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IMPLEMENTATION OF
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FOREWORD

A Technical Committee Meeting (TCM) on Implementation of Burnup Credit in Spent Fuel Management Systems was convened on 10 to 14 July 2000 by the International Atomic Energy Agency in Vienna. The purpose of the meeting was to explore the status of international activities related to the use of burnup credit for spent fuel applications.

The TCM was the second major meeting on the uses of burnup credit for spent fuel management systems held since the IAEA began to monitor the uses of burnup credit 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. Several consultants meetings were held since July 1997 to advise and assist the IAEA in planning and conducting its burnup credit activities. The proceedings of the 1997 AGM, which explored worldwide use and interest in using burnup credit in spent fuel management systems, are reported in the IAEA-TECDOC-1013, Implementation of Burnup Credit in Spent Fuel Management Systems, published in 1998.

Burnup credit (BUC) for wet and dry storage systems is needed in many Member States to allow for increased initial fuel enrichment, and to increase the storage capacity and thus to avoid the need for extensive modifications of the spent fuel management systems involved.

The IAEA wishes to thank all participants of the Technical Committee meeting for their fruitful contributions. 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.

EDITORIAL NOTE

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IMPLEMENTATION OF BURNUP CREDIT IN SPENT FUEL MANAGEMENT SYSTEMS

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Abstract

Improved calculational methods allow one to take credit for the reactivity reduction associated with fuel burnup. This means reducing the analysis conservatism while maintaining an adequate criticality safety margin.
The motivation for using burnup credit in criticality safety applications is based on economic considerations and additional benefits contributing to public health and safety and resource conservation. Interest in the implementation of burnup credit has been shown by many countries.
In 1997, the International Atomic Energy Agency (IAEA) started a task to monitor the implementation of burnup credit in spent fuel management systems, to provide a forum to exchange information, to discuss the matter and to gather and disseminate information on the status of national practices of burnup credit implementation in the Member States. The task addresses current and future aspects of burnup credit. This task was continued during the following years.

The criticality safety analysis of spent fuel management systems has traditionally assumed that the fuel is fresh. This results in significant conservatism in the calculated value of the system's reactivity. Improved calculational methods allow one to take credit for the reactivity reduction associated with fuel burnup. This means reducing the analysis conservatism while maintaining an adequate criticality safety margin.

First a short overview: (Fig. 1)

This diagram shows the network related to the criticality safety acceptance criteria and reflects also the topics of our meeting.

1. On the left hand side you have the validation of depletion codes with input from core measurements, measurements of reactivity coefficients and chemical assays.
2. On the right hand side the validation of the criticality calculation codes through different experimental results (topic “Depletion and Criticality Calculation and Code Validation”).
3. The criticality safety acceptance criteria are also influenced by the regulatory requirements (topic “Regulatory Aspects”) and burnup profiles (topic “Parameters Affecting Burnup Credit”).
4. The verification of the loading procedure is part of the “Implementation Issues”.

The motivation for using burnup credit in criticality safety applications is generally based on economic considerations. Although economics may be a primary factor in deciding to use burnup credit, other benefits are also realized. Many of the additional benefits of burnup credit that are not strictly economic, are contributing to public health and safety, resource conservation, and environmental quality (this results in less transports, smaller storage facilities, use of less gadolinium in dissolvers, etc.).

Interest in the implementation of burnup credit has been shown by many countries.

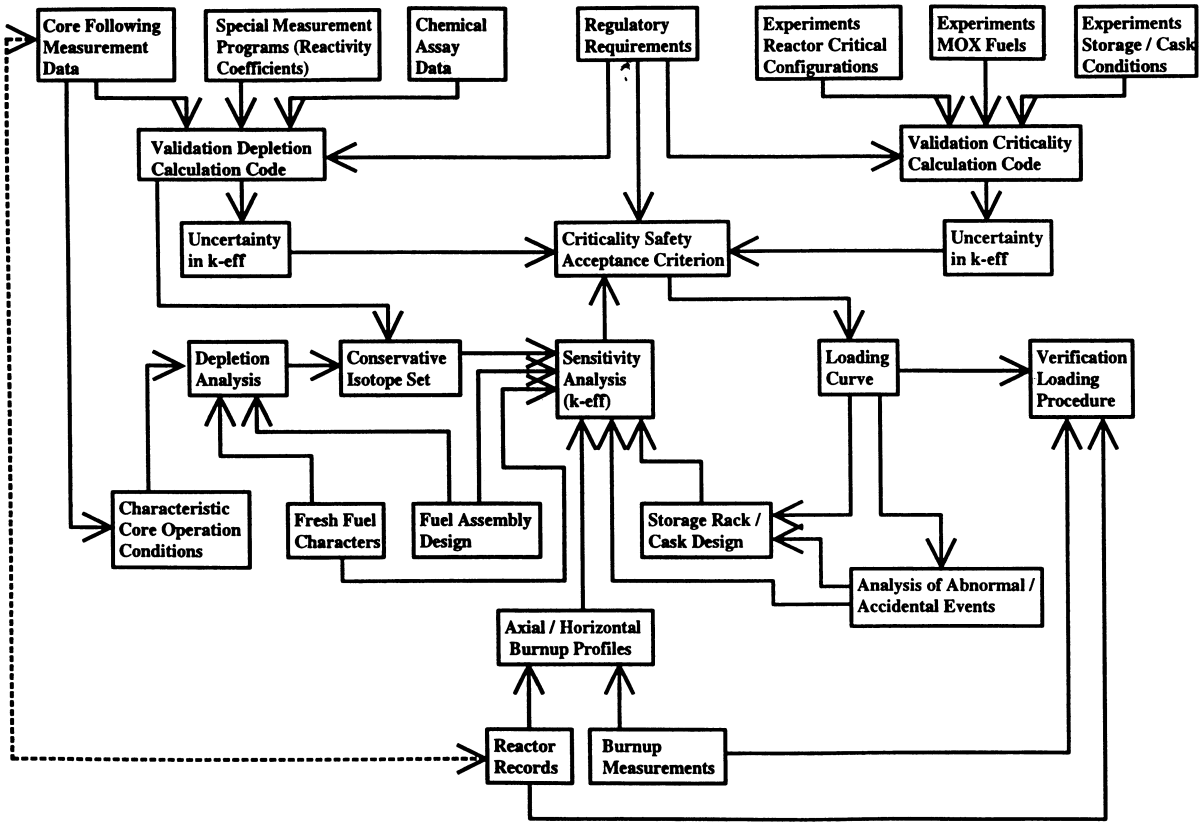

Figure 1. Overview.

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

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

A summary of the information gathered by the IAEA about ongoing activities and regulatory status of burnup credit in different countries is included in Table 1. This table has to be updated in this meeting.

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

In the following years we held some Consultancies to monitor the progress in burnup credit implementation. The results were published in two Working Materials which were widely distributed and are probably known to all the participants.

Table 1. Worldwide uses of burnup credit: National practice and status(as of 19 December 1997).

COUNTRIES	STORAGE									TRANSPORTATION								REPROCESSING				DISPOSAL			
	WET					DRY				WET				DRY											
	PWR	BWR	RBMK	MOX	WWER	PWR	BWR	MOX	WWER	PWR	BWR	MOX	WWER	PWR	BWR	MOX	WWER	PWR	BWR	MOX	WWER	PWR	BWR	MOX	WWER
BULGARIA	-	-	-	-	IC	-	-	-	IC	-	-	-	IC	-	-	-	IC	-	-	-	-	-	-	-	-
CZECH REP.	-	-	-	-	IC	-	-	-	IC	-	-	-	-	-	-	-	IC	-	-	-	-	-	-	-	IC
FRANCE	AP	-	-	IC	-	-	IC	IC	-	AP	IC	IC	IC	AP	IC	IC	IC	AP	RR	IC	-	-	-	-	-
GERMANY	RR	AP[1]	-	IC	NO	IC	IC	IC	IC	-	-	-	-	UD	AP	IC	IC	-	-	-	-	IC	IC	IC	IC
HUNGARY	-	-	-	-	NO	-	-	-	IC	-	-	-	-	-	-	-	IC	-	-	-	-	-	-	-	-
JAPAN	IC	IC	-	UD	-	IC	IC	UD	-	IC	IC	-	IC	UD	UD	IC	-	IC	IC	IC	-	-	-	-	-
KOREA, REP.	AP	-	-	-	-	IC	-	-	-	IC	-	-	-	IC	-	-	-	-	-	-	-	-	-	-	-
LITHUANIA	-	-	AP[2]	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
RUSSIAN FED.	-	-	AP[3]	-	IC	-	-	-	IC	-	-	-	AP[4]	-	-	-	IC	-	-	-	IC	-	-	-	-
SLOVAKIA	-	-	-	-	IC	-	-	-	IC	-	-	-	IC	-	-	-	IC	-	-	-	-	-	-	-	IC
SPAIN	AP	AP[1]	-	-	-	IC	IC	-	-	-	-	-	-	IC	IC	-	-	-	-	-	-	-	-	-	-
SWEDEN	-	AP[1]	-	-	-	-	-	-	-	-	-	-	-	IC	IC	-	-	-	-	-	-	UD	UD	-	-
SWITZERLAN	NO	AP[1]	-	NO	-	IC	IC	IC	-	-	-	-	-	AP[5]	IC	IC	-	-	-	-	-	-	-	-	-
UK	RR	UD	-	UD	-	IC	IC	IC	IC	RR	UD	UD	-	IC	IC	IC	IC	UD	UD	UD	-	-	-	-	-
USA	AP	IC	-	-	-	RR[6]	IC	-	-	RR	-	-	-	RR	IC	-	-	-	-	-	-	RR	RR	-	-

Note: The Table contains information from countries participating in IAEA meetings on burnup credit implementation and from personal communications.

AP = Approved. IC = Interest/Considering, or Applicable. NO = Applicable but not intended.
RR = Regulatory Review. UD = Under Development. - = Not Applicable.

[1] = Credit for the presence of integral burnable absorbers.
[2] = For Ignalina.
[3] = For Smolenskaja.
[4] = For Kola.
[5] = Approved for one case in connection with reprocessing in foreign plants.
[6] = Use of burnup credit for loading single purpose PWR casks is implemented.

The IAEA has determined that worldwide progress has been sufficient in implementation of burnup credit in spent fuel management systems to merit convening a Technical Committee Meeting (TCM) on the subject. The purpose of this TCM is to report on activities and advances that have occurred since the AGM held in October 1997. Matters addressed at this meeting should proceed from the information reported in IAEA-TECDOC-1013 in April 1998 and table 1 as a main outcome reporting and discussing progress in implementing burnup credit for spent fuel management systems.

The IAEA is interested in current use, technical developments, benefits and regulatory matters related to the use of burnup credit. The TCM is expected to address commercial factors that have influenced decisions on the use of burnup credit. The countries already using burnup credit can provide valuable advice to those proceeding toward its use. The TCM will examine the technical information that is already available, that which is being developed, and that needed for future progress. As regulatory approval is needed for any application of burnup credit in spent fuel management, the TCM intends to study regulatory actions of individual national authorities as a way of benefiting practitioners and regulators everywhere.

OECD/NEA is convening a Working Party on Nuclear Criticality Safety which also includes burnup credit. This working party is studying the physical phenomena and parameters relevant to burnup credit while the Agency in its work is taking account of these phenomena and parameters in a licensing evaluation. The working party can give guidance in identifying and ranking the physical parameters affecting burnup credit while our meeting should focus on burnup credit procedures and methodologies used for safety cases and licensing evaluation.
Ms. Brady-Raap will present in her contribution the work done in this OECD/NEA working party.

CONCLUSIONS

The use of burnup credit to demonstrate criticality safety for spent fuel management activities has gained world-wide interest. Burnup credit is recognized as a means of increasing efficiency for storage, transportation, reprocessing, and disposal of spent fuel.

The regulatory status of burnup credit varies from country to country.

Using burnup credit it is important to verify the spent fuel burnup. (Fig. 2)

VERIFICATION OF SPENT FUEL BURNUP

- **horizontal burnup distribution**
- **axial burnup profiles**
- **presence of integral burnable absorbers**
- **decay of radio isotopes.**

Figure 2. Verification of spent fuel burnup.

OVERVIEW OF THE BURNUP CREDIT ACTIVITIES AT OECD/NEA/NSC

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Abstract

This article summarizes activities of the OECD/NEA Burnup Credit Expert Panel, a subordinate group to the Working Party on Nuclear Criticality Safety (WPNCS). The WPNCS of the OECD/NEA coordinates and carries out work in the domain of criticality safety at the international level. Particular attention is devoted to establishing sound databases required in this area and to addressing issues of high relevance such as burnup credit. The activities of the expert panel are aimed toward improving safety and identifying economic solutions to issues concerning the back-end of the fuel cycle.

The main objective of the activities of the OECD/NEA Burnup Credit Expert Panel 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. The method established by the expert panel for investigating the physics and predictability of burnup credit is based on the specification and comparison of calculational benchmark problems. A wide range of fuel types, including PWR, BWR, MOX, and VVER fuels, has been or are being addressed by the expert panel. The objective and status of each of these benchmark problems is reviewed in this article.

It is important to note that the focus of the expert panel is the comparison of the results submitted by each participant to assess the capability of commonly used code systems, not to quantify the physical phenomena investigated in the comparisons or to make recommendations for licensing action.

1. INTRODUCTION

The importance of the safe handling of fissile materials was recognised 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 had occurred in nuclear weapons programmes. 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. Within regions of applicability established using the existing experimental information, new experiments are not always needed. 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.

A series of criticality benchmark studies addressing issues of storage, dissolution and transportation of nuclear materials was carried out several years ago by an OECD/NEA working group established under the leadership of G. Elliott Whitesides. The results of the work have been published both as NEACRP and NSC reports and presented at international conferences. Results from these benchmarks are widely used; this is confirmed by frequent references made in publications.

OECD/NEA has coordinated the activities of this criticality safety benchmark group for more than a decade. The group has addressed criticality safety issues associated with the storage, dissolution and transportation of nuclear materials. Over time the issues tackled by the group have evolved and expanded in line with needs expressed by the international criticality safety community. The technical competence and composition of the group has also evolved through calling in new members with expertise in the specific issues under investigation. The group began to address broad issues such as methods development, experimental needs and international handbook data in the field of nuclear criticality safety. In 1991, the benchmark group elected to add to their agenda a study of burnup credit criticality benchmarks. [Burnup credit is a term that applies to the reduction in reactivity of burned nuclear fuel due to the change in composition during irradiation.] By 1997 the group and scope of work had grown to such proportion that a formal OECD/NEA working party was organised. The Working Party for Nuclear Criticality Safety was chartered to review the activities of the existing working group and to propose establishing task forces (expert panels) corresponding to new demands on methods development, experimental needs and international handbook data in the field of nuclear criticality safety. [1]

2. WORKING PARTY FOR NUCLEAR CRITICALITY SAFETY (WPNCS)

The scope of the WPNCS covers technical away-from-reactor 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 setting up of specific expert panels to address these issues as deemed appropriate.

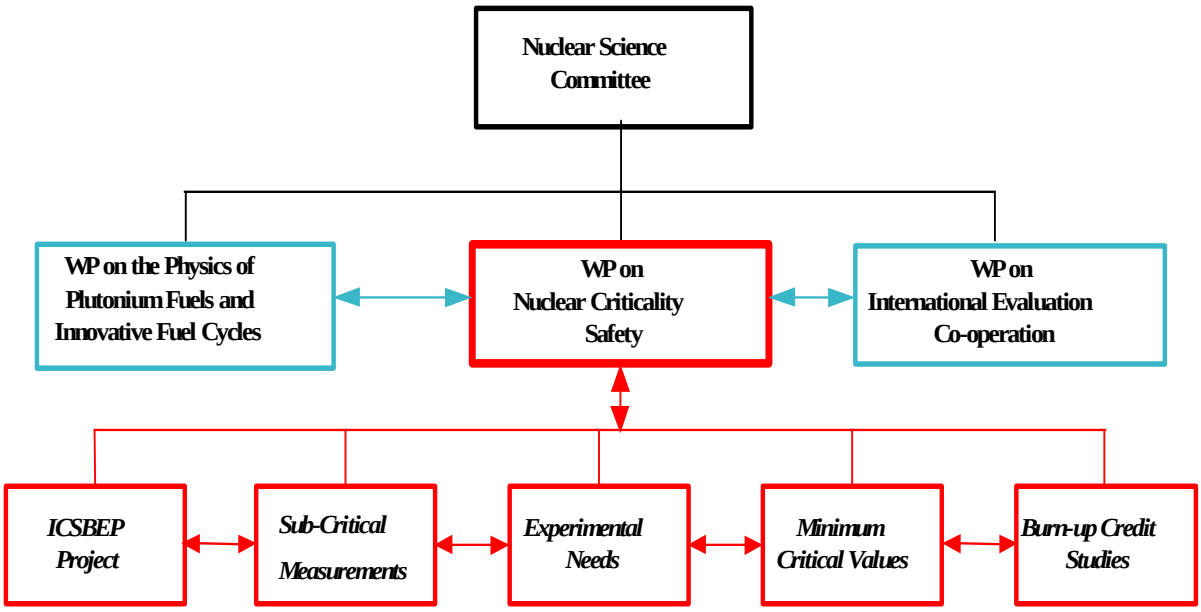

Figure 1. Existing relationship between working parties reporting to the OECD/NEA Nuclear Science Committee and the criticality safety expert groups.

Expert groups have been established for:

1. Developing an experiments database for critical experiments – ICSBEP Project,
2. Developing experiments databases for sub-critical measurements – Sub-Critical Measurements,
3. Identifying needs for critical, subcritical and supercritical experiments – Experimental Needs,
4. Establishing/updating basic criticality condition data – Minimum Critical Values,
5. Verifying the adequacy of existing codes and data for application with burned fuel - Burnup Credit Studies.

