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FOREWORD

The International Atomic Energy Agency convened a Technical Committee Meeting on Requirements, Practices and Developments in Burnup Credit (BUC) Applications in Madrid, Spain, from 22 to 26 April 2002. The meeting was hosted by the Consejo de Seguridad Nuclear at the premises of Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (CIEMAT), in Madrid.

The purpose of this meeting was to explore the progress and status of international activities related to the BUC applications for spent nuclear fuel.

This meeting was the third major meeting on the uses of BUC for spent fuel management systems held since the IAEA began to monitor the uses of BUC in spent fuel management systems in 1997. The first major meeting was an Advisory Group meeting (AGM), which was held in Vienna, in October 1997. The second major meeting was a Technical Committee meeting (TCM), which was held in Vienna, in July 2000. Several consultants meetings were held since 1997 to advise and assist the IAEA in planning and conducting its BUC activities. The proceedings of the 1997 AGM were published as IAEA-TECDOC-1013, and the proceedings of the 2000 TCM as IAEA-TECDOC-1241.

BUC for wet and dry storage systems, spent fuel transport, reprocessing and final disposal is needed in many Member States to allow for increased enrichment, and to increase storage capacities, cask capacities and dissolver capacities avoiding the need for extensive modifications. The use of BUC is a necessity for spent fuel disposal.

The IAEA wishes to thank the Consejo de Seguridad Nuclear and CIEMAT for hosting the meeting and all participants for their fruitful contributions, especially C. Neuber who helped to finalize the summary of the meeting. The IAEA officer responsible for the organization of the meeting and the overall preparation of this report was P. Dyck of the Division of Nuclear Fuel Cycle and Waste Technology.

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SUMMARY

1 INTRODUCTION

Application of burnup credit (BUC) to spent fuel management systems such as wet and dry storage installations, wet and dry transport systems, dissolvers in reprocessing plants and disposal installations consists in implementation of the following key steps:

- Criticality safety assessment of the spent fuel management system of interest including:
  - Estimation of the spent fuel composition under bounding depletion conditions;
  - Criticality calculation and evaluation of the loading criterion that indicates the minimum burnup or the limiting value of a related observable (as for instance the maximum fissile content) necessary or maximum allowable for fuel with a specific initial enrichment to be loaded in the spent fuel management system of interest;
- Application of the loading criterion including:
  - Quantification and verification of the burnup (or a related observable) of the fuel to be loaded in the system of interest;
  - Implementation of a loading procedure assuring compliance with the loading criterion.

Criticality safety is demonstrated with the aid of calculational methods verified by comparison to acceptable standards of known quality. Standards for comparison may be experiments, other accepted codes, or recognized standard problems. Validation of the prediction of the isotopic inventory of spent fuel is typically achieved by comparison with chemical assay data. The meeting addressed, therefore, the issues of:

- Experimental validation of inventory predictions and reactivity calculations including high burnup fuel implications and,
- The quality of nuclear data.

Prediction of the isotopic inventory of spent fuel requires knowledge of the irradiation history of the fuel. Due to the wide variety of irradiation histories it is necessary to look for a bounding history given by those reactor operation conditions resulting in the highest reactivity of the spent fuel under the conditions of the spent fuel management system of interest. It was one of the objectives of the meeting, therefore:

- To specify the depletion and criticality parameters that guarantee a bounding approach, and
- To exemplify the dependence of the importance of these BUC parameters upon the application case.

The evaluation of the loading criterion is based on application of a criticality safety acceptance criterion to the results of the reactivity calculations. A criticality safety acceptance criterion is based on the safety margin required by the regulations for the application case, includes the biases of the applied calculation procedures as obtained from validation of these procedures and depends on the statistical confidence level chosen to express the impact of all uncertainties due to the applied calculation procedures and due to manufacturing tolerances of the system of interest. Therefore, one of the topics of the meeting was directed to the issues of safety margin, bias, uncertainty and statistical confidence.

Because the loading criterion is dependent on the reactor operation conditions assumed in the isotopic inventory prediction and because application of the loading criterion introduces the need for fuel and loading verification and hence the need to rely on complex calculational methods and operator control as well as measurement checks that may be additionally necessary, application of BUC introduces local hazards specific of BUC. The meeting addressed, therefore:

- The issues of risk perspective and perception, and
- The issue of building consensus to present the aims and concerns of operators and regulators in conjunction with the issues of,
- BUC assessment schemes for implementation of BUC, fuel and loading verification, and review of analysis assumptions against changes in the fuel parameters.

The meeting addressed the benefits that have influenced decisions on the use of BUC. Recent developments in BUC and future BUC applications to:

- Wet and dry storage installations,
- Wet and dry transport systems,
- Dissolvers in reprocessing facilities,
- Disposal facilities.

for PWR, WWER, BWR and RBMK fuels, were presented. The meeting addressed, in particular, the issues of:

- BUC application to long term storage and disposal, and
- BUC for MOX and advanced fuel designs.

As an introduction to all the technical issues specified above overviews on the ongoing international activities in BUC were given at the beginning of the meeting. After this introductory part a considerable number of presentations on the above specified technical issues were given, demonstrating the significant advances in BUC implementation. Subsequent to the presentations parallel sessions by four working groups were held in order to discuss the issues in more detail. The topics discussed in the working groups were:

- Validation of codes and methods:
    - Lessons learned from previous experimental programs,
    - Application of results and experience to multiple fuel types,
    - Increase of initial enrichment and burnup.
- Key issues:
    - Depletion parameters,
    - Accounting for variations in plant operations and fuel design,
    - Axial burnup profiles,
    - Variability of parameter importance for different applications.
- Safety assessment and implementation:
    - Safety criteria,
    - Fuel loading and verification,
    - Risk, its perception and building consensus,
    - Continued validity of the analysis assumptions.

- Future applications:
  - MOX and advanced fuel designs,
  - Long term storage and disposal,
  - High burnup fuel.

Each of the groups produced a paper summarizing the discussions conducted and including the conclusions reached. The groups presented summaries of their discussions and conclusions during a panel session at the end of the meeting, and the conclusions reached are therefore the conclusions of the meeting.

2 OVERVIEW ON THE BUC EFFORTS BY COUNTRY

This part provides an overview on the national practices, ongoing activities and regulatory status of using BUC in different countries. The information was mainly gathered from the countries participating in this TM and is divided according to the different areas of BUC application, including:

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

2.1 Wet storage of spent fuel

2.1.1 Wet storage at reactor

Information on the status of BUC applications to wet storage at reactor is presented in Table I.

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

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

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

Table I also provides information on activities ongoing in different countries to get approval for implementing BUC.

Table I. Use of BUC for Wet Storage At Reactor (AR)

Country	PWR	BWR	MOX (PWR)	WWER	RBMK	Reactor Types[1]
Armenia	na	na	na	INT	na	WWER
Belgium	APU-1	na	na	na	na	PWR
Brazil	APU-2	na	na	na	na	PWR
Bulgaria	na	na	na	INT	na	WWER
China	INT	na	na	na	na	PWR
Czech Republic	na	na	na	INT	na	WWER
Finland	na	Gd	na	INT	na	WWER, BWR
France	Nc	na	Nc	na	na	PWR
Germany	APU-2	Gd	APC-2	Nc	na	PWR, BWR, WWER
Hungary	na	na	na	INT	na	WWER
Japan	INT	INT	INT	na	na	PWR, BWR
Korea, Republic of	APU-2	na	na	na	na	PWR
Lithuania	na	na	na	na	Nc	RBMK
Mexico	na	Gd	na	na	na	BWR
Netherlands	APU-2	na	na	na	na	PWR
Russian Federation	na	na	na	INT	APU-1	WWER, RBMK
Slovakia	na	na	na	UD-2	na	WWER
Slovenia	APU-2	na	na	na	na	PWR
South Africa	APU-2	na	na	na	na	PWR
Spain	APU-2	Gd	na	na	na	PWR, BWR
Sweden	Nc	Gd	na	na	na	PWR, BWR
Switzerland	APU-2	Gd	Nc	na	na	PWR, BWR
Ukraine	na	na	na	INT[2]	No	WWER, RBMK
United Kingdom	UD-2	na	na	na	na	PWR
United States of America	APU-2	Gd	UD-2	na	na	PWR, BWR

[1] BUC is not currently envisioned for heavy water or gas cooled reactors so they are not listed.

[2] BUC is allowed by the regulations but actions to implement have not started.

Abbreviations**:**

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

2.1.2 Wet storage away from reactor

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

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

Wet storage facilities away from reactor, that do not take credit for burnup exist in Bulgaria, Germany, Japan, the Russian Federation, Slovakia, Sweden, Ukraine, and the USA. A new facility is planned for Switzerland, which currently is not planned to take BUC.

2.2 Dry storage of spent fuel

Information on the status of BUC applications to dry storage is presented in Table II.

At present only a few countries are using BUC for dry storage. In Armenia the approval for BUC is limited to the use of the *net fissile content BUC level*. In this case credit is taken only for the net fissile content of the fuel. In contrast to this case in the USA application of the *actinide plus fission product BUC level* is permitted if it is ensured that the storage casks are loaded as well as unloaded in a borated pool and the probability of events resulting in penetration of moderator into the cask cavity during dry storage is considered to be very low.

2.3 Transport of spent fuel

Information on the status of BUC applications to transport casks is presented in Table III.

As can be seen from this Table a lot of activities are ongoing in several countries to get approval for application of the actinide plus fission product BUC level to transport casks.

2.4 Reprocessing

France: At La Hague actinide-only BUC is used for 10 years for storage in the pond and reprocessing. For liquids in tanks, some specific authorizations with fission product have been obtained. BUC that utilizes actinides and between 6 and 15 fission products will be submitted to the regulator shortly.

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

Russian Federation: BUC is currently used at the reprocessing facility.

United Kingdom: BUC for the reduction of Gd content in the dissolver is expected in May 2002. This BUC accounts for the change in actinides only.

Table II. Use of BUC for Dry Storage

Country	PWR	BWR	MOX (PWR)	WWER	RBMK	Reactor Types[1]
Armenia	na	na	na	APU-0	na	WWER
Belgium	Nc	na	na	na	na	PWR
Brazil	Nc	na	na	na	na	PWR
Bulgaria	na	na	na	INT	na	WWER
China	INT	na	na	na	na	PWR
Czech Republic	na	na	na	RR-2	na	WWER
Finland	na	nc	na	nc	na	WWER, BWR
France	Nc	na	Nc	na	na	PWR
Germany	APU-1, RR-2	Gd	RR-2	Nc	na	PWR, BWR, WWER
Hungary	na	na	na	INT	na	WWER
Japan	Nc	Nc	Nc	na	na	PWR, BWR
Korea, Republic of	INT	na	na	na	na	PWR
Lithuania	na	na	na	na	INT	RBMK
Mexico	na	Nc	na	na	na	BWR
Netherlands	Nc	na	na	na	na	PWR
Russian Federation	na	na	na	Nc	INT	WWER, RBMK
Slovakia	na	na	na	UD-2	na	WWER
Slovenia	Nc	na	na	na	na	PWR
South Africa	Nc	na	na	na	na	PWR
Spain	INT	INT	na	na	na	PWR, BWR
Sweden	na	na	na	na	na	PWR, BWR
Switzerland	INT	INT	INT	na	na	PWR, BWR
Ukraine	na	na	na	INT	INT[2]	WWER, RBMK
United Kingdom	na	na	na	na	na	PWR
United States of America	APU-M	INT	INT	na	na	PWR, BWR

[1] BUC is not currently envisioned for heavy water or gas cooled reactors so they are not listed.

[2] BUC is allowed by the regulations but actions to implement have not started.

Abbreviations:

APU-0:	Approved and implemented BUC using the net fissile content level.
APU-1:	Approved and implemented BUC using the actinide-only level.
APU-M:	Approved and implemented BUC using the actinide plus fission product level for loading and unloading in a borated pool, exclusion of moderator penetration into the cask cavity during dry storage.
RR-2:	Under regulatory review for taking BUC using the actinide plus fission product level.
UD-2:	Preparing documentation for taking credit using the actinide plus fission product level.
Gd:	Use of the integral burnable absorber level.
INT:	Interested, including some early analysis.
na:	Not applicable.
Nc:	Not being considered but potentially applicable.

Table III. Use of BUC for Transport (Transport Casks)

Country	PWR	BWR	MOX (PWR)	WWER	RBMK	Reactor Types
Armenia	na	na	na	APU-0	na	WWER
Belgium	INT	na	na	na	na	PWR
Brazil	Nc	na	na	na	na	PWR
Bulgaria	na	na	na	INT	na	WWER
China	INT	na	na	na	na	PWR
Czech Republic	na	na	na	RR-2	na	WWER
Finland	na	INT	na	INT	na	WWER, BWR
France	APU-1, UD-2	Nc	UD-1,2	na	na	PWR
Germany	APU-1, RR-2	Gd	RR-2	Nc	na	PWR, BWR, WWER
Hungary	na	na	na	INT	na	WWER
Japan	INT	INT	INT	na	na	PWR, BWR
Korea, Republic of	INT	na	na	na	na	PWR
Lithuania	na	na	na	na	INT	RBMK
Mexico	na	Nc	na	na	na	BWR
Netherlands	APU-1	na	na	na	na	PWR
Russian Federation	na	na	na	APU-1	INT	WWER, RBMK
Slovakia	na	na	na	UD-2	na	WWER
Slovenia	Nc	na	na	na	na	PWR
South Africa	Nc	na	na	na	na	PWR
Spain	INT	INT	na	na	na	PWR, BWR
Sweden	Nc	Nc	na	na	na	PWR, BWR
Switzerland	APU-1	INT	INT	na	na	PWR, BWR
Ukraine	na	na	na	INT[2]	INT[2]	WWER, RBMK
United Kingdom	RR-1	Nc	Nc	na	na	PWR
United States of America	APC-1, UD-1,2	INT	INT	na	na	PWR, BWR

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

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

2.5 Disposal

Reprocessing of fuel eliminates BUC for disposal. Also, fuel consolidation eliminates the need for consideration for BUC while the container is intact. The Czech Republic, Germany, and the Republic of Korea, have performed some analysis of BUC in disposal. The USA and Sweden have actively pursued BUC for disposal to cover failed containers that contain moderated fuel assemblies. The USA has submitted a risk informed topical report that includes actinides and fission products to its regulatory body describing a BUC methodology, and has received approval of the topical report.

3 GROUP DISCUSSIONS

3.1 Validation of codes and methods

Leaders:
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3.1.1. Introduction

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

- Codes and methods:
  - codes schemes and nuclear data libraries used by participants,
  - process for verification and validation,
  - use of validation data to derive methodology/code bias and uncertainty.
- Comparison of validation experiments with BUC validation requirements:
  - status of current validation requirements & current validation data set,
  - comparison of range of applicability with current requirements,
  - future validation requirements and future validation experiment programmes.
- Dissemination of information through criticality community:
  - improvements in nuclear data,
  - developments in code methods,
  - experimental programmes.

The main conclusions of the group discussions are summarized and recommendations arising from the discussions are recorded.

3.1.2. Codes and methods

3.1.2.1 Summary of codes and nuclear data used by participants

A summary of information on codes used by participants is presented in Table IV. Additional information on codes used for BUC calculation can be found in section five of IAEA-TECDOC-1013 and on the following websites:
http://www.nea.fr/html/dbprog; http://www-rsicc.ornl.gov; http://wastenot.inel.gov/icsbep/

Overall it is seen that there is a wide range of techniques including deterministic and Monte Carlo methods and covering a range of energy group schemes. The group discussed the various ways in which these codes are applied to BUC analyses and how the accuracy of their predictions are tested through a process of verification and validation. The results of that discussion are presented below.

Table IV. Codes used for BUC Calculations

Country/Organization	Depletion		Criticality	
	Code	Data	Code	Data
Belgium/Tractebel	LWRWIMS	UKNDL	MCNP	ENDF-BV CE[+]
Belgium/BN	WIMS8	JEF2.2	KEND5A DANTSYS	JEF2.2 172g
Bulgaria/WWER-440	SCALE4.4	ENDF-BV	SCALE4.4	ENDF-BV 44grp
Bulgaria/WWER-1000	NESSEL & NUKO	ENDF-BIV	SCALE4.4	ENDF-BV 44grp
Czech Republic/NRI	SCALE4.4	ENDF-BV	SCALE4.4	ENDF-BV 44grp
Czech Republic/SKODA JS	WIMS 7	JEF2.2	MCNP	ENDF-BV/BVI
France	DARWIN-APOLLO2	JEF2.2	CRISTAL	JEF2.2 172grp
Germany/FRA (PWR)	CASMO3 SCALE4.4	ENDF-BIV ENDF-BV 238grp	SCALE4.4 MCNP	ENDF-BV 44grp ENDF-BV/BVI
Germany/FRA (BWR)	CASMO4	ENDF-BVI	SCALE4.4	ENDF-BV 44grp
Germany/WTI (PWR)	SCALE4.4	ENDF-BV	SCALE4.4	ENDF-BV
Germany/GRS	KENOREST-98	JEF2.2	KENOREST-98	JEF2.2
Japan/JAERI	SRAC	JENDL3..2	VIM	JENDL3..2 CE
Japan/NUPEC	SRAC SCALE4	JENDL3..2 ENDF-BV	MVP SCALE4.4	JENDL3..2 CE ENDF-BV 44grp
Korea, Republic of	*			
Russian Federation	KASSETA	BROND	MCU	BROND
Slovakia	SCALE4.4	ENDF-BV	SCALE4.4	ENDF-BV 44grp
Spain/ENUSA (PWR)	PHOENIX	ENDF-BV	SCALE4.3	ENDF-BV 44grp
Spain/ENUSA (BWR)	TGBLA	ENDF-BV	SCALE4.4	ENDF-BV 44grp
Spain/CSN	CASMO3	ENDF-BIV	SCALE4.4	ENDF-BV 44grp
Switzerland/PSI	BOXER	JEF1	BOXER	JEF1 20grp
United Kingdom/BNFL	WIMS8 FISPIN	JEF2.2	WIMS8 MONK8	JEF2.2 CE
Ukraine	*			
United States of America /Westinghouse	PHOENIX	ENDF-BV	KENO	ENDF-BV
United States of America	SCALE4.4 CASMO4 HELIOS	ENDF-BV ENDF-BV	SCALE4.4 MCNP	ENDF-BV 44grp ENDF-BVI CE

+ CE – Continuous Energy
* No group participant available

3.1.2.2 Verification and validation

The first step in establishing the accuracy of a code/data package is usually based on numerical testing of the code and its nuclear data library. This normally includes checks to ensure that the processed cross-sections accurately represent the information contained in the basic nuclear data file. Verification of the code may then be made by comparison with reference methods, such as continuous energy Monte Carlo. Once these functional tests have been made, the accuracy of the code/data package is validated through a series of calculations for a set of benchmark experiments. In the context of BUC, these generally consist of post-irradiation examination (PIE) data (to validate depletion calculations) and critical benchmark experiments (to validate criticality, i.e. k-effective calculations).

Verification of the codes is usually carried out by the code developer, often in the context of a formal QA framework. Typically the QA programme will include a detailed procedure for identifying and correcting faults and for notifying the user community of such faults and any resulting corrections/improvements.

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

The 'pros and cons' of the two approaches were discussed by the group. Some advantages of the first approach are summarized below:

- Pre-validation of a well-defined route permits the use of specialist expertise, this may be important for verifying and validating some of the more complex calculation functions associated with BUC (e.g. depletion). For BUC these tasks can require a large technical effort.
- A more closely defined scheme may be less vulnerable to user error.
- By adopting a coordinated approach, current developments in codes and data may be introduced efficiently and in a consistent manner.
- Specialists of neutronics codes use the synergy of criticality-safety and reactor physics studies; furthermore, they generally participate to the design of the validation experiments and are consequently well prepared to the measurement result analysis.

On the other hand:

- BUC requirements vary considerably and so will demands on codes' accuracy. Thus a single, tightly defined code scheme is not necessarily the most efficient way of providing a fit-for-purpose calculation capability.
- Not all countries have the resources necessary to maintain a specialist validation team.
- Code licensing arrangements and regulatory requirements vary considerably.

More generally, the following conclusions were drawn:

- As a result of the increased validation requirement for BUC, there are significant advantages in the use of modern codes and data packages that have been verified and validated through coordinated programmes.
- For some BUC application areas validation data is limited, so that the use of approximate methods, which may contain compensating errors, can lead to significant extrapolation uncertainties.

3.1.2.3 Validation experiments & methodology

Participants summarized the types of experiment used to validate their codes and the techniques applied to derive code bias and uncertainty. In general PIE data are used to validate depletion calculations and critical experiments (e.g. zero-power critical benchmarks) are used to validate the criticality calculation code. Participants also noted that evidence of code accuracy might be deduced from comparisons with boron letdown during the fuel cycle in PWRs, although it is recognized that there are difficulties in applying reactivity effects in a hot reactor core to an accurate derivation of code bias for spent fuel environment.

In most countries depletion calculations have been validated against PIE data from public domain or proprietary programmes. An important exception arises in Eastern Europe where there is an acute shortage of PIE data for validation of WWER depletion calculations. The meeting recommended that this type of data should be acquired as a high priority. It was also noted that studies should be made, possibly by making comparisons with reference depletion methods (e.g. continuous energy Monte Carlo), to assess the applicability of ‘square-lattice’ PIE data to WWER methods validation. Potentially, public domain data could then be used to validate these codes.

In discussing validation methods currently employed by various countries and organizations, two basic approaches were identified. In some cases code bias and uncertainty are derived from consideration of global reactivity effects where bias contributions from individual isotopes may be positive or negative (i.e. errors may cancel). Alternatively other countries derive bias by summing contributions by individual isotopes, all of which are treated in a conservative fashion, and reduction of bias by cancellation between isotopes is not permitted. It is noted that the presence of possible canceling errors in the former approach implies a strong reliance on well-correlated benchmarks to avoid extrapolation uncertainty. On the other hand it is also recognized that the ‘isotope-by-isotope’ approach may require more analytical effort and produces larger penalties in safety criteria.

For some applications (e.g. actinide-only credit in spent fuel pools) only part of the reactivity loss with burnup is credited. For these types of case, where there is significant conservatism in the methodology, some participants felt that it is not necessary to employ an overly-cautious method of estimating bias and the global approach may therefore be justified. If fuller forms of BUC were invoked use of the more conservative isotope-by-isotope method, might be applied.

3.1.3. Status of validation programmes

3.1.3.1. Current position

Several of the group participants have been involved in experimental programmes aimed at providing benchmark validation data for BUC. Descriptions of existing and future programmes were presented to the group.

In Europe, current programmes include REBUS (PIE + spent fuel reactivity), ARIANE (PIE for high burnup BWR, PWR, MOX) and PROTEUS (PIE + spent fuel reactivity + individual fission product cross-section) (see Grimm, et al, paper on PROTEUS in Session 2.1, Paper No. 3). Several countries also have their own PIE programmes. Some spent fuel and separated fission product reactivity experiments have been carried out in Japan and similar experiments have been proposed in the USA. Japan has also led an initiative to establish a database of PIE measurements for a wide range of reactors. The database, SFCOMPO is available to members of the OECD through the Nuclear Energy Agency (see http://www.nea.fr/html/dbprog)

Several participants pointed to the use of benchmarks from the ICSBEP handbook. Fresh MOX criticals from the dataset are used to validate criticality calculations for Actinide-Only BUC.

Participants compared existing validation data with current BUC requirements. With the exception of the lack of PIE for WWER systems, most participants considered that their current requirements were reasonably well covered, although in some cases only partial credit is being claimed for a limited range of applications, such as close-to-reactor pool storage. In these cases it is recognized that further development of BUC methodologies will require more extensive validation. In some cases PIE data are a little limited for higher initial enrichment (IE), but it is noted that no trend with IE is seen for many code/data schemes. Generally it is concluded that coverage of burnup is the more important parameter. The most complete validation coverage appears to be in France where existing experiments allow BUC up to 4.5w/o IE and 60GWd/t burnup for PWR fuel.

3.1.3.2. Future requirements

The main future developments in BUC are anticipated to arise in the following areas:

- Increase in initial enrichment and burnup for PWRs,
- Development of BUC for BWRs,
- Development of BUC for MOX fuels,
- Move to full BUC credit, including fission products.

Generally requirements for validation of depletion calculations are being addressed through extensions of existing PIE programmes (e.g. REBUS, ARIANE) or new programmes such as MALIBU. Validation of global reactivity loss and the worth of grouped or individual fission products has been made in the CERES programme. Further measurements are planned in the REBUS and PROTEUS programmes. It is noted that these international co-operations are of a proprietary nature and the data will not, initially at least, be openly available. However feedback of the benchmark results is expected through contributions to international conferences and through international benchmarking programmes such as the JEFF project. By this means, an indication of the performance of some widely used code/data packages for newer applications is made available. Code users not directly involved in the validation programmes also derive benefit from the improvements to methods and data that arise from detailed analysis of the benchmarks by those having full access to the experimental data.

It is noted that validation for BWR and MOX fuels is somewhat more limited than for PWR. Programmes to improve coverage of these fuel types are underway and/or planned, particularly in Europe. At this stage however the immediate BUC requirement for these fuels is limited.

3.1.4. Lessons learnt

Several national and international activities have been identified that provide a valuable means of disseminating information and helping to ensure that important lessons learnt during experimental, code development, nuclear data and benchmarking programmes are communicated through the technical and regulatory community. Some current items have been discussed by the group and the conclusions are presented below. It is noted that BUC is a relatively novel and still developing methodology, so access to (and the support of) these activities can play an important role in developing and maintaining technical expertise and methodologies.

In recent years there has been a continuing reliance on international co-operation in the area of nuclear data evaluation (e.g. OECD-NEA Working Party for International Co-operation on Nuclear Data Evaluation, JEFF Project) [1], partly to pool diminishing resources. As noted above, this provides an important mechanism for feeding back the lessons learnt from new validation programmes (many of which are proprietary), through improvements to nuclear data libraries.

Other international bodies such as the OECD-NEA BUC (see Brady Raap, et al, Paper in Session 1, Paper No. 2) benchmarking group provide a valuable forum where participants can exchange information and experience related to code performance. Information on code development and testing is most often generated by the various code development teams and tends to be disseminated through newsletters, conference papers etc.

In addition to the general dissemination of BUC-related information noted above, some specific items were discussed:

- Design and evaluation of benchmark experiments – the advantage of 'clean', well defined benchmarks was noted. In particular the experiment should be amenable to calculation without significant modelling approximations or assumptions and should include a complete and rigorous assessment of experimental uncertainty. The benchmark evaluation process used in the production of data for the ICSBEP handbook is felt to be an example of good practice in this area. The group also commends the application of the treatment of experimental uncertainty developed as part of ICSBEP activities [2];
- Fission Product PIE – there have been some examples of inconsistent PIE results for some fission products. Chemical separation of spent fuel is known to be a delicate process, particularly for metallic fission products. Independent cross-checks of PIE measurements are therefore desirable;
- Recent studies on depletion calculations for WWER have highlighted the need for condensation spectra specific to that reactor type, rather than reliance on existing standard neutron spectra;
- Experience with benchmarking of JEF libraries [1] has highlighted the importance of energy group structure at high burnup. Resonance shielding effects for Pu-240, Pu-242 and some fission products can cause significant discrepancies if not treated properly.

3.1.5. Conclusions and recommendations

The main conclusions of the group discussion are as follows:

- As a result of the increased validation requirement for BUC, there are significant advantages in the use of modern codes and data packages that have been verified and validated through coordinated programmes;
- For some BUC application areas validation data are limited, so that the use of approximate methods, which may contain compensating errors, may lead to significant extrapolation uncertainties;

- BUC is a relatively novel and still developing methodology; access to (and if possible, the support of) international activities on nuclear data, methods development and code benchmarking can play an important role in establishing and maintaining technical capability;
- The advantage of well-defined benchmarks is noted. In particular experiments should be amenable to calculation without significant modelling approximations or assumptions and should include a thorough assessment of experimental uncertainty. The benchmark evaluation process used in the production of data for the ICSBEP handbook [2] is felt to be an example of good practice in this area. The group also commends the application of the treatment of experimental uncertainty developed as part of ICSBEP activities.

In addition the group recommends that the acquisition of PIE data for the validation of WWER depletion calculations be given high priority.

3.2 Key issues in BUC

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3.2.1 Introduction

The use of BUC in the criticality safety analysis of spent fuel operations is based on a fundamental set of principles. These principles include representing the composition of burned fuel (depletion analyses) and the effect of that composition on the effective multiplication factor (criticality analysis). The sensitivity of these analyses to reactor operations history is to a large degree dependent on the specific application. The key applications include the combination of fuel properties and physical characteristics of the surrounding material (e.g. storage rack, transportation cask/flask, etc.) as well as consideration of operator mistakes and incidents.

This paper reports the results of a working group investigating key issues related to the use of BUC. The primary issues addressed were (1) the impacts of depletion and decay parameters, (2) an approach for incorporating bias and uncertainties, (3) the effects of variations in plant operations and fuel design, including the variability of parameter importance for different spent fuel applications, (4) the importance of axial and horizontal burnup profiles, and (5) impacts to criticality analyses beyond requirements for fresh fuel assumption. The group also discussed whether there is a need to evaluate the impact when there are inconsistencies in the nuclear data used for the two types of analyses (isotopic predictions and criticality evaluations). No conclusions were reached as to the importance of this issue and it was therefore not included as a key issue.

The prioritization of the key issues and the level of detail required are determined by the graded approach to the advantage of BUC being pursued. It has also been suggested that the degree of reliance on BUC to prevent a criticality accident should be considered in determining priorities and the level of detail required in the evaluations. For instance, BUC can be used as a primary or secondary (relative to moderator exclusion or presence of soluble poisons) control in assuring criticality safety.

The primary levels of BUC include the following:

1. Most reactive condition of the fuel over its lifetime (fresh fuel or limited burnup to offset the effect of absorber materials),
2. Net fissile depletion (U-235, Pu-239 and Pu-241),

3. Actinide only,
4. Selected actinide + FPs,
5. Best estimate.

In addition to the levels above, the graded approach is also affected by the method used for assigning the burnup value for assessment and development of the loading curve. These methods include using a burnup value determined from the most reactive region of the fuel (considering axial and horizontal variation), from a fixed length of fuel having the least irradiation, reactor record average value, independent measured value, or based on fuel management calculations for burnup based on operating history information. The burnup value is also a consideration in determining the requirements for verification and measurements during implementation and as such will also be addressed by the working group assigned to investigate *Safety assessment and implementation*.

3.2.1.1 Understand, determine, and consider impacts of depletion parameters

Application of BUC requires calculation of the isotopic inventory of the irradiated fuel for which BUC is taken. The depletion calculation simulates the burnup of the fuel under reactor operating conditions. The result of the depletion analysis is the predicted isotopic composition of the discharged spent fuel. This composition is a necessary input to the criticality analysis. The following factors influence the isotopic inventory:

- Fuel design and degradation and degree,
- Reactor design,
- Reactor operating conditions (including decay time and downtime between cycles).

Major effects resulting from these factors are:

  - Neutron energy spectrum (e.g. hardening increases density of higher actinides),
  - Specific power (including downtime/decay time) influences the production and loss of nuclides.

The following parameters are identified having potential influence on the isotopic predictions:

- Solid absorbers:
  - Control rods,
  - Axial power shaping rods (part-length),
  - Fixed burnable absorbers (removable),
  - Integral burnable absorbers (non-removable from fuel);
- Soluble boron,
- Fuel temperature,
- Moderator temperature and density,
- Specific power and operating history (e.g. decay time between cycles):
  - Magnitude,
  - Time dependence.
- Cooling time after discharge,
- Interaction between MOX and UOX assemblies,
- Assembly and rod deformation:
  - affects the water gap between fuel assemblies-power peaking,

  - impact of assembly water gap variation on nuclide prediction (and subsequent k_{eff} value) has been demonstrated for WWER-1000 reactors (paper by Kovbasenko), not studied in other reactors.
- Assembly material properties and dimensions:
  - Initial enrichment and density,
  - Pu vector for MOX assemblies,
  - Fuel and clad radius,
  - Rod pitch,
  - Axial blankets.
- Determination of important nuclides and selection of nuclear data.

The prioritization and magnitude of the effects of these parameters are dependent on the specific case analyzed (e.g. fuel assembly design, reactor type, nuclides considered, etc.).

Predictions of nuclide inventories as a function of time in repository post-closure conditions are more complicated since they must be coupled with decomposition scenarios and geochemical effects.

3.2.1.2 Incorporating bias and uncertainties

The assumption was made that the working group assigned to investigate *validation of codes and methods* will address how the bias and uncertainties related to validation are determined, the scope of this discussion is limited to how these values are incorporated into the isotopic predictions and criticality evaluations.

The bias and uncertainty considered in the development of the loading curve for BUC consists primarily of two components - the contribution from the depletion validation and the contribution from the criticality validation. Other potential sources of uncertainty noted by the working group were those associated with the burnup value and the depletion parameter values used in the prediction of the nuclide inventory. Each of these areas will be discussed briefly.

Applicable critical experiments have been the traditional means for validating codes used for predicting the neutron multiplication factor, k_{eff}. Incorporating the bias and uncertainty obtained by comparing these experiments to calculated values is normally preferred. However, with BUC, the number of applicable critical experiments may not be as numerous as desired. Thus, the bias and uncertainty established by comparison to worth and subcritical measurements may be needed. Deriving bias and uncertainty values from measurements instead of calculations is particularly important as the degree of BUC moves from a more restrictive state (e.g. actinide only) to a less restrictive state (e.g. actinide plus fission products). Even if not incorporated in the bias and uncertainty estimate, the worth and sub-critical measurements may provide increased confidence in the adequacy of the calculated bias and uncertainty value.

The depletion validation has traditionally been done by comparison with measured PIE data obtained for each nuclide of interest. The approach for incorporating the bias and uncertainty from the depletion validation has varied with the degree of BUC sought. For example, in France the burnup associated with the 50 cm length of least irradiation has been used to establish the burnup value for the safety assessment of PWR fuel. This burnup value in combination with an actinide-only approach was deemed sufficiently conservative that incorporation of a bias and uncertainty for the depletion analysis was not required. Similarly, in past applications in Korea, the burnup from the reactor record was adjusted downward by 15% and the conservatism in this approach was judged sufficient to eliminate the need for incorporation of the bias and uncertainty from the prediction of the nuclide inventory.

In contrast, the current approach being considered by France appears to be always conservative, though in most cases, is less conservative than past practice using the fresh fuel assumption and does incorporate the bias and uncertainty from depletion validation with PIE data. These nuclide bias and uncertainty values are dealt with individually in a bounding manner when preparing the nuclide inventory for the criticality analysis. Other approaches considered, or in use, apply a methodology that seeks to estimate the delta-k impact due to the combined bias and uncertainty of all the nuclides being used in the criticality analysis. These approaches seek to propagate the uncertainty from the total nuclide inventory through to the calculation of k_{eff}. In Korea, such an approach has been used. The more bounding approach to consider the bias and uncertainty individually was not deemed necessary because of the conservative assumptions applied in other portions of the overall BUC approach.

The loading curve for a particular configuration is typically generated using a specified burnup value. The nuclide inventory for this specified burnup level is obtained using depletion parameters established by the analyst. The uncertainty associated with the depletion parameters have been best estimate values or values demonstrated to be bounding relative to prediction of the k_{eff} value. The working group judged that bounding values should be incorporated in the depletion analysis rather than best estimate. However, any uncertainty associated with the selection of a bounding value was felt to be sufficiently small that it could be ignored - assuming proper effort had been made to achieve actual, but realistic bounding values.

If specified burnup values are used to generate the loading curve, then the incorporation of an uncertainty in the burnup value is mostly a matter of implementation, which was considered by another Group. However, this working group did note that the manner in which the uncertainty in the burnup is considered in implementation does vary among Member States. For example, future applications in Korea will use the actual burnup from the reactor record as a satisfactory means for determining whether an assembly can be placed in a particular location in the pool. This approach will be used because it was determined that sufficient conservatism is included in the generation of the loading curve. For pool storage in the US an uncertainty established for the reactor record must be considered. However in France, a measurement is required for higher burnups and the measurement uncertainty is incorporated, but not the reactor record uncertainty. In the US, the guidance for transport and dry cask storage calls for use of a measurement where the uncertainty from the measurement and the records must be considered in establishing acceptability for loading.

3.2.2 Accounting for variation in plant operating history, fuel design and spent fuel operation

Variations in plant operating history, fuel design and the spent fuel operation being evaluated have the potential to introduce perturbations to the neutron energy spectra – either during operations or in the spent fuel environment. These effects can alter the nuclide inventory at a given burnup and consequently the reactivity of the spent fuel condition. Likewise, conditions of the spent fuel application can directly impact reactivity. Any best estimate analysis of BUC will need to assess these affects in detail. Lesser degrees of BUC require analyses detailed enough to ensure that claims of "bounding" or "conservative" can be justified. One can expect that the relative importance of these parameters will change for specific scenarios. Examples of concerns related to problem specific impacts include:

- Assess the most reactive assembly type and condition for each scenario;
- Assess impact of combining different fuel designs as complicated by BUC;
- Assure normal and off-normal effects are included in operating history;
- Assess the relative importance of nuclides for the following applications:
  - UOX,
  - MOX,
  - Storage and transportation,
  - Repository,
  - Reprocessing.
- Depletion parameters (e.g. moderator density):
  - PWR vs. BWR.

It is necessary to demonstrate an in-depth understanding of the system being evaluated and to consider that even local effects may impact some parameters in a non-conservative direction related to generic conditions accepted as bounding.

3.2.2.1 Axial and horizontal burnup profiles

The influence of the axial burnup profiles is generally called the "end effect" because the top and bottom ends of the fuel assembly have lower burnup mainly due to neutron leakage. Differences between the top and bottom are due to the use of control rods and moderator density differences for PWRs. For BWR, the profile is influenced additionally by enrichment and void fraction profiles.

The axial burnup profile must be considered because k_{eff} calculated with a real axial profile may be higher than with the assumption of a flat one (i.e. assuming the average burnup for the assembly). The importance of axial burnup profile is dependent of the specific configuration and could be increased when these less burned ends come close together in an array configuration.

The axial burnup profile depends on the reactor operational conditions. In order not to be overly conservative, a database of axial profiles specific for these reactor-operating conditions should be available from measurements (in-core and/or out-of-core measurements of profiles). A bounding profile or reactivity correlation (based on minimum burnup or preferably reactivity effect) may be derived from this database in such a way that using this bounding condition (i.e. profiles or reactivity correlations) ensures sub-criticality at the required level of confidence. The important issue is to have measurements (either from a database or specific measurements) to verify that the assumed axial distribution used in analyses is satisfied in reality. If not, the most reactive state of the fuel should be assumed. If the assumption about the spent fuel requires the definition of a bounding axial profile (i.e. the assumption of a flat profile is demonstrated not to be conservative), also this axial burnup profile must be shown to be conservative by a pre-shipment measurement. The details of the measurement process and requirements will be addressed by the working group investigating *safety assessment and implementation.*

Some accident conditions may change the geometry of the transport or storage system. During the criticality analysis these changes have to be considered. Particular attention has to be paid to the less burned ends.

Generally the influence of the horizontal variation of the burnup is much less than the influence of the axial variation. Significant horizontal burnup profiles are possible for fuel assemblies that may have been inserted close to the periphery of the reactor core, next to a different type of fuel (MOX or UO_2) or in the case of the WWER, close to a control assembly. The effect of these horizontal distributions should be addressed for these situations.

Every criticality study relies on both a conservative assumption about the composition of the spent fuel and the spatial variation of this composition.

3.2.2.2 Criticality analysis modeling (consideration beyond fresh fuel assumption)

Issues associated with the criticality analysis modeling are principally the same as for the analyses using the fresh fuel assumption for out-of-core spent fuel management. Established good practices for developing nuclear criticality safety evaluations should be followed. There are two notable differences that require special attention in these evaluations. First, due to the axial distribution of burnup and therefore nuclide inventories across the fuel, there is a greater tendency for BUC calculations to suffer complications related to source convergence issues. Since the fuel is underburned at both ends, the problem is one of a loosely coupled system. The standard methods of addressing source convergence should be used as needed. These methods include careful observation of the results to assure convergence, increasing the number of neutron histories and varying the starting distribution of neutrons. Secondly, the increased complexity of the material descriptions required to represent the nuclide inventories require additional modelling detail of fuel regions to account for burnup distributions. In some cases this is best addressed by developing a systematic approach to interface the criticality code with the depletion code. Updates and improvements to the nuclear data libraries may also be required to accommodate use of these nuclides. In fact, even the determination/verification of which nuclides are important to be considered in the analyses may be specific to the application (e.g. curium isotopes are extremely important in evaluating BUC for MOX fuel and have little or no importance for UO_2 fuel in LWRs).

3.2.3 Conclusions and recommendations

BUC is neither the classical out-of-reactor criticality analysis nor is it a typical fuel management assessment. Both of these technical areas are well established and have many lessons learned to offer to the analyst/engineer assessing BUC. Bringing these two engineering disciplines together is a scientific problem requiring study and understanding of the associated physics. The basic requirement is to demonstrate an understanding of the problem that is to be solved. The level of detail required to do this is related to the amount of credit one is seeking to obtain as well as that which can be justified based on independent experimental and operational experience.

The key issues addressed in this document are based on the experience and lessons learned from a number of experts with considerable experience in BUC and/or plant operations. These key issues should provide some insight to the BUC analyst as to the level of detail that would be required in a best estimate analysis of BUC. Analytical experience in specific applications is the best guidance in establishing an acceptable technical defense for conservative models seeking a lower level of BUC (i.e. fissile depletion, or some form of partial BUC such as actinide only, etc.). Based on the experience of the analyst and/or regulator, either qualitative or quantitative arguments are acceptable in the justification of the relative importance of these parameters to the overall assessment of BUC.

The working group concluded that training workshops for BUC should continue to be organized and presented. There was discussion among the group that it might be reasonable to organize a meeting jointly with experts from fuel cycle management and in-core fuel analyses to discuss issues associated with the depletion parameters. Some felt that this could be problematic if one tried to separate the topics of depletion and criticality since qualifying the importance of the depletion parameters must be kept within the context of the criticality analysis.

The working group also concluded that a document should be produced summarizing and making available the BUC regulatory requirements in different member states (e.g. for storage), and transportation including status of any requirement for measurements). This report could be based on the current information compiled in the questionnaire circulated prior to this meeting and confirmed during the working groups that indicates where member countries have approved BUC or are considering BUC.

3.3 Safety assessment and implementation

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3.3.1 Introduction

Application of BUC to a spent fuel management system consists in implementation of the following key steps:

- Safety assessment of the system:
  - Prediction of the spent fuel composition under bounding depletion conditions,
  - Criticality calculation and evaluation of the fuel loading criterion;
- Application of the fuel loading criterion:
  - Quantification and verification of the burnup of the fuel to be loaded in the system,
  - Fuel loading procedure-assuring compliance with the loading criterion.

The evaluation of the loading criterion is based on a criticality safety acceptance criterion. Therefore, the first item discussed by the working group was the topic "*safety criteria*".

The procedure employed to load the fuel in the spent fuel management has to ensure that the loading criterion is met. Thus, the second item discussed by the working group was the topic "*fuel and loading verification*".

It is inherent to BUC that the loading criterion is dependent on the reactor operation conditions assumed in the depletion calculations performed to predict the isotopic inventory of the spent fuel. It is inherent to the implementation of BUC that fuel and loading verification introduces the need to rely on complex calculational methods and operator control as well as complex measurement checks that may be additionally necessary. The need for relying on the knowledge of reactor operation conditions, complex calculational procedures, operator control and possibly additional measurement checks introduces local hazards specific to BUC.

Therefore, the third item discussed by the working group was the topic "*risk, its perception and building consensus*".

The necessity to rely on the reactor operation conditions on which the depletion analysis was based makes it necessary to ensure the "*continued validity of the analysis assumptions*". This item was therefore the fourth topic discussed by the working group.

3.3.2 Safety criteria

The working group has come to the conclusion that there is one *basic* safety criterion:

- The evaluated neutron multiplication factor k_{eff} of the spent fuel management system of interest shall not be greater than an upper limit k_{limit}, including all statistical, mechanical and calculational uncertainties Δk_U.

In order to meet this criterion *safety parameters* have to be derived, i.e. conditions have to be specified with which the spent fuel system must comply. In BUC applications such safety parameters are usually defined by meeting the reactivity equivalence relation

$$k_{eff} + \Delta k_U = k_{limit} \qquad (1)$$

and are usually given in the form of *loading curves.*

A loading curve specifies the loading criterion by indicating, e.g.

- The minimum burnup necessary for the spent fuel with a specific initial enrichment to be loaded in the system of interest, or
- The maximum residual enrichment or the maximum heavy metal mass allowable for the spent fuel with a specific initial enrichment to be loaded in that system, or
- Any other adequate bound for the spent fuel to be loaded in the system.

It is necessary to gain assurance that the spent fuel selection is made to ensure that this fuel satisfies the loading criterion.

The upper limit k_{limit} of the allowable neutron multiplication factor:

- either is an administratively fixed value,
- or depends on conditions given by the spent fuel, the spent fuel management system, the conservatism maintained in the criticality safety analysis, or the method applied to the verification of the spent fuel selection.

In the latter case, there always is, of course, a regulatory upper bound of k_{limit} in order to ensure sufficient sub-criticality.

In case of the application of administratively fixed k_{limit} values the value prescribed for normal operating conditions may be different from the value prescribed for abnormal or accident conditions. Table V gives a survey of the upper limits k_{limit} applied in the different countries represented in the working group. As can be seen, in most of these countries the upper limit k_{limit} is unique, i.e. k_{limit} is specified by a point value. In some countries the k_{limit} value that applies to accident events is case dependent, the applicable upper limit k_{limit} may depend on:

- the probability of occurrence and the consequences of such an event,
- reliance on human actions to maintain safety (e.g. to prevent a misloading event) and, in this conjunction,
- operator experience,
- the quality of the validation of the calculation codes used to evaluate that event,
- the use of a calculational model that is demonstrated to be bounding,
- sensitivity of the reactivity of the system on changes of operational conditions.

In countries without administratively fixed k_{limit} values, the maximum upper bound allowed for the limit k_{limit} depends on:

- the quality of the validation of the calculation codes applied;
- the conservatism maintained in the analysis concerning:
    - the BUC level used (actinide-only or actinides plus fission products level, e.g.),
    - the depletion parameters assumed to bound the variations in the fuel irradiation histories,
    - the design parameters used,
    - the cooling time assumed;
- the quality of the verification of meeting the loading criterion (measurement of burnup before loading, performed in order to check the burnup obtained from the reactor records, might allow to apply a higher upper limit k_{limit}); and
- as regards accident events, the probabilities of occurrence and the consequences of such events.

In France, for instance, application of the actinide-only BUC level may allow the use of a higher value for the upper limit k_{limit} due to the actual presence of fission products in the spent fuel, and it is not required to specify the reactivity margin, which is due to the presence of the fission products. The applicant has to give the proof of sufficient subcriticality by demonstrating that a sufficient *safety margin*

$$\Delta k_{SM} = 1 - k_{limit} \tag{2}$$

is maintained. The k_{limit} value allowed to be used, takes into account the conservatism of the analysis assumptions and the confidence in the codes used for the BUC criticality analysis. Therefore, use of an extended BUC level has to include a re-evaluation of

- The conservatism level of the analysis assumptions; and
- The confidence in the codes (code qualification).

In the United Kingdom, choice of the upper limit k_{limit} is aimed to minimize the risk of a criticality accident. The upper limit k_{limit} is therefore chosen on a case-by-case basis to be appropriate to the nature of the proposed operations, the complexity of the plant and to include other factors such as code validation, operating experience and ease of demonstrating compliance with key safety requirements.

Table V. Upper limits k_{limit} of the neutron multiplication factor in BUC applications in different countries (Figures in italics: non-administrative values)

		Armenia *	Bulgaria *	Czech Rep. *	China	France	Germany	Hungary *	Lithuania *	Slovakia *	Spain	UK	USA
Normal conditions	Wet storage	0.95	0.95	0.95	0.95	*0.95*	0.95	0.95	0.95	0.95	0.95		0.95
	Dry Transport/ Storage	0.95	0.95	0.95	0.95	*SP: 0.95* *PA: 0.98*	0.95	0.95	0.95	0.95			0.95
	Disposal	-	-	-	-	-	0.95 *	-	-	-			0.98 *
	Dissolver (Reprocessing)	-	-	-	-	*0.97 - 0.98* (case dependent)	-	-	-	-		*0.92 – 0.98* (case dependent)	-
Accident conditions	Wet storage	0.95	0.95	0.95 (OM:0.98)	0.98	*0.95 - 0.97* (case dependent)	0.95 – 0.98 (case dependent)	0.98	0.95	0.98	0.95 – 0.98 (case dependent)		< 1
	Dry Transport/ Storage	0.95	0.95	0.95 (OM:0.98)	0.95 (??: 0.98)	*SP: 0.95* *PA: 0.98*	0.95	0.98	0.95	0.98			0.95
	Disposal	-	-	-	-	-	< 1.0*	-	-	-			0.98 or <1*
	Dissolver (Reprocessing)	-	-	-	-	*0.97 - 0.98* (case dependent)	-	-	-	-		*0.92 – 0.98* (case dependent)	-

OM = optimal moderation, * = proposed limits

SP = single package, PA = package array

3.3.3 Fuel loading and verification

3.3.3.1 General principles

Implementation of BUC requires a verification of the burnup level of each individual fuel to make sure it satisfies the loading criterion derived from the analysis. This verification can be performed by:

- using the information from the operational (or reactor) records,
- carrying out a measurement (qualitative or quantitative), or
- some combination of both.

Verification obviously should be performed before loading, storing or reprocessing the spent fuel.

Usually, verification is based on the reactor record information. Unless there are other means to ensure keeping of the loading criterion and prevention of misloading events, measurement may be required for checking:

- the reactor records and
- the assignment of the reactor record information to the fuel identification.

Because the measurement is aimed to minimize the risk of misloading events it must be part of the loading sequence. The technology for performing such a measurement is available [3].

The necessity of performing a burnup measurement to check the reactor record information is related to the conditions for which BUC is taken:

- If BUC is taken for normal operation conditions, the necessity of performing a measurement is high;
- If BUC is taken for abnormal operation conditions only, the necessity of performing a measurement becomes less;
- If BUC is taken for design basis accidents only, the necessity to perform a measurement is low or nil;
- If BUC is only necessary for severe accidents with a low probability of occurrence, no measurement is needed.

However, there may be a demand by the public and/or by the relevant regulators to perform checks of the burnup in order to minimize the risk of misloading events, whatever the conditions may be for which BUC is needed.

The accuracy of the burnup measurement is related to the BUC level needed:

- If the BUC taken is less than the burnup, which a fuel assembly gains in one cycle, only a qualitative measurement (e.g. a gross gamma measurement) is necessary giving the information whether the fuel to be loaded in the spent fuel management system of interest is irradiated or not;
- If a higher BUC level is needed, a quantitative measurement of the burnup is necessary.

The outcome of a quantitative measurement confirms the burnup information from the reactor records when the outcome is falling in an interval, named “acceptance interval” in the following, that depends on the burnup value obtained from the reactor records, the uncertainty of this value, the uncertainty of the measurement result, and the probability density of the difference between measurement result and burnup value from the reactor records. It might be possible, that measurement result and burnup value from the reactor records are correlated due to the fact that, dependent on the measurement procedure used, information from the reactor records is necessary for evaluating the measurement result. Such a correlation, if given, has to be taken into account in the estimation of the “acceptance interval”.

The loading criterion is met when the burnup value obtained from the reactor records (and confirmed by measurement if required) is greater than a lower burnup level which depends on the minimum necessary burnup specified by the loading criterion, the uncertainty of the reactor record burnup value and the probability density of the difference between reactor record burnup value and minimum necessary burnup specified by the loading criterion. In other words, the uncertainty of the reactor record information has to be linked to the safety criterion and hence the loading criterion (e.g. by multiplying the loading curve by a possibly burnup-dependent factor adequately representing the uncertainty of the reactor record information at a given confidence level).

A second possibility is, of course, to compare the outcome of the measurement that confirms the reactor record information, directly with the loading criterion. In that case the uncertainty of the measurement has to be linked to the loading criterion. However, a comparison of the measurement outcome with the loading criterion is usually significantly less powerful than a comparison of the reactor record information with the loading criterion, since:

- The precision of the measurement result is usually significantly smaller than the precision of the reactor record information, and
- Usually the measurement result cannot be treated as independent from the reactor record information due to the fact that the procedure to evaluate the measurement outcome relies on some information from the reactor records, and this may lead to a linking of the uncertainty of the reactor record information to the loading criterion in addition to the uncertainty of the measurement.

Therefore, usually the primary quantity is the burnup information from the reactor records. This quantity has to be compared to the loading criterion. Before that it may be required to confirm this quantity by measurement.

3.3.3.2 Specific fuel handling operations and the need for burnup measurement

As stated in section 3.3.3.1, unless there are other means to ensure keeping of the loading criterion and prevention of misloading events, the necessity of performing a burnup measurement is related to the conditions for which BUC is taken. Thus, the need for measurement depends on the specific spent fuel handling operation to be performed.

Spent fuel movement operations include:

- Movement of fuel within a power plant:
  - Transferring fuel assemblies from the core to the spent fuel pool,

  - Moving fuel between spent fuel pool locations;
- Loading of fuel into storage/transport casks or canisters and transferring fuel from storage/transport systems into another cask or back into wet storage;
- Loading fuel into the head end process (shear/dissolver) at the reprocessing plant;
- Loading fuel into waste packages for final disposal.

Movement of fuel within a power plant

In this case, reliance on knowledge of the fuel assembly basic parameters (initial enrichment, burnup, etc.) is based on reactor records. Administrative controls, such as verification of the identity, the initial enrichment and the burnup of the fuel assembly being moved, using at least two independent checks, are the subject of well-developed and mature procedures. A check of the reactor record burnup information by burnup measurement is not regarded as a practical option.

Administrative and physical controls of the pool water chemistry, such as boron concentration - if applicable - and verification of pool storage rack conditions - if required - are established to maintain a subcritical configuration in the spent fuel pool for all normal and design basis events.

In the USA and many other countries that apply BUC to PWR wet storage systems, the misloading event is usually regarded as a design basis accident event. As with any other design basis accident scenario, the double contingency principle is applied to misloading events. That means that no other *concurrent* event has to be postulated when considering the misloading event, and in particular, the possibility of a boron dilution event in a PWR fuel pool does not need to be considered. As a result, neutron absorption provided by the boron diluted in the coolant of PWR spent fuel pools can be credited to the extent guaranteed by the plant technical specifications.

However, in some countries such as Germany a different approach is used. In these countries the misloading event has to be excluded by application of the double contingency principle, i.e. the interpretation of the principle in this case is that two independent, unlikely and concurrent incidents have to happen before a misloading event can occur. Therefore, fuel handling procedures, based on technical measures employing hardware and software controls, have to be set up in such a way that the misloading event is ruled out by virtue of the double contingency principle [3, pp.274-279]. Thus, there is no need to consider the misloading event in the criticality analysis, irrespective of the presence of neutron absorber diluted in the pool water. This approach is used for the following reasons:

- If the misloading scenario is not ruled out by technical measures employing hardware and software controls, the root cause of a misloading event is always an operational error. Thus, if the event really does occur, there is a high probability of the error remaining undetected. A misloading event that remains undetected and a boron dilution event or any other design basis accident event that takes place at a later time cannot be regarded as “concurrent events”;
- In addition, since no system designed for BUC application can withstand the misloading of more than one fuel assembly if it cannot withstand the misloading of one fuel assembly, this event needs to be ruled out.

Loading of fuel into dry storage/transport systems

Loading of storage and transport casks or canisters is performed in the pool or in an area adjacent to the pool, i.e. in the presence of water (moderator). The presence of a moderator results in criticality considerations. Depending on the reactor type, the pool water can contain soluble boron. Further, storage-only purpose systems and dual-purpose and dry transport systems are considered.

Storage-only purpose systems

In the USA, the following approach is under discussion: For storage-only purpose casks, criticality safety is mostly considered during cask loading, as it is not an issue once the dry storage cask or canister sits on the storage pad. For the latter, the probability of events leading to re-flooding of the cask cavity is considered to be very low (Note: this may not be the case, if the storage area is located in a zone potentially susceptible to flooding). Therefore, in analogy to the movement of fuel within a power plant, when soluble boron is present in the pool water (such as for PWR spent fuel), loading can be accomplished through administrative controls only, and BUC does not trigger requirements for measurements. When soluble boron is not present in the pool water (such as for BWR spent fuel), BUC levels different from the integral burnable absorber level [4, pp.1] have not been implemented, as yet.

In other countries such as Germany, however, re-flooding of the cask cavity has to be considered as a design basis event. Therefore, the misloading event has to be ruled out by virtue of the double contingency principle, as described for the movement of fuel within a power plant. However, due to the fact that fuel-handling procedures designed for movement of fuel within power plants cannot fully be applied to cask loading, a check of the reactor record burnup information by burnup measurement is regarded as a must.

Dual-purpose and dry transport systems

For dual-purpose (storage and transport) and dry transport systems, criticality safety is considered:

- First, during loading because of the presence of water, as outlined above, and
- Second, in the context of potential accident conditions during transport, which would result in re-flooding of the cask cavity.

In France, BUC has been implemented for spent fuel transport to the La Hague facility. The BUC methodology currently used (actinide-only) consists of two levels of requirements depending on the results of the criticality analysis. If the burnup required is less than the minimum burnup that the reactor operator can guarantee after one cycle of irradiation, a qualitative (go/no-go) measurement is sufficient to prove that the fuel has really been irradiated. Else, a quantitative measurement is required. The measurement has to be made on the last 50 cm of the active fuel length that is the least irradiated, and the average burnup of these last 50 cm is allowed to be taken into account.

In Germany, the BUC methodology currently used (actinide-only) is based either on the French approach (for spent fuel transport to the La Hague facility) or a minimum average burnup of the fuel of 5 MWd/kgU for BWR fuel, if the initial enrichment is higher than 4.2

wt% U-235, and 10 MWd/kgU for PWR fuel, if the initial enrichment is higher than 4.05 wt% U-235. The respective minimum burnup value needed is guaranteed using the information from the reactor records and checked through the measurement of the γ dose rate of cesium to demonstrate that the fuel has been irradiated.

In the USA, the NRC Interim Staff Guidance (ISG-8) [5] ties acceptance of a BUC methodology (actinide-only) to the determination by measurement of each fuel assembly burnup before loading. This requirement has caused some concern in the USA that the impact on spent fuel pool operation and costs associated with this requirement may discourage the pursuit of BUC implementation by the U.S. operators. Those who are arguing against this requirement present the following rationales:

Operational experience indicates that reactor records provide fairly reliable information with regard to fuel assembly burnup characteristics. Deficiencies in manual data entry, and record archiving and tracking have been the main cause for observed or detected discrepancies between records and measurements. Therefore, the optimum balance among the operators' desire to minimize operational burden and costs associated with measurement, the benefits deriving from a reduction in the number of shipments, and the risks associated with potential assembly misidentification (misloading) remain to be optimized. Using a risk-informed (what are the risks?), performance based (what is the documented performance?) approach should determine the requirements, at least initially, for any measurement to support the implementation of a specific BUC methodology; any such requirement could be progressively relaxed after a sufficiently long and successful operational experience.

Preparation of spent fuel for reprocessing

Criticality safety is a major consideration in reprocessing facilities using a wet process given the presence of water and organic solvents (moderator), and the resulting separation of the uranium and plutonium species from the fission products. BUC is implemented for reducing analysis conservatism to avoid over-engineering of the facilities, and to keep up processing rates despite increases in initial enrichment of the fuel being reprocessed.

For these reasons, systematic burnup measurement is implemented to validate the data supplied by the reactor operators and set the criticality safety criteria for loading the dissolver. The data supplied by the operators are regarded as verified by burnup measurement when they fall in a specified interval (or "band", cf. Section 3.3.3.3).

Preparation of fuel for disposal

The U.S. research program for final disposal of fuel is using a risk-informed, performance-based approach for criticality (see W.H. Lake paper in Session 2.4, Paper No. 1). This approach includes BUC, and accounts for the key actinide and fission product isotopes important to reactivity. The presence of a moderator (water) cannot be ruled out over the time period during which the safety of the repository is assessed. However, if the probability of criticality falls below a regulatory threshold ($<10^{-4}$ in 10 000 years), no criticality consequence analyses should be performed. At the time of this writing, the applicant, i.e. the U.S. Department of Energy, does not plan to perform measurement when it takes ownership of the spent fuel. The applicant proposes reliance on reactors records.

3.3.3.3 *Uncertainty of the burnup data*

Studies sponsored by EPRI were performed to quantify the magnitude of uncertainties that can be present in burnup value estimates by a PWR utility for their discharged fuel. Uncertainties in average reaction rate in instrumented locations were found to be around 2.2%. The uncertainties of relative assembly power were found to be about 1.8%. Axial evaluation of reaction rate uncertainties indicated that for the top 20%, middle 60%, and bottom 20%, the uncertainty were 6.1%, 2.5%, and 6.7 percent, respectively. After 1 cycle, the assembly average burnup had a 1.5% uncertainty; after 2 cycles, the assembly average burnup had a 1.1% uncertainty, and after 3 cycles, the assembly average burnup had a 1.0% uncertainty (see A. Machiels et al., Paper in Session 2.1, Paper No. 9).

COGEMA’s experience indicates that the value of the burnup as measured at La Hague and the value of the burnup as reported by the French operators differ by a mean average of 5%, which is well within the 15% band constituting COGEMA’s acceptance criterion.

3.3.4 *Risk, its perception and building consensus*

Normal and abnormal operation conditions of a fuel management system as well as design basis accidents are demonstrated to be safe in the criticality safety analysis. Beyond design basis events should be assessed based on risk techniques.

The experience accumulated to-date in the field of fuel assembly handling shows that no criticality event has occurred. Given the high number of fuel assemblies handled every year, that means that the criticality risk has remained limited using the current safety analysis approach. Introduction of BUC may introduce additional local hazards such as the need to rely on knowledge of reactor operating conditions, complex calculational methodologies, operator control, additional measurement checks, etc. This may result in an increased risk that the actual neutron multiplication factor k_{eff} of the system exceeds the upper limit k_{limit}, (cf. Section 3.3.) because:

- The criticality safety analysis of the system might be not bounding with respect to the reactor operation conditions,
- The validation of the calculational methods applied might be insufficient or unsuitable,
- The data from the reactor records might be erroneous,
- The assignment of the reactor record information to the fuel identification might be wrong, or
- A misloading event might occur and remain undetected.

Thus, the *actual* safety margin of the system may be smaller than the *regulatory* safety margin $\Delta k_{SM} = 1 - k_{limit}$, given by equation (2). From that the conclusion might be drawn that the system has the potential of being less safe under the burnup assumption than under the fresh fuel assumption.

Based on such considerations, certain reluctance to BUC implementation has been detected in the past in regulators and other stakeholders. The improvement of the knowledge in the BUC implementation field during the past ten years, including the adoption of advanced calculational techniques, and the experience gained, have provided a solid base to modify this perception.

Of course, a risk analysis of BUC implementation compared with the risk obtained using the fresh fuel assumption would most probably show an increase in risk, because of the reduction in the actually available margin to the critical condition due to the reduction in analysis conservatism. In most regulations, that increase in risk has to be demonstrated to be overweighed by the risk improvement that the BUC implementation can provide in areas other than criticality safety. Thus, the benefits of introducing BUC should be identified and when possible quantified. These may include reduced operator dose, lower transport movements, less environmental impact, lower waste arising, economic advantages etc. A balance of risk arguments should be produced and preferably quantified, to demonstrate that, overall, the advantages outweigh the disadvantages when viewed from global perspective. In other words, an integral risk analysis of the spent fuel management facility or system should be preformed in order to demonstrate that there is an overall improvement.

A direct example of the above is provided by BUC implementation in transport cask design. The increase in cask capacity provided by the use of BUC would allow for a reduction in the number of fuel shipments needed. On top of reducing the dose to workers and the public, the reduction in the number of shipments would mean a similar decrease in the frequency of transportation accidents, hence reducing the criticality risk associated to those scenarios. Risk improvements in these fields might be demonstrated to provide an overall reduction in the risk in case of implementing BUC.

As already mentioned above, there are some items or events associated to BUC implementation that do not apply to criticality safety analysis made using the fresh fuel assumption:

- Errors in the analyst's judgment (analysis not bounding, insufficient validation),
- Systematic errors in the burnup information (erroneous data from the reactor records),
- Fuel misloading (misloading event).

3.3.4.1 Errors in the analyst's judgment

From the analytic point of view, the main difference between a BUC analysis and a traditional analysis lies in its complexity. The need for a depletion calculation heavily increases the number of parameters that need to be considered in the analysis. Most of them are operation conditions of the fuel, which sometimes are not so easy to know in detail. The analyst's judgment to determine the parameter range that covers all the operation conditions (bounding approach) becomes then crucial, and should be exercised with care to avoid underestimation of the reactivity.

The proof that the approach made is really bounding must include sufficient validation of the isotopic inventory used and the criticality calculation code applied. The fact that there is no critical or sub-critical experiment using commercial spent fuel in a configuration of interest (e.g. cask configuration) is certainly one of the main reasons why, to date, the application of the actinide plus fission product BUC level is restricted to PWR wet storage ponds

3.3.4.2 Systematic errors in the burnup information

As already stated above, in many cases the burnup value of each fuel assembly is determined using information from the reactor records. As this information is generated using both measurements of the power distribution and reference core power distribution calculations, it has already occurred that a systematic error is introduced in the process of generating this information. This kind of error might affect the information of a full reload.

3.3.4.3 Fuel misloading

The outcome of a BUC criticality analysis of a spent fuel management system is always a loading criterion the fuel has to comply with in order to be acceptable for inclusion in the system. In most cases this criterion is formulated in the form of a loading curve defining the minimum required burnup as a function of the initial enrichment (cp. Section 3.3.).

A misloading error occurs when a fuel assembly not complying with the burnup and enrichment requirements established by the loading curve of the system is anyway loaded in it. The probability and the consequences of this event are strongly system-specific. In general terms, in the case of wet storage systems the associated risk might be high, whereas in the case of dry storage or transport the importance of the event is either dependent on the probability of re-flooding of the cask cavity or determined by the regulatory requirement to consider re-flooding of the cask cavity as a design basis event (cp. Section 3.3.3.2).

The probability of a misloading event can be reduced by imposing different and independent layers of administrative verifications before loading the fuel into the system. However, procedures based on technical measures employing hardware and software controls are preferable to administrative verifications, cp. Section 3.3.3.2.

The administrative verifications and controls have to be aimed to avoid the possibility of “common mode” errors, which have the possibility to lead to multiple misloadings. Two or more misloading errors do not need to be considered if they can be considered as independent events. However, if the same administrative error can lead to the misloading of more than one fuel assembly, it has to be considered as one event. Since no system designed for BUC application can withstand the misloading of more than one fuel assembly if it cannot withstand the misloading of one fuel assembly, this event needs to be ruled out by administrative controls (i.e. the probability of this event has to be reduced such that this event needs not to be considered as a design basis event).

As already described in Section 3.3.3.2, a different approach is used in some countries, in which the misloading event has to be excluded by application of the double contingency principle. The interpretation of the principle in this case is that two independent, unlikely and concurrent incidents have to happen before a misloading event can occur. With this philosophy, the misloading event is ruled out and needs not to be considered in the analysis.

In those cases in which the misloading scenario is not ruled out, if the event really does occur, there is a high probability of the error remaining undetected. That raises questions about whether or not the double contingency principle may be applied to the misloading event and a different design basis event that may take place at a later time. As mentioned in Section 3.3.3.2, in some countries it is decided that a misloading event, that remains undetected, and any other design basis event, that takes place at a later time, cannot be regarded as “concurrent events”, so that the double contingency principle is not applicable to these events.

3.3.5 Continued validity of the analysis assumptions

Use of BUC relates the criticality safety requirements for the spent fuel management system of interest to core operating conditions. The validity of the loading criterion is tied to the depletion parameters assumed. The values taken for these parameters must be either bounding or conservative with respect to the reactivity of the spent fuel system. This means, that these values are limiting for the validity of the loading criterion. A change in the operational conditions results therefore in a necessity of a check whether the loading criterion is still valid. A change in one parameter such that this parameter is not longer bounding does not necessarily mean that the loading criterion is no longer bounding. In reality, these parameters are correlated. So, whether the loading criterion is still valid or not depends on the changes in the whole set of the depletion parameters.

The depletion parameters to be controlled are:

- Fuel design,
- Initial enrichment, heterogeneous enrichment distributions (MOX, BWR),
- Specific power, cycle length and power histories,
- Fuel temperature,
- Moderator temperature or density, void histories (BWR),
- Boron let down curve (PWR),
- Use of burnable poison rods, axial shaping rods, control rods or blades,
- Use of integral burnable poisons,
- Use of MOX in the core (spectrum hardening),
- Cooling time (if credited).

In addition, the validity of the loading curve is tied to the axial / horizontal burnup profile databases evaluated. These data bases are usually evaluated in such a way that bounding profiles are obtained. Nevertheless, if the fuel design or the core operating conditions are changed in such a way that significant impacts on the burnup shapes are to be expected it will become necessary to check whether the loading criterion is still valid or not.

The parameters impacting axial/horizontal profiles are in particular:

- Reload patterns (determining the interactions between fresh fuel assemblies and fuel assemblies with different burnup shapes at begin of cycle);
- Cycle length;
- Control rod movements;
- The use of burnable poison rods or axial power shaping rods;
- The presence of integral burnable poisons;
- Axial zoning of enrichment or burnable poisons, presence of axial blankets, presence of partial length rods;
- Change in the operational strategies (in-out/out-in strategies, extended low power operation period at end of cycle).

In addition, it has to be taken into account that the end effect due to the axial distribution of the burnup is affected by:

- All the depletion conditions affecting the U-238 capture rate and the U-235 depletion rate (such as initial enrichment, specific power, spectrum hardening due to higher boron concentration or presence of burnable poison rods) and
- The cooling time.

In the WWER 440 case the fuel parts of all the control assemblies have to be checked due to the fact that they are partly outside of the core during the main part of the cycle. It has to be taken into account that the presence of the control assemblies have a strong impact on the flux distribution and hence on the axial profiles of the neighboring assemblies.

Means to perform the checks are:

- Depletion parameters: The decision about which parameter values are bounding can be made on the basis of sensitivity studies on the impact of the depletion parameters on the reactivity of the spent fuel management system of interest;
- In case of parameter changes resulting in spectrum hardening one has to take into account that the burnout of burnable poisons is delayed: A means of checking whether the loading curve is still valid is to re-check the validity of k_{eff} as a function of burnup at a given initial enrichment;
- Spectrum hardening impacts the reactivity worth of the isotopes used in a BUC analysis. Therefore, it might be necessary to re-check the validity of the isotopic validation basis. Means are comparisons of the spectra of the application case with spectra of the validation cases;
- Axial/horizontal burnup profile data bases:
  - Provide the end effect in terms of Δk or ΔB as a function of burnup B and establish a bounding curve for Δk or ΔB for the old case, and calculate the Δk or ΔB for the new shapes;
  - In case that already bounding profiles were used for the generation of the loading curve, compare the reactivity of new shapes with the reactivity of the bounding one under the conditions of the design basis of the loading curve.

3.3.6 Conclusions and recommendations

3.3.6.1 Definitions of terms

During the discussion of the first topic, "safety criteria" (cp. Section 3.3.2), it turns out that there is some confusion in the understanding of the terms "safety margin", "analysis conservatism" and "bounding approach", see also [6]. It is recommended to use the following definitions:

Regulatory safety margin

This margin is the difference between the critical condition ($k_{eff} = 1$) and the maximum allowable neutron multiplication factor k_{limit} as defined by equation (2), $\Delta k_{SM} = 1 - k_{limit}$.

Actual safety margin (of a given fuel configuration)

This margin is given by the difference between the critical condition and the true neutron multiplication factor (corrected on any possible bias).

Analysis conservatism

Analysis conservatism means a *deliberate overestimation* of the neutron multiplication factor of a fuel system due to specific features of the model used to describe the properties of the system in the analysis procedure applied. Analysis conservatism is strictly related to the description of the *physics of the system* and is hence related to an *a priori knowledge*.

A reason for taking analysis conservatism could be the lack of knowledge or tools needed to exactly model a certain effect that would reduce the calculated neutron multiplication factor, but also the desire to simplify the analysis is often a good motivation.

Bounding approach

A bounding approach is a calculation model justified by *physics arguments* demonstrating that the use of the model *does not result in an underestimation* of the neutron multiplication factor of the fuel system of interest.

Bounding approach and analysis conservatism should never be confused. In the limit, assuming that all the knowledge needed is available and the calculation system is perfect, the level of conservatism is nil, but the need for a bounding approach still exists, to guarantee that the calculated reactivity is the maximum that the fuel management system will achieve. For instance, a loading criterion must bound the wide variety of fuel irradiation histories and burnup profiles in order to be applicable to all fuel locations of a spent fuel management system.

3.3.6.2 Regulatory safety margin and analysis conservatism

It has been observed by the working group that in some countries the regulatory safety margin to be used is linked to the conservatism maintained in the analysis (cp. section 3.3.2) such that a higher level of conservatism allows the use of a lower regulatory safety margin.

However, it is more instructive to maintain a higher regulatory safety margin and to reduce the analysis conservatism, since this way improves the knowledge of BUC and hence reduces the risk of errors in the analyst's judgment. Reduced analytical conservatism makes it necessary to study the sensitivity of the neutron multiplication factor of the system to:

- variations in the isotopic concentrations,
- variations in the depletion parameters,
- variations in the isotopic concentrations (due to variations in the depletion parameters).

In order to improve the reliability of the outcomes of such studies it is necessary:

- to improve the knowledge of the nuclear data, in particular of the cross sections of some actinide and fission product isotopes relevant to BUC, and
- to extend the validation range of the codes applied.

This contributes to a better understanding of the physics of the system and makes it easier to come to a bounding approach (i.e. to verify a certain approach as the bounding one).

One should keep in mind that, for example, the actinide-only BUC approach, which is obviously conservative as long as a uniform burnup distribution within the fuel assemblies is assumed, results in a significant underestimation of the end effect. Thus, the level of conservatism maintained by the actinide-only BUC approach can only be quantified when the end effect is analyzed by applying the actinide plus fission product BUC level.

In conclusion, it is recommended to reduce the analysis conservatism instead of using a lower regulatory safety margin.

3.3.6.3 Verification of reactor records and loading criterion

It has been observed by the working group that in some countries it is required to link the uncertainty of the reactor record information *and* the uncertainty of the burnup measurement, performed to verify the reactor record information, to the loading criterion. This requirement is justified only when the measurement result is preferred to the reactor record information (cp. Section 3.3.3.1). However, usually some information from the reactor records is needed to evaluate the burnup measurement. Thus, the outcome of the burnup measurement depends on the reactor record information, but the reactor record information is independent of the measurement results. Therefore, the verification of the reactor record information by the measurement consists in a check whether or not the reactor record information and the measurement results are consistent.

The reactor record information, once verified, can then be compared with the loading criterion. Due to the fact that the reactor record information is independent of the measurement outcome, only the uncertainty of the reactor record information has to be linked to the loading criterion. It is recommended to proceed in this way.

3.3.6.4 Need for integral risk analysis

It has been observed by the working group that application of BUC to a spent fuel management system gives rise to the need to perform an integral risk analysis of the system in order to demonstrate that the use of BUC results in an overall reduction in the risk. In order to get a prime example it is recommended to perform such a risk analysis for a dry storage or transport system under the following conditions:

- Use of the actinide plus fission product BUC level;
- Variation of the probability of events leading to re-flooding the cask cavity in order to develop decision criteria about the importance of the misloading event.

The analysis should give guidance on the establishment of a failure rate database, which is needed for performing the probabilistic risk analysis of the storage or transport system.

3.3.6.5. Monitoring and promoting implementation of buc

The working group concluded that the IAEA program of monitoring and promoting the implementation of BUC which was started in 1997 has been very successful in the worldwide dissemination of knowledge of BUC and its implementation. Due to the fact that:

- The introduction of the actinide plus fission product BUC level to storage and transport, and
- The application of BUC to MOX and WWER fuel.

are still to come, the working group concluded that this BUC program should continue.

3.4. Future applications of BUC

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3.4.1 Introduction

The reduced storage energy in spent nuclear fuel is an intrinsic characteristic of the fuel. Depending on the regulatory environment and/or the level of technical understanding of the spent fuel parameters will allow the reduced stored energy be taken into consideration, BUC. Within the concept of BUC, there are different levels of credit that can be evaluated, actinide and fission product. This chapter performs a quick look at the future application of BUC and how it may be applied to the different reactor types and fuel types noted in Tables VI and VII below.

3.4.2 MOX fuels

There are four types of MOX that can be considered. LWR MOX, weapon grade blend down MOX, over-moderated MOX, and MOX from fast breeder reactors.

3.4.2.1 LWR MOX

The reactivity decrease with burnup is less for LWR MOX than for UOX fuel. This is due to the breeding of Pu-240 to Pu-241. Also, the initial reactivity is significantly lower for MOX than UOX fuel. This is why there is less of a need to claim BUC for MOX fuel than for UOX fuel at this current time. However, the move towards higher MOX enrichments in the future may change this situation.

3.4.2.2 Weapons grade blend-down MOX

MOX derived from weapons grade plutonium has a high initial Pu-239 content and very low Pu-240 isotope poisoning. At the beginning of life, this type of MOX has the same reactivity and nearly the same reactivity change with burnup as UOX fuel. During irradiation, the Pu-239 content of the fuel decreases due to fission and neutron capture to Pu-240 leading to a similar behaviour as reactor grade MOX. The necessity of BUC for this kind of MOX is therefore greater than for LWR MOX.

3.4.2.3 Over-moderated MOX for EPR

One of the possibilities of an EPR core is composed of 100% MOX fuel assemblies that are over moderated. At the start of life and during irradiation, the reactivity behaviour of this type of MOX fuel is similar to that for UOX fuel (Fig. 1). For this reason, there is a greater potential for claiming credit for the burnup of the fuel.

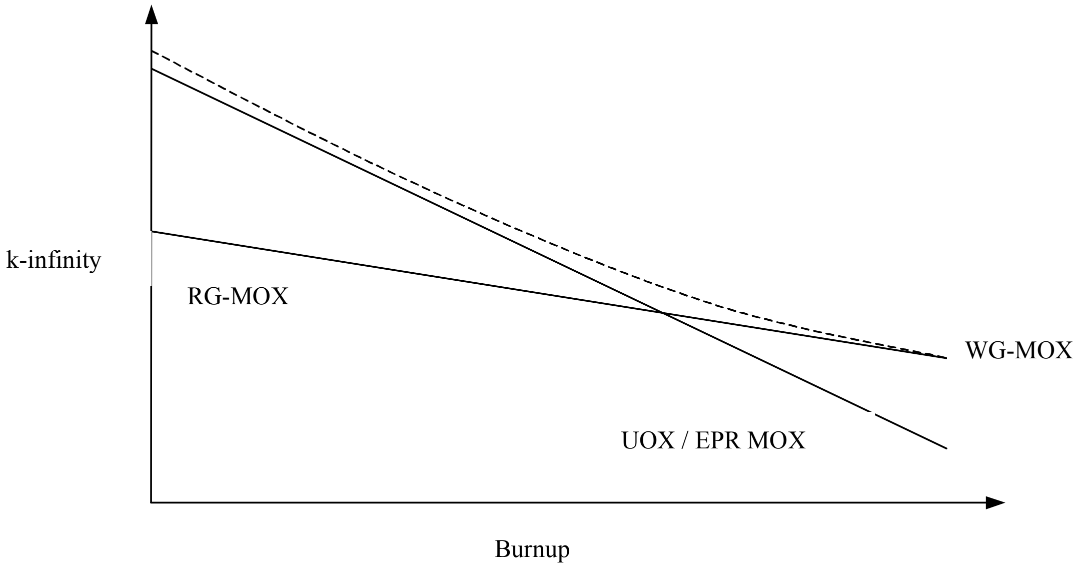

Fig. 1. Reativity behaviour of oxide fuels in EPR.

3.4.2.4 MOX coming from fast breeder reactors

The actinides inventory does not change significantly during irradiation. Therefore, actinide-only BUC is not a viable option. However, the presence of fission products can be considered.

Currently there are only a limited number of fast reactors fuelled with MOX that are currently operating or are in the process of being permanently shutdown. Table VI summarizes the fuel cycle and disposal path for some of the existing fast reactors worldwide. At this time, the amounts of each plant specific fuel assemblies are small; hence, there is little economic incentive in establishing BUC. A generic approach encompassing the wide range of different fuel types may be a possibility however the economic benefits needs to be determine.

3.4.3 BUC applications for MOX

3.4.3.1 Transportation

For MOX fuel, the problem of transportation is mainly that of shielding, rather than that of criticality issue. This is why during transport it is necessary to mix MOX with UOX fuel. Even with this difficulty, BUC can be used for the design of new transport flasks.

3.4.3.3 Disposal

For disposal of MOX fuel, there is no problem of shielding, so it initially appears that one may be able to take greater advantage of BUC than in transport operations. However, over the longer term (greater than 100 years), the Pu-240 and Am-241 content of MOX fuel decreases, which leads to an increase in the reactivity.

Table VI. Fuel Cycle and Disposal Path for Fast Reactors Worldwide

Location	Reactor	Fuel type	Fuel cycle	Disposal Path
USA	FFTF	MOX	Once through	Yucca Mountain
France	Phenix	MOX	Reprocess	Recycle in LWR
France	Super Phenix	MOX	Reprocess	Recycle in LWR
Russia	BN 350	UO_2	Once through	Long term storage Possible reprocessing at Mayak
Russia	BN 600	UO_2 conversion to MOX	Once through	Long term storage Possible reprocessing at Mayak
Japan	MONJU	MOX	Reprocess	Recycle in LWR
USA	EBR II	U and Pu metal	Treatment	Waste forms to geological repository

This is a greater problem for MOX fuel than for UOX fuel as the latter does not contain as much Pu-240. For UOX fuel, the reactivity does not change significantly with time. However for MOX fuel, the reactivity can be greater after 50,000 years than it was initially (Fig. 2). Due to the slightly different fissile isotopes associated with MOX, than UO_2 based systems direct disposal of MOX fuels will force an additional look at the near-field geochemical development. The US has investigated direct disposal of MOX and has concluded that direct disposal for the proposed US MOX design is feasible at the proposed Yucca Mountain site.

3.4.3.3 Storage

BUC may be more easily applied to the storage of MOX fuel in the medium term due to the reduction in reactivity of the fuel over the initial 100 years period. A distinction cannot be made between wet and dry storage. In many countries, dry storage containers must comply with the IAEA recommendations (TS-R-1) [7] for transport. These recommend that the presence of water is considered within the container. For reactor-grade MOX, storage facilities and shipping cask designed for UO_2 at the criticality safety point of view does generally accommodate MOX fuel without any problem. For weapons grade (WG) MOX and/or over moderated MOX fuel assemblies, BUC could be advantageously considered.

3.4.3.4 Reprocessing

As for UOX fuel, the advantage of BUC is to avoid or minimize the need of gadolinium in order to maintain the reprocessing rate. Since the residual fissile content of spent MOX fuel is higher than for UOX fuel, BUC considerations are essential to ensure the capacity and performance of the reprocessing operations.

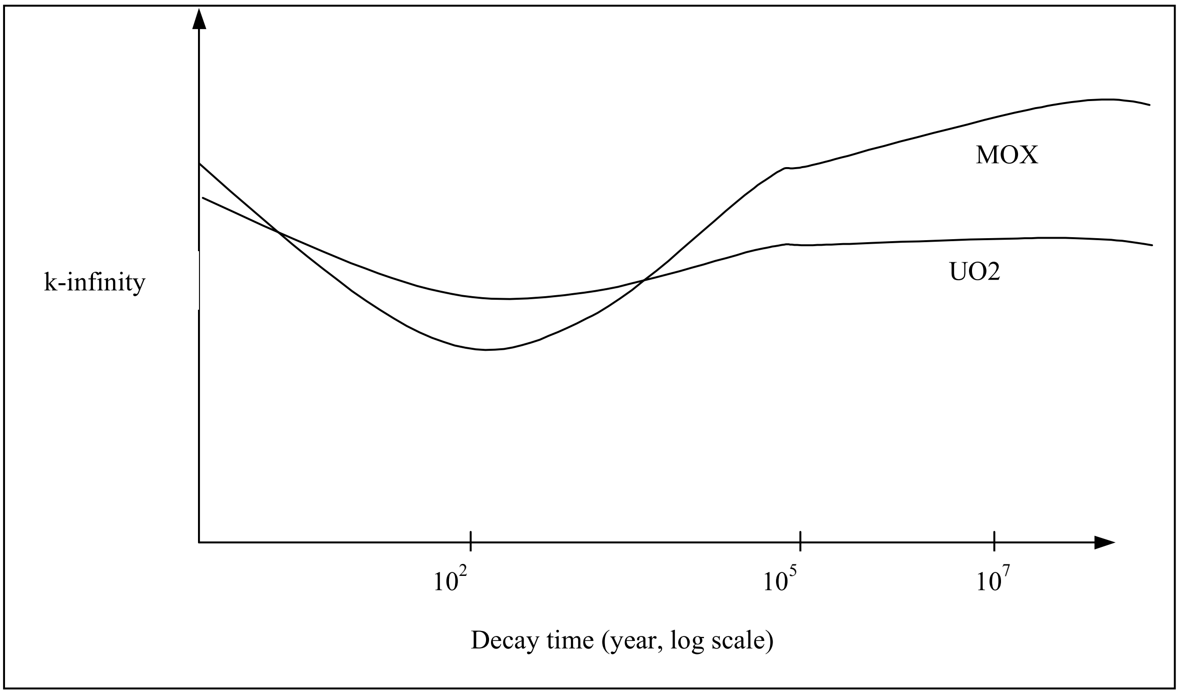

Fig. 2. K-infinity vs. decay time for MOX and UO2 fuels.

3.4.3.5 Important isotopes associated with fast breeder MOX BUC

As actinide-only BUC is not a viable option for fast breeder MOX fuel, the presence of fission products needs to be considered. The most important fission product is Sm-149, which is produced, in large quantities, and contrary to LWR fuel, the thermal capture rate of Sm-149 is negligible during irradiation.

After irradiation, in applications where water is present or needs to be considered, the thermal absorption of Sm-149 will be particularly effective. Consequently the use of 'samarium' BUC will be relevant.

3.4.3.6 Additional research, developments and operational needs for MOX

Research and development concerning BUC of MOX fuel should focus on the following topics:

- Depending on the application of BUC, the calculation methodologies for determination of the spent fuel inventory for MOX and UO_2 may require additional study:
  - Well assessed methods and data sets, concerning particularly shielding of actinides, cross-section sets, nuclide chains,
  - Geometrical modelling taking into account the environment of the fuel during irradiation;
- Improvement of spent fuel isotopic composition database. Compared to UO_2 fuel, for which a wide range of PIE experiments have been performed for years, the MOX databases should be further investigated;
- Special attention should be paid to the minor actinides, especially americium and curium isotopes. The highly fissile minor actinides (Am-242m, Cm-243, Cm-245) could

contribute to several percent to the reactivity at the end of life, especially for high burnup and high Pu-content MOX fuel. Present comparisons between calculation and measurement show large discrepancies (10-20 %) on the inventory of these isotopes;

- If BUC measurements are required, burnup measurement devices for MOX may need to be enhanced in order to be comparable to those used for UO_2 fuel;
- Concerning final disposal of spent MOX fuel, the geochemical behaviour of plutonium and minor actinides must be studied, as well as the long term reactivity evolution of the fuel. Studies performed in the US indicate that little or no impact to the performance of the repository will occur due to direct disposal of MOX fuel [8].

3.4.4 Advanced reactor fuels

3.4.4.1 Types

There are a number of advanced reactor and advanced reactor fuel (ARF) designs being considered by the US DOE Generation IV road mapping Group and the IAEA INPRO Group. Additional road mapping activities have been performed; reports issued include [9] and [10]. A partial list of some of the better-known concepts are summarized in Table VII and listed below.

Table VII. BUC Applicability

Reactor	Fuel Type	Fuel Cycle	Disposal Path	Key Area Motivation
Advanced LWR	UO_2	Once Through Or Reprocessing	Direct Disposal (Geological)	Higher potential credit than LWRs
LWR	MOX	Once Through Or Reprocessing	Recycling or direct disposal	Higher interest in reprocessing, high potentiality for WG MOX and advanced MOX fuels
High Temperature Gas Reactor	UO_2 and ThO_2	Once Through Reprocessing	Direct Disposal Recycle	High potential credit High potential
Pebble Bed	UO_2	Once Through	Direct Disposal Or Recycle	Depending on enrichment and/or significant quantity
Molten Salt	Fluoride - salt	On line-reprocessing	Continues with transuranic byproduct	Understanding of burnup required for neutron control system in processing plant
Advanced Fast Reactor	MOX Na or Gas coolant, Metal, and Metal Pb coolant	Reprocessing	Recycle	Need Fission Product BUC

Pebble bed reactor

The pebble bed modular reactor (PBMR) is a graphite helium cooled reactor, which uses the Brayton direct gas cycle to convert the heat that is generated in the core nuclear fission. The heat is transferred to the coolant gas (helium), where it is converted into electricity by the means of a gas turbine-generator. The PBMR core is based on the German high temperature gas reactor technology and uses spherical fuel elements. The fuel is typically not reprocessed due to its low fissile loading. The fuel typically has an enrichment of about 8 wt% with a typical discharge burnup of 80 GWd/kg U.

Gas turbine-high temperature gas reactor

The Gas Turbine-Modular Helium Reactor (GT-MHR) is an advanced nuclear power system designed to provide very high safety, high thermal efficiency, environmental compliance, and competitive electricity generation costs. The GT-MHR module couples a gas-cooled modular helium reactor (MHR), contained in one vessel, with a high efficiency Brayton cycle gas turbine (GT) energy conversion system contained in an adjacent vessel. The reactor and power conversion vessels are interconnected with a short cross-vessel and are located in underground concrete silo. The MHR refractory coated particle fuel consists of a spherical kernel of fissile or fertile material, as appropriate for the application, encapsulated in multiple coating layers. The overall diameter of the standard particles varies from 650 microns to about 850 microns. For the GT-MHR the particle size vary from 13 mm in diameter to 51 mm in length. The fuel particles are loaded in a “compact” and the compacts are loaded into hexagonal channels. The hexagonal channels are made of graphite, 793 mm long and 360 mm across the flat. The fuel is anticipated to be in the low enrichment range with discharge burnups of greater than 100 GWd/kg U.

Advanced light water reactor

The advanced light water reactor is the next generation LWR, using the once through UO_2 fuel cycle. The joint U.S. DOE and industry funded program resulted in three NRC-certified ALWR designs, the General Electric Advanced Boiling Water Reactor (ABWR), the Combustion-Engineering (now Westinghouse) System 80+, and the Westinghouse AP600. The last design is a passive-safety mid size plant (approx. 600 MW(e), ABWR and System 80+ are large 1300 MW(e) evolutionary designs. The fuel type is similar to the LWRs now in operation, with a few basic improvements. The improvements are based in the materials of construction of the fuel. The materials have migrated to higher and higher purity with the selection of higher temperature alloys. The enrichments for the new generation fuel assemblies are focused on a maximum of approximately 6-wt% U-235.

Advanced fast reactor

The advanced fast reactor (AFR) is an extension of the existing fast reactor technology. Three different coolants are being considered: sodium (reference design), lead and helium. Two different fuel forms are being considered: oxide and metal. The fuel is encased in stainless steel cladding on a triangular array and the enrichment ranges from 20 to 35% plutonium/uranium, with an anticipated burnup in excess of 100 GWd/kg U. It is envisioned that the AFR will be operated under a closed fuel cycle (reprocessing). The oxide fuel can be reprocessed by an aqueous (e.g. PUREX) process or by the combined oxide reduction/electrochemical process. The metal fuel will be electrochemically reprocessed.

Molten salt reactor

The molten salt reactor (MSR) is a fluid fuelled reactor that uses a homogeneous mixture of molten salt as the fuel and coolant. The UF_4 fuel is dissolved in a mixture of fluoride based salts, forming a clear liquid at reactor operating temperatures of about 650^{o}C. MSR can be operated with U-235 or U-233 as the fissile component of the fuel. The fuel salt circulated through the graphite-moderated reactor vessel to a circulating pump and then through a heat exchanger before returning to the reactor vessel. Because of the fluid nature of the fuel some form of on-site or off-site treatment/reprocessing is envisioned. Since there are no traditional fuel assemblies, an online salt slurry treatment process is used to remove the fission products and impurities. BUC is not anticipated for this reactor fuel type.

3.4.4.2 Specific applications

This application of advanced reactor fuels (ARF) does not differ generally from the BUC applications for conventional LWR fuels. The potential applications for BUC include:

- Wet storage at or away from the reactor,
- Dry storage at or away from the reactor,
- Transportation wet or dry,
- Reprocessing,
- Disposal.

The specific application of BUC depends on the specific ARF, as well as technical, economical, political decisions and considerations. In general BUC application can be expected for any large-scale operations.

3.4.4.3 Benefits and motivations

From the point of view of the nuclear industry, the reduction of cost is the driving force. BUC application can cause an increase of storage capacity per cask or and increase of the capacity of a storage pool.

The increase of cask capacity reduces the number of the cask to be handled and shipped. This will result in a decreased impact for the environment from handling and shipping activities. It also decreases the risk of an accident since the number of shipments is less. Furthermore, dose to workers and the public may be reduced.

In critical analyses for long-term storage and disposal, specific aspects may be important that make it necessary to consider burnup. Finally, the investigation of burnup effects on application of the spent fuel may result in an increase of safety margin even if BUC is not applied in any specific case.

3.4.4.4 Status

For the most part the advanced reactor fuel (ARF) types are at the conceptual development stage with limited prototyping. There is little work on BUC for ARF types at this time.

The ongoing development activities for the advanced LWR fuels with very high burnup (100 GWd/tU), include activities that will support BUC application. The material testing, benchmarking and PIE/radiochemical assay work will be useful for BUC applications.

3.4.4.5 Future plans

There are few plans for BUC-activities with the ARF. Until a significant number of AFRs exist, there is no benefit to make definite plans.

3.4.4.6 R&D and operational needs

Implementing BUC for new fuel types requires the following tasks:

(1) Identifying the relevant activities, nuclides, fissile nuclides further actinides, fission products;
(2) Studying the influencing parameters e.g. from reactor operation and cooling, by calculation and measurement;
(3) Validation of codes for inventory and criticality calculations according to specific fuel types;
(4) Development of methods for practical implementation, e.g. verification or measurement, further safety measurement to exclude inadvertent interchange.

Further needs depend on the specifics of the fuel under consideration

3.4.4.7 Regulatory consideration

The regulatory concerns for BUC in ARF are the same as for BUC in current reactor fuels. Protection of the public and workers, health and safety, and protection of the environment are still the underlying considerations.

The potential benefits of BUC to health and environmental protection would continue to be a consideration to be accounted for by regulators.

3.4.4.8 Conclusions and recommendations

MOX - LWR MOX and fast MOX

The type of MOX fuel under consideration determines whether or nor it is advisable to claim for BUC. For standard RG MOX, as used in Europe and in Japan, BUC is of major interest for reprocessing, and to a lesser extent for storage, transportation and disposal. For WG MOX, BUC procures the same advantages as UO_2 fuel. Fast reactor MOX fuel could benefit from fission products credit for wet storage.

Future developments for BUC for MOX should cover calculation methodologies, PIE, for nuclide inventory of the spent fuel and critical experiments.

Advanced reactor fuel designs

There are a large number of advanced reactor fuel designs being considered as concepts. In general, none are sufficiently developed so that there are clear motivations or benefits for implementing BUC at this time.

However, from experience with applications of BUC with current LWR fuel, benefits can be expected if any of the ARF is developed to an operational stage.

Also from experience with implement BUC for current LWR fuels minor additions to developmental activities will allow easier implementation of BUC for the AFRs.

Accordingly, consideration of BUC implementation issues should be included in the development (testing/experimenting) for ARF concepts.

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ABBREVIATIONS

AR	at-reactor
ARF	advanced reactor fuel
ARIANE	an experimental programme of radiochemistry analysis providing data on MOX and UO_2 fuels irradiated in PWR and BWR conditions, managed by Belgonucleaire
BUC	burnup credit
BUC-FP	burnup credit – fission product level
BWR	boiling water reactor
CERES	international programme to validate cross section data and inventory predictions for actinides and fission products important for burnup credit, using the MINERVE reactor, CEA Cadarache.
EFR	European fast reactor
GT-MHR	gas turbine modular helium reactor
ICSBEP	International Criticality Safety Benchmark Experiment Project
IE	initial enrichment
JEF	joint evaluated files
JEFF	Joint Evaluated Data Library based on co-operation between JEF project and European Fusion File (EFF) project.
LWR	light water reactor
MALIBU	PIE programme supporting depletion calculations
MOX	mixed oxide fuel
MSR	molten salt reactor
OM	optimal moderation
PA	package array
PIE	post irradiation examination
PROTEUS	zero-power, fast/thermal mixed critical reactor facility in Switzerland
PUREX	plutonium, uranium extraction (reprocessing) process

PWR	pressurized water reactor
RBMK	Russian graphite moderated reactor
REBUS	reactivity tests for a direct evaluation of BurnUp credit on Selected irradiated LR fuel bundles
SFCOMPO	LWR spent fuel isotopic composition database
SFDS	spent fuel dry storage
SP	single package
UOX	uranium oxide
WG	weapons grade
WWER	Russian type of PWR

INTERNATIONAL ACTIVITIES

(Session 1)

Overview on the BUC activities at the IAEA

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Due to the worldwide interest and broad applicability of BUC for spent fuel management, the International Atomic Energy Agency (IAEA) has taken an active interest in the subject. The IAEA role has been one of an observer and disseminator of appropriate information.

In 1997, the International Atomic Energy Agency (IAEA) started a task to monitor the implementation of BUC in spent fuel management systems, to provide a forum to exchange information, to discuss the matter and to gather and disseminate information on the status of national practices of BUC implementation in the Member States. The task addressed current and future aspects of BUC.

In October 1997, the IAEA organized an advisory group meeting (AGM) to examine and report on the status of BUC for storage, transport, reprocessing, and disposal of PWR, BWR, WWER, RBMK and MOX spent fuel.

Since the proceedings of the AGM were published in April 1998 (IAEA-TECDOC-1013), significant developments have served to advance the use of BUC throughout the world. Experts from countries with nuclear programmes that range from very small to very large have made important contributions. The use of BUC has progressed along the lines of greatest need.

In the following years, several consultancies were held to monitor the progress in BUC implementation. (July and December 1998, July 1999 and July 2001). The results of these meetings were published in three Working Materials.

A Technical Committee Meeting (TCM) on the Evaluation and Review of the Implementation of BUC in Spent Fuel Management Systems was held in July 2000 in Vienna with 35 experts from 17 countries and 2 international organizations. The purpose of this TCM was to survey the progress and status of international activities related to the use of BUC for spent fuel applications.

The current BUC applications status and the BUC level implemented in each application were collected and published in Tables.

Conclusions of the TCM

During the closing session of the TCM, it was concluded that the use of BUC for spent fuel management continues to progress. The TCM recommended continued acquisition of data to support BUC. In particular the need for additional chemical assay to benchmark calculation methods was identified. Further studies of axial effects, and verification methods for fuel burnup values were recommended. The value of a co-operative approach was recognized. Thus co-operation in future experimental programmes and sharing of available data was recommended by the TCM. Implementation of a training course for potential users of BUC and their respective regulators was a stated goal of the participants.

The proceedings of this TCM were published in the TECDOC-1241 in August 2001.

In 2001, a Consultants Meeting was held on the Requirements, Practices and Development in BUC Applications, 23 to 26 July 2001, in Vienna. Besides the review of progress in the field the meeting was used to prepare this meeting we are holding right now.

On 15 to 26 October 2001, a training course on the implementation of BUC in spent fuel management systems, was held at Argonne National Laboratory near Chicago, USA. 25 experts from 12 different countries participated in this training course.

For the next budget cycle 2004/2005, a new task on BUC is foreseen: “Advances in Applications of BUC to Reduce the Number of Transports and Increase Storage Capacity” with a Consultants Meeting in 2004 to prepare for a technical meeting in 2005.

A second task proposed for the next budget cycle 2004/2005 are consultants meetings in 2004 and 2005 to prepare a technical publication on benchmarking depletion codes for BUC using chemical assay data of WWER fuel from a DOE funded programme.

OECD/NEA report

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Abstract. This paper summarizes activities within the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA). There are two groups currently addressing issues associated with BUC: (1) the Expert Group on BUC Criticality, a subordinate group to the Working Party on Nuclear Criticality Safety (WPNCS) working under the Nuclear Science Committee (NSC) and (2) the Working Group on Operating Experience/Fuel Cycle Safety (WGOE/FCS), a subgroup of the Committee on the Safety of Nuclear Installations (CSNI). The WPNCS of the OECD/NEA coordinates and carries out work in the domain of criticality safety at the international level. Particular attention is devoted to establishing sound databases required in this area and to addressing issues of high relevance such as BUC. The activities of the expert group are aimed toward improving safety and identifying economic solutions to issues concerning the back-end of the fuel cycle. The group has established and evaluated a number of calculational benchmarks addressing the physics and modeling needs for performing BUC analyses for light water reactor fuels. The CSNI assists in maintaining and further developing the scientific and technical knowledge base required to assess the safety of nuclear reactors and fuel cycle facilities. The main mission of the WGOE is to promote the development of improved techniques and methods based on a review and analysis of operating experience. The Fuel Cycle Safety subgroup has developed a BUC questionnaire to establish a mutual understanding of BUC from the regulatory view-point and to evaluate the differences from country to country and establish the possibility of standardized BUC in regulation for the future.

1. Introduction

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

2. Working Party for Nuclear Criticality Safety (WPNCS)

OECD/NEA has coordinated the activities of this criticality safety benchmark group for nearly two decades. The Working Party for Nuclear Criticality Safety (WPNCS) was chartered to review and coordinate the activities of the existing expert groups operating under the auspices of OECD/NEA and to propose establishing task forces (expert groups) corresponding to new demands on methods development, experimental needs and international handbook data in the field of nuclear criticality safety.

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

Expert groups have been established for:

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

Validation of codes and data, benchmarking, criticality safety handbooks and standards are common themes among the different expert groups. This overlap often requires integration and coordination that is the responsibility of the WPNCS. The following is a recent example showing the different levels of coordination for this working method. Within the Expert Group on BUC Criticality several issues of wider interest were identified. These have been reported to WPNCS for further investigation, e.g.:

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

The focus of this paper is to report on the activities of the Expert Group on BUC Criticality.

Information about the current activities and links to publications of the OECD/NEA Working Party on Nuclear Criticality Safety may be found at http://www.nea.fr/html/science/wpncs/

3. Expert group on BUC criticality

The main objective of the activities of the OECD/NEA Expert Group on BUC Criticality is to demonstrate that the available criticality safety calculational tools are appropriate for application to burned fuel systems and that a reasonable safety margin can be established. For this purpose the Expert Group established a suite of BUC criticality benchmarks that assess the capability to calculate both spent fuel composition and reactivity of spent fuel.

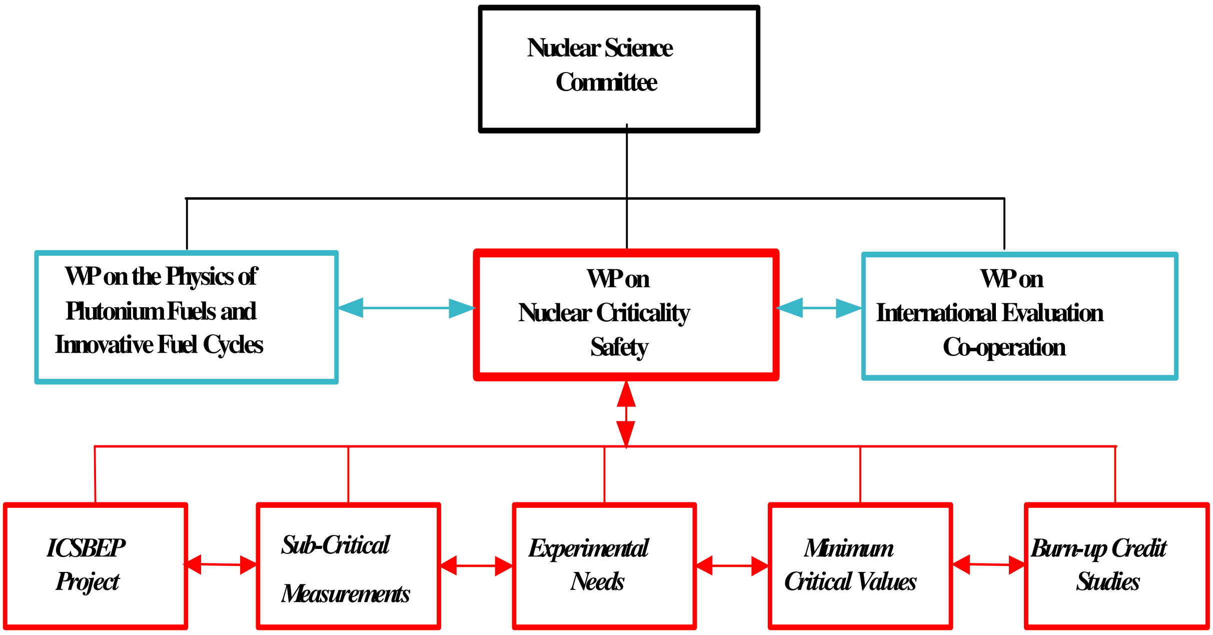
Nuclear Science Committee
WP on the Physics of Plutonium Fuels and Innovative Fuel Cycles
WP on Nuclear Criticality Safety
WP on International Evaluation Co-operation
ICSBEP Project
Sub-Critical Measurements
Experimental Needs
Minimum Critical Values
Burn-up Credit Studies

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

The benchmarks were carefully specified to allow a comparison of results using a wide variety of calculational tools and nuclear data sets. Throughout the tenure of the activities of the Expert Group on BUC Criticality, experts from 17 countries (Belgium, Canada, Czech Republic, Denmark, Finland, France, Germany, Hungary, Italy, Japan, Korea, The Netherlands, Spain, Sweden, Switzerland, United Kingdom and the United States) have participated in various phases of the benchmark exercises. Participants used a wide variety of codes and methods based on transport theory, using SN, nodal and Monte Carlo techniques. Nuclear data (both cross-section and decay data) were taken from a variety of sources: multiple versions of the Evaluated Nuclear Data Files (ENDF/B), the Japan Evaluated Nuclear Data Libraries (JENDL) and the Joint Evaluated Files (JEF). Both multi-group and continuous energy cross-section data were used in the study.

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

Table I. Summary of benchmark problems addressed by the OECD/NEA Expert Group on BUC Criticality

Benchmark	Primary Objective	Status
Phase I-A	Examine effects of seven major actinides and 15 major fission products for an infinite array of PWR rods. Isotopic composition specified at 3.6 wt.% U-235 at 0, 30 and 40 GWd/MTU and at one- and five-year cooled.	Completed (Ref. 1)
Phase I-B	Compare computed nuclide concentrations for depletion in a simple PWR pin-cell model, comparison to actual measurements at three burnups (27.34, 37.12 and 44.34 GWd/MTU).	Completed (Ref. 2)
Phase II-A	Examine effect of axially distributed burnup in an array of PWR pins as a function of initial enrichment, burnup and cooling time. Effects of fission products independently examined.	Completed (Ref. 3)
Phase II-B	Repeat study of Phase II-A in 3-D geometry representative of a conceptual BUC transportation container. Isotopic compositions specified.	Completed (Ref. 4)
Phase II-C	Key sensitivities in criticality safety to burnup profiles.	In progress
Phase III-A	Investigate the effects of moderator void distribution in addition to burnup profile, initial enrichment, burnup and cooling time sensitivities for an array of BWR pins.	Completed (Ref. 5)
Phase III-B	Compare computed nuclide concentrations for depletion in a BWR pin-cell model.	Completed (Ref. 6)
Phase IV-A	Investigate BUC for MOX spent fuel pin-cell for three plutonium vectors (first recycle, fifth recycle, weapons-grade)	Final Draft Approved 2002
Phase IV-B	Compare computed nuclide concentrations for depletion in a MOX super-cell.	In progress
Phase V	WWER BUC. Similar to Phases I and II for PWRs but with hexagonal geometry and WWER fuel specification	Independent/ Parallel Study

Phase I and Phase II included both criticality and depletion benchmarks for pressurized water reactors (PWRs). A set of selected nuclides including 7 major actinides (U-234, 235, 236, and 238; Pu-239, 240 and 241), 5 minor actinides (Pu-238 and 242; Am-241 and 243; Np-237) and 15 fission products (Mo-95; Tc-99; Ru-101; Rh-103; Ag-109; Cs-133; Sm-147, 149, 150, 151 and 152; Nd-143 and 145; Eu-153; and Gd-155) were used in these studies. The results showed no trends in standard deviation among participants with burnup or cooling time in the criticality analyses. Consistently the largest deviations among participants were for the fresh fuel cases. In the depletion analyses, there was evidence of a significant trend in the standard deviation among participants for the residual U-235 (the trend was small for most other isotopes). A number of nuclides have been identified for additional study based on the sensitivity of k to the observed standard deviations: Pu-239, Gd-155, U235, Pu-241, Pu-240 and Sm-151. Much of the differences are assumed to be in the basic nuclear data. Both 2-D and 3-D models have been used to evaluate the impact of axially distributed burnup. It was determined that 70% of the total fissions occur in the upper 40cm of fuel that illustrates the potential importance of this parameter. Good agreement was seen among the participants relative to the calculated “end effect”. It has been noted by the group that the effect on k is strongly a function of the system being evaluated and may be even more important under postulated accident conditions that result in axial heterogeneity. Two remaining issues associated with the axial effect continue to be investigated in the expert group: (1) limited availability of measured axial profile data and detailed power history data in the open literature, and (2) defining/performing analyses to determine the sensitivities due to different axial burnup profiles across the full range of burnups.

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

Mixed oxide (MOX) fuels in PWRs were investigated in Phase IV. The problems included a MOX pincell calculation to identify sensitivities specific to MOX. The primary result of this early benchmark was to identify the need to include curium isotopes in both the criticality and depletion calculations, as Cm contributes up to 1.5% in k. The reports for these analyses are in preparation/review and should be forthcoming this year. Phase V is a completely parallel study being led by L. Markova and addressed in another presentation at this meeting.

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

Information about the current activities and links to publications of the OECD/NEA Expert Group on BUC Criticality may be found at http://www.nea.fr/html/science/buc. The OECD/NEA Secretariat for this work is Mr. Ali Nouri (Ali.NOURI@oecd.org) who may be contacted for additional information.

4. Committee on the Safety Of Nuclear Installations (CSNI)

The mission of the CSNI is to assist member countries in maintaining and further developing the scientific and technical knowledge base required to assess the safety of nuclear reactors and fuel cycle facilities. The Committee is made up of senior scientists and engineers with broad responsibilities for safety technology and research programmes and representatives from regulatory authorities. The Committee constitutes a forum for technical exchange and collaboration between organizations that can contribute to activities related to the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee promotes coordination and the establishment of joint undertakings to achieve these goals. The Committee focuses primarily on power reactors and other nuclear installations currently being build and operated and it also considers the safety implication s of scientific and technical developments. The CSNI is organized in working groups as described in Figure 2. The primary interest in this presentation is the Working Group on Operating Experience (WGOE) and its subgroup on Fuel Cycle Safety (FCS).

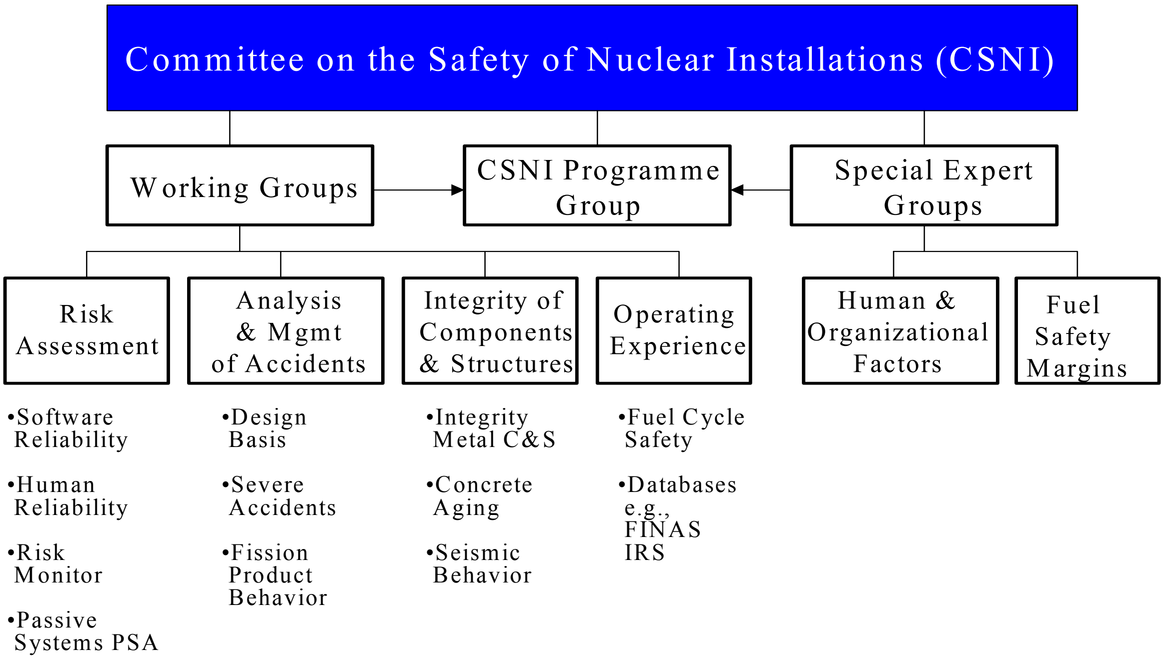
Committee on the Safety of Nuclear Installations (CSNI)
Working Groups
CSNI Programme Group
Special Expert Groups
Risk Assessment
Analysis & Mgmt of Accidents
Integrity of Components & Structures
Operating Experience
Human & Organizational Factors
Fuel Safety Margins
•Software Reliability
•Human Reliability
•Risk Monitor
•Passive Systems PSA
•Design Basis
•Severe Accidents
•Fission Product Behavior
•Integrity Metal C&S
•Concrete Aging
•Seismic Behavior
•Fuel Cycle Safety
•Databases e.g., FINAS IRS

Figure 2. Structure of the Committee on the Safety of Nuclear Installations.

5. Working Group on Operating Experience/Fuel Cycle Safety

The Working Group on Operating Experience is mandated to analyze and develop insights from operating experience and to communicate these insights to the CSNI and interested government and industry bodies. On an annual basis, the WGOE reviews and assesses the safety significance of operating events using information from probabilistic safety assessments of these events when possible. The WGOE is to promote the development of improved techniques and methods for the review of operating events and will utilize these improved practices and methods as they are available. The Fuel Cycle Safety (FCS) expert group has operated since 1976 and since 1999 has functioned as a subgroup to the WGOE. The FCA mission is to advance the understanding of relevant aspects of nuclear fuel cycle safety in OECD Member countries. In pursuing this goal, the FCS meets to:

- exchange information on relevant matters including licensing systems, safety philosophy and safety standards to improve mutual understanding;
- maintain a database on incidents involving fuel cycle facilities (FINAS);
- indicate where further research is needed;
- review and prioritize safety issues;
- prepare state-of-the-art fuel cycle safety reports and
- collaborate with other groups as necessary.

The WGOE/FCS has recently proposed and is currently seeking approval from CSNI for a new activity related to BUC. The proposal is related to the examination of the regulatory situation for BUC and incorporates an internal survey performed by WGOE/FXCS and the CSNI High Level Safety Issues on Spent Fuel Management and Criticality Safety in Fuel Reprocessing. In accordance with the recent increase in burnup of nuclear fuel including increasing initial enrichments, it is apparent that the necessity of adopting BUC in the criticality safety design is becoming larger and larger. The WGOE/FCS has noted that BUC has already been adopted in several countries by the regulatory authorities. The regulatory basis and content appear to differ from country to country. Given the wide-scale research and development activities advancing the understanding of BUC, the WGOE/FCS has determined to make an effort to prepare the regulatory side to incorporate the results of these programs. This concern is similar to that which has prompted the IAEA to organize and host meetings such as this one.

The primary difference between the activity proposed by the WGOE/FCS and that of the IAEA TCM is that the ultimate objective is to determine if it is possible to standardize BUC. Since BUC is complex and will advance in the near future, standardization of the fundamental conditions is preferable. The resulting standardization is expected to be applicable to storage pool, transportation cask, interim storage and reprocessing. The specific activity under consideration is to collect information via the questionnaire given in Table II. The second step will be to use nuclear fuel cycle experts to summarize and analyze the replies to extract items to be used to proceed towards standardizing BUC from a regulatory viewpoint. Care will be taken to proceed with the cooperation of other groups such as the WPNCS/EGBCC and the IAEA TCM to avoid duplication of work.

Information about the current activities and links to publications of the OECD/NEA WGOE/FCS may be found at http://www.nea.fr/html/nsd/csni/wgoe.html. The OECD/NEA Secretariat for this work is Mr. Barry Kaufer(Barry.KAUFER@oecd.org) who may be contacted for additional information.

Table II. WGOE/FCS BUC Questionnaire

Have you adopted the burnup credit already in the regulation?

1.	Facilities where the burnup credit is adopted.	In which facility or equipment is the burnup credit adopted in your country?
2.	Burnup data	There are two kinds of burnup data; one is the reactor management data that is attached to each spent-fuel assembly and the other is the data measured by use of the burnup monitor. Which of these two kinds of data do you use in the regulation for burnup credit implementation.
3.	Precision and accuracy of burnup data	How do you evaluate the precision and accuracy of each burnup data, i.e., reactor management data and measured data? If two data differ beyond the sum of each expected error, how do you treat this assembly?
4.	Maximum allowable multiplication factor	What value is used as the maximum allowable multiplication factor for criticality safety design at facilities where the burnup credit is adopted?
5.	Level of burnup credit adoption	Which level of burnup credit is adopted in your country, actinide only or actinide plus fission products and/or gadolinium effect?
6.	Method to derive multiplication factor from the burnup data	(In order to derive the multiplication factor it is necessary to know the initial enrichment, cooling time and so on of the spent-fuel in addition to the burnup data. How do you know these values and how do you derive the multiplication factor from these values)
7.	Gadolinium effect for burnup credit (especially for countries adopting BWR)	All BWR and some PWR fuel assemblies contain gadolinium. Therefore, in order to evaluate the burnup credit, decreasing effect of gadolinium with increasing burnup may also be evaluated. Do you take into account this gadolinium effect when you evaluate the burnup credit?
8.	Issues to be solved	Do you have any issues for the burnup credit from the regulatory view point?
9.	Adoption or improvement	Do you have any plan to adopt newly or improve the burnup credit in the regulation in the near future? In order to eliminate the back-up confirmation of burnup monitor what conditions do you consider are necessary to be met?
10.	R&D	What kinds of R&D are being carried out to improve the burnup credit?
11.	Other Comments?	

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TECHNICAL TOPICS

(Session 2)

EXPERIMENTAL VALIDATION: ISOTOPIC COMPOSITION AND REACTIVITY CALCULATIONS, HIGH BURNUP FUEL IMPLICATIONS, NUCLEAR DATA QUALITY

(Session 2.1)

Experimental validation of actinide and fission products inventory from chemical assays in French PWR spent fuels

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Abstract. In order to validate fuel inventory calculations, a large experimental programme has been performed in France, based on spent fuel chemical assays. Experimental data are based on chemical analysis measurements from fuel rod cuts irradiated in French PWR reactors and from full assembly dissolutions at the COGEMA/La Hague reprocessing plants. This enables us to cover a large range of UOx fuels with various enrichments in ^{235}U, 3.1% to 4.5%, associated with burnups from 10 GWd/t to 60 GWd/t. Uranium, Plutonium, Americium and Curium isotopes were analysed in PWR samples. Furthermore, Fission Products involved in Burn up Credit studies were measured. Calculation to Experiment comparison has been obtained using the APOLLO2/DARWIN package developed by the CEA. The results are described and discussed in the paper.

1. Introduction

In order to validate fuel inventory calculations, a large experimental programme based on spent fuel chemical analysis has been carried out in France since 1993. Uranium, plutonium, neptunium, americium and curium isotopes have been analysed in PWR samples. Furthermore the 15 fission products selected by the OECD for Burn up Credit (BUC) criticality calculations have been measured.

The available experimental information consists of chemical assays from fuel rod cuts irradiated in French PWR reactors and of solution samples derived from full assembly dissolutions at the COGEMA/La Hague reprocessing plants.

The calculation to experiment comparison have been obtained with the DARWIN package, developed by the CEA and its French industrial partners, and devoted to fuel cycle studies.

2. Presentation of French Tools

The fuel inventory validation is carried out using the DARWIN package described here after, based for PWR studies on the APOLLO2 code and the depletion code PEPIN2.

The DARWIN package

DARWIN is the reference calculation package for the fuel cycle of all types of reactors. It was developed by the CEA and its French partners (COGEMA, EDF and FRAMATOME) to estimate the physics quantities characterizing the burnup fuels from reactors: material balance, decay heat, activity, neutron, γ, α, β sources and spectrum, radiotoxicity.

DARWIN is devoted to all cycle studies, with current fuels (UOx, MOx) or innovative fuels (MIX, APA, PuTh) and for every nuclear road (Pressured Water Reactor, Fast Breeder Reactor, Boiling Water Reactor, Advanced Reactors). DARWIN is also used in the back-end cycle for actinide incineration (SPIN) or long term interim storage studies.

The simplified DARWIN structure, based on new codes and libraries is described in the Figure 1.

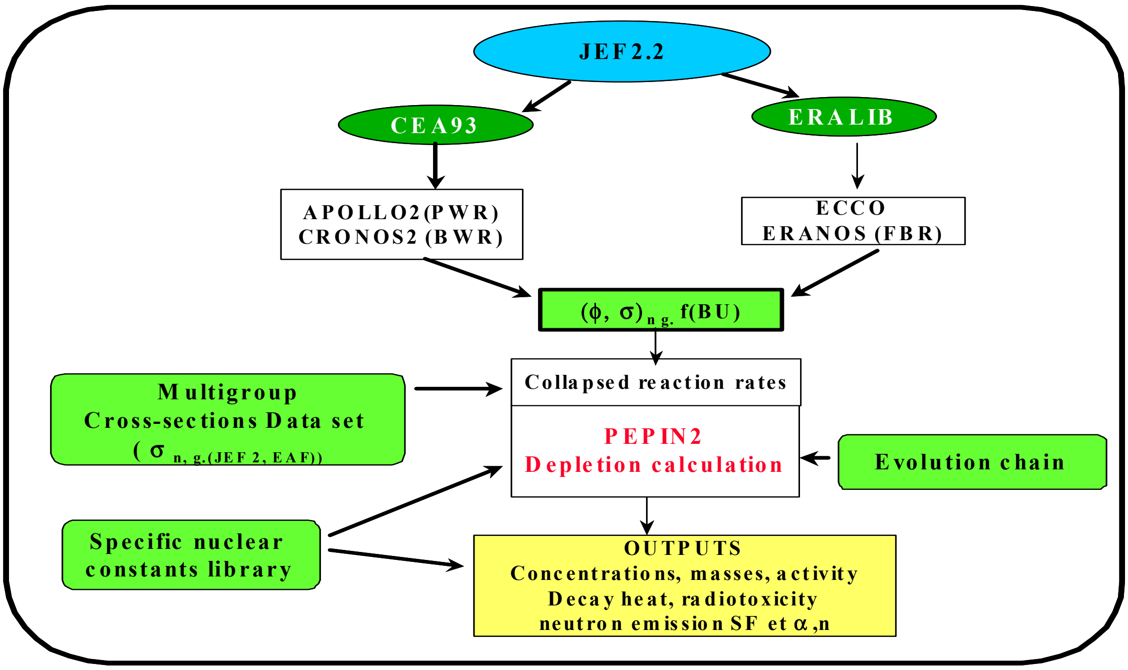
JEF2.2
CEA93
ERALIB
APOLLO2(PWR)
CRONOS2 (BWR)
ECCO
ERANOS (FBR)
(ϕ, σ)n g. f(BU)
Collapsed reaction rates
Multigroup
Cross-sections Data set
(σ n, g.(JEF 2, EAF))
PEPIN2
Depletion calculation
Evolution chain
Specific nuclear
constants library
OUTPUTS
Concentrations, masses, activity
Decay heat, radiotoxicity
neutron emission SF et α,n

FIG. 1. The DARWIN package.

The PEPIN2 program performs the nuclide depletion calculations. Different libraries feed this module

- neutronics data provided by French assembly transport codes APOLLO2 (for PWR studies), ECCO-ERANOS system (for FBR studies) : these data are self-shielded cross sections and neutron spectra;
- nuclear data (decay data, fission and □, n) yields) and evolution chains;
- complementary cross-sections, missing from the transport codes libraries, specially for activation products. They are included in the 'cycle library'.

The basic nuclear data comes from the JEF2.2 European evaluation; in the scope of our study on PWR assemblies, the neutronic data necessary to the depletion module are provided by the APOLLO2 code and its CEA93 library and benefits from its extensive experimental validation achieved in the framework of the PWR lattices.

DARWIN makes possible the retrieval of cumulated reaction rates during irradiation in order to give the origin of every isotope build-up. Furthermore, a "PERTURBATION" of main nuclear data, such as capture cross-section, initial isotopic composition, flux, is also available and allows sensitivity studies.

3. Calculational models

The accuracy of the DARWIN results depends mainly on the APOLLO2 assembly calculation. APOLLO2 solves the integral form of the Boltzmann equation through the collision probability method. We use an APOLLO2 reference calculation route devoted to depletion calculation of 17X17 UO2 PWR assemblies. This calculation route uses the CEA93 cross-section library in 172 groups structure processed from the JEF2 evaluations. Among the themes studied in the optimisation of the assembly calculation, we can mention:

- the radial discretization of the fuel pellet in 4 rings in order to give a faithful representation of the resonant absorption of U238 in the pin and of the actinide and fission product concentration profile;
- the spatial calculation achieved using the UP1 HETE approximation. This enables the probability of leakage PIS and the probability of transmission PSS to be calculated for the true geometry, the interface currents are considered to be linearly anisotropic;
- a grouping of cells with similar flux within a unique 'physical cell';
- the self-shielding of resonant isotopes, with a differentiated treatment for each one depending on the physical characteristics of their cross-sections;
- optimized evolution steps.

The PEPIN2 evolution module then uses the results provided by APOLLO2, self-shielded cross-sections and multigroup spectra - to make up the collapsed library with burnup dependent cross-sections, required in order to characterize the isotopes described in the depletion chains.

4. The experimental database of irradiated PWR fuels

We describe here the main experimental programmes carried out on irradiated UO_X PWR fuels.

The experimental information can be divided into two categories:

- small samples of fuel pins, irradiated in French reactors, with positions in the assembly well characterized. These are sensitive to local irradiation conditions. These time-consuming and expensive experiments provide very accurate results for a limited number of samples;
- dissolution aliquots of entire assembly sets; these are numerous and very different as to the type of assemblies covered, but the information on the irradiation condition is limited.

Four programmes related to fuel samples are used for the experimental validation of actinides and fission products inventory: BUGEY3, FESSENHEIM II, GRAVELINES and CRUAS.

- **BUGEY3** (900 MWe) uses standard fuel with 3.1% initial enrichment and the assembly consists of 17X17 rods with a zircaloy clad. The maximum burnup is 40 GWd/t.
- **FESSENHEIM II** allows the study of UOx fuel (3.1% 235U) with high burnup up to 60 GWd/t.
- **GRAVELINES** is devoted to the extension of the calculation scheme validation for high burnup (five irradiation cycles) with higher enrichment corresponding to 4.5%. This experimental programme on a 900 MWe PWR is the most important being carried out in France today.
- **CRUAS** is devoted to the validation of the URE (Uranium Reprocessed Enriched) fuel, using reprocessed then enriched uranium. It allows the ^{236}U capture validation.

From full assembly dissolutions at COGEMA/La Hague reprocessing plants, we get uranium and plutonium chemical analyses. The assembly involved are 17X17 PWR at 3.1%, 3.25% and 3.45% enrichment with burnup between 25 and 40 GWd/t.

5. Results of the experimental validation

The calculation-experiment comparison (C-E)/E (in %) is summarised below for actinides and important fission products, including the BUC fission products.

The calculation burnup is adjusted using an experimental indicator, namely the ^{145}Nd/^{238}U+^{146}Nd/^{238}U sum for PWR fuel cut analyses and the ^{235}U/^{238}U residual enrichment for the assembly dissolutions (the neodymium chemical analyses are not available).

In the following Table the total uncertainties correspond to the combination of uncertainties on chemical assays and determination of the burnup of the assembly derived from Nd (or ^{235}U) isotopics. When more than one sample is available for the same burnup, the spread of results is also considered.

It is important to remind that for burnup credit studies an underestimation of absorbing fission products buildup or an overestimation of fissile isotopes are conservative, but respectively an overestimation and an underestimation are not.

5.1. Uranium isotope results

Table I points out that uranium isotope concentration are well predicted for Burn up Credit application.

- The ^{235}U depletion is accurately simulated up to very high burnup, even if the quantities involved become very small, and the associated uncertainty and the sensitivity to the cross-section become greater. For the "La Hague" results, the (C-E)/E is negligible because the calculation burnup is adjusted using the ^{235}U/^{238}U residual enrichment.
- The abundance of ^{236}U is underestimated by -4% in standard UOx fuel, principally due to the underestimation of the ^{235}U capture Resonance Integral in JEF2. Many works have been done on this subject and a new evaluation, called Leal-Derrien-Wright-Larson will be introduced in the European JEFF3.0 file. The ^{236}U build-up will be very well predicted: (C-E)/E = -1% ± 1.0% at 60 GWd/t. In URE fuel, the underestimation is no more existing because the ^{236}U concentration is mainly linked, at low burnup, on the initial ^{236}U content and not on the ^{235}U capture.
- The abundance of ^{234}U is well reproduced by calculation if the initial content in the fresh fuel is well known. This is the case, for example for the CRUAS experiment.

Table I. (C-E)/E (%) for 'Uranium' Inventory

Fuel↓	*BU (GWd/t)* ↓	^{234}U/^{238}U	^{235}U/^{238}U	^{236}U/^{238}U
UOx "BUGEY FESSENHEIM" 3.1 % ^{235}U	*20*	2.5 ± 0.6	0.4 ± 1.3	-3.6 ± 1.0
	25	5.2 ± 1.1	0.3 ± 2.4	-3.1 ± 1.2
	40	1.7 ± 0.7	0.04 ± 2.1	-3.2 ± 0.4
	50	-5.9 ± 1.1	0.1 ± 2.7	-4.2 ± 0.1
	60	-2.3 ± 2.3	3.3 ± 7.5	-4.1 ± 0.2
UOx "GRAVELINES" 4.5 % ^{235}U	*30*	-0.3 ± 2.0	-0.7 ± 1.7	-3.4 ± 1.3
	40	0.2 ± 2.2	-0.4 ± 2.8	-3.9 ± 1.0
	50	-4.7 ± 2.2	1.4 ± 4.1	-4.9 ± 0.6
	60	1.4 ± 2.5	0.8 ± 5.3	-4.3 ± 0.3
UOx "La HAGUE" 3.1 % ^{235}U	*30*	-16.6 ± 2.2	-0.01 ± 2.8	-1.4 ± 1.0
	40	-15.7 ± 2.2	0.03 ± 4.1	-1.2 ± 0.6
UOx "La HAGUE" 3.25 % ^{235}U	*35*	-12.5 ± 2.2	-0.01 ± 2.8	-2.4 ± 1.0
	45	-13.7 ± 2.2	-0.01 ± 2.8	-1.8 ± 1.0
UOx "La HAGUE" 3.45 % ^{235}U	*35*	-16.5 ± 2.2	-0.05 ± 2.8	-4.7 ± 1.0
URE "CRUAS" 3.5 % ^{235}U	*15*	0.2 ± 0.5	-0.2 ± 1.0	-0.5 ± 0.4
	25	0.2 ± 0.8	0.8 ± 1.7	-0.3 ± 0.3
	35	-0.5 ± 1.1	0.02 ± 3.1	-1.1 ± 0.2

5.2. *Plutonium isotope results*

Table II. (C-E)/E (%) for 'Plutonium' Inventory

Fuel↓	*BU(GWd/t)* ↓	^{238}Pu/^{238}U	^{239}Pu/^{238}U	^{240}Pu/^{238}U	^{241}Pu/^{238}U	^{242}Pu/^{238}U
UOx "BUGEY FESSENHEIM" 3.1 % ^{235}U	*20*	-12.3 ± 3.2	-0.6 ± 1.1	-0.9 ± 1.6	-4.0 ± 2.1	-6.6 ± 3.9
	25	-9.0 ± 4.6	-0.1 ± 1.5	-0.7 ± 2.1	-3.2 ± 2.9	-5.4 ± 5.0
	40	-6.4 ± 2.0	2.5 ± 0.9	-1.9 ± 0.8	-1.0 ± 1.2	-7.6 ± 2.2
	50	-4.2 ± 1.6	2.0 ± 1.1	-1.2 ± 0.6	-0.4 ± 1.2	-6.1 ± 1.8
	60	-8.7 ± 2.5	3.1 ± 2.4	-0.7 ± 0.9	-0.3 ± 2.6	-8.7 ± 2.9
UOx "GRAVELINES" 4.5 % ^{235}U	*30*	-12.2 ± 5.3	-1.6 ± 0.6	-2.2 ± 2.3	-5.0 ± 2.7	-6.8 ± 5.5
	40	-10.0 ± 5.0	-0.4 ± 0.2	-1.9 ± 2.0	-4.2 ± 2.0	-6.9 ± 4.8
	50	-10.3 ± 4.5	+1.6 ± 0.1	-1.8 ± 1.5	-3.0 ± 1.2	-8.6 ± 4.3
	60	-9.9 ± 4.3	+1.9 ± 0.2	-1.5 ± 1.2	-2.6 ± 1.0	-7.2 ± 3.9
UOx "La HAGUE" 3.1 % ^{235}U	*30*	-14.9 ± 5.0	+0.9 ± 0.2	-1.7 ± 2.0	-2.5 ± 2.0	-11.7 ± 4.8
	40	-15.4 ± 4.5	+0.7 ± 0.1	-1.0 ± 1.5	-6.7 ± 1.2	-11.9 ± 4.3
UOx "La HAGUE" 3.25 % ^{235}U	*35*	-14.1 ± 5.0	1.1 ±0.2	-1.7 ± 2.0	-1.9 ± 2.0	-11.1 ± 4.8
	45	-11.7± 4.5	2.1 ± 0.1	-0.5 ± 1.5	-0.5 ± 1.2	-9.5 ± 4.3
UOx "La HAGUE" 3.45 % ^{235}U	*35*	-18.4 ± 5.0	-0.1 ± 0.2	-2.5 ± 2.0	-2.6 ± 2.0	-11.5 ± 4.8
URE "CRUAS" 3.5 % ^{235}U	*15*	-2.8 ± 3.9	1.7 ± 1.2	-2.2 ± 2.6	0.5 ± 3.6	-5.1 ± 5.8
	25	-2.8 ± 3.4	1.0 ± 1.0	-0.6 ± 2.2	-1.6 ± 2.7	-4.9 ± 5.1
	35	-1.7 ± 3.3	1.9 ± 1.0	-0.5 ± 1.7	-0.6 ± 1.8	-4.2 ± 4.3

— The abundance of ^{238}Pu is underestimated by about 10% in standard UOx fuels. However, in BUC application, this underestimation is not very penalizing because the ^{238}Pu is not a great contributor to the actinide BUC negative reactivity worth (10% of the total minor actinide reactivity worth at 40 GWd/t).

