and ^{238}U during the irradiation,
- the amount of burnup used in the criticality studies was lower than the value reached in the 50-least-irradiated-centimeters,
- the value of the mean burnup in the 50-least irradiated-centimeters was checked by a measurement.

The HTC experiments performed with Apparatus B in Valduc have supported the validation of the calculations using this method. Those subcritical experiments [1] involved fuel pins, representative of a fuel initially enriched at 4.5% and irradiated at 37.5 GWd/t, manufactured only with actinides. The pins arrays were arranged in different types of configurations representative of reprocessing, storage and transport.

This actinide-only method was accepted by the French safety authorities and was, afterwards, used for the transport of irradiated fuel as well as at the designing stage of the UP3 and UP2-800 at La Hague reprocessing plant.

But, since the initial enrichment increased and the needs of interim storage of irradiated fuel grew, it became necessary for the nuclear industry to reduce the conservatisms due to the very pessimistic hypothesis of this actinide-only method.

On an other hand, studies [2] have showed that even if a more accurate consideration of Burnup Credit (BUC) reduces the margins in terms of k-effective (e.g., a maximum allowable mass determined with BUC corresponding to a k-effective equal to the safety criteria will be greater than the allowable mass calculated without BUC), it may globally increase the safety, leading to less waste (e.g. less Gd will be needed in the solution, fewer transport operations…).

For that purpose, a working group was formed in 1997, gathering most of the French nuclear companies and the technical support of the safety authorities, to analyze different propositions of introduction of some fission products plus a more realistic axial profile of burnup in criticality studies.

This paper presents the different questions that need to be answered in order to ensure a safe evaluation of the Burnup Credit in criticality studies.

2. General questions that should be raised

During the last decade, the work achieved in France for UOx irradiated in PWR focused on three new points related to:

- the neutron absorption of some actinides and fission products,
- a more realistic description of the burnup axial profile,
- a more complex k_{eff} calculation that take into account the particular distribution of the isotopes in the irradiated fuel.

As those new considerations will reduce the conservatisms, it seemed necessary to carefully study the validity of the assumptions made to:

- define the axial profile of the burnup,
- determine the composition of the irradiated fuel,
- compute the criticality simulation (particularly regarding the knowledge of the cross-sections of the isotopes that are being taken into account).

For that purpose, the following paragraphs discuss all the above questions; those assumptions have to ensure a global conservatism of the method.

2.1. Considerations related to the profile of burnup

The fuel burnup has axial and horizontal gradients due to the flux distribution during irradiation. This flux shape is mainly related to:

- neutron leakage at the top and the bottom of the assembly and, in some cases, the presence of partially inserted control rods,

- radial leakage of the neutrons, (which depends on the environment of the assembly and its position in the reactor during irradiation).

In some cases the burnup distribution can have significant impact on k-effective and needs to be assessed [3]. Indeed studies pointed out that the assumption of the mean burnup for an irradiated assembly can lead to an under-estimation of the value of the k-effective [4].

Until now, in France, the value of the burnup applied on the whole length of the assembly, was equal to the mean value of the 50-least-irradiated-centimeters of the assembly.

This assumption was very conservative for an "actinide only methodology" and the geometrical configurations studied: the mean burnup in the criticality studies is only 78% of the mean burnup for a standard profile (e.g. 34 GWd/t will be applied to the whole length of the assembly whereas the real mean burnup would be equal to 44 GWd/t).

In order to consider a more realistic profile, some experimental data based on both measured and calculated data have been used:

- more than 3 000 assemblies irradiated in EDF reactors have been measured at La Hague - COGEMA reprocessing plant and were examined,
- different types of profiles were calculated in order to determine a penalizing one.

Those two studies have highlighted the two following points.

- The systematic use of a penalizing profile is very pessimistic (the value of the keff is increased by 12% in Δk compared to a "mean profile") [1][5]. Additionally, the definition of a penalizing profile is tightly linked to the reactor "management".
- The La Hague measurements pointed out that most of the EDF assembly's profiles were quite similar (due to the operating condition). The statistical study of these profiles could provide a conservative axial profile (for ''most of the profiles'' already measured). Figure 1 below gives an example of a conservative axial profile determination. Each point of this profile has a burnup value lower than the existing profile value.

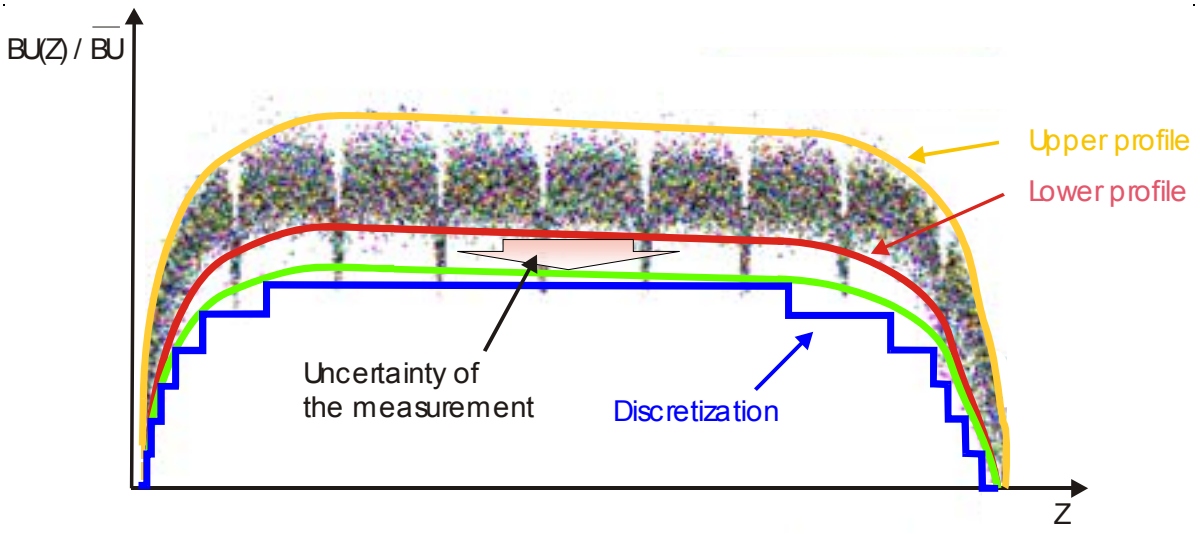

FIG. 1. Example of a conservative axial profile determination.

However, the basic idea is that the conservatism of the axial profile used in the studies will have to be demonstrated. Therefore:

- If the profile have been calculated, the conservatism of this calculation will have to be guaranteed;

- If the profile have been determined by measurements, the user will need to assess the uncertainties due to the method used for measurement (measurement devices uncertainties and validation of the method of measurement based on calculations). The influence of the irradiation history on the measured isotopes concentrations (e.g. ^{154}Eu, ^{134}Cs, ^{137}Cs and ^{148}Nd) may be studied to confirm the measurement uncertainties.

Then, in both cases, it is necessary to consider whether the axial shape used in assessment is suitably conservative relatively to the range of existing profiles.

For that purpose, different approaches are possible, for example:

- Check the profile by a measurement on each assembly; the measured burnup will be greater (at different points along the fuel assembly) than the one used in the criticality studies),
- Define a conservative profile for each operating condition considering the database available, and check by a statistical measurement (every N profile) that there is no deviation of the profiles due to the operating condition of the PWR reactor evolution.

Particular care should be taken when control rods can be partially inserted in the fuel assemblies.

Concerning the radial/horizontal gradients of burnup, some calculations have been carried out; they give the gradients as a function of burnup [6]. Measurements made at La Hague gave indications of values of radial/horizontal gradients as a function of burnup [7] for a large amount of fuel assemblies.

Depending on the case studied this effect may have to be considered, for example with:

- deterministic approach considering the less-irradiated faces close together,
- or, probabilistic approach.

For the criticality calculations, the irradiated fuel assemblies need to be divided in several parts. A first proposal would be to consider, for each part the minimum value of the burnup (BU) in this part. Indeed if the value of BU is equal to the mean value, this approach might be unsafe in some cases where the gradient of BU has a significant impact due to the length of the axial part considered.

The number of zones used for the profile will be determined to give a value of the k_{eff} which is not too conservative: a number around 10 could be sufficient if the part-lengths are adequately chosen (e.g. each part corresponds to a given delta BU). However, if important margin exists for the configuration studied, there is no need to consider a high number of zones (the k_{eff} will decrease as N increases).

2.2. Actinide and fission products concentration in depleted fuels

When the fuel has been divided into several parts, each having a given BU, the concentrations of actinides and fission products have to be determined.

For criticality studies, it is necessary to determine which nuclides are to be included in the evaluation of k-effective (all actinide nuclides, all fission products?); the list depends on the fissile/non fissile properties of the nuclides and on their impact on criticality (cross-section, concentration, stability, non-volatility); it also depends on the accuracy the depletion code can predict their concentration. Moreover, the application context may determine the need to consider a wide range of nuclides. The list of selected nuclides has to be in agreement with the criticality bounding configuration studied (normal and accidental configurations). This list could be very different for storage, transport [8], reprocessing and disposal [9] applications.

For UOx PWR Burnup Credit, the 6 following fission products ^{149}Sm, ^{152}Sm, ^{103}Rh, ^{143}Nd, ^{133}Cs and ^{155}Gd account for 50% of the absorption of all fission products. This list can be extended. For example, 9 other fission products are considered by the OECD BUC working group (^{99}Tc, ^{145}Nd, ^{153}Eu, ^{95}Mo, ^{147}Sm, ^{150}Sm, ^{151}Sm, ^{109}Ag and ^{101}Ru).

Then the evaluation of the fuel composition depends on:

- the fresh fuel characteristics,
- the irradiation and cooling history,
- the depletion code used.

Therefore, the use of Burnup Credit requires:

- the validation of the depletion code with measured post irradiation fuel composition,
- the definition of appropriate values for the parameters of the fuel irradiation.

The depletion codes must be validated with samples of irradiated fuels. Due to the complexity of the depletion calculations, care must be taken for the different options used for the validation (definition of time intervals for re-calculation of the cross-sections during the depletion calculations, self-shielding…). Moreover, it can be noticed that the burnup is determined by the calculated ratio of some isotopes concentrations (e.g. Nd); thus the validation of the method will be tightly linked to the chosen "indicator of burnup" and its own depletion-validation.

Up to now, the depletion codes used in France are CESAR [10] or DARWIN [11] Codes. They have been validated on an experimental basis [12]. This validation relies on comparisons between calculated values of the concentrations and measured ones [13]. Those comparisons have been achieved, on both (i) punctual analysis of irradiated fuel (with initial enrichment of 4.5% and burnup up to 60 GWd/t) (ii) global analysis during the dissolution of irradiated assemblies (with initial enrichment between 3.1% and 3.5% and burnup up to 45 GWd/t).

These comparisons could be used to determine correction factors for the calculated concentrations of each of the actinides and fission products considered in the method.

The concentrations of the actinides and fission products depend on the conditions of irradiation (for a given fresh fuel, e.g. for a given fuel density, pellets diameter, burnable poison):

- Parameters of irradiation leading to a hardening flux spectrum have to be considered [14][15]:
  - boron concentration in the coolant,
  - temperature and density of the coolant,
  - presence of burnable poisons,
  - presence of control rods,
  - presence of mixed oxide (MOX) fuels or poisoned fuels around the assembly of interest.

- Other parameters like specific power, temperature of the fuel, shutdown periods, … need also to be assessed.

Finally a precise knowledge on the range of possible variations of the above parameters in order to define the values of the parameters is required for the depletion calculations. Currently, a French working group considers as conservative, for the depletion calculations, the presence of the control rods, the maximum concentration of the boron, a temperature of the water sets to its out-of-core value. For specific power, the conservative value depends on the nuclides of interest for the BUC and of the cooling time considered (see [14]).

Then, after irradiation and up to a cooling time of 100 years, the reactivity decreases. But, after a cooling time of 100 years, the reactivity starts to increase again (as the Am^{241} and Pu^{240} decay) until around 30 000 years.

For applications corresponding to a cooling time of less than 50 years, it may be acceptable to consider, in the criticality studies, the minimum cooling time that can be justified by the operators.

However, for storage or disposal, this approach raises the problem of which cooling time should be considered to ensure that the k-effective of the storage will not increase compared to the calculated value.

2.3. Criticality calculation step

Common practice for criticality study should be used. Additionally, the following items will be carefully considered:

- geometrical characteristics of irradiated fuel (fuel pitch, rod diameter, thickness of cladding),
- cross-section validation,
- validation of the neutronic calculation scheme,
- use of specific methods to solve possible problems of loosely coupled systems when a Monte-Carlo method is used.

Studies considering fuel burnup require an accurate knowledge of the cross-sections of isotopes (actinides or fission products) that are not commonly used in criticality calculation when a fresh fuel assumption is made. The discrepancies between calculated and measured values of the neutron-absorption rate can give a correction factor that needs to be safely considered in the criticality calculations performed [16][17][18].

The French fission product programs [16] are based on two types of experimental data.

IRSN has carried out experiments in Valduc apparatus B [19] using different fission product isotopes to validate the calculation scheme.

Another program divided into two parts was carried out at CEA/Cadarache. The former one is devoted to fuel inventory validation by chemical analyses and microprobe measurements of irradiated PWR pins. The latter one, involving oscillation experiments, is related to the reactivity worth of the different nuclei selected for the Burnup Credit.

Furthermore, the axial shape of the flux raises the question related to the low-coupled units. Indeed, in the case of storage or transport of irradiated fuel, the level of the reactivity is mainly due to the edges of the assemblies, which are the least irradiated areas. Monte-Carlo codes developers have proposed different methods to deal with the risk of underestimation of the k-effective if neutrons don't visit the fuel highly reactive zones (loosely coupled systems) during the simulation; the need to implement such methods must be considered when burnup is taken into account; For that purpose, a special working group has been created at the OECD to study this problem [20]. Meanwhile different statistical methods have been implemented in the CRISTAL Package such as (i) the super history powering, (ii) the fission matrix (kij matrix) method, (iii) the stratified sampling ; those methods are being studied [21] to solve this particular type of problem.

3. Gain estimation

The comparison between the k_{eff} value obtained with the fresh fuel method and with a "Burnup Credit conservative approach" could be presented as: Δk = k(fresh fuel method) - k(BUC cons. approach). Some studies [14][22] give the values of Δk for different configurations of storage and transportation.

We can notice that, even if the correction factors are pessimistic and the irradiation history is very conservative (control rods inserted during the whole irradiation, pessimistic assumptions regarding the conditions of irradiation…) and no realistic, a conservative approach already gives a gain up to Δk = 19 % for a burnup of 44 GWd/t with 5 years cooling time.

4. Conclusion

A Burnup Credit conservative approach needs the analysis of the three steps of the process to take fuel burnup into account in the criticality-safety studies: (i) the definition of the axial profile of burnup in the studies, (ii) the depletion calculations and (iii) the criticality calculations. But, as soon as we consider a profile of burnup, some actinides and fission products in the studies, it seems necessary:

- to check by a measurement that the profile of burnup used in the criticality-safety studies is actually conservative,
- to have a good estimation of the isotopes concentrations uncertainties,
- to know about the validation of the depletion and the criticality codes, in particular for the fission products cross sections.

A Burnup Credit implementation makes criticality calculation more complicated and can increase the risk of error in modeling. This problem underlines the need of adapted calculation tool development [23][24] to perform automated criticality calculations using Burnup Credit.

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BENEFITS OF BURN-UP CREDIT APPLICATIONS

(Session 2.5)

Partial boron credit implementation assessment relating to pool at reactor of Dukovany NPP

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Abstract. In 2005 the Czech Republic commemorates the anniversary of the commission of Dukovany NPP where four units of VVER-440 have been operating since 1985, when the first unit came into the operation. In between, the working fuel assembly design has been changed several times: from the uniform fuel assembly (FA) of 3.6 wt.% U-235 through the horizontally profiled FA of av. 3.82 wt.% to the horizontally profiled FA of av. 4.38 wt.% with Gd_2O_3 burnable absorber (entitled as 'Gd-1' FA), which has already been working, and finally the horizontally profiled FA of av. 4.25 wt% with Gd_2O_3 burnable absorber (entitled as 'Gd-2' FA), which has been just loaded into Dukovany Unit 3. The step-by-step increasing enrichment of the VVER-440 fuel brings about a general issue of re-licensing pools at the reactors. Although the pools have been partly reracked a new criticality assessment is being performed to open the way of the PBC/BUC storage technology implementation in the near future. First results obtained for the PBC implementation resulted from calculations using KENO VI Monte Carlo code are presented.

1. Introduction

Working fuel assemblies (see Fig.1) of four designs have been loaded into four units of the Czech Dukovany NPP (of the VVER440 type) since the NPP came into operation in 1985 [1][2]:

- uniform fuel assembly (FA) of 3.6 wt.% up to 1998
- horizontally profiled FA of av. 3.82 wt.% from 1998 to 2003
- the horizontally profiled FA of av. 4.38 wt.% with Gd_2O_3 burnable absorber (entitled as 'Gd-1' FA), which has already been working, and finally
- the horizontally profiled FA of av. 4.25 wt.% with Gd_2O_3 burnable absorber (entitled as 'Gd-2' FA), which has been loaded into Dukovany Unit 3 this year.

The fuel supplier for Dukovany NPP is the TVEL company, Russia, the fabrication is made in "Mashinostroitelnyi zavod" in Elektrostal, a city 60 km east from Moscow.

Spent fuel pools at the reactors have been partly reracked. The pool region which remained unchanged is a reserve region that should be used for discharged fuel assemblies incl. those very low irradiated. Up to now there have been some restrictions as for using this pool region for higher enriched FAs than 3.6 wt.%. To assess if new storage technologies based on Partial Boron Credit (PBC) or Burnup Credit (BUC) implementations could allow to rethink/revoke the current inconvenient operational measures a series of scoping calculation for a criticality safety were performed.

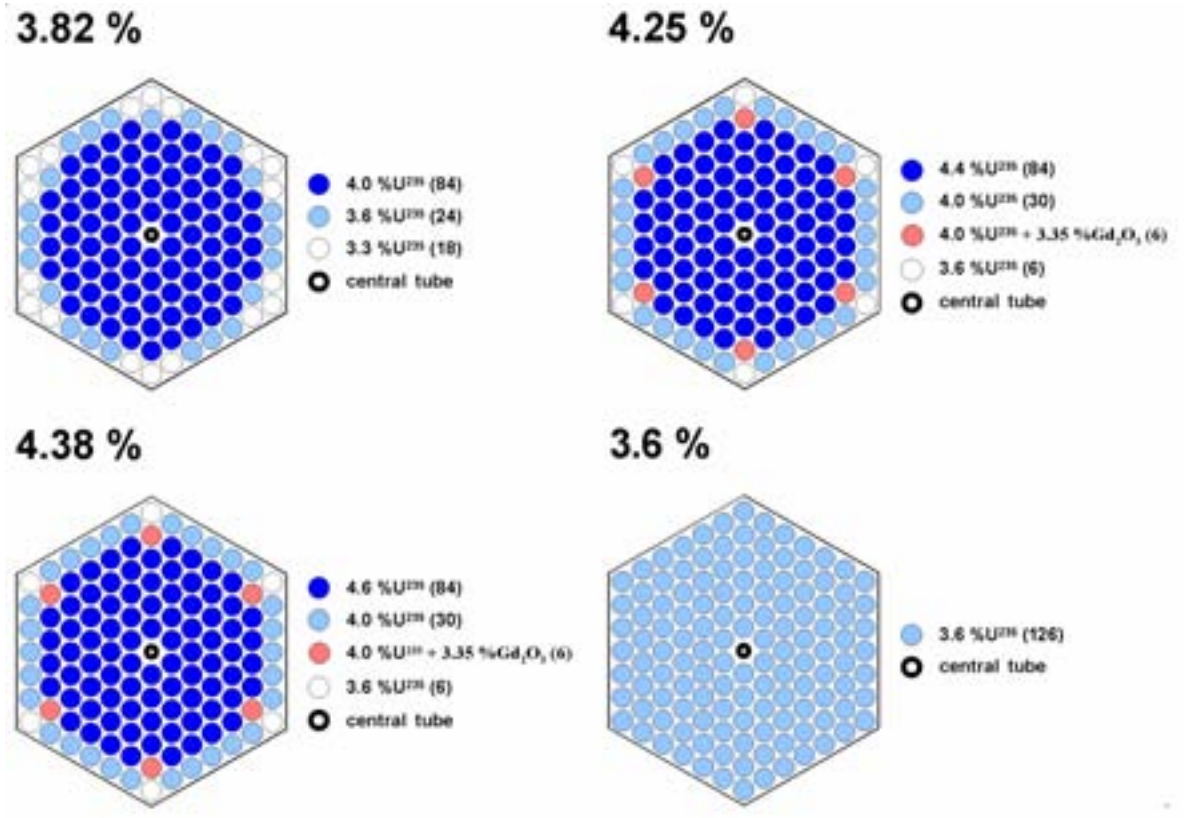

FIG. 1. Working FAs of horizontally uniformed or profiled enrichment (3.6 wt. % or average 3.82/4.25/4.38 wt.%, respectively) used in Dukovany NPP.

2. Calculational model and calculations performed

The calculations were carried out by KENO VI Monte Carlo code (SCALE 4.4a, [3]) with 44groupndf5 for a reasonably detailed finite model of the pool region ([4], Figs 2 & 3) after careful testing the case source convergence [5]. A VVER 440 FA were introduced into the calculations with the conservative uniform pattern [5], which allows interpretation of the results for the individual FAs [5] as if they were of both average and maximum enrichment. The non-reracked pool region contains only the racks for FAs, no other specific storage absorbers are present there. The pitch of FAs in the pool region is 22.5 cm. The lower limit of boron content in the pool is 12g H_3BO_3 per kg of solution and the upper temperature limit is 60°C. However, under normal operational conditions the temperature less than 50°C (usually 25–30°C) is maintained.

In compliance with the goal of the calculations mentioned above, fuel was supposed fresh. Further, in the first scoping calculations, no content of the burnable absorber was taken into consideration in fuel of the Gd-1 and -2 FAs as a conservative approach. Dependence of k_{eff} on fuel enrichment as well as pool temperature in the range 4–100°C (277–373 °K) were computed. For the examined pool region, where design technical features prevent misloading, only a hypothetical drop of a fresh fuel assembly (resulting finally in a horizontal position of the dropped FA) to the top of the others in the pool was studied. However, the evaluation of the case did not show any significant change of the reactivity of the system; it was only the increase of the pool water temperature over the upper limit as credible accident condition that was found more reactive than the nominal case.

The results obtained by the KENO VI calculations are shown in Figs. 4 and 5. As for licensing, the calculational results should enter the criticality safety criterion including mechanical and calculational uncertainties, with a 95 percent probability at a 95 percent confidence level. The estimated standard deviations of the performed scoping calculations were about 0.0005 (2σ=0.001) and methodology bias and its uncertainty were found using the validation package for the KENO VI calculations of the similar systems with the VVER440 fuel of 3.6 wt.% enrichment [6]. Based on the above mention data

and some previous calculations including the tolerance analyses [2], it was estimated, that the resulting multiplication factors should not be more than 3% higher than the results 'as computed' if all the uncertainties are taking into account for the criticality safety assessment. In Figs. 4 and 5 there are shown the results 'as computed'.

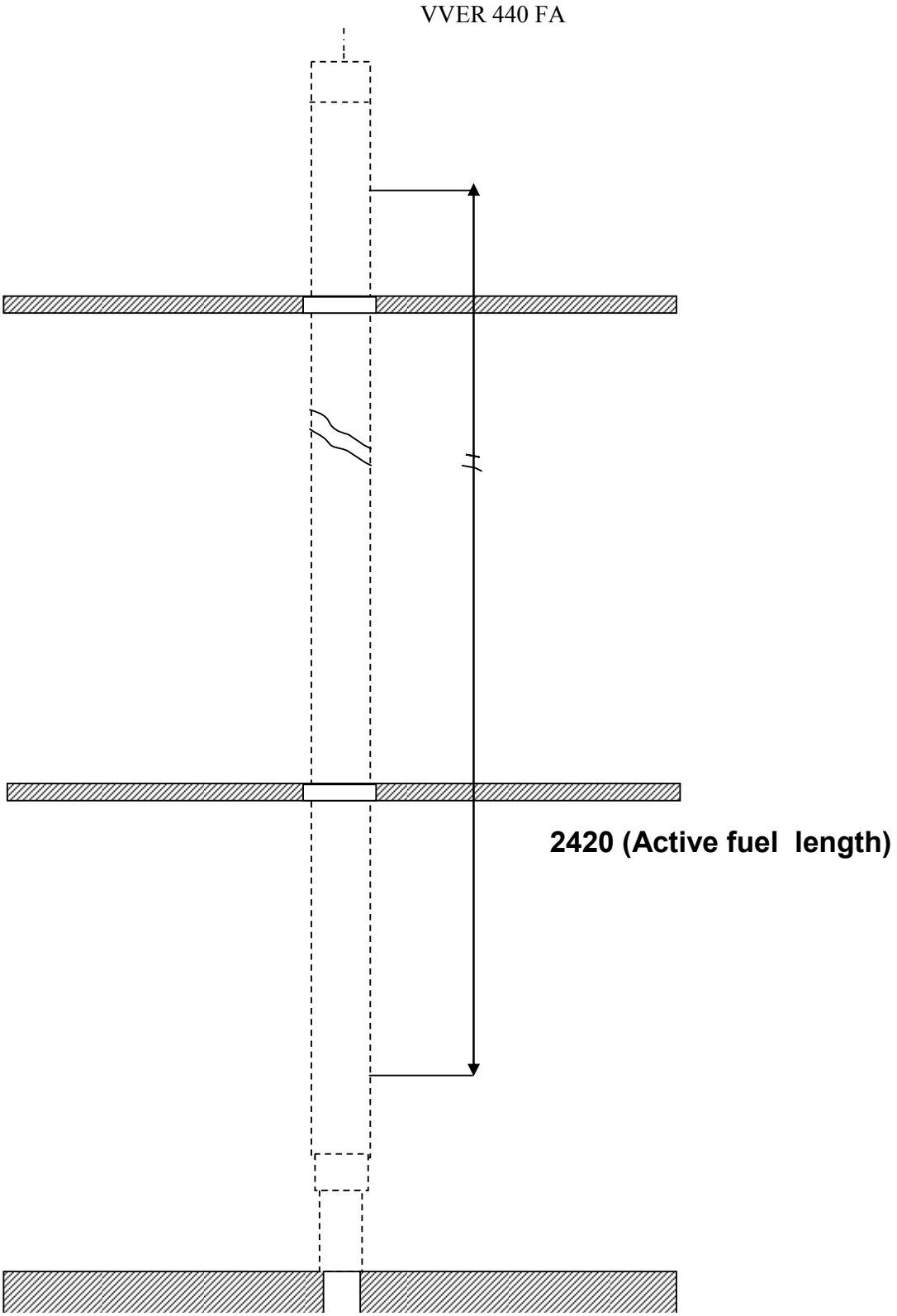

FIG. 2. Vertical section through storage rack.

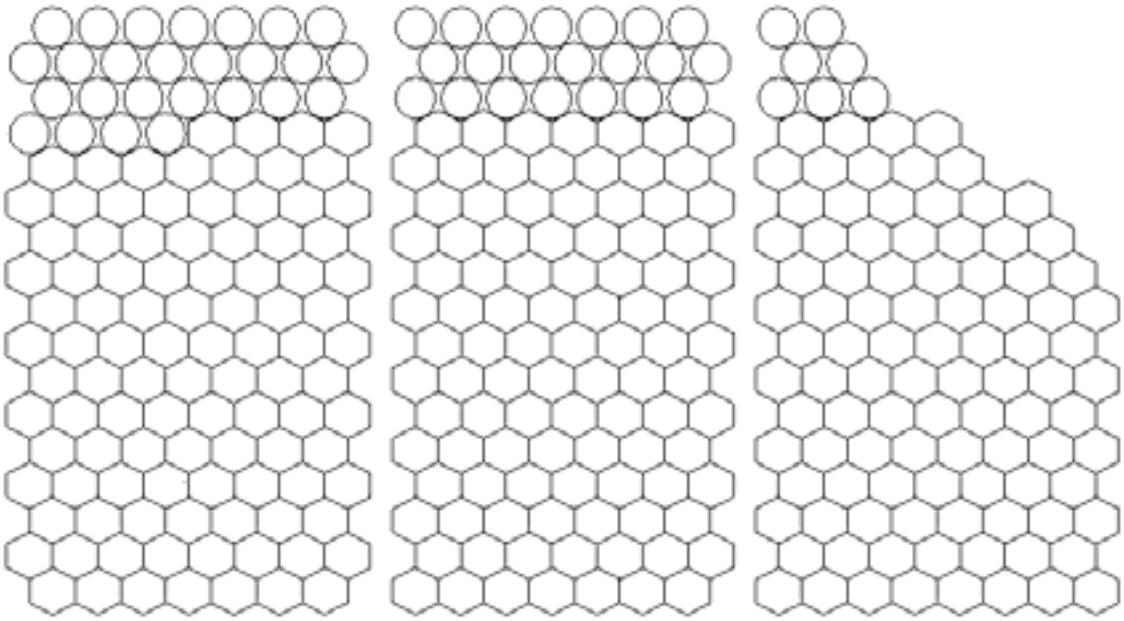

FIG. 3. Array of positions of FAs in non-reracked region of pool at Dukovany NPP.

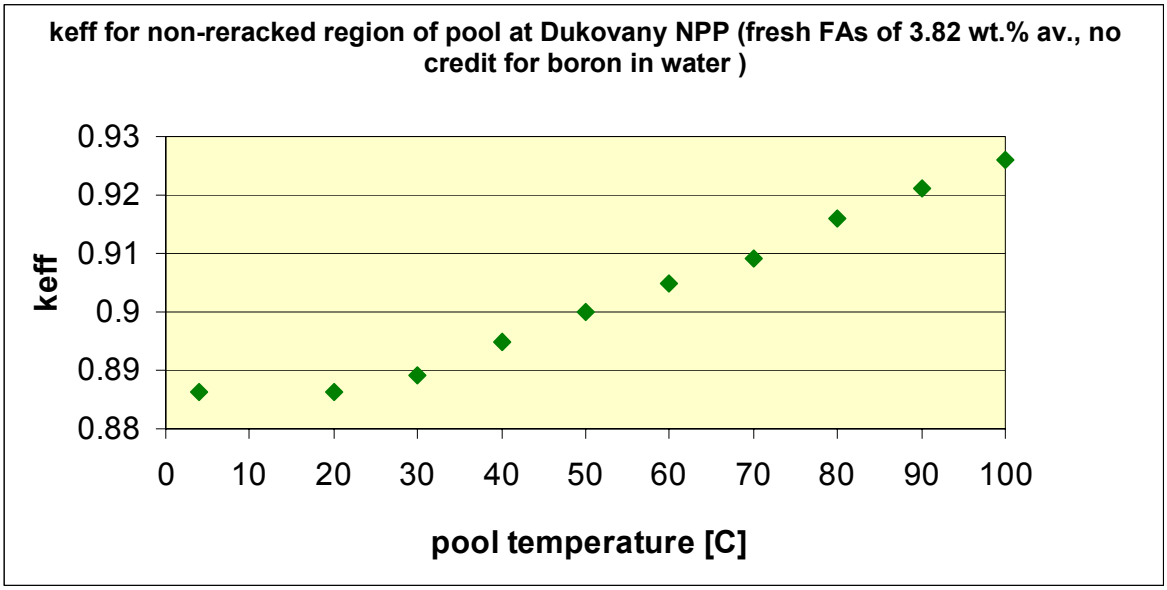

FIG. 4. Criticality calculations for upper non-reracked region of pool at Dukovany NPP with FAs of 3.82 wt.% av. (no credit for boron in water, fresh fuel).

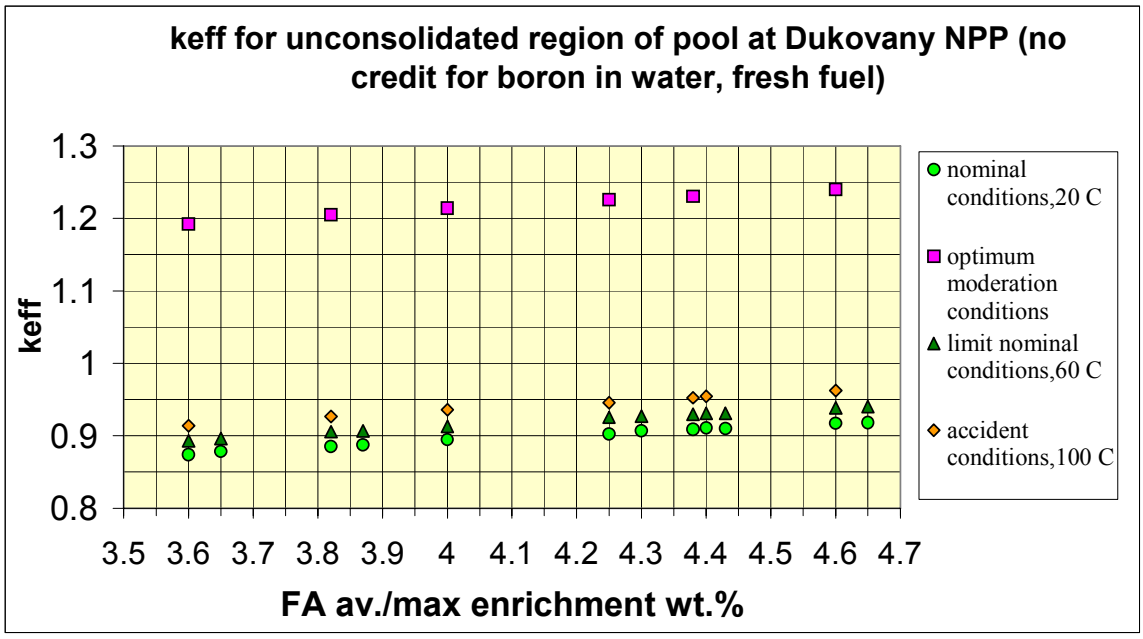

FIG. 5. Criticality calculations for upper non-reracked region of pool at Dukovany NPP (no credit for boron in water, fresh fuel.

3. Acceptance criteria

As in other countries operating nuclear power plants, the Czech regulation [7] requires the 5% safety margin of subcriticality for the case of the full density unborated water flooding the spent fuel storage racks loaded with fuel of the maximum permissible reactivity including mechanical and calculational uncertainties, with a 95 percent probability at a 95 percent confidence level.

Further, according to the Czech regulation [7], another requirement is effective not only for dry storage systems but also for pools at reactors: $k_{eff} < 0.98$ for low-density (optimum) moderation condition.

Regulations in many countries are changing in the course of time due to careful rethinking based on safety considerations, R&D achievements as well as new industry needs. In some countries operating a lot of NPPs, the latter requirement is not applied to pools at reactors (e.g. in US [8], Spain [9],..).

Generally, for fuel with higher enrichment than the original pools were designed for, the requirement could be possibly fulfilled if pools were reracked (thanks to placing absorbers as nests with absorbing tubes) but it is usually impossible to be fulfilled for non-reracked regions. The current Czech regulations insist on the requirement and a discussion on a demonstration that design features and/or administrative controls could prevent such event [9] is very difficult in this case. In reality, the low-density moderation conditions itself is not credible for such a wet system as in NPPs there are very strong measures taken to prevent pools at reactors from water boiling/evaporation not saying about the fact that the pool water level is several meters above the spent fuel racks and water volumes are big enough to give time for reducing temperature and maintaining the water level. If accepted as credible from the criticality point of view such a condition would be practically beyond-design-basis accident from the shielding and heat transfer point of view.

4. Discussion on result of calculations

The results of the conservative calculations (bounding analytical technique) for all the FAs as shown in Fig. 5 (as well as the corrected results after considering estimated increase of 3% due to all uncertainties) fulfill the requirements for the partial boron credit (PBC) implementation as formulated in Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants (US NRC, August 19, 1998) [8]; cited:

If partial credit for soluble boron is taken, the criticality safety analyses for PWRs must address two independent conditions, which should be incorporated into the plant technical specifications:

With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full-density unborated water, the maximum k_{eff} shall be less than 1.0, including mechanical and calculational uncertainties, with a 95 percent probability at a 95 percent confidence level.

With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full density water borated to [*] ppm, the maximum k_{eff} shall be no greater than 0.95, including mechanical and calculational uncertainties, with a 95 percent probability at a 95 percent confidence level.

([*] is the boron concentration required to maintain the 0.95 k_{eff} limit without consideration of accidents)

Unfortunately, the PBC implementation taking into criticality analysis a certain boron concentration to maintain the 0.95 k_{eff} limit is in a direct contradiction with the k_{eff} <0.98 requirement for the low-density moderation conditions (if effective). For the dry storage systems the optimum moderation accidental condition was motivated to postulate a possibility of fog/snow/.../fire-fighting foam intrusion into the system. In contrast with this, in the wet storage system, water first would have to go evaporated so it is water (not any foam) that should go through the phase of the low density. In such a hypothetical transient phase of the low density water (in case of the calculations whose results are shown in Fig. 5 the optimum moderation reactivity peak emerged at water density of about 0.26 g/cm^3) would be impossible to ensure any required amount of boron.

Thus, if the requirement of k_{eff} <0.98 is fulfilled anyway for the case of the non-reracked region of the pool at reactor of Dukovany NPP, the burnup credit (BUC) implementation would have to be considered. In the Czech Republic, research and development related to the BUC implementation in the VVER440 spent fuel management systems is in progress. However, the BUC implementation technology is still under the regulatory review as there are many remaining issues to be solved incl. e.g. a serious decision if the utilities are required to verify FA burnup by measurement.

5. Conclusions

The results of the bounding calculations performed for fresh fuel and unborated water show (see Fig. 5, below) that partial boron credit implementation could solve the issue of the non-reracked region of the pool at reactor of Dukovany NPP for all the new FAs mentioned above (see Fig.1). In spite of the fact that the PBC implementation seems to be nearly tailor-made to solve the issue, the present Czech regulation [7] doesn't allow the realization due to the k_{eff} <0.98 requirement related to low-density moderation condition effective also for pools at reactors. However, an amendment of the Czech Atomic Energy Act (No. 18/1997Coll.) has just been in preparation so it could be used as opportunity for fruitful and matter-of-fact technical discussion on this issue.