Code and data validation and benchmarking and 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. A recent example that shows the different levels of coordination for this working method is the following. Within the Burnup Credit Expert group several issues were identified, which were of wider interest. These were reported to WPNCS for further investigation, e.g.:

1. 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).
2. Effects of geometrical approximations in pin cells, e.g. square versus cylindrical.
3. Mixed configurations of different units with fissionable material.

The focus of this paper is to report on the activities of the Burnup Credit Expert Group.

3. BURNUP CREDIT EXPERT GROUP

The main objective of the activities of the OECD/NEA Burnup Credit Expert Group is to demonstrate that the available criticality safety calculational tools are appropriate for

application to burned fuel systems and that a reasonable safety margin can be established. For this purpose the Expert Group established a suite of burnup credit criticality benchmarks that assess both the capability to calculate both spent fuel composition and reactivity of spent fuel. [2, 3] The benchmarks were carefully specified to allow a comparison of results using a wide variety of calculational tools and nuclear data sets. 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.

Since the objective of the Burnup Credit Expert Group 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 nor 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.

Table I. Summary of benchmark problems addressedby the OECD/NEA Burnup Credit Expert Group.

Benchmark	**Primary Objective**	**Status**
Phase I-A	Examine effects of seven major actinides and 15 major fission products for an infinite array of PWR rods. Isotopic composition specified at 3.6 wt.% ^{235}U at 0, 30 and 40 GWd/MTU and at one- and five-year cooled.	Completed
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
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
Phase II-B	Repeat study of Phase II-A in 3-D geometry representative of a conceptual burnup credit transportation container. Isotopic compositions specified.	Completed
Phase II-C	Key sensitivities in criticality safety to burnup profiles.	Draft specification
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.	Report to be published in 2000
Phase III-B	Compare computed nuclide concentrations for depletion in a BWR pin-cell model.	In progress
Phase IV-A	Investigate burnup credit for MOX spent fuel pin-cell for three plutonium vectors (first recycle, fifth recycle, weapons-grade)	In progress
Phase IV-B	Compare computed nuclide concentrations for depletion in a MOX super-cell.	Draft Specification
Phase V	VVER burnup credit. Similar to Phases I and II for PWRs but with hexagonal geometry and WWER fuel specification	In progress

4. BURNUP CREDIT CALCULATIONAL BENCHMARK PROBLEMS

The benchmark problems were specified to permit the Expert Group to investigate code performance over a variety of physics issues associated with burnup credit. Parameters and effects that have been studied include:

1. The relative contributions from fission products and actinides to the reactivity reduction (k) for burned fuel [light water reactors (LWRs) and VVERs],
2. Trends with burnup and enrichment for LWRs, axially distributed burnup in LWRs,
3. Effects of void distribution for boiling water reactors (BWRs),
4. Identification of sensitive parameters for mixed oxide (MOX) fuels.

It is important to note that the focus of the working group is the comparison of the results submitted by each participant to assess the capability of commonly used code systems, not to quantify the physical phenomena investigated in the comparisons or to make recommendations for licensing action.

4.1. PHASE I/II: PWR Studies

The burnup covered in the PWR studies ranges from fresh fuel to 50 GWd/t and cooling periods from one to five years and varying enrichments.

4.1.1.Phase I-A: Multiplication Factors- PWR Infinite Lattice Studies (1D)

This benchmark consists of 13 cases. Each case is an eigenvalue calculation of a simple infinite lattice of PWR fuel rods. The investigated parameters were burnup, cooling time and combinations of nuclides in the fuel region. The groupings of nuclides include four subgroups: major actinides (U-234, 235, 236, and 238; Pu-239, 240 and 241); minor actinides (Pu-238 and 242; Am-241 and 243; Np-237); major 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) and minor fission products (all others available to participant). The fuel compositions for each case by nuclide were provided as part of the problem specification [4] so that the results could be focused on the calculation of (impacts on) the multiplication factor. In total, 25 sets of results were submitted from 19 institutes in 11 countries. The detailed results are presented in Ref. 5.

Phase I-A is perhaps the most detailed of the benchmark problems in terms of types of data collected and analyzed. Participants provided the following information: codes used, nuclear data libraries, and energy grouping of libraries (group structure or continuous energy); calculated multiplication factor; neutron spectrum in water; neutron spectrum in fuel; absorption rates for all major and minor actinides, major fission products and oxygen; and production rates and neutrons per fission for all major and minor actinides.

Results - Multiplication Factors: Only 17 of the original 25 participants providing solutions were able to execute the problem as specified. Some participants had difficulty incorporating the specified compositions and/or did not have cross section data for all the major fission products. Approximately 23 of the participants were able to successfully complete the actinide-only cases. The results presented in Table II are the average of the results of 17 participants.

Table II. Results of Phase I-A: Average Multiplication Factor, k (2σ), Ref. 5.

Nuclides Set	Fresh Fuel	30 GWd/MTU* 1 yr cooled	40 GWd/MTU* 1 yr cooled	30 GWd/MTU* 5 yr cooled	40 GWd/MTU* 5 yr cooled
All Actinides All Fiss Prod.	1.4378(0.0175)	1.1080(0.0194)		1.0758(0.0185)	
All Actinides No Fiss Prod.		1.2456(0.0107)	1.1885(0.0110)	1.2284(0.0109)	1.1657(0.0099)
Maj Actinides No Fiss Prod.		1.2635(0.0108)	1.2566(0.0109)		
All Actinides Maj Fiss Prod.		1.1402(0.0169)	1.0638(0.0170)	1.1123(0.0164)	1.0240(0.0156)

*Burnup is given in gigawatt days per metric ton initial uranium

An examination of the results in Table II suggest that the largest component of uncertainty originates from the minor fission products as indicated by the larger 2σ values in the cases of "All Fission Products". For all other cases, including those with "Major Fission Products", the 2σ values are smaller than for the case of fresh fuel. The agreement among participants for the "No Fission Product" cases is significantly better than the fresh fuel and fission product cases. No trends in the standard deviation among participants were observed with either burnup or cooling time. Trends in the multiplication factors with burnup and cooling time were as expected; k decreases as both burnup and cooling time increase. The larger 2σ value for the fresh fuel case was expected based on known biases which decrease with fuel depletion [5, 8, 9].

Results - Neutron Spectra: Fourteen participants provided neutron spectra in both the fuel and water. The number of energy groups varied from 27 to 247 and the maximum energy boundaries vary from 20 MeV to 8.2 MeV. Results based on continuous energy data were converted for mutual comparison. The spectra were in quite good agreement. The effects of Pu resonances were clearly seen at approximately 0.3 eV and 1.0 eV in the fuel region and smaller effects at these energies were observed in the moderator region.[5, 9].

Results - Reaction Rates: Seventeen participants supplied the requested reaction rate data. Both the absorption rates and production rates were normalized to unity for comparison. A comparison of absorption rates revealed differences of 0.4 - 0.7% of the total absorption rate for U-238, U-235 and Pu-239. The production rates for these nuclides revealed observed differences among participants of 0.6 to 0.8% of the total production rate. Differences were also observed in the calculated values of neutrons per fission for these nuclides, however there were some discrepancies among participants in the definition of this parameter so the results are not conclusive. Smaller differences in absorption rates (less than 0.1% of the total absorption rate) were observed for Pu-240, Pu-241, Gd-155, Nd-143, Rh-103, Sm-149, Sm-151 and Tc-99.

4.1.2. Phase I-B: Spent Fuel Compositions, PWR Fuel

The purpose of this calculational benchmark problem was to compare computed nuclide concentrations for depletion in a simple pin-cell model. The detailed problem description and results are given in Ref. 6. This benchmark consists of three cases, each with a different burnup. The specific power and boron concentrations for each cycle and cumulative burnup were given in the problem description. Initial isotopic compositions for both the fuel and the moderator were given. Participants were requested to report calculated compositions for the 12 actinides and 15 fission products named in Phase I-A. A total of 21 sets of results were submitted by 16 organizations from 11 countries.

Given that the objective is to ultimately calculate the reactivity of spent fuel, the significance of the differences in nuclide concentrations should be examined from this perspective. As an example of relative importance in the evaluation of multiplication factor, the change in reactivity associated with a change in concentration equal to the observed standard deviation among participants was evaluated.

Table III is a summary of Phase I-B results for important nuclides [have a standard deviation greater than 10% among participants or a change in reactivity greater than 0.01%Δk (10 pcm) per % change in concentration (%ΔN)]. A large standard deviation indicates poor agreement in the calculation of the inventory of a given nuclide. Unlike Phase I-A, trends in the standard deviation with burnup are evident in this study. For many nuclides this trend is relatively small, however the trend of increasing standard deviation with increasing burnup appears to be significant for U-235. A list of nuclides for which further study and comparison of additional information (such as fission product yield data, thermal cross sections, etc.) would be warranted is as follows: Pu-239, Gd-155, U-235, Pu-241, Pu-240, Sm-151, and Sm-149, as these have the largest integral effect on k. Of these nuclides, only Gd-155 and Sm-149 exceed both the 10% standard deviation and a Δk/%ΔN of 0.01%.

4.1.3. Phase II-A: Multiplication Factors-Distributed Burnup Studies (2D)

The configuration considered in this benchmark problem is a laterally infinite array of PWR fuel assemblies with the following characteristics: initial enrichment equal to 3.6 wt % or 4.5 wt %; fuel radius equal to 0.412 cm and array pitch equal to 1.33 cm which leads to a moderation ratio Vmod/Vox = 2.0; different burnups were considered (0, 10, 30 or 50 GWd/MTU) and two cooling times, 1 or 5 years; axially, a symmetrical configuration was adopted including 9 fuel regions (total height = 365.7 cm); and an upper and lower plug and water reflector (30 cm). Specific isotopic compositions were specified for each fuel region and conditions. Cases were analyzed for the axially distributed burnup as well as a uniform burnup assumption equal to the assembly average burnup. The axial burnup profiles used were symmetric about the midplane. As in Phase I-A, the effects of major actinides and fission products were also investigated. Participants were asked to provide calculated multiplication factors and fission densities by axial zone for three cases. Approximately 22 sets of results for the 26 configurations were calculated by 18 different participants from 10 countries.

Details of the problem specification and results for this benchmark are presented in Ref. 7. The average multiplication factors and 2σ values for the 26 cases are summarized in Table IV.

Table III. Summary of Phase I-B Results (Ref. 6).

Nuclide	Case A (27.35 GWd/MTU)[a]		Case B (37.12 GWd/MTU)		Case C (44.34 GWd/MTU)	
	σ[b]	Δk[c] (x100)	σ	Δk (x100)	σ	Δk (x100)
U-235	2.98	0.4410	6.01	0.6485	8.12	0.6285
Pu-238	15.68	0.0329	14.80	0.0562	13.86	0.0679
Pu-239	5.16	0.7085	6.08	1.0611	7.12	1.3962
Pu-240	3.95	0.2054	4.27	0.2404	5.27	0.2772
Pu-241	6.45	0.2219	5.97	0.3248	6.86	0.4583
Am-243	11.31	0.0079	10.41	0.0198	10.40	0.0302
Ag-109	11.03	0.0143	10.61	0.0191	10.21	0.0214
Sm-149	14.14	0.1386	15.01	0.1471	15.61	0.1499
Sm-150	5.30	0.0090	7.07	0.0177	8.50	0.0255
Sm-151	22.41	0.1502	21.72	0.1434	22.31	0.1539
Sm-152	7.20	0.0331	9.01	0.0469	9.68	0.0503
Gd-155	33.45	0.5252	33.28	0.8120	32.97	0.9792

[a] Burnup is given in gigawatt days per metric ton uranium.
[b] The standard deviation among participant results.
[c] Represents an example of the change in multiplication factor times 100 from a one σ change in isotopic composition. The quantity is given as a positive value since the change in composition may be +/-.

Results - Comparison of Multiplication Factors: No significant trends in the agreement among participants (2σ values) were observed with initial enrichment or burnup. As in Phase I-A, the inclusion of fission products results in a greater deviation among participants (larger 2σ values). No clear trends were observed with the inclusion of the axially distributed burnup, although cases with both high burnup (greater than 10 GWd/MTU) and with fission products have some indications of increasing 2σ when axially distributed burnup is considered. At higher burnup (50 GWd/MTU with and without fission products) there is a suggestion of a trend in 2σ with cooling time. Comparisons of multiplication factors from this benchmark with corresponding cases in Phase I-A indicate that the axial leakage is small. Overall, the most interesting result in this benchmark is that the largest discrepancy (2σ) among participants is still seen for the fresh fuel cases.

Results - End Effect: In this study, the "end effect" was defined as the difference in the multiplication factors between the corresponding cases with and without an axial burnup distribution. Tendencies were observed in the multiplication factors that indicate an increase in end effect with increasing burnup. It is very important to note that the end effect is calculated as the difference of two close values and, therefore, has large calculated standard deviations, from 25% to greater than 100% of the value calculated for the end effect (in most cases approximately 75%). Although these tendencies are believed to be representative in general, the effects of both neutron leakage and axial asymmetry of material composition (which was not considered here) may make a considerable difference in the magnitude of the end effect.

Results - Fission Density: The fission density data provided by the participants was found to be in relatively good agreement. The data illustrate the importance of the end regions, approximately 70% of the total fissions occurred in the upper 40 cm of the fuel (representing approximately 22% of the total fuel volume). Therefore, adequate modeling and convergence at the fuel ends are essential to obtain reliable eigenvalues for highly irradiated spent fuel systems.

Table IV. Summary of Phase II-A Results, Average Multiplication Factor (Ref. 7).

Case	Initial Enrichment	Burnup GWd/MTU	Cooling time(yr)	Fission Products	Burnup Profile	k(2σ)
1	3.6 wt %	Fresh	N/A	N/A	N/A	1.4335 (0.0217)
2	3.6 wt %	10	1	Yes	Yes	1.3053 (0.0161)
3	3.6 wt %	10	1	Yes	No	1.3126 (0.0159)
4	3.6 wt %	10	1	No	Yes	1.3607 (0.0175)
5	3.6 wt %	10	1	No	No	1.3665 (0.0174)
6	3.6 wt %	30	1	Yes	Yes	1.1360 (0.0155)
7	3.6 wt %	30	1	Yes	No	1.1358 (0.0138)
8	3.6 wt %	30	1	No	Yes	1.2339 (0.0129)
9	3.6 wt %	30	1	No	No	1.2419 (0.0119)
10	3.6 wt %	30	5	Yes	Yes	1.1160 (0.0144)
11	3.6 wt %	30	5	Yes	No	1.1062 (0.0136)
12	3.6 wt %	30	5	No	Yes	1.2176 (0.0119)
13	3.6 wt %	30	5	No	No	1.2256 (0.0113)
14	4.5 wt %	Fresh	N/A	N/A	N/A	1.4783 (0.0232)
15	4.5 wt %	30	1	Yes	Yes	1.1996 (0.0151)
16	4.5 wt %	30	1	Yes	No	1.2025 (0.0161)
17	4.5 wt %	30	1	No	Yes	1.2972 (0.0145)
18	4.5 wt %	30	1	No	No	1.3064 (0.0139)
19	4.5 wt %	50	1	Yes	Yes	1.0838 (0.0175)
20	4.5 wt %	50	1	Yes	No	1.0584 (0.0136)
21	4.5 wt %	50	1	No	Yes	1.1999 (0.0121)
22	4.5 wt %	50	1	No	No	1.1983 (0.0116)
23	4.5 wt %	50	5	Yes	Yes	1.0543 (0.0156)
24	4.5 wt %	50	5	Yes	No	1.0123 (0.0135)
25	4.5 wt %	50	5	No	Yes	1.1800 (0.0104)
26	4.5 wt %	50	5	No	No	1.1734 (0.0096)

4.1.4. Phase II-B: Multiplication Factors-Distributed Burnup Studies (3D)

In this benchmark problem, a realistic configuration of 21 PWR spent fuel assemblies in a stainless steel transport cask was evaluated. A borated stainless steel basket centered in the flask separates the assemblies. The basket (5x5 array with the 4 corner positions removed) was fully flooded with water. The main characteristics of the fuel assembly are: 17x17 array (289 rods, no guide tubes), water moderated cells with pitch equal to 1.25984 cm; initial fuel enrichment equal to 4.5 wt %; fuel radius equal to 0.4096, fuel rod ID= 0.41785 cm and OD = 0.475 cm which lead to a moderation ratio Vmod/Vox=1.67; as in Phase II-A, the fuel was divided axially into 9 symmetrical zones; burnups of 0, 30 and 50 GWd/MTU and 5 years cooling were used; and the fuel compositions were as specified Phase II-A. Cases were analyzed for the axially distributed burnup as well as a uniform burnup assumption equal to the average burnup. Fourteen participants from 7 different countries submitted partial or complete results (k-eff and fission densities) for the 9 cases specified.

Table V is a summary of the results from this benchmark exercise [8].

Table V. Phase II-B Results - Average Multiplication Factors (Ref. 8).