- The ^{239}Pu is well predicted within 2% accuracy. The slight overestimation increasing with burnup could indicate an underestimation of its (n, γ) cross-section in JEF2. This overestimation is conservative for BUC studies.
- The prediction of ^{240}Pu is very well estimated, confirming the correct modelling of Doppler/self-shielding resonance for 240Pu at 1 eV. The negative reactivity worth of this most poisoning isotope will be very well represented in our BUC calculations.
- ^{241}Pu is slightly underestimated. Its prediction is highly sensitive to ^{239}Pu capture. This result could point out again a slight under-estimation of ^{239}Pu (n, γ) cross-section in JEF2 evaluation. We must taking into account this underestimation leading to a non-conservative BUC calculation.
- ^{242}Pu is underestimated. This result shows that ^{241}Pu capture cross-section in JEF2.2 is under-estimated. Hence, a new evaluation of Pu^{241} will be introduced in JEFF3.0: the increase of the capture will correct the ^{242}Pu, ^{243}Am and ^{244}Cm build-up.

5.3. Minor actinides results

Table III. (C-E)/E (%) for 'Neptunium' Inventory

Fuel↓	*BU (GWd/t)↓*	$^{237}Np/^{238}U$
UOx "BUGEY FESSENHEIM" 3.1 % ^{235}U	*20*	-10.5 ± 4.4
	25	-7.2 ± 4.3
	40	-0.6 ± 2.1
	50	-3.2 ± 2.5
	60	3.0 ± 3.1
UOx "GRAVELINES" 4.5 % ^{235}U	*30*	-0.3 ± 4.7
	40	-2.4 ± 4.2
	50	-4.4 ± 3.4
	60	-3.0 ± 3.1
URE "CRUAS" 3.5 % ^{235}U	*15*	-4.1 ± 3.2
	25	-0.2 ± 3.1
	35	3.9 ± 3.0

The abundance of ^{237}Np is underestimated, due on one hand to the underestimation of the ^{235}U capture Resonance Integral and in the other hand to the underestimation of $(n,2n)^{238}U$ in JEF2.

However, the accuracy of our calculations is conservative for BUC UOx applications.

- The ^{241}Am, formed by decaying of ^{241}Pu, is reasonably estimated for BUC UOx applications.
- ^{242}Am is underestimated by about -20%. This suggests increasing ^{241}Am capture cross-section. We must also improve the knowledge of the branching ratio of ^{241}Am to ^{242}Am in the epithermal range.
- The underestimation in ^{242}Pu generates the underestimation for ^{243}Am.

Table IV. (C-E)/E (%) for 'Americium' Inventory

Fuel↓	*BU (GWd/t)*↓	^{241}Am/^{238}U	^{242m}Am/^{238}U	^{243}Am/^{238}U
UOx "BUGEY FESSENHEIM" 3.1 % ^{235}U	*20*	-6.3 ± 3.4	-25.2 ± 10.1	-10.6 ±7.8
	25	-2.9 ± 3.1	-9.9 ± 3.7	-10.6± 7.5
	40	-3.0 ± 1.2	-17.3 ± 3.0	-4.5 ± 3.4
	50	-13.2 ± 2.6		-10.5 ± 2.9
	60	-3.9 ± 2.4	-21.1 ± 11.4	-8.8 ± 4.6
UOx "GRAVELINES" 4.5 % ^{235}U	*30*	-5.0 ± 2.8	-33.0 ± 5.0	-10.3 ± 7.7
	40	-3.5 ± 2.1	-25.0 ± 3.2	-7.9 ± 7.0
	60	+1.3 ± 1.0	-16.2 ± 2.4	-4.8 ± 5.6
URE "CRUAS" 3.5 % ^{235}U	*35*	-1.3 ± 2.3	-17.0 ± 2.8	-4.8 ± 6.4

Table V. (C-E)/E (%) for 'Curium' Inventory

Fuel↓	*BU (GWd/t)*↓	^{243}Cm/^{238}U	^{244}Cm/^{238}U	^{245}Cm/^{238}U	^{246}Cm/^{238}U	^{247}Cm/^{238}U
UOX "BUGEY FESSENHEIM" 3.1 % ^{235}U	*20*	-42.7 ± 11.3	-33.7 ± 8.0	-27.5 ± 10.4	-81.2 ± 8.7	/
	25	-34.5 ± 8.9	-19.9 ± 9.1	-19.2 ± 10.9	-7.7 ± 17.8	/
	40	-9.6 ± 5.8	-12.1 ± 5.4	-2.3 ± 7.8	-27.8 ± 7.2	/
	50	-28.3 ± 10.2	-21.6 ± 8.1	-18.2 ± 13.4	-31.0 ± 11.3	/
UOX "GRAVELINES" 4.5 % ^{235}U	*30*	-28.1 ± 8.0	-18.5 ± 10.4	-20.0 ± 12.4	-25.0 ± 15.5	-25.6 ± 18.2
	40	-26.5 ± 8.0	-15.4 ± 9.8	-16.8 ± 11.5	-23.0 ± 15.2	-30.2 ± 18.0
	60	-24.5 ± 8.0	-12.4 ± 8.4	-11.4 ± 9.3	-23.1 ± 13.4	-30.2 ± 16.3
URE "CRUAS" 3.5 % ^{235}U	*35*	-5.4 ± 6.3	-2.03 ± 9.3	-0.05 ± 11.2	-12.2 ± 12.8	-12.9 ± 31.7

The curium isotopes are underestimated. The underestimation in ^{242}Pu generates an underestimation for ^{243}Am, and consequently for ^{244}Cm and ^{245}Cm. This underestimation is acceptable for BUC studies on UOx fuels but recent studies on ^{241}Pu JEF2 capture cross-section enable to partially correct this problem.

The underestimation in ^{243}Cm build-up needs also to be corrected. Studies on ^{242}Cm capture cross-section are in progress.

The concentrations of the most important major and minor actinides, poisoning such as ^{236}U, ^{237}Np, ^{240}Pu, ^{241}Am, or fissile as ^{235}U, ^{239}Pu, ^{241}Pu are well calculated and can be used directly in the CRISTAL package for BUC calculations in UOx fuels; the slight underestimation of ^{241}Pu build-up, leading to a non-conservative calculation, needs to be taking into account by the way of a correction factor on its concentration.

5.4. Fission products results

5.4.1. The samarium isotopes

Table VI. (C-E)/E (%) for 'Samarium' Inventory

Fuel↓	*BU (GWd/t)↓*	147Sm/238U	149Sm/238U	150Sm/238U	151Sm/238U	152Sm/238U
UOX "BUGEY" 3.1 % ^{235}U	*20*	-4.9 ± 1.5	4.5 ± 4.7	-3.7 ± 2.6	-3.1 ± 1.2	0.4 ± 1.9
	40	-5.9 ± 1.0	-3.4 ± 23.3	-3.6 ± 2.2	7.3 ± 1.5	1.1 ± 1.7
UOX "GRAVELINES" 4.5 % ^{235}U	*40*	-6.5 ± 1.1	-8.6 ± 8.4	-3.2 ± 2.3	3.6 ± 1.1	1.4 ± 1.8
	50	-7.0 ± 1.0	5.6 ± 4.8	-4.8 ± 2.1	7.9 ± 1.1	3.5 ± 1.6
	60	-7.4 ± 0.9	-3.5 ± 10.0	-4.1 ± 1.8	12.8 ± 1.0	5.6± 1.7

Except the slight overestimation of ^{151}Sm build-up, the samarium isotope calculations are satisfactory.

- The ^{147}Sm is underestimated. This isotope is formed by decaying of ^{147}Pm, itself formed by cumulative fissions, and disappeared by neutron capture. The interpretation of ^{147}Sm samples oscillations in the MINERVE reactor shows that the ^{147}Sm capture cross-section is well calculated (C-E)/E = -0.3% ± 3.2%; furthermore the sensitivity of ^{147}Sm build-up is very low because of the low value of the cross-section (57 barns at 2200m/s); so we can conclude that the underestimation of ^{147}Sm is due to its formation. The capture cross-section of ^{147}Pm and ^{147}Nd cumulative fission yields must be investigated.
- The ^{149}Sm seems to be underestimated but the chemical uncertainties is very high and the ^{149}Sm build-up is very sensitive to the power history mesh. However this slight underestimation is confirmed by rigorous APOLLO2 calculation describing precisely the irradiation history.
- The ^{150}Sm is underestimated, varying in the direct ratio of its father the ^{149}Sm.
- ^{151}Sm is overestimated. This is probably due to an overestimation of ^{151}Pm cumulative fission yields.

— ^{152}Sm is well predicted but we can notice a slow drift toward overestimation when the burnup increases. This could be due to ^{151}Sm calculation overestimation but also to an underestimation of the ^{152}Sm capture cross-section as suggested by the MINERVE oscillation experiment. More studies are needed to conclude.

5.4.2. The neodymium isotopes

Table VII. (C-E)/E (%) for 'Neodymium' Inventory

Fuel↓	*BU (GWd/t)*↓	^{143}Nd/^{238}U	^{144}Nd/^{238}U	^{148}Nd/^{238}U	^{150}Nd/^{238}U
UOX "BUGEY FESSENHEIM" 3.1 % ^{235}U	*20*	0.3 ± 1.3	-2.0 ± 1.8	0.2 ± 1.7	-6.7 ± 1.8
	25	0.3 ± 1.7	-2.2 ± 2.5	1.5 ± 2.2	0.7 ± 2.5
	40	/	/	0.2 ± 1.7	/
	50	2.4 ± 1.8	0.2 ± 2.9	-0.6 ± 2.6	-0.8 ± 3.2
UOX "GRAVELINES" 4.5 % ^{235}U	*40*	0.5 ± 1.4	-1.9 ± 2.5	1.6 ± 2.1	0.7 ± 2.4
	50	1.6 ± 1.0	-2.6 ± 2.6	1.5 ± 2.1	0.5 ± 2.4
	60	1.9 ± 0.4	-2.4 ± 1.5	1.4 ± 1.2	0.4 ± 1.4
URE "CRUAS" 3.5 %	*15*	0.0 ± 1.8	-1.8 ± 2.1	0.7 ± 2.1	-0.3 ± 2.3
	25	0.6 ± 1.6	-2.1 ± 2.2	0.7 ± 2.1	-0.1 ± 2.2
	35	0.7 ± 1.2	-2.3 ± 2.3	1.3 ± 2.1	0.6 ± 2.3

The Nd isotopes are very well calculated.

In Table VII, the ^{143}Nd is the unique FP involved in BUC studies. Its build-up is slightly overestimated for high burnups : +2%. At these burnups, ^{143}Nd concentration is sensitive to its capture cross-section. This result confirms an under-estimation of the thermal capture cross-section of ^{143}Nd. The same conclusion is drawn from the interpretation of ^{143}Nd samples oscillated in Minerve lattices. A new evaluation of ^{143}Nd will be introduced in the future JEFF3 file, in order to increase by 4% the (n, γ) cross-section in the 0-0.2eV energy range

One can notice a slight underestimation of ^{144}Nd concentration due probably to an underestimation of ^{144}Ce cumulative fission yields.

The most absorbing FP, involved in BUC studies, is the ^{133}Cs. A slight and steady under-prediction by -4% is noticed. This is due to an underestimation of ^{133}Xe fission yields for ^{235}U and ^{239}Pu in the JEF2.2 library.

^{137}Cs is not an absorbing FP but it is used as a burnup monitor and must be calculated with about 2% accuracy. Our calculation leads to an under-prediction of the ^{137}Cs concentration ; this is due to underestimation of ^{137}Cs fission yields in JEF2.2.

5.4.3. The cesium isotopes

Table VIII. (C-E)/E (%) for 'Cesium' Inventory

Fuel↓	*BU (GWd/t)*↓	^{133}Cs/^{238}U	^{134}Cs/^{238}U	^{135}Cs/^{238}U	^{137}Cs/^{238}U
UOX "BUGEY FESSENHEIM" 3.1 % ^{235}U	*20*	-2.7 ± 2.0	-8.1 ± 4.6	18.8 ± 0.6	-1.8 ± 2.3
	25	-3.9 ± 1.9	-14.8 ± 4.8	15.2 ± 0.9	-4.0 ± 2.2
	40	-8.5 ± 1.8	-9.3 ± 4.9	3.9 ± 0.7	-1.2 ± 2.3
	50	-1.3 ± 1.6	-3.7 ± 5.6	5.1 ± 0.9	-4.4 ± 2.2
	60	-2.2 ± 1.5	2.1 ± 6.1	1.8 ± 0.9	-3.6 ± 2.3
UOX "GRAVELINES" 4.5 % ^{235}U	*30*	-4.9 ± 2.0	-11.7 ± 4.0	-1.9 ± 1.9	-5.7 ± 2.2
	40	-3.3 ± 2.0	-10.4 ± 3.8	-2.4 ± 2.1	-5.0 ± 2.3
	50	-3.9 ± 1.8	-8.2 ± 3.5	-3.9 ± 2.2	-6.2 ± 2.3
	60	-3.4 ± 1.2	-8.7 ± 2.9	-3.3 ± 1.6	-5.2 ± 1.5

5.4.4. The metallic fission product isotopes

Table IX. (C-E)/E (%) for 'Metallic Fission Products' Inventory

Fuel ↓	*BU (GWd/t)* ↓	^{99}Tc/^{238}U	^{95}Mo/^{238}U	^{101}Ru/^{238}U	^{103}Rh/^{238}U
UOX "BUGEY" 3.1 % ^{235}U	*20*	-7.6 ± 3.1	-3.7 ± 2.7	2.9 ± 2.9	3.5 ± 2.8
	40	4.8 ± 3.0	8.4 ± 2.6	15.2 ± 2.9	13.9 ± 2.6
UOX "GRAVELINES" 4.5 % ^{235}U	*40*	-0.6 ± 3.1	5.5 ± 2.7	5.0 ± 2.9	3.4 ± 2.7
	50	2.6 ± 3.1	2.6 ± 2.6	6.4 ± 2.3	4.6 ± 2.5
	60	3.6 ± 3.0	2.1 ± 2.6	0.9 ± 2.9	3.0 ± 2.3

Metallic fission products isotopic prediction is quite satisfactory in UOx spent fuels since they are in agreement with the experimental standard deviation. However, it should be noted that the dissolution of metallic fission product such as ^{95}Mo, ^{99}Tc, ^{101}Ru and ^{103}Rh could lead to non-soluble deposits. A new programme of dissolution, more accurate, is planned in order to confirm the C/E discrepancies.

5.4.5. The europium and gadolinium isotopes

In order to check ^{155}Gd build-up which is an important poisoning FP in BUC, the ^{154}Eu, ^{155}Eu and ^{154}Gd were also investigated. DARWIN results are presented in Table X.

Table X. (C-E)/E (%) for 'Europium And Gadolinium' Inventory

Fuel↓	*BU (GWd/t)* ↓	^{153}Eu/^{238}U	^{154}Eu/^{238}U	^{155}Eu/^{238}U	^{154}Gd/^{238}	^{155}Gd/^{238}U	^{156}Gd/^{238}U
UOX "BUGEY" 3.1 % ^{235}U	*20*	6.5 ± 3.0	40.9 ± 4.6	8.9 ± 3.8	40.3 ±4.5	-3.1 ± 3.4	
	40	11.8 ± 2.6	80.4 ± 4.2	8.9 ± 4.2	82.4 ±3.8	0.1 ± 3.7	
UOX "GRAVELINES" 4.5 % ^{235}U	*40*	8.9 ± 3.0	54.2 ± 6.4	13.9 ± 4.2	/	4.3 ± 3.8	
	50	11.7 ± 2.8	74.8 ± 6.6	15.4 ± 4.5	72.6 ± 6.0	8.1 ± 3.9	-22.0 ± 3.9
	60	16.4 ± 2.5	94.0 ± 6.2	18.2 ± 4.4	/	11.9 ± 4.0	-15.0 ± 4.0

^{155}Gd is strongly over-predicted but agrees with its father, ^{155}Eu, which is produce by capture on ^{154}Eu and directly by fission. Furthermore the ^{156}Gd, daughter of ^{155}Eu is under-predicted.

^{154}Eu is also strongly overestimated and the same level of overestimation is found on its daughter ^{154}Gd. These remarks point out that modification of ^{154}Eu and ^{155}Eu capture cross-section is needed in JEF2 file.

Recent studies on ^{154}Eu and ^{155}Eu capture cross-section confirm that data used in JEF2.2 file are not satisfactory and will be replaced by ENDFB6 evaluations in the new JEFF3.0 file. The use of these evaluations leads to the following results on Eu and Gd build-up in GRAVELINES UOx fuels.

Table XI. New (C-E/E) (%) results using σc from ENDFB6.7 for ^{154}EU and ^{155}EU in GRAVELINES UOx fuels

Burnup (GWd/t)	*40*	*50*	*60*
^{154}Eu/^{238}U	-7.1%	-4.1%	0.8%
^{155}Eu/^{238}U	13.6%	13.5%	15.5%
^{154}Gd/^{238}U		-3.9%	
^{155}Gd/^{238}U	4.0%	6.2%	9.1%
^{156}Gd/^{238}U		-3.8 %	2.4 %

The use of both europium 154 and europium 155 capture cross-section from the ENDFB6.7 evaluation leads to a better gadolinium 155 calculation build-up but a residual overestimation remains; this could be due to overestimation of ^{155}Sm cumulative fission yields (leading to the overestimation of ^{155}Eu) and/or to an underestimation of the capture cross-section of the ^{155}Gd itself.

The ^{154}Eu concentration, used as a burnup monitor for long cooling time, is well improved with a slight underestimation; the conclusion is the same for ^{154}Gd.

The ^{156}Gd calculation is also well improved by the ^{155}Eu capture cross-section correction.

6. Conclusion

This paper has described the CEA experimental program and associated results devoted to the validation of actinides and fission products build-up in PWR UOx spent fuel cycle. The qualification range extends up to 4.5% ^{235}U enrichment and high burnup fractions up to 60 GWd/t.

The depletion code DARWIN, based on the powerful code APOLLO2, has shown his capability to simulate the fuel inventory versus burnup. So, the DARWIN package is well suited for most of burnup credit nuclides inventory, except a slight underestimation of ^{241}Pu and overestimation of europium and gadolinium isotopes. However, the introduction in the future JEFF3.0 file of new evaluations, such as ^{241}Pu, $^{153\text{-}154}$Eu and ^{155}Eu will highly improved our C/E discrepancies.

An industrial package, named 'CIRACUSE', devoted to burnup credit studies, linking automatically DARWIN and CRISTAL, is in development; it will include correction factors applied to nuclide concentrations deduced from the experimental validation.

The P.I.E. data base is currently being extended to higher burnups with PWR rod cuts extracted after 5, 6 and 7 irradiation cycles (up to 80 GWd/t).

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Improvement of the BUC-FP nuclear data in the JEFF library

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Abstract. The aim of this study is to provide recommendations on the nuclear data and new evaluations for the main BUC fission products. The information given by integral experiments allows to select the best evaluation amongst the most recent one: JEF2.2, JENDL3.2, ENDFB6.7 and BROND2.2. When the evaluations do not follow the trends given by integral experiments, specific modifications of the existing evaluation consistent with the differential measurements are proposed in the thermal and resolved range. A rigorous fit of available differential experiments using the integral information has not been performed in this study. A quick overview of the measurements (generally the thermal capture cross-section at 0.0253 eV and the resonance parameters of the first large resonance) has allowed us to find simple solutions to correct the cross-section.

Trends given by integral experiments

This study uses two kinds of integral experiments: oscillation of separated fission product samples in the center of Minerve thermal reactor and Post-Irradiation Experiment (PIE) performed in French PWRs. The calculation route used to analyse these experiments is based on the deterministic code APOLLO2 [APO99] developed by the CEA. The neutron cross-section library called CEA93 is based on JEF2.2.

A great effort has been made to define an optimised calculation scheme that avoids significant bias [CHA00]. The deterministic flux calculation was compared to a continuous-energy Monte-Carlo calculation (performed with the TRIPOLI4 code [TRI94]). Furthermore, the depletion calculation scheme for P.I.E interpretation has been studied in great detail to assess uncertainties and get a limited calculation bias on isotope build-up and reaction rates at every burnup.

The trends given by the analysis of these integral experiments are described in more detail in references [CHA01], [SAN99], [SAN00], [THI99]. The following sections summarize the main trends obtained for Cs-133, Nd-143, Sm-149, Eu-154, Eu-155 and Gd-155.

In the Minerve reactor, three experiments were performed in the framework of Burnup Credit program [SAN97]: R1-UO2 devoted to BUC investigation in storage pool and PWR-assembly transportation, R2-UO2 with a softer spectrum corresponding to the optimum moderation-ratio in a fuel dissolver and R1-MOX that was carried out to simulate a MOX spectrum.

Table I presents the C/E discrepancies on separated FP sample reactivity worth in the three Minerve configurations for the fission product studied in this work [THI99].

Table I. (C-E)/E Discrepancies in % on Separated FP Sample Reactivity worth in the MINERVE experiments with JEF2.2.

Fission product	R1-MOX	R1-UO2	R2-UO2
Cs133	+6.5 ± 2.5	+5.2 ± 2.5	+4.8 ± 2.8
Nd143	-0.7 ± 4.6	-4.0 ± 2.5	-6.0 ± 3.0
Sm149	-6.2 ± 3.6	-5.3 ± 2.4	-4.2 ± 2.9
Gd155	-4.5 ± 4.3	-2.5 ± 2.9	-6.4 ± 4.0

Fuel irradiated assays are also a powerful tool to validate cross-section averaged in a realistic neutron spectrum. Four experiments are selected within the French PWR PIE programme : Bugey, Fessenheim2, Gravelines (17*17 assembly loaded with UOX fuel) and a MOX assembly from SLB1 900Mwe PWR [CHA-SAN00]. Trends on FP capture cross-section can be derived from the analysis of fission product build-up. The detailed results of the irradiated fuel analysis are described in the reference [CHA01]. For the selected fission product, the trends given by P.I.E are consistent with the Minerve results.

For example, The P.I.E has provided meaningful information about Eu155 isotope (which was not oscillated in Minerve). Large discrepancies were observed on Europium and Gadolinium isotopes build-up with JEF2.2 library as shown in Table II.

Table II. C/E-1 Discrepancies in % from French P.I.E for EU and GD Isotopes with JEF2.2 Neutron Cross-Section Library.

Burnup GWd/t	30 UOX	30 MOX	40 UOX	40 MOX	50 UOX	60 UOX
154Eu/Cs137	+41%		+62%	+60%	+84%	+100%
154Eu/U238	+50%	+13%	+54%	+42%	+75%	+94%
154Gd/U238	+50%	+14%		+45%	+73%	
155Eu/U238	+9%	+62%	+14%	+20%	+16%	+18%
155Gd/U238	-1%	+54%	+4%	+14%	+8%	+12%
156Gd/U238					-22%	-15%

The Minerve experiments and the P.I.E results for the investigated FPs suggest the following trends on JEF2.2 data:

- Cs-133 capture cross-section in the large first resonance is over-estimated (about 6 %);
- Sm-149 capture cross-section in the thermal and resonance range is under-estimated (about 5%);
- Nd-143 thermal capture cross section is under-estimated by about 3-5%;
- Eu-154 and Eu-155 capture resonance integrals are not realistic (strong underestimation of the capture cross-section in the resolved range for both nuclei).

Improvements of JEF2.2 capture cross-section for burnup credit calculations

Cs-133

The Minerve experiments have highlighted an overestimation of the Cs-133capture cross section given in JEF2.2. As shown in Table III and in the left part of Fig.1, the thermal capture cross section has been fixed around 29 barns in all the investigated evaluations. This value is in agreement with the measurement performed by Pomerance in 1951 [POM51]. The capture resonance integrals are also given in Table III and compared to the JEF2.2 result. It can be seen that both JENDL3.2 and ENDFB6.7 follow the trend of the integral experiments, i.e a decrease of the capture cross section. Nevertheless, the decrease proposed by JENDL3.2 seems to be too strong. In addition, since the first resonance (around 5.9 eV) represents about 78% of the capture resonance integral, a special care was devoted to this resonance (see right part of Fig.1).

Table III. Survey of the Thermal Capture Cross Sections and Capture Resonance Integral (RI) given in the investigated evaluations

	σ_{capt}(b)	RI (b)	(RI-RI$_{jef2.2}$) –1 (%)
JEF2.2	29.086	438	0
JENDL3.2	29.0	396	-9.6
ENDFB6.7	29.0	421	-3.9

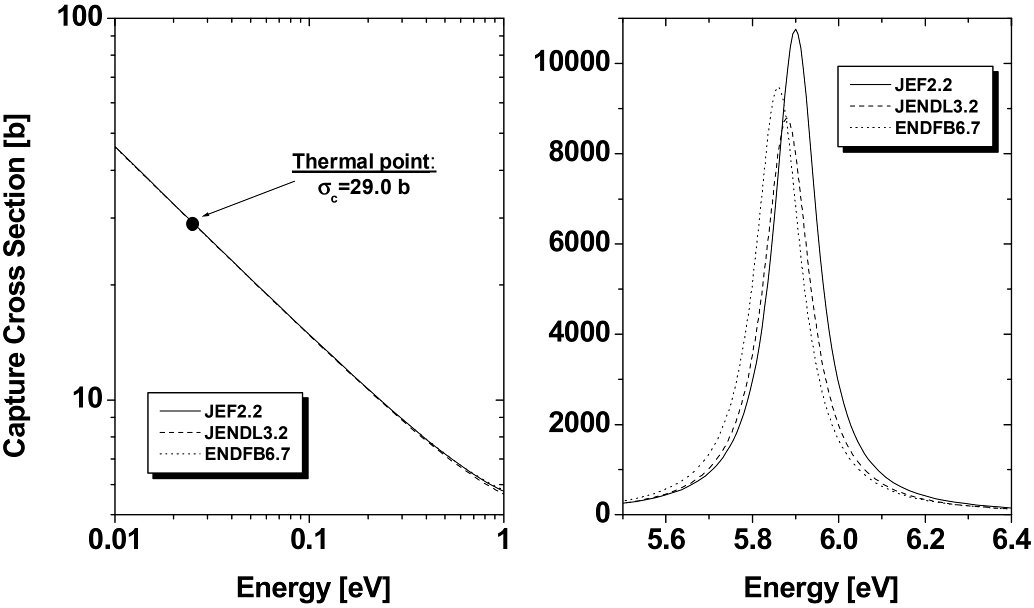

FIG. 1. Cs-133 Capture cross-section given by the evaluations in the thermal region (left) and for the first strong resonance (right).

The resonance parameters for this resonance are mentioned in Table IV and compared to the parameters deduced from the experiment carried out in 1990 by Nakajima et al. [NAK90]. This experiment was performed with a very high energy resolution and is certainly the best capture measurement available up to now in this energy range.

It can be observed from Table IV that Nakajimas's measurement has been incorporated in ENDFB6.7 evaluation (see also Ref. [OH00]). Subsequently, for Cs-133 fission product, we propose to replace JEF2.2 evaluation by the ENDFB6.7 one.

Table IV. Resonance parameters of the first strong resonance given by the evaluations. the parameters obtained from the NAKAJIMA'S measurement [NAK90] are also mentioned

	Eres. (eV)	Spin	Γtot (meV)	Γn (meV)	Γγ (meV)
JEF2.2	5.90	3	122.119	7.119	115
JENDL3.2	5.88	3	130.103	6.103	124
ENDFB6.7	5.86	3	129.562	6.562	123
Nakajima	5.86	3	129.380	6.38 ± 0.26	123 ± 6

Sm-149

The calculation of the Sm-149 absorption rate in a thermal spectrum depends mainly on its thermal capture cross-section at 0.0253 eV and on the resonance parameters of the first strong resonance at about 0.1 eV (see Fig. 2).

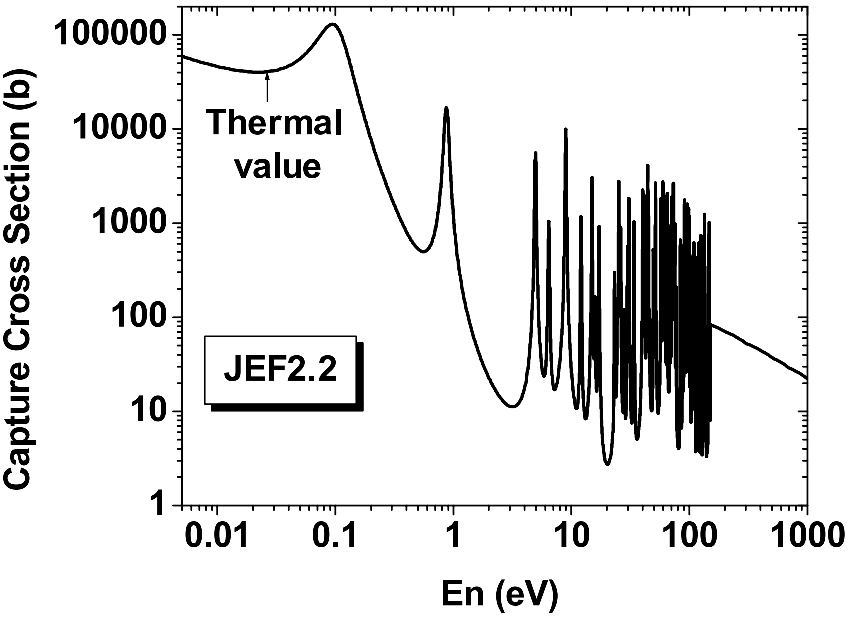

FIG. 2. Sm-149 Capture cross-section from JEF2.2.

Table V presents a comparison of the capture cross-section at 0.0253 eV and the resonance integral in the most recent evaluated files available. There are no significant differences between the evaluations.

Table V. Sm-149 Thermal capture and resonance integral from the evaluations [JEF14].

	Thermal cross-section (barns)	relative difference (%)	Resonance Integral (barns)	relative. difference (%)
JEF2.2	40446	0 %	3487	0 %
ENDFB6.7	40524	+0.19 %	3484	+0.19 %
JENDL3.2	40526	+0.20 %	3494	+0.20 %
BROND2.2	40499	+0.13 %	3466	+0.13 %

We have investigated the experimental measurements of the Sm-149 thermal capture cross-section and the thermal maxwellian spectrum integral. As shown in Table VI, the four thermal capture measurements reported in EXFOR Database (mainly performed by the pile oscillation method) indicate an error weighted average value of $\sigma\gamma$ = (41085 ± 475)b which is greater than the values reported in the evaluation and compatible with the trends given by Minerve. One can notice that the Inghram et al. and Anikita et al. measurements are not consistent with the Aitken et al. and Pattenden experimental values.

Table VI. Sm-149 Thermal capture measurements

Experimental Measurements	Thermal capture at 0.0253 eV (barns)
Inghram et al. (1950) [ING50]	47000 ± 2000
Anikita et al. (1958) [ANI58]	49000 ± 800
Aitken (1961) [AIT61]	42300 ± 1200
Pattenden (1958) [PAT58]	39900 ± 600 b
average value	41085 b ± 475 b

In EXFOR, several measurements of the Thermal maxwellian spectrum Integral are also reported. It is difficult to derive a clear trend from these measurements because the values given by the various experimentalists are very discrepant (from 85000 b as reported by Melaika et al. [MEL55] up to 65000 b as reported by Smither et al. [SMI74]).

In order to modify the evaluation and increase thermal capture cross-section, a consistent fit of the thermal range and the first resonance would be the best solution because the first large resonance is very close to 0.0253 eV. A rigorous fit has not been performed in this work but a close look at the resonance parameters measurements allows us to propose a straightforward correction of the evaluations.

The comparison of the resonance parameters for the first resonance is shown in Table VII. JEF2.2 and ENDFB6.7 follow the Mughabghab recommendation [MUG84]. The resonance parameters from JENDL3.2 and BROND2.2 are slightly different but lead to the same resonance strength value $S = g\,\Gamma n \Gamma\gamma / \Gamma t$.

Table VII. Sm-149 Resonance parameters for the first resonance reported in the evaluations

	Eres (eV)	Γtot (meV)	Γn (meV)	Γγ (meV)
JEF2.2	0.0973	61.03	0.5333	60.5
ENDFB6.7	0.0973	61.03	0.5333	60.5
JENDL3.2	0.0973	63.42	0.525	62.9
BROND2.2	0.09735	62.27	0.5289	61.74

Three measurements of the first Sm-149 resonance parameters are reported [MAR58], [AKY68] and [WIN75]. The measurements are very consistent with the various evaluations as shown in Table VIII. However, compared to the evaluations the slight increase of Γn is compatible with differential measurements. This slight increase could lead to an increase of the thermal capture cross-section.

Table VIII. Sm-149 Resonance parameters for the first resonance from the experiments (source exfor database)

	Eres (eV)	Γtot (meV)	Γn (meV)	Γγ (meV)
Marshak (1958) [MAR58]	0.0976 ± 0.0005	64.1	0.5	63.6 ± 1.0
Akyuez (1968) [AKY68]	0.0973			62.9 ± 1.3
Winiwarter (1975) [WIN75]	0.0973	61.03	0.5333	60.5

To conclude, the thermal capture value suggested by the Minerve experiments is consistent with the average value deduced from differential measurements: $\sigma\gamma = 41085$ b. Therefore, this value is recommended in the evaluated file since it allows improving BUC calculations. A straightforward way to increase σγ in JEF2.2 is to increase Γn of the first resonance because it is consistent with resonance parameter measurements. A consistent fit of the thermal capture value and the first resonance would be more rigorous.

Nd-143

The Nd-143 capture reaction rate in a thermal spectrum is directly linked to the thermal capture cross-section at 0.025 eV. Indeed, the first resonance is located at 55 eV, far from the thermal range (see Fig. 3) and has no significant influence on the burnup credit calculations.

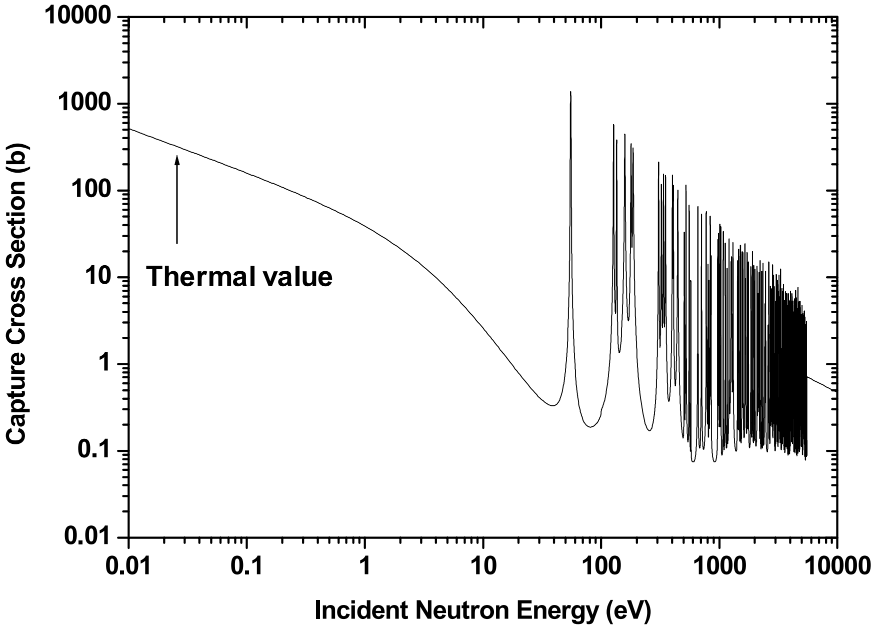

FIG. 3. Nd143 Capture cross-section from JEF2.2.

Table IX shows that the thermal capture values at v = 2200 m/s are very close in the evaluations and consistent with the Mughabghab recommendation [MUG84]. No evaluation can satisfy the trends given by the MINERVE experiments and the P.I.E.

Table IX. Nd-143 Thermal capture cross section reported in the evaluations [JEF14]

	thermal cross-section(barns)	relative difference (%)
JEF2.2	323.3	0 %
ENDFB6.7	325.0	+0.52 %
JENDL3.2	325.1	+0.55 %
BROND2.2	323.2	-0.03 %

The compilation of the experimental measurements of the Nd-143 capture cross-section at 0.0253 eV is presented in Table X. Except the measurement of Hess [HES49], the experimental results listed in Tab.X are consistent. Assuming an error of 5% on the measurements of Lucas and Asghar, and eliminating the Hess's value yields to an error weighted average cross section of (331 ± 4)b.

Table X. Measurements of Nd-143 thermal capture cross-section

Experimental Measurements	Thermal capture cross section (barns)
D.C.Hess et al. (1949) [HES49]	240 ± 50 b
Pomerance et al. (1952) [POM52]	304 ± 25 b
W.H. Walker (1953) [WAL53]	334 ± 12 b
H.J Hay et al. (1958) [HAY58]	343 ± 20 b
Pattenden et al. (1958) [PAT58]	340 ± 20 b
Tattersall (1959) [TAT59]	336 ±- 10 b
Cabell (1968) [CAB68]	318 ± 14 b
Vertebnyj (1973)[VER73]	338 ± 8 b
Lucas (1977) [LUC77]	335 b
Asghar (1978) [ASG78]	325 b

Therefore, an increase of the thermal capture cross-section given by the evaluations is consistent with the measurements. Taking account of these measurements and the information given by the Minerve experiment, we recommend for the evaluated file a thermal capture cross-section at 0.0253 eV equal to: $\sigma\gamma$ = 338 barns. This recommended value is in perfect agreement with the Vertebnyj's result [VER73].

Eu-154 and Eu-155

In JEF2.2, the cross-section in the resolved resonance range for Eu-155 and Eu-154 are not based on differential measurements (the evaluation was performed before the first transmission measurement). The resonance parameters were assigned randomly to respect average properties of these nucleuses (average resonance spacing, average radiative and neutron width and their theoretical distribution). These theoretical methods are not accurate enough to assess cross-section shape and can produce important discrepancies in burnup credit calculations. As shown in Fig. 4, the JEF2.2 Eu-154 cross-section shape is very different from the other evaluations especially in the resolved resonance range. The first resonance for Eu-154 and Eu-155 which must be accurately described for Burnup credit calculation is even not present in the JEF2.2 evaluation.

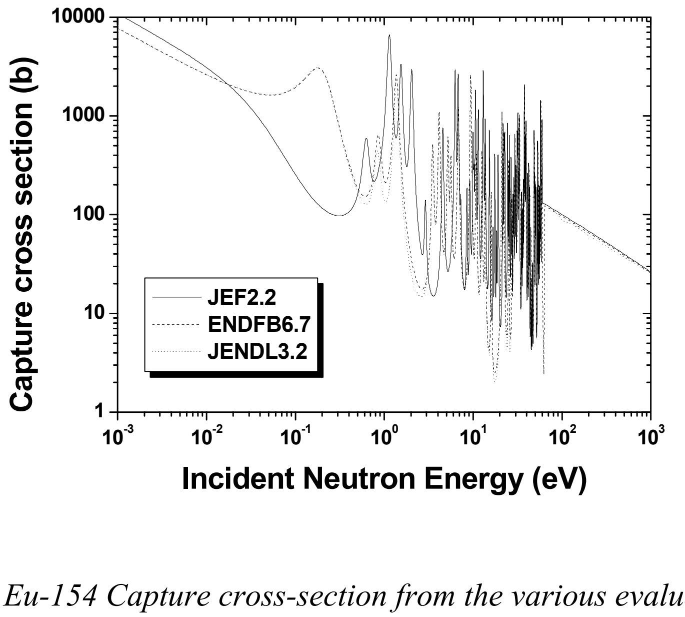

FIG. 4. Eu-154 Capture cross-section from the various evaluations.

The evaluations ENDFB6 and JENDL3.2 are very close in the resolved resonance range (resonance parameters, thermal capture value and resonance integral). They are mainly based on the Anufriev et al. [ANU79] transmission measurement. We have re-analysed the UOX and MOX P.I.E. experiments with the ENDFB6.7 cross-section for Eu-155 and Eu-154.

Table XI. C/E-1 Discrepancies in % from French P.I.E for EU and GD isotopes with JEF2.2 except Eu-154 and Eu-155 that come from ENDFB6.7 library

Burnup GWd/t	30 MOX	40 UOX	40 MOX	50 UOX	60 UOX
154Eu/Cs137		-2.5%		+1%	+4%
154Eu/U238	-8%	-7%	-8%	-4%	+1%
154Gd/U238	-6.5%		-7%	-4%	
155Eu/U238	+17%	+13.5%	+13%	+13.5%	+15.5%
155Gd/U238	+11.3%	+4%	+2%	+6%	+9%
156Gd/U238				-4%	+2.5%

The results presented in Table XI, compared to previous C/E comparison in Table II, allow the following conclusions:

- Eu-154 and Gd-154, Gd-155 and Gd-156 fuel burn up calculation is well improved;
- Eu-155 build-up is still slightly overestimated, but the C/E agreement is no longer the cancellation of two large biases on its formation (Eu-154 capture) and its absorption.

These results confirm that the resolved range evaluation for Eu-154 and Eu-155 based on the available differential measurements, mainly from Anufriev et al. [ANU79] are consistent with integral experiments.

CONCLUSION

In this work, the information given by reactivity worth of separated-FP samples in Minerve and the post-irradiation experiment was used to improve the evaluation of the important fission products in the thermal and resolved range. Our proposed improvements were accepted by the JEFF Group and are currently implemented in the new JEFF3.0 file:

- The Cs-133 capture cross section was decreased according to the results obtained from the Minerve measurements. This trend was satisfied in adopting the ENDFB6.7 evaluation;
- The Sm-149 thermal capture cross-section should be increased, and the value at v = 2200m/s should be set at $\sigma\gamma$ = 41085 b in the new evaluation. To achieve this goal, the Γn value of the first large resonance was increased;
- An increase by 4% of the Nd-143 thermal capture value was needed to correct the discrepancy observed in the Minerve experiment. This was done easily in the evaluation by adjusting the Γn of the bound level in order to get our recommended thermal capture value σ2200 = 338 b;
- Eu-154 and Eu-155 cross-section in JEF2.2 are not realistic in the thermal and resolved range because they are only based on statistical properties of the nucleus. The ENDFB6.7 evaluation was adopted in JEFF3.0, which improves greatly the prediction of Eu and Gd build-up in PWR.

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Measurements of reactivity effects and isotopic composition of highly burnt fuel in LWR-PROTEUS Phase II

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Abstract. The Swiss Nuclear Utilities and the Paul Scherrer Institute (PSI) are conducting the LWR-PROTEUS experimental program at PSI's PROTEUS zero-power facility. The current Phase II is focused on the investigation of the reactivity loss with burnup in highly burnt PWR fuel for code validation in the context of both in-core fuel management and burnup credit applications. Burnt fuel rod samples (~40 cm in length), constituted from fuel irradiated in a Swiss PWR with local burnups of up to 82 GWd/t for UO_2 and 53 GWd/t for MOX, as well as UO_2 samples doped with the 5 most important fission product nuclides (^{149}Sm, ^{155}Gd, ^{103}Rh, ^{143}Nd, ^{133}Cs), are being investigated. The reactivity worths of these samples are measured by repeated insertion and withdrawal in test zones featuring various neutron spectra and compared to those of reference UO_2 samples and calibration samples doped with boron. The burnt samples will also be assayed chemically in the PSI Hotlab. The first criticality of the Phase II configuration was achieved in 2001 and commissioning experiments have been performed using inactive UO_2 samples. The measurements of the burnt samples will be carried out during the year 2002.

1. Introduction

Aiming at improved fuel cycle economy and enhanced operational safety, the Swiss nuclear utilities (viz. Kernkraftwerk Gösgen-Däniken AG, Kernkraftwerk Leibstadt AG, BKW FMB Energie AG and Nordostschweizerische Kraftwerke AG) and the Paul Scherrer Institut (PSI) have been conducting a comprehensive experimental project for the reactor physics investigation of modern light water reactor (LWR) fuel at PSI's PROTEUS facility [1]. In the first phase of the so-called LWR-PROTEUS project, pin-by-pin reaction rate distributions and reactivity effects of the removal of individual pins were investigated in highly heterogeneous BWR fuel assemblies of Westinghouse Atom's SVEA-96 type. The measurements for this first phase were finalized at the end of the year 2000, and the computational analysis is approaching completion. The current Phase II is focused on the investigation of the reactivity loss with burnup in highly burnt PWR fuel with the objective of code validation in the context of both in-core fuel management and burnup credit applications. The measurements consist of the determination of reactivity worths of burnt fuel rod samples from a Swiss PWR as well as UO_2 samples doped with individual fission product nuclides, and chemical assays of the burnt samples. The future phases III and IV will be concerned with advanced UO_2 LWR fuels and fuel designs for actinide reduction. Specific topics currently being considered for Phase III include measurements of temperature coefficients and of in-core instrumentation responses.

2. The LWR-PROTEUS Phase II concept

The goal of the LWR-PROTEUS Phase II experiments is to extend the validation database for both inventory predictions and reactivity calculations to high-burnup fuels of relevance to the Swiss nuclear power plants. Of particular interest are the change in the reactivity of the fuel as a function of burnup and the capability of modern codes to predict this change. Whereas the main interest of the industrial partners (utilities and fuel vendors) lies in the validation of in-core fuel management codes for high-burnup fuel, the same experiments are also applicable to burnup credit for highly burnt PWR fuel in storage and transport configurations. The reactivity worths of burnt fuel rod samples (~40 cm in length), cut from fuel irradiated in the Gösgen PWR with local burnups of up to 82 GWd/t for UO_2 and 53 GWd/t for MOX, are measured by repeated insertion and withdrawal in test zones constituted of PWR fuel rods and featuring various characteristic neutron spectra. Pure fresh UO_2 samples and samples doped with a neutron absorber whose cross-sections are well known (boron) are used as reference and calibration samples, respectively. In order to get a deeper understanding of the source of possible discrepancies between measured and calculated reactivity worths, measurements are also performed using UO_2 samples doped with the 5 most important individual fission product nuclides. The burnt samples are also assayed chemically for 17 actinides and 29 fission products in the PSI hotcells. The combination of reactor physics measurements with chemical analyses of well-characterized power reactor fuel allows the user of the data to validate his entire route (depletion and reactor calculation) whilst at the same time being able to trace discrepancies back to errors, either in the prediction of individual isotopes or in the cross section data used for those isotopes. The experiments are carried out in co-operation with the Gösgen nuclear power plant (KKG) and its fuel vendor Framatome ANP (Germany).

3. Reactor configuration for the experiments

3.1. The PROTEUS facility

PROTEUS is a multi-zone, zero-power critical facility in which a central test zone, which is subcritical by itself, is driven critical by the outer regions of the reactor. The arrangement of the different zones of PROTEUS is shown schematically in Figure 1. For the LWR-PROTEUS programme the test zone consists of actual, full-length power reactor fuel lent to PSI by the utility partners. This fuel is contained in a square (45×45 cm) aluminium test tank. The test zone is surrounded by (in the sequence from the centre to the outside) an unmoderated uranium metal buffer, drivers moderated by D2O and graphite, respectively, and a graphite reflector (for a detailed description of the outer radial regions, see [2]). Since the LWR fuel elements are some 4 m in length and the active height of the PROTEUS driver regions is about 1 m, the test tank can be moved axially, which allows investigations in the different zones of axially heterogeneous assemblies and even axial scans across the zone boundaries to be performed. The reactor instrumentation channels, as well as the control and shutdown rods, are located in the outer regions so that the experiments in the test zone can be performed under "clean" conditions. The unique design of the facility allows a wide range of test zones, both with regard to the nature and moderation condition of the fuel and also to the k□ values, to be accommodated within the almost unchanged driver zones, whilst providing a representative self-generated neutron spectrum of the lattice to be studied in the centre. Many different reactor concepts have been investigated in the course of the years at PROTEUS.

3.2. Test zone for Phase II

The test zone for Phase II consists of a cluster of 11×11 full-length 4.3% enriched PWR fuel pins (incorporating a central guide tube for the samples), surrounded by 8 BWR assemblies (see Figure 2). The PWR fuel pins, which are provided by KKG, with support from Framatome ANP, were extracted from an actual KKG fuel assembly and inserted into a specially designed structure consisting of polyethylene spacers and bottom and top plates, as well as four steel rods connecting these auxiliary structures axially.

The new test configuration with the central PWR region and 8 surrounding BWR elements was optimized to create the adequate spectral conditions for the Phase II investigations, while at the same time minimizing the change from the Phase I test zone (3×3 BWR assemblies) as well as the requirement for new fuel (less than one PWR fuel assembly). Another advantage of this arrangement is that it is less reactive than a test zone constituted entirely of PWR fuel pins. It thus avoids the need for inserting neutron absorbers (either soluble poison or absorber rods) which would arise in the latter case due to the operational constraints of the driven PROTEUS reactor. However, following a request of Kernkraftwerk Leibstadt, the utility providing the BWR fuel, the SVEA-96+ assemblies used in Phase I were replaced by new elements of the SVEA-96 Optima2 design.

The central PWR test region is enclosed in a separate square-shaped stainless steel tank. This allows different moderators to be used in the PWR and BWR parts of the test zone, e.g. a mixture of H_2O and D_2O in the PWR region and regular light water in the BWR assemblies. The steel tank also helps to reduce the reactivity of the test zone. This reactivity is further decreased by steel plates inserted in the gaps between the outer BWR elements.

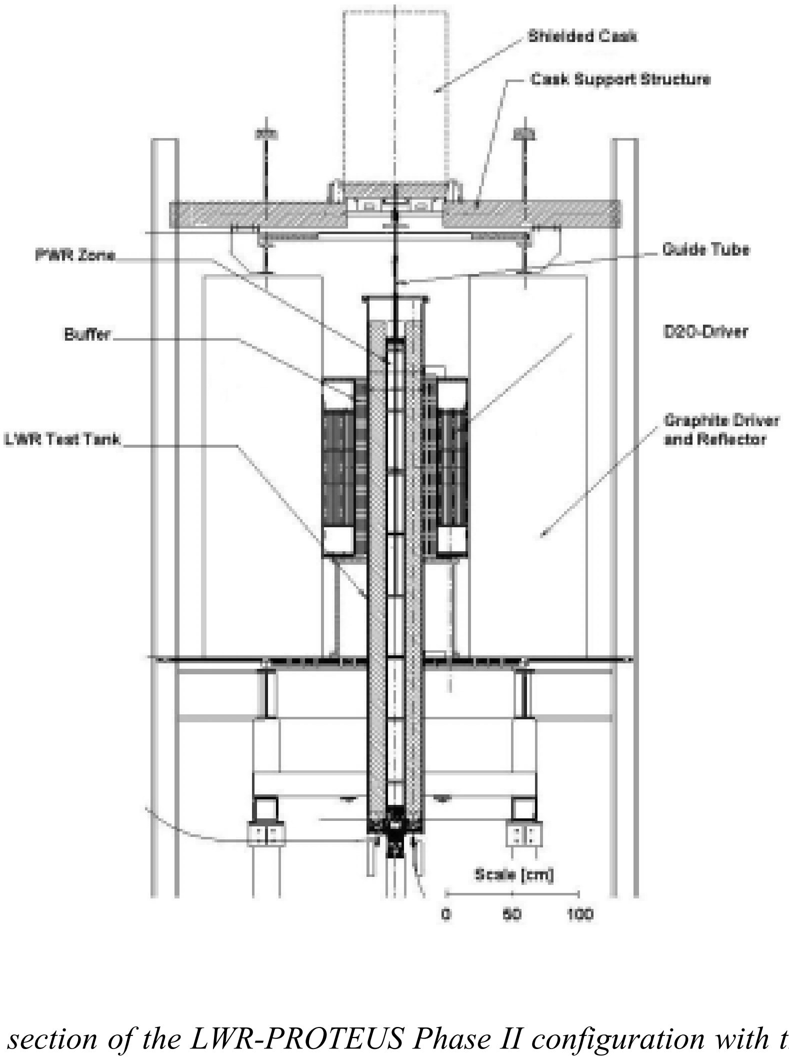
Shielded Cask
Cask Support Structure
PWR Zone
Guide Tube
Buffer
D2O-Driver
LWR Test Tank
Graphite Driver and Reflector
Scale [cm]
0
50
100

FIG. 1. Vertical section of the LWR-PROTEUS Phase II configuration with the shielded cask for the burnt samples.

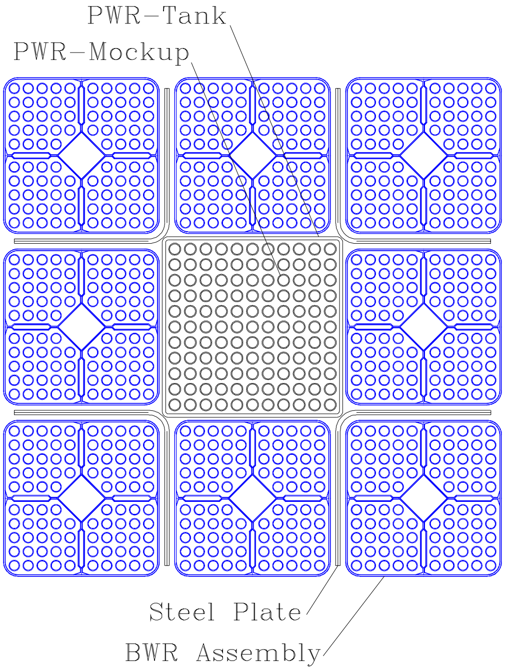

FIG 2. Cross-sectional view of the LWR-PROTEUS Phase II test zone.

Design calculations have shown, as indicated earlier, that an unperturbed neutron spectrum representative of the PWR lattice is achieved in the centre of the test zone in spite of the relatively small size of the PWR region. Spectral indices at the sample location agree with those of a uniform critical PWR configuration within 1%. An inevitable small perturbation of the spectrum is due to the presence of the guide tube surrounding the sample, which locally depresses the thermal flux mostly due to the displacement of water. It has been shown that the reactivity worth ratios of different samples in this test zone are indeed closely representative of the PWR lattice investigated [3]. The interpretation of comparisons between calculated and measured results will be facilitated by inclusion of the guide tube in the computational models.

4. The LWR-PROTEUS Phase II measurements

4.1. Samples

With the aims of Phase II mentioned previously, burnt fuel samples were first identified to constitute a relevant set for systematic analyses. Criteria used for the selection were: (i) to use modern fuel samples (e.g. high ^{235}U enrichment and high discharge burnup values) from the same fuel vendor and extracted from the same nuclear power plant after different cycles, in order to reduce the uncertainties in their characterization (design, construction, irradiation history, etc.); (ii) to cover both UO_2 and MOX fuels; (iii) to have a wide range of burnup values, from low to very high burnups; and (iv) to select the fuel samples from the axial region in the fuel rod where the burnup profile is flat and the burnup level is high, while also avoiding the effect of perturbations due to heterogeneities (e.g. the presence of spacers). Using these criteria, 6 UO_2 and 3 MOX fuel rods were finally identified; their basic characteristics are shown in Table I. From each of these rods, a sample of 40 cm length was cut from an axial section between two spacer locations.

Table I. Selected Burnt Fuel Rods in Phase II

Fuel Type	Initial Enrichment	Discharge Burnup	Discharged
UO_2	4.1% ^{235}U	36 GWd/ton	June 1997
UO_2	3.5% ^{235}U	46 GWd/ton	June 1995
UO_2	3.5% ^{235}U	64 GWd/ton	June 1995
UO_2	3.5% ^{235}U	65 GWd/ton	June 1995
UO_2	3.5% ^{235}U	82 GWd/ton	June 1995
UO_2	3.5% ^{235}U	80 GWd/ton	June 1995
MOX	5.5% Pu_{fiss}	21 GWd/ton	June 1998
MOX	5.5% Pu_{fiss}	37 GWd/ton	July 1999
MOX	5.5% Pu_{fiss}	53 GWd/ton	July 2000

Each of the burnt fuel samples has been overcanned with a special zircaloy cladding and welded tight using a certified procedure to guarantee leak-tightness and absence of contamination. The over-canning is a safety measure usually employed for the return of tested, highly active fuel rods from PSI back to the nuclear power plants. It follows general "defense-in-depth" criteria to prevent contamination in case of an accidental leakage of activity from the primary cladding.

Special fresh UO_2 pellets doped with various additives have been fabricated by Westinghouse Atom, the different pellet types being shown in Table II. The undoped pellets serve as a reference for the experimental determination of the reactivity effect of the replacement of a fresh fuel rod by a burnt one. The enrichment of 4.3% is identical with that of the rods in the PWR test region, whereas 3.5% corresponds to the initial composition of the majority of the burnt UO_2 samples. The fuel pellets doped with ^{10}B are used for the calibration samples against which the relative reactivity worth of the burnt samples is measured. The reactivity worth of the 5 most important fission products (in terms of contribution to the reactivity loss) are investigated directly using pellets doped with the corresponding nuclides.

4.2. Post-irradiation examinations

The first set of experimental results has been obtained in relation to the reception of the full-length burnt fuel rods from KKG and the performance of post-irradiation examinations (PIE) at the PSI Hotlabor. These examinations include: (i) visual inspection, cladding inspection and determination of off-nominal conditions; (ii) measurement of fuel rod elongation and fuel rod diameter as a function of the rod axial length; (iii) axial gamma scans with a resolution in the order of millimetres for the determination of burnup and fission product axial profiles.

The fuel rods were then punctured and cut into a few fuel segments in order to perform extensive analyses including fission-gas release determinations and metallographic studies over the segment cross-sections to study outer and inner oxide thickness growth, pellet grain size and structure, as well as the uniformity of the cladding.

Table II. Special fresh UO_2 samples

Enrichment	Additive	
4.3%	---	
3.5%	---	
2.1%	---	
0.71%	---	
3.5%	^{10}B	2 concentrations
3.5%	^{149}Sm	3 concentrations
3.5%	^{155}Gd	2 concentrations
3.5%	^{103}Rh	2 concentrations
3.5%	^{143}Nd	
3.5%	^{133}Cs	3 concentrations
3.5%	5 fission products	
3.5%	Gd enriched in ^{155}Gd and ^{157}Gd	

4.3. Reactivity measurements

The introduction of the 40 cm long samples, one at a time, into the central PWR test region in PROTEUS allows the experimental determination of the reactivity loss due to burnup. Due to the complex three-dimensional geometry of the driven PROTEUS system, it would be very challenging to calculate the absolute reactivity worth. For the validation of lattice codes however, it is more appropriate to interpret the response of the samples of interest relative to those of well-characterized reference and calibration samples. The design calculations have shown that, particularly for small signals (a few cents), these ratios in the driven configuration are the same as in the corresponding (hypothetical) single-zone system.

The burnt samples are transferred to PROTEUS in a special transport cask. This cask consists of a steel container for shielding, similar to those routinely used for the transport of active samples within PSI, and includes a carousel system and drive mechanisms for the remote-controlled insertion of the selected samples into the reactor. The design of the cask allows the simultaneous transfer of up to 4 burnt fuel samples plus two unirradiated reference samples.

The transport cask is lifted on top of the PROTEUS reactor and located on a specially prepared support structure in the centre of the upper plate (see Figure 1). After the mechanical and electrical connections are made, the PROTEUS shielding doors are closed and the transport cask remains in place above the reactor throughout the reactivity measurements. The samples are then moved, one at a time, into and out of the guide tube in the centre of the reactor.

The measurements are being performed for two different moderation conditions, and hence different neutron spectra, in the PWR test region, viz. full-density light water and a mixture of

H_2O and D_2O in the proportion of about 2/3 to 1/3. The latter mixture simulates the water density in an operating PWR (at 300 ºC) which is about 0.7 g/cm^3

The reactivity effects of the samples are determined by compensation with a calibrated fine control rod, which is moved automatically so as to maintain the reactor exactly critical. In addition, reactivities are also obtained from the observed evolution of the neutron flux (with the automatic control rod in a fixed position) by solving the kinetics equations. The expected reactivity worths are of the order of a few cents. The measurement of the burnt samples along with reference and calibration samples during continuous reactor operation helps to reduce systematic errors considerably; the repeated insertion and withdrawal of the sample (oscillation) allowing to eliminate the effects of possible drifts.

Photoneutrons are emitted in the regions containing heavy water, which influence the time-dependent behaviour of the reactor in an analogous way to the delayed neutrons, but with longer time constants. This effect is taken into account in the analysis of the experiments. Since only parts of the fissions occur in the D_2O-moderated region, a "photoneutron efficiency factor" is determined experimentally for each configuration. Burnt fuel samples, particularly high burnup and MOX fuel, emit neutrons by spontaneous fission and (α,n) reactions. The effect of this external neutron source is discriminated from the reactivity variation by performing measurements at different flux levels.

Experience from previous PROTEUS experiments has shown that reactivity effects of 10 cents and more can be measured to an accuracy of better than 0.1%, whereas effects of about 1 cent can be measured with an accuracy of about 1% using the techniques described above. Calculations made at PSI to estimate the reactivity effect of replacing a central fresh UO_2 rod with different burnt fuel samples showed that rods differing by 2% in k_∞ would yield signals differing by about 0.08 cent per rod. Since signals of one cent can be measured with an accuracy of around 1%, i.e. 0.01 cent, this indicates that the methods proposed will be able to provide a reliable validation base, even for small effects.

4.4. Chemical assays

In addition to the 40 cm long rodlet for the reactivity measurements, a second (contiguous) sample of ~1 cm length (i.e. the size of a pellet) was extracted from each of the burnt fuel rods (see Table I). This is used to conduct destructive chemical assays at the PSI Hotlabor for the accurate quantification of individual isotopes.

The isotopic analysis of the burnt fuel samples is based on chemical assays in the PSI hotcells. The choice of isotopes to be analysed corresponds to the ARIANE "basic programme" [4] and includes the most important actinides,

^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{237}Np, ^{241}Am, ^{242m}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm,

and fission products,

^{142}Nd, ^{143}Nd, ^{144}Nd, ^{145}Nd, ^{146}Nd, ^{148}Nd, ^{150}Nd, ^{133}Cs, ^{135}Cs, ^{137}Cs, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{151}Sm, ^{152}Sm, ^{153}Eu, ^{154}Eu, ^{155}Eu, ^{95}Mo, ^{99}Tc, ^{109}Ag, ^{155}Gd, ^{103}Rh, ^{101}Ru, ^{144}Ce, ^{106}Ru, ^{125}Sb, ^{129}I, ^{147}Pm.

In addition to the fission products important from the neutronics point of view, i.e. for their contribution to the reactivity loss, this list also includes nuclides used as burnup indicators such as ^{137}Cs and the Nd isotopes.

The chemical assays are performed using high performance liquid chromatography/ion chromatography (HPLC-IC) and inductively coupled plasma mass spectrometry (ICP-MS). These methods were applied successfully by PSI in the ARIANE project. The LWR-PROTEUS Phase II programme will thus benefit from available expertise and well-established experimental techniques for obtaining isotopic compositions of burnt nuclear fuel.

5. Calculational analysis

5.1. Nuclide inventory

The compositions of the burnt samples are determined by depletion calculations for the assemblies in which the rods under study were located, following as closely as possible the operational history of the reactor. The KKG reactor was modeled previously using the Studsvik-Scandpower codes CASMO-4 and SIMULATE-3 [5]. The core-follow calculations have been extended to cover the complete history of the plant (22 cycles). This provides the basis for the deduction of parameters (power density, fuel and moderator temperatures, and boron concentration) in some 20 time steps per cycle for the depletion calculations of the assemblies and axial sections from which the samples were cut. Detailed burnup calculations for the samples have been performed using CASMO-4. It is planned to carry out the same kind of calculations also using other codes available at PSI, such as HELIOS, BOXER (PSI-developed assembly code) and MONTEBURNS (combination of MCNP and ORIGEN).

Independent calculations, both for the operational cycles of the KKG reactor and for the detailed history of the measured samples, are being performed by the industry partners using the computational methods of Framatome ANP (CASMO-3, PRISM, PINPOWL). In addition to enabling comparisons to the chemical assays, the analysis thus provides an intercomparison of independent calculational results which, in turn, gives an indication of the “spreads” in current-day predictions of the composition of the investigated samples.

5.2. Sample reactivity worth

The goal of LWR-PROTEUS Phase II is to validate assembly codes for the calculation of the reactivity loss in high-burnup fuel. Typical examples of codes to be used in this context are CASMO-3 for KKG and Framatome ANP, and CASMO-4, HELIOS and BOXER for PSI. However, with many of these codes, it is not possible to model the complex PROTEUS configuration, without employing significant approximations of the actual geometry. The capabilities of such codes are usually limited to the modeling of parts of the PROTEUS configuration, such as a reflected assembly (in the present case, the bundle of 11×11 PWR pins) or the entire test zone (including the 8 BWR elements). Thus, the major contribution of LWR-PROTEUS Phase II to the validation of such codes is expected to be via the comparison of measured relative reactivity worth (with a well-known calibration sample as a reference) to the corresponding values obtained for reduced configurations by the design codes. As mentioned earlier, calculations have shown that these relative worth for the reduced configurations agree well with those estimated for PROTEUS. The best available whole-reactor models will be used to derive corrections for the small deviations of these ratios in the reduced configurations from those in PROTEUS, employing direct calculations and/or a perturbation-theory based methodology. The nuclide densities for the burnt samples in these calculations will be taken, whenever possible, from the assembly code to be tested, so that an integral and consistent validation of both inventory prediction and reactivity calculations by the same code is provided.

6. Status and schedule

The PROTEUS facility has been modified for the Phase II experiments. The first criticality of this configuration, but still with the earlier SVEA-96+ assemblies, was reached in July 2001, followed by a commissioning period for characterization of the new reactor core and for the performance of preliminary measurements aimed at optimizing experimental techniques. The measurements during this commissioning period included:

(i) axial traverses using miniature fission chambers,
(ii) measurements of reactivity worth of geometrically-identical UO_2 samples of different ^{235}U-enrichments for the preliminary validation of numerical calculations,
(iii) measurements of reactivity worth of UO_2 samples with the same ^{235}U-enrichment but with different lengths (i.e. shorter and longer than the 40 cm nominal axial dimension), and
(iv) measurements with a neutron-emitting sample (^{252}Cf source) in order to optimize the experimental technique necessary to discriminate external neutron sources from spontaneous-fission and (α,n) reactions.

The results achieved during the commissioning period, necessary for the preparation of detailed measurement plans, have been very promising in the sense of confirming the accuracy of the planned experiments and the predictability of various effects.

After a certain refurbishment of the facility and the replacement of the SVEA-96+ by the new SVEA-96 Optima2 elements, reactor operation was resumed in April 2002. All the equipment needed for the measurements is ready and has been successfully tested. The burnt samples were prepared and overcanned in winter 2001/2002, and the special UO_2 pellets (reference, calibration and doped samples) were delivered in April 2002. The operation of Phase II, including the insertion of the highly radioactive burnt fuel samples has been authorized by the Swiss regulatory body (HSK).

Measurements employing the entire range of samples are scheduled to be carried out in the period from May to the end of the year 2002. The chemical assays will also be performed during this time.

7. Summary and Concluding Remarks

In the framework of the LWR-PROTEUS Phase II programme, the reactivity worth of UO_2 fuel rod segments with a PWR irradiation history of up to 7 cycles, and of MOX samples with a history of up to 3 cycles, are being measured in various neutron spectra. The chemical assays of the burnt fuel samples provide the isotopic concentrations of 9 major actinide nuclides, 8 minor actinide nuclides and 29 fission products. The strength of the programme lies in the fact that highly burnt fuel discharged from a nuclear power plant is measured for reactivity effects in PROTEUS and analyzed physically and chemically, in parallel, as part of the same programme. Measurements carried out during the approach-to-critical and commissioning period of the reference Phase II configuration in PROTEUS have provided confirmatory evidence for the design of the experiments and of the high accuracies to be expected in the programme.

The current extension of the available validation base for reactor physics codes to very high burnup is essential for regulatory acceptance of future burnup increases. Furthermore, the PROTEUS experiments could provide valuable evidence to justify a significant reduction of the degree of conservatism in present-day PWR core designs. Finally, the experiments will also be important for the validation of codes used to predict the reactivity of spent fuel in storage and transport configurations, i.e. will provide a new extended basis for burnup credit evaluations.

ACKNOWLEDGEMENTS

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Experimental measurement of the isotopic composition of high enrichment and high burnup fuel

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Abstract. Within the frame of the fuel safety and reliability research activities in Spain, a project has been initiated in 2002 to measure the isotopic composition of PWR fuel with a high initial enrichment at high local burnup levels. The fuel to be measured was included in an irradiation demonstration program carried out at the Vandellós 2 NPP, and was fully characterised before irradiation. Poolside inspections were performed at each and every reload. After irradiation, the fuel has been included in several research programs:

- Ramp testing and PIE examinations
- Reactivity insertion accidents research in the CABRI program
- Measurement of the mechanical properties of the cladding in the PROMETRA program
- High burnup fuel safety research program of JAERI (ALPS program)

The isotopic composition measurement program will complete and enhance the outcome of these programs, and will serve as a base for the validation of burnup credit analysis methods to be applied to transport casks in the future. The program will include the measurement of six samples obtained form two rods, with burnup values ranging from 40 to 70 MWd/kgU. The isotopes to be measured have been decided on the basis of their importance for criticality, shielding and residual heat calculations. The preliminary results should be obtained before the end of 2002.

1. BACKGROUND

The participation of Spain in high burnup fuel research activities has been increasing steadily during the last decade. The national activities carried out by the fuel supplier (ENUSA), basically consisting of fuel demonstration and irradiation programs up to high burnup values, are complemented by the participation of different Spanish organizations in a number of international research efforts. Some of those are the CABRI Water Loop Program, the NRC high burnup fuel program, the Halden Reactor Project, the Robust Fuel Program managed by EPRI and the ALPS program of JAERI.

One of the high burnup fuel research fields in which different Spanish organizations are interested is that of the isotopic fuel composition at high discharge burnup values. Isotopic composition data is needed to validate fuel burnup codes, this being an issue specially important for the implementation of burnup credit in criticality safety analysis of both storage and transport systems. However, none of the programs mentioned above are providing isotopic composition data representative of current fuel designs, and with an initial enrichment and irradiation history similar to those now used.

The lack of data in this field has prompted the startup of a co-operative effort to independently measure the isotopic composition of PWR fuel with a high initial enrichment at high discharge burnup values. The program will use fuel irradiated in the so-called irradiation extension program (described in the next paragraph), and the participant organizations are the Consejo de Seguridad Nuclear, ENRESA (responsible in Spain for waste management), ENUSA (Spanish fuel vendor), ENDESA (Spanish utility), Kansai and Mitsubishi Heavy Industries.

2. THE IRRADIATION EXTENSION PROGRAM

The irradiation extension program was performed at the Vandellós 2 reactor, and was sponsored by ENUSA, ENDESA and a group of Japanese organisations. Within this program, 12 fuel rods that had been irradiated during four cycles were loaded into a once-burned fuel assembly to undertake a fifth irradiation cycle. The rod design is the standard 17x17 PWR fuel assembly, with an initial enrichment of 4.5% w/o and Zirlo cladding. The discharge burnup attained is in the range of 65 to 70 Gwd/MtU, and the irradiation history corresponds to high duty fuel.

The rods were transported to the Studsvik (Sweden) hot cell laboratories at the end of the program for ramp testing and PIE examination. One of the advantages of these rods is that they were originally included in a prior demonstration program ran at the same reactor. Hence, they were fully characterised during manufacturing and fabrication, and pool-side inspected during irradiation in all the plant reloads. Together with the post irradiation data later obtained at Studsvik, the availability of this data makes the fuel perfectly suitable for safety research. As a matter of fact, some of these rods have already been included as testing material in the CABRI and ALPS research programs. The outcome of these programs will again add to the knowledge of the behaviour and properties of the rods.

3. PROJECT DESCRIPTION

The isotopic composition measurement project will use fuel samples taken from two of the rods of the irradiation extension program. The main objective of the program is to independently determine the isotopic composition of irradiated PWR fuel at the Studsvik hot cells. Besides the experimental work, analytical tasks to be performed by the participating organisations are included in the project, such as the evaluation of the results obtained and code validation.

3.1. Project scope

The isotopic composition of six fuel samples will be determined, with local burnup values ranging from 55 to 75 Gwd/MtU. Two of the samples will have very similar burnup values, in order to provide for a check of the consistency of the measurements.

The list of nuclides to be measured is included in Table I. The nuclides have been selected on the basis of their importance for criticality safety, shielding and residual heat calculations. Though in some cases, relevant for reactivity calculations, some metallic isotopes have been excluded from the list due to the difficulty in completely diluting the nuclide that would make the results unreliable.

The experimental methods to be used to determine each isotope’s concentration are also indicated in Table I, basically chemical analysis and gamma scanning of the rod. The chemical analyses are performed by isotopic dilution using an inductively coupled plasma mass spectrometer. Ion chromatography must also be performed in order to resolve the isobaric overlaps.

3.2. Analytical works

ENUSA will be responsible for the analytical activities included in the project, namely:

- Evaluation of the experimental results by comparison with depletion calculations using the SCALE system. The expected isotopic composition of each sample will be obtained, and a comparison with the experimental results will be made based on the concentration of the major Actinides.
- The final experimental results for the six samples measured will again be evaluated using the SCALE system.

In addition, other participant organizations will perform their own calculations using tools other than SCALE.

3.3. Schedule

The project has been started at the end of February 2002. The gamma scan of the rods has already been performed, and the position of the samples to be taken is currently being decided. It is expected to have a draft report of the measurement results in November 2002, the final report being expected for January 2003. The analytical work should be finished before the summer of 2003.

Table I.

Motivation	Isotopes	Measurement method
Reactivity	U 234, 235, 236, 238	Chem (HPLC-ICP-MS)
	Pu 238, 239, 240, 241, 242	Chem (HPLC-ICP-MS)
	Am 241, 243	Chem (HPLC-ICP-MS)
	Np 237	Chem (HPLC-ICP-MS)
	Cs 133	Chem (HPLC-ICP-MS)
	Sm 147, 149, 150, 151, 152	Chem (HPLC-ICP-MS)
	Nd 143, 145	Chem (HPLC-ICP-MS)
	Eu 151, 153	Chem (HPLC-ICP-MS)
	Gd 155	Chem (HPLC-ICP-MS)
Residual Heat	Ba-137m (Cs-137)	Gamma-scan
	Pr-144 (Ce-144)	Gamma-scan
	Rh-106	Gamma-scan
	Cs-134	Gamma-scan
	Eu-154	Gamma-scan
	Cm-244	ICP-MS
Shielding	Cm-246	ICP-MS
Gases	Kr-85	FGR gas mass-spectrometry
Volatiles	Cs-134	Gamma-scan
	Cs-135	Chem (HPLC-ICP-MS)
	Cs-137	Gamma-scan
	Ru-103	Gamma-scan
	Ru-106	Gamma-scan
Burnup	Ce-140	Chem (HPLC-ICP-MS)
DETERMINATION	Ce-142	Chem (HPLC-ICP-MS)
	Nd-146	Chem (HPLC-ICP-MS)
	Nd-148	Chem (HPLC-ICP-MS)

The burnup credit experimental programme REBUS

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Abstract. An international programme called REBUS (REactivity tests for a direct evaluation of the Burnup credit on Selected irradiated LWR fuel bundles) for the investigation of the burnup credit has been initiated by the Belgian Nuclear Research Center SCK•CEN and Belgonucléaire with the support of USNRC, EdF from France and VGB, representing German nuclear utilities. NUPEC, representing the Japanese industry, joined the programme recently, making it possible to incorporate the study of MOX fuel. The programme aims to establish a neutronic benchmark for reactor physics codes. This benchmark would qualify the codes to perform calculations of the burnup credit. The benchmark exercise investigates the following fuel types with associated burnup: reference 3.3% enriched UO_2 fuel, fresh commercial PWR UO_2 fuel, irradiated commercial PWR UO_2 fuel (51 GWd/tM), fresh PWR MOX fuel and irradiated PWR MOX fuel (20 GWd/tM). Reactivity effects are measured in the critical facility VENUS. Fission rate and flux distributions in the experimental bundles will be determined. The accumulated burnup of all rods is measured non-destructively in a relative way by gross gamma-scanning, while some rods are examined by gamma-spectrometry for an absolute determination of the burnup. Some rods will be analyzed destructively with respect to accumulated burnup, actinides content and TOP-19 fission products (i.e. those non-gaseous fission products that have most implications on the reactivity). Additionally some irradiated rods have undergone a profilometry and length determination. The experimental implementation of the programme has started in 2000 with major changes in the VENUS critical facility. Gamma scans, profilometry, length determination and gamma-spectrometry measurements on the MOX fuel have been performed. The experiments on the reference configuration have been completed. The irradiation of the fresh PWR MOX configuration has been performed and fission rate distribution measurements are presently executed.

1. INTRODUCTION

Present criticality safety calculations of irradiated fuel often have to model the fuel as fresh fuel, since no precise experimental confirmation exists of the decrease of reactivity due to accumulated burnup. In other occasions only actinide depletion is allowed to be taken into account and the influence of fission products has to be disregarded. The fact that this so-called "burnup credit" cannot (completely) be taken into account has serious economical implications for transport, storage and reprocessing of irradiated fuel. For long-term geological storage it is almost imperative to apply burnup credit.

2. GENERAL DESCRIPTION OF THE REBUS PROGRAMME

The aim of the REBUS programme is to establish an experimental benchmark for validation of reactorphysics codes for the calculation of the loss of reactivity due to burnup, both for UO_2 and MOX fuel bundles.

The reactivity effect of PWR UO_2 spent fuel will be measured in the VENUS critical facility on a bundle of commercial spent fuel with a burnup of 51 GWd/tM. The same will be done on a MOX bundle with a burnup of 20 GWd/tM. Other fuel types can be investigated in future extensions of the programme, like BWR fuel. At the same time we will measure fission rate and flux distributions in the different configurations.

Together with a precise measurement of the reactivity effect, it is also indispensable to have a good characterization of the fuel. The characterization of the spent fuel is performed in two steps.

The first step is non-destructive and is performed before the reactivity measurements. All spent fuel rods will be measured by gross gamma-scanning in order to determine the distribution of total gamma activity in the fuel rods. One specific rod will be investigated by gamma-spectrometry, together with a well-qualified calibration source, to determine the ^{137}Cs content and in this way the burnup of the rod. Via the gross gamma-scans this will give a good picture of the burnup of all rods. This first step is necessary to verify that the selected rods have a similar burnup.

The second step is destructive and is consequently performed after the reactivity measurements. It aims at determining both the actinides content, some burnup indicators (Cs, Nd) and the 19 most important fission products with respect to neutron absorption (representing >80% of the neutron absorption in the spent fuel). The sample for this destructive, radiochemical assay is taken from the same rod, on which gamma-spectrometry has been performed.

The fresh fuel content has been well documented during fabrication.

3. SHORT DESCRIPTION OF THE VENUS REACTOR

The VENUS critical facility is a water-moderated zero-power reactor. It consists of an open (non- pressurized) stainless-steel cylindrical vessel including a set of grids which maintain fuel rods in a vertical position. After a fuel configuration has been loaded, criticality is reached by raising the water level in the vessel.

Parameters that are measured with the VENUS reactor are the critical water level h_c, the reactivity coefficient $\delta\rho/\delta h$, the axial fission rate distribution, the horizontal fission rate distribution, spectrum indices F5/F9, F8/F9, C8/F9, fission rate distribution inside fuel rod, detector response and the delayed neutron fraction β_{eff}. For details of the measurements performed in the VENUS we refer to reference [1], except for the flux measurements that are described below.

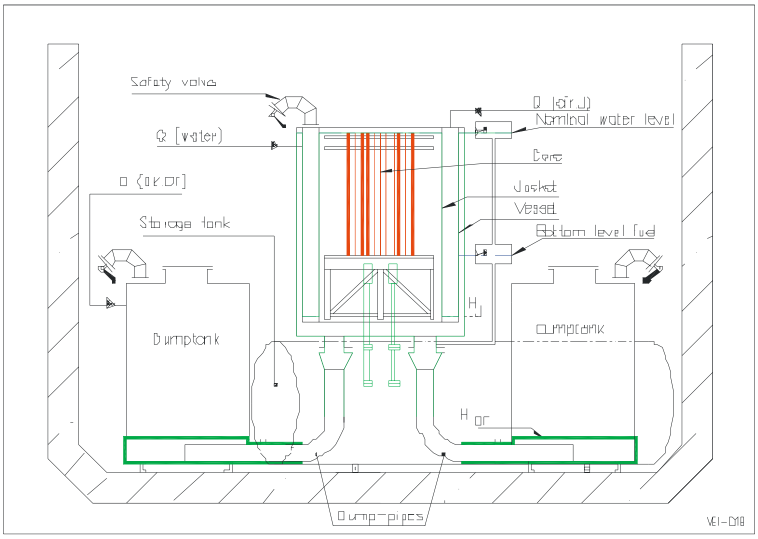


FIG. 1. Vertical cross-section of the VENUS critical facility.

Unlike the fission rate distribution that is measured directly via the fission products created in the fuel rods, the flux distribution is determined by the insertion of activation wires (e.g. Sc) at several positions in the reactor in between the fuel rods and measuring the activity of these wires after irradiation in the VENUS facility. The activation wires typically have a length of 1 cm and are placed between the fuel rods in perspex tubes. The aim of the perspex is to perturb the water environment as little as possible with respect to neutron behaviour.

The flux is derived from the measured activity of the wires. Since this activity is due to both the thermal and epithermal neutron flux, a certain amount of calculational interpretation is required. However, this is limited due to the fact that more than 95% of the activity is caused by thermal neutrons.

The flux distribution measurements will be performed simultaneously with fission rate distribution measurements in order to make a comparison between the results of both methods. In the irradiated fuel bundles only flux distribution measurements will be performed, since fission rate measurements are not possible on irradiated fuel.

4. REACTIVITY MEASUREMENTS PERFORMED AT VENUS IN THE FRAMEWORK OF REBUS

Five fuel bundles or assemblies will be investigated in the framework of the REBUS programme. These five bundles are:

- Reference 3.3% enriched UO_2 fuel
- Fresh commercial PWR UO_2 fuel, constructed by Framatome ANP (formerly SIEMENS)
- Irradiated commercial PWR UO_2 fuel (51 GWd/tM), originating from Neckarwestheim NPP, Germany
- Fresh PWR MOX fuel, originating from the BR3, an experimental Belgian PWR
- Irradiated PWR MOX fuel (20 GWd/tM), also from the BR3 reactor

All test bundles will be loaded as a 7x7 fuel assembly in the center of a 3.3% enriched UO2 fuel driver zone. The 7x7 assembly is chosen because calculations show that such an assembly will result in a reactivity effect that is large enough for benchmark purposes (~1500-2000 pcm) and for practical reasons because the VENUS reactor has removable grids where this assembly fits in.

The reference 3.3% enriched UO_2 fuel bundle (figure 2a) consists of the same 3.3% enriched UO_2 fuel as there is in the driver zone. Its purpose is to validate a k_{eff} calculation.

The fresh commercial fuel bundle (figure 2b) consists of a 5x5 fuel assembly. The fuel is 3.8% enriched UO_2 fuel, fabricated at Framatome ANP (formerly SIEMENS), Germany. It is the original composition of the irradiated commercial fuel.

The irradiated commercial fuel bundle is the same as the fresh bundle, as is obvious for experimental reasons (clean comparison of the fresh and irradiated bundle). The fuel is provided by GKN, the operator of the Neckarwestheim NPP in Germany.

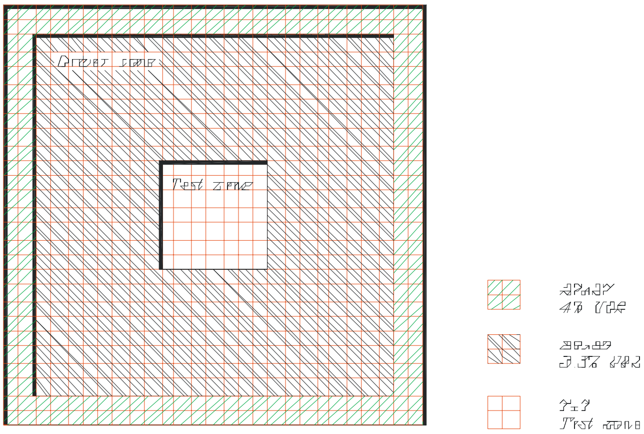

Driver zone
Test zone
7x7
Test zone

FIG. 2a. Reference UO_2 fuel configuration design.

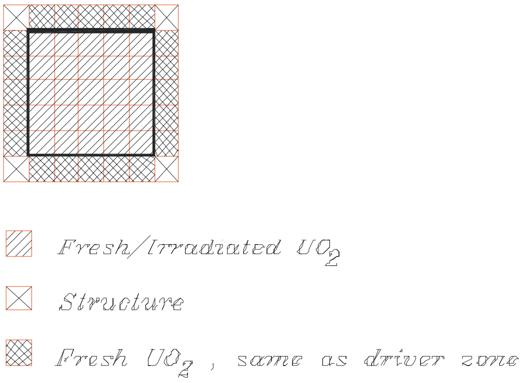

Fresh/Irradiated UO_2
Structure
Fresh UO_2, same as driver zone

FIG. 2b. Central 7x7 UO_2 bundle.

The fresh BR3 MOX fuel bundle (figure 2c) consists of 24 fuel rods (a 7x7 overmoderated assembly). In this way the highest reactivity effect could be obtained with the available rods. The fuel is 6.9% enriched fissile MOX fuel, fabricated at Belgonucléaire, Belgium. It is the original composition of the irradiated BR3 fuel. The irradiated PWR MOX fuel of intermediate burnup originates from the BR3 reactor, Belgium. The fuel is provided by SCK•CEN. The bundle design is the same as for the fresh bundle (figure 2c).

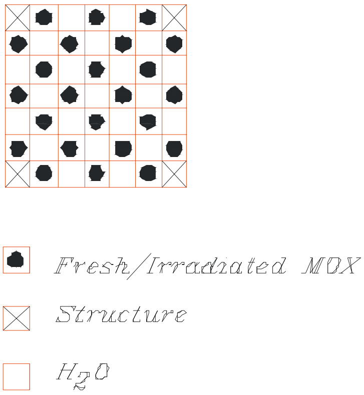

Fresh/Irradiated MOX
Structure
H_2O

FIG. 2c. Central 7x7 MOX bundle.

The reactivity effect will be measured by loading the different bundles in the center of the driver zone and measuring each time the critical water level and the reactivity effect of a change of the water level. From these measurements the reactivity effect can be estimated. However, a more direct way of validating the reactor codes is simply calculating the k_{eff} for the different configurations at the measured critical water levels. In this case there is no need in using the measured reactivity effect, which reduces the measurement uncertainty on the experimental data and increases the performance of the benchmark test.

In addition of the reactivity measurements the fission rate distribution and the flux distribution at the main axes will be measured (see figure 3). Due to the impossibility of measuring this parameter in the spent fuel assembly, wire activation measurements will be performed in this assembly to measure the thermal and epithermal neutron flux.

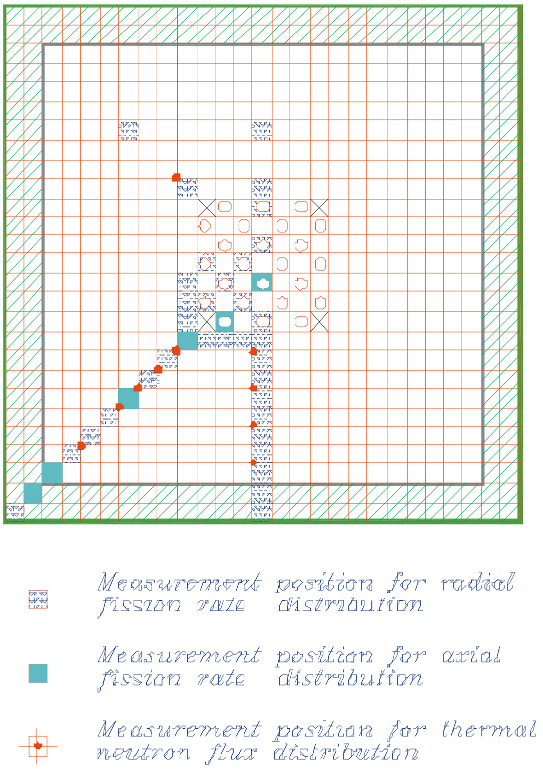

Measurement position for radial fission rate distribution
Measurement position for axial fission rate distribution
Measurement position for thermal neutron flux distribution

FIG. 3. Measurement positions for horizontal fission rate distribution.

Table I. Overview of the measurements in the different configurations.

Table I. Overview of measurements in the VENUS critical facility

Configuration	critical level	reactivity effect	fission rate distribution	flux distribution	validation purpose
reference UO_2	x	x			k_{eff}
fresh UO_2	x	x	x	x	k_{eff} and fission rate
irradiated UO_2	x	x	in driver	x	k_{eff} and fission rate
fresh MOX	x	x	x	x	k_{eff} and fission rate
irradiated MOX	x	x	in driver	x	k_{eff} and fission rate

5. REFABRICATION AND MEASUREMENTS PERFORMED AT HOT CELL LABORATORY, RADIOCHEMICAL MEASUREMENTS

The refabrication of the commercial spent fuel rods from 4 meter rods into 1 meter rodlets will be executed in the SCK•CEN hot cell laboratory. Afterwards the rodlets will be cleaned thoroughly, since the contamination level of the VENUS reactor has to remain very low. Also the BR3 spent fuel rods have to be cleaned, but no refabrication is needed. After cleaning the rodlets will be assembled in the experimental 7x7 bundle. This is performed at the hot cell of the BR2, SCK•CEN's MTR.

Details of the measurement techniques used in the hot cell laboratory and of the radiochemical measurements are given in reference [1].

6. PRELIMINARY CALCULATIONS

Some preliminary calculations have been performed to support the design of the bundles. The main objective of the REBUS programme is validation of reactorphysics codes therefore it is indispensable to create a reactivity effect that is significantly larger than the uncertainty of the calculations. Since the uncertainty of the calculations is estimated to be between 300 and 500 pcm (0.3 to 0.5 % on keff), it is considered necessary to create an experimental reactivity difference of about 1 500 pcm.

Preliminary calculations showed a reactivity difference of 2135 pcm for the high burnup commercial fuel with respect to fresh fuel. For the MOX case the calculated difference is about 1 500 pcm.

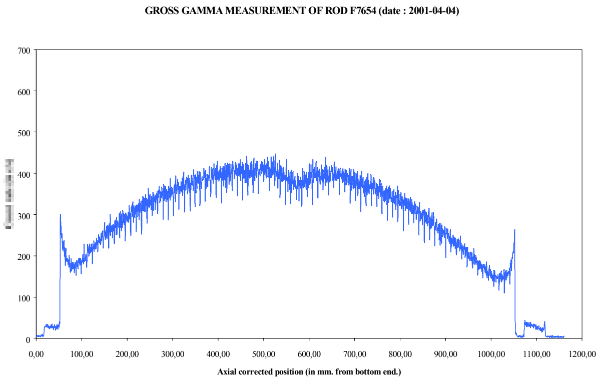
7. RESULTS

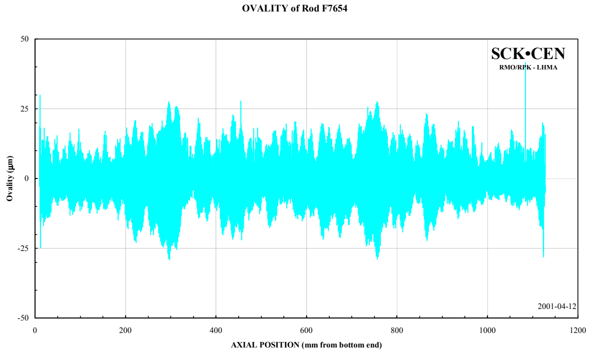
7.1. Characterization of spent fuel rods

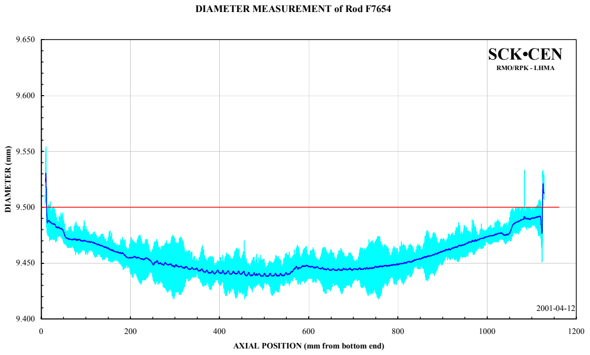
The first results of the REBUS programme deal mainly with the characterization of the fuel. The fresh fuel rods originating from the VENUS and BR3 (driver zone, reference bundle and MOX fuel) have already been used in previous benchmark programmes and their characteristics are well-documented. Additionally, the VENUS rods have been elongated from 50 cm to 1 m and during this elongation we took the opportunity to perform some extra measurements with respect to the cladding inner and outer diameter and the position of the lower end plug.

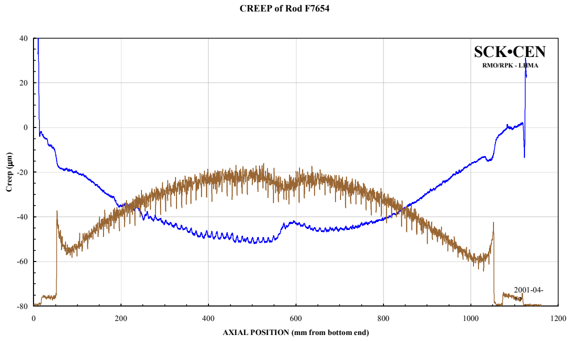
The irradiated MOX fuel from the BR3 has been examined in the hot cell laboratory with respect to gross-gamma scanning, burnup by gamma-spectrometry, profilometry and fuel column length. Typical results are shown in the following figures.

Figures 3a, b, c, d. Measurement of the gross-gamma scan, ovality and diameter of an irradiated MOX PWR rod


GROSS GAMMA MEASUREMENT OF ROD F7654 (date : 2001-04-04)


OVALITY of Rod F7654


DIAMETER MEASUREMENT of Rod F7654


CREEP of Rod F7654

The gross-gamma scan results show for certain MOX rods a rather high degree of Cs migration, indicating a high linear power during irradiation. Since this Cs migration behaviour is not similar for all rods, we have to make some adjustments with respect to the gamma self-absorption for the 660 keV gamma peak of ^{137}Cs in order to compensate the differences.

7.2. Experiments in the critical facility VENUS

The critical height and the reactivity effect of the water level were measured for the reference configuration and the fresh PWR MOX configuration. A comparison with calculated values is given in Table II.

Table II. Comparison of calculated and experimental critical heights, expressed in σK

Configuration	σk_{eff} (calc-exp)
Reference	0.00164
Fresh PWR MOX	- 0.00263

The difference between the measured and calculated values fall well within the uncertainty of the calculations (about 500 pcm or 0.00500).

8. VALUE OF THE REBUS BENCHMARK

The critical water level of configurations in the VENUS reactor is measured with a random uncertainty of 0.02 cm and a systematic uncertainty of 0.05 to 0.07 cm, the latter being mainly due to the uncertainty of the fuel bottom. So the total uncertainty is between 0.05 to 0.07 cm, which results in an uncertainty, expressed in pcm of about 11 pcm (for a core height of 60 cm). The random uncertainty is smaller, about 3 pcm.

For the REBUS configurations the critical level will be in the range of 60 to 80 cm with a corresponding lower reactivity effect of the water level, resulting in an even smaller uncertainty. However, due to the large reactivity effect between the fresh and spent fuel bundle the difference in critical height will be large, resulting in a change of water level reactivity effect.

A simple approximation to determine the total reactivity effect Δρ is given by equation (1):

$$\Delta\rho = \Delta h_c \times \frac{\partial\rho}{\partial h} \tag{1}$$

where Δh_c is the measured difference in critical height for the two compared configurations and ∂ρ/∂h the averaged value of the reactivity effect of the water level.

When due to the large difference in critical height the values of ∂ρ/∂h will be too different for the two configurations, in principle integration should be used for the determination of the total reactivity effect Δρ as shown in equation (2):

$$\Delta\rho = \int_{h_{c1}}^{h_{c2}} \frac{\partial\rho}{\partial h} dh_c \tag{2}$$

The uncertainty related to the measurement of the reactivity effect Δρ in this way is estimated to be 20 pcm, based on the present value for a lower critical value (10 pcm), where the higher critical level will decrease the uncertainty, but where the high value of the reactivity effect will increase the uncertainty due to the change of ∂ρ/∂h in function of the critical height.

The uncertainty of the non-destructive 137Cs burnup measurement is estimated to be 4 to 5%.

The uncertainties of the determination of the actinides, the burnup indicators and the fission products by radiochemical analysis are shown in Tables III, IV and V.

Table III. Uncertainties by Radiochemical Analysis for the different actinides

^{234}U	5-10%	^{238}Pu	3-10%	^{241}Am	2-5%	^{242}Cm	3-7%
^{235}U	0.5-2%	^{239}Pu	0.2-0.5%	^{242}Am	2-10%	^{243}Cm	30-50%
^{236}U	0.5-5%	^{240}Pu	0.2-0.5%	^{243}Am	2-5%	^{244}Cm	3-6%
^{238}U	0.3-0.5%	^{241}Pu	0.2-0.5%			^{245}Cm	3-10%
		^{242}Pu	0.2-0.5%			^{246}Cm	5-15%
^{237}Np	3-10%	^{244}Pu	50%				

Table IV. Uncertainties by Radiochemical Analysis for the different burnup indicators

^{137}Cs	2-4%	^{143}Nd	0.5-1%	^{146}Nd	0.5-1%
^{144}Ce	5-10%	^{144}Nd	0.5-1%	^{148}Nd	0.5-1%
		^{145}Nd	0.5-1%	^{150}Nd	0.5-1%

Table V. Uncertainties by Radiochemical Analysis for the different fission products (TOP-19), representing 80-90% of the anti-reactivity of all long-living fission products

^{147}Sm	0.5-2%	^{153}Eu	0.5-2%	^{103}Rh	10-20%	^{143}Nd	0.5-1%
^{149}Sm	2-4%	^{154}Eu	2-5%	^{109}Ag	15-30%	^{145}Nd	0.5-1%
^{150}Sm	0.5-2%	^{99}Tc	10-20%	^{155}Gd	5-10%	^{133}Cs	2-5%
^{151}Sm	0.5-2%	^{95}Mo	10-20%	^{105}Pd	10-20%	^{135}Cs	2-5%
^{152}Sm	0.5-2%	^{101}Ru	10-20%	^{108}Pd	10-20%		

9. CONCLUSIONS

The REBUS programme will provide an experimental benchmark for burnup credit, taking into account both fissile isotopes depletion and the production of neutron absorbing fission products. However, no distinction between both contribuants is possible within the framework of the programme.

With the results of the REBUS programme a validation of reactor physics codes with respect to burnup credit of PWR UO_2 fuel with intermediate and high burnup will be possible. Based on the results of this validation, and the quoted uncertainties of the experiment, safety margins can be determined that have to be implemented for burnup credit calculations.

The REBUS programme will be executed completely at SCK•CEN in its hot cell laboratory, radiochemistry lab and the VENUS reactor. All these facilities have already proven their capability to execute large-scale international benchmark exercises, like DOMO, HBC, CALLISTO (hot cells, radiochemistry), ARIANE (radiochemistry) and VIP, VIPO, VIPEX (VENUS reactor).

Future extensions of the REBUS programme can deal with BWR and high burnup MOX fuel.

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BUC validation in the UK: Design of experiments and lessons learnt

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Abstract. The history, design, implementation and analysis of experiments related to the validation of BUC in the UK are summarized and reviewed. The experiments include the CERES programme, which was carried out in the UK (at the DIMPLE facility, Winfrith) and in France (at the MINERVE reactor, Cadarache). Measurements of the reactivity of irradiated PWR, BWR and AGR fuel samples were made in a variety of neutron spectra, designed to simulate a range of spent fuel environments. PIE data for actinides and the major fission product absorbers were also obtained.
Lessons learnt during the programme are presented. These include the main conclusions drawn from the code validation analyses and also identify some technical difficulties encountered during the implementation of the programme. Suggestions are made regarding how future programmes might strengthen the validation base.

1. INTRODUCTION

The CERES programme of experiments was carried out from 1991 – 1997 through measurements in the MINERVE reactor at Cadarache in France and the DIMPLE critical facility in the UK. In the latter stages of the programme, the CEA and AEA Technology were joined by BNFL and Sandia National Laboratories. The programme was designed to provide validation data in support of the development of BUC methodologies. Reactivity measurements were performed on a range of well-characterized PWR and BWR samples and included PIE for uranium, plutonium as well as the major fission product absorbers.

This paper describes the design of the experimental programme, mainly for the DIMPLE measurements, and presents a summary of the analysis of the key results. In addition, lessons learnt during implementation of the experiments and from analysis of the measurements are presented. Further details of the experiments can be found in presentations made at the PHYSOR and ICNC international conferences [1,2].

The experimental programme has been successful in establishing an important source of validation data for BUC, but (as always) there were unexpected difficulties encountered during the execution of the experiments, and the analyses of the results have identified areas where further measurements would be useful. The aim of this paper is to summaries these lessons learnt so that other experimental programmes might benefit from the experience gained.

2. HISTORY

At the PHYSOR reactor physics conference in Marseilles in 1991, presentations by the CEA (France) and AEA Technology (UK) highlighted close similarities between the current BUC validation programmes in each country. It was agreed that a collaborative programme, to be known as CERES, would provide an opportunity to carry the experiments further, including measurements on a wider range of spent fuel samples and to provide PIE and reactivity data for individual fission products. Previous programmes at the two facilities had been aimed partly at providing validation for reactor physics codes, but by 1991 BUC had become established as a major research item in the UK and overseas. The CERES programme was designed to provide data specifically for validation of BUC applications. This entailed the inclusion of more detailed PIE, and additional characterization of spent fuel samples through γ spectrometry and neutron source measurements.

Other benefits from the initiation of a collaborative programme included:

- Cross-checks of experimental techniques;
- Exchange of calibration samples and experimentalists;
- Crosschecks of analytical methods.

These provided an important means of increasing confidence in the experimental and analytical programmes as well as encouraging the exchange of ideas and expertise between the participants.

3. REACTIVITY EXPERIMENTS

The DIMPLE experiments were based on reactivity perturbation measurements in three critical assemblies. The assemblies were designed to provide a range of neutron spectra and to emphasize different contributions to spent fuel activity, i.e.:

- High sensitivity to thermal fission in Assembly I;
- Sensitivity to resonance events in Assembly II;
- Sensitivity to thermal events in Assembly III.

Measurements in the MINERVE reactor were made in assemblies designed to simulate the neutron spectra in transport casks and other spent fuel environments. A schematic of the DIMPLE sample handling arrangement is shown in Figure 1.

By this arrangement 10cm long sample of irradiated fuel were introduced into the reactor and the reactivity perturbation was measured by the reactor period technique. Sample reactivity was typically in the range 20 – 50 pcm with an experimental uncertainty of about 0.2pcm. To achieve this accuracy, cycling of the sample, with repeat measurements at the fully-in and fully-out location was required to eliminate reactivity drift associated with temperature change. For Assembly III, which included a Heavy Water annulus around the sample tube, it was also necessary to leave the reactor at power for about an hour before starting measurements, to allow for long-lived photo-neutron groups to build up. These could then be treated as a fixed neutron source in the reactor. Due to this source, it was only possible to use positive reactor periods in this assembly. For Assembly I and II both positive and negative periods were used. For period measurements on irradiated fuel samples a small correction of up to about 0.4pcm was made to allow for the fixed neutron source arising from spontaneous fission and α-n reactions.

A set of fission product samples was constructed from natural UO_2 doped with individual fission product absorbers. The most important fission products in the resonance range are Rh103 and Cs133. These occur naturally at 100% abundance making these samples relatively cheap to manufacture. Absorption in natural samarium is dominated by absorption in Sm149, so this fission product can be ‘simulated’ from the elemental material. Of the other 15 ‘major fission products’ it was not possible to obtain suitable material for Sm150 (too much Sm149 contamination), Sm151 (not available) and Ru101 (too expensive – needs ~17g for suitable reactivity signal).

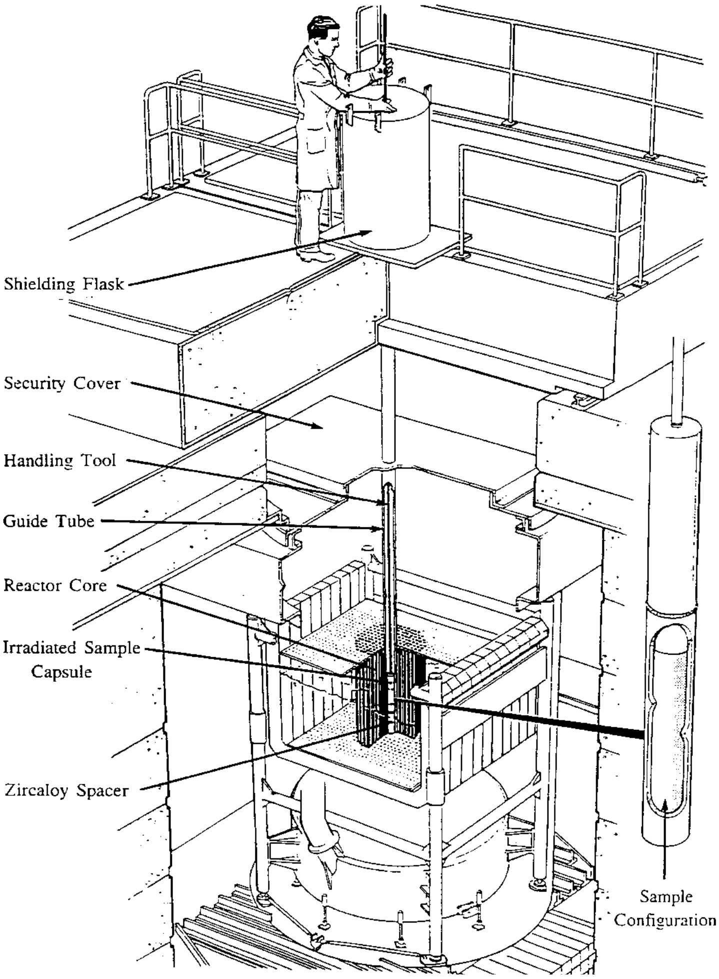

FIG. 1. Schematic of DIMPLE Reactor showing Sample Handling Arrangement.

In addition to the spent fuel and fission product samples, a set of calibration samples was also measured. These samples covered a range of fission and absorber mixes, spanning the range of fission and absorption in the spent fuel and fission product samples. As well as providing the means to calibrate the calculated reactivity scale, these samples also provided a means of establishing the magnitude of modeling uncertainties, which when added to the uncertainty in the measured reactivity gave an overall uncertainty of about 0.4pcm.

4. ANALYSIS OF DIMPLE REACTIVITY MEASUREMENTS

In addition to the spent fuel samples, the first Phase of CERES included measurements on a set of actinide samples where the uranium and plutonium contents were matched to the contents of four irradiated PWR samples. The measured and calculated (WIMS/JEF2.2) results are compared in Figure 2, where good agreement is seen over the whole burnup range. In particular, it is seen that the additional reactivity held in the fission products is much greater than any calculation uncertainty for the actinide-only samples. This provides a simple demonstration of how calculated reactivity loss for Actinides-only BUC will bind real reactivity loss.

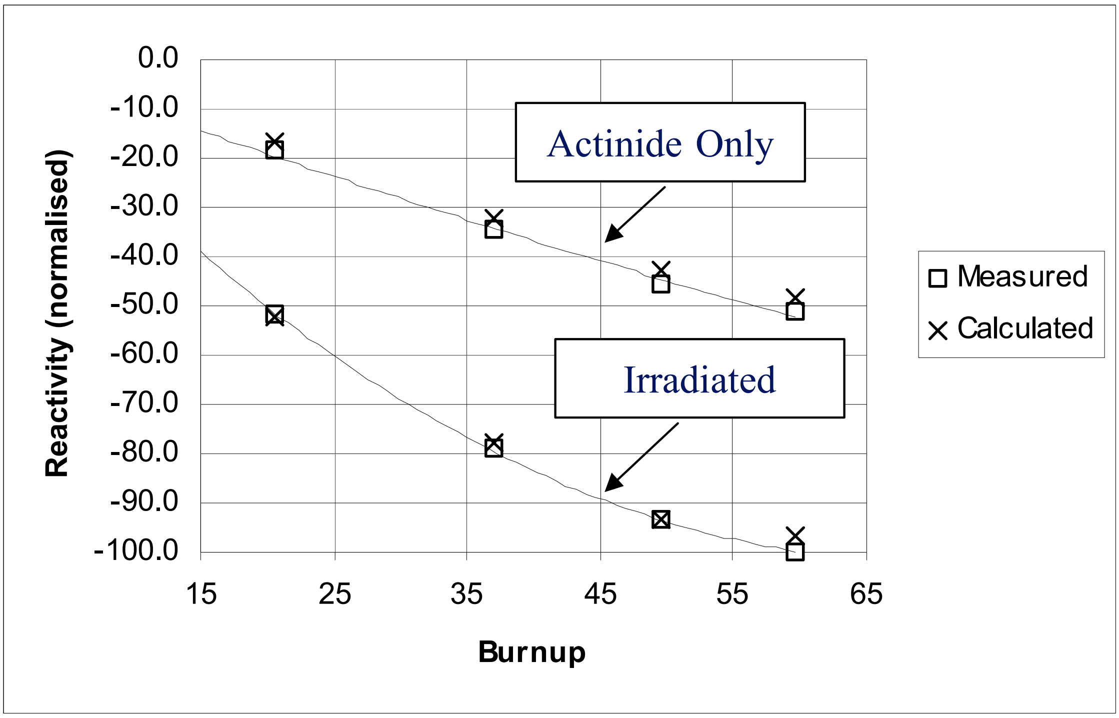

FIG. 2. Comparison of measured and calculated (WIMS7-JEF2.2) reactivity worth.

Results for the fission product measurements are summarized in Table I. It is seen that agreement is good or reasonable for most of the fission products. Exceptions include Rh103, Cs133 and Mo95. The apparent discrepancy for Eu153 is the result of contamination from Eu151 in the sample. Subsequent analysis taking the Eu151 into account shows good agreement. Some inconsistency between the DIMPLE and MINERVE experiment is apparent for Tc99. The other isotopes show good consistency.

4.1 Lessons learnt

In addition to the valuable information gained with respect to criticality code accuracy, several other important lessons were learned. These are summarized below:

- In one of the 'pre-CERES' phases of the DIMPLE programme measurements were made on a high burnup sample from the Zorita reactor. Analysis of the reactivity results for this sample showed very significant discrepancies, much larger than observed for any of the other samples. The PIE results for this fuel were also discrepant. It was concluded that the problem arose from the fact that this sample was taken from a pin which had been pulled from its original element and returned to the reactor in another element to achieve high burnup. It would appear that this creates problems in the depletion calculation, and it was concluded that the results for this sample do not provide a valid test of BUC codes for 'real' fuel elements.
- Once the experiment and the analytical route has been set-up it is relatively cheap to include more samples in the programme. The method provides an accurate measurement of integral cross-section (relative to the calibration sample cross-sections). Advantage can be taken during the BUC measurements to carry out other cross-section validation work.
- For the purposes of simplifying the presentation of the BUC validation argument it is sensible to include an un-irradiated sample of the same fuel type as the spent fuel in the measurements.

- Be prepared for contamination problems handling irradiated fuel samples. The DIMPLE reactor was designated as a low-contamination area and significant decontamination work was needed to bring the samples down to suitable levels.
- There were problems encountered during fabrication of the Tc99 samples, where the technetium dopant interfered with the welding process causing weld failure in the first batch.

Table I. Summary of analysis of fission product sample analysis

Main FP Isotope in Sample	CERES Sample Reactivity (C-E)/E		
	DIMPLE (Assembly II)		MINERVE
	SCALE ENDF/B-V	WIMS7 JEF2.2	WIMS7 JEF2.2
Sm147	-2%	1%	3%
Sm149	-3%	-5%	-2%
Sm152	-1%	1%	-2%
Nd143	-1%	-6%	-4%
Nd145	1%	-1%	0%
Ag109	4%	2%	3%
Gd155	4%	4%	2%
Tc99	5%	7%	-3%
Mo95	19%	6%	11%
Rh103	16%	11%	11%
Cs133 (II)	10%	11%	9%
Cs133 (III)	12%	12%	7%
Eu153	19%	-9%	-10%

Notes:

1. Eu153 Sample to be corrected for Eu151 impurity
2. Two batches of Cs133 samples measured in CERES Phases II & III
3. WIMS/JEF2.2 results consistent with French analysis using APPOLLO2/CEA93

5. ANALYSIS OF PIE MEASUREMENTS

Portions of irradiated fuel taken from close to the reactivity sample position were sent for PIE analysis. This was used to help establish the composition of the reactivity samples (needed for input to the reactivity analysis models) and to provide more general validation of depletion codes. Spent fuel compositions were obtained for uranium, plutonium curium and neodymium for all samples. In Phase III of the programme this was extended to include the 15 major fission products. For PWR samples obtained from the USA (Calvert Cliffs) PIE data was also available from previous analyses. Comparisons with calculated fuel composition is summarized in Table II. The calculations were made using WIMS7 with JEF2.2 data.

For actinides the C/E (calculation/experiment) are consistent with other PIE studies, where a tendency for JEF2.2-based depletion calculations to over-predict residual fissile contents in spent fuel has been noted. For other actinides a tendency to under-predict is seen. Results for fission products show a wider range of C/E values, although in one instance (Sm149) the discrepancy is suspected to arise from problems with the measurement. Further PIE work in the CERES programme shows much better agreement for this isotope, so it may be that the original measurements include interference from some other fission product at this mass number.

Table II. Summary of WIMS7-JEF2.2 depletion analysis for PWR fuel

Isotope	C/E	Isotope	C/E
U234	0.90	Tc99	1.11
U235	1.04	Cs133	0.98
U236	1.00	Cs135	1.03
U238	1.00	Nd143	1.03
Pu239	1.08	Nd144	0.97
Pu240	0.97	Nd145	1.00
Pu241	1.04	Nd146	1.01
Am241	0.94	Sm147	0.91
Pu242	0.89	Sm148	0.90
		Sm149	0.49
		Nd150	1.04
		Sm150	0.88
		Sm151	1.29
		Sm152	1.12
		Eu153	1.05
		Eu155	1.20
		Gd155	1.07

Overall it was concluded from this work that while global fission product worth is well predicted, significant compensating errors between individual isotopes are seen. Uncertainty in the calculated reactivity worth of the major fission products is dominated by uncertainties in the spent fuel inventory, with smaller contributions from uncertainty in the fission product cross-sections. Over-prediction of fissile actinides combined with under-predictions for Pu240 and Pu242 results in conservative estimates of spent fuel composition for Actinide-only BUC.

5.1. Lessons learnt

These may be summarized as follows:

- Fission product separation is a delicate procedure and the first batch of UK PIE measurements produced unreliable data. The problems were apparently solved for a second batch, but confirmation from similar programmes in other laboratories is needed to increase confidence in this type of validation;
- Chemistry on metallic fission products presents particular problems, particularly for Rh103, which is insoluble in nitric acid.

6. GENERAL OBSERVATIONS

The CERES programme has provided key validation for of codes and methods applied to BUC in France and the UK. The data may be used to support BUC application to storage, transport and reprocessing operations. The results may be applied to the validation of both depletion and criticality (k-effective) calculations, although it should be noted that some geometry effects such as axial and radial burnup profile are not addressed by measurements on small samples.

Detailed analysis of the experiments has been made with deterministic methods, which are readily adapted to model the reactivity perturbation techniques used in the measurements. At some stage it might also be of interest to analyze the experiments using Monte Carlo methods, since these are now generally those applied to criticality safety calculations. It is noted that experimental programmes have been planned (e.g. REBUS) where sufficient numbers of spent fuel pins are measured to allow a direct k-difference analysis by Monte Carlo methods.

Since the CERES experiments were completed there has been a continuation of the trend to higher burnup and initial enrichment. Also the use of MOX assemblies has become further established. For these reasons it is likely that further experimental programmes will be carried out, particularly in extending the range of PIE measurements. Perhaps some of the information presented here may help with the planning and execution of those experiments.

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EPRI R&D perspective on burnup credit

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Abstract. "Burnup credit" refers to taking credit for the burnup of nuclear fuel in the performance of criticality safety analyses. Historically, criticality safety analyses for transport of spent nuclear fuel have assumed the fuel to be unirradiated (i.e. "fresh" fuel). In 1999, the U.S. Nuclear Regulatory Commission (NRC) Spent Fuel Project Office issued Interim Staff Guidance – 8 (ISG-8) with recommendations for the use of burnup credit in storage and transportation of pressurized water reactor (PWR) spent fuel. The use of burnup credit offers an opportunity to reduce the number of spent nuclear fuel shipments by ~30%. A simple analysis shows that the increased risk of a criticality event associated with properly using burnup credit is negligible. Comparing this negligible risk component with the reduction in common transport risks due to the reduced number of spent fuel shipments (higher capacity casks for transporting PWR spent fuel) leads to the conclusion that using "burnup credit" is preferable to using the "fresh fuel" assumption. A specific objective of the EPRI program is to support the Goals of the U.S. Industry. These goals are consistent with the original U.S. Department of Energy (DOE) goal defined in 1988: a burnup credit methodology that takes credit for the negative reactivity that is practical (all fissile actinides, most neutron absorbing actinides, and a subset of the fission products that account for the majority of the available credit from all fission products). The determination of the optimum number of fission products to consider in a practical burnup credit methodology validates the approach advocated by researchers from France to first focus on a handful of isotopes that include Sm-149; Rh-103; Nd-143; Gd-155; and Sm-152.

BURNUP CREDIT – U.S. INDUSTRY GOALS

The development of a practical burnup credit methodology must be focused on achievable goals that add value to the use of burnup credit. An NEI/EPRI meeting identified the following goals for a useful commercial burnup credit approach:

(1) Be practical, based on the U.S. Nuclear Regulatory Commission's Interim Staff Guidance 8 (ISG-8).
(2) Support initial enrichments up to 5.0 wt% U-235 without a loading offset.
(3) Include burnable absorbers.
(4) Provide standard axial burnup profiles.
(5) Allow various cooling times.
(6) Include the addition of fission products.
(7) Allow for extension to high burnup (> 40GWd/MTU).
(8) Allow burnup measurements to be replaced by reactor records.
(9) Provide for BWR applications.
(10) Be based upon standard (consensus) parameters
(11) Provide standard benchmark methodologies
(12) Support standard isotopic depletion codes

The first goal is simply that a burnup credit methodology must be practical to be of value to the U.S. commercial nuclear industry. The burnup credit methodology should be simple enough that existing reactor records would allow the determination whether a specific assembly could be stored or transported in a particular cask. Extensive data collection efforts, such as performing cask criticality calculations for each discharged assembly using isotopic data obtained from core-follow calculations, would not be desirable. A go/no-go decision for each assembly should be possible based upon existing reactor records.

The second through fifth goals are to add flexibility to the existing U.S. NRC guidance (Interim Staff Guidance 8, Revision 1, or ISG-8, Rev. 1) [Nuclear Regulatory Commission, Spent Fuel Project Office, Interim Staff Guidance, ISG - 8, Revision 1, Issue: Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks," August 8, 1999]. These have been addressed by an NRC-Research-sponsored PIRT (Phenomena Importance Ranking Table) Panel. If the NRC implements the suggestions of the PIRT Panel, Goals 2 through 5 will have been accomplished.

The sixth goal, the addition of fission products, is key to the value of burnup credit, and is discussed in some detail later in this paper. One important question is just how many fission products are needed.

The seventh goal, higher burnups, is independent from the desire to add fission products, but would likely be effectively addressed by considering fission products, since both the sixth and seventh goals depend upon the extension of current available data.

The eighth goal relates to the NRC's desire for a check of the burnup of each assembly, which is somewhat independent of reactor records.

The ninth goal is the foreseeable need for burnup credit for BWR plants. ISG-8, Rev. 1 does not currently permit the use of burnup credit for BWRs.

Goals 10, 11, and 12 are factors important to the practical implementation of burnup credit. Standard parameters and methodologies, and standard benchmark validation, would streamline the use of burnup credit, and also simplify the NRC review process. These goals may be considered as "cookbooks" for burnup credit calculations for spent fuel casks. For example, some of the parameters required for the burnup credit process are obscure, such as the UO_2 pellet temperature; these parameters are not easily available in existing utility or cask vendor documents. The collection of such parameters and determination of conservative, but reasonable, values would certainly simplify the burnup credit calculations and the NRC review.

Fission products — Determination of optimum number of fission products for a practical commercial burnup credit methodology

The sixth U.S. Industry Goal, the addition of fission products, is a most challenging technical issue because extensive confirmation of our knowledge of nuclear data is required. Nuclear data of two types are required for burnup credit calculations: (i) isotopic build-up and decay data, and (2) nuclear cross-section data. The first type of nuclear data facilitates the calculation of the isotopic concentrations of actinide and fission products in spent fuel. The second type of data allows the reactivity (k-effective) of spent fuel to be determined. This is shown graphically in Figure 1.

At the ICONE 9 meeting, Toubon et al. [Toubon et al. "Current Applications of Actinide-Only Burnup Credit within the COGEMA Group and R&D Programme to Take Fission Products into Account", ICONE 9, Nice, France, April 2001] presented two examples of the potential benefits of making allowance for fission products in criticality studies. For this study, similar types of calculations were performed using on one hand the Yucca Mountain repository project list of 16 fission products that are reasonably stable over geological time scales, and on the other hand five (Sm-149; Rh-103; Nd-143; Gd-155; and Sm-152) of the six fission products selected by Toubon et al. The Cs-133 isotope was not included in this study because of potential volatility considerations.

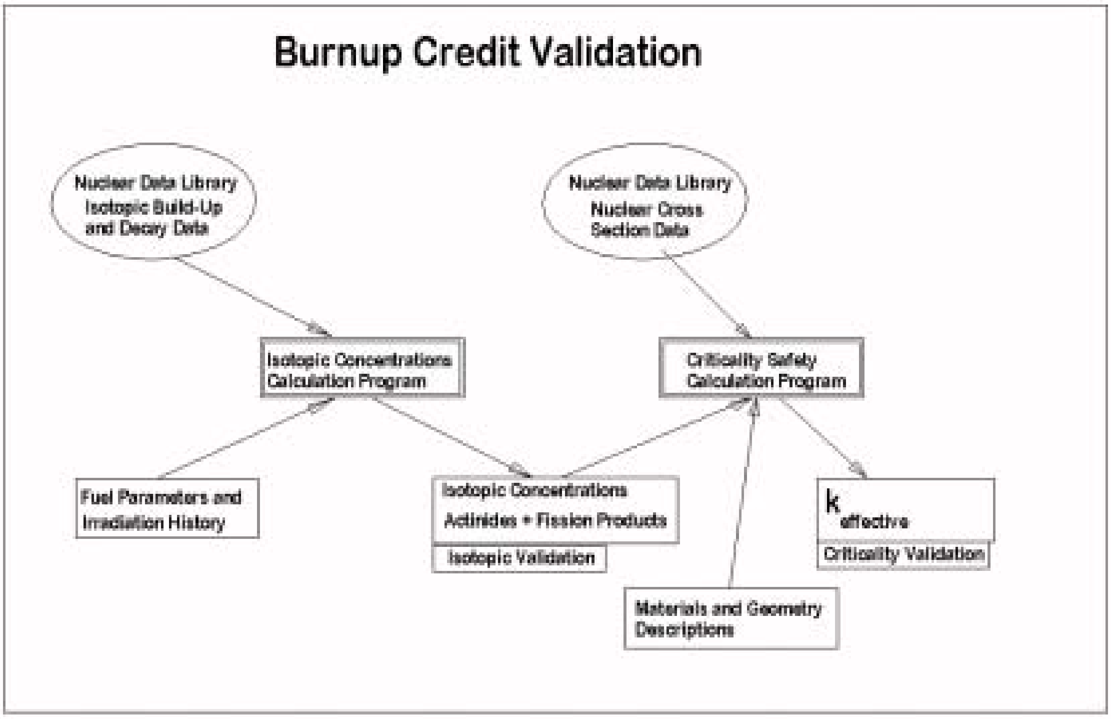
Burnup Credit Validation
Nuclear Data Library
Isotopic Build-Up
and Decay Data
Nuclear Data Library
Nuclear Cross
Section Data
Isotopic Concentrations
Calculation Program
Criticality Safety
Calculation Program
Fuel Parameters and
Irradiation History
Isotopic Concentrations
Actinides + Fission Products
Isotopic Validation
k effective
Criticality Validation
Materials and Geometry
Descriptions

FIG. 1. Burnup credit validation.

The Yucca Mountain approach is very complete and includes all isotopes (except for Cs-133) that make a useful contribution to criticality safety, but the large number of isotopes adds to the level of complexity. The effectiveness of fission product burnup credit was compared, using both the "Yucca Mountain" and "French" (minus Cs-133) approaches, to the actinide-only approach. The commercial VSC-24 storage cask burnup credit loading curve was selected as a means of providing a perspective of what an "acceptable" loading curve might look like.

Calculational method

The Yucca Mountain project had performed a series of studies of the burnup credit loading curves for PWR and BWR fuel assemblies. The PWR studies employed a generalized model of the Crystal River 3 B&W fuel assemblies, and included pre-closure calculations that are analogous to the calculations performed for storage and transport, i.e.the fuel assemblies remain intact. The axial burnup profiles were based upon reactor vendor core following calculations, and had 18 axial nodes. The SCALE 4.3 computer code system SAS2H module was used to obtain the isotopic contents of irradiated fuel for a range of initial enrichments up to 6.0 weight percent U-235 and for burnups up to 60 GWd/MTU. The isotopic data files for the Yucca Mountain loading curve study were extracted and used to prepare a series of input files for the MCNP4 computer code. The spent fuel assemblies were modeled in the VSC-24 cask to allow comparison to a previously existing burnup credit loading curve.

The process of generating a loading curve involves calculation of the required minimum burnup to produce a k-effective of 0.95 for various initial enrichments. The required minimum burnup, expressed as a function of initial enrichment, is the burnup credit loading curve.

Initial calculations were performed for 4.0 wt% U-235 initial, and are shown in Figure 2. The cask basket for these calculations was the actual VSC-24 Multi-Element Sealed Basket (MSB) loaded into the MSB Transfer Cask (MTC). The basket for the VSC-24 consists of 24 square ferritic steel tubes with a 0.200-inch wall thickness. There is no boron neutron absorber in this basket. The k-effective is plotted as a function of burnup in order to determine what burnup provides a k-effective of 0.95, including bias and uncertainty (a bias of 0.02 is assumed for irradiated fuel calculations). Such curves are used to obtain a single data point for the burnup credit loading curves. In Figure 2, the k-effectives are plotted for three different fission product sets: "Actinide-Only" (no fission products), "French Isotopics" (five fission products), and "Principal Isotopes" (Yucca Mountain, 16 fission products). Note that the curves for both the Principal Isotopes and French Isotopics are substantially different from the Actinide-Only curve. The Principal Isotopes and French Isotopics curves require burnups of 44.8 and 49.9 GWd/MTU respectively, while the Actinide-Only curve requires as much as 62 GWd/MTU. The required burnup difference between the Yucca Mountain and French fission product sets is 5 GWd/MTU, which is substantial, but still much less than the 16 GWd/MTU difference between the Yucca Mountain and Actinide-Only burnup requirements. Thus, Figure 2 shows that the French Isotopics list (minus Cs-133) is less effective than the Yucca Mountain isotopic, but much better than Actinide-Only. Figure 2 also shows that the required minimum burnups for all three isotopic selections are excessive, since a 4.0 wt% U-235 initial fuel assembly will typically achieve a burnup of approximately 40 GWd/MTU. The logical conclusion is that some additional means of criticality control is needed, and this is why all large, multi-assembly transport cask baskets contain a supplementary neutron absorber built into the basket structure.

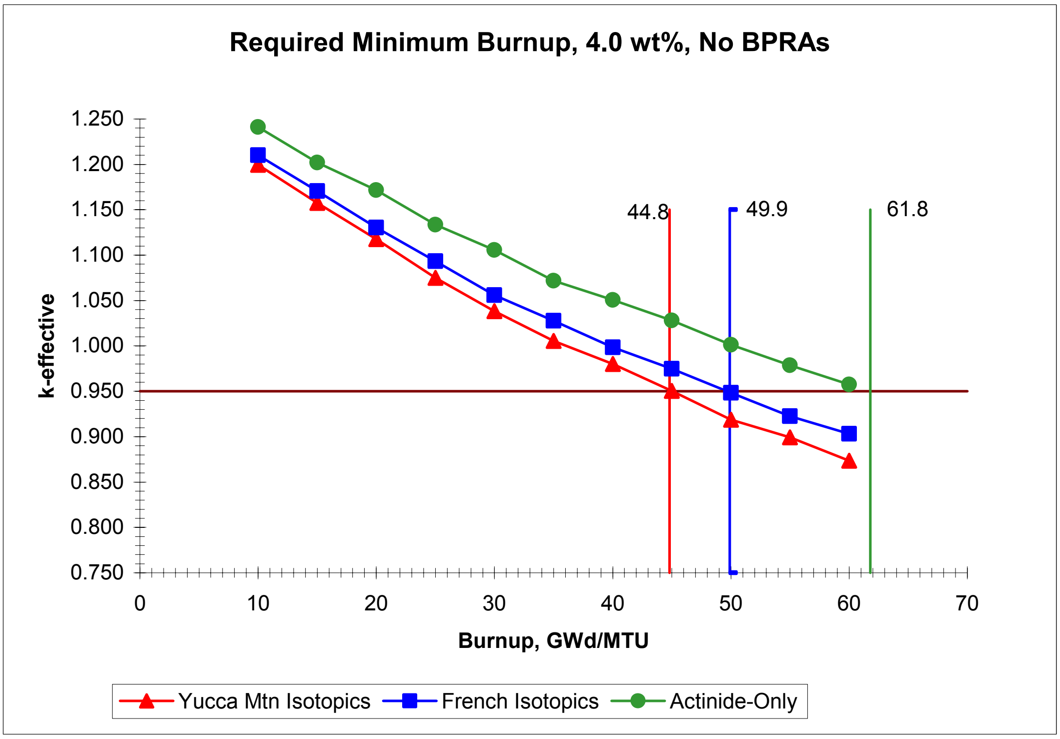

FIG. 2. Comparison for VSC-24 basket.

To investigate the effect of fission product selections in a “poisoned” SNF basket, the square ferritic steel fuel tubes of the VSC-24 basket were changed to stainless steel tubes containing boron. This approach was studied in EPRI TR-102001, “Evaluation of the Transportability of the VSC Basket”. The particular boron/stainless alloy chosen for this new basket was SS-316 B6A, an alloy of stainless steel 316 with 1.6 weight percent of natural boron. Actual basket designs may use a variety of neutron absorber materials, including aluminum/boron carbide composites, boron/aluminum alloys, and stainless steel 304 alloys with up to 2.0 wt% boron. The SS-316 B6A alloy was chosen partly because it was used in conceptual designs for PWR and BWR Waste Packages for Yucca Mountain, and partly because the effective boron areal density is 0.023 g/cm2 B-10. The NRC has allowed credit for only 75 percent of the actual boron content, and the 0.023 g/cm2 B-10 value would require a total boron content of 0.30 g/cm2 B-10. This boron content places the conceptual basket into the middle of the typical PWR neutron absorber content range for transport baskets.

The k-effective calculations were repeated for the Yucca Mountain Principal Isotopes, the French Isotopics, and the Actinide-Only fission product selections. The results of these calculations are shown in Figure 3. Inspection of this figure shows that, as before, the French Isotopics list is closer to the Yucca Mountain fission product set than to the Actinide-Only case. The effect of the addition of boron to the basket is manifested in the expected reduction in the amounts of burnup required for all three fission product sets, and also in a reduction in the spacing between the three curves. The required minimum burnup for 4.0 wt% U-235 initial PWR fuel is less than 30 GWd/MTU for all three fission product lists, which is a much more reasonable burnup requirement. The difference between the Yucca Mountain and French fission product lists is only 1.7 GWd/MTU, one third of the 5 GWd/MTU difference obtained for the non-borated basket calculations. The difference between the French fission product list and Actinide-Only is still a substantial 6.5 GWd/MTU, showing the benefit of the inclusion of fission products. In other words, we obtain most of the benefit of fission products with the simpler French fission product list.