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Assessment of benefits for extended burnup credit in transporting PWR spent nuclear fuel in the United States of America

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Abstract. This paper presents an assessment of the benefits for extended burnup credit in transporting pressurized-water-reactor (PWR) spent nuclear fuel (SNF) in the United States. A prototypic 32-assembly cask and the current regulatory guidance were used as bases for this assessment. By comparing recently released PWR discharge data with actinide-only-based loading curves, this evaluation shows that additional negative reactivity (through either increased credit for fuel burnup or cask design/utilization modifications) is necessary to accommodate the majority of U.S. SNF assemblies in high-capacity storage and transportation casks. Given sufficient data for validation, the most significant component that would improve accuracy, and subsequently enhance the utilization of burnup credit, is the inclusion of fission products (i.e., extended burnup credit). A simple, conservative assessment of the cost benefits of extended burnup credit is also presented. Based on the estimated reduction in the number of shipments, achievable with extended burnup credit in the criticality safety evaluation, the cost savings for the U.S. Department of Energy (DOE) is estimated to be at least US $150M and is most likely in the US $200M–US $300M range. Evaluation of variations in the relevant input assumptions used to develop these estimates provides confidence that the actual cost savings may be much higher but are not likely to be lower.

1. Introduction

Historically, package designs for spent nuclear fuel (SNF) were constrained by weight, thermal loading, external dose, and structural integrity. With the reduced thermal load and dose provided by a minimum 5-year cooling time for transport of SNF, it became apparent in the 1980s that package capacity would often be limited by the conservative, yet simple fuel assumption of un-irradiated fuel (i.e,. no burnup credit) used in criticality safety evaluations [1]. For pressurized-water-reactor (PWR) SNF, burnup credit eliminates the need for the relatively wide basket structures (i.e., flux traps) used for separation and criticality control — thus providing an important degree of flexibility to package designers. For a typical rail-type transportation cask, elimination of the flux-traps between assembly storage cells enables an increase in cask capacity from ~24 to ~32 PWR assemblies. Hence, the potential benefit of using 32-assembly casks with burnup credit is a maximum reduction of ~25% in the number of required shipments. Note that due to the smaller cross-sectional area of some PWR assemblies (e.g. 14×14), assembly-specific canisters can be designed with capacities exceeding 32. However, for simplicity in this discussion, a value of 32 is used for the capacity of PWR burnup credit casks.

The use of higher-capacity packages enables a reduction in SNF packages, a reduction in package handling and loading operations, and fewer package shipments—resulting in a reduction in shipment and operational costs, personnel dose, public exposure, and accident risks [1]. After a decade of exploratory work and regulatory evolution, the U.S. Nuclear Regulatory Commission (NRC) issued Interim Staff Guidance 8 (ISG-8) in May 1999, providing the first allowance of burnup credit for PWR fuel. Subsequently, ISG-8 has undergone two revisions [2], which have eliminated or lessened a number of the restrictions. The initial issuance and subsequent revisions of ISG-8 have provided the impetus for industry to proceed with a new generation of high-capacity rail-type cask designs using burnup credit. However, ISG-8 recommends the burnup credit allowance be limited to that provided

by the change in actinide composition. As shown in the following section, this restriction significantly limits the percentage of the available SNF inventory that can be loaded in a high-capacity cask. To accommodate the majority of the SNF in high-capacity rail casks, extended burnup credit is needed (i.e., credit for the fission product nuclides). This paper presents an assessment of the benefits, in terms of inventory accommodation and cost savings, of extended burnup credit (considering both actinide plus fission product compositions) for transportation of PWR SNF in the United States.

2. Inventory accommodation for PWR SNF

During 2005, the DOE Energy Information Administration released a Microsoft Access™ data base with an updated version of the RW-859 compilation [3] submitted by U.S. commercial nuclear power plant licensees for PWR SNF through the end of 2002 (see Fig. 1). Six of the PWR fuel assembly types—WE 17 × 17, WE 15 × 15, WE 14 × 14, B&W 15 × 15, CE 16 × 16, and CE 14 × 14—comprise about 94% of the 70,290 PWR SNF assemblies in the data base. These six types of PWR assemblies were investigated to assess the benefits that would be provided by full burnup credit.

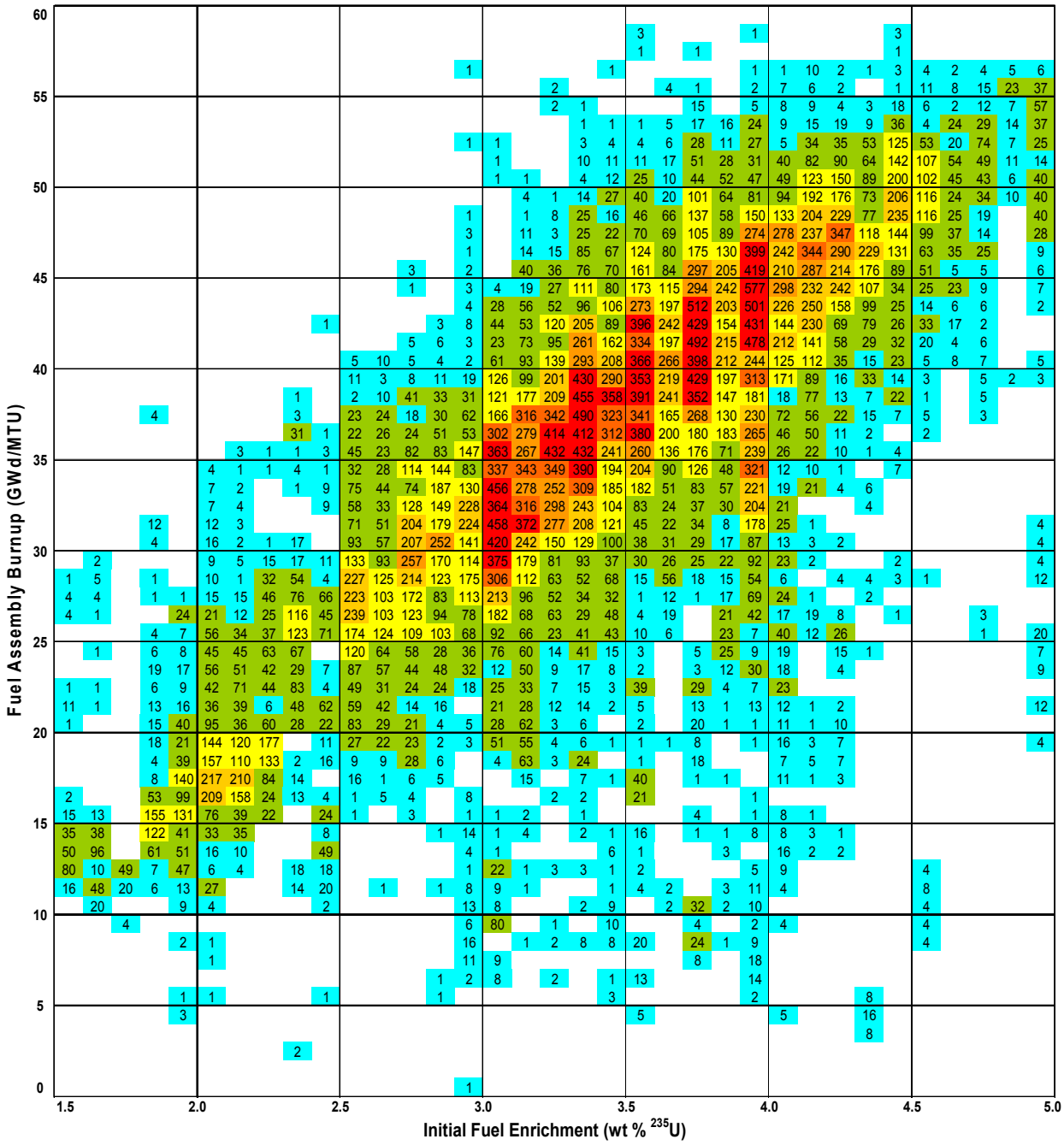
A review of the RW-859 (2002) data reveals that the average burnup of discharged PWR fuel assemblies has risen from around 20 GWd/MTU in 1975 to 45.7 GWd/MTU in 2002. This increase in assembly-average burnup represents a significant increase in the amount of criticality safety margin potentially available through burnup credit. Through 2002, 18.1% of the 70,290 discharged PWR fuel assemblies had burnups greater than 45 GWd/MTU. The average initial ^{235}U enrichment of discharged PWR assemblies has risen from about 2.7 wt % in 1975 to 4.2 wt % in 2002. This trend of increasing initial enrichment has made the fresh fuel assumption typically used in criticality safety analyses a more restrictive approach for cask design.

A generic high-capacity (32-assembly) cask, designated GBC-32, was selected as the reference configuration [4] to assess the benefits of full burnup credit for the RW-859 inventory. The GBC-32 cask is representative of burnup-credit rail casks currently being considered by U.S. industry and is therefore a relevant and appropriate configuration for this evaluation. The loading curves (required burnup and initial enrichment combinations) are generated with the STARBUCS sequence of the SCALE code system [5]. The basic assumptions (reactor operating conditions, bias and uncertainty process, axial profiles, etc.) can be found in Ref. [6].

Loading curves, consistent with the regulatory guidance of Ref. [2], are provided in Fig.s 2 and 3 for two of the six assembly types. The acceptability of the SNF assemblies for each fuel type is summarized in Table 1. Consistent with the regulatory guidance, assemblies that require burnup >50 GWd/MTU are classified as unacceptable. Also, the determination of acceptability does not account for burnup uncertainty, which would reduce the percentage of acceptable assemblies. The results indicate that while burnup credit can enable loading a large percentage of the CE 14 × 14 and WE 14 × 14 assemblies in a high-capacity cask, its effectiveness under the current regulatory guidance is minimal for the other assembly designs considered.

Table 1. Summary of SNF acceptability in the GBC-32 cask with actinide-only burnup credit for the six most prevalent assembly types

Assembly type	Total in discharge data	Number acceptable for loading	Number unacceptable for loading
CE 14 × 14	6,972	4,518 (65%)	2,454 (35%)
CE 16 × 16	6,828	1,731 (25%)	5,097 (75%)
B&W 15 × 15	7,519	166 (2%)	7,353 (98%)
WE 17 × 17	28,704	2,448 (9%)	26,256 (91%)
WE 15 × 15	10,365	475 (5%)	9,890 (95%)
WE 14 × 14	5,448	4,686 (86%)	762 (14%)
Total	65,836	14,024 (21%)	51,812 (79%)



Number of Assy in Group			Total Assy	% of All	Cum. Assy
0	to	20	3,385	4.8	70,128
21	to	100	14,937	21.3	66,743
101	to	200	15,082	21.5	51,806
201	to	250	10,891	15.5	36,724
251	to	300	6,354	9.1	25,833
301	to	350	5,555	7.9	19,479
351	to	577	13,924	19.9	13,924

FIG. 1. PWR spent fuel inventory from RW-859 (2002) nuclear data files.

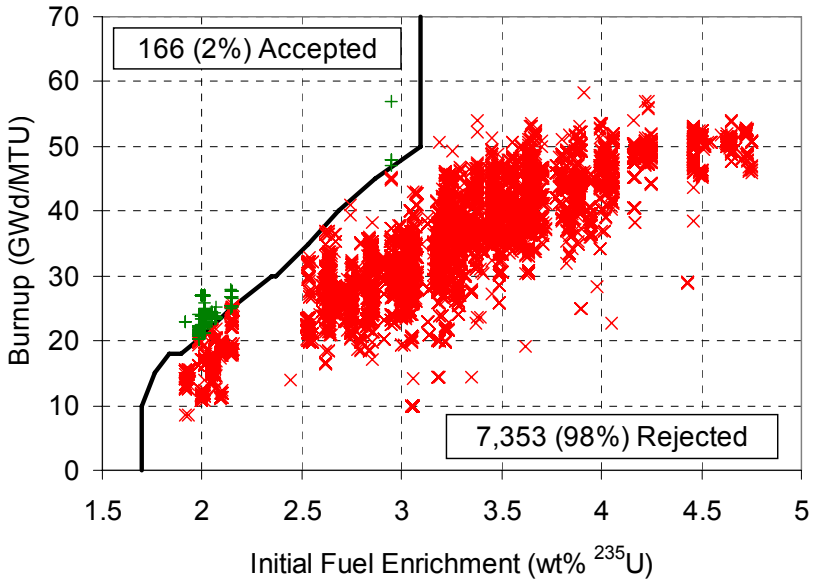


FIG. 2. B&W 15 × 15 inventory shown with ISG-8 burnup credit limit curve.

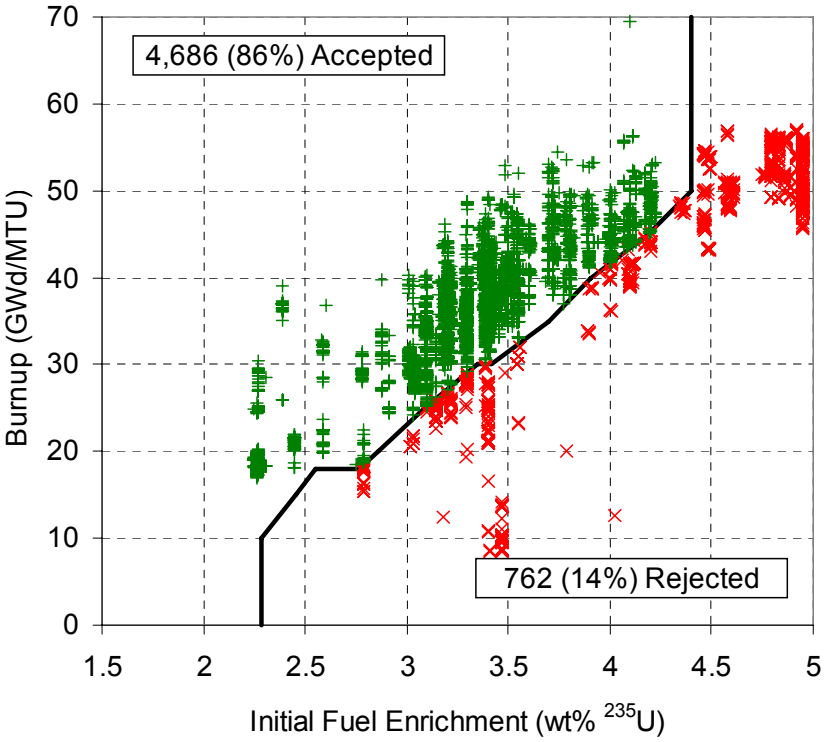


FIG. 3. WE 14 × 14 inventory shown with ISG-8 burnup credit limit curve.

To evaluate the effect of selected calculational assumptions, Fig. 4 compares the reference case loading curve for the WE 17 × 17 assembly with loading curves for the following individual variations:

(1) Inclusion of minor actinides (^{236}U, ^{237}Np, ^{243}Am) and five of the principal six fission products (^{149}Sm, ^{143}Nd, ^{151}Sm, ^{133}Cs, and ^{155}Gd), with isotopic correction factors [7] based on comparisons with available assay data. (The fission product ^{103}Rh is excluded due to insufficient measured assay data.)

(2) Inclusion of minor actinides and five principal fission products with spent fuel composition bias and uncertainty based on a best-estimate approach [7] for bounding isotopic validation.

(3) Inclusion of the principal fission products (95Mo, 99Tc, 101Ru, 103Rh, 109Ag, 133Cs, 147Sm, 149Sm, 150Sm, 151Sm, 152Sm, 143Nd, 145Nd, 151Eu, 153Eu, 155Gd) and minor actinides (236U, 237Np, 243Am), with spent fuel composition bias and uncertainty based on a best-estimate approach for bounding isotopic validation.

(4) Inclusion of the principal fission products and minor actinides without any correction for isotopic validation.

Note that for a few of the relevant fission products (e.g. ^{103}Rh), insufficient measured assay data are available to estimate bias and uncertainty. Thus, with the exception of the final case, no credit was taken for their presence in the SNF.

All of the curves in Fig. 4 were prepared assuming a 5-year cooling time. Extending the cooling time up to 20 years makes only a marginal increase in the allowed inventory [6]. A more effective approach is shown in Fig. 4 where inclusion of fission products and/or the use of more realistic approaches to isotopic validation offers significantly larger increases in allowed inventory. For the GBC-32 cask, the percentage of acceptable assemblies increases from 9 to 38% with the inclusion of the primary five fission products and minor actinides (both cases at 5-year cooling), and from 38 to 78% with the use of a bounding best-estimate approach for isotopic validation [7]. The next case includes the remainder of the principle fission products and uses the best-estimate isotopic validation approach. These assumptions allow the percentage of acceptable assemblies to increase to 90%. The final case shown in Fig. 4 corresponds to full credit for the calculated actinide and principal fission product compositions and, given the conditions considered, represents an unattainable limit in terms of the potentially available negative reactivity. For all the cases with fission products included, no explicit consideration of reactivity bias and uncertainty from comparison with critical experiments is included. However, the loading curves are all based on an upper subcritical limit of 0.94 (as opposed to 0.95), which inherently allows 1% Δk for criticality calculational bias and uncertainty.

Comparison of actinide-only-based loading curves for the GBC-32 cask with PWR SNF discharge data (through the end of 2002) leads to the conclusion that additional negative reactivity (through either increased credit for fuel burnup or cask design/utilization modifications) is necessary to accommodate the majority of PWR SNF assemblies in high-capacity casks. The loading curves presented in this paper are such that a notable portion of the SNF inventory would be unacceptable for loading because the burnup value is too low for the initial enrichment. Relatively small shifts in a cask loading curve, which increase or decrease the minimum required burnup for a given enrichment, can have a significant impact on the number of SNF assemblies that are acceptable for loading. Thus, as the uncertainties and corresponding conservatisms in burnup credit analyses are better understood and reduced, the population of SNF acceptable for loading in high-capacity casks will increase. Given appropriate data for validation, the most significant component that would improve accuracy, and subsequently enhance the utilization of burnup credit, is the inclusion of fission products.

3. Cost benefits for PWR SNF transportation

An initial economic analysis of burnup credit for transportation was prepared for the U.S. Department of Energy (DOE) Office of Civilian Radioactive Waste Management (OCRWM) in 1988 and used a life cycle cost model to estimate a potential savings up to US $900M [8]. Since that time, a portion of this predicted savings has become obtainable via the actinide-only credit allowed by ISG-8. Under this project, a relatively simple, but more current, cost analysis of the potential benefits of burnup credit was initially completed in 2003. The analysis used the current capacity limit for the Yucca Mountain repository [70,000 metric tonnes of heavy metal (MTHM)], the percentage of total MTHM from PWRs at the end of 1998 (~64%), and the average number of PWR assemblies per MTHM to predict that ~100,000 PWR assemblies will need to be transported to the repository. Using representative loading curves and assuming assemblies that cannot be accommodated in a 32-assembly cask are transported in a 24-assembly cask, it was estimated that full burnup credit can reduce the number of shipments by ~22% (~940 shipments), while actinide-only-based burnup credit reduces the number of

shipments by only ~8% (~315 shipments); a difference of ~625 shipments attributable to credit for fission products in the burnup-credit criticality safety evaluation (see Fig. 5).

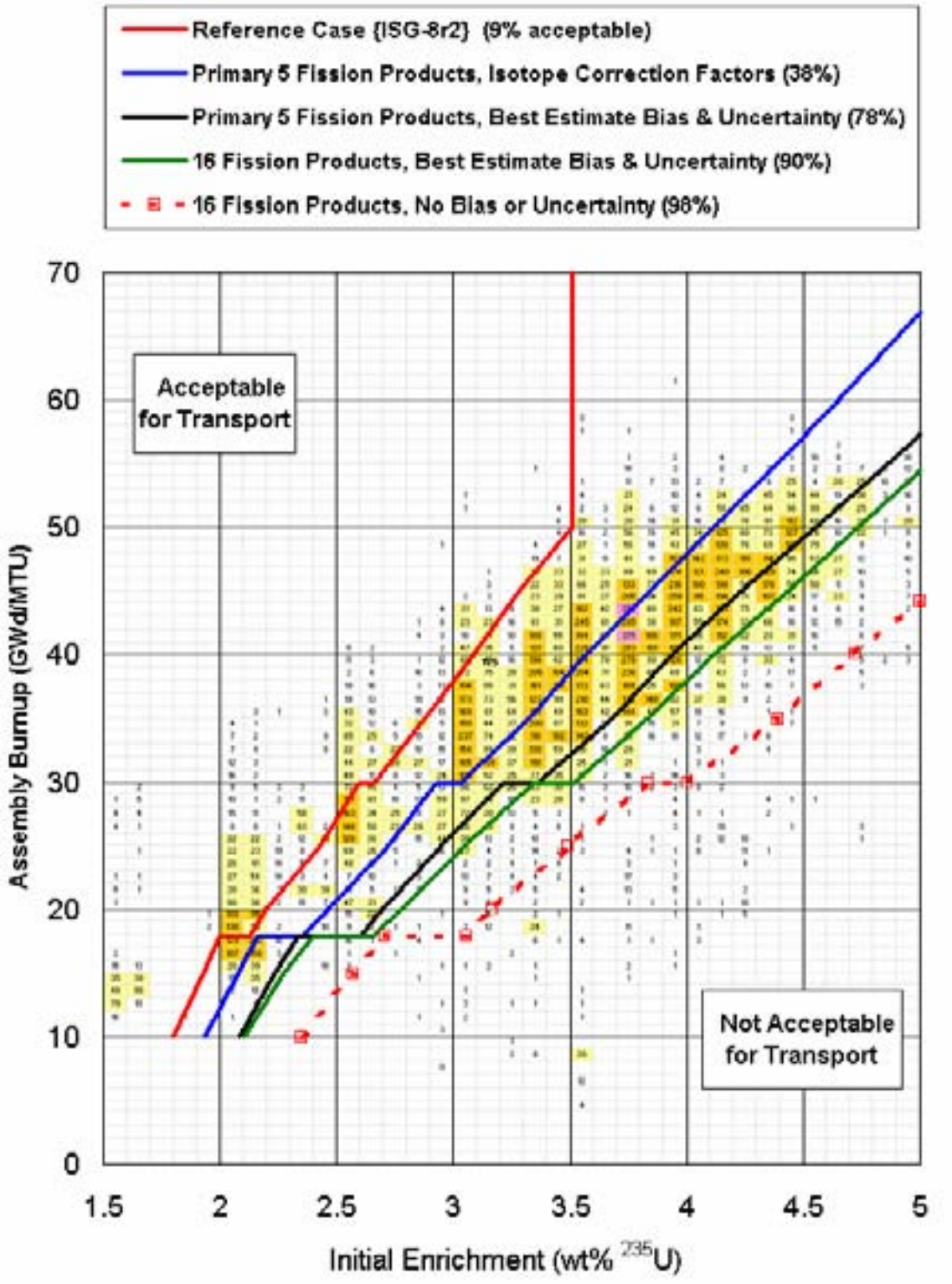


FIG. 4. Comparison of calculational assumptions for WE 17 × 17 fuel assemblies. Percentages of inventory acceptable for the GBC-32 cask are shown in parentheses.

A survey of U.S. industry experts suggested an estimated cost per rail cask shipment (freight and operational costs) ranging from US $200K to US $500K. Although the majority of the experienced opinions supported the US $500K/shipment value, a conservative estimate of US $250K was adopted. The operational and manufacturing costs will be essentially equivalent between the lower-capacity (24-assembly) and higher-capacity (32-assembly) casks. Consequently, the cost savings associated

with burnup credit will be dominated by the reduction in the number of shipments and the cost per shipment. Using the above cost-per-shipment estimate [assuming shipments are reduced by 625 (940 - 315)] provides a resulting costs savings of at least US $156M that can be realized from establishing full burnup credit for SNF transportation. This situation is shown graphically in Fig. 6. Note that the cost-savings estimate

scales linearly with the cost per shipment and does not include the difficult-to-quantify cost savings associated with the reduction in SNF packages required and the reduction in personnel dose, public exposure, and handling and transportation accident risks.

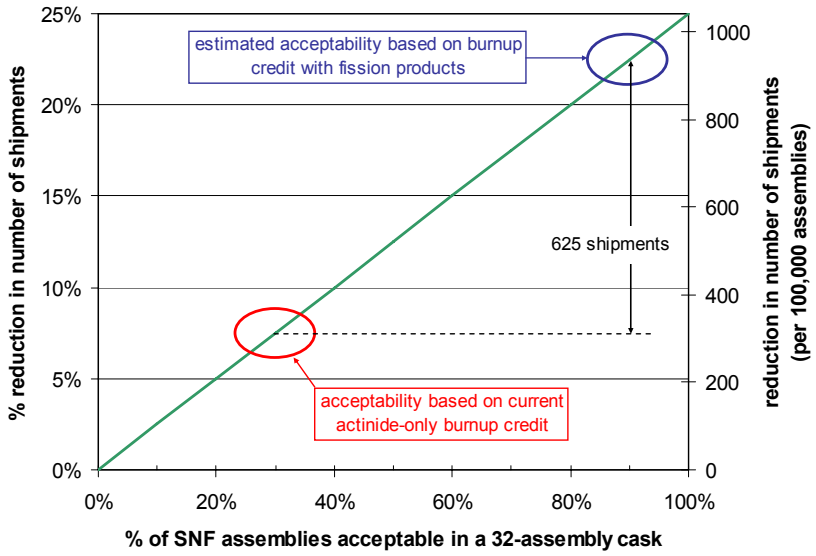


FIG. 5. Graphical representation of the potential reduction in the number of SNF shipments associated with the use of 32-assembly casks, as opposed to the use of 24-assembly casks. (Note that 100,000 assemblies in 24-assembly casks require 4,167 shipments.)

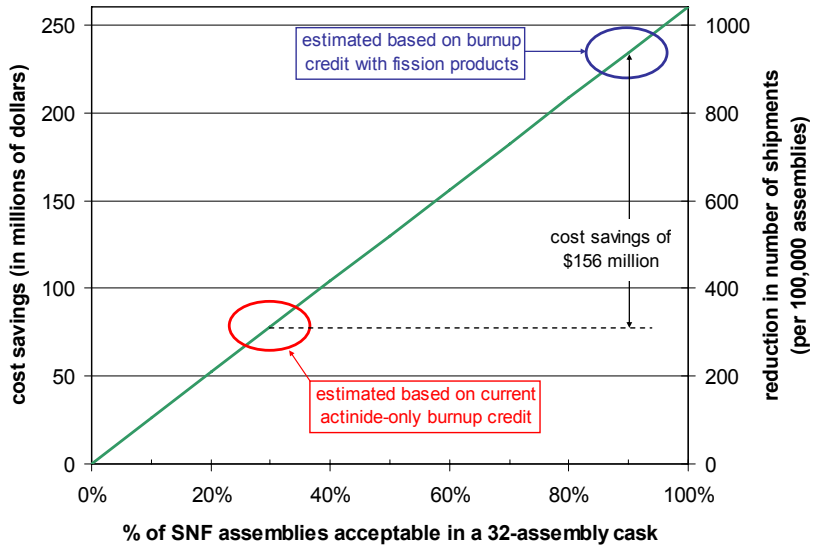


FIG. 6. Graphical representation of the potential cost savings associated with the use of 32-assembly casks, as opposed to the use of 24-assembly casks, assuming a cost of US $250K per cask shipment. (Note that the cost savings scale linearly with the cost per shipment.)

A significant simplifying assumption used in the above cost analysis is that all assemblies would be loaded and transported in large (i.e., 100 to 125- tonne) rail-type casks. In 2005, the cost estimate was updated to remove the simplifying assumption and investigate the impact of using a cask fleet of varying sizes. Discharge data as a function of site capabilities were first obtained (see Table 2). For the various cask sizes that could be used, estimates were developed for (1) cost per cask shipment, (2) cask design capacities with and without burnup credit, and (3) fraction of assemblies acceptable for loading with and without burnup credit. These estimates are listed in Table 3. Using the discharge data from Table 2 and the analysis assumptions listed in Table 3, the cost savings associated with burnup credit for transportation are estimated (see Table 4) to be ~US $638M. Of this total, ~US $235M is attributable to credit for fission products. These estimates are consistent with the previous analysis and demonstrate the significant potential cost savings associated with establishing burnup credit that includes credit for the primary fission product compositions. The results are based solely on cost savings associated with the reduction in the number of shipments for PWR SNF; cost savings associated with reduced personnel dose, public exposure, and accident risks are not included.

Limited sensitivity analyses were performed to evaluate the sensitivity of the cost savings estimates to variations in the input assumptions listed in Tables 2 and 3. In general, it was found that increased use of smaller casks will increase the cost savings. This trend is shown in the last column of Table 4, which lists savings due to fission product burnup credit on a per-assembly basis. This savings is due to the increased shipment cost on a per-assembly basis associated with the use of smaller casks. Assuming all 113,109 assemblies are transported in any one of the various cask sizes yields a range of US $177M – US $424M in estimated cost savings attributable to fission product burnup credit, with the lowest number corresponding to the use of all large rail-type casks and the highest number corresponding to the use of all truck casks. Note that the assumptions listed in Table 3 account for the fact that the increase in the fraction of acceptable assemblies due to fission product burnup credit is much less for smaller casks.

Although this most recent analysis does not specifically address decay heat constraints that could require a reduction in capacity for the large rail-type casks (e.g., if utilities opt to transport hottest fuel first), it does show that the use of smaller casks (e.g., to transport SNF with high decay heat) results in greater cost savings when burnup credit is applied. Also, there is a considerable portion of the discharged SNF inventory that will not present challenges in terms of decay heat, and the ability to use full burnup credit will provide a significant degree of flexibility to the vendors and utilities seeking to optimize their cask loadings.

Table 2. Number of projected discharged SNF assemblies as a function of site capability

Cask size code[a]	Site handling cask weight (tonnes)	Number of assemblies[b]
LWT	LWT ≤ 25	3,234
OWT	25 < OWT ≤ 35	4,734
RC1	40 < RC1 ≤ 75	8,443
RC2	75 < RC2 ≤ 100	52,333
RC3	100 < RC3 ≤ 125	36,426
RC4	125 < RC4	7,939
	Total	113,109

[a] LWT = Legal Weight Truck, OWT = Over Weight Truck, RC1 = Rail Cask 1.

[b] Data correspond to the number of assemblies discharged through 12/31/1998 plus those projected to be discharged through 12/31/2015 (***Source*****:** RW-859).

Table 3. Analysis assumptions for the various cask sizes

Cask size (tonnes)	Cost per shipment (US $K)[a]	Design capacity (number of assemblies)[b]		Fraction of assemblies acceptable for loading[c]		
		w/o BUC	w/BUC	w/o BUC	w/AO[d] BUC	w/AFP[e] BUC
LWT ≤ 25	150	2	4	1	0.9	1
25 < OWT ≤ 35	200	4	6	1	0.8	1
40 < RC1 ≤ 75	200	7	10	1	0.7	1
75 < RC2 ≤ 100	200	12	18	1	0.5	0.9
100 < RC3 ≤ 125	250	24	32	1	0.3	0.9
125 < RC4	250	24	32	1	0.3	0.9

[a] Values are intended to include freight, operational, and security costs and are based on a review of industry experts/experience and information generated during the process of evaluating the use of dedicated trains. The latter source suggested a cost of ~US $200K per cask shipment for freight and security only; no estimate of operational cost was available.

[b] Values developed based on a review of published and unpublished information, as well as consultation with industry experts.

[c] Values based on specific analyses, published results, and analytical experience.

[d] "AO BUC" refers to burnup credit that only accounts for the principal actinide compositions, consistent with current regulatory guidance (ISG-8).

[e] "AFP BUC" refers to burnup credit that includes the actinide and principal fission product compositions. This is also referred to as "full" burnup credit, which is not permitted under current regulatory guidance (ISG-8).

Table 4. Summary of cost savings

Cask size code	Number of assemblies	Number of shipments			Cost savings (US $K)		Additional savings due to FP BUC (US $K)	
		w/o BUC	w/AO BUC	w/AFP BUC	w/AO BUC	w/AFP BUC	Total	Per assembly
LWT	3,234	1,617	889	809	109,200	121,200	12,000	3.71
OWT	4,734	1,184	868	789	63,200	79,000	15,800	3.34
RC1	8,443	1,206	953	844	50,600	72,400	21,800	2.58
RC2	52,333	4,361	3,634	3,053	145,400	261,600	116,200	2.22
RC3	36,426	1,518	1,404	1,176	28,500	85,500	57,000	1.56
RC4	7,939	331	306	256	6,250	18,750	12,500	1.57
Totals	**113,109**	**10,217**	**8,054**	**6,927**	**403,150**	**638,450**	**235,300**	

4. Conclusions

Comparisons of recently released U.S. PWR discharge data with actinide-only-based loading curves, shows that additional negative reactivity (through either increased credit for fuel burnup or cask design/utilization modifications) is necessary to accommodate the majority of SNF assemblies in high-capacity storage and transportation casks. The impact of varying selected calculational assumptions was investigated, and considerable benefits in terms of inventory accommodation were shown to be possible with extended burnup credit (i.e. credit for the principal fission products). A simple, conservative assessment of the cost savings benefits for extended burnup credit in transporting PWR SNF in the United States was also presented. This assessment indicates that the estimated cost savings is greater than US $150M and is most likely in the US $200M–US $300M range. Evaluation of the variations in the relevant input assumptions used to develop these estimates provides confidence that the actual cost savings may be much higher but are not likely to be lower. This estimate of cost savings does not include cost savings associated with the reduction in personnel dose, public exposure, and handling and transportation accident risks.

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Burnup credit for receipt and storage of UOX PWR fuels in COGEMA/La Hague Pools

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Abstract. The current criticality-safety analysis regarding receipt and storage in COGEMA/La Hague pools of spent UOX PWR fuels only considers the decrease in fissile nuclides due to fuel irradiation in reactor: The Actinide-Only Burnup Credit.

In the perspective of the next increase in fuel enrichment, the study described in this paper aims to use the conclusions of the French Working Group on Burnup Credit, and to use the last release – V1 – of the French Criticality Safety Software Package CRISTAL (with its units dedicated to Burnup Credit calculations) in order to build a criticality safety assessment on an "Extensive Burnup Credit".

The calculations made take into account a/ 15 Fission Products (and their associated qualification), b/ burnup axial profile (in a conservative methodology described in this paper) and c/ conservative assumptions for depletion calculations. The results show from a safety point of view the feasibility to extend the field in which burnup measurements are not necessary before the fuels' receipt. This is achieved without changing current pools' baskets.

1. Introduction

Initial authorizations for receipt and storage of spent 17x17 UOX PWR fuel assemblies in COGEMA/La Hague pools were based on a fresh fuel. At present, these authorizations for fuels with an initial enrichment higher than 3.75 wt% are built on "Actinide-Only" Burnup Credit assumptions [1].

Depending on the fuel initial enrichment, two cases appear:

- the burnup noticed by a single irradiation cycle in reactor is sufficient to ensure the safety of the storage; otherwise
- a burnup measurement is necessary to guarantee an irradiation level.

The referring burnup there is the average axial burnup in the 50-least-irradiated-centimetres of the fuel assembly because an uniform axial burnup is assumed.

With these current analysis' hypotheses, the increase in fuel enrichment will require a burnup measurement.

Therefore, this study aims to build a criticality safety assessment on an "Extensive Burnup Credit" according to the three following steps:

(1) Reference calculation (using the current Burnup Credit assumptions).

(2) Extensive Burnup Credit (use of the French Working Group on Burnup Credit assumptions).

(3) Analyse with axial burnup profile (three different profiles are used to determine a conservative methodology).

In order to highlight the relative importance of the main assumptions made for this Extensive Burnup Credit with axial profile, some intermediate studies have also been made.

These calculations are performed using the last release – V1 – of the French Criticality Safety Software Package CRISTAL with its units dedicated to Burnup Credit calculations. The purpose of the calculations is to determine the lower safety burnup of the fuel assembly needed for its receipt and storage in COGEMA/La Hague pools. Then, this lower safety burnup is compared with the burnup noticed by a single irradiation cycle so as to assess the need of a burnup measurement.

This paper focuses on results obtained for fuel assemblies with a 5.0 wt% initial enrichment.

2. Description of the computer codes used

2.1. CESAR point depletion code

The fuel inventory after irradiation is computed by the point depletion code CESAR [2] used in its version 5. This code needs macroscopic cross-sections library built by the APOLLO2 code from the JEF2.2 evaluation [3].

This computer codes linking is the current way of depletion calculation for production studies used by SGN.

For this study, the three following cross-sections library -XSL- for CESAR performed by the SPRC Laboratory of CEA/Cadarache were used:

- a library built on the depleted PWR UOX fuel assembly assumed with control rod out and in a surrounding environment of PWR UOX fuels; called "XSL04". This scheme is the reference used for current actinide-only burnup credit calculations.
- a library built on the depleted PWR UOX fuel assembly assumed with Control Rod Out and in a surrounding environment of PWR MOX fuels; called "XSL71".
- a library built on the depleted PWR UOX fuel assembly assumed with Control Rod Inserted and in a surrounding environment of PWR MOX fuels; called "XSL72".

The use of one or another of these XSL depends on the studied analysis sequence according to section 3.3.

2.2. Criticality safety package CRISTAL V1

The calculations of the neutron multiplication factor (k_{eff}) are performed with the standard way of the criticality-safety package CRISTAL V1 (CIGALES3, APOLLO2 and MORET4 codes) [4][5].

CIGALES 3.0 computes the atomic densities of the fissile mixtures reading the output files of the depletion code CESAR. The CIGALES 3.0 Graphical/User Interface dedicated to burnup credit calculations allows:

- the use of several depletion calculation (for burnup axial profile problems),
- the capture of corrective factors on the fuel inventory computed (to take into account burnup credit nuclides' qualification).

Then, CIGALES generates the calculation file for APOLLO2 code.

APOLLO2 associated with the "CEA93-V6" cross-section library computes the neutronic parameters– material buckling B_m^2 and k_∞ – of the fissile materials and generates, for all fissile and non-fissile

materials, 172-group macroscopic cross-sections usable by MORET code. The CEA93 library is based on JEF2.2 [3].

The MORET 4 code uses cross-sections from APOLLO2 to compute k_{eff} in a three-dimensional geometry through a Monte-Carlo method.

3. Description of the analysis approach

3.1. Reference configuration

This work concentrates on typical 17x17 PWR UO_2 fuel assemblies, commonly used in the EDF French utility nuclear reactors.

Uranium in the fuel rods is assumed to be initially enriched at 5.0 wt% in isotope 235.

The configuration studied here consists of a basket - called NPH basket- with 9 steel cells (one fuel assembly per cell) along with an extra fuel assembly sitting in contact with the basket. Each cell is jacketed by borated steel. This 9+1 fuel assembly system fully reflected by water simulates the accidental and dimensioning situation where a fuel assembly has fallen vertically close to the NPH basket in the pool. This situation is illustrated on the Fig 1.