Case	Initial Enrichment	Burnup GWd/MTU	Cooling time(yr)	Fission Products	Burnup Profile	k (2σ)
1	4.5 wt %	Fresh	N/A	N/A	N/A	1.1257 (0.0135)
2	4.5 wt %	30	5	Yes	No	0.8934 (0.007)
3	4.5 wt %	30	5	No	No	0.9716(0.010)
4	4.5 wt %	30	5	Yes	Yes	0.8953(0.010)
5	4.5 wt %	30	5	No	Yes	0.9647 (0.011)
6	4.5 wt %	50	5	Yes	No	0.7641 (0.005)
7	4.5 wt %	50	5	No	No	0.8737 (0.007)
8	4.5 wt %	50	5	Yes	Yes	0.7933 (0.008)
9	4.5 wt %	50	5	No	Yes	0.8791 (0.010)

Significant differences in the multiplication factors observed for this benchmark relative to Phase II-A are due to differences in the configuration (radially finite, borated stainless basket and stainless steel reflector) and differences in the moderation ratio. There are also significant differences in the calculated standard deviations, which are systematically lower than the corresponding Phase II-A cases. In this benchmark, the trend previously observed indicating increasing dispersion among participant results (higher values of 2σ) for cases including fission products is reversed. In Phase II-B the results with fission products have smaller 2σ values than those cases with no fission products. Consistent with earlier results the highest value of 2σ is for the fresh fuel case. Overall, the agreement among participants is better for Phase II-B than in the Phase II-A benchmark.

The results for the end effects are generally consistent with Phase II-A results. End effect is positive (conservative) for low burnups (below ~30GWd/t) and is increasingly negative as burnup increases above 30GWd/t. The effect of the presence of fission products is illustrated in Figure 2. The "explicit" burnup profile is in reference to the 9-zone symmetrical profile illustrated in Figure 3.

Two additional accident cases highlighted importance of profile if axial heterogeneity present. This means that the importance of the end regions of the fuel is sensitive to the cask geometry, particularly the presence/absence of poison materials under normal conditions at the ends of the cask.

4.1.5.Phase II-C: Proposed Additional Distributed Burnup Studies (3D)

The axial distribution of burnup that is assumed in the evaluation of reactivity for a spent fuel system has been identified as a key parameter for the analysis. Calculations performed to date have shown that the effect varies based on burnup, cooling time and how the burnup distribution and fuel isotopic composition are actually modeled (e.g., number of axial nodes modeled and symmetrical/asymmetrical representations and actinide-only versus including fission products). The reference profile in the above benchmark calculations utilized burnup-dependent profiles based on post-irradiation measurements for a PWR assembly. The profiles conservatively assumed symmetry about the axial midplane using a shape consistent with the bottom half of the fuel assembly. Note that the 'real' profile is clearly asymmetric. Figure 3 illustrates the difference between the real profile (i.e., "continuous") and the 9-zone symmetric representation used in the benchmarks. In this figure, the axial position is the distance from the bottom of the active fuel region of the assembly.

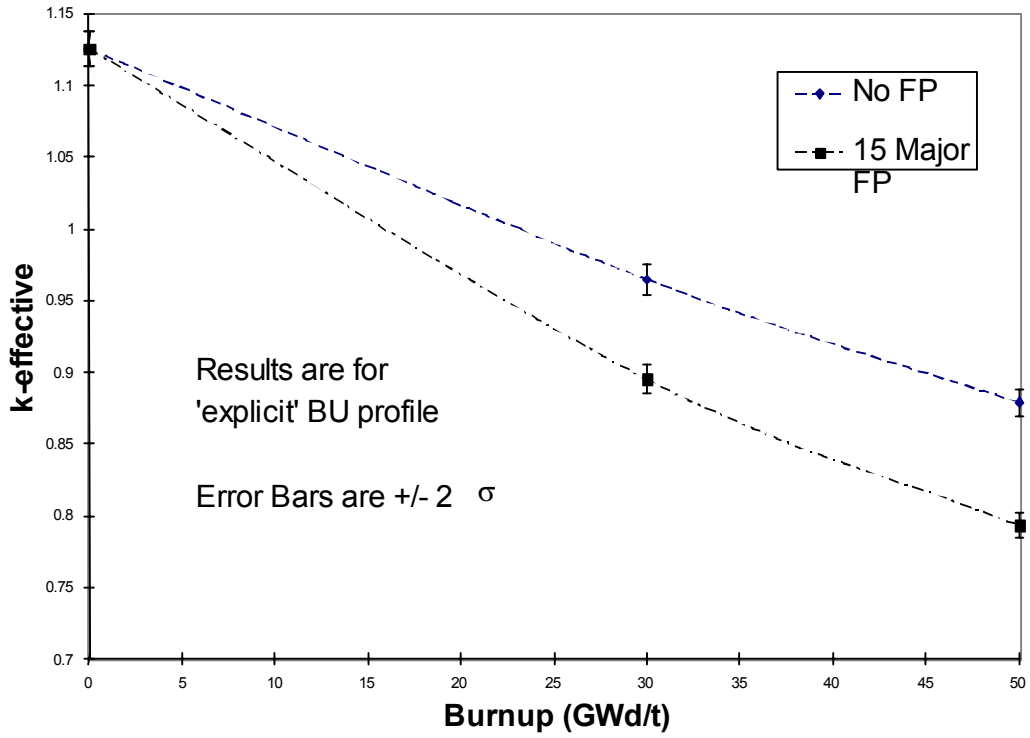

Figure 2. Influence of Fission Products on the Calculated k-effective (Ref. 8).

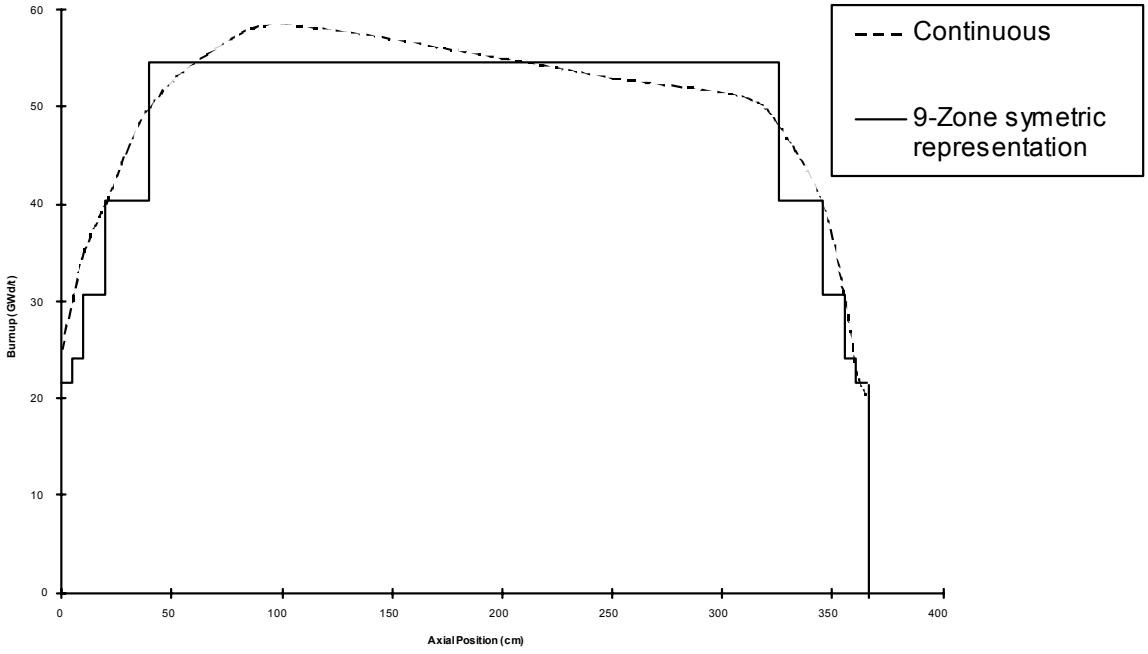

Figure 3. Illustration of Phase II Modeling Assumptions for Axial Burnup Distribution [Burnup(GWd/t) vs Axial Position (cm)].

4.2. PHASE III: BWR Studies

4.2.1. Phase III-A: Reactivity Effects in an Array of BWR pins

This benchmark problem was developed to evaluate the criticality safety of spent boiling water reactor (BWR) fuel in storage facilities or transportation casks. The main features of BWRs important in criticality analyses that differ substantially from PWRs are the moderator void distribution in the core and the complicated composition of a fuel assembly. In BWRs, the moderator void volume fraction is about 70% near the top region of the core and nearly zero near the bottom of the core. The core average void fraction is approximately 40%. A BWR fuel assembly consists of many kinds of fuel rods whose initial enrichments are different from each other. Some fuel rods contain Gd, which is a strong neutron absorber. BWR assemblies also have a large water rod located at their center. For this benchmark problem, the assembly geometry was simplified such that the composition of all the fuel rods in an assembly is considered to be the same. The water rod, cladding, channel box, end plugs and gas plenum are all modeled per the specification. Isotopic compositions for the fuel and water are also given. Twenty-two cases were proposed where burnup varies from 0 to 40 GWd/MTU, fission products are included in some cases, an axial burnup distribution is considered in some cases, an axial void distribution is used in some cases, 40% and 70% uniform void cases are considered and cooling times of 1 and 5 years are specified. Participants were asked to provide calculated multiplication factors and fractional fission densities for five cases.

Results indicate that the largest differences among participants are for the 70% uniform void cases (other than the fresh fuel case). In these cases, the neutron energy spectrum is harder and the plutonium production rate is high compared to the 40% cases and the cases with an axially distributed void fraction.

Table VI. Preliminary Phase III-A Results – Multiplication Factor Trends, based on Average of Participant Results (*all cases for 3.5* wt% U-235).

Burnup GWd/MTU	Cooling Time (yr)	Fission Products	Burnup Profile	Void Profile	Average k
20	1	Yes	Yes	Yes	1.194
20	5	Yes	Yes	Yes	1.182
30	1	Yes	Yes	Yes	1.111
30	5	Yes	Yes	Yes	1.091
40	1	Yes	Yes	Yes	1.027
40	5	Yes	Yes	Yes	0.998
40	5	Yes	No	Yes	0.989
40	5	No	No	No	1.104
40	5	Yes	No	40% (uniform)	0.958
40	5	Yes	No	70% (uniform)	0.998
40	5	No	No	40% (uniform)	1.072
40	5	No	No	70% (uniform)	1.114
40	5	No	Yes	Yes	1.100

4.2.2. Phase III-B: Depletion for BWR Fuel

This benchmark was developed to investigate the ability of evaluation tools to calculate the isotopic composition of irradiated BWR fuel. Unlike the problem specification for Phase III-A, the geometry of the BWR fuel assembly was not simplified for this benchmark. The fuel assembly consists of fuel rods at 5 different initial enrichments and with and without gadolinium (Gd). The initial isotopic composition of each rod and explicit geometry descriptions were specified. As in the Phase III-A specification, the void fraction is varied, cases are evaluated at 0, 40 and 70% uniform void fractions. Number densities for the 12 actinides and 15 fission products of Phase I-A are requested for each of 9 fuel pins in a 1/8 assembly model. The average composition of each of the 5 fuel rod types and assembly average compositions are requested. The calculated burnup for each of the 9 fuel pins is also requested. Participants are also asked to provide neutron multiplication factors for burnups of: 0, 0.2, 10, peak burnup, 20, 30, 40, 50 GWd/MTU for each of the three void fraction cases.

4.3. Phase IV: MOX Studies

4.3.1. Phase IV-A: MOX Pin-Cell Calculations

The first phase of problems to investigate burnup credit for MOX spent fuel pin-cell for three plutonium vectors as indicated in Table VII (first recycle, fifth recycle, weapons-grade). This benchmark problem was started in 1998 and is similar to Phase I-A for PWR UO2 fuel. Isotopics for three types of fresh and burnt MOX assembly were specified in the benchmark specification: single cycle Pu, weapons disposition Pu, and multi-recycle Pu. The isotopics included curium isotopes. Preliminary results from 16 institutions show significantly larger spread than for UO2 cases. The rate of change of k with burnup appears to be strongly related to initial Pu content. Curium isotopes were found to contribute up to 1.5% of the change in reactivity.

Table VII. Initial Distribution of Plutonium Isotopes in MOX Benchmark.

	Isotopic Composition, wt % in Pu_{total}		
NUCLIDE	MOX Case A	MOX Case B	MOX Case C
Pu-238	1.8	0.05	4.0
Pu-239	59.0	93.6	36.0
Pu-240	23.0	6.0	28.0
Pu-241	12.2	0.3	12.0
Pu-242	4.0	0.05	20.0

4.3.2. Phase IV-B: Depletion Calculations for MOX Fuel

The proposed specification has been developed to be consistent with the approach used for studying LWRs. The second phase of the MOX studies are intended to compare computed nuclide concentrations for depletion in a MOX super-cell.

4.4. Phase V: VVER Studies

These studies are similar to Phases I and II for PWRs but with hexagonal geometry and VVER fuel specification. Representatives from the Czech Republic are leading this benchmark.

Preliminary results have been submitted from a number of countries operating VVER reactors.

5. SUMMARY OF CURRENT FINDINGS

The areas that continue to be investigated due to their high importance in the evaluation of burnup credit are isotopic composition and the axial distribution of burnup. The radial distribution of burnup, or flux tilt across an assembly has also been investigated but is not considered a primary effect and can be compensated for in a number of ways including possibly a penalty expressed as a reduction in burnup.

5.1. Isotopic Composition

In support of the burnup credit studies, a Spent Fuel Isotopic Composition Database (SFCOMPCO) has been developed containing data collected from 13 LWRs, including seven PWRs and six BWRs in Europe, the USA and Japan.[10, 11, 12] It is also planned to include measured axial profiles in the database at a later date.

Additionally, several international collaborative efforts have evolved to use to obtain benchmark data to either validate isotopics calculations and/or to validate the reactivity worth of individual fission products. [13, 14, 15, 16,] There are some restrictions on access to the data derived from or to be derived from most of these programs. Examples of these programs are the CERES, ARIANE and REBUS programs. Free access to data continues to be an issue.

5.2. Axial Burnup Distributions

Discussions in the group have led to the conclusion that the effect of the axial distribution of burnup is not well characterized. Also, a "bounding" profile would be dependent on a specific application due to variations in both fuel and flask designs. Accessibility of measured profile data also hinders the determination of a bounding profile. The majority of axial burnup data is derived from in-core measurements. Some participants have stated that they access to some post-irradiation measured profiles that are proprietary and are not generally available. The group reiterated the desire, expressed earlier at the WPNCS meeting, for the WPNCS to ask the NSC to address getting this type of data released for use by the task force.

Phase IIC has been proposed to further study the effect of axial burnup profiles in PWRs. Two separate issues were identified: (1) data needs (measured axial profile data and detailed power history data) and (2) actual problem specification. The purpose of Phase IIC was to be a "complementary study on the sensitivities due to different axial burnup profiles across the full range of burnup". Phase IIC was specifically to include the axial asymmetry. The details of the proposed Phase IIC benchmark are to be presented at this meeting by J.C. Neuber. Measured axial burnup profiles of PWR spent fuel from GKN have been released by Siemens/KWU in the framework of the benchmark studies. Additionally computed axial burnup distributions from 22 PWRs based on data from utilities in the USA have been compiled and are available for use/study by the Expert Group.

5.3. Interface with Other Expert Groups

The group reviews criticality experiments that are applicable to burnup credit to assess their suitability as burnup credit benchmarks. Based on these assessments, the Burnup Credit Expert Group provides feedback to the Expert Group on Experimental Needs.

A liaison is established with the Working Party on International Evaluation Cooperation (as shown in Figure 1) to coordinate data needs. Nuclear data needs in burnup credit primarily concern major and minor actinides as well as fission products. As far as fission products are concerned, 15 fission products that are stable, non-volatile and which contribute to about 75% of the total fission product absorption have been selected for the different studies. They are Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Sm-147, 149,150,151, 152, Nd-143, 145, Eu-153, and Gd-155. Important experimental programs have and are being conducted. They aim at the validation of cross-sections in state-of-the-art evaluations such as ENDF/B-VI.4, JEF-2.2 and JENDL-3.2, and to develop recommendations for needs of re-evaluation. For instance, the JEFF project has set up a specific sub-group on fission product cross-sections addressing these issues.

5.4. Concluding remarks

The issue of burnup credit is of particular importance today as it concerns the different operations involving spent fuel. The number of sessions and papers in ICNC99 (more than 20) was a clear expression of the real need for methods assessment felt in the different countries. The organization of this technical coordination meeting further supports the expectation that burnup credit will play a major role in future licensing evaluations for spent fuel storage, transportation and dissolution.

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BURNUP CREDIT STUDY AND APPLICATION IN SPENT FUEL MANAGEMENT IN CHINA

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Abstract

This paper gives a brief of spent fuel situation of nuclear power plants in China, problems faced with, and measures to be taken. The main research items in this field in next five years are introduced. It is imperative to put burnup credit to use in spent fuel storage, transport and reprocessing.

1. INTRODUCTION

With the rapid development of nuclear electricity in China, two power plants, Daya Bay and Qinshan have come into commercial operation, and other eight units will go into operation during 2003–2005. Spent fuel storage, transport and reprocessing have become an important problem to which regard has been paid in Chinese nuclear industry. The bottleneck of the problem is nuclear criticality safety, as in China all the present design criteria relating to criticality safety of spent fuel storage pool, transport case, and reprocessing technology are in the traditional way, i.e., to design according to the fresh fuel enrichment. With spent fuel accumulating, nuclear power plants and pertaining design and research sections began to consider problems such as how to increase the capacity of storage pool. Adopting burnup credit technology is an effective method to solve the problem. 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 in 1999. CIAE has established basis of theoretical and experimental research in criticality safety, burnup measurement method., and so on.

The design and construction of a pilot plant reprocessing 100tHM per year is underway. It is small and experimental. As no large-scale reprocessing plant is planned to build, long-term storage is the main approach to treat spent fuel in China. The plant has three spent fuel storage pools that can store 1224 spent fuel assemblies altogether.