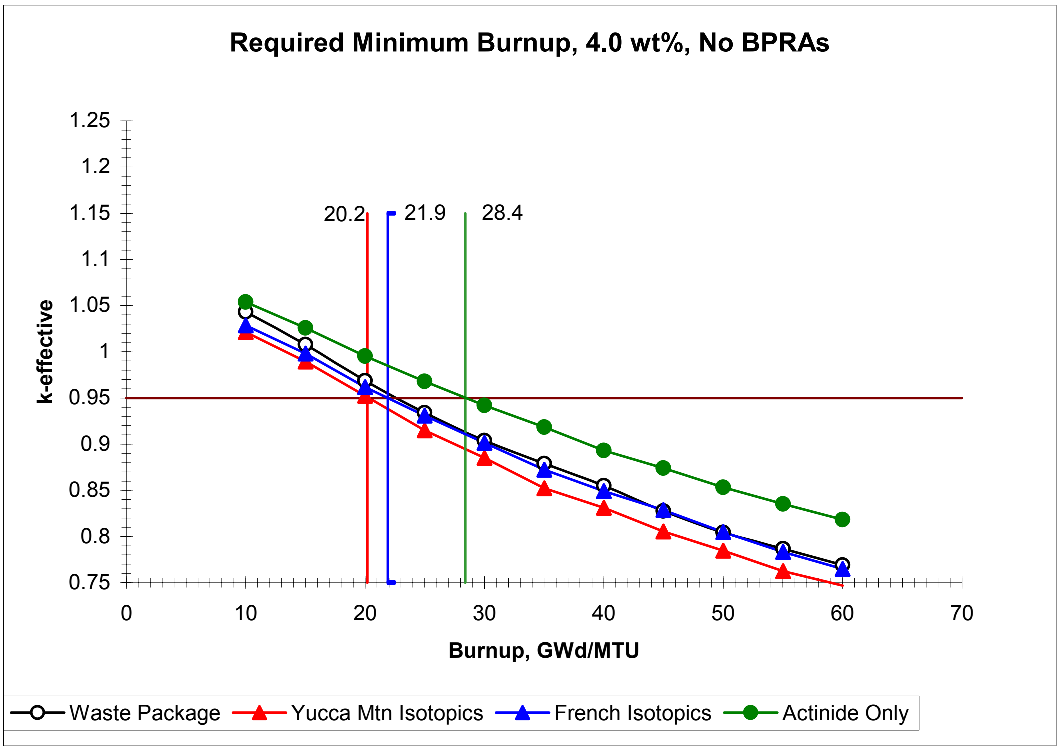

FIG. 3. Comparison for “transportable, poisoned” VSC-24 basket.

These calculations demonstrated that burnup credit alone is not enough for criticality control in a closely packed 24-assembly PWR basket, and that a neutron absorber such as boron must be added to the basket structure to obtain a practical burnup credit loading curve. The calculations for 4.0 wt% U-235 initial showed that the French fission product list (without Cs-133) achieves much of the benefit of the more complex 16-fission product list.

A comparison of the complete loading curves for enrichments up to 5.0 wt% U-235 initial is desirable, so calculations like those shown in Figure 3 were repeated for enrichments of 2.0, 3.0, and 5.0 wt%. In addition, the ability of the "poisoned" VSC-24 basket to accommodate fresh, unirradiated fuel was determined. The results of these calculations are shown in Figure 4. For comparison purposes, the loading curves for the VSC-24 Technical Specifications are also shown in Figure 4, for Westinghouse 14x14 PWR assemblies and for Combustion Engineering 15x15 assemblies. Although the VSC-24 Tech Spec loading curve was not derived with modern burnup credit assumptions (credit was taken for Xenon and Samarium), it has successfully been used by several utilities without undue restriction on the assemblies which can be loaded. An unofficial estimate is that 75 percent or so of the fuel inventory at PWRs could be stored in the VSC-24 based upon the burnup credit loading curve.

Inspection of Figure 4 shows the comparison of loading curves obtained with different fission product isotopic selections. The French Isotopic set (minus Cs-133) is nearly as effective as the 16-fission product set of the Yucca Mountain Principal Isotopes. The Actinide-Only case is not as effective as the French or Yucca Mountain fission product selections. The VSC-24 Tech Spec loading curves are less effective, but it must be recalled that this basket does not contain a neutron absorber and is hence much less expensive to fabricate.

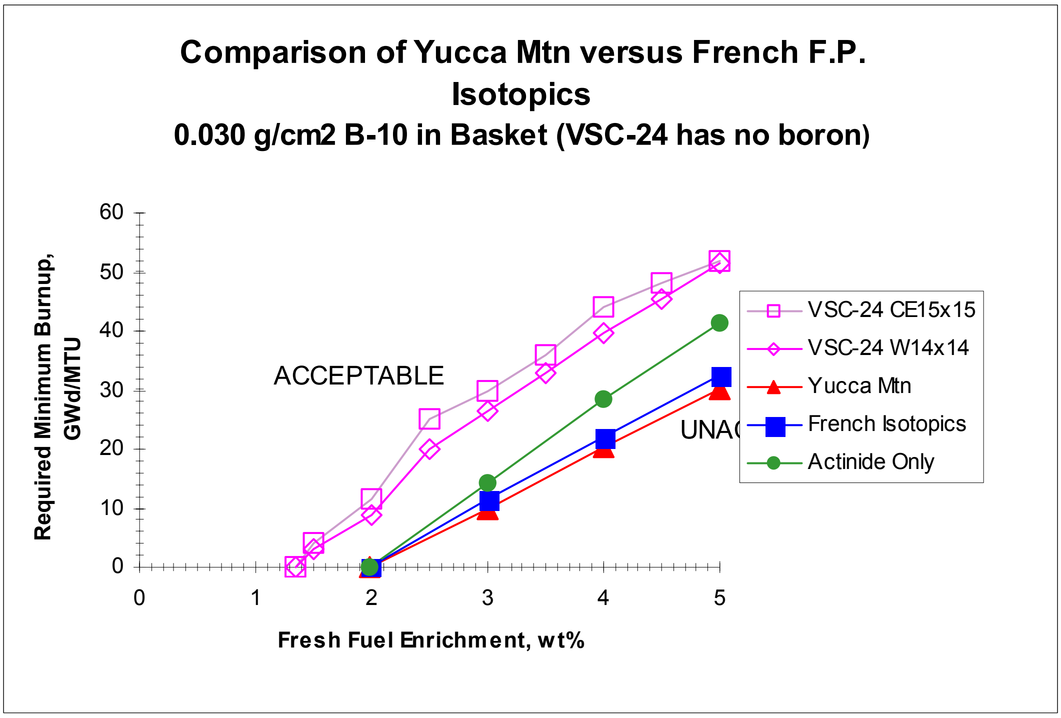

FIG. 4. Comparison of fission product isotopic selections.

In summary, there are three basket criticality solutions based on hardware: no poison basket cells, poison basket cells, and poison with flux trap basket cells. Adding a neutron absorber like boron adds to cost, but adding a flux trap (a neutron absorber/water gap sandwich affair) adds to cost and reduces payload. By itself, burnup credit alone cannot accommodate the spent fuel inventory. With a neutron absorber built into the basket, burnup credit can accommodate most of the fuel inventory (Figure 5). Actinide-only burnup credit captures many of the fuel assemblies, but misses some under-burned higher enrichment fuels. The simpler French Fission Product isotopic approach captures almost all of the assemblies accommodated by the Yucca Mountain isotopic selection.

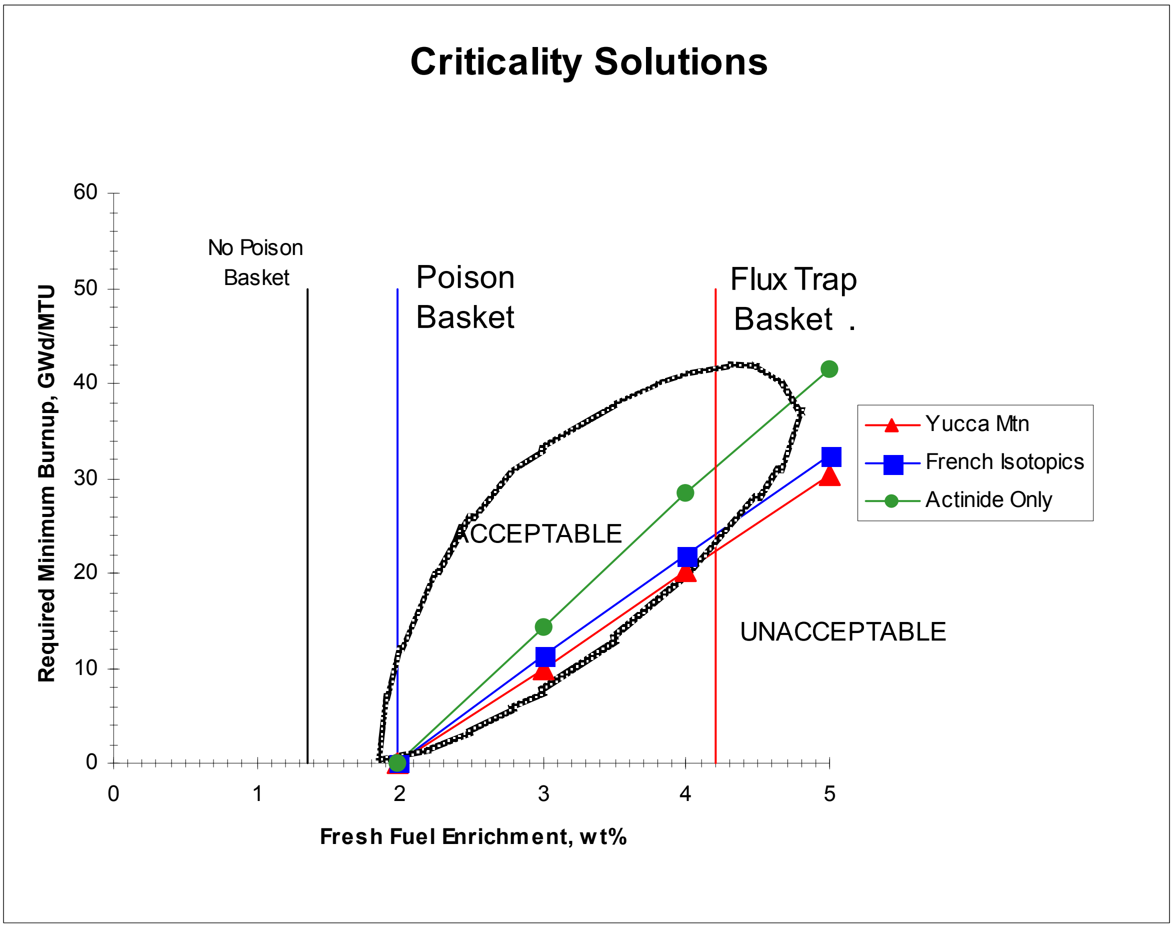

FIG. 5. Effects of built-in neutron absorber relative to fission products.

The simplified approach with five fission product isotopes could result in more rapid approval by the regulators because there are fewer isotopes to defend. Given that the five-isotope approach yields nearly the same benefit as the 16-isotope approach, little is lost by the simplification. This suggests another possibility: that only one or two isotopes might yield nearly the total fission product benefit. Our calculations (not reported in this paper) show that this is unlikely to be the case. Rather, each of the five fission product isotopes contributes a part of the benefit, and the law of diminishing returns has not caused any of the five to be ineffective. Thus the five-isotope approach may be an optimum solution for a commercial burnup credit methodology.

A perspective on risk

This analysis examines the risks associated with using burnup credit. The quantitative estimate of risk associated with using burnup credit is shown to be negligibly small. Although not estimated quantitatively, the overall risk reduction, which is realized by the expectation of fewer shipments when burnup credit is used, is sizable.

The NRC transportation regulations require consideration of the effect of fresh-water in-leakage for criticality analysis of a single package for transport of fissile material [U.S. Nuclear Regulatory Commission, Title 10 Code of Federal Regulations (CFR) Part 71]. The regulations do not preclude the use of burnup credit in demonstrating sub-criticality; they simply require that the package (i.e. cask and contents) be shown to be sub-critical.

Current regulatory practice, which has been applied to all currently approved spent fuel transport casks, does not account for the reduced reactivity of spent fuel in demonstrating sub-criticality under prescribed regulatory conditions. The rules and practices currently applied to NRC-Certified transport casks are listed below:

(1) Sub-criticality is assured, that is, $k_{eff} < 1$.
(2) Moderation by water occurs to the most reactive credible extent.
(3) Full reflection of the system on all side by water occurs.
(4) The system is in its most reactive credible configuration consistent with the chemical and physical form of the material.

Items (1) to (4) are part of the U.S. regulations.

(5) The allowed keff is then reduced from 1 to account for such things as modeling and calculational biases and uncertainties.
(6) Additionally, the allowable keff is further reduced by applying an arbitrary criticality safety margin of 5%, i.e. _keff = -0.05.

Item (5) is simply an acceptable approach to criticality safety analysis. Item (6) is an arbitrary safety factor applied to analysis.

- The fuel is assumed to be in its most reactive state, which is generally unburned.

Item (7), the “Fresh Fuel” assumption is the practice that is modified when “Burnup Credit” (BUC) is used. It should be noted that the word "modified", not "eliminated”, is used. The modified item (7) might read:

(7a) "The fuel is assumed to be in its most reactive arrangement, after credit for the spent fuel's burnup is determined using conservative depletion analysis."

Burnup credit only seeks a change in the fresh fuel assumption.

The issue of “Burnup Credit” vs. the “Fresh Fuel” assumption for the evaluation of PWR spent fuel reactivity in a transportation package involves a trade-off. On one hand, consideration and credit for the reduced reactivity of the spent fuel allows for a better utilization of the package volume; this results in a greater number of assemblies per package, and, in turn, in a smaller number of shipments. On the other hand, the fresh fuel assumption provides additional margin for criticality considerations as it leads to the addition of engineered poisons within the package cavity.

Can any incremental reduction in criticality likelihood (and subsequently risks) be justified against the reduction in transportation risks deriving from using burnup credit?

NRC-sponsored work [NUREG/CR-4829, referred to as the "Modal Study") discusses the likelihood of a rail cask accident with a greater than 2% strain coupled with a concurrent submersion [Modal Study, Subsection 9.3.2.4]. Rail shipping is particularly relevant because it is required for the dual-purpose systems that are or will be implemented at reactor sites. Under the rail-shipping scenario of the Modal Study, " this type of accident is estimated to occur once every ten million years."

The estimated frequency of a criticality event is then obtained by multiplying the frequency of the accident referred to above [i.e. 10-7/year] by the likelihood that the specific package involved in the accident contain enough reactivity under the moderation and geometric conditions of the accident to result in a critical configuration [i.e. 10-x /accident].

Assuming that the package system is a BUC-designed system, such a likelihood, i.e.10-x/accident, would be acceptably low if:

- Non-conservative errors associated with the specific BUC methodology are smaller than the sum of (i) the administrative margin ($\Delta k_{eff} = 0.05$) and (ii) the systematic bias in k_{eff} introduced in the methodology to account for enveloping conditions and uncertainties.
- The potential for human errors is small enough to protect against non-conservative fuel assembly insertion errors (misloadings)

Based on probability data for human error [Homes & Narver in NSS-8191.1, Transportation Accidents Risks in the Nuclear Industry], and given that:

- Misloadings can introduce less reactivity as well as more reactivity,
- Only misloadings in specific cask or canister locations have a marked effect; and
- Two checks are required for every fuel movement

It can then be estimated that the probability of a non-conservative misloading can be as high as 10^{-3} and as low as 10^{-5} for a large package. In addition, past analyses have shown that more than one misloading is required to approach criticality conditions. This brings the likelihood of having to deal with a critical configuration, given a severe enough accident, to an estimated $(10^{-3}$ to $10^{-5})^n$/accident, where "n" is the required number of non-conservative misloadings. Using the conservative assumption that only two non-conservative misloadings are required, the likelihood is (10^{-6} to 10^{-10})/accident.

Given that the frequency of an accident severe enough to result in significant damage to the package (coupled with submersion) is already very low [10^{-7}/year], the expected frequency of a critical configuration under the rail shipping scenario of the "Modal Study" is essentially zero (10^{-13} to 10^{-17}/year is a meaningless number!) Estimates of the consequences of a criticality accident are inconsequential from a risk standpoint.

Therefore, the fresh fuel assumption results in a negligible numerical reduction in critical configuration likelihood. On the other hand, by using a BUC approach, the reduction in the number of shipments is real, and results in a measurable reduction in risk associated with lower exposures to workers and public, lower fatalities and injuries from non-radiological consequences of accidents, and lower property damage.

A risk-informed approach would seek an overall reduction of the risks associated with spent fuel shipments. In a few years, the Department of Energy will begin a thirty-year program of shipping approximately 63 000 metric tons of spent nuclear fuel from commercial nuclear reactors to a first deep geological repository. The use of burnup credit offers an opportunity to reduce the numbers of shipments required to move this material by ~30%. The benefits of reducing risks associated with transport are obvious. This analysis suggests that the increased risk of a criticality associated with properly using burnup credit is negligible. Comparing this negligible risk component with the reduction in overall risk associated with transport actually supports using burnup credit.

DEPLETION CRITICALITY PARAMETERS THAT GUARANTEE A BOUNDING APPROACH. APPLICATION DEPENDENCE OF BUC PARAMETER IMPORTANCE

(Session 2.2)

Impact of the initial enrichment on the end effect

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Abstract. One of the most important issues in a burnup credit criticality safety analysis of a spent fuel management system of interest is the determination of the end effect due to the axial burnup distribution. In the paper on hand the impact of the initial enrichment of the fuel on the end effect of PWR axial burnup profiles is analyzed. It is shown that the end effect tends to increase with decreasing initial enrichment. For low initial enrichments, however, this tendency may be outweighed by the fact, that the higher the burnup is and the lower the initial enrichment is, the lower is the decline of the reactivity with increasing burnup. The impact of the initial enrichment of the fuel on the end effect is analyzed for a wet storage system assumed to be loaded with Westinghouse 16x16-21 Standard fuel assemblies. The analysis is made for normal operation conditions assuming the storage racks to be flooded with unborated water. No cooling time credit is taken.

1. INTRODUCTION

In a fresh PWR fuel assembly the axial flux distribution is more cosine shaped (as reflected by the axial power distributions shown in Reference [1], Figures 3, 5, and 7). The fuel near the axial center of the assembly is therefore depleted at a faster rate than at the ends. With increasing burnup the flux shape flattens out due to the higher fuel depletion and fission product poisoning near the center. However, due to the neutron leakage at the ends of the fuel zone the burnup drops off rapidly near the ends (cf. [1], Figures 4, 6, and 8).

For small average burnups the absolute difference between the burnup of the center zone and the burnup at the ends of the fuel zone is relatively small. The most reactive zone of the fuel is hence more towards the center zone because of the leakage at the ends of the fuel zone. Due to the fact that the burnup in the center zone is slightly higher than the average burnup, the end effect is negative. However, with increasing average burnup the most reactive region of spent fuel is towards the ends, where there is an optimum balance between increased reactivity due to lower burnup and increased leakage due to closer proximity to the fuel ends. The end effect might become positive, therefore.

2. PARAMETERS AFFECTING THE END EFFECT

The optimum balance between increased reactivity due to lower burnup and increased leakage due to closer proximity to the fuel ends is affected by all the parameters, which have an impact on:

- the isotopic concentrations and hence the reactivity of the fuel (depletion parameters, cooling time);
- the neutron spectrum of the spent fuel management system of interest and hence the reactivity worth of the nuclides (design parameters, depletion parameters, cooling time);
- the axial bounding conditions (highly reflective, high leakage) (design parameters).

One of these parameters is the initial enrichment of the fuel, which determines, together with the depletion parameters, the burnup and the cooling time, the isotopic content of the spent fuel. In the paper on hand the impact of the initial enrichment on the end effect is analyzed.

2.1. Initial Enrichment, Fission Product Buildup and End Effect

A higher initial enrichment results in

- a lower U-238 capture rate and;
- a higher U-235 depletion rate and hence in a higher fission product buildup rate (cp. Figure 1, e.g.).

This is true for the center zone of the fuel as well as for the end zones. However, the net buildup rates of plutonium and the fission products are usually lower in the center zone (due to the higher burnup in this zone) than in the end zones. It is to be expected, therefore, that the end effect decreases with increasing initial enrichment.

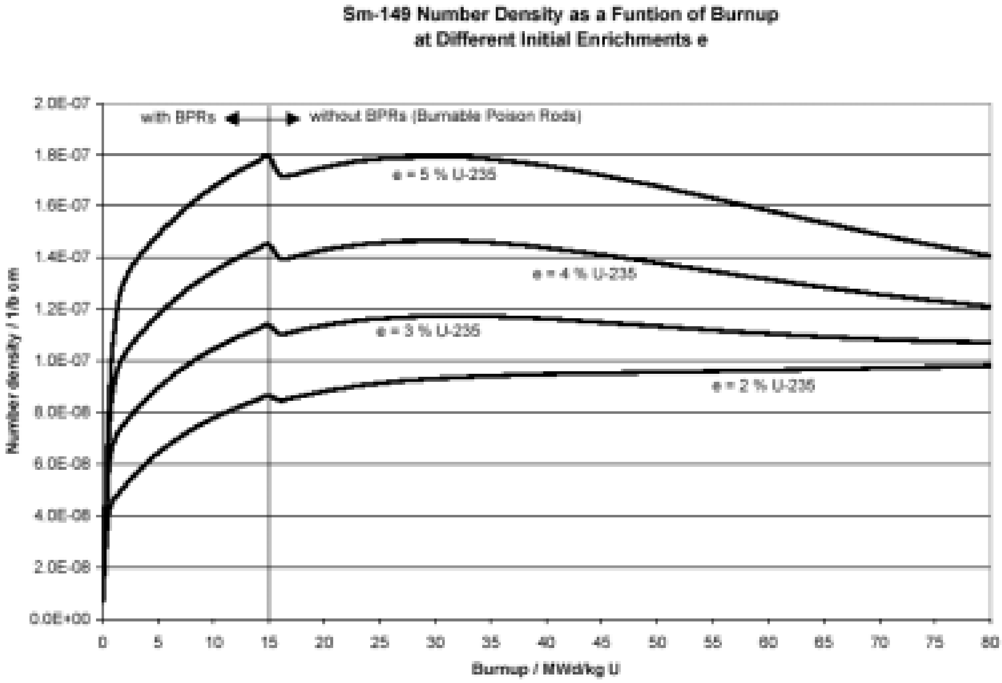

FIG. 1. Depletion of the Westinghouse 16x16-21 Standard Fuel Assembly in the Krzko Core (NEK, Slovenia): Sm-149 Number Density as a Function of Burnup and Initial Enrichment for Bounding Depletion Conditions. (As indicated, up to a Burnup of 15 MWd/kg U Burnable Poison Rods were assumed to be present).

Figure 1 shows for example the number density of Sm-149 as a function of burnup at initial enrichments between 2 wt.-% and 5 wt.-% U-235. It is instructive to observe how the ratio between the number density at a higher burnup and the number density at a lower burnup is changing with increasing initial enrichment:

- At 2 wt.-% initial enrichment this ratio remains always greater than one. With increasing initial enrichment this ratio may become less than one and may decrease with increasing initial enrichment. Accordingly, given any axial burnup distribution, at 2 wt.-% initial enrichment the Sm-149 content of the spent fuel is greater in the center zone of the fuel than at the end zones. With increasing initial enrichment however, the Sm-149 content of the center zone may fall below the Sm-149 contents of the end zones, in particular in case of higher average burnups. Thus, due to the fact that Sm-149 is a very important absorber, it is to be expected that, for higher average burnup values, the end effect tends to decrease with increasing initial enrichment;

- On the other hand, as can be seen from Figure 1, the ratio of the number density at 30 MWd/kg to the number density at any given burnup less than 15 MWd/kg is decreasing with increasing initial enrichment. Therefore, it is to be expected that also for lower average burnup values the end effect decreases with increasing initial enrichment.

However, the end effect is affected by all the parameters mentioned in section 2. Thus, it is not self-evident whether or not the impact of the initial enrichment on the end effect can be really observed.

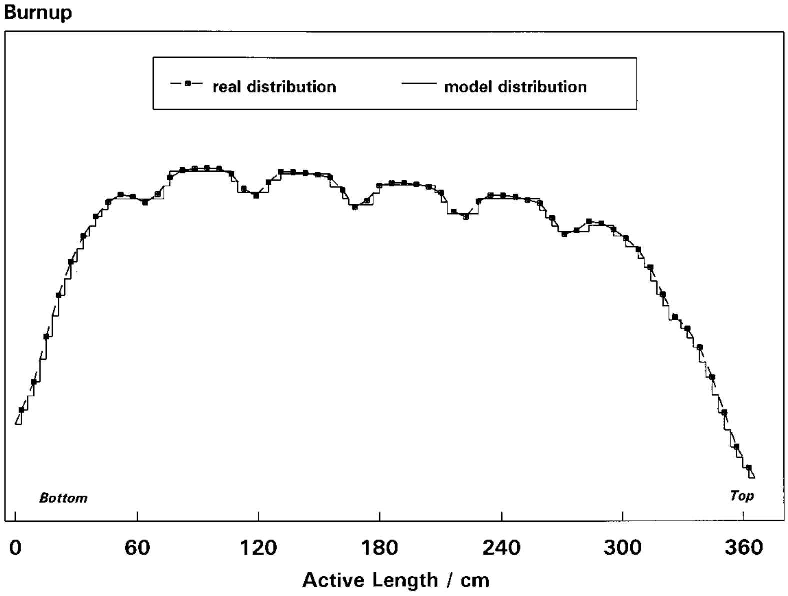

FIG. 2. Example for the Axial Burnup Distributions Obtained from Ref. [3] and Modeling of these Distributions.

2.2. Brief Account of the Parameters Affecting the End Effect

Due to the higher moderator density in the lower half of the core axial burnup shapes are usually asymmetrical, [1]. It is known that the end of the fuel zone, which is burned least, determines the end effect. This end is usually the top end of the fuel zone, cp. [1] and Figure 2. The asymmetry of an axial shape and hence the end effect are strongly dependent on the average burnup and significantly affected by

- reload patterns (determining the interactions between fresh fuel assemblies and fuel assemblies with different burnup shapes at begin of cycle, cp. Ref. [1], Figures 3 and 5),
- control rod movements;
- the use of axial power shaping rods;
- the presence of integral burnable poisons;
- the presence of axial blankets;
- extended low power operations;
- all the depletion conditions affecting the U-238 capture rate and the U-235 depletion rate (specific power, spectrum hardening due to higher boron concentration or presence of burnable poison rods);
- the cooling time.

In addition, the end effect is also dependent on the active length of a fuel assembly. This is often missed. Usually the end effect decreases, at given average burnup, with decreasing active length. This is due to the process of the flattening-out of the axial power distribution with increasing burnup.

2.3. Asymmetry of the Axial Shapes, Initial Enrichment and End Effect

Due to the flattening-out of the axial power distribution with increasing burnup the impact of the initial enrichment on the end effect should become more apparent for lower average burnups. The flattening-out of the axial power distribution with increasing burnup is reflected by the fact that, as shown in Ref. [1], the ratio of the burnup of the top end of the fuel zone to the average burnup increases with increasing burnup. Thus, the higher the average burnup is, the lower should be the impact of a change in the initial enrichment and hence a change in the U-238 capture rate, the U-235 depletion rate and the fission product buildup rate on the end effect. In other words, it is to be expected that the impact of the initial enrichment on the end effect is increasing with decreasing average burnup. On the other hand however, the end effect is decreasing with decreasing average burnup, cp. Ref. [1].

In the following the analysis performed to reveal the impact of the initial enrichment on the end effect is presented.

3. STUDY OF THE IMPACT OF THE INITIAL ENRICHMENT ON THE END EFFECT

3.1. Brief Description of the Spent Fuel Management System Analyzed

The impact of the initial enrichment on the end effect is evaluated for the Westinghouse Fuel Design 16x16-21 Standard used in nuclear power plant Krzko (NEK), Slovenia. The active length of this fuel type is 365.8 cm.

The fuel assemblies are assumed to be loaded in wet storage pool region II racks consisting of storage cells illustrated in Figure 3[1]. The analysis of the end effect is made for normal operation conditions assuming the racks to be flooded with unborated water. No cooling time credit is taken.

3.2. Brief Description of the Method Used for the Analysis

The method used for evaluating the end effect is described in detail in Ref. [2]. As illustrated in Figure 4, the end effect Δk, defined as the difference between the neutron multiplication factor of the spent fuel management system under study obtained by taking account of the axial distribution of the burnup and the neutron multiplication factor obtained by assuming a uniform distribution of the average burnup of this axial distribution, is expressed as difference ΔB between the average burnup of this axial distribution and the so-called “equivalent uniform burnup” [2], which is the uniformly distributed burnup that results in the same neutron multiplication factor as obtained with the real axial burnup distribution. As can be seen from Figure 4, a positive end effect Δk results in a positive difference ΔB.

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[1] Storage racks designed for burnup credit are usually named as “region II” racks.

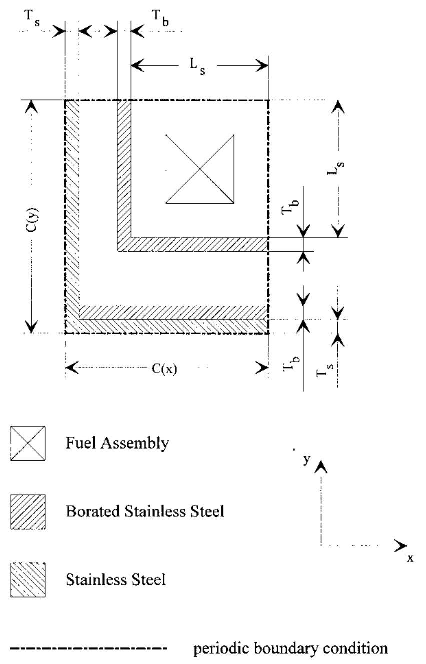
T_s
T_b
L_s
C(y)
C(x)
Fuel Assembly
Borated Stainless Steel
Stainless Steel
periodic boundary condition
y
x

FIG. 3. Illustration of a Storage Cell of the Region II Storage Racks Used for the Analysis (Ts = 3.15mm, Tb = 2.15mm, Ls = 206.5 mm, C(x) = 226mm, C(y) = 231mm).

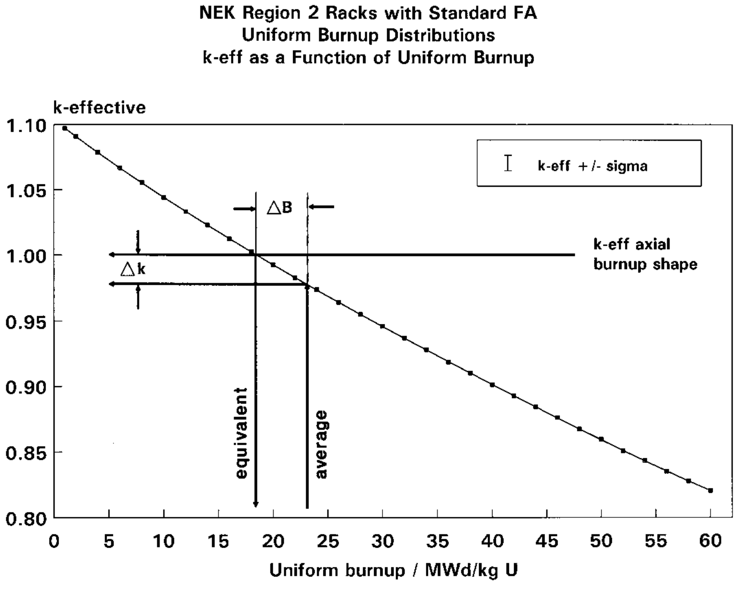

FIG. 4. Illustration of the Method Used for Evaluating the End Effect.

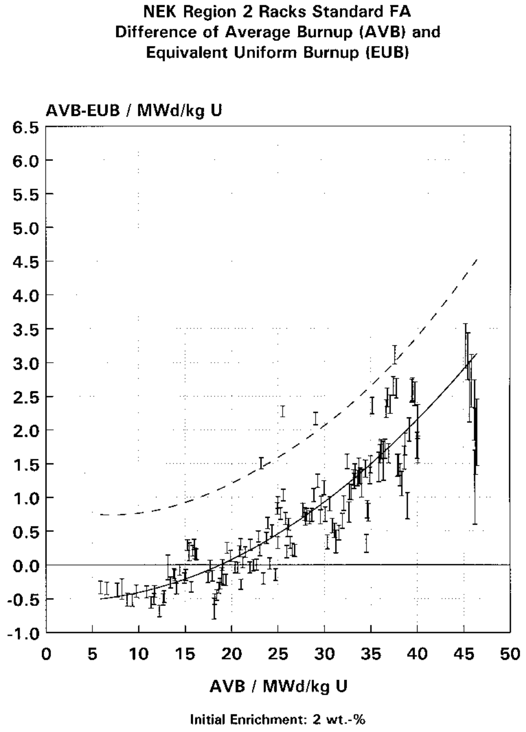

FIG. 5. Analysis of the End Effect at 2 wt.-% Initial Enrichment.

As explained in detail in Ref. [2], due to the fact that the neutron multiplication factors involved are statistics the equivalent uniform burnup assigned to an axial burnup distribution is not given by a point, but by a confidence interval. The difference ΔB between average burnup and equivalent uniform burnup is hence given by an interval as shown in Figure 5. Thus, each bar shown in this figure represents the end effect due to a particular axial burnup distribution. It is obvious, that the method described makes it possible to analyze any desired number of axial burnup shapes.

The axial burnup distributions were generated on the basis of incore measurement data from NEK, cf. Ref. [3]. The distributions are given on 60 axial nodes, and they are modeled as described in Ref. [1] by inscribing first burnup step functions in the shapes and collapsing then steps with minor differences in burnup together. Figure 2 gives an example for the modeling of the axial burnup shapes.

The depletion calculations necessary to obtain the isotope number densities as a function of burnup and initial enrichment for the different axial zones were performed with the aid of CASMO-3 [4]. For the determination of the neutron multiplication factors the criticality analysis portion of the SCALE-4.4a [5] package was used.

As already shown in Figure 5, the results obtained are presented in the form of diagrams showing the difference ΔB between average burnup and equivalent uniform burnup as a function of the average burnup. The curve presented by the solid line in Figure 5 represents a linear least squares fit of the results, and the curve given by the dashed line in this figure represents the one-sided upper 95%/95% tolerance limit of the fit.

3.3. Results

The results obtained are shown in Figures 5 through 8. In Figure 9 the linear least squares fits of the results are summarized. As can be seen, for lower average burnup values the end effect is higher for 2 wt.-% initial enrichment than for 4 wt.-% and 5 wt.-% initial enrichment, as it was expected. However, for higher average burnup values the end effect for 2 wt.-% initial enrichment becomes smaller than for all the other initial enrichments analyzed. This is due to the fact that the decrease in reactivity with increasing burnup (cp. Figure 10) slows down with the depletion of U-235 (cp. Figure 11), the consumption of U-238 (cp. Figure 12) and the buildup of plutonium (cp. Figure 13). Thus, for higher burnup values the reduction in reactivity with increasing burnup decreases with decreasing initial enrichment, cf. Figure 10. This behavior impacts in particular the end effect at 2 wt.-% initial enrichment, as can be seen from Figures 10 and 5. For lower burnups however, where the decline of the reactivity is more or less the same for all the initial enrichments analyzed (cf. Figure 10), the end effect decreases with increasing initial enrichment (cf. Figure 9), as it was expected.

The fact, that for higher burnup values the reduction in reactivity with increasing burnup decreases with decreasing initial enrichment, is still apparent at 3 wt.-% initial enrichment: As can be seen from Figure 6 in comparison to Figures 7 and 8, for higher average burnup values the confidence intervals of ΔB obtained for the individual shapes are larger for 3 wt.-% initial enrichment than for 4 wt.-% and 5 wt.-% initial enrichment. Nevertheless, for the whole range of average burnup values analyzed, the end effect is always higher for 3 wt.-% initial enrichment than for 4 wt.-% and 5 wt.-% initial enrichment, and decreases with increasing initial enrichment (cf. Figure 9), as it was expected. In particular, the end effect, after becoming first negative (cp. section 1) begins to increase again at an average burnup value which is significantly lower in case of 2 wt.-% and 3 wt.-% initial enrichment than in case of 4 wt.-% and 5 wt.-% initial enrichment, cf. Figure 9. This demonstrates that, even though the end effect is lower at lower average burnup, the impact of the initial enrichment on the end effect seems to be higher at lower average burnup than at higher average burnup values.

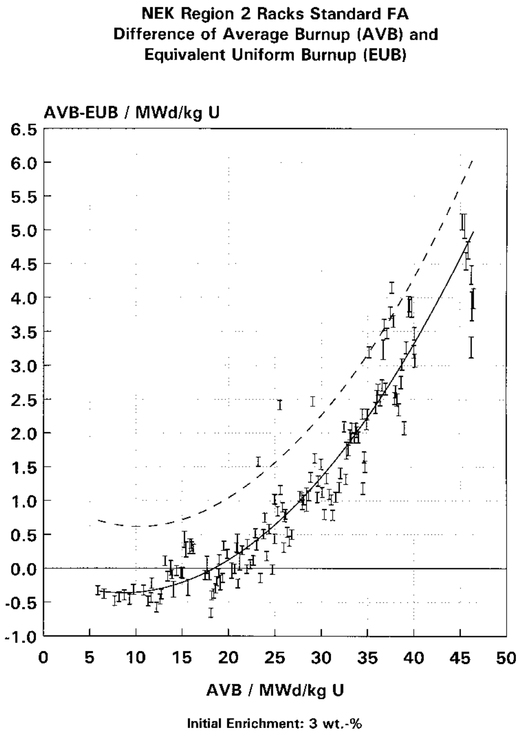

FIG. 6: Analysis of the End Effect at 3 wt.-% Initial Enrichment.

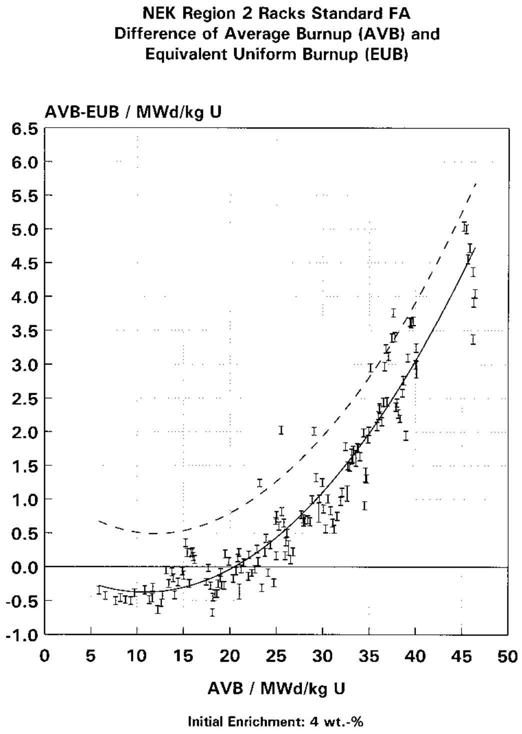

FIG. 7: Analysis of the End Effect at 4 wt.-% Initial Enrichment.

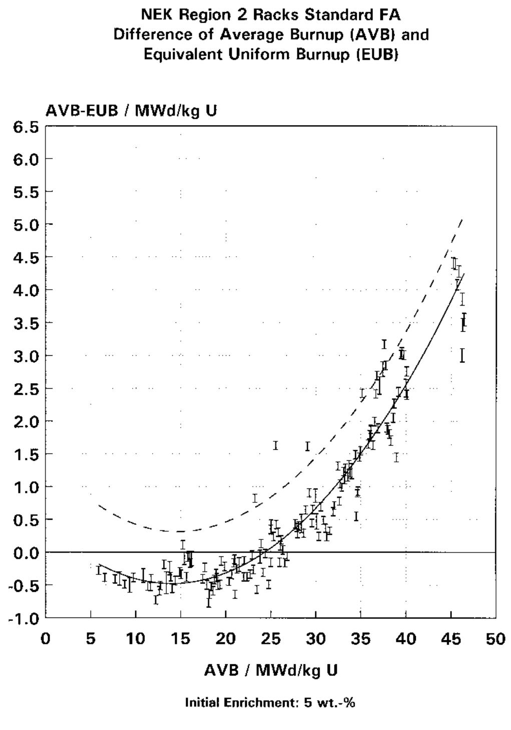

FIG. 8: Analysis of the End Effect 5wt.-% Initial Enrichment.

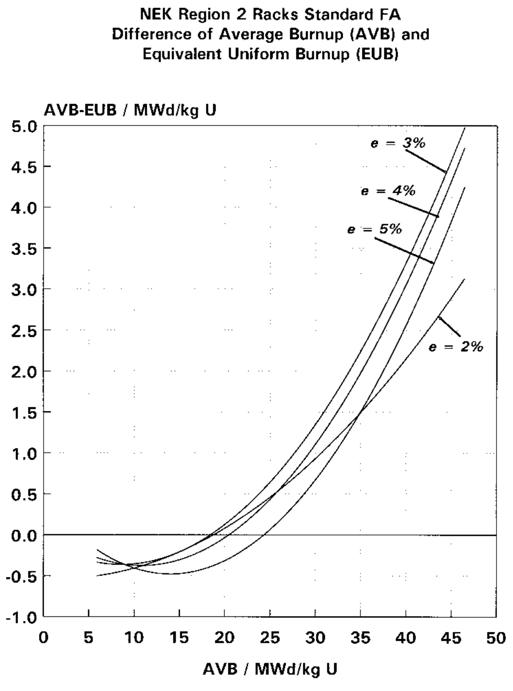

FIG. 9: Summary of the Analysis of the End Effect a (Linear Least Squares Fits from Figures 5 through 8).

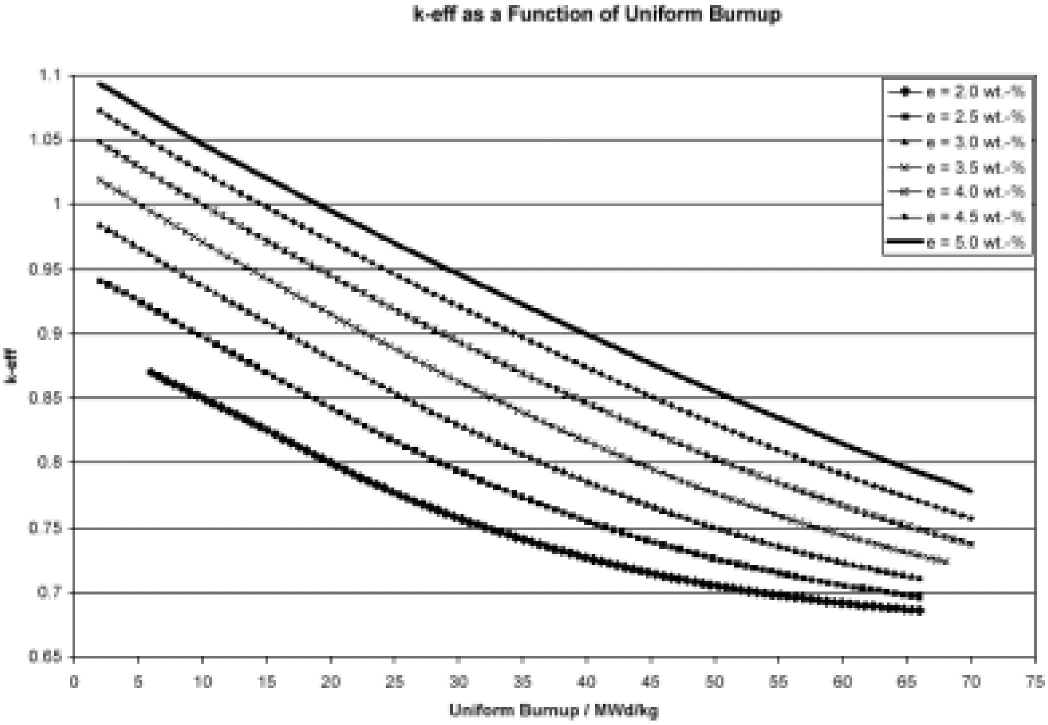

FIG. 10: Neutron Multiplication Factor of the Storage Facility Analyzed (cp. Figure 3) as a Function of Burnup at Different Initial Enrichments.

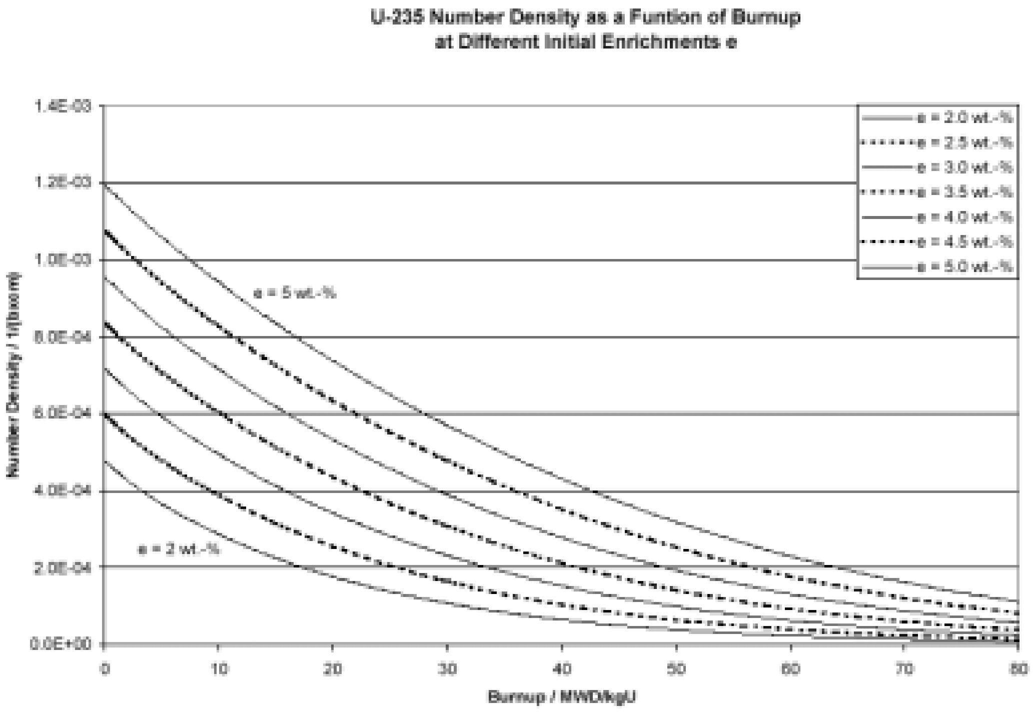

FIG. 11: Depletion of the Westinghouse 16x16-21 Standard Fuel Assembly in the Krzko Core (NEK, Slovenia): U-235 Number Density as a Function of Burnup and Initial Enrichment for Bounding Depletion Conditions.

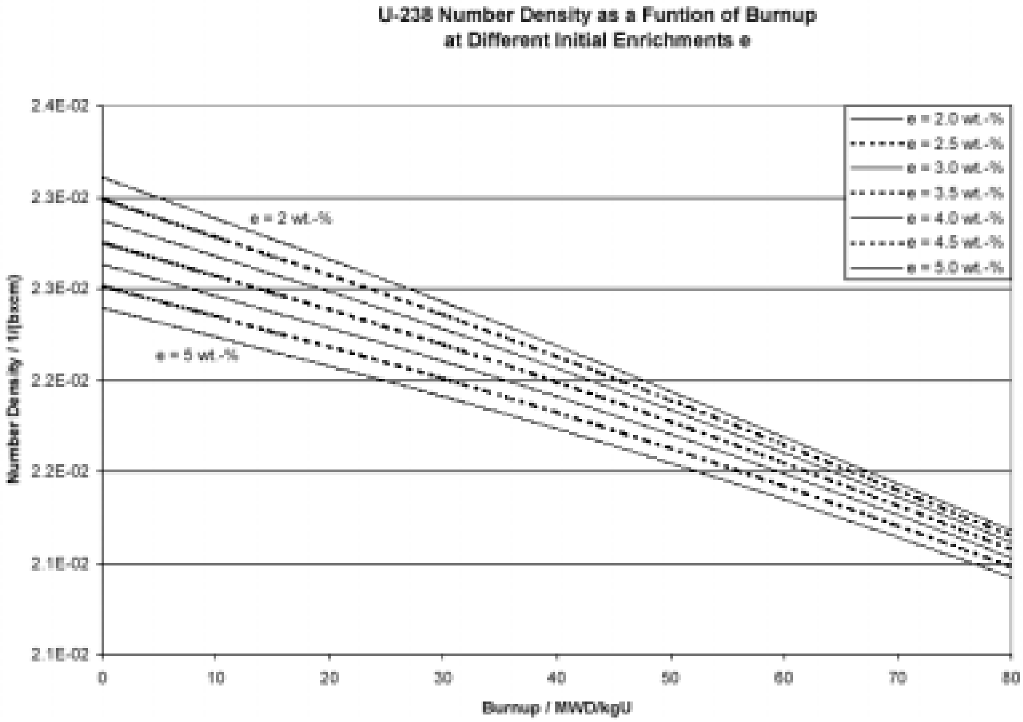

FIG. 12: Depletion of the Westinghouse 16x16-21 Standard Fuel Assembly in the Krzko Core (NEK, Slovenia): U-238 Number Density as a Function of Burnup and Initial Enrichment for Bounding Depletion Conditions.

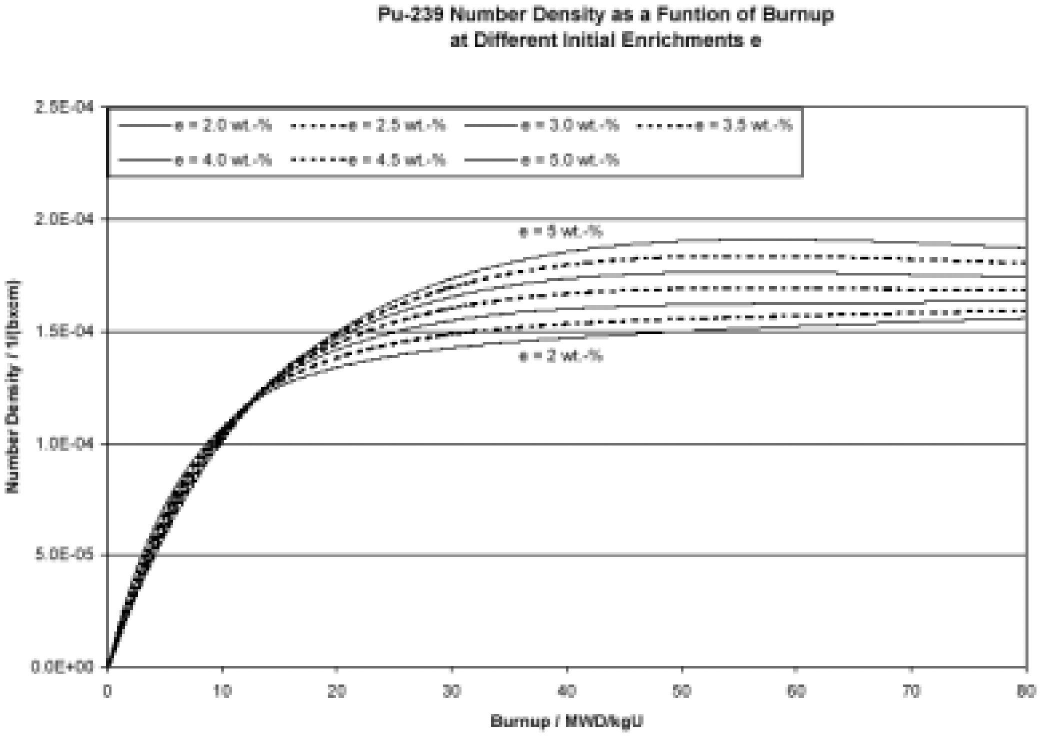

FIG. 13: Depletion of the Westinghouse 16x16-21 Standard Fuel Assembly in the Krzko Core (NEK, Slovenia): Pu-239 Number Density as a Function of Burnup and Initial Enrichment for Bounding Depletion Conditions.

4. CONCLUSION

The outcome of a burnup credit criticality safety analysis is usually given in the form of a loading curve. Such a curve indicates the minimum burnup necessary for the spent fuel with a specific initial enrichment to be placed in the spent fuel management system of interest.

For an initial enrichment of 2.0 wt.-% the minimum necessary burnup usually is zero or at least very low. Thus, for this initial enrichment it is usually not necessary to pay attention to the end effect (cf. Figure 5). Likewise, for 3 wt.-% initial enrichment the minimum necessary burnup usually is relatively low, and so it comes that the impact of the initial enrichment on the end effect usually has less effect on the loading curve. Therefore, as can be concluded from Figures 5 through 9, the impact of the initial enrichment on the end effect can be bounded by modeling the axial burnup profiles sufficiently conservatively (cp. Ref. [1]).

A very different situation is the case of axial zoning of the initial enrichment. The presence of axial blanket zones results in a significant impact on the end effect. This is due to the fact, that the lower the initial enrichment of the axial blankets is, the lower is – for a given axial shape – the reactivity of the axial blanket zones. Therefore, the lower the initial enrichment of the axial blankets is with respect to the center zone of the fuel, the lower is the end effect. This was already demonstrated in Ref. [6].

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Studies of the influence of the spatial change of the fuel burnup on criticality in WWER-440 systems

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Abstract. The vertical and horizontal change of the fuel burnup may have essential influence on the multiplication factor of storage systems. This phenomenon is examined for WWER fuel on the example of a compact storage pool. The axial and horizontal burnup profiles are taken from detailed calculations based on real operational history. The actinides only as well as actinides + fission products approach are examined. The results of the WWER burnup credit benchmark CB3+ investigating the end effect is also presented.

1. Introduction

At present, the Paks NPP, Hungary has four units of WWER-440 type. The electric power of these units is 440 MW/units. The fuel assemblies used in these units have enrichments of 3.6 %, 2.4 % and 1.6 %. However, a decision about the power increase and lifetime extension has been made recently. The power increase will require new fuel assemblies with higher enrichment and profiling and probably with gadolinium.

The introduction of the new fuel assemblies necessitates the re-examination of the criticality safety of the spent fuel storage/transport facilities. A solution for the problems due to the higher enrichment and which might arise in the future is the application of the burnup credit. The study of the implementation of the burnup credit in different systems has been started in the last years. At first, the influence of the spatial change of the fuel burnup was investigated. An essential part of this work was performed within the AER collaboration of the WWER user countries. A burnup credit benchmark series was studied in this cooperation. Two of the benchmarks were devoted to the investigation of the end effect. Beside of these benchmarks, a somewhat more detailed study for a compact storage pool was performed. The systems studied up to now are summarized bellow:

- CB3 benchmark and CB3+ benchmark addition: infinite array of fuel pins in water, no absorber, and no radial leakage. The end effect is to be investigated. The KFKI AEKI defined this benchmark. Summary of the results will be given bellow.
- CB4: conceptual CASTOR cask, no boron in plates between the assemblies, radial leakage is essential. The end effect is to be investigated. L. Markova defined this benchmark; the results up to now are summarized in [1]. The results of the AEKI are also included in this summary.
- A conceptual compact storage pool: absorber plates are present, no radial leakage (infinite array of assemblies). The influence of the axial and radial change of the burnup is investigated. Rough conservative estimation is given for the minimal value of the required average burnup to load an assembly into the pool.

2. Derivation of the burnup distribution

To be realistic, it was considered important that the burnup distributions used in these problems should be real-life distributions. This was ensured by using distributions derived from the KOLA benchmark [2] [3] [4]. This benchmark was defined for the first 12 cycles of Kolskaya Nuclear Power Plant (KOLA NPP) Unit 3. The details of the benchmark were defined in the framework of AER collaboration as core follow benchmark. Detailed operational history and reload patterns were provided. The same set of burnup distributions was used for the investigation of the three problems listed above. The fuel assemblies used in the KOLA NPP do not have not profiling or gadolinium, but this benchmark has special features useful for these studies:

- assemblies with 4.4 % enrichment
- high burnup , some assemblies have burnup more than 50 MWd/kgU
- followers with 3.6 % enrichment

A series of core calculations was performed by the KARATE core design code for the first 12 cycles of KOLA according to the specification. At the end of each cycle, the axial burnup distribution of assemblies was collected from the corresponding core calculations. The distributions used for the CB3+ and CB4 benchmark definitions were selected from this database. The compact storage pool was also investigated using these distributions.

The core calculations, the pin-by-pin calculations and the composition calculations were performed by the different modules of the KARATE-440 core design code system [5]. The calculations were made in 60-degree symmetry sector.

The isotopic compositions of the fuel pins in a node were calculated using the average values of the specific power, moderator density and boron acid concentration (87 Watt/cm3, 557 Kelvin degree and 3.0 g/kg). The shutdown time between cycles was neglected, i.e. only the cooling time after the last cycle was considered.

3. The CB3+ burnup credit benchmark

For the investigation of the end effect in the case WWER fuel, the CB3 benchmark was specified in 1998 [6]. In this benchmark the effect of the axial burnup profile on criticality was investigated in a laterally infinite array of spent fuel rods with initial enrichment of 3.6 %. The multiplication factor of this array had to be calculated using burnup distributions with 10, 30 and 40 MWd/kgU average burnup, assuming 1 and 5 years cooling time and different sets of isotopes. The isotopic compositions of the spent fuel were given. However, because of the selection of the burnup profiles, the sign of ρ was negative for all of the selected cases. To demonstrate the existence of positive end effect for WWER fuel, a benchmark addition called CB3+ was defined in 2000. In this definition only the burnup distributions were replaced with other ones, and all the other details (arrangement, composition of the structural elements, etc.) were kept the same as it was in CB3 definition. Burnup distributions for 4.4 % initial enrichment were selected from the database derived from the KOLA benchmark described above. The detailed specification of the CB3+ is given in [7].

The end effect should be investigated for 5 cases using the selected distributions according to Table I.

The actinides which should be taken into account are the U-235, U-236, U-238, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-243, the corresponding fission products are the Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Nd-143, Nd-145, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Eu-153, Gd-155.

Table I. Cases to be investigated in CB3+

average burnup (MWd/kgU)	cooling time (years)	selected isotopes
10	1	actinides+fission products
30	1	actinides+fission products
40	1	actinides+fission products
40	5	actinides+fission products
40	5	actinides only

Six sets of results were calculated by R.L. Bowden and G.J. O'Connor (Nuclear Technologies plc, United Kingdom and Department for Transport, Local Government and Regions, United Kingdom), V. Chrapciak (VUJE, Trnava, Slovakia), L. Markova (NRI, Rez, Czech Republic), J. Svarny (SKODA, Plzen, Czech Republic), Candan Töre (SEA, Madrid, Spain) and G. Hordósy (AEKI, Budapest, Hungary).

The calculated values of the multiplication factor together with their individual standard deviations are summarized in Table III. In the last two columns of the Table the average multiplication factor and its calculated standard deviation σr is shown. σr is defined as

$$\sigma_r = \sqrt{\sum_{i=1}^{n} (k_i - k_{av})^2 / (n-1)}$$

The investigated case is specified in the first column of the Table in the form of burnup/cooling time/axial distribution. For distribution, u means the uniform, d means the specified axial distribution. NF means that the fission products were not considered, in other cases actinides and fission products were taken into account.

The end effects calculated from the above keff values are given in percent in Table IV.

As it can be seen on Table II, three solutions from all the six were gained by the same code and the same libraries (Markova, Svarny and Hordósy by the MCNP4B and by the identical combination of ENDF/B-VI and ENDF/B-V). Another solution was gained by the same code and a slightly different combination of libraries (Töre). Because of the small number of solutions using different codes/libraries there was no sense to perform a statistical evaluation, but the main tendencies can be seen directly from the Table III and IV. The agreement between the participants is fairly good. The average decrease of the multiplication factor is roughly 0.06 - 0.08 per 10 MWd/kgU burnup calculated by uniform burnup profiles. Some of the tendencies found for PWR fuel can be observed in this case. The end effect is negative at low burnup and positive at high burnup if the fission products are included into the analysis.

Table II. Calculated KEFF values for the CB3+ burnup credit benchmark

	R.L. Bowden, G.J. O'Connor, Nuclear Technologies, Warrington, United Kingdom, MONK8A, JEF2.2, 13193 Point Energy Library		V. Chrapciak, VUJE, Trnava, Slovakia KENO VI, 44GROUPNDF5		G. Hordósy, AEKI, Budapest, Hungary MCNP4B ENDF/B-VI, ENDF/B-V *		L. Markova, NRI, Rez, Czech Republic MCNP4B ENDF/B-VI, ENDF/B-V *		J. Svarny, Skoda, Plzen, Czech Republic MCNP4B ENDF/B-VI, ENDF/B-V *		Candan Töre, SEA, Madrid, Spain MCNP4B ENDFB/VI, ENDFB/V **		average	
B/T/dist	keff	σ(×10-3)	keff	σ(×10-3)	keff	σ(×10-3)	keff	σ(×10-3)	keff	σ(×10-3)	keff	σ(×10-3)	keff	σ_r(×10-3)
10/1/u	1.3111	1.1	1.3085	1.18	1.3069	0.47	1.3054	0.52	1.3052	0.25	1.3059	0.22	1.3072	2.2
10/1/d	1.3040	1.1	1.2981	1.10	1.2989	0.48	1.2972	0.51	1.2969	0.24	1.2974	0.22	1.2988	2.7
30/1/u	1.1436	1.1	1.1380	1.11	1.1388	0.76	1.1388	0.53	1.1380	0.25	1.1376	0.23	1.1391	2.2
30/1/d	1.1358	1.1	1.1314	1.07	1.1363	0.47	1.1353	0.53	1.1362	0.25	1.1368	0.22	1.1353	1.9
40/1/u	1.0695	1.1	1.0685	1.08	1.0665	0.48	1.0667	0.53	1.0667	0.29	1.0656	0.24	1.0673	1.5
40/1/d	1.0781	1.1	1.0776	1.45	1.0796	0.45	1.0778	0.55	1.0780	0.26	1.0795	0.23	1.0784	0.9
40/5/u	1.0400	1.1	1.0351	1.21	1.0355	0.50	1.0352	0.50	1.0351	0.24	1.0347	0.22	1.0359	2.0
40/5/d	1.0541	1.1	1.0521	1.78	1.0567	0.46	1.0557	0.56	1.0555	0.26	1.0572	0.23	1.0552	1.9
40/5/u/NF	1.1700	1.1	-	-	1.1635	0.48	1.1633	0.53	1.1639	0.38	1.1626	0.22	1.1647	3.0
40/5/d/NF	1.1658	1.1	-	-	1.1655	0.42	1.1642	0.52	1.1638	0.38	1.1652	0.22	1.1649	0.8

* Actinides, Ag-109, Eu-153, Gd-155, Mn-155, oxygen and hydrogen were taken from ENDF/B-VI, all other isotopes from ENDF/B-V

** N-237, Pu-239 and oxygen were taken from ENDF/B-VI, all other isotopes from ENDF/B-V

Table III. Calculated values of the end effect in percent

	Bowden O'Connor Nucl. Tech. United Kingdom	Chrapciak VUJE Slovakia	Hordósy AEKI Hungary	Markova NRI Czech Republic	Svarny SKODA Czech Republic	Töre SEA Spain	ave-rage
10/1 A+F	-0.54	-0.79	-0.61	-0.63	-0.64	-0.65	-0.64
30/1 A+F	-0.68	-0.58	-0.22	-0.31	-0.16	-0.07	-0.34
40/1 A+F	0.80	0.85	1.23	1.04	1.06	1.30	1.05
40/5 A+F	1.36	1.64	2.05	1.98	1.97	2.17	1.86
40/5 NF	-0.36	-	0.17	0.08	-0.01	0.22	0.02

The increasing of the cooling time increases the end effect, while decreases the multiplication factor. This latter effect can be explained by the decay of the fissile isotope Pu-241 into the absorber Am-241 and the build-up of absorber Gd-155 from the decay of Eu-155. This behavior of the end effect is close to that of observed at the western type PWR

However, when fission products are excluded at high burnup, there is no definite answer, weather the end effect is positive or it is negative. The individual results have different sign and quite small absolute value. The most likely explanation is that the end effect in that case has a small absolute value, and its sign is masked by the statistical uncertainties of the calculations. This is unlike the case of burnup credit with PWR fuel, where a definite positive end effect was observed for most cases.

4. Investigation of a compact storage pool

The criticality of a conceptual compact storage pool was also examined by the burnup distribution derived from the KOLA benchmark. The characteristics of this pool are close to a real WWER compact storage pool. Because of the large sizes of the pool, the radial leakage has minor importance in such a cases, and the pool was approximated by laterally infinite array of hexagonal units, consisting of a fuel assembly, the surrounding water and boron steel. The schematic cross section of such a unit is shown on the next picture.

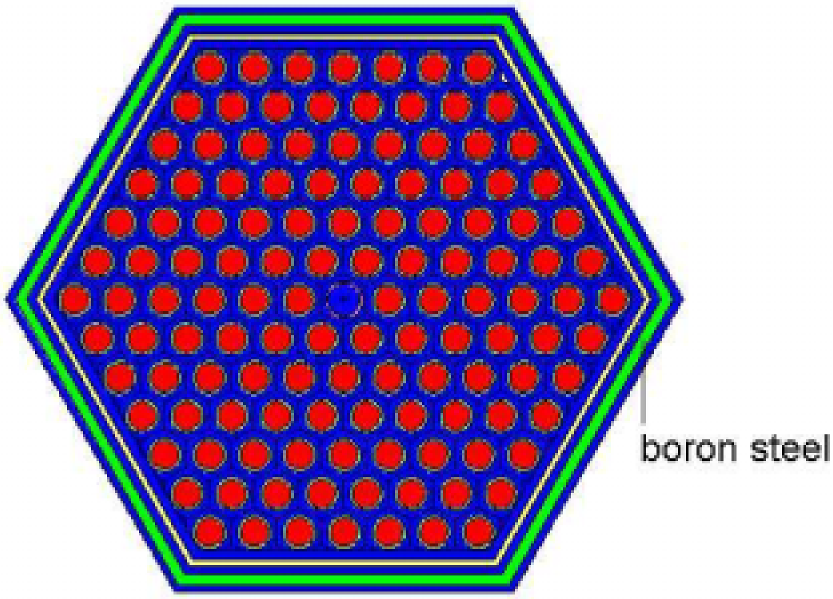

FIG. 1. The cross section of a unit of the compact storage pool.

The aim of this investigation was to get some estimation of the minimum value of the average assembly burnup required to meet the criticality safety criteria for the pool loaded by such assemblies.

This minimum required value of the average burnup could be determined from the well-known criticality safety condition:

and horizontal change of the burnup was investigated.

$$ks\ (B) + \Delta ks \leq 1 - \Delta kc - \Delta kt - \Delta kB - \Delta kI - \Delta km - \Delta ka,$$

where

k_s	is the calculated multiplication factor
Δk_s	is the statistical uncertainty of the calculation
Δk_c	is the uncertainty from nuclear data and validation
Δk_t	is the uncertainty from the technological data
Δk_m	is the uncertainty of the modeling in core design calculation
Δk_I	is the uncertainty from the isotopic calculation
Δk_a	is the administrative safety margin (usually 0.05)
Δk_B	is the uncertainty from using uniform burnup in the criticality analysis instead of spatially changing burnup.

In this analysis the uncertainty of the modeling in core design calculation and the uncertainty from the isotopic calculation was not considered. The influence of the axial and horizontal change of the burnup was investigated.

To examine the end effect, the multiplication factor of the pool was calculated by approximately 200 burnup distributions for 4.4 % and 3.6 % assemblies.

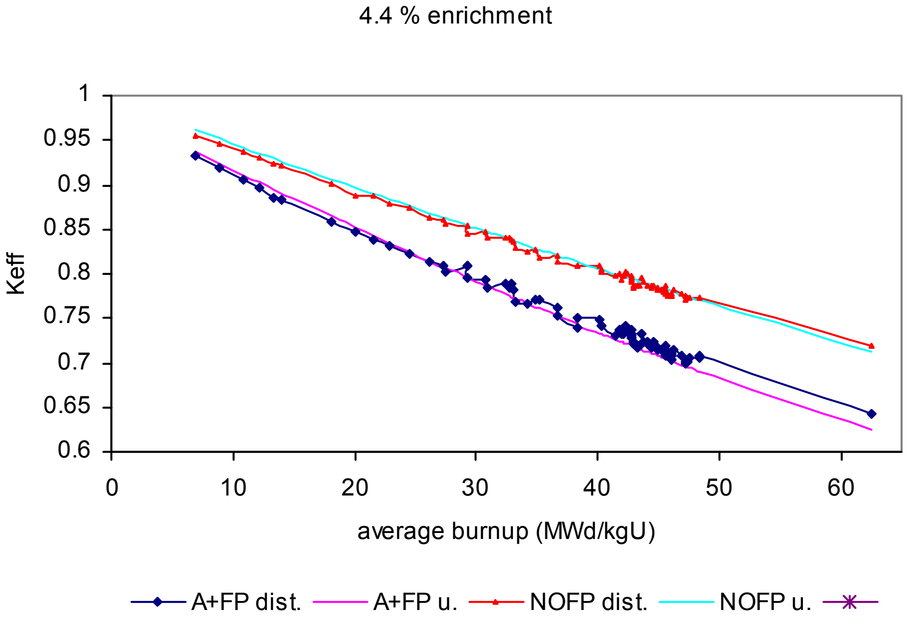
4.4 % enrichment

FIG. 2. Multiplication factor calculated by the uniform burnup distributions and by the actual burnup distributions for the 3.6 % assemblies.

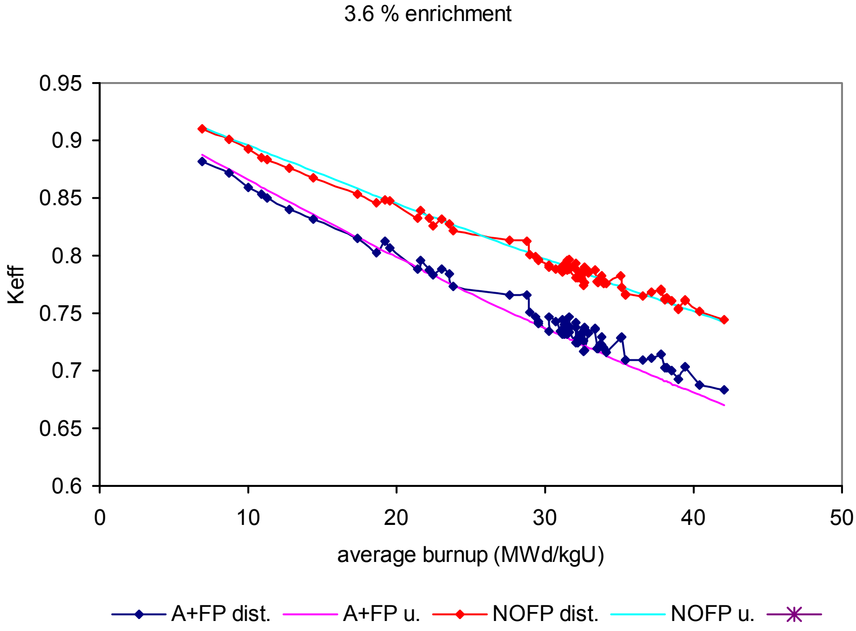

FIG. 3. Multiplication factor calculated by the uniform burnup distributions and by the actual burnup distributions for the 4.4 % assemblies

The general picture that can be seen from the results can be summarized as:

- if fission products are present, the end effect negative at low burnup and positive at high burnup;
- the transition from negative to positive end effect is at about 20 MWd/kgU for 3.6 % enrichment and 28 MWd/kgU for 4.4 % enrichment;
- without fission products the end effect is essentially less.

The maximal end effect found approximately the same for 3.6 % and 4.4 % enrichment (about 3% with fission products and about 1% without fission products). This is different from the observations found in the case of PWR reactors.

Horizontal distribution:

The radial change of the burnup is essential for assemblies near core boundary. However, if such an assembly is positioned into the inner part of the core after a reloading, this change is smoothed. So we examined such assemblies, which were at the core periphery just before their removal from the core. Approximately 20 cases were investigated (all possible cases with 4.4 % or 3.6 % enrichment). The pin-by-pin burnup distribution was calculated for the 126 pins of each assembly by the SADR1 module of KARATE.

SADR1 is a fine-mesh diffusion code with coupled thermohydraulics. It is coupled to the global calculations, in a fine-mesh calculation the surrounding assemblies are taken into account. It utilizes position dependent group constants and the reflector is represented by albedo conditions.

The multiplication factor of the pool was calculated for the cases described above using the average burnup and the pin-by-pin burnup distribution. The difference was within the statistical error for all of the cases.

Minimal required average burnup:

Neglecting the modeling and isotopic uncertainties and using the most conservative axial burnup distribution, a rough estimation of the Bav which satisfy the safety criteria can be determined in a conservative way from usual criticality safety condition cited above. (The influence of the horizontal change is neglected in accordance with the previous observations.) With some rearrangement it can be written in the form

ks (Bav) $(1+\Delta) + \Delta ks \leq 1 - \Delta kc - \Delta kt - \Delta ka = U$.

From the analysis of the results calculated by the different axial distributions the value of Δ can be determined. The resulting values are $\Delta=0.035$ with fission products and $\Delta=0.015$ without fission products for the 4.4 % assemblies. Using these values and the smooth keff (Bav) curve the minimal values of the average burnup required for loading to fuel assembly into the pool are given in the following Table.

Table IV. Minimum required average burnup for 4.4 % assemblies

A+FP	13 MWd/kgU
NOFP	17 MWd/kgU

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Evaluation of horizontal burnup profile for WWER-440 fuel assembly

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Abstract. This paper describes the impact of horizontal burnup profile on the criticality of spent fuel assembly. All calculations were made for WWER-440 fuel assembly with enrichment 3.6w% of U^{235} and burnup up to 60 MWd/kgU.

1. INTRODUCTION

The WWER-440 fuel assembly (Fig.1) is a hexagonal assembly containing 126 fuel pins (in a triangular lattice) and central instrumental tube its symmetry is 30^o. The water volume surrounding each pin is different for pins by central tube, by wall, at corner and inside assembly. This gives different neutron spectrum and different inventory of spent fuel. Analyses of criticality for different burnup profiles are presented in this paper.

2. USED CODES AND LIBRARIES, GEOMETRY MODEL

The inventory for each pin and inventory averaged over all pins were prepared with the WIMS8 code, which is designed for neutronic calculations. The code is supplied with the latest mostly JEF2.2 based validated nuclear data library primarily in 172 group structure condensed into 69 groups structure for standard calculations. Code has an open structure, which comprises a set of methods linked together to form a calculation scheme that can be used to solve most problems associated with Thermal reactor physics. The main advantage for requested purpose is a possibility of exact 2D flux solution in a fuel assembly and calculation of the change of nuclide inventory due to irradiation of materials. It has to be noted that code is not designed for calculation of fuel inventory for long period after irradiation in core. Therefore, the number of isotopes, which are taken into account, is restricted; some reactions are neglected or are substituted. It means that only isotopes important from the point of reactivity calculation are treated fully. This explains some obvious differences comparing with results with codes designed for calculation of isotope inventory after long period after irradiation (WIMS code could be followed with code FISPIN, but these calculations have not be performed).

The criticality calculations were carried out with the SCALE 4.4 system, the KENO VI module and 44GROUPNDF5 library has been used.

The calculations area is an infinitive array of fuel assemblies (= one assembly and mirror boundary condition) in triangular lattice with pitch 14.7 (configuration as in core). The initial enrichment is homogeneous (all pin of 3.6%). Current average assembly burnup is approximately 40 MWd/kgU.

The list of nuclides is according [3].

Results

The inventory was calculated up to average burnup 60 MWd/kgU (by step 3 MWd/kgU). Each pin has its own burnup and inventory. Differences in burnup in pins for average assembly burnup 39 and 60 MWd/kgU are shown in Fig.2. Maximal deviation is in pin at corner.

Maximal differences (average assembly burnup is 39 MWd/kgU) are in Table I:

a) between concentrations in pins for the same burnup (Bp = 39 MWd/kg, prepared by interpolation) and the average concentration. This represents only influence of position pin in assembly (= neutron spectrum).
b) between concentration for real burnup and the average concentration. This represents influence of position and burnup.

Differences in concentration of U235, Pu239 and Gd155 in each pin (case a) and b)) for average burnup assembly 39 MWd/kgU are shown in Fig. 3 – 5.

Concentration of U235, Pu239 and Gd155 in average pin, in pin 16 (at corner, maximal burnup) and in pin 2 (inside, minimal burnup).are shown in Fig. 6 – 8.

Results (Kefprof, Kefflat, and ΔKef = Kefprof – Kefflat) are shown in Table II. They correspond with analyses for western PWR showing that the impact of horizontal burnup profile is very small. For WWER-440 fuel ΔKef is in interval 2 – 3 □$_{\text{Monte Carlo}}$.

Conclusion

The impact of horizontal burnup for homogeneously initially enriched WWER-440 assembly is very small (less than 0.0025 Δk). This result is in good agreement with results for western PWR.

New advanced fuel for WWER-440 has a radial profile of initial enrichment (4.0w%, 3.6w% and 3.3w%). The analyses for this type of fuel will be carried out later.

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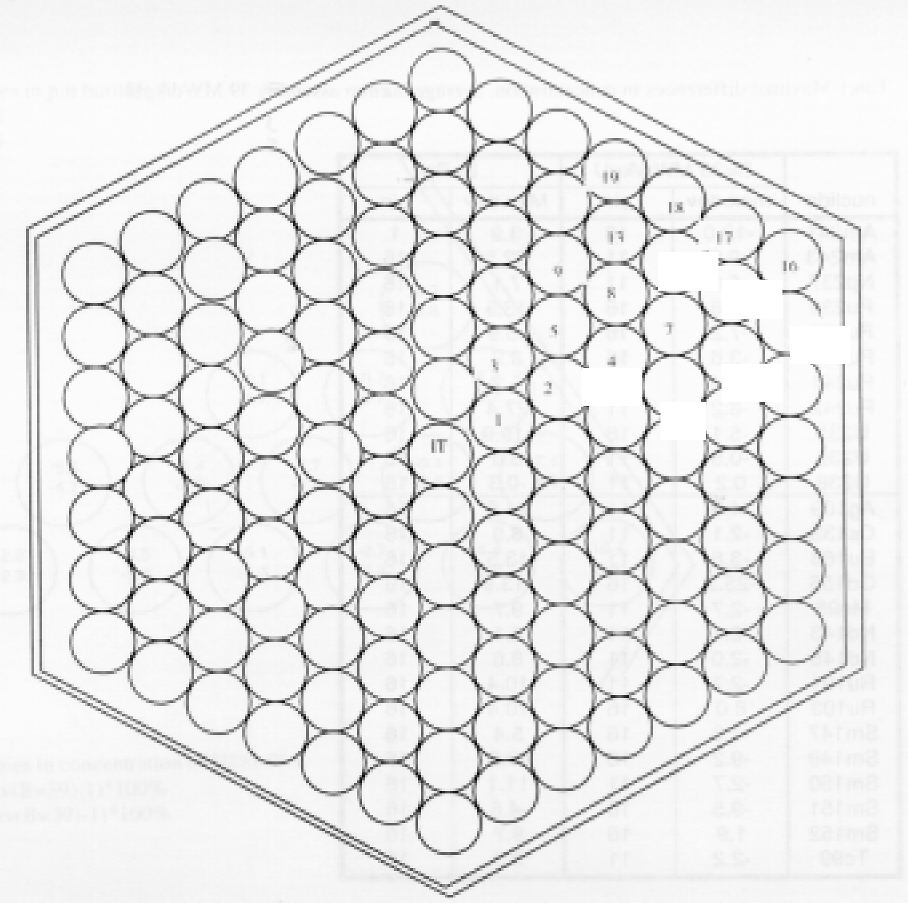
19
18
17
16
9
8
7
5
4
3
2
1
1T

FIG. 1. WIMS's numbering of pins (symmetry 30o).

Table. I. Maximal differences in concentration, average burnup assembly 39 MWD/KGU

nuclide	B = 39 MWd/kgU		B = Bp	
	max.dev	pin	Max.dev	pin
Am241	-12.0	16	3.9	1
Am243	-9.9	11	32.5	16
Np237	-5.0	11	7.1	16
Pu238	-9.8	16	13.3	16
Pu239	-7.2	16	-5.9	16
Pu240	-3.6	16	3.9	16
Pu241	-5.4	11	5.2	16
Pu242	-6.2	11	27.4	16
U235	-5.1	16	-19.9	16
U236	-0.5	11	3.0	16
U238	0.2	11	-0.3	16
Ag109	-4.6	11	14.9	16
Cs133	-2.1	11	8.9	16
Eu163	-3.8	11	13.5	16
Gd155	-25.3	16	-13.9	16
Mo95	-2.7	11	9.7	16
Nd143	-2.1	11	4.1	16
Nd145	-2.0	11	8.6	16
Ru101	-2.7	11	10.4	16
Ru103	8.0	16	10.4	16
Sm147	-4.3	16	5.4	16
Sm149	-9.2	16	-9.5	16
Sm150	-2.7	11	11.1	16
Sm151	-9.5	16	-4.6	16
Sm152	1.9	16	9.7	16
Tc99	-2.2	11	9.3	16

Table. II. KEF

av.bur.	profile		flat		Δkef = Kef$_{prof}$ - Kef$_{flat}$
MWd/kgU	Kef$_{prof}$.	σMonte Carlo	Kefflat	σMonte Carlo	
0	1.39684	0.00066	1.39684	0.00066	0
15	1.24532	0.00057	1.24782	0.00069	-0.00250
30	1.14148	0.00063	1.14309	0.00066	-0.00161
39	1.07640	0.00060	1.07761	0.00059	-0.00121
45	1.05989	0.00061	1.06157	0.00058	-0.00181
60	1.00142	0.00054	1.00347	0.00059	-0.00205

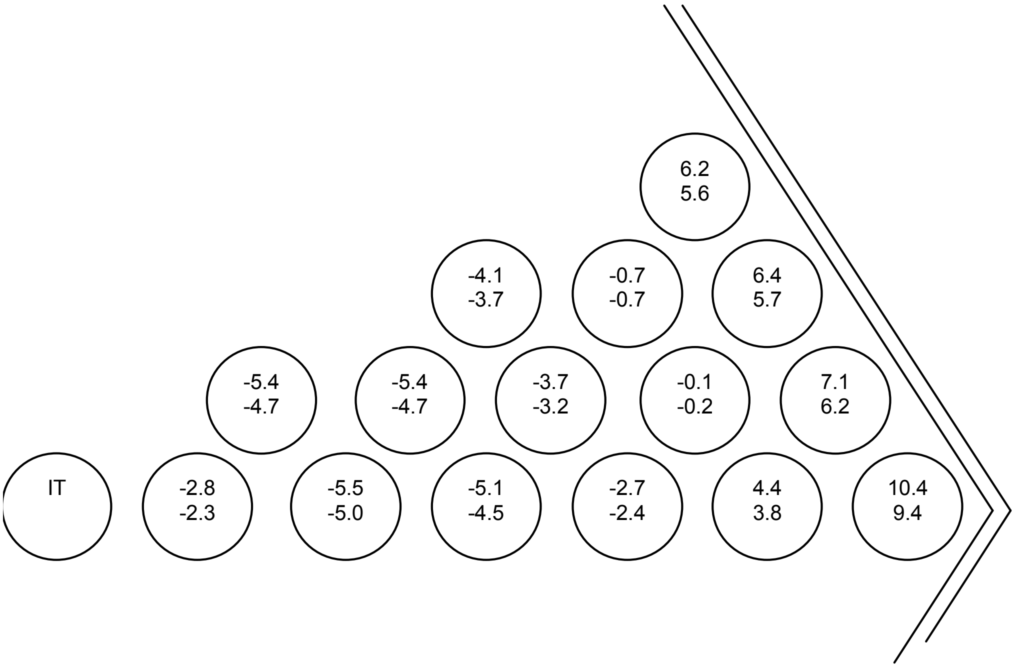
6.2
5.6
-4.1
-3.7
-0.7
-0.7
6.4
5.7
-5.4
-4.7
-5.4
-4.7
-3.7
-3.2
-0.1
-0.2
7.1
6.2
IT
-2.8
-2.3
-5.5
-5.0
-5.1
-4.5
-2.7
-2.4
4.4
3.8
10.4
9.4

FIG. 2. Differences in pin burnup
(Bp/39-1)*100%
(Bp/60-1)*100%

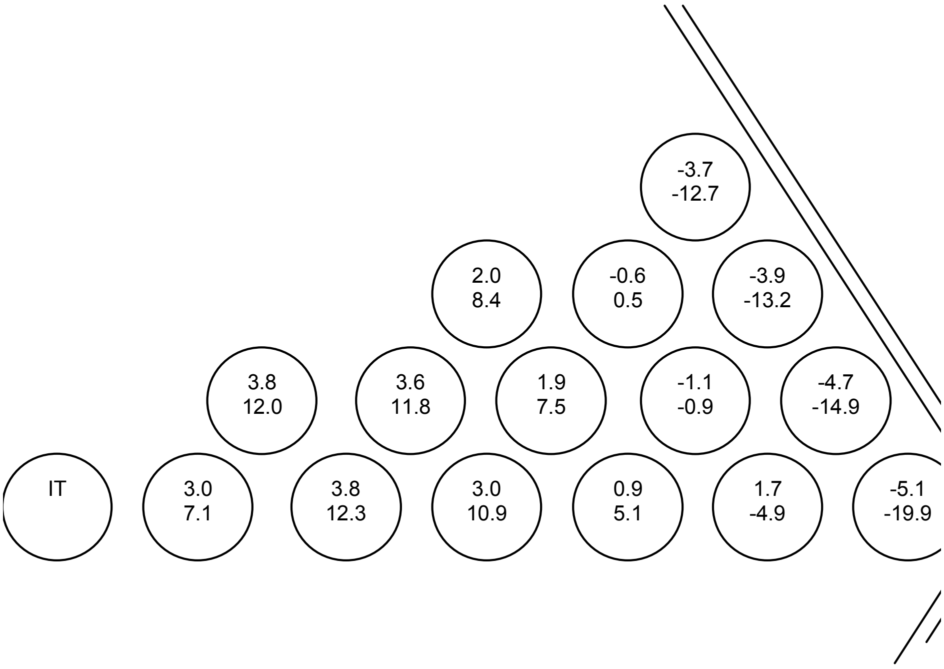
-3.7
-12.7
2.0
8.4
-0.6
0.5
-3.9
-13.2
3.8
12.0
3.6
11.8
1.9
7.5
-1.1
-0.9
-4.7
-14.9
IT
3.0
7.1
3.8
12.3
3.0
10.9
0.9
5.1
1.7
-4.9
-5.1
-19.9

FIG. 3. Differences in concentration of U235 (%)
(Cp(B = 39)/(Cav(B = 39)-1)*100%
(Cp(B = Bp)/(Cav(B = 39)-1)*100%

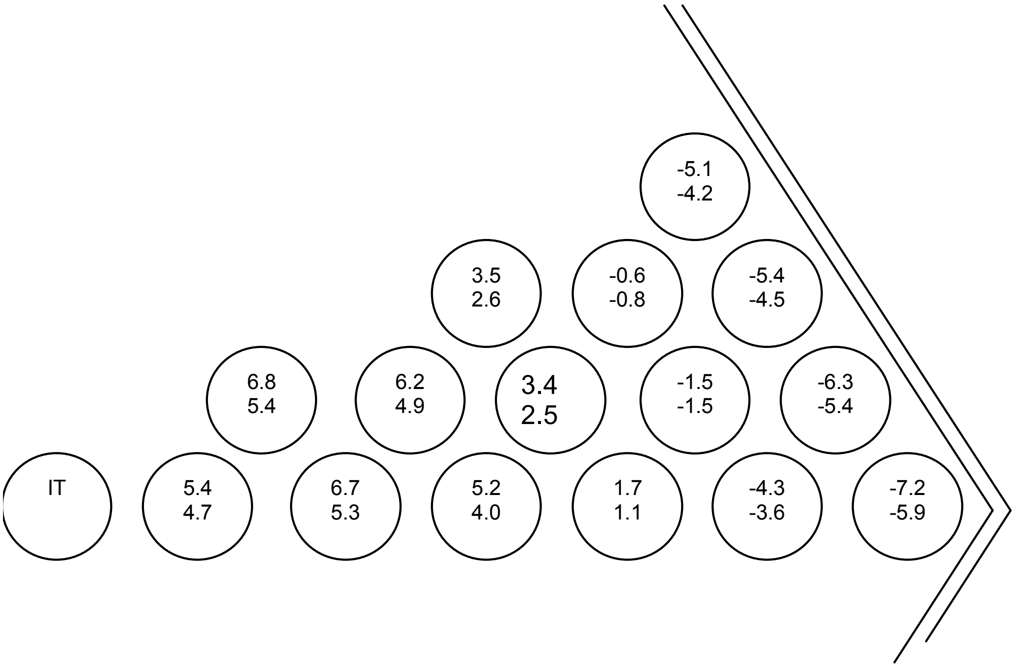
-5.1
-4.2
3.5
2.6
-0.6
-0.8
-5.4
-4.5
6.8
5.4
6.2
4.9
3.4
2.5
-1.5
-1.5
-6.3
-5.4
IT
5.4
4.7
6.7
5.3
5.2
4.0
1.7
1.1
-4.3
-3.6
-7.2
-5.9

FIG. 4. Differences in concentration of Pu239 (%)
(Cp(B = 39)/(Cav(B = 39)-1)*100%
(Cp(B = Bp)/(Cav(B = 39)-1)*100%

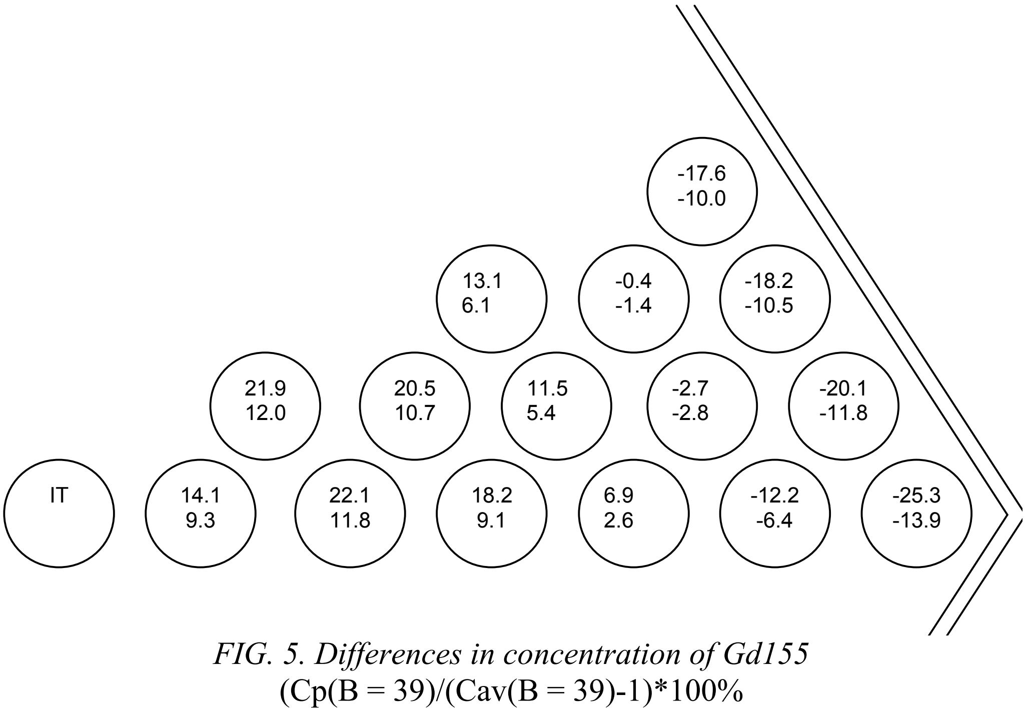
-17.6
-10.0
13.1
6.1
-0.4
-1.4
-18.2
-10.5
21.9
12.0
20.5
10.7
11.5
5.4
-2.7
-2.8
-20.1
-11.8
IT
14.1
9.3
22.1
11.8
18.2
9.1
6.9
2.6
-12.2
-6.4
-25.3
-13.9

FIG. 5. Differences in concentration of Gd155
(Cp(B = 39)/(Cav(B = 39)-1)*100%
(Cp(B = Bp)/(Cav(B = 39)-1)*100%

Burnup credit methodology in the NPP Krzko spent fuel pool reracking project

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Abstract. NPP Krzko is going to increase the capacity of the spent fuel storage pool by replacement of the existing racks with high-density racks. The design, rack manufacturing and installation has been awarded to the Framatome ANP GmbH. Burnup credit methodology, which has been already approved by the Slovenian Nuclear Safety Administration in previous licensing of existing racks, will be again implemented in the licensing process with the recent methodology improvements. Specific steps of the criticality analysis and representative results are presented in the paper showing also the current national practice of the burnup credit implementation.

1. INTRODUCTION

NPP Krzko is a Westinghouse 2-loop PWR plant operating successfully since 1981. The reactor operates with 121 fuel assemblies of a 16x16 type. The plant is currently going through the process of increasing the capacity of the spent fuel storage pool by replacement of the existing racks with high-density racks. This will be the second reracking campaign since 1983 when storage has increased from 180 to 828 storage locations.

The existing racks have been licensed based on analysis performed with the NRC approved Westinghouse methodology. Since the racks exhibit relative large cell center-to-center distance (30.48 cm x 29.64 cm), only a small burnup credit of 4 MWd/kgU is needed to cover the highest enriched allowable fuel (5%). The basic analysis has been performed in a conservative 2-D approximation neglecting axial and radial burnup variations.

The use of borated stainless steel as an additional neutron absorber enables highly compacted storage racks (Fig. 1). With these racks the pool capacity is going to increase in two steps from 828 to expected final 2321 locations, which should be sufficient for the fuel storage during entire plant lifetime. The design, rack manufacturing and installation have been awarded to the Framatome ANP GmbH.

The criticality analysis [1] has demonstrated that evaluated multiplication factor of a fuel storage is less then 0.95 including all uncertainties. No soluble Boron credit has been applied. However, the presence of soluble Boron in the storage pool water was permitted in the analysis of accident conditions.

Application of burnup credit requires calculation of the isotopic inventory of the irradiated fuel for which burnup credit is taken. The depletion calculation simulates the burnup of the fuel under reactor operating conditions. The result of the depletion analysis is the predicted isotopic composition of the discharged spent fuel. This composition is a necessary input to the criticality analysis, which was performed with SCALE-4.4 and 44-group library [2].

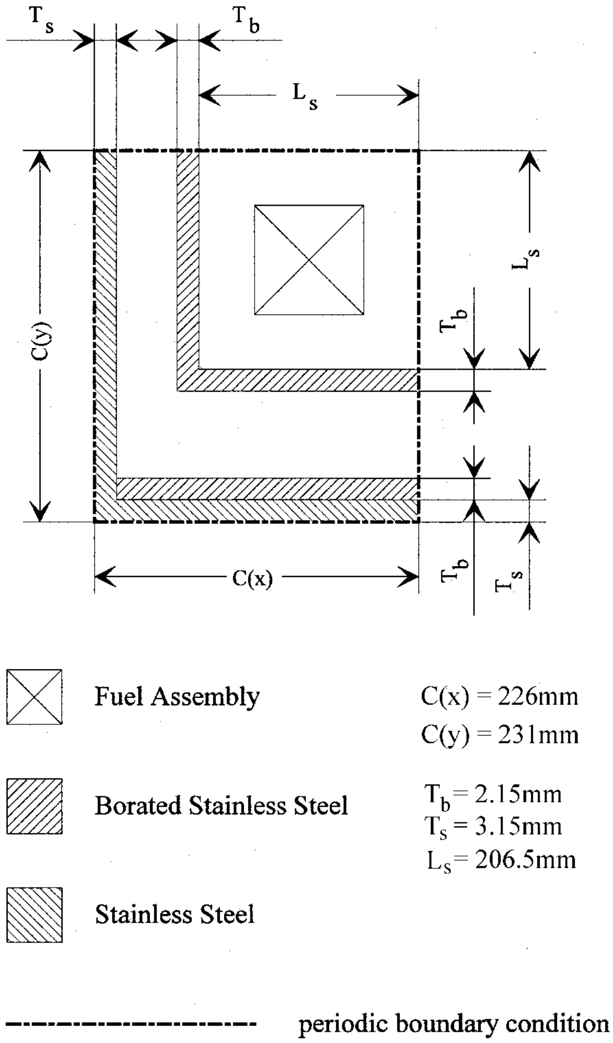
T
s
T
b
L
s
C(y)
L
s
T
b
C(x)
T
b
T
s
Fuel Assembly
C(x) = 226mm
C(y) = 231mm
Borated Stainless Steel
T
b
= 2.15mm
T
S
= 3.15mm
L
S
= 206.5mm
Stainless Steel
periodic boundary condition

FIG. 1. Storage cell of the new rack.

2. NUCLIDES USED IN THE ANALYSIS

In the depletion analysis actinide plus fission products approach has been used. Only a few fission products which are conservative with respect to cooling time but have also significant neutron absorption properties are taken into account – ^{103}Rh, ^{109}Ag, ^{133}Cs, ^{135}Cs, ^{143}Nd, ^{145}Nd, ^{147}Pm, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{151}Sm, ^{152}Sm, ^{153}Eu and ^{155}Eu. No cooling time credit has been used. The accumulation of ^{239}Pu is conservatively covered by increasing the concentration of this nuclide obtained at time of reactor shutdown by 5%.

The depletion code CASMO-3G [3] has been applied to determine the isotopic densities. The code has been validated against several assay measurements. It was demonstrated that for the NPP Krzko case, the uncertainty in isotopic composition could result in underestimation of k_{eff}, but not more then 0.012 for the fuel depleted up to 50 MWd/kgU.

3. PARAMETERS FOR DEPLETION ANALYSIS

In general fuel assemblies are irradiated under a variety of reactor operating conditions (temperature, Boron concentration, etc.). In the depletion calculation assumptions that encompass the known variations must be employed to ensure that the nuclide content of the fuel is conservatively represented. A due attention has to be given specially to the parameters, which contribute to the spectral hardening in the reactor core. Spectral hardening results in an increased production rate of Plutonium from increased fast neutron capture in ^{238}U. Consequently increased Plutonium fission increases the reactivity of discharged fuel.

There is no need to take all of operational parameters in an extreme manner. Operational extreme in one parameter may result in an opposite extreme for a coupled parameter. However, simultaneous use of realistic bounding parameter values in a depletion model provides acceptable approach, since it is unlikely that any fuel would be depleted under all such conditions simultaneously.

Fuel temperature

Average pellet temperature obtained at nominal specific power has been averaged over the permitted fuel burnup range.

Moderator temperature

Core average outlet temperature at nominal power has been considered in the calculations.

Soluble Boron

Cycle average values have been determined from the Boron letdown curves. All cycles of actual reactor operation together with some anticipated future cycles has been taken into account. Maximal obtained value has been conservatively increased by 200 ppm.

Specific power

Core average specific power has been taken.

Fixed burnable poisons

It was assumed that the standard fuel is equipped with the maximum number of the BPR rods during irradiation.

Integral burnable poisons

In the case of Vantage fuel the maximum number of IFBA rods has been considered. However, in the evaluation of the loading curve the credit for the Boron in IFBA rods has not been taken into account.

4. AXIAL BURNUP PROFILES

An extensive database of actual axial burnup shapes has been prepared to properly evaluate the end effects. Burnup shapes are derived from monthly in-core power measurements covering 17 cycles of plant operation [4]. They are provided for the standard fuel and fuel with axial blankets (Vantage fuel) on 60 equidistant axial regions. Each processed burnup profile has been enveloped in conservative manner [5] (see also Fig. 2) and evaluated according to the procedure described in [5], [6]. Actual average burnup of the fuel assembly is correlated to the equivalent uniform burnup. A typical dependence is shown in Fig. 3. All obtained results are bounded with the "correlation curve" which represents the axial end effects.

5. HORIZONTAL BURNUP PROFILES

Since there exists no measurement data specific to the NPP Krzko fuel, conservative generic data have been used. For this purpose a linear model is established for the horizontal burnup tilt which conservatively covers the maximum assembly quadrant deviations from the fuel assembly average burnup determined in [7]. The neutron multiplication factors obtained for the horizontal burnup profiles were evaluated using the same methods described in previous section. The effect was covered by an additional penalty of 0.5 MWd/kgU over the entire burnup range.

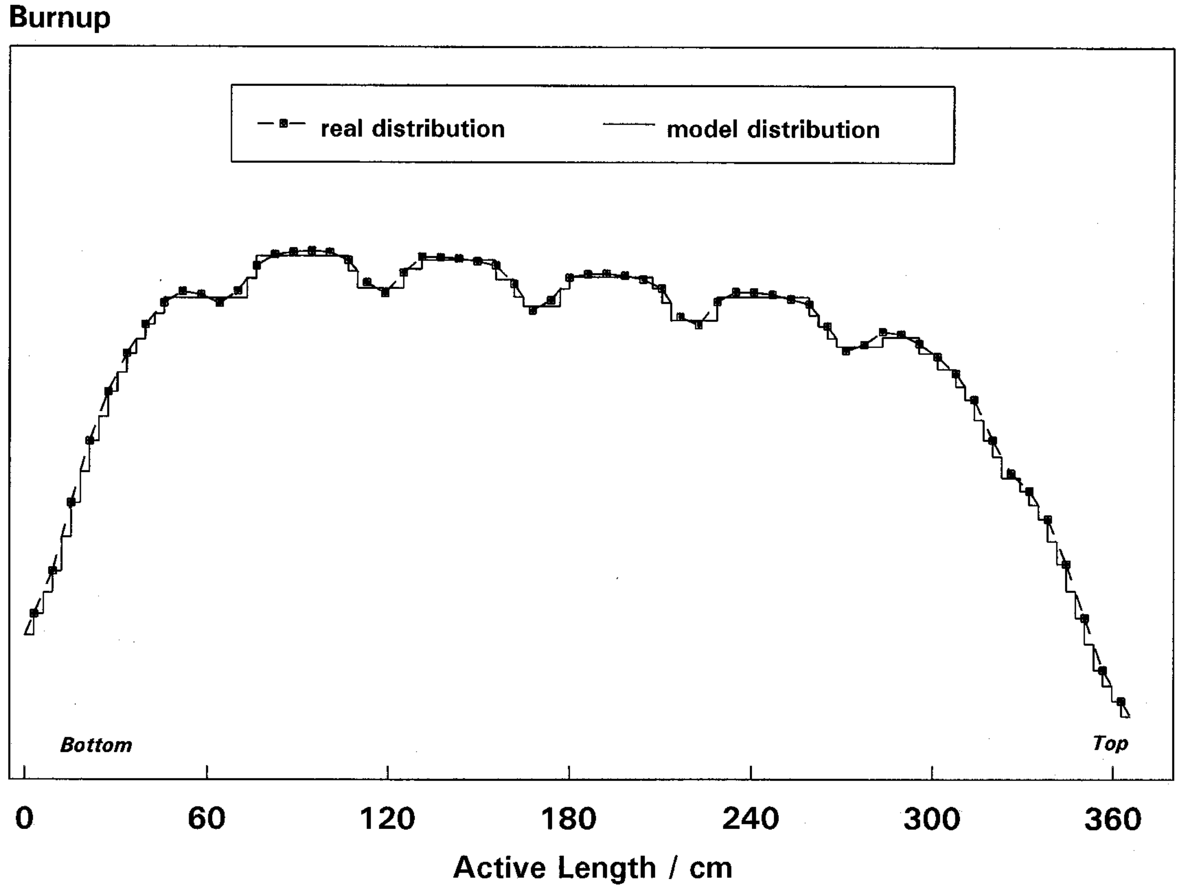

FIG. 2. Modeling of axial burnup distribution.

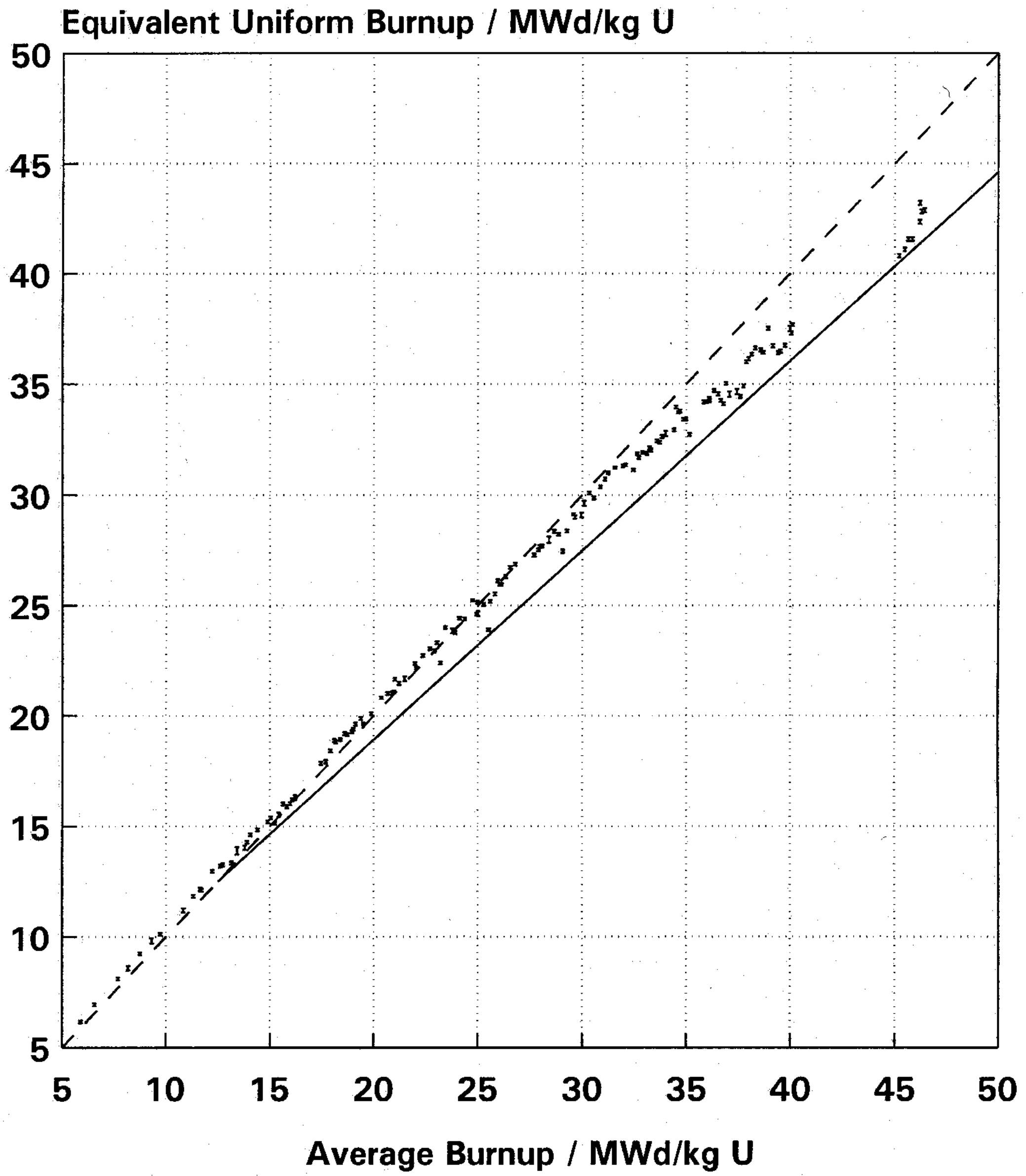

FIG. 3. Correlation of equivalent uniform burnup to average burnup.

6. LOADING CURVE

A series of calculations has been performed to determine correlation between average burnup and initial enrichment of the fuel assembly yielding the neutron multiplication of the rack to its maximum permissible value. The final result of all previously presented calculations is a loading curve, which provides the minimum required discharged burnup as a function of the initial enrichment of the fuel assembly. The loading curves for the new and old racks are presented on Fig. 4.

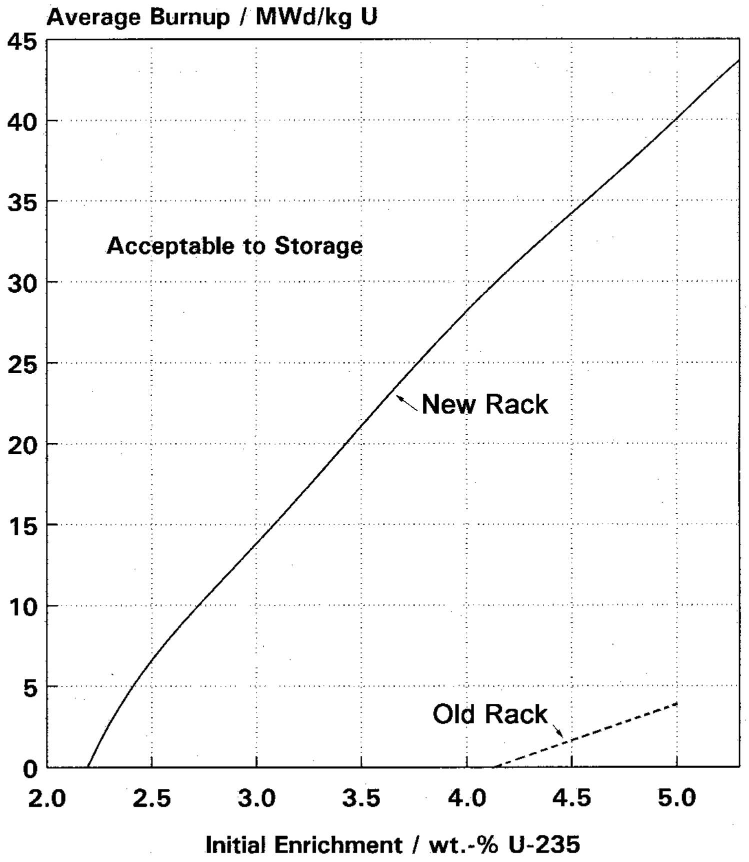

FIG. 4. Loading curves for the new and old storage racks.

7. CONCLUSION

NPP Krzko is going to increase the capacity of the spent fuel storage pool by replacement of the existing racks with high-density racks. The pool capacity will increase in two steps from 828 to expected final 2321 locations, which should be sufficient for the entire plant lifetime. The design, rack manufacturing and installation has been awarded to the Framatome ANP GmbH. Burnup credit methodology, which has been already approved by the Slovenian Nuclear Safety Administration in previous licensing of old racks, is again implemented in the licensing process with the recent methodology improvements presented in the paper.

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WWER fuel rod isotopics by MONTEBURNS 1.0 — Influence on the multiplication factor and comparison with the CB3 benchmark data

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Abstract. Burnup credit has taken a great importance during the last years. Hence, a special effort for improving the reliability of current burnup codes and cross-section libraries has been necessary in order to simulate as accurately as possible the behavior of the irradiated fuel in nuclear power plants and/or storage and transport facilities.

The aim of the work performed at SEA is to compare the new code MONTEBURNS, developed in Los Alamos Laboratory, with ORIGEN-S for the WWER fuel rod isotopics calculation in an infinite array. MONTEBURNS links the Monte Carlo transport code MCNP with the radioactive decay and burnup code ORIGEN-2.1. The results obtained show that MONTEBURNS may become in the future a very important tool for isotopic studies, though its accuracy depends on the quality of the cross-sections used.

OBJECTIVES

Due to the importance, which the burnup credit has taken during the last years, it is necessary to evaluate and validate new codes, which simulate accurately the behaviour of the irradiated fuel.

The aim of the work performed at SEA is to compare the new code MONTEBURNS with ORIGEN-S for the WWER fuel rod isotopics calculation in an infinite array, considering:

- Uniform burnup
- Three different burnup values: 10, 30 and 40 MWd/kgU
- Cooling time: 1 year (for 10 and 40 MWd/kgU) and 1 and 5 years (for 30 MWd/kgU)

Finally, the influence of the isotopic evolution on the criticality of the storage system has also been considered.

MONTEBURNS CODE

MONTEBURNS code has been developed in Los Alamos Laboratory by David Poston and Holly Trellue. Its principal feature is the capability of linking the Monte Carlo transport code MCNP4B with the radioactive decay and burnup code ORIGEN2.1. MONTEBURNS transfers one-group cross-section and flux values from MCNP to ORIGEN, and then transfers the resulting material compositions (after irradiation and/or decay) from ORIGEN back to MCNP in a repeated cyclic fashion. This operation is undertaken only with those isotopes, which are considered as important for the neutronic problem by the user. For the rest of the isotopes the default ORIGEN cross-section libraries are used by MONTEBURNS.

INITIAL DATA (from the CB3 benchmark)

The WWER design is hexagonal. The next two figures correspond to the horizontal and vertical cross-sections, respectively, taken from the MCNP plot function.

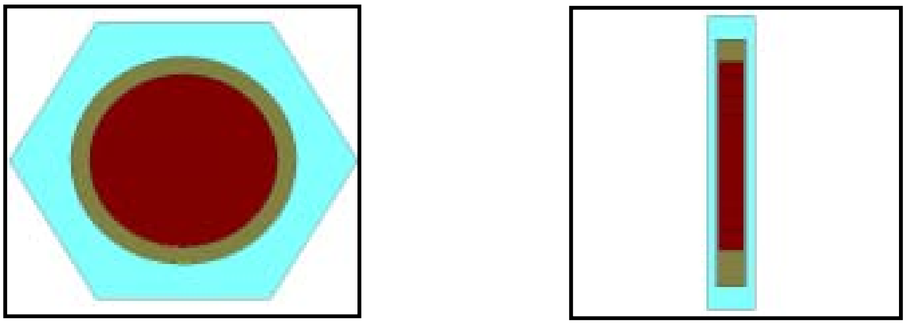

FIG. 1.

- **Fuel specifications**

Fuel enrichment: 3.6% wt. ^{235}U
Fuel cell pitch: 1.22 cm
Fuel radius: 0.38 cm
Pellet density (effective): 10.193 g/cm3
Active fuel length: 244 cm

- **Cladding specifications**

Inner radius: 0.38 cm
Outer radius: 0.455 cm
Material: 1% wt. Nb, 98.97% wt. Zr, 0.03% wt. Hf
Density: 6.0881 g/cm3

- **Moderator**

Water at 0.72 g/cm3
Soluble boron: 500 ppm

- **Operating history data**

Specific power: 4.143 MW/assembly (average value)
Since there are 126 rods/assembly, the specific power for one fuel rod is: 0.0329 MW

Number of cycles: 1, 3, 4 for the 10, 30, 40 MWd/kgU burnup cases respectively
Cycle duration
  - Uptime: 300 days
  - Downtime (between cycles): 65 days

Cooling time
  - 1 year for 10 and 40 MWd/kgU burnup cases
  - 1 and 5 years for 30 MWd/kgU case

HYPOTHESES

MONTEBURNS combines the codes MCNP and ORIGEN. The geometry is built by MCNP and ORIGEN undertakes the burnup simulation. Since MCNP is a three-dimensional code, the WWER fuel rod is modeled in 3-D, but using reflective boundaries to avoid any leakage in any direction in order to approximate ORIGEN-S conditions for an infinite array.

The density of the moderator (light water) is supposed to be constant and the fuel composition is also uniform, radially and axially.

The ENDF/B-VI library is used for most of the isotopes, though the ENDF/B-V is employed for some fission products. Finally, the Watt fission spectrum is considered.

METHODOLOGY

Three input files are usually necessary to define a simulation with MONTEBURNS:

- The MCNP input file defines the geometry of the system and provides the detailed cross-section libraries to ORIGEN for those isotopes considered as important by the user
- By means of the MONTEBURNS input file the power of the system, the isotopes whose results are required and the default ORIGEN library (for the not important isotopes) are determined
- The MONTEBURNS feed file defines the number of times which the ORIGEN libraries are updated by MCNP

The concentration of the next isotopes (actinides + fission products) is required:

^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Rh, ^{109}Ag, ^{133}Cs, ^{143}Nd, ^{145}Nd, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{151}Sm, ^{152}Sm, ^{153}Eu, ^{155}Gd

RESULTS

The next figures represent the ratio MONTEBURNS/ORIGEN-S for the isotopics calculation for the different values of burnup and cooling time. The ORIGEN-S values are taken from the CB3 benchmark specification.

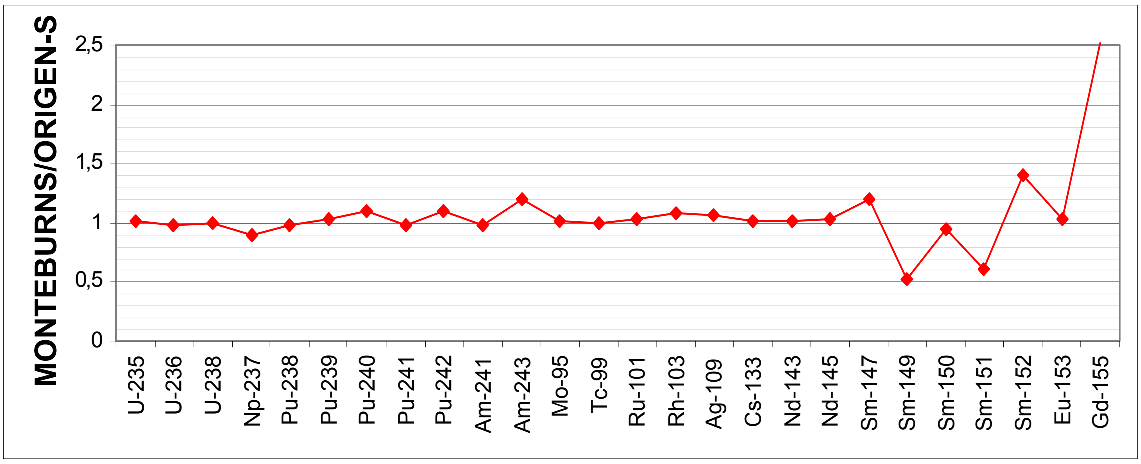

FIG. 2. 10 MWd/kgU and 1 year cooling time.

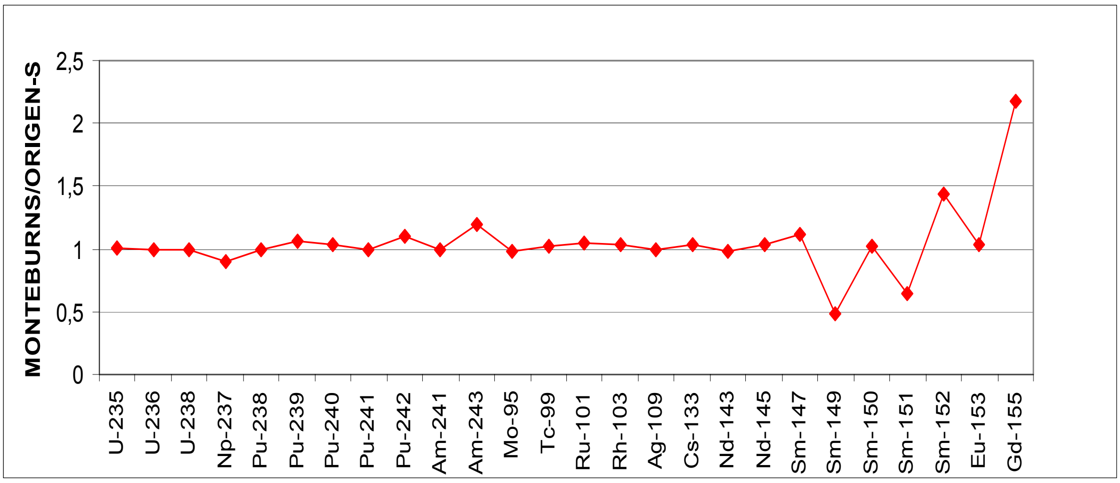

FIG. 3. 30 MWd/kgU and 1 year cooling time.

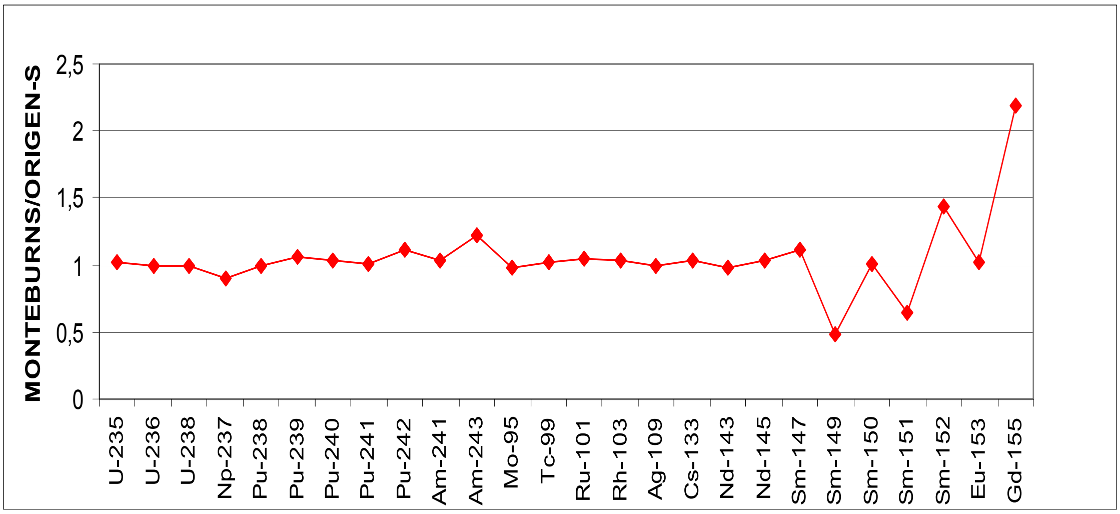

FIG. 4. 30 MWd/kgU and 5 years cooling time.

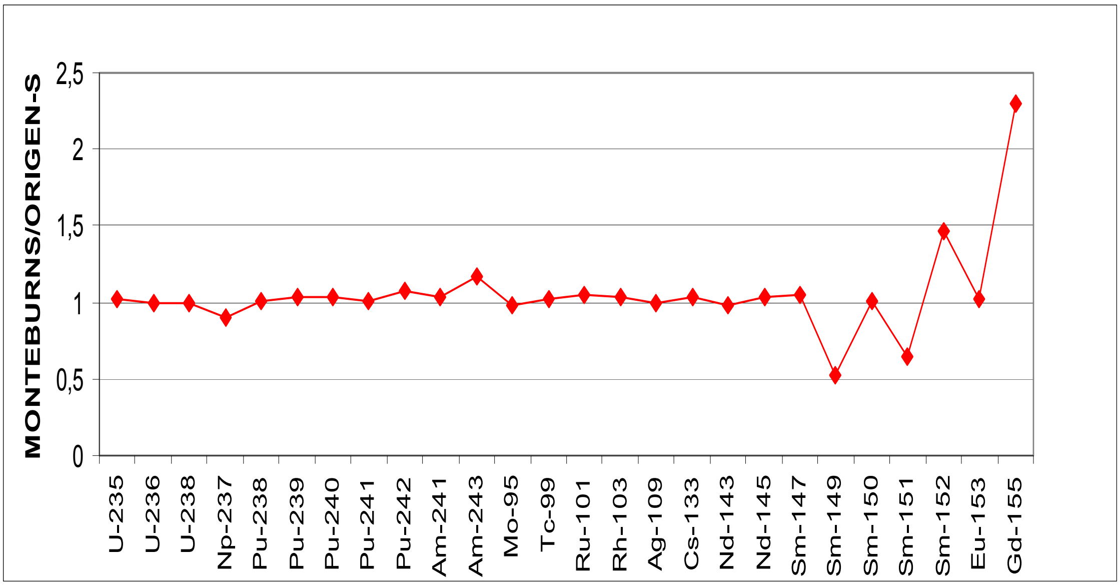

FIG. 5. 40 MWd/kgU and 1 year cooling time.

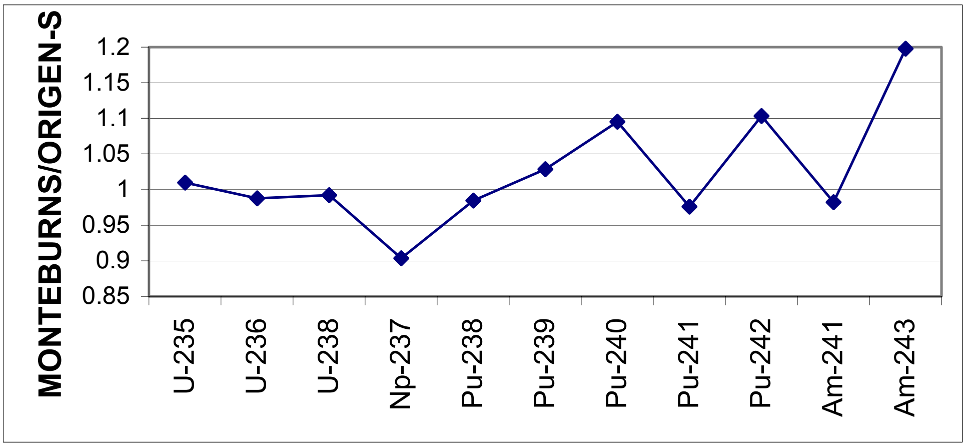

FIG. 6. Actinides.

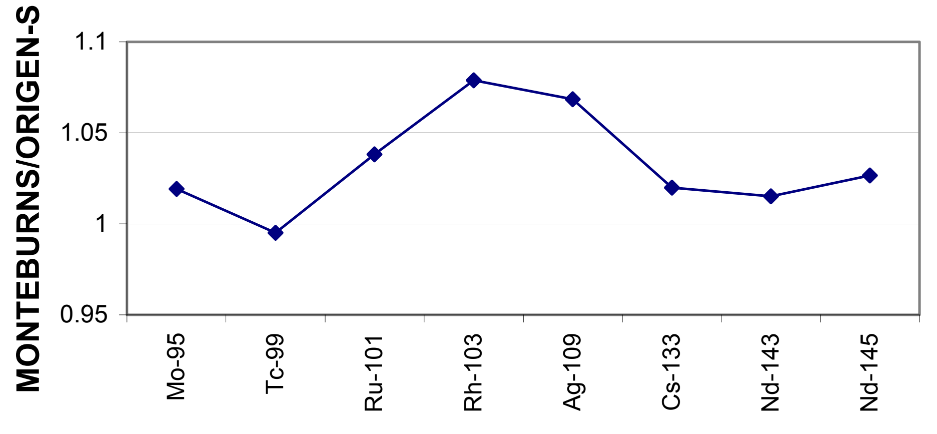

FIG. 7. Fission Products.

It is clear, looking at the figures above, that there are important discrepancies for some isotopes: ^{243}Am, ^{155}Gd, ^{149}Sm, ^{151}Sm and ^{152}Sm. In the next two figures (one for the actinides and a second one for the fission products except for the samarium family and ^{155}Gd) the ratio MONTEBURNS/ORIGEN-S for 10 MWd/kgU and 1 year cooling time (the rest of the cases are similar to this one) is represented in a more detailed scale, in order to appreciate with more clarity the differences between the results of the two codes.

Influence on the multiplication factor

The different isotopic compositions calculated with MONTEBURNS and ORIGEN-S are introduced into the MCNP input which models the WWER fuel rod in cold conditions, in order to calculate the keff of the system in all the burnup and cooling time cases. For both calculations the same libraries (ENDF/B-VI and ENDF/B-V) are used. Hence, the deviations observed are only produced by the differences in the concentrations. These are the results:

Burnup (cooling time)	10(1)	30(1)	30(5)	40(1)
k-eff (monteburns)	1.255	1.102	1.086	1.028
k-eff (origen-s)	1.216	1.068	1.050	1.006
deviation (%)	3.2	3.2	3.4	2.2

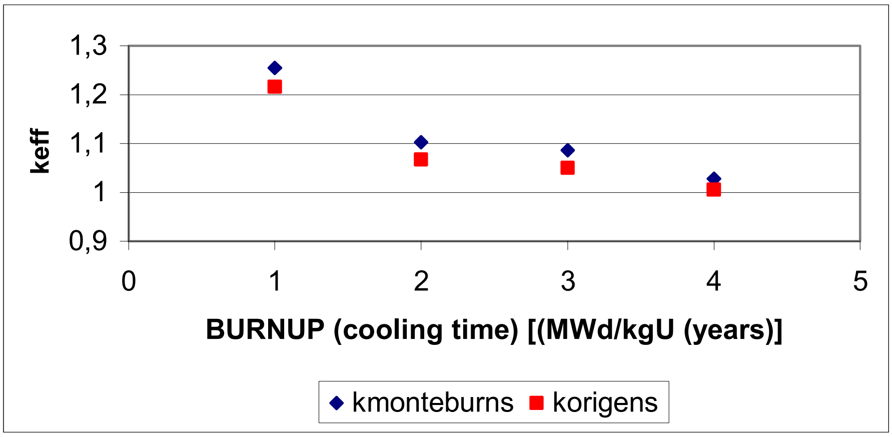

FIG. 8.

CONCLUSIONS

An isotopic calculation has been undertaken for the WWER fuel rod in order to compare the results provided by two different codes, ORIGEN-S and MONTEBURNS, and for evaluating the capabilities of prediction, which the latter has in burnup calculations.

In most of the isotopic calculations a relative error lower than 10% is achieved between MONTEBURNS and ORIGEN-S results, except for the next nuclides: ^{243}Am, ^{155}Gd, ^{149}Sm, ^{151}Sm and ^{152}Sm. For the Sm family and ^{155} Gd there are some undefined resonance regions in the ENDF-B-V cross-section libraries, which may be the reason of the big discrepancies. In the ^{243}Am case, the relative error is about 20%, because there are insufficient resonance shielding calculations for ^{242} Pu and its progeny (^{243} Am and ^{244}Cm).

In the criticality calculations, the differences in the isotopic compositions calculated with MONTEBURNS and ORIGEN-S produce a constant overestimation of the keff calculated with the MONTEBURNS composition with respect to the ORIGEN-S one. The average value of this difference is about 3%, which is due to the overestimation of the ^{235}U and ^{239} Pu isotopics by MONTEBURNS. This discrepancy may be important in spent fuel storage or transport facility.

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Research to support expansion of U.S. regulatory position on burnup credit for transport and storage casks

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Abstract. In 1999, the United States Nuclear Regulatory Commission (U.S. NRC) initiated a research program to support the development of technical bases and guidance that would facilitate the implementation of burnup credit into licensing activities for transport and dry cask storage. This paper reviews the following major areas of investigation: (1) specification of axial burnup profiles, (2) assumption on cooling time, (3) allowance for assemblies with fixed and removable neutron absorbers, (4) the need for a burnup margin for fuel with initial enrichments over 4 wt %, and (5) evaluation of experimental data. Recommendations resulting from the research program are presented.

1. INTRODUCTION

The concept of taking credit for the reduction in reactivity due to irradiation of nuclear fuel (i.e., fuel burnup) is commonly referred to as burnup credit. The reduction in reactivity that occurs with fuel burnup is due to the net reduction of fissile nuclides and the production of parasitic neutron-absorbing nuclides (non-fissile actinides and fission products). Historically, criticality safety evaluations for transport have assumed the fuel contents to be unirradiated fuel compositions. In July 1999, the U.S. NRC Spent Fuel Project Office (SFPO) issued Revision 1 of Interim Staff Guidance 8 (ISG8) to provide staff recommendations for the use of burnup credit for storage and transport of pressurized-water reactor (PWR) spent fuel [1]. Subsequently, the recommendations of ISG8 were included in the staff Standard Review Plan for transport casks [2].

Since the issuance of ISG8 Rev. 1, in July 1999, the U.S. NRC Office of Regulatory Research (RES) has sponsored Oak Ridge National Laboratory (ORNL) to help develop expanded guidance relative to selected elements of ISG8, to develop a technical basis for staff consideration of potential revisions of ISG8, and to implement software enhancements that can facilitate the use of computational methods in safety analyses. A baseline report [3] was prepared to review the status of burnup credit and to provide a strawman prioritization for areas where additional guidance, information, and/or improved understanding were considered to be beneficial to the effective implementation of burnup credit in transport and dry storage casks. As a result of the initial review and input from industry and licensing staff, four focus areas for the NRC research program were established. The NRC SFPO will discuss each of these four focus areas together with the research recommendations for consideration.

2. AXIAL BURNUP PROFILE

It is well established (e.g., see [3]) that the axial burnup profile in a spent fuel assembly is an important component of the criticality safety analysis for burnup credit. Although ISG8 notes the importance of the axial profile, an acceptable approach for modeling axial burnup is not

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[*] Managed by UT-Battelle, LLC, under contract DE-AC05-00OR22725 with the U.S. Department of Energy.

provided. The research program has sought to develop and propose initial guidance that can be readily implemented by industry and readily reviewed by NRC staff. Thus, a review and evaluation [4] of the existing, publicly available U.S. database of axial burnup profiles was performed. This database [5] of 3169 axial burnup profiles from ~1 700 different assemblies was developed using information from 20 different U.S. PWRs representing 106 cycles of operation through the mid-1990s. Although the database represents only 4% of the assemblies discharged through 1994, the review indicates the database provides a good statistical representation of discharged assemblies in terms of fuel vendor/reactor design, types of operation (i.e., first cycles, out-in fuel management and low-leakage fuel management), burnup and enrichment ranges, and use of burnable absorbers. For burnup and enrichment values beyond the current limits of ISG8 (40 GWd/MTU and 4.0 wt %), expansion of the existing database would be desirable to increase the number of profiles representing that regime. However, Ref. 4 indicates that the bounding profile from intermediate burnup ranges do bound the available profiles at higher burnups. Consequently, the existing database may be adequate for burnups beyond 40 GWd/MTU; additional work is needed to better understand the phenomena.

