TECHNOLOGICAL INNOVATIONS FOR
SPENT FUEL STORAGE

(Session 7)

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THE ROLE OF TECHNOLOGICAL INNOVATIONS FOR DRY STORAGE OF USED NUCLEAR FUEL

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Abstract

We cannot predict the recovery from the financial crisis, but regardless of whether it is slow or quick, the global need for energy and the growth of electricity consumption have been confirmed. Many countries throughout the world are pursuing or have publicly expressed their intention to pursue the construction of Nuclear Power Plants or to extend the life of existing nuclear reactors and to address the back end of the fuel cycle. As always in history, when economic constraints become more severe, the answer is often innovation. Maintaining the high level of performance of nuclear energy and increasing safety with an attractive cost is today's challenge. It is true for reactors, true also for fuel cycle and in particular for the back end: recycling and interim storage. Interim storage equipment or systems of used fuel are considered in this presentation. The industry is ready to provide support to countries and utilities for the development of radioactive material transportation and storage, and is striving to develop innovative solutions in wet or dry storage systems and casks and to bring them to the market. This presentation will elaborate on the two following questions: Where are the most crucial needs for technological innovations? What is the role of innovation? The needs of technological innovation are important in 3 domains: storage equipment design, interfaces and handling of used fuel and safety justification methodology. Concerning the design, continuous effort for optimisation of used fuel storage equipment requires innovations. These designs constitute the new generation of dry storage casks. The expectations are a higher payload thanks to new materials (such as metal matrix composites) and optimised geometry for criticality-safety, better thermal evacuation efficiency to accept higher fuel characteristics (more enrichment, burnup, shorter cooling time), resistance to impact of airplanes. Designs are also expected to be optimised for sustainable development. Innovative production methods with combination of economic and reliable technologies are also a key factor for an acceptable used fuel storage system by the customers. Concerning interfaces for handling, loading and transfer, improvements and optimisation are expected for fuel transfer or cask transfer or tie down systems. As the competitiveness relies also upon the optimal fuel utilization, a target for R&D is quick evacuation. Some key processes like drying to avoid gas build up in cask cavity also requires innovative solutions. Regulatory requirements for safe storage are still progressing, specifically to reduce dose exposure. To match these expectations, safety justification methods need to be updated and the knowledge of safety margins must be continuously improved. In this improvement process we observe several innovations, for example for criticality-safety with burnup credit, moderator exclusion, evaluation of fuel integrity, analysis of accident conditions with new shock absorbing covers and new calculation models, and also new evaluation methods of material behaviour for longer periods of interim storage up to 100 years. The role of innovation is to help development and acceptance of an industry or activity which is critical to limit CO_2 emissions. In the new landscape of the nuclear renaissance, new technical developments will be associated with innovative features and designs, which will offer both high performance systems to customers who have the responsibility of storage and safety and reduction of radiation exposure benefits to all the stakeholders.

1. INTRODUCTION

Today, innovation is everywhere: for companies as a project, in order to provide new products, obtain a market advantage, be competitive in the global economy and to overcome current difficulties. The nuclear industry offers electricity in a sustainable way, without CO_2 emissions, to answer to ever-growing needs. But, in an environment still uncertain, the nuclear industry and its perception are changing. The management of used fuel is searching especially for new ideas, innovations and emerging solutions. The purpose of this paper is to present the framework of such new innovation challenges and the role of technological innovation for used fuel management.

2.   THE INNOVATION MANDATE

Appropriate management of used fuel is a key issue for the strategy of the back end of the nuclear fuel cycle. Today nuclear utilities and storage system vendors share the same objective: reduce the cost of the back end of the fuel cycle. Innovation is a keystone for the strategy of the back end, it is of vital importance and storage system vendors must face this challenge. The needs of nuclear utilities evolve, becoming more and more demanding. Innovation is necessary not only to satisfy customer needs but also to anticipate evolutions and help to stay competitive. Moreover, thanks to the incorporation of improvements brought by new technologies, innovation is also a powerful tool to avoid obsolescence of storage solutions.

Nuclear utilities require advanced storage solutions, with additional payload, acceptance of higher discharge burnups and easier licensing processes. Actually, the expectations from customers for used fuel storage affect five key performances:

2.1. Storage capacity and economical performance

The first key performance of used fuel management is a higher payload. Added value expected from innovation comes from more capacity, a greater quantity of spent fuel in the cavity of a dry storage cask. The flexibility of the system can bring added value: for example, a better adjustment to the fuel specification. What is expected is the adaptation of the system to modifications of spent fuels: higher initial enrichment, higher fuel burnup, evolving fuel designs and geometry and acceptance of Mox fuel.

2.2. Safety and ease of licensing

The safety and justification methodology must take into account the latest scientific developments and publications. Therefore, there is a need for access to updated expertise. Experts from industry and from regulatory authorities should be involved in innovation process, be familiar with new ideas, and be prepared to evaluate them. Typically, an important issue is the knowledge of long term behaviour of materials and components. As projects and licensing of final repositories are postponed and decisions concerning reprocessing are delayed, extended duration of used fuel storage is now considered; member states have referred to storage periods of 100 years and even beyond. New materials or surface treatments and new experimental or theoretical approaches on the assessment of material behaviour are now being considered. These innovations are necessary and should be verified by authorities.

2.3. Ease of operation and reduction of doses of operators

The feedback of dose intake from more than 30 years of experience of dry storage operation has led to recommendations of simple and efficient system and procedures for closure and shipment. At the same time the standard requirements for acceptable dose issued by regulators request constantly improvement. Solutions having good efficiency for dose reduction are therefore very attractive for the customers.

2.4. Impact of selected technology for sustainable development

Very often recent dry storage systems have had to comply with objectives of sustainable development. Selection of materials and energy consumption of solutions is compared and innovations are welcomed.

2.5. Impact of selected technology on proliferation issues

Even with this sensitive subject on non-proliferation, the evolution of technology brings improvements to existing systems or develops new interesting systems.

3. INNOVATION PROCESS

In order to permanently provide good and economical solutions through innovation, the first action is to define a structured innovation process. Designer teams involved in innovation always implement an innovation process. The myth of individual innovation dictates that innovation is natural and that nothing else is necessary. Experience shows on the contrary that a structure is beneficial [1]. The willingness to structure an innovation process is important. This innovation process allows the maintenance of a high performance standard. A typical current process most often includes four steps:

(1)   Regular interviews with customers and utilities;
(2)   Access, capture and reuse of experience feedback and knowledge;
(3)   Creativity and idea generation;
(4)   Selection of ideas.

New technologies, improved manufacturing processes or simple ideas should be screened for significant added value.

It is also important to have a communication system (innovation is creation, collaboration, communication).

Methods for creation and expression of ideas:

For individual and spontaneous ideas, an *idea management data bank* is very often implemented in companies, and sometimes they are shared with suppliers, or customers. It is a current bottom-up process which is very effective (see §5 for the idea data bank of TN international for example).

Often the best ideas come from exchanges between colleagues, discussions, collaborative meetings. That is why it is a interesting to establish *creativity groups* and collaborative tools, (see next § for the methods implemented at TN international for example);

Success in innovation process needs involvement of top management. The top management role is to make the means available by deciding to appoint people dedicated to stimulate and coordinate innovation (innovation catalysts), to galvanize creativity groups and a budget for the development of innovative projects, and to put in place incentives (events, awards).

4. INNOVATION AREAS

What are the innovation areas which should be emphasised for a storage management designer? We consider 3 main technological innovation areas: storage equipment design, interfaces and handling of used fuel, and safety justification methodology.

Concerning storage equipment design, permanent effort is made in the technology of baskets for dry storage or racks for wet storage, aiming at a higher payload. Metal matrix composites or new materials allow more compact solutions and optimized geometry for criticality-safety. In the case of dry storage, designers are looking for a better thermal evacuation efficiency to accept higher fuel characteristics, and also for resistance to impact of airplanes. Designs should be optimised for sustainable development considering availability of raw materials,

energy consumption and costs of dismantling. For both dry and wet systems, innovative fabrication methods with a combination of economical and reliable technologies are also a key factor for an acceptable used fuel storage system by customers.

Concerning second innovation area interfaces for handling, loading and transfer, improvements and optimization are expected for fuel transfer or cask transfer or tie-down systems. As competitiveness also relies upon optimal fuel utilization, a target is quick evacuation. Some key processes like drying for avoiding gas build-up in cask cavities also require innovative solutions.

The third innovation area is the justification methodology. Often it is difficult to admit that innovative methods can be used to demonstrate the safety. Proven methodology is preferred. Yet, recent developments in science or modelization have to be considered. On the other hand, regulatory requirements for safe storage continue to progress, specifically to reduce dose exposure. To match these expectations, safety justification methods need to be updated and knowledge of safety margins must be continuously improved. In this improvement process we observe several innovations, for example for criticality-safety with burnup credit, moderator exclusion, evaluation of fuel integrity, analysis of accident conditions with new shock absorbing covers, new calculation models, and also new evaluation methods of material behaviour for longer periods of interim storage up to 100 years.

When ideas in each area have been selected through this process, then it is a significant step to establish an innovation portfolio and to concentrate effort on the selected ideas, to be implemented in actual used fuel management systems as a first phase.

5.   EXAMPLE OF INNOVATION POLICY IN TN INTERNATIONAL

Many engineers are naturally creative but the context is not always favourable. We give hereafter a few suggestions to obtain more innovations. These suggestions come from experience at TN International.

A first suggestion is to find or to create an open space for innovation: for example the ID SCHOOL^{TM}. Since September 2009, the management of TN international has decided to set up an ID SCHOOL^{TM}, to make available a special room dedicated to innovation. Designers, engineers or anyone from the company who wish to innovate through brainstorming or by simply changing place, out of his everyday environment (usual desk, usual meeting rooms), can organize an innovation group and gather in the ID SCHOOL^{TM}. Small tools, games, web connections, mock-ups, supports for creativity are provided. In the ID SCHOOL^{TM} the creativity groups gather for brainstorming, invent solutions, check their validity in a creative ambiance.

FIG. 1. ID SCHOOL^TM.

An innovation policy calls for methods to stimulate creativity. Proposed by consultants different creativity tools and methods are available on the net, already experienced in various industries and their efficiency has been proved. The industries of telecommunication and internet have shown many successful examples. It is also well known in the automobile industry. In a sector as sensitive to safety issues as nuclear industry and especially the management of used fuel, with some inspectors reluctant to evaluate innovations, it is sometimes difficult but these methods should not be rejected; instead they should be adapted to the time scale and the specific regulatory environment of nuclear energy. We know many innovation challenges in the field of used fuel management: acceptance of damaged fuels, resistance to airplane crash. It is possible to challenge an innovation team with such issues. TN international has developed and implemented a special method for an innovation team called *method EFICA*. This method is alternating diverging and converging phases to give at the end a set of innovative ideas associated with an action plan to develop them. In this method the brainstorming is stimulated by an innovation catalyst. These catalysts are taken among the group of engineers. These engineers get a special training. More than 20 EFICA projects have been carried out very successfully, allowing many new ideas to emerge. These ideas are often implemented in cask designs and for some of them a patent application is underway.

The EFICA method was originally developed in AREVA fuel business unit is expanding in whole AREVA group with a lot of success. Description of EFICA can be seen in Fig. 2.

EXPLORE (Exploration, Impregnation):

Identify all the aspects of the problem without paying too much attention to the aspect which brought the problem to light; look at it from all angles without any prejudices or

FORMALIZE (Expression, Analysis, Formulation, Incubation):

Analyze and structure in detail the formulation of the problem and all its components and break it down into targeted areas of research; specify objectives, requirements and selection criteria

IDEAS (Creation, Ideas, Enrichment, Inspiration):

Put together and produce for each area of research a large number of solutions and

CONSTRUCTION (Prioritization, Choice, Valorisation):

Range, prioritize, select ideas following the previously defined selection criteria;

ACTION (Application, Organisation, Action Plan) :

For any solution, build an action plan, i.e. a program with detailed facts and

Every step is the result of a divergent and a convergent phase

FIG. 2. Description of EFICA method.

Idea Management Data Bank:

To keep all interesting ideas available, whether they come from groups or from individuals and sometimes from organised methods or spontaneously, an *idea management data bank* has been set up at TN international. The tool is very interactive, each idea is processed, by a committee and if added value is shown, it is implemented. Figure 3 shows the portal of idea management data bank IDHALL.

FIG. 3. IDHALL portal.

After implementation and use of these innovation tools, innovation in TN international has become a company-wide skill.

Some results of this innovation policy for the design and manufacturing equipment for storage are described:

5.1. High performance design solutions for sub-criticality [2]

The general trend towards high burnups for LWR fuels (typically 60 000 MWd/tHM for the EPR) leads to higher fissile contents, which means either higher U-235 enrichments (5%), or higher plutonium contents for MOX. Sub-criticality is guaranteed by the basket geometry and the material. Compactness is one of the major criteria for the design of evolutionary casks: as the mass and volume of packaging are generally bounded due to various interfaces (transport limitations or facility interfaces) there is a real challenge to design high capacity baskets. Therefore, a family of borated alloys has been developed for use in the baskets: borated stainless steel plates or Metal matrix composites, formed by casting, powder metal processes. All characteristics (composition, mechanical) have been studied, including the homogeneity of Boron content and the resistance to corrosion in borated water; they are satisfactory. Boralyn™ with 15% B_4C is an example of high performance materials for sub-criticality: it can be used for the structural resistance of the baskets. There is also the new Boron Metal Matrix Composite (MMC) material with an aluminium matrix and up to 25% B_4C.

5.2. Innovation in containment

A new type of fluorocarbon O-ring gaskets has been developed and qualified [3] to keep the guaranteed leak rate for a large range of temperatures. The long term behaviour at high temperature of EPDM O-ring gaskets has been studied with innovative methodology; it is now possible to obtain a curve of temperature versus time-limit for EPDM O-rings.

5.3. Mitigation of hydrogen risk.

 For the mitigation of hydrogen risk in the cavity of dry storage casks, a catalytic recombiner has been developed and qualified, with a sufficient capacity to stabilise the hydrogen concentration bellow its flammability limit [4]. Cooperation with French research institute IRCELYON has permitted to develop this solution.

5.4. Complete range of high performance neutron shielding materials [5–7]

TN has developed high performance neutron shielding materials (formulation and manufacturing methods) resisting to fire tests (self-extinguishing): TN Vyal B™, TN HYPOP™ and BORA™ for sub-criticality. These materials are adapted to different thermal environments. Depending of the temperature of use, the designer can choose the most adequate product.

FIG. 4. Neutron shielding (TN Vyal B^{TM}).

6. SOLUTIONS FOR THERMAL AND STRUCTURAL MANAGEMENT

For a given metallic containment vessel containing a given number of used fuels, the necessary thickness of neutron shielding material increases 20% when burnup of uranium fuel increases 15% and 50% if we change from uranium fuel to Mox fuel. Innovations have brought a better heat evacuation system to compensate the negative effect of thermal insulation of neutron shielding material (polymers are generally low heat-conductive materials): thermal conductors, fins, new aluminium heat exchangers, special surface treatments, and gap reduction (gap between cask inner wall/basket).

7. SPENT FUEL DRY STORAGE SYSTEMS

The most significant results of the innovation policy of TN international is the new generation of dry storage systems which include most of the above described innovations. These new dry storage systems TN® DUO, TN® NOVA will be presented in another paper this publication (See Garcia, J., Session 7).
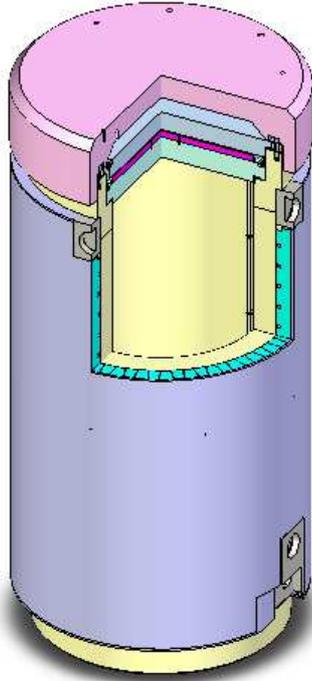
FIG. 5. TN® DUO dry storage.

8.   CONCLUSION

The role of innovation for the management of used fuel is to bring important benefits in term of performance, safety and public acceptance and at the same time to impulse a dynamic perspective for the whole nuclear industry. To show that we have solutions for the back end issue is beneficial for the whole nuclear activity.