In China, burnup credit concept has been widely accepted, but technology study in this area is just at its beginning. We lack experience of operable method for concrete project, establishment of regulations and rules, and so on. Thus we are in urgent need of technology assistance and more international academic exchange. [1]

2. SITUATION OF SPENT FUEL IN TWO COMMISSIONED NUCLEAR POWER PLANT 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. With the advanced refueling method the fuel enrichment will

further increase, up to the projected 4.5%. As spent pool storage will soon reach the design capacity; the power plant is prepared for transporting out the spent fuel. Meanwhile, as mentioned early, feasibility of employing burnup credit is investigated to increase the storage capacity. [2] The spent fuel storage pool of Linao Nuclear Power Station under construction will be designed using burnup credit technology to increase its capacity.

The following is the data of Daya Bay Nuclear Power Station 6 batches unloaded spent fuel:

Unit No.	Enrichment	Number	Average Burnup (MWd/tU)
1	1.8%	52	23064
	2.4%	48	23587
	3.1%	44	29802
	3.2%	164	29892
2	1.8%	52	23064
	2.4%	40	23587
	3.1%	52	29802
	3.2%	168	29892

Qinshan Nuclear Power Plant has a 30MW unit. The reactor has undergone four 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). The unloaded spent fuel data is as follows:

Enrichment	Number	AverageBurnup(MWd/tU)
2.4%	40	17000
2.67%	40	20000
3.0%	80	25000

3. MAIN RESEARCH ACTIVITIES FOR THE NEXT FIVE YEARS

In the next five years, the research project on the burnup credit will engaging in the following major aspects:

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. Combining with Chinese actual management and technology level, specific steps and technical nodi to be solved further can be determined then.
2. 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 several years and built some 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.

3. To establish nuclear criticality experiment facility mocking up spent fuel storage to develop 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. However no experimental investigation aiming at NPP spent fuel storage and transport has been developed. It is expected to obtain some criticality data in this area.
4. To investigate burnup measurement methods and develop spent fuel assembly burnup measuring devices, and develop subcriticality system measurement methods. Besides complete theoretical basis and precise calculation program, 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.

In the above mentioned items of research we are hoping to have exchange with IAEA member countries and support from IAEA.

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TCM IMPLEMENTATION OF BURNUP CREDIT IN SPENT FUEL MANAGEMENT SYSTEMS

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Abstract

The paper tries to describe the existing legislative and administrative requirements for the Spent Fuel Facilities licensing in the Czech Republic concerning the Burnup Credit Implementation. It also briefly mentions recent situation in the Spent Fuel Management and the future tasks connected with BUC.

1. INTRODUCTION

Nuclear power represents a significant source of Czech Republic electricity production. The four WWER 440 units at NPP Dukovany contribute to the electricity production in Czech Republic by about 215%. NPP Temelín will increase the ratio of nuclear power on total electricity production about 40%. In May '99, after long evaluations based both on economical and safety features of the Temelín project, the Czech government decided positively about the completion of NPP Temelín. The first fuel assembly was loaded in the core of NPP Temelín Unit 1 in July 2000.

Spent Fuel from the WWER 440 NPP Dukovany after discharging from reactors spends from five to six years cooling period in NPP's at reactor pools. This spent fuel is than stored in a dry interim storage - Interim Spent Fuel Storage Facility (ISFSF) at Dukovany using dual - transport and storage CASTOR - 440/84 casks. The ISFSF Dukovany was commissioned in January 1997. ISFSF consists of a light storage building and its capacity is approximately 600 Mg of heavy metal (60 casks). Its storage capacity could cover spent fuel arisings from the operation of NPP Dukovany only until the year 2005. A storage facility will be built next to the existing one. This facility will also use dual-purpose metallic casks- for storage and transport. The bid has not been finished yet, but there are shortlisted three vendors: GNB, SKODA and Transnucleaire. The initial conditions for these cask licensing were set according usual conservative practice (fresh fuel, fresh water) without Implementation of Burnup Credit (BUC).

2. REGULATION ON SPENT FUEL STORAGE LICENSING

Parliament of the Czech Republic passed new act in January 1997 under No. 18/1997 Coll., on Peaceful Utilization of Nuclear Energy and Ionizing Radiation (the Atomic Act) [1]. The Act was developed with the objective to re-codify utilization of nuclear energy and ionizing radiation, and, especially to modify so far insufficiently regulated issues such as radioactive waste management, liability for nuclear damage, emergency preparedness.

State guarantees safe disposal of all radioactive waste, and a Radioactive Waste Repositories Authority has been set up for this purpose by the Ministry of Industry and Trade. Activities of the Authority are financed from nuclear account with main income represented by payments from radioactive waste generators. This act also establishes the basic principles of spent fuel safe management. The Atomic act does not predict whether to reprocess or dispose the spent

fuel. According to this law spent fuel is not considered the waste, but both the operator and the State Office for Nuclear safety have right to declare spent fuel the waste.

The Atomic Act authorized the SÚJB, and in some specific cases - also other State Administration Bodies, to issue a set of implementing regulations (full text of the regulations can be found in the SÚJB Website www.sujb.cz).

In order to license a spent fuel storage facility there is a complex procedure which can be seen the in Figure 1. Such procedure includes EIA process and According Czechoslovak Act No. 50/1976 Coll., on Civil Construction [2] there are three stages of the licensing process of any construction with a nuclear facility:

1. Siting permit,
2. Construction permit,
3. Operational permit.

The power to issue respective permits is in the hands of a local Construction Authority.

But for each step an approval shall be issued by the SUJB. Basic condition to SUJB approval represents SAR evaluation. Initial, Preliminary and Pre – operational (final) SARs shall be gradually submitted to the SUJB. The contents of them as well as requirements are set by the Atomic Act.
More detailed safety requirements are set namely by Regulation No 195/1999 Coll on Requirements on Nuclear Installations, which in its §47 (Irradiated and Spent Nuclear Fuel Handling and Its Storage) requires.

The installation for the handling with the irradiated and spent nuclear fuel and its storage, and for the handling and storing the other substances containing the fissile products and radioactive substances shall be designed in a such way, in order that it may be possible:

1. To prevent with reserve the achievement of criticality even under conditions of the most effective deceleration of neutrons (optimum moderation) by area arrangement or by other physical means and procedures, and by this to prevent:
   1.1. The exceeding the 0.95 value of effective coefficient of multiplication of neutrons under the assumed accident situations (including the flooding by water),
   1.2. the exceeding the 0.98 value of effective coefficient of multiplication of neutrons under the conditions of optimum moderation,
2. To assure the sufficient remove of residual heat under normal and abnormal operations and under accident conditions,
3. To assure the capability for performance of periodic inspections and tests,
4. To prevent the fall of irradiated fuel during the transport,
5. To reduce to the minimum the possibility of fuel damage, i.e. namely to prevent the exposure of irradiated element or fuel set to the non-allowable load during the handling,
6. To prevent the fall of heavy objects on the fuel set, i.e. the objects with the mass greater than the mass of fuel set is,
7. To enable the storage of damaged fuel elements or damaged fuel sets at the constructions and operational units, the part of which is a nuclear reactor,
8. To assure the radiation protection of nuclear installation personnel,
9. For wet storage with a water charge to assure

9.1. The check-up of chemical composition and of radioactivity of all water, inside of which the irradiated fuel is stored or in which there is a handling with it,
9.2. The monitoring and controlling the height of water level in the spent fuel pool and the leakage detection.

From the text above could be easily derived that no explicit rules for BUC implementation exist in the CR. Anyway the licensing process has been until now based on conservative approach, without BUC.

3. IMPLEMENTATION OF CRITICALLITY EVALUATIONS IN THE STORAGE SYSTEMS AND THE FUTURE TASKS

During licensing Interim Spent Fuel Storage Facility (ISFSF) at Dukovany the requirement on maintaining subcriticality naturally belonged to basic safety criteria. For the SAR consideration the fresh fuel its maximum allowed enrichment 3,6% U235, in fresh water were calculated. Upon the SUJB request were calculated the situations taking into an account optimum moderation, namely the decreasing density of water. No credit was taken for BUC.

During licensing Spent Fuel Pools at Reactors – NPP Dukovany was the situation almost the same as above (ISFSF). There was only one exemption. Partial credit was allowed for one (not compacted yet) upper rack (for emergency unloading) the validity of this decision was limited until re-racking. The SUJB decision was conditioned - prior to its eventual use was necessary to calculate subcriticality and the result consult with the regularor.

Because there appeared several indications, especially from the cask vendors to ask for the future storage-transport cask licenses with the BUC implementation, the intensive studying of the BUC issue in the CR has started. SUJB opened a state supported research project, with the goal to prepare the metrology and detailed safety criteria for the evaluation of BUC in transport and storage systems. The team of experts at the Nuclear Research Institute Rez realizes this project.

4. CONCLUSIONS

Existing experience in the field of the BUC implementation can be briefly summarized as follows:

1. Conservative approach based No significant experience with the BUC yet,
2. Next transport/storage cask to be licensed – indication of the licensee – BUC,
3. A state programs focused on BUC (namely the regulatory aspects) started this year.

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CURRENT APPLICATIONS OF ACTINIDE-ONLY BURNUP CREDIT WITHIN THE COGEMA GROUP AND R&D PROGRAMME TO TAKE FISSION PRODUCTS INTO ACCOUNT

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Abstract

The paper gives a brief overview of use of actinide only burnup credit in spent fuel storage, transport and reprocessing issues in France. Situation is presented at which allowance is made for burnup credit. Various systems used in France to measure burnup are described. The potential benefits to be gained from making allowance for fission products and the future R&D programme are outlined.

1. BURNUP CREDIT BACKGROUND AND CONTEXT

Burnup credit can be defined as making allowance for absorbent radioactive isotopes in criticality studies, in order to optimise safety margins and avoid over-engineering of nuclear facilities.

In the 1980s, when a new reprocessing plant (UP3) was being designed for the site at La Hague, burnup credit had to be taken into account so that the dimensions of the new, higher capacity facilities could be optimised. At the time, however, only the major actinides were considered as being sufficiently qualified to be taken into account.

In the mid-nineties, a joint R&D programme was set up by the French Atomic Energy Commission, the Institute for Nuclear Safety and Protection and COGEMA, aimed at qualifying between six and fifteen stable, non-volatile fission products with the highest neutron capture capability by 2002 or 2003.

This qualification would make it possible to take into account the presence of 6 to 15 fission products in safety analyses, depending on the case.

A programme to develop a burnup measurement system is also underway, so that allowance can be made for the burnup of UOX and MOX fuel assemblies by 2003.

Taking a 17x17 PWR type fuel assembly with a cooling time of five years, Table I below shows that the major actinides currently taken into account in safety analyses account for approximately 19,000 pcm, i.e. 60% of the negative reactivity of all actinides for a burnup of 40 GWd/tU.

TABLE I. MAJOR ACTINIDES TAKEN INTO ACCOUNT

BU	**20 GWd/tU**	**40 GWd/tU**
U236	720	910
Pu238	60	310
Pu240	5720	8370
Pu242	160	710
Am241	610	1290
Actinide total	1000	19000
5 Actinides	*7270 (73%)*	*11590 (61%)*

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. TOTAL NEGATIVE REACTIVITY OF FISSION PRODUCTS

BU	**20 GWd/tU**	**40 GWd/tU**
Sm-149	980	1030
Rh-103	790	1360
Nd-143	530	900
Cs-133	420	750
Gd-155	390	1550
Sm-151	350	500
Sm-152	250	490
Tc-99	240	440
Nd-145	230	410
Eu-153	150	390
Mo-95	150	290
Sm-147	150	230
Sm-150	120	270
Ag-109	100	250
Ru-101	100	220
Total for 200 fission products	6120	11500
Total for 6 fission products	*3460 (56%)*	*6090 (53%)*
Total for 15 fission products	*4950 (81%)*	*9080 (79%)*

2. ADVANTAGES OF BURNUP CREDIT FOR THE COGEMA GROUP AND ITS APPLICATIONS

As far as the COGEMA Group is concerned, the three areas in which burnup credit proves to be an advantage are the transport of spent fuel assemblies, their interim storage in spent fuel pools and spent fuel reprocessing.

In the case of transport, burnup credit means that cask sizes do not need to be altered, despite an increase in the initial enrichment of the fuel assemblies.
Burnup credit also makes it possible to offer new cask designs with higher capacity.

Burnup credit means that fuel assemblies with a higher initial enrichment can be put into interim storage in existing facilities and opens the way to the possibility of more compact ones.

As far as reprocessing is concerned, burnup credit makes it possible to keep up current production rates, despite an increase in the initial enrichment of the fuel assemblies being reprocessed.

3. HOW ALLOWANCE IS MADE FOR BURNUP ON AN INDUSTRIAL SCALE TODAY

3.1. The two levels at which allowance is made for burnup credit

The methodology currently used for making allowance for burnup credit on spent fuel assemblies comprises two levels, summarised in the table below.

TABLE III. LEVELS OF ALLOWANCE FOR BURNUP CREDIT

Levels	Constraints	Profits
1	Simple irradiation check Guaranteed quality monitoring	Credit of a minimum cycle in core (3200 MWd/tU today)
2	Quantitative measurement of the minimum burnup	BUC in the least irradiated 50 cm

The first level at which allowance is made for burnup credit allows for a value of 3200 MWd/tU only, corresponding to the minimum irradiation of a fuel assembly in a reactor. A simple fuel irradiation check suffices.

The second level authorises the taking into account of the burnup of the least irradiated 50 cm in criticality safety studies. This burnup must be guaranteed by a suitable burnup measurement.

3.2. Using burnup credit on an industrial scale today

3.2.1. Transport and interim storage in spent fuel pools

In the case of spent fuel assemblies from EDF reactors, a simple irradiation check is currently carried out, since a burnup credit of 3200 MWd/tU is enough to ensure that the fuel assemblies can be transported and put into interim storage at La Hague.

However, a suitable burnup measurement is required for fuel assemblies removed from power plants outside France (Germany, Switzerland and Belgium), since they are either more highly enriched or have larger cross-sections (this is the case of spent fuel assemblies used in 1300 MWe reactors in Germany). This measurement is made in the reactor spent fuel pool.

3.2.2. Reprocessing

A systematic burnup measurement is made before fuel assemblies are reprocessed at the La Hague complex. But even though this measurement is not strictly necessary (since no allowance is made for burnup credit), it is made to validate the data supplied by the reactor operator.

3.3. Link between criticality safety studies and burnup measurement

The following table summarises the methodology used for making allowance for burnup credit in terms of criticality safety studies and the associated burnup measurements.

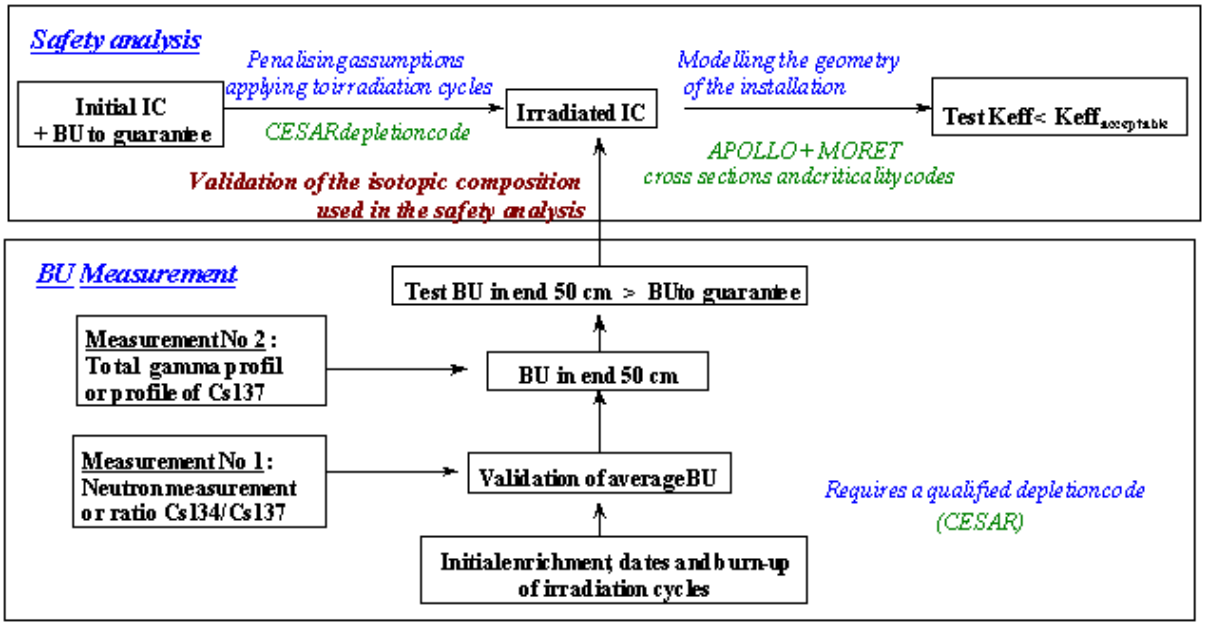

FIG. 1.Link between safety studies and measurement.

3.3.1 Criticality safety analysis

When the initial enrichment of the fuel is such that no adequate guarantee of sub-criticality can be provided with the geometry studied (K_{eff} > K_{eff} acceptable), a burnup value has to be used. Allowance for this burnup value and the use of a qualified depletion code (CESAR code, [1]) gives the isotopic composition of the spent fuel. This isotopic composition comprises only isotopes which have been sufficiently qualified (currently major actinides and eventually the main fission products). The isotopes in question are taken into account in the facility modelling study, which is carried out using the APOLLO and MORET codes, allowing the CRISTAL criticality safety package to be used on an industrial scale [2].

3.3.2. Burnup measurement

The aim of burnup measurement is to validate the isotopic composition of the fuel used in the safety analysis. The burnup measurement used makes it possible to validate the mean burnup and thereby determine end burnup (i.e. the burnup in the least irradiated 50 cm).

Regardless of the system used, measurement is made in two steps.

The first step consists in validating the mean burnup of a fuel assembly using a certain amount of data supplied by the reactor (the initial enrichment and various irradiation cycles need to be known).