Previous work [6] identified the axial profiles within the database that provide the highest neutron multiplication factors (k_{eff}) over selected burnup ranges. This information was used to propose artificial bounding profiles for each burnup range. Figure 1 shows the spread of keff values that result from the set of profiles available from a selected burnup range, together with the actual bounding profile from the database and the proposed (artificial) bounding profile from Ref. 6. Note that the profiles are in arbitrary order and they display the discrete values associated with each of the 228 profiles. The figure shows the mean k_{eff} value and indicators for 1, 2, and 3 standard deviations. An examination of the calculated k_{eff} values reveals that, for each of the 12 burnup ranges, the keff value associated with the actual bounding axial profile is more than 3 standard deviations above the mean and, in most cases, is more than 5 standard deviations above the mean. In other words, the limiting profiles can be considered statistical outliers that appropriately bound the range of actual profiles, as opposed to being representative of typical spent nuclear fuel (SNF) profiles. Consequently, one can infer that there is a very small probability for the existence of other profiles that are notably more reactive than the limiting profile determined from the database. When one considers that the limiting profiles are based on statistical outliers and that these limiting profiles will be applied to all assemblies in a burnup credit cask, it appears that such an approach provides adequate bounding of the realistic condition in a cask with respect to the axial burnup issue. A study [4] to investigate the impact of loading one assembly with a significantly more reactive profile (cask system worth up to 6% Δk more than a system with limiting profile) indicates the cask multiplication factor would increase by less than 0.5% Δk. These analyses have led to the recommendation that this publicly available database is an appropriate source for selecting bounding axial burnup profiles to be used in a safety analysis. The rationale for this recommendation is: (1) the axial profile database provides an adequate representation of discharged U.S. PWR SNF; (2) the bounding profiles, as determined from the database, are statistical outliers, and thus, the probability that more reactive profiles exist is small; (3) the bounding profiles will be applied to all assemblies in a burnup credit cask; and (4) the low consequence associated with the loading of an assembly with a more extreme profile.

3. COOLING TIME

ISG8 recommends that safety analyses be performed at a fixed cooling time of 5 years. Figure 2 shows the trend of keff with cooling time for a 32-element, generic burnup-credit cask design [7] (GBC-32). For burnup-credit criticality safety analyses performed at 5 years, increased cooling times result in an increasing conservative safety margin out to ~50 years. The additional benefit for cooling times between 50 and 100 years is insignificant. A cooling time of 40 years provides a keff value that approximately equates to the keff value at 200-year cooling, which might be considered a practical lifetime for dry storage and transport casks. Thus, this rationale leads to a conclusion that cooling times up to 40 years can be assumed in developing the safety basis. To address concerns with use of storage casks beyond the assumed 200-year storage time and to lay a consistent foundation that enables future extension beyond the ISG8 actinide-only recommendation, a value of 10 years could be assumed as the cooling time limit for safety analysis. The rationale is that the best-estimate results (lower curve of Fig. 2) for keff at a 10-year cooling time are always greater than the maximum keff in the secondary peak (10 000-to-30 000-year time frame). The incentive in moving from 5 years to 10 years is approximately 1% Δkeff, with the largest benefit seen for the major actinides case.

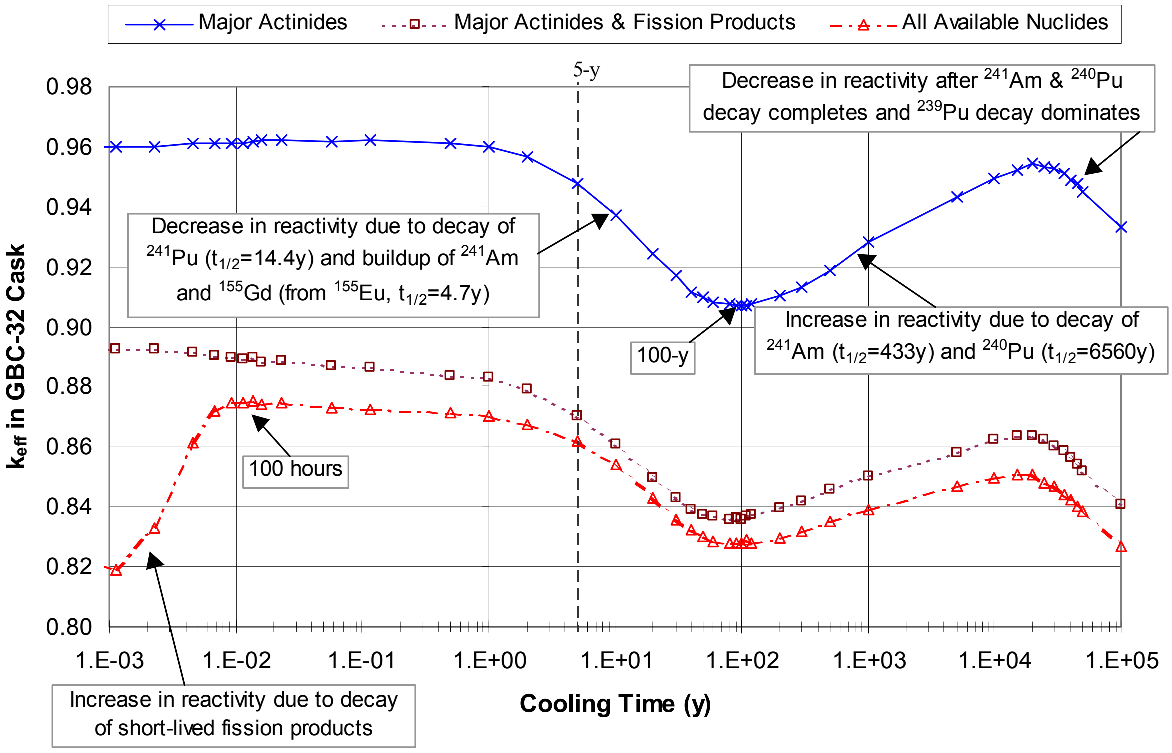

FIG. 1. Values of k_{eff} for an infinite planar array as a function of database axial profiles for 38-42 GWd/MTU.

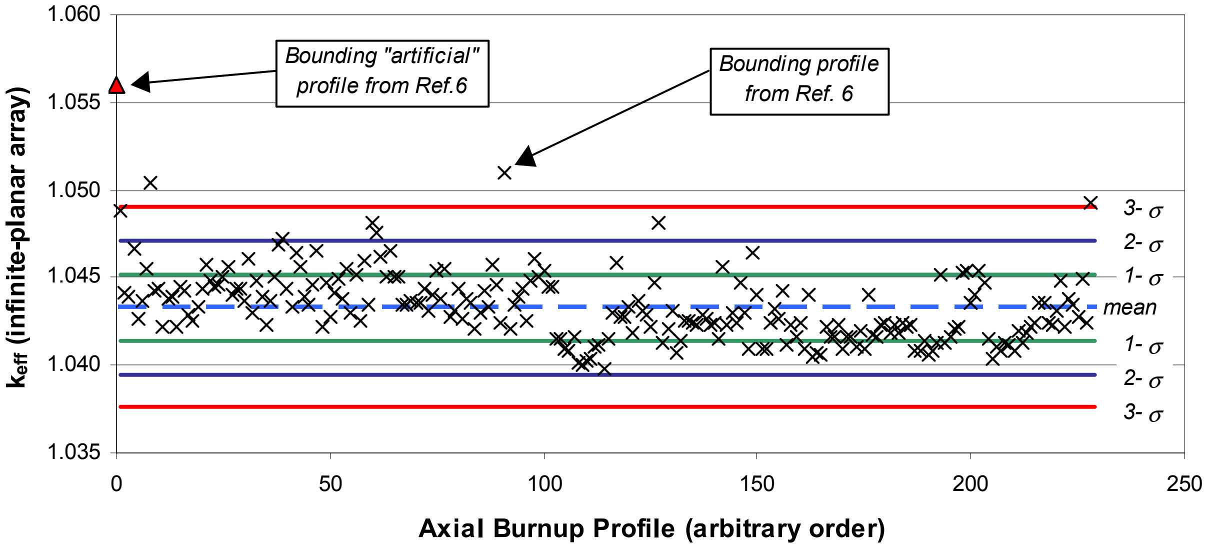

FIG. 2. Values of keff in the GBC-32 cask as a function of cooling time for 4.0-wt % fuel burned to 40 GWd/MTU.

4. FIXED AND REMOVABLE ABSORBERS

Assemblies exposed to fixed neutron absorbers [integral burnable absorbers (IBAs)] and removable neutron absorbers [burnable poison rods (BPRs) and control rods (CRs)] can have higher k_{eff} values than assemblies which are not exposed because the presence of the absorber will harden the spectrum and lead to increased ^{239}Pu production and reduced ^{235}U depletion. In addition, when removable neutron absorbers are inserted, the spectrum is also hardened due to moderator displacement. Since this effect had not been fully quantified at the time ISG8 was issued, the NRC recommendation in ISG8 was to restrict the use of burnup credit to assemblies that have not contained IBAs or BPRs during any part of their exposure. In addition, guidance for licensees to consider the impact of CRs was noted in ISG8.

The restriction on burnable absorbers (IBAs and BPRs) eliminates a large portion of the spent fuel from being loaded in a burnup credit cask. To provide a technical basis for potential change to this guidance, investigations [8–10] have been performed to quantify how the k_{eff} of a discharged assembly would change due to exposure to BPRs, IBAs, and CRs. A comprehensive range of assembly designs, absorber loadings, and exposure history (for BPRs and CRs) was used to determine the impact on the k_{eff} of spent fuel. The studies show that exposure to BPRs can cause the k_{eff} to increase a maximum of 3% when the maximum number of BPRs and/or the maximum absorber loading is assumed for the maximum exposure time. More typical absorber loadings and exposures (1-cycle of 20 GWd/MTU) lead to increases of < 1% Δk (e.g., see Fig. 3). By comparison, except for one IBA type where the increase was a maximum of 0.5% Δk (i.e., see Fig. 4), the IBAs actually provide a decrease in k_{eff} relative to assemblies not exposed to IBAs. References 8–9 provide a base characterization for the effect of burnable absorbers on spent fuel and indicate that a depletion analysis with bounding BPR loadings and exposure limits should provide an adequate bounding safety basis for fuel with or without burnable absorbers.

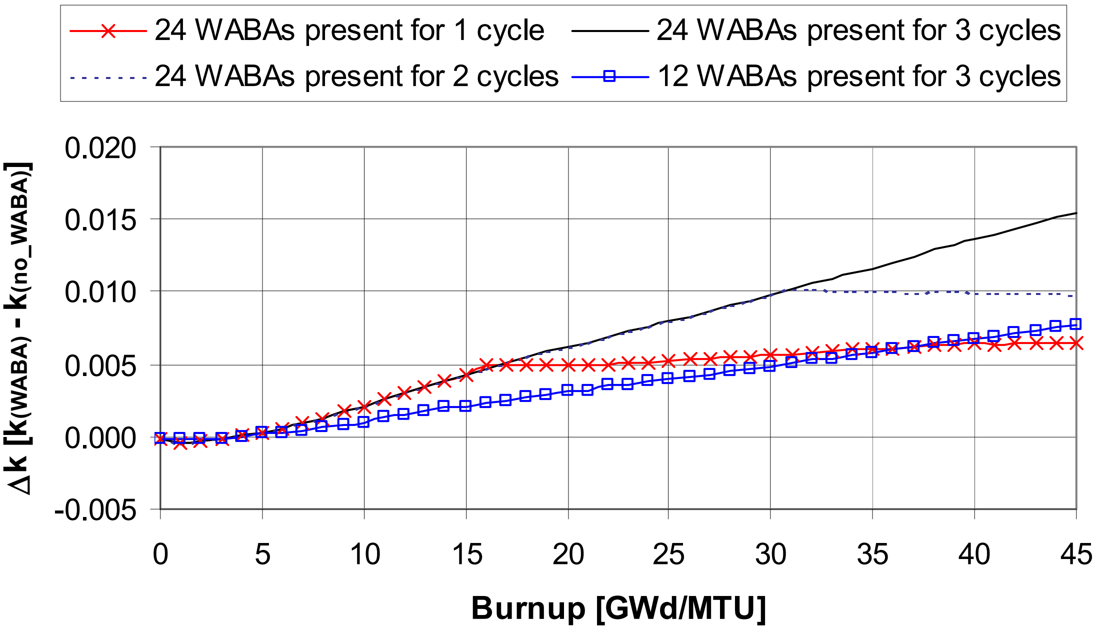

FIG. 3. Comparison of Δk values, as a function of burnup, for assemblies exposed to Wet Annular Burnable Assembly (WABA) rods. Results correspond to Westinghouse 17 × 17 assemblies with 4.0 wt % ^{235}U initial enrichment.

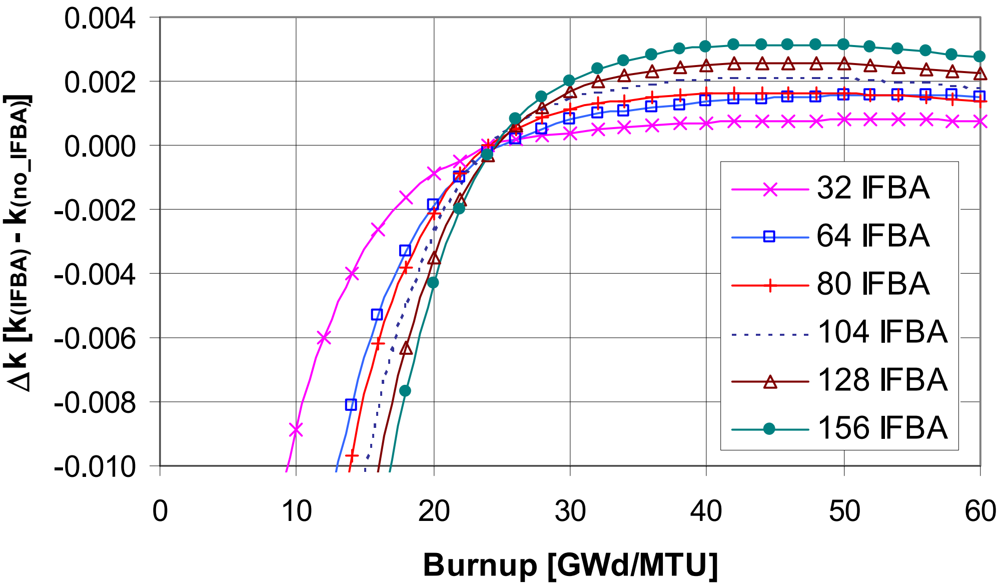

FIG. 4. Comparison of Δk values, as a function of burnup, between assemblies with and without Integral Fuel Burnable Absorber (IFBA) rods present. Results correspond to Westinghouse 17 × 17 assemblies with 4.0 wt % ^{235}U initial enrichment.

The results of a parametric study [10] to quantify the effect of CR exposure are summarized in Fig. 5, where it can be seen that, even for significant burnup exposures (up to 45 GWd/MTU), minor axial CR insertions (e.g., < 20 cm) result in an insignificant effect (< 0.2% Δk) on the k_{eff} of a burnup credit cask. Full insertion for burnups up to 5 – 10 GWd/MTU increase cask k_{eff} on the same order as seen for BPRs. Since BPRs and CRs can not be inserted in an assembly at the same time and since CRs, if inserted, are normally placed in first cycle assemblies, it follows that a bounding consideration of BPRs should bound the potential for SNF to have been exposed to CRs during irradiation.

5. LOADING OFFSET FOR HIGH INITIAL ENRICHMENTS

Currently, ISG8 limits credit for burnup to 40 GWd/MTU and initial enrichments to 4 wt %, although allowance for initial enrichments up to 5 wt % is permitted with an added burnup margin applied at loading. The major reason for these recommended limitations is the lack of chemical assay data for higher burnups and enrichments. When ISG8 was issued the experimental database of public domain actinide assay data in the U.S. consisted largely of samples from older fuel assembly designs with enrichments below 3.5 wt %, and contained only one measurement for fuel above 3.4 wt % (a 3.89 wt % sample with a low burnup of 12 GWd/MTU). Only seven of the approximately 50 samples had BPRs present during irradiation.

The loading offset of ISG8 provides a means of extending the usefulness of ISG8 to include spent fuel with initial enrichments above 4 wt % using an engineering approach to compensate for potentially larger uncertainties. Several studies [11, 12] suggest, however, that the effect of enrichment on isotopic uncertainties is minimal. Published French results [11] for Gravelines spent fuel using French computational methods and JEF cross-section data indicate a level of agreement that is comparable to that of lower-enrichment fuel. In addition, sensitivity-based methods have been applied at ORNL to assess the influence of nuclear data bias and uncertainties on the isotopic compositions and the k_{eff} of a spent fuel storage cask [12]. These studies indicate that there is a strong correlation between spent fuel systems with a constant enrichment-to-burnup ratio. The results suggest that existing isotopic assay data may be highly applicable to regimes well beyond that of the data and that the basic depletion phenomena do not change significantly with relatively minor increases in enrichment (i.e., from 4 to 5 wt %).

Assay data for SNF from the Takahama PWR now extends the enrichment/burnup range to 4.1 wt % and 47.3 GWd/MTU [13]. The Takahama measurements include an extensive number of burnup credit actinides and fission products. Including calculated-to-experiment (C/E) ratios of this assay data into the existing database of C/E data indicates there is no significant increase in the uncertainty of the neutron multiplication factor as the SNF burnup increases. Reference 14 provides a discussion of the trends seen using this expanded C/E database.

6. BURNUP CREDIT ANALYSIS SEQUENCE

ISG8 highlights the need for applicants employing burnup credit in criticality safety assessments to account for the axial and horizontal variation of the burnup within a spent fuel assembly. In practice, the axial burnup variation (e.g., the axial burnup profile) is commonly modeled in a criticality calculation using a finite number of axial segments or zones (10 to 20 is typical) to represent the burnup profile, each zone having a uniform average burnup for that segment. Consequently, implementation of burnup credit using this approach requires separate fuel depletion calculations for each axial zone, and the subsequent application of these spent fuel compositions in the criticality safety analysis. Implementation of this approach requires that numerous spent fuel depletion calculations must be performed, and potentially large amounts of data must be managed, converted, and transferred between the depletion and criticality codes.

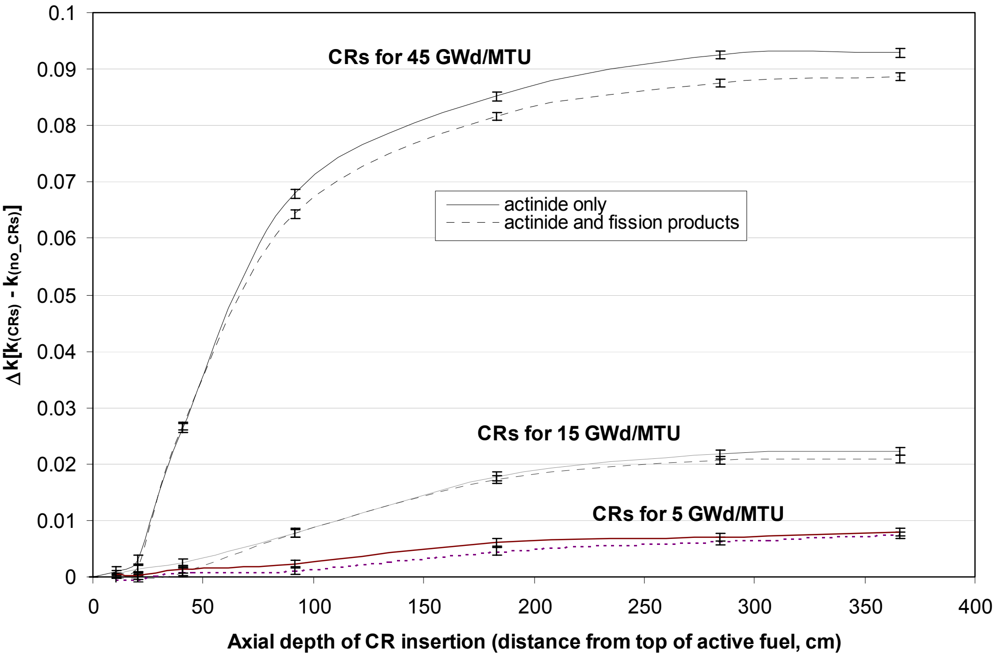

FIG. 5. Impact of CR insertion during irradiation on SNF in the GBC-32 cask.

To simplify this analysis process and assist the NRC staff in their review of criticality safety assessments of transport and storage casks that apply burnup credit, a new SCALE control sequence, STARBUCS (Standardized Analysis of Reactivity for Burnup Credit using SCALE) has been created [15]. STARBUCS automates the generation of axially varying isotopic compositions in a spent fuel assembly, and applies the assembly compositions in a three-dimensional (3-D) Monte Carlo analysis of the assembly in a cask environment. The STARBUCS control sequence uses the new ORIGEN-ARP methodology [16] of SCALE to perform automated and rapid depletion calculations to generate spent fuel isotopic inventories in each axially-varying burnup zone of a fuel assembly. The analyst need only specify the average assembly irradiation history, the axially varying burnup profile, the actinides and, optionally, the fission products that are to be credited in the criticality analysis. An arbitrary number of axial zones may be employed, or the user may select from several pre-defined profiles. This series of calculations is used to generate a comprehensive set of spent fuel nuclide compositions for each axial zone of the assembly. The STARBUCS sequence uses the SNF inventories provided for each zone to automatically prepare cross sections for the criticality analysis. A 3-D KENO V.a criticality calculation is performed using cask geometry specifications provided by the user. Isotopic correction factors (ICFs) may also be applied to correct the criticality calculation for known bias and/or uncertainty in the prediction of the isotopic concentrations.

This new STARBUCS sequence has been used at ORNL to support the study of the impact of various assumptions that might be applied in the development of a loading curve. Figure 6 illustrates three loading curves highlighted against the 1998 inventory of U.S. discharged fuel. The loading curves show how the assumptions relative to selected nuclides and associated ICFs can lead to significant increases in the spent fuel inventory that can be loaded in a burnup credit cask. The curves indicate that, as discharge burnups and initial enrichments increase, efforts to incorporate fission products and/or reduce the ICFs will be needed to assure a burnup credit cask can carry a significant portion of the fuel anticipated for future discharge.

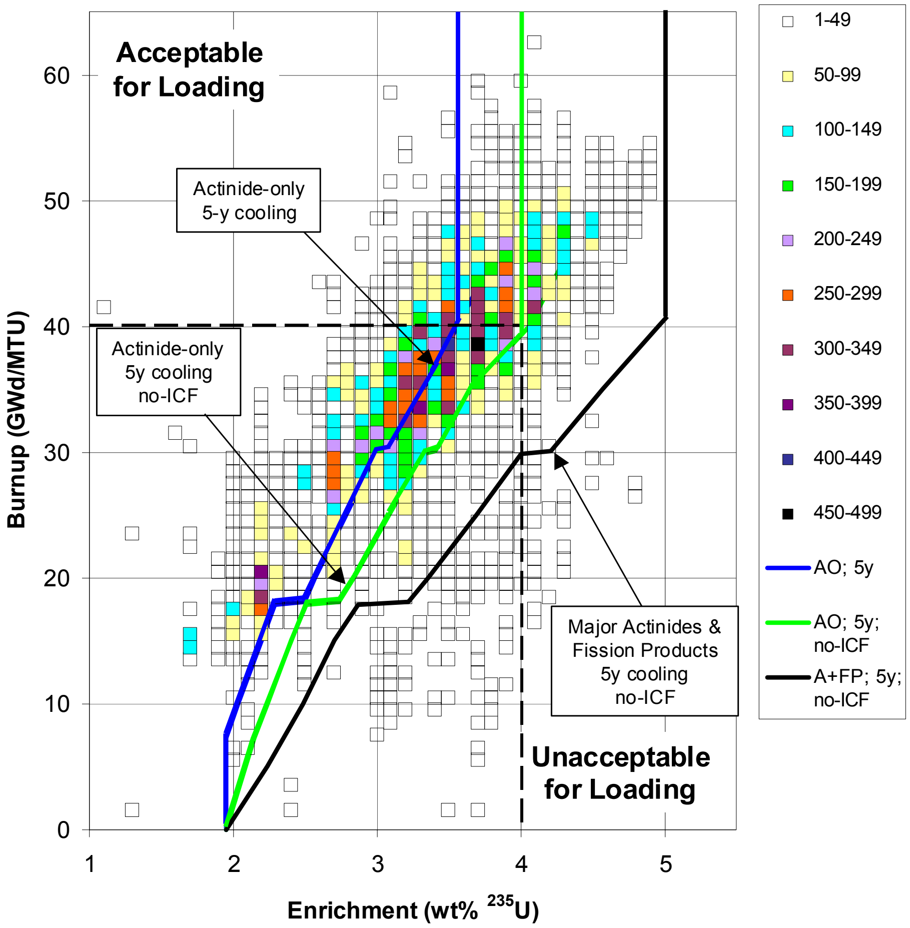

FIG. 6. Illustrative loading curves for GBC-32 cask shown with PWR SNF discharge data through 1998 (numbers in legend indicate number of assemblies). Dashed lines represent current burnup and enrichment limits of ISG8. ICF refers to the Isotopic Correction Factors.

7. SUMMARY

The technical bases needed to help improve and expand the U.S. regulatory guidance for burnup credit in transportation casks have been developed at ORNL under the direction of the U.S. NRC research staff. *The goal has been to develop criteria and/or recommendations that are technically credible, practical, and cost effective while maintaining needed safety margins*. The technical work performed at ORNL is now undergoing final review by NRC staff and it is anticipated that changes to the recommendations of ISG8 will be forthcoming.

ACKNOWLEDGEMENTS

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Bounding approach in burnup credit criticality safety analysis

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Abstract. In this paper some guidance is given how to find a bounding approach in a burnup credit criticality safety analysis of a spent fuel management system of interest. For this purpose first of all the term "bounding approach" is defined and distinguished from the term "analysis conservatism". It is shown that even if the level of conservatism is nil, the need for a bounding approach still exists. Then, the main considerations of the key steps and key parameters of burnup credit criticality safety analysis implementation are discussed: The impact of the fuel characteristics on the spent fuel management system's reactivity is briefly described. The depletion conditions –characterized by the parameters specific power and operating history, fuel temperature, moderator temperature and density as well as by the presence of soluble boron and fixed neutron absorbers in the core and the presence of integral burnable absorbers in the fuel – are briefly discussed with particular attention being given to the sensitivity of the system's reactivity to variations in these conditions. The main steps of the criticality calculation procedure – isotopic selection and validation, validation of the criticality calculation code used, sensitivity studies on the reactivity effect of the system's tolerances and the reactivity effect of axial and horizontal burnup profiles, as well as analysis of abnormal and accidental events – are briefly discussed with particular attention focused on effects which may affect the reactivity worth of the spent fuel under the conditions of the spent fuel management system of interest. The outcome of a burnup credit criticality safety analysis is usually given in the form of a loading curve. This curve is based on a reactivity equivalence relation given by the criticality safety acceptance criterion to be applied to the system of interest. Any such criterion includes a safety margin, and therefore, at the very beginning, the term "safety margin" is defined and its relation to the terms "analysis conservatism" and "bounding approach" is clarified.

1. INTRODUCTION

The criticality safety acceptance criteria to be applied to burnup credit analysis are the same as those used for analysis based on the fresh fuel assumption. No regulation, whether national or international, has been found that prescribe different criteria for burnup credit cases. The approach to be taken in demonstrating sufficient subcriticality based on numerical calculation of the neutron multiplication factor is prescribed in all the regulations in the same way: The evaluated neutron multiplication factor of a spent fuel management system under study

- shall include all the calculation uncertainties and all the uncertainties arising from the fuel and the system manufacturing tolerances and
- shall not be greater than an adequate upper bound $(1-\Delta \mathrm{k}_{\mathrm{m}})$ of subcriticality, under all normal operation conditions as well as all anticipated abnormal or accident conditions.

Which value of the "safety margin" $\Delta \mathrm{k}_{\mathrm{m}}$ is acceptable depends on the problem or system under study, i.e. on the conditions to be analyzed. $\Delta \mathrm{k}_{\mathrm{m}}$ is usually lower for low-probability events.

2. SAFETY MARGIN, ANALYSIS CONSERVATISM AND BOUNDING APPROACH

2.1. Safety Margin

The safety margin Δk_m is given by the value necessary to ensure an accepted margin of *subcriticality*. Acceptable $\Delta \mathrm{k}_{\mathrm{m}}$ values are laid down in the regulations (cp. References [1] through [8], e.g.), but may vary with the conditions to be analyzed.

In general,

- $\Delta k_m = 0.05$ is applied to normal operation conditions as well as abnormal or accident conditions,
- $\Delta k_m = 0.02$ (cf. [4], e.g.) or $\Delta k_m = 0.03$ (cf. [6], e.g.) is included in some regulations for accidental events of very low probability of occurrence or related risk.[1] [2]

Whichever value is used for Δk_m, the safety margin is, by definition, a *fixed* value.

It is often thought and said that application of burnup credit may result in a reduction of the safety margin. However this view does not comply with the definition of the safety margin. And due to the fact that safety margins applied in burnup credit applications are the same as those used for analysis based on the fresh fuel assumption, *there is no reduction in the safety margin*. What is really meant is that application of burnup credit results in a reduction of the analysis conservatism because the hypotheses used are more realistic, and in fact, in practice the safety margin is always used as a fixed value, in accordance with its definition, [10].

2.2. Analysis Conservatism

Analysis conservatism means a deliberate overestimation of the neutron multiplication factor of a fuel system due to specific features of the model used to describe the properties of the system in the analysis procedure applied, [10].

Analysis conservatism is strictly related to the description of the *physics of the system* and is hence related to an *a priori knowledge*. Due to physics it is a priori known, for instance,

- that ignoring of neutron absorbing materials present in the system results in an overestimation of the neutron multiplication factor of the system,
- that the traditional fresh fuel assumption results in significant conservatism in the calculated neutron multiplication factor of a spent fuel system,
- that the actinide-only burnup credit maintains a lot of conservatism in comparison to the actinide plus fission product burnup credit level.

A reason for taking analysis conservatism could be the lack of knowledge or tools needed to exactly model a certain effect that would reduce the calculated neutron multiplication factor, but also the desire to simplify the analysis is often a good motivation.

Analysis conservatism refers to the difference between the neutron multiplication factor k_{CC}, obtained for the model used and assumed to be corrected on any possible bias, and the actual neutron multiplication factor k_S of the real system. It is a priori known that k_{CC} is greater and not equal to k_S, $k_{CC} > k_S$.

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[1] In Ref. [3] only $(1-\Delta k_m) < 1$ is required for the analysis assumption that, due to a dilution event, the boron content of the coolant of a PWR wet storage pond fades away down to 0 ppm, provided that this assumption is purely hypothetical.

[2] In some countries the requirements for spent fuel transport prescribed in Ref. [9], paras 671 through 682, are interpreted as follows: $\Delta k_m = 0.05$ is used for the criticality safety assessment of an individual package in isolation, $\Delta k_m = 0.02$ is used for the criticality safety analysis of package arrays under normal and accident conditions of transport.

Removal of an analysis conservatism would result in k_{CC} being closer to the real system's neutron multiplication factor, and can only be allowed when the conservatism can be quantified, i.e., when

- the state of knowledge allows for an adequate understanding of the physical phenomena involved and
- the calculation tools can properly model these phenomena.

Burnup credit implementation eliminates a conservatism (fresh fuel assumption) that was originally included, among other reasons, due to the lack of sufficient validation of the spent fuel isotopic calculation.

2.3. Bounding Approach

A bounding approach is a calculation model justified by *physics arguments* demonstrating that the use of the model *does not result in an underestimation* of the neutron multiplication factor of the fuel system of interest, [10].

Bounding approach and analysis conservatism should never be confused. In the limit, assuming that all the knowledge needed is available and the calculation system is perfect, the level of conservatism is nil, but the need for a bounding approach still exists, to guarantee that the calculated reactivity is the maximum that the fuel management system will achieve.

Which approach is in fact the bounding one this depends on the fuel management system and the requirements one makes on this system. Obviously, the *minimum* bound is achieved when the individual reactivity of each and every spent fuel assembly is determined and the system's reactivity is calculated using the real position of each assembly. This would however mean that all the parameters that define the reactivity state of each fuel assembly need to be known and verified for each assembly throughout its life, and a depletion calculation would be performed for each assembly. Then a reactivity calculation using the real loading scheme would need to be performed, and revised every time this layout is modified. In other words, approaching the minimum bound usually results in an increased volume of calculation, to specifically address the specific characteristics of the problem, and results in more detailed verifications during implementation.

Therefore, in order to maintain a rational level of analysis amount and verification, there is need for a bounding approach. For instance, the outcome of a burnup credit criticality safety analysis is often presented in the form of a loading curve indicating the minimum burnup (or a related parameter) necessary for fuel with a specific initial enrichment to be loaded in the spent fuel management system designed for burnup credit. The loading curve applies to any fuel position of this system. (In fact, one can generate a loading curve for the whole system only then, if no credit is taken for the real loading scheme.) A loading curve must bound therefore the wide variety of fuel irradiation histories to be taken into account, i.e., the generation of a loading curve makes it necessary to look for a bounding history given by those fuel operation conditions that lead to the highest reactivity of the spent fuel under the conditions of the spent fuel management system under study.

3. KEY STEPS AND PARAMETERS IN BURNUP CREDIT IMPLEMENTATION

Application of burnup credit to a spent fuel management system consists in implementation of three key steps:

- estimation of the spent fuel composition,
- criticality calculation and evaluation of the loading criterion,
- quantification and verification of the fuel burnup.

The main considerations of the first and the second step will be discussed in the following sections.

3.1. Estimation of the Spent Fuel Composition

The isotopic inventory of spent fuel is predicted with the aid of depletion calculations. To perform a depletion analysis one has to define:

- the fuel characteristics
- the fuel depletion conditions (reactor operation conditions) and
- the cooling time.

3.1.1. Fuel Characteristics

The fuel characteristics may affect the decision on choosing an appropriate depletion code. E.g., when integral burnable absorber bearing fuel rods are present in a fuel design application of a two-dimensional depletion code certainly has an advantage over the use of a point-depletion code.

If the spent fuel management system of interest (such as a transport or storage cask, e.g.) is intended to accept different fuel designs, it has to be ensured that the calculated isotopic inventory is bounding with respect to the reactivity of the spent fuel management system considering all the fuel types.

Tolerances in the fuel design parameters shall be taken into account (cf. section 1). The tolerances in the pellet density and diameter are usually bounded by using the upper tolerance limit of the pellet density and by ignoring the dishing of the pellets. What regards the impact of the tolerances in the fuel rod outer diameter, the fuel rod pitch and the guide tube or water rod diameters (or water channel dimensions) on the spent fuel reactivity, less moderation results in neutron spectrum hardening and has therefore the effect of increasing the reactivity of the fuel during depletion and after that (see section 3.1.2). However, in a PWR core where soluble boron is present, more moderation results in the presence of more boron in the fuel assembly lattice, hence in spectrum hardening and therefore in a higher reactivity of the spent fuel. A higher moderation of the fuel in a spent fuel management system usually leads to a higher neutron multiplication factor of this system.

3.1.2. Parameters for the Depletion Analysis

The depletion conditions are characterized by the following parameters:

- specific power and operating history,
- fuel temperature,
- moderator temperature/density,
- presence of soluble boron in the core,
- presence of fixed neutron absorbers in form of control rods, burnable poison rods, axial power shaping rods,

- presence of integral burnable absorbers in the form of Gd or Er bearing fuel rods or IFBA rods (rods containing pellets with burnable absorber coating, e.g. B-10 coating).

These parameters will briefly be discussed in turn in the following sections with particular attention being given to the sensitivity of the neutron multiplication factor of the spent fuel management system of interest to variations in the parameters. Apart from the specific power and operating history effects all the other parameters are related to *neutron spectrum hardening*. Spectrum hardening results in an increased build-up rate of plutonium due to the increased neutron capture in U-238. The increased plutonium production concurrently leads to a decrease in the U-235 fission rate due to an increase in the plutonium fission rate and has therefore the effect of increasing the reactivity of the fuel at shut-down and after that.

3.1.2.1. Specific Power and Operating History

The production rate of each fission product is related to the specific power via the fission rate. The higher the specific power is, the higher is the production rate, the higher is therefore the equilibrium level of unstable nuclides where the decay rate approaches the production rate [10]. It was in fact observed [11] that calculations *with actinides plus fission products* show a slight downward trend in the neutron multiplication factor with increasing specific power level, whereas calculations on the *actinide-only* level show the opposite trend. And both trends increase with increasing burnup due to spectrum hardening.

An operating history is simply a time varying specific power profile. To understand the effect of different operating histories on the reactivity of spent fuel, model histories in the form of specific power histograms were developed to represent the key aspects of operating histories (e.g. high power operation early or late in the fuel's life, extended downtimes early or late in the fuel's life, short or long inter-cycle downtimes etc.), cf. Ref. [11]. In general, low power operation near end of cycle results in a higher reactivity worth of the fuel due to the decreased production rate of the fission products. And the opposite is true again when only actinides are considered for burnup credit.

However, all the effects studied, those which are related to the downtimes as well as those which are related to low power operations or higher power operations at end of cycle, are really very small: On the actinide plus fission product scale the relative change in the neutron multiplication factor k is not greater than 0.2% [11]. The conclusions are, therefore:

- Downtimes can be ignored in depletion calculations. This is conservative in case of actinides-only. In case of actinides plus fission products the impact on k is insignificant (i. e., negligible) for typical downtimes, and for extended downtimes in the late life ignoring of the downtimes is conservative.
- Usually full power can be assumed in the depletion analysis. Whether or not extended low/ high power operation should be included, that should be decided on the basis of the specific case of interest and the burnup credit level that will be chosen.

3.1.2.2. Fuel Temperature

Resonance neutron absorption in U-238 is increased at higher fuel temperatures due to Doppler broadening. This leads to spectral hardening and hence to a higher reactivity worth of the fuel. This effect increases with increasing burnup due to the fact that increasing burnup results in spectral hardening too.

Due to the fact that the impact of fuel temperature variations on the neutron multiplication factor remains fairly small (cf. [11] and [12]), not overly conservative estimating but reasonable bounding of the fuel temperature effect during depletion is recommended.

3.1.2.3. Moderator Temperature/Density

In a PWR, as the moderator temperature increases, the moderator density decreases. This leads to reduced moderation and hence spectral hardening. The effect on the neutron multiplication factor is significant (cf. [11] and [13]); a bounding temperature value must be used, therefore.

Due to the physics of an operating BWR the moderator temperature changes very little axially once the height were boiling begins is reached, but the moderator density significantly changes axially since the *void fraction* increases with increasing height. Due to the variations in the axial power peaking in an operating BWR the void fraction can change significantly both axially and as a function of time. It is obvious, therefore, that depletion effects have to be studied as a *function of moderator density or void fraction* instead of moderator temperature.

3.1.2.4. Presence of Soluble Boron

Neutron absorption by boron diluted in the PWR reactor coolant results in spectral hardening. Studies performed to assess the effect of soluble boron concentration used during depletion show a clear increase in reactivity with increased boron concentration, cf. [11], [13] and [14]. Due to the fact that in a real operating PWR the boron concentration decreases during the cycle due to the increasing core burnup, it has been found that use of the *cycle averaged boron concentration* in the depletion analysis results in a reactivity worth of the fuel which is at least bounding, if not even conservative, cf. [13] and [14].

3.1.2.5. Fixed Neutron Absorbers

Fixed poison rods are commonly used for reactivity control as well as enhanced fuel utilization. The net effect of such poison rods is the same as that of soluble boron, but is more localized and impacts, therefore, the axial and horizontal distribution of the burnup within the fuel assemblies. The effect on reactivity is significant [10], in particular when the rods are inserted for a long time.

In an operating BWR the use of control rods (or control blades) tends toward an increase in spectral hardening on the one hand, but results on the other hand in a reduction of local power and hence a decrease in the void fraction, which counteracts spectral hardening.

Burnable poison rods and axial power shaping rods are depleted usually in one cycle (mostly in the first third of the life of the assemblies). Therefore, the effect of these rods on the spent fuel reactivity is dependent on the fuel exposure prior to depletion of these rods, the subsequent burnup, and the possibly remaining poison loading within the rods. In any case, the effect on reactivity is significant, even after removal of these rods [10].

3.1.2.6. Integral Burnable Poisons

Integral burnable absorbers are also used for reactivity control and improved fuel utilization. The effect of integral burnable absorber fuel rods is similar to that of fixed neutron absorbers, but the integral burnable absorbers are depleted in the first third of the life of the fuel assemblies. The reactivity of the fuel assemblies may therefore increase with increasing burnup to a maximum and decrease after that. The effect of burnable absorber fuel rods on the spent fuel reactivity depends on their initial poison loading, their positions in the fuel assembly, the distribution of the poison within the pellets (pellets coated with poison or poison homogeneously mixed with the fuel), the burnup, and the possibly remaining poison loading within the rods [10].

The presence of integral burnable poisons results in spectrum hardening. However, spectral hardening not only leads to an increase of plutonium build-up and a delay to U-235 depletion but also results in a delay to the burnout of the integral burnable absorber, and this delay tends toward a decrease in the reactivity.

3.1.2.7. Impact of Specific Features of BWR Designs on the Spent Fuel Reactivity

In a BWR the issue is complicated further by the fact that the presence of the integral burnable absorber results in a decrease of the local power and thus leads to a local decrease in the void fraction and hence a reduction of the spectral hardening.

In addition, things are complicated by the fact that *partial length fuel rods* are usually present in modern BWR designs, and *axial zoning* of the enrichment and the integral burnable absorber loadings is often used. Moreover, in some modern designs the fuel rods differ in outer diameter and rod pitch.

The absorber loading of the integral burnable absorber bearing fuel rods is often varied horizontally within a fuel assembly, whereas *the initial enrichment is* always *varied horizontally* in order to achieve lower gradients in the thermal neutron flux distribution within the fuel assembly and to maintain thus the limitations of the local power peaking factors (form factors) needed for avoiding dry out. An *inhomogeneous* enrichment distribution usually results, just because of these objectives, in a more or less slightly lower neutron multiplication factor of a spent BWR fuel management system of interest than a homogeneous distribution with the same average enrichment. But the enrichment distribution depends on the core lattice type and one therefore has to take account of the sensitivity of the spent fuel management system's neutron multiplication factor to different spatial orientations of neighboring fuel assemblies.

3.1.3. Cooling time

Promptly after reactor shutdown, Xe-135 builds up thereby reducing the reactivity of the spent fuel, but due to the rapid decay of I-135 and Xe-135 the reactivity of spent fuel will increase within the first four days after shutdown. In the same time, there is an accumulation of Pu-239 due to the decay of Np-239. The maximum Pu-239 inventory is at about 20 days after shut-down (half-life of Np-239 is about 2.5 days). Four days after shutdown, the production of strongly absorbing nuclei like Sm-149 through β-decay of Pm-149 leads to a slight decrease in reactivity. After about 100 days, the production of Gd-155 through β-decay of Eu-155 and the production of Am-241 through β-decay of Pu-241 become predominant and cause a stronger reactivity decrease for time periods covering all spent fuel management systems with the exception of disposal.

For periods of times which are to be considered for disposal, it has to be taken into account that the half-life of Pu-238 and Pu-240 as well as of Am-241 are significantly shorter than the half-life of Pu-239, and that Pu-239 decays via α-decay to U-235. Thus the reactivity increases after a time of about 100 years until a peak is reached around 30 000 years. The minimum reactivity at about 100 years after shut down depends on the burnup and the initial enrichment of the fuel because the U-238 capture rate decreases with increasing enrichment.

Due to the production and decay processes after irradiation, not only the neutron multiplication factor of a spent fuel management system changes with cooling time, but also its bias arising from the uncertainties in the depletion calculations [15]. This has to be taken into account when credit is taken for some cooling time.

3.2. Criticality Calculation and Evaluation of the Loading Criterion

The criticality calculation procedure consists of the following main steps:

- isotopic selection and validation,
- validation of the criticality calculation code applied,
- sensitivity studies on the reactivity effects of variations and tolerances in the parameters describing the characteristics of the spent fuel management system of interest,
- sensitivity studies on the reactivity effects of axial and horizontal burnup profiles,
- determination of the criticality safety acceptance criterion, estimation of the loading criterion, and analysis of abnormal and accident conditions.

Some of these steps will be briefly discussed in the following sections with particular attention focused on effects, which may impact the reactivity worth of the spent fuel under the conditions of the spent fuel management system of interest.

3.2.1. Isotopic Selection and Validation

The decision about which isotopes are to be used to represent the spent fuel reactivity is strongly related to the *decision on the burnup credit level* to be applied. The decision depends on the spent fuel management system to be analyzed and the accuracy that can be attained in the validation of the isotopic inventory estimated and the criticality calculation code used.

The isotopes to be used in a criticality calculation should be 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, [10], [16], [17].

Isotopic validation can be achieved

- either by correcting the isotopic concentrations with the aid of correction factors derived from comparisons with chemical assay data [11]
- or by estimating the bias in the neutron multiplication factor of the spent fuel facility of interest due to the bias of the concentrations of the individual isotopes used.

Whilst the application of correction factors to individual isotopes within a predicted isotopic inventory can lead to analyses with potential over-conservatism (see however also section 3.2.4.1), the second way to achieve validation is more instructive since it includes studies on the sensitivity of the facility’s neutron multiplication factor to variations in the concentrations of the individual isotopes. A big isotopic correction factor is of less importance when the change in the reactivity worth of the isotope obtained by applying this correction factor is insignificant under the conditions of the facility analyzed.

The reactivity worth of the isotopes is affected by the depletion conditions as well as by the characteristics of the spent fuel management system analyzed. Depletion conditions resulting in spectrum hardening cause not only an increased build-up rate of plutonium and a decrease in the U-235 fission rate due to an increase in the plutonium fission rate, but also a decrease in neutron absorption in the fission products. Therefore, bounding depletion conditions usually results in a higher fission product concentration (in particular in a higher Sm-149 content) than more realistic conditions. Nevertheless, the isotopic inventory obtained under the bounding conditions – and this makes these conditions bounding - results in a higher spent fuel reactivity due to the increased build-up of plutonium and the decrease in the U-235 fission rate due to an increase in the plutonium fission rate.

The use of neutron absorbers in a spent fuel management system usually results in spectral hardening and hence in less neutron absorption in the fission products in particular. The relative reactivity worth of the neutron absorbing (non-fissile) actinides and the fission products of an isotopic composition is thus dependent on the system's characteristics.

3.2.2. Criticality Calculation Validation

Available experimental data include ([10], [16] and [18])

- critical experiments with fresh fuel in systems similar to spent fuel configurations of interest,
- reactivity worth measurements, and
- reactor critical configurations.

The fact that there is no critical or subcritical experiment using commercial spent fuel in a configuration of interest (e.g. cask configuration) is certainly one of the main reasons why, to date, the application of the actinide plus fission product burnup credit level is restricted to PWR wet storage ponds [7].

3.2.3. Characteristics of the Spent Fuel Management System of Interest

It was already stated that the presence of neutron absorbers in the spent fuel management system might affect the reactivity worth of the spent fuel due to spectral hardening. It has to be added that, vice versa, spectral effects due to plutonium and fission product build-up may reduce – when compared to the fresh fuel case – the efficiency of the neutron absorbers used in the system of interest.

The variations and tolerances in the parameters describing the characteristics of the spent fuel management system shall be considered (cp. section 1). This can be done by

- either constructing the worst case (which has to be confirmed by analysis)
- or by statistical analysis of the impact on the neutron multiplication factor of the system.

A statistical analysis can be performed if and only if the real distribution of the deviations of the actual values of a parameter under study from the nominal value of this parameter is known.

3.2.4. Axial and Horizontal Burnup Profiles

3.2.4.1. PWR Axial Burnup Profiles

In a fresh fuel assembly the axial flux distribution is more cosine shaped. The fuel near the axial center of the assembly is therefore depleted at a faster rate than at the ends. With increasing burnup the flux shape flattens out due to the higher fuel depletion and fission product poisoning near the center. However, due to the neutron leakage at the ends of the fuel zone the burnup drops off rapidly near the ends.

Axial burnup shapes are mostly asymmetric due to the higher moderator density in the lower half of the core [13]. The asymmetric component is strongly dependent on the average burnup of the fuel assembly and strongly affected by control rod movements, use of axial power shaping rods, presence of integral burnable poisons as well as reload patterns used (determining the interactions between fresh fuel assemblies and fuel assemblies with different burnup shapes at begin of cycle).

The fact that the axial burnup distribution is non-uniform must be taken into account in the criticality safety analysis of a spent fuel system of interest. The difference between the system's neutron multiplication factor obtained by using an axially varying burnup profile and the system's neutron multiplication factor obtained by assuming a uniform distribution of the averaged burnup of this profile is known as the "*end effect*". It has been shown that the end effect may become positive for averaged burnups greater than about 15 MWd/kg U, cf. [13], [19], [20]. In view of the vast variety of axial profiles due to the vast variety of irradiation histories that are occurring already in one core alone, the task is given to find a bounding description of the end effect, [13], [21].

Since the fission product concentration is higher in the center zone, application of the actinide-only burnup credit level results in an underestimation of the end effect because the reactivity worth of the center zone is more overestimated than the reactivity worth of the end zones.

Likewise, since the fission product concentration as well as the plutonium concentration is higher in the center zone, application of isotopic correction factors (cp. section 3.2.1) may result in an inadvertent underestimation of the end effect. This is an additional rationale to prefer studies on the sensitivity of the spent fuel system's neutron multiplication factor to variations in the concentrations of individual isotopes.

Since the plutonium and fission product concentration is higher in the center zone, the reactivity of this zone decreases faster with cooling time than the reactivity of the end zones. The end effect therefore increases with cooling time for cooling periods less than about 100 years.

In addition, due to spectral effects the end effect is affected by the characteristics of the spent fuel management system of interest. It is therefore obvious that *the end effect has to be evaluated under the conditions of the specific system*.

3.2.4.2. PWR Horizontal Burnup Profiles

Radial variations in the neutron flux in an operating reactor, which are mainly due to leakage at the core periphery and to burnup differences between neighboring assemblies, result in a non-uniform horizontal burnup distribution over the radial extent of the core. It has been shown (cf. [13], e.g.) that non-uniform horizontal burnup profiles might result in a slight increase of the neutron multiplication factor of the spent fuel management system of interest. The magnitude of this increase is affected, due to spectral effects, by the characteristics of this system and depends on the fuel assembly type. Thus, finding a bounding description of the impact of horizontal profiles on the system's reactivity is one of the tasks that have to be done in the criticality safety analysis of the system.

3.2.4.3. BWR Burnup Profiles

In view of the specific BWR designs features described in section 3.1.2.7 and in view of all the effects specific to an operating BWR (cf. sections 3.1.2.3, 3.1.2.5, and 3.1.2.6), it is an extremely challenging task to find a real irradiation history that bounds all other irradiation histories, i.e., which leads to the highest neutron multiplication factor of the spent fuel management system of interest. Things are additionally complicated by the fact that various fuel designs are usually present in the same BWR core.

Burnup credit criticality safety analysis of BWR spent fuel is therefore restricted to application of the *integral burnable absorber burnup credit level*. For a given fuel design it is assumed that the burnup is uniformly distributed both axially and horizontally, and the neutron multiplication factor k_{eff} of the spent fuel management system of interest is analyzed as a function of burnup at given void fractions. The objective is to find the void fraction value and the burnup value which together result in the highest value $(k_{eff})_{max}$ of the system's neutron multiplication factor, cp. Ref. [22].

3.2.5. Analysis of Abnormal and Accident Conditions

The loading criterion for a spent fuel management system designed for burnup credit is usually given in the form of a loading curve. Due to the increase of spectral hardening with increasing burnup it might be necessary to base the analysis of abnormal or accidental conditions on the upper end-point of the loading curve because the efficiency of neutron absorbers used in the system decreases with increasing spectral hardening, cf. Ref. [23].

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Comparative analysis of multiplicating properties of WWER-1000 spent fuel depending on assembly layout in the reactor core and their operation conditions

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Abstract. Spent nuclear fuel with the same burnup value can have different isotope composition, and, therefore, different multiplicating properties. This work has analyzed the impact on WWER-1000 spent fuel multiplicating properties of different operation conditions, such as the presence or absence of absorbers-rods in an assembly, changes in the concentration of the boric acid dissolved in the moderator (water) during the campaign, oscillations of fuel and/or moderator temperature during the campaign in different areas of the core, and changes in water amount at the periphery of an assembly due to its location in the central or periphery part of the core during the fuel campaign and/or due to changes in inter-assembly gaps.

1. INTRODUCTION

The use of the burnup as a nuclear safety parameter in assessment safety of spent fuel management systems (burnup credit principle) can be divided into three main stages. There are:

- determination of fuel burnup;
- determination of fuel isotope composition depending on its burnup;
- determination of multiplicating properties of spent fuel depending on its isotope composition.

The isotope composition of spent fuel depending on its burnup is determined, as a rule, by means of reactor cell programs, such as CASMO, HELIOS, WIMS, NESSEL, KASSETA, etc., or by means of specialized calculation programs for spent fuel isotope composition such as ORIGEN, NUKO, etc.

It is evident that the isotope composition of spent fuel is determined not only by its burnup value, but also by burnup conditions, in other words by that neutron spectrum in which this burnup took place.

Let us separate the main factors, which can affect changes in the neutron spectra during the campaign, and along with it changes in spent fuel isotope composition.

1) The presence of absorbers in an assembly, such as rods of burnable absorbers or control rods (CR) clusters.
2) The change in the concentration of the boric acid dissolved in the moderator (water) during the campaign.
3) Oscillations of fuel and/or moderator temperature in different areas of the core during the campaign.
4) Changes in water amount at the assembly periphery due to its location in the central or periphery part of the core during the fuel campaign and/or due to changes in inter-assembly gaps.

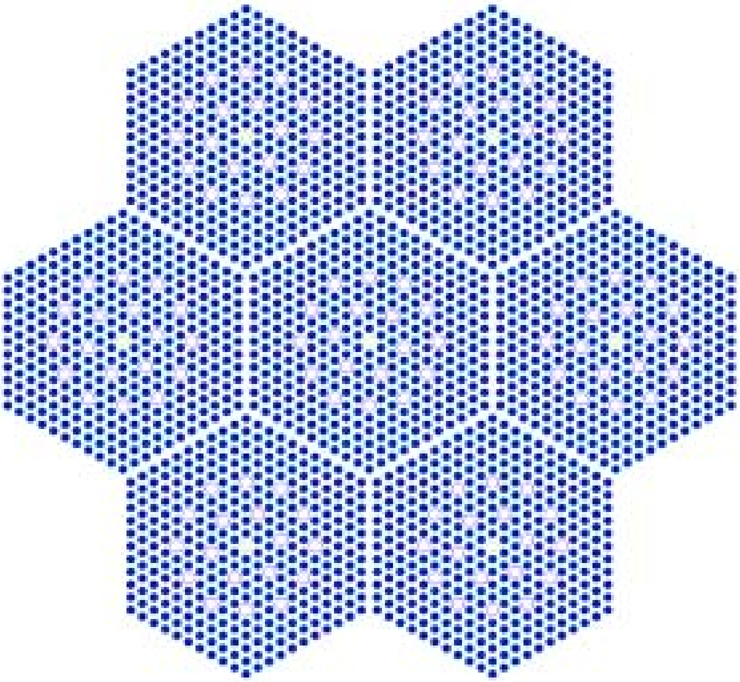

FIG. 1. WWER-1000 reactor cell.

The calculations have been conducted on an example of the FA with the maximum multiplicating properties. Taking into account the manufactory tolerances, WWER-1000 reactor FA with 4.45-% enrichment and fuel (UO_2) mass 460.02 kg/FA was selected as such FA, figure 1.

The isotope composition of spent fuel has been determined for rated (or normal) operation parameters of WWER-1000 reactor.

All calculations have been performed by the program NESSEL developed by a firm for calculating WWER reactor cell.

Figure 2 presents the calculation results for multiplicating properties of WWER-1000 reactor cell mesh under the rated operation parameters of the reactor for several possible conditions:

- when there are no additional absorbers in the FA (f445h2o curve);
- when burnable absorber rods with different boron concentration are located in Control and Protection System (CPS) guide tubes (f445ba20, f445ba36, and f445ba65 curves);
- when CR clusters are located in CPS guide tubes (f445cl curve).

The behavior of the curves is completely explainable and logical. The impact of the burnable absorbers on FA multiplicating properties becomes negligibly small at the level in 20 GW*day/t approximately, CR clusters work in the whole burnup range, at that, their effectiveness slowly decreases due to boron burnup.

In following section multiplicating properties of spent FA have been determined for infinite layout of FA located with 23.6-cm pitch in unborated water with temperature 293 K (without any absorber rods in CPS guide tubes). Fuel temperature was also supposed 293 K, Figure 1. This conditions was denoted in following as storage conditions.

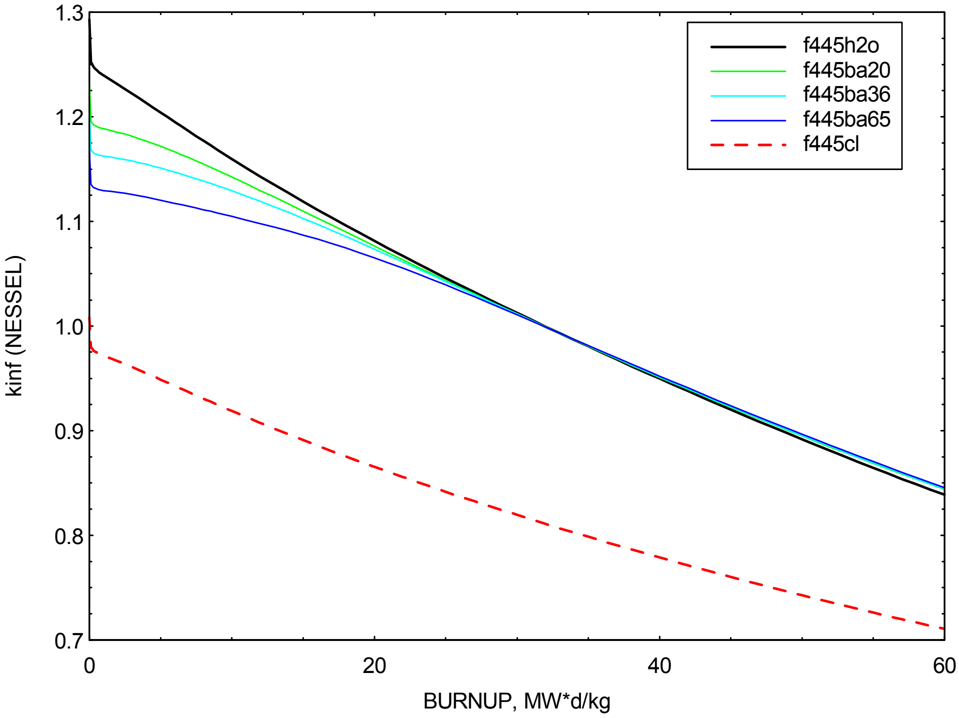

FIG. 2. Multiplicating Properties of WWER-1000 Reactor Cell under Rated Reactor Operation Parameter.

2. THE PRESENCE OF ABSORBERS, SUCH AS BURNABLE ABSORBER RODS OR CR CLUSTERS, IN THE FUEL ASSEMBLY

Based on the calculations of WWER-1000 reactor cell burnup under the rated reactor operation parameters, the isotope composition of spent fuel has been determined in the cases when water, burnable absorber, and CR clusters are in CPS guide tubes. Based on this isotope composition, multiplicating properties of the reactor mesh have been calculated in storage conditions (Tf = Tmod = 293 K, unborated water, no removable absorbers, such as burnable absorber rods or CR cluster). The results of these calculations are given in Figure 3, three lower curves (nes445h2o, nes445h2o(ba65), nes445h2o(cl)).

It is evident that multiplicating properties of spent fuel as applied to storage conditions should be calculated without credit of short-lived isotopes. Therefore, in the next series of these calculations we did not credit the impact of the isotopes associated with lines Xe135 on multiplicating properties of the reactor cell, three middle curves in Figure 3 (nes445h2os, nes445h2os(ba65), nes445h2os(cl)).

The last series of the calculations, three upper curves in Figure 3 (nes445h2os(10is), nes445h2os(ba65)(10is), nes445h2os(cl)(10is)), presents the results of spent fuel multiplicating property calculations which have been performed with the credit of changing during burnup in the concentration of 10 isotopes that are used in the "burnup credit" methodology the most frequently – there are basic fuel isotopes U235, U236, U238, Pu239, Pu240, and Pu241, some actinides and fission products such as Pu242, Am241, Sm149, and Sm151. In all cases the spent assemblies whose burnup took place with the presence of CR clusters have the maximum multiplicating properties.

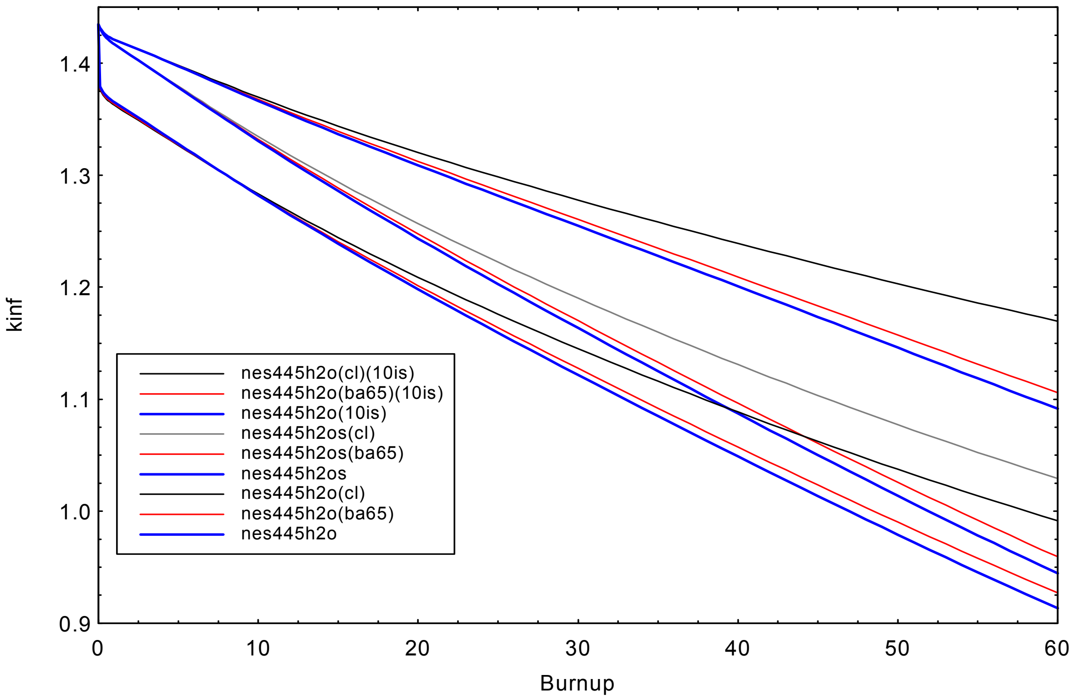

FIG. 3. Multiplicating Properties of WWER-1000 Reactor Cell under Storage Conditions.

3. COMPARISON OF THE NESSEL CODE WITH THE CASMO-4 AND SCALE-4.3 CODES

The possibility of comparing the results obtained with the similar ones obtained applying other codes is emerged already at this stage of calculations.

Figure 4 demonstrates the comparison of the results of the calculations presented in Figure 2 by f445h2o curve with the similar results obtained by means of the well-known code CASMO-4 (performed within the framework of BMU-project SR 2331, German).

The close coincidence of the results by neutron multiplication factor can be considered as the integral characteristic, which demonstrates the close coincidence in these calculations of the changes during burnup in concentrations of the basic isotopes affecting assembly multiplicating properties.

Figure 5 demonstrates the comparison of the results of the calculations presented in Figure 3 by three upper curves (nes445h2os(10is), nes445h2os(ba65)(10is), nes445h2os(cl)(10is)) with the similar results obtained by means of the well-known Monte-Carlo SCALE-4.3 (KENO-VI) code.

It is evident that the multiplication factor result discrepancies have the systematic nature, and is 0.015 under all burnup values (from 0 to 60 GW*day/t), and for different possible types of neutron spectra (the presence of burnable absorbers, CR clusters, or their absence), in the range of the multiplication factor from 1.0 to 1.5. These results completely confirm the applicability of the code NESSEL for such studies, at least as regards the comparative analysis of different possible variants of changes in WWER-1000 assembly multiplicating properties during fuel burnup.

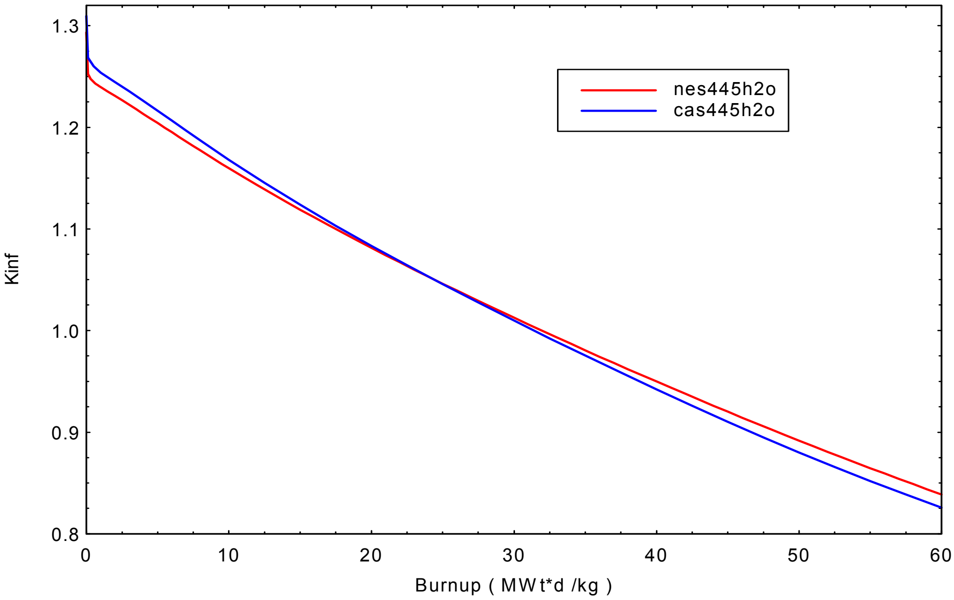

FIG. 4. Comparison of the Calculation Results Obtained with the NESSEL and CASMO-4 Codes.

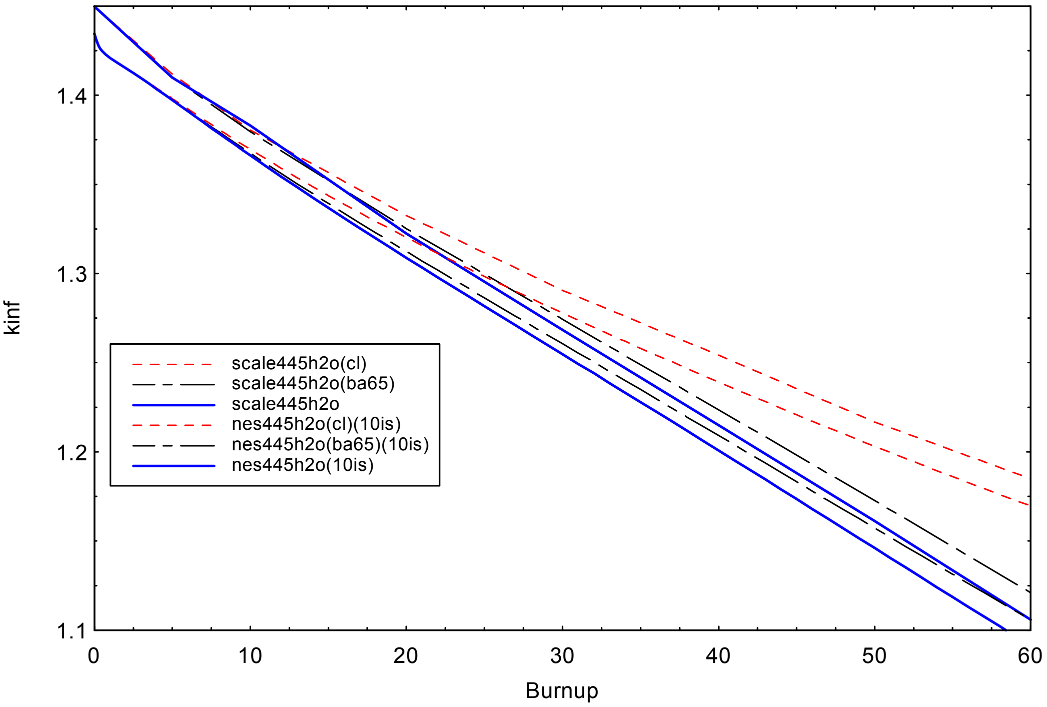

FIG. 5. Comparison of the Calculation Results Obtained with the NESSEL and SCALE-4.3 Codes.

4. THE CHANGE IN THE CONCENTRATION OF THE BORON ACID DISSOLVED IN THE MODERATOR (WATER) DURING THE CAMPAIGN

In the calculations whose results are presented in the previous sections the isotope composition in burnup fuel has been determined for the 525 ppm value of the boric acid concentration, which is average by the reactor campaign. The concentration of the boron acid dissolved in the moderator (water) is changed during the WWER-1000 reactor campaign in the range from 1 050 ppm at the beginning of the campaign to approximately 0 ppm at the end. Presently, each assembly passes 3 – 4 such cycles during operation. To determine the impact of these oscillations on spent fuel multiplicating properties the calculations of changing during burnup in the fuel isotope composition have been performed when the concentration of the boron acid dissolved in water is 0 and 1 050 ppm (with CR clusters and without them). Multiplicating properties of the assemblies in storage conditions have been determined based on this isotope composition data. The results of these calculations are presented below in Figure 6.

Two lower solid lines characterize the spent fuel multiplicating properties in storage conditions (without credit of Xe135 lines) for two cases when the fuel burnup took place without CR clusters (the lower curve) and with them. Two upper solid lines correspond to the same two cases but with the credit of changing during burnup in 10 the above-listed isotopes. The solid lines in all four cases correspond to the fuel isotope composition obtained for the average boron acid concentration – 525 ppm . The lower dot lines have been obtained for the case of fuel burnup in unborated water, and the upper ones correspond to the boron acid concentration in the moderator – 1 050 ppm .

Hence, the spent assemblies whose burnup took place with the presence of CR clusters, in water with the maximum boron acid concentration (in our case – 1 050 ppm) have the maximum multiplicating properties.

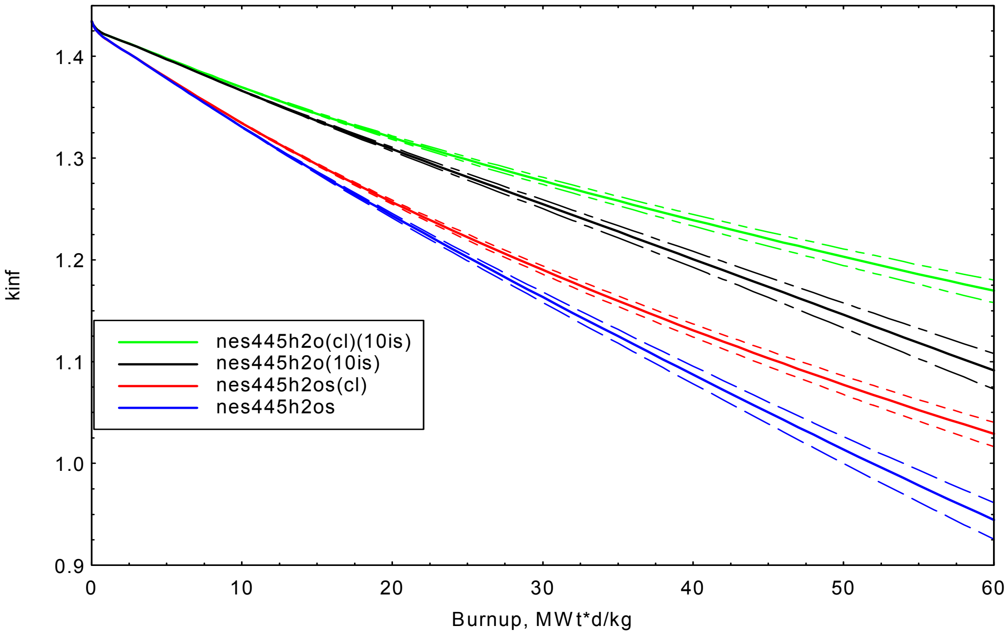

FIG. 6. Multiplicating Properties of WWER-1000 Reactor Cell in Storage Conditions for Different Boron Acid Concentrations During Fuel Burnup.

5. THE CHANGE IN FUEL AND/OR MODERATOR TEMPERATURE

The calculation studies performed demonstrated that the oscillations of average temperature of fuel in the range from 900 to 1 100 K and moderator temperature in the range from 540 to 600 K do not impact, within the accuracy of the program applied in the calculations, on fuel multiplicating properties depending on its burnup.

6. THE CHANGE IN WATER AMOUNT AT THE PERIPHERY OF AN ASSEMBLY DUE TO ITS LOCATION IN THE CENTRAL OR PERIPHERY PART OF THE CORE DURING THE FUEL CAMPAIGN AND/OR DUE TO CHANGES IN INTER-ASSEMBLY GAPS

The location of an assembly can be changed during the fuel campaign. The assembly can be located both in the core periphery and its central part. Furthermore, great attention is paid during the last years to the effect of assembly deformation during their operation. Both these events lead to changing the amount of water around the assembly, and therefore, to changing neutron spectrum in the assembly.

To study the impact on spent fuel multiplicating properties of the water amount at the assembly periphery during its operation the isotope composition has been determined for the case of operation of assemblies with CR clusters and without them when the thickness of the surrounding water layer is changed from –2 mm to +2 cm in relation to the rated value. The boron acid concentration in the moderator is assumed equal to 1050ppm.

The results obtained are presented in Figures 7 and 8. As in the previous cases, spent fuel multiplicating properties have been determined for the assemblies operated without absorber rods (Figure 7) and with CR clusters (Figure 8), with the credit of the complete change of its nuclide composition, except for lines Xe135 (the lower bundle of curves in both Figures), and with the credit of only 10 the most significant isotopes. In both Figures the solid lines and the dot lines adjoining them from above and below have been taken from Figure 6 and correspond to different boron acid concentrations in the moderator – 525 ppm (the solid lines), and 0 and 1 050 ppm (the lower and upper dot lines appropriately).

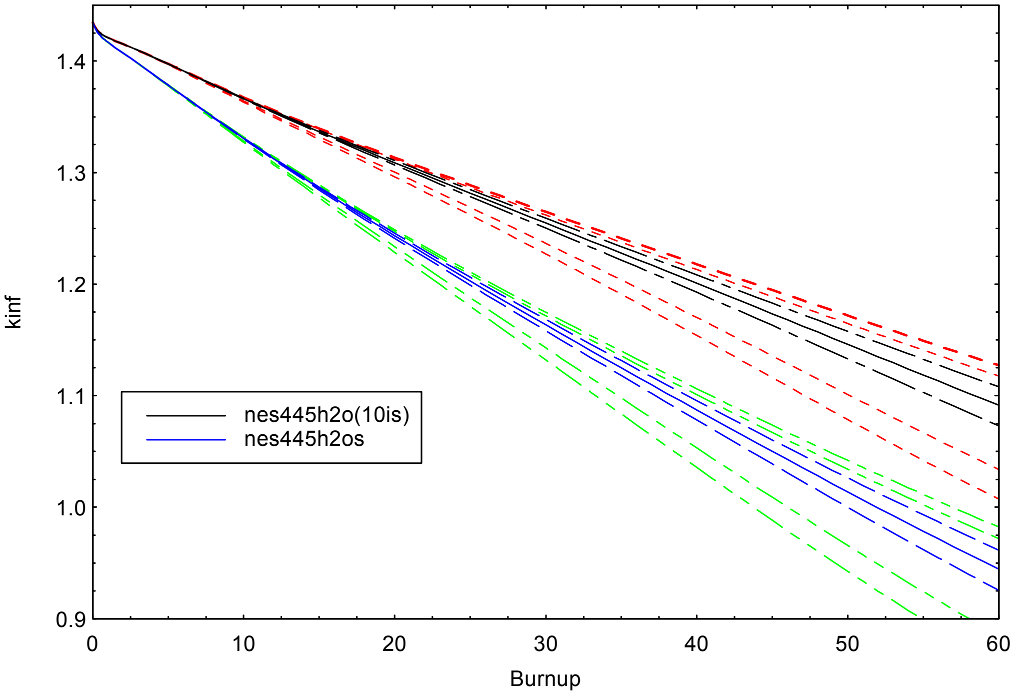

FIG. 7. Multiplicating Properties of WWER-1000 Reactor Cell in Storage Conditions for Different Boron Acid Concentrations during Fuel Burnup and Different Water Amount at the Assembly Periphery.

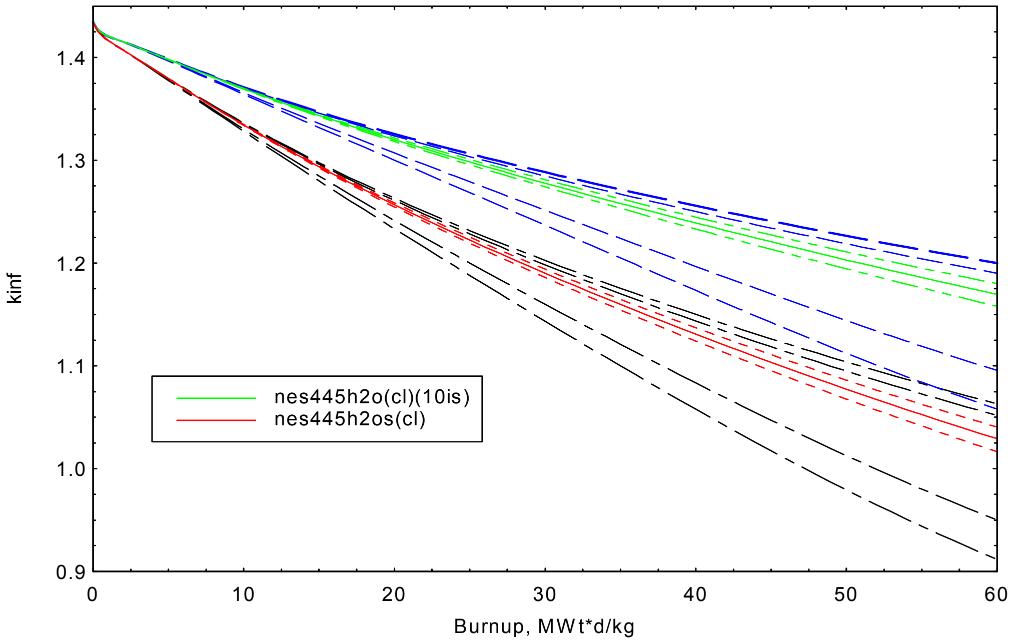

FIG. 8. Multiplicating Properties of WWER-1000 Reactor Cell in Storage Conditions during Fuel Burnup and Different Water Amount at the Assembly Periphery with Credit of CR Cluster Presence in the Assembly.

Then, in each set of curves two lower dot lines correspond to increasing the water layer around the assembly in 1 and 2 cm. Two upper dot lines correspond to decreasing the water layer around the assembly in 1 and 2 mm in relation to its rated value.

Hence, the spent assemblies whose burnup took place with the presence of CR clusters, in water with the maximum boron acid concentration (in our case – 1 050 ppm), under the minimum water layer thickness around the assembly (in our case it is –2 mm in relation to the rated value) have the maximum multiplicating properties.

7. CONCLUSIONS

The studies conducted demonstrated that WWER-1000 spent assemblies with 4.4% enrichment have the maximum multiplicating properties at all levels of the burnup range from 0 to 60 GW*day/t provided that during the operation:

- Absorber rods of CPS clusters are located in assembly guide tubes;
- Fuel burnup takes place in the moderator with the maximum boron acid concentration;
- Water layer around the assembly has the minimum thickness.

This conclusion is correct both for the case of determining spent fuel multiplicating properties in the whole isotope spectrum (except for lines Xe135) and for the case when multiplicating properties are determined only taking into account the bounded number of the most significant nuclides (U235, U236, U238, Pu239, Pu240, Pu241, Pu242, Am241, Sm149, and Sm151).

Practical issues with implementation of burnup credit in the USA for storage and transportation

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Abstract. The US NRC issued an interim staff guidance (ISG8 rev1) allowing for burnup credit applications for storage and transport casks in July of 1999. In over two and a half years there has still not been a license submittal using burnup credit. ISG8 rev1 does not provide sufficient burnup credit to allow loading of 5 wt% enriched fuel in a 32 PWR assembly cask without the addition of absorber rod inserts. Pressure to allow all assemblies to contain inserts from the utility, force continued investigation into alternative levels of burnup credit. Utilities do not wish to measure to confirm burnup. This measurement costs, which range form $10 000 to $50 000 per cask and must be done prior to loading. Since burnup credit is actually only needed for transport, and transport is not expected for many years, many utilities are considering keeping the money in the bank until the time of transport. In order to address the need perceived for additional burnup credit beyond actinide-only burnup credit (ISG8), investigations have moved beyond into assuming moderator exclusion during transport and the use of burnup credit to cover a beyond design basis accident assumption of flooding. Burnup credit analysis requirements for a beyond design basis accident should be less than that for criticality control for normal operation. It is proposed that burnup credit analysis to cover the beyond design basis accident of flooding should be consistent with the beyond design basis dilution event in PWR spent fuel pools. The US NRC precedence for this type of burnup credit allows for all isotopes, a 5% reduction in the delta k of burnup, and an allowable k_{eff} of less than 1.0 after biases and uncertainties.

INTRODUCTION

In criticality licensing of spent nuclear fuel containment systems, taking credit for the decrease in fuel reactivity due to incore burnup is known as burnup credit. Burnup credit has been successfully used in spent fuel pool criticality licensing and has been used as a loading criterion in dry fuel storage in the USA. It has also been used in Europe. Burnup credit has not yet been licensed for transport of commercial spent nuclear fuel in the USA. There has been much work performed to allow application for burnup credit for transport [1,2] but no license applications have been submitted by the time of this writing (April 2002).

The current position of the US NRC is given in ISG8 Rev.1 [2]. This position gives credit only for the change in actinide concentration and requires conservatism in a number of independent steps. ISG8 Rev.1 rest heavily on the limited clean experimental data of critical experiments using MOX and fresh UO_2 fuel and on chemical assays of spent nuclear fuel to justify the isotopic content of fuel as a function of burnup.

Unfortunately, “actinide-only” burnup credit appears to be insufficient for fuel that is being placed in the cores today. Figure 1 shows a loading curve for a typical 32 assembly transport cask. Although the loading curve is acceptable for much of the older fuel with enrichments of about 3.2 wt% U-235, it is clear that it does not cover fuel with initial enrichments of greater than 3.6 wt%. Refinements in the method may allow loadings of up to 4.0 wt% but then significant changes in methods would have to be considered. The current solution to the loading curve problem is to add absorber rod inserts into the fuel assemblies. The number of assemblies that would need the these inserts is fairly limited, 4 to 16 of the 32 assemblies in the cask. These add cost and decrease the number of available positions for disposal of the utilities spent fuel inserts such as burnable absorbers. An alternative solution that has been investigated is to derate the capacity of the cask from 32 assemblies to 30, 28, or 24 assemblies[3].

Current US Cask Vendor Positions

Although no cask vendor has yet submitted a burnup credit application to the US NRC, two of the vendors, Holtec and TN, are expecting submittals shortly. Both of these vendors are currently not planning on following an actinide-only approach but rather plan to utilize fission products in their burnup credit applications. Other cask vendors with burnup credit plans, GNB, GA, and NAC, will follow Holtec and TN and expect to benefit from NRC comments prior to the need to commit to their burnup credit methods.

Both Holtec and TN believe that addition of control inserts to cover for inadequate burnup credit is not acceptable.

Current US Utility Positions on Burnup Credit

Although a number of utilities have already purchased burnup credit dual purpose canisters, it is not clear they intend on performing a burnup credit verification measurement prior to loading the canister. The burnup up verification measurement is required by the US NRC in ISG8 and is also found as a requirement in IAEA ST-1. The utilities project the cost of measurement to be up to \$50 000 per cask. (Measurement experts suggest that a much lower cost of about \$10 000 per cask would be reasonable.) Since the utilities only need burnup credit for transport, and since transport is not expected for about 20 years, the utilities have suggested that it may not be cost effect to measure now even if it is required 20 years from now. If \$50 000 were to be compounded for 20 years at 10% per year, this would be greater than \$350 000. \$350 000 would be sufficient to pay to return the cask to the pool, open it up, and perform the measurement when needed. But in 20 years the measurement may not be required.

The Law Allowing Moderator Exclusion

The law in 10CFR71.55b requires that the criticality analysis for a transport cask assume that the cask is flooded with pure water. However, 10CFR71.55c states:

"(c) The Commission may approve exceptions to the requires of paragraph (b) of this section if the package incorporates special design features that ensure that one single packaging error would permit leakage, and if appropriate measures are taken before each shipment to ensure that the containment system does not leak."

Many of the cask designs now have double lids so it should be possible to meet the "special design feature" requirement of the law. The cask systems also generally have two independent containers (the canister and overpack) or a thick walled cask body.

Leak tests are already generally performed prior to shipping.

The casks filled with commercial fuel but no water are very subcritical. One could argue that no burnup credit is needed. However, it is may be a reasonable position to consider a beyond design basis accident of flooding with water since this would allow fuel loading licensing requirement to be consistent with the fuel pool licensing requirements.

Burnup Credit with Moderator Exclusion

Currently, PWR spent fuel pools are subcritical due to the dissolved boron in the pool. They are however required to show that if there were no dissolved boron there would still not be a criticality. [4] The burnup credit proposed here uses the same type of argument. It will be required to show that if the cask were flooded with pure water it would not be critical. Current spent fuel pool analysis allows for the criticality criteria a k_{eff} of 1.0. This is often referred to as boron credit of about 500 ppm. For transport burnup credit using moderator exclusion, this proposal assumes the criticality criterion is k_{eff} must also be less than 1.0.

The burnup credit analysis for transport burnup credit should follow the methods currently used for spent fuel pools. [4] This method is known as full burnup credit and uses all the isotopes in the spent nuclear fuel.

Figure 1 shows the results of this method applied to the same typical 32 assembly cask. As can be seen on Figure 1, fuel with a reasonable discharge burnup can be placed in the cask without addition of absorber inserts or derating the cask.

Discussion and Conclusion

In order for moderator exclusion with burnup credit for the beyond design basis accident to be applicable the cask cannot be loaded or unloaded in an unborated pool. This clearly eliminates BWRs. It also requires that Yucca Mountain allow for dry unloading of the cask.

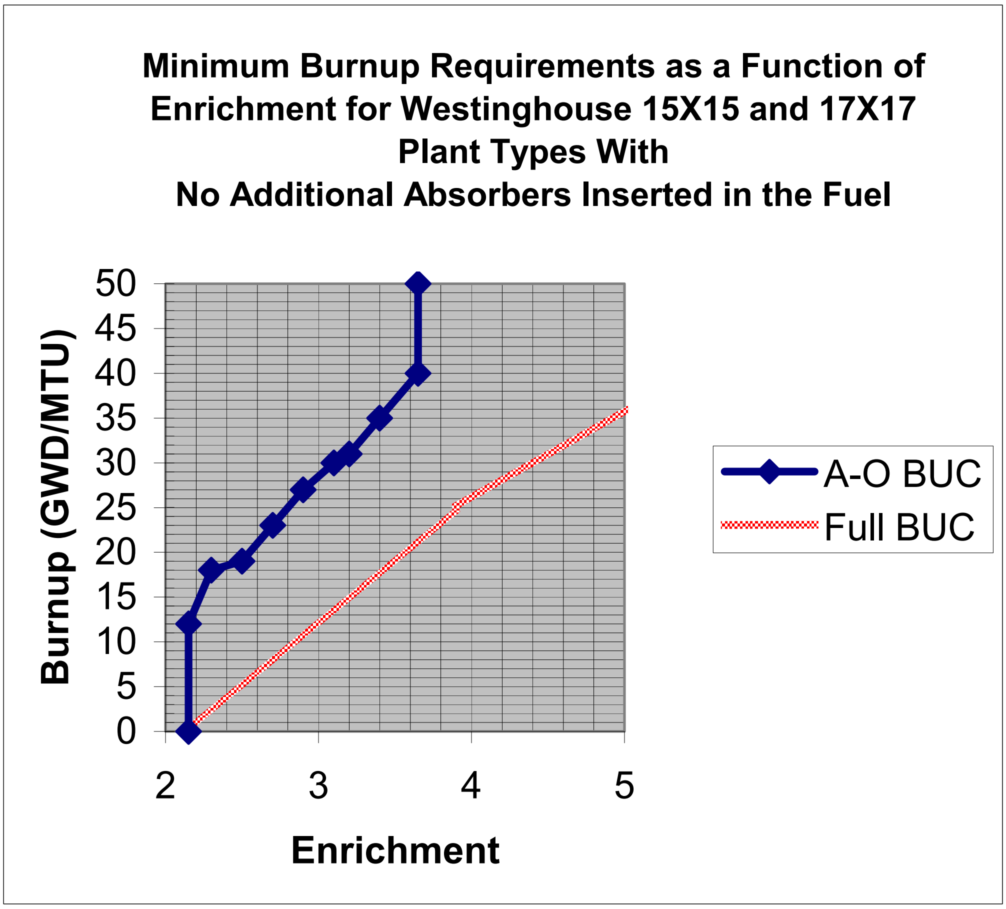

FIG. 1. Loading curve for a typical 32 assembly transport cask using Actinide-Only Burnup Credit (A-O BUC) or Moderator Exclusion with Burnup Credit.

Moderator exclusion with burnup credit does not utilize burnup credit as its main method of being subcritical. Therefore, the fuel does not require a burnup verification measurement prior to shipment as required by the NRC and IAEA. This is consistent with current use of burnup credit in spent fuel pools.

From a risk informed licensing point of view, moderator exclusion with burnup credit has a small but positive benefit. Moderator exclusion with burnup credit requires less fuel handling than current burnup credit approaches since no burnup measurement is needed and no additional inserts are needed. Less fuel handling is a small safety benefit along with a small reduction in operational radiation exposure. The risk of flooding is extremely low. The recent study of the risks associated with transport [5] shows that the risk of train accident resulting in the cask landing in significant water to be about $2*10^{-6}$ per shipment. Further, it is clear that impact with water would not be a reasonable mechanism to fail the cask. Also remember that if the cask is flooded, it will not go critical but have a best estimate k_{eff} of less than 0.95.

Casks that are expected to be unloaded wet in unborated pools as normal operations should require more margin in their burnup credit method than cask that are expected to be dry unloaded. For this reason, moderator exclusion plus burnup credit may not apply to some countries.

In conclusion, moderator exclusion with burnup credit is a safe approach for licensing transport casks. Criticality is not credible with this approach of belts and suspenders.

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SAFETY MARGIN, BIAS, UNCERTAINITY AND STATISTICAL CONFIDENCE

(Session 2.3)

Risk, confidence, tolerance and bias — brief outlines of the basic concepts

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Abstract. The proof of sufficient subcriticality of a fuel management system under study consists in estimating the neutron multiplication factor of the system and all the uncertainties of this factor and applying then a criticality safety acceptance criterion to the calculational results in order to reach a decision about whether or not it is acceptable that the *true* value of the neutron multiplication factor is below the upper bound usually laid down by criticality safety standards. In the paper on hand a basic approach to decision theory is therefore given. The decision problems in estimation of a parameter value, estimation of a confidence interval for this parameter, and testing of hypotheses about the true value of a parameter are pointed out. It is exemplified that for these problems optimum solutions exist which minimize the risk of having made a wrong decision. And it is shown, that safety acceptance criteria are based on such solutions. The paper on hand is intended for criticality safety analysts who lack deeper knowledge of statistics. The paper on hand outlines the basic concepts, therefore. This is done from the point of view of the modern (Bayesian) school of statistics because this school takes account of the fact that the a priori knowledge of a criticality safety analyst impacts the procedure which the analyst applies to perform the analysis.

1. INTRODUCTION

Irrespective of whether one uses burnup credit or not the approach to be taken in demonstrating sufficient subcriticality based on numerical calculation of the neutron multiplication factor of a fuel system of interest is prescribed in all national and international regulations in the same way (cp. References [1] through [8], e.g.). The evaluated neutron multiplication factor

- shall include all uncertainties arising from the applied calculation procedure and from manufacturing tolerances and
- shall not be greater than an adequate upper bound $(1-\Delta k_m)$ of subcriticality, under all conditions, normal operation conditions as well as all anticipated abnormal or accident conditions.

These requirements are expressed, in terms of mathematics, in the inequality

$$k+\lambda s \le k_{limit} = (1-\Delta k_m) - \Delta k_U \tag{1.1}$$

where k is the calculated value of the neutron multiplication factor, λs represents the statistical "uncertainty" of k (if a statistical calculation method is used) or numerical error of k (if a non-statistical calculation procedure is applied). Δk_U represents the bias in the applied calculation procedure as obtained from benchmark calculations and includes the "uncertainty" arising from manufacturing tolerances concerning dimensions, construction materials etc.

One is confronted with the problem, therefore,

- to give estimates for k, λs and Δk_U,
- to reach a decision about whether the true value of the neutron multiplication factor is below k_{limit} or not, and
- to derive a criterion telling us the risk of having made a wrong decision.

The objective of the paper on hand is, therefore, to show

- that estimation as well as testing of a hypothesis about the true value of a parameter are usual decision problems in Statistics,
- that methods of estimation and testing relevant to criticality safety analysis can be optimized (meaning that the risk to make a wrong decision can be minimized).

For this purpose References [9] through [11] are used as guidelines.

2. ESTIMATION

2.1. Basic Definitions

Estimation my be considered as the "measurement of a parameter" which is assumed to have some fixed but unknown value based on a limited number of *observations*. Given these observations, estimation consists in determining either a single value (*point estimation*) or a range of values (*interval estimation*).

Estimator, estimate: To estimate a parameter, one first chooses a function of the observations (i.e., a method for proceeding from the observations to the estimate) which is called *estimator*. The numerical value yielded by the estimator for a particular set of observations is the *estimate*.

Consistency and convergence: An estimator is called *consistent* if its estimates converge toward the true value of the parameter as the number of observation increases.

Bias and consistency: Let $\hat{\Theta}$ be an estimator of a parameter Θ, based on N observations. The *bias* b of the estimator $\hat{\Theta}$ is defined as the deviation of the expectation of $\hat{\Theta}$ [1] from the true value Θ_0,

$$b_N(\hat{\Theta}) = E[\hat{\Theta}] - \Theta_0 = E[\hat{\Theta} - \Theta_0] \ . \tag{2.1}$$

Thus, an estimator is unbiased, if, for all N,

$$b_N(\hat{\Theta}) = 0 \ \text{ or } E[\hat{\Theta}] = \Theta_0 \, . \tag{2.2}$$

It may seem that unbiasedness and consistency are related, but neither one implies the other. For example, an estimator may be biased, but consistent. The fact that the estimates converge towards Θ_0 as N increases does not imply that (2.2) holds for all N.

Information content or efficiency: An estimator $\hat{\Theta}$ is *efficient* when its variance V[$\hat{\Theta}$] attains its minimum variance bound.

In addition to unbiasedness, consistency and efficiency there is a further desirable property characterizing the goodness of estimators, namely *robustness.* For further discussions of these desirable properties see [9] and [10].

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[1] For the definition of "expectation", "variance" etc. see Appendix A

2.2. Interval Estimation

2.2.1. Introduction

Whilst in point estimation single values of unknown parameters are estimated, it is the objective in interval estimation to find the range

$$\Theta_L \le \Theta \le \Theta_U \tag{2.3}$$

which contains the true value Θ_0 with a probability $(1-\alpha)$. Among all such ranges the problem is to determine, in some sense, the *optimal* range, given $(1-\alpha)$. For instance, a possible criterion may be that the interval chosen has minimal length among all intervals $[\Theta_L, \Theta_U]$ with the same probability content. Such intervals are called *confidence intervals* for Θ with *probability content* $(1-\alpha)$.

Since one wants to be reasonable confident that the interval indeed contains the true value Θ_0, one chooses $(1-\alpha)$ to be large, for example 95% or 99%. (Thus α, which is the probability that Θ_0 is not contained in the interval, is small.) In practice it is often standard to choose $(1-\alpha) = 68.3\%$ or 95.5%, and the corresponding "errors" (confidence intervals) are called "1 standard deviation errors" or "2 standard deviation errors". This is a confusing language because it is true, in general, only for the normal distribution, cf. [11].

Given an observation x (or a set of observations) with a probability density function $f(x|\Theta)$ depending on Θ (*conditional* density function of x, given the parameter Θ), the probability content $(1-\alpha)$ in the region $[x_L, x_U]$ in the x space is

$$1-\alpha = P(x_L \le x \le x_U) = \int_{x_L}^{x_U} f(x|\Theta)\,dx\,. \tag{2.4}$$

When the density f is known, as well as the parameter Θ, one can always calculate $(1-\alpha)$, given x_L and x_U. When the parameter is unknown, one has to find another variable

$$z = z(x, \Theta) \tag{2.5}$$

such that its probability density function is independent of the unknown Θ. If this can be found, it may be possible to re-express equation (2.4) as a problem of interval estimation: Given $(1-\alpha)$, find the optimal range $[\Theta_L, \Theta_U]$ in Θ space such that

$$P(\Theta_L \le \Theta \le \Theta_U) = 1-\alpha\,. \tag{2.6}$$

2.2.2. Examples

Let $f(x|\Theta)$ in (2.4) be the normal distribution $N(\mu, \sigma^2)$ with expectation μ and variance σ^2,

$$N(\mu, \sigma^2) = \frac{1}{\sigma\sqrt{2\pi}} \exp\left(-\frac{1}{2}\cdot\frac{(x-\mu)^2}{\sigma^2}\right). \tag{2.7}$$

1. When μ and σ^2 are known, equation (2.4) can be evaluated.
2. Suppose that the mean μ is unknown, but that one has N observations x_i from $N(\mu,\sigma^2)$.

The sample mean

$$\bar{x} = \frac{1}{N}\sum_i x_i \tag{2.8}$$

can be used to find a variable (2.5) such that its probability density function is independent of the mean μ. Due to the fact that all the observations x_i are from $N(\mu,\sigma^2)$ the expectation and the variance of x_i are

$$E[x_i] = \mu \text{ and } V[x_i] = \sigma^2 \text{ for all i.} \tag{2.9}$$

Therefore, taking into account that the expectation is a linear operator (see Appendix A) the expectation of the sample mean becomes

$$E[\bar{x}] = \frac{1}{N}\sum_i E[x_i] = \frac{1}{N}\cdot N\mu = \mu, \tag{2.10}$$

i.e., $\bar{x}$ is an unbiased estimator of the mean μ (cp. equation (2.2)). (It can also be shown, that $\bar{x}$ is a consistent estimator, cf. [9]).

The variance $V[\bar{x}]$ becomes

$$V[\bar{x}] = \frac{\sigma^2}{N} + \frac{2}{N}\sum_{i=1}^{N}\sum_{j=i+1}^{N} \operatorname{cov}(x_i, x_j). \tag{2.11}$$

If the observations are independent, $\operatorname{cov}(x_i, x_j) = 0$ for every pair (i,j), $i \neq j$, and the double sum term drops out. One gets then the very known result

$$V[\bar{x}] = \frac{\sigma^2}{N}. \tag{2.12}$$

It can be shown that a set of variables, each of which is a linear function of a set of normal variables, has itself a many-dimensional normal distribution (cf. [11]). In particular, because the N variables x_i are assumed to be from a normal distribution with expectation μ and variance σ^2 the sample mean (2.8), which is a linear function of the N variables x_i, has a normal distribution with expectation μ and, if the x_i are independent, variance σ^2/N. The variable

$$z = \sqrt{N}\cdot\frac{\bar{x}-\mu}{\sigma} \tag{2.13}$$

has a probability density function which is independent of μ, namely the so-called standard normal distribution (expectation 0, variance 1)

$$N(0,1) = \varphi(z) = \frac{1}{\sqrt{2\pi}} \exp\left(-\frac{1}{2} \cdot z^2\right). \tag{2.14}$$

Due to the symmetry of N(0,1) one gets

$$P\left(-\Phi_{1-\frac{\alpha}{2}} \le z \le \Phi_{1-\frac{\alpha}{2}}\right) = 1 - \alpha . \tag{2.15}$$

$\Phi_{1-\alpha/2}$ is the so-called "α/2-point" of the standard normal distribution given by

$$\frac{\alpha}{2} = \int_{-\infty}^{-\Phi_{1-\frac{\alpha}{2}}} \varphi(z)\, dz = \int_{\Phi_{1-\frac{\alpha}{2}}}^{\infty} \varphi(z)\, dz . \tag{2.16}$$

With the aid of (2.13) it is possible to re-express (2.4) as confidence interval for μ:

$$P\left(\bar{x} - \Phi_{1-\frac{\alpha}{2}} \cdot \frac{\sigma}{\sqrt{N}} \le \mu \le \bar{x} + \Phi_{1-\frac{\alpha}{2}} \cdot \frac{\sigma}{\sqrt{N}}\right) = 1 - \alpha . \tag{2.17}$$

3. Suppose that the mean μ and the variance σ^2 are unknown, but that one has N observations x_i from $N(\mu, \sigma^2)$.

It can be shown (cf. [9] and [11]) that the so-called "sample variance"

$$\hat{\sigma}^2 = \frac{1}{N-1} \sum_i (x_i - \bar{x})^2 \tag{2.18}$$

is an unbiased and consistent estimator of the variance σ^2, and it can be shown that the variable

$$(N-1)\frac{\hat{\sigma}^2}{\sigma^2} \tag{2.19}$$

is distributed as a χ^2 variable with (N-1) degrees of freedom. From that it follows (cf. [11]) that the variable

$$t = \frac{z}{\hat{\sigma}/\sigma} = \frac{\sqrt{N} \cdot \frac{\bar{x} - \mu}{\sigma}}{\hat{\sigma}/\sigma} = \frac{\sqrt{N}(\bar{x} - \mu)}{\hat{\sigma}} \tag{2.20}$$

has a Student's t distribution with (N-1) degrees of freedom. Due to the symmetry of this distribution and because of (2.20) one gets from

$$P\left(-t_{N-1;1-\frac{\alpha}{2}} \le z \le t_{N-1;1-\frac{\alpha}{2}}\right) = 1-\alpha \tag{2.21}$$

a confidence interval for μ:

$$P\left(\bar{x} - t_{N-1;1-\frac{\alpha}{2}} \cdot \frac{\hat{\sigma}}{\sqrt{N}} \le \mu \le \bar{x} + t_{N-1;1-\frac{\alpha}{2}} \cdot \frac{\hat{\sigma}}{\sqrt{N}}\right) = 1-\alpha . \tag{2.22}$$

$t_{N-1;1-\alpha/2}$ in (2.21) and (2.22) denotes the $\alpha/2$-point of the Student's t distribution with (N-1) degrees of freedom.

Point estimation as well as interval estimation are usual decision problems in Statistics. Classical statistics serves to summarize the outcomes (observations) of a measurement with a *minimum loss of information*. The purpose of the measurement may, however, be to reach a *decision* about the acceptability of a hypothesis or the true value of a parameter. A decision theory is needed, therefore.

3. DECISION THEORY

3.1. Introduction

In decision theory a technique is developed for calculating an optimum decision (on a parameter Θ, e.g.), given a set of observations $\vec{x} = (x_1, ..., x_N)$.

The probability density $f(\Theta|\vec{x})$ of Θ given $\vec{x}$ is proportional to the probability density $h(\vec{x}|\Theta)$ of the observations under the condition Θ and the prior knowledge $g(\Theta)$ about Θ (cf. Appendix A),

$$f(\Theta|\vec{x}) \propto h(\vec{x}|\Theta)\, g(\Theta). \qquad \textit{(Bayes theorem)} \tag{3.1}$$

Accordingly, $f(\Theta|\vec{x})$ is named as the posterior knowledge about Θ given $\vec{x}$.

The Bayes theorem provides the link between the observations $\vec{x}$ and the parameter Θ about which the decision is made.

3.2. Definitions and Terminology

- *Observable space* Ω, in which all observations $\vec{x}$ fall.
- *Parameter space* Π, contains all possible values of the parameter Θ or the parameters $\vec{\Theta} = (\Theta_1; ..., \Theta_p)$.
- *Decision space* D, contains all possible decisions.
- *Decision rule* δ, specifies what decision d is to be taken given the observations[2], $d = \delta(\vec{x})$. (3.2)

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[2] Here we shall limit ourselves to non-random decision rules, which define d completely when the observations are given.

- *Loss function* $L(\Theta, d)$: To choose between decision rules one needs a loss function, defined as the loss incurred by taking the decision d, when Θ is assumed to be the true value of the parameter. $L(\Theta, d) = L(\Theta, \delta(\vec{x}))$ is a random variable, being a function of $\vec{x}$.

The introduction of a loss function is an essential part of decision theory. *Any rational choice between decisions must be based on a calculation of the loss.*

- *Risk function* $R_\delta(\Theta)$*: The risk function for the decision rule* δ *is defined as the average loss over all possible observations,*

  $$R_\delta(\Theta) = E_{\vec{x}}[L(\Theta, \delta(\vec{x}))] = \int_\Omega L(\Theta, \delta(\vec{x}))h(\vec{x}|\Theta)dx. \qquad (3.3)^{3)}$$

- *Posterior risk* $r_g(\delta)$ *of using decision rule* δ*, given the prior density* $g(\Theta)$*, is the expected risk over Θ,*

  $$r_g(\delta) = E_\Theta[R_\delta(\Theta)] = \int_\Pi R_\delta(\Theta)g(\Theta)d\Theta. \qquad (3.4)$$

With (3.1) and (3.3) the posterior risk can be written

$$r_g(\delta) = E_\Theta\left[E_{\vec{x}}[L(\Theta, \delta(\vec{x}))|\Theta]\right] = E_{\vec{x}}\left[E_\Theta[L(\Theta, \delta(\vec{x}))|\vec{x}]\right]. \qquad (3.5)$$

The quantity

$$E_\Theta[L(\Theta, \delta(\vec{x}))|\vec{x}] = \int_\Pi d\Theta\, L(\Theta, \delta(\vec{x}))f(\Theta|\vec{x}) \qquad (3.6)$$

is called the posterior loss, given the observations $\vec{x}$. It is the average loss over the posterior density incurred by using the decision $\delta(\vec{x})$.

3.3. Choice of Decision Rules

3.3.1. Classical Choice

The classical choice of a decision rule δ is based on the risk function (3.3). Obviously, the best decision rule δ is the one which gives the smallest risk.

In general, it will not be possible to find a single best rule for the whole range of Θ, but for every value of Θ there will be an optimal rule. However, the basic uncertainty in the classical choice is that the true value of Θ is unknown.

A decision rule $\tilde{\delta}$ is called admissible if there is no rule δ such that the inequality

$$R_\delta(\Theta) < R_{\tilde{\delta}}(\Theta) \qquad (3.7)$$

is true for all values of Θ.

[3)] Subscript at operator E refers to the variable over which the average is taken.

Clearly one has to choose decision rules among the admissible class. The two problems of decision theory are to find admissible solutions, and to choose between them. A useful tool for this purpose is the Bayesian family discussed in the next section.

3.3.2. Bayesian Choice

The Bayesian choice of a decision rule δ for a prior density $g(\Theta)$ is based on the posterior risk (3.4). The Bayesian decision rule is the one which gives the smallest posterior risk, that is

$$r_g(\delta) \leq r_g(\tilde{\delta}) \text{ for any } \tilde{\delta}. \tag{3.8}$$

Thus, the Bayesian approach is to locate the uncertainty about the true value of Θ in a prior distribution of beliefs, $g(\Theta)$. By averaging the risk over the prior density $g(\Theta)$ (cp. equation (3.4)) the basic uncertainty of the classical approach can be avoided.

As can be seen from equation (3.5), if a decision rule δ exists, which minimizes the posterior loss (3.6) for all $\vec{x}$, it obviously minimizes also the posterior risk (3.4), and it is therefore a Bayesian solution. Similarly, as follows from equation (3.4), if there is a decision which minimizes the risk (3.3) for all Θ, it is a Bayesian solution. It can be shown under fairly general conditions that all admissible solutions are Bayesian solutions, cf. [12] and [13]. Therefore, if δ is an admissible decision rule, then there exists some prior density $g(\Theta)$ such that δ is the Bayesian solution for $g(\Theta)$. Conversely, given any decision rule, there exists some Bayesian rule which is equivalent or preferable. The class of Bayesian rules is complete.

3.4. Usual Decision Problems in Statistics

3.4.1. Point Estimation

As already stated at the beginning of section 2.1, estimation my be considered as the "measurement of a parameter" which is assumed to have some fixed but unknown value based on a limited number of observations. Thus, the decision problem in point estimation is what value $\hat{\Theta}$ to choose for a parameter Θ, given N observations $\vec{x} = (x_1, ..., x_N)$ from a probability density $h(\vec{x}|\Theta)$. The decision space D is therefore in one-to-one correspondence with the parameter space Π: Corresponding to each value Θ there is a decision d to assign this value to the parameter Θ.

Let $d = \delta(\vec{x}) = \hat{\Theta}$ be the decision rule that maps the observable space ΩN onto D = Π. The loss is a function $L(\Theta - \hat{\Theta})$ of the distance between the estimate $\hat{\Theta}$ and the true value Θ. The loss function is taken to have a minimum at $\Theta = \hat{\Theta}$. Because a regular function may be approximated at a minimum by a quadratic form, one usually chooses

$$L(\Theta - \hat{\Theta}) = (\Theta - \hat{\Theta})^T W (\Theta - \hat{\Theta}) \tag{3.9}$$

and hence, in the on-dimensional case considered here,

$$L(\Theta - \hat{\Theta}) = \omega(\Theta - \hat{\Theta})^2 \tag{3.10}$$

With this loss function, using the properties of the expectation operator E (cp. Appendix A), the posterior loss (3.6) becomes

$$\begin{aligned} E_\Theta\left[L\left(\Theta-\hat{\Theta}\right)\middle|\vec{x}\right] &= E_\Theta\left[\omega\left(\Theta-\hat{\Theta}\right)^2\middle|\vec{x}\right] = \omega\left\{\underbrace{E_\Theta\left[\left(\Theta-E_\Theta\left[\Theta\middle|\vec{x}\right]\right)^2\middle|\vec{x}\right]}_{\text{variance, cp. Appendix A}}+\left(\hat{\Theta}-E_\Theta\left[\Theta\middle|\vec{x}\right]\right)^2\right\} \\ &= \omega\left\{V_\Theta\left[\Theta\middle|\vec{x}\right]+\left(\hat{\Theta}-E_\Theta\left[\Theta\middle|\vec{x}\right]\right)^2\right\} \end{aligned} \tag{3.11}$$

Clearly, the posterior loss is minimal when the decision rule is

$$\hat{\Theta} = E_\Theta\left[\Theta\,\middle|\vec{x}\right] = \int_\Pi \Theta\, f\left(\Theta\middle|\vec{x}\right) d\Theta\ . \tag{3.12}$$

Thus, the optimum value to chose for Θ is the mean of the posterior density $f\left(\Theta\middle|\vec{x}\right)$.

To gain some insight into the solution (3.12) let us return to the second example of section 2.2.2.: Assume the observations to be mutually independent, let the probability density $h\left(\vec{x}\middle|\Theta\right)$ of the observations be the normal distribution $N\left(\Theta,\sigma^2\right)$ with expectation Θ and variance σ^2, and assume σ^2 to be known:

$$h\left(\vec{x}\middle|\Theta\right) = \frac{1}{\left(2\pi\sigma^2\right)^{N/2}}\cdot\exp\left\{-\sum_{i=1}^{N}\frac{\left(x_i-\Theta\right)^2}{2\sigma^2}\right\}. \tag{3.13}$$

Assume the prior density $g(\Theta)$ to be the normal distribution $N\left(\Theta_0,\sigma_0^2\right)$ with expectation Θ_0 and variance σ_0^2,

$$g(\Theta) = N\left(\Theta_0,\sigma_0^2\right). \tag{3.14}$$

It is easily seen that, due to (3.13) and (3.14), the posterior density (3.1) is also normal. Ignoring terms not involving Θ and using the sample mean (2.8) the posterior density becomes

$$f\left(\Theta\middle|\vec{x}\right) \propto h\left(\vec{x}\middle|\Theta\right) g(\Theta) \propto \exp\left(-\frac{1}{2\tau^2}\cdot\left(\Theta-\tau^2\kappa\right)^2\right) \tag{3.15}$$

with expectation

$$E_\Theta\left[\Theta\,\middle|\vec{x}\right] = \tau^2\kappa = \frac{\dfrac{N\,\bar{x}}{\sigma^2}+\dfrac{\Theta_0}{\sigma_0^2}}{\dfrac{N}{\sigma^2}+\dfrac{1}{\sigma_0^2}} \tag{3.16}$$

and variance

$$V_\Theta[\Theta\,|\vec{x}] = \tau^2 = \frac{1}{\dfrac{N}{\sigma^2} + \dfrac{1}{\sigma_0^2}}. \tag{3.17}$$

Thus, the expectation (3.16) is the weighted average of the prior mean Θ_0 and the observed mean $\bar{x}$, the weights being proportional to the reciprocal of the variances. The precision τ^{-2} of the posterior density is the sum of the precision of the prior density and the precision of the observed mean.

According to the result (3.12), from the decision point of view, the optimum value to choose for the estimate $\hat{\Theta}$ is the mean (3.16),

$$\hat{\Theta} = E_\Theta[\Theta\,|\vec{x}] = \tau^2\kappa = \frac{\dfrac{N\bar{x}}{\sigma^2} + \dfrac{\Theta_0}{\sigma_0^2}}{\dfrac{N}{\sigma^2} + \dfrac{1}{\sigma_0^2}} \tag{3.18}$$

The case that there is no prior knowledge is represented by $\sigma_0 \to \infty$. Then (3.18) becomes

$$\hat{\Theta} = \bar{x}, \text{ for } \sigma_0 \to \infty, \tag{3.19}$$

as one might expect. Thus, in case of no prior knowledge the sample mean (2.8) is the optimum value to choose for the estimate $\hat{\Theta}$.

3.4.2. Interval Estimation

Since the posterior density $f(\Theta|\vec{x})$ summarizes one's knowledge of Θ, the decision problem in interval estimation may be to choose an interval (a, b) in the range of Θ which best describes $f(\Theta|\vec{x})$. The loss function is then a function $L(\Theta; a, b)$ of Θ and the interval:

$$L(\Theta; a, b) = \begin{cases} \omega_1 (b-a)^2, \text{if } \Theta \in (a, b) \\ \omega_2 (\Theta - a)^2, \text{if } \Theta < a \\ \omega_3 (\Theta - b)^2, \text{if } \Theta > b \end{cases} \tag{3.20}$$

The posterior loss (3.6) becomes

$$E_\Theta[L(\Theta; a, b)|\vec{x}] = \omega_1 \int_a^b (b-a)^2 f(\Theta|\vec{x}) d\Theta + \omega_2 \int_{\Theta_{\min}}^{a} (\Theta - a)^2 f(\Theta|\vec{x}) d\Theta + \omega_3 \int_b^{\Theta_{\max}} (\Theta - b)^2 f(\Theta|\vec{x}) d\Theta \tag{3.21}$$

The solution of the decision problem is to choose the interval (a, b) to minimize the posterior loss (3.21).

To gain some insight in this solution let us return to the example given by equations (3.13) and (3.14), i.e., let us assume that the posterior density $f(\Theta|\vec{x})$ is given by the normal distribution (3.15),

$$\mathrm{f}\left(\Theta\middle|\vec{x}\right) \propto \mathrm{h}\left(\vec{x}\middle|\Theta\right)\mathrm{g}(\Theta) \propto \exp\left(-\frac{1}{2\tau^2}\cdot\left(\Theta-\tau^2\kappa\right)^2\right)$$

with expectation (3.16) and variance (3.17).

With

$$z \equiv \frac{\Theta-\tau^2\kappa}{\tau} = \frac{\Theta - \mathrm{E}_\Theta\left[\Theta\middle|\vec{x}\right]}{\sqrt{\mathrm{V}_\Theta\left[\Theta\middle|\vec{x}\right]}},\ z_1 \equiv \frac{a-\tau^2\kappa}{\tau},\ z_2 \equiv \frac{b-\tau^2\kappa}{\tau} \tag{3.22}$$

the posterior loss (3.21) becomes

$$\mathrm{E}_\Theta\left[\mathrm{L}(\Theta;a,b)\middle|\vec{x}\right] \propto \sqrt{2\pi}\cdot\tau^3\left\{\omega_1\left(z_2-z_1\right)^2\int_{z_1}^{z_2}\varphi(z)\,dz + \omega_2\int_{-\infty}^{z_1}\left(z-z_1\right)^2\varphi(z)\,dz + \omega_3\int_{z_2}^{\infty}\left(z-z_2\right)^2\varphi(z)\,dz\right\} \tag{3.23}$$

where $\varphi(z)$ is the standard normal density (2.14).

Without any loss of generality one may choose for normalization

$$\omega_1 \propto \frac{\omega}{\sqrt{2\pi}\cdot\tau^3},\ \omega_2 \propto \omega_1\cdot\left(\frac{z_2}{z_1}-1\right)^2,\ \omega_3 \propto \omega_1\cdot\left(1-\frac{z_1}{z_2}\right)^2 \propto \omega_2\cdot\left(\frac{z_1}{z_2}\right)^2. \tag{3.24}$$

Expression (3.23) becomes, therefore,

$$\begin{aligned}\mathrm{E}_\Theta\left[\mathrm{L}(\Theta;a,b)\middle|\vec{x}\right] &\propto \omega\left(z_2-z_1\right)^2\times\\ &\times\left\{1+2\cdot\left(\frac{1}{z_1}e^{-\frac{1}{2}z_1^2}-\frac{1}{z_2}e^{-\frac{1}{2}z_2^2}\right)+\frac{1}{z_1^2}\int_{-\infty}^{z_1}z^2\,\varphi(z)\,dz+\frac{1}{z_2^2}\int_{z_2}^{\infty}z^2\,\varphi(z)\,dz\right\}\end{aligned} \tag{3.25}$$

For $z_1 = -z_0$ and $z_2 = z_0 = -z_1$ this expression is minimized because

$$2\cdot\left(\frac{1}{z_1}e^{-\frac{1}{2}z_1^2}-\frac{1}{z_2}e^{-\frac{1}{2}z_2^2}\right) = -\frac{4}{z_0}e^{-\frac{1}{2}z_0^2},\ \text{for } z_1 = -z_0 \text{ and } z_2 = z_0, \tag{3.26}$$

and

$$\frac{1}{z_1^2}\int_{-\infty}^{z_1}z^2\,\varphi(z)\,dz+\frac{1}{z_2^2}\int_{z_2}^{\infty}z^2\,\varphi(z)\,dz = \frac{2}{z_0^2}\int_{z_0}^{\infty}z^2\,\varphi(z)\,dz << \frac{2}{z_0^2},\ \text{for } z_1 = -z_0 \text{ and } z_2 = z_0. \tag{3.27}$$

Thus, the optimum choice is

$$z_1 = -z_0 \text{ and } z_2 = z_0. \tag{3.28}$$

Using the definitions (3.22) and the optimum choice (3.18) the interval (a, b) sought-after becomes, therefore,

$$\begin{aligned} a &= \tau^2\kappa - z_0\tau = E_\Theta\left[\Theta\middle|\vec{x}\right] - z_0\sqrt{V_\Theta\left[\Theta\middle|\vec{x}\right]} = \hat{\Theta} - z_0\sqrt{V_\Theta\left[\Theta\middle|\vec{x}\right]} \\ b &= \tau^2\kappa + z_0\tau = E_\Theta\left[\Theta\middle|\vec{x}\right] + z_0\sqrt{V_\Theta\left[\Theta\middle|\vec{x}\right]} = \hat{\Theta} + z_0\sqrt{V_\Theta\left[\Theta\middle|\vec{x}\right]} \end{aligned} \tag{3.29}$$

Due to the fact that z_0 is given by the standard normal distribution (2.14),

$$P(z < -z_0) = \int_{-\infty}^{-z_0} \varphi(z)dz = \int_{z_0}^{\infty} \varphi(z)dz = P(z > z_0) \equiv p, \tag{3.30}$$

the probability content of the interval (3.29) amounts to

$$P(a \le \Theta \le b) = 1 - 2p. \tag{3.31}$$

Choosing the $\alpha/2$-point (2.16) of the standard normal distribution for z_0,

$$z_0 = \Phi_{1-\frac{\alpha}{2}} \text{ and hence } p = \frac{\alpha}{2} = \int_{\Phi_{1-\alpha/2}}^{\infty} \varphi(z)dz \tag{3.32}$$

the probability content (3.31) of the interval (3.29) becomes $(1-\alpha)$.

In case of no prior knowledge (cp. equations (3.16) through (3.19)) one obtains from (3.29), using (3.32), the confidence interval (2.17),

$$a = \bar{x} - \Phi_{1-\frac{\alpha}{2}} \cdot \frac{\sigma}{\sqrt{N}}, \; b = \bar{x} + \Phi_{1-\frac{\alpha}{2}} \cdot \frac{\sigma}{\sqrt{N}} \tag{3.33}$$

as one might expect.

3.4.3. Tests of Hypotheses

3.4.3.1. Basic Concepts

Suppose that one has to decide between two hypotheses, H_0 and H_1, which uniquely correspond to values Θ_0 and Θ_1, respectively, of a parameter Θ. Usually one chooses a loss function which has the value zero (no loss) for the good decision, cp. Table 3.1. Given some decision rule δ, there is a finite probability $\alpha(\delta)$ of choosing H_1 when H_0 is true, and $\beta(\delta)$ of choosing H_0 when H_1 is true.

Table I. Loss function and probabilities of decisions

State of nature	Prior probability	Loss function		Probabilities $p(H_i \mid \Theta)$ of decisions	
		Decision: choose Θ_0	Decision: choose Θ_1	Decision: choose Θ_0	Decision: choose Θ_1
H_0 true	μ	0	L_0	$1-\alpha(\delta)$	$\alpha(\delta)$a
H_1 true	1-μ	L_1	0	$\beta(\delta)$a	$1-\beta(\delta)$

a. In the classical approach α is called "significance" or "significance level" (or "error of the first kind"), and β is called "contamination" (or "error of the first kind").

In the Bayesian approach one attributes prior probabilities μ and 1-μ to H_0 and H_1, respectively. And the Bayesian solution is to choose the decision rule δ that gives the smallest posterior risk, cf. section 3.3.2.

According to equation (3.4) the posterior risk is given by

$$r_\mu(\delta) = L_0\,\alpha(\delta)\mu + L_1\,\beta(\delta)(1-\mu). \tag{3.34}$$

The decision rule δ which minimizes (3.34) is found by considering the posterior loss, given the observations $\vec{x}$. The posterior loss in choosing H_0 is given by

$$L_1 P(H_1 \mid \vec{x}) = L_1(1-\mu)P(\vec{x} \mid H_1), \tag{3.35}$$

$P(H_1 \mid \vec{x})$ is the posterior probability that H_1 is true4). The posterior loss in choosing H_1, on the other hand, is given by

$$L_0 P(H_0 \mid \vec{x}) = L_0\,\mu\,P(\vec{x} \mid H_0), \tag{3.36}$$

$P(H_0 \mid \vec{x})$ is the posterior probability that H_0 is true.

Thus the minimum risk decision rule is:

- Choose H_0 (i.e., $\Theta = \Theta_0$) if

$$L_1(1-\mu)P(\vec{x} \mid H_1) < L_0\,\mu\,P(\vec{x} \mid H_0) \;\Rightarrow\; \frac{P(\vec{x} \mid H_1)}{P(\vec{x} \mid H_0)} < \frac{L_0\,\mu}{L_1(1-\mu)}. \tag{3.37}$$

- Otherwise choose H_1 (i.e., $\Theta = \Theta_1$).

---

[4] (3.35) is an application of the Bayes theorem (3.1), but for discrete variables: (1-μ) is the prior probability for H_1 (cp. Table I.), P($\vec{x}$ |H_1) is the probability to get the observations $\vec{x}$ under the hypothesis H_1.

3.4.3.2. Power of the Test and Neyman-Pearson Lemma

Classically, one would not speak of loss L_0 or L_1. One would choose a significance level α arbitrarily, and one would then minimize the contamination β. Due to the fact that α is the probability of choosing H_1 when H_0 is true (cp. Table 3.1) one intuitively chooses a small α. That means that one would choose H_1 only when the observations make H_0 quite impossible. This line of reasoning, which makes no sense in the classical approach, is justified in the Bayesian interpretation by the fact that the smaller α is, then, the bigger should the prior probability for H_0 (or the loss L_0) be. It can be shown rigorously [13] that the classical approach corresponds to a Bayesian solution with a particular choice of the ratio $L_0\mu/L_1(1-\mu)$ in (3.37).

According to its definition the significance level α is given by

$$\alpha = \int_{\omega(\alpha)} dx\, P(\vec{x}|H_0) \tag{3.38}$$

where $\omega(\alpha)$ is the critical region in the observable space Ω. I.e., the space Ω is divided into this critical region $\omega(\alpha)$ and the region of acceptance $\Omega-\omega(\alpha)$, such that observations $\vec{x}$ falling into $\omega(\alpha)$ are regarded as suggesting that the hypothesis H_0 is not true. Thus, α is the probability that $\vec{x}$ falls in $\omega(\alpha)$ even though H_0 is true.

According to Table 3.1 the following expression, usually called "power of the test of significance α", holds:

$$1-\beta(\alpha) = \int_{\omega(\alpha)} dx\, P(\vec{x}|H_1). \tag{3.39}$$

We want to find a region $\omega(\alpha)$ which maximizes $1-\beta(\alpha)$ (i.e., minimizes β): Equation (3.39) can be rewritten

$$1-\beta(\alpha) = \int_{\omega(\alpha)} dx \frac{P(\vec{x}|H_1)}{P(\vec{x}|H_0)} P(\vec{x}|H_0) = E_{\omega(\alpha)}\left[\left.\frac{P(\vec{x}|H_1)}{P(\vec{x}|H_0)}\right| H_0\right]. \tag{3.40}$$

Clearly, this will be maximal if and only if $\omega(\alpha)$ is that fraction of the observable space that contains the largest values of the ratio $P(\vec{x}|H_1)/P(\vec{x}|H_0)$. Thus, the best critical region $\omega(\alpha)$ consists of points satisfying

$$t(\vec{x};\Theta_0,\Theta_1) \equiv \frac{P(\vec{x}|H_1)}{P(\vec{x}|H_0)} \geq c_\alpha, \tag{3.41}$$

c_α being so chosen that the significance level condition (3.38) is satisfied.

Thus, the decision criterion is:

- Choose H_0 (i.e., $\Theta = \Theta_0$) if

$$t(\vec{x};\Theta_0,\Theta_1) \equiv \frac{P(\vec{x}|H_1)}{P(\vec{x}|H_0)} \le c_\alpha \,. \tag{3.42}$$

- Otherwise choose H_1 (i.e., $\Theta = \Theta_1$).

This criterion is known as *Neyman-Pearson lemma* (which is optimal in the classical sense).

To gain some more insight in the solution (3.42) let us take again the example that the observations $\vec{x}$ are from a normal distribution with known variance σ^2, but unknown expectation Θ:

- Hypothesis H_0: $\Theta = \Theta_0$, distribution $P(\vec{x}|\Theta_0) = N(\Theta_0, \sigma^2)$
- Hypothesis H_1: $\Theta = \Theta_1$, distribution $P(\vec{x}|\Theta_1) = N(\Theta_1, \sigma^2)$

Using the sample mean (2.8), the quantity (3.41) becomes

$$\begin{aligned} t(\vec{x};\Theta_0,\Theta_1) = \frac{P(\vec{x}|H_1)}{P(\vec{x}|H_0)} &= \frac{\exp\left\{-\frac{1}{2\sigma^2}\sum_{i=1}^{N}(x_i - \Theta_1)^2\right\}}{\exp\left\{-\frac{1}{2\sigma^2}\sum_{i=1}^{N}(x_i - \Theta_0)^2\right\}} \\ &= \exp\left\{-\left[\frac{N}{2\sigma^2}(\Theta_1 - \Theta_0)^2 - \frac{N}{\sigma^2}(\bar{x} - \Theta_0)(\Theta_1 - \Theta_0)\right]\right\} \end{aligned} \tag{3.43}$$

Thus, t is a monotonic function of

$$z = \frac{\sqrt{N}\,(\bar{x} - \Theta_0)}{\sigma} \tag{3.44}$$

which is distributed according to

- $N(0,1)$ if H_0 is true,
- $N(\Theta_1 - \Theta_0, 1)$ if H_1 is true.

The Neyman-Pearson test is therefore equivalent to a cut on z: If α is the significance level, the critical region is that region of the space of $\vec{x}$ where

$$z \begin{cases} < -\Phi_{1-\alpha}, \text{ if } \Theta_1 < \Theta_0 \\ > \Phi_{1-\alpha}, \text{ if } \Theta_1 > \Theta_0 \end{cases} \tag{3.45}$$

$\Phi_{1-\alpha}$ the α-point of the standard normal distribution (cp. equation (2.16)).

Thus H_0 will be chosen when

$$z = \frac{\sqrt{N}\left(\bar{x}-\Theta_0\right)}{\sigma} \begin{cases} \ge -\Phi_{1-\alpha}, \text{ if } \Theta_1 < \Theta_0 \\ \le \Phi_{1-\alpha}, \text{ if } \Theta_1 > \Theta_0 \end{cases} \tag{3.46}$$

The power of the test (3.40) becomes

$$1-\beta = \begin{cases} \dfrac{1}{\sqrt{2\pi}} \displaystyle\int_{-\infty}^{-\Phi_{1-\alpha}} dz \, \exp\left(-\frac{(z-d)^2}{2}\right), \text{ if } \Theta_1 < \Theta_0 \\ \dfrac{1}{\sqrt{2\pi}} \displaystyle\int_{\Phi_{1-\alpha}}^{\infty} dz \, \exp\left(-\frac{(z-d)^2}{2}\right), \text{ if } \Theta_1 > \Theta_0 \end{cases} \tag{3.47}$$

with

$$d = \frac{\sqrt{N}\left(\Theta_1-\Theta_0\right)}{\sigma} \ . \tag{3.48}$$

Using (2.14) with u = z – d, (3.47) becomes

$$1-\beta = \left\{ \begin{matrix} \displaystyle\int_{-\infty}^{-\Phi_{1-\alpha}-d} du \ \varphi(u) = \int_{-\infty}^{-(\Phi_{1-\alpha}-|d|)} du \ \varphi(u), \text{ if } d<0 \\ \displaystyle\int_{\Phi_{1-\alpha}-d}^{\infty} du \ \varphi(u) = \int_{\Phi_{1-\alpha}-|d|}^{\infty} du \ \varphi(u), \text{ if } d>0 \end{matrix} \right\} = \int_{\Phi_{1-\alpha}-|d|}^{\infty} du \ \varphi(u). \tag{3.49}$$

The power is an increasing function of $\Theta_1-\Theta_0$ and N. Also $1-\beta=\alpha$ for $\Theta_1=\Theta_0$, as one would expect.

Tests may be compared on the basis of their power functions:

- The best test of H_0 against H_1 at the significance level α is the test with maximum power at $\Theta=\Theta_1$.
- If for a given value of Θ, a test is at least as *powerful* as any other test, it is called most powerful test at the value of Θ.
- A test which is most powerful for all values of Θ under consideration is called *Uniformly Most Powerful* (UMP) test.

The Neyman-Pearson lemma gives the best test provided that H_0 and H_1 are completely specified simple hypotheses, i.e.

$$\begin{aligned} H_0 &: \Theta=\Theta_0 \\ H_1 &: \Theta=\Theta_1 \end{aligned} \tag{3.50}$$

However, the case of *composite hypotheses* is the more common one. One example of composite hypotheses is

$$\begin{aligned} &H_0: \Theta = \Theta_0 \\ &H_1: \ \Theta < \Theta_0 \end{aligned} \tag{3.51}$$

In this case H_0 and H_1 belong to one continuous family, i.e. the hypotheses consist in the specification of different regions of the parameter space. In this case the following theorem holds [14]:

- If the observations $x_1, ..., x_N$ are independent, identically distributed random variables with a probability density of the form

$$p(\vec{x}|\Theta) = F(\vec{x})G(\Theta)\exp\{A(\vec{x})B(\Theta)\} \qquad \text{(``}\textit{exponential family}\text{'')} \tag{3.52}$$

where $B(\Theta)$ is strictly monotonic, then there exists a UMP test of $H_0: \Theta = \Theta_0$ against $H_1: \ \Theta < \Theta_0$ 5).

Obviously the normal distribution $N\left(\Theta, \sigma^2\right)$ belongs to the exponential family (3.52). Thus, our example (3.43) becomes now

$$t = \exp\left\{-\left[\frac{N}{2\sigma^2}\Delta^2 - \frac{N}{\sigma^2}(\bar{x} - \Theta_0)\Delta\right]\right\} \tag{3.53}$$

with

$$\Delta \equiv \Theta - \Theta_0 . \tag{3.54}$$

Again, t is a monotonic function of

$$z = \frac{\sqrt{N}\,(\bar{x} - \Theta_0)}{\sigma} \tag{3.55}$$

which is distributed according to

- $N(0,1)$ if H_0 is true,
- $N(\Delta,1)$ if H_1 is true.

If α is the significance level, the critical region is that region of the space of $\vec{x}$ where

$$z = \frac{\sqrt{N}\,(\bar{x} - \Theta_0)}{\sigma} < -\Phi_{1-\alpha} . \tag{3.56}$$

Thus, H_0 will be rejected when

$$\bar{x} + \frac{\sigma}{\sqrt{N}}\Phi_{1-\alpha} < \Theta_0 . \tag{3.57}$$

---

5) This is also true for the "one-sided" test of H_0: $\Theta = \Theta_0$ against H_1: $\Theta > \Theta_0$. But there exists no UMP test for the "two-sided" test of H_0: $\Theta = \Theta_0$ against H_1: $\Theta \neq \Theta_0$.

3.4.3.3. Safety Acceptance Criteria

The result (3.57) may be interpreted as a decision that, on the basis of the significance level α, it is rejected that the parameter Θ exceeds a maximum allowable value Θ_0. Thus, inequality (3.57) may be understood as a safety acceptance criterion.