With innovation, which is a long lasting process, the nuclear industry, and especially the back end is looking towards the long term and engaged in preparing a future with less CO_2 emissions. That is our commitment at AREVA TN international.

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ADVANCES IN BURNUP CREDIT CRITICALITY SAFETY ANALYSIS METHODS AND APPLICATIONS

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Abstract

An International Workshop on "Advances in Applications of Burnup Credit for Spent Fuel Storage, Transport, Reprocessing, and Disposition" organized by the Nuclear Safety Council of Spain (CSN) in cooperation with the International Atomic Energy Agency (IAEA) was held at Córdoba, Spain, on October 27–30, 2009. The objectives of this workshop were to identify the benefits that accrue from recent improvements of the burnup credit (BUC) analysis methodologies, to analyze the implications of applying improved BUC methodologies, focusing on both the safety-related and operational aspects, and to foster the exchange of international experience in licensing and implementation of BUC applications. In the paper on hand the attention is focused on the improvements of BUC analysis methodologies.

1.   INTRODUCTION

SNF management is a key issue for many IAEA member countries. Since the storage facilities in many countries reach their design capacities it becomes increasingly necessary to optimize the SNF storage designs in order to control fuel cycle costs. Because an adequate criticality safety margin has to be maintained under all conditions that have to be considered in the storage design criticality calculations the most economical way to optimize SNF storage designs is to model the SNF in the design criticality calculations as realistically as possible by using, in particular, credit for the burnup of the SNF. In fact, most of the storage designs used at present are based on the assumption that the fuel is just at the maximum reactivity point of its lifetime. For spent PWR fuel, in particular, this results in a significant overestimation of the fuel's reactivity and hence in a significant "over-designing" of the storage facilities.

Although economics is the primary factor in deciding to use BUC there are additional benefits contributing to public health and safety (e. g., less SNF transports due to higher transport capacities), resource conservation and environmental quality.

Application of BUC to SNF management systems for storage, transport, reprocessing, or disposition of SNF consists in implementation of three key steps:

(1)   Estimation of the SNF isotopic composition by means of depletion calculations, selection of BUC isotopes and validation of their concentrations calculated as functions of initial enrichment and burnup of the SNF;

(2)   Criticality analysis and evaluation of the loading criterion indicating the minimum burnup (or a related parameter) necessary for SNF with a specific initial enrichment to be loaded in the SNF management system of interest;

(3)   Quantification and verification of the fuel burnup (or a related parameter) before the fuel is loaded in the SNF management system.

The methodologies and procedures used in these three steps have been significantly improved in recent years. In the paper on hand the attention is focused on the improvements in the validation of the depletion calculations and the validation of the criticality calculations.

2.   VALIDATION OF DEPLETION CALCULATIONS

The isotopic inventory of SNF is predicted with the aid of depletion calculations considering the fuel characteristics, the fuel depletion conditions (reactor operation conditions) and the cooling time. The isotopes to be used in a BUC criticality analysis are selected on the basis of their reactivity worth and their nuclear and chemical stability. The change in reactivity due to burnup and cooling time can be adequately represented by a relatively small set of isotopes which meet the requirement of nuclear and chemical stability [1].

Due to possible biased errors in the nuclear data applied to the depletion calculations and due to algorithmic and numerical weaknesses in the employed depletion calculation code the calculated concentrations of the BUC isotopes, i. e. of the SNF isotopes to be used in the BUC criticality analysis, may be biased. To validate the depletion calculations and to eliminate the biases in the isotopic number densities, comparisons between predicted and measured isotopic concentrations are made. The measured isotopic concentrations are obtained from chemical assays of samples from irradiated fuel.

The availability and reliability of publicly available chemical assay data have been significantly improved in recent years. Under the auspices of the OECD NEA Data Bank Working Party on Nuclear Criticality Safety (WPNCS) an expert group on assay data has been formed three years ago. The objectives of this group include:

— Expanding the SFCOMPO experimental data base of SNF isotopic measurements [2];
— Making the data accessible through the SFCOMPO website;
— Sharing best practices on radiochemical analysis methods;
— Identifying input data and modeling requirements, and
— Evaluating uncertainties associated with the measurements and deficiencies in documented design and reactor operating history information.

The comparison of predicted to measured concentrations inevitably introduces:

— All the uncertainties and correlations in the measured concentrations arising from the applied assay methods and;
— All the uncertainties in the information about the depletion conditions required to predict the isotopic concentrations by means of re-calculating the irradiation history of the assayed fuel sample.

The uncertainties in the parameters characterizing the depletion conditions lead to uncertainties and correlations in the calculated concentrations.

From the statistics of comparisons between predicted and measured isotopic concentrations isotopic correction factors (ICFs) are derived. Due to the uncertainties and correlations in the predicted and measured concentrations the ICFs have uncertainties and correlations which have to be considered when the ICFs are applied to the isotopic number densities calculated for an application case, i. e. for the SNF to be used in a design analysis of a SNF management system of interest. The importance of considering the correlations between the ICFs has been demonstrated in [3].

As described in detail in [4] the most convenient way to consider all the uncertainties and correlations related to the ICFs is to use Bayesian Monte Carlo sampling of ICF values $ICF_{(s)}$

on the basis of the observed ICF values ICF_{obs} obtained from the comparisons of predicted to measured isotopic concentrations,

$$ICF_{(s)} \sim P(ICF \,|\, ICF_{obs}) = \int P(ICF \,|\, \Theta)\, P(\Theta \,|\, ICF_{obs})\, d\Theta \qquad (1)$$

where $P(ICF|\Theta)$ denotes the joint probability density distribution of the ICFs for the BUC isotopes. The parameter set Θ characterizing the density $P(ICF|\Theta)$ is unknown and is therefore treated as a random parameter set the probabity density of which is given, according to Bayes' theorem [5], by the so-called "posterior density"

$$P(\Theta \,|\, ICF_{obs}) \propto P(ICF_{obs} \,|\, \Theta)\, P(\Theta) \qquad (2)$$

which represents the knowledge about Θ gained from the observed data ICF_{obs}. $P(\Theta)$ in Eq. (2) represents the knowledge prior to the observation of the values ICF_{obs}. If there is no prior knowledge available, a so-called "non-informative-prior" is chosen for $P(\Theta)$ [6–7]. $P(ICF_{obs}|\Theta)$ in Eq. (2) represents the Likelihood function of the observations ICF_{obs} under Θ.

So, applying the n Monte Carlo samples $[ICF_{(s)}]_i$, $i = 1, \ldots, n$, to the set N of isotopic number densities calculated for the application case results in n bias-corrected sets $[N_{corr}]_i$, $i = 1, \ldots, n$, of isotopic number densities for the application case. Performing with each of these sets a criticality calculation gives, as illustrated in Figure 1, a distribution of the neutron multiplication factor k_{eff} which covers all the uncertainties related to the estimation of the ICFs.
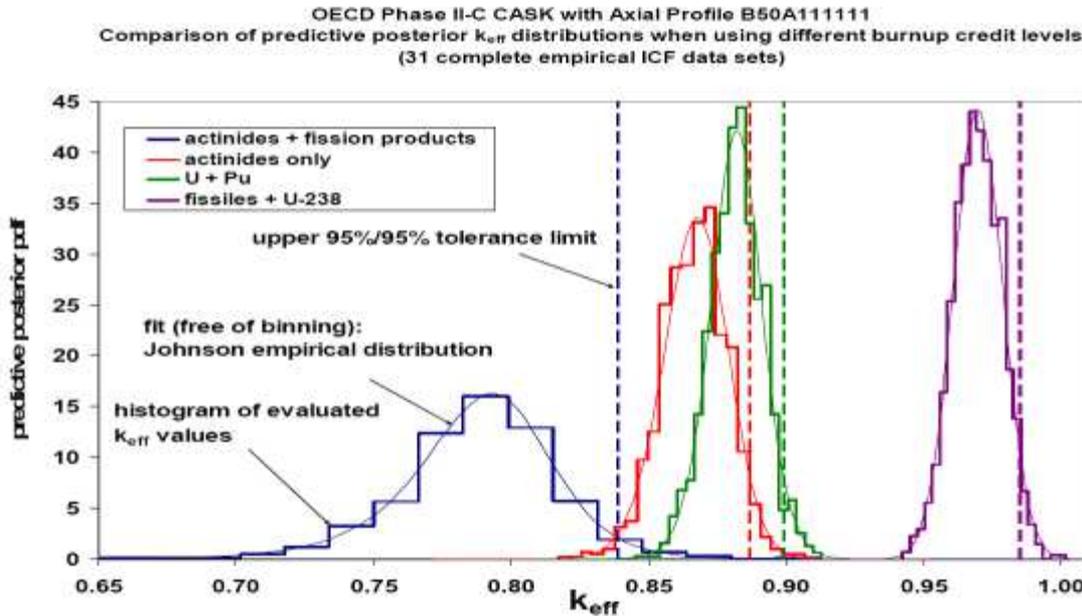
FIG. 1. Examples of k_{eff} distributions reflecting the uncertainties associated with the estimation of Isotopic Correction Factors ICFs and demonstrating the potential economical benefit of the actinide-plus-fission-product BUC level [3] (pdf:= probability density function; each distribution based on n = 1000 Monte Carlo samples $ICF_{(s)}$).

As described in detail in Ref. [4] the missing data problem illustrated in Fig. 2, i. e., the problem that the set of BUC isotopes is not always completely assayed in all the isotopic concentration measurements can be solved within the Bayesian Monte Carlo sampling framework without any loss of empirical information [7].

Up to now only the biased errors but not the uncertainties in the nuclear data, i. e. the variances and correlations of these data, have been considered. In fact, there is no need to consider these uncertainties in the evaluation of the ICFs because the isotopic biases to be corrected by applying the ICFs are those which are characteristic of the combined use of the given nuclear data library and the given depletion calculation code with respect to the neutron spectrum related to the application case, i. e., to the depletion conditions to which the SNF to be used in a design analysis of a SNF management system of interest has been exposed.

FIG. 2. Examples of the variations of the Isotopic Correction Factors (ICFs) obtained from different chemical assay measurements. The isotope ^{149}Sm has not been observed in all of the evaluated measurements, i.e. the ICF data set is incomplete with respect to the total number of evaluated measurements.

However, the uncertainties in the nuclear data have to be considered, in principle, in the calculation of the isotopic number densities N of the application case. To the knowledge of the author this has never been done up to now. However, studies on the reactivity impacts due to the uncertainties in the nuclear data employed for the depletion calculations are of fundamental interest in BUC criticality safety analysis. In the Bayesian framework such studies can be performed by drawing Monte Carlo samples $[N_{(s)}]_i$ from a distribution $p(N|\Psi)$ of the BUC-application-specific number densities N where Ψ is related to the variation of the nuclear data due to their variances and covariances. This variation can be simulated by drawing Monte Carlo samples on the nuclear data as described in section 4.

Use of the set $[N_{(s)}]_i$, i = 1, …, n, of uncorrected number densities in subsequent criticality calculations delivers a set of k_{eff} values the distribution of which reflects the uncertainty in k_{eff} due to the uncertainties in the nuclear data employed for the depletion calculations.

In a design criticality safety analysis one uses, of course, the bias-corrected set of number densities

$$[N_{corr}]_i = [ICF_{(s)}]_i \cdot [N_{(s)}]_i , \ i = 1, \ …, \ n; \tag{3}$$

and the distribution of the k_{eff} values then obtained reflects both:

— the uncertainties in the nuclear data used in the depletion analysis of the SNF of the application case and

— the uncertainties in the validation of the depletion analysis procedure applied to perform the depletion calculation for this SNF.

3.   VALIDATION OF CRITICALITY CALCULATIONS

Possible biased errors in the applied nuclear data used for the criticality analysis of a SNF management system of interest as well as algorithmic und numerical weaknesses of the employed criticality calculation code may result in a non-zero bias Δk_B in the calculated k_{eff} value obtained for the system of interest. This bias Δk_B is characteristic of the employed nuclear data library and the used criticality calculation code with respect to the application case, i. e. the SNF management system of interest. To be able to determine the bias Δk_B which applies to the application case A under the given nuclear data library and the given calculation code, it is necessary to analyze benchmark configurations which are similar to the application case with respect to neutron physics properties and hence representative for this case. The representativeness of a benchmark B with respect to A can be measured by the correlation between the neutron multiplication factors k_B of the benchmark and k_A of the application case: The higher the correlation is the more representative is B for A with respect to the neutron physics properties. The correlation $corr(k_B, k_A)$ between k_B and k_A is given by the covariance $cov(k_B, k_A)$ and the variances $\sigma^2(k_B)$ and $\sigma^2(k_A)$ of k_B and k_A, respectively, [5]:

$$corr(k_B, k_A) = \frac{cov(k_B, k_A)}{\sqrt{\sigma^2(k_B) \cdot \sigma^2(k_A)}} .$$ (4)

$\sigma^2(k_B)$ and $\sigma^2(k_A)$ are the diagonal elements of the covariance matrix

$$\mathbf{V(k)} = \begin{pmatrix} \sigma^2(k_B) & cov(k_B, k_A) \\ cov(k_A, k_B) & \sigma^2(k_A) \end{pmatrix}$$ (5)

of the vector $k = (k_B, k_A)^T$, and $cov(k_B, k_A)$ gives, due to $cov(k_B, k_A) = cov(k_A, k_B)$, the off-diagonal elements of V(k). V(k) is given by (cf. Ref. [5])

$$\mathbf{V(k)} = E\left[\left(\mathbf{k}(\xi) - E[\mathbf{k}(\xi)] \right) \left(\mathbf{k}(\xi) - E[\mathbf{k}(\xi)] \right)^T \right]$$ (6)

$E[\dots] :=$ expectation (expectation operator); e. g., the expectation of $k(\xi)$ as a function of the nuclear data ξ is given by the vector

$$E[\mathbf{k}(\xi)] = \begin{pmatrix} \int_{\Omega(\xi)} k_B(\xi)\varphi(\xi)d\xi \\ \int_{\Omega(\xi)} k_A(\xi)\varphi(\xi)d\xi \end{pmatrix}$$ (7)

where $\varphi(\xi)$ is the joint probability density of the nuclear data ξ, and $\Omega(\xi)$ denotes the total space of ξ.

In first-order perturbation theory [8] it is assumed that $k(\xi)$ can be expanded about the expectation values $E[\xi]$ of the nuclear data in a Taylor series neglecting terms of order greater than 1:

$$\mathbf{k}(\xi) \approx \mathbf{k}(E[\xi]) + (\xi - E[\xi]) \left. \frac{\partial \mathbf{k}(\xi)}{\partial \xi} \right|_{\xi = E[\xi]} .$$ (8)

The expectation vector Eq. (7) thus becomes

$$E[\mathbf{k}(\xi)] \approx \mathbf{k}(E[\xi]) , \tag{9}$$

and the covariance matrix Eq. (6) becomes, therefore,

$$\mathbf{V}(\mathbf{k}) \approx \mathbf{Z}\,\mathbf{V}(\xi)\,\mathbf{Z}^{\mathrm{T}} \tag{10}$$

with the matrix

$$\mathbf{Z} = \frac{\partial \mathbf{k}(\xi)}{\partial \xi}\bigg|_{\xi=E[\xi]} \tag{11}$$

and the covariance matrix of the nuclear data

$$\mathbf{V}(\xi) = E\Big[\big(\xi - E[\xi]\big)\big(\xi - E[\xi]\big)^{\mathrm{T}}\Big]. \tag{12}$$

So, in first-order perturbation theory $\mathrm{cov}(k_B, k_A)$ becomes

$$V_{BA} \equiv \mathrm{cov}(k_B, k_A) \approx \sum_{\nu,\mu} \frac{\partial k_B}{\partial \xi_\nu}\, \mathrm{cov}(\xi_\nu, \xi_\mu)\, \frac{\partial k_A}{\partial \xi_\mu} \tag{13}$$

where $\mathrm{cov}(\xi_\nu, \xi_\mu)$ denotes the elements of $V(\xi)$.