Two types of measurement can be used for this first step: neutron measurement or gamma spectrometry.

The second step consists in measuring end burnup. This is defined as being the ratio between the count rate in the end 50 cm and the average count rate, multiplied by the mean burnup of the fuel assembly, once all uncertainties have been removed.

4. THE VARIOUS BURNUP MEASUREMENT SYSTEMS

There are two burnup measurement systems. The easiest to use is called PYTHON (installed in some European reactors, see Section 3.2.1) and includes passive neutron measurement for determining mean burnup and a total gamma probe which, through acquisition of the gamma profile of the fuel assembly, can be used to determine burnup in the end 50 cm.

A second method is used at the La Hague plant. The first stage is to extract ^{134}Cs and ^{137}Cs rays by gamma spectrometry (using a germanium detector), the ratio of activity of these two isotopes being proportional to burnup. The second stage is then carried out directly by the same detector which extracts the ^{137}Cs profile.

Gamma spectrometry has the advantage of requiring no initial calibration and it also validates the irradiation history (since any error in history introduced as initial data results in non-validation of the burnup). However, this method applies only to the first few rows of fuel rods. Neutron measurement has the advantage of being representative of all the fuel rods in the fuel assembly and exhibiting a higher degree of sensitivity to high burnups. It is unaffected by irradiation history. But it does require to be calibrated using one or more fuel assemblies with known characteristics.

The interpretation principle for both measurement methods is shown in Figure 2 below.

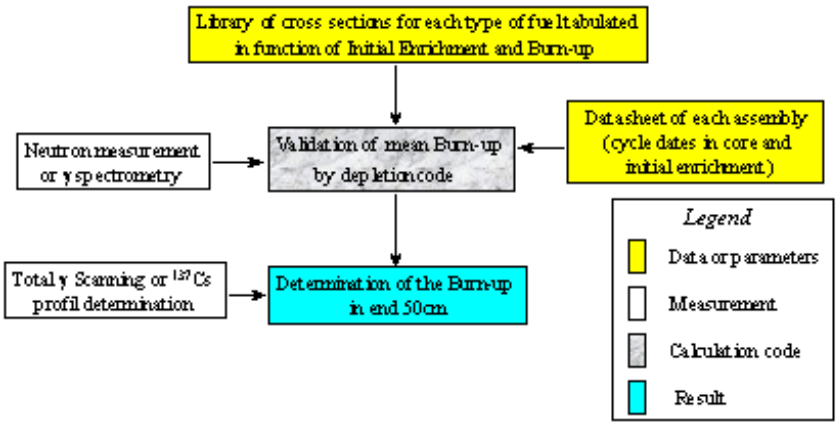

FIG. 2.Interpretation principle for both measurement methods.

An effective cross-section library adapted to the fuel to be measured must be created beforehand. The library is then integrated into the measurement system. In conjunction with a simplified depletion code, the library can be used to determine a correlation between the burnup to be validated and the parameters measured (fuel assembly neutron emission or ratio between ^{134}Cs and ^{137}Cs activities). Burnup can then be calculated using these measurements.

4.1. Mean burnup validation by neutron measurement

In the case of neutron measurement, the principle for determining burnup is as follows:

where

NE neutron emission (n/s/tU),
NC neutron count per second,

we have NE = a.BU b and NC = c.NE where a and b are established by a depletion code and c is established by calibration.

FIG.3.Burnup determination.

4.2. Validation of mean burnup by gamma spectrometry

Burnup is determined using the following formula:

$BU = a.\ A(^{134}Cs) / A\ (^{137}Cs) + b$

where $A(^{134}Cs)$ and $A\ (^{137}Cs)$ are the areas under the ^{134}Cs and ^{137}Cs peaks respectively, obtained when the fuel assembly is passed in front of the germanium diode.

Constants a and b are determined using a depletion code.

4.3. Determining end burnup

End burnup is determined by total gamma counting or extraction of the ^{137}Cs profile when spectrometry is used to validate mean burnup.

End burnup is obtained using the following formula:

$BUend. = BU\ .\ (\ A - \varepsilon\ )\ /\ (\ A + B\ )$

BU: mean burnup validated by neutron or gamma spectrometry measurement,
ε: all uncertainties,

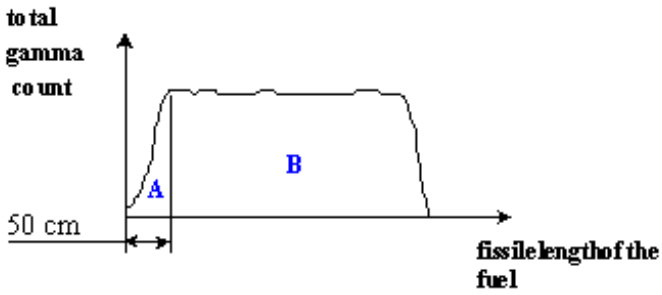

FIG.4.Determining and burnup.

5. POTENTIAL BENEFITS TO BE GAINED FROM MAKING ALLOWANCE FOR FISSION PRODUCTS

In this section, two examples are given of the potential benefits of making allowance for fission products in criticality safety studies. These are simply prospective studies and in no way anticipate the results of the qualifications in hand or any new hypotheses that might be used in future criticality safety studies. The results for the two analyses where allowance was made for fission products were obtained using current criticality safety methodology.

The first analysis involved the interim storage racks currently used in the spent fuel pools at La Hague, which measure 1010 x 1010 mm. The second analysis gives results for smaller racks measuring 900 x 900 mm. In both cases, the racks contain nine compartments coated with a 1.5 mm thick layer of 1% borated stainless steel. The calculation considers an infinite array of racks containing off-centre fuel assemblies (the worst case).

5.1. Interim storage racks at La Hague

Figure 5 shows the results for 17 x 17 PWR type fuel assemblies. The analysis is carried out at optimum moderation, then with a set number of fuel rods (10 fuel rods missing) and a burnup credit making allowance for 0, 6 and finally 15 fission products.

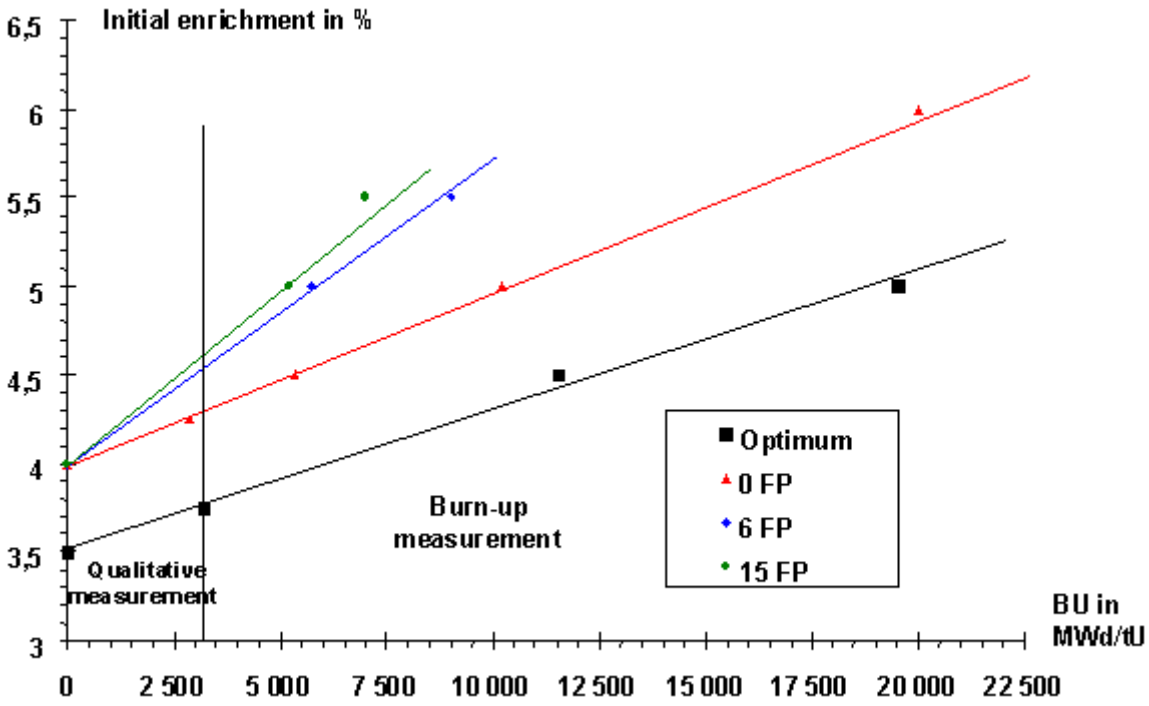

FIG.5.Calculation results for 17 x 17 PWR type fuel assemblies.

Thus, at optimum moderation and with no burnup credit, the maximum allowable enrichment is 3.5%. This increases to 3.75% with a burnup credit of 3200 MWd/tU requiring a simple irradiation check only (Level 1 burnup credit, see Section 3.1). A burnup credit of around 20 GWd/tU is required if fuel assemblies with an initial enrichment of 5% are to be stored.

If it can be guaranteed that less than 10 fuel rods will be missing from the fuel assembly, then the maximum allowable enrichment would be 4% with no burnup credit. With an actinide-

only burnup credit, 10 GWd/tU has to be guaranteed if fuel assemblies with an initial enrichment of 5% are to be stored. When 6 (or 15) fission products are taken into account, a burnup credit of around 6 GWd/tU (or 5 GWd/tU) is required if fuel assemblies with an initial enrichment of 5% are to be stored.

5.2. Smaller racks

As for the previous analysis, Figure 6 gives the results obtained for 17 x 17 PWR type fuel assemblies with 10 missing fuel rods and a burnup credit that makes allowance for 0, 6 and 15 fission products.

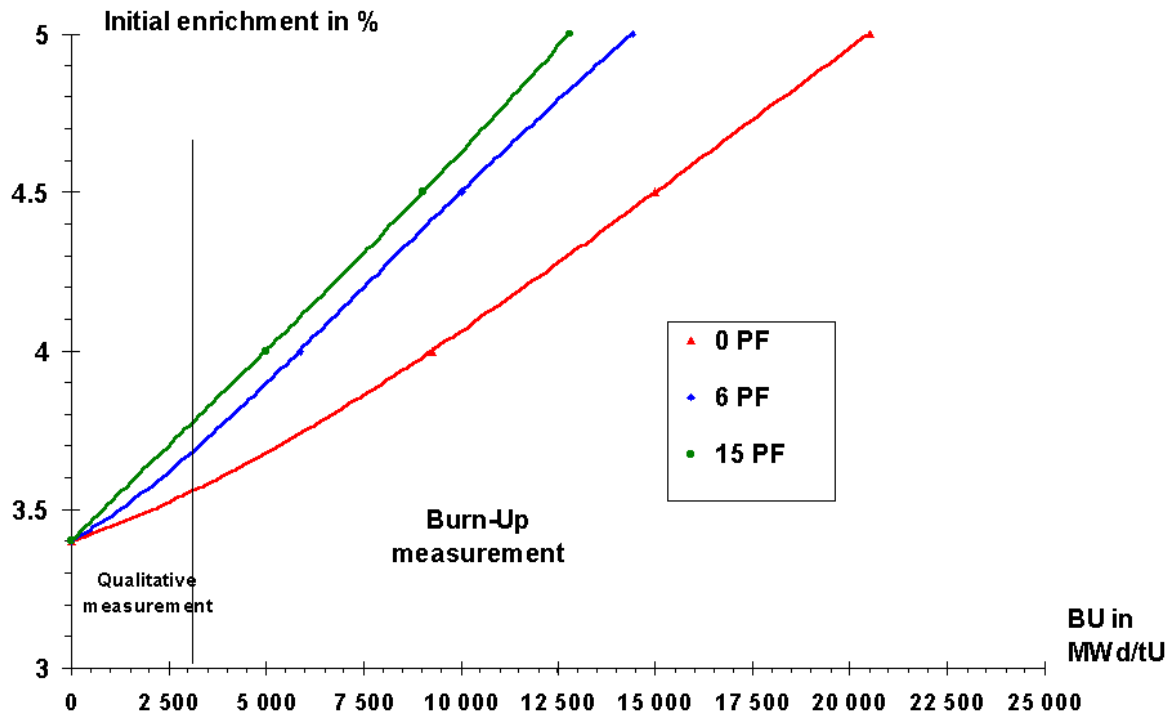

FIG.6. Results for 17 x 17 PWR assemblies with 10 missing rods and burnup credit.

It can be seen that in this case, the maximum allowable enrichment decreases from 4% to 3.4% when there is no burnup credit.
In order to be able to store fuel assemblies with an initial enrichment of 5%, around 20 GWd/tU of burnup credit are required when fission products are not taken into account. Likewise, when allowance is made for six (or fifteen) fission products, a burnup credit of 15 GWd/tU (or 13 GWd/tU) is required.

The second case involving an array of smaller racks therefore requires a higher burnup credit. This is why, even more than in the first case, making allowance for fission product burnup credit is an extremely interesting option.

6. R&D PROGRAMME SET UP

The R&D programme financed by the COGEMA Group as one of its joint interest programmes run in collaboration with the French Atomic Energy Commission and the

Institute for Nuclear Safety and Protection has three objectives. The first is to develop a new burnup measurement system suitable for MOX fuel.

The second is to qualify the source term (abundance and level of capture of fission products) and the last is to qualify the use of fission products for geometrical configurations whose characteristics are close to those of industrial configurations.

6.1. Development of a system for measuring MOX fuel assembly burnup

The objective of this development programme is to be able to estimate the burnup of a MOX fuel assembly with the same level of validation as the PYTHON system used for UOX fuel assemblies. It combines research into the qualification of effective cross-section libraries for MOX fuel assemblies and developments in Cd-Te detector technology. The programme is being run by the SSAE Department at the French Atomic Energy Commission.

6.2. Qualification of the source term

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 oscillation of individual samples of the fission products in the reactor. Oscillation of spent UOX and MOX fuel samples are also underway. These measurements are being carried out by the SPEX Department.
2. The measurements and analysis of samples of spent fuel from French and German pressurised water reactors will allow the SPRC Department of the Atomic Energy Commission at Cadarache to qualify the APOLLO-DARWIN package and the CESAR code [4].

6.3. Qualification of codes for industrial configurations

The programme involves experiments and qualification of the French criticality package CRISTAL.

The experiments are carried out on the Institute for Nuclear Safety and Protection's B apparatus at Valduc by the SRSC Department. 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 [5].

The experiments are interpreted by the Criticality Study Department of the Institute for Nuclear Safety and Protection at Fontenay-aux-Roses with a view to qualifying the industrial version (APOLLO-MORET) of the CRISTAL package [2].

7. CONCLUSION

Actinide-only burnup credit has been an industrial reality for the COGEMA Group for ten years now. The Group is in possession of experience feedback from over 10,000 fuel assemblies whose mean and end burnups have been measured.

In collaboration with the French Atomic Energy Commission and the Institute for Nuclear Safety and Protection, the COGEMA Group is participating in an extensive experimental programme and working to qualify criticality and fuel depletion computer codes. The research programme currently underway should mean that by 2003, allowance will be made for fission products in criticality safety analyses.

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PRESENT STATUS AND FUTURE DEVELOPMENTS OF THE IMPLEMENTATION OF BURNUP CREDIT IN SPENT FUEL MANAGEMENT SYSTEMS IN GERMANY

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Abstract

This paper describes the experience gained in Germany in implementing burnup credit in wet storage and dry transport systems of spent PWR, BWR, and MOX fuel. It gives a survey of the levels of burnup credit presently used, the regulatory status and activities planned, the fuel depletion codes and criticality calculation codes employed, the verification methods used for validating these codes, the modeling assumptions made to ensure that the burnup credit criticality analysis is based on a fuel irradiation history which leads to bounding neutron multiplication factors, and the implementation of procedures used for fuel loading verification.

1. INTRODUCTION

Burnup credit methodologies are implemented in Germany due to significant increases in the initial enrichment of different fuel assembly types:

1. The PWR Nuclear Power Plants (NPP) of the KONVOI type, NECKARWESTHEIM II (GKN II) and EMSLAND (KKE) are increasing their enrichment from 4.0 wt.-% to 4.4 wt.-%,
2. The PWR NPP of the KONVOI type ISAR II (KKI II) is increasing the initial enrichment from 4.0 wt.-% up to 4.6 wt.-%,
3. The BWR NPP GUNDREMMINGEN B and C (GUN B/C) have increased their initial enrichments up to 4.6 wt.-%.

The spent fuel management systems concerned are:

1. The spent fuel storage ponds at plant site,
2. The dry transport casks, meant to be used in future also for interim dry storage at plant site.

The fuel assembly type of the KONVOI plants has a 18x18 lattice with 24 guide thimbles in case of UO_2 fuel and 4 additional water rods in case of MOX fuel. An isolated, water-flooded, unirradiated and unpoisoned fuel assembly of the UO_2 type attains the neutron multiplication factor k_{eff} of 0.95 at an initial enrichment of 4.4 wt.-%. Therefore, all the 4.6 wt.-% enriched fuel assemblies for KKI II are poisoned with a certain number of Gadolinium (Gd) bearing fuel rods.

As is usual with BWR plants GUN B/C has different fuel assembly types in operation. In the criticality safety analysis made by Siemens AG for the increase of the initial enrichment to

4.6 wt.-% an ATRIUM™ 10 design was taken as a basis and it was assumed that the averaged initial enrichment of this type amounts to 5 wt.-%.

2. REGULATORY STATUS AND ACTIVITIES

2.1. Wet Storage of LWR Fuel

In Germany criticality safety design criteria are laid down in Regulatory Guides of the German nuclear technology committee KTA ("Kerntechnischer Ausschuß") and Safety Standards of the German society of standardization DIN ("Deutsches Institut für Normung).