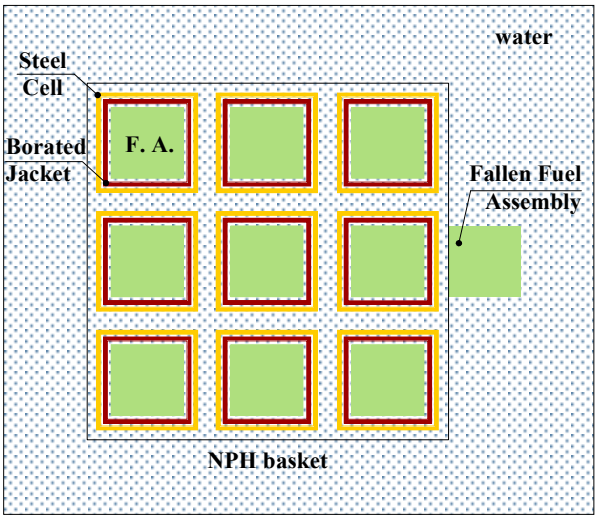

FIG.1. Plan view of the studied configuration.

3.2. Calculation methodology

3.2.1. Principles of the calculations

The purpose of the calculations is to determine the minimal burnup of the fuel assemblies which respects the safety criterion on k_{eff} of the above configuration.

Then, this lower safety burnup is compared with the burnup noticed by a single irradiation cycle. For 17x17 PWR UO_2 fuel assemblies with an initial enrichment higher than 4.25 wt%, the fuel managements performed allow the use of a minimal irradiation during a single cycle in reactor of:

- 8 500 MWd/t (average axial burnup),
- 4 100 MWd/t in the 50-least-irradiated-centimetres (axially) of fuel rods.

The use of the irradiation in the end of fuel rods is the current safety condition for actinide-only and axially constant burnup credit. On respect of this condition, a burnup measurement for safety reasons is not necessary before the receipt of the fuel assemblies.

3.2.2. Extensive burnup credit assumptions

The "Extensive Burnup Credit" hypotheses assumed here are based on the studies of the French Working Group on Burnup Credit: FWG-BUC. This group leaded by IRSN gathers experts from IRSN, CEA, EDF and AREVA Group (*via* COGEMA, COGEMA-Logistics, SGN, and FRAMATOME) [6].

The nuclides considered for an Extensive BUC are the 8 actinides used for the current actinide-only BUC (^{235,236,238}U and ^{238,239,240,241,242}Pu) to which are added ^{241}Am and the 15 following fission products: ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Rh, ^{109}Ag, ^{133}Cs, ^{143,145}Nd, ^{147,149,150,151,152}Sm, ^{153}Eu and ^{155}Gd.

Concerning the depletion calculation, assumptions have to be made relatively to the fuel irradiation history in the reactor and to the direct environment of the depleted fuel assembly during its irradiation.

The following conditions have been proved by the FWG-BUC to provide conservative burnt fuel isotopic inventory, with regard to burnt fuel criticality calculations [7][8].

- a MOX surrounding environment for a UOX fuel assembly depletion calculation,
- Control Rod Inserted in the depleted UOX fuel assembly,
- a specific set of reactor's conditions (boron concentration, temperatures…).

These conditions are implemented in the CESAR's cross-section library XSL72 (see section 2.1) thus defined as the conservative library for BUC -with fission products- calculations.

Furthermore, additional studies by the FWG-BUC[8] show that the irradiation power has to be maximized in the depletion calculation. Then, for this work, the spent fuel is assumed to be continuously irradiated during at least 285 days -minimal duration for a single irradiation cycle- and with a 40 W/g power upper limit. The cooling time after the reactor shutdown is minimized at the guaranteed value of 6 months.

Finally, the qualification of the depletion calculation scheme and the qualification of the absorption cross-section of the fission products considered must be taken into account.

Then, the FWG-BUC determines a set of Corrective Factors -CFs- [8] to be applied to nuclides balance computed by the point depletion code CESAR. These CFs are mainly based on results of fission products experiments performed in CEA/Cadarache and CEA/Valduc Facilities [9].

Moreover, one can notice that the fuel burnup here is low (about a single irradiation cycle) consequently the impact of the irradiation conditions simulated in the depletion calculation on nuclides balance is attenuated.

3.2.3. Burnup axial profile

Current authorizations for receipt and storage of spent fuels in COGEMA/La Hague pools are based on an uniform (axially constant) fuel burnup. This burnup is compared with the 50-least-irradiated-centimetres of fuel rods to ensure the criticality safety of the receipt operations.

This work aims to analyze the credit of an axial burnup profile within the fuel assembly. The three different profiles presented on the Fig. 2 are then studied:

- The "standard profile" (referred by "3a") is representative of an irradiation excluding particular events.
- The "distorted profile" (referred by "3b") is representative of an irradiation with Control Rod Inserted in the fuel assembly during its last irradiation cycle.

- The “extremely distorted profile” (referred by “3c”) is a theoretical profile computed assuming Control Rod Inserted at their maximal position during all the fuel irradiation (three cycles in reactor).

These three profiles can be characterized by their “distortion level”, as presented in section 4.2.1.

In the Monte-Carlo k_{eff} calculations, these profiles are discretized into axial zones according to the following principles:

- the number of axial zones is set around 10 (because of the compromise between computation time and results’ accuracy),
- the fixed burnup of a zone is equal to the average axial burnup of the studied profile in this zone,
- the burnup variation between two consecutive zones is constant.

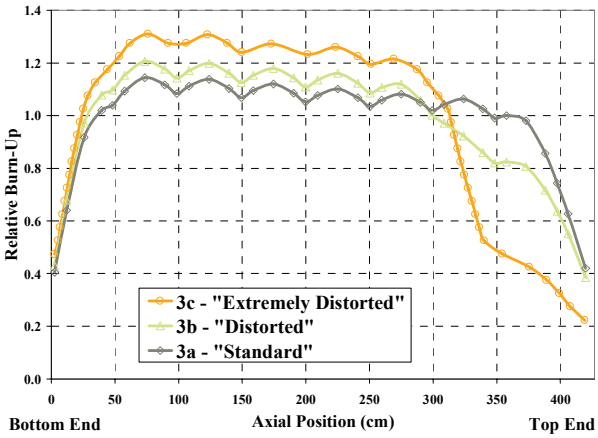

FIG. 2. Axial burnup profiles studied.

3.3. Burnup credit sequences

The analysis approach is based on the three following sequences:

(1) Reference calculation.

(2) Extensive Burnup Credit.

(3) Analyse with axial burnup profile.

The first sequence “Reference calculation” aims to compute the lower safety burnup with the current BUC assumptions: 8 actinides which balances are computed by the “current way” (with CESAR’s XSL04, see section 2.1) and with an uniform burnup.

The second sequence “Extensive Burnup Credit” also assumes an uniform burnup but considers the assumptions presented in the section 3.2.2, that is to say: 9 actinides and 15 fission products which balances are computed by the “conservative way” (with CESAR’s XSL72, see section 2.1) and are corrected by CFs.

In addition to these two sequences, the two following intermediate calculations (called “i1” and “i2”) are made:

i1 – Reference calculation taking into account the additional BUC due to the ^{241}Am.

i2 – In addition to calculation “i1”, evaluation of the BUC due to the 15 fission products considered for the Extensive BUC. In comparison with results of the second sequence, this calculation aims to

quantify (in term of lower safety burnup) the conservatisms introduced by the Extensive BUC assumptions.

The third sequence purpose is to analyse the impact of a non-uniform axial burnup profile within the fuel assembly. The three axial profiles presented in the section 3.2.3 are studied with the Extensive BUC assumptions: calculations referred by 3a, 3b and 3c.

In each axial zone, the nuclides' balance for the "standard" profile is computed using the "penalizing" CESAR's library XSL72 and CFs (calculation 3a).

For both "distorted" (calculation 3b) and "extremely distorted" (calculation 3c) profiles representing an irradiation with a control rod insertion, the isotopic inventory of the criticality calculation is computed from the Control Rod Inserted obtained Library XSL72 for the upper part of the fuel assembly, while the lower remaining part of the assembly is fed with isotopic concentrations resulting from the Control Rod Out obtained Library XSL71. The CFs are still applied to the nuclides' inventory of each axial zone.

4. Main results and analyses

4.1. Uniform axial burnup

For calculation sequences 1 and 2 (and intermediate i1 and i2 calculations), the lower safety burnup of each fuel assembly in the reference configuration is given on the Fig. 3. As these calculations assume an uniform axial burnup, the results are presented in comparison with the single irradiation cycle burnup in the 50-least-irradiated-centimetres (axially) of fuel assemblies: 4 100 MWd/t (see section 3.2.1).

In the reference calculation (sequence 1), the lower safety burnup obtained is 9 600 MWd/t, higher than the burnup noticed by a single irradiation cycle. Then, a burnup measurement of the fuel assemblies is necessary to allow their receipt operations in pools.

The effect of ^{241}Am (calculation i1) is negligible regarding the lower safety burnup because of the very short cooling time assumed (6 months).

The Extensive BUC calculation (sequence 2), as defined in the section 3.2.2, leads to a lower safety burnup reduced to 5 900 MWd/t. This conservative methodology allows a 3 700 MWd/t credit (almost 40% of the reference lower safety burnup) but is not sufficient to ensure the safety of the receipt operations without a previous burnup measurement.

With regard to the calculation i2 (an "Extensive BUC" without the safety penalizing assumptions), it can be noticed that the safety assumptions made for the Extensive BUC represent a 900 MWd/t margin (around 10% of the reference lower safety burnup).

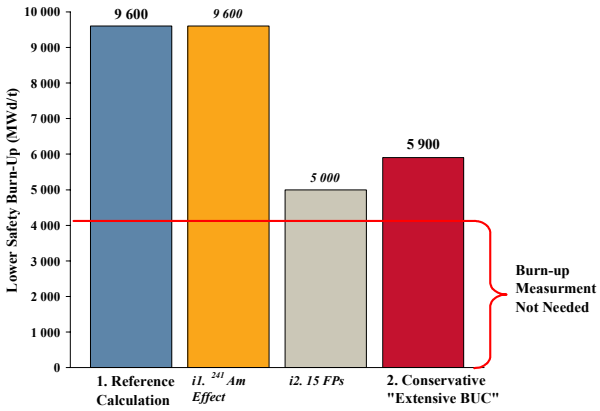

FIG. 3. Lower safety burnup of fuel assemblies for calculation sequences 1 and 2.

4.2. Burnup axial profile

4.2.1. Particularity of axial profile problems

When the burnup axial profile is taken into account, the single irradiation cycle burnup in the 50-least-irradiated-centimetres is no more the only criterion to judge the need of a burnup measurement. The average axial burnup guaranteed for assemblies irradiated during a single irradiation cycle must be also considered.

In order to characterize the different axial profiles, a Distortion Factor -DF- is then defined, for a burnup axial profile, as the ratio between the burnup in the 50-least-irradiated-centimetres of this profile and the average axial burnup of this profile. The Table 1 below gives the DF for the three axial profiles studied and for the axially uniform burnup, which can be seen as a non-distorted profile.

Consequently, the results of the calculation sequence 3 are given in both terms of:

- lower safety **end axial burnup** (that is to say the burnup in the ending 50 cm of the assembly),
- lower safety **average axial burnup**.

These results have to be compared with the corresponding burnup criterion to appreciate the need of a burnup measurement.

One can note that depending on the axial profile's DF, only one or another of the burnup criteria is the dimensioning one: If the DF is lower than 0,482 (4 100/8 500, ratio between the criteria), the lower safety average burnup is dimensioning.

Table 1. Distortion factor -DF- for the three axial profiles studied and for the axially uniform burnup profile

Burnup axial profile (see section 3.2.3)	DF
Uniform	1.000
"Standard"	0.673
"Distorted"	0.580
"Extremely Distorted"	0.312

4.2.2. Results obtained for the end axial burnup

For the three axial profiles of the calculation sequence 3, the lower safety end axial burnup of each fuel assemblies in the reference configuration is given on the Fig. 4. These results are then presented

in comparison with the single irradiation cycle burnup in the 50-least-irradiated-centimetres of fuel assemblies: 4 100 MWd/t (see section 3.2.1).

These results show that the more distorted the axial profile is, the lower the end burnup must be to ensure the safety of the reference configuration. This effect is easily explained while looking at the Fig. 5 on which are represented the three studied axial profiles normalized to the same end burnup. It can indeed be noticed that a greatly distorted profile presents a central axial zone clearly more irradiated (thus with more BUC) than a less distorted profile with the same end burnup. Therefore, with regard to the end burnup, the bounding profile is the least distorted one.

It is interesting to notice that both “standard” and “distorted” profiles give equivalent results.

Furthermore, for the three axial profiles studied, the lower safety end burnup could be guaranteed without a burnup measurement but only with the irradiation noticed by a single irradiation cycle in reactor.

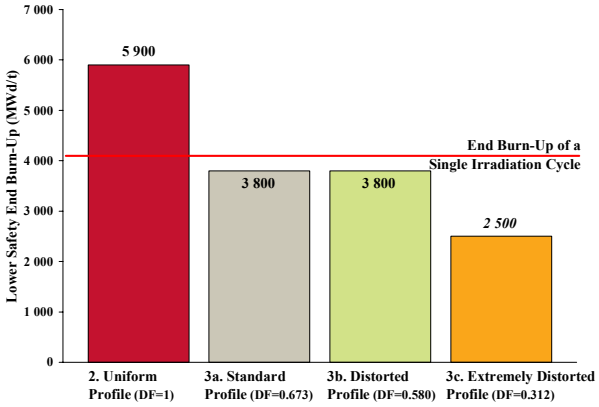

FIG. 4. Lower safety end burnup of fuel assemblies for calculation sequence 3.

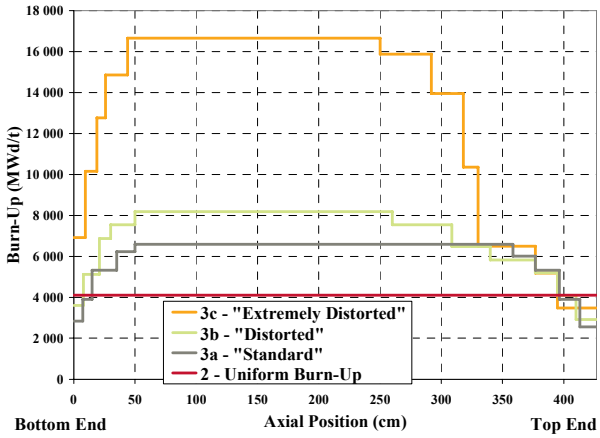

FIG. 5. Comparison of the three studied axial profiles normalized to an end burnup equal to 4 100 MWd/t.

4.2.3. Results obtained for the average axial burnup

For the three axial profiles of the calculation sequence 3, the lower safety average axial burnup of each fuel assemblies in the reference configuration is given on the Fig. 6. Then, the results are presented in comparison with the single irradiation cycle average burnup: 8 500 MWd/t (see section 3.2.1).

These results show that the more distorted the axial profile is, the higher the average burnup must be to ensure the safety of the reference configuration, because of the lower irradiation in the fuel

assembly's end. Therefore, with regard to the average axial burnup, the bounding profile is the most distorted one.

One can note on the Fig. 6 that the axially uniform profile, the "standard" profile and the "distorted" profile are almost equivalent in terms of lower safety burnup.

As in the previous section, the lower safety average burnup could be guaranteed without a burnup measurement but only with the irradiation noticed by a single irradiation cycle in reactor for the three axial profiles studied.

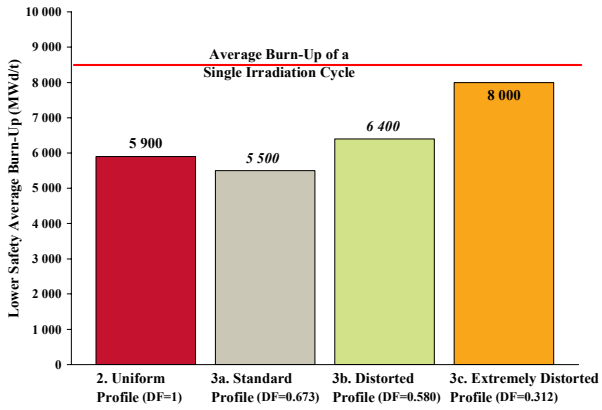

FIG. 6. Lower safety average burnup of fuel assemblies for calculation sequence 3.

4.2.4. Synthesis

The studies performed with an axial burnup profile leads to introduce a "Distortion Factor" -DF- to characterize axial profiles. This factor permits to know if the studied profile should be analyzed according to the assembly's end burnup or according to the assembly's average burnup.

The three axial profiles studied present a DF between 0.31 (theoretical extremely distorted profile) and 0.67 (standard profile). This wide range of DF should bound "real" profiles.

The results obtained show the feasibility of the extension to fuel assemblies initially enriched at 5.0 wt% of the field in which burnup measurements are not necessary for safety reasons before the fuels' receipt.

The question is now about the safety validation of the DF of the received fuel assemblies. This validation may consist in:

- a theoretic study of axial profiles giving the bounding DFs,
- an additional safety factor on the burnup of a single irradiation cycle,
- a previous measurement of the DF of the received fuel assemblies.

5. Conclusion

From a criticality-safety point of view with the current approach of safety analysis, the receipt and the storage in COGEMA/La Hague pools of spent UOX PWR fuels with a high initial enrichment require a burnup measurement of the fuel assemblies to be made prior to their reception.

This study makes use of an "Extensive Burnup Credit" method (9 actinides and 15 Fission Products) based on conservative assumptions (with regard to depletion calculations and nuclides' qualification) according to the French Working Group on Burnup Credit and considering an axial burnup profile (with a safety methodology). This calculation approach for burnup axial profile problems presented here should nevertheless be supported by additional studies [10].

The calculations made points out the effect of the main hypotheses assumed for this Extensive Burnup Credit. Moreover, the results show the feasibility of the extension to fuel assemblies initially enriched at 5.0 wt% of the field in which burnup measurements are not necessary for safety reasons before the fuels' receipt.

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Swedish Nuclear Fuel and Waste Management, SKB — burnup credit in the Swedish system for management of spent nuclear fuel

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Abstract. Swedish Nuclear Fuel and Waste Management, SKB, is performing studies concerning burnup credit with the objective to increase the capacity in the transport system and in the interim storage for spent nuclear fuel (CLAB). In CLAB the enrichment is presently limited to 4.2 % U235. Burnup credit could be used to increase this limit to 5 % U235. Burnup credit could also reduce the rigorous administrative control of fuel parameters that has to be performed before storage at CLAB.

For final disposal of spent nuclear fuel, burnup credit is needed in the disposal canisters to assure safe storage. A preliminary study has shown that burnup credit will give sufficient margin to meet the criticality criteria in the encapsulation plant and in the final disposal.

1. Introduction

SKB is managing the radioactive spent nuclear fuel, SNF, from the Swedish nuclear power plants. The system for management and disposal SNF includes:

Transportation system

M/s Sigyn – a specially built ship for transportation of the SNF.

Transport casks TN17/2

Land transport vehicles for TN17/2

Interim storage facility

CLAB – Central interim storage facility for SNF. CLAB was completed and ready for operation in 1985. In CLAB the spent fuel is stored in water pools. The storage capacity is

8000 tons HM.

Encapsulation plant

SKB's plans for management and disposal SNF include building an encapsulation plant. In the encapsulation plant the fuel will be placed in copper canisters prior to final disposal. This facility is planned to be in operation 2017.

Deep repository

The disposal canisters will be deposited a deep repository in the bedrock. The site selection process is ongoing.

2. Nuclear programme

In Sweden there are presently 10 reactors in operation, 7 BWRs and 3 PWRs. Two BWR-plants have been shut down due to government decision.

Assuming 40 years operation time for the reactors this programme will produce around 37000 BWR-assemblies and around 5000 PWR-assemblies.

Presently there are 18351 BWR-assemblies and 2321 PWR-assemblies in storage in CLAB (June 2005). There are 27 different fuel designs stored in CLAB.

3. Limits in the present system

From a criticality standpoint the following limits exist:

- In the transport casks the enrichment is limited to 4.5 % enrichment assuming fresh fuel
- In CLAB the limiting enrichment is 4.2% assuming fresh fuel for PWR-fuel and BA-credit for BWR-fuel

In disposal canisters for final storage the present enrichment limit is 4.2 % with burnup credit.

It should be noted that the criteria for transport and storage require control of several parameters besides from the enrichment. The system for control of the criteria gets more complicated as the number of different fuel designs is increasing.

4. Requirements

The power plants indicate the need for increase in enrichment up to 5 % U235. Development in fuel design gives very complex fuel designs.

5. Burnup credit

In 1990-91 different strategies to increase the storage capacity in CLAB were studied. The main strategies were to make the storage pattern denser using burnup credit or using borated steel in the storage canisters.

The result of the evaluation at that time was that burnup credit would not give a sufficient reactivity margin.

The conclusion of the study was that burnup credit alone was not a suitable way to control reactivity in CLAB. Instead borated steel was used in the storage canisters.

Presently burnup credit is studied for the Swedish system in the following steps:

Transport system

Burnup credit has been used for TN –casks several years. A preliminary assessment from Cogema indicates that the enrichments up to 5% could be handled with actinide only burnup credit in TN17/2. The required burnup is estimated to be less than 10 000 MWd/tU.

CLAB

Today the enrichment limit in CLAB is 4.2%. A preliminary study shows that with small amount of BU-credit enrichments up to 5% could be accepted and the administrative controls significantly reduced. Verification of burnup by measurement will probably be required.

Preliminary limit curves for BWR- and PWR fuel are presented in Figs. 1 and 2.

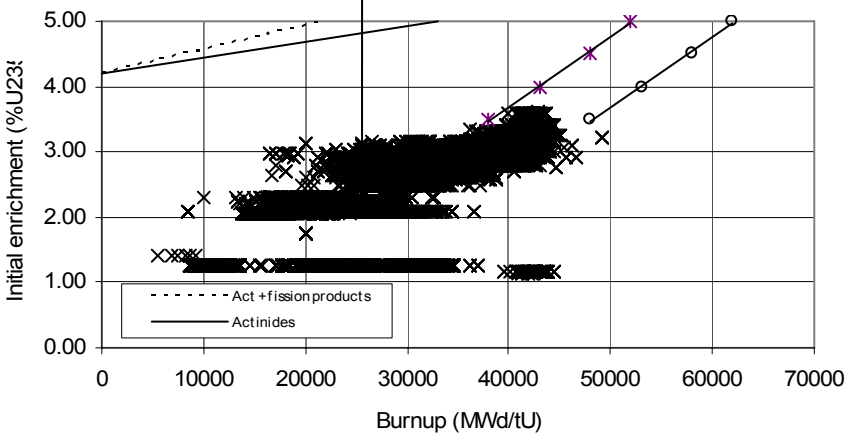

FIG. 1. Limit curves for BUC in CLAB compared to 17512 PWR assemblies.

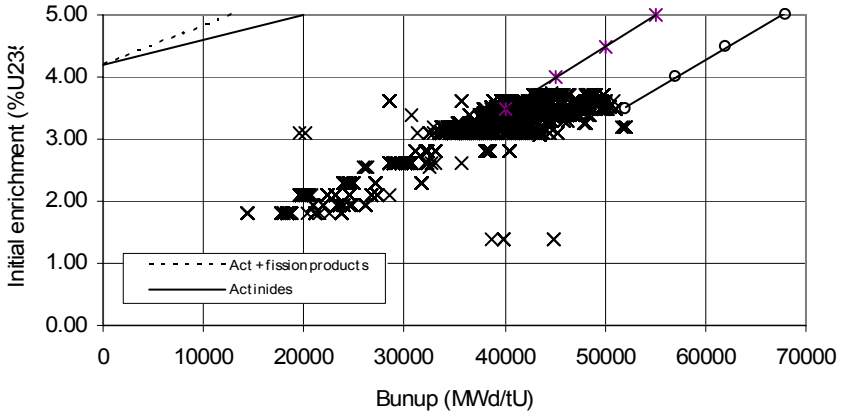

FIG.2. Limit curves for BUC in CLAB compared to 2021 PWR assemblies.

In the diagrams all fuel assemblies (combinations of initial enrichment and burnup) that appear on the right side of the limit curves will result in a $k_{eff} < 0{,}95$. It can be seen that all assemblies stored in CLAB at the end of 2004 meet the burnup criteria. This is evident since all fuel assemblies in CLAB meet the criteria for fresh fuel. More interesting is that the margin remains for enrichments over 4.2 % up to 5 % compared to the projected enrichments and burnups for future fuel assemblies.

Canister for final disposal

For the canister for final disposal burnup credit is necessary to maintain subcriticality with the current design.

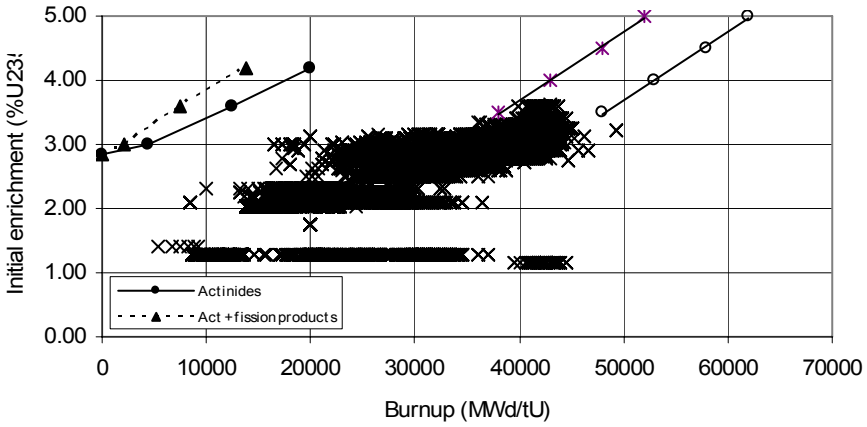

FIG. 3. Limit curves for BUC in final storage canister compared to 17512 BWR assemblies.

In Figure 3 all BWR - fuel assemblies (combinations of initial enrichment and burnup) that appear on the right side of the limit curves will result in a keff < 0,95. It can be seen that all BWR-assemblies stored in CLAB at the end of 2004 could be accepted for storage in canisters for final storage using burnup credit for actinides only. This conclusion is valid for projected enrichments and burnups for future fuel assemblies.

If additional actinides and fission products are included more margin is obtained.

In Figure 4 most of the PWR fuel assemblies appear on the right side of the limit curves and will result in a k_{eff} < 0,95. All PWR-assemblies stored in CLAB at the end of 2004 except for 20 could be accepted for storage in canisters for final storage using burnup credit for actinides only. If additional actinides and fission products are included all assemblies meet the criteria. The same conclusion is valid for projected enrichments and burnups for future fuel assemblies.

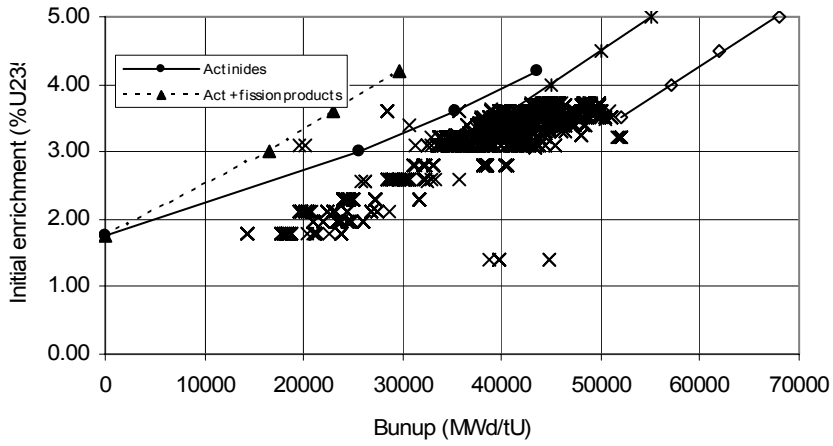

FIG. 4. Limit curves for BUC in final storage canister compared to 2021 PWR assemblies.

6. Conclusion

Burnup credit is judged to be a feasible way to handle fuel with enrichments up to 5% in the transport system and in CLAB. This strategy would also simplify the system for criteria control before transport.

For the canister for final disposal of spent fuel burnup credit is necessary to control the reactivity. Based on preliminary studies it is judged that actinide only would give sufficient margin.

Spent fuel management in the Slovak Republic

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Abstract. Presentation describes the spent fuel management in the Slovak Republic with reference to possibility of burnup credit using. First experiences with spent fuel were gained in the seventies. Spent fuel form A-1 NPP was handled at Jaslovske Bohunice site, in order to prepare the spent fuel for the transport to the former USSR. After shut down of the A-1 NPP, all spent fuel was transported to the USSR. In 1978 first unit of V-1 NPP was set into operation. Actually there are six NPP units of the WWER-440 type at Jaslovske Bohunice and Mochovce sites in operation in the Slovak Republic. In 1988 an Interim Spent Fuel Storage Facility was build at Jaslovske Bohunice site. In 2004 Nuclear Regulatory Authority of the Slovak Republic approved transport container C-30 for transport of forty-eight spent fuel assemblies partially using burnup credit in decision making for emergency conditions criticality calculations. Following the development in spent fuel storage area Nuclear Regulatory Authority of the Slovak Republic started support programs in order to verify and validate burnup credit and its components. Burnup credit will be an important element in solving future spent fuel transport and storage tasks.

1. Introduction

Usage of nuclear energy started in the Slovak republic in1972 by commissioning of the first NPP A-1 at Jaslovske Bohunice site. Fuel assemblies of this heavy water moderated gas cooled reactor were assembled from fuel roads transported from the former USSR. A-1 NPP was definitely shut-down after serious accident in 1977.

After construction of WWER-440 V-1 in 1978-1980 and V-2 in 1984-1985 NPPs the spent fuel was stored in the Interim Spent Fuel Storage Facility Jaslovske Bohunice (ISFSF) for 10 years before transportation to the Soviet Union. ISFSF is in operation at Jaslovske Bohunice site since 1987. Before 1987 a small number of spent fuel assemblies from WWER-440 reactors were also transported to the Russian Federation. During 1997-2000, the ISFSF was subject of a reconstruction and seismic upgrade.

The at-reactor spent fuel storage pools are used for temporary storage of the spent fuel after its terminate reloading from the reactor core. The spent fuel is stored in a grate and cooled by water with presence of the boric acid. In Mochovce NPP, the at-reactor spent fuel storage pools were made more compact, and the lower grate capacity is almost double compared to the V-1 or V-2 NPP.

After at least 2.5 years of storage in the at-reactor spent fuel storage pools, the spent fuel is removed to the C-30 transport container and transported to the ISFSF. The assemblies can be put into the C-30 container either in the compact cask KZ48 or in the T-12 or T-13 casks.

In 2001, Slovak Electric Joint Stock Company decided on the dry ISFSF construction on Mochovce site. This ISFSF will accommodate the spent fuel from the Mochovce NPP and shall be commissioned in about 2010.

2. Burnup credit

2.1. C-30 spent fuel transport container licensing

Transport of spent nuclear fuel started in Slovakia in seventies. Spent nuclear fuel was transported to the USSR. The original enrichment of nuclear fuel used in WWER-440 units was 1.6, 2.4 and 3.6 % of U235. Since end of nineties a new nuclear fuel with average enrichment 3.82 % of U235 started to be used for refueling. For the future the enrichment up to 4.38 % of U235 is planned for use in WWER-440 power reactors. C-30 transport container was licensed only for transport of 30 irradiated spent nuclear fuel assemblies with original enrichment up to 3.6 % of U235. A new license became necessary.

In 2001 Slovak Electric joint-stock company applied for a new license. Application was thoroughly reviewed by the Nuclear Regulatory Authority of the Slovak Republic. As a result of this review additional requirements were submitted. In 2004 all requirements were met. A new license — Decision Nr. 123/2004 — has been issued.

Chapter 9 of the Preliminary Safety Report of C-30 transport container describes the results of criticality calculations for spent fuel, both for operational and emergency conditions. KZ 48 cask consists of 48 hexagonal boron steel tubes with a pitch of 168 mm. The presence of boron in the tubes ensures satisfactory subcriticality in all conditions for the initial enrichment up to 4.45 % of U235. Construction of T-12 and T-13 casks differs from KZ-48. Both casks consist of grate with a pitch of 225 mm. For emergency conditions of the cask T-12, namely fall-out of the cask and grate deformation, the k_{ef} will be greater than 1. Therefore new criticality analyses with new parameters have been made. The initial enrichment was determined to 3.87 % of U235, average burnup to 19.265 $MWd.kgU^{-1}$, and in order to have more conservative results only some actinides were taken into account (U235, U236, U238, Pu239, Pu240, Pu241 and Pu242). The other actinides and fissile products were excluded. Using burnup credit the subcriticality condition has been met.

3. Research and Development

Nuclear Regulatory Authority of the Slovak Republic warrants various research tasks under the R&D program. The Division of Nuclear Materials prepared a task of the BUC application in the criticality calculation of the WWER-440 fuel assemblies in cooperation with Nuclear Power Plants Research Institute (VUJE). VUJE will perform this task in 2005 through 2007. The following subtasks will be addressed under this research task:

Verification of SCALE 5.0 calculation system

The aim is to verify applicability of the latest version of the SCALE 5.0 calculation system to the WWER-440 spent fuel storage and transport. It will consist of the SCALE 5.0 system testing during the calculations of criticality nuclide composition and residual heat of the WWER-440 fuel and verification of the system applicability by means of the results comparisons with the ones of the numerical models.

Methodology of the burn-up credit for the WWER-440 fuel

The aim is to develop appropriate methodology of the burn-up credit application for the WWER-440 fuel. It will consist of the proposal of the calculation analyses range in order to ensure sufficient subcriticality during the WWER-440 spent fuel storage and transport.

Application of the burn-up credit for the dry storage conditions of the WWER-440 fuel

This task should demonstrate that when the burn-up consequences are partially taken into account, it significantly decreases requirements on the WWER-440 spent fuel storage under dry conditions. The

task puts an emphasis on the burn-up credit analysis for the dry storage of the WWER-440 spent fuel. The results will serve for validation of the basic parameters of the Mochovce dry store.

Application of the burn-up credit for the wet storage conditions of the WWER-440 fuel

The aim is to examine possibilities of the WWER-440 spent fuel storage and transport with higher original enrichment in the existing storage and transport facilities. It will consist of the analysis of the possibility to transport and store the WWER-440 spent fuel with original enrichment up to 5% U235 in the existing C-30 transport container with T-12 or KZ48 casks and in the at-reactor spent fuel storage pools.

In 2004 Nuclear Regulatory Authority of the Slovak Republic started in co-operation with Department of Nuclear Chemistry of Komensky University in Bratislava a new scientific program. The program should compare isotopic composition of WWER-440 spent fuel calculated by SCALE 5.0 software with isotopic composition received by chemical analysis. The program should be finished at the furthest in 2007.

With reference to above mentioned information we hope that in 2007 all conditions for burnup credit using will be fulfilled. The amount of spent fuel increases, today we have about 10.000 spent fuel assemblies stored in ISFSF Jaslovske Bohunice and in at-reactor pools of six working units. Each year we produce other 500 assemblies. In 2004 more than 56 % of all electricity produced in Slovakian power plants was produced by nuclear power. In the future the proportion of nuclear power in all electricity produced will be rather stable. The usage of burnup can and will be an important element in solving future spent fuel storage problems.

4. Conclusions

In 2004 for the first time the burnup credit has been used in decision making for the spent fuel transport criticality calculation in the Slovak Republic.

The most recent information obtained mainly thanks to the cooperation with IAEA convinced us that the application of the burnup credit for the spent fuel is not only possible but even desirable in Slovakia. We hope this fruitful cooperation will continue.

Implementation of burnup and control rod credit for storage of spent nuclear fuel in Ukraine

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Abstract. Preliminary analysis of the regulations in force in Ukraine concerning nuclear safety of spent nuclear fuel management systems shows that some regulatory requirements in force are too conservative in view of current international practice. The extent of conservatism can be determined and reduced, if necessary, only using calculated studies for analyzing the criticality of spent nuclear fuel management systems. Such activity is consistent with the requirements posed by state-of-the-art production requirements. However, this can be only based on improving our level of understanding the processes occurring in nuclear dangerous systems and improving our capabilities as regards accuracy, correctness, and reliability in numerical modeling these processes. This work was intended to demonstrate that the excessive conservatism laid previously into the requirements on nuclear safety in Ukraine due to insufficient development of means for modeling processes in nuclear fuel can be considerably decreased through using more real modeling fuel systems. If such modeling is performed with the use of state-of-the-art software and computers, based on more complete understanding the processes in fuel systems, then removal of the excessive conservatism does will not reduce the safety of nuclear dangerous systems.

1. Ukraine approach to criticality risk assessment

Nuclear safety of fresh and spent fuel is assessed in compliance with current technical regulations, among which the following documents should be singled out:

«Safety Rules for Storage and Transportation of Nuclear Fuel at Nuclear Power Facilities, PNAEG-14-029-91».

«Basic Rules for Spent Nuclear Fuel Intermediate Dry Storage Facilities Safety Evaluation, NP 306.2.105-2004».

According to this documents, the effective neutron multiplication factor K_{eff} must remain below 0.95 in normal operation and design-basis accidents. Taking this into account, nuclear safety analysis — or more specifically, calculations to justify nuclear safety — should be based on the following conservative initial conditions:

(1) Optimal density of the moderator (water):

- Equipment must be designed so that the effective neutron multiplication factor should not exceed 0.95 even if the equipment is filled with water and in case of such water amount, distribution and density resulting from initiating events which lead to the maximum value of K_{eff}

- Such amount, distribution and density of the moderator (water in particular) should be taken into account which leads to the maximum value of the effective neutron multiplication factor as a result of initiating events.

(2) In case of storage facilities with homogenous absorber (for example, borated water), it should be assumed that the absorber is absent

(3) Absence of removable absorbers:

- In case of fuel assemblies containing burnable absorbers, it should be assumed that absorbers are absent.

- The presence of absorbing elements in fuel assemblies or storage racks should be neglected if these absorbers are not fixed or if their effectiveness decreases as a result of initiating events.

(4) The nuclear safety analysis should consider the maximum possible fuel enrichment:

- If there is fuel with different enrichments, fuel with the maximum enrichment should be considered.

(5) Nuclear safety analysis should be performed taking into account the inaccuracy of calculation methods, concentration and isotope composition of absorbers, tolerances in the production.

(6) It is necessary to tolerate the presence of the reflector and to consider the state which cause maximum K_{eff} in case of temperature change in normal operation conditions and due to initiating events.

(7) The potential for increase in the neutron multiplication factor in the process of nuclear fuel burnup as a result of nuclide composition changes associated with accumulation of nuclear fissile nuclides should be taken into account.

(8) Spent nuclear fuel should be regarded as fresh if the neutron multiplication factor decreases in burnup excluding cases when burnup is used as a nuclear safety parameter and is monitored with special instrumentation.