Usually sensitivity coefficients

$$S_{c\lambda} \equiv \frac{1}{k_c}\left(\frac{\partial k_c}{\partial \xi_\lambda}\,\xi_\lambda\right) \tag{14}$$

are introduced [8–10], so that one gets from Eq. (13):

$$C_{BA} \equiv \frac{\mathrm{cov}(k_B, k_A)}{k_B \cdot k_A} \approx \sum_{\nu,\mu} S_{B\nu}\, \frac{\mathrm{cov}(\xi_\nu, \xi_\mu)}{\xi_\nu \cdot \xi_\mu}\, S_{A\mu}. \tag{15}$$

The correlation coefficient Eq. (4) thus becomes

$$c_k \equiv \mathrm{corr}(k_B, k_A) = \frac{C_{BA}}{\sqrt{C_{BB} \cdot C_{AA}}}. \tag{16}$$

So, the correlation coefficient Eq. (4) is related to sensitivity coefficients Eq. (14) indicating the sensitivities of the neutron multiplication factors k_B and k_A to variations in the nuclear data. The more similar the sensitivity coefficients $S_{B\lambda}$ and $S_{A\lambda}$ as functions of the neutron energy are the higher is the representativeness of the benchmark B with respect to the application case A. According to [11] a benchmark B is regarded as representative with respect to an application case A if the c_k value given by Eq. (16) is not less than 0.9.

Let us assume that we have found N_B representative benchmarks, and let k now be the vector $k = (k_1, k_2, \ldots)^{\mathrm{T}}$ of the k_{eff} results k_i, $i = 1, \ldots, N_B$, obtained for these benchmarks. So, the vector Eq. (8) is N_B–dimensional now, and the covariance matrix Eq. (10) becomes a symmetric $N_B \times N_B$ matrix with elements obtained by replacing B and A in Eq. (13) with i and j, $i = 1, \ldots, N_B$ and $j = 1, \ldots, N_B$. Using

$$\boldsymbol{\delta k} \equiv \mathbf{k}(\xi) - \mathbf{k}(E[\xi]) \equiv \mathbf{k}(\xi) - \mathbf{k} \quad \text{and} \quad \boldsymbol{\delta \xi} \equiv \xi - E[\xi] \tag{17}$$

in Eq. (8) it follows from equations (8) and (14)

$$\frac{\boldsymbol{\delta k}}{\mathbf{k}} = \mathbf{S} \frac{\boldsymbol{\delta \xi}}{\xi} . \tag{18}$$

In References [8] and [9] the k_{eff} results k_i, $i = 1, \ldots, N_B$, obtained with a given nuclear data library are interpreted as experimental information which increases the information on the nuclear data ξ. Accordingly, Bayes theorem

$$p(\xi \,|\, \mathbf{m}) \propto L(\mathbf{m} \,|\, \xi)\,\pi(\xi) \tag{19}$$

is used in Ref. [9] in combination with nuclear data adjustment achieved by means of the Maximum Likelihood procedure [5].

In Eq. (19) m denotes the set of observed k_{eff} results k_i, $i = 1, \ldots, N_B$. $p(\xi)$ is the probability density of ξ prior to the experimental information contained in the data m, and $L(m|\xi)$ is the Likelihood of the data m under ξ. Due to the first-order evaluation Eq. (8) and hence to the linearity of the transformation Eq. (18) the Maximum-Likelihood estimator provides best estimates [5]. Assuming that the prior density $p(\xi)$ is given by a Normal distribution and using the functional form of a normal distribution for the Likelihood $L(m|\xi)$ one gets

$$\frac{\boldsymbol{\delta \xi}}{\xi} = \mathbf{R}(\xi)\,\mathbf{S}^{\mathrm{T}}\,(\mathbf{C_{kk}} + \mathbf{C_{mm}})^{-1}\,\frac{\boldsymbol{\delta k}}{\mathbf{k}} \tag{20}$$

with the relative covariance matrix $R(\xi)$ having the elements $\mathrm{cov}(\xi_\nu, \xi_\mu)/(\xi_\nu \cdot \xi_\mu)$, cf. Eq. (15), and with

$$\frac{\boldsymbol{\delta k}}{\mathbf{k}} = \frac{\mathbf{k} - \mathbf{m}}{\mathbf{k}} \tag{21}$$

where k represents the benchmark k_{eff} values and δk represents the observed deviations of the results m from the benchmark k_{eff} values. The covariance matrix of these deviations is given by the sum $C_{kk} + C_{mm}$ in Eq. (20); C_{kk} and C_{mm} are of the form of Eq. (15).

The posterior covariance matrx, i.e. the covariance matrix of the adjusted nuclear data is given by

$$\mathbf{R}(\xi)^{\mathrm{posterior}} = \mathbf{R}(\xi) - \mathbf{R}(\xi)\,\mathbf{S}^{\mathrm{T}}\,(\mathbf{C_{kk}} + \mathbf{C_{mm}})^{-1}\,\mathbf{S}\,\mathbf{R}(\xi) \tag{22}$$

showing that the uncertainty in ξ is reduced due to the additional experimental information contained in m.

The adjustment solution Eq. (20) can now be used in Eq. (18) to estimate the computational bias of the application case. Due to the linearity of Eq. (18) one simply gets

$$\frac{\delta k_A}{k_A} = \mathbf{S}_A \frac{\boldsymbol{\delta \xi}}{\xi} \tag{23}$$

for the computational bias with a variance of

$$\sigma^2\left(\frac{\delta k_A}{k_A}\right) = \mathbf{S}_A \, \mathbf{R}(\xi)^{\text{posterior}} \, \mathbf{S}_A^T .\tag{24}$$

In Ref. [8] a different way is taken by using the Generalized Linear Least Squares (GLLS) procedure minimizing the quadratic form

$$Q^2(\psi, \varphi) = (\psi, \varphi)^T \begin{pmatrix} \mathbf{C_{mm}} & \mathbf{0} \\ \mathbf{0} & \mathbf{R}(\xi) \end{pmatrix} (\psi, \varphi)\tag{25}$$

with

$$\psi \equiv \frac{\mathbf{k}(\xi^{\text{adjusted}}) - \mathbf{m}}{\mathbf{k}(\xi)} \ \text{ and } \ \varphi \equiv \frac{\delta\xi}{\xi} .\tag{26}$$

However, due to the linearity of Eq. (18) the GLLS procedure leads to the same expressions Eq. (20) for $\varphi = \delta\xi/\xi$ and Eq. (22) for $R(\xi)^{\text{posterior}}$ as were obtained by means of the Likelihood method used in Ref. [9].

The question naturally arises if the adjusted nuclear data have a reasonable meaning with respect to physics. If the above described adjustment procedure were really capable to improve the nuclear data due to the increase of experimental information then one should not hesitate to use the adjusted data in the next criticality calculation coming along. However, nobody is doing that. In fact, it has been observed that the adjustment procedure can lead to data values which are incompatible with physics. For this reason a so-called "χ^2-filter" has been introduced in the GLLS procedure used in [12] in order to maintain the outcomes of the GLLS adjustment procedure consistent with physics. The application of such a filter can result in the exclusion of benchmarks from the GLLS adjustment procedure even though these benchmarks are representative for the application case. The exclusion of representative benchmarks is not understandable, since the decision criterion for excluding these benchmarks is a purely statistical one, whereas the representativeness of these benchmarks is based on physics properties. It is a fundamental principle that benchmarks can safely be discarded only on physical arguments. So, the only possibility to exclude benchmarks, which have been identified as representative, is to demonstrate that one cannot rely on the performance, the evaluation and the results of these benchmarks because of physics reasons.

So, in conclusion, whereas the characterization of the representativeness of a benchmark with respect to an application case by means of the correlation coefficient Eq. (16) is in compliance with that what is to be expected due to physics, the combination of the first-order perturbation evaluation with the described nuclear data adjustment procedure seems to be a typical case of overfitting leading to an artificial biasing of nuclear data which only serves as a means to determine a computational bias for the application case, but which has no further meaning.

The nuclear data are basic data, and so they have to be treated as basic data. As stated at the beginning of section 3, the bias Δk_B in the k_{eff} value of the application case is characteristic of the combined use of the given nuclear data library and the given criticality calculation code with respect to the application case. So, there is no need for considering the uncertainties in the nuclear data ξ when calculating the set of k_{eff} values k_i, $i = 1, \ldots, N_B$, for the N_B representative benchmarks.

The uncertainties in the material and design data x_B of the benchmarks have to be taken into account, since they lead to uncertainties in the observed k_{eff} values k_i and hence to uncertainties in the deviations $(\Delta k_B)_i$, $i = 1, \ldots, N_B$, of the k_i values from the respective benchmark k_{eff} values.

Since the correlation coefficient Eq. (16) is less than 1 for each of the N_B benchmarks, i. e., since the benchmark configurations $i = 1, \ldots, N_B$ are similar but not equal to the application case A with respect to neutron physics properties, a model is required which is capable to derive the bias Δk_B related to the system A from the results $(\Delta k_B)_i$, $i = 1, \ldots, N_B$, taking into account the uncertainties in $(\Delta k_B)_i$ due to the uncertainties in the data x_B. For this purpose a set z of explanatory variables is required which characterizes the benchmarks $i = 1, \ldots, N_B$ as well as the system A: $z = z_i$ and $z = z_A$, respectively. The required model therefore consists in a trending analysis procedure $\Delta k_B = \Delta k_B (z)$ which has the capability to demonstrate the significance of the chosen set z of explanatory variables and to conclude then from the points z_i in the z-space to the point z_A in this space. This trending analysis is performed by means of a Bayesian Monte Carlo regression analysis procedure described in detail in Ref. [13]. Examples for results obtained by means of this procedure are presented in Ref. [14].

4. CONSIDERATION OF NUCLEAR DATA UNCERTAINTIES

The procedure used for estimating the neutron multiplication factor k_A of a SNF management system of interest must include the consideration of the uncertainties in the nuclear data ξ. As already mentioned in section 2, variation of the nuclear data due to their variances and covariances can be simulated by drawing Monte Carlo samples on the nuclear data. This can be performed as follows [4]:

Let $\zeta(E_n)$ be the mean vector of the applied nuclear basis data $\xi_{BD} = \xi(E_n)$ ($E_n := $ neutron energy), and let $\Sigma(\xi_{BD})$ be the related covariance matrix of these data. Monte Carlo samples $\xi(E_n)_{(s)}$ can be drawn from the Normal distribution with expectation vector $\zeta(E_n)$ and covariance matrix $\Sigma(\xi(E_n))$. The resulting samples $\xi(E_n)_{(s)}$ are used for generating continuous cross-section libraries $L_{(s)}$ by means of a basis-data evaluation code. With each of these libraries a depletion calculation can be performed (cf. section 2) or a calculation of the neutron multiplication factor k_A can be carried out. The variations in the results, i. e., the variation in the obtained isotopic number densities and the variation in the obtained k_A values, respectively, reflect the uncertainties in the nuclear data.

The criticality safety analysis group PEEA8-G of AREVA NP GmbH, Germany, has already installed the Monte Carlo sampling procedure on nuclear data for criticality calculations with the MCNP code [15]. For depletion calculations the procedure will be installed in the near future.
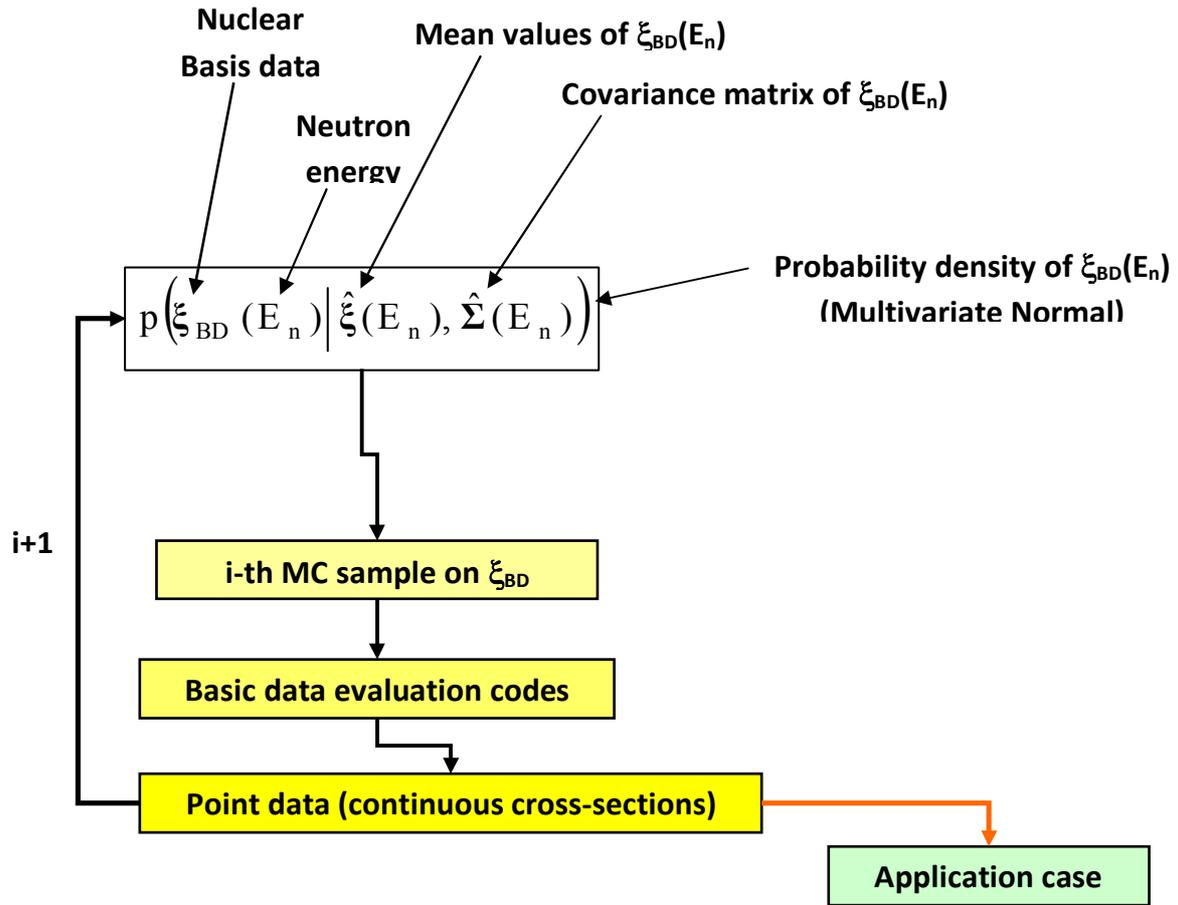
FIG. 3. Monte Carlo (MC) sampling on nuclear data.

5. CONCLUSIONS

It has been shown that calculation procedures are under develoment and partially already installed which make it possible by means of Bayesian Monte Carlo procedures to consider all the uncertainties in

— The depletion calculations performed for the application case;
— The chemical assay data evaluated for validating the depletion calculations;
— The critical experiments evaluated for estimating the bias of the criticality calculation code applied, and
— The application case.

Application of these calculation procedures to BUC criticality safety analysis does not only result in significant improvements of methodologies used in BUC analysis, but leads to a complete BUC calculation route.

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EXPERIMENTAL METHODS AND MONTE CARLO SIMULATIONS FOR BURNUP ASSESSMENT OF SPENT FUEL ELEMENTS

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Abstract

SCK•CEN is currently investigating spent fuel experimental methods and devices to assess their capability for burn up determination. This assessment is performed from an operator's point of view, meaning that certain information like initial enrichment of the fuel element, the irradiation history and material composition can be taken into account for the assessment. The aim of this investigation is to make a choice for a few methods that will be explored further in detail. Although these methods are meant for burn up determination of the spent fuel, the results of this investigation can also give valuable information about the applicability of these methods for safeguards purposes. In a general framework of spent fuel investigations, two methods are being considered: the so-called Fork detector [1], and the Self-Interrogation Neutron Resonance Densitometry (SINRD) [2]. In order to understand and characterize the performances of these techniques Monte Carlo simulations were carried out using the MCNPX code [3]. Preliminary results for the Fork detector are presented in this paper, considering its use in wet and dry conditions.

1. INTRODUCTION

The management of irradiated nuclear fuel consists in either going for reprocessing or for direct disposal in a safe habitat that depends on the availability in the State, and actually considers clay, bedrock, salt. Although direct disposal in clay is still considered a viable option, only the reprocessing route has been implemented so far in Belgium.

The transport of spent fuel to France and the reprocessing of the fuel are submitted to some safety related restrictions. To avoid a criticality accident during the chopping of the assembly a minimum burnup at the fuel extremities is needed. This minimum burnup is 11 GWd/tHM in the extreme 50 cm, for fuel with an initial enrichment of 4.20% and higher.

An experimental verification of the burnup of a spent fuel assembly to guarantee a minimum burnup in the 50 cm extremities of the assembly with the lowest irradiation is therefore important. A lot of effort has been devoted in the past to investigate methods to determine the burnup of spent fuel assemblies in a non-destructive way.