Let $x_1, ..., x_N$ be *Monte Carlo* estimates of the neutron multiplication factor of a fuel system of interest, x_i the outcome of the i-th batch evaluated. The outcomes $x_1, ..., x_N$ may be taken as being independent when a sufficiently large number of initial batches is skipped (excluded from evaluation). Because of the *Central Limit theorem of statistics* the results $x_1, ..., x_N$ may be taken as normally distributed [9]. Monte Carlo estimates of the neutron multiplication factor belong to the exponential family (3.52), therefore. Thus, the criticality safety acceptance criterion (1.1) may be taken to be given by the decision criterion (3.57) related to the UMP test of the hypotheses (3.51), where Θ_0 represents the maximum allowable neutron multiplication factor k_{limit}.

However, some criticality safety standards (as [1], [4] and [5], e.g.) require a decision criterion which is more pessimistic than the criterion (3.57). Such standards prescribe that a fraction of at least

$$1-\gamma = \int_{x_{\min}}^{T_u(\alpha)} dx\, P(\vec{x}|\Theta) \tag{3.58}$$

of the distribution $P(\vec{x}|\Theta)$ shall be below the upper limit $T_U(\alpha) < \Theta_0$, at significance level α.

This upper limit $T_U(\alpha)$ is often called "upper $(1-\gamma)/(1-\alpha)$ tolerance limit", telling that this limit is based on a probability content of at least $(1-\gamma)$ at a confidence level of $(1-\alpha)$.

To gain some more insight let us again assume that the observations $\vec{x}$ are from a normal distribution with known variance σ^2, but unknown expectation Θ. From the above UMP test of the hypothesis $H_0 : \Theta = \Theta_0$ against the alternative $H_1 :\ \Theta < \Theta_0$ it is known that for all

$$\bar{x} \geq \Theta_0 - \frac{\sigma}{\sqrt{N}} \Phi_{1-\alpha} \equiv \Theta_L\,. \tag{3.59}$$

the hypothesis $H_0 : \Theta = \Theta_0$ is accepted at a confidence level $(1-\alpha)$. One is able therefore to determine a factor λ such that

$$\int_{-\infty}^{\Theta_L+\lambda\sigma} dx\, P(\vec{x}|\Theta) = \int_{-\infty}^{\Theta_L+\lambda\sigma} dx\, N(\Theta_0,\sigma^2) = 1-\gamma\,. \tag{3.60}$$

Using

$$z = \frac{x-\Theta_0}{\sigma} \text{ and } \bar{z} = \frac{\bar{x}-\Theta_0}{\sigma} \tag{3.61}$$

one gets with (3.59)

$$\int_{-\infty}^{\Theta_L+\lambda\sigma} dx\, N\left(\Theta_0,\sigma^2\right) = \int_{-\infty}^{\lambda-\frac{1}{\sqrt{N}}\Phi_{1-\alpha}} dz\,\varphi(z) = 1-\gamma = 1-\int_{\Phi_{1-\gamma}}^{\infty} dz\,\varphi(z) = \int_{-\infty}^{\Phi_{1-\gamma}} dz\,\varphi(z). \tag{3.62}$$

From that it follows

$$\lambda - \frac{1}{\sqrt{N}}\Phi_{1-\alpha} = \Phi_{1-\gamma} \;\Rightarrow\; \lambda = \Phi_{1-\gamma} + \frac{1}{\sqrt{N}}\Phi_{1-\alpha} \tag{3.63}$$

and the upper limit $T_U(\alpha)$ sought-after becomes, thus,

$$T_U = \overline{x} + \lambda\sigma = \overline{x} + \left(\Phi_{1-y} + \frac{1}{\sqrt{N}}\Phi_{1-\alpha}\right)\cdot\sigma. \tag{3.64}$$

The safety acceptance criterion becomes, therefore,

$$\overline{x} + \lambda\sigma = \overline{x} + \left(\Phi_{1-y} + \frac{1}{\sqrt{N}}\Phi_{1-\alpha}\right)\cdot\sigma < \Theta_0. \tag{3.65}$$

A comparison of this criterion to the criterion (3.57) shows that the sample mean $\overline{x}$ in (3.65) must be smaller than the sample mean $\overline{x}$ in (3.57) by at least $\Phi_{1-\gamma}\sigma$ in order to be acceptable.

Note that σ in (3.65) as well as in (3.57) refers to the variance σ^2 of the outcomes $x_1, \ldots, x_N$ of the N evaluated batches of one Monte Carlo calculation. As apparent from the case $N \to \infty$, the upper limit (3.64) does refer to the distribution $P(\vec{x}|\Theta)$ of the outcomes x_i of the batches evaluated, it does not refer to the distribution of the sample mean $\overline{x}$. This is often confused in practice.

If one wants to determine an upper $(1-\gamma)/(1-\alpha)$ tolerance limit that refers to the distribution of the sample mean $\overline{x}$, one has to perform m independent Monte Carlo calculations for one and the same case, varying the starting random number. One gets thus m independent outcomes

$$\overline{x}_j = \frac{1}{N_j}\sum_{i=1}^{N_j} x_{ji}\,,\quad V\left[\overline{x}_j\right] = \frac{\sigma^2}{N_j}, \tag{3.66}$$

where Nj is the number of batches evaluated in the j-th calculation run. For the sake of simplicity it is assumed in the following that

$$N_j = N \text{ for all } j = 1, \ldots, m\,. \tag{3.67}$$

The variance of all the means $\overline{x}_j$ becomes, therefore,

$$V\left[\overline{x}_j\right] = \frac{\sigma^2}{N} \text{ for all } j = 1, \ldots, m\,. \tag{3.68}$$

Instead of (3.57) and (3.59) one gets now

$$\hat{\Theta} + \frac{\sigma/\sqrt{N}}{\sqrt{m}} \Phi_{1-\alpha} < \Theta_0 \tag{3.69}$$

and, respectively,

$$\hat{\Theta} \geq \Theta_0 - \frac{\sigma/\sqrt{N}}{\sqrt{m}} \Phi_{1-\alpha} \equiv \Theta_L \tag{3.70}$$

where, as follows with (3.67) from (3.18),

$$\hat{\Theta} = \frac{1}{m} \sum_{j=1}^{m} \overline{x}_j \, . \tag{3.71}$$

Instead of (3.60) on has now

$$\int_{-\infty}^{\Theta_L + \lambda \frac{\sigma}{\sqrt{N}}} d\overline{x} \, N\left(\Theta_0, \frac{\sigma^2}{N}\right) = 1 - \gamma . \tag{3.72}$$

Using now

$$z = \frac{\sqrt{N}\,(\overline{x} - \Theta_0)}{\sigma} \text{ and } \overline{z} = \frac{\sqrt{N}\,(\hat{\Theta} - \Theta_0)}{\sigma} \tag{3.73}$$

one gets with (3.70)

$$\int_{-\infty}^{\Theta_L + \lambda \frac{\sigma}{\sqrt{N}}} d\overline{x} \, N\left(\Theta_0, \frac{\sigma^2}{N}\right) = \int_{-\infty}^{\lambda - \frac{1}{\sqrt{m}}\Phi_{1-\alpha}} dz \, \varphi(z) = 1 - \gamma = 1 - \int_{\Phi_{1-\gamma}}^{\infty} dz \, \varphi(z) = \int_{-\infty}^{\Phi_{1-\gamma}} dz \, \varphi(z) . \tag{3.74}$$

From that it follows

$$\lambda - \frac{1}{\sqrt{m}} \Phi_{1-\alpha} = \Phi_{1-\gamma} \;\Rightarrow\; \lambda = \Phi_{1-\gamma} + \frac{1}{\sqrt{m}} \Phi_{1-\alpha} \tag{3.75}$$

and instead of (3.64) one gets now

$$T_U = \hat{\Theta} + \lambda \frac{\sigma}{\sqrt{N}} = \hat{\Theta} + \left(\Phi_{1-y} + \frac{1}{\sqrt{m}} \Phi_{1-\alpha} \right) \cdot \frac{\sigma}{\sqrt{N}} \, . \tag{3.76}$$

It should be kept in mind that this result holds only then, if (3.67) holds. In fact, choosing

$$N_j = N \text{ for all } j = 1, \ldots, m$$

may be realized in practice.

However, for the sake of convenience it was assumed up to now that the variance σ^2 is known. In practice this is usually not true, and the formulae for the criteria (3.57) and (3.69), respectively, as well as the formulae for the upper tolerance limits (3.64) and (3.76), respectively, will become more complex, therefore, cf. References [11] and [15]. Nevertheless, all the statements made above about the principles remain true.

4. CONCLUSIONS

In the paper on hand a basic approach to decision theory was given by pointing out the decision problems in point estimation, interval estimation, and test of hypotheses. The Bayesian approach was taken, and it was mentioned that this approach comprises the so-called “classical” (or “Anti-Bayesian”) approach.

In the Bayesian approach all prior knowledge is represented by a probability law on the parameter Θ under study. Some observations $\vec{x} = (x_1, ..., x_N)$ (experimental or calculational results) modify this knowledge, transforming the prior law into a posterior law. A decision on Θ can be made according to this posterior law.

It was shown that the Bayesian solution in point estimation as well as interval estimation consists in an optimum point estimate, an optimum confidence interval respectively, meaning that the posterior loss incurred by choosing this point estimate, confidence interval respectively, given the observations $\vec{x}$, is minimized, and hence the posterior risk is minimized.

It was shown that safety acceptance criteria are based on a Uniformly Most Powerful (UMP) test procedure providing a decision rule that represents a particular Bayesian decision rule and hence minimizes the posterior risk, given the observations $\vec{x}$. And it was demonstrated that safety acceptance criteria consisting in applying $(1-\gamma)/(1-\alpha)$ tolerance limits are based on a more pessimistic decision rule.

Even though the Bayesian approach is more complete, it is not undisputed. The problem is in the prior knowledge introducing induction in the data analysis. However, the average criticality safety analyst behaves like a Bayesian. In fact, his prior knowledge enters already in the modeling of the fuel system of interest and affects therefore the outcomes $\vec{x} = (x_1, ..., x_N)$. His prior knowledge leads to the decision on the codes (depletion and criticality calculation) to be applied and the cross-section sets to be used. In addition, his prior knowledge impacts the decision about which fuel configurations and which events will be analyzed by him and which will be not considered. In other words, his prior knowledge results in pre-choosing a particular region in the space of parameters describing the reactivity of the fuel system of interest. And, last not least, his prior knowledge determines which set of experiments or benchmarks will be selected to determine the contributions to the term Δk_U of the safety acceptance criterion (1.1) which arise from

- the bias in the depletion calculation results, i.e. the bias in the calculated isotopic number densities, and
- the bias of the criticality calculation code.

The evaluation of such contributions may be performed with the aid of point and interval estimation methods as well as testing procedures. The decision on the experiments or benchmarks to be considered may be based on sensitivity analysis “measuring” the distances between parameter values characterizing the system of interest and parameter values describing the experiments or benchmarks. This is often performed by employing estimation methods which are based - for reasons of consistency – on application of the law of large numbers [9], such as for instance the maximum likelihood method or least squares methods (which consist in fact in minimizing a quadratic form like (3.9)), [9] through [11]. Application of this methods of constructing estimators to finite samples of experiments or benchmarks makes it necessary to determine the properties of the estimates yielded by these estimators. The estimates are usually biased, and the bias is not always removable. It is often necessary to rely on the asymptotic properties of the estimators. A summary of the properties of these estimators is given in Ref. [9], Tables 8.1 through 8.4 and 9.2; for further discussion see Ref. [10].

APPENDIX A

Glossary and Definition of some Basic Terms used in Statistics

Probability:

Let Ω be the set of all possible elementary events x_i which are exclusive, i.e., the occurrence of one of them implies that none of the others occurs. The probability of the occurrence of x_i, $P(x_i)$, is defined to have the following properties:

$$P(x_i) \geq 0 \text{ for all } i,\ P(x_i + x_j) = P(x_i) + P(x_j),\ \text{and} \sum_{\Omega} P(x_i) = 1. \tag{A.1}$$

A set S of elementary events x_i is a non-elementary event. The occurrence of S is defined by occurrence of at least one event x_i in the set S.

Random variable and probability distribution:

A *random event* S is an event which has more than one possible outcome. The outcome of a random event is not predictable, only the probabilities of the possible outcomes are known.

A random variable x takes different possible numerical values x_1, x_2, ... corresponding to the different possible outcomes of a random event. The corresponding probabilities $P(x_1)$, $P(x_2)$, ... form a probability distribution.

The probability distribution is given by the function

$$F(x_0) = P(x < x_0). \tag{A.2}$$

The value $F(x_0)$ of the distribution F at x_0 is the probability $P(x < x_0)$ that the random variable x has a value less than x_0.

Continuous random variable:

The probability distribution of a continuous random variable is given by

$$F(x) = \int_{x_{min}}^{x} f(\tilde{x})\, d\tilde{x} \tag{A.3}$$

with $\Omega = (x_{min}, x_{max})$:= space of all possible values of x,

$$F(x_{min}) = 0,\ F(x_{max}) = 1. \qquad (A.4)$$

f(x) is called probability density function or density.

Continuous random vector:

A n-dimensional random vector x is a set of n random variables with a probability distribution

$$F(x_1, ..., x_n) = P(\mathbf{x} \in R) = \int_R ... \int dx_1 \cdot ... \cdot dx_n f(x_1, ..., x_n) \qquad (A.5)$$

with

$$\int_\Omega ... \int dx_1 \cdot ... \cdot dx_n f(x_1, ..., x_n) = 1, \qquad (A.6)$$

Ω the definition space of x. $f(\mathbf{x}) = f(x_1, ..., x_n)$ is the joint probability density function of x.

Statistic:

Suppose a new random variable, y, is defined by

$$y = g(x_1, ..., x_n). \qquad (A.7)$$

Any such function g is called a statistic.

Expectation, mean and variance:

If g(x) is some function of a random variable x with density f(x), the expectation of g(x) is the number

$$E[g] = \int_\Omega g(x) f(x)\, dx \quad . \qquad (A.8)$$

Note: The expectation E is a linear operator

$$E[a \cdot g(x) + b \cdot h(x)] = a \cdot E[g(x)] + b \cdot E[h(x)]. \qquad (A.9)$$

The expectation of the random variable x itself is called the mean of the density f(x),

$$\mu \equiv E[x] = \int_\Omega x \cdot f(x)\, dx. \qquad (A.10)$$

The expectation of the function $(x - \mu)^2$ is called the variance of the density f(x):

$$\sigma^2 \equiv E\left[(x - \mu)^2\right] = \int_\Omega (x - \mu)^2 \cdot f(x)\, dx. \qquad (A.11)$$

Covariance and correlation:

The covariances $\mathrm{cov}(x_i, x_j)$ of the joint density $f(\mathbf{x}) = f(x_1, ..., x_n)$ of a random vector $\mathbf{x} = (x_1, ..., x_n)$ are the expectations

$$\mathrm{cov}(x_i, x_j) = E[(x_i - \mu_i)(x_j - \mu_j)] = \int_\Omega (x_i - \mu_i)(x_j - \mu_j) \cdot f(\mathbf{x})\, d\mathbf{x} \tag{A.12}$$

with

$$\mu_i = \int_\Omega x_i \cdot f(\mathbf{x})\, d\mathbf{x}\,.$$

The correlation coefficient of x_i and x_j is given by

$$\mathrm{corr}(x_i, x_j) = \frac{\mathrm{cov}(x_i, x_j)}{\sqrt{\sigma_i^2 \cdot \sigma_j^2}} \tag{A.13}$$

with

$$\sigma_i^2 \equiv \mathrm{cov}(x_i, x_i) = E\left[(x_i - \mu_i)^2\right]\,. \tag{A.14}$$

If the variables are independent, covariance and correlation vanish. The converse statement is not necessarily true.

Conditional distributions:

Sections through distributions are called conditional distributions. The section through the density function $f(x, y)$ at $x = x_0$ gives the conditional density of y, given that $x = x_0$, which is denoted by $f(y|x_0)$,

$$f(y|x_0) = \frac{f(x_0, y)}{\int f(x_0, y)dy} = \frac{f(x_0, y)}{g(x_0)} \tag{A.15}$$

From that it follows

$$f(x, y) = f(y|x)g(x) = h(x|y)q(y) \tag{A.16}$$

and from this follows *Bayes theorem* for continuous variables

$$h(x|y) = \frac{f(y|x)g(x)}{q(y)} \tag{A.17}$$

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Strategies for applying isotopic uncertainties in burnup credit[*]

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Abstract. Uncertainties in the predicted isotopic concentrations in spent nuclear fuel represent one of the single largest sources of overall uncertainty in criticality calculations that use burnup credit. The methods used to propagate the uncertainties in the calculated concentrations to the uncertainty in the predicted neutron multiplication factor (k_{eff}) of the system can have a significant effect on the uncertainty in the predicted safety margin and ultimately can affect the potential capacity of spent fuel storage casks. This report surveys several different best estimate strategies for considering the effects of isotopic uncertainties in burnup credit analyses and illustrates the impact of these strategies on the predicted k_{eff} for a prototypical burnup credit cask design. The reactivity margin associated with the more realistic best estimate strategies is discussed in comparison to the reactivity margin associated with conventional bounding methods of uncertainty propagation. The experimental database of isotopic uncertainties used in the study has been significantly expanded to include new high-enrichment and high-burnup spent fuel assay data recently published for burnup credit nuclides. Expanded rare earth fission product measurements that contain the only known publicly available measurement for ^{103}Rh have also been included. This expanded experimental database is reviewed together with the bias and uncertainties obtained using the SCALE code system.

INTRODUCTION

Over the past decade, there has been a concerted effort in the United States and other countries to use more accurate and realistic estimates of the reactivity worth of spent nuclear fuel (SNF) in licensing of spent fuel storage and transportation systems by applying burnup credit. Criticality safety analyses have traditionally assumed that the fuel is unirradiated, which has clearly led to considerable safety margins. The process for performing criticality calculations in a burnup credit model requires two distinct steps — the first to predict the spent fuel isotopic concentrations using burnup calculations; the second, to perform a criticality calculation using the isotopic concentrations estimated in the first step. Consideration of the depletion phenomena in the criticality assessment significantly increases the overall complexity of a criticality safety analysis, places increased demands and reliance on computational tools and methods, and necessitates consideration of many additional uncertainties associated with the fuel compositions that are not encountered in analyses that assume the fuel is unirradiated.

The U.S. Nuclear Regulatory Commission (NRC) issued Revision 1 of Interim Staff Guidance 8 (ISG-8) in July 1999, to provide guidance on the application of limited burnup credit in criticality safety analyses for pressurized-water reactor (PWR) spent fuel in transportation and storage casks [1]. This ISG-8 guidance states,

> *"The applicant should ensure that the analysis methodologies used for predicting the actinide compositions and determining the neutron multiplication factor (k-effective) are properly validated. Bias and uncertainties associated with predicting the actinide compositions should be determined from benchmarks of applicable fuel assay measurements."*

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The analyst is ultimately required to assess the impact of the nuclide bias and uncertainty on the predicted neutron multiplication factor (k_{eff}) for the system. Unfortunately, there is currently no guidance or consensus on how the bias and uncertainties should be propagated to the k_{eff} in a burnup credit analysis. The different approaches to considering the nuclide uncertainties in this step of the analysis can have a significant effect on the predicted criticality safety margin and ultimately impact the number and types of spent fuel assemblies that may be considered acceptable for loading in a transport or storage cask.

This report reviews and illustrates several different potential strategies for considering the effects of nuclide uncertainties in burnup credit. These strategies include conventional bounding and best estimate methods to estimate the net effect of nuclide uncertainties. The reactivity margins associated with the different methods are compared for PWR spent fuel loaded in a prototypic burnup credit cask. The nuclide uncertainties used in these studies are based on a SCALE code system analysis of revised and expanded isotopic assay data publicly available in the United States. Recent publication of radiochemical assay measurements for the Japanese Takahama-3 reactor fuel has significantly enhanced the database in terms of both the number of measurements and the enrichment and burnup range of the experimental data. Recently published fission product measurements performed by the Khlopin Radium Institute in Russia for rare earth fission products are also included. The Khlopin measurements include the only known publicly-available measurement for the major fission product absorber ^{103}Rh.

METHODS OF UNCERTAINTY PROPAGATION

The different approaches used for treating uncertainties in complex calculational models are generally grouped as either “bounding” methods that use statistically bounding values to conservatively account for individual parameter uncertainty, or “best estimate” methods that use average parameter values and attempt to estimate realistically the effects of parameter uncertainties on the final results by using Monte Carlo (probabilistic) methods or other techniques (e.g., sensitivity analysis). Best estimate methods can provide a more accurate representation of the effects of nuclide uncertainty in a subcritical burnup-credit spent fuel system. However, using a best estimate approach is more complicated because of the additional effort required to determine accurately the uncertainty in the calculated k_{eff} for the system. The available criticality safety margin is underestimated because bounding methods tend to significantly overestimate k_{eff},. Thus, the subcritical margin predicted using best estimate methods would actually be larger than that obtained using a bounding approach. The uncertainty predicted using such realistic methods may be used to derive tolerance intervals to achieve any desired level of statistical confidence. Best estimate methods provide an improved understanding of the uncertainties and can help reduce the subcritical margin needed in burnup credit safety analyses, resulting in fewer casks needing to be transported and decreasing the regulatory burden on licensees while maintaining safety for transporting SNF.

Bounding Approach

In a conventional bounding approach, the analysis assumptions and input parameters are simultaneously set to their limiting (maximum or minimum) values to produce the most conservative final result. As applied to nuclide uncertainties in burnup credit, this approach applies conservatively bounding values for the predicted concentration of each nuclide. In the bounding model, the calculated concentrations are adjusted in a way that always leads to a more reactive system. In other words, the concentration of fissile nuclides is always increased, while the concentration of absorbing nuclides is always decreased, in order to maximize the k_{eff} of the system. Each nuclide is adjusted for the average bias in the burnup calculation, and a statistically bounding estimate of uncertainty based on the nuclide standard deviation (from calculated-to-experimental ratio) and a multiplier on the uncertainty to achieve the desired level of statistical confidence for the number of samples available. This approach ensures that the estimated uncertainty associated with the prediction of the nuclide inventories will be bounding. Such a conservative approach was proposed by the United States Department of Energy (U.S. DOE) in support of transportation and storage burnup credit [2]. Although this approach assures that the uncertainty in the predicted k_{eff} will be very conservative, simultaneously adjusting all nuclide concentrations to their statistical extremes is not realistic and masks the true importance of the uncertainties on the criticality calculation. However, such an approach is simple, easy to defend, and yields concentrations that when used in a criticality calculation, will provide an upper bounding estimate of the k_{eff} for the system.

Best Estimate Approaches

Several best estimate methods can be used to propagate uncertainties from individual parameters contributing to the global k_{eff} uncertainty. Three best estimate methods are discussed and illustrated in this paper:

1. Monte Carlo uncertainty sampling,
2. Sensitivity/Uncertainty method, and
3. a "direct difference" method.

Monte Carlo Sampling Method

The total uncertainty in a computed quantity may be estimated using a technique that involves Monte Carlo (stochastic) sampling of the uncertainty probability distributions of the different parameters used in a calculation. Unlike a bounding calculation performed using a single set of conservative parameter values, the Monte Carlo (MC) approach undertakes multiple calculations with changes to the input parameters that reflect the random uncertainty distributions of the parameters. These multiple calculations yield a distribution of results from which the mean and probability of exceeding a particular value or threshold can be determined statistically.

For burnup credit calculations, the technique involves stochastically varying the nuclide concentrations according to the uncertainty in the predicted concentration of each burnup credit nuclide (i.e., related to the standard deviation). The shape of the probability distributions (typically normal) and uncertainties are established by comparing measured and computed nuclide concentrations in SNF.

The practical implementation of this approach requires automating the Monte Carlo stochastic sampling to determine the needed nuclide concentration sets and apply each set in the specified criticality analysis model. To illustrate this approach, a Monte Carlo sampling method was implemented at Oak Ridge National Laboratory (ORNL) to estimate the effects of nuclide uncertainties in burnup credit analyses using a computer program called KRONOS [3] designed to interact within the SCALE code system. The criticality calculations are repeated automatically, until the mean k_{eff} and probability distribution in k_{eff} from the nuclide uncertainties have converged. The average bias and uncertainty for each nuclide, used to define the sampling distribution, are input directly to the code. KRONOS will perform either a SCALE CSAS1X one-dimensional (1-D) XSDRNPM calculation or a CSAS25 three-dimensional (3-D) KENO V.a criticality calculation for the system. Due to the large number of criticality calculations required by this method to provide statistically reliable results, the code was developed for parallel processing on a distributed network environment to minimize the time to complete an analysis.

Sensitivity/Uncertainty Method

Sensitivity/uncertainty (S/U) analysis techniques have been widely used as a means of quantifying the effect of input data and other parameters on computer model predictions. These methods are used to develop sensitivity coefficients for the system that reflect the change in the calculated response to a change in an input parameter. To a first order, a sensitivity coefficient of 1.0 means that a 1% change in the parameter will cause a 1% change in the result. Combined with uncertainty information, sensitivity techniques can provide a powerful tool to estimate the global system uncertainty.

With this approach the relative change in the k eigenvalue due to a change in the concentration of nuclide N is expressed through first-order accuracy by the linear relationship

$$\frac{dk}{k} = S_N \frac{dN}{N} ,$$

were the proportionality constant S_N is the sensitivity coefficient of k to the nuclide concentration N. This technique provides a straightforward method of predicting the uncertainty in the k_{eff} given the variation in nuclide concentration that is attributed to the nuclide uncertainty. The nuclide uncertainty, expressed as a relative change in the concentration, dN/N, may be obtained for the important burnup credit nuclides by comparing predicted and measured nuclide concentrations in SNF. The uncertainty from multiple nuclides may be combined to provide a measure of total uncertainty. If the uncertainties are assumed to be uncorrelated or random, they may be combined using a root-mean-square (RMS) approach. The total uncertainty was calculated using the expression

$$\frac{dk}{k} = \sqrt{\sum_{i=1}^{n} \left(\frac{dN}{N} \{S_N\} \right)^2} ,$$

where the sum is performed over all n burnup credit nuclides in the criticality analysis. Note that if the uncertainty from each nuclide is combined additively (using the absolute values of the sensitivity coefficients S_N), then the result is equivalent to the uncertainty predicted using the conventional bounding approach.

Sensitivity coefficients used in this work were generated for a generic burnup-credit cask model using SEN35, a prototypic SCALE code sequence that implements sensitivity analysis techniques for 3-D Monte Carlo criticality calculation [4]. The methods used to generate the sensitivity information are based on the widely-used perturbation theory approach [5]. SEN35 calculates forward and adjoint neutron fluxes using an enhanced version of the KENO V.a Monte Carlo criticality code. Once the fluxes are obtained, the SAMS module (Sensitivity Analysis Module of SCALE) produces flux moments and calculates the sensitivity coefficients from these data and the cross-section data. SAMS also calculates the uncertainty in the sensitivity coefficients due to the Monte Carlo uncertainties. The motivation and principal application of the SEN35 sequence has been the need for modern computational tools that can generate sensitivity data necessary to gauge the applicability of validation experiments used for criticality studies [6]. SEN35 provides sensitivities for the various partial and total cross sections in the criticality calculations. However, the sensitivity coefficient for the nuclide total cross section is equal to that for the nuclide concentration.

This allows sensitivity coefficients for each nuclide (at a given concentration) to be obtained directly. To generate burnup-dependent sensitivity coefficients, multiple KENO V.a input files were created for SEN35, with each file containing the burnup-dependent compositions of the burnup credit nuclides in spent fuel. Sensitivity coefficients were generated for a uniform axial burnup and an axial burnup profile for the spent fuel assemblies.

Direct-Difference Method

Another best estimate technique has been explored recently at ORNL. Instead of evaluating the bias and uncertainty for individual nuclides used in a burnup credit calculations, the measured nuclide concentrations from radiochemical assay experiments are applied directly in a criticality calculation for the spent fuel configuration used in the intended burnup credit application. The keff calculated for the system is then compared to the value predicted using calculated nuclide concentrations for the same set of burnup credit nuclides. The difference (Δk) is a measure of the net bias in the keff calculation associated with the predicted nuclide concentrations, and the variation of the results obtained using multiple experimental data sets provides a direct measure of the uncertainty. In this paper, this technique is called the direct difference method.

Unlike the other methods, this approach deals with the aggregate effect of the nuclide uncertainties on k_{eff} directly, and does not required an analysis of the bias and uncertainty for any individual nuclide. Rather, the net effect of the uncertainty from all nuclides is determined simultaneously. This technique can be used to estimate the bias and uncertainty in the neutron multiplication factor for a given set of burnup credit nuclides. The bias and uncertainty in the neutron multiplication factor are obtained from a statistical analysis of the distribution of Δk values obtained using measured and predicted nuclide concentrations. The variance in the Δk results is related to the uncertainty in the predicted nuclide compositions and can be used to estimate the net effect of the reactivity margin to any desired level of statistical confidence. The method does not require any *a prior* assumptions about the shape of the uncertainty distribution of individual nuclides, and does not assume that uncertainty in different nuclides is uncorrelated. In addition, the method may yield trends in the k_{eff} bias with fuel enrichment and/or burnup that may not be evident by analyzing individual nuclides. However, this method requires a comprehensive database of measured isotopic data for a common set of burnup credit nuclides. That is, an experimental data set must contain measurements for all nuclides selected for the burnup credit analysis in order to be used in this method. A sufficient number of measurements is needed to allow statistically reliable observations to be made about the uncertainty and trends in the k_{eff}. As the number of nuclides in a burnup-credit analysis increases, the number of appropriate experiments tends to decrease. Another limitation of the method is that only spent fuel with a uniform axial burnup can be simulated since assay data are not available for the range of axial burnup values required to simulate an axial profile.

Review and Analysis of Experimental Data

Several benchmark studies have been published using measured data available in the U.S. to establish the bias and uncertainties in the predicted nuclide concentrations using specific code systems for PWR fuel. As part of this current study, a comprehensive review of publicly-available PWR isotopic assay data was undertaken with a re-analysis of the nuclide bias and uncertainties using the current version of the SCALE code system. The benchmark results provide a common set of nuclide uncertainty values that were applied to the different uncertainty propagation techniques. The data sets selected for this study include all PWR assays from Refs. [7] and [8], and the Yankee Rowe reactor assay data used in Ref. [9]. Details of the reactor descriptions, fuel descriptions, laboratories, and experimental methods can be found in the citations to the original work.

In addition to previous studies, recently published radiochemical assay data for spent fuel from the Japanese PWR Takahama-3 reactor were analyzed and added to the experimental database. These data represent an important addition to the available radiochemical assay data. The Takahama-3 data provide 10 new samples with extensive burnup credit nuclide measurements. These samples have an enrichment of 4.11 wt % ^{235}U and relatively high burnups, significantly extending the enrichment and burnup range of publicly available data. The updated radiochemical assay database includes a total of 56 individual spent fuel assay samples from seven different reactors. Table 1 summarizes the reactors, assembly designs, and basic fuel parameters. Nuclide measurements are available for all of the major burnup credit actinides [10]. Spent fuel assay data for the Mihama reactor, used previously in Ref. [9], were not selected for this study. A review [11] of the Mihama data indicated there was high variability in the measurements for fuel having similar burnups and fuel assembly locations, and erratic behavior was also observed for fuel samples taken from different axial positions of the same rod. Since the enrichment and burnup range of the Mihama measurements did not extend beyond the range already available from other data sets, these data were not added to the present study.

The amount of publicly available experimental fission product data is currently very limited. Measurements for fission products important to burnup credit are available in only a small subset of the fuel samples. The majority of fission product data come from measurements of the Calvert Cliffs Approved Test Material (ATM) samples performed at the Pacific Northwest Laboratory (PNL) Materials Characterization Center (MCC). Subsequent to the original ATM-series fission product measurements, the V. G. Khlopin Radium Institute in Russia performed an independent analysis of the three ATM-104 samples, and one sample from ATM-106. The isotopic measurements included Nd, Sm, Eu, and Gd for all samples. In addition, the ^{103}Rh content of the ATM-106 sample was measured. This measurement represents the only known publicly-available data for ^{103}Rh in spent fuel. The Khlopin results were only recently published [12] in an effort supported by the NRC for burnup-credit validation studies. In this study the Khlopin results were used for all Nd, Sm, Eu, and Gd isotopes, except ^{145}Nd. Results for the important burnup credit Nd and Sm isotopes were also available for the Takahama-3 samples, significantly augmenting the number of measurements available for these isotopes. To date, no results for ^{95}Mo, ^{101}Ru, or ^{109}Ag for LWR fuel have been published in open literature.

The radiochemical assay database was used to benchmark the computational methods of SCALE to provide a measure of the mean bias and uncertainty of each nuclide required for the different uncertainty propagation methods. All nuclide inventory calculations were performed with the SCALE 1-D depletion analysis sequence SAS2H [13] and the SCALE 44-group ENDF/B-V-based cross-section library.

Table I. Summary of selected pwr spent fuel radiochemical assay data

Reactor	Lattice type	Enrichment (wt %)	Burnup (GWd/t)	Absorbers	No. of samples
Trino Vercellese	WE 15 × 15	3.13 3.897	11.5 – 24.5 12.0	CR[a]	13 1
Turkey Point	WE 15 × 15	2.556	30.5 – 31.5		5
Obrigheim	CE 14 × 14	3.13	25.9 – 29.5		6
H. B. Robinson-2	WE 15 × 15	2.561	16.0 – 31.7	BPR[b]	4
Yankee Rowe	WE 17 × 18	3.4	16.0 – 36.0	CR	8
Calvert Cliffs	CE 14 × 14	3.038 2.72 2.453	27.4 – 44.3 18.7 – 33.2 31.4 – 46.5	 BPR	3 3 3
Takahama-3	WE 17 × 17	4.11	14.3 – 47.3	BPR	10
Total		2.56 – 4.11	11.5 – 47.3		56

[a] CR = Assemblies exposed to control rods
[b] BPR = Assemblies with burnable poison rods

Uncertainty ANALYSIS Results

This work deals only with the uncertainties in the predicted spent fuel nuclide compositions used as input to the criticality calculation. Additional uncertainties associated with other depletion phenomena (i.e., spatial variation of the burnup), operating history, and the actual criticality calculation itself (e.g., cross-section uncertainties, etc.) are not considered. The results in this report are intended to illustrate the typical uncertainty margins due to the predicted SNF inventory that can be expected in burnup credit analyses using different uncertainty analysis methods. In these current studies, the uncertainty in the reactivity margin was determined as the 2 standard deviation ($\pm 2\sigma$) uncertainty interval for the results. Tolerance factors that account for the additional uncertainty associated with limited sample sizes may also be applied. In this work tolerance factors were not applied for simplicity. The effect of tolerance factors will be very small for most of the actinides that have relatively large amounts of measured data, but would result in larger fission product uncertainties than shown in this current study.

The criticality calculations were performed using a common cask design, based on a conceptual generic rail-type burnup credit cask that would accommodate 32 fuel assemblies [14]. The spent fuel assembly was assumed to be a Westinghouse 17 × 17 OFA design. Burnup credit calculations were performed for a flat axial burnup and for an 18-axial-zone burnup profile derived for assemblies with an average burnup greater than 30 GWd/MTU [2].

Actinide-Only Burnup Credit

The uncertainty in the calculated k_{eff} for the spent fuel cask due to the uncertainties in the predicted nuclide concentrations using the MC and S/U best estimate methods are compared to the conventional bounding approach in Fig. 1 for actinide-only burnup credit with a uniform (flat) axial burnup distribution. The results for the axially-varying burnup profile are shown in Fig. 2. The uncertainty is expressed in units of reactivity ($\rho = \Delta k/k$). All of the important burnup credit actinides [10] were included in the criticality calculations. In all results presented here, the nuclide uncertainties were based on code validation results using the SCALE 4.4a code system using the updated radiochemical assay database. In the bounding analysis method the burnup-credit nuclide concentrations were simultaneously adjusted for uncertainty by an amount equal to the 2 sigma value determined from the comparisons of measured and calculated concentrations for each nuclide. For the Monte Carlo sampling calculations, the concentration of each nuclide was sampled according to a normal distribution defined by its 1 sigma uncertainty value, and the uncertainty in the k_{eff} value determined from the $\pm$ 2 σ uncertainty in the computed distribution of k_{eff} values calculated by the KRONOS code. For the KRONOS results 100 KENO V.a criticality calculations were performed to determine the k_{eff} value uncertainty associated with each point. For the S/U calculations the effects on the k_{eff} value due to a 2 σ perturbation in the nuclide concentrations was determined from the sensitivity coefficients calculation using SEN35, and propagated to the k_{eff} value using the RMS combination method. These methods were used to quantify the estimated uncertainty in an illustrative example using fuel with an initial enrichment of 3.5 wt % ^{235}U and burnup values from 10 to 60 GWd/MTU.

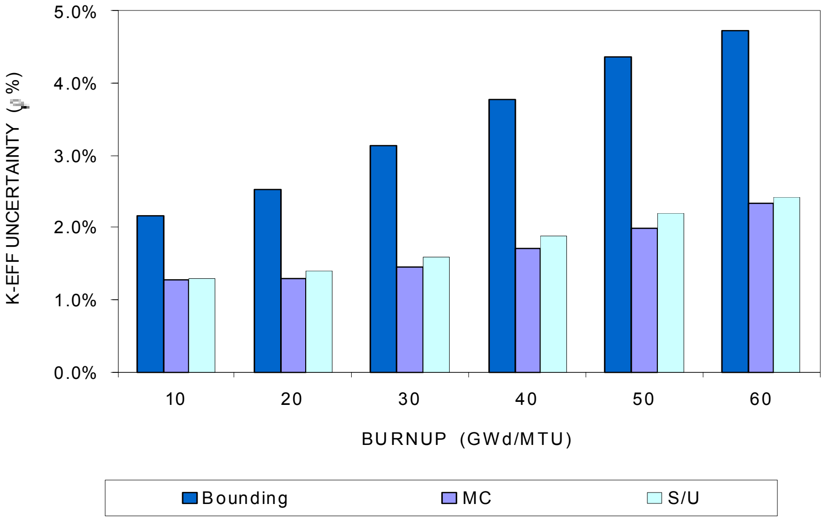

FIG. 1. Estimated uncertainty (2 sigma) in k_{eff} value, in units of reactivity, associated with predicted actinide compositions for a burnup credit cask and flat axial burnup distribution (3.5 wt % ^{235}U).

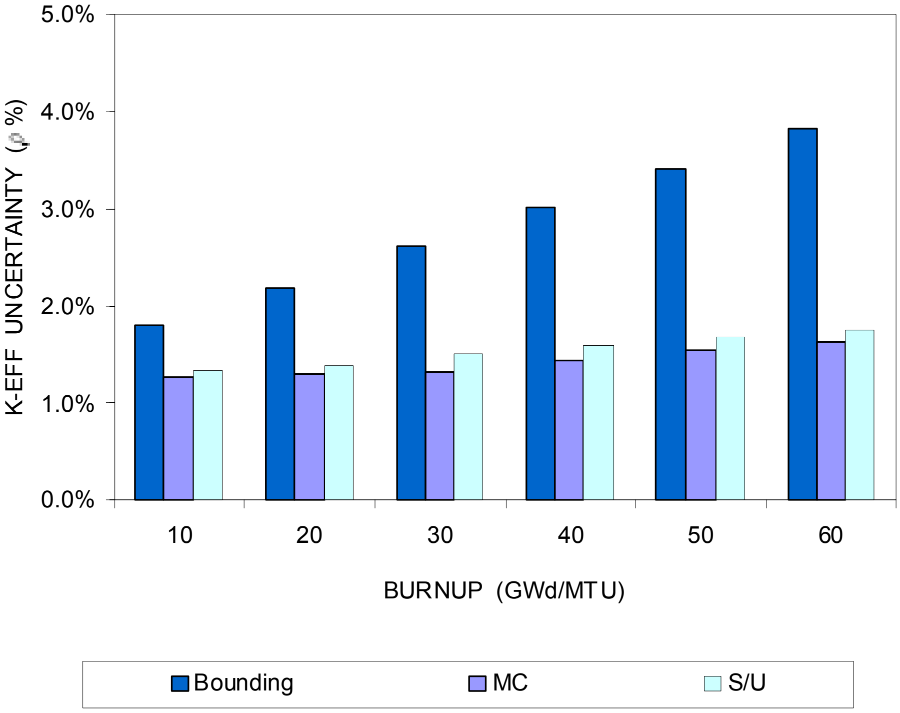

FIG. 2. Estimated uncertainty (2 sigma) in k_{eff} value, in units of reactivity, associated with predicted actinide compositions for a burnup credit cask and axially-varying burnup distribution (3.5 wt % ^{235}U).

The results in Figs. 1 and 2 show that the predicted levels of uncertainty (2 σ) are similar for the two different independent best estimate methods. The uncertainty in the k_{eff} value, expressed as reactivity, is observed to increase with increasing burnup, from about 1.2% at 10 GWd/MTU to 2.3% at 60 GWd/MTU for uniform axial burnup. The uncertainty decreases when an axial burnup profile is used due to the lower burnup at the ends of the assembly, which become increasing important to the reactivity of the fuel in burnup credit. The uncertainty predicted using the bounding approach is seen to be about double the value obtained using the best estimate methods.

Fig. 3 shows the results obtained using the direct difference method for the major uranium and plutonium isotopes ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, and ^{241}Pu plotted as a function of sample burnup. Radiochemical assay data for these actinides were available from all 56 experimental data sets evaluated. Combined, these nuclides represent typically > 90% of the relative reactivity worth for all actinides in spent fuel 5 years after discharge. Additional subsets of burnup credit nuclides have also been evaluated but are not shown here. However, as the number of actinides increases, the number of available measurements typically decreases. Consequently this approach attempts to balance the number of actinides available in the criticality calculation with the number of experiments in order to ensure an adequate number of comparisons are available to allow a reliable statistical interpretation of the results. The criticality calculations performed using this approach assumed a uniform axial burnup profile, since nuclide measurements were available for only a limited set of burnup values. The differences in the calculated k_{eff} value using the predicted nuclide inventories from SCALE and the measured inventories, expressed in units of reactivity[*] ρ, yield information about the bias and uncertainty associated with the predicted nuclide concentrations. Fig. 3 shows the data points for each fuel sample evaluated, and shows the linear regression fit and the ± 2 σ uncertainty interval from the values. The results indicate a small negative bias trend (increasing with burnup) and an average 2 σ uncertainty interval of ± 1.76%. A similar trend in the bias and uncertainty was found when the results are plotted as a function of sample enrichment.

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[*] ρ % defined as $(k_m - k_c)/k_m \times 100$, where k_m and k_c are the k_{eff} values based on measured and calculated nuclide concentrations.

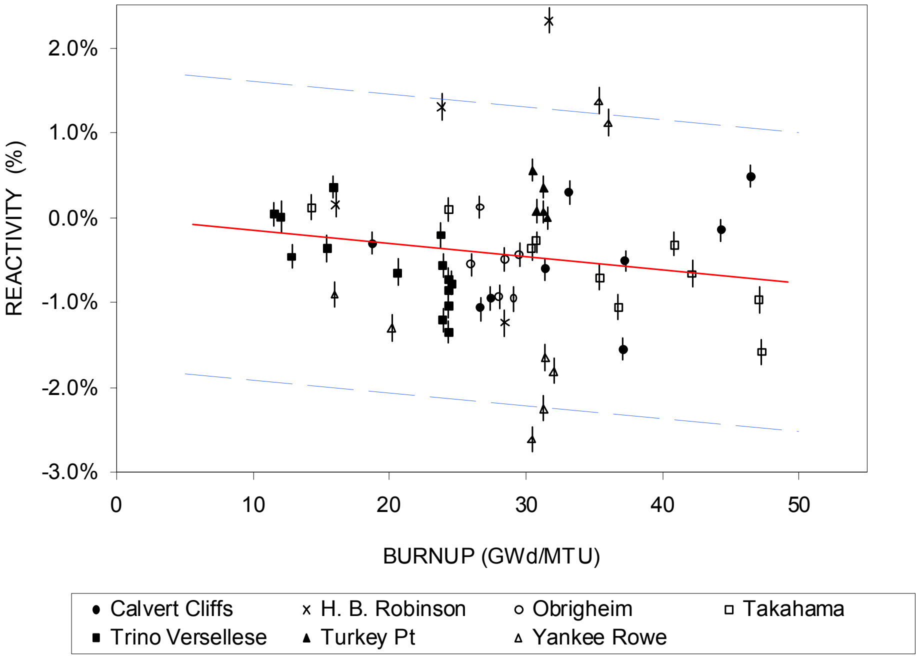

FIG. 3. Estimated reactivity effect (average bias and 2 sigma uncertainty interval) associated with predicted nuclide concentrations for six major actinides, ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, and ^{241}Pu. Results are plotted as a function of sample burnup, for a generic burnup credit cask.

The increase in the uncertainty with increasing burnup found in the Monte Carlo sampling and the S/U results is not evident in the direct difference results. However, there is an important distinction between the direct difference method and the other analysis methods. The direct difference results are based on experimental data with a wide range of enrichments and burnup values. In most cases (but not all) the sample burnup is commensurate with the initial enrichment. That is, as the enrichment increases, so does the discharge burnup. The results, therefore, reflect a proportionality between the enrichment and burnup. However, the Monte Carlo and S/U calculations were performed for variable burnup and a *fixed* initial enrichment (see Figs. 1 and 2). Therefore, the results in the low burnup regime reflects the uncertainty for fuel that has not achieved a typical burnup, while the results in the high-burnup regime reflects fuel that is overburned with respect to the initial enrichment, as compared to typical discharged fuel. The uncertainty determined using the Monte Carlo uncertainty approach for a burnup of 40 GWd/MTU, a value commensurate with the initial enrichment of 3.5 wt % ^{235}U used in the analysis, is about 1.7%. Note that this value is in good agreement with the results derived from the direct difference method of about 1.8% over the range of the experimental data.

Actinide and Fission Product Burnup Credit

Uncertainty calculations were repeated using both actinide and fission product credit. The quantity of measured radiochemical assay data for the fission products is considerably less that that available for the actinides, and for some major fission products (e.g., ^{95}Mo, ^{109}Ag, and ^{101}Ru) there are no known publicly available measurements. The limited quantity of radiochemical assay data makes a statistical evaluation of fission product nuclide uncertainties difficult and increases the uncertainty associated with the predicted concentrations. The lack of a comprehensive fission product database precludes the use of the direct difference method with fission products. Uncertainties were estimated using the MC and S/U methods using nuclide uncertainties for ^{99}Tc, ^{103}Rh, ^{133}Cs, ^{143}Nd, ^{145}Nd, ^{147}Sm, ^{150}Sm, ^{151}Sm, ^{152}Sm, ^{151}Eu, ^{153}Eu, and ^{155}Gd. The one ^{103}Rh measurement provides an estimate of the calculational bias, but precludes an estimate of the uncertainty. For this study the uncertainty in ^{103}Rh was assumed to be nominally 30%. Nuclides with no measured data, ^{95}Mo, ^{109}Ag, and ^{101}Ru, were excluded from the criticality analysis. As noted previously, the addition of tolerance factors to the fission product results would have the effect of significantly increasing the estimated nuclide uncertainty for those nuclides with few measurements.

The uncertainty results for the best estimate methods are compared to the results for the bounding method in Fig. 4 for actinide and fission product credit. The bounding uncertainty predicted for the k_{eff} value is considerably larger when fission products are included. For calculations with a uniform (flat) axial burnup, the 2 σ uncertainty interval is about ± 8.4% at 60 GWd/MTU, compared to about ± 4.2% for actinide-only burnup credit. The uncertainties are somewhat less when an axial profile is included. The results using the best estimate methods are again observed to be similar to each other. The increase in the uncertainty when fission products are included is seen to be significantly less using the best estimate methods compared to the bounding approach. The best estimate 2 σ uncertainty interval for a uniform axial burnup was ± 3.0% at 60 GWd/MTU, compared to ± 2.2% when only the actinides are credited. The uncertainty in calculations that use an axial burnup profile are again observed to be smaller than those for the uniform axial burnup cases.

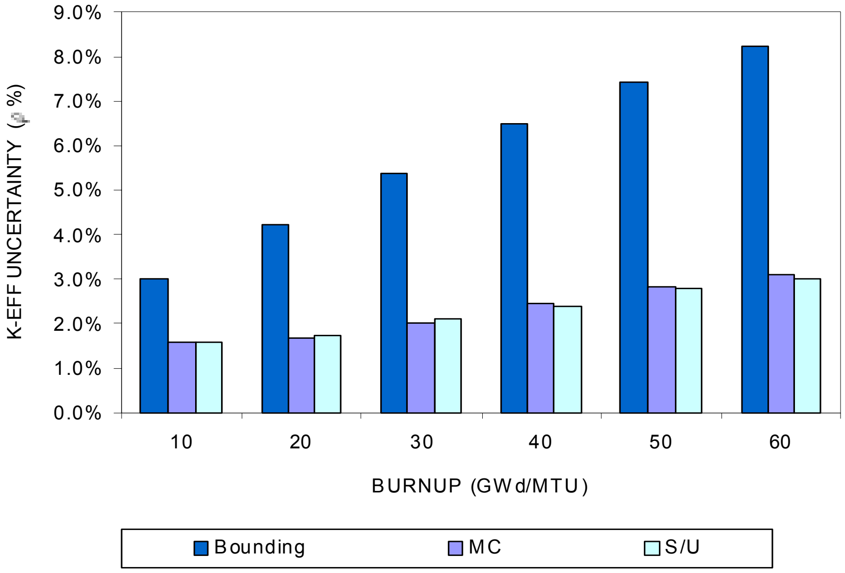

FIG. 4. Estimated uncertainty (2 sigma) in k_{eff} value, in units of reactivity, associated with predicted actinides and fission products compositions for a burnup credit cask and flat axial burnup distribution (3.5 wt % ^{235}U).

Summary and Conclusions

The results indicate that use of best estimate uncertainty methods will significantly reduce the level of uncertainty in the predicted k_{eff} compared to conventional bounding methods for actinide burnup credit calculations. The reduction is observed to be even greater when both actinide and fission product credit is considered. Bounding methods, while clearly conservative and easy to defend, are not realistic and result in higher uncertainties and an underestimation of the true safety margin in a criticality evaluation. The best estimate methods yield more realistic estimates of the effect of nuclide uncertainties, but require considerable more computational effort. An improved understanding of the uncertainties can help reduce the subcritical margin needed in burnup credit safety analyses, thus resulting in fewer casks needing to be transported and reducing regulatory burden on licensees while maintaining safety for transporting SNF.

The uncertainty values presented in this paper are intended for illustrative purposes only since the results are based on the evaluation of a specific set of experiments using the SCALE code system. Also, the nuclide uncertainties have not included tolerance factors to account for the uncertainty from the sample size. The addition of tolerance factors would lead to larger fission product uncertainties than shown in this work but would not have a significant effect on the actinide results since the number of actinide measurements is relatively large.

The direct difference approach illustrated in this report provides a relatively straightforward and definitive method of predicting the net effect of nuclide uncertainties. The results are also consistent with the other independent best estimate methods. The direct difference method requires no ***a priori*** assumptions about the potential enrichment or burnup dependence of the nuclide uncertainties, correlation between different nuclides, or statistical shape of the nuclide uncertainties. In short, the method requires no evaluation of the individual nuclide uncertainties. In practice, this method could be used to estimate the reactivity associated with the nuclide uncertainties, as predicted using a given code system for a specific SNF configuration, and this reactivity would be incorporated into the subcritical margin. This is significantly different than conventional bounding approaches that have been proposed in the U.S. that apply uncertainties directly to the predicted individual nuclide concentrations.

One of the important findings from this work to burnup credit in the U.S. is that the accuracy of the calculated actinide concentrations for the high burnup Takahama-3 fuel samples is comparable to that observed in previous studies involving lower enrichment and lower burnup samples. The ISG-8 guidance on burnup credit recommends limiting the amount of credit for burnup to 40 GWd/MTU or less, and recommends a loading offset for fuel with an initial enrichment between 4.0 and 5.0 wt %. These restrictions on burnup and enrichment are based largely on the lack of sufficient radiochemical assay data above 40 GWd/MTU and 4.0 wt % (the majority of enrichments were under 3.4 wt %) for code validation at the time the guidance was issued. The recent publication and analysis of the Takahama-3 PWR radiochemical assay data significantly extends the range of the database for code validation. The Takahama-3 results indicate that the nuclide uncertainties for fuel exceeding 4 wt % and 40 GWd/MTU are not significantly different than lower enrichment and burnup fuels analyzed previously. This finding has potential implications for the current ISG-8 guidance and may provide a technical basis to support extending burnup credit beyond the present limits.

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RISK PERSPECTIVE AND PERCEPTION, BUILDING CONSENSUS

(Session 2.4)

Probabilistic assessment of dry transport with burnup credit

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Abstract. The general concept of probabilistic analysis and its application to the use of burnup credit in spent fuel transport is explored. Discussion of the probabilistic analysis method is presented. The concepts of risk and its perception are introduced, and models are suggested for performing probability and risk estimates. The general probabilistic models are used for evaluating the application of burnup credit for dry spent nuclear fuel transport. Two basic cases are considered. The first addresses the question of the relative likelihood of exceeding an established criticality safety limit with and without burnup credit. The second examines the effect of using burnup credit on the overall risk for dry spent fuel transport. Using reasoned arguments and related failure probability and consequence data analysis is performed to estimate the risks of using burnup credit for dry transport of spent nuclear fuel.

INTRODUCTION

Burnup credit is the practice of accounting for the reduced reactivity of spent nuclear fuel (SNF) when demonstrating subcriticality of fissile systems. Traditionally, subcriticality is demonstrated for SNF management applications using the so-called "fresh fuel" assumption. Under the fresh fuel assumption, SNF is assumed unburned or fresh, i.e., in its most reactive state. The use of burnup credit can result in various benefits. The burden of using burnup credit may include additional analysis, data collection, experiments, procedural checks, and verification measurements. From a risk perception standpoint, we face the fear that comes from using new practices that seem to reduce criticality safety margins.

For dry transport of SNF the use of burnup credit results in efficiencies in SNF management. Burnup credit allows closer packing of SNF for transport, which results in more fuel in each cask, less exposure, and fewer transports. The use of burnup credit further results in less reliance on external criticality control (e.g., poison plates).

The demonstration of criticality safety for SNF transport using the fresh fuel assumption is the current practice, and its credibility is established. Familiar and accepted criticality analysis methods are available, experiments are available to benchmark these methods, and loading the SNF according to approved specifications is uncomplicated. When burnup credit is used, the criticality analysis for burnup credit is complicated by the need to consider isotopes that vary according to the burnup history of the SNF. Identification of the burnup credit isotopes requires depletion calculations, adding new data requirements to the analyst’s needs. The library of fresh fuel critical benchmark experiments must be supplemented with new critical experiments, and chemical assays. For the fresh fuel assumption, proper loading of a cask is easily assured; only the initial enrichment of the fuel must be known and verified. For burnup credit, proper loading requires knowledge of the initial enrichment of the SNF and its burnup characteristics, which are a function of the fuel’s burnup history.

The use of burnup credit, although it is generally recognized as a potential means of improving efficiencies for SNF management, introduces new challenges, which face the designer, analyst, operator, and regulator. To evaluate these challenges, this paper uses probabilistic analysis.

GENERAL PROBABILISTIC ANALYSIS

The probabilistic analysis methods described in this paper assess the performance of systems that are comprised of arrays of organized components, which may be objects or actions. The objective is to assess the probability of a system's likelihood of not performing as designed. The term fault will be used to describe the failure to perform as designed.

The systems considered consist of two types of component arrangements with statistically independent fault probabilities. The system is a fault tree construction with elements referred to as "AND" and "OR" Gate arrangements.

For a system comprised of two events, A and B, the AND Gate expression for fault probability, P(F), is given by Eq. (1). Fault probability expressions for the OR Gate are given by Eqs. (2) and (3). Eq. (2) applies for independent random events. Eq. (3) applies to independent random events that are mutually exclusive. Eq. (3) is a good approximation for an OR Gate for independent random events of low probability that are not mutually exclusive, since the product of small probabilities are much less than their sum.

$$P(F) = P(A) \times P(B) \tag{1}$$

$$P(F) = P(A) + P(B) - P(A) \times P(B) \tag{2}$$

$$P(F) = P(A) + P(B) \tag{3}$$

When decisions are required about systems, events, and activities having different outcomes, probability estimates are useful, but not always sufficient. The limits associated with probability comparisons alone for informing decisions lies in the lack of weighting for the predicted outcomes. That is, not all faults have the same affect. The risk concept is introduced as a means of weighting such probabilities. The basic definition of risk is the product of a consequence and an associated fault probability. Using probabilistic risk analysis (PRA) shows that high probability low consequence events have risks that can be equivalent to low probability events that have high consequences.

The concept of risk has been introduced as a logical, quantitative measure. That is, the product of probability and consequence. Ideally, probability and consequence are determined rigorously, using observation, or rationally estimated data. In contrast, the concept of risk perception suggests an emotional based view of risk. The emotional based perception may overestimate or underestimate either the probability or the consequence associated with risk.

PROBABILISTIC ANALYSIS OF BURNUP CREDIT FOR SPENT NUCLEAR FUEL TRANSPORT

Two cases will be considered for probabilistic analysis of burnup credit for spent nuclear fuel transport. The first tests the perception that replacing the fresh fuel assumption with burnup credit would increase the likelihood, and thus, the risk of exceeding a specified criticality safety limit. The second looks at the transport risks associated with using burnup credit.

Case 1: Relative Risk - Fresh Fuel Assumption Versus Burnup Credit

Perceived risk often suggests an emotionally based view of what a true risk might be. Case 1 will provide a rational, probabilistic basis for the comparison of the risk of exceeding a specified criticality limit for burnup credit and the fresh fuel assumption. For case 1, a ratio of the risk factors is used. When consequences are equal, relative risk is the same as relative probability.

The relative risk of exceeding the specified criticality limit, RR_{cl}, is the ratio of the probability of exceeding the specified criticality limit for burnup credit, $P_{cl, bc}$, and for the fresh fuel assumption, $P_{cl, ff}$. Both of these probabilities involve the occurrence of three events. That is, the probabilities of critical configuration faults, $P_{cc, bc}$ and $P_{cc, ff}$; the probabilities of water in the cask cavity $P_{w, bc}$ and $P_{w, ff}$; and the probabilities that a final verification of the subcriticality of loaded cask is faulty, $P_{v, bc}$ and $P_{v, ff}$.

The introduction of $P_{w, bc}$ and $P_{w, ff}$ is not intended to challenge the regulatory requirement of a fully flooded transport cask, which applies even to a dry cask design [1, 2]. The flooded cask requirement, which might appear to be an artificial condition, enables the demonstration of subcriticality for a dry cask with light water reactor fuel. Although the fully flooded cask may be an appropriate regulatory condition, accounting for the probability of water in a dry cask is suitable for probabilistic risk analysis. It is further noted that the probability of having water in the cask cavity is independent of the criticality approach used (i.e., fresh fuel versus burnup credit), and $P_w = P_{w, bc} = P_{w, ff}$.

The probabilities, $P_{v, bc}$ and $P_{v, ff}$, pertain verification of loaded casks. These verifications should not be confused with verifications conducted during loading of fuel into a cask. For the fresh fuel assumption, this type of verification is not performed; therefore, it can be neglected by setting $P_{v, ff} = 1$.

The probabilities of exceeding the specified criticality safety limit for burnup credit and the fresh fuel assumption are given by Eq.s (4) and (5). These fault events require critical configurations, the presence of water, and failure to verify subcriticality of the system as prepared for transport. These are AND Gate arrangements. The probabilities are used to form the relative risk given by Eq. (6).

$$P_{cl, bc} = P_{cc, bc} \text{ x } P_w \text{ x } P_{v, bc} \tag{4}$$

$$P_{cl, ff} = P_{cc, ff} \text{ x } P_w \tag{5}$$

$$RR_{cl} = P_{cl, bc} / P_{cl, ff} = (P_{cc, bc} \text{ x } P_{v, bc})/(P_{cc, ff}) \tag{6}$$

To develop the probability models for $P_{cc, bc}$ and $P_{cc, ff}$ the component steps involved in each activity are identified. The steps, which reflect typical procedures for using the fresh fuel assumption and burnup credit, are presented in Table I [3].

The first seven component steps in Table I relate to calculation and design. These steps are considered similar, having the same fault probabilities. The eighth step, manufacturing, is independent of the use of the fresh fuel assumption or burnup credit, and is the same value for each. The last three are verification activities.

Table I. Components of criticality safety and error probabilities

Component Steps	Fresh Fuel	Burnup Credit
Perform criticality analysis	p_1, ff	p_1, bc
Benchmark criticality analysis with fresh fuel experiment	P_2, ff	P_2, ff
Benchmark criticality analysis with spent fuel experiment	-	P_3, ff
Include burnup credit parameters (e.g., axial distribution)	-	P_4, ff
Determine isotopics of spent fuel	-	P_5, ff
Benchmark depletion analysis	-	P_6, ff
Develop loading curve	-	P_7, ff
Manufacture criticality safety devices (e.g., basket)	P_8, ff	P_8, ff
Verify initial enrichment for loading	P_9, ff	P_9, ff
Verify burnup for loading	-	p_{10}, ff
Additional verification of burnup for loading	-	p_{11}, ff

Steps 1-10 provide fault opportunities, and any single fault could cause a system fault; therefore, they are represented by OR Gates. The component probabilities, $p_{1,\ ff}$, … $p_{11,\ bc}$, represent activities and checks by the originating organization and regulatory authorities. The actions and subsequent checking activities are represented by AND Gates.

Each of the steps identified involves objects and human actions. In assigning values to the probabilities, it is assumed that scientific data and computer codes (objects) have been thoroughly reviewed using rigorous internal checks and peer review processes with high reliability (i.e., low fault probability). Similarly, tools and instruments (objects) used in manufacturing and loading operations have high reliability. The remaining elements of these activities are done by humans, and are subject to human faults. For human faults, probabilities that range from 10^{-4} to 5×10^{-3} are used [4].

For the seven components that apply to analysis, $p_{1,\ ff}$, … $p_{7,\ bc}$, the elemental fault probability for human actions is assumed to be 10^{-4}. The fault probabilities for the component is the product of these three equal fault probabilities for action, check, and recheck, and: $p_{1,\ ff} = \ldots = p_{7,\ bc} = 10^{-12}$. Likewise, manufacturing has high reliability, and multiple independent checks. Again, we find $p_{8,\ ff} = p_{8,\ bc} = 10^{-12}$, for probabilities for manufacturing faults that affect criticality safety.

Loading verification is represented by two major elements, the reactor records and the verification prior to loading fuel into a cask. The verification checks the records, and so, the two are represented by an AND Gate. Although checks may be applied to this activity, only one verification check is assumed. The check for initial enrichment involves purchase records and matching those with fuel (e.g., reading serial numbers). Assuming the records and matching faults for verifying initial enrichments are 10^{-4} and 5×10^{-3}, respectively, $p_{9,\ ff} = p_{9,bc} = 5 \times 10^{-7}$. For the burnup verification, records and matching faults for the initial enrichment are assumed to be 5×10^{-3}, then $p_{10,bc} = 2.5 \times 10^{-5}$. For the additional check of burnup in step 11, a variable value is used.

The fault probabilities used in this paper are found in Table II.

Table II. Components of criticality safety and error probabilities values

Component Steps	Fresh Fuel		Burnup Credit	
	Symbol	Value	Symbol	Value
Perform criticality analysis	$p_{1, ff}$	10^{-12}	$p_{1, bc}$	10^{-12}
Benchmark criticality analysis, ff experiment	$P_{2, ff}$	10^{-12}	$P_{2, bc}$	10^{-12}
Benchmark criticality analysis, SF experiment	-	-	$P_{3, bc}$	10^{-12}
Include burnup credit parameters	-	-	$P_{4, bc}$	10^{-12}
Determine isotopics of spent fuel	-	-	$P_{5, bc}$	10^{-12}
Benchmark depletion analysis	-	-	$P_{6, bc}$	10^{-12}
Develop loading curve	-	-	$P_{7, bc}$	10^{-12}
Manufacture criticality safety devices	$P_{8, ff}$	10^{-12}	$P_{8, bc}$	10^{-12}
Verify initial enrichment for loading	$P_{9, ff}$	$5x10^{-7}$	$P_{9, bc}$	$5x10^{-7}$
Verify burnup for loading	-	-	$p_{10, bc}$	$2.5x10^{-5}$
Additional verification of burnup for loading	-	-	$p_{11, bc}$	range

From Tables I and II, the criticality configuration fault probabilities for the fresh fuel assumption and burnup credit are developed, $P_{cc, ff}$ and $P_{cc, bc}$, respectively.

$$P_{cc, ff} = p_{1, ff} + p_{2, ff} + p_{8, ff} + p_{9, ff} \tag{7}$$

$$P_{cc, bc} = p_{1, bc} + p_{2, bc} + p_{3, bc} + \ldots + p_{7, bc} + p_{8, bc} + p_{9, bc} + p_{10, bc} \text{ x } p_{11, bc}$$

$$P_{cc, bc} = P_{cc, ff} + p_{3, bc} + \ldots + p_{7, bc} + p_{10, bc} \text{ x } p_{11, bc} \tag{8}$$

The ratio of criticality configuration fault probabilities, RR_{cc}, is given by:

$$RR_{cc} = P_{cc, bc} / P_{cc, ff} \tag{9}$$

$$RR_{cc} = (P_{cc, ff} + p_{3, bc} + \ldots + p_{7, bc} + p_{10, bc} \text{ x } p_{11, bc}) / P_{cc, ff}$$

$$RR_{cc} = 1 + (p_{3, bc} + \ldots + p_{7, bc} + p_{10, bc} \text{ x } p_{11, bc}) / P_{cc, ff} \tag{10}$$

Substituting the fault probability values from Table II into Eq. (10), yields:

$$RR_{cc} = 1 + (5 \text{ x } 10^{-12} + 2.5x10^{-5} \text{ x } p_{11, bc}) / (3 \text{ x } 10^{-12} + 5x10^{-7}) \tag{11}$$

Consistent with the previous assumptions about verification procedures, which rely on human capabilities, we assume the following human fault probability range: $10^{-4} < p_{11, bc} < 5 \text{ x } 10^{-3}$. Substituting into Eq. (11), and eliminating small terms, yields the following range for RR_{cc}:

$$RR_{cc, min} = 1 + (5 \text{ x } 10^{-12} + 2.5x10^{-9}) / (3 \text{ x } 10^{-12} + 5x10^{-7}) = 1.005$$

$$RR_{cc, max} = 1 + (5 \text{ x } 10^{-12} + 1.25x10^{-7}) / (3 \text{ x } 10^{-12} + 5x10^{-7}) = 1.25$$

$$1.005 < RR_{cc} < 1.25 \tag{12}$$

The result in Eq. (12) suggests that for burnup credit the fault probability is always greater than it is for the fresh fuel assumption if there is no final verification on a loaded cask. Adding an independent verification for burnup at loading can reduce the fault ratio, RR_{cc}, so that it is close to 1.0, but not less than 1.0. To see how to reduce the fault ratio RR_{cc} to less than 1.0, we use Eq. (6), which we combine with Eq. (9), to get:

$$RR_{cl} = RR_{cc} \times P_{v,\ bc} \quad (13)$$

Recall that the device whose fault probability, which is represented by $P_{v,\ bc}$, is an independent check that verifies that the specified criticality limit is satisfied for the loaded cask. We will not speculate on what such a check might be, but only look at its reliability requirements. Reliability, R, is defined as the complement of the fault probability, P, i.e., $R = 1 - P$. The requirement is that $RR_{cl} < 1$. Since the ratio, RR_{cc}, is given as a range, and we want to find the value P(v, bc) that will limit RR_{cl} to less than 1.0, the upper limit, i.e., $RR_{cc,\ max}$ is substituted into Eq. (13):

$$RR_{cl} = 1.25 \times P_{v,\ bc} < 1, \text{ and } P_{v,\ bc} < 1/1.25, \text{ so,} \quad (14)$$

$$P_{v,\ bc} < 0.8, \text{ which gives a reliability of } R_{v,\ bc} > 0.2 \quad (15)$$

Case 2: Burnup Credit and Transport Risk

For case 2, the affect of using burnup credit on the transport risk is evaluated. Risks may be expressed in terms of health effects. The unit of risk used in this paperfor radiological risk is the latent cancer fatality (LCF), and for non-radiological risk, fatalities.

The results of the evaluation of case 1 will be used along with results of a study of severe accidents conducted by the U.S. Nuclear Regulatory Commission (USNRC) in 1987 [5]. Although the USNRC studied highway and rail accidents, their evaluation of subcriticality was only done for rail. The USNRC study assumed criticality if water leaked into a cask, and gave a conditional probability for such an event happening, if a rail accident has occurred, as 7.8×10^{-9}. The accident rate for rail was given as 1.19×10^{-5}/mile (7.39×10^{-6}/km). So, the conditional probability of water leaking into a cask is: $P_w = 5.76 \times 10^{-14}$/km. Substituting this value for P_w into Eq.(4) and assuming that no verification is performed on the loaded cask (i.e., $P_{v,\ bc} = 1$) yields:

$$P_{cl,\ bc} = 5.76 \times 10^{-14} \times P_{cc,\ bc}\ /\text{km} \quad (16)$$

Substituting the values from Table II into Eq.(8), and using the maximum value for the additional verification for burnup performed during loading (i.e., $p_{11,\ bc} = 5 \times 10^{-3}$) gives a value for $P_{cc,\ bc} = 6.25 \times 10^{-7}$, and the following value for the probability frequency of exceeding the specified criticality limit for a burnup credit cask:

$$P_{cl,\ bc} = 3.6 \times 10^{-20}\ /\text{km} \quad (17)$$

The probability frequency in Eq. (17) can be applied to estimates of shipments required for the repository development program in the USA, which are found in the U.S. Department of Energy (USDOE) Draft Environmental Impact Statement (DEIS) [6]. For a case in which mostly rail shipments are used for the USA repository program a total of 10,800 rail cask shipment are expected over 24 years. The average shipment distance (D_{avg}) for a cask loaded with SNF is about 3,864 km, which results in about 42 million km (4.2×10^{7} km) over 24 years. So, the expectation of exceeding the specified criticality limit for a burnup credit cask per average trip, and over the 24 years of shipment for a repository, are given by Eqs. (18) and (19):

$$E_{cl} = \mathrm{P}_{\mathrm{cl,\,bc}} \times \mathrm{D}_{\mathrm{avg}} = 3.6 \times 10^{-20}\ /\mathrm{km} \times 3.86 \times 10^{3}\ \mathrm{km} = 1.39 \times 10^{-16} \quad (18)$$

$$\mathrm{E}_{\mathrm{cl}} = E_{cl} \times 10{,}800 = 1.39 \times 10^{-16} \times 10{,}800 = 1.5 \times 10^{-12} \quad (19)$$

Because the expectation of exceeding the specified criticality limit for a burnup credit cask is so small it is considered incredible, and can be ignored. Additionally, it should be recognized that for highway transport the expectation of exceeding the specified criticality limit is also incredible.

Having determined the incredibility of a criticality event during transport, the benefits of using burnup credit are addressed. The benefit from using burnup credit is due to reducing the number of shipments and transport risks. In the USA, the expected numbers of cask shipments are reduced to 75% for rail and at least 50% for highway transport when burnup credit is used. Burnup credit in the USA will be done for PWR shipments, which comprises about 67% of the total. These reductions are used to estimate potential reduction in health effects reported in the USDOE DEIS for the USA repository program. The DEIS and burnup credit (BUC) results are presented in Table III.

Table III. Estimated transport impacts with burnup credit

Impacts	Mostly Rail[1]		Mostly Highway[2]	
	DEIS	BUC	DEIS	BUC
Radiological Impacts of Normal Transport				
Involved Worker, LCF	3	1.5	11	3.6
Public, LCF	3	1.5	18	6
Radiological Impacts of Accidents, LCF	0.02	0.01	0.07	0.02
Non-Radiological Fatalities	4	2	4	1.3

CONCLUSIONS

Analysis in case 1 demonstrates that using burnup credit for SNF transport is not necessarily riskier than using the fresh fuel assumption for criticality safety. By applying estimates of human fault probabilities for analysis, engineering review, and procedures, it was shown that the increase in probability of exceeding a specified criticality limit could be held to a range of 0.5% and 25%. This was accomplished by applying an additional verification check when loading a burnup credit cask. It was further shown that if a verification method of modest reliability were devised and used for a loaded cask, the risk associated with burnup credit could be made less than that for the fresh fuel assumption.

For case 2, transportation risk was considered. The analysis showed that the probability of exceeding a specified criticality limit is incredibly small, and that the risk reduction from burnup credit, which is dependent on the reduced number of shipments, is significant when compared to the small impacts already predicted.

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[1] Mostly rail includes 10,800 rail and 2,600 truck shipments, but the reduction factor assumes all rail.

[2] Mostly truck includes 49,500 truck and 300 rail shipments, but the reduction factor assumes all truck.

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UK regulatory perspective on the application of burnup credit in plant criticality safety cases

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Abstract. This paper reports recent regulatory progress in the development of a UK regulatory position for assessing licensees' plant safety cases which apply the use of burnup credit for criticality applications. This has been prompted by an increasing awareness amongst UK nuclear operators of the possible advantages in the use of burnup credit.

INTRODUCTION

In the UK, the safety of nuclear installations is regulated by the Health and Safety Executive (HSE) under the Health and Safety at Work etc Act 1974(1), which, inter alia, places a general duty of care on employers and employees, and the Nuclear Installations Act 1965 which provides a permissioning regulatory structure based upon standard licence conditions attached to nuclear site licences. HM Nuclear Installations Inspectorate (NII) is that part of HSE which regulates nuclear safety, including criticality safety, for a range of operations including the design, construction, commissioning, operation, modification, decommissioning and delicensing of nuclear plant. All matters associated with the transport of nuclear materials outside licensed sites is the regulatory responsibility of the Department of Transport, Local Government and the Regions (DTLR) - the UK Competent Transport Authority. Currently in the UK there are no approved applications of burnup credit for criticality purposes covering civil operational plant or transport applications.

Taking credit in criticality assessments for the reduction in spent fuel nuclear reactivity as a result of irradiation, "burnup credit", is a complex issue involving many aspects of nuclear criticality safety. There has been extensive debate and development of methods, data, and operational procedures over an extended timescale including international programmes comparing national methodologies. The general trend is an acceptance of the principle, followed by a move towards progressive application, usually starting with spent fuel storage in reactor ponds, followed by spent fuel transport, and finally spent fuel process plant. Each progressive application may arguably be more complex and involves more issues. The USA currently applies burnup credit for reactor pond storage, is considering transport issues and may eventually include burnup credit in long term repository storage / disposal. France already applies limited burnup credit in areas of the nuclear industry using conservative assumptions, and may extend this approach to other applications in the future.

Burnup Credit

HSE has published guidance (2) for its inspectors on assessing the adequacy of operator's safety cases. The Safety Assessment Principles recognised at an early stage the possible inclusion of burnup credit (SAP 284) in criticality safety cases and NII has recently reconsidered and re-endorsed the principle of burnup credit as a possible tool for inclusion in safety case methodology. This is seen as being complementary to traditional methods of criticality which employ the double contingency principle - our regulatory principles being based upon the safety requirements for adequate redundancy and diversity with defence in depth. Other relevant principles included reliance on the hierarchy of operator based controls (SAP P61/P62) validation and verification of codes and methods. (SAP P87). This paper only discusses the general regulatory principles which are applicable to burnup credit applications.

As regulators we have been aware for some time of the international developments in this area of criticality safety and, where appropriate, have encouraged UK participation in collaborative initiatives. From time to time we have been approached by operators to give a regulatory steer on our views on the application of burnup credit as industry has developed its own ideas and arrangements. Our agreement to implement this methodology has been delayed because of potential conflict between commercial and safety related applications so far as UK operations are concerned. Our regulatory priorities are generally focused towards the inspection and assessment of safety issues relating to operational nuclear plant. Until recently burnup credit cases which had been presented to us appeared to contain significant elements which were largely of a commercial nature and therefore attracted low regulatory priority within the constraints of limited regulatory resources. This situation changed towards the end of last year when a UK licensee presented a case which appeared to have significant safety implications. As a result HSE/NII has reviewed its regulatory priorities to accelerate its development of a position with respect to regulatory acceptance or otherwise of burnup credit.

Issues of possible Regulatory Concern

At first sight the application of burnup credit appears to offer a number of potential disadvantages when viewed from a regulatory safety assessment perspective. These might include the move away from the conservative 'fresh fuel' assumption, removal of some layers of defence in depth for contingency purposes, greater reliance upon codes and supporting experiments and the potential for additional reliance upon operator control in preference to engineered safeguards. One of the key issues to emerge has been the development of an adequate safety case by the operator which considers nuclear safety in a holistic manner in order to demonstrate that when implemented on the plant the overall safety of the proposed operation (plant or site) is at least as safe with the inclusion of burnup credit rather than less safe.

When assessing the adequacy of specific applications we might expect to find some or all of the following aspects addressed:

(i) Case by case assessment

In the UK, the nuclear regulatory regime for operational plants is based upon a non-prescriptive permissioning licensing system - in essence the operator has the flexibility to develop safety cases suited to local applications but has the responsibility of demonstrating the adequacy of this approach to the regulator if challenged. As regulators NII have seen a number of different proposals and suggestions which might invoke burnup credit for criticality control. NII believes that there are advantages in adopting flexibility in its regulatory approach, in contrast to setting prescriptive rules and guidance. This is in order to provide flexibility in assessment of different applications, for example:

(a) use of burnup credit for normal operations in contrast to applications which mitigate the consequences of remote faults (which may place less demands on the burnup assumption),
(b) differences in regulatory frameworks, transport and on-site operations,
(c) application such as long-term waste storage vs current reprocessing operations
(ii) Operational versus Engineered Control

NII's Safety Assessment Principles [2] in SAPs P61/P62 identify a clear preference for robust engineered controls which employ diversity and redundancy for use in criticality safety cases in contrast to reliance upon operational or administrative controls. The application of burnup credit arguments in criticality provides opportunities for including engineered safety systems, for example, gamma and neutron radioactive assay systems - but early experience in the UK has highlighted some additional necessity for operator controls, for example the need to change software constraints periodically, identify and apply correct correlation algorithms between reactor records and fuel elements in process. NII's regulatory expectation is that whilst some additional reliance upon operator control may almost be inevitable, such aspects should be robustly addressed in the safety case - perhaps by inclusion of appropriate human factors analysis or probabilistic risk assessment to assist in the balancing of plausible fault sequences.

(iii) Safety Margins/Phased Approach

Although the principle of applying burnup credit may appear to be straightforward, there is little current operational experience of applying this methodology in the UK. NII sees advantages in adopting a prudent and cautious approach when applying burnup credit to reflect the need to gain confidence in this technical area. Areas which may benefit from a period of 'bedding-in' are:

(b) validation of methods particularly where extrapolation of experiment results have been used
(c) compliance arrangements with safety case parameters such as 'residual enrichment' or fuel element burnup;
(d) increased reliance on safety engineered and/or administrative systems. This could be reviewed with additional operator experience.
(e) the measurements of fuel parameters for compliance testing, particularly the development and application of appropriate calibration data for equipment used for this purpose, in other words equipment should be fit for intended use.

For these reasons NII sees advantages in using an additional margin of safety on reactivity than might otherwise be the case.

In adopting a phased approach, NII sees advantages in, for example, adopting an 'actinide only' assumption rather than 'actinide plus fission products'. This follows a similar line to that suggested in recent USNRC guidance [3].

As regulators we also expect licensees to adopt a prudent and cautious approach to numerical technical areas such as safety margins, validation and verification since there is little operational experience from which to gain confidence in this new area. In view of this, it may be prudent to consider applying burnup credit in an incremental manner. This would allow for the build up of operational experience with more easily characterised fuel, such as standard PWR, before considering more complex cases, such as BWR with mixed enrichments and burnable poisons, or eventually MOX fuel.

(iv) Justification / Risk Assessment

As part of its arrangements, NII has a need to prioritise its regulatory activities based upon its perception of risk to workers, the public and other social/political factors. Sometimes commercially driven applications from licensees are not matched by the same regulatory priority because of a necessarily constrained limitation on regulatory resources. This may be relevant to burnup credit. On a local plant process level, it is almost certainly the case that the quantified risk using a burnup credit assumption will be higher i.e. higher risk than using a fresh-fuel assumption even though k-eff may be nominally unchanged - this reduction in safety margin is because of the additional compliance aspects invoked by using burnup credit. Simply put another way, a storage pond with a limiting average density of X fuel elements per square metre is almost certainly more safe when assessed against the fresh fuel assumption than a limiting storage density of f.x where f is greater than one assuming burnup credit all other things being equal. In the UK, this is sometimes termed 'reverse ALARP' where proposals for modification run contrary to principles embedded in UK legislation [1] requiring employers to seek improvements in safety where reasonably practical. NII has suggested that it might be helpful for operators to produce safety arguments which consider the risk detriment at a local level balanced by the wider risk gain or offset in other areas such as waste minimisation, worker exposure, conventional safety hazards which may be used as part of wider regulatory considerations.

(v) Compliance

It is important that the plant safety case of the licensee identifies appropriate parameters which adequately characterise the fuel burnup or reactivity credit, examples might be 'residual enrichment' or 'burnup'. There are clear advantages for operators in establishing arrangements which provide a clear and understandable audit trail which demonstrates compliance with the safety case and ownership by the plant operators. We have seen examples where avoidable non-compliances have resulted in a loss of stakeholder confidence.

Our regulatory oversight would expect to find consideration by the operator of key determinants of the burnup credit assumption including the measurement of fuel burnup and residual fissile content. It is also necessary for the operator to demonstrate rigorous control of fuel identification and fuel element handling operations. We would expect licensees to be able to take account of prior experience of plant operations, for example - a good track record relating to measurement and control regimes, as part of the detailed justification for adjusting any safety margins to take credit for burnup.

The robustness of such parameters is of key importance. For example, a strong safety case may be expected to show such parameters being routinely measured as part of compliance testing. Additionally, good evidence would be given that a firm calibration base has been established which is not vulnerable to potential common systematic errors in fuel parameter predictions and measurements.

CONCLUSION

The Health and Safety Executive has accepted the application of burnup credit methodology for use in criticality applications in UK nuclear plants. It has advocated the use of prudence and caution when considering the application of burnup credit in safety cases which may be assessed for adequacy on a case by case basis. The Health and Safety Executive has concluded that providing an adequate safety case can be made which justifies an overall improvement in nuclear safety then the application of burnup credit might be applied. It is expected that a regulatory view on a specific UK plant safety case application will be given this year.

The views expressed in this paper are those of the authors and not necessarily the Health and Safety Executive or R M Consultants.

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Risk informed criticality process

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Abstract. This paper will introduce and demonstrate the 'Risk-Informed' criticality process developed to investigate the pre and post closure potential for criticality at a high-level waste repository. This process is based on the US Nuclear Regulatory Commission (NRC), Title 10 Code of Federal Register (CFR) Part 63. The risk-informed, performance-based methodology presented is consistent with the Code of Federal Regulations (CFR) Title 10, Part 63 (64 FR 8640). Part 10 CFR 63 specifies the overall performance objectives of the potential repository at Yucca Mountain prior to closure and during postclosure. The overall performance of the repository is specified for postclosure (10 CFR 63.113) in terms of expected annual dose to the average member of the critical group. There are no specific design criteria for postclosure criticality control in 10 CFR 63. This regulation is a risk-informed, performance-based regulation, which treats criticality as one of the processes or events that must be considered for the overall system performance assessment.

INTRODUCTION

Limiting the potential for, and consequences of, criticality during the postclosure phase of the geologic repository relies on multiple barriers, both natural and engineered. The natural barrier system consists of the climate around, and the rock formations of, the repository, and includes the geologic, mechanical, chemical, and hydrological properties of the site. As defined within 10 CFR 63, the engineered barrier system (EBS) comprises the waste packages and the underground facility in which they are emplaced. A waste package is the generic term for describing the waste form (radioactive waste and any encapsulating or stabilizing matrix) and any containers, shielding, packing, and other absorbent materials immediately surrounding an individual package. The underground facility consists of the underground structure, backfill materials, if any, and openings that penetrate the underground structure (e.g., ramps, shafts, and boreholes, including their seals). The EBS will work in concert with the natural barrier system to minimize the potential for conditions that would be conducive to a criticality event after the repository has been permanently closed.

As is shown in Figure 1, if the total probability of criticality falls below the regulatory threshold established in 10CFR63.114(d), no criticality consequence analyses should be required. This position is supported by the third paragraph of Section 3.2.2 in the Topical Report.[1] Two probability thresholds are discussed in the Topical Report. The higher probability threshold, or design threshold, is 1.0 over the 10 000 years regulatory period, or an average frequency of 1×10^{-4} criticalities per year[2]. The design threshold triggers a waste package redesign if the probability of a criticality event sequence exceeds this value. The lower probability threshold, or credibility threshold, is 1×10^{-4} over the 10 000 years regulatory period, or an average frequency of 1×10^{-8} criticalities per year. This threshold is established by 10CFR63.114(d). If the total probability of criticality (sum of all criticality even sequences) is below this threshold, criticality is screened out from further evaluation. If the total probability of criticality falls between these two thresholds, the Topical Report commits to the performance of consequence calculations for each event sequence leading to a potential criticality.

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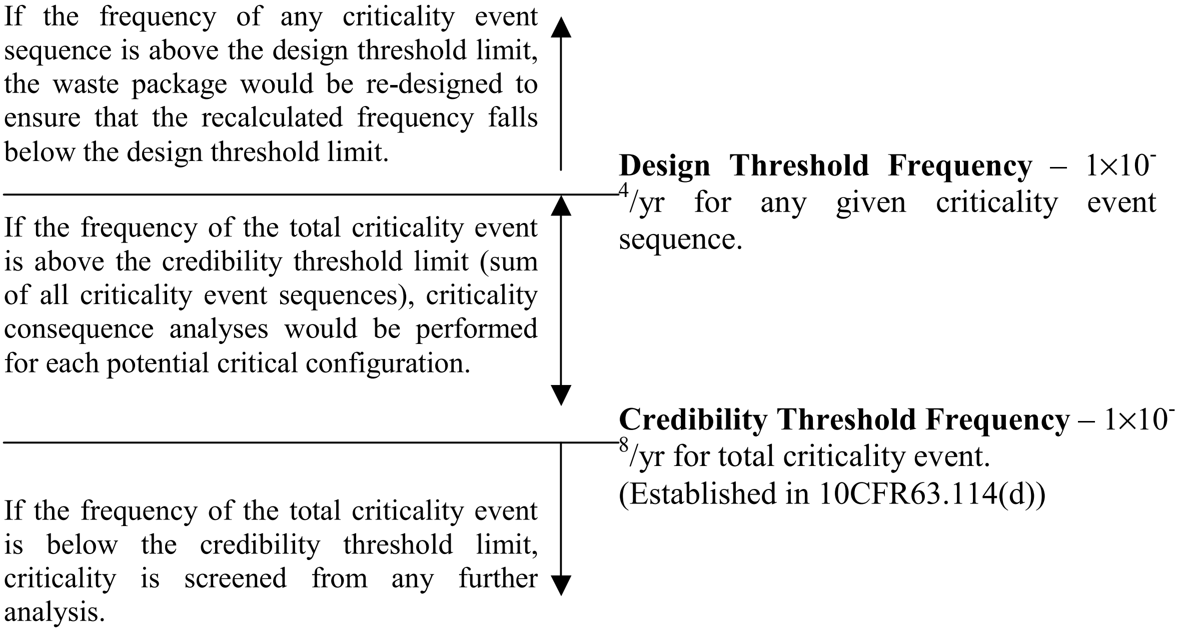
[1] "If any configurations were determined to be capable of supporting criticality events and found to have an estimated probability of occurrence below the design probability criterion, but contribute to a total probability of criticality for the entire repository inventory above the 10 CFR 63.114(d) screening probability threshold of 10^{-4} in 10 000 years, consequence analyses would be performed. Only the criterion in 10 CFR 63.114(d) will be used for screening criticality events from further consideration in the TSPA. The probabilities tested against this screening threshold will be the sum of the probabilities for all the scenarios that can lead to an individual criticality FEP."

[2] Although this is how this criteria is established in the text of the Topical Report, the intent was to have the design threshold probability the same as the credibility threshold probability established in 10CFR63.114(d). It is anticipated that the Topical Report will be revised to correct this discrepancy.



If the frequency of any criticality event sequence is above the design threshold limit, the waste package would be re-designed to ensure that the recalculated frequency falls below the design threshold limit.

Design Threshold Frequency – 1×10^{-4}/yr for any given criticality event sequence.

If the frequency of the total criticality event is above the credibility threshold limit (sum of all criticality event sequences), criticality consequence analyses would be performed for each potential critical configuration.

Credibility Threshold Frequency – 1×10^{-8}/yr for total criticality event. (Established in 10CFR63.114(d))

If the frequency of the total criticality event is below the credibility threshold limit, criticality is screened from any further analysis.

FIG. 1. Threshold Limits as Currently Published

The following example was provided to the US NRC for review and support of the Yucca Mountain Site Recommendation documentation and is intended as an example of the process not the final evaluation. So as the technical work is developed for License Application it is anticipated that additional refinements to the process will be implemented, while following the methodology described in the post closure criticality topical report. (CRWMS 2001a)

Scenario Development

As a demonstration of this process, the following example is presented based on the potential for waste package criticality prior to 10 000 years assuming the early failure of three waste packages. A probability of unity is used for this event although the probability of three non-mechanistic, early waste package failures is predicted as 0.002 in *Yucca Mountain Supplemental Science and Performance Analyses* (BSC 2001a, Section 7.3.6). Subsequent to the early waste package failures, several independent events must occur in order to have a criticality event in the failed waste packages. These events include:

The potential for sufficient water flow to enter the waste package failure location either through (1) a flow path from the mountain surface or (2) condensation from the underside of the drip shield.

For scenario (1) above, the potential for a drip shield failure to allow water flow from the drift overhead to the failed waste package.

The potential that the water source is horizontally and vertically aligned to allow flow into the waste package failure location. For scenario (1) above, this includes the potential that a failed drip shield is aligned over a failed waste package and that seepage from the drift overhead strikes the drip shield and waste package in a location that would allow the water to enter the waste package failure location. For scenario (2) above, this includes the potential that the condensation is released above the failed waste package and that it strikes the failed waste package in a location that would allow the water to enter the waste package failure.

The potential that the waste package failure location and geometry is such that water could enter and be retained in the waste package in a sufficient volume and for a sufficient period of time to allow for waste form degradation and moderation. Other issues to be considered in the evaluation of this issue include whether the failure location becomes plugged with corrosion products or water impurities and whether the heat generation rate of the waste form is below the evaporation rate of the water inflow.

The potential that the waste form degradation process would allow for a critical configuration. This includes the potential that: (1) the waste form contains sufficient fissile material to become critical; (2) the neutron absorber material is flushed from the waste form matrix; (3) the corrosion products are flushed from the waste form matrix; and (4) the waste form degrades to a critical configuration.

The following sections discuss these events in detail. In some instances, a preliminary quantification of these events is presented.

Availability of Water

The presence of water is essential to the occurrence of waste package criticality. This necessitates the availability of a sufficient and focused water source entering the failed waste package's inner shell. Water flow is surmised to be potentially available from two independent sources: (1) focused flow from the mountain surface to the drift overhead directly above the failed waste package and (2) condensation from the underside of the drip shield directly above the failed waste package.

The availability of a focused flow of water from the mountain surface to the drift directly above the failed waste package in a sufficient quantity to flood a waste package is dependent on two factors. The first factor is the probability that the climate of southern Nevada could support such a flow rate. This probability is given as unity in *Total System Performance Assessment for Site Recommendation* (BSC 2001d, Table 3.2-1). It is based on an analysis that a glacial transition climate is expected to occur within the next 2,000 to 10,000 years. The second factor accounts for seepage from the mountain surface to the emplacement drift above any given waste package. The most likely seepage fraction is predicted to be 0.17.

Therefore, the estimated probability of focused flow infiltrating into the mountain surface to the drift directly above the failed waste package is estimated to be 0.17 (1.0 x 0.17).

The second water source considered is condensation from the underside of the drip shield directly above the failed waste package. The probability is conservatively assumed to have a value of unity until an evaluation establishing this value is performed. However, this water source is considered unlikely. In order for water to physically drip from the underside of the drip shield onto the failed waste package, the drip must occur at or near the apex of the drip shield. Condensation flow from other parts of the drip shield surface would either flow down the sides of the drip shield due to gravity or impinge on the waste package surface at a point where flow would not be accessible to the waste package failure location.

Drip Shield Failure

In the water source scenario in which there is a focused water flow from the mountain surface to the drift overhead directly above the failed waste package, it is necessary for the drip shield above the failed waste package to also be failed. This is necessary to allow the water to flow from the drift overhead, through the drip shield failure location, and onto the failed waste package. A drip shield failure can occur due to the following three factors:

Emplacement Error——For the continuous drip shield configuration, a drip shield segment is misaligned during emplacement resulting in a gap between two segments. Drip shield segments are designed to interlock to provide a continuous protective umbrella for the waste packages along the length of the emplacement drift.

Fabrication Error—The drip shield is not fabricated to specifications due to the use of improper materials or bad welds.

Rock Fall Event—A rock fall event can cause the drip shield to breach either through direct impact or via stress corrosion cracking. The residual stresses in the drip shield resulting from a rock fall as small as 1 MT (metric ton) have been calculated to be sufficient to allow for stress corrosion cracking.

Drip Shield Emplacement Error

The probability of drip shield emplacement error is calculated using the binomial model (Walpole et al., 1998, p. 118). The binomial model assumes a given number of drip shield emplacement errors will occur in the repository during the preclosure period and that each drip shield emplacement is an independent event. The binomial model is defined as:

$$P_n(x) = \binom{n}{x} p^x (1-p)^{n-x}$$

where: p is the probability of a single emplacement error
n is the total number of emplacements
x is the expected number of emplacement errors

The probability of having at least one drip shield emplacement error is calculated by subtracting the probability of having no drip shield emplacements errors from one. To calculate this probability, the following values are input into the equation above: a single drip shield emplacement error probability (variable p) of 9.0×10^{-5} (CRWMS M&O 2000a, p. 62); and the emplacement of 11 538 drip shields (variable n) [11 538 = 60 000 m of total drift length (DOE 2001c, Table 2-2) divided by 5.2 m per drip shield segment (CRWMS M&O 2000b, p. II-1)]. Using these inputs, the probability of having at least one drip shield emplacement error is calculated to be 6.5×10^{-1}.

$$1 - P_{11,538}(0) = \{1 - [(9.0 \times 10^{-5})^0 (1\text{-}9.0 \times 10^{-5})^{11,538}]\} = 6.5 \times 10^{-1}$$

Drip Shield Fabrication Error

A probability of 2.4×10^{-3} is estimated for drip shield failure due to fabrication errors. This probability is based on the waste package fabrication error flaw information obtained from

CRWMS M&O (2000a, p. 44). Using the lognormal cumulative flaw distribution equation of CRWMS M&O (2000a, p. 34), the probability of a weld flaw 7.5 mm deep (half the depth of the weld on the drip shield) is estimated to be 1.04×10^{-3}. An arbitrarily doubling of this weld flaw probability results in 2.08×10^{-3}. Multiplying by the probability of weld inspection failure (1×10^{-4}) presented in CRWMS M&O (2000a, p. 34) and the number of drip shield segments fabricated (11,538 [60 000 m total drift length/5.2 m per drip shield segment]) results in a probability of drip shield fabrication error of 2.4×10^{-3}. Verification or replacement of this value is dependent on future analyses.

Rock Fall Event onto Drip Shield

A probability of 7.4×10^{-3} is estimated for drip shield breach due to a rock fall event. This probability is calculated using the expected number of key blocks greater than 1 MT. It is estimated that there are 60 key blocks of this magnitude based on information in CRWMS M&O (2000e, p. 13, Equation 6). All key blocks identified are assumed to fall within the study period (10,000-years) as no rock fall frequency is available. The total effective length of the key blocks is estimated to be 329 m [length per rock mass obtained from CRWMS M&O (2000d, Table IX-2)]. In addition to the key block length, residual stresses in the drip shield that may result in stress corrosion cracking can occur up to 1 m on either side of the key block impact point (BSC 2001c). This results in a potential drip shield failure length of 442 m out of a total drip shield length of 60 000 m, or a probability of 7.4×10^{-3}. This estimate assumes that plugging of the drip shield breach by precipitates or corrosion products will not occur.

Cumulative Drip Shield Failure Probability

The cumulative probability of drip shield failure is estimated to be 6.5×10^{-1}. This probability is dominated by the drip shield emplacement error probability. The probability that a failed drip shield is located at a specific location that would allow the focused flow to impinge on a failed waste package is estimated to be one divided by the total number of drip shields. This probability is estimated to be 8.7×10^{-5} (5.2 m / 60 000 m; where 5.2 m is the length of a drip shield segment [CRWMS M&O 2000b, p. II-1] and 60,000 m is the total length of the emplacement drifts [DOE 2001c, Table 2-2]). The probability that a failed drip shield is located at a specific location is calculated by multiplying the cumulative drip shield failure probability (6.5×10^{-1}) by the drip shield location probability (8.7×10^{-5}). This value is calculated to be 5.7×10^{-5}. To date, no dependency has been identified for a common mode of drip shield failure and focused flow onto the failed drip shield events.

Focused Flow Onto Failed Waste Package

In order for the water source to enter the failed waste package through the failure location, the water flow must be horizontally and vertically aligned in a specific configuration. The following subsections discuss the necessary alignments. Both the horizontal and vertical alignment evaluations assume that the waste package is orientated (tilted) in a manner that is optimal for the water flow to enter the waste package failure location.

Horizontal Alignment of Flow

In either water source scenarios presented in Section 3.1, the water source must impact on the failed waste package in order to allow flow into the waste package failure location. Because of the configuration of the waste package outer barrier trunnion collar, the flow from either source (drip shield breach or condensation from the underside of the drip shield) must impact no more than 165 mm from the closure end of the waste package (CRWMS M&O 2000c, Att. I, SK-0175 Rev 02, Sheet 2 of 2). Otherwise the raised profile of the trunnion collar will prevent flow from reaching the failure location. This results in a probability that the failed waste package will be under the failed drip shield of 8.4×10^{-6}. This probability estimate assumes an average waste package length of 5 m and the three failed waste packages out of a total of 11 770 waste packages emplaced.

$8.4 \times 10^{-6} = (0.165 \text{ m} / 5 \text{ m})(3 / 11\,770)$.

Vertical Alignment of Flow

For the water source scenario from the mountain surface to the drift overhead, it is estimated that the flow from the drift overhead onto the drip shield and from the drip shield breach onto the waste package must occur within 3.5° of the drip shield or waste package apex. A 7° arc (2 × 3.5°) is based on a maximum waste package tilt angle of 3.3° that can be achieved due to failure of one end of the waste package pallet.

$3.3° = \sin^{-1} (230 \text{ mm} / 4024.8 \text{ mm})$

where: 230 mm is the height of the waste package above the intact end of the waste package pallet

$230 \text{ mm} = 1012 \text{ mm} – (1564 \text{ mm} / 2)$

where: 1012 mm is the height from emplacement drift floor to the 21-PWR wastepackage radial center (CRWMS M&O 2000b, Att. II, SK-0144 Rev 01, p. II-3)

1 564 mm is the outer barrier diameter of the 21-PWR waste package (CRWMS M&O 2000c, Att. I, SK-0175 Rev 02, p. 1 of 2)]

4 024.8 mm is the length of the waste package from the collapsed pallet end to the intact pallet support

$4\,024.8 \text{ mm} = 430 \text{ mm} + (4\,147.2 – 552.4 \text{ mm})$

where: 430 mm is the length of bottom trunnion collar on the 21-PWR waste package (CRWMS M&O 2000c, Att. I, SK-0175 Rev 02, p. 2 of 2)

4 147.2 mm is the length of the waste package pallet (CRWMS M&O 2000b, Att. III, SK-0144 Rev 01, p. III-1)

552.4 mm is the length of the pallet waste package support (CRWMS M&O 2000b, Att. III, SK-0144 Rev 01, p. III-1)

Otherwise, the flow will roll off the curved sides of either the drip shield or waste package due to gravity rather than flowing horizontally along the drip shield or waste package surface to the failure location. This would further necessitate that the flow source from the drift overhead be within 3.5°of the drift overhead apex. However, the radial arc of the flow from the drift overhead could be greater than 7° if a rock fall results in a drip shield surface depression that could funnel the focused flow from the drift overhead to the drip shield failure location.

The probability that the failure location is on the top half of the waste package is estimated as 0.5 (180°/360°). Together, these conditions result in a flow vertical alignment probability of 2.9×10^{-5} $[(7°/180°)^3 \times (180°/360°)]$. It should be noted that even if the flow impact area is increased to 5° on either side of the structure apex (drift, drip shield, and waste package), the vertical alignment probability would only increase to 8.6×10^{-5}.

The probability that condensation from the underside of the drip shield will enter the waste package failure location has some of the same events listed above. These events are the waste package impingement radius (within 3.5° of the waste package apex) and the waste package failure location (top half of the waste package lid weld). The flow vertical alignment probability for this scenario is estimated to be 1.9×10^{-2} [(7°/180°) x (180°/360°)]. If the flow impact area is increased to 5° on either side of the waste package apex, the vertical alignment probability would only increase to 2.8×10^{-2}.

Waste Package Water Accumulation

The water has to enter the waste package in such a manner that the water can accumulate to sufficient depth and have sufficient flow to allow absorber material and corrosion product removal and provide sufficient moderation to allow criticality to occur. A conservative, but unrealistic, failure configuration would be a discrete failure point on the waste package outer lid weld which is located on the top half of the horizontally emplaced waste package and a complete circumferential failure of the middle and inner lid welds. This configuration is considered unrealistic because "…only the weld region of the outer-lid of the outer barrier would be affected by potential improper heat treatment . . . the inner lid of the outer barrier is not likely to be affected" (DOE 2001a, Section 5.2.4.2). Also, the lid of the inner stainless steel shell would be expected to provide an additional barrier for some period of time. This proposed failure configuration would allow water to enter the waste package and flood the internals up to the height of outer lid weld failure location. The probability that the failed weld will be located on the top half of the emplaced waste package is given in Section 3.3.2 as 0.5.

Once the water is in a position to enter the waste package failure location, four sequential conditions must exist to allow water to accumulate in the waste package. These conditions are:

1. The ability of the flow to enter the waste package failure location given the geometry of the waste package at the weld location.
2. The waste package failure (i.e., weld failure) must of sufficient size to allow the water to penetrate into the waste package internals.
3. Corrosion products and water impurities do not plug the waste package failure.
4. The waste package's waste form decay heat load is sufficiently low that the inflow does not evaporate.

To date, the probabilities of these conditions have not been evaluated and are therefore conservatively assumed to have a value of unity. In addition, the probability of water accumulation in the waste package due to complete failure of the circumferential weld and waste package tilt has not been considered in this evaluation.

Potential for Criticality

Once water has accumulated inside the waste package inner shell, four additional conditions must exist before a criticality event can occur. These conditions are:

1. The waste package internals and/or waste form must degrade into a configuration conducive to criticality.
2. Corrosion products resulting from the waste package and waste form degradation processes must be removed or segregated from the degraded configuration. Corrosion products would displace water in the degraded waste form matrix. Removal or segregation of the corrosion products would therefore allow for greater neutron moderation and a higher potential for criticality.
3. Neutron absorber materials contained in the basket assembly of the waste package internals and within the waste form matrix itself must be removed from the degraded waste form configuration in order to increase the potential for criticality.
4. The waste form (e.g., spent fuel) contained in the waste package has to have sufficient potential to allow criticality to occur.

The probability that the waste form contained in a failed waste package has the potential for criticality is conservatively estimated to be 1.6×10^{-2}. This estimate is based on the annual criticality probability per waste package of pressurized water reactor spent nuclear fuel of 1.4×10^{-10} per year (DOE 1998, Table C-13). The annual criticality probability per waste package is multiplied by 11,770 (the total number of emplaced waste packages [BSC 2001a, Sect. 7.3.6]) and by the 10,000-year regulatory period.

Other Considerations

Several issues have not been considered in the above evaluation. These issues include:

The rate at which water must flow into the waste package in order to accumulate sufficient water to support waste form degradation. The repository is not a closed system and an inadequate or inconsistent inflow could allow for evaporation of accumulated water over a period of time.

The loss of water when the flow splashes as it impacts on the drip shield and waste package surfaces.

Waste package type effects (i.e., some waste packages without an adequate heat generation rate to keep water out do not have a criticality potential and visa versa).

Surface tension effects for the flow of water that would account for the necessary failure size and flow rate necessary to enter failure.

Quantification of Criticality Probability

An event tree with supporting fault trees has been developed to quantify the events and conditions leading to potential for waste package criticality as described above. These trees were developed using the probabilistic risk assessment code SAPHIRE V6.69 (CRWMS M&O 2001). The event tree presenting the evaluation logic for the potential for criticality is given in Figure 2. This event tree consists of eight top events. Each top event represents a separate condition or event that must be met in order to have criticality within the failed waste package. The branching under each top event follows standard event tree convention that requires “success” or “YES” answers to branch up and “failure” or “NO” answers to branch down. The probability of each top event failure branch is provided on the event tree. If a branch has not been quantified, the failure probability is assumed to be unity. The end state of each branching sequence (located in the second to last column of the figure) is either defined as “OK” (sequence has no criticality potential) or “CRITICALITY” (sequence has a potential for criticality). The estimated probability of each “CRITICALITY end state is provided in the last column of Figure 2. A brief definition of each top event follows:

WATER-FLOW	Estimates the probability that an adequate water source will be provided from the mountain surface to the drift overhead
CONDENSATION	Estimates the probability that an adequate water source will be provided from condensation on the underside of the drip shield above the failed waste package
DRIP-SHIELD	Estimates the probability of drip shield failure and the probability that the failed drip shield will be under a water source (this top event is only evaluated in conjuncture with the WATER-FLOW top event)
DRIP-WP-FF	Estimates the probability that the water source will take a pathway from the drift overhead through the drip shield to the failed waste package and enter the waste package failure location (this top event is only evaluated in conjuncture with the WATER-FLOW top event)
DRIP-WP-COND	Estimates the probability that the water source will take a pathway from the underside of the drip shield to the failed waste package and enter the waste package failure location (this top event is only evaluated in conjuncture with the CONDENSATION top event)
FLOW-ACCUM	Estimates the probability that the water striking the failed waste package will enter the failure location and accumulate inside the waste package
FUEL-CRITICAL	Estimates the probability that the waste form contained in the failed waste package will degrade into a configuration favorable to criticality

Each of the top events of Figure 2 and defined above are supported by fault trees. Fault trees are constructed to physically represent the system or event logic. The probability of the basic events comprising the fault trees are obtained from the evaluations are noted; “AND” and “OR” gates are used in the fault trees to represent the dependencies between the basic events. During event tree processing, the fault trees are evaluated to obtain an overall probability for each of the top events. The fault trees for each of the top events are presented in Figures 3 through 9.

From the evaluation, the probability of criticality prior to 10 000 years has been preliminarily estimated for both focused flow and condensation water source scenarios. The probability of criticality for these scenarios is estimated to be 3.9×10^{-17} and 2.7×10^{-9}, respectively. Although both probability values are below the credibility threshold for the postclosure period, the probability for the condensation water source scenario approaches this threshold value. However, the probability for a number of the events identified in this evaluation have not been quantified and it is anticipated that once the probability for these events are quantified the probability of criticality for the condensation water source scenario will also be well below the threshold value.

Summary

This paper has demonstrated the ‘Risk Informed’ process that is being applied in support of determination of important scenarios associated with potential criticality. As is indicated throughout the demonstration, a clear understanding of the processes that will cause the engineered system to degrade and allow a moderator to become intermixed with the radionuclides is the key to developing a logical event tree and in developing the probabilities of occurrence. By simply looking at the probability of occurrence, one can draw a conclusion that the potential for criticality is already extremely remote. As noted throughout the paper additional work is still underway to better quantify some of the values set to unity. It is anticipated that with the further development of the values set to unity the probability of a criticality occurring will decrease further. Coupling the Risk Informed process with the additional understanding of the residual stored, burnup credit, with the probability of moisture being present in sufficient quantity to degrade the engineered barrier one can almost draw the conclusion that criticality is an improbable event. This conclusion is based on the fact that the natural system is working with the engineered system to drive the probabilities sufficiently low as to allow us to draw this conclusion. As has been stated in other documents the design philosophy is based on a ‘defense-in-depth’ process that supports the process demonstrated in this paper. A final note, the process demonstrated above is also applicable to storage and transportation issue. Using the U.S. Nuclear Regulatory Commission’s Risk Informed process will allow a better definition of the important scenarios and focus the design on the key issues that will help reduce any probability of a criticality event from occurring.

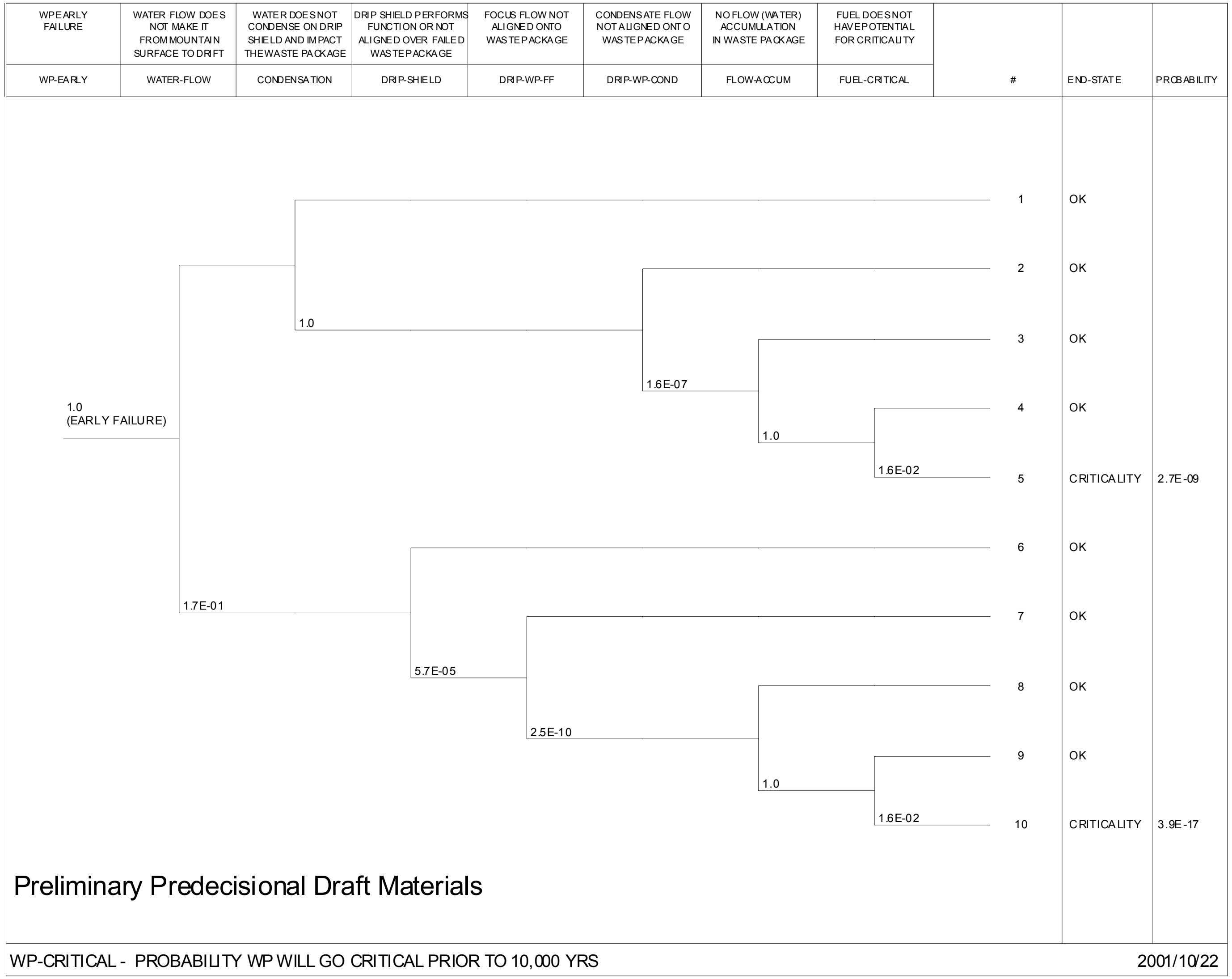
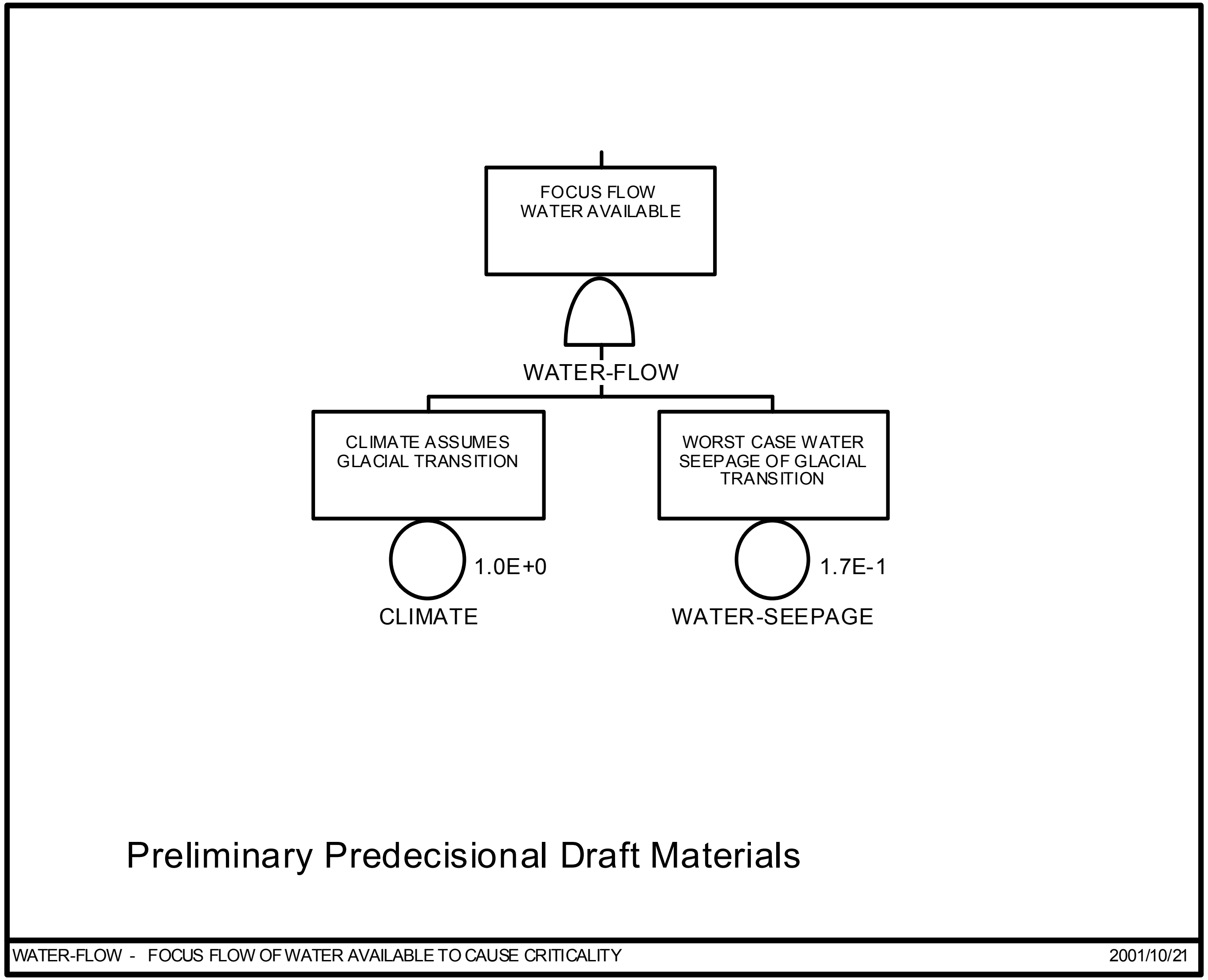

WP EARLY FAILURE
WATER FLOW DOES NOT MAKE IT FROM MOUNTAIN SURFACE TO DRIFT
WATER DOES NOT CONDENSE ON DRIP SHIELD AND IMPACT THE WASTE PACKAGE
DRIP SHIELD PERFORMS FUNCTION OR NOT ALIGNED OVER FAILED WASTE PACKAGE
FOCUS FLOW NOT ALIGNED ONTO WASTE PACKAGE
CONDENSATE FLOW NOT ALIGNED ONTO WASTE PACKAGE
NO FLOW (WATER) ACCUMULATION IN WASTE PACKAGE
FUEL DOES NOT HAVE POTENTIAL FOR CRITICALITY
WP-EARLY
WATER-FLOW
CONDENSATION
DRIP-SHIELD
DRIP-WP-FF
DRIP-WP-COND
FLOW-ACCUM
FUEL-CRITICAL
#
END-STATE
PROBABILITY
1.0
(EARLY FAILURE)
1.0
1.6E-07
1.0
1.6E-02
1.7E-01
5.7E-05
2.5E-10
1.0
1.6E-02
1 OK
2 OK
3 OK
4 OK
5 CRITICALITY 2.7E-09
6 OK
7 OK
8 OK
9 OK
10 CRITICALITY 3.9E-17
Preliminary Predecisional Draft Materials
WP-CRITICAL - PROBABILITY WP WILL GO CRITICAL PRIOR TO 10,000 YRS
2001/10/22

FIG.2. Event Tree for Evaluating a Potential Criticality Event.


FOCUS FLOW
WATER AVAILABLE
WATER-FLOW
CLIMATE ASSUMES GLACIAL TRANSITION
WORST CASE WATER SEEPAGE OF GLACIAL TRANSITION
1.0E+0
CLIMATE
1.7E-1
WATER-SEEPAGE
Preliminary Predecisional Draft Materials
WATER-FLOW - FOCUS FLOW OF WATER AVAILABLE TO CAUSE CRITICALITY
2001/10/21

FI.G. 3. Focused FloW.

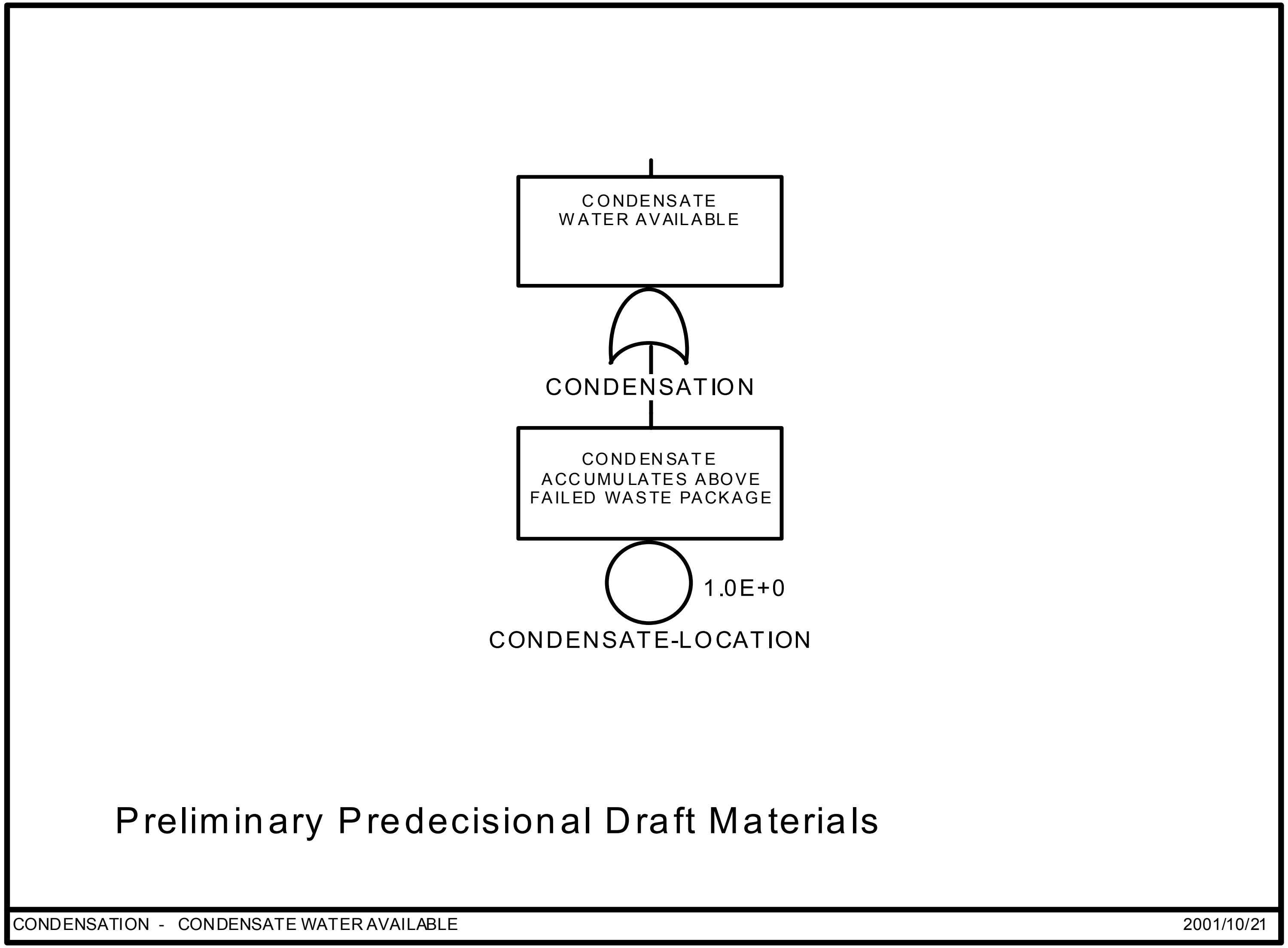

CONDENSATE WATER AVAILABLE
CONDENSATION
CONDENSATE ACCUMULATES ABOVE FAILED WASTE PACKAGE
1.0E+0
CONDENSATE-LOCATION
Preliminary Predecisional Draft Materials
CONDENSATION - CONDENSATE WATER AVAILABLE
2001/10/21

FIG. 4. Probability of Condensation Under Drip Shield Fault Tree.

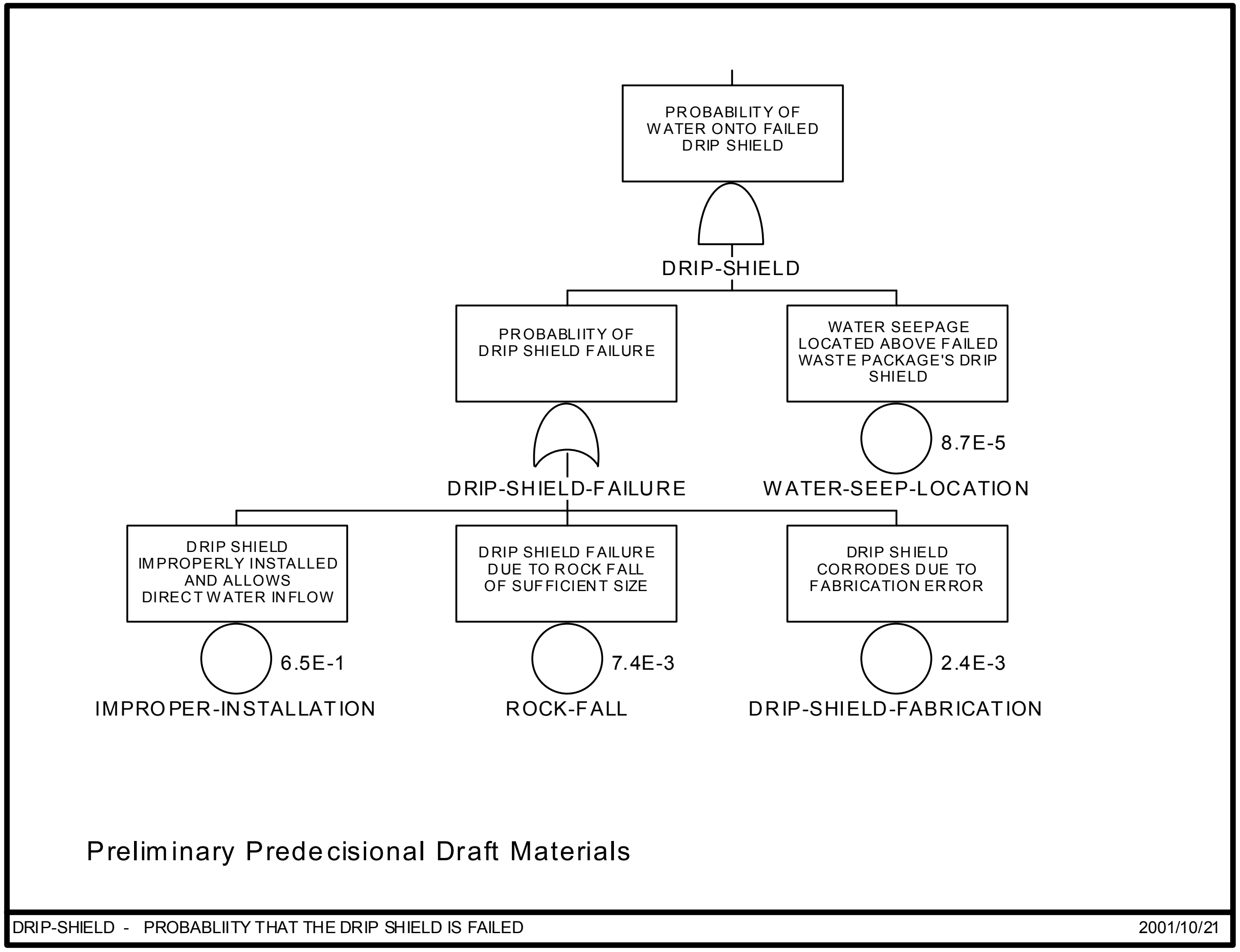

PROBABILITY OF WATER ONTO FAILED DRIP SHIELD
DRIP-SHIELD
PROBABLIITY OF DRIP SHIELD FAILURE
WATER SEEPAGE LOCATED ABOVE FAILED WASTE PACKAGE'S DRIP SHIELD
8.7E-5
DRIP-SHIELD-FAILURE
WATER-SEEP-LOCATION
DRIP SHIELD IMPROPERLY INSTALLED AND ALLOWS DIRECT WATER INFLOW
DRIP SHIELD FAILURE DUE TO ROCK FALL OF SUFFICIENT SIZE
DRIP SHIELD CORRODES DUE TO FABRICATION ERROR
6.5E-1
7.4E-3
2.4E-3
IMPROPER-INSTALLATION
ROCK-FALL
DRIP-SHIELD-FABRICATION
Preliminary Predecisional Draft Materials
DRIP-SHIELD - PROBABLIITY THAT THE DRIP SHIELD IS FAILED
2001/10/21

FIG. 5. Probability of Drip Shield Failure Fault Tree.

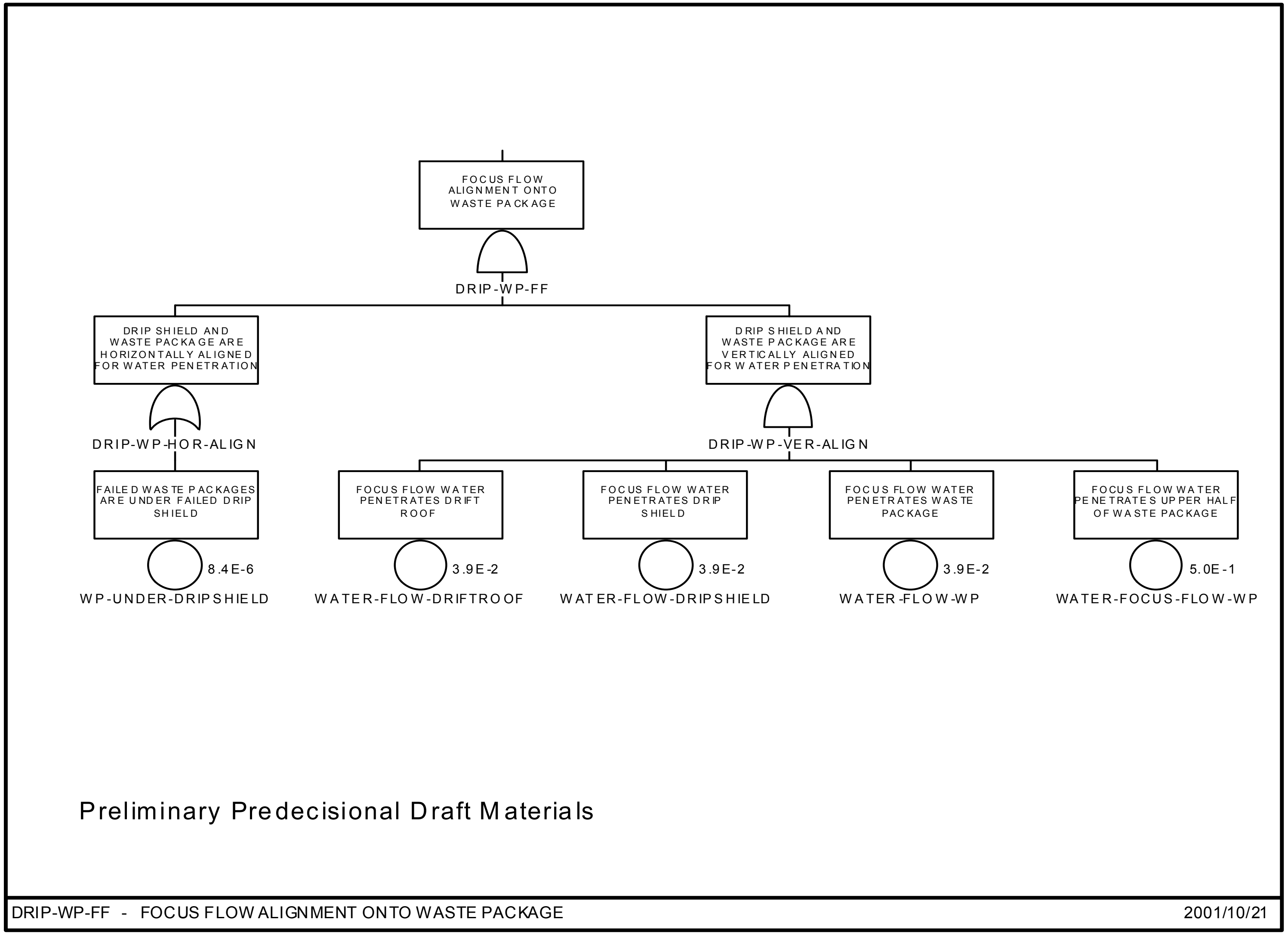

FOCUS FLOW ALIGNMENT ONTO WASTE PACKAGE
DRIP-WP-FF
DRIP SHIELD AND WASTE PACKAGE ARE HORIZONTALLY ALIGNED FOR WATER PENETRATION
DRIP-WP-HOR-ALIGN
DRIP SHIELD AND WASTE PACKAGE ARE VERTICALLY ALIGNED FOR WATER PENETRATION
DRIP-WP-VER-ALIGN
FAILED WASTE PACKAGES ARE UNDER FAILED DRIP SHIELD
8.4E-6
WP-UNDER-DRIPSHIELD
FOCUS FLOW WATER PENETRATES DRIFT ROOF
3.9E-2
WATER-FLOW-DRIFTROOF
FOCUS FLOW WATER PENETRATES DRIP SHIELD
3.9E-2
WATER-FLOW-DRIPSHIELD
FOCUS FLOW WATER PENETRATES WASTE PACKAGE
3.9E-2
WATER-FLOW-WP
FOCUS FLOW WATER PENETRATES UPPER HALF OF WASTE PACKAGE
5.0E-1
WATER-FOCUS-FLOW-WP
Preliminary Predecisional Draft Materials
DRIP-WP-FF - FOCUS FLOW ALIGNMENT ONTO WASTE PACKAGE
2001/10/21

FIG. 6. Probability of Focused Flow From Drift Overhead Onto Failed Waste Package Fault Tree.