We should specially note the following fact, the neglect of which often leads to incorrect understanding of the above conditions. These conditions are not associated with modeling of any emergencies and, for this reason, cannot be analyzed in terms of the probability of their occurrence.

These are initiating conditions which should be used in the nuclear safety analysis for spent nuclear fuel dry storage facility.

The conservatism laid in these conditions is intended to compensate for the differences between the computer model and actual physical object.

2. Software used by SSTC NRS for criticality assessment

One of the most well-known programs for calculational analysis of nuclear safety for spent and fresh nuclear fuel management systems is the US SCALE program complex. At the US NRC request, this program has been developed and upgraded for more than 20 years by researchers at the US Oak-Ridge National Laboratory.

The SCALE program package was developed and validated, first of all, for calculations of PWR and BWR fuel systems.

During several recent years the program package SCALE has been actively used in our organization for expert calculations on nuclear safety. Criticality calculations, whose results are presented in this report, have been obtained with the SCALE-4.3 and SCALE-4.4.

Before using these codes in our activity they were successfully tested for calculation of nuclear safety (k_{eff}) as to fuel systems for VVER, and RBMK as soon as isotopic content of spent nuclear fuel from VVER. These results are published in:

> ***Y.Kovbasenko, V.Khalimonchuk, A.Kuchin, Y.Bilodid, M.Yeremenko, O.Dudka "Validation of scale control module CSAS26 for criticality safety analysis of VVER and RBMK fuel designs", NUREG/CR-6736.***

3. Zaporizhya NPP interim dry storage system for spent nuclear fuel interim dry storage VSC-VVER

The first Ukrainian interim storage system of spent fuel assemblies dry storage in ventilated concrete casks (VSC-VVER system) used at the Zaporizhya NPP is the direct modification of the VSC-17 and VSC-24 interim storage system used at US NPPs and has a license of supervisory bodies of the USA (NRC).

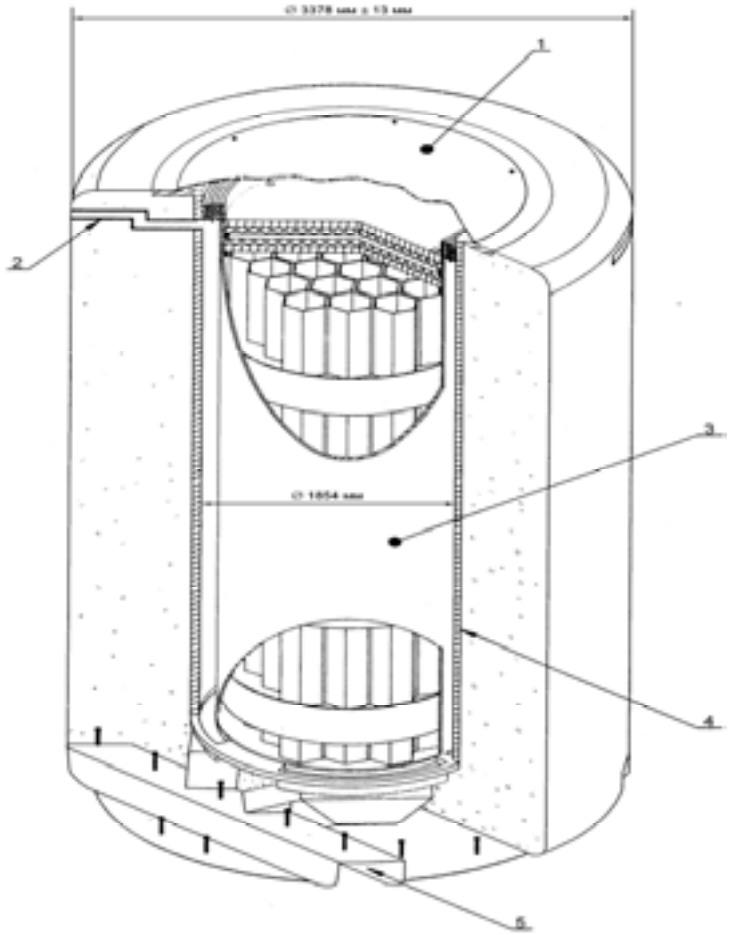
Spent fuel assemblies of PWR reactors are being stored in the USA for more than ten years in such systems. Now six US NPPs have such dry cask storage facilities. Canada, Germany, Switzerland, Great Britain are also the countries with the formed practice of spent nuclear fuel dry cask storage. Despite the comparatively short period of spent fuel assemblies storage in VSC-17 and VSC-24 casks there is already a significant scope of scientific and technical developments and results of the operational experience which demonstrate the safety of spent nuclear fuel long-term storage in such systems. The first VSC was commissioned at the Palisades NPP on May 1993. The correctness of computer calculation programs for thermal hydraulic, radiation analyses and strength analyses for the VSC was verified by experiments in the USA. The expansion of this positive experience to the VSC-VVER system for SFA of VVER-1000 reactors is an urgent question for Ukrainian NPPs.

Twenty-four spent fuel assemblies are stored in hexagonal tube covers (guide tubes) located in a cylindrical multi-place sealed storage basket made of the carbon steel. The multi-place cask storage basket is also a radiator which removes overheat from the spent fuel assemblies to the volume of the ventilated concrete shield cask. Cask basket filling with helium creates and maintains the dry inert media transmitting heat during the whole storage period.

The sealed steel cask basket is installed in a ventilated concrete cask which fulfills the following functions: removal of overheat from the cask basket; cask basket protection against climatic, mechanical, and thermal impacts; biological protection of the working personnel; assurance of stable vertical position of the cask basket with spent fuel assemblies during transportation and storage. The loaded ventilated concrete cask is installed on a special storage site which is located at the NPP territory with the proper security.

Safety assessment calculations VSC-VVER covered the following areas:

- assess the impact of boundary conditions. Select boundary conditions for subsequent calculations;
- calculate ventilated storage cask (VSC) criticality for normal operation;
- calculate VSC criticality on condition that the basket is filled with water-air mixture with various moisture concentration (mixture density varied from 0 to 1 g/cm3);
- calculate VSC criticality in case of displacement of assemblies from central position;
- calculate VSC criticality on condition that spent control rod clusters are located in assemblies and can be displaced in axial direction.



1 – cask lid; 2 – air outlet; 3 – sealed storage basket; 4 – coating; 5 – air inlet and guides for transportation

FIG. 1. VSC-VVER dry storage cask with VVER-1000 assemblies.

Conservative account of process tolerances for fuel enrichment (0.05%) and weight (4.5 kg) have been used in criticality calculations. Design-basis values were used for other characteristics of the assembly since account of other process tolerances does not result in noticeable changes in $K_{eff.}$

For designed parameters of VSC and normal operation (in particular, absence of moisture inside VSC), its criticality is as follows:

$$K_{eff} \pm \sigma = 0.3622 \pm 0.0009$$

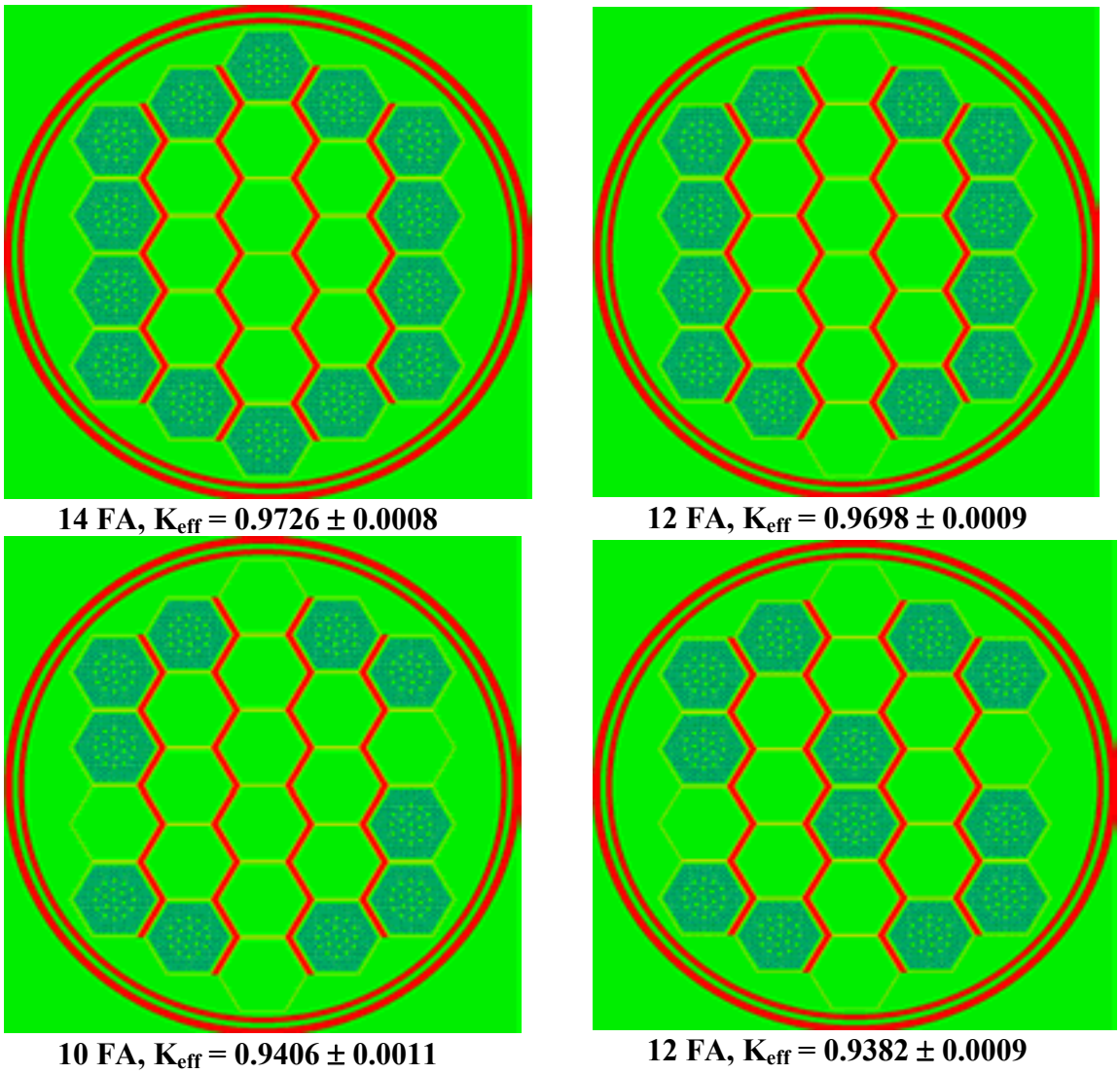
For optimal water density in VSC (which constitutes 1.0 g/cm^3 as confirmed by calculations), the neutron multiplication factor reaches the following value:

$$K_{eff} \pm \sigma = 1.1930 \pm 0.0008$$

In this regard, the calculations were intended to find the maximum fuel enrichment, when VSC is loaded with one-type fuel, at which the requirement of the regulatory document [3] $K_{eff}<0.95$ is met. The results presented in Table 1 show that the condition $K_{eff}<0.95$ with the optimal water density is only met when the cask is loaded with FA whose fuel enrichment does not exceed ~1.86%.

Table 1. K_{eff} depending on initial enrichment of loaded fuel (for optimal water density)

Initial enrichment, %	K_{eff}
1.6	0.9030 ± 0.0007
2.0	0.9761 ± 0.0008
3.0	1.0943 ± 0.0007
3.6	1.1426 ± 0.0009
4.4	1.1930 ± 0.0008



14 FA, K_{eff} = 0.9726 ± 0.0008

12 FA, K_{eff} = 0.9698 ± 0.0009

10 FA, K_{eff} = 0.9406 ± 0.0011

12 FA, K_{eff} = 0.9382 ± 0.0009

FIG. 2. Schemes of VSC partial loading with 4.4% fuel.

To ensure principle safety factors, a series of calculations were conducted to determine the VSC safe loading. Some combinations of such partial loadings and results obtained are shown in Figure 2 for water density of 1 g/cm^3 and boundary conditions of complete reflection.

Figure 2 show that the needed subcriticality level is achieved when the VSC are nearly half loaded. Therefore, for VSC successful safety substantiation, secondary means for criticality reduce must be maximally benefited and credited, such as use of control rods clusters, burnup credit, etc.

Two type of credit was analyzed:

(1) control rod credit and
(2) burnup credit

With optimal water density in VSC (which constitutes 1.0 g/cm^3 as confirmed by calculations) and without taking into account fuel burnup the neutron multiplication factor in case of VSC loading with fuel assemblies with control rod clusters reaches the following value:

$$K_{eff} = 0.9851 \pm 0.0009$$

The credit of fuel burnup results in reducing the neutron multiplication factor in the cask from 1.1936±0.0008 (fresh fuel without CPS rods) to 0.9416±0.0007 (burnup – 50 MW·d/kgU).

The results show that spent fuel assembly with initial enrichment 4.4% may not be installed without additional absorbers, such as control rods etc., even with burnup credit

Acceptable nuclear safety substantiation for the cask without its modernization is possible only if two thisauxiliary possibilities are credited in the nuclear safety analysis: burnup of fuel loaded and presence of control rods absorbers (Fig. 3).

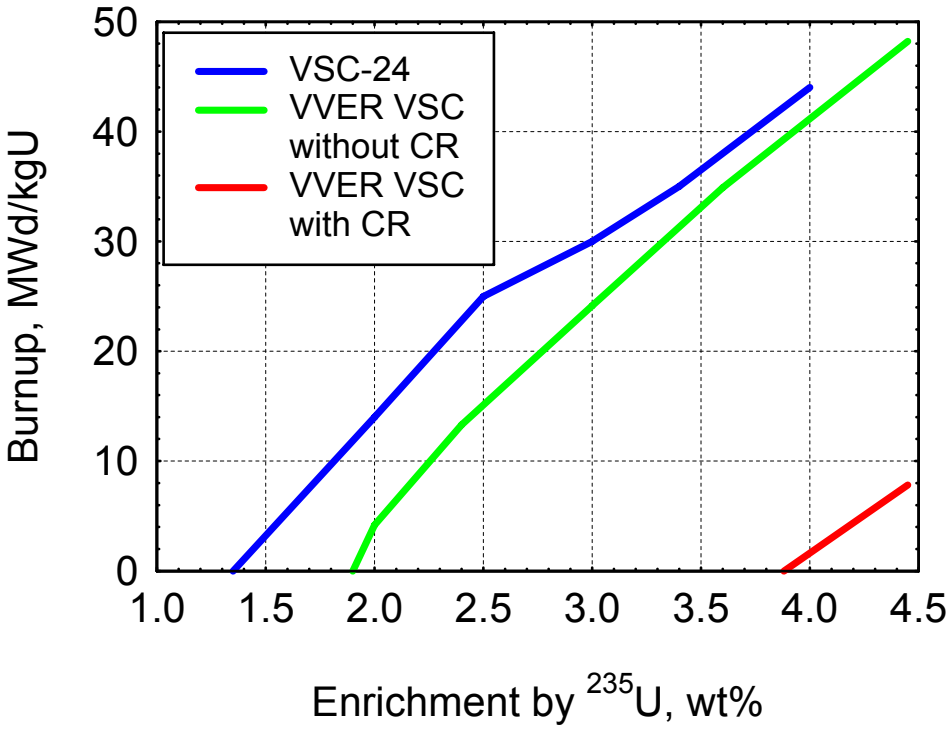


FIG. 3. Loading curve for VSC.

Based on results of the research, a part of which is presented in this work, two regulatory documents have been prepared and implemented.

The first one: ***"Branch Regulatory Document «Storage of Spent Fuel in VVER-1000 Ventilated Cask of Spent Fuel Dry Storage Facility. Authorization Procedure, Requirements on Documentation and Calculations of Neutron Physical Characteristics of VSC Loading at ZNPP Spent Fuel Dry Storage Facility»"***

The document applies to ventilated storage casks of the dry storage facility for VVER-1000 spent fuel at Zaporizhya NPP. The document establishes requirements on nuclear safety justification for loadings of VSC with sent nuclear fuel regarding:

- procedure for obtaining authorizations for loading of ventilated casks of the dry storage facility for VVER-1000 spent fuel;
- content of documents that justify the safety of VSC operation;

- scope of nuclear safety calculations for VSC loading.

The second document: ***«Methodology of Spent Nuclear Fuel Burnup Credit as Nuclear Safety Parameter for VSC Fuel Loadings at Spent Fuel Dry Storage Facility»***

The methodology establishes the work procedure and requirements in justification of nuclear safety of VSC spent fuel loading with taking into acount burnup credit:

- Procedure for spent fuel assembly burnup calculation. Procedure for calculation of spent fuel assembly conservative burnup profile;

- Selection and determination of concentration of radionuclides that are the main contributors to Keff

- Nuclear safety justification of VSC loading with credit of spent fuel isotopic composition;

- Requirements on mahematical model of VSC. Procedure of Keff calculation for fuel loading of VSC;

- Requirements on documentation and finalization of calculation results;

- The work procedure and requirements in monitoring of spent fuel burnup at the stage of VSC loading.

As known, burnup is not uniformly distributed along FA height (Fig. 4). To implement a conservative approach to nuclear safety analysis, in particular calculation of multiplication factor for VSC fuel loadings, two methods are applied to consider non-uniform fuel assembly burnup by height:

- burnup is accepted unchanged by height for each spent fuel assembly and equal to average burnup between lower and upper (least burnup)assembly parts;

- for spent fuel assembly of the same initial enrichment, conservative burnup profile is formed so that burnup of each part is equal to the least burnup of associated assembly parts.

In burnup of 50 MW day/kgU, difference in the multiplication factor for fuel assemblies infinite lattice with initial enrichment 4.4% calculated with account of only fuel isotopes and all isotopes (but without Xe) constitutes 14%. This value actually determines the additional safety margin which we incorporate in our justification in connection with possible errors in determining the concentration of U or Pu isotopes.

If there are no additional justifications, decrease of B_{10} concentration in control rods is conservatively accepted equal to 25% by the moment of their loading in multi-seat storage basket.

Human error should be considered in K_{eff} calculations by replacement of one spent fuel assembly by fresh one with 4.4% enrichment in loading – the replacement should lead to the maximum increase of K_{eff}.

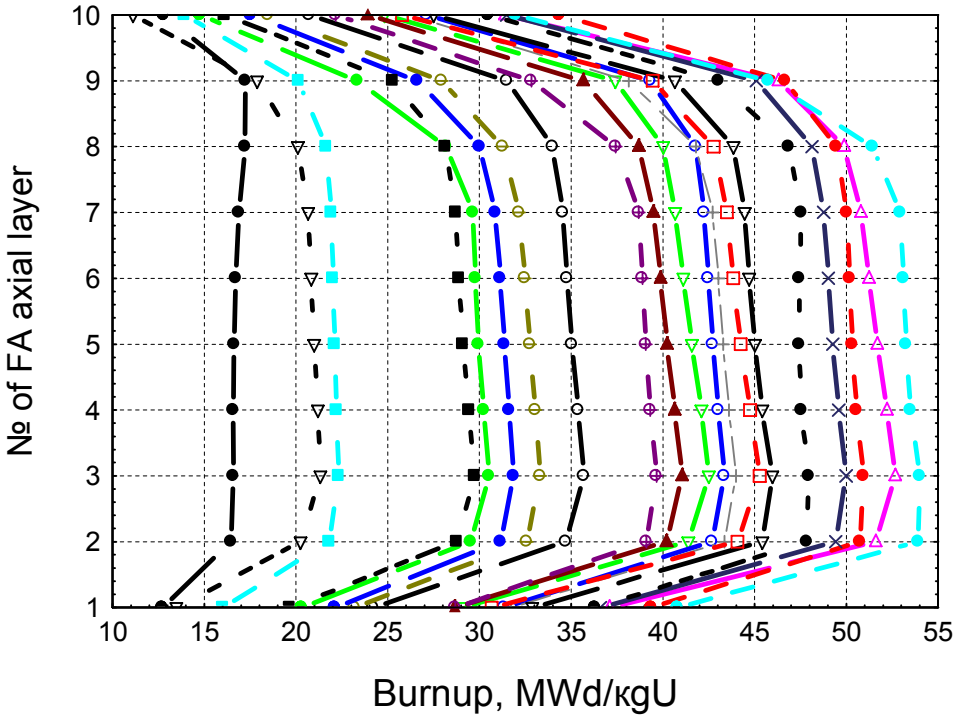


FIG.4. Typical distribution of VVER-1000 fuel burnup.

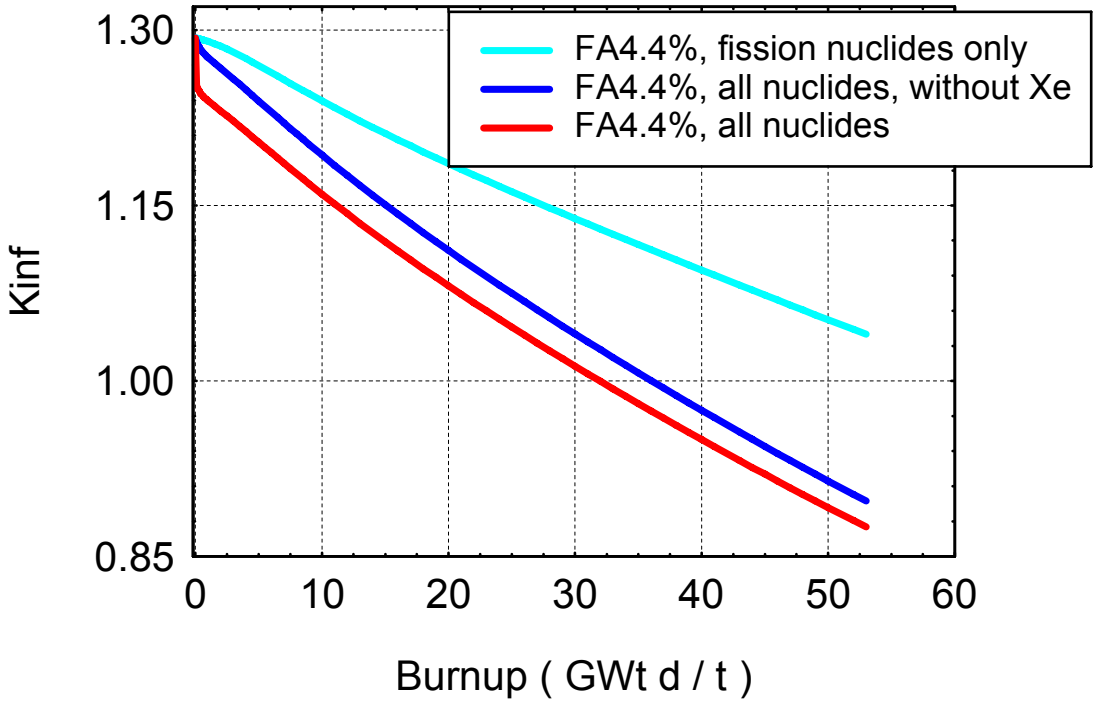


FIG. 5. Multiplication purposes of irradiation VVER-1000 fuel.

For successful safety substantiation VSC complete loading with spent fuel assemblies with control rods clusters, fuel burnup must be not lower than 10 MW·d/kgU (Fig. 3). For partial loading control rods, the admissible fuel burnup must be determined individually for each particular case.

In the framework of this methodology, changes in concentration of only fuel isotopes U-235, U-238, Pu-239, Pu-240, Pu-24 are taken into account in VVER-1000 spent fuel depending on its burnup (Fig.5).

4. Conclusions

This work was demonstrated that the excessive conservatism laid previously into the requirements on nuclear safety in Ukraine due to insufficient development of means for modeling processes in nuclear fuel can be considerably decreased through using more real modeling fuel systems. If such modeling is performed at the state-of-the-art level, based on more complete understanding the processes in fuel systems, then removal of the excessive conservatism will not reduce the safety of nuclear dangerous systems.

The burnup credit approach is one of the most advanced and widely used in international practice. But analyses and calculations related to its introduction mainly pertain to PWR and BWR fuel. This approach formally complies with Ukrainian standards in force; however, its application is greatly complicated in view of absence of necessary research results and regulatory practice.

The possibility of taking burnup credit into account for establishment of licensing requirements

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Abstract. Use of burnup credit (BUC) in compliance with appropriate regulatory requirements is an actual possibility that is taken into consideration in criticality safety analysis of transportation and dry storage of Spent Nuclear Fuel (SNF) from the Armenian Nuclear Power Plant (ANPP) under operation. The working-out of a relevant regulatory standard including a standardized BUC methodology is underway at the present time.

Problems and issues to be considered are:

- The political situation in the area of transportation of the SNF from the Republic of Armenia (RA)
- Storage issues:
  (a) the operational safety issues of wet storage,
  (b) the Independent Spent Fuel Storage Installations (ISFSI),
  (c) the decommissioning planning,
  (d) the final disposal of SNF.

1. Introduction

Issues which have to be taken into account are as set forth below:

- Storage capacity of the reactor pool (affected by the amount of discharged spent fuel assemblies and the cooling time applied)

- Installation of a new ISFSI (using the NUHOMS system for instance)

- Technical issues:
  (a) cladding integrity under normal (loading and storage) and abnormal or accident conditions,
  (b) thermal creep cracking and cladding damage,
  (c) mechanical failure: gas density, leakage of alpha-emitters, destruction of cladding (i.e. accidental condition and deviation from normal behavior for irradiated nuclear fuel assemblies),
  (d) long-duration thermal regimes.

2. The experience of the NUHOMS® Horizontal Storage Module (HSM) operating

A description of Armenian NUHOMS® HSM system is given in Ref. [1]. The system is in operation since 2000. Eleven Dry Storage Casks (DSC's) have been delivered and are loaded with spent WWER-440 fuel assemblies at ANPP site, each having a capacity for 56 Fuel Assemblies (FA's). The dates of loading the systems are given in Table 1.

The criticality safety analysis for these 11 systems was based on the "fresh-fuel-assumption". To maintain the maximum neutron multiplication factor below 0.95 under the worst condition (flooding of DSC with pure water), as required by ANRA, it was necessary to equip 24 of the 56 FA positions

per DSC with borated stainless steel channels [2]. The remaining positions are equipped with non-borated stainless steel.

Table 1. Schedule of NUHOMS-56 loading with ANPP spent fuel

DSC No.	Start of loading operation	End of loading operation	Date of loading to HSM
1	02.08.2000	09.08.2000	23.08.2000
2	25.08.2000	26.08.2000	06.09.2000
3	07.09.2000	07.09.2000	15.09.2000
4	16.09.2000	17.09.2000	29.09.2000
5	30.09.2000	30.09.2000	11.10.2000
6	03.05.2002	05.05.2002	16.05.2002
7	04.11.2000	04.11.2000	Loading from Reactor Pool 2 (RP-2) to Reactor Pool (RP-1)
	09.11.2000	09.11.2000	19.11.2000
8	29.05.2002	31.05.2002	11.06.2002
9	14.06.2002	14.06.2002	24.06.2002
10	29.06.2002	29.06.2002	09.02.2002
11	02.04.2003	03.04.2003	18.04.2003

3. Use of burnup credit

Instead of using borated stainless steel channels burnup credit may be introduced to comply with ANRA's criticality safety acceptance criteria. If burnup credit is introduced, then the question naturally arises as to what level burnp credit can be used (net fissile content level, actinide-only level etc, cf. Ref. [3]). In other words, what is the minimum burnup credit level required and what is the maximum burnup credit allowable with respect to ANRA's criticality safety acceptance criteria? To be able to give a sound answer to this question it has first to be clarified what the criticality safety acceptance criteria have to include when burnup credit is used. For this purpose it has

- to be determined which level of burnup credit is required to replace the borated stainless steel channels with non-borated ones,

- to be clarified whether the required burnup level is accessible to validation including

  (a) validation of the calculation methodology (depletion and criticality calculation tools)
  (b) validation of the procedure used for generating the information about the burnup of the FA's
  (c) verification of the FA's burnup information by measurement using a validated measurement procedure.

The criticality safety acceptance criteria which have to be worked out must include all steps of the spent fuel loading and storage operation:

- loading of a DSC in the borated wet storage pool of ANPP
- DSC in drying process
- dry DSC inside transfer cask [1]
- DSC inside HSM (normal operation conditions)
- DSC flooded with pure water (accidental condition).

Note that the Russian regulatory guide PNAEG G-14-029-91 allows application of burnup credit to storage of spent FA's in reactor pools, if the burnup of each FA is monitored by means of a validated measurement procedure.

If it turns out that use of burnup credit is not acceptable to ANRA then an alternative possibility may be to chose long-term wet storage and increase the storage capacity of the reactor pool therefore (e.g., by applying the second-floor-shelves technique or the use of partial boron credit for realizing appropriate changes of the storage geometry).

Whatever storage technique will be chosen, acceptance criteria for approval of long-term storage under dry or wet conditions must include, as mentioned in section 1 as well as in Ref. [2], consideration of the condition of the cladding material. Note that since the re-start of ANPP Unit 2 in 1995 all spent FA's were checked on cladding condition by applying a special technology and measurement equipment.

4. Conclusions

The process of working-out of a regulatory safety standard for applying Burnup Credit (BUC) includes clarification

- whether BUC can be applied to transport, storage (long-term dry storage in particular) or final disposal of the SNF,
- which level of BUC (fissile-only, actinide-only or actinide + fission products) can be used for transport, storage and final disposal of the SNF?
- what are the licensing requirements for transport, storage and disposal; in particular, what are the requirements for

  (a) validating the chosen BUC calculation procedure and calculation models?
  (b) validating the techniques used for determining and verifying the burnup of the SNF?

- and what are the criteria for approval?

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REGULATORY ASPECTS IN BURN-UP CREDIT

(Session 2.6)

U.S. regulatory recommendations for actinide-only burnup credit in transport and storage casks

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Abstract. In July 1999, the U.S. Nuclear Regulatory Commission (NRC) Spent Fuel Project Office (SFPO) issued Interim Staff Guidance 8 Revision 1 (ISG8R1) to provide recommendations for the use of burnup credit in storage and transport of pressurized-water reactor (PWR) spent fuel. Subsequent to the issuance of ISG8R1, the NRC Office of Regulatory Research (RES) has directed an effort to investigate the technical basis for extending the criteria and recommendations of ISG8R1 to allow improved implementation of burnup credit. This work sponsored by NRC/RES provided the reference material used by the NRC/SFPO to prepare Revision 2 of ISG8 (ISG8R2) that was released in September 2002. This paper discusses each of the six recommendations within ISG8R2 with specific emphasis on the changes implemented with ISG8R2 and the technical basis for the changes.

1. Introduction

The concept of taking credit for the reduction in reactivity due to irradiation of nuclear fuel (i.e. fuel burnup) is commonly referred to as burnup credit. The reduction in reactivity that occurs with fuel burnup is caused by the net reduction of fissile nuclides and the production of parasitic neutron-absorbing nuclides (nonfissile actinides and fission products). Historically, criticality safety analyses for transport and dry cask storage of spent nuclear fuel (SNF) assumed the fuel contents to be unirradiated (i.e. "fresh" fuel) compositions. In July 1999, the U.S. Nuclear Regulatory Commission (NRC) Spent Fuel Project Office (SFPO) issued Interim Staff Guidance 8, Revision 1 (ISG8R1), to provide recommendations for the use of burnup credit in storage and transport of pressurized-water reactor (PWR) spent fuel [1]. These recommendations were subsequently included in the Standard Review Plan for transportation casks and dry storage cask facilities [2][3]. Subsequent to the issuance of ISG8R1, the NRC Office of Nuclear Regulatory Research (RES) directed an effort to investigate the technical basis for extending the criteria and recommendations of ISG8R1 to allow improved implementation of burnup credit. The work sponsored by NRC/RES provided the reference material used by the NRC/SFPO to prepare Revision 2 of ISG8 (ISG8R2) [4], which was released in September 2002.

Similar to ISG8R1, the recommendations provided in ISG8R2 cover six areas:

(a) general information on limits for the licensing basis,
(b) guidance on code validation,
(c) guidance on licensing-basis model assumptions,
(d) guidance on preparation of loading curves,
(e) the process for assigning a burnup loading value to an assembly, and
(f) the benefit derived in demonstrating any additional reactivity margin beyond that which can be substantiated through the validation process.

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[1] Managed by UT-Battelle, LLC, under contract DE-AC05-00OR22725 with the U.S. Department of Energy

The remainder of this paper discusses each of these six recommendations, with specific emphasis on the changes implemented with ISG8R2 and the technical basis for the changes.

2. Limits for licensing basis

Similar to ISG8R1, the recommendations of ISG8R2 restrict burnup credit to actinide compositions associated with UO_2 fuel irradiated in a PWR. However, ISG8R2 provides additional ranges for the burnup, initial enrichment, and cooling times that can be considered in the safety analysis performed for the licensing basis. ISG8R1 recommended that burnup credit should only be taken for assembly-averaged burnups up to a value of 40 GWd/MTU and that fuel with initial enrichments between 4.0 wt % and 5.0 wt % have an additional margin of burnup (1 GWd/MTU for every 0.1% enrichment above 4.0 wt %) beyond that for which credit is taken. Since the issuance of ISG8R1, additional radiochemical assay data for PWR fuel have become available. Figure 1 shows that the range of existing radiochemical data that are readily available for validation now extends up to 47.3 GWd/MTU and 4.1 wt % initial enrichment. Risk-informed technical judgement indicates that trends in the calculational bias and uncertainty derived from this database (see Sect. 3) can be extended for use with SNF having initial enrichments up to 5.0 wt % and average assembly burnups limited to 50 GWd/MTU. Fuel with an average assembly burnup greater than 50 GWd/MTU can be loaded into a burnup-credit cask; however, based on the limited assay data available for validation, credit should only be taken for the reactivity reduction up to 50 GWd/MTU.

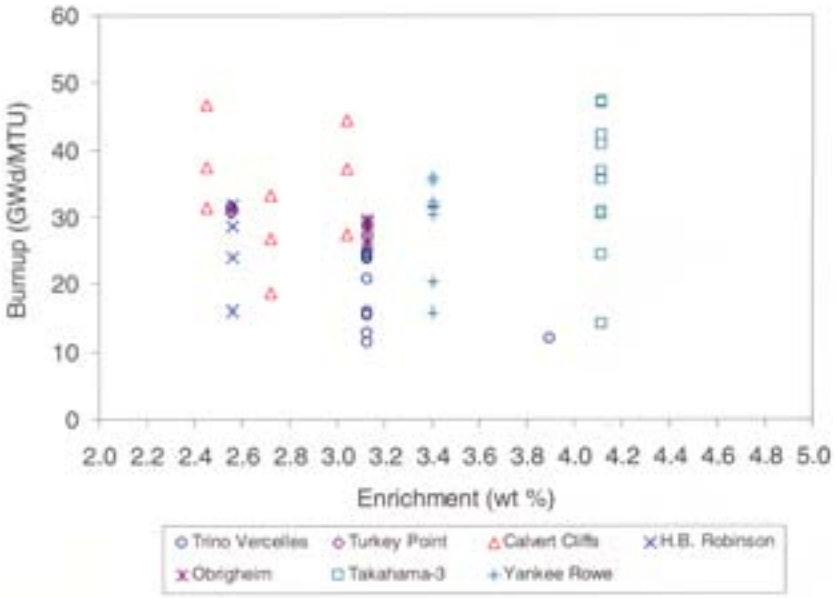

FIG. 1. Enrichment and burnup of 56 PWR assay samples available for burnup-credit isotopic validation.

Figure 2 illustrates the expected reactivity behavior for SNF in a hypothetical 32-element General Burnup Credit (GBC-32) cask, assuming use of major actinide concentrations in the calculation of k_{eff}. The fact that the reactivity begins to rise around 100 years after discharge means that the time frame for interim SNF storage should be considered in the evaluation of acceptable cooling times. The curve indicates that the reactivity of the fuel at 40 years is about the same as that of the fuel cooled for 200 years. The low probability that fuel in a storage or transportation cask would remain in place for more than 200 years led to the recommended limiting cooling-time criterion of 40 years (i.e. no credit for cooling time beyond 40 years should be taken). Approval of a cooling time longer than 5 years for burnup credit in dry storage or transportation casks does not automatically guarantee acceptance for disposal without repackaging. Reference [5] provides a comprehensive study of the effect of cooling time on burnup credit for various cask designs and SNF compositions.

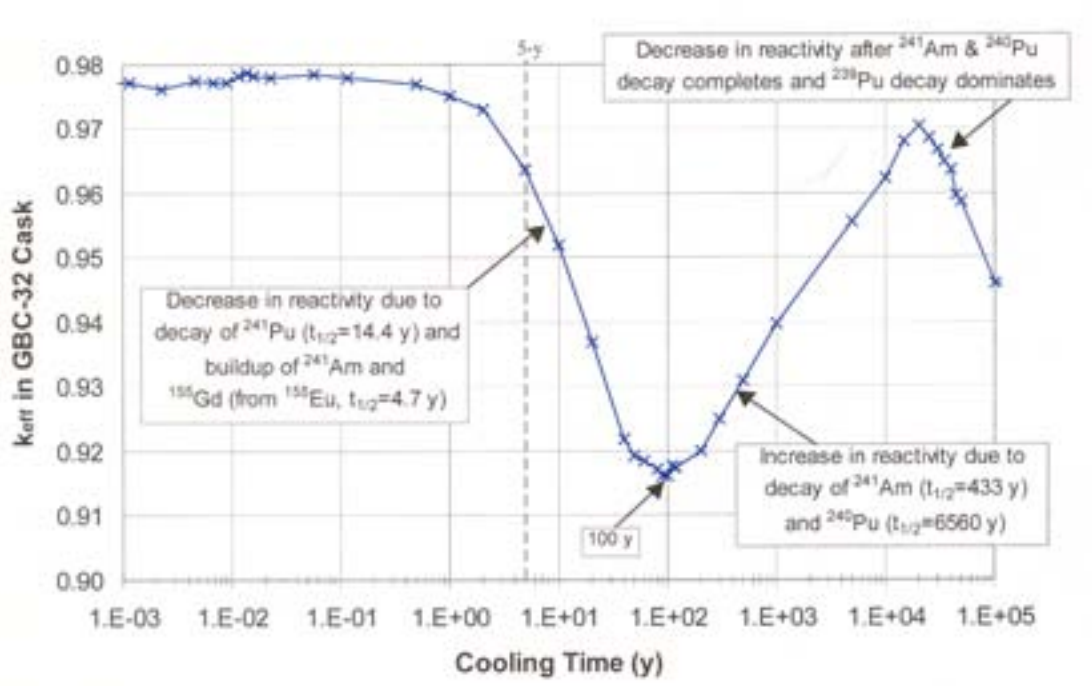

FIG. 2. Plot of k_{eff} in the prototypic GBC-32 cask using actinide-only assumptions for 40 GWd/MTU fuel with a 4.0 wt % initial enrichment.