One of the methods applied is by isotopic correlation [4, 5], based on the measurement of a high resolution gamma spectrum and the determination of the ^{137}Cs, ^{134}Cs, ^{154}Eu content, where either the isotope activities or ratios of these have specific correlations to the burnup of the considered irradiated fuel assembly.

Another method consists in the use of the Fork detector [1]. The choice to use the Fork detector for spent fuel burnup measurement was made following a request from the Belgian utilities to provide an instrument capable to assess the burnup of irradiated fuel assemblies. The equipment was used to provide the information requested by AREVA and the French Authorities upon shipments of spent fuel from Belgian Nuclear Power Plants (BNPP) to the reprocessing plant of La Hague.

Performance characteristics of the Fork detector are elaborated in this paper, based on Monte Carlo simulations.

2. FORK DETECTOR

2.1. General description

The Fork detector is a measurement device for burnup determination of spent fuel assemblies under water, where the spent fuel assembly is placed between the Fork arms. The stainless steel housing has two arms, each containing two radiation detectors (sensors), one to detect neutrons (fission chamber) and one to detect gamma-rays (ionization chamber), contained in a polyethylene housing which is surrounded by a layer of cadmium. For safeguards inspections purposes a Fork detector that differs slightly from device described is used.

The stainless steel housing is mounted on a stainless steel support, made compatible with the cell dimensions of the storage racks: the support is installed in one cell while the housing is adjusted over the adjacent cell, in which the fuel can be lowered for the axial scan. The housing (with the sensors) is adjusted over the fuel for optimal neutron and gamma signal. The neutron and gamma ray signals are correlated with the burnup, as explained in [4–6].

The combined housing and support are installed on the rack by using a long tool at every measurement campaign. The support is supposed to remain fixed to the Fork body, while a long tool is made installation specific, and removed during the measurement. The Fork detector and its use are shown in Fig. 1.
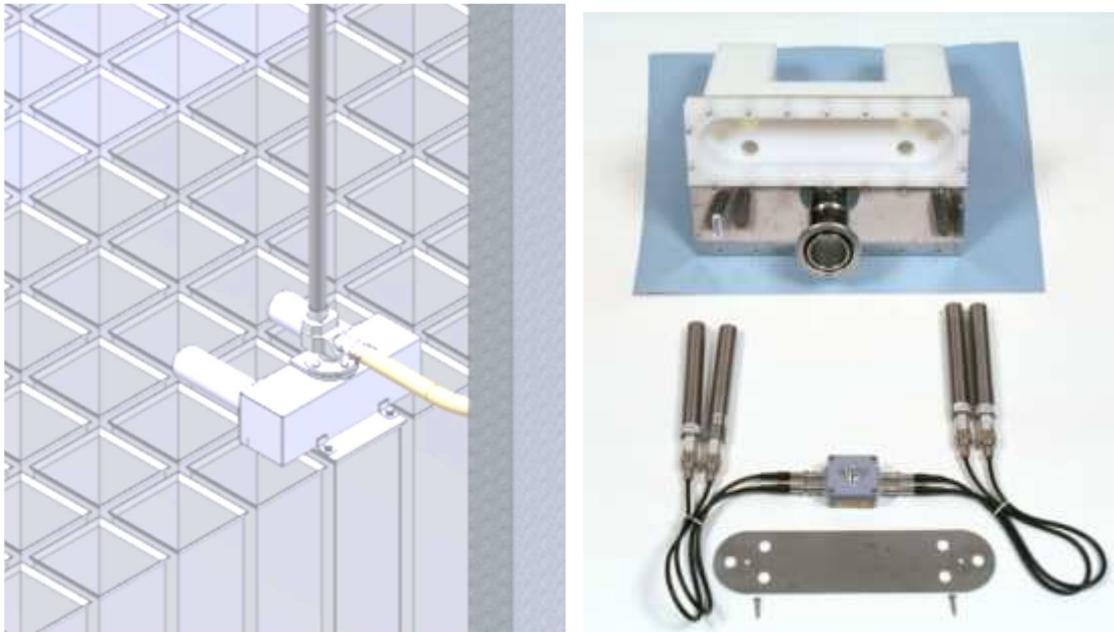
FIG. 1. Use and components of the Fork detector.

The link between detector and electronics is assured by cables that pass through a silicon flexible tube that keeps the system water tight. The cables end in an interim transfer box, from which the connection can be made to the electronics rack, for signal handling and processing. The electronic part contains all components required for the treatment of signals. The operation of the system is managed by a personal computer, which is connected to the

electronic components and provides data processing, but does not affect the movement of spent fuel.

2.2. Monte Carlo simulations

In order to better understand the characteristics of the Fork detector and determine if and how such a system could be improved, a set of Monte Carlo simulations was carried out. First a careful modelling of the detector and fuel element was done based on the design information available.

The geometry and composition of the Fork detector developed by SCK•CEN and in use at the BNPP was carefully modelled with MCNPX. It consists of a stainless steel body on which two arms are mounted. Each arm consists of a polyethylene part with two cavities where a fission chamber and an ionization chamber are placed. The polyethylene is wrapped by a 1 mm thick Cd sheet and is housed in a stainless steel cover.

The fuel element modelled was a 17×17 PWR with 264 fuel pins and 25 water channels. The isotopic composition of the fuel was obtained from Origen-ARP [7] calculation. In this calculation a LEU fuel with an initial enrichment of 4% was considered. A burnup of 44 GWd/tU spread over 4 irradiation cycles of 330 days followed each by 30 days decay, a cooling time of 1000 days after the end of irradiation was considered. The resulting isotopic composition was kept constant during the simulations. The total neutron emission is about 6.6×10^8 n/(s × tU) where more than 97% is due to decay of ^{244}Cm. The gamma-ray spectrum is dominated by the ^{137}Cs, emitting 4.0×10^{15} photons/(s × tU) with an energy of 662 keV, and ^{134}Cs, emitting 2.8×10^{15} photons/s × tU with an energy of 605 keV and 2.6×10^{15} photons/s × tU with an energy of 796 keV. Considering a uranium mass of about 610 kg for the fuel element considered, the resulting neutron emission is 4×10^8 n/s.

The geometry of the detector and the fuel element as modelled in MCNPX [3] is shown in Fig. 2. The distance between the fuel element and the Fork detector arms is 1.49 cm and the distance between the fuel element and the detector body is 4.34 cm. The Fork arm has a diameter of 8.9 cm and a length of 25.4 cm.

The neutron source was supposed to be uniformly distributed throughout the pins. The energy of the emitted neutrons followed the shape of ^{244}Cm fission spectrum. For gamma-ray the 661.7 keV from ^{137}Cs was used.

The Fork detector was studied in wet conditions, submerged in borated water, with an amount of 2270 ppm w/o boron. The quantities of interest are the gamma-ray induced ionization current and the expected neutron counts. These quantities are described as the so-called "tally" in MCNP terms. The current was obtained with an energy deposition F6 tally on the ionization chamber active area and neutron counts with a flux F4 tally in the fission chamber active layer multiplied by the ^{235}U fission cross section and the layer thickness. These quantities (tallies) are representative of the measured signals, being the current and the neutron counts respectively.

It is useful to determine which parts of the source (e.g. the fuel element) are contributing the detected signal. This kind of information is given by the so-called "importance function". The importance function can be split into a radial and axial component. The radial component provides information about which of the pins are contributing to the quantity of interest, while the axial component is indicative on the detector sensitivity to particles emitted along the vertical direction (Z axis in Fig. 2). In ideal measurement conditions, the radial importance

function would be uniform throughout the plane XY. If an axial scan of the fuel element is made, the axial importance function is peaked with a width that limits the sensitive zone to an area around the detector axial position. The quantitative determination of the radial and axial importance was one of the goals of the simulations.
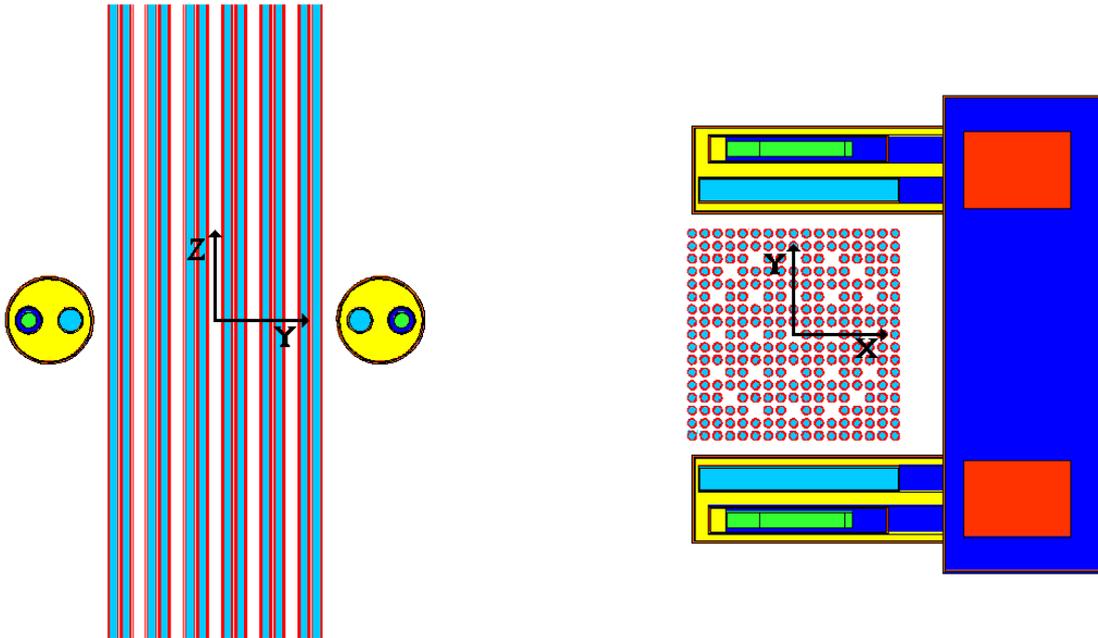
FIG. 2. The geometry as modelled in the MCNPX input files. The cross sections in the X=0 (left) and Z=0 (right) planes are shown. The origin is set in the middle of the fuel element.

The importance functions for the neutron counts were studied in different configurations. They were obtained with the so-called "mesh-based weight window" variance reduction technique [3] in order to limit the computation time. The number of simulated source particle histories was 4×10^7.

Similar simulations were carried out to verify the feasibility to use the Fork detector in dry conditions. In these simulations the water was replaced by air. Spent fuel measurements with the Fork in air could be of interest when dry storage is applied.

Additional simulations aimed to determine the impact of the boron amount on the expected neutron counts and the impact of the detector displacement on the expected counts were also carried out.

3.   RESULTS

This paper reports the results obtained for neutrons. Simulations for gamma-rays are on-going together with simulations for the SINRD method.

3.1.   Radial importance function

In case of wet conditions with 2270 ppm w/o boron, the obtained radial importance function for neutrons is shown in Fig. 3. The detector arms are placed close to rows 1 and 17. Figure 3 reveals that the central pins of these rows contribute mostly to the neutron counts in the fission chamber and their contribution is set to unity. The importance function exhibits the shape of a saddle with the contribution of the pins slowly decreasing as their distance from the sensors increases.

The radial importance function was also determined in wet conditions without boron in the water. For an easy comparison of the results, the radial importance function was projected along the row number. In this projection the importance data have been summed and divided by the effective number of pins in each row, in this way accounting for the fact that there are water channels from which no source neutrons originate. The data obtained were then normalized to their maximum value. The obtained projections are shown in Fig. 4, indicating that the highest sensitivity to inner rows of pins is achieved in the case without Cd and without boron.
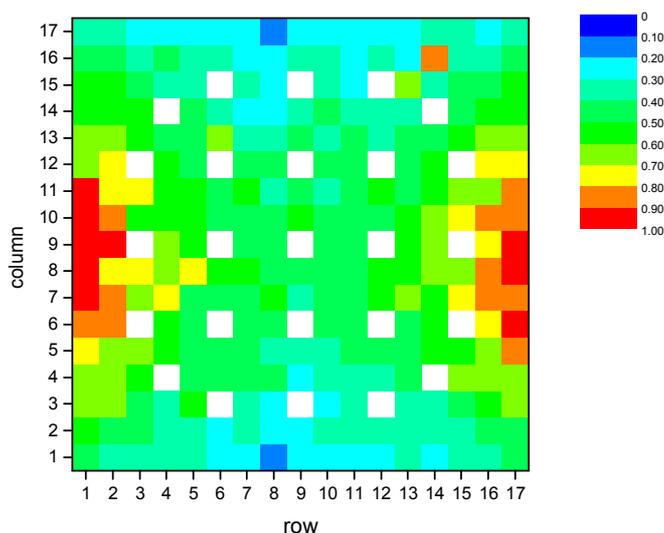
FIG. 3. The radial importance function for the neutron counts in both fission chambers with Cd sleeve and 2270 ppm w/o boron.
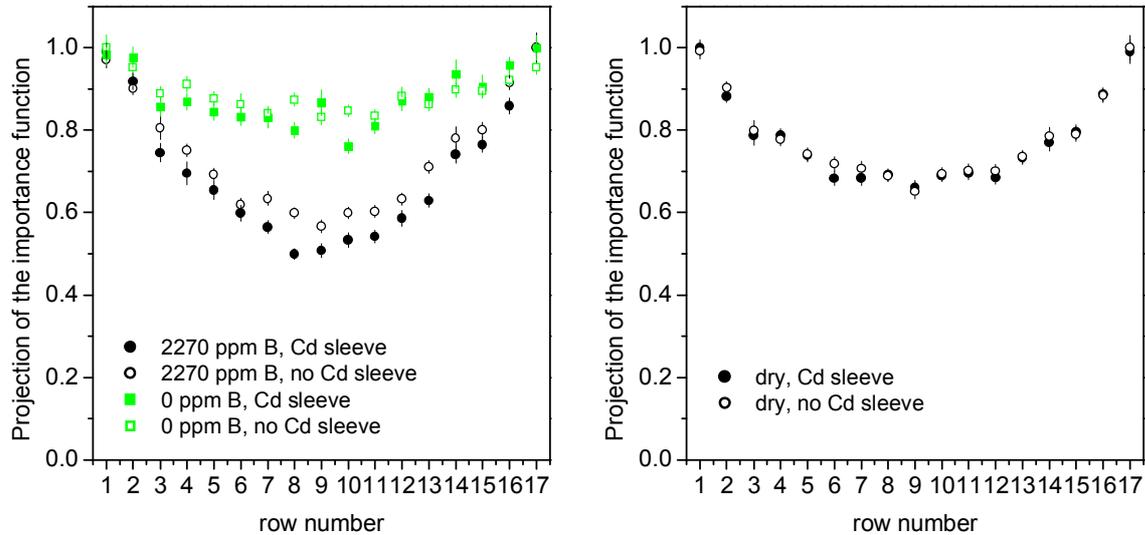
FIG. 4. The radial importance function projected along the row number for the neutron counts. The data with and without boron and with and without Cd sleeve are shown for wet (left) and dry (right) conditions.

The results of simulations of the radial importance function in dry conditions are also shown in Fig. 4 and reveal that the sensitivity to neutrons emitted from the inner rows of pins is about 10% higher, when compared to one obtained in wet conditions with 2270 ppm w/o boron.

Table 1 compares the total neutron fluence and the expected counts in the fission chambers, as tallied by MCNPX, in "dry" and "wet" conditions, with and without boron, with and without Cd layer around the detector arms. The results were normalized to the ones obtained in wet conditions without boron and Cd layer. The total neutron fluence is the neutron fluence on the whole neutron energy range.

TABLE 1. CALCULATED NEUTRON COUNTS AND TOTAL NEUTRON FLUENCE IN THE FISSION CHAMBERS IN DIFFERENT CONFIGURATIONS. THE VALUES ARE GIVEN RELATIVE TO THE ONES OBTAINED IN WET CONDITIONS WITHOUT BORON AND CD LAYER

Quantity	wet 2270 ppm w/o boron		wet no boron		dry	
	Cd	no Cd	Cd	no Cd	Cd	no Cd
Neutron Counts (counts per source neutron)	0.18	0.35	0.33	1.00	0.11	0.12
Total Neutron Fluence (n/cm^2 per source neutron)	0.49	0.52	0.90	1.00	0.71	0.71

Table 1 reveals that, in wet conditions:

- The Cd layer strongly affects the expected counts. This is due to the fact that the neutrons thermalized in the water do not reach the detector as they are absorbed by the Cd.;

- While the total neutron fluence is reduced by about 5–10% we observe a reduction of at least 50% in the counts. This phenomenon can be explained considering that the detector counts mainly come from the thermal neutron component due to the 1/v shape of the neutron cross section;

- The presence of boron in the water has an impact which is different from the effect of the Cd sleeve. The boron presence affects significantly both the total neutron fluence and the total neutron counts.