Burnup credit for wet storage of LWR fuel at nuclear power plants has to comply with the newly developed safety standard DIN 25471 [1] passed in November 1999 and to be published shortly. This standard establishes the safety requirements for burnup credit criticality safety analysis of LWR fuel storage ponds and gives guidance on meeting these requirements. In particular, methods acceptable to validation of fuel depletion and criticality calculation codes are specified, parameters affecting the burnup credit are characterized (e.g., axial and horizontal burnup profiles), and methods acceptable to the verification of the fuel loading procedure are specified.

DIN standards are industry codes. Even though not laid down by the law the KTA regulations are commonly regarded as superior to the DIN standards. The basic criticality safety requirements for wet storage of LWR fuel are laid down in the standard KTA 3602 [2]. This standard does not prohibit burnup credit, but requires to give reasons for deviating from the fresh fuel assumption if burnup credit is employed. Therefore, a working group was set up which has the task to revise the standard KTA 3602 in such a way that this standard endorses the new safety code DIN 25471. This working group will probably have finished its work at the end of the year.

2.2. Dry Transport and Storage of LWR Fuel

Licensing evaluations of dry transport systems are based on the application of the IAEA Safety Standards Series No.ST-1 [3].

There are no national regulations that prohibit application of burnup credit to dry-cask transport and storage. However, because of the fact, that burnup credit for dry-cask transport becomes more and more inevitable due to increasing initial enrichment of the fuel, and because of the increasing importance of dry-cask storage in Germany, the necessity of giving regulatory guidance on applying burnup credit to dry-cask transport and storage is seen. It is planned, therefore, to work out criticality safety standards for burnup credit in dry-cask transport and storage on the analogy of the safety standard DIN 25471.

3. CURRENT AND INTENDED LEVELS OF BURNUP CREDIT

3.1. Storage Ponds at the KONVOI Plants

The existing spent fuel storage racks at the KONVOI plants GKN II, KKE, and KKI II are designed to accommodate fresh and spent fuel with a maximum enrichment of 4 wt.-% U-235.

3.1.1. Storage Ponds at NPP GKN II and NPP KKE

In order to minimize the costs of the reracking necessary due to the increase of the enrichment to 4.4 wt.-% only 320 storage cells (5 racks) are equipped with new absorber channels suitable for accommodating fresh 4.4 wt.-% U-235 enriched fuel. These 320 storage positions suffice to accommodate one full core (193 fuel assemblies) plus one reload batch plus all the fuel assemblies which haven't attained the end of their life time. The remaining 448 storage positions (7 racks) are left unchanged and are used as storage region II. Accordingly, in the criticality safety analysis of this region full burnup credit (actinide plus fission product burnup credit, cp. [4]) was applied. Due to the fact that the storage positions of this region are designed to accommodate fresh and spent fuel with a maximum enrichment of 4 wt.-% U-235 only a low burnup credit of 5 MWd/kg U is required. However, the criticality safety analysis includes already plans for a future increase of the storage capacity of this region from 448 positions to 732 positions. The burnup credit required then is given by the loading curve shown in Figure 1.

The cost savings due to application of burnup credit are about 4 million € per storage pond for the present reracking stage (exchange of 320 absorber channels). This amount includes the material savings due to the reduction of the number of absorber channels to be replaced as well as the cost savings due to the reduction of the waste to be managed (decontamination and disposal of the absorber channels replaced).

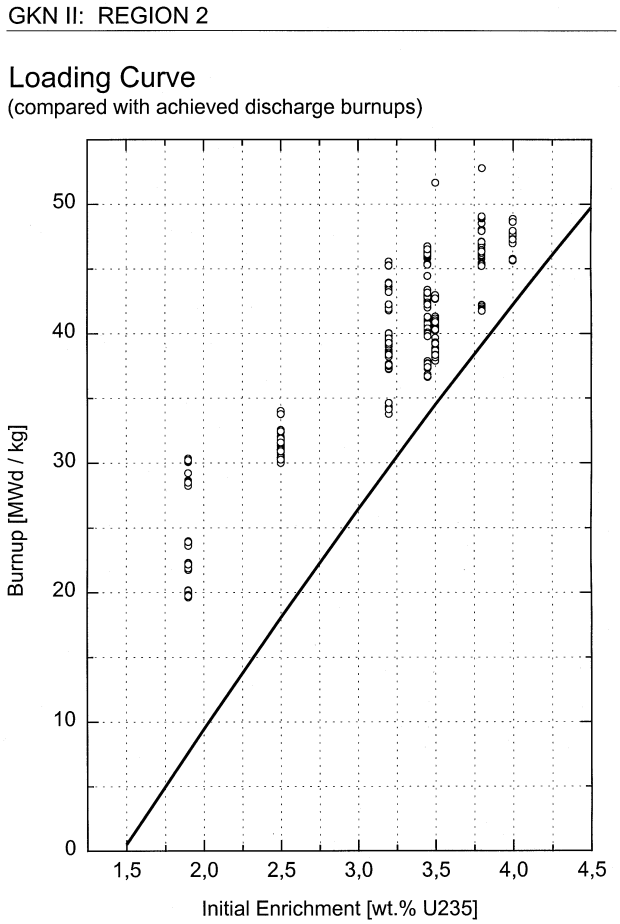

FIG. 1. Storage Pond at NPP GKN2: Loading Curve Referring to the Planned Future Increase of the Region II Storage Capacity.

3.1.2. Storage Pond at NPP KKI II

Due to the fact that each and every 4.6 wt.-% U-235 enriched fuel assembly is poisoned with Gd bearing fuel rods no reracking is required for the storage pond at NPP KKI II. In order to be able to demonstrate this it was necessary to apply the "integral burnable absorber burnup credit level" [4] as an "actinide plus fission product burnup credit level". The following isotopes were included:

1. Actinides: U-235, U-236, U-238, Np-237, Pu-239, Pu-240, Pu-241, Pu-242, and Am-243.
2. Fission Products: Mo-95, Tc-99, Rh-103, Cs-133, Cs-135, Nd-143, Nd-144, Nd-145, Nd-146, Nd-148, Nd-150, Pm-147, Sm-149, Sm-150, Sm-151, Sm-152, Sm-154, Eu-153, Gd-155, Gd-156, Gd-157, as well as the isotopes of the burnable absorber of course.

The methodology applied is similar to the methodology used for wet storage of BWR fuel (see below). Approval of this methodology was obtained in 1999.

3.2. Storage Pond at NPP GUN B/C

In the criticality safety analysis made under the assumption of 5 wt.-% U-235 enriched fuel the "integral burnable absorber burnup credit level" [4] was applied as an "actinide plus fission product burnup credit level". The following isotopes were included:

1. Actinides: U-234, U-235, U-236, U-238, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, and Am-241.
2. Fission Products: Rh-103, Cs-133, Cs-135, Nd-143, Nd-145, Sm-149, Sm-150, Sm-151, Sm-152, Eu-153, as well as the isotopes of the burnable absorber (Gd) of course.

The methodology applied is described in more detail in Reference [5]. Approval of this methodology was obtained in 1999.

3.3. Dry Transport and Storage of LWR Fuel

Limited burnup credits based on the "actinide only level" [4] have been approved for dry transport of LWR fuel. The licensing is based on either the French approach or a fuel assembly minimum average burnup as set forth below.

The standard casks used for shipping and dry storage of spent LWR fuel are the CASTOR casks developed by the Gesellschaft für Nuklear-Behälter (GNB), Essen, Germany.

The cask CASTOR V/52 is licensed to accommodate spent BWR fuel with average initial enrichments up to 4.6 wt.-% ^{235}U. The licensing evaluation of this cask is based on:

1. The fresh fuel approach for initial enrichments less or equal to 4.2 wt.-% ^{235}U,
2. The "uranium plus plutonium isotopes only" burnup credit for initial enrichments greater than 4.2 wt.-% ^{235}U.

If the initial enrichment is greater than 4.2 wt.-% ^{235}U it has to be ensured that the fuel to be loaded is irradiated (this is ensured by checking the cesium γ dose rate) and has a minimum

average burnup of 5 MWd/kg U (this is ensured through the analysis of each fuel assembly's exposure history).

GNB applies this "uranium plus plutonium isotopes only" burnup credit concept also to the CASTOR V/19 cask used for shipping and dry storage of spent PWR fuel. This cask is licensed for a maximum averaged enrichment of 4.45 wt.-% U-235 with:

1. The fresh fuel approach for initial enrichments less or equal to 4.05 wt.-% ^{235}U,
2. The "uranium plus plutonium isotopes only" burnup credit for initial enrichments greater than 4.05 wt.-% ^{235}U.

For an enrichment in excess of 4.05 wt.-% the required minimum averaged burnup of 10 MWd/kg U must be ensured on the basis of the irradiation history of the fuel assemblies from the reactor records. Additionally prior to cask loading γ dose rate measurement on each fuel assembly must be performed to ensure loading of irradiated fuel only.

4. CALCULATION CODES

The standards which have to be applied to the calculation codes used and the verification of these codes are laid down in the safety codes KTA 3101.2 [6], DIN 25471 [1], and DIN 25478 [7].

4.1. Depletion Codes and Verification Methods Applied

The depletion code systems mainly used in Germany in conjunction with burnup credit are:

1. The Siemens KWU standard design procedures SAV90 [8] and SAV95 [9] including the depletion codes FASER, MICBURN/CASMO-3G [10-11], and KORIGEN [12],
2. CASMO-4 [13-14] (as well as earlier versions of CASMO – those earlier versions make use of MICBURN to independently perform the integral burnable absorber burnup calculations),
3. The sequence controller SAS2H of the SCALE system [15] running the codes BONAMI-S, NITAWL-II, XSDRNPM-S, and ORIGEN-S.

The Siemens procedures SAV90 and SAV95 are usually used for burnup credit applications to PWR wet storage systems [16]. The depletion codes FASER and MICBURN/CASMO-3G within SAV90 and SAV95, respectively, are applied to determine the isotopic densities at the time of shutdown. To get cooling time dependent isotopic densities the depletion code KORIGEN is used with the cross-section sets generated by FASER or CASMO. The broad and comprehensive verification of the SAV90 and SAV95 procedures is based on observation and evaluation of normal power operation, special measurement programs (reactivity coefficients and equivalents describing the behavior of the reactor, short-term and long-term transients), and analysis of chemical assay data [9, 16]. Among the numerous chemical assay data against which SAV90 and SAV95 were verified are the data from the ARIANE programme [16-17]. It is intended to verify SAV95 also against the outcomes of the REBUS programme [18].

The code CASMO-4 or versions of CASMO prior to version 4 are usually employed for burnup credit applications to BWR wet storage systems [5]. CASMO is a widely used code

which is extensively verified against in-core measurement data and chemical assay data (as for instance from the ARIANE programme [17]) as well as critical experiments.

The SAS2H sequence of the SCALE system [15] is usually used for burnup credit applications to dry transport casks. This sequence was verified against numerous experimental data resulting from dissolution experiments and in-core measurement data [19-20]. It is intended to verify SAS2H also against the outcomes of the REBUS programme [18].

Some other depletion codes used in Germany (for instance for WWER fuel depletion calculations) are described in [21-22].

4.2. Criticality Calculation Codes and Verification Methods Applied

The criticality calculation codes mainly employed in Germany are:

1. The KENO module of the SCALE package [15] (usually used with the aid of the criticality safety sequences CSAS25, CSAS2X, CSAS26, and CSAS26X running the codes BONAMI-S, NITAWL-II, and in case of CSAS2X and CSAS26X also XSDRNPM-S for cross-section processing and then the Monte Carlo code KENO V.a or, in case of CSAS26 and CSAS26X, KENO VI),
2. The MCNP code [23],
3. The CASMO code already described above.

The KENO code as well as the MCNP code are verified against a large number of critical experiments and critical configurations:

1. Verifications of fresh fuel and net fissile content burnup credit calculations [4]: Evaluation of critical experiments covering a broad range of systems and fissile material types including homogeneous high- and low-enriched U-235 systems, heterogeneous low-enriched U-235 systems, U-233 systems, and Pu-systems: See [24-26],
2. Verifications of actinide-only burnup credit applications: Evaluation of critical experiments on mixed uranium-plutonium systems: See [24-28],
3. Verifications of integral burnable absorber burnup credit applications: Evaluation of critical experiments on integral burnable poisons: See [26-27],
4. Verification of actinide plus fission product burnup credit applications: Evaluation of reactor critical configurations: See [27, 29],
5. Verification of the temperature dependence of the neutron multiplication factor: See [30],
6. Verification of LWR spent fuel assembly storage pool and cask analysis: See [26, 28].

To evaluate the impact of different cross-section processing methods on the neutron multiplication factor of a spent fuel system comparisons between KENO - used with CSAS25 in conjunction with the ENDF/B-V derived 44-group library 44GROUPNDF5 of the SCALE system [15] - and the MCNP version MCNP4B2 [23] – used with continuous-energy neutron cross-section data available from several libraries [23] – were drawn using the burnup credit benchmark problems specified in [31]. As can be seen from Figures 2 through 4 [16] and in more detail from [31], the KENO and MCNP results are in good agreement.

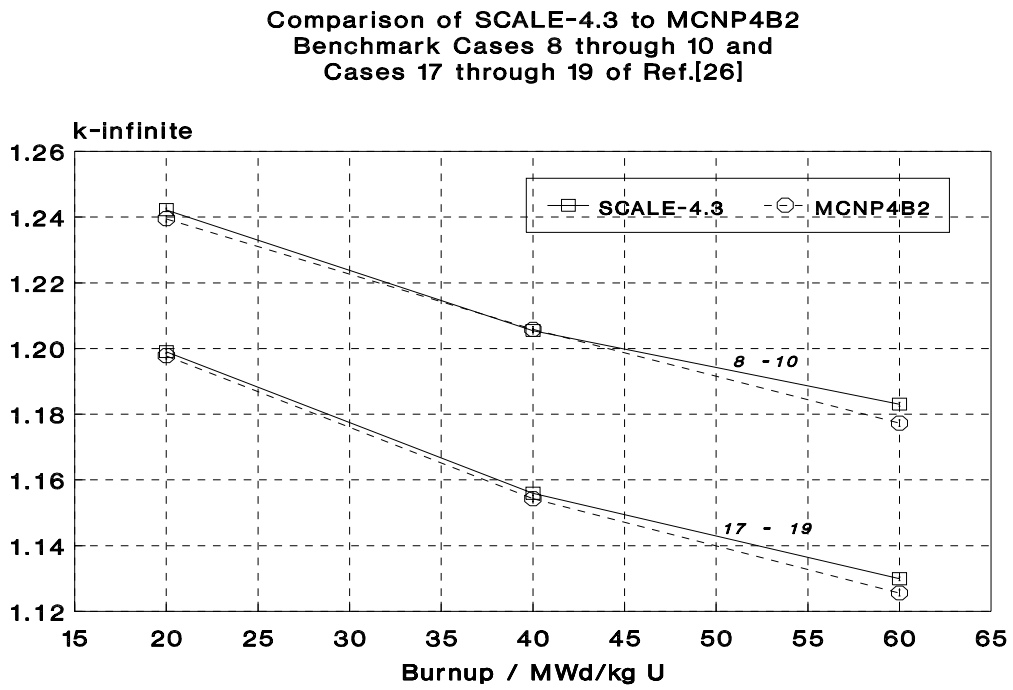

FIG. 2. Comparison of SCALE-4.3 to MCNP4B2 for Benchmark Problems Specified in [31].

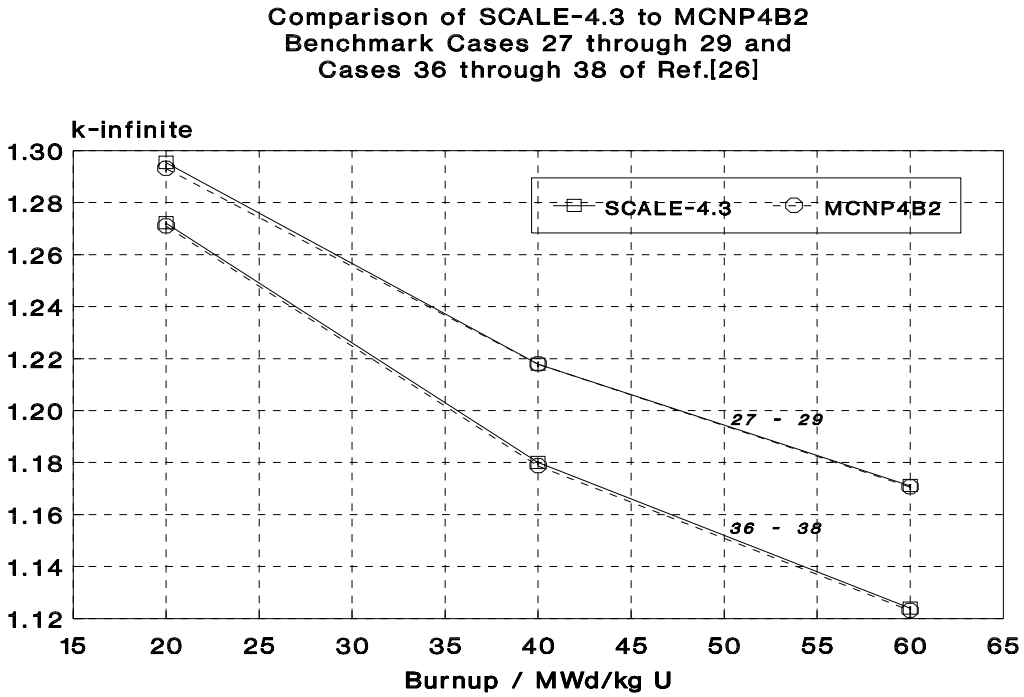

FIG. 3. Comparison of SCALE-4.3 to MCNP4B2 for Benchmark Problems Specified in [31].