The recommended acceptance criteria for burnup credit were set based on the characteristics of SNF discharged to date, the parameter space considered in the predominance of technical investigations, and the experimental data available to support development of a calculational bias and uncertainty. A safety analysis that uses parameter values outside those recommended by ISG8R2 will need to (a) demonstrate that the measurement or experimental data necessary for proper code validation have been included and/or (b) provide adequate justification that the analysis assumptions or the associated bias and uncertainty have been established in such a fashion as to bound the potential impacts of limited measurement or experimental data.

3. Code validation

ISG8R2 provides no substantive change in the guidance for code validation; the recommendation calls for validation of the analysis tools using measured data to determine appropriate bias and uncertainties. However, it was an examination of the available measured data and an evaluation of that data as it would apply to cask licensing that led to the extended burnup and enrichment limits of ISG8R2. The recommended credit for burnup is limited to 50 GWd/MTU because the assay data (e.g. Fig. 1) are not available to support development of a bias and uncertainty beyond this burnup without unwarranted extrapolation. From Fig. 1 it can be seen that the primary source of readily available assay data in the regime above 4.0 wt % and 40 GWd/MTU is from the Takahama PWR in Japan. Work reported in Ref. [6] has demonstrated that the standard deviations of the calculated-to-experimental nuclide ratios for the Takahama data are comparable with those observed for previous lower-enrichment and lower-burnup assay data. This lack of trending with burnup and enrichment was confirmed using different techniques for assessing the uncertainty and trends in the uncertainty. These findings are consistent with independent published results [7], in which use of French computational methods and JEF cross-section data to analyze assay data for PWR fuel with 4.5 wt % initial enrichment indicates a calculated-to-measured ratio comparable with that of lower-enriched fuel.

The methodology used to combine the biases and uncertainties for individual isotopes can have a significant impact on the final k_{eff} value and needs to be properly explained and justified. Reference [6] contains a description of various approaches that can be used to obtain estimates of the bias and uncertainty in the SNF compositions. The simplest approach is to individually adjust the concentration of each nuclide based on the results of the validation against radiochemical assay data. This adjusted

set of nuclides can then be used in the analysis of k_{eff} needed for the Safety Analysis Report (SAR). However, this process is conservative because each adjustment should be made so as to always create a more reactive system (e.g. fissile nuclides only adjusted to increase concentration and parasitic absorber nuclides only adjusted to decrease concentration).

A more realistic but more complex approach to incorporating bias and uncertainty from the SNF compositions is to use methods [6] that demonstrate how the uncertainty in the *combined* nuclide inventory propagates to an uncertainty in the k_{eff} value. The simplest way to implement this approach would be to first obtain the set of Δk values associated with separately changing the concentration of each SNF nuclide (only those used in the k_{eff} analysis) by the value of the bias and uncertainty in the prediction. Reference [6] indicates that a root-mean-square (RMS) summation of these individual Δk values provides an estimate of the uncertainty in the k_{eff} value due to the combined uncertainties in the inventory prediction. The impact on k_{eff} of the bias and uncertainty from the SNF concentrations is system dependent; thus, if a fixed Δk value (RMS-combined value of Δk for all nuclides) is used to account for the nuclide inventory uncertainties, the value must be obtained based on the cask design and contents specified. Propagation of the calculated inventory uncertainties into the criticality calculation representative of the cask configurations used in the SAR is the reason this approach is more complex and time-consuming to implement and review.

The RMS approach assumes the uncertainty for each nuclide is independent (i.e. random) and does not consider potential correlated uncertainties in transmutation and decay chains. However, the work of Ref. [6] shows that the use of several independent "best-estimate" approaches to predicting the uncertainty (e.g. use of RMS, use of Monte Carlo sampling from inventory calculated-to-measurement distributions, and direct use of measured and predicted assay data) provides similar estimates of the bias and uncertainty. This consistent estimation of the bias and uncertainty using various realistic approaches provides risk-informed confidence that the correlated uncertainties in the transmutation and decay chains have a minor impact.

The applicant is responsible for demonstrating that the experiments selected for the validation process are representative of the system (cask) of interest and that the code-to-experiment comparative information is utilized to estimate bounding values for the bias and uncertainty.

4. Licensing-basis model assumptions

This recommendation indicates that the actinide compositions used to determine a value of k_{eff} for the licensing safety basis should be calculated using fuel design and in-reactor operating parameter values that appropriately encompass the range of design and operating conditions for the proposed contents. Furthermore, the calculation of the k_{eff} value should be performed using cask models, appropriate analysis assumptions, and code inputs that allow adequate representation of the physics. This aspect is no different from the recommendation of ISG8R1. However, ISG8R2 goes further and provides additional guidance on selecting axial-burnup profiles and consideration of the impact of both burnable absorbers and control rods. In contrast, ISG8R1 included a restriction that assemblies exposed to burnable absorbers during irradiation not be considered eligible for loading in a cask designed for burnup credit.

4.1. Axial profiles

To support added guidance in ISG8R2, a review and evaluation of the publicly available U.S. database [8] of axial-burnup profiles were performed [9]. Although the database represents only 4% of the assemblies discharged through 1994, the review indicates that the database provides a good representation of discharged assemblies in terms of fuel vendor/reactor design, types of operation (i.e. first cycles, out–in fuel management, and low-leakage fuel management), burnup and enrichment ranges, and use of burnable absorbers. The primary deficiency in the database of Ref. [8] is the number of profiles associated with assembly burnup values greater than 40 GWd/MTU and initial enrichment values greater than 4.0 wt %. However, Ref. [9] indicates that a high probability exists that profiles providing the highest reactivity in intermediate burnup ranges will also provide the highest

reactivity at higher burnups. Consequently, by using risk-informed judgement along with the margin presented by isotopes not included in the analysis, the existing database should be adequate for burnups beyond 40 GWd/MTU and initial enrichments above 4%, if appropriate care is taken to select profiles that include a margin for the potential added uncertainty in moving to higher burnups and initial enrichments.

However, given the finite nature of the available database (4% of the inventory through 1994 discharge), a low probability likely exists that some discharged SNF would have a higher reactivity than the limiting profiles identified for the same burnup group. Using a generic burnup-credit cask model, Ref. [9] investigated the impact of loading single assemblies with a significantly more reactive profile and found the consequence to be small. Thus, the characterization of the limiting profiles from the database as statistical outliers, the use of a limiting profile for all assemblies loaded in the cask, and the low consequence associated with the loading of an assembly with a higher reactivity (beyond the selected limiting profile for that burnup group) have led to the recommendation that this publicly available database be accepted as an appropriate source for selecting axial-burnup profiles that will encompass the SNF anticipated for loading in a burnup-credit cask.

4.2. Burnable absorbers

Assemblies exposed to fixed neutron absorbers [integral burnable absorbers (IBAs)] and removable neutron absorbers [burnable poison rods (BPRs)] can have higher k_{eff} values than assemblies that are not so exposed, because the presence of the absorber will harden the spectrum and lead to increased ^{239}Pu production and reduced ^{235}U depletion. In addition, when removable neutron absorbers are inserted, the spectrum is further hardened due to displacement of the moderator. The lack of quantitative information on the effect of removable neutron absorbers caused the NRC to exclude assemblies irradiated with burnable absorbers as candidates for loading in a burnup credit cask.

Under the NRC/RES research program, investigations [10][11][12] have been performed to quantify how the k_{eff} value of a discharged assembly would change due to irradiation with BPRs and IBAs included in the assembly. A comprehensive range of assembly designs, absorber loadings, and exposure history was used to determine the impact on the k_{eff} value of SNF. The studies show that exposure to BPRs can cause the k_{eff} to increase up to 3% when the maximum absorber loading is assumed for the maximum exposure time. More typical absorber loadings and exposures (one cycle of 20 GWd/MTU) lead to increases of < 1% Δk (e.g. see Fig. 3). By comparison, except for one IBA type, where the increase was as much as 0.5% Δk, the IBAs actually provide a decrease in k_{eff} relative to assemblies not irradiated with IBAs. References [10][11][12] provide a general characterization of the effect of burnable absorbers on spent fuel and indicate that a depletion analysis with a maximum realistic loading of BPRs (i.e. maximum neutron poison loading) and maximum realistic burnup for the exposure should provide an adequate bounding safety basis for fuel with or without burnable absorbers. This result led to the recommendation included in ISG8R2 allowing assemblies exposed to burnable absorbers to be loaded in a burnup-credit cask.

4.3. Control rods

As with BPRs, control rods (CRs) fully or partially inserted during reactor operation can harden the spectrum in the vicinity of the insertion and lead to increased production of ^{239}Pu. In addition, CRs can alter the axial-burnup profile. In either case, the CR would have to be inserted for a reasonable fraction of the total irradiation time for these effects to be seen in terms of a positive Δk for the SNF cask. Domestic PWRs typically do not operate with CRs inserted, although the tips of the rods may rest at the fuel ends. However, some older domestic reactors and certain foreign reactors may have used control rods in a more extensive fashion such that the impact of CR insertion would be significant.

The results of a parametric study [12][13], to quantify the effect of CR exposure are summarized in Fig. 4, where it can be seen that even for significant burnup exposures (up to 45 GWd/MTU), minor axial CR insertions (e.g. < 20 cm) result in an insignificant effect (less than 0.2% Δk) on the k_{eff} value of a burnup-credit cask. However, Ref. [13] shows that full insertion for burnups up to

5–10 GWd/MTU provided an increase in cask k_{eff} values on the same order as seen for BPRs. Thus, since BPRs and CRs cannot be inserted in an assembly at the same time, it follows that the inclusion of BPRs in the assembly irradiation model (up to burnup values that encompass realistic operating conditions) should adequately account for the potential increase in k_{eff} that may occur for SNF exposed to CRs during irradiation.

Insertion of CRs (or use of axial power-shaping rods, APSRs) during reactor operation can also lead to a distorted, or nontypical, axial-burnup profile. However, as noted in the discussion of axial profiles, the existing database of axial-burnup profiles [8] includes a representative sampling of assemblies exposed to CRs and APSRs. In fact, many of the limiting profiles that exist in the database are from assemblies exposed to CRs and APSRs. Thus, the appropriate selection of a limiting axial profile(s) from the available database (or a similar one) would, in a risk-informed fashion, adequately encompass the potential impact for axial-profile distortion caused by CRs and APSRs.

5. Loading curve

A loading curve is a plot that specifies, as a function of initial enrichment, the assigned burnup value above which fuel assemblies may be loaded in the cask. Typically the personnel responsible for loading an SNF cask have ready knowledge of the average assembly burnup and initial enrichment values. Thus, a loading curve that provides the burnup and initial enrichment combination associated with the upper subcritical limit for the cask will provide a rapid means to assess whether a specific assembly is acceptable for loading in the cask. Separate loading curves should be established for each set of applicable licensing conditions. For example, a separate loading curve should be provided for each minimum cooling time to be considered in the cask loading. The applicability of the loading curve to bound various fuel types or burnable absorber loadings should be justified. To limit the opportunity for misloading, only one loading curve should be used for each cask loading. Each loading curve should be clearly marked relative to key assembly characteristics (e.g. assembly design type, cooling time, etc.).

6. Assigned burnup loading value

In Regulatory Guide 3.71, NRC endorsed the recommendations of ANSI Standard 8.17-1997, with the exception that credit for fuel burnup may be taken only when the amount of burnup is confirmed by physical measurements. Like ISG8R1, the new guidance of ISG8R2 indicates that a measurement to confirm the average burnup recorded for an assembly is needed prior to or during cask-loading operations. The administrative procedures for cask loading should include such a measurement and note that the uncertainty in the measurement and the uncertainty in the reactor records should both be included in adjusting the reactor record burnup to an assigned burnup loading value. The burnup measurement approaches proposed to date use measurements of numerous assemblies and comparisons with reactor record values to self-calibrate the system. Thus, the measurement and record for these types of systems are not independent, and the uncertainty in both should be considered in order to mitigate the potential for a systematic error in the reactor records. An assessment of the uncertainty of the burnup values provided in reactor records has been performed [14], indicating that uncertainties should be less than 5% for PWR assemblies.

ISG8R2 does indicate that procedures confirming the reactor records using measurement of a sampling of the fuel assemblies will be considered if a database of measured data is provided to justify the adequacy of the procedure in comparison with procedures that measure each assembly.

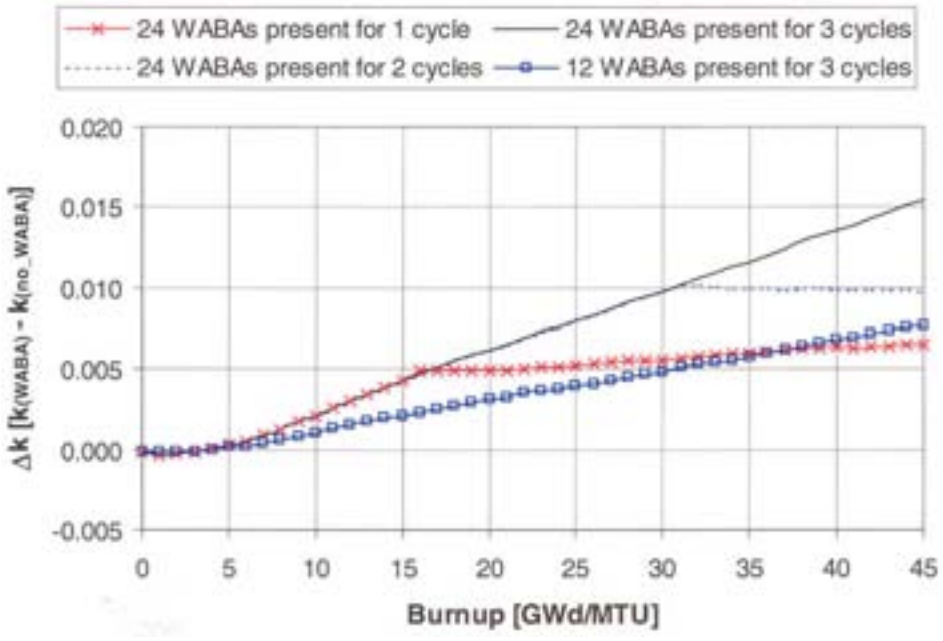

FIG. 3. Comparison of Δk values, as a function of burnup, for assemblies exposed to wet annular burnable assembly (WABA) rods. Results correspond to Westinghouse 17 × 17 assemblies with 4.0 wt % ^{235}U initial enrichment[11].

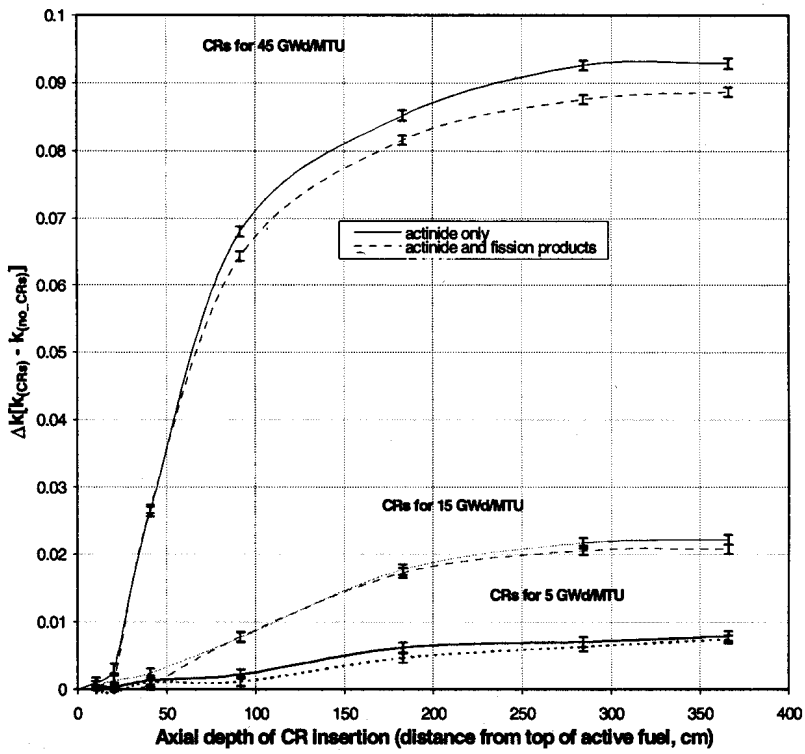

FIG. 4. Impact of CR insertion during irradiation on SNF in the GBC-32 cask [13].

7. Estimate of additional reactivity margin

As indicated in Ref. [6], the assay data available for fission-product nuclides are scarce relative to the data available for major actinides. In addition, the types of experiments (critical experiments, worth experiments, etc.) that may be needed to validate the reactivity effect from fission products are generally not publicly available and/or are difficult to use (e.g. reactor critical measurements and differential worth measurements). Thus, until additional data are available to validate the quantity of

the fission-product worth for a specific cask, the NRC staff has not recommended that the fission-product inventory be considered in the licensing basis safety analysis for burnup credit.

The fact that the neutron-absorbing properties of fission products are known to reduce the k_{eff} value beyond the actinide-only assumption indicates that the actinide-only assumption is conservative. However, the quantity of the conservatism cannot be well substantiated given the existing experimental and measurement data. Until additional experience is gained with the uncertainties associated with actinide-only burnup credit, an estimate of the additional reactivity margin that is available from nuclides not considered in the safety analysis may be used to compensate for uncertainties not readily understood or quantified in the safety analysis using only actinides. The estimate should be specific to the cask design because the margin will vary depending on the external absorbers in the cask basket. The estimation of additional reactivity margin should not be used to reduce the level of validation or realistic bounding assumptions used as a basis for safety. However, the information can be used to help justify that difficult-to-quantify uncertainties are adequately covered within the safety envelope of the cask design. Other easily identified conservative assumptions that may have been used in the licensing basis model can also be considered.

8. Summary

Revision 2 of the Interim Staff Guidance 8 expands the ranges of SNF parameters that can be considered in the safety analysis of a burnup-credit cask. Fuel with average assembly burnups to 50 GWd/MTU and initial enrichments to 5.0 wt % can be considered for loading in a burnup-credit cask. Cooling times from 1 to 40 years can be considered. In addition, ISG8R2 allows assemblies exposed to burnable absorbers to be considered for loading and recommends a methodology for accounting for CR insertions.

The six recommendations provided in ISG8R2 were developed with intact PWR fuel as the basis. An extension to damaged fuel may be warranted if the applicant can demonstrate that any additional uncertainties associated with the irradiation history and structural integrity (both during and subsequent to irradiation) of the fuel assembly (or parts thereof) have been adequately addressed. In particular, an appropriate model that bounds the uncertainties associated with the allowed fuel inventory and fuel configuration in the cask must be applied. Such a model should include the selection of appropriate burnup distributions and any potential rearrangement of the damaged fuel during normal and accident conditions. The applicant should also strive to apply each of the recommendations provided in ISG8R2 and discuss or justify any exceptions taken due to the nature of the fuel (e.g. the use of the recommended axial-profile database may not be appropriate).

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Guide for nuclear criticality safety analysis and review — Accounting for neutron irradiation and radioactive decay

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Abstract. The paper and presentation refer to work on a Swedish guide but limits the discussions to differences and news compared with guides published elsewhere. Accounting for irradiation and decay has been considered in criticality safety during transport and storage in Sweden during at least 30 years. Sweden has participated in OECD/NEA studies, IAEA technical meetings as well as in initiations of ANS and ISO standards for irradiation credit. In early 2005, the Swedish industry expressed intentions to prepare for irradiation credit in CLAB, at a reactor site and in transport. There is no time to wait for international standards being completed. A recent revision of a statute related to the Swedish nuclear energy law gives guidance on criticality safety. Defence-in-depth as well as deterministic and probabilistic safety analyses is required. Consideration of irradiation and decay does not change the fundamental criticality safety principles. A primary incentive for irradiation credit in CLAB is reduction of administrative controls. The criticality safety related fuel specifications are based on fuel as well as reactor design and operation. A safety analysis covering routine and normal operation together with human error, incidents and accidents can be made only when these fuel specifications are coupled with the proposed storage or transport designs and operations. Realistic fuel specifications, including fission products, need to be considered during irradiation simulation, k_{eff} validation and verification measurements. Integral reactivity effects of irradiation and decay during different conditions need to be evaluated. A licensing process of the expected application for CLAB storage will determine specific criteria and requirements that may be useful in international standards development.

1. Introduction

Accounting for reactor irradiation and radioactive decay in nuclear criticality safety during transport and storage of nuclear fuel in Sweden has been practiced or considered during at least 30 years [1]. Examples are irradiation credit for irradiated MTR fuel in transport to the U.S., gadolinium (a burnable absorber) credit in BWR storage, conceptual studies of final disposal of fuel and finally intermediate storage in CLAB (away-from-reactor underground storage facility) of BWR and PWR fuel. Sweden has been represented in OECD/NEA studies, IAEA technical meetings as well as in initiations of ANS and ISO standards for irradiation credit. The preparedness for new challenges is high.

In early 2005, the Swedish industry expressed solid intentions to prepare for irradiation credit in CLAB [2]. During an SKI seminar on irradiation credit [3], one of the reactor sites indicated strong interest in irradiation credit in the storage pools in the future. Further, the French package design used to ship irradiated fuel between Swedish power plants and CLAB includes some contents based on irradiation credit (not applicable to current CLAB shipments). There is no time to wait for international standards being completed; the Swedish authority SKI is prepared now for applications involving irradiation credit in LWR fuel storage and transport.

Literature of direct relevance to the issue of accounting for irradiation and decay has been compiled during SKI licensing and research during the years. The recently expressed interest for irradiation credit in Sweden has motivated a compilation and extension of this literature base. Some references

are still missing. Many documents in printed and electronic forms that did not motivate direct studies in the past are now found valuable.

A guide with criticality safety criteria for Swedish implementation of irradiation credit is being prepared. It will primarily be based on Swedish, often quite unique, conditions with consideration of standards, guides, experience and research in other countries and international organisations. This paper is not intended as a complete guide itself; it will focus on differences and new ideas for the planned guide compared with those expressed in many foreign documents.

2. Legal foundations

Safety is an important target, but not the only one. Laws and regulations as interpreted by various authorities also must be complied with.

A recent revision of a statute related to the Swedish nuclear energy law [4] gives guidance also on criticality safety. Defence-in-depth as well as deterministic and probabilistic safety analyses is required. The safety analysis shall cover prevention and mitigation (in case of failed prevention) of a criticality accident. Consideration of irradiation and decay does not change the fundamental criticality safety principles.

The legal requirements on criticality safety during transport are based on recommendations by IAEA [5]. Consideration of irradiation has always been required if the neutron multiplication factor k_{eff} can be increased due to irradiation (e.g. breeding and burnable absorber effects). Irradiation credit has also been allowed, but recently additional requirements were added to the regulations.

3. Focus on the issue: criticality safety

Criticality safety relies on many different types of control, often in combination. Examples are geometry, mass, moderation, reflection, density (covers separation of fissionable units), fixed and soluble neutron absorbers, isotope distribution (e.g. ^{235}U enrichment) and other concentration limits, irradiation (burnup is a crude measure of this), radioactive decay, etc.

Accounting for irradiation and decay is not fundamentally different from other criticality safety considerations. Identification of routine, normal (within limits), incident, accident and human error conditions and variations is required. Combinations need to be considered according to the combined probability. Some reference scenarios that are of general interest in irradiation and decay account are shown in Table 1.

Table 1. Some reference scenarios for determination of k_{eff}

No	Scenarios for determining k_{eff}	Comments
1	Fresh fuel without BA	Conservative for non-breeding fuel
2	Fresh fuel with BA	When applicable.
3	Routine conditions at various irradiation levels	Real fuel conditions, measurements, emergency: Best estimate.
4	If BA credit is not considered, missing BA must be accounted for	
5	Incident conditions at various irradiation levels, with and without BA	Credible enough to consider other incidents and human error
6	Accident conditions at various irradiation levels, with and without BA	Sufficiently unlikely to exclude other unlikely events or human error

An important issue is the representation of fission products and actinide nuclides that are not so common in nuclear criticality safety evaluations. Some are more or less volatile. Some may be redistributed within the fuel rod, even during routine reactor operation or subsequent storage and transport. A volatile fission product may leak from the system completely. Incidents and accidents will not necessarily involve the same effects.

Since scenario 3 in the table is very informative, it should always be evaluated. This requires some kind of representation of all fission products and actinide nuclides. It is a normal task for reactor core design and management software. Long-term storage requires additional representation of radioactive decay to what typical reactor operation requires.

Volatility and other properties of fission products and actinide nuclides under normal and accident conditions are subjects of many studies.

Another issue is the geometrical distribution of actinide nuclides and fission products. How much detail is required for different applications? Is the information from the reactor operation records sufficient, does it need to be increased or should it be condensed into a format that covers criticality safety requests?

Information about the real properties of the irradiated fuel, as opposed to conservative approximations, is important when measurements are carried out. Neutron and gamma irradiation levels and spectra as well as heat generation may be useful to verify the data used in the criticality safety assessment.

4. Examples of criticality safety control

4.1. Burnable absorber credit is a fixed neutron absorber credit

Burnable absorber (BA, e.g. gadolinium) credit is a credit for a fixed absorber in the system. In a BWR fuel assembly only a small fraction of the rods are of the BA type. Removal of those rods is possible at the site (BWR) and must be prevented by administrative measures. It would be an event that could lead to a criticality accident on its own. The BA rods may behave differently to normal rods during incidents at sites and during transport. In a PWR fuel assembly, some or all rods may contain BA, not necessarily in the full length of the assembly. This could cause sensitivity to certain configurations or incidents.

As a fixed neutron absorber credit, gadolinium can be combined with other fixed neutron absorbers such as boron in the storage rack or packaging.

4.2. Burnable absorber credit when irradiation is credible

Burnable absorber credit (gadolinium) in Sweden is applied to fresh fuel fabrication (BWR and PWR), transport of fresh fuel (BWR and a few times PWR) and in dry storage of fresh fuel (BWR) at power plants. It is not necessary to account for irradiation in those applications. However, in the storage pools at the power plants and in CLAB, irradiation must be accounted for. Irradiation depletes the burnable absorber together with the fissile material in the fuel. Irradiation is an additional parameter that must be accounted for, it is not an option. Removal of the BA from fresh or almost fresh fuel will still be the most serious event. This is independent on whether irradiation can increase k_{eff} or not.

The BA credit in the dry fresh fuel storage and transport packages is larger than in the irradiated fuel storage pools or transport packages. Transfer of irradiated fuel assemblies from a pool or a transport package to the dry fresh fuel storage or to a fresh fuel transport package would violate the criticality safety control. It is of course not a credible event.

The BA credit applied in Sweden so far has been quite conservative. The intent is to use the most reactive node of any credible fuel assembly under representative conditions and to apply this to all nodes in the assemblies of the application. All credible irradiation conditions must be accounted for.

This means that there is no need for control of the irradiation level. There is certainly no credit taken or given for burnup.

Even though the BA credit is smaller when irradiation is credible, the major hazard is still removal of BA rods from BWR fuel assemblies through human error or events.

4.3. Credit for irradiation and decay (burnup credit)

Average burnup is not a good measure of the irradiation of a fuel assembly. Even if it is known in every part of the assembly (axial and horizontal distributions) it may not be sufficient. Spectrum effects can make two assemblies with identical burnups quite different. Irradiation credit is a more proper term but burnup credit is so established that its use is difficult to avoid. The additional influence of radioactive decay after irradiation can be integrated with the terms irradiation credit and burnup credit. However, radioactive decay can also be a separate control by itself (e.g. based on measurements).

5. Irradiation simulation

Irradiation simulation is preferred to the more common terms depletion (depletion of uranium does not result in depleted uranium) or burnup (other irradiation aspects such as neutron energy spectrum are also important) calculations. Determination of the irradiated fuel properties is a very complicated task. Detailed information about the operation of the reactor is required. Information from various measurements is used by the reactor operator to support and validate computer simulations.

All nuclides that can influence the neutron flux significantly or that can be converted or decay into such nuclides need to be represented in the irradiation simulation. This representation may be achieved by “lumping” nuclides into artificial materials with appropriate properties. Lumping of fission products that works for reactor operation may not be sufficient for long-term storage.

The interface between reactor operation and criticality safety is very important. Irradiation accounting is one of the areas where reactor physicists and criticality safety specialists really need to understand each other. Since criticality safety is the issue, it is the responsibility of criticality safety specialists to learn about reactor operation and associated calculations and measurements. Human error, code and data errors, uncertainties, normal variations and documentation inaccuracies that are not important to reactor operation or safety may lead to significant errors in criticality safety applications.

Can the methods routinely tested in reactor core design and fuel management be used directly in criticality safety applications? Simplified criticality application models may be sufficient to evaluate the fuel properties in the real applications. Can the methods commonly used by criticality safety specialists be used to simulate the reactor operation? How much information about the fuel irradiation history is required in order to calculate its final contents after irradiation? Simplified irradiation models that give adequate results are requested.

6. Verification of the fuel properties

The issue of continuous verification of the actual conditions of the operation to assure that they comply with the specifications and limits is very important in all criticality safety control. Accounting for irradiation and decay may not necessarily be complicated but each application needs to be evaluated on its own merits. The need for verification is not new to irradiation credit. Chemical forms, enrichments, absorber distributions, etc. are parameters that need to be verified at many plants and shipments containing fissionable materials.

Accounting for irradiation in BA credit can be very simple. The behaviour of the fuel during reactor operation is well known from experience and from validated software simulation. If there is an increase in k_{eff} due to irradiation, it can be determined in a conservative way using simple methods. No verification of the irradiation level is needed in such simple applications.

In irradiation credit, some control will be needed. It can be extremely simple, such as avoiding taking fresh fuel from the dry storage and putting it in the transport cask intended for irradiated fuel. If fresh fuel is stored together with irradiated fuel in ponds, the verification required depends on how accurately the irradiation level needs to be known and how the storage and transfer of assemblies is controlled. Most complicated are cases where there are many levels of irradiation, many design limits for different racks and fuel assemblies and where frequent transfers are occurring.

At reactor sites all levels of irradiation must be accounted for (different cycles and interrupted cycles). At away-from-reactor storage it may be possible to separate the relatively few assemblies that may be outside the limits. On the other hand, there may be assemblies from many different types of reactors. The irradiation histories of the assemblies may not be so easy to verify at away-from-reactor sites.

Even if criticality safety can be assured by verifying the reactor records administratively, measurements may be justified if the administrative requirements are reduced significantly. Liquid or fixed neutron absorbers may be used for questionable fuel until measurements are carried out. An example of this could be unloading of a transport cask that has been involved in a transport accident.

7. Validation of methods for irradiation, decay and safety

Simulation of irradiation and decay needs validation. It requires accurate information about the reactor core and the reactor operating history. The real fuel composition needs to be determined as a function of space and time. The reactor operator will have methods that are validated concerning reactivity and various safety-related variables during operation of the reactor. Cross-section data and decay constants for individual nuclides are not necessarily accounted for or correct. Before the fuel specifications from such methods are applied to criticality safety, their accuracy needs to be validated. Further, they must be adequately understood by the criticality safety specialists in order to identify potential abuse or errors.

Even if the irradiation and decay simulation gives accurate compositions, the issue of validation of the criticality method remains. Results from critical experiments and reactor critical configurations with irradiated fuel are rarely published. Measurement information from the reactor operator may be required in order to get a irradiation credit license for the associated fuel.

8. Irradiation credit guides, studies and discussions

Even though irradiation credit as a concept has been applied a long time both in storage and in transport, serious and wide-spread discussions seemed to start about twenty years ago. The progress since then has been slow and not always straight-forward. Many of the problems that have shown up during the years are not restricted to irradiation credit. However, the effects are often larger and clearer than in previous applications using fresh fuel assumptions. Some of the problems were discussed in detail at the previous IAEA burnup credit meeting [1].

8.1. Combination of uncertainties – Independence, linearity?

The common view by criticality safety specialists on combinations of uncertainties in k_{eff} seems to be that they can be independent. This is not possible in a single system at a given time. An uncertainty represents a range of probabilities of occurrence combined with influences on k_{eff} and on the neutron flux. Even if the occurrences of uncertainties are independent, the influences on the neutron flux are never independent (there is only one neutron flux in a system). Thus the uncertainties are not independent. This is easy to realise by considering an extremely large uncertainty. It will obviously influence other uncertainties. A small uncertainty will have smaller influences on other uncertainties but independence is not possible.

The ANS 8.17 standard has recently been revised (2005). The 1984 edition stated that uncertainties can be combined statistically (square root of sum) if they are independent. Correlated uncertainties had to be combined additively. This requirement can be extremely conservative and was removed in the

2005 revision. However, the first option is still supported. This option is being discussed in the preparations of a new standard; ANS 8.27 concerning irradiation credit.

Without a very strong emphasis on linearity of uncertainties, even the additive combination of uncertainties can be totally incorrect, possibly leading to a criticality accident [7]. The whole issue if combining uncertainties in k_{eff} usually seems to be misunderstood. It is really the influence of the parameter uncertainty on the neutron flux that is the key to the effect of combined uncertainties. It is not necessarily the magnitude of the reactivity, not even the sign of the reactivity that determines the effect of a parameter change when combined with the effects of uncertainties in other parameters. The combined effect on k_{eff} can be much larger than the sum of the reactivities.

8.2. Large approximations in the fuel assembly composition — Axial distribution

The idea of using average irradiation or burnup levels in the whole assembly is still considered in some organisations. It seems to be either a very dangerous approach or a very conservative approach (if all problems are considered). The additional cases of Phase IIB of the OECD/NEA study [8] shows for PWR fuel how a credible human error or accident condition could cause criticality when the average burnup level would indicate safety. This has later been confirmed to be valid also for BWR fuel [1].

Since all irradiated fuel assemblies are different, using a single model for the fuel is an approximation. Whether it is conservative or not depends on the model and the application. In early studies, the uncertainty (a reactivity) of this approximation was discussed [9]. It was soon found that there was very likely a non-conservative bias in the approximation.

Even if the bias is zero or negative and the uncertainty very small, many other uncertainties are influenced very strongly by such an extreme and non-linear approximation. The idea that those uncertainties remain unchanged (independent), as discussed in 8.1 above, may explain surviving plans for using average uniform axial distributions with a bias correction rather than more realistic distributions.

8.3. Increase of k_{eff} due to realistic conditions (all fuel assemblies are different)

If the safety assessment is based on identical fuel assemblies in each calculation, it is possible that k_{eff} is seriously underestimated. When peak reactivity fuel is assumed, conservative models are easy to create. A fresh PWR assembly is more reactive than a fresh BWR assembly without BA rods. When irradiation is accounted for, this may not be true anymore. A fresh BWR assembly without BA can be more reactive than a highly irradiated PWR assembly. With irradiation credit, BA credit and fresh fuel mixed together, there are many possible variations. It is more difficult to find optimum configurations.

After all these years, the issue of mixed array problems [1][10] has still not been generally recognised for irradiation credit. In a Swedish application for irradiation credit, some consideration will be needed. In current safety evaluations for CLAB (fresh fuel for PWR, peak reactivity for BWR), the limiting case is a mixture of BWR and PWR fuel canisters under incident conditions. The effect may be small but the principle that mixed configurations need to be covered remains and needs to be emphasized in irradiation credit.

8.4. K_{eff}, fission density, fission density distribution, source convergence

The issue of source convergence and fission density distributions in connection with irradiation credit has been discussed previously [1]. A figure with examples of adequate normalised fission density distributions for uniform axial fuel distributions was shown. With consideration of the axial distribution, the normalised fission density distributions change drastically. Source convergence can be much more complicated and slower in irradiation credit than in many other criticality safety applications. Source convergence is covered in studies [11] by OECD/NEA Expert Groups (Burnup Credit and Source Convergence).

An adequate solution to a calculation of k_{eff} relies on an adequate convergence of k_{eff} itself and of source convergence of the fission density distribution. In a Monte Carlo calculation, the active tracking of neutrons and tallying of events (e.g. fissions) should only be started when source convergence has been obtained. How this source convergence is obtained is of some interest (time, cost, test of evaluator's understanding of the problem, test of the method) but it is not relevant for the result.

It is an engineering judgment rather than a numerical or statistical evaluation that determines if the source convergence is acceptable. A statistical or numerical test that depends on the geometry region size specifications (assuming the system is unchanged) is not reliable. Use of symmetry or division of each region into ten sub-regions should not influence source convergence judgments.

There seems to be a wide gap in the interpretation of source convergence between many specialists. Even more surprisingly, there also seems to be a gap in the interpretation of k_{eff}. The requested k_{eff} is a property of a system under specific conditions. It is not the expected result of a real event. The normalised fission density probability in a zone is also a property of the system. The fission density in a zone has no importance in itself (unlike in reactor operation). The normalised fission density distribution is very important and should approach the normalised fission density probabilities. The difference is accuracy and precision.

Even without a single fission anywhere in the system, k_{eff} remains the same physical constant. The normalised fission density probabilities are also pre-determined. In a calculation, since convergence the fission density distribution is already obtained before tallying starts, fission density per region is only depending on the precision requested in k_{eff}.

The fission density can be zero in one or more regions if the fission density probabilities are very low in those regions. Specialists in the OECD/NEA Burnup Credit Expert Group, Phase IIC and Phase IIE [11] consider use of the fission density in the least important region as a primary criterion for source convergence acceptability, while leaving the criteria for source convergence in the most important regions for later discussion.

In Phase IIC, 17 different axial burnup distributions, each with an average burnup 50 GWd/tU were specified. The axial distribution of the active part of the 21 identical fuel assemblies in the calculation cases was divided into 21 zones (different lengths). The total number of fissions was not given in the specifications. Calculations were recently completed of the 17 cases using SCALE 5.0 with KENOVa and the 238-group cross-section library and at least 475 million active neutrons (5 000 neutrons per generation, 100 000 generations, 5 000 skipped generations, neutrons started at the top of the fuel). The results using normal and logarithmic scales are given in Figures 1 and 2.

The fission densities in the figures are normalised to the sum of fission densities in the 21 different zones. The distribution is of interest, not the zone values in themselves. The normalised fission fractions were also requested in Phase IIC. Multiplication of these fractions with the net number of fissions, after source convergence is obtained, would give the observed number of fissions per zone. An observed fission is an integer (true also for deterministic calculations).