In dry conditions, one can conclude that:

- The Cd layer has negligible impact;
- Compared to wet conditions, lower counts are expected. In dry conditions the moderation is due only to the polyethylene in the detector arm. In addition there is no water acting as a moderator or reflector of neutrons;
- Higher total neutron fluence is obtained. The corresponding lower expected counts are due to the fact that the share of thermal neutrons in dry conditions is lower than in wet conditions.

3.2. Axial importance function

The axial importance function was determined for both the wet and dry configuration. The fuel was cut in slices along the Z direction and the contribution of each slice to the expected neutron counts in the fission chambers was scored. In Fig. 5 the importance function obtained in wet conditions and 2270 ppm w/o boron is shown. The data obtained were then normalized to their maximum value which was obtained for the slice in the plane of the detectors. The importance profile can be well fitted with a Gaussian as shown in Fig. 5. The FWHM (Full Width at Half Maximum) is 15.6±0.3 cm. The FWHM gives an indication of the sensitivity of the detector along the axial direction. About 90% of the expected neutron counts is due to neutrons emitted within 12 cm from the detector axial position.
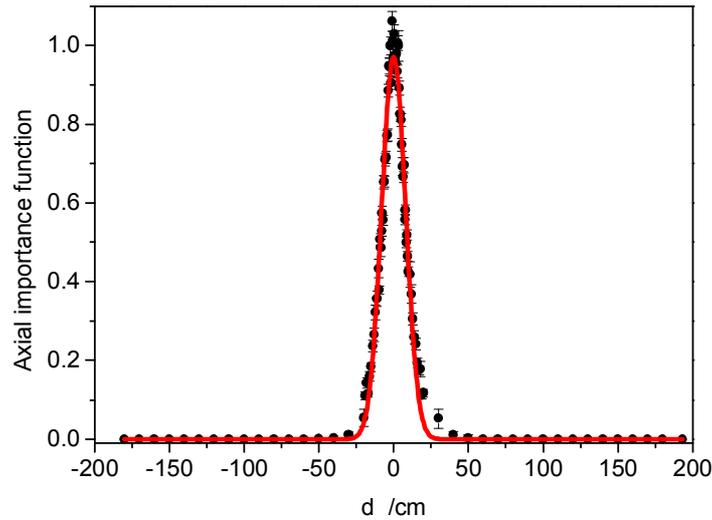
FIG. 5. The axial neutron importance function for neutrons in wet conditions.

In dry conditions, the shape could be fitted with a Gaussian curve with a FWHM of 21.5±0.3 cm. This value is 6 cm higher than the value obtained in wet conditions. This increase can be explained considering the fact that the borated water acts as a sort of neutron shield in "wet" conditions. In "dry" conditions there is practically no shielding and neutrons originating from a further distance can be detected. In "wet" conditions without boron a FWHM of 19.5±0.4 cm was obtained.

The impact of the Cd sleeve on the axial importance function is significant only in the configuration "wet" without boron, when a FWHM of 21.5±0.4 cm was obtained for the configuration without Cd sleeve.

3.3. Additional simulations

Additional sets of simulations were carried out to investigate the neutron counts as a function of the boron amount, with and without Cd sheet. The obtained counts as a function of the boron content for the configuration with and without Cd are shown in Fig. 6. One can conclude that a Fork detector without Cd sheet is more sensitive to the boron content and its variations. These results are consistent with the experimental results given in [8].

Simulations to determine the sensitivity of the neutron counts to the relative position Fork detector-fuel were also carried out. Displacement between 0 cm and 4 cm in the X direction and between 0 cm and 1.4 cm in the Y direction were considered. The XY directions are shown in Fig. 2.
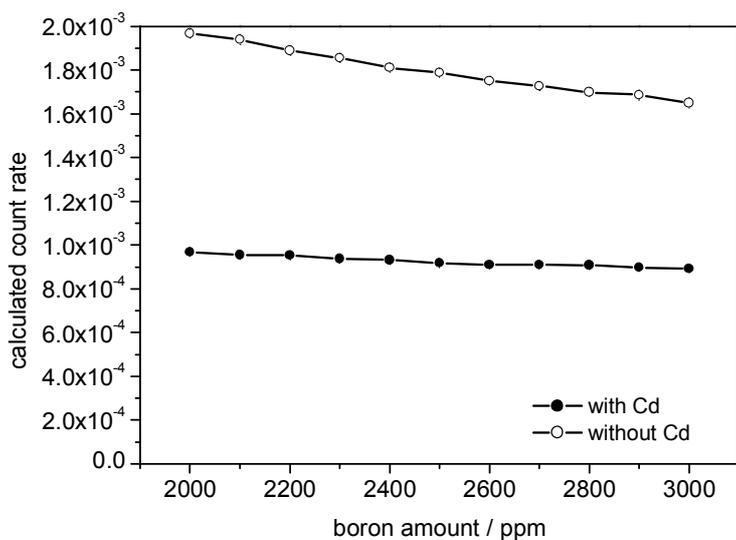
FIG. 6. Sensitivity of the expected neutron counts to the boron content with and without Cd sleeve.

The results in wet conditions and 2270 ppm w/o boron are shown in Fig. 7. Averaging the response of both detectors allows limiting the spatial dependence of the detector response. These results reveal that the detector is almost not sensitive to variation of the relative position along the Y direction. Variation along the X direction affects the detector response. The detector response is sensitive to positive variation along the X direction (the fuel element becomes closer to the Fork Box). However, if the distance between the fuel element and the Fork Box increases (negative values of the displacement) the expected counts change more rapidly. An accurate positioning of the fuel along the X direction is important for the reproducibility of the results.

The impact of the relative position detector-fuel on the expected neutron counts was investigated also in case for dry conditions. The results are shown in Fig. 8. The sensitivity to the displacement along X is slightly lower in dry conditions than in wet conditions.

Anyhow, performing measurements with the fuel kept against the back side of the detector body, does not affect the results, as the lateral movement is much less important.

4. CONCLUSIONS

Monte Carlo based calculations were carried out for a better understanding of the parameters that influence the response to neutrons of a Fork detector. The results of these Monte Carlo simulations were presented. According to these results, the sensitivity of the neutron detector to the inner rows of pins can be increased removing the Cd sleeve and the boron in the water. However, these conditions result in an increased sensitive zone along the axial direction.

Simulations oriented to the feasibility of measurements in dry conditions were also carried out. According to the results obtained measurements in dry conditions are feasible and would result in a decrease of efficiency of about 30%. The axial importance function would be

increased by about 35% in dry conditions. Therefore, a worse resolution for a fuel axial scanning is expected.
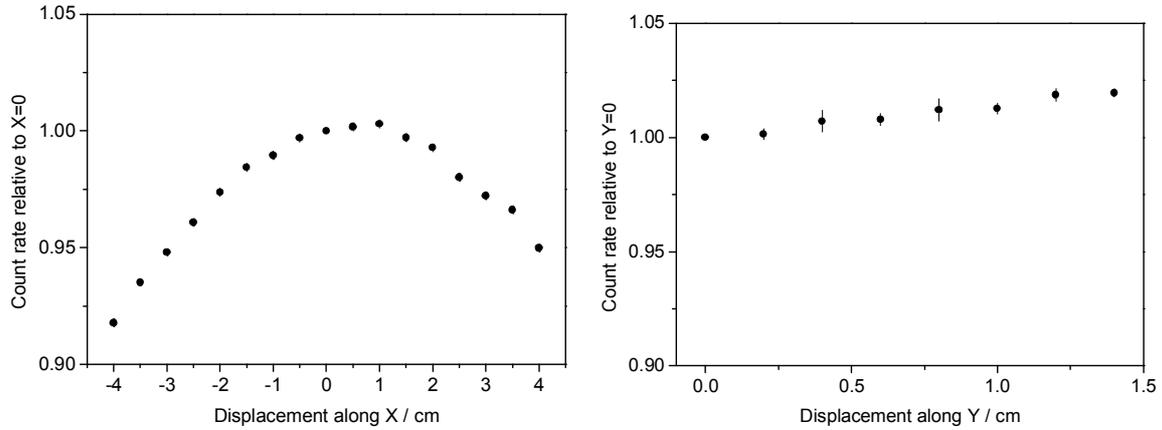
FIG. 7. Sensitivity of the neutron counts to displacement in the X (left) and Y (right) position in wet conditions.
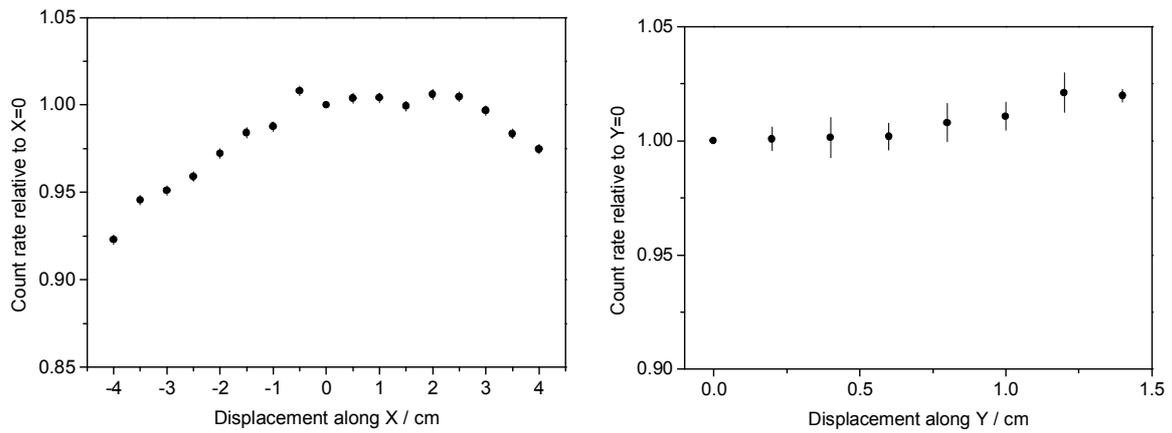
FIG. 8. Sensitivity of the neutron counts to displacement in the X (left) and Y (right) position in dry conditions.

ACKNOWLEDGEMENTS

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MEET THE CHALLENGES OF SPENT FUEL INTERIM STORAGE BY USING INTENSIVE INNOVATION

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Abstract

AREVA Logistics Business Unit, through its entities TN International in France, Transnuclear Inc. in the USA and Transnuclear Ltd. in Japan, has proposed for more than 2 decades the leading dry storage systems of spent fuel in use today. These systems have mainly been sold in Europe, in the US and in Japan. The PWR, BWR or VVER fuel characteristics may have various enrichment values up to 5%, various cooling time down to 2 years and various burnups up to 65,000 MWd/tU. Facing the current international trend towards expanding Spent Fuel Interim Storage capabilities and the unpredictable market prices of steel large forged components, AREVA Logistics Business Unit has launched an extensive innovation process to create the new generation of dry interim storage systems: i) the TN®DUO cask is an innovative and cost effective dual purpose cask; and ii) the TN®NOVA system is an innovative canister system based on the NUHOMS® cask system, the US industry leading spent fuel storage solution. These two innovative solutions can naturally be transported to the storage facilities as well as other sites such as reprocessing facilities or geological repositories depending of the national strategy for the back-end of the nuclear fuel cycle. In addition to these innovative dry interim storage systems and based on 40 years experience in design, licensing and fabrication of baskets for transportation cask, AREVA Logistics Business Unit has developed new innovative designs for Underwater Fuel Storage Racks which includes the use of Metal Matrix Composite (MMC) material as a neutron absorbing material. This kind of material allows proposing a cost efficient solution with a reduced rack weight and a significant improvement of the criticality performance. Furthermore, AREVA Logistics Business Unit Rack Design remains flexible and evolutionary linked to fuel characteristics evolution and it can include other neutron absorbing materials commonly used in the nuclear industry as borated stainless steel (BSS). The purpose of this paper is to present these experiences, and furthermore to underline our know-how and ability to provide highly efficient interim storage systems.

1. INTRODUCTION

As from the 1990s, AREVA Logistics Business Unit (BU) has developed the TN®24 family of cask for both transport and storage of spent nuclear fuel. Many different versions have been designed to accommodate different quantities and types of fuel, ranging from 21 PWR assemblies to 97 BWR assemblies. These dual-purpose casks have been sold in Europe (Belgium, Switzerland, Germany and Italy), in the United States and in Japan to safely perform the interim storage of used fuel elements. In parallel, another type of interim storage system so called the NUHOMS® system has been developed mainly for the US market: the spent fuel assemblies are stored inside a canister which is placed inside a concrete storage overpack. These two types of storage products were perfectly suited to their respective markets.

The increasing demand for massive forged pieces such as the ones used for the TN®24 body brought uncertainties as to the availability and price of the TN®24 body. More generally, the overall cask needed to be redesigned to optimize all the costs.

Hence, in order to be more cost effective and to take into account new safety constraints AREVA Logistics BU decided to launch an intensive innovation process using design to cost and creativity methods. This process has led to two storage solutions: a new family of dual-purpose cask, which is called TN®DUO and a new line of canister solution, which is called the TN®NOVA.

In addition to these innovative dry storage systems, AREVA Logistics BU has developed new innovative designs for Underwater Fuel Storage Racks. Thanks to a specific technology developed for the dual-purpose cask high capacity baskets, AREVA Logistics BU designed in only 18 months 3 different racks dedicated to European, Chinese and US markets.

2. TN®DUO CASK: THE NEW LINE OF DUAL-PURPOSE CASK

The TN®DUO dual-purpose cask was first developed for the BWR spent fuel storage of a Swiss operator. Considering its customer's needs, AREVA Logistics BU decided to adapt this concept to PWR fuel assemblies as well. The work performed for this new solution development is an excellent example of AREVA Logistics BU ability to offer adapted solutions to its customers.

The TN®DUO dual-purpose cask is presented in the Figure 1.
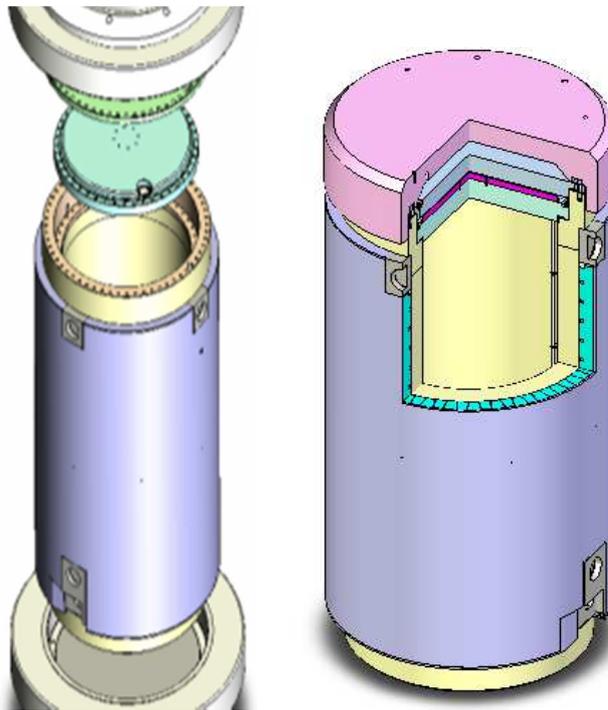
FIG. 1. View of the TN®DUO in transport configuration and in storage configuration.

2.1. TN®DUO concept

The TN®DUO cask has been designed to ensure the safety of the spent fuel during their storage period and their transportation. The transport cask has been designed to meet type B package requirements of the transport regulations issued by IAEA, 2005 regulations.

The TN®DUO dual-purpose cask is mainly constituted of:

- A cask body constituted of several forged pieces, closed by 2 bolted lids;
- A basket to host the spent fuel assemblies;
- The shock absorbing covers in transport configuration or an anti-aircraft crash cover in storage configuration.

The safety of the cask is mainly ensured by the mechanical properties of the cask body and the lids equipped with their shock absorbing covers in transport conditions. The cask body is forged carbon steel, which has good ductility at low temperature. It also insures the gamma shielding and the transfer of the decay heat.

Aluminium heat exchangers are set up between the body and the cask outer surface to dissipate the fuel decay heat. Resin blocks are inserted in these heat exchangers to insure neutron shielding capacities.