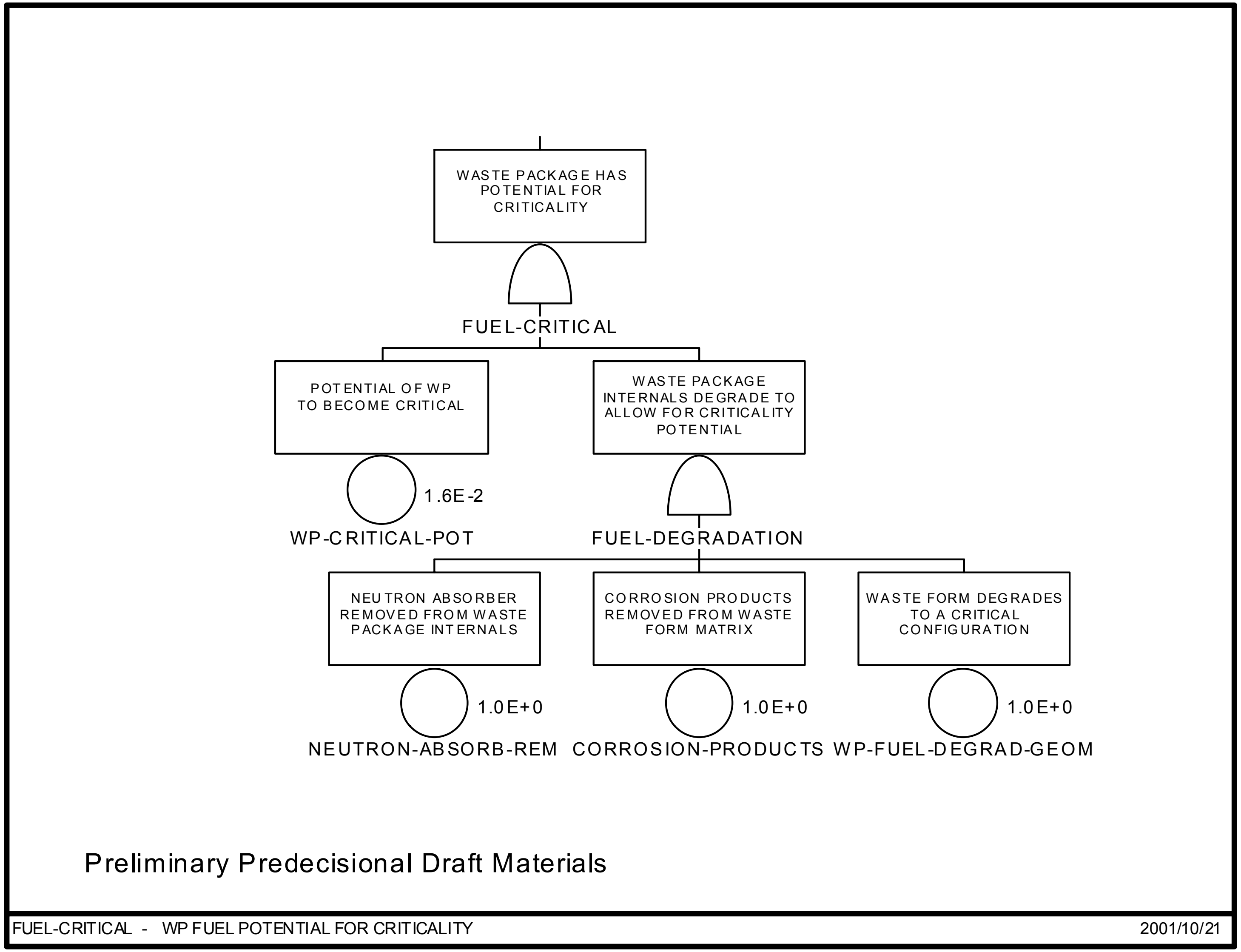

WASTE PACKAGE HAS POTENTIAL FOR CRITICALITY
FUEL-CRITICAL
POTENTIAL OF WP TO BECOME CRITICAL
1.6E-2
WP-CRITICAL-POT
WASTE PACKAGE INTERNALS DEGRADE TO ALLOW FOR CRITICALITY POTENTIAL
FUEL-DEGRADATION
NEUTRON ABSORBER REMOVED FROM WASTE PACKAGE INTERNALS
1.0E+0
NEUTRON-ABSORB-REM
CORROSION PRODUCTS REMOVED FROM WASTE FORM MATRIX
1.0E+0
CORROSION-PRODUCTS
WASTE FORM DEGRADES TO A CRITICAL CONFIGURATION
1.0E+0
WP-FUEL-DEGRAD-GEOM
Preliminary Predecisional Draft Materials
FUEL-CRITICAL - WP FUEL POTENTIAL FOR CRITICALITY
2001/10/21

FIG. 7. Probability of Focused Flow from Condensation Under Drip Shield Onto Failed Waste Package Fault Tree.

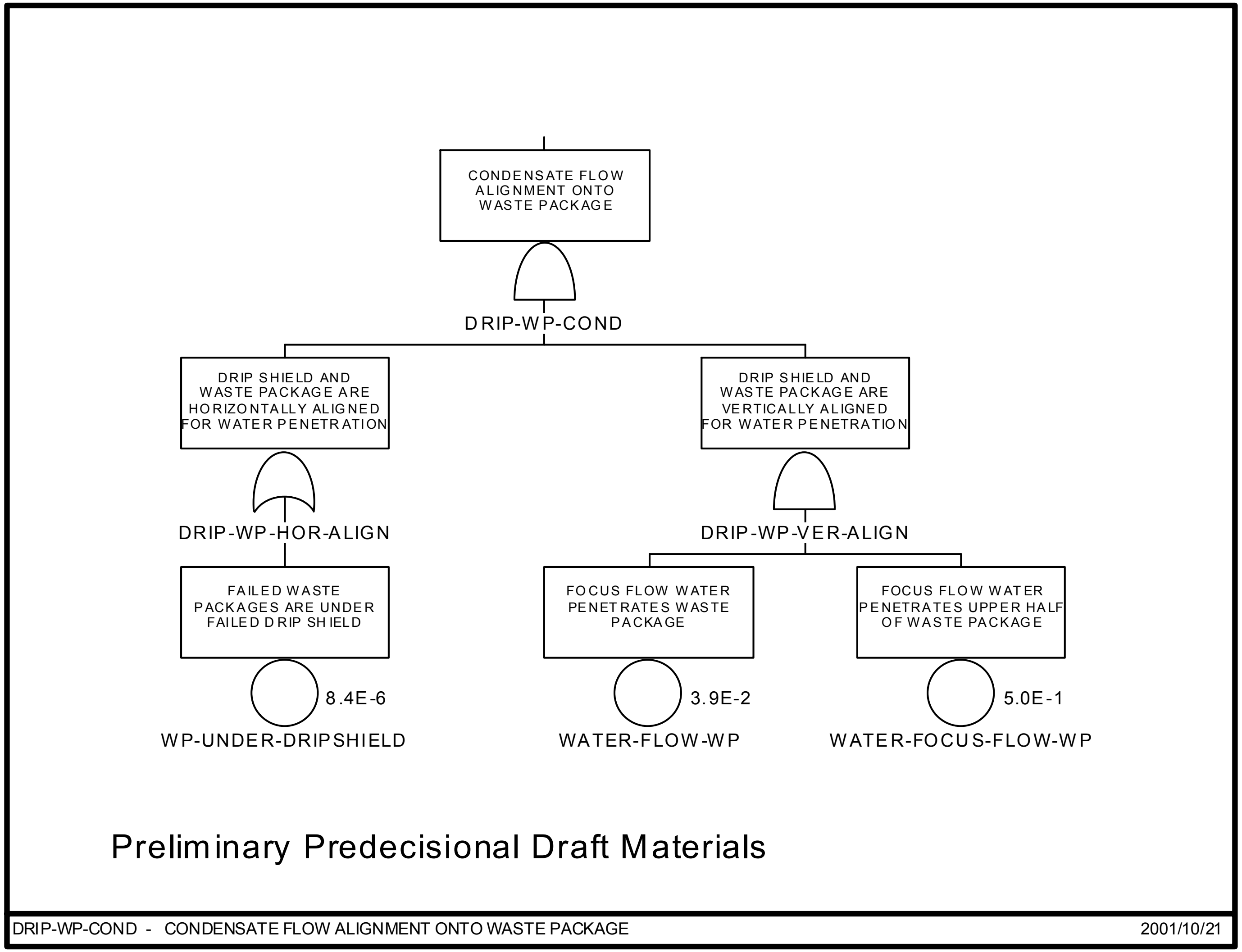

CONDENSATE FLOW ALIGNMENT ONTO WASTE PACKAGE
DRIP-WP-COND
DRIP SHIELD AND WASTE PACKAGE ARE HORIZONTALLY ALIGNED FOR WATER PENETRATION
DRIP-WP-HOR-ALIGN
DRIP SHIELD AND WASTE PACKAGE ARE VERTICALLY ALIGNED FOR WATER PENETRATION
DRIP-WP-VER-ALIGN
FAILED WASTE PACKAGES ARE UNDER FAILED DRIP SHIELD
8.4E-6
WP-UNDER-DRIPSHIELD
FOCUS FLOW WATER PENETRATES WASTE PACKAGE
3.9E-2
WATER-FLOW-WP
FOCUS FLOW WATER PENETRATES UPPER HALF OF WASTE PACKAGE
5.0E-1
WATER-FOCUS-FLOW-WP
Preliminary Predecisional Draft Materials
DRIP-WP-COND - CONDENSATE FLOW ALIGNMENT ONTO WASTE PACKAGE
2001/10/21

FIG. 8. *Probability of Water Flow Into Failed Waste Package Fault Tree.*

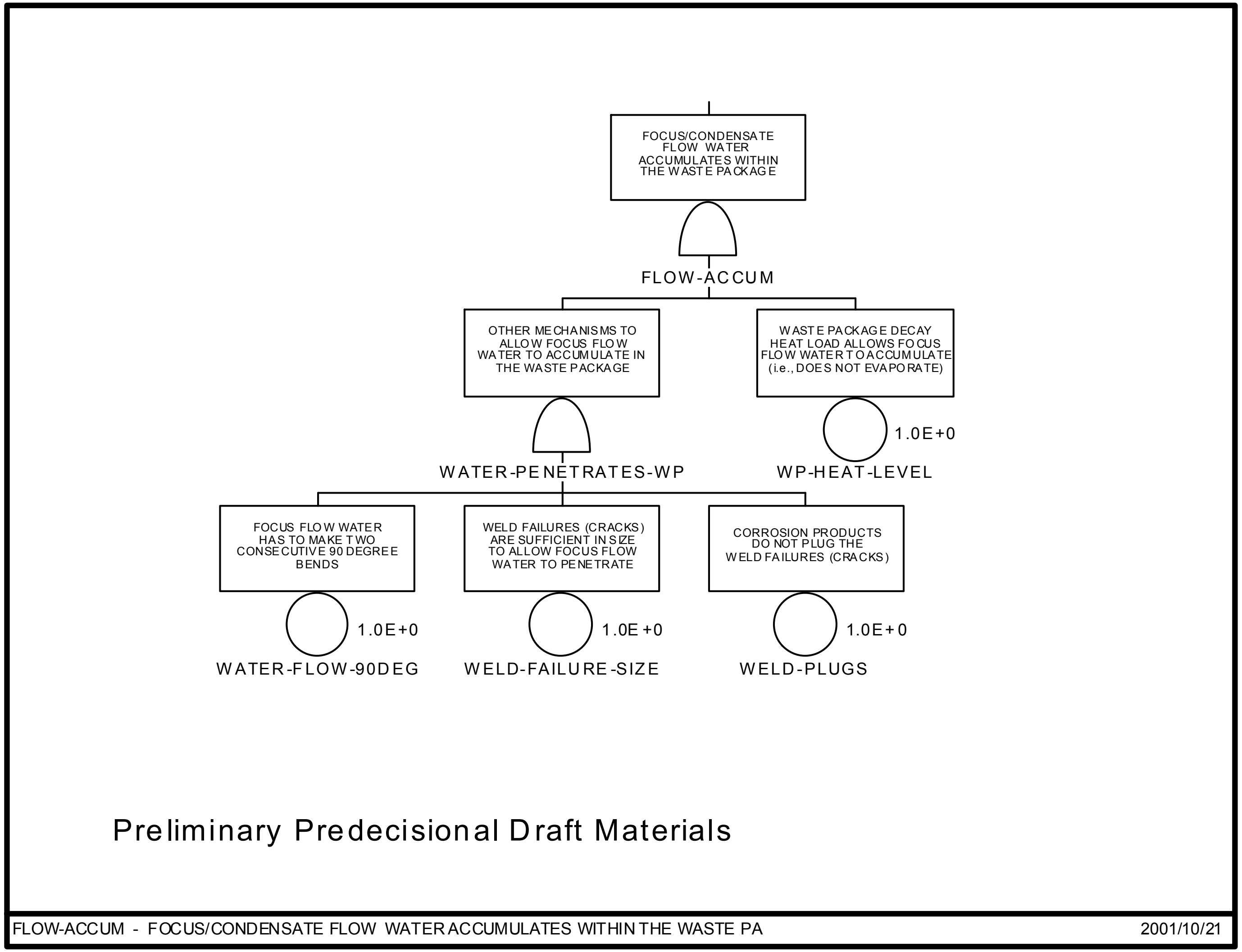

FOCUS/CONDENSATE FLOW WATER ACCUMULATES WITHIN THE WASTE PACKAGE
FLOW-ACCUM
OTHER MECHANISMS TO ALLOW FOCUS FLOW WATER TO ACCUMULATE IN THE WASTE PACKAGE
WASTE PACKAGE DECAY HEAT LOAD ALLOWS FOCUS FLOW WATER TO ACCUMULATE (i.e., DOES NOT EVAPORATE)
1.0E+0
WATER-PENETRATES-WP
WP-HEAT-LEVEL
FOCUS FLOW WATER HAS TO MAKE TWO CONSECUTIVE 90 DEGREE BENDS
WELD FAILURES (CRACKS) ARE SUFFICIENT IN SIZE TO ALLOW FOCUS FLOW WATER TO PENETRATE
CORROSION PRODUCTS DO NOT PLUG THE WELD FAILURES (CRACKS)
1.0E+0
1.0E+0
1.0E+0
WATER-FLOW-90DEG
WELD-FAILURE-SIZE
WELD-PLUGS
Preliminary Predecisional Draft Materials
FLOW-ACCUM - FOCUS/CONDENSATE FLOW WATER ACCUMULATES WITHIN THE WASTE PA
2001/10/21

FIG. 9. Probability of Failed Waste Package Criticality Fault Tree.

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M&O), September 3, 1999, OL&RC:SB-1714, with enclosure, "Interim Guidance Pending Issuance of New NRC Regulations for Yucca Mountain (Revision 01)." ACC: MOL.19990910.0079.

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BUC ASSESSMENT SCHEME: IMPLEMENTATION, FUEL AND LOADING VERIFICATION, REVIEW OF ANALYSIS ASSUMPTIONS AGAINST CHANGES IN THE FUEL PARAMETERS

(Session 2.5)

Burnup factor in the licensing of Armenian NPP SFDS

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Abstract. The Armenian NPP Spent Fuel Dry Storage (SFDS) facility of NUHOMS-56 project was constructed for storing the spent fuel assemblies remained in reactor pools after NPP shutdown in 1989. The criticality value was taken as for the fresh fuel assemblies and was required as 0.95. Basic criticality analysis assumptions should demonstrate the nuclear safety during all operations for normal, abnormal and accidental situations, and was based on fresh fuel elements. No credit was taken from burnup, which is conservative since only burned fuel will be stored. The licensing requirements were established by the ANRA with respect to the nuclear safety criticality limits set in IAEA rules and USNRC 10CFR codes. [1][6][7][8][9][11][14].

1. INTRODUCTION

Before 1989 the spent nuclear fuel was transported from the Armenian NPP by railway in Special Transport Cask (STC), for which the main criteria is the heat-decay and the fuel cladding. On the moment of the NPP shutdown in 1989 in the reactor pools there where 612 spent fuel assemblies, which where not transportable because of high heat-decay. The approximate storing time in reactor pool before transporting by STC is about 3-5 years. All the spent fuel assemblies where exposed through cladding control, such as on gas density or presence of fusion products (alpha-emitters) on surface.

The maintained Spent Fuel (SF) assemblies were different by Burnup Credit and by primary enrichment by U-235. The distribution of SF assemblies by primary enrichment and dept of burn was up to assemblies U-235 3.60 % w/o with working time in reactor core 3-4 months to assemblies U-235 1.60% w/o 3-4 years. The main requirements for SF, such as:

i. the quantity of heat-decay,
ii. the condition of fuel cladding,
iii. reactivity value,

during the stay in the reactor pools or in the Soviet design Special Transport Cask have some difference from NUHOMS project. [15][16][17].

That’s why the licensing requirements were established in accordance with USNRC 10CFR.

However, the calculation of the criticality value for 56 Spent Fuel assemblies in Dry Shield Cask (DSC) at accidental deviations (such as the flow of “pure water’’) in Primary Safety Analysis Report (PSAR) was reported *Keff* = 0.98, regarding the spent fuel as ‘’fresh’’, while the required safety criteria was established *Keff* = 0.95.

2. CRITICALITY CALCULATION APPROACHMANTS

The criticality analysis was carried out by using the standard computer codes APOLLO-MORET, mainly on MORET-III, based on code Monte-Carlo. [2][3][4]

The maximum reactivity resulting was analyzed by two different ways:

The French rules recommended by the CEA IPSN;
The American ones agreed by the USNRC.

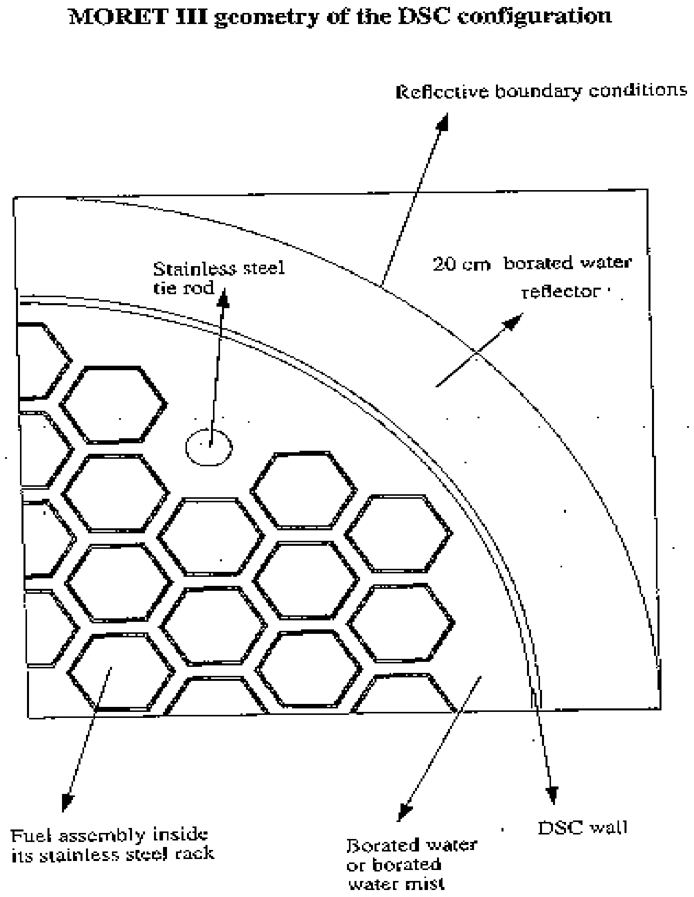
MORET III geometry of the DSC configuration
Reflective boundary conditions
Stainless steel tie rod
20 cm borated water reflector
Fuel assembly inside its stainless steel rack
Borated water or borated water mist
DSC wall

FIG. 1. MORET III geometry of the DSC configuration

2.1. The French approach

The French way is to choose all the parameters at the most reactive dimensions. It was done for all parameters except for nominal enrichment and the cells (pitch and internal dimension)(see the figure 1). Considering the worst dimensions is roughly equivalent to combine all the sensitivity results, as additive manufacturing bias, and the global bias is 2.518ΔK.

The maximum reactivity is given by the following formula:

$$K\max = Kcalc + \Delta Kcalc + \Delta Kbias + \Delta Kmetod \qquad (1)$$

$Kcalc$: Monte-Carlo code result

. $Kcalc$: statistical uncertainty on Monte-Carlo result ($3\sigma = 0.5\%\Delta K$)

$\Delta Kbias$: summation of the reactivity effects obtained between nominal and worse penalized dimension: 2.518ΔK.

$\Delta Kmetod$: 0

$$K_{MAX}^{IPSN} = 0.886 + 0.005 + 0.2518$$

$$K_{MAX}^{IPSN} = 0.9162$$

2.2. The American approach

The American way of determining the K_{MAX}^{USNRC} is to consider the entire nominal dimension to compute a *Keff* "nominal". All the manufacturing tolerances effects are statistically combined, based on the independence of the effects. The worst reactivity difference between calculations and a special lot is applied as a methodology bias.

The maximum reactivity is given by the following formula:

$$K_{MAX}^{USNRC} = Kcalc + \Delta Kbias + \Delta Kmetod + \Delta Kuncer \qquad (2)$$

$Kcalc$: Monte-Carlo code result

$\Delta Kbias$: no bias is applied

$\Delta Kmetod$: 1% ΔK

$\Delta Kuncert$:

$$\sqrt{\left[(\text{statistical uncertainty})^2 + \sum(\text{independant manufactoring tolerance effect})^2\right]}$$

$$K_{\max}^{USNRC} = 0.88565 + 0.01 + \sqrt{\left[(0.005)^2 + (0.01986)^2 + (0.00363)^2 + (0.00169)^2\right]}$$

$$K_{\max}^{USNRC} = 0.9165$$

Note that the level of confidence required to statistical results is usually limited to 95/95 (1.763□for a sufficient number of particles). So, the statistical uncertainty is reduced from 0.5%(□) to 0.3%(1.763□) this minor difference leads to a strictly equal $K\max$ between both French and American approaches. So, the global penalty to apply on the calculations result is 0.03, which covers all the uncertainties.

The operations taking place from the reactor fuel storage pool through the transportation to the final disposal of spent fuel assemblies in the SFDS were divided into several phases, which are identified as normal situations (11cases), abnormal situations (1case) and accidental situations (2 cases).

All the phases of the SFDS operations, were computed or fully documented when not performed. The performed analysis leads to the main following results.

*DSC in the borated water pool with 56 WWER U-235 3.60% w/o: *Keff* = 0786
*DSC during the drying process: *Keff* = 0.891
*Dry DSC inside the transfer cask: *Keff* = 0.437
*DSC inside the HSM: *Keff* = 0.415
*DSC in the borated water pool: *Keff* = 0.950

And at least, the accidental situation:

*DSC in "pure" water with 56 WWER U-235 3.60% w/o: *Keff* = 0.973

with a value of 0.166 % ΔK as a bias to calculation result.

This value is not in conformity with ANRA requirements, which are:

at the normal situation: *Keff* <0.95
at the accidental and abnormal situation: *Keff* <0.95

To fulfill the ANRA requirements based on the USNRC rules, and to ensure the sub-criticality during such accidental situation, fixed neutron absorbing materials have been placed in the canister. The optimized solution consists of 56 fresh assemblies inserted in 24 Borated Stainless Steel (BSS) racks (1w/o of boron) and 32 stainless steel (SS) racks according to a specific distribution as it shown on figure 2.

At this case the calculated result is: *Keff* = 0.942

with included uncertainties 3.0% ΔK.

NUHOMS® 56 V Dry Shielded Canister Cross-Section

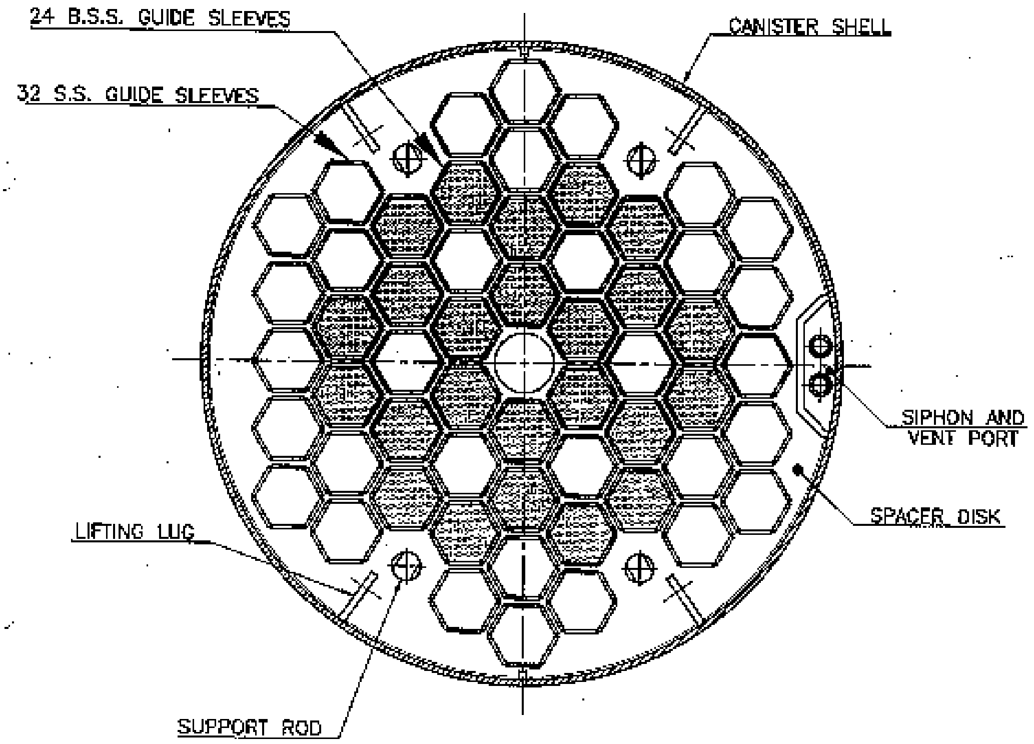
24 B.S.S. GUIDE SLEEVES
32 S.S. GUIDE SLEEVES
CANISTER SHELL
SIPHON AND VENT PORT
SPACER DISK
LIFTING LUG
SUPPORT ROD

FIG. 2.

3. CONCLUSIONS

Regardless of numerous warnings and objections from the part of Armenian Nuclear Regulatory Authority (ANRA), the Company (Framatome) produced and brought to Armenian NPP Dry Shielded Canisters (DSC) not corresponding in construction and technology to safety criteria, suggesting to regard the factor of burnup while calculating Spent Fuel assemblies reactivity.

However, Framatome never presented any confirming calculations or arguments on the above-mentioned problem and hade to change some constructive peculiarities of DSC in accordance with Final Safety Analysis Report (FSAR), which naturally increased the starting price of the project.

Especially, its time consuming during the licensing the Armenian NPP Spent Fuel Dry Storage Facility.

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Criticality calculations of WWER spent fuel casks implementing burnup credit

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Abstract. A methodology for criticality safety analysis of spent fuel casks with burnup credit implementation is presented. It includes the world well-known and applied program systems: NESSEL-NUKO for depletion and SCALE-4.4 for criticality calculations. The possibilities of this methodology to analyze storage and transportation casks with different type of spent fuel are demonstrated on the base of various tests. The depletion calculations have been carried out for the WWER-440 and WWER-1000 fuel assemblies. The criticality calculation models have been developed on the basis of real fuel casks for WWER-440 and WWER-1000 spent fuel assemblies. The results obtained show that the criticality safety criterion K_{eff} less than 0.95 are satisfied for both: fresh and spent fuel. Besides the implementation of burnup credit allow to account for the reduced reactivity of spent fuel and to evaluate the conservatism of the fresh fuel assumption.

1. INTRODUCTION

In this paper a methodology for criticality safety analysis of spent fuel casks with burnup credit implementation is presented. This methodology gives a possibility for criticality evaluation of various storage and transportation facilities with different type of spent fuel. It includes the world well-known and applied program systems: NESSEL-NUKO for depletion and SCALE-4.4 for criticality calculations. The possibilities of this methodology to analyze criticality safety of storage and transportation casks with different type of spent fuel are demonstrated on the base of several test models. The methodology allows also accumulation of detailed information for the spent fuel characteristics, needed for the computerized system for management of nuclear materials in Bulgaria, according to the international requirements and regulations.

2. COMPUTER CODES AND NUCLEAR DATA

The application of burnup credit requires detailed knowledge of the irradiated fuel isotopic concentrations. The isotopic inventory and post decay power of the irradiated fuel has to be calculated by a depletion code, depending on the initial state and irradiation history of the fuel. The results of the depletion analysis are a necessary input to the criticality analysis of the system for which the burnup credit is taken.

The methodology is based on the two world well known and used code systems for depletion and criticality calculations, respectively:

NESSEL-NUKO - for depletion calculations;
SCALE-4.4 - for criticality calculations.

2.1. NESSEL-NUKO code system

NESSEL-NUKO is a complex code system intended especially to WWER depletion analysis. The NESSEL-4 code [1] calculates the local neutron physics characteristics and depletion of different nuclear fuel assemblies in the core. The NUKO code [2] calculates the isotopic inventory and post decay power of spent fuel assemblies – the concentrations of actinides and fission products with medium decay period important for practice.

2.2. SCALE- 4.4 code system

The SCALE-4.4 modular code system [3] is verified and world-widely used for criticality safety analyses of PWR spent fuel storage facilities. The system has been recently in process of international testing for WWER applications. It was verified also at the INRNE, BAS for analyses of WWER spent fuel storage and transportation facilities [4,5]. The analytical sequence CSAS6 has been applied for the criticality calculations. It includes the modules BONAMI, NITAWL-II and XSDRNPM for neutron data preparation, as well as the 3D multigroup Monte Carlo criticality code KENO-VI. The 44-group neutron data library 44GROUPNDFB5 based on evaluated data file ENDF/B-V has been used [3].

3. RESULTS

3.1. Depletion calculations

In order to prepare nuclear inventory data for implementation of burnup credit in the criticality calculations of spent fuel storage casks, the nuclear densities of the basic fuel isotopes from the Uranium - Plutonium chain, the actinides and the fission products under consideration have been determined by the NESSEL-NUKO code system for three different fuel assembly types, given in Table I.

Table I. Fuel assemblies, calculated by the NESSEL-NUKO code system

Type of reactor under consideration	Assembly type (enrichment)	Burnup reached [MWd/kg]	Cooling time [d] after discharge
WWER-440	3.6%	30.0	2575.0
WWER-1000	3.3%	27.53	3220.0

These fuel assemblies are shown in Figs.1 and 2 respectively. The real geometry data and material content of each assembly have been modeled corresponding to the homogenization procedure of the NESSEL code, generating binary data files with the isotopes number densities, the fission and capture microscopic cross sections and the 34-group neutron spectrum in dependence on fuel burnup. The real operational power history and the outages for reloading of the given assembly has been taken as follows: for the WWER-440 assembly – from the benchmark task [6] and for the WWER-1000 assembly - from the operational benchmark for the Unit 6, Kozloduy NPP [7].

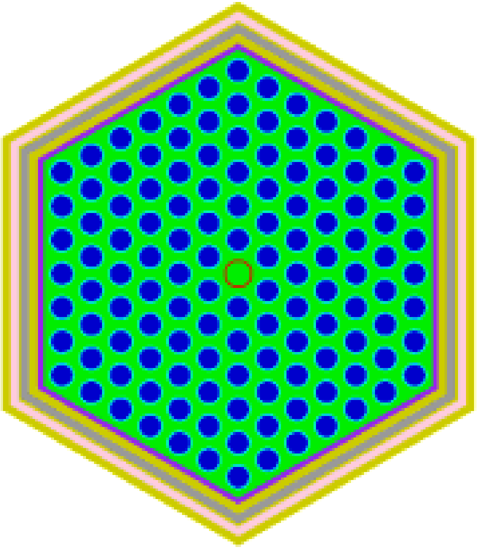

FIG. 1. WWER-440 fuel assembly. Visualisation by the KENO-VI.

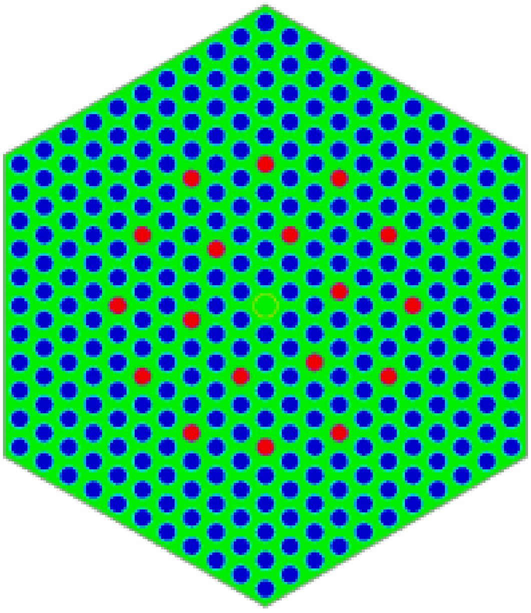

FIG. 2. WWER-1000 fuel assembly. Visualisation by the KENO-VI.

3.2. Criticality calculations

3.2.1. WWER spent fuel cask models

The first model (Fig.3) is developed on the basis of a real WWER-440 spent fuel cask for storing 84 fuel assemblies, designed by the SKODA Nuclear Machinery [8]. The geometry and material data needed for modeling the fuel assemblies are taken from Ref. [9]. The internal rack for loading the fuel is manufactured from hexagonal aluminum tubes shielded by stainless steel (containing 1.0% of natural boron), each of 0.3 cm thickness. Correspondingly, the stainless steel plate is 7.5 cm and the aluminum plate is 7.9 cm away from the center. The fuel assemblies are located in a triangular lattice with pitch 17 cm, in the stainless steel cask with 90 cm inner radius and 28 cm thick wall.

The second model (Fig.4) is based on a real cask, designed by the Izorskie zavody (Russia) [10]. The inner radius of the cask is 77 cm; the cask wall is 20 cm thick. It contains 54 WWER-440 fuel assemblies, located in a triangular lattice with pitch 16.5 cm, in compact rack made of stainless steel containing 1.1% of natural boron, with 0.3 cm thickness and at distance 7.5 cm from the center of the assembly.

The third model (Fig.5) is developed on the basis of a real WWER-1000 spent fuel cask for storing 18 fuel assemblies, designed by the Izorskie zavody (Russia) [10]. The calculations have been carried out for both fresh and spent fuel assemblies in casks filled with distilled water. Actinides and fission products credit has been also taken into account. The number of neutrons in every generation is 600 and the number of generations is 300.

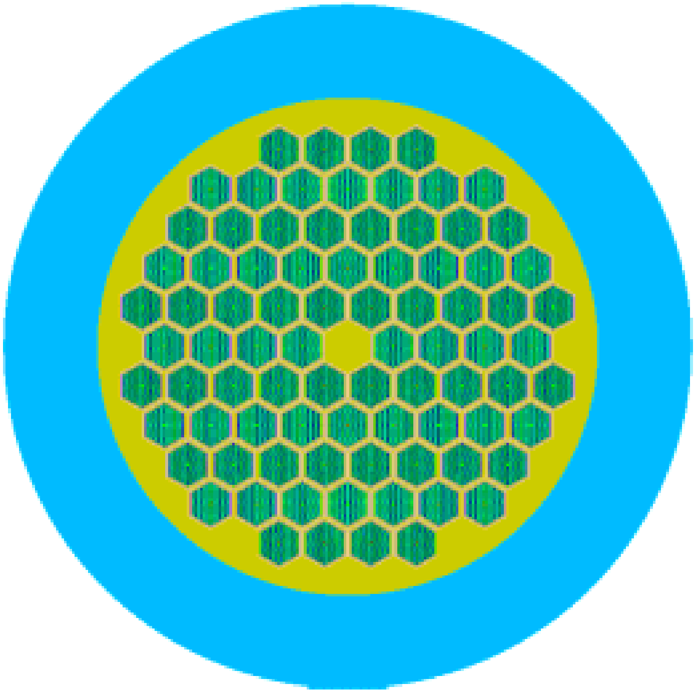

FIG. 3. SKODA spent fuel cask model (84 WWER-440 fuel assemblies). Visualisation by the KENO-VI.

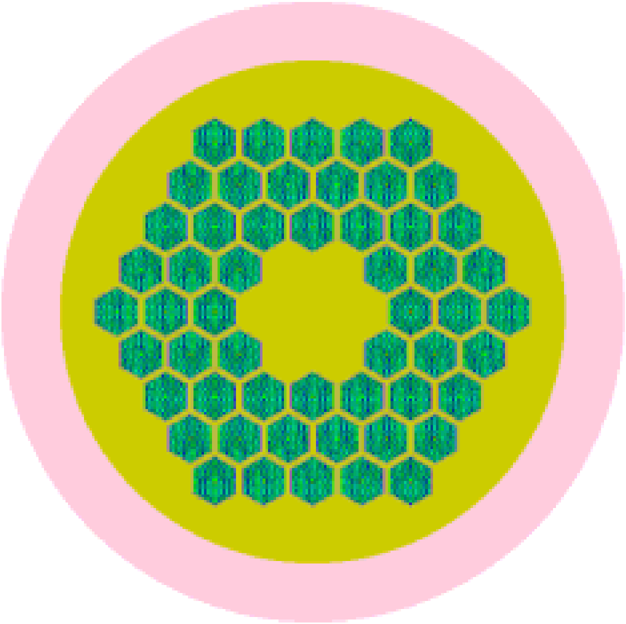

FIG. 4. Izorskie Zavody cask model (54 WWER-440 fuel assemblies). Visualisation by the KENO-VI.

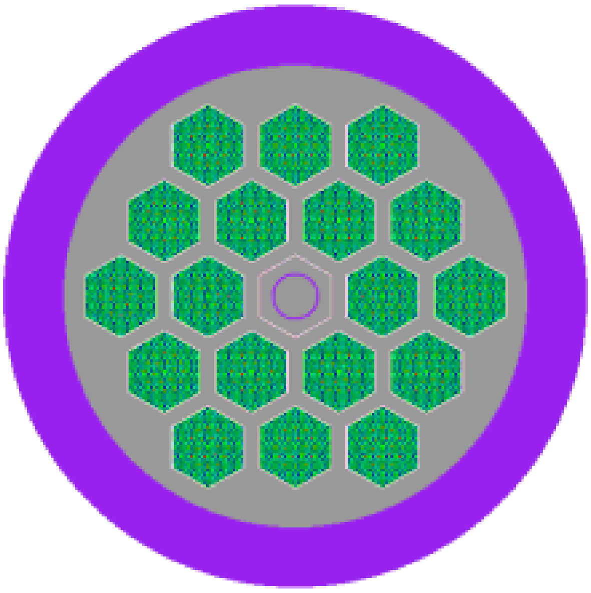

FIG.5. Izorskie Zavody cask model (18 WWER-1000 fuel assemblies)(x-y plane). Visualisation by the KENO-VI.

3.2.2. Results

The calculated values of multiplication factors K_{eff} and their statistical errors 1σ for both fresh and spent fuel at different cooling times for various spent fuel assemblies and cask models are presented in Table II. They show that the reduction of the K_{eff} value due to the burnup credit implementation especially for higher cooling times is of about 23 %.

The calculated values of average K_{eff} for fresh and spent fuel of several types at different cooling times are presented graphically in Figs.6÷8. The K_{eff} values for both fresh and spent fuel are less then the criticality safety limit $K_{eff} = 0.95$. The conservatism of the “fresh fuel assumption” compared to the real criticality state of the spent fuel cask can be clearly seen from these figures. The reduction of K_{eff} value due to the burnup credit implementation varies from 16% to 23% for the presented models with WWER fuel assemblies.

Table II. Effective multiplication factor K_{eff} for various spent fuel cask models and for different cooling times, calculated by the SCALE- 4.4 code system

Configuration	Fresh fuel	Spent Fuel			
		At discharge	Tcool 1 year	Tcool 3 years	Tcool n*) years
Model of SKODA WWER-440 cask for 84 fuel assemblies	0.8316± 0.0018	0.6850± 0.0015	0.6829± 0.0014	0.6779± 0.0015	0.6768± 0.0015
Model of Izorskie zavody WWER-440 cask for 54 fuel assemblies	0.7858± 0.0021	0.6454± 0.0015	0.6447± 0.0015	0.6401± 0.0015	0.6340± 0.0015
Model of Izorskie zavody WWER-1000 cask for 18 fuel assemblies	0.8205± 0.0019	0.6868± 0.0015	0.6750± 0.0015	0.6717± 0.0016	0.6684± 0.0018

*) n = 6 for the first and second configurations and n = 5 for the third configurations.

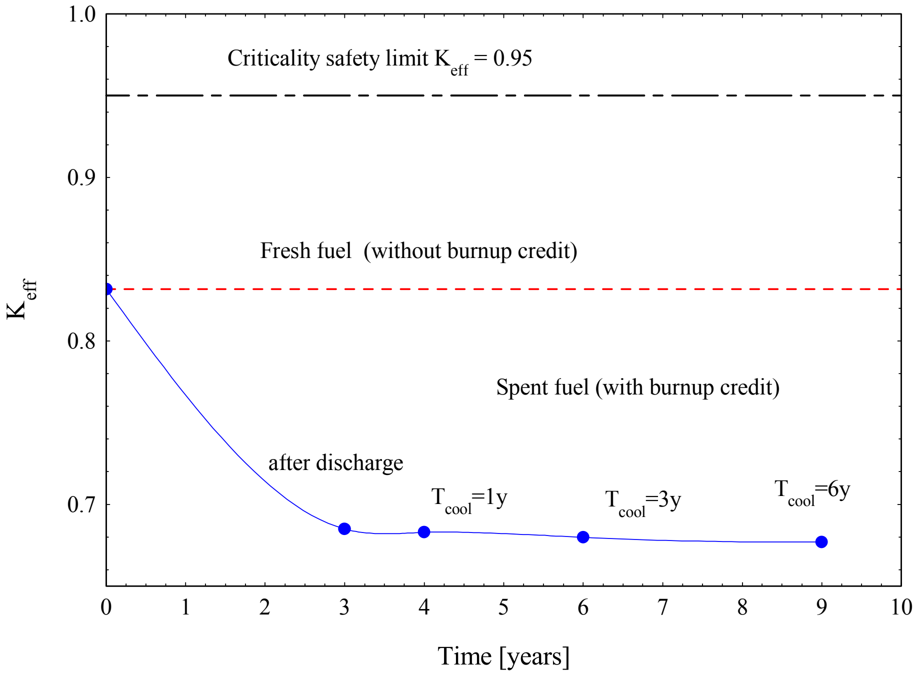

FIG.6. Time dependent criticality of the CASTOR WWER-440 fuel cask.

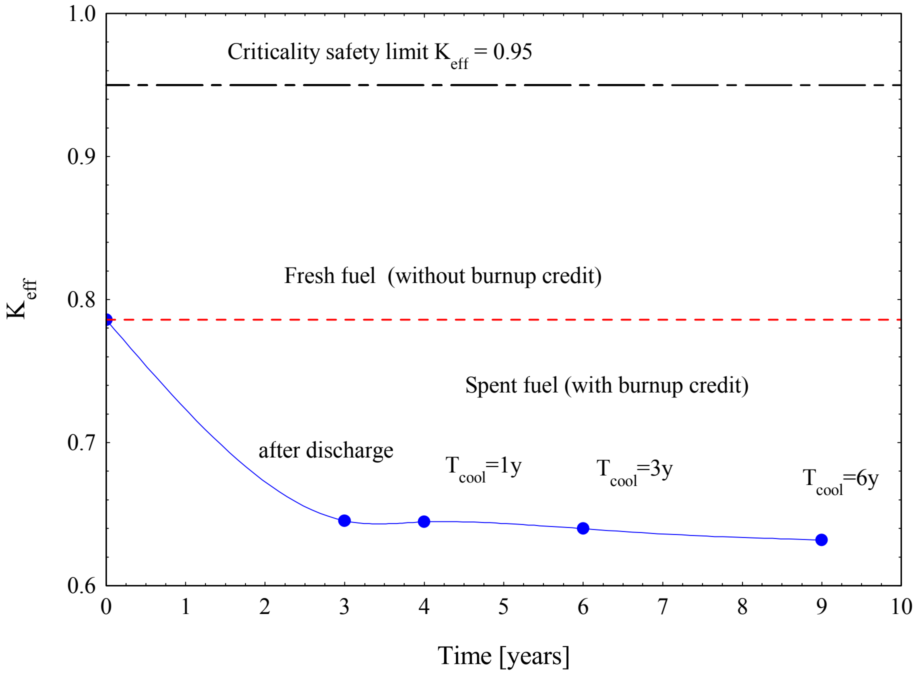

FIG.7. Time dependent criticality of the Izorskie zavody WWER-440 fuel cask.

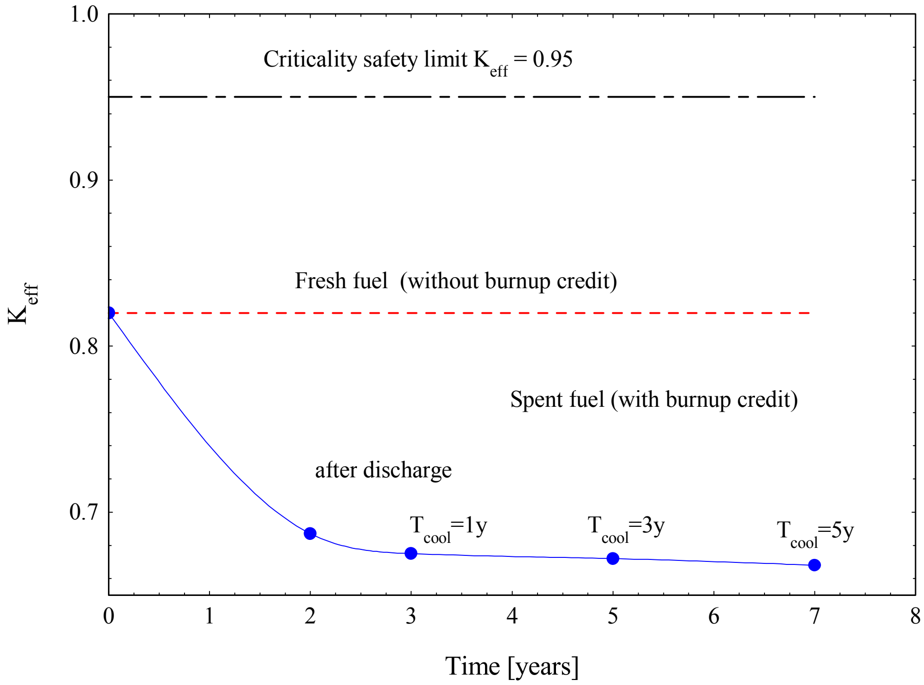

FIG.8. Time dependent criticality of the Izorskie zavody WWER-1000 fuel cask.

4. CONCLUSIONS

On the basis of the obtained results the following conclusions could be drawn:

The ability of the presented methodology for burnup credit criticality safety analysis for spent fuel cask of several designs and different fuel types is demonstrated.

The results for K_{eff} obtained by the SCALE-4.4 code system (CSAS6 analytical sequence) for the WWER-440 and WWER-1000 cask models confirm that the modeled real casks with Russian and Czech design satisfies the criticality safety requirements $K_{eff} < 0.95$ for both fresh and spent fuel. The implementation of burnup credit accounts for the reduced reactivity of spent fuel and allows decreasing the conservatism of criticality evaluation in comparison with the fresh fuel assumption.

The NESSEL-NUKO and SCALE-4.4 code systems can be used for the WWER spent fuel casks safety analyses (criticality and dose), including thermal analysis (by SCALE) and implementing burnup credit. The SCALE code system should continue to be verified for WWER on the basis of both WWER benchmark problems and WWER experimental data. Some verification has already been completed by a contract with the Bulgarian Regulatory Body.

The results obtained for the basic characteristics of the spent fuel assemblies of WWER power reactors could be applied for the computerized system for management of nuclear materials. As a country, ratified the Treaty for nonproliferation of nuclear weapons and Treaty for safeguards INFCIRC/178 Bulgaria is obliged to build up such a system and put it into operation.

ACKNOWLEDGEMENT

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Research and application of burnup credit technology in China

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Abstract. The status of the spent fuel storage of Daya Bay Nuclear Power Station is been described. Preliminary calculations adopted burnup credit on Daya Bay Nuclear Power Station spent fuel storage pool have been performed, it is shown that use of burnup credit in the criticality analysis and design of spent fuel storage pool could result in considerable benefits. The k_{eff} of the storage pool could be decreased about 20% if only the main actinides are taken into account, and decreased about 30% if the actinides and the major fission product poisons are taken into account. That means the capacity of the Daya Bay Nuclear Power Station spent fuel storage pool could be increased about one times. Chinese government and the owner of power station plant are very interested in burnup credit technology, and a 5-year project is launched, it mainly includes: 1) Research and development of criteria, calculation model, computer codes. 2) Analysis of burnup uncertainty. 3) Criticality experiment. 4) Measurement of fuel burnup.

INTRODUCTION

Up to now, two power plants, Daya Bay and Qinshan, have come into commercial operation in China, and the other eight units will go into operation during 2003~2005.

Table I. Nuclear Power Plants in China

Plant	Unit No.	Power, MWe	Remark
Daya Bay	1	900	Constructed
	2	900	Constructed
Qinshan	1	300	Constructed
	2	600	Constructing
	3	600	Constructing
	4	700	Constructing
	5	700	Constructing
Lingao	1	1000	Constructing
	2	1000	Constructing
Tianwan	1	1000	Constructing
	2	1000	Constructing
Total		8700	

Spent fuel storage, transport and reprocessing have become an important problem to which regard has been paid in Chinese nuclear industry. Because no large-scale reprocessing plant is planned to build, long-term storage is the main approach to treat spent fuel in China. With spent fuel accumulating, nuclear power plants and relative sections of design and research began to consider the means to increase the capacity of storage pool. Apparently, adopting burnup credit technology is an effective resolvent. Theoretical research has been done preliminarily in this area since 1996 in China Institute of Atomic Energy (CIAE), and feasibility of dense storage for Daya Bay Nuclear Power Plant spent fuel storage pool was investigated from 1999. CIAE has established basis of theoretical and experimental research in criticality safety, burnup measurement method. and so on. Although the preliminary investigations are performed for several years, technology study in this area is just at its beginning now. In order to implement burnup credit for spent fuel, a programme has been developed in CIAE in 2001.

Requirements of burnup credit in China

Daya Bay Nuclear Power Station in Guangdong province has two 900Mw units, put into commercial operation in February and March 1994 respectively. The reactor core employs 1/3 refueling, with refueling batch of 52 spent fuel assemblies each refueling cycle of one year. The reactor has undergone 6 refueling up to now. The spent fuel storage pool has storage capacity of 695 fuel assemblies, and the design capacity is for ten years. At present the improvement of reactor core fuel management is underway for an 18-months-refueling plan from the ninth or the tenth cycle. In the advanced refueling method the fuel enrichment will further increase, up to the projected 4.5%.

As the in-site spent fuel pool will soon reach the design capacity; the power plant is prepared for transporting out the spent fuel in the next year. The transportation will be performed cooperating with an American company. Because railway transportation is further more expensive than highway, the last alternative is determined. One transport cask can ship 26 fuel assemblies. All the spent fuel assemblies of a year can be put into 2 casks.

Qinshan Nuclear Power Plant has a 300MW unit. The reactor has undergone 6 refueling cycles since the plant connection the electricity network in December 1991. The plant has two spent fuel storage pools with total capacity of 756 assembly boxes. The design capacity is 15 years (on the basis of 40 assemblies each refueling batch).

Burnup does not credit for either in-site wet storage or transport dry storage in Daya Bay Nuclear Power Plant now. As the increase of the fuel initial enrichment in the advanced refueling method, BUC application has become a need in order to keep the facilities in operation as well as due to economics. In spent fuel storage and transport area, Qinshan Nuclear Power Plant faces the similar problems as Daya Bay Nuclear Power Plant.

Spent Fuel Pool operator and its designer are urgent to employ BUC technology due to economics. Specialists demand some requests on three aspects: history record and calculation, measurement, management. Because there is not regulatory guidance in this area yet, the implementation of BUC is very difficult. Otherwise, the spent fuel storage pool of Lingao Nuclear Power Station that is under construction will be designed using burnup credit technology to increase its capacity. But it have not got the license yet.

Preliminary FEASIBILITY STUDY FOR BURNUP CREDIT

Whereas the situation of spent fuel in China, preliminary feasibility study for BUC in Daya Bay spent fuel storage has been performed from 1999. The purpose of our analysis was to evaluate the criticality safety and heat residual by taking burnup credit technique for improved schemes of Daya Bay Nuclear Power Station spent-fuel storage pool.

Analysis models

Principles of all the analyses are based on the facts of Daya Bay Nuclear Power Station. So any approximation about storage pool geometric description wasn’t made, and all the fuel assemblies with various discharge burnup and different enrichment that could be loaded in the storage pool were considered in criticality calculation. Full core discharge adopted by Daya Bay Nuclear Power Station was also considered in our analysis.

The capacity of current spent-fuel storage pool is 695 fuel assemblies. Our analyses of spent-fuel storage pool dealt with the three improved schemes (the capacity is 1287, 1455, 1520 fuel assemblies respectively) of Daya Bay Nuclear Power Station spent fuel storage pool. The construction of three new storage pool is similar each other. In order to allow high-density rankings, it is necessary to consider storage in two regions. Cadmium was adopted as the absorbers wrapped with 304 stainless steel in the basically racks for the fresh fuel, the pitch is 280 mm. The second type of racks is the borated steel as the absorbers for the spent fuel assemblies; the pitch is 232 mm. Faulted fuel assembly storage racks was the same as that adopted cadmium.

Our analyses were based on the actual discharge fuel assemblies of Daya Bay Nuclear Power Station. Unit 1 and unit 2 employ UO_2 fuel assemblies with different enrichment. During the first 8 cycle for both unit, unit 1 employed the fuel assemblies with 1.8%, 2.4%, 3.1%, 3.2%, 3.7% enrichment, and unit 2 employed the fuel assemblies with 1.8%, 2.4%, 3.1%, 3.2% enrichment. After that, the 4.45% fuel assembly is the only choice because the 18 months fuel management will be adopted. The discharge burnup of the 4.45% fuel assembly is about 45000MWD/TU expectedly. The fuel assembly was made in 17×17 rods array.

Although nearly 1700 nuclides were included in the output result, only the most important 12 nongaseous fission products and 13 actinides were considered in our criticality analysis according to the reference [1] and our calculation results. The 12 fission products were ^{99}Tc, ^{103}Rh, ^{131}Xe, ^{133}Cs, ^{143}Nd, ^{145}Nd, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{152}Sm, ^{153}Eu, ^{155}Gd. The 13 actinides were ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, $^{\text{and 244Cm}}$.

Table II. The quantity and discharge burnup of fuel assemblies with different enrichment

Unit 1					
Enrichment, %	1.8%	2.4%	3.1%	3.2%	3.7%
No.	53	52	60	256	100
Burnup, MWD/TU	23064	23587	29802	29892	42000
Unit 2					
Enrichment, %	1.8%	2.4%	3.1%	3.2%	
No.	53	52	56	264	
Burnup, MWD/TU	23064	23587	29802	29892	

In analysis models, the following assumptions were taken into account:

- The last fuel assembly being unloaded 7 days after reactor shutdown.
- Pure water in the spent fuel pit (more than 20000 pcm negative reactivity due to the borated water is not taken into account).
- Criticality safety criteria to be respected: k_{eff}<0.95
- After the 8th cycle for both Unit 1 and Unit 2, every 1.5 year discharge of 52 fuel assemblies with 45000MWD/TU discharge burnup.
- 72 new fuel assemblies with 4.45% enrichment being loaded
- Except the position occupied by spent-fuel assemblies from the first 8 cycles, others were fully filled by spent-fuel assemblies with 4.45% enrichment.

Calculation Results

After the termination of each burnup history, ORIGEN-2 was used to calculate the inventories of radioactive actinides, fission products and decay heat power within a total of 25 year. These fission products provide about 70% contributions to the total fractional neutron absorption rates of all the fission products, and this rates were increasing with the extend of the cooling time.

Based on the calculation results of heat residual power, the actual quantities and the time loaded in the storage pool for every type assembly, we obtained the maximum heat residual power of the pool. If the full core discharge conditions were not taken into account, heat residual power is 3.5 MW. If the full core discharge conditions were taken into account, heat residual power is 8.8 MW.

Three improved schemes of unit 1 and unit 2 were analyzed for both fresh fuel and irradiated fuel. The effect of actinides only and the combined effect of actinides and fission products were considered separately.

In our analysis, the following principle for assembly configuration was abided by. New fuel assemblies with 4.45% enrichment were arranged in the original rack region of the pool. Different types of spent-fuel assemblies were arranged in the new rack region from the periphery to the core as the fissionable nuclides concentration within spent-fuel assemblies, while assemblies with a same enrichment were arranged in the same rack as far as possible. So that spent-fuel management and actual operation of nuclear power station are easier, while the k_{eff} value of the pool is lower. The results given by M-C codes were summarized in Table III. In addition, we analyzed the k_{eff} sensitivity at various burnup of 4.45% enrichment assemblies, because of uncertainty of discharge burnup in the future. The results are given in Figure 1.

Table III. KEFF of criticality calculation for three improved schemes

Unit	BUC Level	Scheme 1	Scheme 2	Scheme 3
	Fresh Fuel	1.2095±0.0011*	1.02091±0.0010	1.2094±0.0012
1	Actinide Only	0.9285±0.0014	0.9315±0.0013	0.9274±0.0014
	Actinide+Fission Product	0.8694±0.0016	0.8680±0.0015	0.8731±0.0017
2	Actinide Only	0.9268±0.0014	0.9327±0.0013	0.9320±0.0016
	Actinide+Fission Product	0.8715±0.0016	0.8680±0.0015	0.8702±0.0018

* Results were based on 3000 neutrons/generation, 100 generations. 10 generations were skipped.

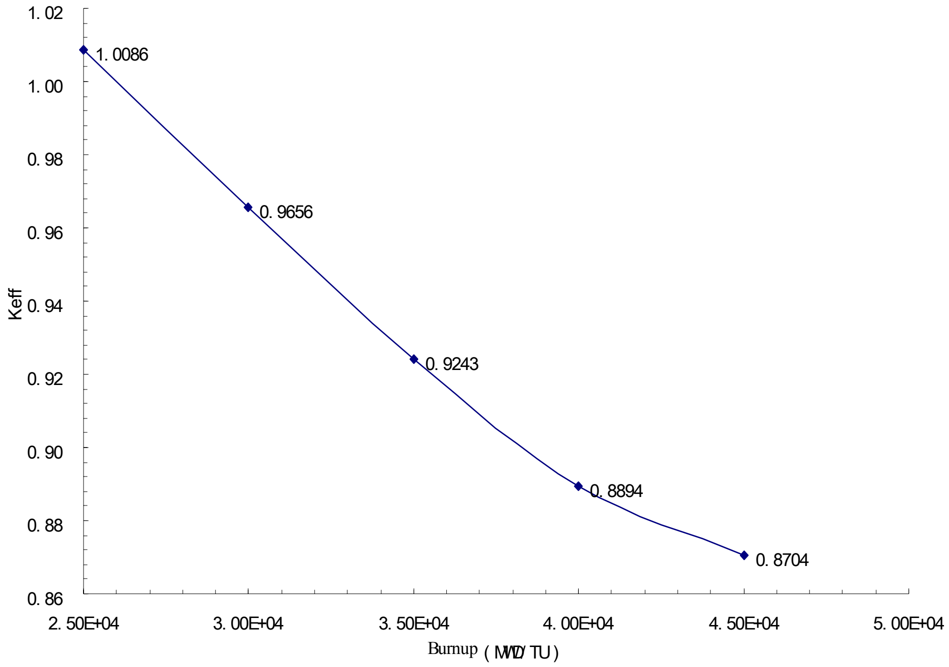

FIG. 1. The k_{eff} at various discharge burnup of assemblies with 4.45%enrichment

The analysis results for Daya Bay Nuclear Power Station three improved schemes of spent-fuel storage pool demonstrated that use of burnup credit in the criticality analysis and design of spent fuel storage pool could result in considerable benefits. The k_{eff} of the storage pool could be decreased about 20% if we only consider the effect of main actinide, and decreased about 30% if we consider the combinative effect of actinide and the major fission product poisons. The capacity of the Daya Bay Nuclear Power Station spent fuel storage pool could be increased about one times, such as the using time of storage pool for loading spent-fuel assemblies could be extended to 25 years. However it is worth to pay more attention on the discharge burnup of 4.45% enrichment assemblies. From Figure 1, the k_{eff} will be great than the criticality criteria 0.95 if the discharge burnup is less than about 32000MWD/TU.

Research programme on BUC for future

Chinese government and the owner of power station plant are very interested in burnup credit technology. To push into the implementation of burnup credit, a 5-year official project is launched in CIAE, it mainly includes:

1. To keep track of international development and practice of burnup credit technology, including advancement, method, standard and management. Since work in these aspects is preliminary and at theoretical study stage, it is necessary to strengthen international exchange and systematically learn the overall circumstance, especially those problems to be solved in engineering practice.
2. To push into the legislation of the regulation and criteria relative to burnup credit. The lack of regulations retards seriously the implementation of BUC. Cooperating with the related section, we will do our best to push into the legislation of the regulation and criteria relative to BUC.
3. To establish comparatively complete theoretical analysis software system for burnup credit technique. Research scientists in the Nuclear Reactor Physics Laboratory in CIAE have engaged in reactor physics, nuclear criticality safety theoretical and experimental research for many years and built essential foundation for the proposed project.

Although we have several software packages for conventional reactor physics and criticality safety calculations, as for the burnup credit technique, the calculation precision is probably expected to be increased and computation software to be improved.

4. To perform some nuclear criticality experiment to simulate the spent fuel storage to master experimental technique and obtain fundamental criticality data. There are several zero power reactors and a uranium solution criticality experiment facility in the Reactor Physics Laboratory, which can be utilized for the related experimental researches.
5. To investigate burnup measurement methods and develop spent fuel assembly burnup measuring system. Besides complete theoretical basis and precise calculation codes, measurement method and devices are also necessary for the enforcement of burnup credit technique. Preliminary research work has been done in recent years in CIAE. It is expected to master the measurement method through the project.

Conclusion

With the rapid development of nuclear electricity in China, spent fuel storage, transport and reprocessing have become an important problem to which regard has been paid in Chinese nuclear industry. Theoretical research has been done preliminarily in this area since 1996 in China Institute of Atomic Energy (CIAE), and feasibility of dense storage for Daya Bay Nuclear Power Plant spent fuel storage pool was investigated from 1999. CIAE has established basis of theoretical and experimental research in criticality safety, burnup measurement method., and so on. But it is just at its beginning for the implementation of burnup credit. And now, Chinese government and specialists are interested and active in burnup credit. The regulations relative to BUC should be legislated as soon as possible.

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Burnup credit implementation in WWER-440 dual purpose cask

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Abstract. Based on worldwide research community consensus related to the framework of the burnup credit (BUC) implementation methodology, a transfer of the latest state-of-the-art knowledge to the WWER-440 application environment initiated and encouraged national research and development on this topic in the Czech Republic. The project entitled the 'Development of Technology Related to Burnup Credit Implementation in Storage and Transport Spent Fuel Casks' was proposed; this has been in development since late 1999 and comes to completion this year. Two collaborators, the Nuclear Research Institute (NRI) at Rez and SKODA JS company in Plzen, the Czech Republic, were developing a WWER spent fuel dual-purpose cask design based on the burnup credit implementation. The SKODA JS company, which is the only WWER spent fuel cask and component manufacturer in the Czech Republic, has only produced dual-purpose casks of the classical design so far. In this newly proposed design, the enormously conservative reactivity margin, which is connected with the spent fuel burnup, was converted into a cask material and capacity improvement due to the BUC implementation, keeping approximately the same price of the cask at the same time. Original design features enabled the designers to overcome difficulties connected with the heat transfer and removal solution as well as to meet the standards put on the transport casks.

The paper is available via ftp://nri.cz/pub/OECD/BUC

INTRODUCTION

In 1999 the project on development of technology related to burnup credit implementation in dual purpose WWER-440 spent fuel casks was proposed by NRI at Rez and funded by the Czech Ministry of Industry and Trade [1]. Two collaborators, the analysts and developers of NRI and SKODA JS company, have been working on the project for three years. The project itself was the first attempt to form and examine a basis for the BUC implementations in WWER-440 spent fuel management systems in the Czech Republic.

The project was divided into six stages as follows:

1. Development of BUC Methodology Implementation
2. Methodology Validation
3. Spent Fuel Record Database
4. Cask Loading Procedure
5. Cask Design Study
6. License Documentation Procedure

One of the goals of the project was recognition of what pieces of knowledge related to the BUC methodology could be transferred from the PWR application area into the WWER environment and what part of knowledge must be developed. The further goal was to start with validation effort aimed to the depletion and criticality codes and libraries supposed to be used for the BUC calculations in the WWER application area. WWER-440 spent fuel record database was proposed and compiled to support parameter modeling and sensitivity analyses because of finding out the bounding values of the key WWER-440 reactor parameters which enter depletion calculations. The project dealt mainly with the WWER-440 spent fuel (enrichment of 3.6 wt.% ^{235}U, non-profiled, without any burnable absorbers) having been discharged from the four WWER-440 units of the Czech Dukovany NPP since 80's. The design study and cask-loading curve developing, which were focused on concrete depletion and criticality calculations as well as design developing from the structural point of view, could not avoid solving technical conflicts between the heat transfer/removal requirements and the new design parameters. The last stage of the project reviewed the regulatory issues and outlined items, which should be involved in the cask documentation and be checked up in the course of licensing.

METHODOLOGY DEVELOPMENT

BUC implementation methodology development related to the WWER-440 application area was based on the approach using actinides plus major fission products [2] to describe the spent fuel for the criticality analyses. The sequence of the individual steps of the methodology agreed with the earlier actinides-only approach as described in [3] for PWR spent fuel packages. Results of PWR studies (e.g. [4], [5]) were used for a comparison.

Big effort was devoted to validation work (as mentioned below) related to the depletion and criticality calculational methodologies, which is essential for establishing the safety margin of the BUC implementation.

Basic sensitivity analyses related to the BUC implementation in the WWER – 440 application area were performed [6], [7] and presented at [8] as [14]. They provided the trends in k_∞ due to the variations of the key reactor operating parameters, such as boron concentration in moderator, fuel temperature, coolant temperature, specific power and length of downtimes between cycles as well as specific power and history effects during the burning of the fuel. Results (some of them can be seen in Figs. 1 and 2, below) enabled to find out bounding values of parameters subsequently used for depletion analyses. All the calculations were performed by SCALE4.4a (sas2, 44-group library) and MCNP 4A with DLC-189 library; no corrections of the isotopic compositions were made (isotopics were used as computed by sas2).

Another part of the BUC methodology, a perturbation method of searching for axial bounding fuel burnup profiles due to the end effect assessment, was developed ([21], [22]). Using this method as well as a spent fuel parameterized isotopic composition library ([21], [23]) the bounding values of the fuel depletion parameters were selected and used for cask subcriticality calculations and finding out the bounding values of the end effect for the given designs of spent fuel.

VALIDATION EFFORT

Critical experiments that could be applicable to validate the away-from-the-reactor criticality codes and libraries in the WWER-440 environment were selected. These experiments were performed in the past at ZR-6 and LR-0 research facilities in KFKI Budapest, Hungary, and LR-0, Rez, Czech Republic, respectively. Huge validation work in the WWER-440 environment, which has recently made in VTT Finland (e.g. see [9]) and KFKI Budapest [10] was taken as a basis and extended. Beside the recalculation of the selected critical experiments, a series of LR-0 critical experiments were prepared, carried out and recalculated ([11], [12], [13]) in NRI by MCNP. The experimental cores patterns of the WWER fuel assemblies were not too sophisticated but rather typical and focused on good accuracy of the performance. As a result of this effort, not only the idea on quality and number of applicable critical experiments were formed but also MCNP4A/B+DLC-189 methodology bias and uncertainty were derived for some specific subsets of the experiments. The benchmarking analyses established both a bias (defined as the mean difference between experiment and calculation) and an uncertainty of the mean with a one-sided tolerance factor for 95-percent probability at the 95-percent confidence level. Unfortunately, it turned out that the spread of the differences between the experiments and calculations is rather big, which led to methodology bias + uncertainty of a relatively big value (such as 0.00590 + 0.00845 for one of the best specific subsets of the experiments). Since all these experiments were performed at critical research facilities working with fresh fuel assemblies the applicability of the results is restricted.

On the other hand, in the WWER environment there is a persisting urgent need of any validation work so such results are definitely a valuable contribution.

As far as the methodology for the depletion calculation is concerned, the validation level is even worse. In the WWER application area there are not any radiochemical assays the results of them could be applicable to validate depletion calculations related to the BUC implementation. That is why big effort was also aimed at obtaining a moral, financial and technical support for the future in this area.

It is expected that in the future the validation will continue not only in the framework of a national project of one country but will be supported by the whole research community of the countries operating WWERs. To start with this, a series of international calculational benchmarks on WWER-440 BUC methodology have been proposed, coordinated, calculated and evaluated, most of them under this project. The overview of the results of the CB1, CB2, CB3 and CB4 benchmarks were presented at [8] as [15]. These calculational benchmarks contributed to the methodology understanding as an illustrative and didactic tool giving a view of conservatism and reactivity margins in the WWER environment.

DATABASES

This stage of the project was focused on establishing an efficient collaboration with the NPP Dukovany resulted in the creation of the MS Access Database of spent fuel records [16]. Having been proposed and compiled the database was continuously used to support the BUC implementation methodology development. The database has been filled with data of spent fuel, which have been discharged, from the four Dukovany NPP units. Spent fuel axial burnup profiles and operational ranges of key reactor parameters having influence on fuel depletion could be checked using the database. Further, using this database a special study was devoted to finding out how many and how frequently the fuel assemblies were placed close to the HRK-control rods and what were their axial burnup profiles after each cycle as well as time coincidences of placing these 'affected' fuel assemblies to the same cask. Another database specifically focused on burnup profiles of spent fuel was created by SKODA JS developers [17]. Based on the operational data of the four Dukovany NPP units, spent fuel burnup profiles given as 20 nodes were calculated using the fuel management code.

DESIGN STUDY

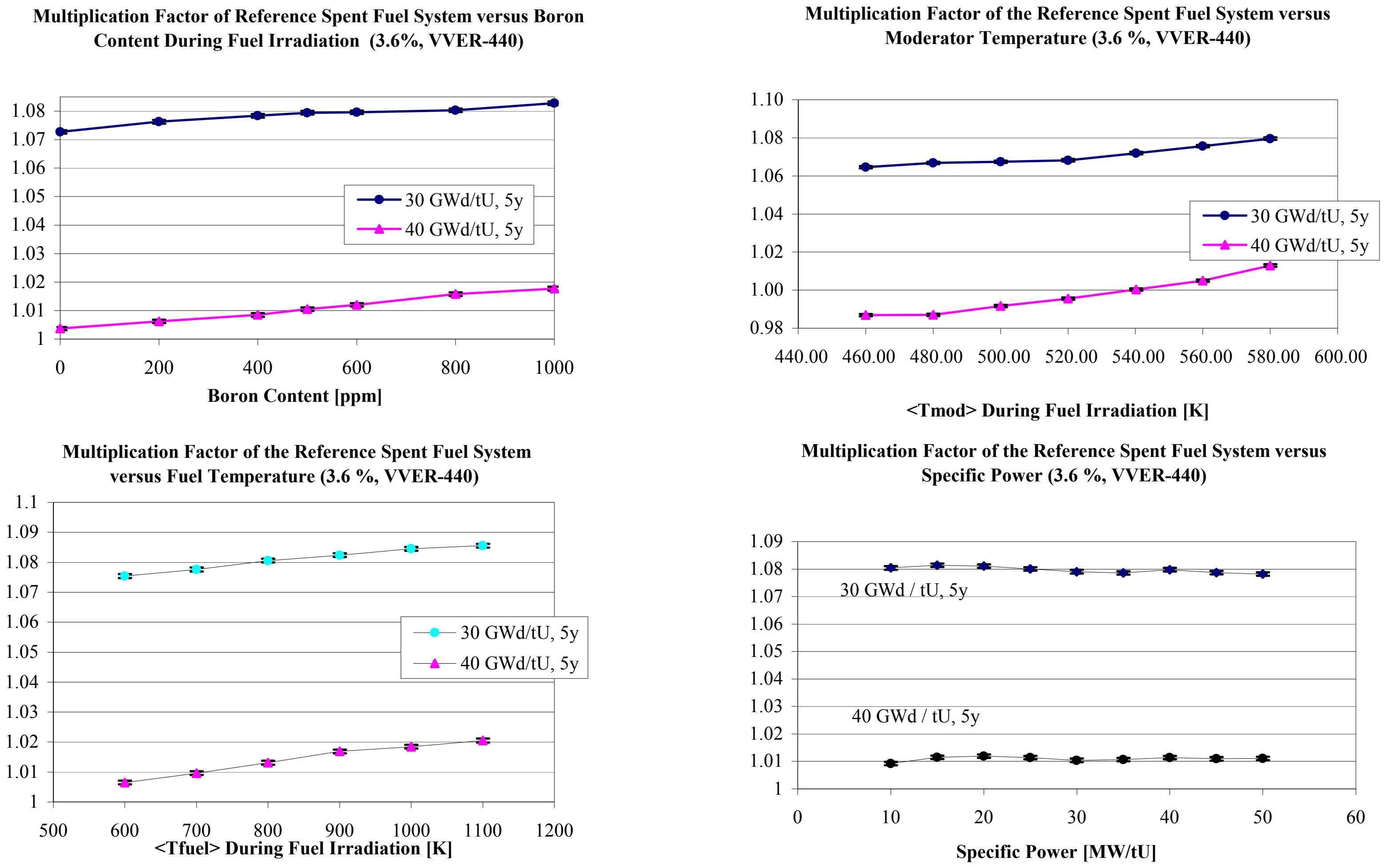
The newly developed design of the dual-purpose all-metal cask (entitled 'SKODA BUC' as a working name) was based on the current WWER-440 dual cask design SKODA 440/84 (licensed in 1999 for (railway) transport and storage in the Czech Republic), which did not use BUC.

A long series of depletion and criticality calculations related to varying heat transfer and removal strategies were made to reselect materials and change dimensions to fix upon new performance features. Finally, the SKODA BUC cask was designed for 120 WWER-440 fuel assemblies (Fig. 3) in comparison with 84 fuel assemblies of SKODA 440/84. According to specific loading curves ([18], [19], one of them can be seen in Fig. 4), the new cask design gives a possibility to accommodate either the former WWER-440 spent fuel of 3.6 % enrichment and 34 GWd/tU minimal burnup or the newer profiled fuel of 3.82 % average enrichment and 36.5 GWd/tU minimal burnup as well as the new advanced fuel with Gd burnable absorber of 4.38 % average enrichment and 43 GWd/tU minimal burnup (of the three possibilities the latter two are still in development).

From a structural point of view, the final conception [20] is based on a ribbed cast steel body and stiffed copper basket. The mass of the empty/fully-loaded cask is 105/130 t, respectively. The cask is filled up with He of 0.07 MPa. The outer dimensions of the cask cylinder (of about 4m height and 2.8 m outer diameter) were limited by a manipulation space at NPP and the way of transportation using railway wagons. The use of borated steel, which is a significant material of the SKODA 440/84 cask performance, turned out unnecessary for the SKODA BUC cask. Due to higher heat transfer and removal requirements not only steel but also copper was used as a structural material. The inner container, which creates the first seal barrier in the case of the current SKODA 440/84 cask, were not used any more, which improved heat transfer conditions. The introduction of copper called for a lot of sophisticated structural improvements (e.g. the use of longitudinal stiffeners) due to meeting the safety requirements and performance-oriented standards for the transport casks. In the end, the design development resulted in meeting all the transport standards given by IAEA and the Czech regulatory body regulations (142/97 Sb.). Temperature of easy accessible cask surface should not exceed 85 °C on condition that surrounding temperature is between – 40 °C and +38 °C. As for the storage, the cask meets ANSI/ANS-57.9-1992 Design Criteria for an Independent Spent Fuel Storage Installation (dry type), Safety Series 118 – Safety Assessment for Spent Fuel Storage Facilities (IAEA, 1994) as well as Safety Series 116 – Design of Spent Fuel Storage Facilities (IAEA, 1994)).

CONCLUSIONS

Despite focusing prevailingly on the BUC implementation methodology development, the project needed large experience from calculational as well as structural points of view for the new cask design to be developed. The BUC implementation gave a possibility of a simplification of the structural conception of the cask basket. On the other hand, the whole cask design had to fulfill all the safety requirements related to transport which are put on dual purpose casks as well as to cope with the necessity of a higher heat transfer and removal. The successful conception resulted in about 50% increase of the cask capacity (the change of 84 into 120 fuel assemblies) and, which is very important at the same time, even in about 10% decrease of the estimated price of the whole cask in comparison with the current design of the SKODA 440/84 cask.



Multiplication Factor of Reference Spent Fuel System versus Boron Content During Fuel Irradiation (3.6%, VVER-440)

Multiplication Factor of the Reference Spent Fuel System versus Moderator Temperature (3.6 %, VVER-440)

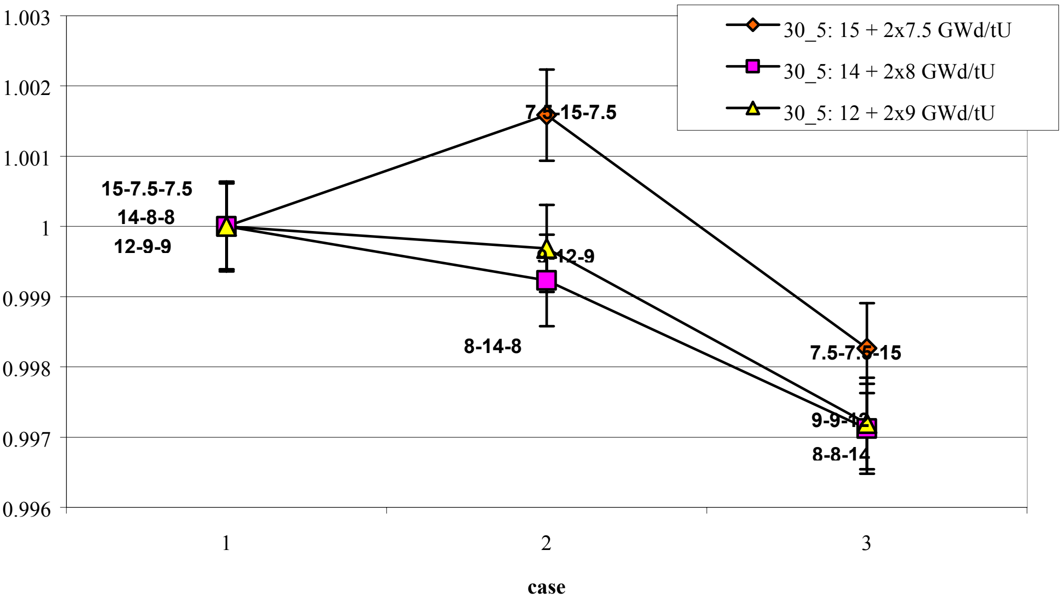
Multiplication Factor of the Reference Spent Fuel System versus Fuel Temperature (3.6 %, VVER-440)

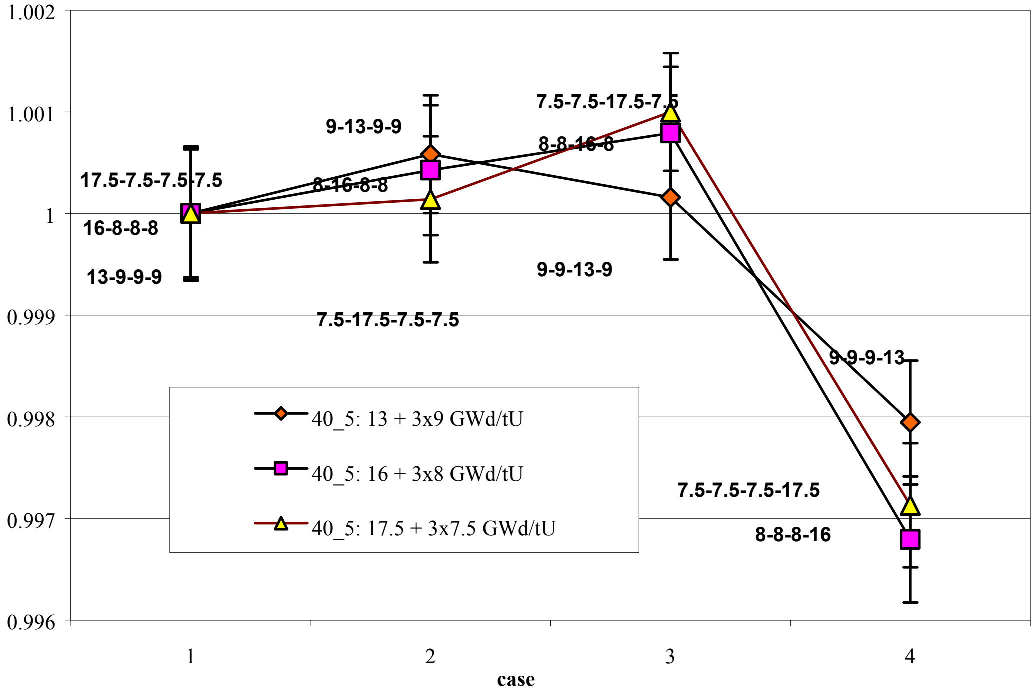
Multiplication Factor of the Reference Spent Fuel System versus Specific Power (3.6 %, VVER-440)

1.08
1.07
1.06
1.05
1.04
1.03
1.02
1.01
1
0
200
400
600
800
1000
Boron Content [ppm]
30 GWd/tU, 5y
40 GWd/tU, 5y
1.10
1.08
1.06
1.04
1.02
1.00
0.98
440.00
460.00
480.00
500.00
520.00
540.00
560.00
580.00
600.00
<Tmod> During Fuel Irradiation [K]
30 GWd/tU, 5y
40 GWd/tU, 5y
1.1
1.09
1.08
1.07
1.06
1.05
1.04
1.03
1.02
1.01
1
500
600
700
800
900
1000
1100
1200
<Tfuel> During Fuel Irradiation [K]
30 GWd/tU, 5y
40 GWd/tU, 5y
1.09
1.08
1.07
1.06
1.05
1.04
1.03
1.02
1.01
1
0
10
20
30
40
50
60
Specific Power [MW/tU]
30 GWd / tU, 5y
40 GWd / tU, 5y

FIG. 1. Trends in k

Fig. 2 Normalized Multiplication Factor of the Refference Spent Fuel System versus SpecificPower and History Variations
(3.6 %, VVER-440)





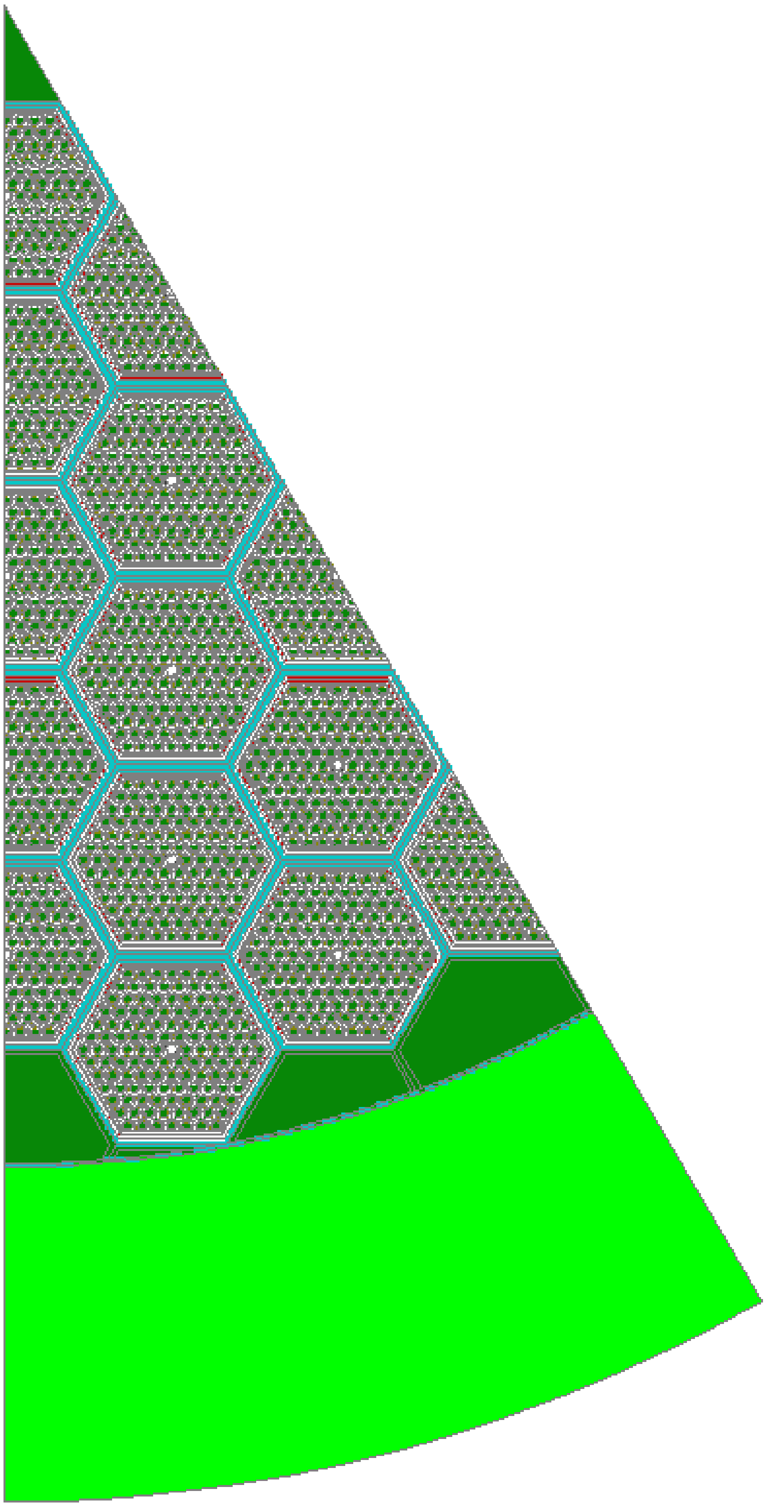


FIG. 3. 30° symmetry segment of SKODA BUC dual purpose cask

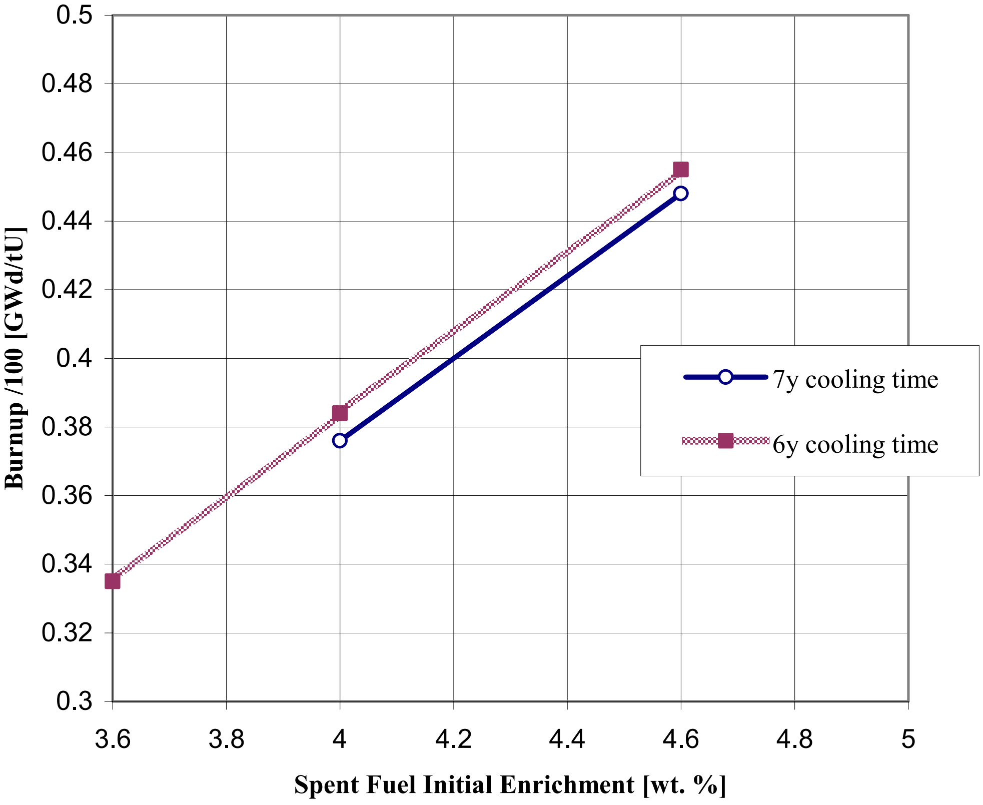


FIG. 4 SKODA BUC Cask Loading Curve for VVER-440 Spent Fuel

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Computer codes qualification in the Czech Republic

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Abstract. SUJB (Regulatory Body in the Czech Republic) approach to the appraisal of the computer codes, which were applied for nuclear installation safety evaluation, is given. SUJB is governing the whole appraisal process according to the given rules. The aim is to establish a recorded complex and consistent set of computer codes and a simplification of safety documents review and evaluation process conducted by SUJB. As the computer codes (databases, libraries, correlations) used for the nuclear safety assessment belongs to professionally different areas, seven Technical Appraisal Committee of experts are set up in order to perform the appraisal (evaluation) according to the computer code orientation. The task of Technical Appraisal Committees, principles of the activities of Technical Appraisal Committees, documentation provided to technical evaluation committees, uniformity of the computer code evaluation, appraisal process in the Technical Appraisal Committee are discussed.

1. INTRODUCTION.

SUJB considers the quality and the suitability of the computer codes, which are used for the safety assessments, as very important part of decision-making process if these codes are used to provide evidence of compliance with the regulatory requirements and safety standards i.e. for demonstration that the nuclear power plant can achieve the acceptable level of safety (new plant) or that the safety of the nuclear power plant in all aspects is maintained (for design changes).

There is a requirement in Regulation No. 195/1999 Sb. of the State Office for Nuclear Safety of August 21, 1999 on Requirements on Nuclear Installations for Assurance of Nuclear Safety, Radiation Protection and Emergency Preparedness experessed in § 4 section (3) “The quality and suitability of computational programs used for the analyses important for nuclear safety shall be verified”. The Czech legalistic requirement is in compliance with similar IAEA recommendation. According to paragraph 732 of the IAEA Safety Standards Series, Draft NS 248 Review and assessment by regulatory body for Nuclear Facilities “… As part of its review and assessment, the regulatory body should ensure that the computer codes are based on well-understood principles. Computer codes should be validated against experience or experiment that the coding has been done accurately and the input data have been correctly assigned…”

Only qualified codes, accepted for these purposes by the SÚJB, can be applied for the calculational analyses.

In the Report of the International Regulatory Review Team (IRRT) Mission to Czech Republic held from 4 to 15 June 2001 is stated as Good Practice: “SÚJB has formed a Commission for validating computer codes, which are used in the safety assessments. Use of validated computer codes in the licensing process increases its efficiency and reduces the pressure to the regulator and licensee.”

With respect to the scope demands complexity nuclear installations safety aspects, it is reasonable, from the SUJB point of view, to perform an independent quality evaluation and the applicability of calculating codes used for the development of safety documents. The evaluation of computer codes should be considered as a continuous activity. The computer codes are continuously improved, which is enabled by the increase of knowledge in the relevant area of problems and the continues improvement in computer technology and it results from the necessity to describe the phenomena and events taking place in the NPP better and in more detail and in more complex way.

The regulatory guide VDS 030 gives the SUJB approach to the appraisal of the computer code for nuclear installations safety assessment. Main purpose of the computer codes appraisal is verification of the adequacy of the code used to perform calculation to assess properties of the plant. But it has got some other very useful side effects. It enables to maintain a systematic overview on usable computer codes of good quality, which are used in CR for the safety evaluation of NPP. It also initiates the development and the completion of programs in areas, which have not been sufficiently covered so far. It initializes along with the computer technology development the switchover from the fast engineering methods (imperfect) to the sophisticated and more accurate methodologies. It creates a condition for the transferring of high quality computer codes.

The result should be a complex and consistent set of computer codes under a record keeping of SUJB and a simplification of safety documents review process conducted by SUJB, in course of which the positively evaluated computer codes will be used.

2. CATEGORIZATION OF THE COMPUTER CODES

Computer codes (databases, libraries, correlations) used for the nuclear safety assessment belong to professionally different areas. For the above reason seven technical committees are set up in order to perform the corresponding evaluation according to the computer code orientation. The technical committee members including its chairmen are appointed by SUJB chairman. The technical committees will be staffed with Czech republic experts working in the major development facilities involved in the areas of nuclear installments safety evaluation. In all the technical committees staff will be SUJB representatives.

The categorization of computer codes for the purpose of evaluation in particular technical committees and more detail scope of activities are as follows:

2.1. Neutron-physical calculations.

This group of computer codes covers the following categories of computer codes:

a. Transport multigroup codes or codes using point kinetics nuclear data, applicable for:
   - preparation of a few group diffusion and transport constants for core analysis,
   - direct solution of the above mentioned tasks,
   - criticality analysis, heat sources development and irradiational effects in nuclear fuel storage facilities.

b. Diffusion and transport few group computer codes designated for:
   - determination of a steady state neutron-physical core characteristics (reactivity, reactivity effects, critical parameters, neutron flux and power distribution, etc.) of a commercial and experimental reactors and its time dependence (fuel burnup, time dependent fuel poisoning) and on the changes of primary circuit parameters,
   - analysis of neutronic and power distribution behavior and reactivity during short term transient processes,
   - reload cores optimization and optimization in general

These computer codes can contain thermohydraulic and thermomechanic moduls. In cases when such modules prevail the computer code is categorized as belonging to the following categories 1.2 and 1.3.

c. Special codes creating on-line core monitoring systems (power distribution reconstruction, predictive calculation)
d. Database
    - of the data needed for neutron-physical characteristics calculation,
    - of the neutron-physical experiments data (banks, testing problem)
e. Libraries of a few group, multi group or point data

2.2. Thermohydraulic analysis of the Reactor and Plant circuits Transient and Accident events

This wide area covers analysis of steady state, transient and accident NPP evens within a design basis accident concept:

- computer codes for analysis of steady state, transient and non-steady state phenomenons within fuel assembly eventually with possible occurring two-phase flow including inter channels heat and quantity of motion exchange
- overall thermohydraulic core analysis for various operational conditions(different number of main coolant pumps in service, different burnup etc.) with possible inclusion of system of computer codes for analysis of reactor and plant control system and for system stability overcheck
- computer codes for transient and accident events (processes), pressurizer pressure description, main coolant pumps, steam generators, etc. Apart from computer codes for analysis of automatic control actions, here a strongly non linear processes are concerned
- computer codes for analysis of phenomenon occurring in plant including secondary circuit (turbine trip, steam line break, steam generator feedline pumps malfunction, etc.)
- computer codes for analysis of thermal-hydraulic processes resulted from loss of coolant type accidents
- thermohydraulic analysis of conditions existing in hermetical boxes, pressure relieve system and containment during loss of coolant type accident processes

2.3. Fuel rod behaviour calculations (analysis)

As a result of fission process there occurs under long time operational loads fuel swelling, volume growth, pressure increase within a rod internal volumes. The pellet - clad interaction and cladding - spacer interaction occurs, pellet-clad gap heat conductivity changes. Substantial therefore is to arrange for data for the evaluation of transient and accident conditions (states) modeling uncertainties from point of view fuel safety criterions maintenance.

Evaluation of a fuel rods behaviour at such conditions includes:

- development and verification of the methods for operational reliability evaluation
- fuel safety criterions accuracy improvement and development of potential proposal for modifications
- evaluation and verification of methods designated for determination of thermal-mechanical loads in operational and accident modes

- methods for evaluation of fuel damage degree and methods for evaluation of radiologically significant fission product leakage from fuel under the conditions of design accident development to the beyond design basis accidents

2.4. Beyond design basis accidents

For the purpose of evaluation of beyond design basis accident time behavior and consequences, this area includes:

- set of computer codes for thermohydraulic and strength analysis of plant beyond design basis accident time behavior and consequences
- determination of a source term of activity leaking from plant as basis for development of emergency plans and liquidation of consequences of such accidents
- evaluation of accident scenarios risk on the basis of calculational analysis results using probabilistic safety assessment (thermohydraulic eventually strength-deformation, radiation, radiological point of view)
- performance of an analysis of the possibilities of design basis accident to develop into beyond design basis accidents

2.5. Strength calculations of a components and piping systems

This group of computer codes covers:

- strength calculations of a components and systems
- calculations of a residual life time of components and plant apparatus
- calculations within the approach "leak before break" applied to the primary circuit piping system
- evaluation of a compatibility with reputable world standards
- Within the scope of strength calculation primarily the following is a subject:
- temperature field and temperature tension
- static and dynamic analysis
- low frequency and high frequency fatigue
- break - mechanical analysis (strength intensity coefficient, fatigue crack growth, crack stability)
- coolant leakage through the circumferential and axial throughole
- probability of failure
- processing of measurement results
- supporting graphic means

2.6. Radionuclides transport and Source term calculation

In the primary circuit released radioactive substances (fission products, activated coolant, corrosion products) propagates through untightness into the environment. Released products chemically react with environment, a part is binded with coolant, part is absorbed on the structural surfaces, a part is removed by the filtration systems, a part escapes into environment via different paths at continuous decay. There is scattering in atmosphere and after deposition, these products can get back to the man via various food chains. Separate processes can be modeled such that the circumstances can be understood and influence of different conditions on the leakage propagation can be checked. The complex scatter into environment is divided into the following phases:

- calculation of the activity transport in the air, water and other substances from nuclear energy sources
- calculation the fission and corrosion products transport within technological circuits and facilities
- evaluation of the plant staff and population irradiation loads in the plant surrounding area during normal and emergency conditions
- fission products balance in spent fuel
- preparation of a libraries and input data for these calculations
- shielding calculation and irradiation doses calculation
- database for determination of radiation term

2.7. Probabilistic safety and reliability analysis

Probabilistic safety assessment identifies and ties the combination of events resulting in severe accidents, determines probability of occurrence of such combination and determines its consequences. The overall risk value is a quantitative measure of accomplished safety level of NPP. It can be compared with another technical risks of society. In course of its establishment can be revealed the most significant contributions to such risk. It is reasonable to focus on minimization of such risks.

Computer codes for such analysis can be either complex (they include several of the following areas) or specialized to certain area only.

The respective areas to be considered are:

- construction of the failure trees and their evaluation
- construction of the event trees and their evaluation
- reliability data collection and treatment, database preparation
- analysis of human factor influence
- uncertainty analysis
- sensitivity analysis
- calculation of the reliability indicators
- special programs (e.g. "Risk monitor")

3. COMPUTER CODES QUALITY REQUIREMENTS

In the past studies on the SW quality level were made, these defined a variety of incongruous aspects of quality which resulted in the need for unification of computer codes evaluations.

Because of this, the ISO/IEC technical committee began work on the development of the needed consent and supported the normalization on a worldwide basis. The development of the international standard ISO/IEC 9126 has began in 1985.

In the June of 1994 a standard CSN ISO/IEC 9126 "Evaluation of a software product - quality characteristics and the guide for their utilization". In appendix to this standard a statement is made that the current status of the software technology still does not represent well established and generally accepted approach for software product evaluation. This existing standard (CSN ISO/IEC 9126) is followed by another standard ISO/IEC 12119 Information technology "software packages - quality requirement and testing".

It appears from SUJB point of view as most convenient to accept the rules for computer code evaluation based on the standards CSN ISO/IEC 9126" Evaluation of software product - quality characteristics and the guide for their utilization" and ISO/IEC 12119 Information Technology - " Software packages Quality requirement and testing", modify and develop the rules taking into account specifics of technical calculation.

Quality requirements include:

- Requirement, to have a product description for each software package (software package properties) and users documentation
- Requirements for the product description including designated data and taking into account, that computational conditions which can occur have to be testable
- Requirements for the users documentation
- Requirements for the computer codes - eventually data too
- Testing conditions
- Requirements of the staff and workplace qualification, if requested by Technical Appraisal Committee

4. SUJB MANAGING OF THE COMPUTER CODES APPRAISAL PROCESS

There are basic principles for accepting computer codes

- The computer codes can be evaluated in case they are obtained legally only
- For the evaluation process are accepted only codes developed by organization with QA procedure
- The application for the evaluation of the computer code by Committee has to be in written form approved by developing (author) organization.

4.1. Principles of the activities of Technical Appraisal Committees

The principles for technical evaluation committees (Committees) activities are set as follows:

1. Committees are the SUJB chairmen’s advisory organ for evaluation of computer codes belong to given area,
2. Committee chairman calls the session
3. Committee task is to secure the computer code evaluation process in the given area and to suggest including the evaluated computer code into the set of evaluated codes.
4. The suggestion of the computer code for the evaluation by Committee shall be in written approved by developing organization.
5. The foreign computer codes can be evaluated in case they are obtained officially only
6. From the Committee membership does not follow any commitments for the organization sending a delegate. The Committee members act as independent experts; their membership is non-replaceable. At the Committee meeting can be invited another experts.
7. The majority expert opinion is important for the acceptance of Committee meeting conclusions. The different positions shall be documented.
8. Committee chairmen via SUJB representatives to the Committee hand over the results of evaluation of particular computer code to SUJB for approval.
9. The changes made in the bases used for computer code evaluation shall be provided to all holders of relevant documentation in accordance with appendix No 5.

10. Committees will suggest the way to reevaluate or set aside from the set of evaluated computer codes those, which until now passed the so called Standardization according to the rules of this Regulatory guide.
11. The decision to include the computer code into the set of evaluated codes is subject to reconsideration after three years.
12. The legal responsibility of developing - author organization for the quality of the computer code and its users is not after evaluation performed by technical evaluation committees transferred on SUJB.

4.2. The task of Technical Appraisal Committees

The technical committee main tasks include:

- Selection of computer codes for appraisal within the appraisal procedure based on their utilization in course of safety documentation development for Czech nuclear installations.
- Collecting the relevant documentation from the organization suggesting the code for appraisal process,
- Selection of opponents who can elaborate technical position on the inclusion of actual computer code into the set of evaluated codes,
- Reaching the position on applicability range of the computer code under appraisal,
- Transfer the appraisal result and the relevant documentation to SUJB.

4.3. Documentation provided to Technical Appraisal Committees

The organization asking for evaluation (the applicant) will provide to the Technical Appraisal Committee the documentation covering the following information:

a. Abstract of computer code, which contain a set of computer code relevant information.
b. QA document and declaration of organization
c. Certification of legal purchase, leasing agreement or similar document giving rights to use the computer code in case of adaptation of this code from other organization.
d. Certificate for adopted computer code, attesting regulatory body of the country of origin approval to use the computer code for nuclear safety evaluation (the position of the regulatory body of computer code country of origin with respect to the suitability of the code to be used for nuclear safety evaluation). Needed in case of "shortened procedure for computer code appraisal"
e. Technical reports related to the computer code under appraisal:
   - *Introduction* - overview and the status of the given problems solution in the world with providing the connections to the computer codes and experiments. Specification of the area of applicability of the assumed computer code. The main parameters characterizing the NPP safety, which are followed (monitored) by the computer code.
   - *Physical model* -detail description of the physical model with the list and discussion of all the used assumptions and limitations.
   - *Numerical model* - the description of the used mathematical model with the analysis of compatibility and verification of the convergence and stability of the numerical process.
   - *Computer code* - brief characterization of the computer code (demands on the computer, type and the configuration of the computer, programming language, structure, demands on the memory, speed of calculations and so on). The comparison of the program with currently significant computer codes. To provide comments on further developments and its eventual further validation/verification.

- *Calculation results* - a report with results of the computer code validation based on the experimental, operational or other acceptable reference data, eventually comparison to another computer codes of the same category.
- *Conclusion* - Comparison of the computer code with current foreign computer codes, Outline of further possible development and its eventual further validation/verification
- *Manual* - a guide how to use the computer code (In case of its unavailability it can be replaced by description of QA for inputting of the input data, computer code utilization).

4.4. Uniformity of the Computer Code Appraisal

In order to assure uniform level of evaluation of the computer codes, eventually correlations, databases, libraries it shall be the computer codes developer′s effort and technical committees too to arrange for the comparable (if possible identical) form of documentation and bases provided for the opponent proceedings, methods of controlling and testing, criteria for evaluation and testing, computer code abstract, opponent opinion report technical evaluation committee opinion on the evaluated computer code use for development of bases for safety documentation data for computer code record keeping.

The position of the technical evaluation committee will be elaborated based on the results of the opponent process. The chairman of the technical appraisal committee will fill out the form "Position of the Technical Appraisal Committee to the use of the evaluated computer code" (Computer Code Appraisal Certificate) after the opponent procedure will be closed. The "Position of the Technical Appraisal Committee" is signed by the chairman and the opponents.

A new method to take burnup into account in criticality studies considering an axial profile of burnup plus some fission products

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Abstract. In order to avoid criticality risks, a large number of facilities using spent fuels have been designed considering the fuel as fresh. This choice has obviously led to considerable safety margins. In the early 80's, a method was accepted by the French Safety Authorities allowing operators to consider the changes in the fuel composition during depletion with some very pessimistic hypotheses: only actinides were considered and the amount of burnup used in the studies was equal to the mean burnup in the 50-least-irradiated centimeters. As many firms still want to optimize their processes (e.g. transportation, storage, fuel reprocessing), the main companies involved in the French nuclear industry and IPSN set up a Working Group in order to define the way burnup could be taken into account in the criticality calculations, considering some fission products plus a more realistic axial profile of burnup. The first part of this article introduces the current French method used to take burnup into account in the criticality studies. The second part is devoted to the studies achieved by the Working Group to improve this method, especially concerning the consideration of the neutron absorption of some fission products and of an axial profile of burnup: for that purpose, some results are presented related to the depletion calculations, the definition of an axial profile for the calculations, the knowledge of the fission products absorption and the calculation tools. In the third part, the results (keff) obtained with the new method are compared to those obtained with the current one. The conclusions presented are related to the present state of knowledge and may differ from the final conclusions.

INTRODUCTION

Up to the 80's, the nuclear facilities, which were dealing with spent fuel, were designed with the assumption of fresh fuel. This assumption led to considerable safety margins.

In the early 80's, in order to use the existing devices at La Hague reprocessing plant for some irradiated fuel initially enriched at 4.4% (this enrichment was higher than the highest enrichment considered at the designing stage); a method was proposed by COGEMA to enable them to consider a certain amount of burnup in the criticality studies. However some pessimistic assumptions were made to guarantee some safety margins:

- only uranium and plutonium were considered after the depletion of ^{235}U and ^{238}U during the irradiation;
- the amount of burnup used in the criticality studies was lower than the value reached in the 50-least-irradiated-centimeters;
- the value of the mean burnup in the 50-least irradiated-centimeters was verified by a measurement.

The calculations were supported by the HTC experiments, which were achieved in Apparatus B in Valduc. Those subcritical experiments [1] involved fuel pins, representative of a fuel initially enriched at 4.5% and irradiated at 37.5 GWd/t, manufactured only with actinides. The pins arrays were arranged in different types of configurations, which were representative of reprocessing, storage and transport.

This actinide-only method was accepted by the French Safety Authorities and was, afterwards, used for the transport of irradiated fuel and, also, at the designing stage of the UP3 and UP2-800 at La Hague reprocessing plant.

But, since the initial enrichment is increasing and the needs of interim storage of irradiated fuel is growing, it is becoming necessary for the nuclear industry to reduce the conservatisms due to the very pessimistic hypotheses of this actinide-only method.

For that purpose, a Working Group was created in 1997, gathering most of the French nuclear companies, to analyze different propositions for a new method where some fission products plus a more realistic axial profile of burnup are considered.

A NEW METHOD TO TAKE BURNUP CREDIT INTO ACCOUNT

The new points introduced by the method are related to:

- the neutron absorption of some fission products; the method proposes to first take into account the 6 following fission products 149Sm, 152Sm, 103Rh, 143Nd, 133Cs and 155Gd, which are responsible for 50% of the absorption of all fission products. This list should eventually be extended to the 9 other fission products which are considered by the OECD (99Tc, 145Nd, 153Eu, 95Mo, 147Sm, 150Sm, 151Sm, 109Ag and 101Ru);
- a more realistic description of the axial profile of the burnup.

As those new considerations will reduce the conservatisms, it seems necessary to guaranty the validity of the assumptions made to:

- determine the composition of the irradiated fuel (especially for fission products);
- define the axial profile of the burnup;
- and compute the simulation (particularly regarding the knowledge of the cross-sections of the isotopes that are being taken into account).

For that purpose, the following paragraphs describe all the assumptions made during (i) the depletion calculations (ii) the definition of the axial profile of burnup in the studies, (iii) the criticality calculations; those assumptions have to ensure the conservatism of the method. Moreover, the amount of burnup (its mean value over the fuel assembly and its axial profile) will have to be firmly guaranteed by some precise measurements. Thus the safety of the method will be assured.

The Working Group has been divided into 4 sub-groups to study each of those subjects. The following paragraphs give the proposed recommendations that have already been issued by the Group and the studies, which are on-going for each subject.

Depletion calculations

First of all, it seems necessary to guarantee in the method that, for a given amount of burnup, the depletion calculations lead to a conservative inventory of the irradiated fuel.

It can be noticed that, in the depletion calculations, the burnup is determined by the calculated ratio of some isotopes concentrations (e.g. Nd); thus the qualification of the method will be tightly linked to the chosen "indicator of burnup".

The conservatism of the depletion calculations depends on:

- the conditions of irradiation [2]; the Group decided to quantify its effect considering (i) the spectrum hardening, (ii) the fuel temperature variations (Doppler effect), (iii) the temporal variation of the flux level (variations of specific power), (iv) the cooling time;
- the qualification of the depletion codes which are used for the calculations i.e. CESAR [3] or DARWIN [4]

IRRADIATION HISTORY

The main parameters that are responsible for changes in the fuel inventory (neutron spectrum, fuel temperature, specific power, cooling time) are discussed in this paragraph.

Spectrum effect

It has already been shown [1] that, for a given burnup, the fuel is more reactive when it is irradiated with a hardened neutron spectrum. Consequently, the different conditions of irradiation that might harden the spectrum have been sought.

During the irradiation, the spectrum may be hardened when:

- control rods are inserted; this situation leads (for UO_2 initially enriched at 4.5% and irradiated at 44 GWd/t) to a penalty up to $\Delta k = 3.5\%$,

- ➢ the UO_2 assembly has been irradiated near some MOX assemblies; the penalty with a very pessimistic assumption (the MOX assemblies were present on the 4 sides of the UO_2 fuel during 4 cycles) leads to a Δk near 1.5%,
- ➢ the temperature of the water (moderator) is increased (the temperature is set to its maximum value at the edge of the reactor core),
- ➢ the concentration of the boron during the cycle is increased,
- ➢ the presence of burnable poison is considered.

The Working Group proposes to consider, for the depletion calculations, the presence of the control rods, the maximum concentration of the boron, a temperature of the water set to its out-of-core value.

Studies are still on going to determine how the spectral effect of the presence of some MOX fuels around the UO2 assembly should be taken into account.

Doppler effect

The temperature of the fuel used during the depletion calculations has a slight effect on the reactivity. Actually, it is conservative to consider a high value of the temperature as it leads to more absorptions on the ^{238}U (and then to more productions of ^{239}Pu). Consequently, the temperature of the fuel may be set to the maximum value of the effective temperature of the fuel.

Temporal effect

The temporal effects are linked to the variation of the specific power (SP). Actually, this effect is the consequence of the competition between neutron absorptions (which are directly related to the value of the burnup) and decay reactions (related to the time spent in the reactor).

Some studies [3] showed that an increase of the specific power is conservative (when no cooling time is considered) even if fission products are taken into account. The results obtained for a single assembly surrounded by 20 cm of water (initially enriched at 4.5% and irradiated at 44 GWd/t) are given below.

	keff (σ<0.095%)				
	SP ↓	11 GWd/t	22 GWd/t	33 GWd/t	44 GWd/t
cooling time: 0 year	5 W/g	0.8639	0.8150	0.7757	0.7385
	20 W/g	0.8648	0.8217	0.7837	0.7508
	40 W/g	0.8664	0.8209	0.7832	0.7515
	60 W/g	0.8668	0.8215	0.7820	0.7525
cooling time: 5 years	5 W/g	0.8626	0.8094	0.7603	0.7233
	20 W/g	0.8623	0.8062	0.7637	0.7264
	40 W/g	0.8589	0.8039	0.7634	0.7238
	60 W/g	0.8587	0.8045	0.7595	0.7194

Single assembly (depleted with the control rods inserted - 15 Fissions products)

It emerges from the table that, for a cooling time of 5 years, the highest specific power does not give the most penalizing result. This has already been shown in some previous publications [5] and can be explained by the behavior of some isotope concentrations when the specific power varies.

While considering the isotope concentration variations, the calculations pointed out that, when the specific power increases, the concentrations of the absorbant isotopes decrease and the fissile ones increase, except for the following isotopes:

- ^{149}Sm and ^{152}Sm which increase respectively by 52% and 5.5% when the specific power grows from 5W/g to 60W/g (at a burnup of 40 GWd/t),
- ^{242}Pu which slightly increases by 2.8% when the specific power grows from 5W/g to 40 W/g (at a burnup of 45 GWd/t),
- ^{155}Eu (father of ^{155}Gd), which increases by 26% (at 20 GWd/t) and 44% (at 44 GWd/t) when the specific power grows from 5 to 60 W/g.

Then, the different behaviors observed in the table can be explained. In fact, the reactivity changes when the specific power grows from 5 to 60 W/g could be written as follows:

$\Delta\rho_{(5W/g \rightarrow 60W/g)} = \Delta\rho({}^{149}\mathrm{Sm}, {}^{152}\mathrm{Sm} \text{ and } {}^{155}\mathrm{Gd}) + \Delta\rho({}^{239}\mathrm{Pu}) + \Delta\rho({}^{241}\mathrm{Pu}) + \Delta\rho({}^{241}\mathrm{Am}) + \Delta\rho(\text{others})$

$\Delta\rho(X)$ represents the changes in reactivity when the concentration of the isotope X is modified by the change in specific power. The previous paragraph gives the isotopes whose concentrations change with the specific power; thus:

- $\Delta\rho({}^{239}\mathrm{Pu}) < 0$, $\Delta\rho({}^{241}\mathrm{Pu}) < 0$ and $\Delta\rho({}^{241}\mathrm{Am}) < 0$,
- $\Delta\rho({}^{149}\mathrm{Sm}, {}^{152}\mathrm{Sm} \text{ and } {}^{155}\mathrm{Gd})>0$ and $\Delta\rho(\text{others})<0$.

Moreover, the isotope concentrations, which undergo significant changes during 5 years of cooling time, are: ^{149}Sm (+50%), ^{155}Gd (>1500%), ^{239}Pu (+2%), ^{241}Pu(-20%) and ^{241}Am (+700%). If $\Delta\rho_5$ stands for cooling times of 5 years and $\Delta\rho_0$ stands for no cooling times:

$\Delta\rho_5({}^{149}\mathrm{Sm}, {}^{152}\mathrm{Sm} \text{ and } {}^{155}\mathrm{Gd})\ \Delta\rho_0({}^{149}\mathrm{Sm}, {}^{152}\mathrm{Sm} \text{ and } {}^{155}\mathrm{Gd})\ (> 0)$.

Then the assumption of a maximized specific power, which led, for no cooling time, to a conservative value of the keff is no more conservative when a 5 year cooling time is considered.

Finally the specific power may be set to its maximum value; for the 4 isotopes described above, some correction factors should be used to consider their non-conservative increase when the specific power is over-evaluated.

Cooling time

Many studies have shown [2] [6] that, after the irradiation and up to a cooling time of 100 years, the reactivity decreases. This decrease is mainly due to the decay of the ^{241}Pu (to ^{241}Am) plus the increase of ^{155}Gd.

After 100 years of cooling time, the reactivity starts to increase again (as the ^{241}Am decays) until around 30 000 years [7].

The needs, in terms of cooling time, for the French Working Group participants mainly concern some devices, which are to be used for cooling times of less than 50 years. For those applications, it should be acceptable to consider, in the criticality studies, the minimum of the cooling time that can be justified by the operators[1].

However, for interim storage, this approach raises the problem of how operators can guarantee that, after 100 years of cooling time, a solution will exist to move the fuel to some other adapted storage? For this type of storage and for cooling times greater than 100 years, some special studies will have to be done to determine the maximum of the reactivity when the cooling time is growing. It should be decided not to take into account the ^{241}Am in those studies.

QUALIFICATION OF THE DEPLETION CODES

The depletion codes that will be used are CESAR [4] or DARWIN [5] Codes. They have been qualified on an experimental basis recalled in appendix. This qualification rests on comparisons between calculated values of the concentrations and measured ones. Those comparisons have been achieved, on both (i) punctual analysis of irradiated fuel (with initial enrichment of 4.5% and burnup up to 60 GWd/t) (ii) global analysis during the dissolution of irradiated assemblies (with initial enrichment comprised between 3.1% and 3.5% and burnup up to 45 GWd/t).

These comparisons will give correction factors for the calculated concentrations of each of the actinides and fission products, which are considered in the method.

Definition of an axial profile of burnup

Until now the value of the burnup, which was applied on the whole length of the assembly, was equal to the mean value of the 50-least-irradiated-centimeters of the assembly.

This assumption was very conservative, giving a reduction of 22% of the mean burnup for a standard profile (e.g. 34 GWd/t will be applied to the whole length of the assembly whereas the real mean burnup would be equal to 44 GWd/t).

In order to consider a more realistic profile, it was decided to determine a profile for the criticality studies by:

- examining the profiles measured at La Hague reprocessing plant (more than 3000 irradiated assemblies have been measured),
- calculating different types of profiles to determine a penalizing one.

Those two studies pointed out that:

- the systematic use of a penalizing profile is very pessimistic (the value of the keff can raise of 12% in Δk) [2, 7] and it seems difficult to ensure that the profile will always be penalizing for every possible reactor management,

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[1] For no cooling time, the amount of ^{239}Pu has to be increased with the amount of ^{239}U and ^{239}Np.

- for a given range of burnup, the measured profiles are between two curves $P_{mini}(z)$ and $P_{maxi}(z)$ ($P(z) = \int_H BU(z)\,dz / H$).

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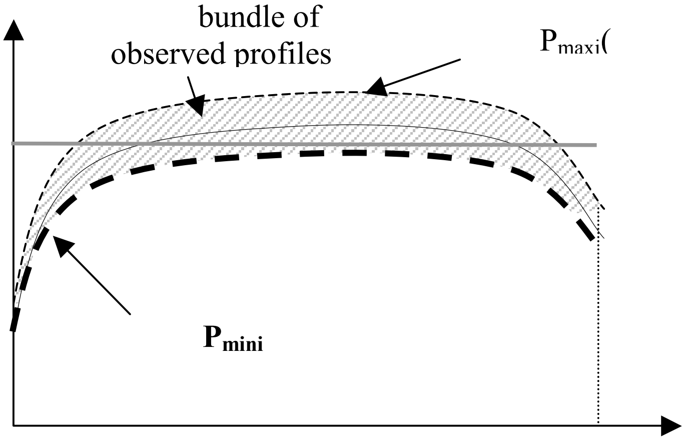
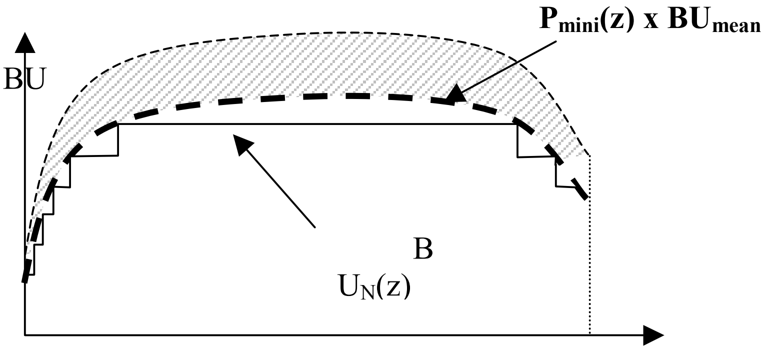


The Working Group is currently studying, for a given mean burnup BU_{mean}, whether a profile $P_{mini}(z)$ should be used in the studies: the burnup at the altitude z will then be equal to:

$BU_{study}(z) = P_{mini}(z) \times BU_{mean}$.

Then the number N of zones in the calculations will be determined using a Burnup curve $BU_N(z)$ where: For $z \in [z_I, z_{I+1}]$ $BU_N(z) = \mathrm{Min}_{z \in [ZI, ZI+1]} (P_{mini}(z) \times BU_{mean})$.



The number of zones (N is around 12) will be determined to give a value of the keff which is not too conservative (the keff will decrease as N increases) but with a number of zones which is not too high.

Criticality Calculations

The criticality calculations will be achieved with CRISTAL V.1 package [8]. This package will determine the keff with a Monte Carlo Method; two codes are available in CRISTAL: MORET 4 [9] (using cross sections from 172 energy-groups libraries, calculated with APOLLO 2 [10]) or TRIPOLI 4 [11]] (point wise cross sections).

The specificities of the calculations corresponding to the new method are (i) the use of some fission products (ii) the loosely coupled units (the top and the bottom of the irradiated assemblies are the most reactive zones).

Qualification of the Fission Products Cross Sections

The qualification of the criticality calculations with the 6 chosen fission products relies on two types of experimental data which come from:

- the MINERVE (Cadarache/France) experiments which validate the cross section of every individual fission product for 3 different types of spectrum,
- the APPARATUS B (Valduc/France) experiments, which test the capacity of the codes to calculate some critical situations representative of dissolution and storage.

The experiments performed at MINERVE [12] showed that the calculated cross sections were in good agreement with the measured ones or that the cross sections of thermal absorbers tended to be underestimated (which is satisfactory in the point of view of safety), except for the ^{133}Cs (the calculated value is up to 8.5% greater than the measured one) and for the ^{103}Rh (the calculated value is up to 11% greater than the measured one); the uncertainties (1ρ) of the measurements are respectively equal to 3.5% and 4% .

The program in Valduc is still on going. The first results concerning ^{149}Sm have been published [13]. The experiments have been described [14] and the calculations give little over-estimations of 0.1% in Δk for the critical configurations (over-estimations are satisfactory in terms of safety).

The completion of the Valduc program will make it possible to determine whether some correction factors have to be considered for the cross-sections (for that purpose the corrections factors will be applied to the concentrations in the criticality studies) of the fission products (possibly, in order to take into account an over-estimation of the absorption of a given fission product) or, eventually, if a margin must be added to reduce the value of the keff limit in the criticality studies.

Loosely coupled units

In the case of storage or transport of irradiated fuel, the level of the reactivity is mainly due to the edges of the assemblies, which are the least irradiated areas.

It raises the problem of the calculations of loosely coupled units with Monte Carlo codes; a special Working Group has been created at the OECD to study this problem. Meanwhile different statistical methods have been implanted in the CRISTAL Package such as (i) the super history powering, (ii) the fission matrix (kij matrix) method, (iii) the stratified sampling; those methods are being studied [15] to solve this particular type of problem.

Until the completion of the studies in that field, the R&D studies should be achieved with the following prescriptions:

- at the beginning of the calculations a certain amount of neutron sources has to be distributed in all of the different fissile units (the sources can be shared so that every fissile unit will include the same amount of starters at the first stage of the calculations),
- at the beginning of the calculations, some additional neutron sources will be placed at the top and the bottom of the fissile assemblies.

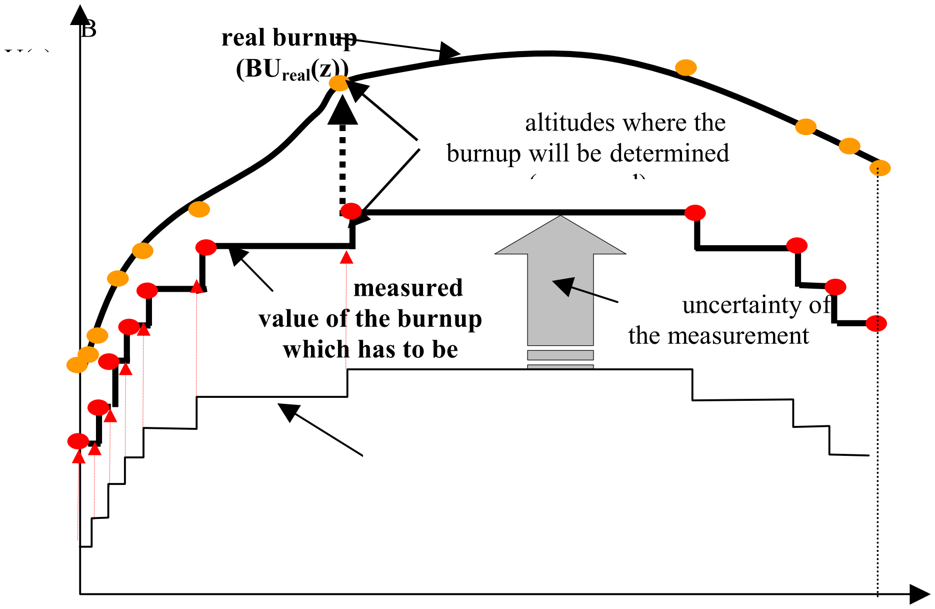
Needs for Measures

As far as a very conservative method was used in the criticality studies to determine the burnup profile (the value of the burnup was set constant on the whole length of the assembly and equal to the mean burnup of the 50-least-irradiated-centimeters) and as an additional safety margin existed due to the disregarded fission products, the measurements were realized to confirm the value of the mean burnup on the 50 least-irradiated-centimeters (for that purpose a measurement of the mean burnup and a measurement of the profile were realized).

For some special configurations where only 3 200 MWd/t on the 50 least-irradiated-centimeters were needed, the only measurement required was a YES/NO measurement which determined whether the fuel was irradiated or not; if it was irradiated, it would then have undergone at least 1 cycle of irradiation which corresponds to around 11 GWd/t for the mean burnup (this led at least to a value of 3 200 MWd/t on the 50 least-irradiated-centimeters).

Considering the margins due to the different other hypothesis, this type of measurements were sufficient to guarantee the safety of this actinide-only method. The French Safety Authorities accepted this method.

But, as soon as we consider a profile of burnup in the studies, which should be quite similar to the real profile, it seems to be necessary to determine precisely the uncertainty linked to the known (measured) value of the irradiated assemblies.



For that purpose, the Working Group will define the possible causes of uncertainties due to:

- the devices used for the measurements,
- the dependence on the conditions of irradiation for the isotopic concentrations of the measured isotopes.

It is important to insist again that the knowledge of the burnup is related to the isotopes that will be measured (e. g. ^{145}Nd, ^{146}Nd or ^{235}U, etc.). Moreover, it will be necessary to define for the whole method an "indicator of the burnup". Thus, the measured value of the burnup will be related to that indicator (and its uncertainty will be linked to the changes of the concentrations of that indicator with the conditions of irradiation), but also the qualification of the depletion codes will have to be done with the same indicator of burnup.

However, it can be pointed out that, an accurate knowledge of the uncertainties is only necessary if the amount of burnup needed to demonstrate the safety of a system is very close to the real burnup of the fuel: this is not actually the case for the facilities where the fission products should be taken into account (e.g. La Hague interim storage).

PRELIMINARY RESULTS

The comparisons between the keff values obtained with the actinide-only method and with the new method will be given in this part. The values of k and Δk, which are given below, correspond to different configurations of storage (e. g. interim storage at La Hague) and transportation (e. g. TN17-2).

For those calculations 17 axial zones were used. Some pessimistic correction factors were used for the fission products and for the actinides to take into account, (i) the discrepancies observed between the results of the depletion calculations and the experimental data, (ii) the differences in the cross sections (condensate to 1 group of energy, for a given spectrum) noticed during the MINERVE program. Moreover, the pessimistic assumptions, regarding the conditions of irradiation described in paragraph 2.1.1., were taken into account (as all of the studies have not been finished yet, the correction factors which have been used may still be very pessimistic).

Preliminary results (actual qualification of the tools): k_{eff} ρ<0.1%

No cooling time	fresh fuel ei=4.5%	11 GWd/t penalizing profile	11 GWd/t standard profile	22 GWd/t standard profile	33 GWd/t standard profile	44 GWd/t standard profile
Assembly surrounded by 20 cm of water	0.9339	0.9005(1)	0.8799(1)	0.8330(1)	0.7926(1)	0.7511(1)
		0.8995(2)	0.8851(2)	0.8476(2)	0.8172(2)	0.7861(2)
		0.9197	0.9004	0.8661	0.8335	0.8011
Storage (La Hague)	0.9366	0.8984(1)	0.8822(1)	0.8339(1)	0.7912(1)	0.7498(1)
		0.9028(2)	0.8894(2)	0.8487(2)	0.8192(2)	0.7846(2)
		0.9247	0.9005	0.8684	0.8345	0.7980
Storage (La Hague) - accident	0.9825	0.9490(1)	0.9271(1)	0.8749(1)	0.8312(1)	0.7865(1)
		0.9499(2)	0.9321(2)	0.8905(2)	0.8603(2)	0.8253(2)
		0.9680	0.9430	0.9064	0.8715	0.8375
TN 17-2 cask (7 assemblies)	1.0412	1.0015(1)	0.9827(1)	0.9264(1)	0.8757(1)	0.8282(1)
		1.0050(2)	0.9921(2)	0.9458(2)	0.9084(2)	0.8755(2)
		1.0308	1.0033	0.9648	0.9267	0.8878
Δk*		1.8 % to 2.6%	1.1% to 1.5%	1.6% to 2 %	1.1% to 1.8%	1.2% to 1.5%

[1] values obtained with the control rods not inserted during irradiation.

[2] values obtained with the control rods inserted during irradiation.

values in italics correspond to the current method (actinide only and "50-least-irradiated centimeters" assumptions - "standard" irradiation history - control rods are not inserted).

* Δk = k(current method) - k(new method with control rods inserted)

Finally, we can notice that, even if the correction factors are pessimistic (they have been deduced with a code version which is not the one that will be implemented in the CRISTAL package - they should therefore be reduced) and the irradiation history is very conservative, the new method already gives a gain up to Δk = 2.6 %.

CONCLUSIONS

The French Working Group for Burnup Credit has to analyze different propositions for a new method to consider (i) the neutron absorption of some fission products (ii) an axial profile of the burnup.

In order to define a conservative method many studies have been performed related to:

- the depletion calculations; the Group proposes to consider some pessimistic conditions of irradiation plus some correction factors corresponding to the qualification of the depletion codes,
- the definition of an axial profile; it has been decided to study the use of a conservative profile, determined from the bundle of profiles observed at La Hague reprocessing plant and then to verify by a measurement that, for each assembly, the measured burnup is above the one used in the criticality study,
- the calculation tools used to determine the value of the k_{eff}; those studies are mainly based on experimental results obtained for the qualification of the cross sections of the fission products.

Finally, the preliminary calculations performed with the proposed recommendations and a standard profile (initially enriched at 4.5% and irradiated up to 44 GWd/t) gives a gain in reactivity up to Δk = 2.6%, compared to the current method.

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APPENDIX

Source of experimental measurements for validation of depletion codes

EXPERIMENTS REACTOR	FUEL	Enrichment	Burn up	Experimental analysis
La Hague	PWR 17x17 UOx 1350 sub assemblies	1.8% to 3.5%	11 GWd/t to 44.5 GWd/t	HN, Nd
Tihange 1	PWR 15x15 UOx	3.10%	10 to 40 GWd/t	HN, MA, Nd
Bugey 3	PWR 17x17 UOx	2.10% 3.10%	19 to 38 GWd/t	HN, MA, Nd, Cs, 15 FP BUC.
Fessenheim 2	PWR 17x17 UOx	2.60%	27 to 30 GWd/t	HN, AM, Nd
Fessenheim 2	PWR 17x17 UOx	3.10%	45 to 60 GWd/t	HN, MA, Nd, Cs
Gravelines 3+2	PWR 17x17 UOx	4.50%	25 to 62 GWd/t	HN, MA, Nd, Cs, 15 FP BUC
Cruas 4 / URE	PWR 17x17 Enriched Uranium from reprocessing	3.61% U6 /U8 = 1.2%	11 to 34 GWd/t	HN, MA, Nd
Cruas 2 / HTC	PWR 17x17 UOX	4.50%	55 to 65 GWd/t	HN, MA, Nd, Cs, 15 FP BUC
Saint Laurent B1	PWR 17x17 MOx 3 initial Pu contents	Average $(\frac{Pu}{U+Pu}) \cong 4.5\%$	10 to 45 GWd/t	HN, MA, Nd, Cs, 15 FP BUC
Gravelines 4	PWR 17x17 MOx	Average $(\frac{Pu}{U+Pu}) \cong 4.5\%$	40 to 50 GWd/t	HN, Nd
Dampierre 2 / HTC	PWR 17x17 MOx	Average $(\frac{Pu}{U+Pu}) \cong 5.3\%$	10 to 65 GWdj/t	HN, MA, Nd, Cs, 15 FP BUC

HN = Heavy nuclides (U, Pu); **MA** =minor actinides (Np, Am, Cm); **FP BUC** = Fission products burn up credit

☐ Available results

☐ Experiments in progress

Utilization of burnup credit in Slovakia

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Abstract. Nuclear Regulatory Authority of the Slovak Republic (ÚJD SR), as a central body of the state administration, performs the state supervision over nuclear safety of nuclear installations in accordance with Act No. 130/1998 Coll. I. on peaceful use of nuclear energy. Based on its commission, Department of Nuclear Materials of ÚJD SR supervises over nuclear safety in facilities for storage and transportation of nuclear spent fuel. In accordance with above mentioned Act and relevant Regulations ÚJD SR performs the state supervision during designing, construction, commissioning, operation, and decommissioning of nuclear installations. Basic engineering of the dry interim spent fuel storage facility (ISFSF) construction has begun at Mochovce nuclear power plant (NPP) in 2001. The casks for the ISFSF will be designed taking into account the BUC. Department of Nuclear Materials is also a guarantee for research and development project called "Utilization of Burnup Credit (BUC) in the Criticality Calculation of the WWER-440 Fuel Assemblies".

1. SURVEY OF THE FACILITIES FOR SPENT FUEL STORAGE AND TRANSPORTATION

At present there are 6 nuclear power units of WWER type in operation in the Slovak Republic. Out of this number, 2 are the V 230 model, and 4 are the V 213 model. 4 units are located on the Jaslovské Bohunice site, 2 units are located on the Mochovce site. A wet ISFSF is also located on the Jaslovské Bohunice site. The C-30 transport container is used for the spent fuel transport.

1.1. Interim spent fuel storage facility Jaslovské Bohunice

The Jaslovské Bohunice ISFSF was commissioned in 1988. During 1997-2000, the ISFSF was subject to a reconstruction and seismic upgrade. The original capacity was increased from 5040 to 14112 spent fuel assemblies by replacement of the T-12 casks with the capacity of 30 fuel elements by the compact ones KZ48 with the capacity of 48 fuel elements.

The criticality calculation was executed using the SCALE 4.3 code. The calculation results are shown in Table I. for the store filled with fresh fuel enriched to 4.4 % U235, and absorption assemblies' pitch of 168 mm with 1.05 % of boron.

Table. I. K_{EFF} for the compact cask KZ48 [6]

	Infinite rack	1 cask
k_{eff}	0.910	0.849

1.2. At-reactor pools

The at-reactor pools are used for temporary storage of the spent fuel. The subcriticality is ensured by the storage grate pitch of 225 mm and presence of the boric acid in the coolant at the V-1 and V-2 NPPs [2] and [1].

In Mochovce NPP, the at-reactor pools were made more compact, and the lower grate capacity is almost double compared to the V-1 or V-2 NPP. The main grate pitch is 162 mm, the back-up grate pitch is 225 mm, original fuel enrichment is 3.82 % U235, average burnup is 43 MWd/kg U, maximum burnup is 45.1 MWd/kg U. The criticality calculation was performed using the Russian code MCU using the Monte-Carlo method. The Table II. presents the calculated values of k_{eff} for the fresh fuel with the enrichment of 3,82 % U235 and boron free coolant.