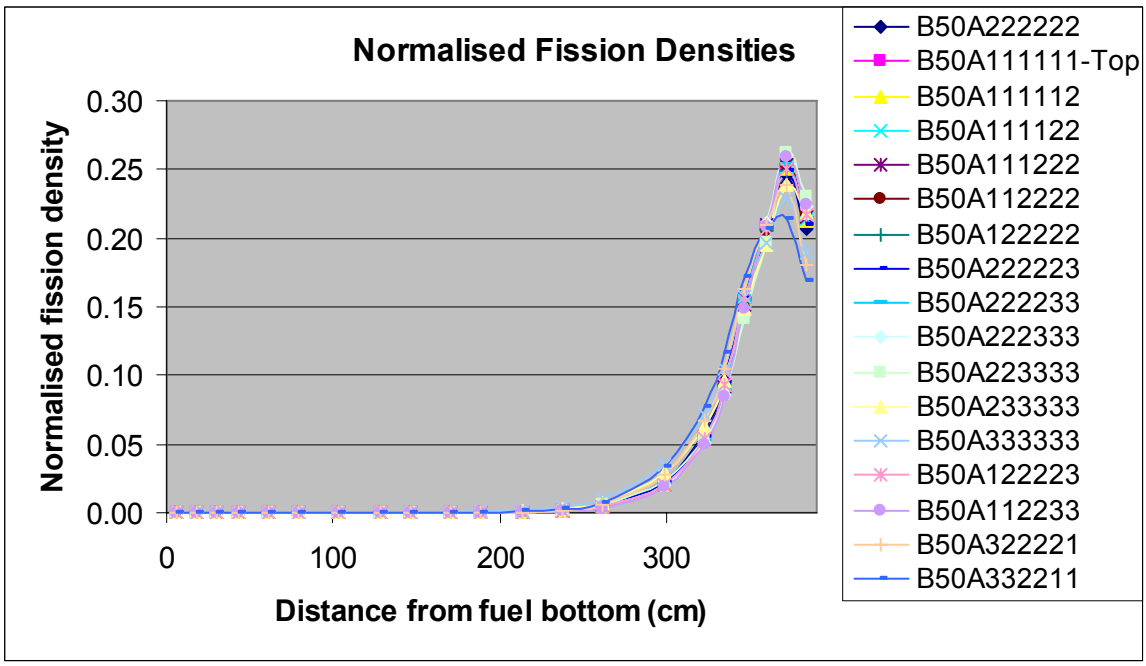

FIG. 1. Normalised fission densities, linear scale.

The logarithmic scale in Fig. 2 is only shown to emphasize the non-relevant regions while hiding the more important differences near the peaks.

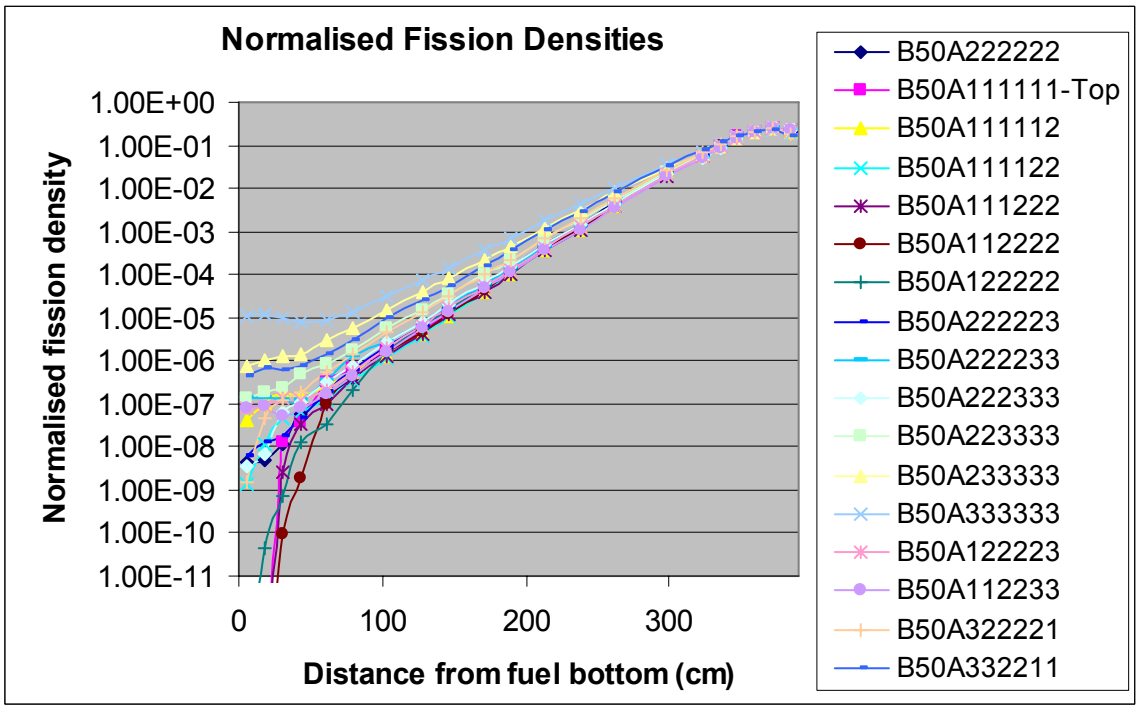

FIG. 2. Normalised fission densities, logarithmic scale.

A Monte Carlo calculation for criticality safety applications historically has been based on less than a million total fissions, often much less (several presentations in the 1988 burnup credit workshop 0 were based on calculations using 30 000 active neutrons). This determines the maximum precision of the observed fission fractions and the distribution since a single fission corresponds to a normalised fission fraction of 10^{-6}. If all source fissions are equally biased, a fission density (same order of

magnitude as a fission fraction) much less than 10^{-6} should not be expected (lower probabilities can be calculated using weighting).

If there are a million fissions in a source converged calculation, the fission densities should be zero in regions with fission probabilities much lower than 10^{-6}. This applies also to deterministic code calculations. The fission density probabilities may be calculated correctly to much higher precision than 10^{-6} but when multiplied by the number of fissions, the density should be zero (fissions are discrete events, fractions of fissions are not possible).

If the fissions in non-relevant regions are required at this precision (10^{-6}), at least the same precision should be requested for the most important regions. It means six digits, e.g. 0.114672 in an important region and 0.000001 in a region with one fission.

If the fission probability in one of the regions is already low, how low would it be if each fuel pellet was modelled individually. Multiplying the number of fuel assemblies in the cask with the number of rods per assembly and the number of pellets per rod, the number of regions would become in the order of two million or a factor 10^5 more than the current number of regions. Assuming that the horizontal fission density distribution is flat over the whole cask, the fission density will be reduced by a factor at least 10^5 in a pellet compared with the current values. To get non-zero fission densities in all regions using traditional Monte Carlo calculation methods would require enormous computer resources.

8.5. Nuclide and material categorisation: fissile, fissionable, decay groups

The introduction of irradiation credit makes new fissionable and fissile nuclides and elements significant in the calculation of k_{eff}.

The identification of fissile in the group of fissionable nuclides/elements is really useful in irradiation credit (and in other applications involving varying plutonium isotope distributions). Definition of fissile is here that addition of the nuclide/element to a thermal neutron energy region in a system moderated with water (excludes a reflector and the internal of a thick unmoderated fissile material lump) can increase k_{eff}.

All plutonium isotopes can support criticality in a fast energy spectrum. Recognised fissile isotopes are ^{236}Pu, ^{237}Pu, ^{239}Pu and ^{241}Pu. In irradiation credit for LWR fuel, the plutonium as an element will always be fissile. For curium, the situation is not so obvious. The isotopes ^{243}Cm, ^{245}Cm and ^{247}Cm are fissile. As an element, curium is normally non-fissile for uranium dioxide fuel. For LWR irradiated mixed oxide fuel, curium can be a fissile element. It would be non-conservative to select only a fissionable isotope like ^{244}Cm in the irradiation credit.

A useful way of lumping large numbers of fission products is by decay time. Other ways of categorisation is by volatility, solubility, etc. depending on the application and the length of time considered in the safety analysis. Decay and build-up (due to decay from other fission products) as a function of time must be considered.

9. Conclusions

There is a need for the Swedish industry and SKI to get more involved in the issue of irradiation credit. The guidance available from international or national standards, guides, regulations and published studies is not sufficient. Effective solutions need to be based on evaluation of conditions and scenarios that are more or less unique for Sweden.

A complete irradiation credit assessment should cover a range of fuel conditions, including fresh fuel with and without burnable absorbers, peak k_{eff} evaluations (normal for BWR) as well as k_{eff} evaluations for a range of other burnups. Routine (reality simulation), normal (including some variations and minor incidents), accident and human error conditions should all be evaluated.

In the late 1980s, irradiation credit was mainly seen as an operational issue rather than an evaluation issue. However, issues that still don't seem to be sufficiently represented in national and international criticality safety studies and developments of guides and standards include determination of a combined k_{eff} uncertainty due to various parameter uncertainties (individual k_{eff} uncertainties in the same system cannot be independent), evaluation of combined axial and other geometrical variations on k_{eff}, impact of fuel assembly differences (all assemblies are different) in a storage or cask, determination of source convergence and finally definition of nuclide and material categories (fissile, fissionable, volatility, solubility, lumping of fission densities considering decay and build-up effects).

The issue of what actinide nuclides and what fission products that can be taken credit for depends on the need for more credit, verification of the presence of the nuclides and validation of the integral effect of verified nuclides. If there are techniques such as lumping of nuclides that are effective in a specific application, data on individual nuclides may not be needed. Verification of presence and validation of integral effects is always required.

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The German burnup credit regulatory standards

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Abstract. In Germany burnup credit criticality safety analysis of nuclear spent fuel wet storage as well as transport and dry storage systems has to comply with the German burnup credit regulatory safety standards DIN 25471 and DIN 25712, respectively. The paper on hand presents the main requirements laid down in these regulatory criticality safety standards.

1. Introduction

In Germany burnup credit criticality safety analysis of spent nuclear fuel *wet storage systems* has to demonstrate compliance with the regulatory requirements laid down in the criticality safety standard DIN 25471 [1]. Application of burnup credit methods to the criticality safety analysis of *transport and dry storage* of spent nuclear fuel has to meet the requirements laid down in the criticality safety code DIN 25712 [2].

Each of the safety standards consists of two parts. In the first part the regulatory requirements are given, the second part includes explanatory and advisory remarks which are non-compulsory. The regulatory requirements are addressed to

- implementation and validation of the depletion calculations and isotopic selection,
- implementation and validation of the criticality calculations,
- determination of the reactivity effects of axial and horizontal burnup profiles,
- determination of the criticality safety acceptance criterion and the loading criteria (loading curves),
- determination of zone-specific loading curves or loading criteria for transport or storage casks (DIN 25712),
- quantification and verification of the burnup of the fuel assemblies to be loaded in the spent fuel pool or the transport or dry storage casks,
- the fuel handling procedures applied to the loading operations.

The second part of the safety standards include guidance for performing

- the depletion calculations (DIN 25471),
- evaluation of axial and horizontal profiles (DIN 25712),
- determination of 95%/95% tolerance limits (both standards),
- determination of loading curves and evaluation of burnup verification data (DIN 25712).

The main regulatory requirements will be briefly described in the following sections.

2. Regulatory requirements

2.1. Implementation and validation of the depletion calculations and isotopic selection

The depletion calculations shall be performed on the basis of plant-specific fuel designs and plant-specific irradiation histories and reactor operation strategies.

The depletion code applied shall be validated in particular against chemical assay data.

The selection of isotopes for the criticality calculations shall observe the following requirements:

- All nuclides with significant positive reactivity worth shall be included.
- Nuclides with negative reactivity worth are allowed to be included then and only then, if their contribution to the isotopic bias of the k_{eff} value of the spent nuclear fuel system of interest can be validated.
- Radionuclides with negative reactivity worth, but with half lives not significantly greater than the cooling time of the fuel are allowed to be included then and only then, if justified by the absorption properties of the daughter products or by an analysis of k_{eff} as a function of the cooling time.
- Nuclides with negative reactivity worth, which are isotopes of elements or relevant compounds being volatile under normal operation or accident conditions to be considered, *must not be included*.

The k_{eff} value of the spent nuclear fuel system of interest shall be analyzed as a function of cooling time:

- To take credit for some cooling time is allowed (DIN 25712), but it shall be shown in any case that k_{eff} *does not become greater at a later time* (both standards).
- A correction of calculated number densities such that a reactivity increase at a later time is covered is allowed (e.g. correction of Pu-239 number density due to Pu-239 accumulation after shut-down).

2.2. Implementation and validation of the criticality calculations

The criticality calculation code has to be validated against an adequate set of experiments. The adequateness of the chosen experiments shall be demonstrated.

The most reactive state of the spent nuclear fuel system which can be reached under the operation and accident conditions of the spent nuclear fuel system shall be determined, including

- uncertainties in keff due to mechanical tolerances in fuel design and spent nuclear fuel system's design and
- reactivity impacts of axial and horizontal burnup profiles.

2.3. Determination of the reactivity effects of axial and horizontal burnup profiles

Determination of the *end effect*, i.e. the reactivity effect due to the axial distribution of the burnup, shall be performed on the basis of the evaluation of a *sufficient number* of *plant-specific* axial burnup profiles. As laid down in the safety standard DIN 25712, if fuel assemblies from different plants are intended to be inserted into a transport or storage cask, sufficiently large sets of axial burnup profiles received from the *different plants* shall be analyzed.

It shall be demonstrated that the model distributions, which are used to describe the actual axial burnup profiles in the input to the criticality calculations (Fig. 1), do not result in an underestimation of the k_{eff} value of the spent nuclear fuel system.

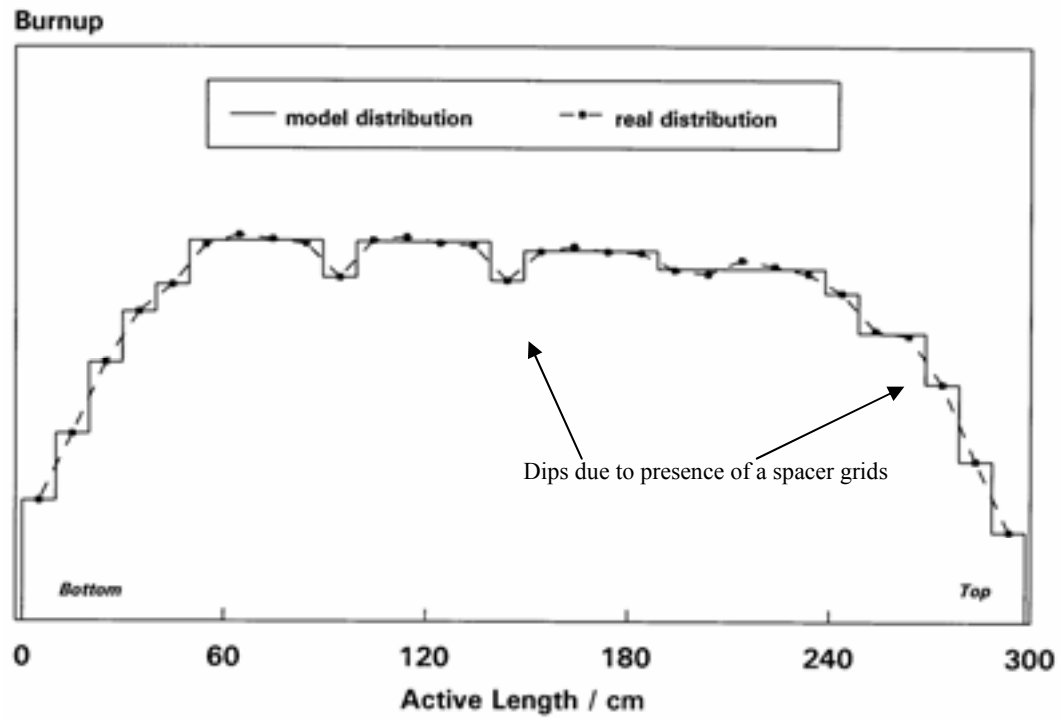

FIG. 1. Comparison of a burnup model distribution (step function) as usually used in burnup criticality calculations to an actual end-of-cycle axial burnup profile received from the German nuclear power plant Neckarwestheim I.

Two procedures for determining the end effect are described in the informatory part of the standard DIN 25712 [2].

- A sufficiently large set of plant-specific axial burnup profiles is selected. The end effect is estimated for each of the profiles separately. From the sample of end effects thus obtained a bounding correlation between end effect and average burnup of the profiles is extracted [2][3].

- All plant-specific profiles available are compiled in a database. A bounding profile, which includes or covers the dependence of the end effect on the average burnup of the actual profiles, is extracted from this database ([4] for instance). The end effect is considered by using this bounding profile in the criticality calculations.

As described in the informatory part of the standard DIN 25712, the end effect Δk can be described, according to its definition (Fig. 2), by Δk as a function of the average burnup of the spent fuel. The end effect can also be characterized, as illustrated in Fig. 2, by the difference ΔB between the average burnup and the so-called "equivalent uniform burnup" which is the burnup that leads to the same neutron multiplication factor as obtained with the actual axial burnup profile [3]. Therefore, the end effect can also be described by a correlation between equivalent uniform burnup and average burnup of the axial profiles (Fig. 3).

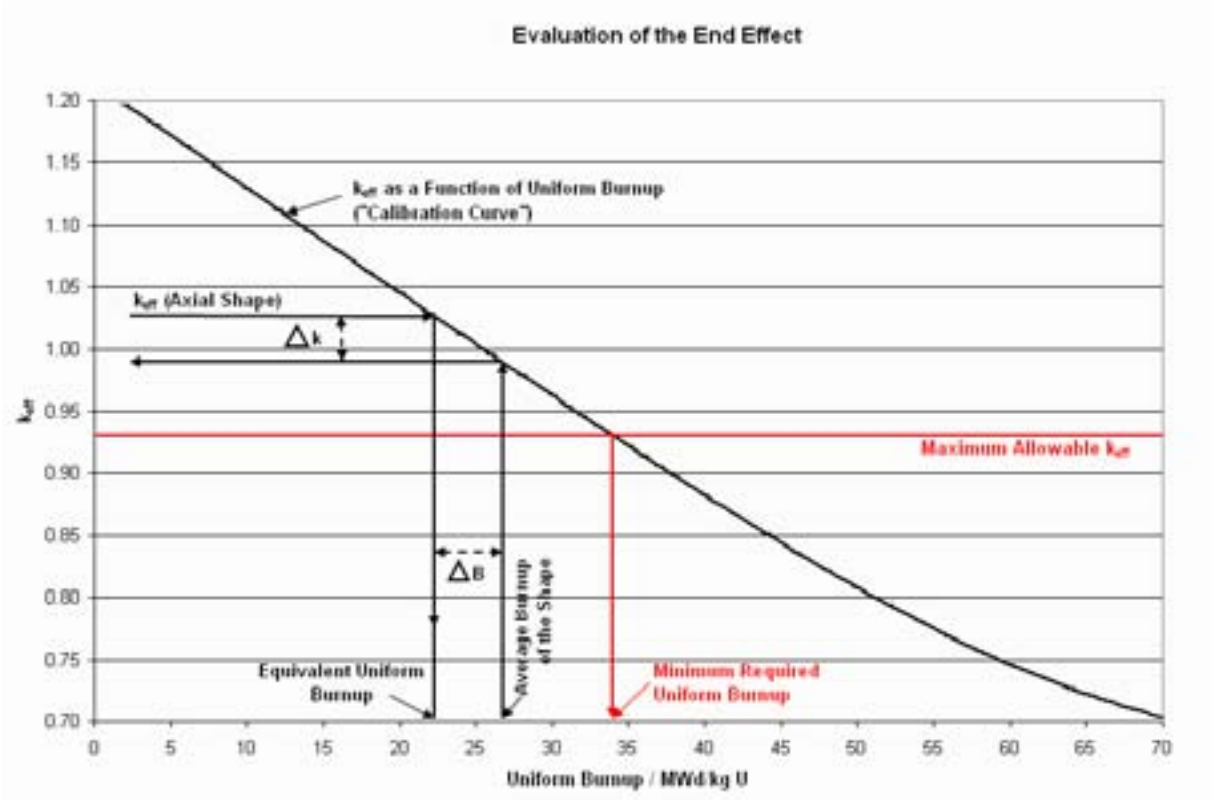

FIG. 2. The end effect Δk is the difference between the spent nuclear fuel system's neutron multiplication factor obtained with an axial burnup profile (axial shape) and the neutron multiplication factor obtained for the system by assuming the average burnup of the profile uniformly distributed over the active zone of the spent fuel.

As illustrated, the end effect can also be expressed by the difference ΔB between average burnup of the profile and the "equivalent uniform burnup" of the profile given by the neutron multiplication factor obtained with the profile. The end effect can therefore be expressed by a correlation between equivalent uniform burnup and average burnup (Fig. 3).

It is not required by the standards DIN 25471 and DIN 25712 to calculate the amount of the end effect in terms of Δk, ΔB or a correlation between equivalent uniform burnup and average burnup. The end effect can directly be covered by using a bounding profile which includes or covers the fact that the end effect changes with the average burnup. However, in any case it has to be demonstrated that the procedure used does not result in an underestimation of the neutron multiplication factor k_{eff} of the spent nuclear fuel system of interest.

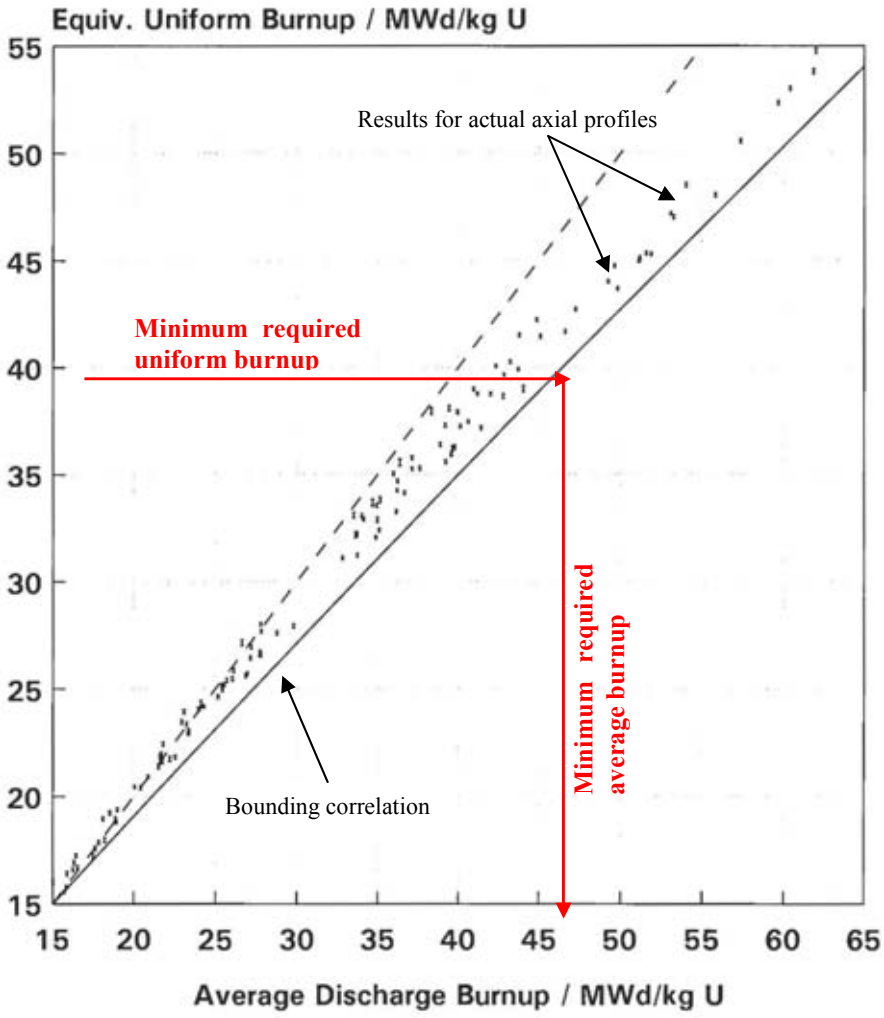

FIG. 3. Correlation between equivalent uniform burnup (Fig. 2) and average burnup.

The points shown in this figure represent the results obtained for axial profiles analyzed according to Fig. 2. From these results a bounding correlation between equivalent uniform burnup and average burnup can be derived. This bounding correlation can be used for determining the loading curve at given initial enrichment of the fuel: The intersection of k_{eff} as a function of the uniform burnup with the maximum allowable neutron multiplication factor gives the minimum required uniform burnup (as indicated in Fig. 2). This minimum required uniform burnup can be transformed into the minimum required average burnup with the aid of the bounding correlation between equivalent uniform burnup and average burnup (for details see section 2.4).

Determination of the *horizontal effect*, i.e. the reactivity effect due to horizontal gradients in the burnup of the fuel, shall be performed on the basis of *bounding* horizontal profiles, considering bounding spatial orientations of neighboring fuel assemblies.

If credit for some cooling time is taken it shall be taken into account that the reactivity effects due to axial and horizontal burnup distribution changes with the cooling time (Fig. 4).

FIG. 4. Examples for cooling time effects on the end effect for different numbers of fission products included in the criticality calculation (CT:= cooling time, FP:= fission product).

2.4. Determination of the criticality safety acceptance criterion and the loading criterion (loading curve)

If burnup credit is taken the maximum neutron multiplication factor evaluated

- shall not be greater than 0.95 under all conditions (normal operation as well as accident conditions) and
- shall include all calculational and mechanical uncertainties with a 95% probability at a 95% confidence level.

Therefore, a loading curve is given by the reactivity equivalence relation (Fig. 5)

$$k_c(e,B) + \lambda_c s_c(e,B) + \Delta k_U(e,B) = (1 - \Delta k_m) \text{ with } \Delta k_m = 0.05 \qquad (2.1)$$

where k_c is the neutron multiplication factor of the spent nuclear fuel system calculated at given initial enrichment e and uniform or average burnup B, $\lambda_c s_c$ is the 95%/95% tolerance of k_c (if statistical methods are used for the calculation of k_c) or the numerical error of k_c (if deterministic methods are used for calculating k_c). Δk_U is the uncertainty in k_c, expressed at the 95%/95% tolerance limit, due to

- bias and uncertainty in the depletion calculation,
- bias arising from the criticality calculation code,
- variances due to manufacturing tolerances, and
- axial end effect and horizontal effect if these effects are expressed in terms of Δk.

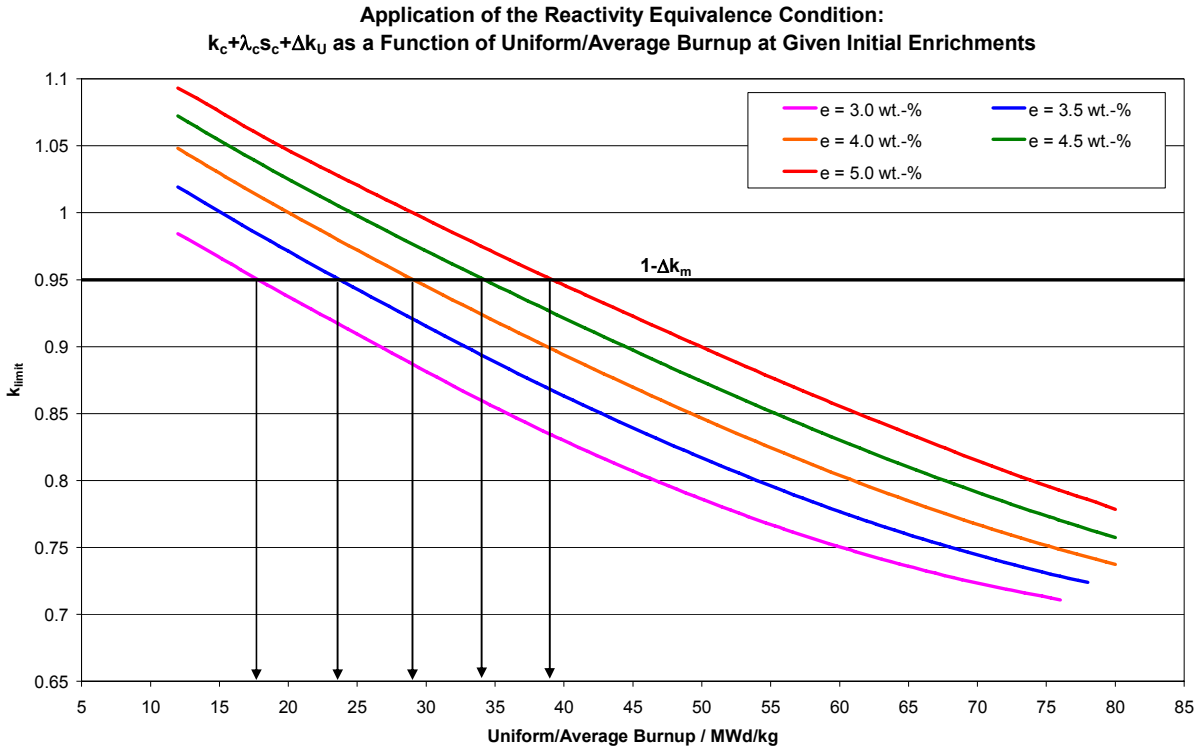

FIG. 5. Illustration of determining a loading curve (If the left-hand side of relation (2.1) is presented as a function of the uniform burnup then end effect and horizontal effect have still to be included according to section 2.3.).

2.5. Determination of zone-specific loading curves or loading criteria for transport or storage casks

The safety standard DIN 25712 allows *optimization* of the loading of transport or storage casks with respect to criticality safety. This means that one can choose among the following procedures:

- "*Standard*" burnup credit: Determination of a loading curve which applies to all the fuel positions of the cask of interest.

- "*Optimized*" burnup credit: The storage positions of the cask of interest are grouped together in several zones (but the number of different zones remains significantly lower than the total number of storage positions). For *each zone a separate loading curve* is determined taking account of the fact that all the *loading curves are mutually correlated*.

- "*Individualized*" burnup credit: A burnup credit analysis of a *given individual loading* of a cask of interest is performed. The one-sided lower 99%/99% tolerance limits of the real burnups of the individual fuel assemblies are used (as obtained by taking account of the uncertainty inherent to the procedure used to determine the individual burnup, see below).

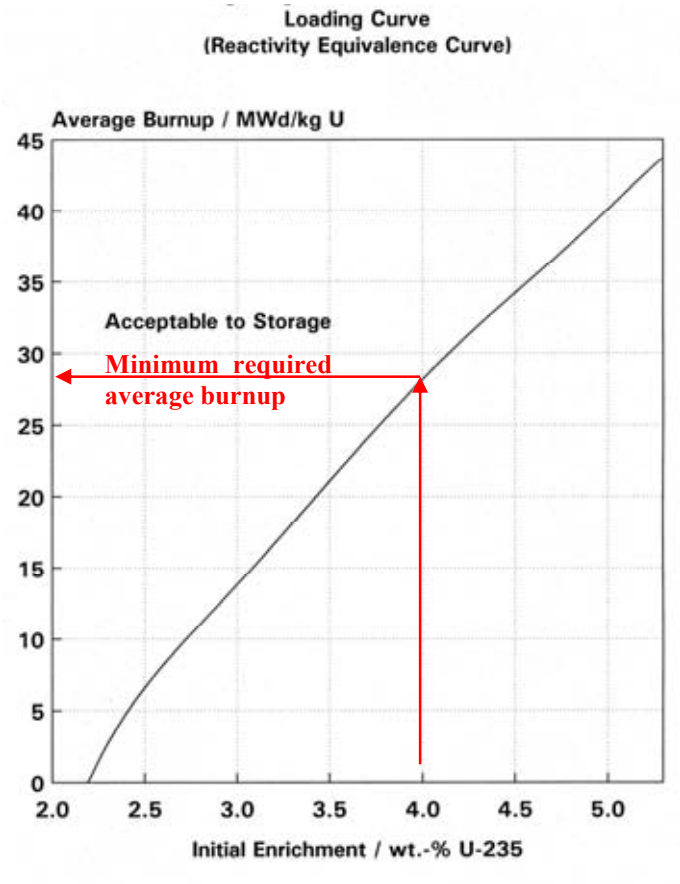

FIG. 6. Example for a loading curve. A loading curve indicates the minimum average burnup necessary for a fuel assembly with a specific initial enrichment to be loaded in the spent fuel management system of interest.

2.6. Quantification and verification of the burnup of the fuel assemblies to be loaded in a spent fuel pool or a transport or dry storage casks

2.6.1. Quantification and verification of the burnup of the fuel assemblies to be loaded in a spent fuel pool (DIN 25471)

As laid down in the safety standard DIN 25471, quantification and verification of the burnup of the fuel assemblies shall be based on the *evaluation of the reactor records*. Determination of the fuel's burnup and its verification shall be performed in compliance with the quality assurance requirements laid down in the German safety code KTA 1401 [5].

The key elements of the quality assurance requirements are standard. They require planning, identification of inputs, identification of assumptions, thorough analysis by qualified analyst, checking and documentation. Analyses performed shall be sufficiently detailed as to the purpose, method, assumptions, input, and references such that a person technically qualified in the subject can understand the analysis and verify its adequacy without recourse to the originator. Technical document reviews shall be performed to ensure that the assumptions are described and the inputs are correctly selected for their incorporation into the analysis. Technical outputs shall be reasonable compared to the inputs. Quality assurance records shall be indexed for ease in retrieval and shall be distributed, handled and controlled in accordance with the licensee's quality assurance procedures. This includes proper identification, classification, distribution, storage, retrieval and disposition. The process is subject to quality assurance audits to ensure compliance with the applicable procedures.

2.6.2. Quantification and verification of the burnup of the fuel assemblies to be loaded in a transport or dry storage casks (DIN 25712)

As laid down in the safety standard DIN 25712, quantification of the burnup of the fuel assemblies shall be based on the *evaluation of the reactor records*. Verification of the burnup shall be based on the *evaluation of the reactor records* as well and, additionally, on a *consistency check by means of a measurement* (such as a gamma scanning, a measurement of passive neutron emission, or a combination of both measurement procedures). Determination of the fuel's burnup and its verification shall be performed in compliance with the quality assurance requirements laid down in the German safety code KTA 1401 (cf. section 2.6.1).

The only reason that the requirement for a consistency check by means of a measurement has been included in the safety standard DIN 25712 is that this requirement is laid down in the IAEA regulation IAEA TS-R-1 [6]. If this were not the case this requirement would not have been included in the standard DIN 25712 since the evaluation of a measurement requires information from the reactor records [7] and since studies like [8] have shown that the utility-supplied data on burnup are of greater accuracy and reliability than could be provided by additional radiation measurement of spent fuel. The following rules are therefore laid down in the safety standard DIN 25712:

- The check whether the burnup of a fuel assembly fulfills the loading criterion (loading curve, cf. Fig. 6 for example) is carried out by using the reactor record information only: The loading criterion is met when the assembly's burnup obtained from the reactor records does not fall below an upper discrimination limit which is calculated, at a significance level of 5%, from the minimum required burnup given by the loading criterion and the uncertainty of the burnup value obtained from the reactor records. An example for determining such an upper discrimination limit is given in the informatory part of the standard DIN 25712.

- The consistency of a measurement result and the reactor record information is proven when the measurement result falls into an interval which is given by the lower and upper discrimination limit calculated, on the basis of a 5% significance level, from the reactor record information, the uncertainty of this information and the uncertainty of the measurement result. An example for determining such an interval is given in the informatory part of the standard DIN 25712.

- If it is obvious that the actual burnup of a fuel assembly is much greater than the minimum required burnup given by the loading criterion, then, instead of the consistency check, a measurement procedure may be used which is capable of demonstrating compliance of the assembly's burnup with the loading criterion. (Such a measurement procedure can be based for instance on the measurement of the intensity of the 662 keV gamma-transition following the decay of the burnup indicator Cs-137 [7][9].

2.7. Fuel handling procedures applied to the loading operations

It is obvious that fuel handling procedures have to ensure keeping of the loading criterion and prevention of misloading events. A misloading event (also named as "misloading error") occurs when a fuel assembly which does not comply with the loading criterion of the spent fuel management system of interest is anyway loaded in this system.

Both the standard DIN 25471 and the standard DIN 25712 require that *the misloading event has to be excluded in compliance with the double contingency principle*. I.e. at least two independent, unlikely and concurrent incidents have to happen before a misloading event can occur. With this application of the double contingency principle the misloading event is ruled out and needs not to be considered in the criticality safety analysis as a design basis event.

2.7.1. Fuel handling in wet storage pools

As laid down in the safety standard DIN 25471, to ensure exclusion of a misloading event in compliance with the double contingency principle the fuel handling procedure applied in the plant of interest has to meet the relevant requirements laid down in section 4.4 of the German safety code KTA 3602 [10]:

- The actual loading of the reactor core(s), all the storage installations, and all the fuel handling, inspection and repair facilities has to be documented in compliance with the quality assurance requirements laid down in the German safety code KTA 1401 [5].
- This documentation shall include identification, initial isotopic content, and burnup (cf. section 2.6.1) of each of the fuel assemblies as well as the identification of the actual positions of the fuel assemblies.
- Without an authorized working order, e.g. in form of an authorized fuel handling sequence plan, any fuel handling operation is forbidden.
- Concurrent handling of more than one fuel assembly in the wet storage pool is forbidden. (A special regulation can only be allowed on the basis of a special safety proof.)
- During the transfer of unirradiated fuel from the new fuel storage installation to region I of the wet storage pool (Fig. 7) any transfer of fuel from region I to a position outside of region I is forbidden.
- Fuel assemblies unloaded from the core have to be placed in region I of the storage pond. During re-shuffling of the core any transfer of fuel from region I to a position outside of region I or the core is forbidden.
- During the transfer of fuel between region I and region II of the storage pond any transfer of fuel from positions outside of region I or region II is forbidden. In addition, if the storage pond consists of more than two zones (e.g. one region I but two region II zones with different loading curves), concurrent fuel transfers between more than two zones are forbidden.