The inner basket provides compact spacing of the spent fuel according to the type of fuel. It is set in the cavity and:

- Mechanically supports fuel assemblies;
- Maintains sub-criticality in transport conditions, during fuel loading and unloading operations, and in storage conditions;
- Transfers the fuel decay heat to the cask body.

The typical content of the TN®DUO dual-purpose cask is presented in the Table 1.

TABLE 1. TN®DUO SPENT FUEL CONTENT

TN®DUO cask	BWR content	PWR content
Capacity	52–68 BWR SFA w/ control components	24–37 PWR SFA w/ control components
Fuel Enrichment	5%	5%
Burnup	70 GWd/tU	65 GWd/tU
Decay Heat	32 kW	32 kW

In storage configuration, a monitoring system including three pressure sensors are located on the side of the cask.

An anti-aircraft crash cover is fitted on the top of the cask and protects the package in case of airplane crash.

For the transport, the TN®DUO is equipped with shock absorbers that are suitable for rail and road utilisation.

2.2. TN®DUO advantages

While having high technical performances, the TN®DUO brings cost effectiveness to every design features and provides significant advantages:

- **A dual-purpose cask** (transport and storage) following IAEA 2005 regulations;
- **No massive forged body** but several forged pieces;
- **New aluminium heat exchangers** to simplify the manufacturing: corrugated aluminium plates for the conductors are optimized to simplify cask assembly. Thermal

tests run on a mock-up have already validated the calculated heat rejection capacities of this design;

- **A new basket design** has been developed for the TN®DUO to optimize the existing 28 PWR assemblies basket. Another basket is currently pre-designed to reach 32 PWR assemblies with the burn up credit methodology. This would ease the manufacturing and the assembly while improving the level of performance. A basket dedicated to the BWR assemblies has been also developed with a capacity of 68 assemblies;

- **Resin blocks** are inserted into the heat conductors to optimize the overall assembling phase;

- **Efficient shock absorbers are designed** to minimize acceleration during drops using the latest technology developed by AREVA Logistics R&D teams;

- **The TN®DUO incorporates the latest advances** of the AREVA Logistics BU and subcontractors innovation concepts. For instance, for the basket, metal matrix composites contain the highest B4C ratio, providing the latest technologies in the field;

- **Smart design features** have been integrated in the TN®DUO to ease storage operations. The monitoring system is now located on the side of the cask and thus does not require removing the anti-aircraft crash lid.

3. TN®NOVA SYSTEM: THE NEW LINE OF CANISTER SOLUTION

The TN®NOVA system is evolved from the NUHOMS® cask system, the US industry leading innovative spent fuel storage solution. The TN®NOVA System differs from the NUHOMS® system by using a vertical metallic storage overpack instead of a horizontal concrete module which surrounds the spent fuel placed inside a canister. The TN®NOVA System can position itself as one of the most technologically advanced, cost effective and operationally convenient system in today's dry storage metal casks market.

The TN®NOVA system is presented in Fig. 2.
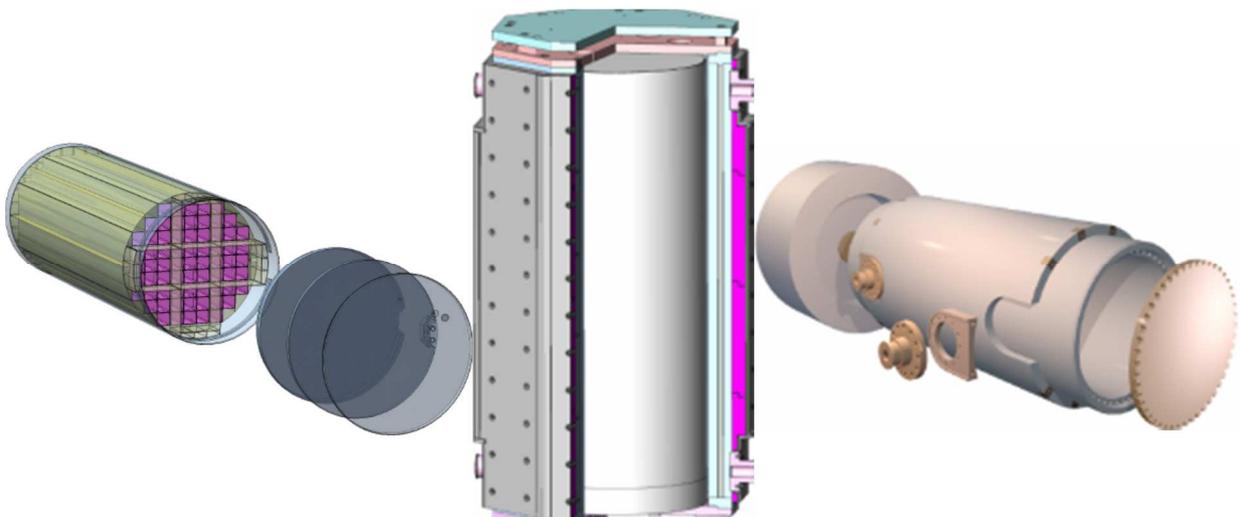
FIG. 2.View of the different components of the TN®NOVA system: the canister, the storage overpack and the transport cask.

The TN®NOVA system was developed in 2007 and offered to the Swiss utility Axpo. The current design has been developed for both PWR and BWR fuel assemblies. More generally, this solution can be adapted to comply with the requirements of any interested operator. This is the perfect illustration of the dedication of AREVA Logistics Business Unit teams to provide tailor-made solutions to their customers.

3.1. TN®NOVA concept

The TN®NOVA system is comprised of three different elements:

- **The canister:** the fuel assemblies are loaded in the canister, while the canister is in the transfer or transport cask in the fuel pool. Once welded, the canister is transferred horizontally from the cask into the TN®NOVA storage overpack. The 69BTH and 37PTH dry shielded canisters ensure primary containment for 69 BWR and 37 PWR fuel assemblies;
- **The storage overpack:** the TN®NOVA provides for horizontal transfer and vertical storage inside a storage facility. The TN®NOVA provides protection for the canister during storage and provides for passive cooling of the loaded spent fuel canister. The TN®NOVA overpack is constructed primarily of carbon steel and is well shielded to keep doses ALARA. Additionally, the TN®NOVA provides aircraft impact and earthquake resistance;
- **The transport or transfer cask:** the MP197HB Transport Cask is designed to transport the fully loaded canister and would be licensed in accordance with the US requirements of 10CFR Part 71 in October 2010. This transport cask is designed to meet IAEA requirements and maintain compatibility with the La Hague reprocessing facility. For the transfer and loading operations, the OS197 Transfer cask could be used instead of the transport cask.

TN®NOVA operations: The fuel assemblies are loaded into the canister (which is placed inside the transport or transfer cask) in the fuel pool at the reactor site or in a hot cell. The transfer cask containing the loaded canister is then prepared for the transfer or transport operations (lid sealing, cavity draining and drying, helium backfilling). At the storage location, the cask is unloaded of the canister: alignment of the cask with TN®NOVA overpack, which is in horizontal position, push of the canister using the hydraulic ram, placement of the lid on the storage overpack, uprighting of the loaded TN®NOVA in a vertical position for storage. Once inside the TN®NOVA overpack, the canister is in safe, passive dry storage. This equipment and process allows simple and low risk transfer operations.

The typical content of the TN®NOVA dual-purpose cask is presented in Table 2.

TABLE 2.TN®NOVA SPENT FUEL CONTENT

TN®NOVA System	BWR content	PWR content
Capacity	69 BWR SFA w/ control components	37 PWR SFA w/ control components
Fuel Enrichment	5%	5%
Burnup	70 GWd/tU	65 GWd/tU
Decay Heat	32 kW	32 kW

3.2. TN®NOVA advantages

The TN®NOVA System is one of the most technologically advanced, cost effective and operationally convenient system and it provides significant advantages:

- **Separation of the transport and storage functions** brings flexibility to the operator: The TN®NOVA system uses two different casks for storage and transportation. As a result, the TN®NOVA system dispatches the stringent criteria of transport and storage regulations. Future changes to transport regulation will have little or no effect on the canister solutions. At the time of de-storage, the canister can be transferred to a transport cask meeting future licensing requirements at very limited cost;
- **The procedure to transfer horizontally the canister from the transfer or transport cask to the storage overpack** by using a hydraulic RAM will be the same as the one currently in use with the NUHOMS® system in the US. Once loaded, the TN®NOVA is uprighted to a vertical position and transferred to its storage location. To date more than 460 canisters have been loaded and welded by AREVA Logistics Business Unit in the US without incident;
- **Efficient low constraint cooling of the TN®NOVA system:** the TN®NOVA storage overpack removes spent fuel decay heat using natural air circulation: through bottom ventilation inlet openings, the air is guided to the inner cavity where it circulates around the canister before exiting through top outlet openings. Most of the heat is rejected this way and the storage cask materials can thus be dedicated to improve shielding and mechanical performance;
- **High adaptability of the system:** the TN®NOVA system offers high adaptability and efficiency for spent fuel pool unloading in 69BTH or 37PTH canisters. Depending on the evolution of the spent fuel, the loading plan can be adapted to optimize fuel quantity, heat load and burn up. Aluminium inserts can be placed at the centre of the canister to allow high heat loads and high burnups;
- **Cost effectiveness:** the TN®NOVA system provides cost effectiveness and the security of supply and manufacture due to the use of common raw goods and standard manufacturing process.

4. UNDERWATER FUEL STORAGE RACKS

Leapfrogging the current 1980s technology, the underwater fuel storage racks developed by AREVA Logistics BU are qualified for use at new EPR™ reactors and have the versatility to support Spent Fuel Pool re-racks and the international market.

AREVA Logistics BU has developed an innovative design for underwater fuel storage racks which includes the use of metal matrix composite (MMC) material as a neutron absorbing material. Furthermore, AREVA Logistics Business Unit Rack Design remains flexible and can include other neutron absorbing materials commonly used in the nuclear industry as borated stainless steel (BSS).

4.1. Functions of AREVA logistics business unit racks

The main functions of Fuel Storage Racks are the following ones:

- To allow the storage of the corresponding quantity of spent fuel assemblies;
- To have a sufficient mechanical resistance under normal conditions but also to withstand the design earthquake;

- To maintain the fuel assemblies in a sub-critical configuration (even in case of total dilution of the pond water (no more boric acid present in the water);
- To guarantee the proper cooling of the fuel assemblies (the natural circulation of the water through the lodgements has to be guaranteed);
- To allow the insertion and removal of the fuel assemblies from the storage racks in all conditions and to avoid inducing any damage namely during the insertion or extraction of the fuels in their lodgements (namely by using plane surfaces and smooth angles).

All these main functions have to be guaranteed for the long term.

The view of the fuel storage rack is presented in the Fig. 3.
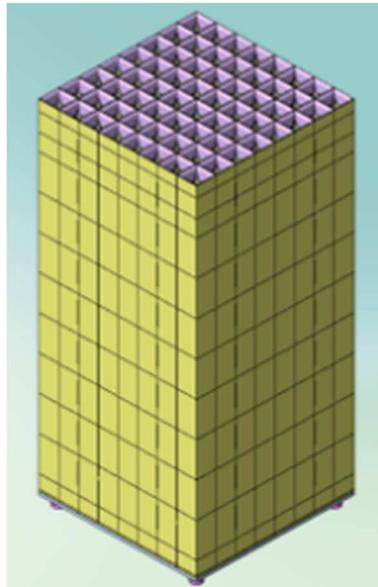
FIG. 3. View of the fuel storage rack.

4.2. Structure design

The designs of the underwater fuel storage racks have taken advantage of the AREVA Logistics BU expertise in the field of the design, licensing and fabrication of the dual-purpose cask high capacity baskets. Thus 3 Underwater Fuel Storage Racks designs have been developed:

- Designs using BSS or MMC as poison material in accordance with the European and Chinese requirements;
- Design using MMC as poison material (NUSTOR™) in accordance with the US requirements of 10CFR Part 50 and 52.

The design proposed to European and Chinese markets consists in:

- A stainless steel structure defining adjacent cells (called "central structure");
- "Individual tubular structures" (hereafter called "sleeves") inserted in these adjacent cells.

The stainless steel central structure is composed of stainless steel plates (A240 gr 304L stainless steel). The stainless steel plates are linked together at the intersections by welds placed between two perpendicular stainless steel plates.

The sleeves inserted in each different cell of the central stainless steel structure are composed of the following elements:

- Four angle beams which define the lodgement (cross section to be adapted to the need of the nuclear power plant operator) in which one the fuel assembly is inserted;
- "Peripheral spacers" surrounding these angles beams (and welded to these beams), these spacers are called further in this document "hoops";
- Borated plates placed behind the corner beams linked to these beams by stainless steel mounting brackets welded onto the angle beams.

In order to propose the most optimized and competitive rack design while fulfilling the criticality requirements, the best compromise between the boron content and the thickness of the borated plates is determined. Besides, the hoops are positioned in front of the grids of the fuel assemblies in order to optimize the load transfers to the stainless steel central structure (namely in case of earthquake). Openings are machined on the hoops to ensure a natural water circulation in the rack modules and avoid trapping air bubbles. The connection between the lead-in and the corner beams insures a continuous guidance of the fuel assembly in the sleeve.

4.3. Design advantages

The design proposed by AREVA Logistics Business Unit presents the following advantages:

- The use of the high performance neutron absorbing material "boron MMC" allows a significant cost reduction;
- The mechanical and criticality functions are separated (no mechanical requirement on the boron MMC plates);
- The use of the MMC material allows a significant reduction in the mass of the modules (and therefore of the global mass of the rack) which are consequently easier to transport and handle (which is important in the frame of this offer, the handling capacity of the power plants being limited to 10 tons including the mass of the handling devices);
- The sleeve structure allows a great flexibility, and an adaptation of the Borated Plates features (possibility to use BSS) to the nuclear power plant operator's needs.

A view of the Fuel Storage Rack manufactured by AREVA Logistics BU is presented in the Fig. 4.
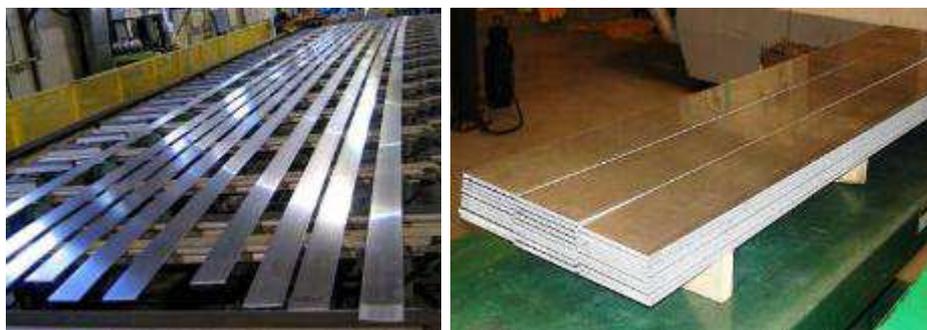
FIG. 4. View of the rack manufactured by AREVA logistics BU & views of MMC plates.

5.   CONCLUSIONS

The AREVA Logistics BU solutions are designed to be the most economical and versatile storage products available based on proven designs and ease of fabrication, simplicity of use and the availability of materials. And with our history of supplying satisfied clients with quality fuel related products for over 40 years, utilities can rest assured that the AREVA Logistics BU solutions will meet their needs.

EVALUATION AND SELECTION OF BOUNDARY ISOTOPIC COMPOSITION FOR BURNUP CREDIT CRITICALITY SAFETY ANALYSIS OF RBMK SPENT FUEL MANAGEMENT

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Abstract

The on-site wet-type spent fuel storage facility ISF-1 is currently used for interim storage of spent nuclear fuel removed from Chernobyl NPP power units. The results of ISF-1 preliminary criticality analyses demonstrated the need for using the burnup credit principle in nuclear safety analysis. This paper provides results from the selection and testing of computer codes for determining the isotopic composition of RBMK spent fuel. Assessment is carried out and conclusions are made on conservative approaches to fuel burnup credit in subsequent ISF-1 safety assessment.

1. INTRODUCTION

After the Ukrainian Government made the decision on early decommissioning of Chernobyl NPP power units 1, 2 and 3, these units have been at the stage of operation cessation.

The on-site wet-type spent fuel storage facility ISF-1 is used for interim storage of spent nuclear fuel removed from Chernobyl NPP power units.