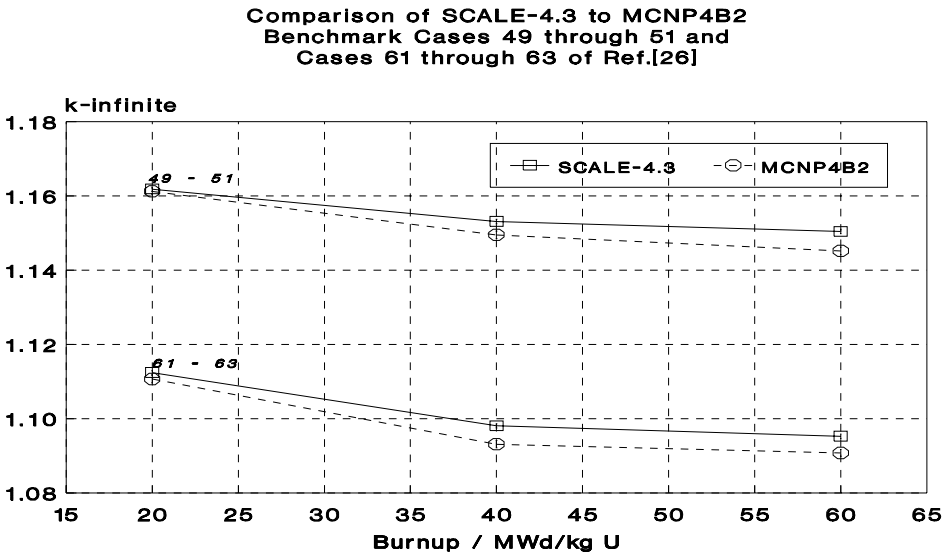

FIG. 4. Comparison of SCALE-4.3 to MCNP4B2 for Benchmark Problems Specified in [31].

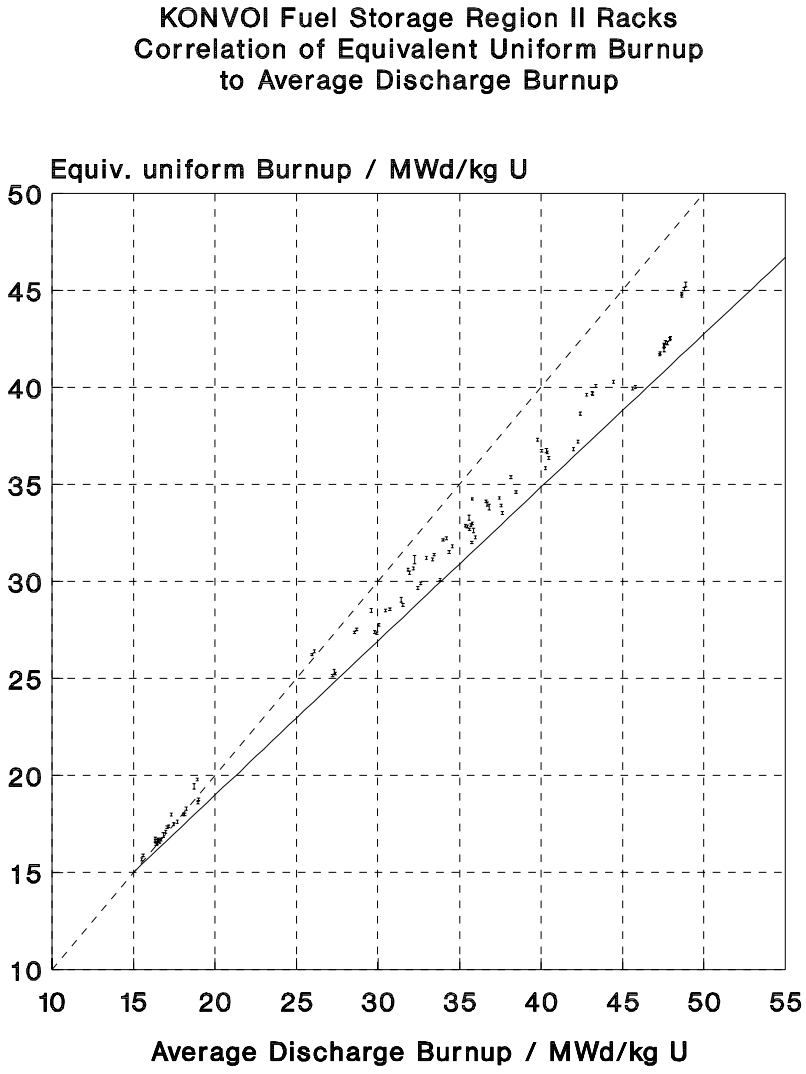

FIG. 5. Effect of Axial Burnup Shapes on Burnup Credit.

5. PARAMETERS AFFECTING BURNUP CREDIT

5.1. Reactivity Effect of Axial Burnup Profiles of PWR Fuel Assemblies

Siemens KWU has collected a big lot of axial burnup profiles based on in-core measurement data from different PWR NPP all over the world. All these data have been evaluated – for each plant separately – using the methods described in [32]. Figure 5 shows for example results obtained for NPP GKN II. This figure shows the "equivalent uniform burnup" as a function of the average burnup of the fuel assemblies. The equivalent uniform burnup is the uniform burnup (uniform burnup means constant burnup over the full active length of the fuel assemblies) which leads to the same neutron multiplication factor of the spent fuel management system of interest as obtained by considering the real axial burnup profile. If the equivalent uniform burnup is less than the average burnup of the profile then the difference Δk between the neutron multiplication factor obtained with the profile and the neutron multiplication factor obtained by assuming a uniform distribution of the average burnup of the profile is positive. If this difference, known as the "end effect", is positive it has to be covered by the loading curve Figure 1. To get an enveloping loading curve the equivalent uniform burnups of a big lot of axial profiles are estimated – each small bar in Figure 5 represents one analyzed profile (the fact that an analyzed profile is represented by a bar in Figure 5 is due to the fact that a Monte Carlo criticality calculation code was applied to determine the neutron multiplication factors). From the results obtained an enveloping correlation between equivalent uniform burnup and average burnup can be derived. This correlation represented in Figure 5 by the solid line (the dashed line corresponds to zero end effect) can be used:

1. First, to correct a loading curve based on the assumption of uniform burnups,
2. Secondly, to segregate non-acceptable axial profiles from acceptable ones: An axial profile is acceptable only then if the related equivalent uniform burnup is not beneath the correlation curve.

5.2. Reactivity Effect of Horizontal Burnup Profiles

Horizontal burnup profiles are covered by the linear model shown in Figure 6. This model gives the difference ΔB between the horizontally averaged burnup of one half of the fuel assembly and the horizontally averaged burnup B_{av} of the entire fuel assembly as a function of B_{av}. The averaged burnup B_{avH} of the higher burned half and the averaged burnup B_{avL} of the lower burned half of the fuel assembly are bounded by the equation $\Delta B = B_{avH} - B_{av} = B_{av} - B_{avL}$. The linear model shown in Figure 6 covers the horizontal profiles presented in [33] as well as horizontal profiles calculated with the SAV90 system described above [8-9].

Results obtained for the difference Δk between the neutron multiplication factor obtained with the model Figure 6 and the neutron multiplication factor obtained for the averaged burnup B_{av} are shown in Figure 7. These results refer to the KONVOI storage region II represented by the loading curve shown in Figure 1.

6. VERIFICATION OF FUEL LOADING PROCEDURES

6.1. Wet Storage of LWR Fuel

According to the safety code DIN 25471 [1] fuel assembly burnup determination based on reactor records without any additional measurements is acceptable. A fuel handling error has to be excluded by virtue of the double contingency principle (i.e., at least two unlikely, independent and concurrent incidents must occur before a misplacement of a fuel assembly that does not meet the region II loading criterion into a region II storage cell can occur).

To meet the double contingency principle NPP GKN II, e.g., is establishing the following procedure: To prevent fuel handling errors an interlock logic protected against malfunction is used for the loading machine hindering this machine from handling operations which are not laid down in a "handling sequence plan" established by an authorized person and checked by an empowered person according to the quality assurance requirements. The "handling sequence plan" is generated with the computer code ALFA [34] which uses appropriate interlock logic schemes to prevent fuel handling errors already at the planning stage. Fuel handling operations cannot be executed until the "handling sequence plan" is installed in the control unit of the fuel handling machine by an authorized person.

A similar procedure is intended to be used for the wet storage pond of NPP KKE.

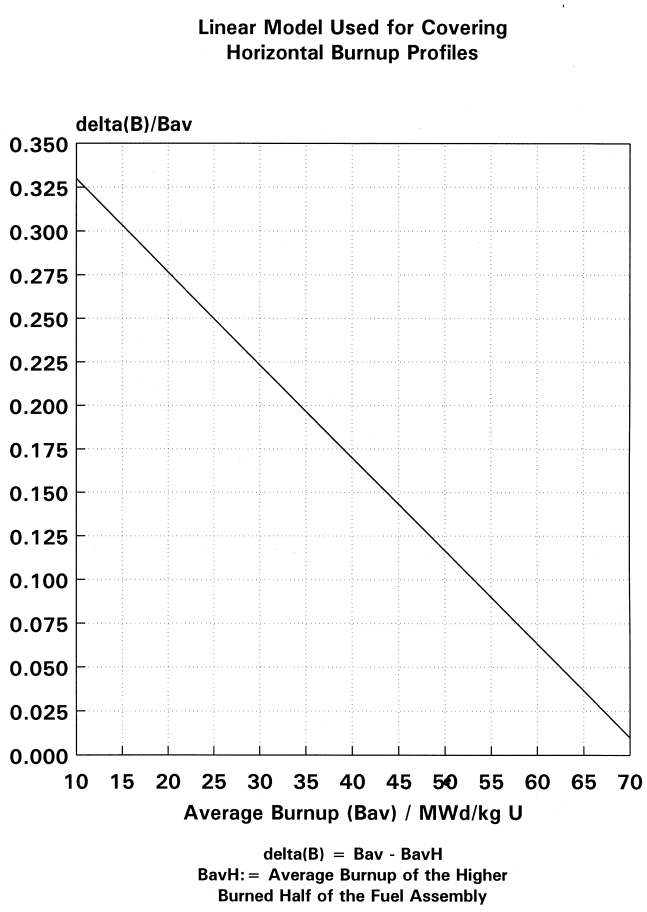

FIG. 6. Linear Model Used for Horizontal Burnup Profiles (See Equation (16) in Reference [32]).

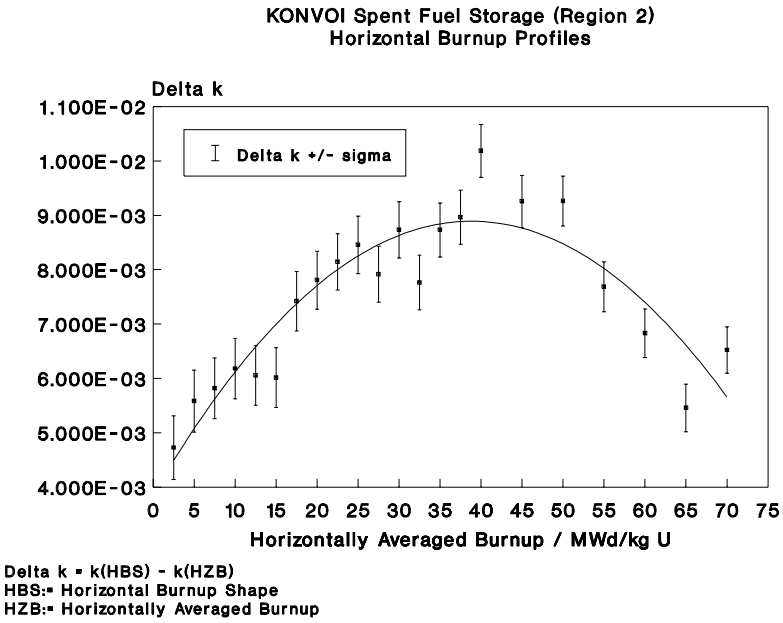

FIG. 7. KONVOI Storage Region II: Effect of Horizontal Burnup Profiles (cp. [32]).

6.2. Dry Transport of LWR Fuel

Verification of the loading procedure by measurement is required as set forth below:

1. If the licensing is based on the fuel assembly minimum average burnup approach described in section 3.3 only a qualitative burnup verification measurement is necessary, as already described in section 3.3. The required minimum level of burnup is verified from the reactor records,
2. If the licensing is based on the French approach a quantitative burnup verification measurement is required. The NPPs Grohnde and Brokdorf, e.g., use the French PYTHON device [35] for this purpose.

7. SUMMARY

Application of the “actinide plus fission product burnup credit level” to wet LWR storage ponds at plant site is now introduced in Germany. A criticality safety standard is established in this matter, the methodologies used are well established and approvals of these methodologies are obtained.

Further development of the burnup credit methodologies applied to dry-cask transport is required due to increasing initial enrichment of the fuel. Dry-cask storage has to be included due to its increasing importance in Germany. One of the most important steps on the road to full burnup credit will be the evaluation of the outcomes of the REBUS programme [18].

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BURNUP CREDIT IN SPAIN

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Abstract

The status of development of burnup credit for criticality safety analyses in Spain is described in this paper. Ongoing activities in the country in this field, both national and international, are resumed. Burnup credit is currently being applied to wet storage of PWR fuel, and credit to integral burnable absorbers is given for BWR fuel storage. It is envisaged to apply burnup credit techniques to the new generation of transport casks now in the design phase. The analysis methodologies submitted for the analyses of PWR and BWR fuel wet storage are outlined. Analytical activities in the country are described, as well as international collaborations in this field. Perspectives for future research and development of new applications are finally resumed.

1. INTRODUCTION

Burnup credit has been taken in Spain for the criticality safety evaluation of spent fuel pools since 1990. The need for increasing the storage capacity of the on-site spent fuel pools by means of high-density storage racks led to the implementation of burnup credit, originally at PWR plants.

At present, credit to burnup is included in the criticality safety analysis of all the PWR plants in operation, 7 units, and credit to the presence and burnout of burnable absorbers (Gd) is given in the two BWR units in operation. The analyses refer to wet on-site storage in all cases, and cover a wide variety of fuel types for each plant. Full burnup credit is taken in all the licensed analysis, meaning that the neutron absorption due to fission product buildup is taken into account.

There is an ongoing dual-purpose cask design program in Spain, of which the first cask for PWR fuel has already been licensed assuming fresh fuel will be loaded. The possibility of taking credit for fuel burnup in the design of the new dry casks generation is currently under consideration, and will most probably be adopted. This decision will likely condition whether burnup credit will be taken for the design of the Interim Storage Facility that should be built in Spain to cover the time gap until the final repository is available. No decision has been yet taken about the latter.

2. REGULATORY STATUS

The use of burnup credit and partial credit for Boron in the criticality safety analysis of spent fuel pools is allowed by specific national regulations, published in 1990. These regulations have never since been revised, if they are scrutinised at the light of the current burnup credit knowledge and of the improvement of the codes and analytical methods, it has to be acknowledged that the technical details and other parts of their contents are now outdated. A revision of this rule is expected to be published in 2001.

The acceptance criteria now applied for recent analysis follow the technical positions of the NRC, that demand that the K-eff of the storage has to be below 1 with unborated water, including all the uncertainties, and below 0.95 assuming borated water, again including uncertainties.

No reference to the use of burnup credit for the design of spent fuel storage/transport casks, of the dry storage of spent fuel, or of the final repository, can be found in the national regulation. Given the fact that a new generation of burnup credit casks will be submitted for licensing in a few years, there is a clear need to define technical positions and acceptance criteria on schedule. It is expected that this work will be included in the revision of the regulation mentioned above.

3. BURNUP CREDIT ANALYSIS OUTLINE AND IMPLEMENTATION

Criticality safety analyses licensed up to date for the storage of PWR fuel follow the usual procedure of determining a reactivity equivalence curve. The curve relates the initial (fresh) enrichment of a given fuel bundle with the minimum discharge burnup needed to guarantee that the reactivity of that bundle is within the allowable limits for the system design. In those cases where different fuel types are (or have been) used in the plant, the most reactive fuel type under storage conditions is first determined, and the reactivity equivalence curve finally applied refers to the most reactive fuel type.

The calculation is performed in two steps. First, a fuel depletion calculation is performed to determine the evolution of the nuclide's concentration as a function of burnup, using a bounding irradiation history in the sense of maximising the spent fuel reactivity. Second, a criticality calculation is performed using the specific characteristics and tolerances of the spent fuel storage system.

Fuel depletion calculations necessary for burnup credit criticality safety analysis introduce new parameters and effects that should be taken into account to obtain an adequate (in the sense of bounding) reactivity result. The first effect comes from the spatial distribution of burnup in the fuel assembly. Depletion calculations are performed with 2-D lattice codes. The isotopic composition obtained is then used to calculate the reactivity of the spent fuel assuming that the fuel composition is uniform in both the radial and axial directions.

However, although a fuel assembly is characterised by an average burnup value, there is an axial burnup distribution in the fuel assemblies, the burnup level being higher in the centre and lower at the ends of the assembly. Whether this axial burnup distribution has an impact on reactivity or not, and how big the impact is, depends on the specifics of the fuel assembly type and on the details of the plant's fuel management scheme. So, it has to be determined on an application specific basis. As a consequence, burnup credit applications for storage have all taken into account the specific burnup distributions of the fuel operated at the specific plant.

For PWR fuel storage applications, the axial burnup shape effect on reactivity has been determined using burnup profiles coming from core follow data of the specific plant. A reactivity bias due to this effect is obtained for each burnup value on a 95/95 basis, and the reactivity of the uniformly burnt fuel assembly is corrected at each burnup value. The reactivity equivalence curve finally applied already includes this bias.