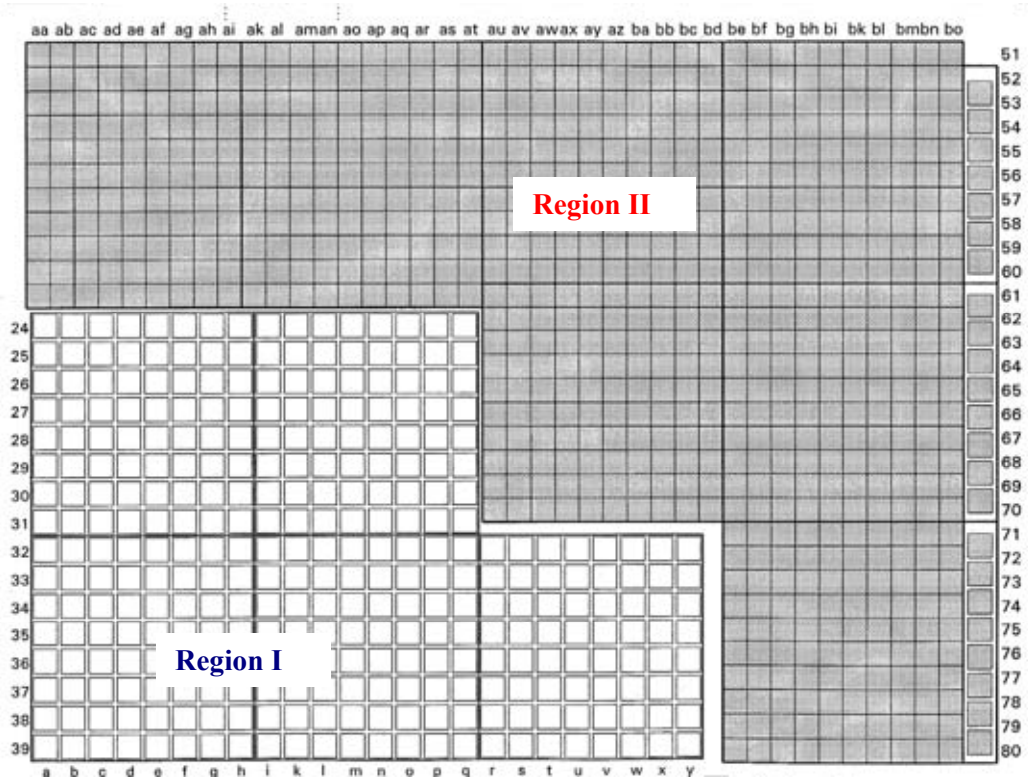

FIG. 7. Application of burnup credit to wet storage facilities results in dividing the storage pool into two regions at least [1][10].

Region I with storage racks usually designed to accommodate fuel which is at the maximum reactivity point of its life, and region II with storage racks designed to accommodate fuel for which burnup credit is taken. The German safety standard DIN 25471 allows splitting of region II in several zones with different loading *curves* [1].

2.7.2. Loading of transport or storage casks

According to the safety standard DIN 25712 the following requirements have to be met:

- For each cask loading a *loading plan* has to be made up which specifies the fuel assemblies to be loaded in the cask and the specific arrangement of these fuel assemblies within the cask.
- It has to be ensured that the fuel assemblies to be loaded in the cask meet the loading criterion (cf. section 2.6.2).
- The fuel assemblies to be loaded in the cask shall be transferred to special positions of the wet storage pool. Each of these positions has to be assigned to a particular fuel location within the cask (Fig. 8).
- For the transfer of the fuel assemblies to the special positions in the storage pit a fuel handling sequence plan has to be drawn up.

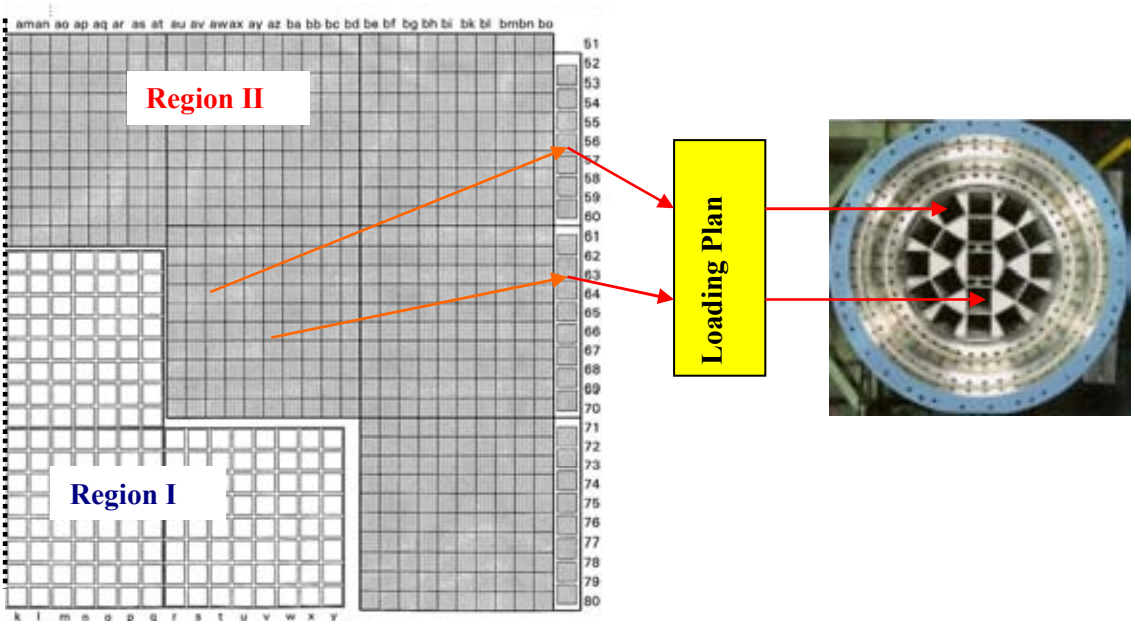

FIG. 8. A loading plan makes up a definite assignment of a storage position of the storage pit where a fuel assembly is coming from to the location inside a cask where the fuel is destined for. The first step in loading a cask is the transfer of the fuel to special positions in the pit each assigned to a particular location in the cask.

- Before a fuel assembly is transferred to the cask location, to which it is assigned, the identification of the fuel assembly has to be verified and its burnup has to be proven by means of the check procedure described in section 2.6.2. If the fuel passes these checks it has to be loaded in its cask location without delay.
- For the transfer of the fuel assemblies to their cask positions a fuel handling sequence plan has to be made up.

- Each step of the loading procedure shall comply with the quality assurance requirements laid down in the German safety code KTA 1401 [5]. In addition, the relevant requirements laid down in section 4.4 of the German safety code KTA 3602 [10] shall be met.

- To each of the steps of the loading procedure the double contingency principle shall be applied, i.e. in each step of the loading procedure it has to be ensured, that at least two independent, unlikely and concurrent incidents have to happen before an error can occur which may result in a misloading event.

2.8. Application of burnup credit to arbitrary fuel rod configurations

The safety codes DIN 25471 and DIN 25712 allow, *to some extent*, the use of burnup credit for some arbitrary fuel rod configurations or clusters as obtained for instance in fuel assemblies under repair.

As laid down in the safety code DIN 25471 use of burnup credit for arbitrary fuel configurations handled or stored in a *wet storage pond* has to meet the requirements laid down in section 4.2.6.2 of the German safety code KTA 3602 [10]. Accordingly, burnup credit can be applied to fuel assemblies under repair, but use of burnup credit is not permitted for fuel rod quivers, receptacles, boxes or containers.

For *transport and dry storage*, however, the safety code DIN 25712 allows use of burnup credit for any arbitrary fuel rod configuration. However, burnup credit as applied to complete and integer fuel assemblies (cf. section 2.5) may be applied to individual fuel rods only then if these rods are arranged in a geometrically stable configuration. Otherwise the following restrictions have to be observed:

- The burnup credited for all the fuel rods of the rod configuration of interest shall not be greater than the smaller one of the two following burnup values:

  - the lower 99%/99% tolerance limit of the burnup of the least burnt fuel rod of the rod configuration,

  - the highest average burnup value for which the axial end effect is not positive under the assumption of realistic axial burnup profiles and the conditions to be analyzed.

- Taking credit for an axial zoning of the initial fissile content is forbidden. The initial fissile content which is the most reactive one under the conditions to be analyzed has to be considered.

- If the fuel rods of the rod configuration of interest have different initial fissile contents the initial fissile content which is the most reactive one under the conditions to be analyzed has to be considered.

- Taking credit for neutron poisons initially present in the fuel is forbidden. However, spectrum hardening effects due to the presence of neutron poisons during depletion have to be taken into account.

- In case of flooding of the rod configuration of interest the most reactive arrangement of the fuel rods within this configuration has to be considered.

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Some words about the 95%/95% tolerance limit

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Abstract. It is required by many criticality safety regulations from all over the world that the maximum k_{eff} value of a fuel system under study shall be less than or equal to a prescribed value (e.g., 0.95), including mechanical and calculational uncertainties, *with a 95% probability at a 95% confidence level*. This upper limit of k_{eff}, usually named as "upper 95%/95% tolerance limit", gives a confidence statement about the probability content of the true, but unknown distribution of the sample mean of k_{eff} of the fuel system under study calculated with the aid of a statistical code. Unfortunately, there are a lot of confusing ideas of the 95%/95% tolerance limit in practice. The objective of the paper on hand therefore is to:

- give the correct definition of the 95%/95% tolerance limit and to show the difference to a confidence statement about the expectation value of the sample mean of k_{eff},
- demonstrate the necessity for applying the 95%/95% tolerance limit to k_{eff} calculated with the aid of statistical codes, and
- give some hints of calculating the 95%/95% tolerance limit.

1. Introduction

In many criticality safety regulations, codes and guides from all over the world ([1][2][3][4][5][6][7] for instance) it is laid down that the evaluated neutron multiplication factor k_{eff} of a nuclear fuel configuration of interest

- shall include all calculational and mechanical uncertainties with a 95% probability at a 95% confidence level and
- shall not be greater than an adequate upper bound $(1 - \Delta k_m)$ of subcriticality under normal operation as well as all anticipated abnormal or accident conditions.

These requirements are expressed, in terms of mathematics, in the inequality

$$k_c + \Delta k_U \le (1 - \Delta k_m). \qquad (1.1)$$

k_c represents the calculated neutron multiplication factor k_{eff} as obtained for a configuration under given conditions. Since statistical criticality calculation codes are usually applied today k_c is based on sample means $\overline{k}_{eff}$ each of which results from sequences of $N = (N_t - N_s)$ evaluated k_{eff} values k_i, $i = 1, \ldots, N$, obtained from a total of N_t computed neutron generations (batches, cycles) with an adequately chosen number of neutrons per generation; N_s denotes the number of generations skipped before sampling on k_{eff}.

Δk_U in inequality (1.1) comprises the calculational and mechanical uncertainties with a 95% probability at a 95% confidence level. In everyday work Δk_U is usually splitted into several terms due to different characteristics of the contributing uncertainties,

$$\Delta k_U = \lambda_c s_c + \sum_j \Delta k_j \ . \qquad (1.2)$$

Every single term, if based on a statistics, covers the respective uncertainty with a 95% probability at a 95% confidence level.

The term $\lambda_c s_c$ in eq. (1.2) represents that uncertainty of the computed value k_c of k_{eff} which is entirely due to the applied criticality computation technique on its own. Therefore, inequality (1.1) is usually rewritten into the expression

$$k_c + \lambda_c s_c \le (1 - \Delta k_m) - \sum_j \Delta k_j \ . \qquad (1.3)$$

So, if k_c is based on results calculated with the aid of a statistical code, $\lambda_c s_c$ covers the purely statistical uncertainty, which is related to the sample mean k_c, with a 95% probability at a 95% confidence level; and $k_c + \lambda_c s_c$ is often named as "upper 95%/95% tolerance limit of the computed sample mean k_c of k_{eff}". If a non-statistical calculation code is used $\lambda_c s_c$ represents the numerical error of the computed k_c value [5].

The terms Δk_j, j = 1, 2, …, in expressions (1.2) and (1.3) represent

- the bias in the applied criticality calculation procedure as obtained from benchmark calculations,
- the uncertainty arising from manufacturing tolerances in construction materials, dimensions etc,
- the uncertainty due to uncertainties in the isotopic composition of the fuel, the uncertainty arising from specific reactivity effects, e.g. due to non-homogeneous enrichment and burnup distributions within fuel assemblies.

Note that most of these uncertainties are statistical uncertainties even if they are estimated with the aid of a non-statistical criticality calculation code. In fact, the bias of the applied calculation procedure is obtained from an evaluation of a statistics of k_{eff} values calculated for a set of experiments. The "as-built" values of parameters characterizing construction materials, dimensions etc follow some statistical distributions within the respective tolerance intervals of the parameters. The uncertainty due to uncertainties in the isotopic composition of the fuel is estimated by evaluating statistics of deviations between predicted and measured isotopic concentrations. The evaluation of specific reactivity effects is also based, in many cases, on statistical methods ([8] for instance). So therefore, the terms Δk_j in expressions (1.2) and (1.3) have usually to be expressed at the upper 95%/95% tolerance limit; and they can thus be rewritten into expressions

$$\Delta k_j = \overline{\Delta k}_j + \lambda_j s_j \ , \ \forall j \text{ (i.e., for all j)}, \qquad (1.4)$$

where $\overline{\Delta k}_j$ represents the estimate of the mean of the uncertainty Δk_j and $\lambda_j s_j$ represents the uncertainty of this estimate with a 95% probability at a 95% confidence level. To be able to estimate $\overline{\Delta k}_j$ and $\lambda_j s_j$ one has to evaluate some sample on the uncertainty Δk_j.

So therefore, whether one uses a statistical or a non-statistical criticality calculation code, in any case one has to evaluate some sample of observations (z_1, …, z_N) of a random variable z following some statistical distribution which is usually unknown; and, to comply with the rules, one has to determine the upper 95%/95% tolerance limit

$$z_{95/95} = \bar{z} + \lambda_z s_z \qquad (1.5)$$

related to the sample mean $\bar{z}$.

Unfortunately, there are a lot of confusing ideas of the 95%/95% tolerance limit in practice. The objective of the paper on hand therefore is

- first, to give the correct definition of the (upper) 95%/95% tolerance limit $z_{95/95}$,
- second, to show the difference to a confidence statement about the expectation value of the sample mean $\bar{z}$,
- third, to demonstrate the necessity for applying the 95%/95% tolerance limit to the evaluation of samples of observations (z_1, …, z_N) of a safety parameter z such as k_{eff},
- fourth, to give some hints of calculating the 95%/95% tolerance limit.

As stated above, even if one uses a non-statistical criticality calculation code one is forced to deal with statistics to evaluate the bias of the applied calculation code and the uncertainties due to manufacturing tolerances, uncertainties in the isotopic composition of the fuel etc. In the following the attention is therefore focused on the case that a statistical criticality calculation code is used since this case is the more general case with respect to the idea of the 95%/95% tolerance limit.

2. Brief review of some basic principles

2.1. Sample mean and the laws of large numbers

Statistical criticality calculation codes solve the neutron transport equation as an eigenvalue problem through employment of Monte Carlo techniques. Such techniques are probably the most common examples of the use of the laws of large numbers of mathematical statistics.

The laws of large numbers state that the *sample mean* (*average*)

$$\bar{z} = \frac{1}{N}\sum_{i=1}^{N} z_i \tag{2.1}$$

of N *mutually independent* observations (z_1, …, z_N), each having the *same expectation* (*mean*) value μ,

$$\mu_i = E[z_i] = \int_{\Omega_i} z\, dF_i(z) = \mu, \forall i, \tag{2.2}$$

and variances σ_i^2

$$\sigma_i^2 = V[z_i] = E\left[(z - E[z_i])^2\right] = \int_{\Omega_i} (z - \mu_i)^2\, dF_i(z), \tag{2.3}$$

converges to the expectation μ (the "true value") when $N \to \infty$, provided that

$$\lim_{N\to\infty}\left[\frac{1}{N^2}\sum_{i=1}^{N}\sigma_i^2\right] = 0 \qquad \text{(weak law of large numbers)} \tag{2.4}$$

or

$$\lim_{N\to\infty}\left[\sum_{i=1}^{N}\left(\frac{\sigma_i}{i}\right)^2\right] \text{ is finite} \qquad \text{(strong law of large numbers)} \tag{2.5}$$

$F_i(z)$ in equations (2.2) and (2.3) denotes the probability distribution (cumulative distribution) of z_i [1) *)], and Ω_i denotes the range of possible values z_i.

The usual case is that the z_i are N different trials of the same experiment and hence observations on one and the same probability distribution. In this case one has

$$\sigma_i^2 = \sigma^2, \forall i, \tag{2.6}$$

and conditions (2.4) and (2.5) are obviously met: Eq.(2.4) becomes

$$\lim_{N\to\infty}\left[\frac{\sigma^2}{N}\right] = 0 \tag{2.7}$$

which is obviously true (provided, of course, that σ^2 exist), and eq.(2.5) becomes (cf. Ref. [11])

$$\lim_{N\to\infty}\left[\sigma^2\sum_{i=1}^{N}\left(\frac{1}{i}\right)^2\right] = \sigma^2\frac{\pi^2}{6} < \infty. \tag{2.5}$$

Both laws of large numbers imply convergence in probability. The sequence $(z_1, \ldots, z_N)$ is said to *converge in probability* to z if for any $\varepsilon > 0$ and any $\eta > 0$ a value of M can be found such that

$$P\left[\left|z_N - z\right| > \varepsilon\right] < \eta \text{ for all } N \geq M, \tag{2.9}$$

where P denotes the probability that the "distance" $|z_N - z|$ is greater ε.

Convergence in probability is a weaker type of convergence than the types of convergence to which the laws of large numbers refer [9]. Therefore, in general, convergence in probability does not imply the types of convergence to which the laws of large numbers refer.

However, convergence in probability implies convergence in distribution, whereas the reverse is not generally true since convergence in distribution is the weakest kind of convergence in statistics (cf. ibid.). The sequence $(z_1, \ldots, z_N)$ of random variables with cumulative distribution functions $F_1(z), \ldots, F_N(z)$ is said to *converge in distribution* to z of cumulative distribution F(z), if, for every point where F is continuous, one has

$$\lim_{N\to\infty} F_N(z) = F(z). \tag{2.10}$$

Let us come back to the mean μ and the variances σ_i^2 of the observations $(z_1, \ldots, z_N)$. Note that μ is *not* the same as the sample mean $\bar{z}$ given by eq.(2.1). In general, μ is *not known*. The sample mean $\bar{z}$ is an *unbiased* estimator of μ [2) 3)] and converges to μ when $N \to \infty$, provided that condition (2.4) or (2.5) holds [4)].

Likewise, the variances σ_i^2 are *unknown* in general. In the usual case specified by eq.(2.6) the expression

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*) Notes are given in Appendix A

$$\hat{\sigma}^2 = \frac{1}{N-1}\sum_{i=1}^{N}(z_i - \bar{z})^2 \tag{2.11}$$

gives an *unbiased* and *consistent* estimator of σ^2 [9]. An estimator is said to be consistent if it converges in probability.

However, in real life $N \to \infty$ cannot be realized. Therefore a *confidence statement* about the difference between the sample mean $\bar{z}$ of N independent observations z_i of one and the same experiment and the unknown mean μ of the z_i is required. Such a confidence statement is related to the *probability content* of the "distance" of $\bar{z}$ from μ. However, the probability

$$P(\bar{z} - a \le \mu \le \bar{z} + b) = (1-\gamma),\ a, b \in \Re^+ \tag{2.12}$$

that $\mu \in [\bar{z} - a,\ \bar{z} + b]$, a and b some parameter values to be evaluated for the given probability content of $(1 - \gamma)$, can only be calculated if a variable

$$t = t(\bar{z}, \hat{\sigma}^2; \mu, \sigma^2) \tag{2.13}$$

can be found the probability distribution F(t) of which is independent of the unknown parameters μ and σ^2. If this can be found, it may be possible to find, with the aid of F(t), a range $[\bar{z} - a,\ \bar{z} + b]$ such that (2.12) holds. However, since eq.(2.13) denotes a change of variables knowledge about the probability distributions of $\bar{z}$ and $\hat{\sigma}^2$ with respect to μ and σ^2 is required for determining F(t). In the following section it is presupposed, just for the sake of studying the consequences, that the z_i, $i = 1, \ldots N$, are independent observations on the normal distribution

$$\Phi\left(\frac{z-\mu}{\sqrt{\sigma^2}}\right) = \int_{-\infty}^{\frac{z-\mu}{\sigma}} dt\, \varphi(t,1) \tag{2.14}$$

with μ and σ^2 unknown and

$$\varphi(t,1) = \frac{1}{\sqrt{2\pi}}\exp\left(-\frac{t^2}{2}\right). \tag{2.15}$$

2.2. Sample mean and confidence interval estimation for observations on a normal distribution

2.2.1. Confidence interval of the mean

If it is assumed that the z_i are N independent observations on the normal distribution (2.14), it can be shown that the variables

$$y_N = \frac{\bar{z} - \mu}{\sqrt{N^{-1} \cdot \sigma^2}} \tag{2.16}$$

and

$$q^2 = (N-1)\frac{\hat{\sigma}^2}{\sigma^2} \tag{2.17}$$

are *independently* distributed; y_N follows a standard normal distribution, i.e., the probability that y_N is less than y is given by

$$P(y_N < y) = \int_{-\infty}^{y} dt\, \varphi(t,1)\,, \tag{2.18}$$

and q^2 follows a χ^2-distribution with (N-1) degrees of freedom [9][13]. From that it follows that the variable

$$t = \sqrt{(N-1)}\,\frac{y_N}{\sqrt{q^2}} = \frac{\bar{z} - \mu}{\sqrt{N^{-1} \cdot \hat{\sigma}^2}} \tag{2.19}$$

has, irrespective of μ, a central Student's t-distribution F(t; N-1) with (N-1) degrees of freedom (cf. ibid.).

Due to the symmetry of the central Student's t-distribution density (cf. ibid.) and because of eq.(2.19) one gets from

$$P\left(-t_{N-1;1-\gamma/2} \le t \le t_{N-1;1-\gamma/2}\right) = 1 - \gamma = \int_{-t_{N-1;1-\gamma/2}}^{t_{N-1;1-\gamma/2}} dF(t; N-1) \tag{2.20}$$

a confidence interval for μ:

$$P\left(\bar{z} - t_{N-1;1-\gamma/2} \cdot \sqrt{\frac{\hat{\sigma}^2}{N}} \le \mu \le \bar{z} + t_{N-1;1-\gamma/2} \cdot \sqrt{\frac{\hat{\sigma}^2}{N}}\right) = 1 - \gamma\,. \tag{2.21}$$

Even though the confidence interval estimation formula (2.21) does not meet the requirements laid down in the criticality safety regulations, codes or guides mentioned at the beginning of section 1, it is very often used in criticality safety analysis practice for evaluating the results k_i, i = 1, …, N, obtained for the neutron multiplication factor k_{eff}. Usually only one calculation run per case of interest is made. N is thus the number of evaluated neutron generations, $N = (N_t - N_s)$, cf. section 1; and k_i denotes the result obtained in the i-th evaluated neutron generation. The sample mean $\bar{k}_{eff}$ and the sample variance $\hat{\sigma}^2$ are evaluated according to equations (2.1) and (2.11), respectively. Since the Student's t-distribution converge to the standard normal distribution when $N \to \infty$ [13], a value of 2 is usually taken for the so-called "(1 - γ/2) point" $t_{N-1;1-\gamma/2}$ of the Student's t-distribution F(t; N-1) to get a confidence level (probability content) of about $(1 - \gamma)\cdot 100\% \approx 95\%$ [5]. In some countries a value of 3 is used for $t_{N-1;1-\gamma/2}$ resulting in a probability content of more than 99% [6].

2.2.2. Comments on the use of the confidence interval of the mean in the evaluation of the neutron multiplication factor

Throwing a glance at the criticality safety acceptance criterion (1.3) the question naturally arises why one calculates a bounded confidence interval. With respect to the criterion (1.3) the lower limit of the confidence interval given by eq.(2.21) is of no interest. In reality one uses, with respect to the left-hand side of inequality (1.3) the unbounded confidence interval

$$-\infty \le \mu \le \bar{z} + t_{N-1;1-\gamma} \cdot \sqrt{\frac{\hat{\sigma}^2}{N}} \tag{2.22}$$ [7]

which has a probability content of

$$(1-\gamma) = \int_{-\infty}^{t_{N-1;1-\gamma}} dF(t; N-1) \tag{2.23}$$

which amounts to $(1 - \gamma)\cdot 100\% \approx 97.5\%$ if one keeps at $t_{N-1;1-\gamma} = 2$. If one wants to keep the probability content at about 95% one can reduce $t_{N-1;1-\gamma}$ to a value of about 1.67 [8)].

If one uses a confidence interval for the left-hand side of the criterion (1.3) one can expect that the terms Δk_j on the right-hand side of the criterion (1.3) are estimated in a consistent way. Unfortunately, practice does not come up very often to this expectation. If $\Delta z \equiv \Delta k_j$ is the difference of two sample means $\bar{z}_1$ and $\bar{z}_2$ of independent observations on normal distributions with unknown expectations μ_1 and μ_2, respectively, and unknown variances σ_1^2 and σ_2^2, respectively, then one can find a Student's t-distributed variable which allows, first, to check whether the observed difference $\bar{z}_1 - \bar{z}_2$ is significant or merely random and, second, if it is found that this difference is significant, to estimate a confidence interval of the unknown "true" difference $\delta = \mu_1 - \mu_2$ [9][13]. If $\Delta z \equiv \Delta k_j$ has to be calculated from a whole set $(\Delta k_{1j}, \ldots, \Delta k_{nj})$ of Δk_{ij} values as it is the case for instance in the evaluation of benchmark experiments, then it is adequate and efficient to use the linear least squares method [8][9]. This method has optimal properties due to its linearity alone; and if the data Δk_{ij}, i = 1, …, n, follow normal distributions then it follows from the linearity of the method that the least squares estimates are also normally distributed. This allows the estimation of a confidence interval of the expectation of Δk_j, [9] [9)].

The question naturally arises as to what extent a series $(z_1, \ldots, z_N)$ of observations can be considered *a priori* to be normally distributed. In everyday criticality safety analysis practice this question is either not asked or (mostly successfully) evaded by giving an *a posteriori* answer:

- First the observations k_i, i = 1, … N, on the neutron multiplication factor k_{eff} are sampled in a calculation run employing $N = (N_t - N_s)$ evaluated neutron generations (cf. section 1).

- Then it is checked with the aid of an appropriate test procedure (e.g., Pearson's χ^2-test, cf. Ref. [9] [10)] whether the null hypothesis H_0 (i.e., the hypothesis under test), that the results k_i, i = 1, … N, of the evaluated generations can be assumed to be normally distributed, is rejected or not.

If H_0 is rejected then it is often assumed in practice, that there is some source convergence problem, even though normality of the k_i and source convergence are *not* related, *neither one implies the other*. However, in practice usually the neutron starting distribution is changed and/or the number N_s of initial neutron generations skipped is increased and/or the number of evaluated neutron generations is increased till non-rejection of H_0 is reached [11)].

The test procedures employed to check H_0 include the use of the sample mean (2.1) and the sample variance (2.11). Non-rejection of H_0 does not mean that the distribution underlying the observations k_i is known by the outcome of the test. The expectation μ and the variance σ^2 remain unknown since $N \to \infty$ cannot be realized. Non-rejection of H_0 does not even mean that the observations k_i are really normally distributed [9].

However, from the viewpoint of practice, non-rejection is acceptance, and it is therefore not rejected that the variable defined by eq.(2.19) follows a Student's t-distribution. It is thus justified to estimate a confidence interval for the true value $\mu = E[k_{eff}]$ according to eq.(2.21).

So, everything seems to be sound, apart from the fact that use of eq.(2.21) does not comply with the requirements laid down in the criticality safety regulations, codes or guides mentioned at the beginning of the paper on hand.

2.3. Definition of the 95%/95% tolerance limit

That use of eq.(2.21) does not meet the requirements laid down in the criticality safety regulations, codes etc this is already apparent from the differences in terminology:

- By definition, eq.(2.21) yields a confidence interval of μ. A confidence level of 95% can be chosen meaning that the 95%-confidence-interval contains the true value μ with a probability of 95%.

- What the criticality safety regulations, codes etc require is to estimate an upper limit which covers all uncertainties with a 95% probability at a 95% confidence level. This requirement is related to the fact that the neutron multiplication factor k_{eff} estimated with the aid of statistical methods is a random variable. A random variable is completely defined by its cumulative distribution [9]. The cumulative distribution $F(k_{eff})$ of k_{eff} is not known. It is therefore required to find an upper limit $L_u(\alpha,\gamma)$ of k_{eff} such that a part (i.e., a probability content) of at least $(1 - \gamma)$ of $F(k_{eff})$ is below $L_u(\alpha,\gamma)$ with a confidence of $(1 - \alpha)$. One has therefore to evaluate the expression

$$P\left[\int_0^{L_u(\alpha,\gamma)} dF(k_{eff}) \geq (1-\gamma)\right] = (1-\alpha) \tag{2.24}$$

with $(1 - \gamma)\cdot 100\% = (1 - \alpha)\cdot 100\% = 95\%$ as laid down in the regulations.

By definition,

$$P[k_{eff} < L_u(\alpha,\gamma)] = \int_0^{L_u(\alpha,\gamma)} dF(k_{eff}) \tag{2.25}$$

is the value of the cumulative distribution $F(k_{eff})$ at $k_{eff} = L_u(\alpha,\gamma)$ and gives the probability that $k_{eff} < L_u(\alpha,\gamma)$. This probability amounts to a minimum of $(1 - \gamma)$. Expression (2.24) is the probability that the probability $P[k_{eff} < L_u(\alpha,\gamma)]$ amounts to a minimum of $(1 - \gamma)$. So, expression (2.24) is a confidence statement about the unknown distribution $F(k_{eff})$ whereas eq.(2.21) is only a confidence statement about the expectation of k_{eff}.

The fact that k_{eff} is a random variable is entirely due to the use of statistical methods for the calculation of k_{eff}. Therefore, the distribution $F(k_{eff})$ and hence the upper limit $L_u(\alpha,\gamma)$ are dependent on the amount of information given by the observations made with the aid of the statistical method employed. So, what is the amount of information minimum required to solve eq.(2.24)?

Let k_i denote, as before, the k_{eff} result obtained in the i-th evaluated neutron generation of a calculation run employing $N = (N_t - N_s)$ evaluated neutron generations and n_g neutrons per generation. Because n_g is a finite number the results k_i will follow some distribution with a finite width. Let us assume, that the expectation of this distribution

$$E[k_i] = \mu, \forall i, \tag{2.26}$$

exists, and that the variance of this distribution

$$V[k_i] = E[(k_i - \mu)^2] = \sigma^2, \forall i, \tag{2.27}$$

does not diverge. These assumptions are justified by the fact that k_{eff} is a bounded variable, i.e.,

$$k_{eff} \in [k_{min}, k_{max}] \tag{2.28}$$

where $k_{min} \geq 0$ and k_{max} is determined by the neutron physics properties of the fuel configuration of interest.

Obviously, k_i converges to μ when $n_g \to \infty$, and σ^2 converges to zero when $n_g \to \infty$. However, $n_g \to \infty$ cannot be realized. Therefore, μ as well as $\sigma^2 = \sigma^2(n_g)$ remain unknown, and $\sigma^2 > 0$.

The sample mean $\bar{k}$ given by eq.(2.1) (with $z_i \equiv k_i$) converges, according to the laws of large numbers, to μ when $N \to \infty$ (cf. section 2.1). And the sample variance $\hat{\sigma}^2$ given by eq.(2.11) (with $z_i \equiv k_i$ and $\bar{z} \equiv \bar{k}$) converges in probability to σ^2 when $N \to \infty$ (cf. section 2.1). However, $N \to \infty$ cannot be realized. So therefore, μ and σ^2 remain unknown.

As follows from eq.(2.1) (with $z_i \equiv k_i$) [12)]

$$E[\bar{k}] = \mu. \tag{2.29}$$

Let us assume that the k_i can be considered as *mutually independent* observations. Then it follows from eq.(2.1) and eq.(2.27)

$$V[\bar{k}] = \frac{\sigma^2}{N}. \tag{2.30}$$

In spite of the fact that k_{eff} is a bounded variable let us assume for the moment that the distribution underlying the k_i can be well approximated by the *normal distribution* (2.14) (with $z = k_{eff}$). This assumption results in a significant increase in information about the distribution $F(k_{eff})$ because of the following reasons:

- First, a normal distribution is completely defined by its expectation and its variance, cf. eq.(2.14).
- Second, the sample mean $\bar{k}$ is a linear function of the k_i, cf. eq.(2.1). If the k_i are assumed to be normally distributed then it follows from the linearity of eq.(2.1) that $\bar{k}$ is normally distributed with expectation (2.29) and variance (2.30).

So, under the condition that the k_i can be considered as observation on a normal distribution with expectation μ and variance σ^2 the distribution $F(k_{eff})$ in eq.(2.24) is given by a normal distribution with expectation μ and variance σ^2/N. In other words the standardized variable [13)]

$$y_N = \frac{\bar{k} - \mu}{\sqrt{N^{-1} \cdot \sigma^2}} \tag{2.31}$$

follows a standard normal distribution

$$F(k_{eff}) \equiv F(\bar{k}) = \Phi\left(\frac{\bar{k}-\mu}{\sqrt{N^{-1}\cdot\sigma^2}}\right) = \int_{-\infty}^{\frac{\bar{k}-\mu}{\sqrt{N^{-1}\cdot\sigma^2}}} dt\,\varphi(t,1). \tag{2.32}$$

This was already used in the derivation of the confidence interval (2.21), cf. equations (2.16) through (2.20). So, the distribution $F(k_{eff})$ in eq.(2.24), given by eq.(2.32), is just the same as is "behind" eq.(2.21). This was to be expected because of consistency reasons. Eq.(2.21) gives a confidence statement about the expectation μ of the distribution $F(k_{eff})$ given by eq.(2.32) but cannot exclude that a significant part of $F(k_{eff})$ can be located above the upper limit $\bar{k} + t_{N-1;1-\gamma/2}\cdot\sqrt{N^{-1}\cdot\hat{\sigma}^2}$ of the confidence interval of μ. This is excluded, with a confidence of $(1 - \alpha)$, by eq.(2.24).

μ and σ^2 in eq.(2.32) are still unknown. So, how one can solve the equation (2.24)? Obviously, all the information given by the observations $(k_1, \ldots, k_N)$ of one calculation run is already used up in the determination of the sample mean $\bar{k}$, the sample variance $\hat{\sigma}^2$ and the confidence interval of μ. So, to solve eq.(2.24) more information about $F(k_{eff})$ is required, i.e., a sample on $F(k_{eff})$ has to be drawn. This means that a sample $(\bar{k}_1,\ldots,\bar{k}_M)$ on the sample mean $\bar{k}$ has to be taken.

A Monte Carlo calculation run is started with a certain starting random number which induces a certain pseudo random number sequence. If the calculation run is repeated without any change in the input parameters one gets the same pseudo random number sequence and hence the same results. However, if one uses a different starting random number and repeats the calculation run without any change in all the other input parameters, one gets a different pseudo random number sequence and hence different results. So, if one repeats the calculation M times, varying the starting random number only (all the other input parameters remain unchanged) one gets a distribution of the sample means $(\bar{k}_1,\ldots,\bar{k}_M)$ which is just a sample on $F(k_{eff})$. Since the starting random number is the only parameter that has been changed one has because of equations (2.29) and (2.30)

$$E[\bar{k}_j] = \mu, \forall j = 1,\ldots,M, \tag{2.33}$$

and

$$V[\bar{k}_j] = \frac{\sigma^2}{N}, \forall j = 1,\ldots,M, \tag{2.34}$$

respectively. So the set of sample means $(\bar{k}_1,\ldots,\bar{k}_M)$ is in fact a set of M independent observations on the distribution (2.32).

The sample mean

$$k_c = \frac{1}{M}\sum_{j=1}^{M}\bar{k}_j \tag{2.35}$$

of the sample means $\bar{k}_j$ converges, according to the laws of large numbers, to μ when $M \to \infty$ [14], and the sample variance

$$s_c^2 = \frac{1}{M-1}\sum_{j=1}^{M}(\bar{k}_j - k_c)^2 \tag{2.36}$$

converges, in probability, to expression (2.34) [9].

It is instructive to throw a glance at an example. In Table 1 a sample of M = 100 sample means $\bar{k}_j$ is given which has been obtained for an isolated, unirradiated and unpoisoned 4.45 wt.-% ^{235}U-enriched Convoy Series fuel assembly (immersed in pure light water). In addition, the estimated standard deviation $\sqrt{N^{-1} \cdot \hat{\sigma}_j^2}$ ($\hat{\sigma}_j^2$ given by eq.(2.11)) and the upper limit $\bar{k}_j + t_{N-1;1-\gamma/2} \cdot \sqrt{N^{-1} \cdot \hat{\sigma}_j^2}$ of the confidence interval eq.(2.21) (with $t_{N-1;1-\gamma/2} = 2$) are given in Table 1 for each sample j. Each sample is based on N = (N_t - N_s) = 1600 evaluated neutron generations (N_t = 2000, N_s = 400) with n_g = 2000 neutrons per generation.