The results of ISF-1 preliminary criticality analyses demonstrated the need for using the burnup credit principle in nuclear safety analysis under the optimum moderation or some accident conditions provided for in the Ukrainian regulatory documents [1]. The principle is widely used in international practice (IAEA recommendations [2], regulations of the USA [3], Germany [4], etc.) in criticality analysis of spent fuel transportation and interim storage systems.

During the first stage of implementing the burnup credit principle, experimental and calculation data on the isotopic composition of RBMK spent fuel were collected and analyzed, computer codes to identify the spent fuel isotopic composition were selected and tested.

Then, the impact of uncertainties of fuel burnup operating characteristics on the concentration of individual isotopes was analyzed in view of ensuring the acceptable level of conservatism in calculating system criticality (the so-called sensitivity analysis). The conclusions were made: how the impact of fuel burnup operating characteristics can be conservatively considered in subsequent ISF-1 nuclear safety analysis with fuel burnup credit.

1.1. Parameters of RBMK-1000 fuel assemblies

Main type of fuel assemblies that are stored at or planned to be placed into ISF-1 is regular fuel assemblies (FA). Figure 1 shows schematic view RBMK-1000 FA and their basic components. The geometrical and material parameters of FA used in the calculations are presented in Table 1 and correspond to data provided in [5–7].

1 – suspension bracket; 2 – top plug; 3 – adapter;
4 – connecting rod; 5 – fuel rod; 6 – carrier rod;
7 – end sleeve; 8 – end cap; 9 – nut

1 – bottom plug; 2 – fuel pellets; 3 – claddin
4 – spring; 5 – insert; 6 – end piece
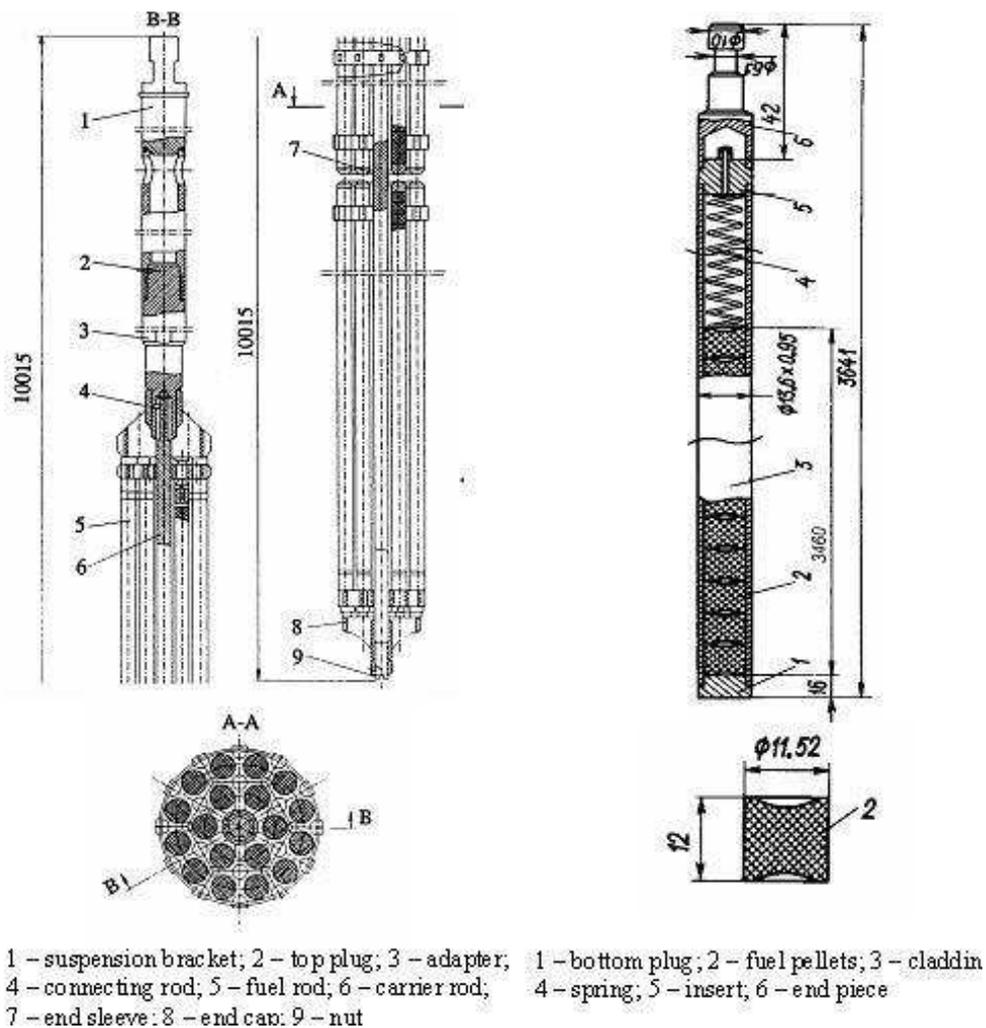
FIG. 1. Schematic of RBMK-1000 FA and fuel rod.

1.2. List of isotopes and experimental data

To calculate the isotopic composition of spent nuclear fuel, it is needed to determine the isotopes which are most significant in terms of accounting multiplication properties of storage systems (K_{eff}).

A list of isotopes for the burnup credit criticality analysis of spent fuel storage systems is based on validation results of codes for calculating the nuclide composition of spent fuel. The list includes only those isotopes whose calculated concentration has been proved to be reliable after comparing with experimental data.

To test the computer codes, experimental data provided in [8] were selected. Measurement isotopic composition for 3 RBMK FA of the Leningrad NPP was used. The initial enrichment of RBMK FA is 1.8% and 2.0% (regular fuel assemblies). RBMK FA was unloaded from the core in the 1976–1989s. The selected assemblies had operated for 1–5 years. The isotopic composition of 18 samples with different burnups and height along the assembly was

determined; this was significant in terms of inhomogeneous composition of the RBMK core along the height (coolant density, moderator temperature etc.).

The conditions of fuel burnup and subsequent holding are not known. It was accepted in the calculations that fuel burnup occurred at rated power of the reactor.

The concentrations of fission products were not measured. The data relate only to the concentration of fuel isotopes and actinides in spent fuel. The error of measurement is indicated as about 1–7%. Therefore, we limited the subsequent analysis to the following fuel isotopes and actinides:

^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{237}Np, ^{243}Am, ^{242}Cm, ^{244}Cm
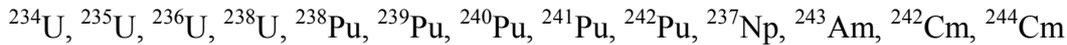
TABLE 1. THE GEOMETRICAL AND MATERIAL PROPERTIES OF RBMK-1000 FA

RBMK fuel element	
Total / Active fuel length in cold state	3640 / 3460 mm
Pellet (FA enrichment by ^{235}U, %)	UO$_2$ (1.8, 2.0, 2.4±0.05)
External fuel pellet radius	0.575 cm
Internal / external fuel cladding radius	0.5975 / 0.68 cm
Density	6.45157 g/cm^3
Composition	Zr - 98.97, Nb - 1, Hf - 0.03 wt.%
Central tube	
Internal / external tube radius	0.625 / 0.75 cm
Density	6.45157 g/cm^3
Composition	Zr–97.47, Nb-2.5, Hf–0.03 wt.%
Regular FA	
Number of segments / fuel elements in a segment	2 / 18
Total weight of steel/zirconium, kg	2.34/ 40
Weight of uranium in FA, kg	114.7±1.6
Spacer grid	
Composition	Steel
Weight of grids / Thickness	1.22 g / 2.0 mm

1.3. Description of the computer codes and models. Results of computer modeling of experimental data

RBMK fuel is quite difficult for developing a geometrical model. This is due to the following factors that make RBMK fuel complex for modeling:

- The presence of two moderators. The rbmk fuel assembly (Fig. 1) is located inside the graphite block (Fig. 2). The distance between the assemblies is quite large and is much greater than that between the fuel rods. The channel with the assembly also contains water, or more exactly, steam-water mixture with different densities along the channel height;
- The position of fuel rods in the assembly is not regular. The rods are arranged in two circles;
- The presence of a large number of structural elements in the fuel assembly — central tube and tube of the channel with the fuel assembly.

At the first stage, for a comparative analysis and selection of the most appropriate solution for the problem, the codes were tested on the basis of data from [8].

The uncertainty of input data was preliminary assessed as to their impact on calculation results. Impact of the following parameters was assessed:

- Power level, loading curve;
- Density and temperature of the graphite (moderator) and water (coolant);
- Temperature of the fuel cladding, central tube and fuel channel;
- Location of the fuel rod in the assembly.

The water density in the fuel channel and power level has the greatest impact on the isotopic composition of spent FA. The location of the fuel rod in the assembly (internal or external circles) also has a significant impact.
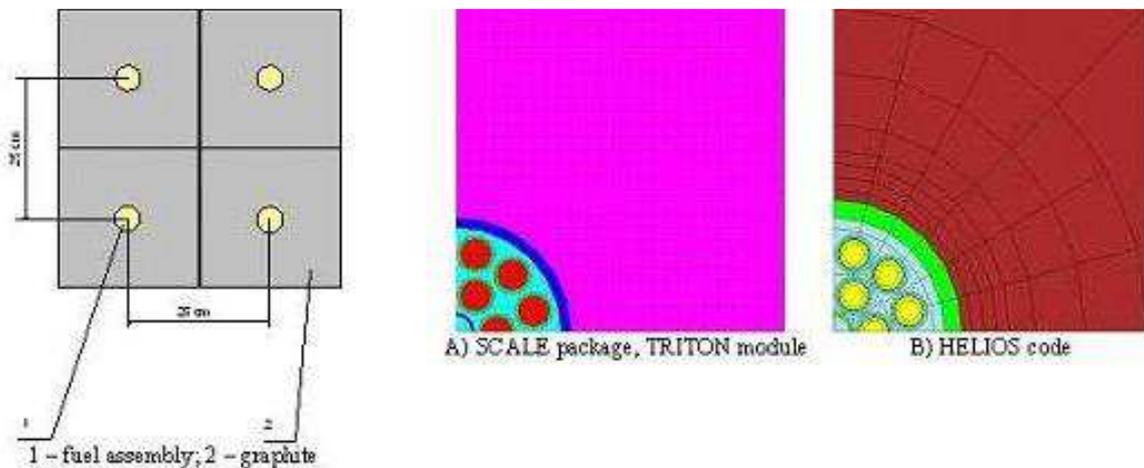
FIG. 2. Location of fuel assemblies in the RBMK-1000 core and RBMK-100 FA computer models.

The concentration of isotopes was determined separately for the internal and external fuel rod circuits in compliance with data from [8].

To determine the isotopic composition of spent RBMK fuel assemblies, 2 codes were selected: US SCALE code package and HELIOS code of the SCANDPOWER Company.

The SCALE code package includes computer modules combining programs to calculate one or another problem (criticality analysis, radiation safety, heat transfer, isotopic composition distribution depending on burnup). The most complete description of the programs included in the SCALE code package is provided in —].

The SCALE code package was first of all developed and validated for PWR and BWR fuel systems. In recent years, SCALE has been widely used to model WWER and RBMK fuel management systems. The applicability of the SCALE code package and its libraries of neutron-physical constants for modeling WWER and RBMK fuel management systems is considered in [12]. To calculate the isotopic composition of spent fuel depending on burnup, we used the TRITON code, which models fuel burnup processes in complex 2D geometry.

The HELIOS code [9–10] is intended to calculate spatial energy distribution of neutron flow density in transport approximation for the cell (this may be: fuel assembly for WWER-1000 or fuel assembly and graphite moderator for RBMK-1000) of the nuclear reactor. The cell is calculated in 2D geometry; this permits adequate representation of the geometry and

composition of the assembly (cell) in the cross-section (location of different assembly components is taken into account, such as: central tube, fuel elements with different enrichments, fuel elements containing a burnable absorber with different concentrations, fuel channel tube). The HELIOS code is now widely used in the world practice for all types of research and power reactors [10].

Based on the RBMK-1000 geometry, computer models were developed for the TRITON and HELIOS codes. The computer cell (see Fig. 2) shows some part of the assembly and graphite moderator with mirror reflection at the boundaries.

All FAs were modeled pin-by-pin (Fig. 2). The location of FA elements, their geometry and material composition correspond to the description provided in Section 1. The effective density of the cylindrical uranium core used to model the fuel part of the rod was based on the total weight of fuel in FA taking into account tolerances. The top and bottom end pieces of the FA were not modeled; they were replaced by the moderator (water), which is obviously a conservative assumption.

Table 2 and Fig. 3 summarize the results from computer modeling of experimental data from [8] using the TRITON and HELIOS codes for 18 samples from three FA (No. 5-8,10–17, 20–25).

1.4. Analysis of results from experimental data modeling

Analysis of the results from computer modeling of experimental data immediately shows that the SCALE (TRITON module) and HELIOS codes give a large root-mean-square error (more than 20%) in determining the concentrations of ^{238}Pu, ^{242}Pu, ^{237}Np, ^{243}Am, ^{242}Cm and ^{244}Cm. Hence, these isotopes can be excluded from further consideration.

Regarding the remaining 7 isotopes ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu and ^{241}Pu, we should exclude ^{234}U and ^{236}U from further analysis. The certificates for RBMK and WWER FA do not indicate the initial concentrations of ^{234}U and ^{236}U isotopes in fresh fuel (in contrast to the certificates for Western PWR and BWR). These isotopes are present in fresh fuel, and change in their concentration with burnup substantially depends on the initial values.

Therefore, it can be recommended for burnup credit nuclear safety analysis that the following 5 isotopes are used for burnup credit:

^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu
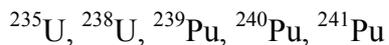
In the licensing process for Ukraine's first dry-type interim storage facility for WWER-1000 spent fuel at Zaporizhya NPP, in 2003 the National Operator «NAEK Energoatom» developed and implemented the document [13] to determine the burnup credit procedure for analyzing the loading of storage casks. Based on a preliminary analysis, this document identifies the same list of isotopes that can be used in burnup credit nuclear safety analysis, namely: ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu and ^{241}Pu. Therefore, it can be stated that the list of isotopes that was earlier approved for WWER-1000 fuel is proposed for further analysis of RBMK-1000 fuel.