Table. II. K_{EFF} for the AT-Reactor pools at Mochovce NPP

	Main grate	Back-up grate
k_{eff}	0.948	0.900

1.3. Transport container C-30

After at least 2.5 years of storage in the at-reactor pools, the spent fuel is removed to the C-30 transport container. The assemblies can be put into the C-30 container either in the compact cask KZ48 (see 1.1.) or in the T-12 cask with the grate pitch of 225 mm. The criticality calculation was executed using the SCALE 4.3 code for the transport using the T-12 cask. The Table III. presents the values of k_{eff} for the fresh fuel enriched to 4.4 % and boron free coolant.

Table. III. K_{EFF} for the C-30 transport container with T-12 cask

	Infinite rack	1 cask
k_{eff}	0.970	0.904

According to [4] the subcriticality condition (k_{eff}<0.95) is met for the infinite rack for the fuel enrichment less than 3.9 % U235.

2. BUC APPLICATION IN THE CRITICALITY CALCULATION OF THE WWER-440 FUEL ASSEMBLIES

ÚJD SR warrants various research tasks under the R&D program. The Nuclear Materials Division prepared a task of the BUC application in the criticality calculation of the WWER-440 fuel assemblies in cooperation with Nuclear Power Plants Research Institute (VÚJE). VÚJE will perform this task in 2003 through 2005. The following subtasks will be addressed under this research task:

2.1. Verification of SCALE 5.0 calculation system

The aim is to verify applicability of the latest version of the SCALE 5.0 calculation system to the WWER-440 spent fuel storage and transportation facilities. It will consist of the SCALE 5.0 system testing during the calculations of criticality nuclide composition and residual heat of the WWER-440 fuel and verification of the system applicability by means of the results comparisons with the ones of the numerical models.

2.2. Methodology of the Burnup Credit for the WWER-440 fuel

The aim is to develop appropriate methodology of the BUC application for the WWER-440 fuel. It will consist of the proposal of the calculation analyses range in order to ensure sufficient subcriticality during the WWER-440 spent fuel storage and transportation.

2.3. Application of the Burnup Credit for the dry storage conditions of the WWER-440 fuel

This task should demonstrate, that when the burnup consequences are partially taken into account, it significantly decreases requirements on the WWER-440 spent fuel storage under dry conditions. The task puts an emphasis on the BUC analysis for the dry storage of the WWER-440 spent fuel. The results will serve for validation of the basic parameters of the Mochovce dry store.

2.4. Application of the Burnup Credit for the wet storage conditions of the WWER-440 fuel

The aim is to examine possibilities of the WWER-440 spent fuel storage and transportation with higher original enrichment in the existing storage and transportation facilities. It will consist of the analysis of the possibility to store and transport the WWER-440 spent fuel with original enrichment up to 5% U235 in the existing C30 containers with T-12 or KZ48 casks and in the at-reactor pools.

3. DRY INTERIM SPENT FUEL STORAGE FACILITY AT MOCHOVCE

In 2001, Slovak Electric joint-stock company decided on the dry ISFSF construction on Mochovce site. This ISFSF will accommodate the spent fuel from the Mochovce NPP. According to [9], it is assumed in the project that the spent fuel from 40 years of operation of two Mochovce NPP reactor units will be stored in it. It represents 6552 fuel assemblies in total. It is assumed that the majority of the fuel assemblies will have the original enrichment higher than the highest currently used fuel assemblies, i.e. 3.82 % U235 (profiled fuel).

According to the basic engineering [9], 78 casks in total, each with capacity of 84 assemblies, will be required to store the spent fuel. The BUC will be used in the criticality calculation.

The ISFSF shall be commissioned in 2009 when the at-reactor pools will be full.

CONCLUSIONS

The BUC has not been used for the spent fuel storage and transportation criticality calculation up to now in the Slovak Republic. Up to now, ÚJD SR has requested the criticality calculations for the fresh fuel in the licensing process of such facilities.

The most recent information obtained mainly thanks to the cooperation with IAEA convinced us that the application of the BUC for the spent fuel storage and transportation is not only possible but also even desirable in the Slovak Republic.

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Criticality analysis with burnup credit for APR1400 in the Republic of Korea

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Abstract. Criticality analysis for the spent fuel pool storage racks of APR1400, Advanced Power Reactor with 1400 MWe capacity for the next generation reactors in Korea, has been performed with the introductions of burnup credit and neutron-absorbing material to maximize spent fuel storage capacity in a given area. The main purpose of this analysis is to determine the minimum burnup necessary for a fuel assembly with a specific initial enrichment and minimum cooling time to be placed in the spent fuel storage rack. In this analysis, the SCALE 4.4 code system has been used. Also, the biases and uncertainties evaluated through validation calculations for both isotopic predictions and criticality calculation have been taken into account in the criticality evaluation. All of the key reactor operating parameters, such as moderator density, soluble boron level, fuel temperature, specific power, and operating history have been selected in a conservative way for the criticality analysis. The analysis results showed that the spent fuel storage rack of APR1400 is designed to achieve the high storage density by using burnup credit concept and poison material.

1. INTRODUCTION

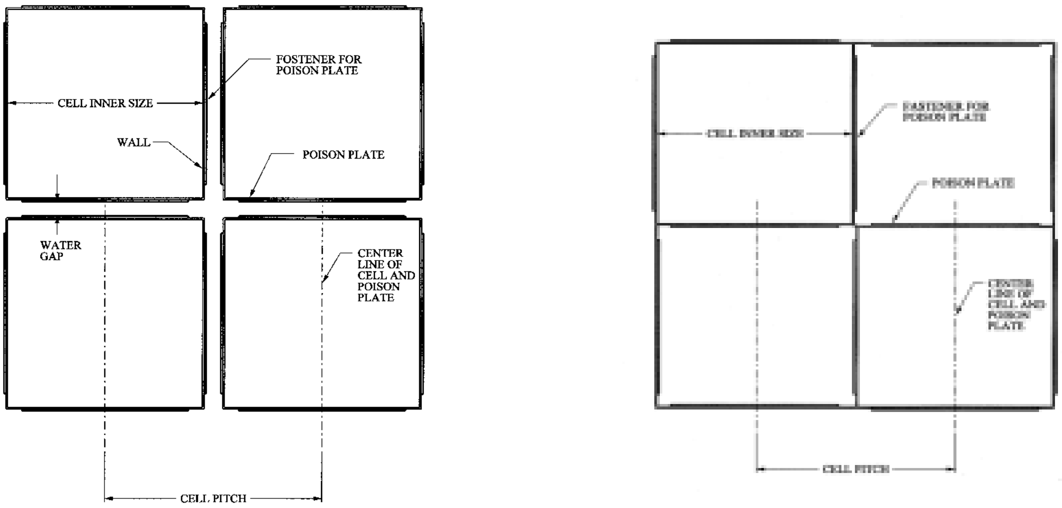
The traditional assumption used in criticality analysis for the spent fuel pool is that the spent fuel has the same reactivity as the fresh fuel. This assumption ignores the decrease in reactivity as a result of irradiation and results in significant conservatism in the calculated reactivity values. Improved calculational methods allow one to take credit for the reactivity reduction associated with fuel burnup, hence reducing the analysis conservatism while maintaining an adequate safety margin.

A criticality safety analysis has been performed for the high-density spent fuel storage rack of the Advanced Power Reactor 1400 (APR1400), while maintaining the criticality safety margins and related requirements. For this work, the SCALE 4.4 code system[1] has been used through the following process. The process begins with the validation of the calculational method and the calculated bias and uncertainty through the validation analysis are used to establish the subcritical safety limit to be applied to the spent fuel rack design. For the validation calculation, the criticality safety analysis sequences (CSAS1X and CSAS25) have been used. The sensitivity analysis was also performed for various operating parameters to ascertain the sensitivity of k_{eff}. After the sensitivity analysis, the criticality analyses have been performed via CSAS25 and the shielding analysis sequence (SAS2H) considering the parameters conservatively. Finally, the minimum burnup required for a fuel assembly with a specific initial enrichment and minimum cooling time has been decided for the high-density spent fuel storage rack.

Section II briefly describes the spent fuel storage rack facility. In Section III, the calculational tools, validation of calculational method and sensitivity analysis for design parameters are provided. The calculational results are provided in Section IV and the conclusion in Section V.

2. FACILITY DESCRIPTION

The spent fuel pool storage racks of APR1400 are designed of twelve 11x12 individual modules containing 132 storage cells each with poison material for high-density storage. The storage racks are stainless steel honeycomb structures with rectangular fuel storage cells and poison (Boral) plates attached to their outside surface. Boral as a neutron absorbing material is mechanically attached to the cells such that the fuel assemblies are fully surrendered by the neutron absorbing material. The neutron absorbing material extends the full height of the active fuel. The spent fuel storage rack is divided into two regions. Region I provides core off-load capability for approximately 396 spent fuel assemblies (equivalent to more than 1.64 cores, 241 assemblies per core). This is achieved with a half storage capacity in a checkerboard array marked in alternating fuel storage cells. The remaining half of Region I is provided for the fully burned fuel assemblies. Region II provides 100% density storage for up to 792 spent fuel assemblies. The spent fuel storage cells are installed in rows with center spacing of 9.8 inches for Region I and 8.925 inches for Region II. The geometry of the storage cell is shown in Figure 1. Both Region I and II are designed to accommodate fuel assemblies with initial enrichment of up to 5.0 wt% U-235. For the fuel assemblies in Region II, there is a burnup limit dependent on the initial enrichment for each fuel assembly. The maximum total spent fuel storage capacity is approximately 1584 storage cell locations (storage capacity for ~ 20 years), which represent approximately 656% of a full core.


FOSTENER FOR POISON PLATE
CELL INNER SIZE
WALL
POISON PLATE
WATER GAP
CENTER LINE OF CELL AND POISON PLATE
CELL PITCH
CELL INNER SIZE
FASTENER FOR POISON PLATE
POISON PLATE
CENTER LINE OF CELL AND POISON PLATE
CELL PITCH

Region I Region II

FIG. 1. The Geometry of the Spent Fuel Storage Cell

3. METHOD OF CRITICALITY SAFETY ANALYSIS

3.1. Calculational Tools

The computational tools used in this analysis are the shielding analysis sequence SAS2H and the criticality safety analysis sequence CSAS from SCALE 4.4 computer code system. The base cross-section library used in this study is the SCALE 44-neutron-group ENDF/B-V library (44GROUPNDF5).

3.2. Validation of Calculational Methods

3.2.1. Selected Nuclides for criticality analysis

To make the analysis practical, a subset of important nuclides that can adequately represent the behavior of spent fuel in criticality analysis models is selected. Isotopes to represent the spent fuel composition are selected under a conservative consideration. A set of 12 actinide isotopes, 15 fission products, and O-16 is considered for this work as shown in Table I.

Table I. Set of isotopes used in criticality calculations

Actinides	Fission Products	Other
U-234, U-235, U-236, U-238, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am- 241, Am-243	Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Nd-143, Nd-145, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Eu-153, Gd-155	O-16

3.2.2. Isotopic Prediction Validation

For the validation of predicting isotopic contents, a deterministic CSAS1X (XSDRNPM) [1] calculation has been performed to evaluate the difference in calculated k_∞ based on the calculated and the measured isotopic values. The 38 spent fuel samples from ORNL documents [2,3] have been used for this calculation. Evaluated bias and 95/95 uncertainties for fuel depletion calculation are 0.00073 Δk and 0.02689 Δk, respectively.

3.2.3. Criticality Calculation Validation

A set of 95 critical experiments (48 UO_2 and 47 MOX experiments) data from reference [4] through reference [12] has been benchmarked using CSAS module with 44GROUPNDF5 cross section library to demonstrate its applicability to criticality analysis and to establish the bias and uncertainty (2σ). Calculated results are shown in Table II.

Table II. The results of the validation calculation

Experiment	Average k_{eff}	Bias(average k_{eff} - 1)	Deviation(σ)
UO_2	0.99992	-7.92E-05	0.003751
MOX	1.00029	0.000292	0.004422
TOTAL	1.00010	0.000104	0.004101

3.2.4. Sensitivity Analysis for Various Operating Parameters

Sensitivity analyses have been performed to determine the input parameters that have a significant effect on the depletion and criticality analyses. The considered parameters are initial enrichment (1.7, 3.0, 4.6 wt%), discharge burnup (18, 36, 54 GWD/MTU), core moderator temperature (T_{avg}, T_{hot}), core specific power level (P_{avg}, P_{peak} MWd/MTU), boron concentration in the coolant (C_{avg}, C_{max}), cooling time (0, 1, 5 years), and axial burnup profile of spent fuel (actual, flat). The following effects and trends for operating parameters and axial burnup distribution are observed.

The higher moderator temperature in depletion calculation yields the higher reactivity of spent fuel assembly for all other operating conditions such as boron concentration, specific power level, cooling time, initial enrichment and discharge burnup. There is no significant difference in the calculated k_{eff} at different power levels when considered no cooling time. However, for 1- and 5- year cooling time, the lower power yields slightly higher k_{eff} than the higher one. The higher boron concentration in depletion calculation yields the higher reactivity of spent fuel assembly for all operating conditions. The longer cooling time after discharge of fuel yields the lower reactivity of spent fuel assembly.

To evaluate the effect of axial burnup profile of spent fuel, a flat burnup and an actual axial burnup distribution have been considered. The top half of fuel rods has been only modeled for axial burnup distribution. The results show different trends with the initial enrichment, discharge burnup, and cooling time after discharge of fuel and they are summarized as:

(a) For high initial enrichment and high burnup, a flat burnup distribution causes lower k_{eff} than actual burnup distribution and this trend is more severe at a long cooling time.
(b) For relatively low initial enrichment and high burnup, a flat burnup distribution yields lower k_{eff} than actual burnup distribution and this trend is also magnified with the long cooling time. However, the degree of difference is less than the high enrichment and high burnup case.
(c) For high initial enrichment and relatively low burnup, a flat burnup distribution yields higher k_{eff} than an actual burnup distribution at no cooling time and 1-year cooling time. However, a flat burnup distribution gives lower k_{eff} than an actual burnup distribution at 5-year cooling time.

In all 5-year cooling time cases, a flat burnup distribution gives lower k_{eff} than an actual burnup distribution.

For APR1400 design, typical expected burnup and initial enrichment of spent fuel to be stored are 36 GWD/MTU and 4.31 wt% with no cooling time. For this case, the criticality calculation with a flat burnup distribution is more conservative than an actual axial burnup distribution. Thus, a flat burnup profile is assumed in the criticality analysis of spent fuel storage rack for APR1400.

4. CRITICALITY ANALYSIS FOR THE SPENT FUEL STORAGE RACK

4.1. Criticality Calculation

The CSAS25 module and 44GROUPNDF5 library of SCALE 4.4 have been used to calculate the k_{eff} in the spent fuel storage rack Region I and II under normal conditions. The k_{eff} is calculated for the checkerboard array of 5.0 wt% fresh fuel and depleted fuel assemblies in Region I and for infinite array of depleted fuel in Region II. Following operating parameters are used in this case study.

(a) Initial fuel enrichment: 2.0, 2.5, 3.0, 3.5, 4.0, 4.5, 5.0 wt%
(b) Fuel burnup: 18, 36, 54 GWD/MTU
(c) Pellet stack density: 10.114 g/cm^3 (92.28% of theoretical density)
(d) Moderator density: 1.0 g/cm^3 (maximum density assumed)
(e) Boral data: 0.075" thickness, 7.5" width, 150" height (active fuel height)
(f) Rack wall and Fastener thickness: 0.075", 0.025"

4.2. Bias and Uncertainty Evaluation

For the evaluation of uncertainties on the criticality analysis, the following parameters have been considered:

(a) Bias and uncertainty for burnup calculation
(b) Bias and uncertainty for criticality calculation
(c) Individual calculational uncertainty of Monte Carlo method

(d) Tolerance and uncertainty due to the deviation from nominal condition (temperature, fuel rack pitch, rack steel thickness, most reactive position of fuel assembly in cells, etc.)

The total uncertainty has been evaluated as 0.02714 Δk for Region I and 0.03783 Δk for Region II, respectively for APR1400 spent fuel rack design.

4.3. Calculational Results

CSAS25 has been run for various combinations of initial fuel enrichments and burnups to determine k_{eff} as a function of burnup for the given initial fuel enrichments. Figure 2 shows the minimum allowable burnup for a fuel assembly, which could be stored in the spent fuel storage rack. The minimum allowable burnup is the value satisfying the criticality safety limit of 0.95 k_{eff}. Only spent fuel assemblies with minimum burnup and initial enrichment above the curve are qualified for loading into the spent fuel storage rack. The minimum allowable burnups for the maximum allowable initial enrichment (5.0 wt% U-235) are approximately 39 GWD/MTU for Region I and 42 GWD/MTU for Region II, respectively.

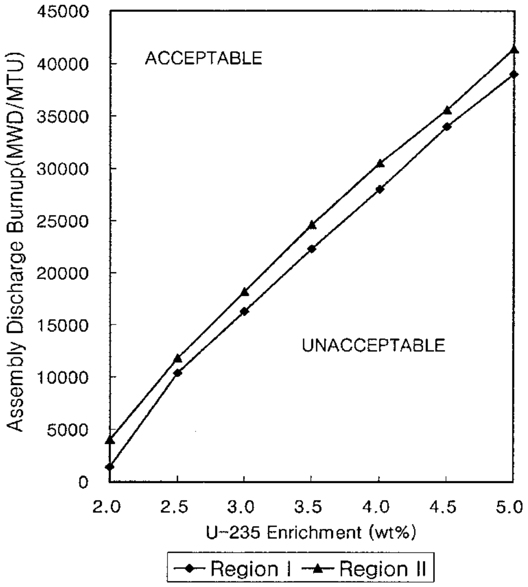


FIG. 2. Minimum Allowable Burnup for Region I and II

5. CONCLUSION

The criticality analysis for the spent fuel storage rack of APR1400 has been performed with burnup credit. The isotopic prediction and criticality analysis method has been validated, and the corresponding biases and uncertainties have been evaluated and reflected in the final calculational results. Also, all of the key reactor operating parameters have been conservatively selected for the criticality analysis. Therefore, it is concluded that the spent fuel storage rack of APR1400 is designed to achieve the high storage density by using burnup credit concept and poison material.

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Possibility for BUC implementation in RBMK-1500 fuel dual purpose storage casks

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Abstract. The paper presents dummy calculations of effective neutron multiplication factor for dual purpose CONSTOR RBMK-1500 cask. This cask is designed for interim dry storage (and transportation) of 102 spent nuclear fuel half-assemblies. It was assumed in the calculations that internal basket tubes are consolidated. In this case 156 half-assemblies could be placed in the cask. K_{eff} was calculated for different fuel initial enrichments and for various cases when: a) only actinides; b) actinides plus fission products (FP) are taken into account. Obtained results have showed that if burnup credit is taken into account even if basket tubes are consolidated, k_{eff} values are less then allowable value 0.95.

1. INTRODUCTION

After final shutdowns of the Ignalina NPP Unit 1 and Unit 2 total amount of spent nuclear fuel (SNF) will be about 20 thousands of fuel assemblies (FA). All these assemblies should be stored about 50-100 years and after that disposed of. At present unloaded FA from the reactor are stored in the water pools at least for 5 years. In 1993 it was decided to use dry storage technology for interim storage of SNF at the INPP. GNB dual-purpose storage casks have been chosen. Part of them is ductile cast iron CASTOR RBMK-1500 casks and the remaining ones are metal-concrete CONSTOR RBMK-1500 casks. Both type casks are designed to load 102 half-assemblies, which are placed in the basket of special configuration. Capacity of existing SNF dry storage facility at the Ignalina NPP is 72 casks, but totally about 400 casks will be needed. So expansion of existing SNF dry storage facility or construction of new facility is under consideration.

For the meantime there are no specifications for the burnup credit implementation (what actinides and fission products must be taken into account, what burnup profile, etc.) in the criticality safety analysis of any storage system (water pool or storage cask) in Lithuania. It is evident that implementation of burnup credit would have significant economical effect. For example, if consolidation of basket tubes is accepted, then about 250 storage casks will be enough. So the possibility of the implementation of burnup credit in SNF storage or disposal must be under consideration.

2. RBMK-1500 FUEL ASSEMBLY

The RBMK-1500 fuel consists of sintered, cylindrical UO_2 pellets with a density between 10.4 and 10.5 g/cm^3. The pellet shape is adapted to an intensive, high-temperature operating mode, so they have spherical indentation, in order to reduce the thermal expansion of fuel column and thermo-mechanical interaction with the cladding. The pallets are placed into a clad tube with an outside diameter 13.6 mm and a wall thickness 0.9 mm. Two screen pellets (enrichment 0.4%) are placed in each clad tube in the area of the plug. Cladding material is an alloy of zirconium (99 wt-%) and niobium (1 wt-%). This alloy has good anti-corrosive properties and a low neutron absorption coefficient. The initial gap between UO_2 pellets and cladding varies from 0.22 to 0.38 mm. The tubes are pressurized with helium and sealed. A schematic representation of the fuel rod is shown in Fig. 1.

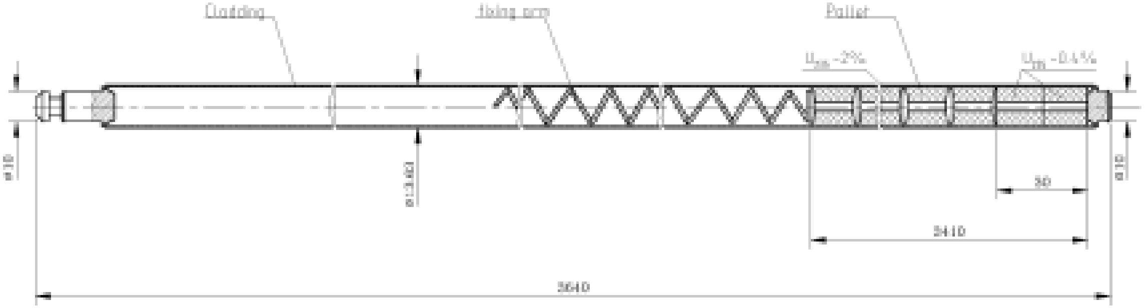

FIG.1. RBMK-1500 fuel rod

A fuel assembly consists of 18 fuel rods arranged in two concentric circles around a central carrier rod (Fig. 2). In the inner circle, with a diameter of 3.2 cm, there are 6 equally spaced rods. In the outer circle with a diameter of 6.2 cm, there are 12 equally spaced rods. The fuel assembly is made up of two segments (half-assemblies) that are joined axially with a total length of almost 7 m active fuel.

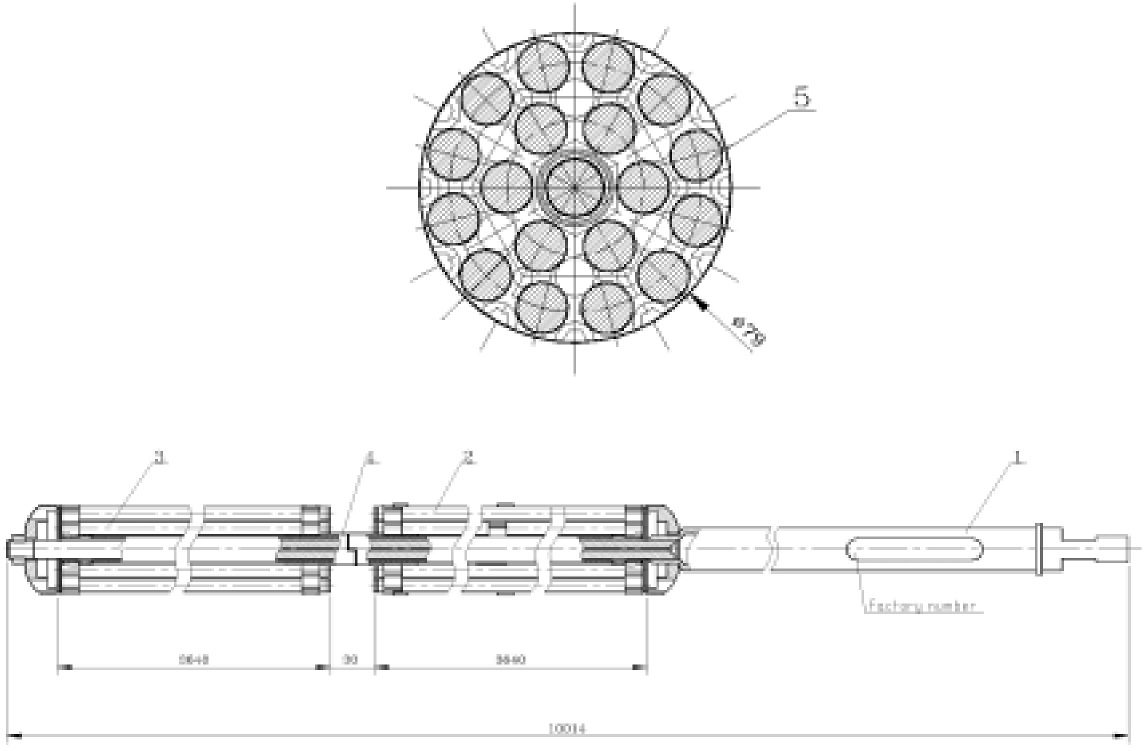

FIG.2. RBMK-1500 fuel assembly (1-suspension bracket, 2-upper half-assembly; 3-bottom half-assembly; 4-carrier rod; 5-fuel rod)

Reactor RBMK-1500 was design for nuclear fuel with U-235 2% enrichment from the outset. Ignalina NPP used fuel only with 2% enrichment till June 1995. Implementation of the new uranium-erbium fuel at Ignalina NPP has been started in 1995. Improvement of reactor nuclear safety is the aim of the reactor conversion to the uranium-erbium fuel. Loading of such fuel into Unit 1 has been started in December 1996. In May 1999, there were 759 FA with uranium-erbium fuel in the core. Average fuel burnup increased by 23% and was 1038 MWd/FA.

At present, at the Ignalina NPP the process of a wider use of the uranium-erbium fuel is going on. The loading process of the uranium-erbium FA into the reactor cores is accompanied by experiments and analytical investigation of the behaviour of the main neutron-physical characteristics of the reactors. Average burnup of the uranium-erbium fuel increased by 42-46% in comparison with regular RBMK fuel. Calculations have shown that augmentation of burnup can reach 50%. Fair quantity of FA with uranium-erbium fuel is unloaded now and average produced energy is 2200 MWd/FA and more. At the end of 2001 first batch of experimental fuel with 2.6% enrichment and 0.5 percentage of erbium has been loaded into reactor. Investigations have shown that use of such fuel will allow additionally to reduce a steam factor of reactivity and to increase burnup of fuel.

3. CALCULATION METHOD AND ASSUMPTIONS

The SCALE 4.3 computer codes package was used for calculations. In the [1], it was shown that this package is applicable for criticality calculations of systems with the fresh RBMK fuel. There is a lack of experimental data of the determination of isotopic inventory for the irradiated RBMK fuel assembly, so it is quite difficult to validate SCALE package for depletion analysis. However, some results of experimental activity measurements of the samples polluted with Chernobyl NPP nuclear waste after the accident are available. Using SCALE 4.3 computer codes package modeling of isotopic characteristics of Chernobyl NPP nuclear fuel after accident was performed and comparison of calculation results with experimental investigation is presented in [2]. A good agreement between the experimentally measured activity ratio values of plutonium isotopes and fission products and computer calculated values enables to use SCALE codes system for modeling of the spent RBMK nuclear fuel characteristics.

The CONSTOR cask has cylindrical shape with diameter more than 2 meters, height more than 4 meters, total thickness of the sidewall 0.43 meter. The sidewall of the cask consists of two carbon steel cylinders and the heavy concrete layer between them. The bottom consists of the same materials and arrangement as the sidewall. The lid of the cask is made from carbon steel. Radial cross-sections of CONSTOR RBMK-1500 storage cask with the normal and consolidated internal basket are shown in Fig.3.

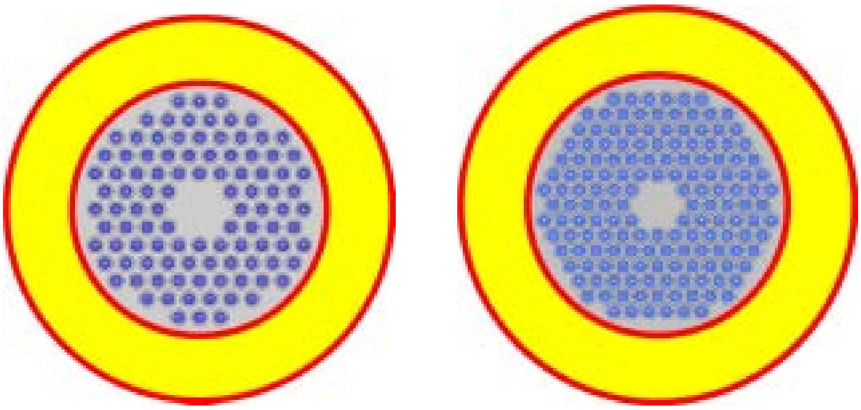

FIG.3. Normal and consolidated internal basket in the CONSTOR storage cask

Sequence CSAS25 from SCALE 4.3 package was used to calculate effective neutron multiplication factor k_{eff} for CONSTOR storage cask. Depletion/decay sequence SAS2H was used to define nuclide inventory of irradiated fuel.

The following isotopes were included in criticality calculations according to [3]:

Actinides: U-234, U-235, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-243, and Np-237;

Fission products: Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Nd-143, Nd-145, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Eu-153, Gd-155, Er-166, and Er-167.

The conditions that were accepted for nuclear fuel depletion/decay calculations are listed below:

Five different types of nuclear fuel: 2.0%, 2.4%, 2.4%+Er, 2.6%, and 2.6%+Er;

Burnup varies from 0 to 25 MWd/kgU;

Spent fuel cooling period varies from 0 to 100 years;

Burnup shape is uniform and doesn't vary axially.

The following conditions and assumptions were accepted for the criticality calculations:

Existing internal basket of the cask contains 102 stainless steel tubes each with fuel half-assembly inside. The steel tube has an outer diameter of 100.2 mm and nominal wall thickness of 2 mm. Nominal pitch (center-to-center distance) of the basket tubes is 124.13 mm;

Consolidated basket tubes in the cask have triangular pitch of 100.2 mm. In this case all basket tubes will be in contact and 156 fuel half-assemblies can be placed;

Discrete representation of the fuel rods is used in calculations. This means that each half-assembly consists of 18 fuel rods;

Inner region of the cask is homogeneously filled with water. A water density inside the cask varies from 0.0 g/cm^3 to 1.0 g/cm^3;

Two sets (for fresh fuel and for irradiated fuel) of k_{eff} calculations for different fuel enrichments and its dependence on burnup and cooling time were done;

There is not neutron leakage from the system. This means that setting reflection condition is fulfilled on outside surface of the cask;

The central basket tube (it is in the center of the cask) is replaced by water, but it is assumed that basket tubes can't appear in this region.

4. RESULTS

The variation of k_{eff} with water density for existing CONSTOR RBMK-1500 cask with the normal internal basket and fresh fuel half-assemblies inside is presented in the Fig.4. In this case neutron multiplication factor k_{eff} reaches maximum (~0.85) when water density is approximately 0.35 g/cm^3. The same k_{eff} variation for the cask with consolidate basket with 156 fresh fuel half-assemblies is presented in the Fig.5. As can be seen permissible k_{eff} value 0.95 is exceeded at certain water density interval for fuel with 2.4% and 2.6% enrichment when burnable absorber erbium is not taken into account.

A set of graphical charts of k_{eff} dependence on the burnup level is presented in the Fig.6 - Fig.9. At first k_{eff} variation with burnup for different fuel enrichments (with and without erbium) and for optimal moderation in the cask is presented in the Fig.6. In this case only actinides are taken into account. K_{eff} has maximum at the certain burnup value when burnable absorber erbium is taken into account. As can be seen difference between k_{eff} values when fuel is with and without erbium is decreasing with increasing of the burnup value.

The effects of the presence of the major fission products on the k_{eff} value for fuel with different enrichments are presented in the Fig.7, Fig.8, and Fig.9. The difference of k_{eff} values when only actinides and actinides plus FP are taken into account is marginal at the low burnup values. This difference increases with the increase of burnup level.

Finally, k_{eff} variation with cooling time is presented in the Fig.10. In all cases there is monotonous decrease of k_{eff} values for cooling time period of 100 years. For longer cooling times, it is believable that k_{eff} will increase in certain time interval due to decay of Am-241 and Pu-240. But this effect is important for disposal, not for interim storage of SNF.

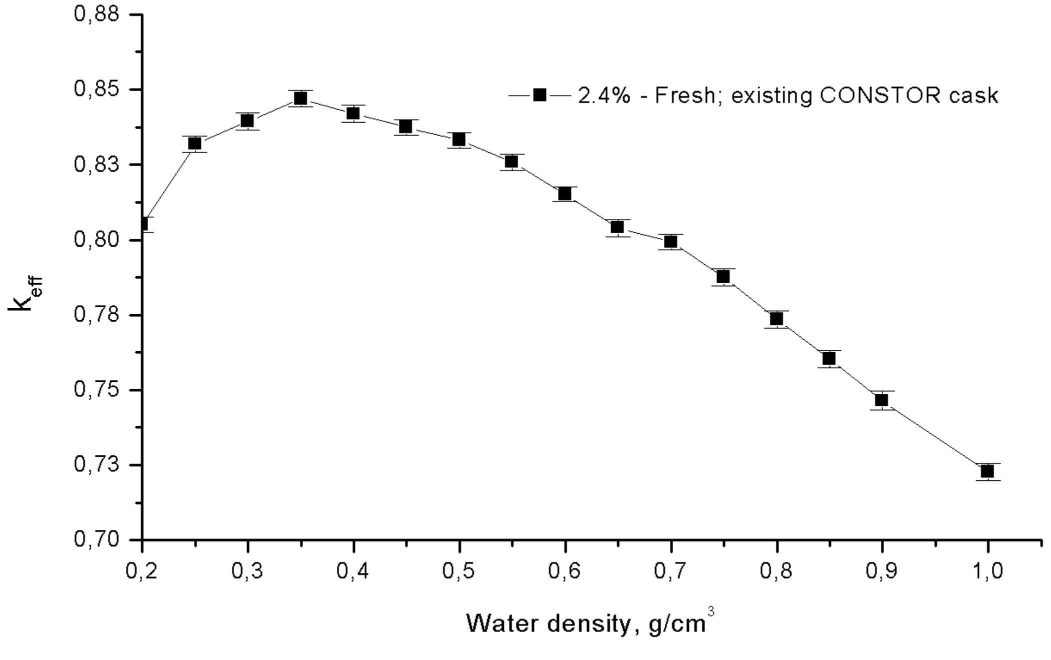

FIG.4. Variation of k_{eff} with the water density for CONSTOR cask with 102 half-assemblies

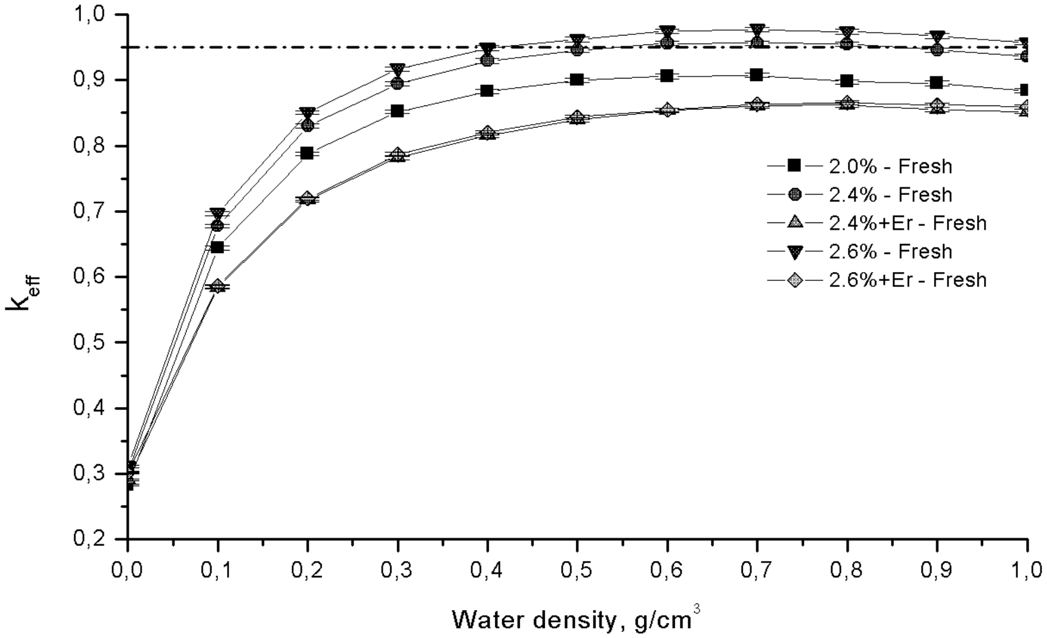

FIG.5. Variation of k_{eff} with the water density for CONSTOR cask with 156 half-assemblies

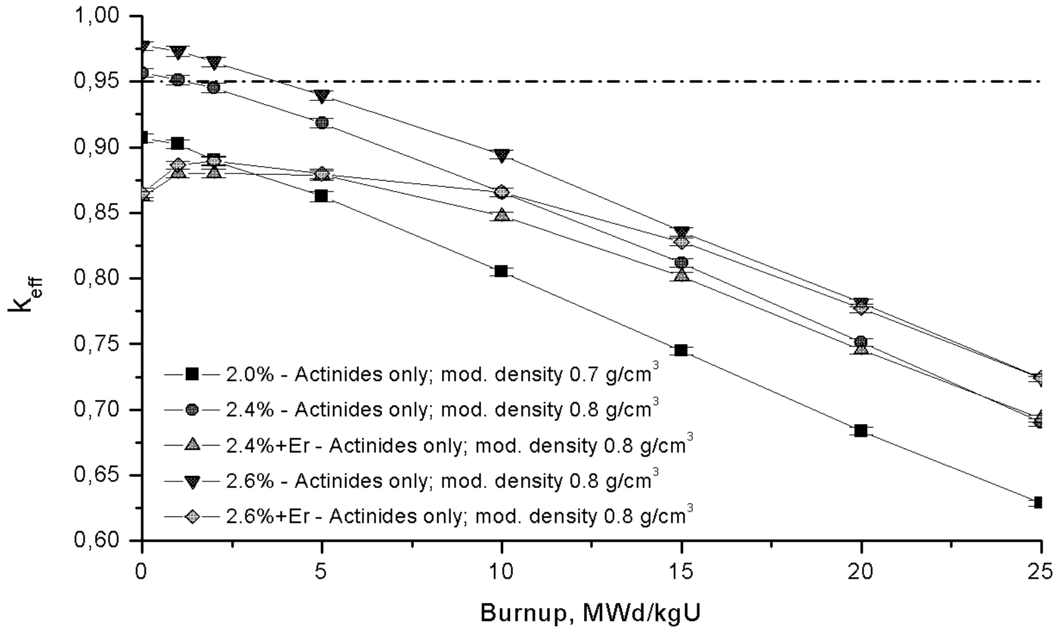

Fig.6. Variation of k_{eff} with the burnup for CONSTOR cask with 156 half-assemblies when only actinides are taken into account

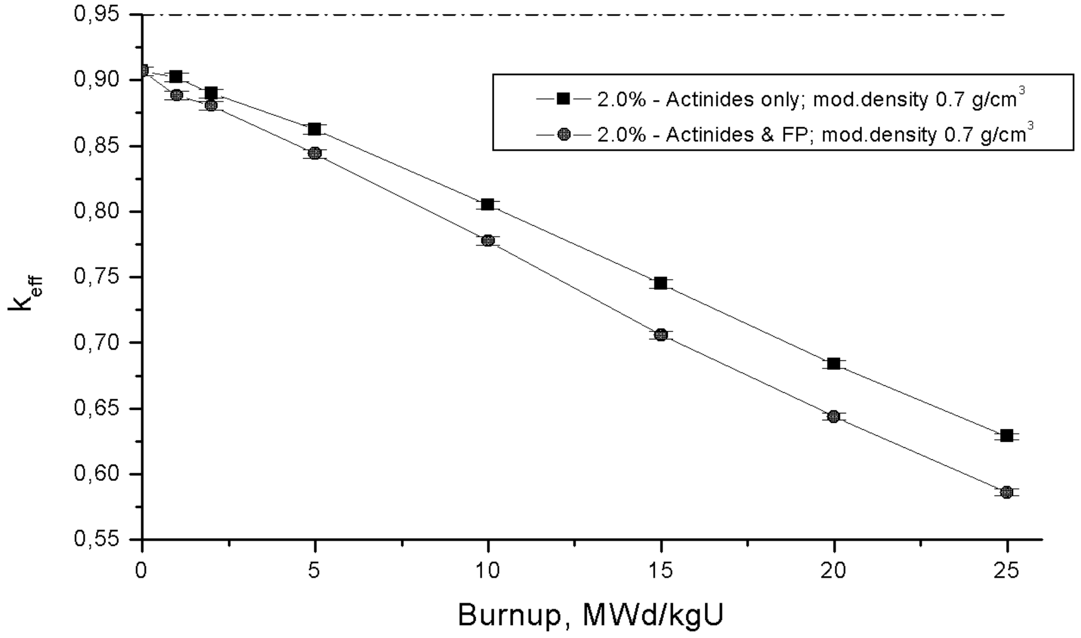

FIG.7. Variation of k_{eff} with the burnup for CONSTOR cask with 156 half-assemblies when actinides and fission products are taken into account

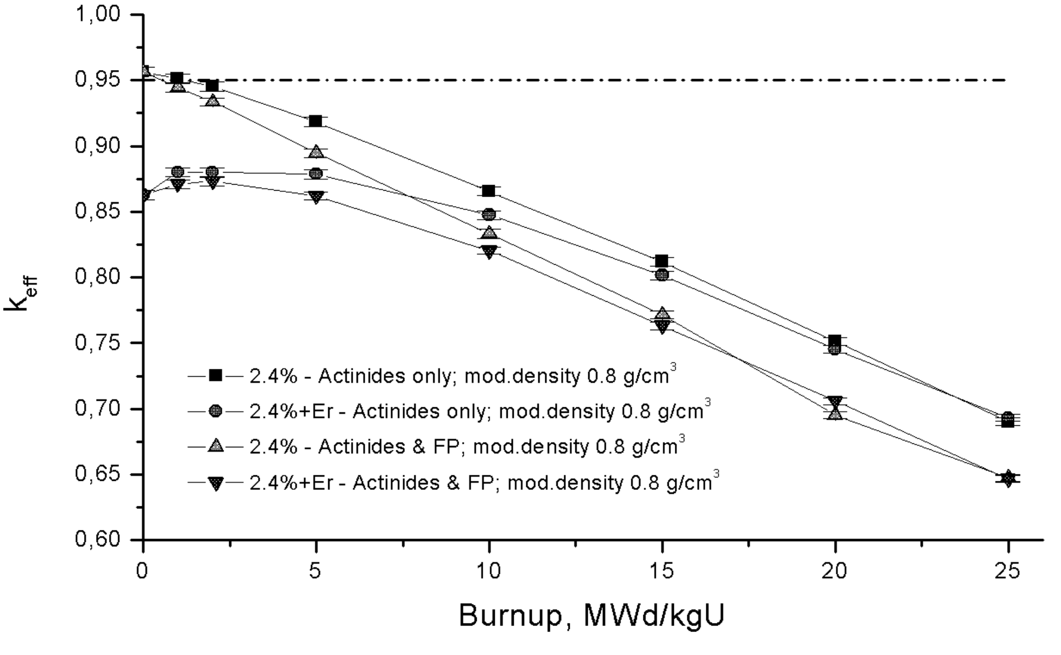

FIG.8. Variation of k_{eff} with the burnup for CONSTOR cask with 156 half-assemblies when actinides and fission products are taken into account

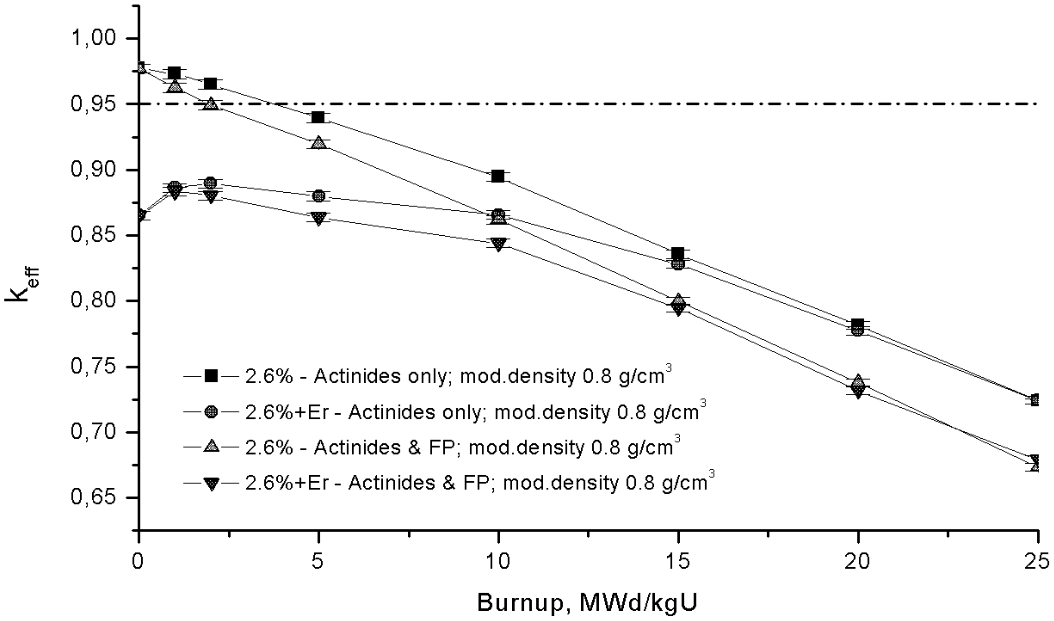

FIG.9. Variation of k_{eff} with the burnup for CONSTOR cask with 156 half-assemblies when actinides and fission products are taken into account

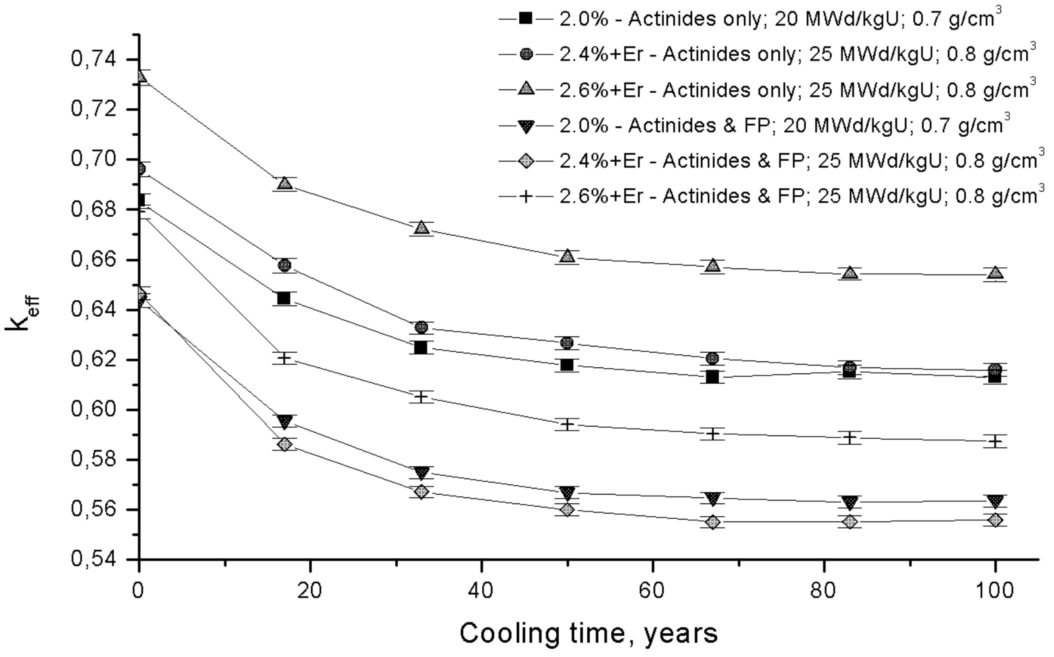

FIG.10. Variation of k_{eff} with the cooling time for CONSTOR cask with 156 half-assemblies when actinides and fission products are taken into account

5. FUTURE PLANS

1. It is expected to evaluate various parameters (e.g., axial and radial shapes, fuel irradiation history, etc.) that affect neutron multiplication factor when burnup credit is taken into account.
2. It is planned to have results of experimental investigation of irradiated RBMK fuel assembly and to perform detailed validation of the code.

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BUC — A simple nuclear criticality safety concept that can be very difficult to implement

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Abstract. New safety studies have been carried out. There is no general burnup threshold below which the burnup distribution is not significant. There is no general active fuel length threshold below which axial variations of the fuel or in the surroundings are insignificant. The axial burnup distribution effect for BWR fuel can be very large, even if the real void profile is accounted for or assumed to be flat at 70 %. The horizontal burnup distribution effect for PWR fuel can be significant, several reactivity percent. A mixed configuration of different fuel types can lead to a high reactivity increase, more than five percent. A combined effect of approximations, tolerances, incidents and human error effects can be very complicated to evaluate. Two individually conservative approximations can become non-conservative when combined. On the other hand, adding bias corrections and uncertainties from a large number of sources can be extremely conservative. Starting about 40 years ago, burnup credit was applied to a Swedish transport cask for irradiated MTR fuel elements. The fresh ^{235}U loading per fuel element was increased with time. Horizontal and axial burnup profile distributions as well as burnup verification became important. The transport casks for irradiated LWR fuel from Swedish reactors are licensed for some burnup credit in France but not in Sweden. The neutron absorbers cover the active lengths of the fuel assemblies only if special hardware supports are added before each shipment. The safety report allows some active fuel outside the neutron absorber region. A useful measure for various effects is the burnup credit reduction. Accurate information on individual fuel assemblies, even fuel rods, is valuable for many purposes. In burnup credit such information is important to determine realistic k_{eff} values but also to interpret verification measurements. The burnup distribution influences the neutron sources for consideration in shielding as well as the actinide build-up for consideration in final disposal (criticality and radiation protection) and non-proliferation issues.

1. INTRODUCTION

A nuclear criticality safety assessment requires methods, information on the fissionable material and its environment as well as identification of potential incidents and influences of human error. Most important, somebody needs to understand the methods and the information. Somebody needs to determine the importance of various facts and whether any information is missing. It is on this level that the influence of human error can be most severe.

Burnup credit is a concept restricted to nuclear criticality safety applications. Burnup credit is not a physics term. It is possible that credit for burnup can be obtained in some other area than criticality safety. The reactor operator requests low burnup and high productivity (electricity, heat, etc.). Credit will not be obtained for higher burnup at a given productivity. The radiation protection staff will not propose credit for shipping irradiated fuel in a transport package intended for fresh fuel. Credit for burnable absorbers is a completely different concept that should not be confused with burnup credit.

Nuclear criticality safety can be more complicated than most specialists realize. Many criticality specialists are or have been experts in limited areas and deserve respect for that. However, some of those specialists and others have extrapolated valid conclusions into areas where they don’t apply. Burnup credit is an area where insufficient nuclear criticality safety methods and understanding have been more clearly revealed than ever.

2. FISSIONABLE MATERIAL

Unirradiated nuclear fuel can be quite complicated but the fuel specifications, including material compositions are well defined. Simultaneous irradiation and decay make the material compositions difficult to determine. Both measurements and calculations are required. Fortunately, reactor management generates measurement and calculation results that are directly applicable to burnup credit. The reliability of these results with respect to burnup credit needs to be demonstrated. Incorrect information for a few fuel assemblies may be sufficient to cause a criticality accident if it is not corrected in time.

It has been established that axial and horizontal variations in the fuel material compositions can be important to burnup credit. However, the evaluations have not been extensive and have often been restricted to symmetrical arrangements of identical fuel assemblies. The environment has often been simple such as in a large storage pool or transport cask without incidents, human error or irregular features.

In reality, all irradiated fuel assemblies are different. Each fuel rod, even each fuel pellet is different. The level of detailed information that is required for burnup credit applications depend on the scenario.

Published methods for dealing with mixed configurations are not validated. Worse, they are clearly incorrect. Examples are the index method used in the IAEA Transport Regulations [1] and the Law of Substitution [2].

Mixed configurations of fresh fuel assemblies are common but it is easy to make conservative assumptions to reduce the number of required safety assessments. Burnup credit can lead to a large number of variations involving fuel assemblies with different characteristics. Potential applications involve independent storage facilities where both PWR and BWR fuel assemblies are stored. Transport casks may hold PWR and BWR fuel assemblies in the same baskets. When only one reactor type is involved, the fuel types may still be different. Even for a specific fuel design, the initial ^{235}U enrichments may vary, making the limiting burnups and fuel materials different.

The purpose here is to demonstrate potential complications with burnup credit. To find clear examples, some freedom has been allowed in creating different types of fuel assemblies. Criteria used in this creation are value of k_{eff}, fission density distribution, etc. The figures 1 and 2 show in quarter symmetry horizontal cuts through 3x3 arrays of fuel assemblies in water together with some of the iron reflection (water reflection would not change the conclusions). Figure 1 shows a potential fuel type containing 12x12 rods. It could be similar to a damaged (expanded) BWR fuel assembly but there is no need to find a corresponding existing design. The water separation between assemblies corresponds to 6 fuel rod cells. Figure 2 shows an 18x18 PWR fuel assembly type with 24 guide tubes. The fuel assemblies in figure 2 are tightly packed. The fuel rod dimensions and pitches are identical in both figures. The fuel compositions may vary axially as well as horizontally inside the fuel assemblies and axially inside the fuel rods. Fuel with integral burnable absorbers can be found in some rods and also replacing some of the guide tubes. The burnable absorbers may cover only parts of the active fuel length. Such designs are common for BWR fuel and exist for PWR fuel.

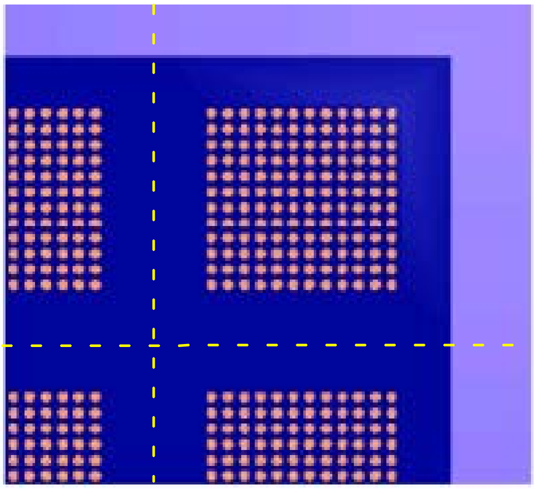


FIG. 1. 3x3 array of 12x12 assemblies.

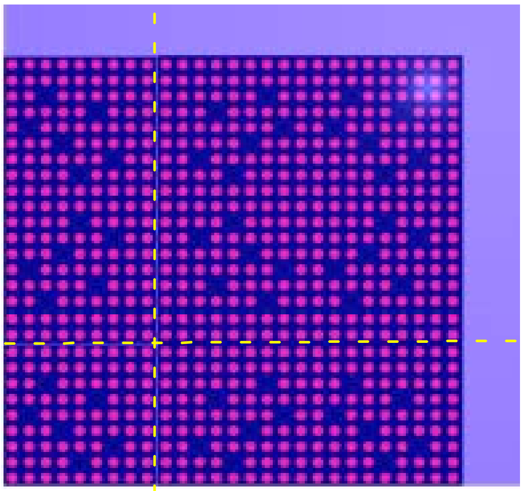


FIG. 2. 3x3 array of 18x18 assemblies.

3. MIXED CONFIGURATIONS OF FISSIONABLE MATERIAL

The arrays in figures 1 and 2 are adjusted to give values of k_{eff} close to 0.949. The rods in figure 1 contain unirradiated UO_2 with uranium containing 4.0 wt-% ^{235}U. The rods in figure 2 contain irradiated UO_2 with uniform 40 GWd/tU burnup (flat axial and horizontal burnup profiles). The guide tubes are replaced with fuel rods with uniform burnup of 11 GWd/tU. The irradiated fuel rods in figure 2 initially contained 4.0 wt-% ^{235}U. What can happen if the two fuel assembly types are stored or transported together? Many control methods that are accepted by criticality safety specialists are based on safety assessments of identical fuel assemblies. Reactivity equivalences are used to determine k_{eff} for a mixed configuration. In this case, where the two arrays of identical assemblies give almost identical values of k_{eff}, the mixed configuration should give essentially the same value.

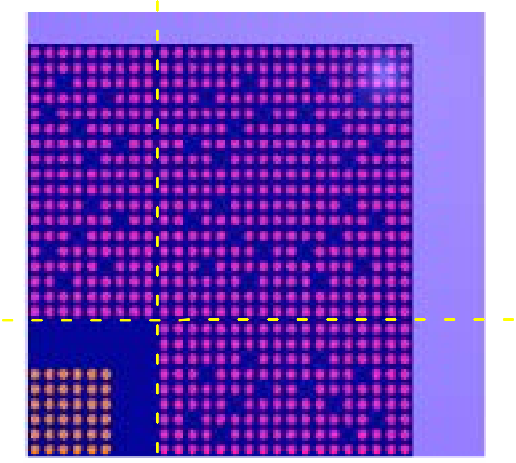


FIG. 3. Mixed configuration 1.

With a single fresh fuel assembly in the center, k_{eff} increases to 1.003. This simple demonstration is not restricted to burnup credit applications. The effect is general and often easy to predict and understand. If burnup credit is applied, there will be many more parameters that need to be considered simultaneously. Axial and horizontal burnup profiles, geometry variations in the surroundings, etc. The mixed configuration effect can be larger than 5 % in k_{eff}. It may be more difficult to realize in complicated scenarios.

4. HORIZONTAL ASSEMBLY BURNUP DISTRIBUTIONS

Only one compilation of horizontal burnup profiles has been considered [3]. The data in [3] is collected per fuel rod and per assembly. The extreme horizontal burnup profiles with more than 20 % deviation from the average burnup only occur for a few out of thousands of assemblies. Axial profiles within the fuel rods are not described. This means that local variations can be larger. For simplicity, the fuel rods are here assumed to have similar axial profiles as the average for a fuel assembly. Half of the rods are assumed to have average horizontal burnups, a quarter of the rods have 25 % higher burnup while the remaining quarter of the rods have 25 % lower burnup.

A PWR fuel assembly with 32 GWd/tU average burnup was selected for some calculations. The fuel compositions for this assembly with 21 axial zones were obtained from [4]. An artificial fuel assembly with the horizontal profile mentioned above was created from this assembly. Figure 4 shows a quarter symmetry model of a simplified preliminary final disposal storage canister. Four assemblies are embedded in iron. Calculations are also made with water between the assemblies.

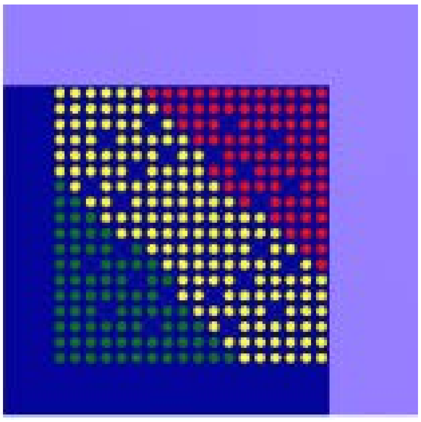


FIG.4. Horizontal, diagonal burnup profile.

When the axial burnup profile is kept, k_{eff} increases up to 2 % compared with a flat horizontal profile. The maximum increase is obtained for a small water separation. The increase is larger, almost 3 % if there is no axial burnup profile. However, these are artificial models and the results may be different with a more realistic horizontal burnup profile.

Calculations were also made for the larger geometry model with 9 fuel assemblies used earlier. To reduce k_{eff}, the 12 inner guide tubes were replaced with fuel rods with integral burnable absorbers, see figure 5 for a model without horizontal burnup profiles. Different horizontal burnup profiles were then introduced, see figure 6. The central assembly has a flat horizontal burnup profile while the others have fuel rods with low burnup facing the central assembly. All assemblies have similar axial burnup profiles. This is a mixed configuration. For the model in figure 5, k_{eff} was calculated as 0.960. There was an increase of k_{eff} to 0.974 for the model shown in figure 6. Without axial profiles, the increase is expected to be slightly larger. The central unit is not optimized. More burnup in the center, where the BA rods are, would probably increase k_{eff} somewhat. Also, it is expected that the horizontal effect becomes smaller for larger arrays.

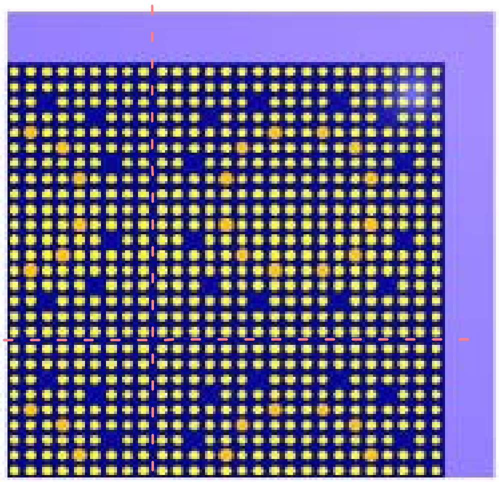


FIG. 5. PWR assemblies with BA rods

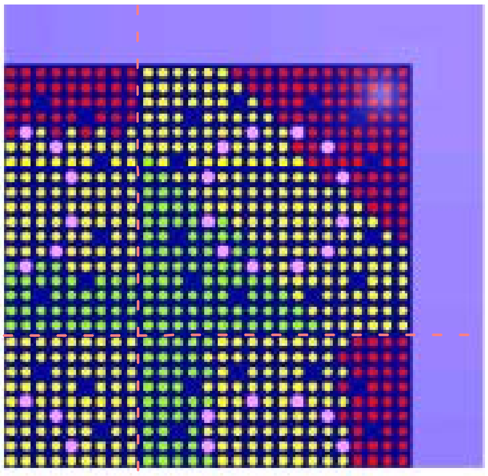


FIG. 6. Different horizontal profiles.

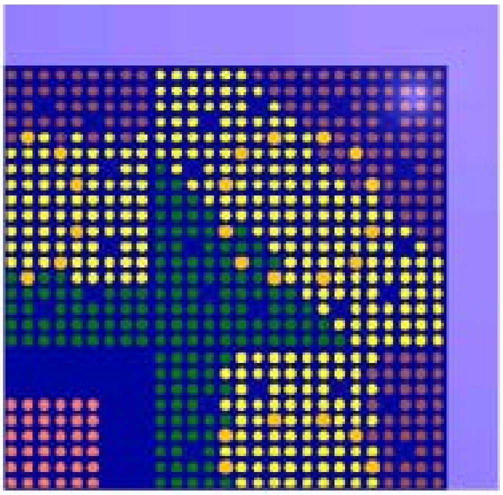


FIG. 7. X-Y cut of mixed configuration.

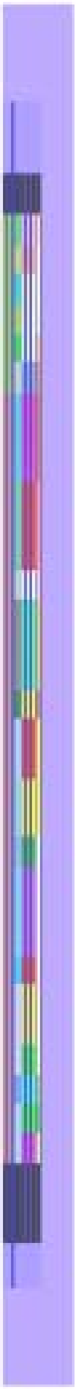


FIG. 8. X-Z cut.

Figure 7 (X-Y cut) and figure 8 (X-Z cut) show a mixed configuration based on the models shown in figures 1 and 5. However, the assembly in figure 1 was modified to contain uranium with 6 wt-% ^{235}U and burnable absorbers, except in the top 6 zones. This resulted in a similar k_{eff} as for figure 5, or 0.973. The mixed configuration shown in figures 7 and 8 resulted in a k_{eff} of 1.043. This is close to the sum of reactivity increases due to the mixed configuration and the horizontal burnup profile effects.

6. AXIAL BURNUP PROFILE EFFECTS FOR PWR FUEL

Uniform burnup profiles can cause serious under-estimations in k_{eff}. This was shown for PWR fuel in [5] for an average burnup of 30 GWd/tU. The reactivity effect is larger at 50 GWd/tU but significant also for 10 GWd/tU. Typical reactivity values are 8.4 % (30 GWd/tU), 15.4 % (50 GWd/tU) and 3.0 % (10 GWd/tU). These cases include fission products. Without fission products the corresponding reactivity values are 5.4 %, 8.8 % and 2.1 %.

The burnup distributions in [5] are simplified by being symmetrical around the mid-plane of each assembly. In [4] the burnup profiles are more realistic. A similar study of irregular geometry as in [5] has been made recently [6]. For 32 GWd/tU, the maximum axial profile reactivity effect was 12.2 %. For 50 GWd/tU, the corresponding effect was 19.6 %. The results are shown in figure 9.

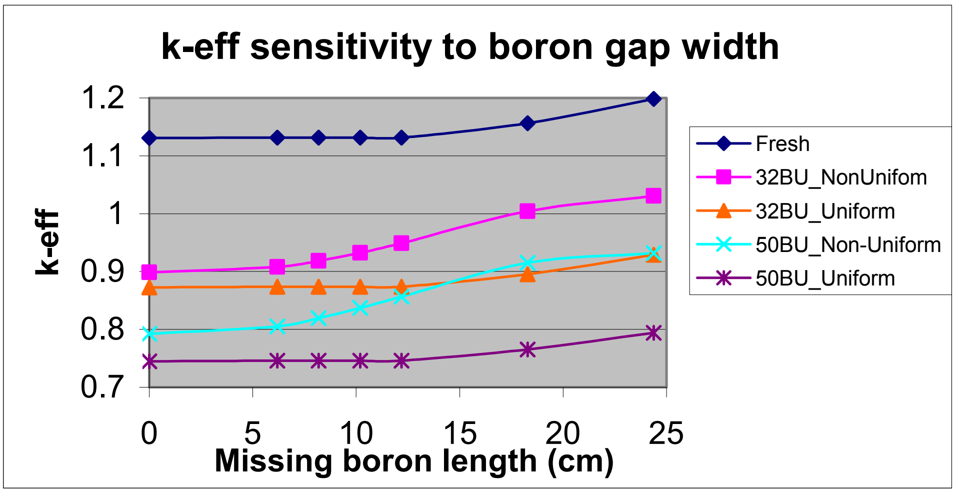


FIG. 9. k_{eff} for PWR fuel in transport cask with missing boron steel sections.

This figure shows that a realistic burnup profile at 50 GWd/tU can be more dangerous than a uniform burnup profile at 32 GWd/tU. In the past, some studies have proposed bias corrections to cover up the crude approximations of uniform burnup. Neither a few percent in k_{eff} nor 5 GWd/tU is enough to cover the potential errors in this scenario. Bias correction is not a suitable method to cover the lacking safety assessment caused by insufficient understanding of the fuel and its environment. The results are presented later (figures 14 and 15) in different formats: as reactivity values and as burnup credit reductions (BUCRs).

These reactivity effects are examples, not maximum possible. The more irregularities, the more difficult it is to predict the maximum combined effect.

7. AXIAL BURNUP PROFILE EFFECTS FOR BWR FUEL

In [7], burnup credit influences of void fraction and axial void distribution during reactor operation as well as of axial burnup distribution were studied. The chosen geometry can easily lead to incorrect conclusions. It appears as if the axial burnup profile is not important if an average void fraction of 70 % is selected. This cannot be extrapolated to the general case of BWR burnup credit.

To demonstrate this issue, a similar study as for PWR fuel was made recently [6]. Varying sections of boron plates between the fuel assemblies were removed, see figures 10 and 11. Fuel with the three burnups 20, 30 and 40 GWd/tU were included as well as different void and axial burnup profiles. The most interesting comparisons are between a case with a realistic burnup profile generated from a realistic void profile and a case with a uniform burnup profile generated from a uniform 70 % void.

The maximum axial profile reactivity effects are 6.0 %, 9.3 % and 11.0 % for the three burnups 20, 30 and 40 GWd/tU respectively. These are obtained with boron missing from about 10 cm of the active fuel zone. As mentioned above, these errors are obtained with a uniform burnup profile and with 70 % uniform void. Figures 12 and 13 show the results.

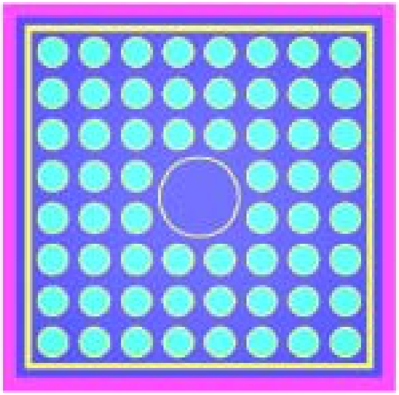


FIG. 10. BWR assembly in boron box.

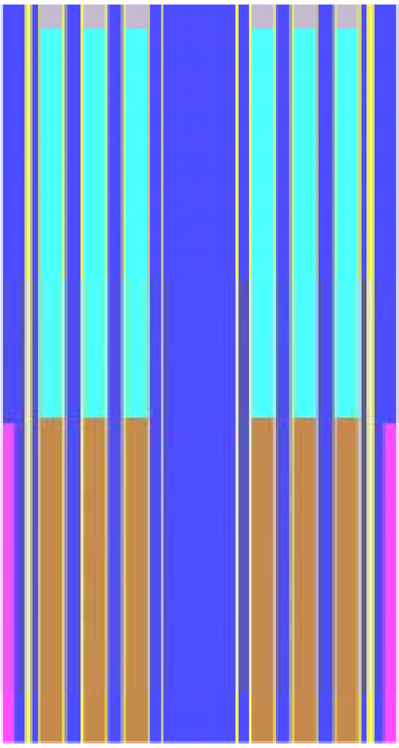


FIG. 11. X-Z cut (some boron removed).

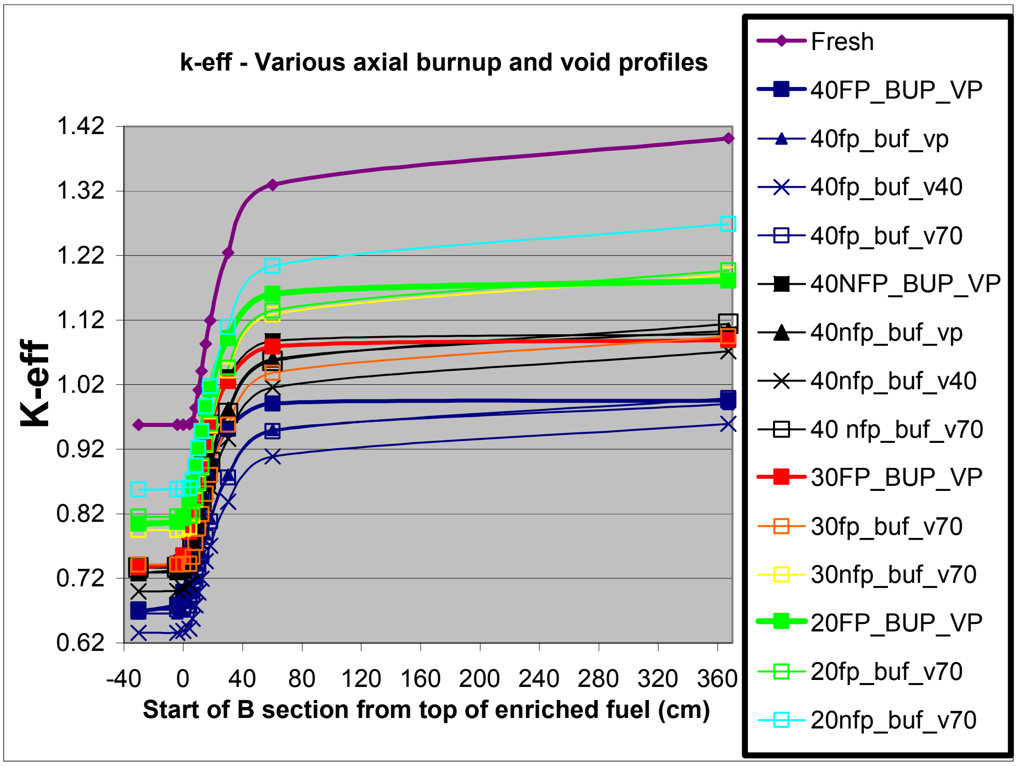


FIG. 12. BWR fuel assembly array with missing sections of boron – Full length.

Important conclusions can be drawn from the busy figure 12. The blue curve with solid blue squares is a best estimate (marked 40FP_BUP_VP). It covers fuel with 40 GWd/tU, fission products, realistic axial burnup profile and realistic axial void profile. It should be compared primarily with the blue curve with hollow blue squares (marked 40fp_buf_v70). This curve covers fuel with 40 GWd/tU, fission products, flat burnup profile and 70 % uniform void. It is almost identical to the blue curve with blue triangles (marked 40fp_buf_vp), which has a realistic void profile but a flat burnup profile. At the far left (no boron missing) and at the far right (no boron at all) all these curves coincide. If comparisons are only made for such scenarios, one may think that the flat burnup profile approximation is adequate together with 70 % void or a realistic void profile. That this is not the case in general can be seen in the enlarged figure 13, which uses the same legend as in figure 12.

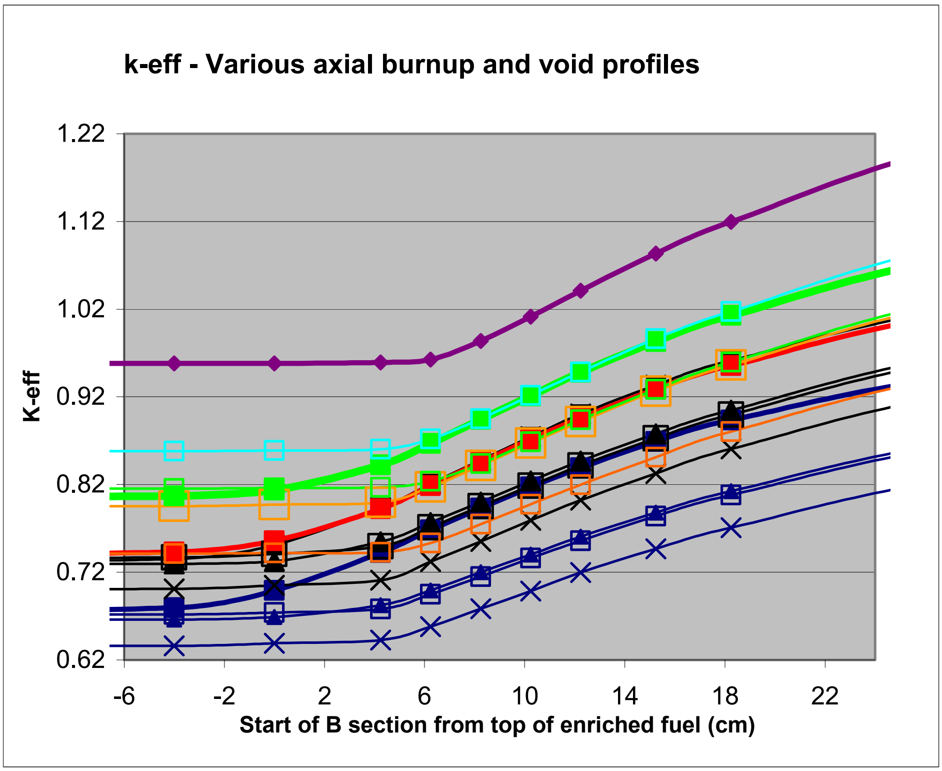


FIG. 13. BWR fuel assembly array with missing sections of boron – Selected region.

The reason for the importance of the boron in the region outside the active (enriched) fuel zone is that it contains a blanket with natural uranium. The results based on uniform burnup profiles do not show this effect correctly. The results are also shown later in figures 16 and 17 in different formats (reactivity values and BUCRs).

8. COMBINED REACTIVITY EFFECTS

It is normal procedure for criticality safety specialists to combine reactivity effects from approximations, incidents, human error, process variations, tolerances, cross section uncertainties, statistical uncertainties, etc. The reactivity effects are often determined individually with reference to a base case. Statistical uncertainties are often combined statistically as the square root of the sums of the squares if the causes of the uncertainties are independent.

A problem with these approaches is that even if the causes of the effects are independent, the consequences are not. In some cases this is obvious, but in others the question does not even seem to be raised. Burnup credit is an area where many biases and uncertainties are expected. It is important that the combined reactivity effects are not underestimated.

The most striking mistakes can be found when biases from two individually conservative approximations are added when the real combined effect is non-conservative. Two examples are given here. One is dramatic and easy to predict. The other is more difficult to predict.

First, consider a system consisting of two storage pools with different storage racks. One pool is filled with fresh water, the storage rack is poisoned, designed for fuel with higher enrichment and also contains such fuel. The other storage pool is dry, the storage rack is not poisoned and is designed for fuel with lower enrichment, but contains no fuel. The reactivity effect of moving the water from one pool to the other is negative. The separate reactivity effect of moving the fuel from the wet pool to the dry pool is also negative. If both actions were carried out at the same time, the conclusion obtained from adding the reactivity values would be completely wrong. Criticality may be possible.

The second case involves PWR fuel with 30 GWd/tU burnup. The base case is fuel with actinide changes, fission products and a uniform axial profile of both. k_{eff} is 1.106. The first reactivity effect is the influence of a realistic axial profile of the actinides without fission products. Δk_{eff} is -0.006. The second reactivity effect is the reverse; a realistic profile for the fission products without actinide changes. Δk_{eff} is -0.005. If both reactivity effects are added, the result is a k_{eff} of 1.095. A direct calculation with axial actinide changes and axial fission products result in a k_{eff} of 1.116. The two individually negative bias corrections of –0.005 and -0.006 become a positive bias correction of 0.010 when combined. The statistical uncertainties from the Monte Carlo calculations are much smaller and the calculations appear to have converged. Similar effects have been noted in other studies and should not be a surprise to anybody.

The point to be made here is that all potential sources for deviations from the correct solution should be evaluated in combination for each scenario. Conservative approximations are sometimes discarded without documentation. The problem is that they may not be conservative in another scenario or in combination with other approximations.

Burnup credit adds many new uncertainties to a nuclear criticality safety assessment. As mentioned above, adding bias corrections as if they were independent can be a danger to safety. However, more often it may lead to extremely conservative results. Cross section data for each fission product and actinide will have substantial uncertainties. Treating these individually will probably overestimate the total uncertainty. Bias corrections from other approximations may be derived for different scenarios that are incompatible. An ambitious evaluation of the combined effects of uncertainties and biases will lead to increased safety while avoiding punishing and extreme conservatisms.

9. BURNUP CREDIT REDUCTION – AN ALTERNATIVE CONCEPT

The safety influence of a specific effect is often measured in Δk_{eff} or in reactivity $\Delta k_{eff}/k_{eff}$. A different measure is to relate the effect to the nominal Burnup Credit Reduction (BUCR). The nominal burnup credit is defined as the difference in k_{eff} between the peak reactivity case (fresh for PWR fuel and slightly irradiated for BWR fuel) and the case with uniform burnup (based on the average burnup per assembly). The BUCR for a specific effect can be expressed as

$$BUCR = (k_c - k_u)/ (k_{pr} - k_u)$$

k_c is the result for a specific effect, k_u is the result for uniform burnup and k_p is the peak as a function of burnup. Consideration of accidents, human error, uncertainties, mixed configurations, etc., will increase the total BUCR.

Figure 9 shows the influence on k_{eff} of various missing boron absorber lengths for PWR assemblies with fresh fuel, different axial burnup profiles and different burnups.

Figure 14 shows the reactivity differences based on the same calculations. The maximum error caused by the uniform approximation is found at a missing boron length of about 18 cm.

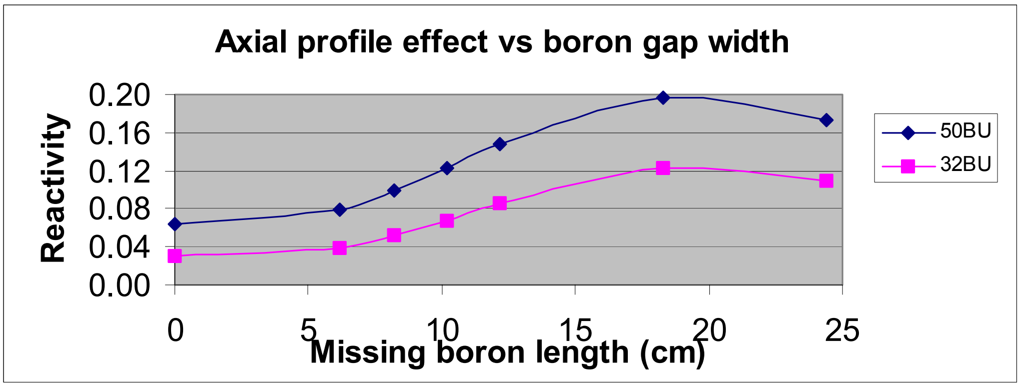


FIG. 14. Reactivity for PWR fuel in transport cask with missing boron steel sections.

Figure 15 shows the same results in the form of BUCR. An advantage of the BUCR concept is that it is easy to see the penalty of a specific effect on the potential burnup credit. Another advantage appears to be that the BUCR is not so dependent on the burnup. However, all three parameters k_{eff}, reactivity and BUCR are useful to focus on different points.

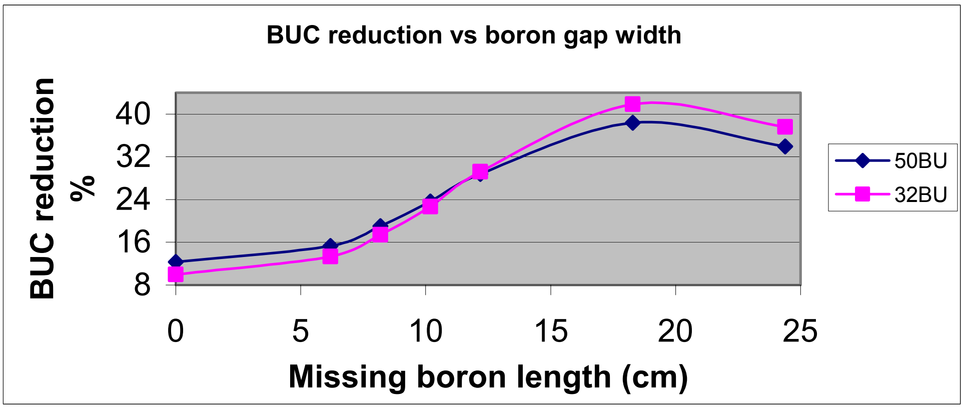


FIG. 15. BUCR for PWR fuel in transport cask with missing boron steel sections.

Similar curves for the BWR assembly results shown earlier in figures 12 and 13 are shown in figures 16 and 17.

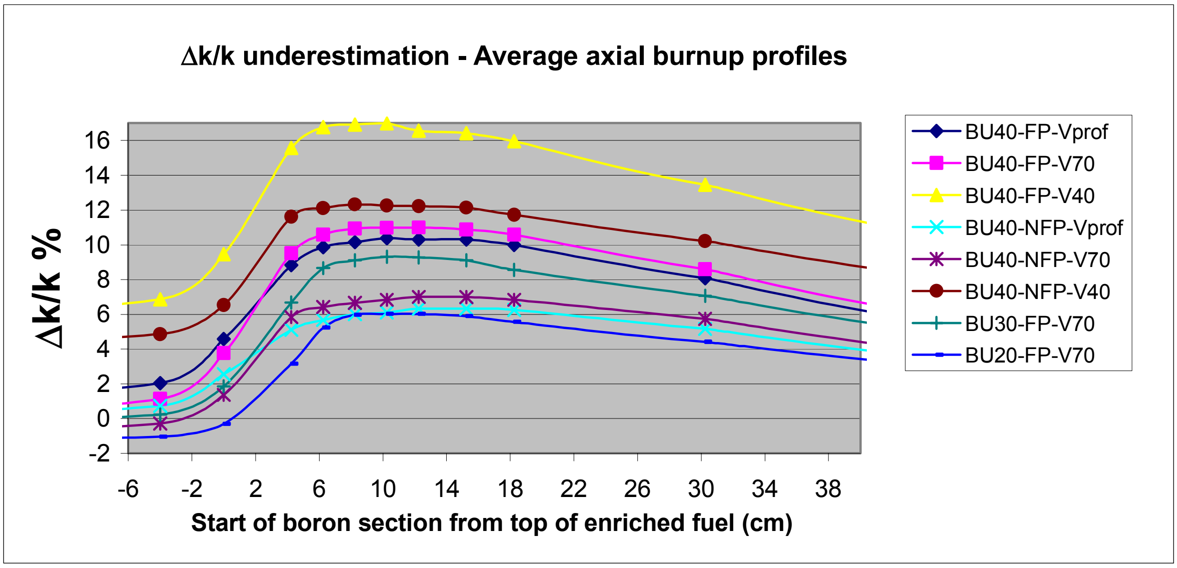


FIG. 16. Reactivity for BWR fuel in storage geometry with missing boron steel sections.

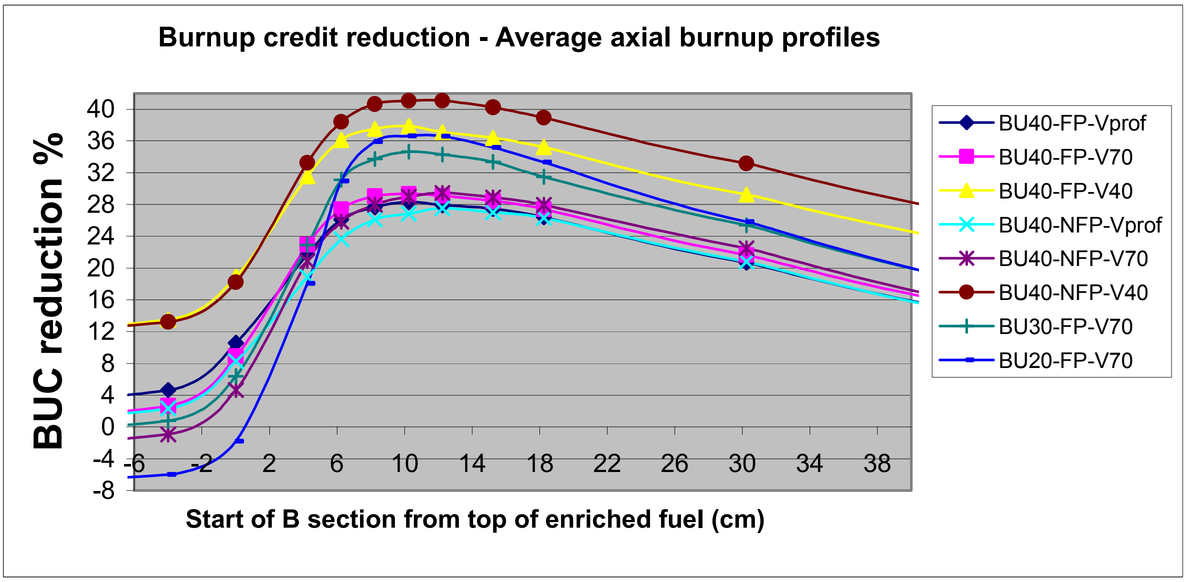
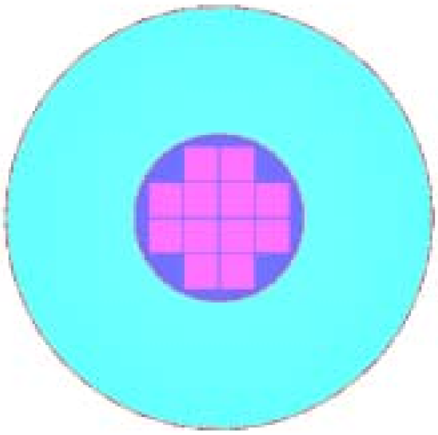


FIG. 17. BUCR for BWR fuel in storage geometry with missing boron steel sections.

10. BURNUP CREDIT IN SWEDEN – TRANSPORT CASKS FROM 1963

Burnup credit for transport of irradiated MTR fuel from Sweden to the U.S. has been applied during at least 20 years, starting about 1963. It started with a simple application, where the burnup credit was needed to obtain a substantial safety margin (80 % of the critical mass at a time when over-batching was not possible) but not to avoid criticality. The critical mass was obtained from sub-critical experiments in one of the actual casks without the lid and with real MTR fuel elements. Figures 18 and 19 show a model of the lead cask. The non-active parts of the fuel elements were cut off before transport. Loading was made with water in the cask but during transport the contents were dry. The lead lid reduced the estimated critical mass, something that was neglected for a long time.



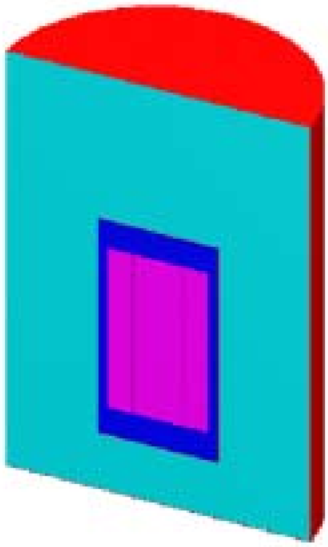


FIG. 18. MTR fuel element cask. *FIG. 19. Vertical cut through MTR cask.*

In 1979, a revision of the transport certificate was issued [6]. The initial ^{235}U loading per fuel element had been increased from 200 grams to 250 grams. Burnup credit was now necessary since the experiments showed that the cask would be critical otherwise. The safety report supporting the application had not been changed since 1969. Before the transport certificate was issued, the maximum ^{235}U mass per fuel element was restricted to 200 grams (horizontal burnup distribution control). Further, requirements were made for the applicant to assess the influence of axial burnup distribution and human error during loading. The result was that realistic burnup reduced k_{eff} compared with the case of uniform burnup.

11. REAL COMPOSITIONS OF IRRADIATED FUEL

The smallest component with fissionable material in a fuel assembly is a fuel pellet. Irradiation makes the pellet composition non-uniform. In depletion calculations, this non-uniformity should be considered. If the average composition after irradiation is known, it is probably sufficiently accurate for burnup credit application to UO_2 fuel. This needs to be confirmed, and also further evaluated for MOX fuel and for fuel with burnable absorbers. An average burnup will not give correct average material composition. Figure 20 shows how different nuclide densities in the fuel change as a function of burnup. Each density function is normalized to the density at full burnup (37.6 GWd/tU). The chart program is slightly confused with some values at the higher burnups. This is because the same burnup in different parts of the assembly causes different nuclide densities (neutron energy spectrum effects).

The non-linear effect for ^{243}Am (bottom curve) clearly shows that burnup averages will not give correct actinide densities. Half of the ^{243}Am is created in the top 20 % of the burnup. Higher actinides (not calculated) should behave similarly. Higher burnups will emphasize this trend.

The next level of detail after the pellet is the fuel rod. This may be axially non-uniform already before the irradiation. After the irradiation there will be significant variations even if the composition was uniform before irradiation. These variations may be very important for a correct criticality safety assessment.

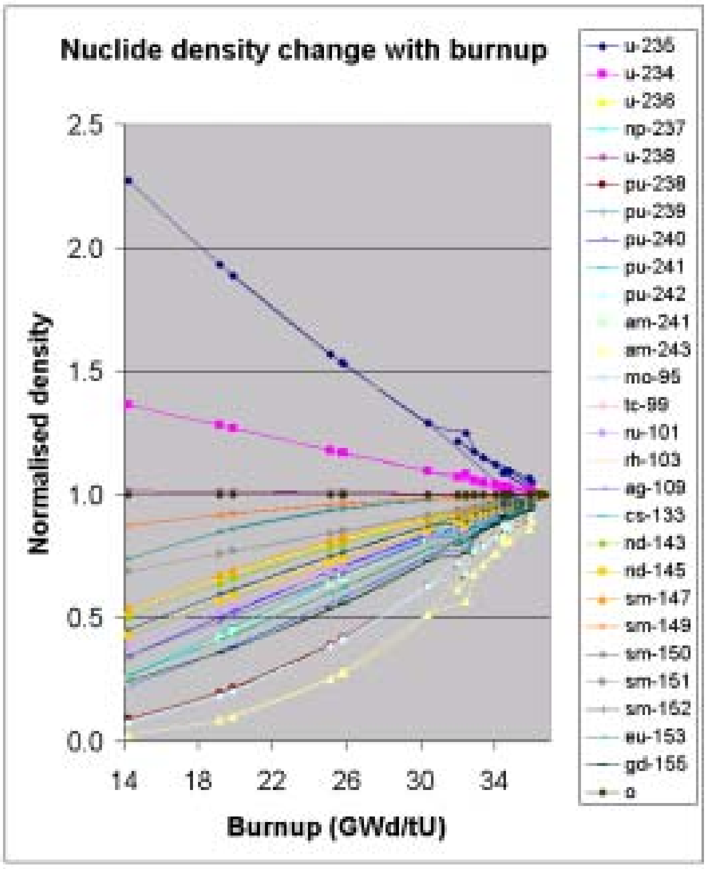


FIG. 20. Nuclide density changes as a function of burnup.

A fuel assembly is made of fuel rods, water holes, guide tubes and other objects. As long as the fuel assembly stays intact, advantage can be taken of simplified specifications of the material variations in the assembly. The axial burnup profiles can probably vary significantly between different fuel rods in the same assembly. However, I have not seen realistic axial burnup profile data for fuel rods, only for fuel assemblies.

The data generated by reactor simulation codes and supported by reactor measurements appears to be valuable in many later operations. Adequately correct material compositions, as functions of time and space in individual fuel rods, are probably achievable. Besides in burnup credit, this information is valuable in other areas such as burnable absorber credit, non-proliferation control, shielding and final disposal (including criticality safety aspects).

The methods used for reactor simulation can probably be used directly in connection with burnup credit. Besides generation of nuclide densities, the codes can also calculate k_{eff} directly, at least for fuel assemblies in water and in the presence of neutron absorber materials. These conditions are similar to reactor operation for which the codes and data are continuously validated. In burnable absorber credit, a common approach is to use the specifications of the node that generates the maximum k_{eff} if all nodes were identical. This and similar information could be given for each fuel assembly. Administrative requirements to check this information before each operation can be simplified with computer software. Without such controls, the likelihood increases that a future fuel design or a reactor operating condition changes outside the limits used in the safety assessment.

12. SOURCE CONVERGENCE

Early studies of burnup credit showed that the traditional statistics used in Monte Carlo calculations were not sufficient. Where less than 100 000 neutrons used to give adequate results, now more than 1 million neutrons are required. With poor guesses of the neutron fission distribution, many more neutrons are required and convergence cannot even be assured.

What does source convergence mean? It depends on what parameter is requested and its sensitivity to variations in the fission distribution. If the maximum reactivity influence of a particular material is searched, a scenario should be chosen where this material is important.

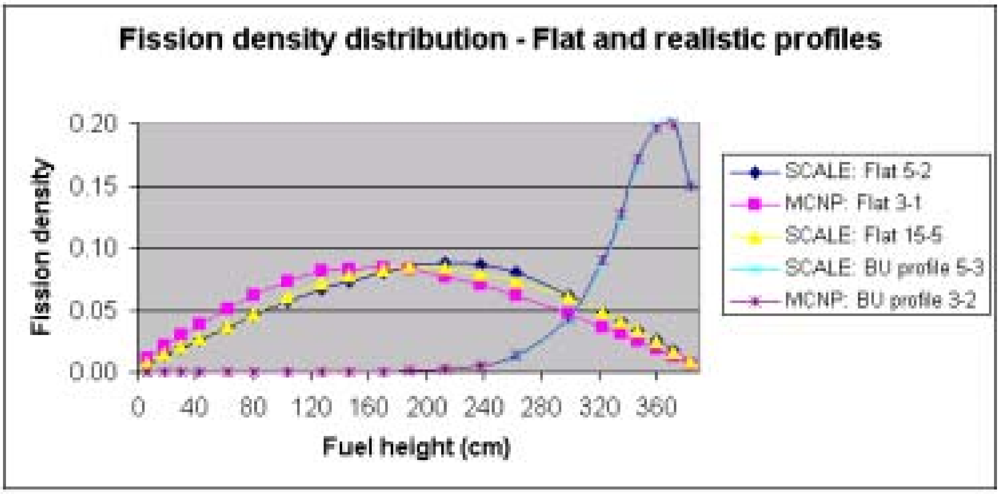


FIG. 21. Fission density distributions and source convergence.

All of the curves in figure 21 correspond to adequate source convergence related to k_{eff}. They refer to the same problem but are based on different approximations (flat or realistic burnup profiles), statistics (5-2 indicates 5 million neutrons of which 2 millions are skipped) and different codes (SCALE [8] or MCNP [9]).

The differences between the three “flat” curves are not significant. The fissionable material is identical and the only thing that should be expected is a noticeable neutron leakage at the ends. The radial source convergence may be more important to study since it involves strong neutron flux gradients through boron absorbers between the fuel assemblies. For the realistic burnup profiles similar shifts in the curves would indicate non-convergence. However, the agreement for these cases is very good.

Figure 22 shows a number of fission density distributions. They refer to different burnups (32 or 50 GWd/tU), different axial burnup profiles (uniform or non-uniform) and to various missing boron lengths. These curves show that the reactivity influence of a local variation or a material such as grids (fuel rod spacers) depends strongly on the position and the scenario.

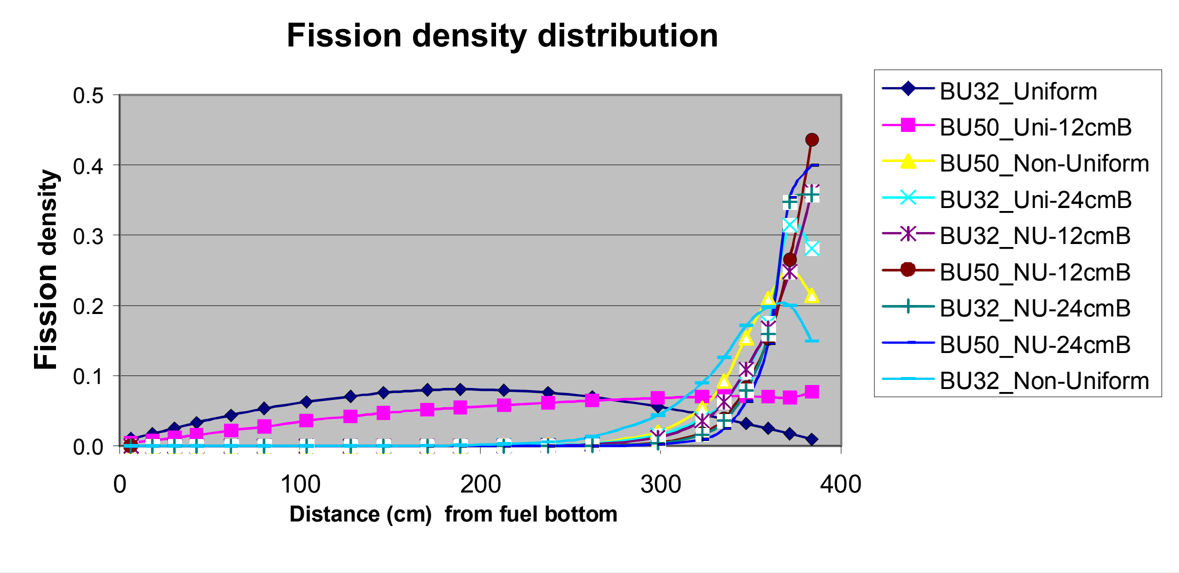


FIG. 22. Various fission density distributions.

13. CONCLUSIONS

There are serious flaws in current methods for criticality safety assessment and in the ways they are applied. Together with all the variations and influences of human error related to the properties of irradiated fuel, this would make general burnup credit an unpredictable risk.

Transport as well as storage in dry casks or storage in pools with borated water may not be seen as general burnup credit and safety can probably be credibly assured. Credit for burnable absorbers is not burnup credit in any way and the depletion calculations needed are simple. Depletion and radioactive decay calculations are needed for final disposal of irradiated fuel, whether burnup credit is applied or not.

In the long term, all countries will benefit from the efforts made today to learn more about depletion calculations and verification. Burnup credit can be used to improve efficiency and to reduce consumption of natural energy resources. Capability of assessing the real hazard during an operation is an accident preparedness resource that can reduce unnecessary worries and lead to correct prioritization during incidents. Further studies related to methods involving realistic (mixed) configurations and to compilation of data on irradiated fuel are recommended.

ACKNOWLEDGEMENTS

The Swedish Nuclear Power Inspectorate (SKI), in particular Jan In de Betou, has supported my participation in the OECD/NEA studies on burnup credit, Swedish studies on mixed configurations of fissionable materials as well as other nuclear criticality safety studies. The results of these studies, I believe, show that the support has been motivated.

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BUC APPLICATION TO LONG TERM STORAGE AND DISPOSAL

(Session 2.6)

Future disposal burnup credit process and effort

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Abstract. The United States Department of Energy's Office of Civilian Radioactive Waste Management has developed a risk-informed, performance based methodology for disposal criticality analyses. The methodology is documented in the *Disposal Criticality Analysis Methodology Topical Report*, YMP/TR-004Q (YMP 2000). The methodology includes taking credit for the burnup of irradiated commercial light water reactor fuel in criticality analyses, i.e., burnup credit. This paper summarizes the ongoing and planned future burnup credit activities associated with the methodology.

BACKGROUND

The United States Congress charged the U.S. Department of Energy (DOE) with managing the deep geologic disposal of high-level radioactive waste (HLW) and spent nuclear fuel (SNF) through the Nuclear Waste Policy Act of 1982 and the Nuclear Waste Policy Amendments Act of 1987. The DOE intends to directly dispose of the majority of its commercial SNF, without modifying the SNF by reprocessing or rod consolidation. A risk-informed, performance based approach is to be used to demonstrate that the health and safety of the public will be protected for the licensed lifetime of the geological repository. One criterion used in that demonstration is that an inadvertent criticality be classified as a very unlikely event (i.e., less than one chance in 10,000 over the first 10,000 years) (10 CFR 63).

The criticality approach is documented in the *Disposal Criticality Analysis Methodology Topical Report*, YMP/TR-004Q. This approach has been reviewed by the U.S. Nuclear Regulatory Commission and accepted in their issuance of the *Safety Evaluation Report for Disposal Criticality Analysis Methodology Topical Report, Revision 0* (Reamer, 2000). The methodology includes taking credit for the burnup of irradiated commercial light water reactor fuel in criticality analyses, i.e., burnup credit. Burnup credit is used in the methodology to determine the fraction of commercial SNF with the potential to achieve criticality.

The burnup credit used for disposal accounts for the key actinides and fission products important to reactivity (criticality potential). The isotopes used are called the *Principal Isotopes*. A list of the Principal Isotopes is provided in Table I.

Table I. Principal isotopes for disposal burnup credit

Fission Products				
Mo-95	Tc-99	Ru-101	Rh-103	Ag-109
Nd-143	Nd-145	Sm-147	Sm-149	Sm-150
Sm-151	Sm-152	Eu-151	Eu-153	Gd-155
Actinides				
U-233	U-234	U-235	U-236	U-238
Np-237	Pu-238	Pu-239	Pu-240	Pu-241
Pu-242	Am-241	Am-242m	Am-243	

INTRODUCTION

The burnup credit portions of the disposal criticality methodology are made up of two main models, the criticality model and the isotopics model. The criticality model consists of the MCNP computer code with the point-wise/continuous energy cross-section library. The isotopic model consists of the SAS2H sequence of the SCALE computer code system using the 44 Group Library cross-sections and selected depletion parameters.

The major ongoing efforts in disposal burnup credit consist of validating the two models for Pressurized Water Reactor (PWR) fuel. Model validation reports for the two models are scheduled to be completed in 2002. Revisions of the model validation reports for Boiling Water Reactor (BWR) fuel are to follow in 2003. The major future planned effort in disposal burnup credit consists of using the validated models in analyses of a waste package design for the planned Yucca Mountain Repository License Application. These evaluations are to start before the end of 2002.

CRITICALITY MODEL VALIDAITION

The validation of the Criticality Model involves three main steps. The first step involves comparing how well the Criticality Model can compute k_{eff} for a number of critical benchmark cases. The second step involves performing statistical analyses on the results to establish the bias, uncertainty, and any trends in the bias. The last step involves defining Critical Limits (CL) based on the statistical analyses and defining the range of applicability for the CL.

Defining of CL values, determination of ranges of applicability, and general information about the statistical analyses are discussed at length in the Topical Report (YMP 2000). The DOE has automated the statistical analysis in a computer code called CLREG (BSC 2002). A general discussion of the benchmark selection process is also provided in the Topical Report. An expanded discussion on aspects of the benchmark data being used for the validation will follow.

Three main types of benchmark data are being used in the validation of the Criticality Model for disposal burnup credit. The three types are as follows:

1) Commercial Reactor Criticals (CRCs);
2) Laboratory Critical Experiments (LCEs);
3) Reactivity Worth Experiments (RWEs).

CRCs are considered irradiated fuel critical experiments where measurements are taken at specific snapshots in time during the startup operations of a commercial power reactor. CRCs will be used to determine the CL for intact configurations of SNF. Over 40 CRC state points for PWR reactors are used in determining the CL.

LCEs are small critical experiments performed in laboratories. They are used in determining the CL for various geometries (lattices to solutions) that SNF may experience over the time period of disposal. Over 300 different LCEs have been evaluated.

RWEs are experiments that measure small changes in reactivity due to the presence of known quantities of different actinides and fission products. They provide an additional means of validating the cross section libraries used in the MCNP code.

The use of LCEs and RWEs for validating criticality models is somewhat common in criticality safety applications. The use of CRCs is common for reactor physics applications, but is not a common benchmark for criticality safety evaluations. Some additional discussion of what CRCs are will be provided.

Commercial Reactor Criticals

The CRC snapshots in time are referred to as state points. The CRC state points are selected at critical configurations, under hot zero-power conditions, and where sufficient time has passed since the reactor shut down for Xenon-135 to have sufficiently decayed (2 days or longer). Some of the PWR CRC state points being used in the validation of the disposal criticality model come from the Crystal River Unit 3 (CR3), Three Mile Island Unit 1 (TMI1), McGuire Unit 1 (MG1), and Sequoyah Unit 2 (SQ2) reactors. Figure 1 shows some CRC results. Results are shown for four different sets of isotopes. The Principal Isotope (PI) set includes the 29 isotopes listed in Table I. The Best-Estimate (BE) set includes up to 84 isotopes available in the MCNP continuous energy cross-section library. The Principal Actinide (PA) set includes the 14 actinides from the Principal Isotopes list shown in Table I, and the Actinide-Only (AO) set includes the 9 isotopes from the Topical Report on Actinide Only Burnup Credit (DOE 1998), plus U-236.

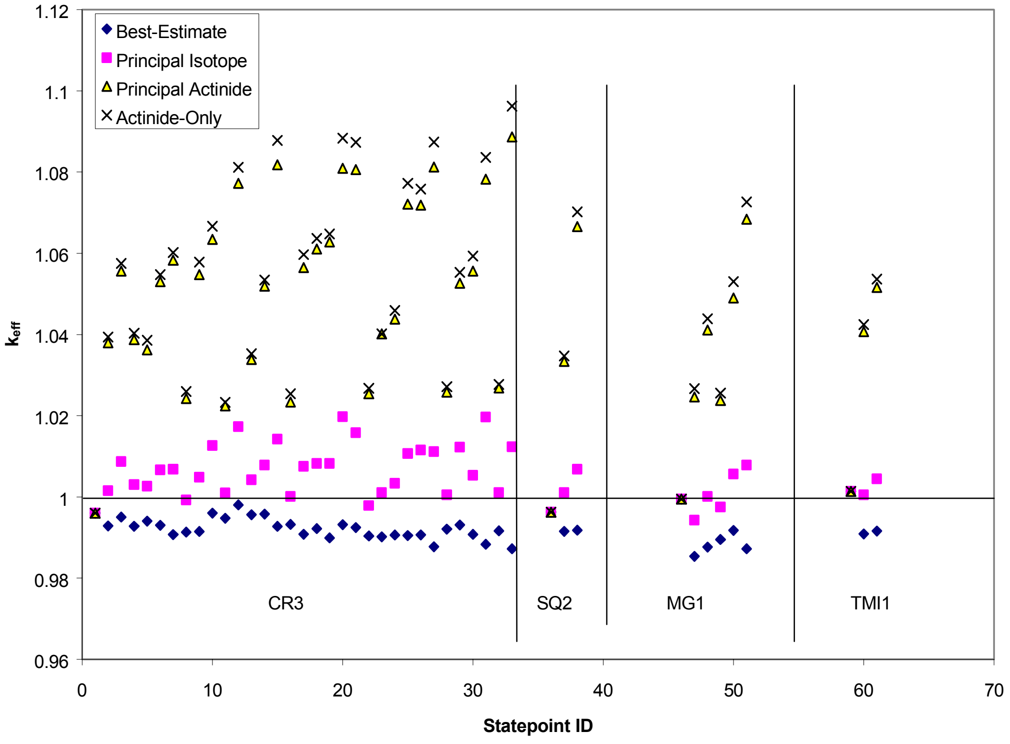

FIG. 1. PWR CRC Eigenvalues.

The CRC results shown in Figure 1 demonstrate how well the Criticality Model can model a critical configuration of irradiated fuel. A point to note is how the variability or uncertainty in the calculated values increases when fission products are ignored. The average k_{eff} values and the 2-sigma uncertainty for the different isotope sets shown in Figure 1 are as follows:

Best-Estimate	0.9916 ∀ 0.00298
Principal Isotopes	1.0062 ∀ 0.00291
Principal Actinides	1.0520 ∀ 0.00298
Actinide-Only	1.0545 ∀ 0.00295

A common concern about using CRCs in determining a CL for waste packages is the applicability of CRCs. A number of statistical and trend evaluations have been perform regarding the applicability of CRCs, some of these are presented in Sapyta (2000). DOE has worked to address this issue and has presented this work in a paper at the 2001 winter ANS meeting (Scaglione, 2001). A brief summary of the paper is provided here.

The applicability of using CRCs as a benchmark for waste packages can be addressed by looking at the fundamental neutronic parameters of the systems. The fundamentals include materials, geometry, and spectrum. The materials and geometry of fuel assemblies in a CRC are the same as will be in waste packages. The fuel pin dimensions, lattice spacing, and structural materials are the same. CRCs and waste packages both contain irradiated fuel isotopics. The reflectors and moderators in both systems are water. The geometry and material differences are for criticality control purposes. There are differences in the inter-assembly pitches and the interstitial material. These items are addressed by looking at the neutron spectra of the two systems.

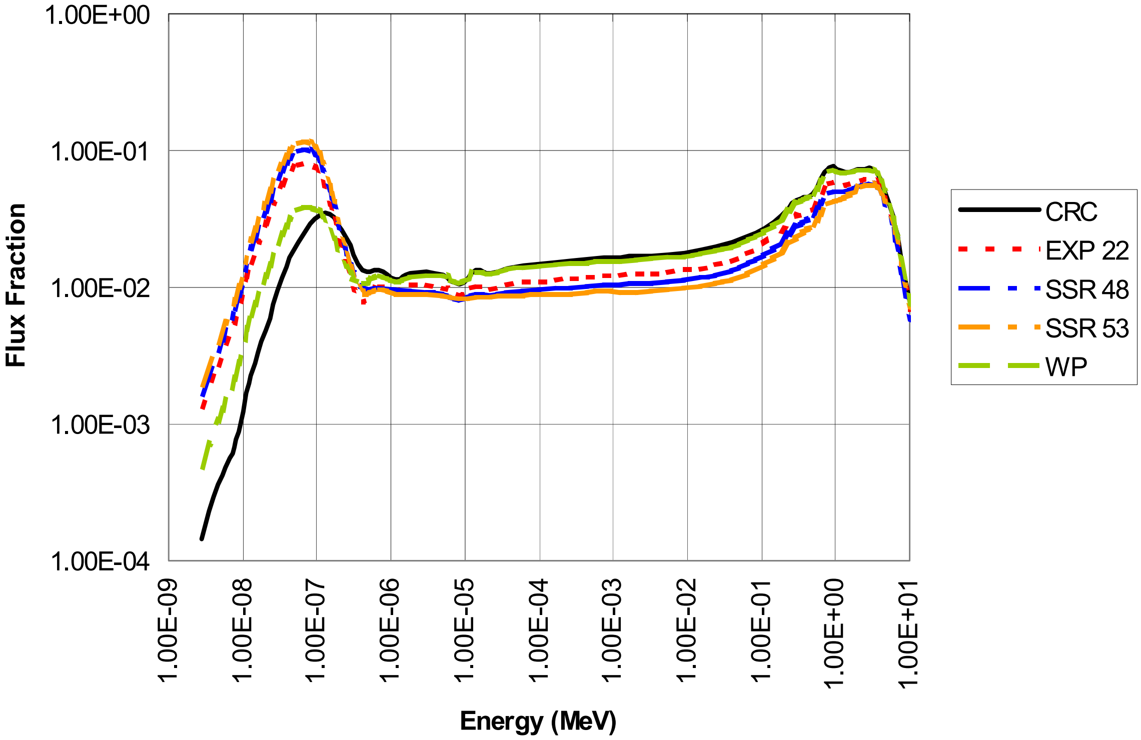

FIG. 2. Neutron Energy Spectra Comparison.

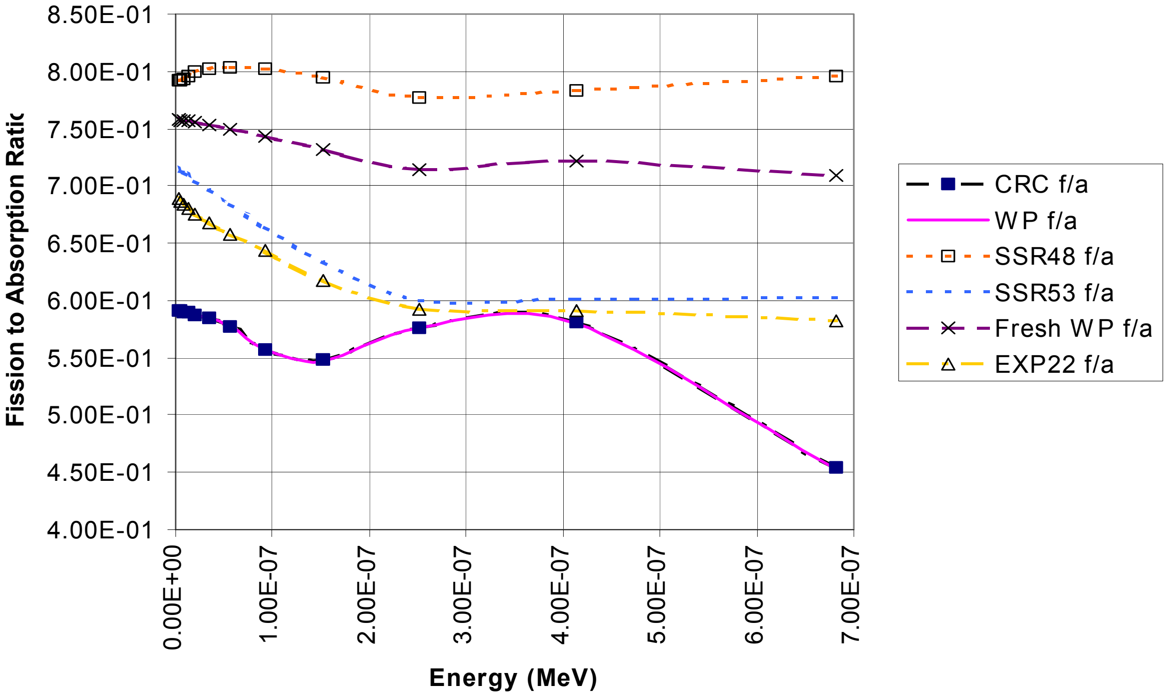

FIG. 3. Fission to Absorption Reaction Ratios in the Thermal Region.

A comparison of the neutron energy spectra between a CRC, a waste package with irradiated fuel, and three LCEs was performed. The three LCEs evaluated include: 1) the Saxton53 (SSR53) MOX LCE which has 6.6wt% PuO_2-Natural U UO_2-90wt% Pu-239 fuel, 2) the EPRI22 (EXP 22) MOX LCE which has 2.0wt% PuO_2-Natural U UO_2-90wt% Pu-239 fuel, and 3) the Saxton48 (SSR48) LEU LCE which has 5.74wt% U-235 UO_2 fuel. Figures 2 and 3 show the comparisons.

From Figure 2 it can be seen that the CRC neutron spectrum is very close to that of a waste package. The CRC and waste package neutron spectra are closer than the LCE and waste package neutron spectra. Part of the reason for the differences with the LCE is the greater leakage for the small LCEs. The small shift in the CRC spectrum compared to the waste package is primarily due to temperature differences, which cause a slightly harder spectrum in the CRCs.

Another neutron spectra comparison is shown in Figure 3. In Figure 3, the fission to absorption reaction ratios in the thermal region are shown for the five cases in Figure 2 (CRC, waste package, EXP 22, SAX48, and SAX53) plus a case for a waste package with fresh, unirradiated fuel. The comparison of the ratios for the CRC and the waste package with irradiated fuel shows good agreement and indicates that the ratios are primarily material composition dependent.

These comparisons show that the fundamental neutronic parameters for CRCs are very similar to those for a waste package with SNF. Because of this similarity, it can be concluded that CRCs are applicable benchmarks for validating the Criticality Model for waste packages.

It is recognized that there is also an issue of the difference in temperatures between the CRCs and waste packages. Temperature differences have been addressed using code-to-code comparisons between MCNP and CASMO results (Anderson, 2000).

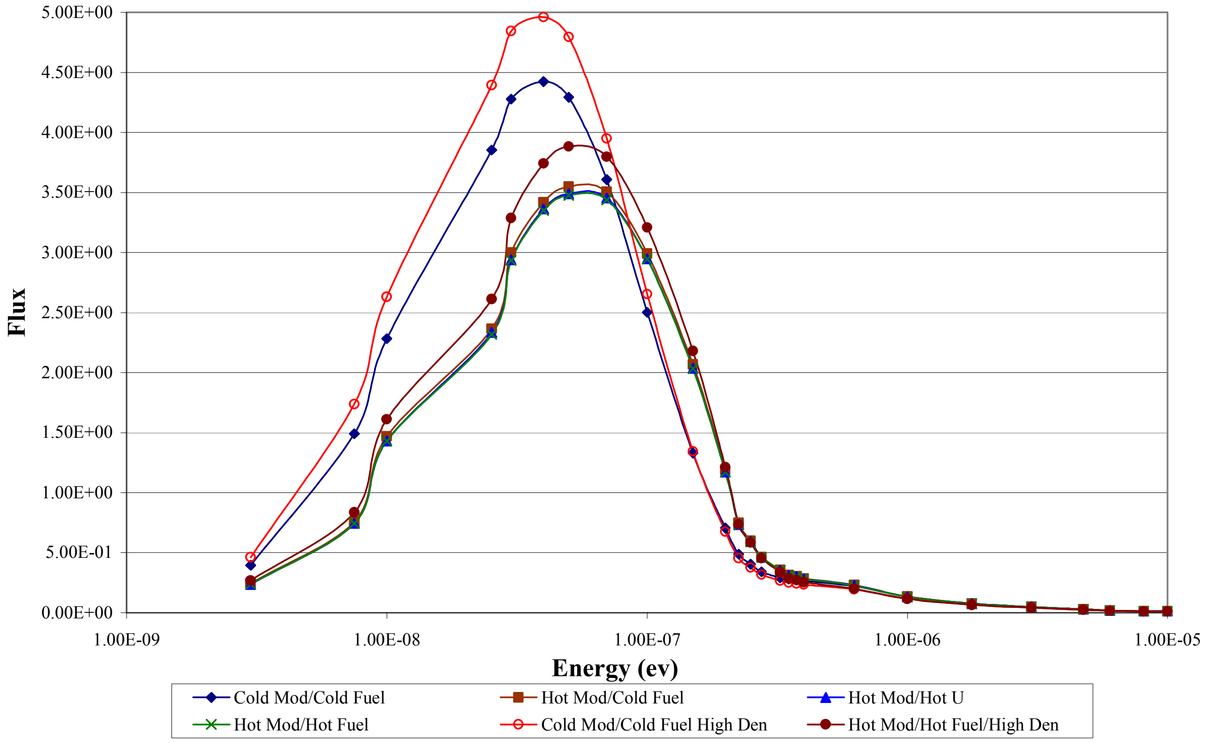

FIG. 4. Spectral Shift Related to Temperature Differences.

Temperature differences have also been addressed by comparing cross section sets at temperature for select isotopes (Anderson, 2000). Figure 4 shows the spectral shift related to temperature differences between waste packages and CRCs. The bulk of the fuel-temperature effects in SNF appear to be accounted for by using the uranium “at-temperature” cross sections.

ISOTOPIC MODEL VALIDATION

The Isotopic Model is being validated using radiochemical assay (RCA) data for the benchmark and comparisons to the 2-D lattice physics code *CASMO* (Davis and Mays, 2002). The radiochemical assay data is applied in model validation to establish the bias in k_{eff} values predicted by the isotopic model and to establish the uncertainty in the principal isotope concentrations predicted by the isotopic model. The bias in k_{eff} values will be incorporated in CL values established for commercial SNF.

The bias in k_{eff} values are established by comparing reactivity calculations performed using measured isotopic concentrations from assay samples with calculations performed using calculated isotopic concentrations for the assay samples obtained from the isotopic model. Calculations for the assay samples are used in establishing the reactivity effects due to irradiation of the fuel samples in a commercial reactor. The length of the irradiation time affects the reactivity of the fuel sample because of changes in uranium isotopic concentrations and the buildup of higher actinides and fission products. These changes in isotopic concentrations will, in general, increase with increasing irradiation time. The bias in k_{eff} established for the isotopic model is based on the capability of the model to predict the changes in the isotopic concentrations with increasing irradiation time (burnup) in the commercial reactor.

The isotopic concentrations from the assay samples will be compared to the isotopic concentrations calculated using the isotopic model. The uncertainty in the calculated concentrations will be established for each of the principal isotopes. In addition, the time-integrated effect of uncertainties in half-lives and branching fractions are quantified in terms of reactivity (CRWMS M&O 1999).

The analyses of the assay samples will use burnup history parameters that are based on three-dimensional neutron diffusion-depletion analyses. Some of the three-dimensional analyses will be based on core-follow calculations where the fuel assembly that contains the assay sample is followed through its entire irradiation history in the core. This level of detailed core-follow data is not available for some of the assay samples. However, the operating history data that is available will be used to reconstruct burnup history parameters based on representative three-dimensional diffusion-depletion calculations. Sensitivity analyses will then be performed to provide an estimate of the uncertainty introduced by the use of reconstructed burnup history parameters such as demonstrated in O'Leary and Scaglione (2001) and O'Leary and Pitts (2000). The process for reconstructing burnup history parameters and establishing the uncertainty introduced by the process will be documented in the validation report for the isotopic model.

Radiochemical assay samples are generally taken from a single fuel pellet in a burned fuel assembly. This fuel pellet may not be representative of the many fuel pellets contained in the fuel assembly. Thus, the one-dimensional neutron transport-depletion model will contain additional uncertainty because of the limited capability to represent individual fuel pellets and the neutron spectrum associated with fuel pellet samples. Limitations in the capability of the one-dimensional model will be addressed through the use of a two-dimensional neutron transport-depletion model as discussed in Henderson (2000). Sensitivity analyses will be performed to estimate the uncertainty associated with the approximations made in the one-dimensional model. The uncertainties established from the approximations in the burnup history parameters and the approximations in the one-dimensional model will be compared to the uncertainty established from analysis of the radiochemical assay data with the one-dimensional neutron transport-depletion model. These uncertainties will be documented in the validation report for the isotopic model.

Since the Isotopic Model is inherent in CRCs, a process to reconcile the uncertainty in k_{eff} in the integral benchmarks for burned fuel (CRCs) with the measured isotopic data to ensure that the CL accounts for uncertainties indicated by local isotopic concentration measurements is discussed in Kochendarfer and Henderson (2000). This is so that the uncertainties associated with the bias are not double counted in establishing the CL.

APPLICATIONS

Results from many example applications of the disposal burnup credit models have been presented during the development of the disposal criticality analysis methodology. The example applications of the models have been used to test the models and designs during their development. The most informative example applications of the burnup credit models are loading curves. An example-loading curve is shown in Figure 5. Current plans call for generating loading curves for the 21 PWR waste package design with the disposal burnup credit models starting late in 2002.

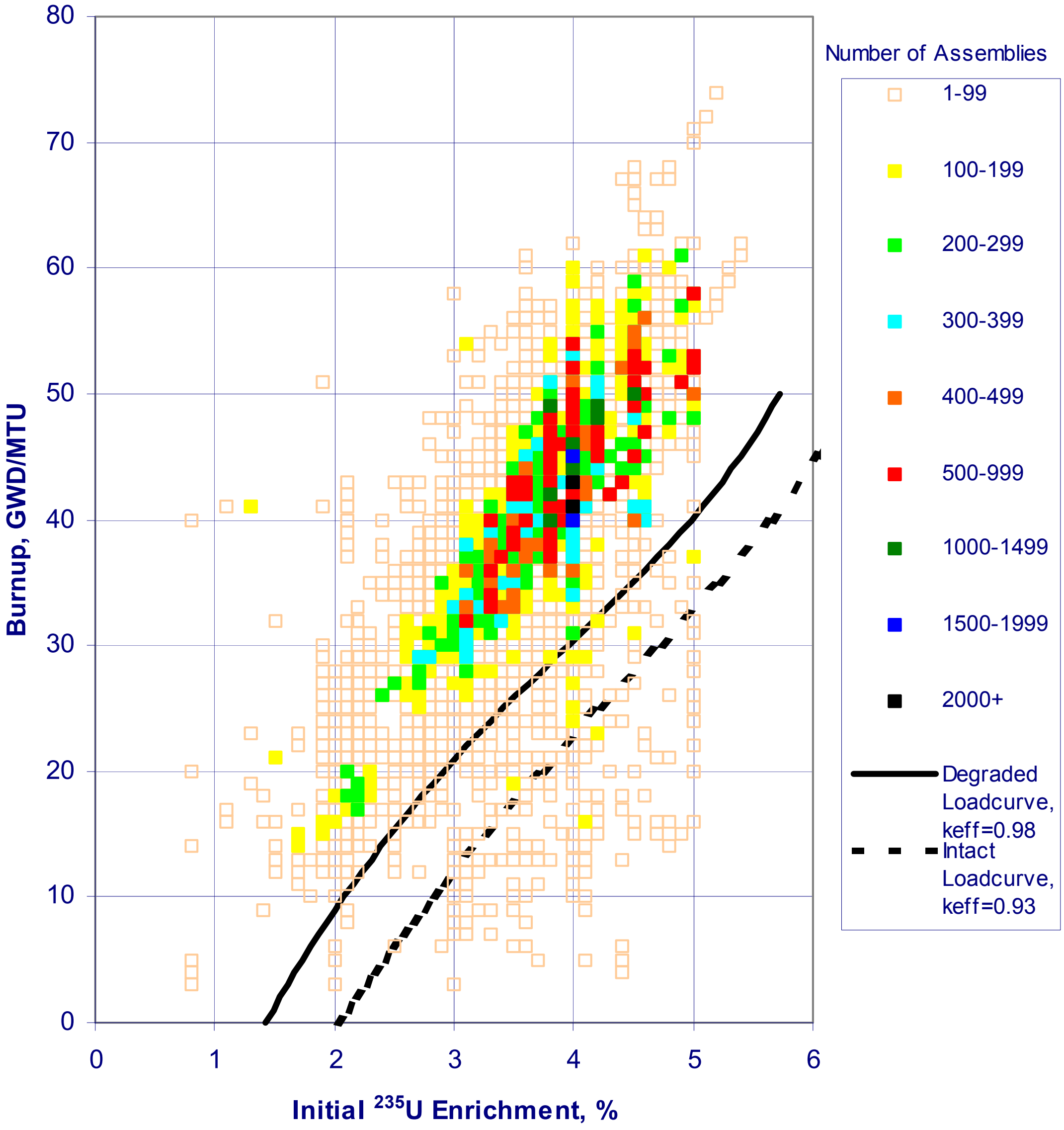

FIG. 5. Example Loading Curve for the 21 PWR Waste Package

SUMMARY

The DOE plans for criticality analyses on the Yucca Mountain Project to include Principal Isotopes Burnup Credit. The DOE has a significant effort underway in 2002 developing disposal burnup credit model validation reports. The criticality and isotopic model validation reports for PWR SNF are planned to be completed in 2002. The model validation reports for BWR SNF are planned for 2003. Applications of the model validation reports to generate loading curves for a 21 PWR waste package is also planned to start in late 2002.

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Consideration of burnup in criticality analyses for long term storage and final disposal of spent nuclear fuel in Germany

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Abstract. For the back-end of the nuclear fuel cycle in Germany long term interim storage before final disposal is the concept favored by the federal government and actually pursued by the utilities. Under the viewpoint of increasing initial enrichment of advanced LWR fuel the application of burnup credit is of interest not only for reactor pond storage but also for dry interim storage using for example dual-purpose casks. Furthermore fuel burnup is taken into account in current criticality analyses for final disposal. This paper addresses basic issues, the actual status and future tasks of considering burnup for long term interim storage and final disposal of spent nuclear fuel.

1. INTRODUCTION

The present situation of the back end of the nuclear fuel cycle in Germany is primarily characterized by increasing interim storage capacities. One of the federal government's priority objectives on the field of nuclear energy is minimizing the number and frequency of shipments of spent nuclear fuel on public roads. In this regard the federal government and the utilities agreed in long term interim storage of spent fuel at the reactor sites before final disposal. Licensing procedures for new storage facilities at the NPP sites are underway. The current contracts for reprocessing up to 2005 will be fulfilled. The time period for interim storage of spent fuel may increase due to the fact, that the program and the targets for final disposal will be reconsidered. On the investigation program of the salt rock formation of gorleben, which has been pre-selected for final disposal already in the eighties, a moratorium was issued and a new site selection process was launched. An expert group has been commissioned to develop criteria for a scientific based evaluation of the suitability of different rock formations for final disposal of spent nuclear fuel. In this context long term safety analyses including criticality studies for final disposal will be performed for different host rock types for providing additional information to the site selection process. There are three areas arising from this situation, where burnup credit (BUC) for criticality safety may be applicable and useful: (1) wet storage in pools, (2) dry cask interim storage and (3) long term criticality analyses of final disposal. Item (1) burnup credit in wet pool storage is being applied presently and has been discussed earlier in detail [1]. Some aspects and examples of burnup credit applications to items (2) and (3) are discussed below.

2. BURNUP CREDIT IN LONG TERM STORAGE

2.1. Basic considerations

Most of the interim storage facilities, which are actually planned on site of German nuclear power plants are designed for dry cask storage. The basic safety requirements including criticality safety of the stored fuel are to be observed by the cask, which is used for storage. On the one hand the increasing enrichment of $\geq$ 4 % of modern LWR fuel assemblies makes it more difficult to proof criticality safety under the 'fresh fuel assumption' and requires additional measures like insertion of fixed neutron absorbers or redesigning of container baskets. On the other hand the 'margin of conservatism' increases with the initial enrichment for assuming fresh fuel in criticality analysis. As the back-end is already the most costly part of the nuclear fuel cycle, there will be an intention from the industry side for using BUC in dry cask storage. The possible benefits are

- Accommodation of fuel with higher initial enrichment than the cask/basket was initially designed for without changing the system;
- Increase of the storage capacity for lower enriched fuel;
- Avoiding extensive and costly use of fixed neutron absorbers in the basket design.

In general dual-purpose casks will be used for transport and interim storage. If burnup credit is approved for transport of a cask and fuel load, it can also be used for storage. The requirements for the criticality safety proof for storage are the same as for transport. The criticality safety criterion keff ≤ 0.95 has to be observed for accidental flooding condition. For the dry case the keff value usually is < 0.8. The licenses for dry storage facilities actually are limited to 40 years. If in the future long term storage for about 100 years applying BUC is envisaged the reactivity change for this cooling time must be considered. Taking into account only U and Pu isotopes LWR fuel shows a decrease in reactivity during this time period, due to the decay of Pu-241. With increasing burnup the residual content of U-235 decreases while the fraction of Pu isotopes increases. This may require additional safety margins or efforts for validation of criticality codes for Pu systems. The MOX pin cell calculations of the OECD BUC benchmark iv-a for example show a spread of up to 2.5 % of the results of participants [2]. This spread arises from criticality codes alone, without uncertainty from inventory calculations.

One should be aware also that in dry cask storage besides criticality safety further requirements e. g. limitation of cladding temperature and surface dose rates for neutron and gamma radiation must be observed and may narrow the possible range of benefits from burnup credit.

2.2. Regulatory Issues and Status of Application

Safety guidelines for dry cask storage have been written by a working group of the German Reactor Safety Commission (RSK), which is the principal advising committee of the Federal Minister for Environment, Nature Conservation and Reactor Safety (BMU). The guidelines were issued in 2001 by BMU [3]. Herein the use of burnup credit for dry cask storage is basically allowed. If BUC is applied, the minimum required burnup has to be verified by measurement for every fuel assembly before being loaded into the cask. Further details of BUC application will be ruled in a technical DIN standard on burnup consideration for dry storage and transport casks, which is currently in preparation.

To date BUC in dry cask storage in Germany has been approved in particular cases e. g. for the central storage facility Gorleben for use of the cask type CASTROR V designed for fuels of initial enrichment higher than 4 %. For UO_2 PWR fuel of initial enrichment above 4.05 wt-% a minimum burnup of 10 GWd/tHM is required. For BWR fuel of initial enrichment higher than 4.05 wt-% and burnable neutron poison a minimum burnup of 5 GWd/tHM is required. For criticality analyses only actinides were considered. Before cask loading a verification of the burnup by measurement is required. According to the prescriptions of the licensing procedure the application of burnup credit, if applied, has to be examined and approved in connection with the cask permission. Further approvals of BUC in dry cask storage are to be expected from the current licensing procedures for storage facilities at the NPP sites.