Table 1. Isolated, unirradiated and unpoisoned 4.45 wt.-% ^{235}U-enriched Convoy Series fuel assembly: k_{eff} results of 100 calculation runs performed with the aid of code system [14] using the 44-group neutron cross section library of this system

Sample No.	Starting random number (last three digits)	k_{eff} (average)	Standard deviation of the average	Upper limit of the 95% confidence interval
1.000000E+00	E96	9.476200E-01	4.600000E-04	9.485400E-01
2.000000E+00	E97	9.472900E-01	5.400000E-04	9.483700E-01
3.000000E+00	E98	9.474700E-01	5.300000E-04	9.485300E-01
4.000000E+00	E99	9.478500E-01	4.900000E-04	9.488300E-01
5.000000E+00	E9A	9.484700E-01	5.700000E-04	9.496100E-01
6.000000E+00	E9B	9.485300E-01	5.200000E-04	9.495700E-01
7.000000E+00	E9C	9.480100E-01	4.900000E-04	9.489900E-01
8.000000E+00	E9D	9.480300E-01	5.300000E-04	9.490900E-01
9.000000E+00	E9E	9.471300E-01	4.800000E-04	9.480900E-01
1.000000E+01	E9F	9.473700E-01	5.000000E-04	9.483700E-01
1.100000E+01	EA0	9.484700E-01	4.900000E-04	9.494500E-01
1.200000E+01	EA1	9.484100E-01	5.100000E-04	9.494300E-01
1.300000E+01	EA2	9.471000E-01	5.000000E-04	9.481000E-01
1.400000E+01	EA3	9.471600E-01	5.400000E-04	9.482400E-01
1.500000E+01	EA4	9.479200E-01	5.000000E-04	9.489200E-01
1.600000E+01	EA5	9.476300E-01	5.100000E-04	9.486500E-01
1.700000E+01	EA6	9.480700E-01	5.800000E-04	9.492300E-01
1.800000E+01	EA7	9.474300E-01	4.800000E-04	9.483900E-01
1.900000E+01	EA8	9.479200E-01	4.900000E-04	9.489000E-01
2.000000E+01	EA9	9.483700E-01	5.300000E-04	9.494300E-01
2.100000E+01	EAA	9.468400E-01	5.200000E-04	9.478800E-01
2.200000E+01	EAB	9.486700E-01	4.900000E-04	9.496500E-01
2.300000E+01	EAC	9.481300E-01	4.800000E-04	9.490900E-01
2.400000E+01	EAD	9.488500E-01	5.300000E-04	9.499100E-01
2.500000E+01	EAE	9.474300E-01	5.100000E-04	9.484500E-01
2.600000E+01	EAF	9.475700E-01	4.900000E-04	9.485500E-01
2.700000E+01	EB0	9.479000E-01	5.100000E-04	9.489200E-01
2.800000E+01	EB1	9.482000E-01	5.200000E-04	9.492400E-01
2.900000E+01	EB2	9.481000E-01	5.100000E-04	9.491200E-01
3.000000E+01	EB3	9.478000E-01	4.900000E-04	9.487800E-01
3.100000E+01	EB4	9.482200E-01	5.000000E-04	9.492200E-01
3.200000E+01	EB5	9.475600E-01	5.200000E-04	9.486000E-01
3.300000E+01	EB6	9.484500E-01	4.700000E-04	9.493900E-01
3.400000E+01	EB7	9.478500E-01	5.000000E-04	9.488500E-01
3.500000E+01	EB8	9.479500E-01	4.800000E-04	9.489100E-01
3.600000E+01	EB9	9.477000E-01	4.800000E-04	9.486600E-01
3.700000E+01	EBA	9.480600E-01	5.700000E-04	9.492000E-01
3.800000E+01	EBB	9.469300E-01	5.200000E-04	9.479700E-01
3.900000E+01	EBC	9.478000E-01	5.200000E-04	9.488400E-01
4.000000E+01	EBD	9.482100E-01	5.800000E-04	9.493700E-01
4.100000E+01	EBE	9.480200E-01	5.000000E-04	9.490200E-01
4.200000E+01	EBF	9.474100E-01	5.000000E-04	9.484100E-01
4.300000E+01	EC0	9.472100E-01	4.300000E-04	9.480700E-01
4.400000E+01	EC1	9.474000E-01	4.800000E-04	9.483600E-01
4.500000E+01	EC2	9.481600E-01	5.100000E-04	9.491800E-01

Sample No.	Starting random number (last three digits)	k_{eff} (average)	Standard deviation of the average	Upper limit of the 95% confidence interval
4.600000E+01	EC3	9.480700E-01	5.500000E-04	9.491700E-01
4.700000E+01	EC4	9.479300E-01	3.900000E-04	9.487100E-01
4.800000E+01	EC5	9.485500E-01	5.300000E-04	9.496100E-01
4.900000E+01	EC6	9.475300E-01	5.200000E-04	9.485700E-01
5.000000E+01	EC7	9.478600E-01	5.200000E-04	9.489000E-01
5.100000E+01	EC8	9.484900E-01	5.000000E-04	9.494900E-01
5.200000E+01	EC9	9.476100E-01	5.000000E-04	9.486100E-01
5.300000E+01	ECA	9.482200E-01	4.800000E-04	9.491800E-01
5.400000E+01	ECB	9.476600E-01	5.100000E-04	9.486800E-01
5.500000E+01	ECC	9.485900E-01	4.200000E-04	9.494300E-01
5.600000E+01	ECD	9.475200E-01	4.900000E-04	9.485000E-01
5.700000E+01	ECE	9.482600E-01	4.700000E-04	9.492000E-01
5.800000E+01	ECF	9.475200E-01	5.000000E-04	9.485200E-01
5.900000E+01	ED0	9.485900E-01	4.900000E-04	9.495700E-01
6.000000E+01	ED1	9.487300E-01	5.000000E-04	9.497300E-01
6.100000E+01	ED2	9.483600E-01	4.900000E-04	9.493400E-01
6.200000E+01	ED3	9.482700E-01	4.900000E-04	9.492500E-01
6.300000E+01	ED4	9.474700E-01	5.100000E-04	9.484900E-01
6.400000E+01	ED5	9.481900E-01	5.000000E-04	9.491900E-01
6.500000E+01	ED6	9.482400E-01	5.200000E-04	9.492800E-01
6.600000E+01	ED7	9.485100E-01	5.700000E-04	9.496500E-01
6.700000E+01	ED8	9.471200E-01	4.700000E-04	9.480600E-01
6.800000E+01	ED9	9.481400E-01	4.900000E-04	9.491200E-01
6.900000E+01	EDA	9.474200E-01	5.900000E-04	9.486000E-01
7.000000E+01	EDB	9.473300E-01	4.900000E-04	9.483100E-01
7.100000E+01	EDC	9.484900E-01	5.100000E-04	9.495100E-01
7.200000E+01	EDD	9.467600E-01	5.000000E-04	9.477600E-01
7.300000E+01	EDE	9.476100E-01	5.700000E-04	9.487500E-01
7.400000E+01	EDF	9.472800E-01	5.100000E-04	9.483000E-01
7.500000E+01	EE0	9.478600E-01	4.900000E-04	9.488400E-01
7.600000E+01	EE1	9.488000E-01	4.800000E-04	9.497600E-01
7.700000E+01	EE2	9.479100E-01	5.100000E-04	9.489300E-01
7.800000E+01	EE3	9.479900E-01	3.600000E-04	9.487100E-01
7.900000E+01	EE4	9.473600E-01	4.900000E-04	9.483400E-01
8.000000E+01	EE5	9.476700E-01	4.100000E-04	9.484900E-01
8.100000E+01	EE6	9.491600E-01	6.000000E-04	9.503600E-01
8.200000E+01	EE7	9.477500E-01	5.200000E-04	9.487900E-01
8.300000E+01	EE8	9.478200E-01	5.200000E-04	9.488600E-01
8.400000E+01	EE9	9.466400E-01	4.800000E-04	9.476000E-01
8.500000E+01	EEA	9.486600E-01	5.200000E-04	9.497000E-01
8.600000E+01	EEB	9.463400E-01	4.800000E-04	9.473000E-01
8.700000E+01	EEC	9.480200E-01	5.200000E-04	9.490600E-01
8.800000E+01	EED	9.480900E-01	5.200000E-04	9.491300E-01
8.900000E+01	EEE	9.484200E-01	4.900000E-04	9.494000E-01
9.000000E+01	EEF	9.468400E-01	4.800000E-04	9.478000E-01
9.100000E+01	EF0	9.482800E-01	4.800000E-04	9.492400E-01
9.200000E+01	EF1	9.483200E-01	5.200000E-04	9.493600E-01
9.300000E+01	EF2	9.477400E-01	5.300000E-04	9.488000E-01
9.400000E+01	EF3	9.476700E-01	5.400000E-04	9.487500E-01
9.500000E+01	EF4	9.489100E-01	5.000000E-04	9.499100E-01
9.600000E+01	EF5	9.475400E-01	4.800000E-04	9.485000E-01
9.700000E+01	EF6	9.479600E-01	5.000000E-04	9.489600E-01
9.800000E+01	EF7	9.474200E-01	5.700000E-04	9.485600E-01
9.900000E+01	EF8	9.481100E-01	4.800000E-04	9.490700E-01
1.000000E+02	EF9	9.469700E-01	5.200000E-04	9.480100E-01

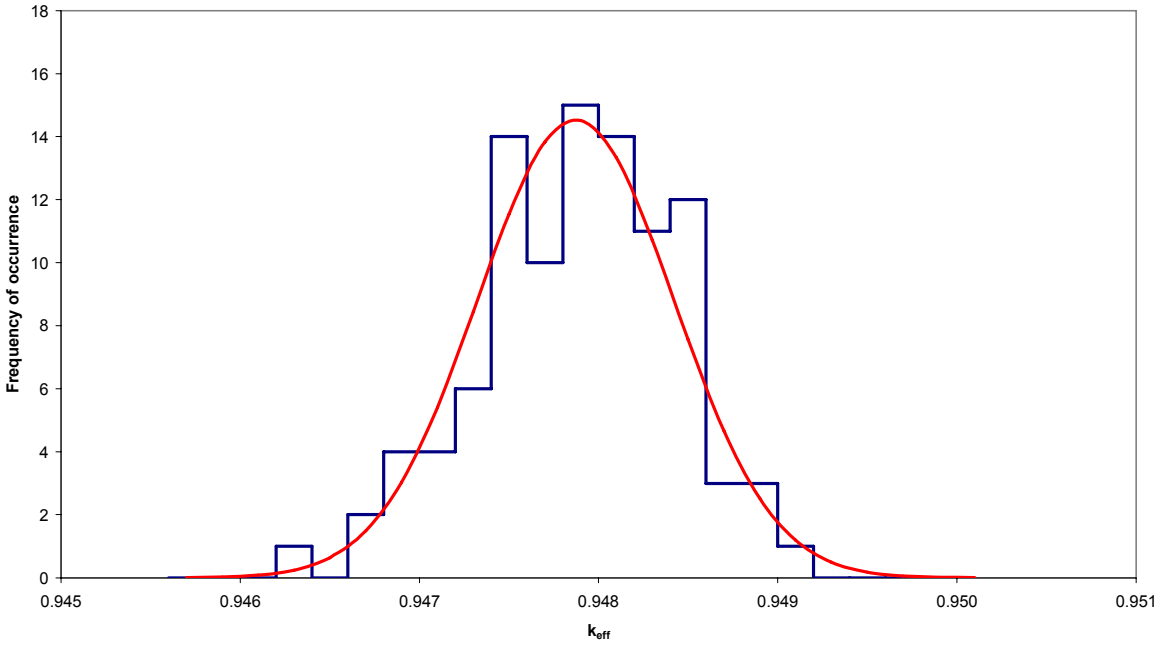

FIG. 1. Histogram of the k_{eff} sample means listed in Table 1.

In Figure 1 the $\bar{k}_j$ results are combined into histogram bins. The number of results observed in bin i ranging from $k_{eff} = \kappa_{i-1}$ to $k_{eff} = \kappa_i$ is compared with the number

$$< m_i > = M \cdot \int_{\kappa_{i-1}}^{\kappa_i} d\kappa \; f(\kappa) \tag{2.37}$$

of k_{eff} values expected in this bin when the normal distribution density

$$f(\kappa) \equiv f(\kappa; k_c, s_c^2) = \frac{1}{\sqrt{2\pi s_c^2}} \exp\left(-\frac{(\kappa - k_c)^2}{2 s_c^2} \right) \tag{2.38}$$

is used for f(κ), where k_c is the sample mean (2.35) and s_c^2 is the sample variance (2.36). The distribution of the numbers $<m_i>$ thus obtained is given by the solid red line in Figure 1. In Figure 2 this distribution is compared to the distribution of the upper limits $\bar{k}_j + 2 \cdot \sqrt{N^{-1} \cdot \hat{\sigma}_j^2}$ of the confidence intervals (2.21) which are analogously combined into histogram bins (solid blue line in Fig. 2). The dashed lines in red color give the result obtained for the sample mean k_c and the result obtained for $k_c + 2 \cdot \sqrt{s_c^2}$, respectively.

The comparison shown in Fig. 2 is really instructive. It appears from this figure that there are in fact some results $\bar{k}_j + 2 \cdot \sqrt{N^{-1} \cdot \hat{\sigma}_j^2}$ which fall below the sample mean k_c. As experience proves, very likely an applicant comes in with the lowest $\bar{k}_j + 2 \cdot \sqrt{N^{-1} \cdot \hat{\sigma}_j^2}$ value observed (0.9473, cf. Table 1, sample no. 86), and very likely a regulator comes out with the highest value $\bar{k}_j + 2 \cdot \sqrt{N^{-1} \cdot \hat{\sigma}_j^2}$ observed (0.95036, cf. Table 1, sample no. 81). So, the applicant expects the regulator's thumbs up, but the regulator will give him the thumbs down. This can become the beginning of a wonderful friendship then and only then, if a decision is made with the aid of eq.(2.24).

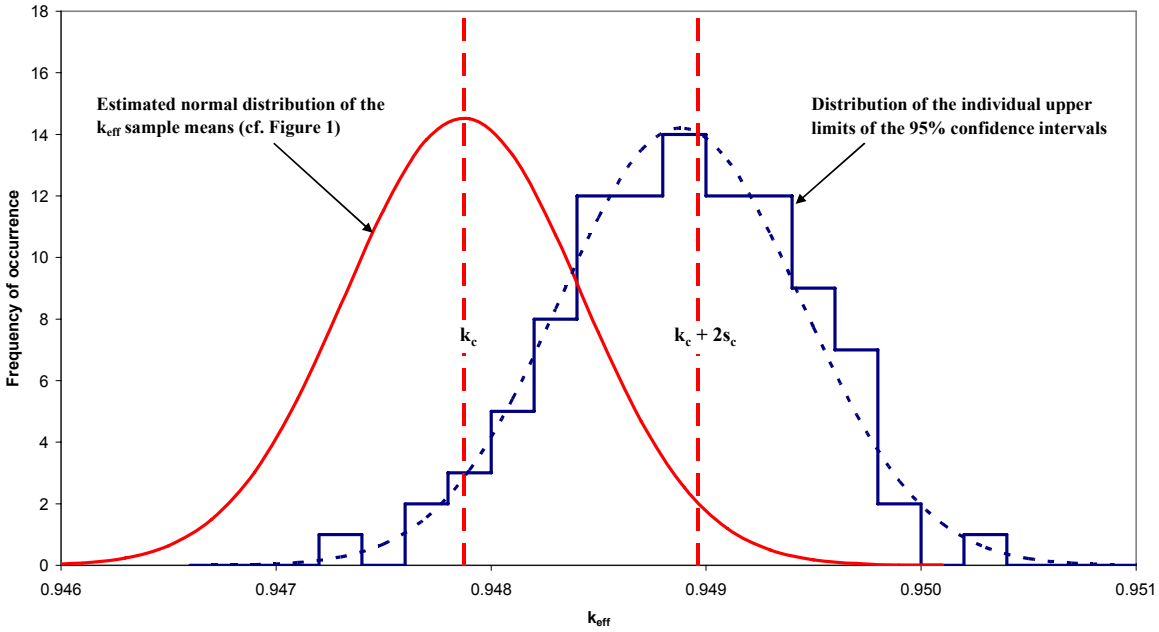

FIG. 2. Comparison of the distribution of the individual upper limits of the 95% confidence intervals given in Table 1 with the $k_o + 2s_o$ value of the distribution of the k_{eff} sample means.

The given example demonstrates in fact that it is necessary to exclude with the aid of eq.(2.24) that a significant part of the unknown distribution $F(k_{eff})$, given by eq.(2.32), is above the maximum allowable neutron multiplication factor given by the right-hand side of the criterion (1.3) [15]. μ and σ^2 in eq.(2.32) are still unknown, but with the aid of the sample mean (2.35) and the sample variance (2.36) equation (2.24) can be solved now. As appears from Fig. 2, the upper limit $L_u(\alpha,\gamma)$ in eq.(2.24) must be given by an expression of the form

$$L_u(\alpha,\gamma) = k_c + \lambda_c(M;\alpha,\gamma)\cdot\sqrt{s_c^2}\ . \tag{2.39}$$

In fact, as will be shown below, a unique solution can be derived for the factor λ_c by evaluating the distribution properties of the estimators k_c and s_c^2.

Before this is shown it is reasonable to drop the assumption that the results $(k_{1j}, \ldots, k_{Nj})$ of each and every sample $j = 1, \ldots, M$ are normally distributed. As follows from eq.(2.32), this assumption is not required. It is only required that the sample means $\bar{k}_j$ of the samples $j = 1, \ldots, M$ can be assumed to be normally distributed.

2.4. Sample mean and Central Limit theorem of statistics

Let $(z_1, \ldots, z_N)$ be a sequence of *independent* random variables each from a probability distribution $F_i(z)$ with expectation

$$\mu_i = \int_{\Omega_i} z\, dF_i(z) \tag{2.40}$$

and variance

$$\sigma_i^2 = \int_{\Omega_i} (z-\mu_i)^2\, dF_i(z)\ . \tag{2.41}$$

Irrespective of the distributions $F_i(z)$, the distribution of the sum

$$S_N = \sum_{i=1}^{N} z_i \tag{2.42}$$

will have the expectation

$$E[S_N] = \sum_{i=1}^{N} \mu_i \tag{2.43}$$

and the variance

$$V[S_N] = \sum_{i=1}^{N} \sigma_i^2 \equiv r_N^2 \tag{2.44}$$

provided that the z_i are independent, the individual expectations μ_i exist, and the individual variances σ_i^2 exist and are all finite or, at least, do not approach infinity as fast as i.

The Central Limit theorem of statistics states that, independent of the distributions $F_i(z)$, the standardized variable

$$y_N = \frac{S_N - E[S_N]}{\sqrt{V[S_N]}} = \sum_{i=1}^{N} \frac{z_i - \mu_i}{r_N} \tag{2.45}$$

converges *in distribution* (cf. section 2.1) to a standard normal distribution variable when $N \to \infty$,

$$F(y_N) \underset{N\to\infty}{\longrightarrow} \Phi(y_N) = \int_{-\infty}^{y_N} dt\, \varphi(t,1), \tag{2.46}$$

provided that [15]

$$\lim_{N\to\infty} \int_0^1 d\tau \sum_{i=1}^{N} \int_{\mu_i - r_N\tau}^{\mu_i + r_N\tau} \left(\frac{z - \mu_i}{r_N} \right)^2 dF_i(z) = 1. \tag{2.47}$$

From the viewpoint of applications in practice, the meaning of the condition (2.47) is as follows: Because the integration over z in eq.(2.47) is restricted to the range $|z - \mu_i| \le r_N\tau$ and because τ is restricted to $\tau \in (0, 1)$, eq.(2.47) put limits on the summands $(z_i - \mu_i)/r_N$ that contribute to y_N (cf. eq.(2.45)); the summands shall have some uniformity.

This is met, in particular, in the usual case that the z_i are N different trials of the same experiment. In this case one has (cf. equations (2.2) and (2.6))

$$\mu_i = \mu, \forall i = 1,..., N, \tag{2.48}$$

$$\sigma_i^2 = \sigma^2, \forall i = 1,..., N, \tag{2.49}$$

and hence

$$r_N^2 = N\sigma^2. \tag{2.50}$$

The range $|z - \mu_i|$ becomes therefore $|z - \mu_i| = |z - \mu| \le r_N\tau = \tau\cdot\sqrt{N\sigma^2}$, i.e., the integration over z in eq.(2.47) extends, for $\tau > 0$, to $(-\infty, +\infty)$ as $N \to \infty$. Eq.(2.47) thus becomes (with $\varepsilon > 0$)

$$\lim_{N\to\infty}\lim_{\varepsilon\to 0}\int_{\varepsilon}^{1} d\tau\ \frac{N}{r_N^2}\int_{\mu_i-\tau\cdot\sqrt{N\sigma^2}}^{\mu_i+\tau\cdot\sqrt{N\sigma^2}}(z-\mu)^2\,dF(z)=\lim_{\varepsilon\to 0}\int_{\varepsilon}^{1} d\tau\ \frac{1}{\sigma^2}\int_{-\infty}^{+\infty}(z-\mu)^2\,dF(z)=\lim_{\varepsilon\to 0}\int_{\varepsilon}^{1} d\tau = 1. \qquad (2.51)$$

With equations (2.1), (2.48) and (2.50) the variable (2.45) becomes

$$y_N = \frac{\bar{z}-\mu}{\sqrt{N^{-1}\cdot\sigma^2}}. \qquad (2.52)$$

This expression is well-known in the meantime. So, *independent* of the distributions underlying the observations $(z_1, \ldots, z_N)$ the variable (2.52) converges in distribution to the standard normal distribution variable when $N \to \infty$. Thus, the sample mean $\bar{z}$ is *asymptotically* normally distributed with expectation μ and variance σ^2/N.

Since $N \to \infty$ cannot be realized the question naturally arises as how "far away" the real life distribution $F(y_N)$ of the variable (2.52) is from the standard normal distribution $\Phi(y_N)$. There is obviously no general answer on this question, since the "difference" between $F(y_N)$ and $\Phi(y_N)$ depends on the distribution underlying the observations $(z_1, \ldots, z_N)$ and the number N.

So what can be done in practice? First, one can check, as was already described in section 2.2.2, the hypothesis H_0 that the observations $(z_1, \ldots, z_N)$ can be assumed to be normally distributed. If H_0 is not rejected one is through because of the linearity of eq.(2.1) [16]. If H_0 is rejected, then one can draw a sample $(\bar{z}_1, ..., \bar{z}_M)$ on the sample mean $\bar{z}$ and check the re-formulated hypothesis H_0 that the distribution of the results $\bar{z}_j$, j = 1, …, M, follow a normal distribution. If H_0 is rejected by the outcome of the test even though the number M is sufficiently large [17] then the number N has to be increased since the Central Limit theorem says that there is a number N_0 such that for all $N \ge N_0$ the distribution of the sample mean $\bar{z}$ can be well approximated by a normal distribution.

This brings us back to the previous section where it was discussed that a sample of M normally distributed sample means $\bar{z}_j$, j = 1, …, M, is needed to solve eq.(2.24).

3. Estimation of the one-sided 95%/95% tolerance limit

Let us assume that N in eq.(2.31) is sufficiently large so that the variable (2.31) can be assumed, because of the Central Limit theorem, to be standard normally distributed. $F(k_{eff})$ in eq.(2.24) is thus given by eq.(2.32). It follows therefore from eq.(2.24)

$$\int_{-\infty}^{L_u} dF(k_{eff}) = \int_{-\infty}^{\frac{L_u-\mu}{\rho}} dt\,\varphi(t,1) \ge \int_{-\infty}^{\Phi_{1-\gamma}} dt\,\varphi(t,1) = (1-\gamma) \qquad (3.1)$$

with

$$\rho = \sqrt{N^{-1}\cdot\sigma^2}\ . \qquad (3.2)$$

It follows from eq.(3.1)

$$\frac{L_u - \mu}{\rho} \geq \Phi_{1-\gamma} \tag{3.3}$$

and hence

$$\mathrm{Min}(L_u) = \mu + \rho \cdot \Phi_{1-\gamma} . \tag{3.4}$$

As can be seen from eq.(3.1), $\Phi_{1-\gamma}$ is that value of the standardized variable t that results in the probability content $(1 - \gamma)$; $\Phi_{1-\gamma} = 1.645$ for $(1 - \gamma)\cdot 100\% = 95\%$ [13].

As appears from expressions (3.1) and (3.4), if L_u falls below $\mathrm{Min}(L_u)$ the probability (2.25),

$$P[k_{eff} < L_u] = \int_0^{L_u} dF(k_{eff}), \tag{3.5}$$

falls below $(1 - \gamma)$. As follows from eq.(2.24) the probability that the probability (2.25) falls below $(1 - \gamma)$ is α. So therefore,

$$P[L_u < \mathrm{Min}(L_u)] = \int_0^{\mathrm{Min}(L_u)} dF(L_u) = \alpha . \tag{3.6}$$

It is therefore necessary to determine the probability distribution $F(L_u)$ of L_u.

With eq.(2.39) the expectation $E[L_u]$ and the variance $V[L_u]$ of L_u become

$$E[L_u] = E[k_c] + \lambda_c(M;\alpha,\gamma)\cdot E[s_c] \tag{3.7}$$

and

$$V[L_u] = V[k_c] + \lambda_c^2(M;\alpha,\gamma)\cdot V[s_c] + 2\cdot\lambda_c(M;\alpha,\gamma)\cdot \mathrm{cov}(k_c, s_c) \tag{3.8}$$

respectively, where $\mathrm{cov}(k_c,s_c)$ denotes the covariance of k_c and s_c [18]; s_c is given by eq.(2.36).

As follows from eq.(2.35) with equations (2.33), (2.34), and (3.2)

$$E[k_c] = \mu \tag{3.9}$$

and

$$V[k_c] = \frac{1}{M^2}\sum_{j=1}^{M} V[\bar{k}_j] = \frac{N^{-1}\cdot\sigma^2}{M} = \frac{\rho^2}{M} . \tag{3.10}$$

Since k_c is the sample mean of M independent samples $(\bar{k}_1,...,\bar{k}_M)$ which can be assumed as normally distributed because of the Central Limit theorem, k_c is normally distributed with expectation (3.9) and variance (3.10). Due to the normality of the $\bar{k}_j$, j = 1, …, M, the quantity

$$q^2 = (M-1)\cdot\frac{s_c^2}{\rho^2} \tag{3.11}$$

is distributed *independently* from k_c and follows a χ^2-distribution with (M-1) degrees of freedom which has the expectation

$$E[q^2] = (M-1), \tag{3.12}$$

[9][13]. The quantity

$$q = \sqrt{(M-1)} \cdot \frac{s_c}{\rho} \tag{3.13}$$

follows then a so called "χ-distribution" [12] with expectation [13]

$$E[q] = \frac{\sqrt{(M-1)}}{\rho} \cdot E[s_c] = \sqrt{(M-1)} \cdot \beta_M, \tag{3.14}$$

$$\beta_M \equiv \sqrt{\frac{2}{(M-1))}} \cdot \frac{\Gamma\left(\frac{M}{2}\right)}{\Gamma\left(\frac{M-1}{2}\right)} \tag{3.15}$$

where Γ denotes the Γ-function [12].

Eq.(3.14) can be re-written

$$E[s_c] = \beta_M \rho. \tag{3.16}$$

By definition, the variance of s_c is

$$V[s_c] = E\left[(s_c - E[s_c])^2\right] = E[s_c^2] - (E[s_c])^2. \tag{3.17}$$

It follows from equations (3.11) and (3.12)

$$E[s_c^2] = \frac{\rho^2}{(M-1)} \cdot E[q^2] = \rho^2. \tag{3.18}$$

Thus, with this expression and with eq.(3.16) equation (3.17) becomes

$$V[s_c] = \left(1 - \beta_M^2\right) \cdot \rho^2. \tag{3.19}$$

It can be shown (cf. Ref. [13]) that, for M > 10, the distribution of the standardized variable

$$\frac{s_c - E[s_c]}{\sqrt{V[s_c]}} = \frac{s_c - \beta_M \rho}{\sqrt{\left(1 - \beta_M^2\right)} \cdot \rho} \tag{3.20}$$

can be approximated very well by the standard normal distribution. So, for M > 10 the quantity s_c is close to normal with expectation (3.16) and variance (3.19). For M > 10 expression (3.15) becomes

$$\beta_M \approx 1-\frac{1}{4\cdot(M-1)} \text{ and hence } \left(1-\beta_M^2\right)\approx \frac{1}{2\cdot(M-1)}. \tag{3.21}$$

So, for M > 10 equations (3.16) and (3.19) are well approximated by

$$E[s_c]=\rho, \tag{3.22}$$

$$V[s_c]=\frac{\rho^2}{2(M-1)}. \tag{3.23}$$

Since q^2, hence q and therefore s_c are independently distributed from k_c the covariance term in eq.(3.8) vanishes. With equations (3.9), (3.10), (3.22) and (3.23) expressions (3.7) and (3.8) are very well approximated, for M > 10, by

$$E[L_u]=\mu+\lambda_c\rho, \tag{3.24}$$

$$V[L_u]=\frac{\rho^2}{M}+\lambda_c^2\cdot\frac{\rho^2}{2\cdot(M-1)}=\frac{\rho^2}{2M}\cdot\left(2+\frac{M}{M-1}\cdot\lambda_c^2\right). \tag{3.25}$$

Since k_c is normally distributed and s_c is, for M > 10, very well approximated by a normal distribution, the upper limit L_u given by eq.(2.39) is, for M > 10, close to normal with expectation (3.24) and variance (3.25). Expression (3.6) becomes, therefore,

$$\int_{-\infty}^{Min(L_u)} dF(L_u)=\int_{-\infty}^{\frac{Min(L_u)-E[L_u]}{\sqrt{V[L_u]}}} dt\,\varphi(t,1)=\int_{-\infty}^{\Phi_\alpha} dt\,\varphi(t,1)=\int_{-\infty}^{-\Phi_{1-\alpha}} dt\,\varphi(t,1)=\alpha. \tag{3.26}$$ [19)]

From this it follows

$$Min(L_u)=E[L_u]-\Phi_{1-\alpha}\cdot\sqrt{V[L_u]}. \tag{3.27}$$

So, with equations (3.4), (3.24) and (3.25) expression (3.27) becomes

$$\mu+\rho\cdot\Phi_{1-\gamma}=\mu+\lambda_c\cdot\rho-\Phi_{1-\alpha}\cdot\frac{\rho}{\sqrt{2M}}\cdot\sqrt{2+\frac{M}{M-1}\cdot\lambda_c^2}. \tag{3.28}$$

The quantities μ and ρ drop out, and one gets for λ_c:

$$\lambda_c(M;\alpha,\gamma)=\frac{2\cdot(M-1)}{2\cdot(M-1)-\Phi_{1-\alpha}^2}\cdot\left[\Phi_{1-\gamma}+\frac{\Phi_{1-\alpha}}{\sqrt{2\cdot(M-1)}}\cdot\sqrt{\left[2\cdot(M-1)-\Phi_{1-\alpha}^2\right]\cdot\frac{1}{M}+\Phi_{1-\gamma}^2}\right]. \tag{3.29}$$

Figure 3 shows λ_c as a function of M for (1 - α) = 0.95 and (1 - γ) = 0.95 and 0.975.

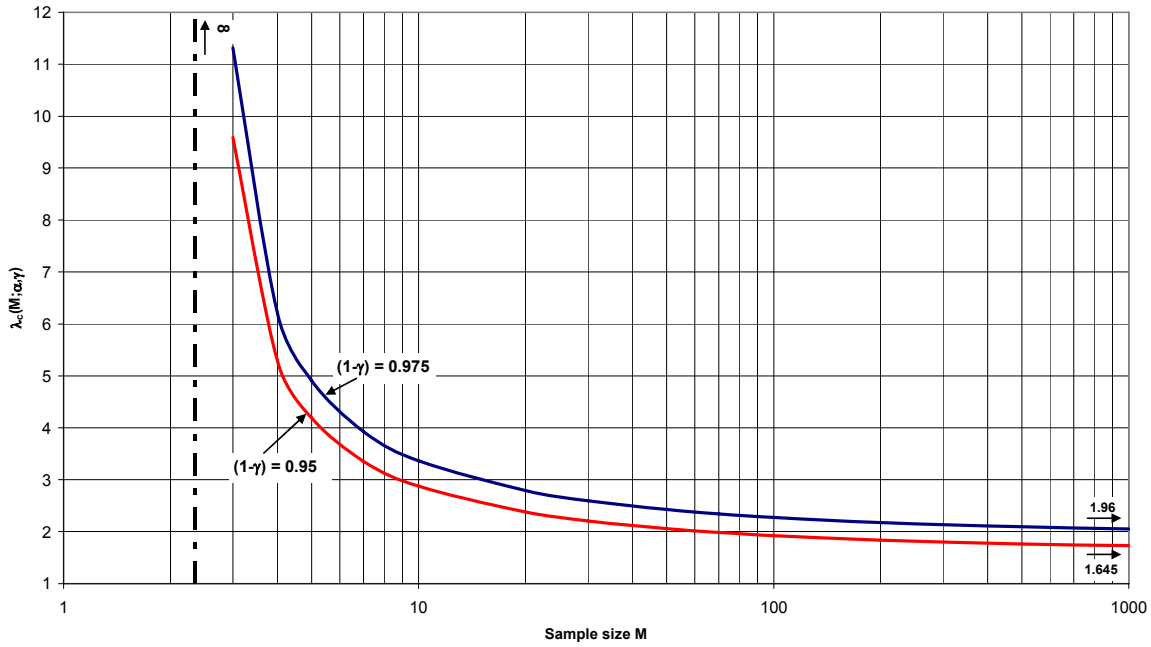

FIG. 3. Factor $\lambda_c(M; \alpha, \gamma)$ for one-sided $(1-\gamma)/(1-\alpha)$ tolerance limits with $(1-\alpha) = 0.95$ (cf. equation (3.29)).

As appears from eq.(3.29), as $M \to \infty$ λ_c converges to $\Phi_{1-\gamma}$. This was to be expected since, due the Central Limit theorem, as $M \to \infty$ k_c is normal with expectation μ and variance $\rho^2 = \sigma^2/N$ (cf. discussion of equations (2.35) and (2.36)). μ and ρ^2 are known for $M = \infty$ since k_c converges, according to the laws of large numbers, to μ when $M \to \infty$, and the sample variance s_c^2 given by eq.(2.36) converges, in probability, to ρ^2 when $M \to \infty$. So, as $M \to \infty$ the limit $L_u = k_c + \lambda_c s_c$ given by eq.(2.39) converges to the expression given by the right-hand side of eq.(3.4).

As can be seen from Figure 3, if one wants to keep λ_c at a value of about 2 for $M \to \infty$ one has to choose $(1 - \gamma) = 0.975$ instead of $(1 - \gamma) = 0.95$ for $(1 - \alpha) = 0.95$. However, this is not required by the criticality safety regulations, codes and guides mentioned at the beginning of the paper.

In real life M is finite. Note that λ_c is not defined for $2 \cdot (M - 1) \leq \Phi_{1-\alpha}^2$. It follows therefore

$$M > \frac{\Phi_{1-\alpha}^2}{2} + 1 \tag{3.30}$$

and hence $M \geq 3$ for $(1 - \alpha) = 0.95$. This confirms what was already stated in section 2.3: *The information from one calculation run, i.e. from one sample, is not sufficient to estimate a 95%/95% tolerance limit* for the distribution of the sample mean.

As appears from Fig. 3, if one wants to keep λ_c at a value of about 2, one needs a sample $(\bar{k}_1, ..., \bar{k}_M)$ of size $M \approx 70$ on $\bar{k}$ [20].

The expressions (2.35), (2.36) and (3.29) given for the upper tolerance limit eq.(2.39) refer to the case that the neutron multiplication factor k_{eff} is not analyzed as a function of any parameter such as initial enrichment, burnup etc. The respective expressions for the upper tolerance limit of k_{eff} as a function of a parameter x, $k_{eff} = k_{eff}(x)$, are more complex. The solution of this case can be found in Ref. [8]. In Ref. [13] solutions for linear regression applications are given.

4. Appendix A: Notes

(1) For details see Ref. [9]. Readers who have the feeling that Ref. [9] is too comprehensive can find a glossary and definitions of the basic terms of statistics used in the paper on hand in Ref. [10].

(2) An estimator is a function of the observations leading from the observations to an estimate. The estimate is the numerical value yielded by the estimator for a particular set of observations.

(3) An estimator $\hat{x}_N$ of a parameter x based on N observations is unbiased if its expectation $E[\hat{x}_N]$ does not deviate from the true value x_0 for all N: $E[\hat{x}_N] - x_0 = 0, \ \forall N$. Otherwise $\hat{x}_N$ is biased.

The unbiasedness of the sample mean (2.1) is obviously due to the linearity of the expectation operator E:

$$E[\bar{z}] = E\left[\frac{1}{N}\sum_{i=1}^{N} z_i\right] = \frac{1}{N}\sum_{i=1}^{N} E[z_i] = \frac{1}{N}\sum_{i=1}^{N}(\mu) = \mu \text{ , cf. eq.(2.2).}$$

(4) Note that the unbiasedness and convergence of an estimator are not related, neither one implies the other. Assume, for instance, that only positive values are possible for zi. Then expression (2.1) is absolutely convergent (when $N \to \infty$) with sum μ provided that (2.4) or (2.5) is met [11]. Then, as follows from Cauchy's theorem for multiplication of absolutely convergent numerical series (cf. Ref. [12]), $\bar{z}^2$ is absolutely convergent (when $N \to \infty$) with sum μ^2. However, $\bar{z}^2$ is not an unbiased estimator of μ^2 as can easily be verified by using the fact that the expectation E is a linear operator.

(5) $t_{N-1;1-\gamma/2} \leq 2$ for $(1 - \gamma) = 0.95$ and $N \geq 60$ [13].

(6) $t_{N-1;1-\gamma/2} < 3$ for $(1 - \gamma) = 0.99$ and $(N - 1) > 60$ [13].

(7) Note that the Student's t-distribution is defined on $t \in (-\infty, +\infty)$.

(8) $t_{N-1;1-\gamma} < 1.67$ for $(1 - \gamma) = 0.95$ and $(N - 1) \geq 70$ [13].

(9) An example is given in Ref. [8] where a Student's t-distributed variable is derived which allows to estimate a confidence interval of the axial end effect as a function of the average burnup of axial burnup profiles.

(10) Most of the today's statistical criticality calculation codes include such a test procedure.

(11) The reason that in practice rejection of H_0 is often put down to a supposed source convergence problem is probably the following: There is an a-priori-belief that repeating of an observation results in a set of normally distributed results (z_1, …, z_N) provided that N is sufficiently large. It is therefore assumed that, if source convergence is reached, then the results should be normally distributed.

This a-priori-belief seems to be very firm and ineradicable. It is often assumed, for example, that a person making repeated measurements of the distance between two fixed points will obtain a set of measurements that is normally distributed with mean equal to the "true" distance and width given by the precision of the method used. Well, the a-priori-belief that repeating of a measurement results in a set of normally distributed measurements is, *to some extent*, empirically supported. And there are suggestions from theory that normality is at least a good approximation in many cases.

(12) Cf. note 3)

(13) A standardized variable y is a variable which has E[y] = 0 and V[y] = 1.

(14) This can easily be proved: In conditions (2.4) and (2.5) N has only to be replaced with M and σ_i^2 has then to be identified with the expression (2.34).

(15) In the discussion of the given example the terms Δk_j of the right-hand side of inequality (1.3) have been disregarded. However, because of the obvious purpose in giving this example, this does not result in any loss of generality.

(16) It was already stated several times that a linear function of normally distributed variables is normally distributed.

(17) Test procedures of the type needed here require to decide how to bin the observations. Too few bins carry too little information, but too many bins lead to too few events per bin. There are procedures [9][16] to choose an optimum number n of bins which have *equal probability contents* under the hypothesis H_0. n increases with M, so that M should be sufficiently large to get a sufficiently large number of bins. On the other hand the number of events per bin expected under H_0 should not fall below 5.

So, by the way, the histograms shown in Figs 1 and 2 are not those which are used in tests.

(18) The covariance of two random variables x and y is defined as

$$\mathrm{cov}(x,y) = \int_{\Omega_{x,y}} (x - E[x])(y - E[y])\, dF(x,y).$$

If x and y are mutually independent then cov(x,y) = 0, cf. Ref. [9].

(19) $-\Phi_{1-\alpha} = \Phi_\alpha$ follows from the symmetry of $\varphi(t,1)$, cf. eq.(2.15).

(20) As can be seen from Fig. 3, for M = 100 λ_c is less than 2. So, as appears from Fig. 2, the applicant mentioned in section 2.3 will be happy.

However, in everyday work the ordinary applicant is not keen to perform 100 calculation runs for one case. So, let us assume he has performed the first seven runs the results of which are listed in Table 1. The upper 95%/95% tolerance limit based on these seven runs amounts to 0.94940. Our applicant will remain happy.

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