TABLE 2. RELATIVE ERROR IN DETERMINING THE CONCENTRATION OF ISOTOPES

Sam-ple N.	Isotope												
	^{234}U	^{235}U	^{236}U	^{238}U	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{237}Np	^{243}Am	^{242}Cm	^{244}Cm
	Error in determining the concentration, (experiment-calculation)/experiment*100%												
Relative error in determining the concentration of isotopes using the TRITON code													
5	-10.60	2.10	-4.33	0.04	14.47	12.57	-6.16	11.88	1.35	17.73	-2.89	36.74	-7.21
6	-11.34	-9.71	-3.97	0.10	19.90	5.56	-2.75	5.81	5.27	-1.41	1.01	39.47	4.26
7	-3.15	-3.16	1.74	-0.17	36.29	11.31	-0.30	13.00	10.19	27.19	11.03	41.02	5.45
8	-7.91	10.15	-5.21	0.08	13.89	1.51	-15.22	-1.39	-17.63	35.63	4.29	36.42	-10.89
10	-14.84	-2.94	-7.69	0.02	32.60	16.36	4.64	19.74	8.63	49.00	15.15		
11	-11.87	5.13	-17.27	0.03	11.65	8.13	-15.11	-6.14	-47.90	33.43	-53.57		
12	-11.96	-5.81	-4.93	0.06	38.63	13.93	2.47	14.89	15.87		3.57		
13	-12.80	-2.36	-5.77	0.05	37.94	9.47	5.58	8.73	3.15	46.58	0.92		
14	-9.56	2.14	-8.30	0.06	30.36	8.16	-0.64	0.53	-3.33	-1.63	-21.23		
15	-0.63	22.40	-15.69	-0.08	15.06	22.91	0.38	9.55	-22.26	14.25	-81.39		
16	-13.03	-3.43	-5.64	0.03	42.28	17.51	4.71	20.14	18.29	37.36	1.85		
17	-11.59	-1.60	-5.37	0.08	35.08	8.03	-0.47	10.66	7.26		-0.33		
20	-4.49	0.42	4.56	0.01	-2.17	-4.82	-19.02	1.78	-11.20	30.28	56.98	57.85	43.95
21	-4.01	1.15	2.57	0.01	-1.84	4.92	-7.47	3.80	-11.90	26.43	21.52	58.44	-14.21
22	-5.15	1.63	2.06	-0.02	3.95	8.41	-1.91	8.19	1.09	27.43	20.12	59.44	57.40
23	-63.07	-4.22	17.72	0.10	38.55	9.76	12.48	23.01	28.55	32.67	76.39	77.50	79.43
24	-58.98	2.08	7.29	-0.01	17.08	5.74	-4.83	2.88	-9.04	35.25	53.38	59.83	7.52
25	-59.66	2.17	8.85	0.03	10.99	-11.36	-25.24	-10.93	1.66	24.06	81.98	48.75	65.61
Error in determining the concentration of isotopes using the HELIOS code													
5	-1.46	-8.80	1.65	-0.42	-2.72	-0.50	-3.16	-7.89	-3.73	12.64	-8.13	12.71	-6.51
6	-3.22	-18.59	-0.14	-0.41	12.43	1.87	0.49	0.69	8.46	0.99	10.92	25.34	22.64
7	3.87	-21.39	6.05	-0.83	25.77	1.47	3.11	0.38	10.77	24.26	11.54	20.54	12.89
8	0.86	6.01	-1.26	-0.44	9.72	-1.75	-12.45	-6.02	-12.72	38.19	16.75	26.39	12.76
10	-3.59	-2.86	1.57	-0.51	16.81	1.02	-2.39	-10.73	-11.26	43.35	-7.57	-	-
11	-2.80	-1.44	0.22	-0.55	17.24	1.22	-2.24	-8.49	-18.12	37.88	-12.28	-	-
12	-1.97	-8.61	2.32	-0.44	28.84	3.37	3.13	-2.09	9.98	-	-2.03	-	-
13	-5.69	-17.07	7.28	-0.58	43.79	3.14	15.46	10.44	25.62	51.60	32.75	-	-
14	-2.90	-11.60	4.51	-0.59	37.99	4.11	9.72	5.07	21.93	10.19	21.00	-	-
15	7.84	20.76	-8.44	-0.59	2.20	16.75	2.09	-1.21	-25.32	13.42	-79.17	-	-
16	-2.74	-6.83	2.26	-0.46	29.70	4.72	4.85	-0.44	9.75	32.84	-9.24	-	-
17	-1.87	-1.64	1.34	-0.48	29.70	2.69	0.68	3.12	8.02	-	7.64	-	-
20	7.50	-0.81	13.39	-0.50	-20.27	-24.69	-27.15	-31.56	-28.69	22.96	48.73	21.75	31.08
21	7.74	-0.07	10.13	-0.50	-15.52	-8.30	-10.27	-15.74	-17.33	21.49	18.79	25.62	-16.97
22	6.88	0.69	9.09	-0.52	-12.92	-5.88	-6.05	-14.88	-8.62	20.74	11.80	23.88	52.72
23	4.28	-10.41	26.97	-0.49	42.54	7.30	25.63	32.30	52.27	34.64	86.41	73.17	89.50
24	6.60	-4.32	16.84	-0.59	20.14	1.33	9.00	10.16	22.54	36.42	70.92	48.01	48.48
25	6.52	-3.48	18.49	-0.54	12.44	-18.36	-9.06	-8.29	26.88	23.69	87.91	39.90	78.82

The use of the neutron multiplication factor based on the concentration of only five transuranium isotopes in calculations overestimates the results (Fig. 3). For example, this overestimation is more than 5% for spent FA with the initial enrichment of 2.0% (taking into account fuel weight and enrichment tolerances) for burnup of 20 MWd/kgU. Actually, this conservatism that results from the credit of only five isotopes in nuclear safety assessment is intended to compensate for potential errors in determining spent nuclear fuel burnup and compensate for errors in determining the variation in the spent fuel isotopic composition.

The deviations of the calculated results for the SCALE (TRITON module) and HELIOS codes from the experimental data for different samples are shown in Fig. 3. For comparison, the same figures show results obtained earlier with the well-known WIMS British code by experts from the Russian Scientific Center «Kurchatov Institute» [8].

Table 3 summarizes the results obtained with different codes for the selected isotopes. The table shows that both codes provide approximately the same errors in determining the selected isotopes. As integral parameters that characterize the difference between the experimental and calculated data, we selected the maximum and minimum errors for each isotope and the root mean square deviation (RMSD) for all samples.

The figures show that the HELIOS code gives the most conservative results for all considered isotopes than the SCALE code.

TABLE 3. COMPARISON OF MODELING RESULTS FOR 5 ISOTOPES OF U AND PU

		Isotope				
		^{235}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu
		Error in determining the concentration, (experiment-calculation)/experiment*100%				
TRITON	min	-9.70	-0.17	-11.36	-25.24	-10.93
	max	22.40	0.102	22.91	12.48	23.01
	RMSD	6.44	0.065	11.24	9.56	11.69
HELIOS	min	-21.39	-0.83	-24.69	-27.15	-31.56
	max	20.76	-0.41	16.75	25.63	32.30
	RMSD	10.67	0.53	9.02	11.20	13.16

1.5. Computer analysis to determine dependence of the multiplication properties of spent fuel in ints burnup conditions

One of the important steps in implementing the burnup credit principle is to analyze how variation in operational characteristics of fuel burnup influences the concentration of individual isotopes in spent fuel. This analysis should take into account acceptable conservatism to calculate the criticality of the system; i.e., sensitivity analysis is needed. For this purpose, the concentrations of isotopes in RBMK spent fuel are calculated for different operational conditions and then the criticality of the ISF-1 storage system is calculated for these concentrations of isotopes.

Based on the results, a conservative set of RBMK burnup conditions is determined. It should take into account the differences in the material composition of the system for irradiated fuel and fuel stored at ISF-1. Fuel is irradiated in the RBMK cell where graphite is the main moderator and water as the coolant has smaller effect on the moderating properties of the system. Air-water mixture with the optimal density is considered in the ISF-1 cell in criticality assessments in compliance with Ukrainian regulations.

The limiting values of variation in operational burnup parameters of RBMK fuel are represented in Table 4. The characteristics of the RBMK reactor and fuel are selected in compliance with [5–7].
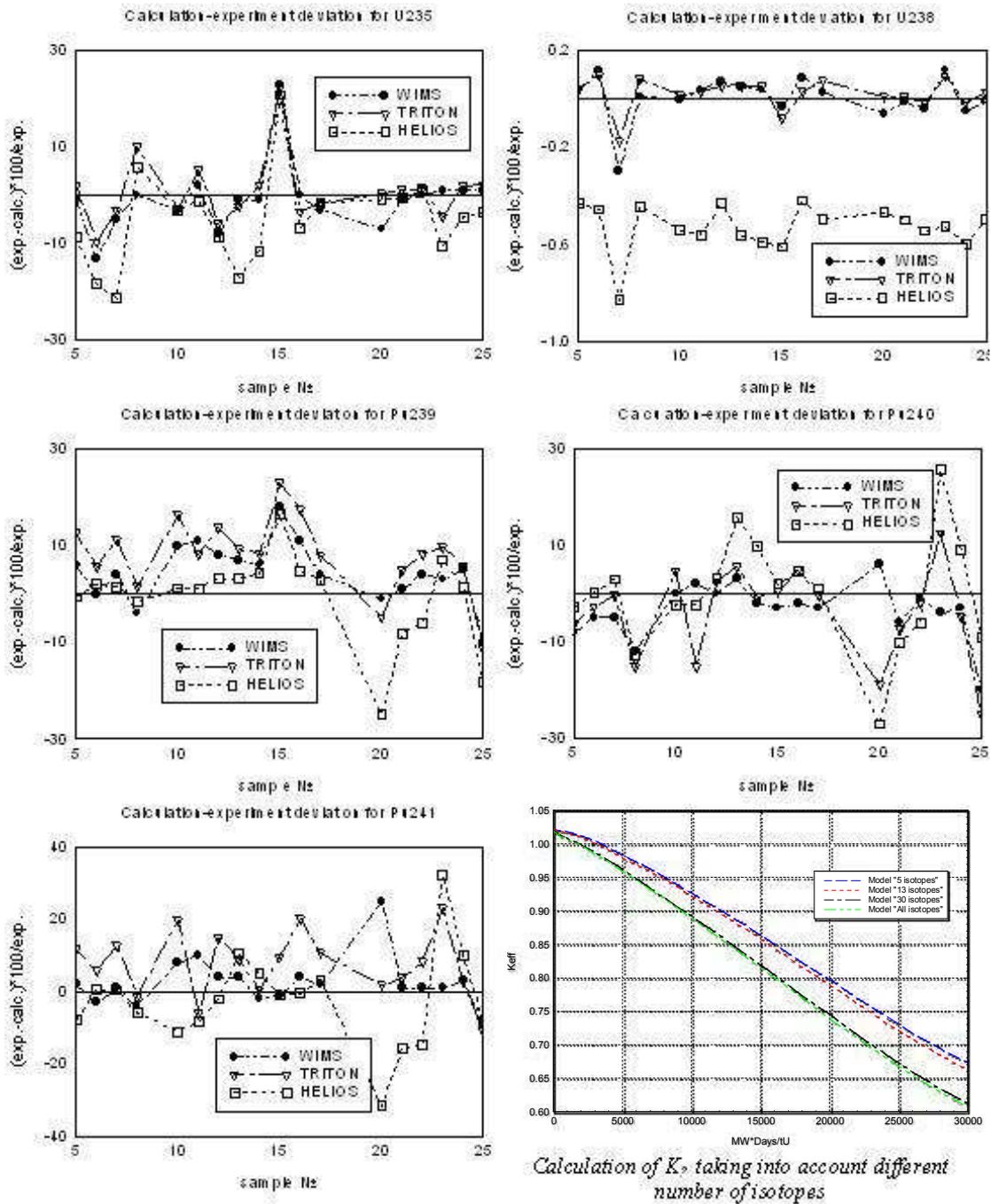
FIG. 3. Results from experimental data modeling.

TABLE 4. OPERATIONAL PARAMETERS OF RBMK FUEL BURNUP

Operational parameter	Range
Coolant (water) density, g/cm^3	0.2–0.8
Coolant (water) temperature, K	543–563
Moderator (graphite) density, g/cm^3	1.65
Moderator (graphite) temperature, K	693–853
Fuel (UO2) temperature, K	1020
Assembly power, W/gU	6.6–26.2

The impact of uncertainties of operational characteristics of fuel burnup on ISF-1 criticality was assessed using the SCALE code package (CSAS26 control module). The infinite lattice of ISF-1 cells was modeled (Fig. 4). Table 5 shows the geometrical characteristics of the computer model of the ISF-1 cell.
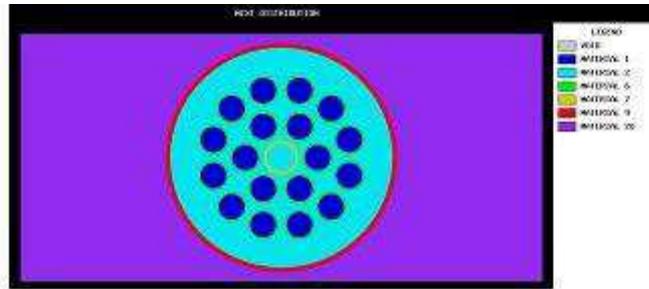
MATERIAL 1 – fuel, MATERIAL 2 – water in canister, MATERIAL 6 – fuel cladding,
MATERIAL 7 – central tube, MATERIAL 9 – canister,
MATERIAL 20 – water in storage facility (between canisters)

FIG. 4. Computer model of the ISF-1 cell.

Water temperature was accepted equal to 20°C. Water density in the canister was 0.8 g/cm^3 and in the space between the canisters 0 g/cm^3. These densities are optimal in terms of the maximum multiplication factor K_{eff} of the system and are selected based on the conclusions of [14].

Impact of the following operational parameters of RBMK burnup was assessed in criticality calculations:

- Change in coolant density during RBMK assembly irradiation (Fig. 5);
- Change in coolant temperature during RBMK assembly irradiation;
- Change in moderator temperature during RBMK assembly irradiation (Fig. 5);
- Change in fuel temperature during RBMK assembly irradiation;
- Change in assembly power during RBMK assembly irradiation;
- Credit of radial irregularity of RBMK assembly burnup.

The radial irregularity of RBMK assembly burnup was credited as follows. In the first case, the isotopic composition was accepted equal for all fuel rods. In the other case, the isotopic composition was set individually for the inner and outer rings of the fuel rods in RBMK assembly. In doing so, the isotopic composition along the fuel rod rings was formed in the burnup process in HELIOS cell calculation.
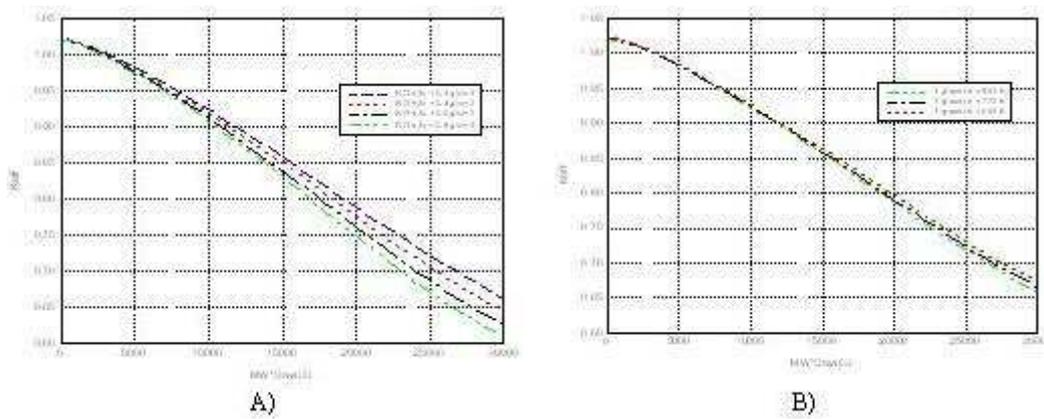
FIG. 5. Dependence of the multiplication properties of the ISF-1 cell changes in RBMK assembly during irradiation: A) coolant density and B) moderator temperature.

TABLE 5. GEOMETRICAL CHARACTERISTICS OF THE COMPUTER MODEL OF THE ISF-1 CELL

Operational parameters	Values for calculation
Inner / outer radius of the fuel channel, cm	4.0 / 4.4
Size of the computer cell of ISF-1, cm	230 × 110
Water density in canister / inter-canister space, g/cm^3	0.8 / 0.0

TABLE 6. OPERATIONAL PARAMETERS OF RBMK THAT ENSURING THE GREATEST MULTIPLICATION PROPERTIES OF THE ISF-1 CELL

Operational parameter	Values for calculation
Significant impact on K_{eff} of the ISF-1 cell	
Coolant (water-steam mixture) density, g/cm^3	0.2
Moderator (graphite) temperature, K	693
Insignificant impact on K_{eff} of the ISF-1 cell	
Coolant (water)/fuel (UO2) temperature, K	553/ 1020
Moderator (graphite) density, g/cm^3	1.65
Assembly power, W/gU	16.0, constant

Analysis of the results permits a conclusion that only two parameters have a noticeable impact on the RBMK assembly isotopic composition in terms of the multiplication properties of the ISF-1 cell. These parameters are changes in the coolant density and moderator temperature. The impact of other operational parameters on K_{eff} of the ISF-1 cell is insignificant. In this regard, a set of operational parameters for RBMK fuel burnup, which ensures the greatest multiplication properties of the ISF-1 cell, is selected and presented in Table 6. This set will further be used to calculate the concentration of isotopes in RBMK spent fuel in criticality assessments using the burnup credit principle.

2.   CONCLUSIONS

(1)   Analysis of the computer data obtained with the SCALE (TRITON module) and HELIOS codes permits the following conclusions:
   —   The developed models of the RBMK fuel assembly allow adequate modeling of changes in the isotopic composition of fuel during its burnup;
   —   The error of determining the concentration of uranium and plutonium can be assessed as ±10% and this value can be recommended for further use. The error of determining the concentration of other isotopes exceeds this value.

(2)   It is recommended that changes in the concentration of the following 5 isotopes are used in burnup credit nuclear safety analysis for ISF-1:
$$^{235}U, \quad ^{238}U, \quad ^{239}Pu, \quad ^{240}Pu, \quad ^{241}Pu.$$
This list of isotopes has been currently used for WWER-1000 fuel in nuclear safety analysis of the Zaporizhya NPP spent fuel dry-type cask storage system.

(3)   The dependence of the isotopic composition of spent fuel on its operational conditions has been analyzed and Chornobyl NPP RBMK operational parameters at which fuel stored in ISF-1 has the highest multiplication properties (i.e., conservative conditions in terms of nuclear safety) have been selected (Table 6).

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