3. BURNUP CONSIDERATION IN CRITICALITY ANALYSES FOR FINAL DISPOSAL

3.1. Criticality safety approach for final disposal

According to the IAEA Safety Principles and Technical Criteria for Underground Disposal of High Level Radioactive Waste [4] provisions shall be met for future human health and the environment on the base of what is acceptable today. For disposal of fissile material the maintenance of sub-criticality is required. Criticality analyses are performed for the operational phase and the post closure phase of the facility. During the operational phase criticality safety is achieved by appropriate design of the disposal containers and waste packages. For example the multipurpose cask POLLUX designed for storage, transport and disposal of spent LWR fuel has to comply with all basic safety requirements including criticality safety in the licensing procedure. For the current design of the disposal canister POLLUX no burnup credit was applied. The cask is designed for accommodating consolidated rods of up to 10 disassembled PWR fuel assemblies. Borated steel is used as neutron absorber in the inner structural components [5]. The calculated k-eff value is less than 0.92 for assuming fresh UO_2 and MOX fuel of 4.5 % enrichment respectively 4.2 % fissile Pu under flooding conditions. For fuel of higher fissile content the application of limited BUC may be necessary.

For the post closure phase of the repository is assumed, that the canisters will loose their safety function under the influence of geological and geo-chemical conditions and the arrangement and distribution of the fuel loads will change. The goal of the criticality analysis for the post closure phase is to demonstrate, that under the probable site-specific performance of the waste packages and the disposal mine no critical system will be formed. For long-term criticality studies scenarios describing the possible behaviour and degradation of the waste packages in the geologic environment are screened to identify the formation of potential critical configurations of disposed fissile material. Typical scenarios are for example

- Distortion and fracture of containers caused by external pressure;
- Intrusion of water or brine into containers;
- Local accumulation of the inventory of several containers;
- Leaching, migration and precipitation of fissile material by hydro-geologic processes.

The last type of scenario may support selective accumulation of different nuclides e.g. Uranium and Plutonium due to different geo-chemical behaviour. A more detailed description of long-term scenarios is given in the paper presented at the ANS NCSD Meeting in Reno [6]. For the configurations of fissile materials arising from the scenarios, criticality is investigated. The following aspects argue for considering burnup in long-term criticality analyses:

(1) The depletion of the fissile content is favorable for the intended proof of sub-criticality in the post closure phase. As a rule the fissile content (U-235 + Pu_{fiss}) of finally burned LWR fuel is about 1.5 wt-% for UO_2 and 2 wt-% for MOX fuel.

(2) For long term scenarios including the degradation and leaching of fuel material a realistic assumption for the major actinide composition must be applied to take into account the content of Plutonium, which has been built during burnup.

(3) Possibly in some cases stable and long-lived fission products may be taken into account if their physical presence can be assured for very long time.

3.2. Burnup Consideration in Current Disposal Criticality Studies

The studies done so far were directed to evaluate the reactivity (loss) of burned fuel and the change of reactivity with cooling time. A time period of 1 million years was considered. Criticality calculations were mainly based on U and Pu isotopic compositions determined from burnup and decay calculations. Comparative studies were performed to evaluate the principal magnitude of burnup and decay time reactivity effects as well as the influencing parameters and to compare this effect with reactivity changes arising from other parameters relevant for the long term behaviour of a geologic disposal, for example disintegration of canisters and fuel rods or increased moderation.

The curves in Figure 1 represents the calculated multiplication factor k-inf of spent fuel in a water moderated pin cell. The criticality calculations are based on calculated isotopic compositions of Uranium, Plutonium and Americium, additionally Np-237 and Th-232 were taken into account. The curves show a typical variation k-inf with cooling time arising from the decay of the isotopes Pu-241 and Pu-240. The decay of Pu-241 causes a decrease for the first 100 years and the decay of Pu-240 is responsible for the subsequent increase. The decrease beyond 10 million years is determined by the U-235 decay. The variation of k-inf with decay time is more significant for fuel compositions with a higher fraction of Plutonium e.g. MOX fuel and high burned UO_2 fuel. It should be pointed out, that the pin cell calculations were performed for a fixed moderation ratio $V_{mod}/V_{fuel} = 1.8$. As the fissile nuclide composition varies from U+Pu to U with cooling time, the optimum moderation ratio is shifted towards a lower value. Therefore in a more accurate investigation for each time step the k-value corresponding to the optimum moderation ratio shall be determined. Figure 2 compares the reactivity changes arising from cooling time and variation of moderation. The calculation model was a pin cell of burned fuel and water. The calculated burnup was 40 GWd/tHM. The moderation ratio of 0.3 corresponds to hexagonal close package of pins. The effect of moderation on k is significantly larger than that of the decay of Plutonium. If low moderation can be achieved for the post closure phase for example by consolidating the fuel rods before disposal and backfilling free space in the disposal container long term criticality safety can be increased substantially.

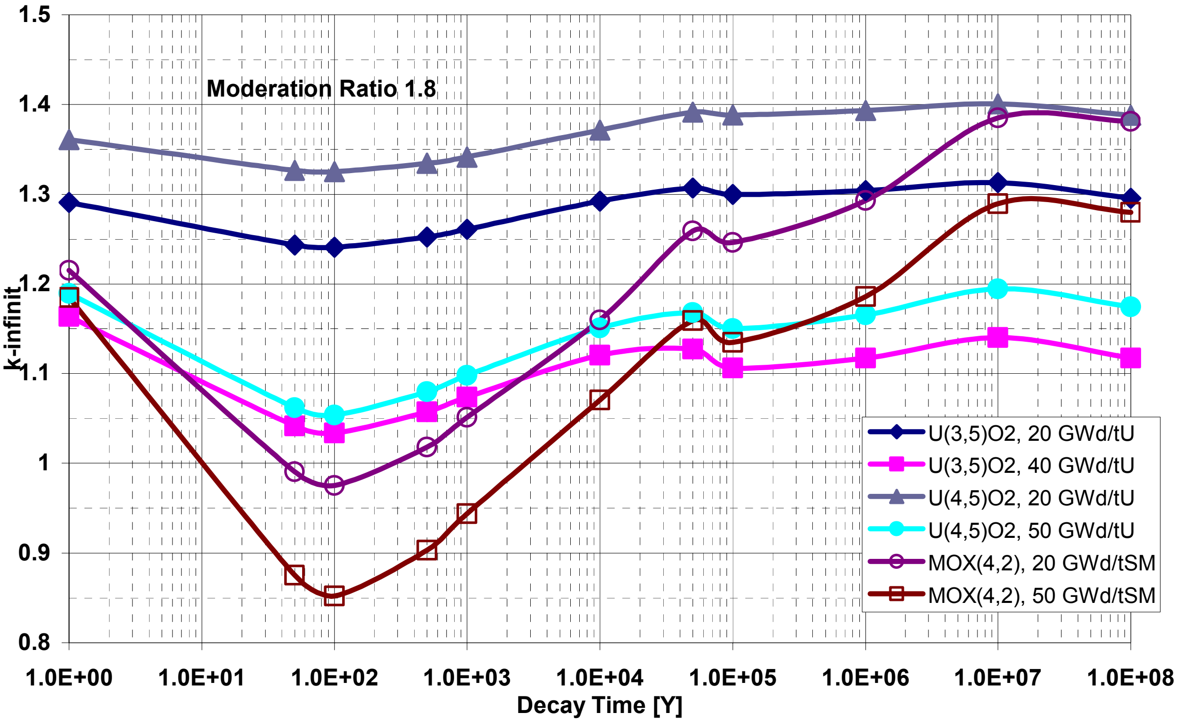

FIG. 1. Variation of k-inf of spent fuel with cooling time. Calculations performed for a water moderated pin cell for different types and burnup of spent fuel.

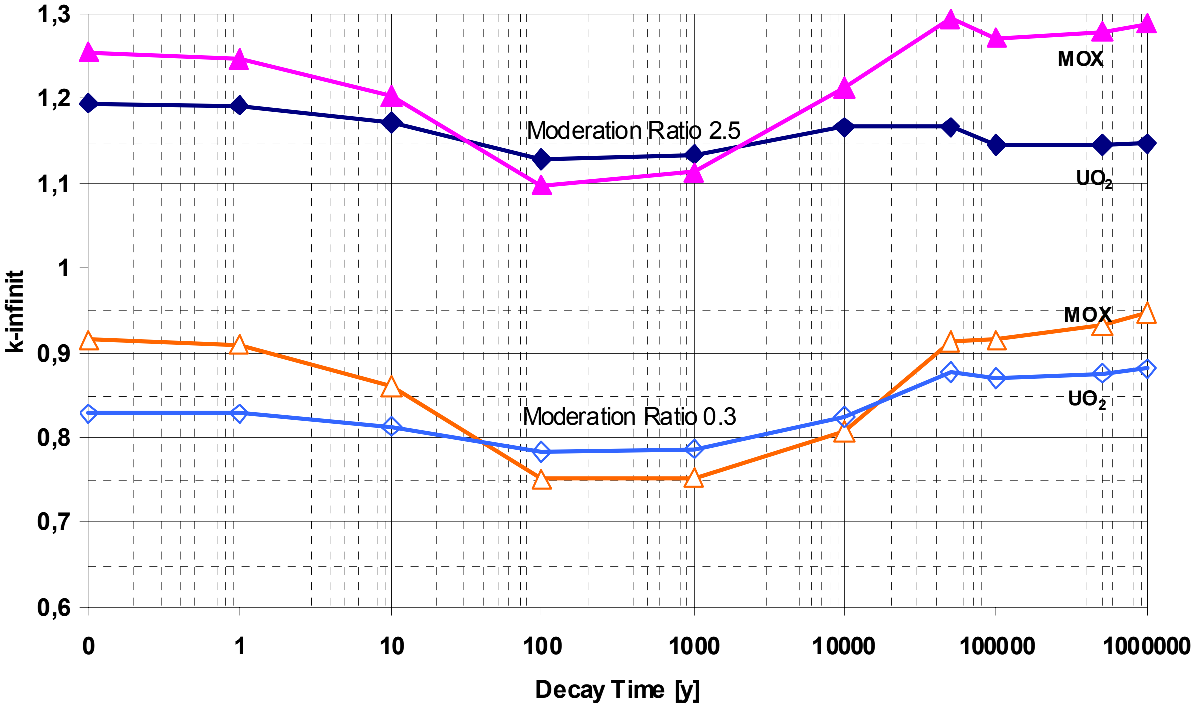

FIG. 2: Pin cell calculations of spent fuel of 40 GWd/tHM in water in dependence of the cooling time for two different moderation values. For UO_2 3.6 wt.% initial enrichment was assumed, for MOX 3.7 wt.% Pu-fiss.

From the studies so far can also be concluded, that under the criticality safety point of view salt rock has an advantage compared to other rock formations [6]. Due to the neutron absorption of Cl-35 criticality can not occur for any water moderated mixture of spent fuel and salt rock if at least the salt concentration of saturated brine is sustained and the fissile content in the fuel is not higher than 2 wt-%. This is true for burned commercial LWR fuel. For fuel with higher fissile content a concept for isotopic dilution with depleted Uranium may be developed [7].

3.3. Future Tasks

According to the present situation the following tasks are to be addressed.

1. Presently the selection of a disposal site is reconsidered. Salt rock as well as granite and further available rock formations suitable for deep geologic disposal are taken into account. Therefore long term criticality studies for different potential host rock formation shall be performed considering burnup for the spent fuel.
2. For burnup considerations a systematic approach must be developed. The objective is to define representative and sufficient conservative sets of spent fuel composition as a base for criticality calculations. For this purpose further and more detailed parameter studies on burnup and very long cooling times will be necessary.
3. The GRS developed calculation tool OREST for burnup and decay calculations has been improved [8]. An updated version KENOREST 01 is available and is actually in an extensive test and validation process. Further planned improvements concern the flexibility of the application range (e. g. WWER, MTR fuel types), the processing and evaluation of output data and the update of cross-section libraries.

4. CONCLUSION

The consideration of burnup in criticality analysis may be applied to dry cask storage as well as in long term analyses for final disposal of spent fuel. The benefits in dry cask storage are increased storage capacity and cost saving in fabrication of casks. The risk of inadvertent mix-up of fuel assemblies in cask loading shall be minimized according to the double contingency principle. A verification of the minimum required burnup by measurement for every fuel assembly before loading into the cask is prescribed in the German regulations. In criticality analyses for final disposal the fuel burnup is considered under two aspects, firstly the depletion of the fissile content, which is important for demonstrating long term sub-criticality, secondly the build up of Plutonium, which may affect the long term sub-criticality under certain scenarios. The variation of spent fuel reactivity for very long cooling times is significant if the Plutonium content is higher than about 1 wt-% as for MOX and highly burned UO_2 fuel, but in general the effect is noticeable smaller than that possibly caused by geologic and hydro-geologic processes. Tasks for the future are developing a systematic approach for burnup consideration in long term criticality analyses and further validation of the inventory calculation tool.

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BUC FOR MOX AND ADVANCED FUEL DESIGNS

(Session 2.7)

Burnup credit methodology for UO_2 and MOX fuel assemblies in AREVA/COGEMA

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Abstract. For the last five years, Areva/Cogema has been implementing the second phase of its burnup credit programme (including fission products). Since the early nineties, major actinides have been taken into account in criticality analyses. Next year (2003), COGEMA will be making allowance for the six main fission products (^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{152}Sm and ^{155}Gd) that contribute to 50% of the negative reactivity of all fission products. The experimental programme is nearing completion. The new burnup credit methodology is in progress. After giving a general overview, this paper will focus on the new burnup measurement system for MOX fuel assemblies. It will describe the measurement instrumentation and the measurement experiment that is planned for early 2003 at La Hague.

1. CONTEXT OF BURNUP CREDIT (BUC)

For over ten years, burnup credit (actinides only) has been used at the Cogema plant in La Hague for spent fuel transport, interim pool storage and reprocessing. More than five years ago, an intensive research programme was begun jointly with the French Atomic Energy Commission (CEA) and the French Institute for Radiological Protection and Nuclear Safety (IRSN). This research programme will make it possible to take the major actinide neutron absorbers into account in criticality studies. These are listed in Table II below. The research as a whole will be used to qualify the new criticality package known as Cristal.

After an introduction (chapter 2), the first part (chapter 3) of this article describes the research programme, which is nearing completion, and the second part (chapter 4) gives details of the new method used to interpret burnup measurements, which will be applied to both UO_2 and MOX fuel assemblies.

2. EXPECTED GAINS IN BUC

For a 17x17 PWR type fuel assembly with a cooling time of five years, Table I below shows that the five major actinide neutron absorbers currently taken into account in safety analyses account for approximately 60% of the total negative reactivity of all actinides (i.e. 19,000 pcm for a burnup (BU) of 40 GWd/tU).

Table I: Negative reactivity values in pcm

BU	20 GWd/tU	40 GWd/tU
^{236}U	720	910
^{238}Pu	60	310
^{240}Pu	5720	8370
^{242}Pu	160	710
^{241}Am	610	1290
Actinide total	10,000	19,000
5 Actinides	*7270 (73%)*	*11,590 (61%)*

$$1\ pcm = \frac{Keff - 1}{Keff}.10^5$$

The fifteen major fission products (listed below in Table II) account for around 80% of the total negative reactivity of all fission products, i.e. 9000 pcm for a burnup of 40 GWd/tU. The first six fission products alone account for half the total negative reactivity of all the fission products.

Table II: Negative reactivity values in pcm

BU	20 GWd/tU	40 GWd/tU
^{149}Sm	980	1030
^{103}Rh	790	1360
^{143}Nd	530	900
^{133}Cs	420	750
^{155}Gd	390	1550
^{152}Sm	250	490
^{151}Sm	350	500
^{99}Tc	240	440
^{145}Nd	230	410
^{153}Eu	150	390
^{95}Mo	150	290
^{147}Sm	150	230
^{150}Sm	120	270
^{109}Ag	100	250
^{101}Ru	100	220
Total for 200 fission products	6120	11,500
Total for 6 fission products	*3460 (56%)*	*6090 (53%)*
Total for 15 fission products	*4950 (81%)*	*9080 (79%)*

3. FINALISING THE CRITICALITY R&D PROGRAMME

3.1. Minerve experimental and qualification programme

The aim of this programme is to:

1. Measure the effective capture cross-sections of the 15 major fission products in the Minerve reactor operated by the French Atomic Energy Commission at Cadarache. These measurements are made by oscillations of individual samples of the fission products in the reactor. Oscillations of spent UOX and MOX fuel samples is also under way. These measurements are being carried out by the SPEX Department to qualify the APOLLO2 Sn option of the Cristal package ;
2. Measure and analyse samples of spent fuel from French and German pressurised water reactors which will allow the SPRC Department of the Atomic Energy Commission at Cadarache to qualify the Darwin fuel cycle package and the Cesar code (see References [1] and [2]).

Some results have been published (see Reference [3]).

3.2. VALDUC experiments and interpretation

The programme involves experiments and qualification of the French criticality package CRISTAL.

The experiments are carried out by IRSN at Valduc. These experiments represent a sub-critical approach involving raising of the water level. Various configurations are studied, ranging from individual fission products to more general experiments on mixtures of products (see Reference [4]).

The experiments are interpreted by the Criticality Study Department of IRSN at Fontenay-aux-Roses with a view to qualifying the industrial version (Apollo-Moret) of Cristal (see Reference [5]).

3.3. CRISTAL, the French criticality package

CRISTAL comprises three calculation routes:

1. The Monte Carlo route with the Tripoli code developed by the CEA's Department for the Study of Reactors and Advanced Modelling (SERMA) in Saclay;
2. The Apollo 2 SN route that is followed to make 1D and 2D calculations (calculations of standards); it is qualified by the CEA's Department of Reactor Physics in Cadarache;
3. The industrial route with coupling between the Apollo 2 code used to assess multi-group cross-sections in infinite surface areas and the Moret 4 code used to calculate complete 3D geometries using the Monte Carlo method but with cross-sections calculated by Apollo, which requires a considerably shorter calculation time than the direct Monte Carlo method. This route has been qualified by the Criticality Study Department of the CEA, in Fontenay-aux-Roses.

Figure 3 below shows the various codes and sequences making up the CRISTAL package.

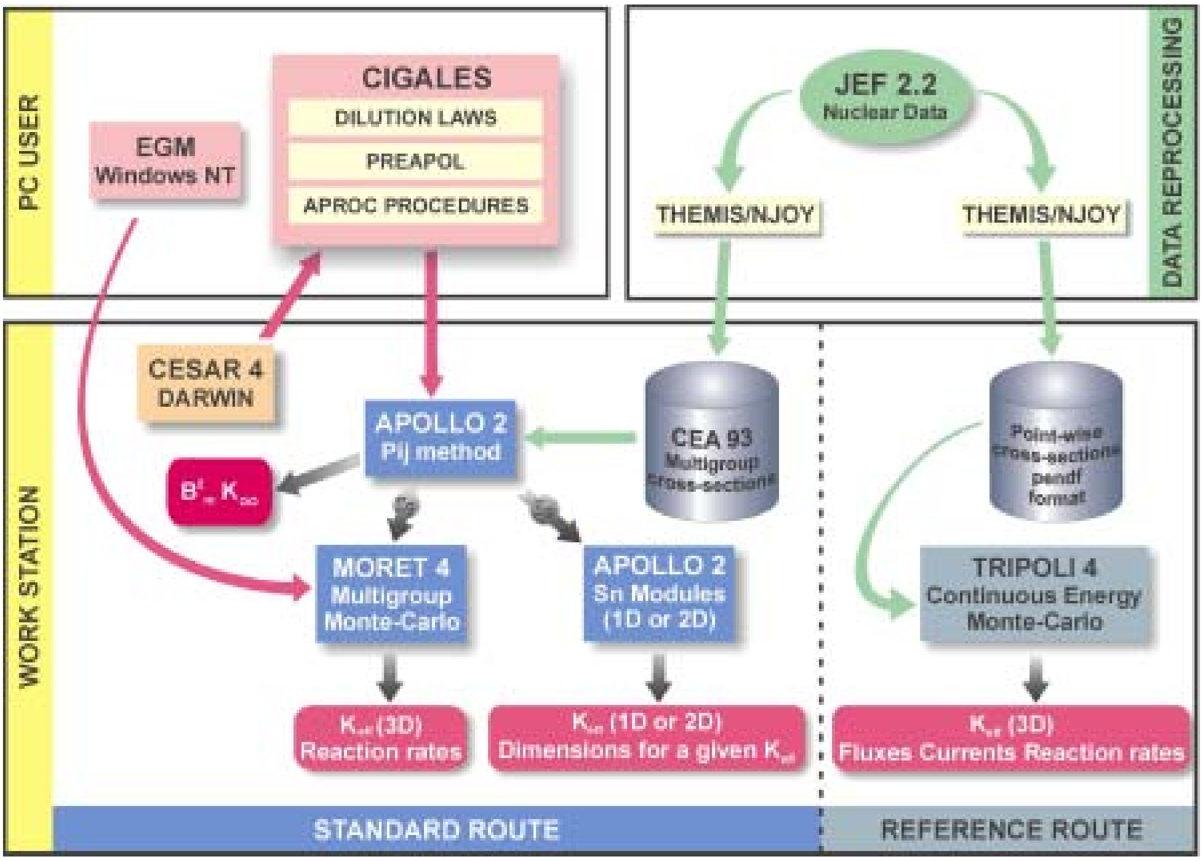
PC USER
EGM
Windows NT
CIGALES
DILUTION LAWS
PREAPOL
APROC PROCEDURES
JEF 2.2
Nuclear Data
THEMIS/NJOY
THEMIS/NJOY
DATA REPROCESSING
WORK STATION
CESAR 4
DARWIN
APOLLO 2
Pij method
CEA 93
Multigroup
cross-sections
Point-wise
cross-sections
pendf
format
B^2_m K_∞
MORET 4
Multigroup
Monte-Carlo
APOLLO 2
Sn Modules
(1D or 2D)
TRIPOLI 4
Continuous Energy
Monte-Carlo
K_{eff} (3D)
Reaction rates
K_{eff} (1D or 2D)
Dimensions for a given K_{eff}
K_{eff} (3D)
Fluxes Currents Reaction rates
STANDARD ROUTE
REFERENCE ROUTE

Figure 3

4. A NEW BU MEASUREMENT FOR UO_2 AND MOX FUELS

4.1. Industrial and development objectives

COGEMA currently uses two methods for measuring burnup, one based on gamma spectrometry and the other on passive neutron measurement. Both measurement systems are located in the shearing and dissolution facilities of both UP2 and UP3 reprocessing plant at La Hague.

There is also a portable system known as Python which uses a single neutron measurement and a total gamma measurement to obtain a burnup profile along the axis of a fuel assembly. This system has been installed in various reactor spent fuel pools in Europe, making it possible to use burnup credit for the transport of spent fuel casks and their interim storage in the COGEMA pools at La Hague. It does not involve gamma spectrometry since the technique currently used at La Hague requires a germanium detector with a liquid nitrogen cooling system that cannot easily be installed underwater in a reactor pool.

These two systems can be used to measure the burnup of UO_2 fuel assemblies but not that of MOX fuel assemblies.

The studies currently under way are therefore aimed at developing a system that could be used to measure the burnup of UO_2 and MOX fuel assemblies as simply as possible, without the need for a liquid nitrogen cooling system.

Furthermore, the combination of neutron measurements and gamma spectrometry would simply involve modifying the software of the existing systems at La Hague, without any effect on the hardware.

4.2. Measurement system

In view of the industrial objectives mentioned previously, preference was given to a system that combined gamma spectrometry and neutron measurements and yet remained portable. There was no alternative but to use Cd-Zn-Te detectors which operate at room temperature. A considerable amount of development work was required nonetheless to optimise these detectors and the associated electronics for this type of application, since they are currently used almost exclusively in the field of nuclear medicine to measure low energy rays at low count rates.

The detectors were extended, making it possible to measure all the gamma rays of interest (661 keV for ^{137}Cs, 796 keV for ^{134}Cs and most importantly 1274 keV for ^{154}Eu). The electronics were optimised to handle count rates of 50,000 counts/second. Dual parameter analysis (energy and pulse shape) results in improved resolution and makes it possible to separate the ^{154}Eu line from those of ^{60}Co at 1173 keV and 1332 keV. To date, a resolution of the order of 1.5% has been obtained, i.e. 10 keV at 661 keV and 20 keV at 1.4 MeV.

4.3. Interpretation of the measurements

A considerable amount of work has also been carried out as regards analysis of the measurement results. A combined measurement analysis algorithm has been developed. It comprises four major stages that occur after the fuel assemblies pass in front of the detector.

1. Blind interpretation to be able to know with no data on the fuel assembly measured whether the fuel is MOX or UO_2 fuel;
2. Validation of the average burnup of the fuel assembly by analysis of neutron measurement (use of certain data such as irradiation history, burnup of fuel cycles, initial isotopic composition);
3. Determination of burnup at one end of fuel assembly due to acquisition of the axial profile of ^{137}Cs;
4. Validation of data provided by reactor operator (use of gamma spectrometry ray ratio to detect any possible inconsistencies in the data provided).

The first stage ensures that MOX fuel assemblies are not mistaken for UO_2 ones. Indeed, for a given burnup, a MOX fuel assembly can have a far higher reactivity than a UO_2 fuel assembly. Likewise, the data validation module (Stage 4) guarantees that any significant error regarding the initial enrichment (or Pu content in the case of MOX fuel assemblies) will be detected. In addition to the average burnup validation stage (Stage 2), Stage 3 makes it possible to set a minimum burnup value, which must be higher than the burnup used for the criticality-safety study.

All of these stages use laws of correlation between a physical quantity (gamma peak ratio or neutron measurement) and burnup. An evolution code is therefore required and is used on-line. This code is known as CESAR (see Reference [1]). A large amount of research and development work has been carried out (especially for MOX fuel assemblies) to create neutron effective cross-section libraries that are used to accurately characterise entire fuel assemblies.

This new measurement system can be used to measure both UO_2 and MOX fuel assembly burnup and is far more accurate than existing burnup measurement systems thanks to its data validation module.

4.4. Qualification Programme

Since the R&D work is nearing completion, a programme has been devised for qualifying the system.

Hot testing of MOX fuel assembly burnup measurement is scheduled to take place early 2003 in the COGEMA plant at La Hague.

Transversal section of the burn-up measurement well

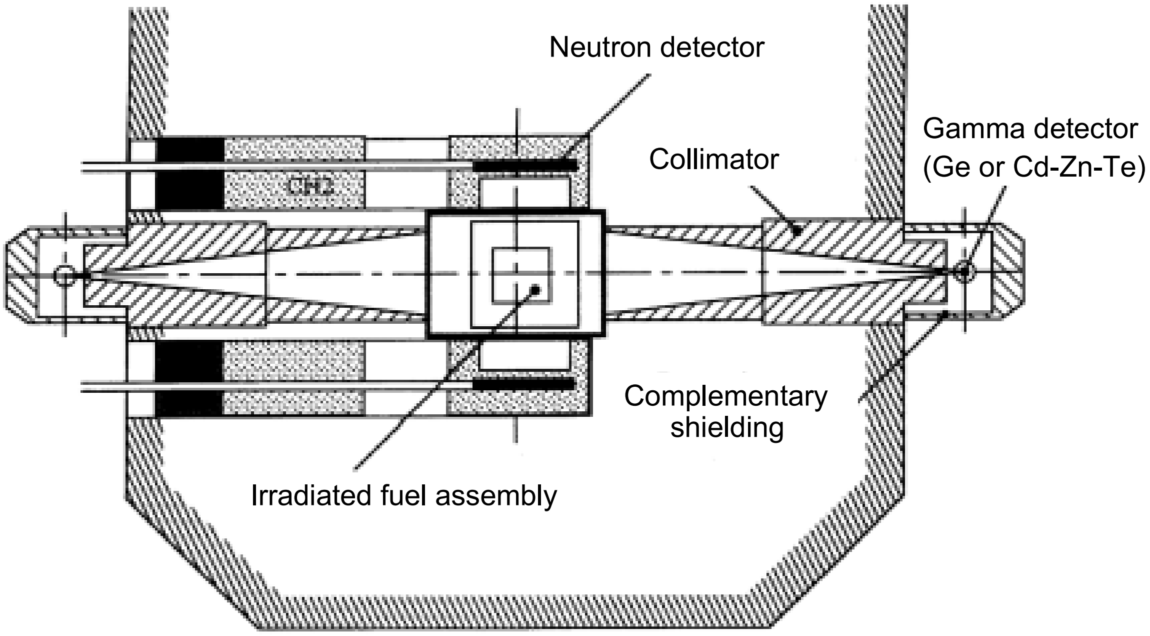

Figure 4

Figure 4 shows the positions of the detectors on the burnup measurement stations in the shearing and dissolution facilities in COGEMA's UP2 and UP3 plants at La Hague.

The COGEMA site at La Hague also has a database containing exploitable gamma spectrometry and neutron measurements for over 10,000 fuel assemblies, dating back to over ten years ago. In 2003, these measurements will be re-analysed using the new algorithm (see Section 4.3).

This should mean that, in 2004, it will be possible to submit a report on the new UO_2 and MOX fuel assembly measurement system to the French safety authorities for approval.

5. CONCLUSION

The R&D work being carried out by COGEMA on the subject of burnup credit is now entering Phase II (with allowance being made for the major fission products as well as minor actinides). It is also intended to make allowance for the burnup credit of MOX fuel assemblies (actinides only to begin with). To this end, a new device for measuring the burnup of UO_2 and MOX fuel assemblies has been successfully developed. The system is now at the qualification stage.

All this R&D (in the fields of criticality and nuclear measurements) should ensure that COGEMA will be in a position to handle, in its La Hague facilities, the different types of fuel currently envisaged by electricity companies and fuel designers without any major problems.

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Cross-checking of the operator data used for burnup measurements

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Abstract. Taking into account of the loss of reactivity of fuels at the end of their irradiation is known under the term burnup credit (BUC). It is a question of dimensioning in a less penalizing way the devices of transport, storage or of processing with respect to the risk of criticality. In the context of nuclear criticality safety a better realism cannot be obtained at the price of conservatism. As a result the regulator requires measurements make it possible to validate the adequacy between real fuels and the design assumptions. The sophistication of the measures recommended by the regulator can vary from a simple control of the irradiation to a complete characterization of fuels. It is within the latter framework that this study fits of which the goal is to describe the methods by which some of the data necessary to interpretation of the measurements and provided by the operator could be validated. The interest is that if the method of measurement makes it possible to detect a basic data error, its effects on the result do not have to be integrated into the assessment of uncertainty of measurement.

This paper exposes the mechanisms making it possible to validate the data input used by calculations associated with interpretation in term with burnup rate with passive neutron measurements. In the majority of the cases, the time of cooling as well as the history of irradiation could be validated in a reliable way if the precision of measurements of gamma spectrometry makes it possible to consider the relationship isotopic with a precision of about 10%. It remains that in practice, the reliability of the methods described in this document depends primarily on the performances of the instrumentation, which will be used. A work of evaluation of these performances on the basis of characteristic of the instrumentation remains to be made for each project. A study of sensitivity based on sets of incorrect data but plausible will be carried out in order to specify the performances of the measuring instruments necessary to the detection of each anomaly. Contrary to the time of cooling and history of irradiation, the checking by gamma spectrometry of the initial composition of fuels MOX and UOX remains problematic because, on the one hand, the transmitters gamma are not sensitive to the initial isotope of the plutonium used in fuels, and on the other hand, the sensitivity to the initial content plutonium or initial enrichment in ^{235}U remains low. In any event, very accurate gamma spectrometry measurements should be used for checking initial contents. It is however significant to note that the initial composition can be checked independently of the operator and that thus the probability of an error is less than for the operating data.

INTRODUCTION

Taking into account of the loss of reactivity of fuels at the end of their irradiation is known under the term burnup credit (BUC). One distinguishes the BUC "actinide only" which takes into account only the disappearance of uranium and the formation of plutonium and from major actinides of the BUC "FP" which integrates in more the formation of 6 (or 15) fission product neutron absorbents. According to the nature of the objects and the type of burnup credit "actinide only" or "FP", the sophistication of the measures recommended by the regulator can vary from a simple control of the irradiation to a complete characterization of fuels. It is within the latter framework that this study fits of which the goal is to describe the methods by which some of the data necessary to interpretation of the measurements and provided by the operator could be validated. The interest is that if the method of measurement makes it possible to detect a basic data error, its effects on the result do not have to be integrated into the assessment of uncertainty of measurement. More directly the results of this study are intended to be integrated into the future system of characterization of fuels UOX and MOX intended for measurements of the burnup rate and the axial profile of burnup rate in reactor ponds. That being, this paper attempts to describe the whole of the techniques of validation available without prejudging their applicability in term of measuring accuracy according to technological choices' which will be carried out.

POSITIONING OF THE PROBLEM

The system evoked above must take measurements of burnup rate on assemblies UOX as well as MOX in reactor ponds over fuels between 10 and 60 GWd/t and a time of cooling ranging between 15 month and 20 years. The feasibility of this measurement was the subject of a feasibility study [1] whose results made it possible to release a methodology of measurement which is articulated around two instrumental techniques: a passive neutron counting which is interpreted in term of average burnup rate by taking of account initial enrichment, the time of cooling and the history of irradiation, a gamma spectrometry - at ambient temperature in the case of underwater measurements whose interpretation is used to establish the axial profile of burnup rate but also to validate the data provided by the operator for interpretation of neutron measurement.

The measurable isotopes by the gamma spectrometry are ^{144}Ce, ^{106}Ru, ^{137}Cs, ^{134}Cs and ^{154}Eu. According to the measurement technique used (spectrometry at ambient temperature (CdZnTe) or spectrometry of high resolution (HPGe)) and times of cooling considered, all these isotopes will not be measurable at the same time. This paper attempts to in general define the methods applicable to the validation of the data without prejudging conditions of measurements. It results from this that according to cases, certain isotopic reports/ratios will not be usable. For example and as indicates it Table I and Figure 1, the ^{106}Ru and ^{144}Ce will not be measurable with a spectrometry at ambient temperature after a long time of cooling. Same manner, the measurement of the ^{154}Eu seems difficult on little irradiated and little cooled fuels. Fortunately, each method rests on the use of several isotopic reports/ratios, which one can reasonably think that at least one will be measurable. The characteristics of the detectable isotopes by gamma spectrometry are recalled in Table 1.

Table I. Nomenclature and properties of the transmitting isotopes gamma

Isotope	Energy (keV)	Period (days)	Disintegration Yield (γ/Bq)	Approximate fraction of the total gamma emission from an irradiated MOX fuel (Initial Pu 5,62 %)		
				BU 48 GWd/t CT 450 d	BU 12 GWd/t CT 450 d	BU 60 GWd/t CT 5500d
^{144}Ce-^{144m}Pr	2185	285	0,7 %	0,3 %	0,7 %	$<10^{-4}$ %
^{106}Ru-^{106m}Rh	621	372,6	9,9 %	7,6 %	14,3 %	$<10^{-2}$ %
	1050		1,5 %	1 %	2 %	$<10^{-3}$ %
^{134}Cs	604	754	97,6 %	20,8 %	6,5 %	1,7 %
	796		85,5 %	20 %	6,2 %	1,7 %
	801		8,7 %			
^{137}Cs	662	11005	85,2 %	17,3 %	16,6 %	95,6 %
^{154}Eu	1004	3135	18,2 %	0,5 %	0,1 %	0,16 %
	1274		35 %	0,76 %	0,20 %	0,3 %

The following isotopic reports/ratios are usable: ^{134}Cs/^{137}Cs, ^{154}Eu/^{137}Cs and (^{106}Ru ^{137}Cs)/^{134}Cs^{2} for the determination of the burnup rate, ^{106}Ru/^{137}Cs and ^{144}Ce/^{137}Cs for the determination of short times of cooling, ^{134}Cs/^{154}Eu for the determination of long times of cooling.

EVALUATION OF UNCERTAINTIES

The determination of uncertainties is fundamental within the framework of the validation of the data input since they are directly related to the development of the decision criterion.

Uncertainties are of two types:

- uncertainties of measurements (instrumentation, interpretation of the spectra);
- uncertainties of calculations (precision of the code of evolution, and the code of transport).

Measuring accuracy

The spectrometry software strips the spectra and delivers isotopic reports/ratios together with an evaluation of the error of measurement.

Precision of depletion calculations

In the current state of knowledge on fuels UOX, it appears that significant variations exist between the measurements of references and the predictions carried out by calculations of evolution. The situation concerning fuels MOX is appreciably equivalent. Under these conditions, the taking into account of corrective factors calculation - experiment is inescapable. The difficulty comes owing to the fact that skews observed are strongly dependent on the burnup rate. The results of parametric calculations of evolution will have to thus take into account for each isotope a law of correction calculation – experiment.

PRINCIPLE OF THE METHOD OF VALIDATION

Correlations

Correlation with the time of cooling

During cooling the decrease of an isotope i is described by the relation:

$$A_i(CT) = A_i(CT_0) \cdot e^{-\lambda_i \cdot CT}$$

Because of the differences in periods, the various isotopic reports/ratios indicated above strongly vary according to the CT. Moreover, during the irradiation, ^{134}Cs and ^{154}Eu vary according to the square of BU whereas ^{144}Ce, ^{106}Ru and ^{137}Cs linearly vary according to TC. As a consequence, at the end of the irradiation ^{134}Cs/^{154}Eu, ^{144}Ce/^{137}Cs, ^{106}Ru/^{137}Cs are constants independent of TC. During cooling, the following relations describe the evolution of the isotopic reports/ratios independently of the burnup rate:

$$Ln\left(\frac{^{134}Cs}{^{154}Eu}\right)_{CT} = \left(\lambda_{^{154}Eu} - \lambda_{^{134}Cs}\right)CT + Ln\left(\frac{^{134}Cs}{^{154}Eu}\right)_{CT_0}$$

$$Ln\left(\frac{^{144}Ce}{^{137}Cs}\right)_{CT} = \left(\lambda_{^{137}Cs} - \lambda_{^{144}Ce}\right)CT + Ln\left(\frac{^{144}Ce}{^{137}Cs}\right)_{CT_0}$$

$$Ln\left(\frac{^{106}Ru}{^{137}Cs}\right)_{CT} = \left(\lambda_{^{137}Cs} - \lambda_{^{106}Ru}\right)CT + Ln\left(\frac{^{106}Ru}{^{137}Cs}\right)_{CT_0}$$

The term of slope (λi-λj) depends on the differences of the constants of disintegration and represented the variation of the report/ratio according to time of cooling based on different speeds of decrease. The term of ordinate in the beginning depends on the fractions relating between isotopes to the exit of the irradiation and thus of the history of irradiation.

Correlation with the burnup

Various isotopic reports/ratios indicated below vary according to BU the influence of the time of cooling are obviously not negligible but can be defined in an analytical way:

$$\frac{^{134}Cs}{^{137}Cs} = \left(\alpha_1 BU + \beta_1\right)\cdot e^{\left(\lambda_{^{137}Cs} - \lambda_{^{134}Cs}\right)CT}$$

$$\frac{^{154}Eu}{^{137}Cs} = \left(\alpha_2 BU^2 + \beta_2 BU + \delta\right)\cdot e^{\left(\lambda_{^{137}Cs} - \lambda_{^{154}Eu}\right)CT}$$

$$Ln\left(\frac{^{106}Ru\,^{137}Cs}{^{134}Cs^2}\right) = \left(\alpha_3 .Ln(BU) + \beta_3\right) - \left(\lambda_{^{106}Ru} + \lambda_{^{137}Cs} - 2\lambda_{^{134}Cs}\right).CT$$

Validation of the cooling time validation

Principe

On the basis of the initial composition (IC), the BU and the history of irradiation (HI) provided by the operator, the cooling time is validated using the correlations given in section 4.1

The general step is as follows:

- Taking into account of the data provided by the operator other than time of cooling (HI, IC, BU);
- Calculation of evolution parametric according to single variable CT;
- Adjustment of the coefficients of correlation taking into account of the coefficients of calculation-experiment correction;
- Interpretation of the measured isotopic reports/ratios and of their uncertainty;
- Calculation of correlated Cooling Times $CT_{Correlated}$

$$CT_1 = \frac{1}{a_1}\left(Ln\left(\frac{^{134}Cs}{^{154}Eu}\right) - b_1\right)$$

$$CT_2 = \frac{1}{a_2}\left(Ln\left(\frac{^{144}Ce}{^{137}Cs}\right) - b_2\right)$$

$$CT_3 = \frac{1}{a_3}\left(Ln\left(\frac{^{106}Ru}{^{137}Cs}\right) - b_3\right)$$

- and their uncertainties $\Delta CT_{Correlated:}$

$$\Delta CT_1 = \frac{1}{a_1}\frac{\Delta\frac{^{134}Cs}{^{154}Eu}}{\frac{^{134}Cs}{^{154}Eu}} \quad \Delta CT_2 = \frac{1}{a_2}\frac{\Delta\frac{^{144}Ce}{^{137}Cs}}{\frac{^{144}Ce}{^{137}Cs}} \quad \Delta CT_3 = \frac{1}{a_3}\frac{\Delta\frac{^{106}Ru}{^{137}Cs}}{\frac{^{106}Ru}{^{137}Cs}}$$

- Validation using the value of CTCorrelated associated with weakest CTCorrelated uncertainty according to the following relation then CTOperator is validated.

If $CT_{Operateur} - \Delta CT_{Correlated} \leq CT_{Correlated} \leq CT_{Operator} + \Delta CT_{Correlated}$ then CT_{Operator} is validated.

If the time of cooling obtained is coherent with the CT operator, the history of irradiation is then controlled.

Example

Let us consider as an example a fuel irradiated MOX (initial Pu 5,62%) to 60 GWd/t according to the history of irradiation of reference declared cooled during 2000 days and suppose that the time of real cooling is in fact 450 days. Under these conditions the interpretation of neutron measurement would lead to a burnup rate over-estimated of 7 %. The following Table recapitulates the results relating to the case BU.

Table II. Example of detection of an erroneous cooling time given by operator.(MOX Pu 5,62 %, BU 60 GWd/t)

	Calculated size Equations of correlation (CT in days)	TC	
Neutron emission (NE)	• $NE=1,8.10^{10}.e^{-0,0001.CT}+1,07.10^{10}.e^{-0,0042.CT}$	$NE_{450}=1,89.10^{10}$ n/s/t $NE_{2000}=1,47.10^{10}$ n/s/t	$^{\Delta NE}/_{NE}$ 22 %
	• $NE_{CT}\ True=1,32.10^{7}.TC^{1,699}$ • $NE_{CT\ Operator}=1,32.10^{7}.TC^{1,78}$	$TC_{CT\ True}$=60 GWd/t $TC_{CT\ Operator}$=64GWd/t	$^{\Delta TC}/_{TC}$= 7 %
Ln(^{134}Cs/^{154}Eu) Calculated	• Ln(^{134}Cs/^{154}Eu)=-0,000705.CT + 0,99	$CT_{Operator}$ =2000 days $CT_{Correlated}$= 425 days	$^{\Delta CT}/_{CTOperator}$=78 %
Ln(^{144}Ce/^{137}Cs) Calculated	• Ln(^{144}Ce/^{137}Cs)=-0,00239 .CT – 1,823	$CT_{Operator}$ =2000 days $CT_{Correlated}$= 442 days	$^{\Delta CT}/_{CTOperator}$=77 %
Ln(^{106}Ru/^{137}Cs) Calculated	• Ln(^{106}Ru/^{137}Cs)=-0,00182 .CT – 1,748	$CT_{Operator}$ =2000 days $CT_{Correlated}$= 442 days	$^{\Delta CT}/_{CTOperator}$=77 %

Validation criterion of CT : $CT_{operator} - \Delta CT_{correlated} \leq CT_{correlated} \leq CT_{operator} + \Delta CT_{correlated}$ |

The example developed above shows under which conditions could be detected a time of erroneous cooling leading to a skew of 7 % on the burnup rate resulting from the interpretation of neutron measurement. In this case, the detection of the time of cooling would be effective as long as the error of measurement (ΔCT) would be lower than 1558 days is a relative error of 350 %. The relative error associated the measurement of the isotopic report/ratio R expressed by a relation of the form:

$$Ln.r = a.CT + b \text{ is } \frac{\Delta r}{r} = a.\Delta CT$$

In the studied case, acceptable uncertainties of measurements are calculated in Table III.

Table III. Acceptable experimental errors to validate the time of cooling in the example (MOX FUEL, BU 60 GWd/t, CT(True) = 450 days, CT(Operator) = 2000 days)

Isotope ratio (r)	Maximal ΔCT to avoid erroneous anomaly detection	Correlation slope Ln(r)=a.CT + b	Relative measurement error acceptable on the isotope ratio r $^{\Delta r}/_{r}$
^{134}Cs/^{154}Eu	1575	0,000705	111%
^{144}Ce/^{137}Cs	1558	0,00239	372%
^{106}Ru/^{137}Cs	1558	0,00182	284%

The control of the irradiation history

Irradiation history is the sequence of the cycles of irradiation and the periods of inter cycle, which lead to a given burnup. By convention, the history of irradiation ends in the end of the last cycle of irradiation.

It does not appear possible to construct a whole of parametric calculations likely to cover the whole of the possible histories of irradiation. Consequently, the validation of the history can be only indirect. The method consists in using the fact that a significant variation of the history of irradiation does not have the same effect on the various correlations between isotopic reports/ratios and burnup rate.

The general step of detection of an erroneous history of irradiation is as follows:

- Taking into account of the data provided by the operator other than the burnup rate. The initial composition and the history of irradiation declared by the operator are used as well as the time of cooling checked beforehand;
- Parametric evolution calculations according to the single variable BU;
- Extraction of the results of calculations (isotopic reports/ratios mass) and taken into account of the coefficients of calculation-experiment correction;
- Adjustment of the coefficients α1, β1, α2, β2, δ2, α3, β3. of the correlation equations without the term of CT since this one validated beforehand is used for parametric calculation:

$$\frac{^{134}Cs}{^{137}Cs} = \alpha_1 BU_1 + \beta_1$$

$$\frac{^{154}Eu}{^{137}Cs} = \alpha_2 BU_2^2 + \beta_2 BU_2 + \delta_2$$

$$Ln\left(\frac{^{106}Ru\,^{137}Cs}{^{134}Cs^2}\right) = \alpha_3.Ln(BU_3) + \beta_3$$

- Interpretation of the calculation and results of measurement of the mass isotopic reports/ratios starting from measurements of gamma spectrometry to obtain the mass isotopic reports/ratios:

$$\frac{^{134}Cs}{^{137}Cs}_{mes}, \frac{^{154}Eu}{^{137}Cs}_{mes} \; and \; \frac{^{106}Ru\,^{137}Cs}{^{134}Cs^2}_{mes}$$

- Calculations of the burnup associated with the measured isotopic reports/ratios and the correlations. According to times of cooling and the instrumentation considered, the results of spectrometric measurements make it possible to determine at least two of three values BU_1, BU_2 and BU_3 such as:

$$BU_1 = \frac{1}{\alpha_1} \frac{{}^{134}Cs}{{}^{137}Cs}_{mes} - \frac{\beta_1}{\alpha_1}$$

$$BU_2 = -\frac{\beta_2 \pm \sqrt{\beta^2 - 4\alpha_2(\delta_2 - \frac{{}^{154}Eu}{{}^{137}Cs})}}{2\alpha_2}$$

$$BU_3 = e^{-\frac{1}{\alpha_3} Ln\left(\frac{{}^{106}Ru\,{}^{137}Cs}{{}^{134}Cs^2}_{mes}\right) - \frac{\beta_3}{\alpha_3}}$$

- Calculation of the uncertainties of measurement expressed in term of burnup ΔBU_i. The relative error associated the measurement of the isotopic report/ratio R expressed by a relation of the form is the values of absolute uncertainties on the burnup rates are thus, are checked: $Ln.r = \alpha.BU + \beta$ is $\frac{\Delta r}{r} = \alpha.\Delta Bu$, whereas that associated to a relation of the type $r = \alpha.BU^2 + \beta \cdot BU + \delta$ is $\frac{\Delta r}{r} = \frac{\Delta BU}{BU} \cdot \frac{2\alpha.BU^2 + \beta.BU}{\alpha.BU^2 + \beta.BU + \delta}$

  the values of absolute uncertainties on the burnup rates are thus:

$$\Delta Bu_1 = \frac{\Delta \frac{{}^{134}Cs}{{}^{137}Cs}}{\frac{{}^{134}Cs}{{}^{137}Cs}} \frac{\alpha_1 BU_1 + \beta_1}{\beta_1}$$

$$\Delta BU_2 = \frac{\Delta \frac{{}^{154}Eu}{{}^{137}Cs}}{\frac{{}^{154}Eu}{{}^{137}Cs}} \frac{\alpha_2 BU_2{}^2 + \beta_2 BU_2 + \delta_2}{2\alpha_2 BU_2 + \beta_2}$$

$$\Delta BU_3 = \frac{1}{\alpha_3} \frac{\Delta \frac{{}^{106}Ru\,{}^{137}Cs}{{}^{134}Cs^2}}{\frac{{}^{106}Ru\,{}^{137}Cs}{{}^{134}Cs^2}}$$

- Diagnostic: the history of irradiation is declared validated if the following inequalities are true ΔBU_i. indicates the uncertainties brought by measurement for each isotopic report/ratio.

$$|BU_1 - BU_2| \leq \Delta BU_1 + \Delta BU_2$$
$$|BU_2 - BU_3| \leq \Delta BU_2 + \Delta BU_3$$
$$|BU_1 - BU_3| \leq \Delta BU_1 + \Delta BU_3$$

Example

This paragraph through example shows how an inaccurate history of irradiation can be to detect thanks to method indicated above. It is first of all significant to note that only two very different histories of irradiation are likely to lead by the neutron method to values significantly different from the burnup rate.

The following Table makes it possible to illustrate this point with the comparison of the neutron emissions of 2 fuels MOX and UOX for the same burnup rate obtained during 4 different histories.

Table IV. Influence of the irradiation history on the emission neutron (BU 60 GWd/t, MOX Pu 5,62 % and UOX 4 % ^{235}U)

Irradiation History	BU by cycles	MOX fuel 5,62 % Pu CT 450 days		UOX fuel 4 % ^{235}U CT 450 days	
		Neutron Emission (n/s)	Relative variation	Neutron Emission (n/s)	Relative variation
Nominal History	Cycle 1 12GWd/t Cycle 2 12GWd/t Cycle 3 12GWd/t Cycle 4 12GWd/t Cycle 5 12GWd/t	$1,892.10^{10}$		$2,736.10^{9}$	
One cycle miss during the beginning of the fuel life	Cycle 1 15GWd/t Cycle 2 0 GWd/t Cycle 3 15GWd/t Cycle 4 15GWd/t Cycle 5 15GWd/t	$1,927.10^{10}$	1,85%	$2,746.10^{9}$	0,37%
One cycle miss during the end of the fuel life	Cycle 1 15GWd/t Cycle 2 15GWd/t Cycle 3 15GWd/t Cycle 4 0 GWd/t Cycle 5 15GWd/t	$1,898.10^{10}$	0,32%	$2,796.10^{9}$	2,39%
2 very low power cycle during the end of the fuel life	Cycle 1 20 GWd/t Cycle 2 20 GWd/t Cycle 3 18GWd/t Cycle 4 1GWd/t Cycle 5 1GWd/t	$1,703.10^{10}$	**-10%**	$2,476.10^{9}$	**-9,43%**

Table IV shows that only a history very different from the nominal history could lead to a value of erroneous burnup rate. It is noted in addition that concerning the neutron emission, the fuel MOX reaction of is very similar to that of fuels UOX.

Table V shows that the variations induced by the variation of history are much larger for certain isotopic reports/ratios than for others. It is also noted that the influence on the burnup rate measured by the method neutron remains limited to 8 %. The comparison of the BU values associated with the correlations "history disturbed" reveals significant variations from 11 to 27 GWd/t are 22 and 54 % respectively. Consequently, in this case, if the instrumentation makes it possible to measure the isotopic reports/ratios ^{134}Cs/^{137}Cs, ^{154}Eu/^{137}Cs, Ln((^{106}Ru. ^{137}Cs)/^{134}Cs^{2}) with a precision better than 10%, the detection of the inaccurate history of irradiation could be assured.

Table V. Influence of an extreme history (5 cycles including 2 cycles with very low power at the end of the lifetime) on the isotopic transmitters gamma and their reports/ratios

Calculated value	Correlation equations	BU (GWd/t)	
	• nominal History • perturbed History (2 cycles with low power during the end of fuel fife)		
Neutron Emission	• EN=1,3200.10^7.$BU^{1,780}$ (n/s/t) • EN=1,3289.10^7.$BU^{1,743}$ (n/s/t)	BU_{True}50 BU_1 54	 8 %
$^{134}Cs/^{137}Cs$	• $^{134}Cs/^{137}Cs$=0,00146 BU + 0,00159 • $^{134}Cs/^{137}Cs$=0,000924 BU + 0,00198	BU_{True} 50 BU_1 78	 57 %
$^{154}Eu/^{137}Cs$	• $^{154}Eu/^{137}Cs$=-1,002.10^{-5} BU^2 + 0,00143 BU – 0,00531 • $^{154}Eu/^{137}Cs$=-0,869.10^{-5} BU^2 + 0,00127BU – 0,00441	BU_{True} 50 BU_2 62	 24 %
Ln(($^{106}Ru.$ ^{137}Cs)/ $^{134}Cs^2$)	• Ln(($^{106}Ru.$ ^{137}Cs)/ $^{134}Cs^2$)=-1,844.Ln(BU) + 9,859 • Ln(($^{106}Ru.$ ^{137}Cs)/ $^{134}Cs^2$)=-1,846.Ln(BU) + 9,906	TC_{True} 50 BU_3 51	 2 %

$|BU_1-BU_2|$ = 16 GWd/t, $|BU_1-BU_3|$ = 27 GWd/t, $|BU_2-BU_3|$ = 11 GWd/t

Checking of the initial composition

The initial composition is the parameter most difficult to check by stepping of passive measurements. The case of fuels UOX is notably different from that of fuels MOX since the initial enrichment expressed by the content Uranium 235 conditions fuels UOX completely whereas the isotope of the plutonium and its rate of incorporation are necessary to the complete description of the MOX. The principle of a possible checking of initial enrichment or initial content Pu is similar to that of BU: it rests on the assumption of a variation different from the isotopic reports/ratios according to isotopes' according to initial composition IC (initial enrichment for UOX or initial content Pu for the MOX).

$$\frac{^{134}Cs}{^{137}Cs}_{(CT,BU)} = \left(\delta_1 IC_1 + \delta_1\right)$$

$$\frac{^{154}Eu}{^{137}Cs}_{(CT,BU)} = \left(\delta_2 IC_2 + \delta_2\right)$$

$$\frac{^{106}Ru\,^{137}Cs}{^{134}Cs^2}_{(CT,BU)} = \left(\delta_3 IC_3 + \delta_3\right)$$

The method would thus consist in adjusting correlations above and applying a method similar to that retained for the validation of the history of irradiation. The following paragraphs contribute to the evaluation of the sensitivity of a method of this type based on measurements of gamma spectrometry.

Checking of the initial enrichment of fuels UOX

The variation of the measurable isotopic reports/ratios by gamma spectrometry according to initial enrichment is weak.

As an example, Figure 3 indicates the relative variations of the isotopic reports/ratios for a burnup rate of 60 GWd/t between 2 and 5 % of enrichment in ^{235}U.

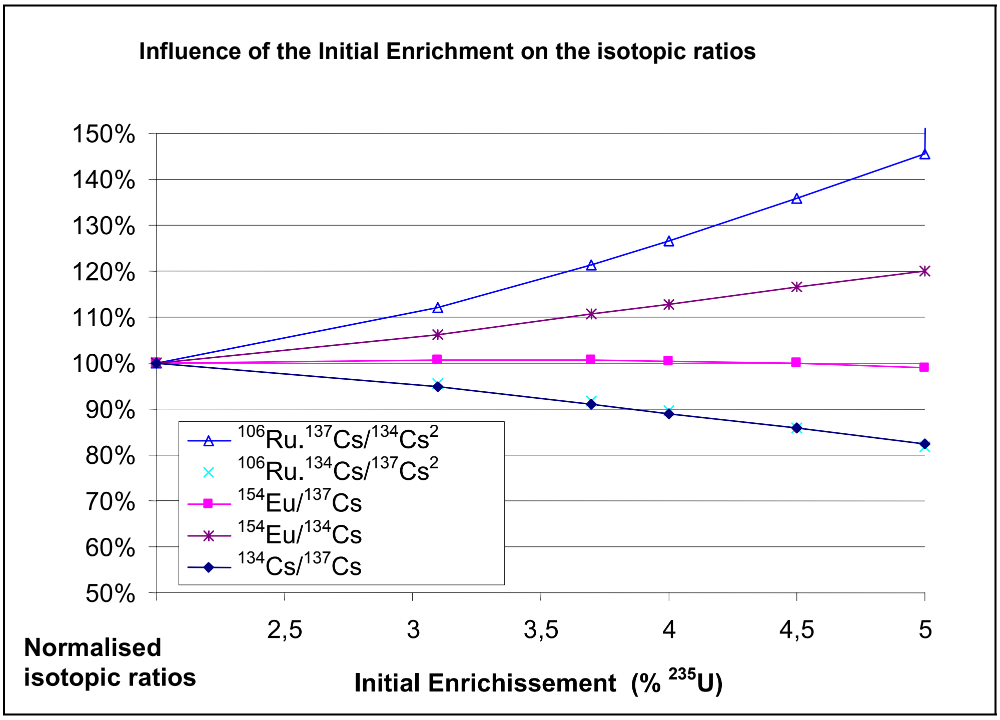

FIG. 1. Variation of the isotopic reports/ratios according to the initial content of ^{235}U between 2 and 5 % and a rate burnup of 60 GWd/t.

The examples illustrated by figure 1 show that the variations of the isotopic reports/ratios resulting from measurements of gamma spectrometry remain weak. Among evoked measurable isotopic reports/ratios above the most significant variation is obtained for the report/ratio. This isotopic report/ratio varies approximately 45 % on the whole of the range of enrichment, the interpretation of this measurement in term of confirmation of initial enrichment in ^{235}U does not appear easy to used and would imply a very precise instrumentation.

Checking of the initial content plutonium of MOX fuels

The variation of the measurable isotopic reports/ratios according to the initial content plutonium is weak. As an example the following Table indicates the relative variations of the isotopic reports/ratios for a burnup rate of 48 GWd/t between 5 and 14 % of initial content Pu.

Just like in the case of fuels UOX, the examples illustrated by figure 2 show that the variations of the isotopic reports/ratios resulting from measurements of gamma spectrometry remain weak. vary approximately 40 % on the whole of the field what would return an interpretation of this delicate and vague measurement in term of initial content Pu.

CONCLUSION

This paper exposes the mechanisms making it possible to validate the data input used by calculations associated with interpretation in term with burnup rate with passive neutron measurements. In the majority of the cases, the time of cooling as well as the history of irradiation could be validated in a reliable way if the precision of measurements of gamma spectrometry makes it possible to consider the relationship isotopic with a precision of about 10%.

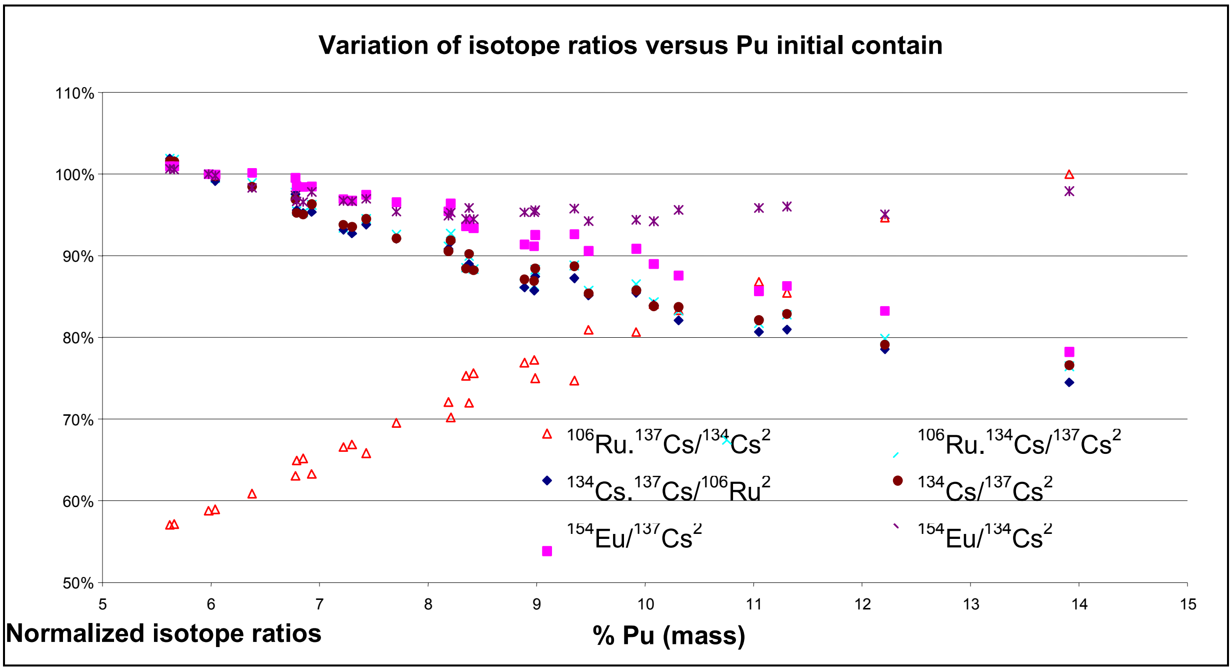

FIG. 2. Variation of the isotopic reports/ratios according to the initial content Pu from 5 to 14 % for equivalences ^{235}U between 3,1 and 4,5 % and a burnup rate of 48 GWd/t.

It remains that in practice, the reliability of the methods described in this document depends primarily on the performances of the instrumentation, which will be used. A work of evaluation of these performances on the basis of characteristic of the instrumentation remains to be made for each project. A study of sensitivity based on sets of incorrect data but plausible will be carried out in order to specify the performances of the measuring instruments necessary to the detection of each anomaly.

Contrary to the time of cooling and history of irradiation, the checking by gamma spectrometry of the initial composition of fuels MOX and UOX remains problematic because, on the one hand, the transmitters gamma are not sensitive to the initial isotope of the plutonium used in fuels, and on the other hand, the sensitivity to the initial content plutonium or initial enrichment in ^{235}U remains low. In any event, very accurate gamma spectrometry measurements should be used for checking initial contents. It is however significant to note that the initial composition can be checked independently of the operator and that thus the probability of an error is less than for the operating data.

An experimental campaign at la Hague, will be done on MOX fuel at the end of 2002 to confirm the global methodology of MOX burnup measurement and the specific data validation module performances detailed in this paper.

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Burnup credit in the evaluation of MOX fuel storage in the USA

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Abstract. In support of the United States Department of Energy Mixed Oxide (MOX) fuel irradiation program, Duke Energy Corporation has performed criticality evaluations of MOX fuel storage in the spent fuel pools at McGuire nuclear station. The boron credit (sic burnup credit) methodology approved by the US Nuclear Regulatory Commission for use with all uranium fuel at McGuire was used for these analyses. This presentation will review the analysis methodology and assumptions used for the evaluation of the MOX fuel. The criticality analyses demonstrate that the proposed MOX assembly designs, containing weapons-grade plutonium with concentrations between 4.07 and 4.37 wt % Pu, can be stored without any modifications to the existing fuel storage racks. Calculations indicate that, for normal storage conditions, MOX fuel assemblies are effectively equivalent to uranium fuel assemblies enriched to between 4.25 and 4.75 wt % U-235. Under postulated accident conditions with MOX fuel stored in the spent fuel pools, adequate subcriticality is maintained without requiring enrichment of the soluble boron in the spent fuel pool coolant.

INTRODUCTION

The objective of the U.S. Department of Energy (DOE) MOX fuel irradiation program is to disposition approximately 33 tonnes of weapons-grade plutonium, by irradiating MOX fuel assemblies in Duke Energy's McGuire and Catawba reactors. The current plans will require a total of around 1300 MOX fuel assemblies to accomplish the disposition goal. Following the successful performance of a MOX lead test assembly campaign, batch irradiation of MOX fuel at McGuire and Catawba is scheduled to begin around 2007, and continue for another 15 years. During the campaign plans call for loading MOX fuel alongside conventional low-enriched uranium fuel in a manner similar to current practice in European pressurized water reactors. Overall core loadings of up to 40% MOX fuel are envisioned [1].

An initial phase of the DOE MOX irradiation program involves identifying necessary facility modifications to allow MOX fuel to be received, stored, burned in the reactor, and discharged to the spent fuel pool or dry storage, while complying with all safety limits and regulatory requirements. As part of this program phase, Duke Energy has analyzed the criticality implications of MOX fuel storage at McGuire and Catawba. The scope of the MOX fuel criticality analysis reported here are evaluations of MOX fuel storage in the McGuire spent fuel pools. The focus of this report is to describe the burnup credit methodology used to evaluate the multiregion spent fuel pool at McGuire approved for partial boron credit. The analyses are performed using the industry-standard CASMO-4 / SIMULATE-3 and SCALE 4.4 / KENO V.a nuclear analysis codes.

MCGUIRE SPENT FUEL POOLS

McGuire Nuclear Station operates two (2) 3411 MWth Westinghouse 4-loop PWRs. Figure 1 shows an overhead view of the pertinent fuel storage areas in one of the McGuire fuel buildings. This layout is typical of the two fuel buildings at McGuire. Fresh fuel is first received in the New Fuel Receiving Area and stored temporarily prior to being removed from the shipping container. Upon removal from the shipping container the assembly is placed in a NFV location for inspection and then is either kept in the NFV or transferred to the SFP for storage prior to reactor irradiation. Fresh fuel and irradiated reload fuel are transported to the reactor via the water-filled Fuel Transfer Area. Discharged fuel assemblies from the reactor are also returned to the Spent Fuel Pool through the Fuel Transfer Area. Qualified spent fuel assemblies may be loaded into dry storage casks in the Cask Area. Once the dry storage casks are drained, sealed, and decontaminated, they are taken to the on-site independent spent fuel storage installation (ISFSI) for interim storage. McGuire began ISFSI operations in 2001.

The SFPs are designed to store fresh and irradiated fuel assemblies in a wet, borated environment. The McGuire SFPs have been divided into two regions: Region 1 and Region 2. Figures 2 and 3 depict the storage layout of fuel assemblies in Region 1 and Region 2 cells. The Region 1 storage racks have a flux trap design, with the rack cell walls composed of stainless steel. Boraflex poison panels are attached to the outsides of each of the Region 1 rack cell walls. McGuire Region 1 is normally used for storage of fresh fuel and irradiated fuel that will be reloaded into the reactor core. Region 2 is designed to store fuel assemblies that have been permanently discharged from the reactor. This design is called the “cell / off-cell” pattern, because it consists of a tight checkerboarded cluster of stainless steel rack cells. The holes in this pattern are the off-cells, and fuel assemblies are stored in these off-cells as well. As with Region 1, Boraflex poison panels are attached to each of the cell walls in the Region 2 racks.

Table I provides McGuire SFP rack data important to the criticality modeling of these storage regions. Note that, as a result of measured and projected degradation of the Boraflex panels in the McGuire SFP storage racks, McGuire Region 1 currently is only allowed credit for a maximum of 25% of its original Boraflex loading (as listed in Table I), while McGuire Region 2 can take credit for up to 50% of it. In fact, some areas of the McGuire SFPs have experienced so much Boraflex loss that the storage regions have been further subdivided as follows:

McGuire Region 1A – 25% of original Boraflex loading
McGuire Region 1B – 0% of original Boraflex loading
McGuire Region 2A – 50% of original Boraflex loading
McGuire Region 2B – 0% of original Boraflex loading

It is likely that the McGuire SFP racks will undergo further Boraflex degradation prior to batch irradiation of MOX fuel, and this will necessitate remedial measures such as rack absorber inserts or re-racking in one or both SFPs. Such actions would be required to allow continued efficient pool storage of either uranium or MOX fuel.

Table I. General design information for the MCGUIRE SFP storage racks

	McGuire Region 1	McGuire Region 2
# of storage locations in each SFP	286	1177
Storage cell pitch (cm)	26.4	23.2 (Avg)
Original Boraflex Loading (g/cm^2)	0.020	0.006
Storage cell wall thickness (cm)	0.19	0.19
Normal SFP water temperature range (degrees F)	68 - 150	68 - 150
Minimum required SFP boron concentration (ppm)	2475	2475

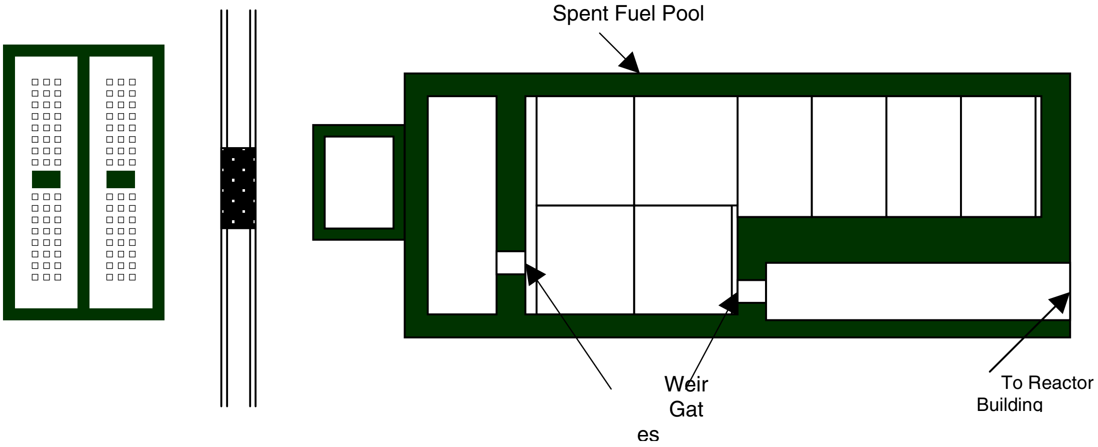


FIG. 1. Overhead View of McGuire Fuel Building.

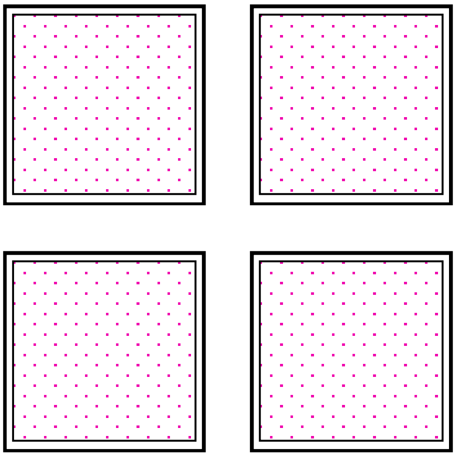


FIG. 2. McGuire SFP Region 1 Flux Trap Arrangement.

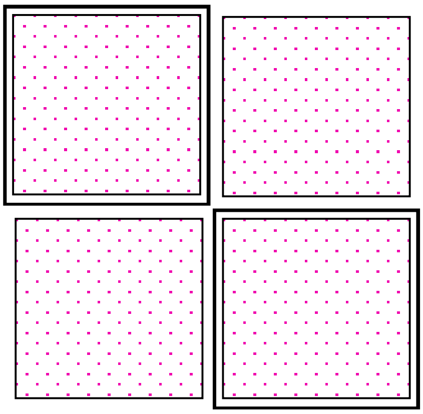


FIG. 3. McGuire SFP Region 2 "Cell / Off-Cell" Arrangement.

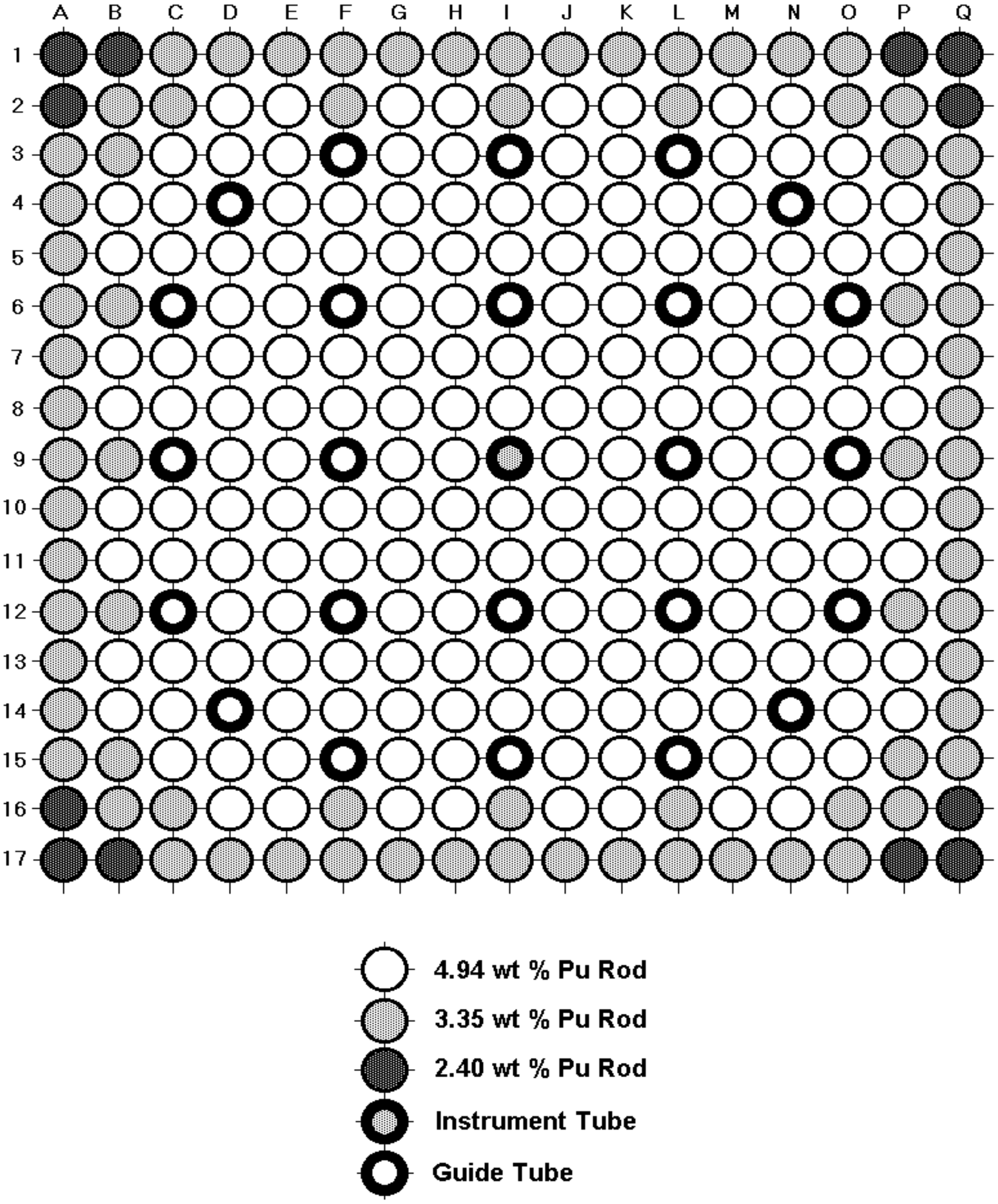
MOX FUEL ASSEMBLY DESIGN AND NEUTRONIC FEATURES

The reference MOX fuel assembly is a 17 x 17 array of fuel rods that are radially zoned by plutonium concentration as shown in Fig. 4. The zoning scheme is designed to control power peaking associated with the high thermal flux gradient between adjacent MOX and uranium fuel assemblies that exists during reactor irradiation. The corner rods in the MOX lattice contain the lowest plutonium loading, while the edge rods have an intermediate plutonium concentration. Fuel with the highest plutonium concentration is used in the interior rods. Two "enrichments" (assembly-average plutonium loadings) are envisioned with the reference design – a "High MOX" assembly (average 4.37 wt % Pu) and a "Low MOX" assembly (average 4.07 wt % Pu). The majority of the weapons-grade plutonium in the reference MOX assembly is Pu-239 (~ 93%) and Pu-240 (~ 6%). The balance of the fuel material is depleted uranium oxide.

Table II. Design information for the reference MOX and uranium fuel assemblies

	Mark-BW/MOX1 Assembly	RFA Uranium Assembly
Fuel Pellet Diameter (cm)	0.8192	0.8192
Fuel Cladding Inner Diameter (cm)	0.8356	0.8356
Fuel Cladding Outer Diameter (cm)	0.9500	0.9500
Average Fuel Stack Density (g UO_2/cc or g PuO_2/cc)	10.30	10.30
Active Fuel Stack Length (cm)	365.8	365.8
Fuel Rod Pitch (cm)	1.26	1.26
# of Fuel Rods per Assembly	264	264

As compared with the current 17 x 17 uranium assemblies that fuel the McGuire reactors, the reference MOX assembly is almost identical in the neutronically-important design variables such as fuel mass, fuel rod dimensions, and fuel pellet dimensions. Because the water to-fuel ratio is the same for both the MOX and uranium assemblies, it makes the observed reactivity differences between them easier to assess. Such differences are essentially attributable to the actinide material fission and absorption characteristics, and their resulting spectral effects. The reference MOX fuel assembly is more undermoderated than the uranium fuel lattices currently used at McGuire. The near-thermal absorption resonances of Pu-239 and Pu-241 cause the reference MOX assembly to have a harder neutron spectrum and a smaller migration area than a uranium assembly [2]. As a result, if strong thermal absorbers such as boron (whether in the coolant or in lumped burnable poison rods) are present in the MOX fuel lattice, they are less effective, as compared with a uranium fuel lattice, in reducing system reactivity. The difference in absorber effectiveness between MOX and uranium fuel assemblies is less severe when SFP rack panel poisons (such as Boraflex) are considered, since these absorbers are outside of the fuel lattices.


A B C D E F G H I J K L M N O P Q
1
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13
14
15
16
17
4.94 wt % Pu Rod
3.35 wt % Pu Rod
2.40 wt % Pu Rod
Instrument Tube
Guide Tube

FIG. 4. Fuel Rod Layout for High MOX Fuel Assembly (Avg 4.37 wt % Pu Loading).

BURNUP CREDIT EVALUATION FOR MOX FUEL

Current regulations allow partial credit for soluble boron in maintaining adequate subcriticality in SFPs. The requirements for adopting this method are provided very generally in 10CFR50.68 (b), with more specific guidance in Reference 3. The following criteria must be met in using this methodology:

- With the SFP racks loaded with fuel of the maximum permissible enrichment and flooded with full-density unborated water, the maximum 95/95 k_{eff} shall be less than **1.0**, including all pertinent mechanical and calculational uncertainties.
- With the SFP racks loaded with fuel of the maximum permissible enrichment and flooded with full-density water at a boron concentration of "X", the maximum 95/95 k_{eff} shall be less than **0.95**, including all pertinent mechanical and calculational uncertainties. ["X" represents the maximum amount of boron required to keep the k_{eff} below 0.95, without consideration of accidents].
- The boron concentration "X" determined above must be less than the boron concentration remaining in the SFP following a worst-case credible dilution event.

This boron credit methodology has been approved for use with all uranium fuel at McGuire.

Given the above regulatory requirements, the MOX fuel criticality analysis for the SFPs comprises the following general steps:

- The design information is obtained for the different fuel assemblies in use, as well as for the fuel storage racks in the spent fuel pools.
- The analyst builds computer models of the fuel assemblies and performs several fuel depletions for the Low MOX (4.07 wt % Pu avg) and High MOX (4.37 wt % Pu avg) fuel designs, mimicking in-reactor irradiation conditions.
- From these depletion cases, low-temperature, no-boron branch cases are performed in which the fuel assembly is put into the spent fuel racks with various fuel storage configurations (Unrestricted, Restricted / Filler, and Checkerboard / Empty). These branch case computer calculations yield k_{eff} results for several enrichment/burnup combinations. To each k_{eff} result various reactivity penalties are added to account for mechanical uncertainties and code methodology biases/uncertainties, which gives the no-boron 95/95 k_{eff} for that enrichment/burnup/storage configuration combination.
- Using these 95/95 k_{eff} results, for each evaluated enrichment the lowest burnup is determined for which the no-boron 95/95 k_{eff} is less than 1.00. In this manner the minimum burnup limits are developed for a given enrichment, fuel pool, and storage configuration.
- The analyst ascertains a bounding amount of soluble boron credit that reduces the previously determined no-boron 95/95 k_{eff}s from less than 1.00 to less than or equal to 0.95, and accounts for burnup-related biases and uncertainties as well. This amount of soluble boron credit required is verified to ensure it does not exceed the amount remaining following a worst-case credible boron dilution event.
- Several potential spent fuel pool accident scenarios are also evaluated, including an assembly-misloading event, accidents that increase or decrease the fuel pool water temperature, and a heavy load drop (weir gate) event. The amount of soluble boron needed to keep the 95/95 k_{eff} at or below 0.95 is determined for each of these accidents, and the maximum amount required is verified to ensure it does not exceed the minimum spent fuel pool boron concentration for normal operations (2475 ppm).

The following assumptions are used in the SFP criticality analysis for MOX fuel:

- All conditions are modeled at both 68 and 150 ^{o}F, the normal operating bounds for the SFPs. Only the most reactive temperature is used to set the storage requirements.
- All calculations are performed in 2-D; i.e. no axial effects are directly modeled for the nominal rack criticality cases. Additional computations are performed to determine reactivity biases that account for differences between 2-D and 3-D modeling. These biases, if positive, are included in the overall 95/95 k_{eff} calculations.
- "No xenon" conditions are assumed in the storage racks. This assumption conservatively bounds the highest fuel reactivity observed following irradiation (~ 100 hrs after shutdown).
- No credit is taken for the spacer grid material. A slight reactivity penalty is applied for spacer grids in the heavy load drop accident evaluation, since this is analyzed in highly borated (2475 ppm) conditions, where the water displacement caused by the presence of spacer grids can actually increase system reactivity.

Each of the biases and uncertainties are listed in Table III and discussed in more detail in the following paragraphs.

Table III. Pertinent biases and uncertainties in the criticality calculations for the MCGUIRE SFP storage racks (with MARK-BW/MOX1 fuel)

Biases	Uncertainties
Benchmark Method Bias	Benchmark Method Uncertainty
BP-Pull Bias	Enrichment Manufacturing Uncertainty
Axial Burnup / 3D Bias	Fuel Density Manufacturing Uncertainty
	Plutonium Isotopic Variation Uncertainty
	Storage Rack Cell Wall Thickness Manufacturing Uncertainty
	Storage Rack Center-to-Center Cell Spacing Uncertainty
	Boraflex Uncertainties
	Burnup Measurement Uncertainty
	Burnup Computational Uncertainty

Benchmark Method Bias

This bias is determined from the benchmarking of the code system used (SCALE / KENO V.a or CASMO-4 / SIMULATE-3), and represents how much the code system is expected to overpredict (negative bias) or underpredict (positive bias) the "true k_{eff}" of the physical system being modeled. Although CASMO-4 benchmarking is not yet completed, calculations indicate that the final bias should be about the same, or slightly lower than, the preliminary SCALE / KENO V.a benchmark bias, which is comparable to previously determined biases for uranium-fueled systems.

BP-Pull Bias

This bias accounts for the potential increase in assembly reactivity associated with that assembly having operated in the reactor with a burnable poison (absorber) insert. To compute a BP-Pull bias using CASMO-4, a nominal depletion for a particular MOX assembly (Low MOX or High MOX) is performed, with up to a 4.0 wt % B_4C burnable poison assembly inserted in the fuel assembly. The burnable poison assembly is removed at 20 GWD/MTHM, and then further depleted with no component inserts in the fuel assembly.

The BP-Pull bias is thus the difference in reactivity between a MOX fuel assembly that had operated with a burnable poison inserted and the nominal, "never-BP'ed," fuel assembly.

Axial Burnup / 3D Bias

The axial burnup / 3D bias represents the maximum reactivity difference expected between a 2-D (such as CASMO-4) model of a fuel assembly at its average burnup, and a detailed 3-D axial model of that fuel assembly. The 3-D model considers a conservative axial burnup profile, and includes a realistic assumption of operational moderator density history along the axial length of the fuel assembly. This bias thus accounts for potential non-conservatisms in the simple 2-D average-burnup model.

Previous analysis has shown that for high-enriched RFA uranium fuel, the Axial Burnup / 3D Bias becomes positive only at burnups around 25 GWD/MTU and higher. The Mark-BW/MOX1 fuel experiences a much smaller reactivity gradient as a function of burnup than does uranium fuel. For a given conservative axial burnup profile, then, MOX fuel should have a lower Axial Burnup / 3D Bias than uranium fuel. Indeed, calculations with burnup profiles from earlier analyses show that up to an average burnup of nearly 45 GWD/MTHM (the current MOX program limit), the Axial Burnup / 3D bias for MOX fuel is less than zero.

Benchmark Method Uncertainty

This uncertainty is determined from the benchmarking of the code system used (SCALE / KENO V.a or CASMO-4 / SIMULATE-3), and is a measure of the expected variance (95/95 one-sided uncertainty) of predicted reactivity from the "true k_{eff}" of the physical system being modeled. The critical experiment benchmarks were MOX criticals taken from the International Criticality Safety Benchmark Experiments Project handbook. Since the CASMO-4 benchmarking is not yet complete, for this evaluation a conservative assumption of twice the previously determined uranium fuel method uncertainty (~ $\pm 0.02\ \Delta k$) was employed.

Enrichment Manufacturing Uncertainty

An "enrichment" uncertainty of ± 0.05 wt % Pu is assumed for the MOX fuel analysis.

Fuel Density Manufacturing Uncertainty

For both uranium and MOX fuel, tolerances on pellet dishing, pellet diameter, and pellet densification can increase the effective fuel density from the nominal values reported in Table II. No manufacturing tolerances for the Mark-BW/MOX1 are currently available. However, previous uranium fuel data show tolerances on pellet dishing of up to 0.4% reduction, a tolerance on pellet diameter of up to a 0.0007-inch increase in OD, and a tolerance of up to a 1.5 % increase in fuel pellet densification. If these three variations are taken together to maximize a MOX or uranium assembly fuel loading, this maximum loading can be used with the nominal pellet dimensions to determine a maximum fuel density. Maximizing the tolerances above yields up to 10.50 g UO_2 or PuO_2 /cc.

Plutonium Isotopic Variation Uncertainty

The worst-case plutonium isotopic fractions of 95% Pu-239 and 5% Pu-240 are assumed.

Storage Rack Cell Wall Thickness Manufacturing Uncertainty

Although no tolerance on the nominal thickness of the storage rack cell walls is indicated in the McGuire SFP rack drawings, previous criticality calculations that have considered these SFP racks have assumed a conservative maximum tolerance of 0.01 inches on the cell wall thickness.

Storage Rack Center-to-Center Cell Spacing Uncertainty

This uncertainty accounts for possible variations in the SFP rack geometry. The rack cells in the McGuire Region 1, McGuire Region 2 SFP are brought together as close as the tolerances allow on center-to-center spacing.

Boraflex Uncertainties

Applicable to the McGuire Region 1A and 2A SFP racks, the Boraflex uncertainties are bundled uncertainties that account for gaps in the Boraflex panels, axial and radial material shrinkage, and physical self-shielding effects. For MOX fuel these uncertainties are slightly less than those previously computed for uranium fuel. This is expected, since strong thermal absorbers such as Boraflex are less effective in the presence of MOX fuel.

Burnup Measurement Uncertainty

Previous studies have determined that a burnup measurement uncertainty on the order of ~ 2% appears to be adequate to account for potential inaccuracies in spent fuel burnup records.

For the burned-fuel evaluations in this MOX fuel criticality analysis, this reactivity uncertainty is quantified by running branch cases from the nominal model for a particular pin pitch, fuel assembly design, and enrichment, with the pertinent assembly burnup reduced by 2%.

Burnup Computational Uncertainty

This uncertainty essentially accounts for deviations from criticality code predictions to actual burned-fuel isotopics. This uncertainly is typically quantified by benchmark evaluations of reactor operational data and/or isotopic measurements of burned fuel. This uncertainty has not yet been quantified by Duke Power for CASMO-4 with MOX fuel, but vendor benchmarking seems to indicate that the burnup computational accuracies for both MOX and uranium fuel are about the same. Therefore, for this analysis, a conservative burnup computational uncertainty (developed for a uranium-fueled system) is employed here for all MOX fuel 95/95 k_{eff} computations.

The mechanical and calculational biases and uncertainties described above add between 0.04 and 0.06 Δk to the overall 95/95 k_{eff} for MOX fuel, depending on the SFP rack. This is slightly higher, but still quite comparable, to the totals for uranium fuel.

Using the burnup credit methodology, and the evaluation procedure outlined at the beginning of this section, criticality calculations were carried out for the McGuire SFPs. Results for unrestricted storage are presented in this paper. Unrestricted storage means that a fuel assembly that meets the enrichment / burnup limits for “Unrestricted Storage” can be stored with other Unrestricted fuel assemblies. The entire stored array meets the regulatory SFP subcriticality criteria.

Both the normal and accident SFP conditions (described at the beginning of this section) were analyzed for the McGuire SFP storage regions. The boron concentration required for MOX fuel to achieve a 95/95 k_{eff} of 0.95 was noticeably higher than that required for uranium fuel. Nevertheless, the MOX fuel boron credit requirements were still well below the concentration resulting from the worst-case credible boron dilution events, so all the normal storage regulatory subcriticality requirements were met for MOX fuel.

For three of the accident conditions that needed to be evaluated for fuel storage (fuel assembly misload, dropped fuel assembly, and abnormal SFP temperature changes), the required boron concentrations to maintain the 95/95 k_{eff} below 0.95 are far below the minimum available in the SFP (2475 ppm), even with MOX fuel.

The other accident condition is the heavy load drop onto the SFP racks. The largest loads that can be carried over the McGuire SFPs are the weir gates (see their locations in Figure 1). These 3000 – 4000 lb steel gates, if dropped onto the SFP racks, are capable of crushing up to seven (7) fuel assemblies. In accordance with NUREG-0612 (Reference 20), heavy load drop evaluations must assume the racks and the fuel assemblies within them are crushed uniformly to an optimum pin pitch. The crushed-rack evaluation with MOX fuel in the McGuire racks determined a worst-case 95/95 k_{eff} well below the 0.95 limit, with 2475 ppm boron in the SFP. The MOX fuel k_{eff} was similar to the highest uranium fuel k_{eff} in the McGuire weir gate drop analysis.

Table IV documents the computed minimum burnup requirements for unrestricted MOX fuel storage in the McGuire 1A, McGuire 1B, McGuire 2A, and McGuire 2B, SFP storage racks, respectively. Also included are the previously calculated uranium fuel burnup limits. For the McGuire SFPs, both the Low MOX fuel and the High MOX fuel are effectively equivalent to uranium fuel enrichment between 4.25 to 4.75 wt % U-235.

Table IV. MCGUIRE minimum burnup requirements for uranium and MOX fuel storage

McGuire SFP Fuel Storage		Region 1A Unrestricted Storage	Region 1B Unrestricted Storage	Region 2A Unrestricted Storage	Region 2B Unrestricted Storage
Fuel Assembly Type	Fuel Enrichment (wt % U-235 or wt % Pu)	Min Burnup GWD/MTHM	Min Burnup GWD/MTHM	Min Burnup GWD/MTHM	Min Burnup GWD/MTHM
Uranium	3.00	0.0	0.0	21.5	19.8
Uranium	3.50	0.0	0.0	27.4	25.9
Uranium	4.00	1.6	0.0	33.0	31.5
Uranium	4.50	4.9	0.0	38.3	36.9
Uranium	4.75	6.7	0.0	40.9	39.5
Low MOX	4.07	3.0	0.0	37.0	34.5
High MOX	4.37	4.0	0.0	40.0	37.5

CONCLUSION

The results of the SFP criticality analyses demonstrate that MOX fuel can be stored without any modifications to the existing SFP storage racks. All MOX fuel in the McGuire SFPs can be treated as the equivalent of uranium fuel at 4.75 wt % U-235 or less. Finally, the SFP criticality evaluations show that failed MOX fuel pins can be stored in available failed fuel rod canisters, and these canisters can be placed essentially unrestricted in the McGuire SFP storage cells.

REFERENCES

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Burnup credit calculation route for PWR MOX assemblies and experimental validation in Minerve R1-MOX and SLB1 P.I.E.

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DRN/DER/SPRC,
Saint Paul lez Durance, France

Abstract. The current increase of French MOX fuel cycle enhances the need for criticality-safety design studies linked to plutonium-fueled assemblies. Therefore, the Burnup Credit in MOX fuel is an important challenge in France and is investigated in this paper. In a first step, we developed a calculation scheme for MOX lattices; specific neutronics models were elaborated and validated. Then, the Burnup Credit Nuclides, Actinides and Fission Products for MOX fuel Criticality calculation were selected. The validation of our calculational tools against international codes is summarised, and the experimental validation of Burnup Credit analysis in LWR-MOX fuel, throughout experimental programmes launched in CEA facilities, is presented.

1. INTRODUCTION

In most countries, criticality analysis of light water reactor (LWR) fuel stored in racks and casks, or dissolved, has assumed that the fuel is fresh. This assumption leads to significant conservatism in the calculated value of the system's reactivity. The concept of allowing reactivity credit for spent fuel, referred to as "Burnup Credit", offers economic incentives. French design studies pointed out that a load increase in several fuel cycle devices (casks, storage pools, dissolvers) is possible when Fission Products (FP) are accounted for in criticality studies.

In France, the plutonium from the La Hague reprocessing plant is recycled in PWRs. Currently twenty-four 900 MWe PWRs are devoted to the Pu recycling in 30% mixed core loading. It is foreseen to enlarge the Mixed Oxide (MOX) loading to the 28 CP1-CP2 French Reactors. Furthermore, the current 'hybrid' mixed core fuel management based on 3 irradiation cycles of the MOX assemblies will be replaced in 2005 by 4 irradiation cycles (equivalence of UOX and MOX assemblies), leading to high burnup of the MOX fuels.

This increase of French MOX fuel cycle emphasizes the need for criticality-safety design studies linked to plutonium-fueled assemblies. Therefore, the Burnup Credit in MOX fuel is an important challenge in France and is investigated in this paper.

2. PRESENTATION OF THE FRENCH CALCULATIONAL TOOLS

The two first steps of BUC methodology consist in validation of a computer code system to calculate isotopic concentrations in Spent Fuel and in validation of a code package to predict the Keff of a Spent Nuclear Fuel package. In France, these code systems are respectively DARWIN [1] and CRISTAL [2, 3]. They are both based on the code APOLLO2 [4] used for this study on Burnup Credit in LWR-MOX assemblies.

APOLLO2 is a modular code which solves both the Boltzmann integral equation and the integro-differential equation (Sn method). It allows the use of several collision probability methods to solve the integral equation: exact-2D Pij, multicell Pij based on the interface current method. The one used for the Benchmark is the exact-2D Pij method: no hypothesis of pincell cylindrization has been made. The Neutron Data Library used is the 'CEA93' library in a 172-group structure. These multigroup cross-sections and effective cross-sections were processed by NJOY from the JEF2 European File [5]. The CEA93 library was extensively benchmarked against specific French integral experiments.

DARWIN is the reference calculation package for the fuel cycle of all types of reactors. It was developed by the CEA and its French partners (COGEMA, EDF and FRAMATOME) to estimate the physics quantities characterizing the burnup fuels from reactors: material balance, decay heat, activity, neutron, γ, α, β sources and spectrum, radiotoxicity.

DARWIN is devoted to all cycle studies, with current fuels (UOx, MOX) or innovative fuels (MIX, APA, PuTh) and for every nuclear road (Pressurized Water Reactor, Fast Breeder Reactor, Boiling Water Reactor, Advanced Reactors). DARWIN is also used in the back-end cycle for actinide incineration (SPIN) or long term interim storage studies. The simplified DARWIN structure, based on new codes and libraries is described in Figure 1.

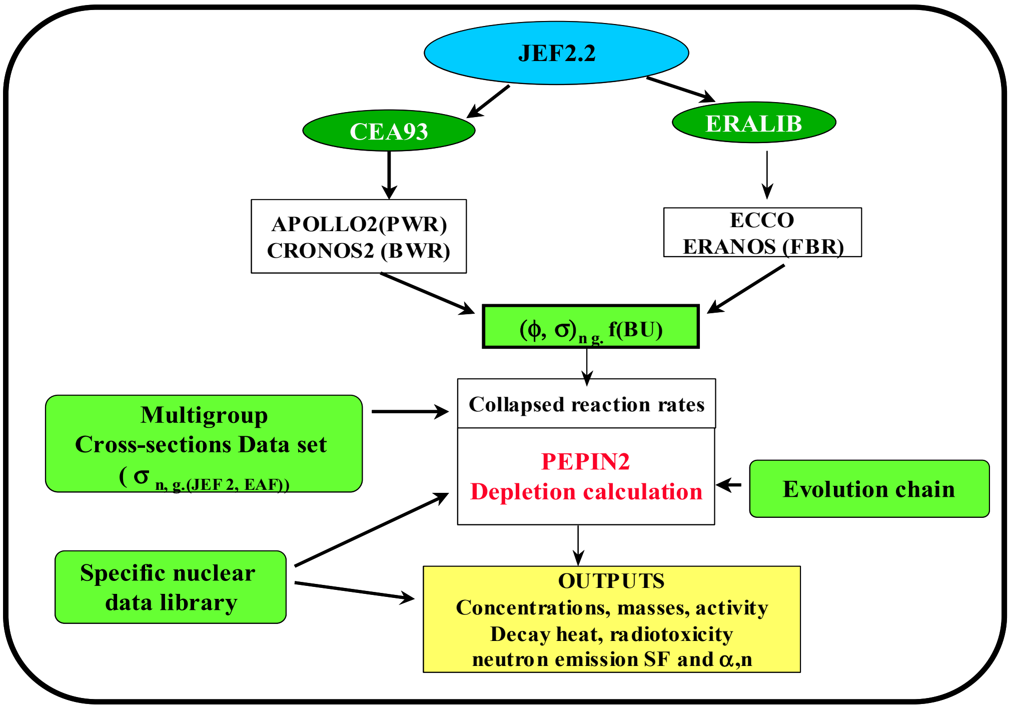
JEF2.2
CEA93
ERALIB
APOLLO2(PWR)
CRONOS2 (BWR)
ECCO
ERANOS (FBR)
$(\phi, \sigma)_{n\,g.}$ f(BU)
Collapsed reaction rates
Multigroup
Cross-sections Data set
($\sigma_{n,\,g.(JEF\,2,\,EAF)}$)
PEPIN2
Depletion calculation
Evolution chain
Specific nuclear
data library
OUTPUTS
Concentrations, masses, activity
Decay heat, radiotoxicity
neutron emission SF and α,n

Figure 1: the DARWIN Package

The PEPIN2 program performs the nuclide depletion calculations. Different libraries feed this module

- neutronics data provided by French cell or transport codes APOLLO2 (for PWR studies), ECCO-ERANOS system (for FBR studies): these data are self-shielded cross sections and neutron spectra,
- nuclear data (decay data, fission and α, n) yields) and evolution chains
- complementary cross-sections, missing from the transport codes libraries, specially for activation products. They are included in the 'cycle library'.

DARWIN makes possible the retrieval of cumulated reaction rates during irradiation in order to give the origin of every isotope build-up. Furthermore, a "PERTURBATION" of main nuclear data, such as capture cross-section, initial isotopic composition, flux, is also available and allows us to perform sensitivity studies.

The CRISTAL system presented on Figure 2 includes two routes:

- a design route with the CEA93 multigroup cross-sections and the APOLLO2/MORET4 codes
- a reference route using the continuous Monte-Carlo code TRIPOLI4 [6]

Both routes are based on the latest version of the Joint European nuclear data file JEF2.2. Accurate calculation procedures were elaborated for the design route/2/ of the criticality safety package CRISTAL. These procedures were validated against reference calculations, such as the CRISTAL reference route based on the continuous energy Monte Carlo code TRIPOLI4. Both calculation routes are consistent due to their common library processed from JEF2.

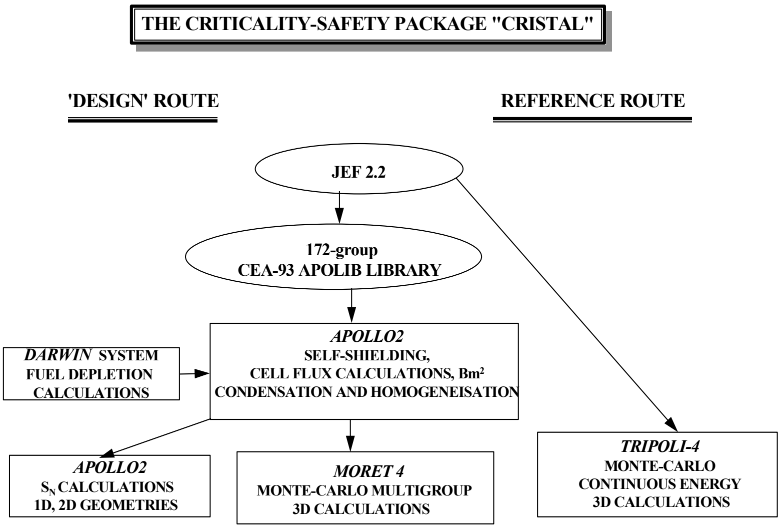
THE CRITICALITY-SAFETY PACKAGE "CRISTAL"
'DESIGN' ROUTE
REFERENCE ROUTE
JEF 2.2
172-group
CEA-93 APOLIB LIBRARY
DARWIN SYSTEM
FUEL DEPLETION
CALCULATIONS
APOLLO2
SELF-SHIELDING,
CELL FLUX CALCULATIONS, Bm²
CONDENSATION AND HOMOGENEISATION
APOLLO2
S_N CALCULATIONS
1D, 2D GEOMETRIES
MORET 4
MONTE-CARLO MULTIGROUP
3D CALCULATIONS
TRIPOLI-4
MONTE-CARLO
CONTINUOUS ENERGY
3D CALCULATIONS

Figure 2: the CRISTAL Package.

3. CALCULATION SCHEME FOR MOX

In a first step, specific APOLLO2 neutronics models to implement in MOX lattice calculations were elaborated, then validated against reference continuous-energy TRIPOLI4 [7] Monte-Carlo results.

The geometry described is the actual geometry of the lattice: square cell with specular boundary. We applied the exact-2D Pij method to solve the Boltzmann integral equation: no cylindrization, nor isotropic interface angular flux assumption is used. Thus, studies on MOX lattices [7] have shown that the standard cylindrical pin-cell model leads to a reactivity underestimation of 700 pcm, as shown in Table X. This Table refers to a 7% MOX lattice; the discrepancies between the APOLLO2 and TRIPOLI4 reference calculations are expressed in Ln (K_∞ /K_∞ ref)x10^5. Kinfinity is split into the "six" factors: Kinf = $\chi.\varepsilon fast.\varepsilon epi.p.f.\eta$ in agreement with the Fermi neutron balance breakdown.

Table I: 7% Pu MOX Lattice; Discrepancies between TRIPOLI4 and APOLLO2 calculations (in pcm)

APOLLO2 Pij modelling	Cylindrical (cell calculation)	Interface anisotropic angular flux (multicell calc.)	Pij 2D (lattice calc.)	TRIPOLI4 (Reference calc.)
χ(n,2n)	24	24	23	1.00122
εfast (U 238, Pu240 fissions)	187	162	81	1.09146
εepi. (U235,Pu239,Pu 241 fis.	812	311	277	1.41465
p	-1597	-332	-165	0.46013
f	183	40	6	0.96549
η	-78	16	18	1.67188
Kinfinity	-560	131	149	1.14924

Table I shows that the standard pincell model (Pij 1D cylindrical) induces strong compensation of errors between the resonance escape factor 'p' and the epithermal fission factor 'εepi' taking into account U235, Pu239 and Pu241 fissions in the resonance range. These compensations are clearly reduced in multicell calculations, provided anisotropic angular fluxes are used in the interface current method (column 3): the escape factor is underestimated by 300 pcm (instead of 1600 pcm in cylindrical calculation), due to accurate Dancoff modelling by the UP1 assumption. Table I shows the exact 2D Pij calculation is the most accurate, with reduced biases on resonant capture and fission rates (i.e. εepi and p in column 4).

In our calculations, the MOX fuel pin is divided in six concentric zones. Self-shielded cross-sections are calculated for U238 in each concentric zone: an accurate 'background matrix' formalism [8] is used for resonant reaction rate calculation, allowing space-dependent resonance self-shielding and rim effect modelling. For the other Actinides, Pu239, Pu241, Pu242, Pu238, Am241, U235, U236 and Zr we used a pin averaged self-shielding' formalism.

It was pointed out that a refined energy mesh is needed in the 1 eV resonance of the Pu240 isotope. The XMAS 172-group structure, conceived for MOX fuel and used in the CRISTAL design route was demonstrated accurate within ± $40*10^{-5}$ in $\Delta k/k$ (while the 99-group structure of the previous criticality-safety CRIBLE package was unsatisfactory). Furthermore, the XMAS energy mesh allows the accurate modelling of the mutual shielding between the Pu240 1.0 eV resonance and the Rh103 1.3 eV resonance arising in spent MOX fuel.

4. SELECTION OF BUC NUCLIDES FOR MOX FUEL CRITICALITY CALCULATIONS

The Burnup Credit in MOX fuels is different from the Burnup Credit in UO2 assemblies, particularly for the Actinide component which is strongly decreased in MOX fuel due to the high initial Pu240 amount and consequently to a better conversion factor during irradiation. On the other hand, the Am241 poisoning is enhanced in MOX fuel owing to the Pu241 initial content. The BUC reactivity loss components in MOX fuel are summarised in Table II:
BUC = ρ(BU) - ρ(fresh) = Ln Keff(BU)/Keff(fresh)

Table II: |BUC| reactivity worth in pcm (10-5 in $\Delta k/k$) in PWR MOX assembly Pu/U+Pu = 5.6 %, Pu initial isotopic vector from reprocessed 33 GWd/t UO2

Burnup		20 GWd/t	40 GWd/t	60GWd/t
T = 1 year	Actinide Component	4540	7550	9720
cooling time	F.P. Component	5190	8330	11200
	Total BUC	9730	15880	20920
T = 5 years	Actinide Component	8220	11900	14360
cooling time	F.P. Component	5650	9300	12840
	Total BUC	13870	21200	27200

These results confirm that the Burnup Credit in transportation and storage of MOX fuel assemblies is reduced compared to UO2 assemblies for short cooling times ($T \leq 1$ year). This difference is mainly linked to the strong reduction of the Actinide component: $|\Delta\rho| = 7550$ pcm at 40 GWd/t and T=1 year, instead of $|\Delta\rho| = 19000$ pcm in UO2 lattices.

The FP poisoning worth in Table II corresponds to FP Burnup Credit component of the 15 FPs BUC nuclides selected for LWR-UOX assemblies [9, 10]. The FP component in MOX fuel is equivalent to the FP Burnup Credit in UOX fuel: $|\Delta\rho| = 8330$ pcm in MOX fuel (Table II) to be compared with $|\Delta\rho| = 8400$ pcm in UOX fuel at BU = 40 GWd/t (cooling time =1 year).

4.1 Selection of BUC-Actinides in MOX fuel

In agreement with UOX fuel, the U234-U235-U236-U238 must be selected. However, the absorbing U236 isotope is less important due to the depleted uranium used in MOX fuel: the U236 reactivity worth at 40 GWd/t decreases from -900 pcm in a 3.3 wt% U235 assembly to -100 pcm in a MOX fuel.

For short cooling times, the poisoning of Np237 isotope could be neglected in MOX fuel. Nevertheless, owing to the α decay Am237 →Np237, characterised by a half-period T=432 y, the Np237 negative worth is useful in criticality analysis of interim storage and disposal. For instance, the Np237 BUC component in a MOX fuel irradiated up to 45 GWd/t amounts to $\Delta\rho = -270$ pcm after 25 years of cooling.

Consistently with UOX fuel, the main Pu isotopes (Pu238, Pu239, Pu240, Pu241, Pu242) and the absorbing Am241, Am243 isotopes are requested in Burnup Credit analysis of MOX fuel. The reactivity worth of poisoning isotopes in MOX and UOX fuels are compared in Table III (BU = 40 GWd/t, cooling time T = 5 years).

Table III: Burnup Credit Component of absorbing actinides (BU = 40 GWd/t, T=5 years)

Nuclide	UOX	MOX
^{238}Pu	-310 pcm	-1100 pcm
^{240}Pu	-8300 pcm	-13000 pcm
^{242}Pu	-710 pcm	-1500 pcm
^{241}Am	-1290 pcm	-3400 pcm
^{243}Am	-280 pcm	-1900 pcm

The first difference in the selected BUC Actinides in MOX fuels compared to UOX fuels arises from Am242m. The weak positive reactivity introduced by Am242m build up was neglected in spent LWR-UO2 fuel (ρ= +27 pcm at 40 GWd/t [10]). In MOX spent fuel this fissile isotope, σ_f^{2200} = 6886 barns in JEF2 library, must be accounted for: ρAm242m = +500 pcm in the 20-50 GWd/t burnup range.

The second difference in the requested BUC Actinides is linked to Curium isotopes. These isotopes are neglected in UOX burnup studies due to their small concentrations (curium worth: -30 pcm in 40 GWd/t spent UOX assemblies [10]). Due to the increasing amount of Curium in MOX fuel, particularly of the fissile isotopes Cm243 and Cm245 (σ_f^{2200} = 500 barns and 2130 barns respectively), the BUC component of Cm isotopes is non negligible; it becomes positive beyond 30 GWd/t due to Cm245 build up at high burnup as shown in Figure 3. The BUC nuclides are Cm243, Cm244 and Cm245. The Cm242 (T1/2 =163 d) and Cm246 absorbing isotopes are rejected due to their negligible poisoning worth.

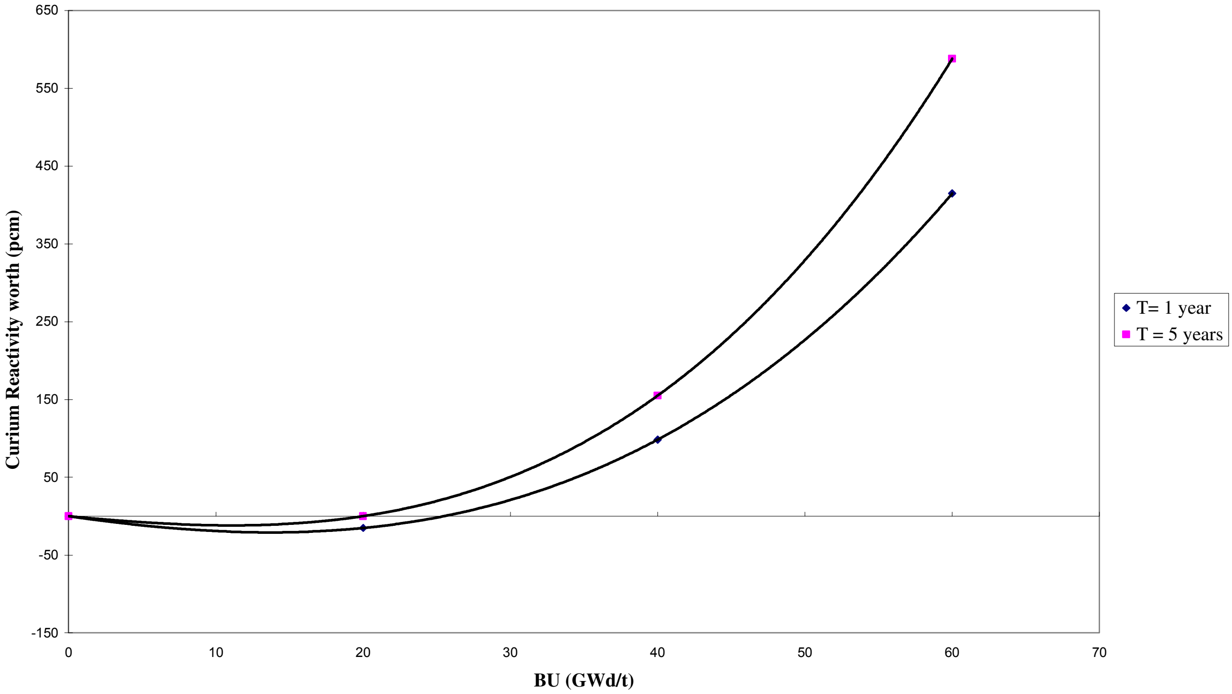

Figure 3: Curium Reactivity worth versus Burnup.

4.2 Selection of Fission Products in MOX fuel

Table IV lists the fifteen most absorbing FP nuclides in MOX fuel (stable and non volatile for safety-criticality purpose). These FP isotopes contributes up to 78 % of the total poisoning in the MOX pellet and are identical to the BUC-FPs selected in UOX studies. However, the relative importance of some nuclides is strongly modified.

Table IV: Reactivity worth (pcm) of the main FPs in LWR assemblies Burnup = 40GWD/T - Cooling time = 5 years

BUC-FP	MOX	UOX
Rh103	1800	1380
Sm149	1490	1300
Gd155	1110	1500
Sm151	830	640
Nd143	690	1100
Cs133	680	780
Ag109	540	260
Sm152	480	540
Eu153	450	390
Tc99	440	470
Nd145	280	420
Ru101	230	230
Sm147	210	230
Mo95	200	290
Sm150	190	270
Total	9620 pcm	9800 pcm

The first difference of the FP poisoning in UO2 assemblies and in MOX assemblies is linked to the spectrum hardening induced by the plutonium fuel. Thus, the FP poisoning worth is reduced in MOX assemblies, particularly for the thermal absorber such as Nd143, Gd155, Sm149, Sm151 (reactivity worth decrease by 35%-40% in MOX lattice). Therefore, the strong decrease of the Nd143 poisoning in MOX fuel, pointed out in Table IV, is the direct consequence of the neutron capture reduction in the thermal energy range. However, due to the poisoning saturation during irradiation, the huge absorbers such as Sm149 give an equivalent Burnup Credit component in MOX fuel (the Sm149 concentration is twice in a MOX fuel).

The second difference is linked to the modification of the FP yields associated with U235 fissions or Pu239 fissions. On one hand, the yields of Mo95, Nd143, Nd145 are reduced with Pu239-Pu241 fissions, leading to the decrease of the poisoning of these FPs. For instance, the negative worth of Mo95 isotope decreases from 290 pcm in UOX fuel to 200 pcm in MOX fuel due to a reduction by 20% of its concentration (γ_{Zr95}= 6.5% for U235 thermal fissions, γ_{Zr95} = 4.9% and 3.9% respectively for Pu239 and Pu241 thermal fissions). On the other hand, the cumulative yields of Rh103, Ag109 and Sm151, Sm152, Eu153, Eu155 decaying in Gd155, are strongly increased in Pu239 and Pu241 fissions. Table IV shows that the poisoning worth of Rh103, Ag109, Eu153 is enhanced in MOX fuel by 30%, 210%, 20% respectively, due to an increase of their amount in the MOX pellet (respectively +70%, +240%, +40% at 40 GWd/t).

5. CRISTAL VALIDATION IN THE FRAMEWORK OF THE OECD BUC-MOX BENCHMARK (PHASE IVA)

We participated in the last Burnup Credit Exercise, sponsored by the OECD/NEANSC Criticality Working Group, devoted to the calculation of infinite PWR fuel pincell reactivity for fresh and irradiated MOX fuels [11].

This exercise confirms that APOLLO2 calculations are very satisfactory. Figure 4 shows the agreement in international calculation results [12]; our CEA results are just lying on the mean value.

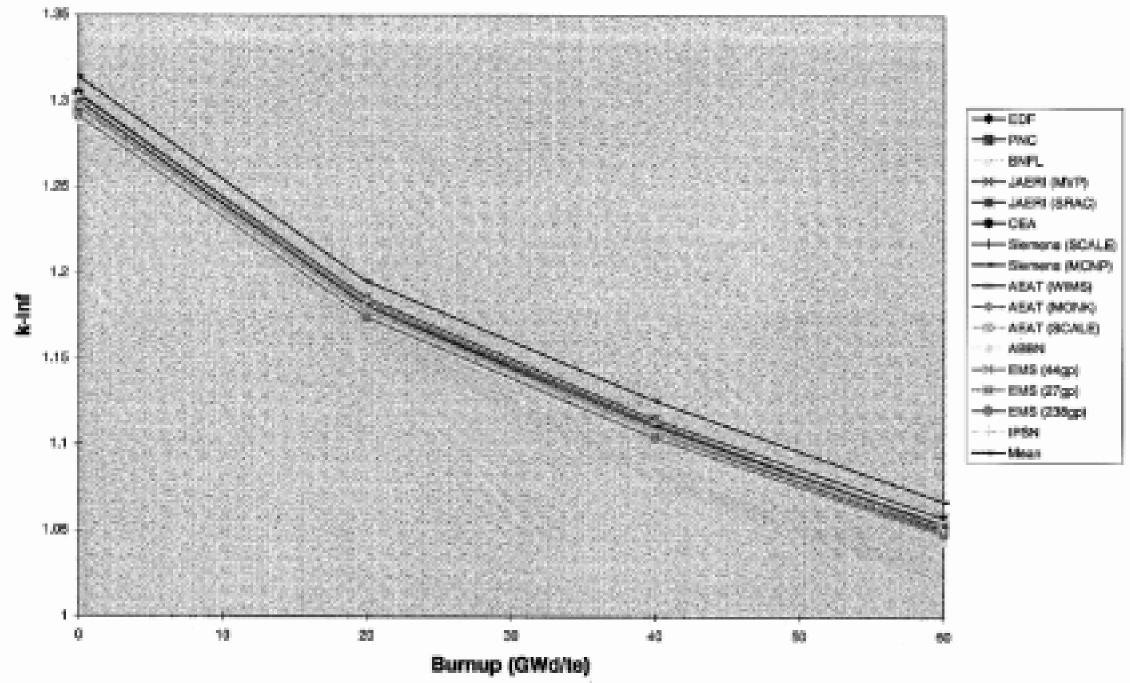

Figure 4: Kinfinity versus Burnup for a standard MOX fuel (5.6 % Pu/U+Pu).

6. THE FRENCH EXPERIMENTAL PROGRAM AND ASSOCIATED VALIDATION

To investigate Burnup Credit in MOX assemblies burnt in French PWRs, a large experimental programme [9] was started-up in 1996 within the framework of CEA-COGEMA collaboration. This programme includes:

- an experimental validation of BUC nuclide build-up, based on Actinide and FP chemical analyses in MOX fuel cuts from PWR assemblies,
- a qualification of the 15 selected BUC-FPs capture cross-sections achieved from reactivity worth measurements by oscillation of specific samples in the MINERVE experimental reactor test zone. The available FPs samples are: Sm147, Sm149, Sm152, Nd143, Nd145, Rh103, Eu153 Gd155, Cs133, Tc99, Ru101, Mo95 and Ag109. Each sample, containing a separated FP, was oscillated in the centre of a 4 % MOX lattice (R1-MOX), 1.26 cm square pitch.

These two programs have been fully accomplished and their results are presented below.

6.1 Experimental validation of BUC nuclides buildup

The fuel cuts come from MOX assemblies irradiated in the French Saint Laurent B1 reactor, corresponding to an initial Pu amount in the central zone of 5.6 % and a maximum burnup of 45 GWd/t.

The fuel inventory is calculated by the cycle code DARWIN. Calculation to Experiment (C/E) comparison is summarized in Tables 6, 7 and 8 for the major actinides, the minor actinides and the BUC fission products respectively.

^{145}Nd + ^{146}Nd is used as a burn up indicator because of its quasi-linear and well-known build-up with the burnup. Conclusions and trends are detailed for each fission product composition calculated.

The recent implementation in the DARWIN package of the "INVERSION" module enables the analysis of isotopes main process of formation, for each kind of fuels and at various burnup. A "PERTURBATION" of main nuclear data, such as capture cross-section, initial isotopic composition, flux, is also available and enables sensitivity studies.

The trends observed from experimental validation, analysis of isotope formation ways and perturbation studies allow us to propose some revisions on nuclear data in order to improve the DARWIN calculation-experiment discrepancies and the future nuclear data files.

In the following Tables, we indicate the uncertainties linked to chemical analyses and burnup determination (assumed to be 2%).

It is important to remind that for burnup credit studies an underestimation of absorbing fission products buildup or an overestimation of fissile isotopes are conservative, but respectively an overestimation and an underestimation are not.

Table VI: Calculation-experiment discrepancies for major actinides (in %) in MOX fuel

	1 cycle (≈ 11 GWd/t)	2 cycles (≈ 28 GWd/t)	3 cycles (≈ 42 GWd/t)	Remarks
U234/ U238	4.40 ± 2.05	-5.04 ± 1.28	-3.12 ± 1.17	Satisfactory
U235/ U238	0.30 ± 0.54	1.19 ± 0.87	1.70 ± 1.37	Satisfactory
U236/ U238	-7.56 ± 1.03	-5.75 ± 0.90	-4.78 ± 0.64	Satisfactory, new U5 LDWL evaluation (JEFF3.0) improve the C/E discrepancy by +4%
Pu238/ U238	-6.25 ± 0.57	-6.31 ± 0.45	-6.32 ± 0.34	JEF2 ^{241}Am capture cross-section needs to be reviewed
Pu239/ U238	0.05 ± 0.70	-0.24 ± 1.13	-1.61 ± 1.44	Satisfactory
Pu240/ U238	0.58 ± 0.17	0.88 ± 0.23	0.66 ± 0.45	Satisfactory
Pu241/ U238	-5.18 ± 0.53	-2.96 ± 0.46	-3.46 ± 0.73	To be improved in JEFF3.0 /13/
Pu242/ U238	-1.39 ± 0.49	-2.71 ± 0.81	-3.90 ± 0.77	To be improved (for ^{243}Am and curium build up) /13/

Table VI points out that major actinide concentrations are well reproduced for Burn up Credit application.

Table VII: Calculation-experiment discrepancies for minor actinides (in %) in MOX fuel

	1 cycle (≈ 11 GWd/t)	2 cycles (≈ 28 GWd/t)	3 cycles (≈ 42 GWd/t)	Remarks
Np237/ U238	-11.27 ± 2.46	-11.06 ± 2.04	-7.44 ± 1.85	(n, 2n) ^{238}U too low in JEF2.2
Am241/ U238	-4.33 ± 1.51	2.30 ± 1.33	1.56 ± 1.63	Satisfactory; σ_c needs to be reviewed
MaM242/ U238	-27.76 ± 1.86	-23.30 ± 1.96	-22.65 ± 2.17	Branching ratios to be reviewed
Am243/ U238	-11.14± 2.05	-7.07 ± 1.85	-5.99 ± 1.60	Improved in JEFF3.0
Cm243/ U238	-39.76 ± 2.14	-22.30 ± 2.07	-16.52 ± 1.51	Underestimation of σc ^{242}Cm (10 to 15%)
Cm244/ U238	-29.54± 2.23	- 6.64 ± 2.33	-5.63 ± 2.12	Improved in JEFF3.0
Cm245/ U238	-30.62 ± 3.23	-8.30 ± 3.00	-7.59 ± 2.42	"
Cm246/ U238	-59.43 ± 4.49	-16.56 ± 4.27	-14.64 ± 3.71	"

The minor actinides are generally underestimated:

- The ^{237}Np underestimation will be soon corrected by the use in the forthcoming JEFF3.0 file of a new ^{238}U(n,2n), higher than the JEF2 one.
- The underestimation in ^{242}Pu generates an underestimation for ^{243}Am, and consequently for ^{244}Cm and ^{245}Cm. This underestimation is embarrassing for BUC studies because of the positive effect of ^{245}Cm on reactivity. Hence, a new evaluation of Pu^{241} will be introduced in JEFF3.0 [14]: the increase of the capture will correct the ^{242}Pu, ^{243}Am and ^{244}Cm build-up.
- The underestimation in ^{243}Cm build-up needs also to be corrected. Studies on ^{242}Cm capture cross-section are in progress.

Table VIII: Calculation-experiment biases (in %) for BUC fission products in MOX fuel

	1 cycle (≈ 11 GWd/t)	2 cycles (≈ 28 GWd/t)	3 cycles (≈ 42 GWd/t)	Remarks
Sm147/ U238	-5.7 ± 3.2	-3.9 ± 2.2	-3.8 ± 1.0	Satisfactory
Sm149/ U238	-7.1 ± 1.3	2.6 ± 2.4	-7.7 ± 1.2	Strongly dependant on power history
Sm150/ U238	-7.1 ± 4.4	-6.3 ± 4.4	-5.7 ± 1.2	Depending on ^{149}Sm calculation
Sm151/ U238	-8.1 ± 1.4	-1.2 ± 1.6	1.9 ± 0.6	Satisfactory
Sm152/ U238	-5.3 ± 3.9	-1.4 ± 2.9	2.9 ± 1.5	Satisfactory
Nd143/ U238	0.0 ± 3.8	0.2 ± 3.4	1.1 ± 1.7	Satisfactory
Nd145/ U238	0.3 ± 3.9	0.2 ± 3.8	0.4 ± 2.1	Satisfactory
Eu153/ U238	-2.7 ± 5.3	4.0 ± 4.7	8.7 ± 2.7	Overestimation of ^{151}Pm and ^{153}Sm FY
Eu154/ U238	-0.9 ± 8.7	13.2 ± 9.9	41.5 ± 7.4	Revision of JEF2 resonance parameters
Eu155/ U238	83.8 ± 4.0	61.6 ± 7.6	24.4 ± 4.3	Revision of JEF2 resonance parameters
Gd154/ U238	-0.3 ± 9.0	14.2 ± 9.8	38.5 ± 6.7	Depending on 154Eu calculation
Gd155/ U238	68.8 ± 3.7	53.9 ± 5.4	14.4 ± 4.6	Depending on 155Eu calculation
Rh103/ U238	28.0 ± 6.2	57.2 ± 5.7	67.8 ± 3.3	Chemical analyses will be remake
Mo95/ U238	3.8 ± 5.7	16.2 ± 6.1	27.3 ± 3.6	Chemical analyses will be remake
Tc99/ U238	15.4 ± 6.6	34.9 ± 7.0	46.2 ± 4.1	Chemical analyses will be remake
Ru101/ U238	20.4 ± 6.1	53.1 ± 7.3	70.4 ± 4.4	Chemical analyses will be remake
Ag109/ U238	87.8 ± 9.0	36.351 ± 6.6	22.8 ± 3.6	Chemical analyses will be remake
Cs133/ U238	-2.2 ± 4.2	-1.0 ± 3.6	0.1 ± 1.9	Satisfactory

In order to check Gd155 build-up, Eu154, Eu155 and Gd154 isotopes were also investigated (see results in Table VIII).

The DARWIN calculation is satisfactory for the main samarium isotopes. The fluctuation observed in the calculation to experiment comparison for Sm149 amounts might be due to the extreme sensitivity to the Sm149 build-up to the power history mesh. The BUC Nd and Cs isotopes are also very well predicted.

For metallic fission products such as Mo95, Tc99, Ru101 and Rh103 the chemical dissolution process is not yet optimized; non-soluble deposits yield to inaccurate analyses. New results should be obtained at the end of 2002 with a new process of dissolution.

Gd155 is strongly over-calculated but agrees with its father, Eu155, which is produce by capture on Eu154 and directly by fission (20% at 50 GWd/t). Recent studies on Eu154 and Eu155 capture cross-section show that the sections used in JEF2.2 file are not satisfactory and will be replaced by ENDFB6 evaluation in the new JEFF3.0 file [15]. The use of this evaluation leads to the following results on Gd155 build-up.

Table IX: Gd155 (C-E/E) (%) results using □$_c$ from ENDFB6.7 for Eu154 and Eu155

BU (MWd/t)	*13200*	*29200*	*43000*	*46000*
Gd155/U238 Eu154 and Eu155 ***JEF2.2***	68.8 ± 3.7	53.9 ± 5.4	9.7 ± 6.4	19.4 ± 6.6
Gd155/U238 Eu154 and Eu155 ***ENDFB6.7***	19.3 ± 3.7	11.3 ± 5.4	2.1 ± 6.4	2.0 ± 6.6

The use of both europium 154 and europium 155 capture cross-section from the ENDFB6.7 evaluation leads to a satisfactory gadolinium 155 calculation build-up: the C/E discrepancy is now 2% at 45 GWd/t.

6.2 Qualification of BUC-FPs capture cross-sections

1. Description of Reactivity Worth Measurements in Minerve

The goal of the reactivity worth measurements by the oscillation technique in the MINERVE reactor is the qualification of the calculation tools and data used to predict the poisoning worth of the selected burn up credit nuclides.

The oscillation technique consists in oscillating periodically the central pin of the MELODIE lattice through the core, so the sample under study is alternatively in and out of the core. A rotating control rod is automatically operated so as to maintain the count rate of a flux detector. The corrected rotation amplitude is in a very close linear relationship with the sample reactivity.

One measurement for each sample corresponds to 30 oscillations with a period of 60 seconds. Each sample is measured at least three times in order to avoid systematic errors. The fission products poisoning worth is directly derived by subtracting the fission product sample reactivity from a reference sample reactivity that differs just by its miss of fission products. The uncertainty on the measured worth is roughly ± 0.7 % in one standard deviation.

Effective fission products cross-sections are obtained in terms of the boron effective cross-section, from the comparison between the measured poisoning worth for the fission product and boron samples; thus a supplementary uncertainty must be added due to this reactivity worth calibration (± 1 % systematic uncertainty).

The oscillation samples have identical geometries, specifically 100 mm-long sections of PWR pins contained in thin, water tight zircaloy sleeves with an external length of 103.5 mm. Two kinds of PWR-type samples were manufactured:

➥ *Calibration samples*: fresh UO_2 with increasing U enrichment and borated samples to relate experimental signal and calculated reactivity,

➥ *Separated fission product isotopes added with different matrices (natural UO_2 or inert)* to validate every BUC Fission Product poisoning worth.

2. Analysis of the Minerve Experiments

The MINERVE reactivity worth measurements were simulated with the French criticality calculation package CRISTAL

The CRISTAL package is based on the recent APOLLO2 code and its CEA-93 library. CEA-93 library is a 172-group data set based on the European JEF-2.2 File.

Calculation to experiment comparisons on separated fission products reactivity worth in MOX configurations are summarized in Table X.

This Table shows a satisfactory agreement using JEF2.2 library in APOLLO2 calculations. This agreement is mostly due to the refined 172 energy-groups structure in the epithermal resonance range. Therefore, the resonance self-shielding was rigorously calculated, for all fission products presented on Table X, through effective cross-section formalism.

However, the capture cross-section of thermal absorbers is slightly underestimated
(- 6 %±3 % for ^{149}Sm and - 5 %±4 % for ^{155}Gd).

^{103}Rh (n,γ) cross-section and ^{133}Cs resonant capture tends to be overestimated by approximately 8 %±4 % and + 6 %±3 % respectively. Studies on these nuclear data are in progress in the JEFF3 project, with an improvement of the ^{133}Cs evaluation in the JEFF3.0 file [16].

^{95}Mo and ^{145}Nd capture cross-sections seem to be overestimated. These trends are not consistent with those obtained in the UOX configuration [17] (see Table X); however oscillation measurements in MOX lattice are less accurate than UOx lattice measurements (decrease of neutronic Importance at the center of the MINERVE MOX lattice).

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Table X: Fission products reactivity worth (C-E)/E and experimental uncertainty (1σ) in % in MOX configuration

Fission Product	MOX (C-E)/E ± 1σ
^{149}Sm	-6.2 ± 3.6
^{103}Rh	+7.7 ± 5.3
^{143}Nd	-0.7 ± 4.6
^{133}Cs	+6.5 ± 2.3
^{155}Gd	-4.5 ± 4.3
^{152}Sm	-2.6 ± 5.1
^{99}Tc	+4.1 ± 5.3
^{145}Nd	+7.0 ± 6.4
^{153}Eu	-1.6 ± 4.1
^{95}Mo	+9.2 ± 3.0
^{147}Sm	-0.3 ± 2.8
^{109}Ag	-0.7 ± 3.4

Table XI: Fission products reactivity worth (C-E)/E and experimental uncertainty (1σ) in % in UOx configuration

Fission Product	UOX (C-E)/E ± 1σ
^{149}Sm	-4.8 ± 1.6
^{103}Rh	+10.0 ± 2.8
^{143}Nd	-7.6 ± 2.4
^{133}Cs	+5.5 ± 1.5
^{155}Gd	-3.9 ± 2.3
^{152}Sm	-1.7 ± 2.3
^{99}Tc	+3.7 ± 2.6
^{145}Nd	+0.7 ± 2.9
^{153}Eu	-2.9 ± 3.0
^{95}Mo	-0.7 ± 1.8
^{147}Sm	+1.9 ± 3.2
^{109}Ag	-2.8 ± 2.0

7. CONCLUSION

The methodology to account for Burnup Credit in spent MOX fuel has been defined, with the selection of the BUC nuclides and the validation of the calculation route in the CRISTAL safety-criticality package. A large experimental programme, based on chemical assays and MINERVE reactivity worth measurements, has been undertaken to support the application of Burnup Credit in the MOX fuel cycle. This programme allowed the validation of DARWIN and CRISTAL packages involved in burnup credit calculation. The depletion code, DARWIN, has shown its capability to simulate the fuel inventory versus burn up for most of burn up credit nuclides. Nevertheless, isotopic measurement has to be improved to get precise values for metallic fission products. Concerning the reactivity measurements performed in the MINERVE reactor, the analysis of the MOX configurations was achieved using the recent CRISTAL criticality package. The results demonstrate the ability of CRISTAL to account for burn up credit, originating in the quality of the nuclear data (JEF-2.2) and the accuracy of the modern calculation methods.

The Burnup Credit implementation in MOX fuel storage should allow rack tightening. Furthermore, BUC-FPs in safety criticality analysis would authorize spent MOX assembly dissolution at La Hague reprocessing plant without gadolinium soluble poisoning.

We can conclude that the DARWIN-CRISTAL system is well suited for BUC calculations on MOX assemblies. The experimental validation of this system is completed therefore BUC strategy could be implemented in MOX fuel cycle. It is foreseen to use BUC in French MOX facilities around 2005.

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