A direct measurement of the discharge burnup is not required for storage implementation of burnup credit. As a result, no burnup measurement method is available at the Spanish plants, and the discharge burnup is determined based on core following data coming from the reactor records. A problem associated to this prcatice is that core following methods are very different in quality among the plants, and still rather crud in some cases in which the older methods and procedures have not been replaced by modern on-line monitoring systems.

In the case of BWR fuel, the level of credit to fuel burnup given is in fact the consideration of the neutron absorption in the Gadolinia. Hence, it is not precisely burnup credit, although the calculations processes and needs are very similar. The approach followed is to first determine the most reactive lattice of all the specific designs of every fuel type used at the plant. This process can be complicated, given the wide variety of fuel types, and the extreme design flexibility of BWR fuel.

All the fuel is then assumed to be axially uniform with that lattice design. The burnup value at which the lattice reactivity is at its maximum (due to the decrease of the Gadolinia contents with burnup) is calculated, and the maximum lattice average enrichment that fulfils the reactivity limit for the storage is obtained. Hence, the analysis is inherently conservative because the maximum reactivity of the most reactive lattice is assumed to be the condition of all the fuel present in the pool.

The radial enrichment distribution of BWR fuel is not uniform, in order to adequately shape the radial power distribution and reduce the local peaking factor. This distribution has an effect on the lattice reactivity that has to be taken into account, because the enrichment distribution can be modified from cycle to cycle without changing other fuel bundle characteristics. It has been observed that the assumption of a uniform radial enrichment distribution is more reactive than any radial distribution with the same average value. Hence, a uniform enrichment distribution is used for the analysis.

Consideration of the axial burnup shape in BWR fuel is a rather difficult task. A process equivalent to that described for PWR fuel cannot easily be formalised for BWR fuel. The axial burnup shape of BWR fuel assemblies depends on too many factors, including the specific axial zoning design and the important axial variation of the depletion parameters (void fraction, temperature, flow…). Hence, fuel with the same burnup level can have very different axial burnup shapes, with a remarkable effect on reactivity. That is the basic reason why a bounding approach such as the one described above (limiting lattice at worst time in life) has to be used. A less restrictive approach has never been submitted for licensing.

4. BURNUP CREDIT RELATED ACTIVITIES IN SPAIN

There is no experimental program in Spain related to burnup credit, neither ongoing nor projected for the near future. All the activities performed up to now have been analytic in nature, and the experimental data needs have been covered in different ways.

As a result of an ongoing rethinking of the CSN research priorities, burnup credit related research has been identified as one of the fields were more regulatory work is expected in the coming years, specially that related to spent fuel transport applications. Therefore, it is expected that this issue will be rated as high priority, meaning that the participation of Spain in international experimental programs and activities could be better funded in the future.

Three Spanish organisations (two engineering companies and one University) have developed burnup credit analysis methodologies for spent fuel storage systems. All of them have already been licensed, and the regulatory authority has accepted their applications at least once. The oldest methodology of them all, applied to PWR fuel storage for the first time in 1989, is being now revisited and updated to include the developments attained in the field during this period, before additional applications are submitted. These new applications are anticipated

due to the need of the plants to increase the fuel enrichment to face the power uprate process in which most of them are involved.

During licensing evaluations performed in the past years, it was observed that the specific characteristics of advanced fuel designs (axial zoning, enrichment distribution, integral burnable absorbers) could not be modelled in the SCALE system with the same level of detail used for traditional fuel. This lack of detail could have an impact on the quality of the depletion calculation, affecting both the spent fuel calculated reactivity and the source term obtained. A development work co-funded by ENRESA (state owned company responsible for radioactive waste management) and the CSN is going on at Oak Ridge National Laboratory to put in place a new calculation sequence (SAS-2D) in the SCALE system.

Spain also maintains an active participation in the Working Party on Nuclear Criticality Safety (formerly Burnup Credit Criticality Benchmark Group) of the OECD-NEA, and has been collaborating with IAEA activities in this field since they were started. Also, some collaboration with the US-NRC has recently been started, as well as bilateral cooperation with some South-American countries.

IRRADIATED FUEL STORAGE AND TRANSPORT: A SWEDISH PERSPECTIVE

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Abstract

This paper gives the views of the author and may not correspond to the views of the Swedish industry or the licensing authority. The views are based on experience from consultation to the Swedish licensing authority and from participation in international cooperation, in particular in the OECD/NEA NSC Working Group on Burnup Credit (ref. 1).

1. LESSONS TO BE LEARNED FROM THE TOKAIMURA CRITICALITY ACCIDENT ON SEPTEMBER 30, 1999

If not before, the Tokaimura criticality accident in Japan on September 30, 1999 should make it clear to decision makers and criticality safety specialists in the world that the formulation of defence-in-depth, risk-informed, double-contingency principles etc. are not sufficient for adequate safety. Adequate understanding of all individual issues of importance as well as of the total "picture" is required.

Contrary to conclusions in many assessments of the Tokaimura accident, it was not primarily caused by incredible human errors by the operators or by criminal negligence of licensing conditions by the company. Such errors and negligence should have been expected, based on experience from previous industrial operations.

The accident was caused by incorrect safety assessments, in particular the risk assessments. Not only the plant operator, but also the licensing authority made those incorrect assessments. Criticality was not considered a realistic event. After the emergency initiation in 1984 (licensing), a long sequence of mistakes was made by the operator and by the authorities. It seems as if the original incorrect assessments in 1984 were not questioned later. "Criticality can't happen, don't waste time, money, training or other resources on worrying about it" seems to have been the conclusion by the operator and licensing authority from the first incorrect assessments.

The emergency condition at the JCO plant started about 1984 when the license was issued. Unfortunately, there was no alarm system for this type of emergency. This is a major lesson to be learned. Criticality safety specialists in the industry and at licensing authorities should never assume that existing safety assessments are correct without checking at frequent intervals. The original assessments (methods or application of methods) could be incorrect. Another accident source is that the real world has changed since the original assessments were made.

The incorrect safety assessments for the JCO plant must have been the result of lack of training. Not of the operators but of criticality safety specialists. Incredible mistakes are frequently made by operators, by plant management, by safety departments, by licensing authorities and by criticality safety specialists. A trained criticality safety specialist knows this.

One of the most promising consequences of the Tokaimura criticality accident is that communication between plant operators on significant events will be improved (NS-Net in Japan, NEI in the United States). Also authorities and international organisations (OECD/NEA, IAEA) will participate in collecting and presenting such information. This new communication is a recognition that the lack of training of safety specialists was a significant source for the Tokaimura accident.

Only time will tell if the new interest in better communication on events will help to train criticality safety specialists. A later section contains some examples of events and of special designs that could be useful to train criticality safety specialist in burnup credit assessments and to validate the methods.

2. CAUTION IS MOTIVATED IN INTRODUCING NEW METHODS IN NUCLEAR CRITICALITY SAFETY

Burnup credit involves new methods for criticality safety control. Often in the past, criticality safety specialists have found that incorrect methods have been introduced. Once they have been introduced, accepted and approved by the authorities, they have often been difficult to correct. A license application that has been approved once is often used as a template for the next application. Some of the general limitations may not be obvious during the first application. All considerations may relate to the current plans only. If the general limitations of the method are not clearly defined in the first application, it is possible that they may be neglected later.

Burnup credit requires good understanding of the possible variations in the fuel assembly isotopic compositions and distributions. There are efforts made by different organisations to compile such data. Reactor parameters that influence the composition are void, temperature, pressure, burnable absorbers, position in the reactor, etc. The isotopic distribution varies within a fuel assembly and between similar positions in different fuel assemblies. Axial and radial (or horizontal) variations need to be considered. An OECD/NEA NSC Working Group on Burnup Credit has made various calculation studies during 10 years. Some participants have drawn general conclusions on methods for burnup credit, but the truth is that real fuel has not yet been introduced (horizontal burnup variations). Many interesting results have been obtained, but the subject has turned out to be more complicated than originally expected.

Burnup credit also requires good understanding of potential scenarios in which the fuel assemblies may be involved. This is an area where past experience could be very important for safety. It is not enough to have a method that predicts the neutron multiplication factor correctly in a specific scenario. To be able to evaluate parameter variations, approximations and incidents correctly, the real fission distribution, energy spectrum, etc. must be modelled adequately by the method.

To identify all scenarios that could lead to criticality, it is necessary to understand both the fuel assembly variations and the variations in the volumes surrounding the fuel assemblies.

Validation of burnup credit methods involves not only cross section validation but also computer code convergence and other tests. Specifications of the fuel assembly history in the reactor and the resulting fuel characteristics need to be verified.

Early proposals for burnup credit methods were based on a flat burnup profile model, using the average burnup for the assembly, in all sections of the fuel assemblies. This is probably adequate for specific scenarios but not for a general method. The limitations of the flat profile methods have been noticed during recent years, but there may still be unexpected complications. The potentially strong horizontal tilts of the burnup in a single assembly were discovered quite late. New PWR fuel designs with axial burnable absorber zones create other complications.

3. CREDIT FOR FUEL BURNUP AND/OR FOR GADOLINIUM PRESENCE

Fig. 1. shows how gadolinium credit and fuel (uranium and gadolinium) burnup credit are two different concepts that may be combined. This figure is slightly modified from a previous presentation.

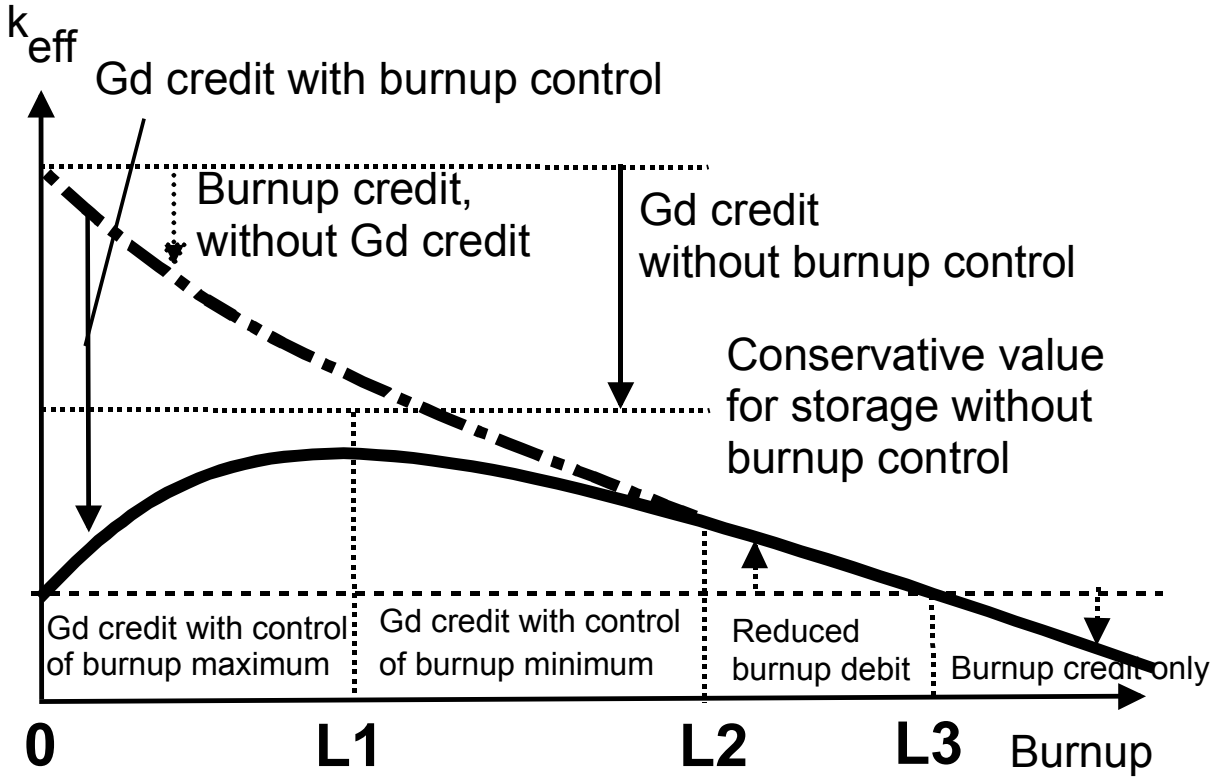

Figure 1. Gadolinium and burnup credit and control.

4. VERIFICATION OF IRRADIATED FUEL CHARACTERISTICS

It is important to recognize the potential isotopic and geometric distribution variations in irradiated fuel. The verification methods need to be designed with such potential variations in mind.

5. STORAGE OF IRRADIATED LWR FUEL

Storage of LWR (both PWR and BWR) spent fuel in Sweden is of the water pool type. This applies both to the nuclear power plants and to the independent spent fuel storage facility CLAB. Burnup credit has been seriously considered more than 10 years ago (ref. 2) during the

planning and design of compact storage canisters for CLAB. A decision was made not to try to get burnup credit licensed. Instead the canisters were made of boron steel to reduce k_{eff}, the neutron multiplication factor.

Gadolinium credit is applied at the nuclear power plants and at CLAB. This means that some consideration of possible changes in the fuel (including depletion of uranium and gadolinium and buildup of plutonium) specifications as a consequence of irradiation is required. However, no credit is taken for burnup of the fuel. Depletion of gadolinium without simultaneous depletion of uranium is not considered credible. There is no need for control of burnup. However, presence of gadolinium rods must be controlled administratively. In the past, rods have been removed from fuel assemblies at the nuclear power plants. Gadolinium credit is also applied to fresh fuel fabrication and transport.

6. TRANSPORT OF IRRADIATED LWR FUEL

Transport of irradiated fuel is normally made with the fuel in dry condition but wet transports have also been made. The most commonly used transport package for spent fuel is French, with the original safety reports and authority certificates written in the French language. Recently, some of the allowable contents of the French certificate have been based on burnup credit. Those contents have not been approved in Sweden and there is no request to get them approved. The current transport certificate does not take credit for gadolinium in irradiated fuel.

The French transport package design used for irradiated fuel transport in Sweden has another feature of interest to burnup credit evaluation. There is a potential for some of the fuel at the ends to reach regions where there is no boron neutron absorber. The safety reports specify that the fuel section outside the boron region must not be more than about 15 cm. This type of evaluation is more complicated if burnup credit is applied. The scenario is similar to the accident condition that was discussed during Phase IIB of the OECD/NEA NSC Working Group study on burnup credit.

A peculiar situation has existed during the last year. There has been some discussion in France on the integrity of the fuel during a postulated transport accident. The package design has been approved in France on the condition that it is always dry inside, even in the case of a postulated accident. First of all, this means that there must not be any potential for leaving water in the package before transport. Second, if there is a transport incident that could result in fuel damage, the package must not be unloaded in clean water. In Sweden, the normal procedure is to unload the irradiated fuel in clean water. In France, the normal procedure is to unload the fuel dry or with borated water. The safety situation is more complicated in Sweden.

7. EVALUATION OF SERIOUS INCIDENTS AND ACCIDENTS

If there is a serious incident that is believed to be a criticality threat, a burnup credit method could be very useful to assess the realistic threat and to avoid unnecessary confusion. Without established burnup credit methods, such evaluations are not very reliable. If a criticality accident has already occurred, a burnup credit method can be used to evaluate how long the criticality excursion may continue and also how to stop it. It may be important to know if the fission centre is near the bottom of the fuel assemblies, near the middle or near the top.

Burnup credit is not currently used to design storage pools or transport packages in Sweden. However, the knowledge base resulting from national and international studies on burnup credit is valuable when the potential scenarios and consequences of a criticality accident involving irradiated fuel are evaluated. Emergency planning is usually based on "beyond design basis" accidents.

8. EVALUATION OF REAL SAFETY MARGINS

Both the authorities and the industry in Sweden have shown interest in determining the real safety margins and the real risks for irradiated nuclear fuel storage and transport. There are probably several different reasons for such interest. Information to management, safety departments and to the public is one reason. Another reason is that realistic safety margins and risks can be used to assign priorities to safety controls carried out by the authorities and safety departments. Often the evaluation of real safety margins and risks are carried out informally using methods that are not validated. Burnup credit is necessary to evaluate real safety margins and risks in the storage and transport of irradiated nuclear fuel. The lessons learned from burnup credit studies during the last 10 years show that the previous informal evaluations of real safety margins and risks were not adequate in a non-conservative direction. An incorrect risk assessment can lead to loss of respect for formally established safety controls.

9. COLLECTION OF SCENARIOS AND EVENTS OF IMPORTANCE TO VALIDATION OF BURNUP CREDIT METHODS

To evaluate the general suitability of a proposed burnup credit method, it is essential to have a number of typical scenarios available. This will help in determining the limitations of the method.

The typical scenarios could be based on past experiences, on postulated events or on design variations. Some examples follow:

1. BoraFlex Panel (BFP) shrinking in storage ponds. Gaps up to 15 cm have been noticed in the neutron poison sheets. A common problem at many nuclear power plants in the United States,
2. Axial regions without boron in transport packages. Administrative errors or transport accidents could lead to this situation in the packages used for transport of irradiated fuel between Swedish nuclear power plants and the central storage facility CLAB. The package design is also used in other countries,
3. Monte Carlo source convergence problems. A local variation in a very large storage pond (CLAB) was not seen by the computer code due to insufficient sampling. Overlapping of the top part of a stored canister with the bottom part of a canister in transfer involved about 60 cm length of active fuel. The neutron multiplication factor for the pool was determined by only four fuel assemblies. With burnup credit the increase in the multiplication factor would have been much larger due to axial and horizontal burnup distributions,
4. Insertion of a single fresh fuel assembly in an array with irradiated fuel assemblies. This is a common requirement when there is a potential for administrative errors. A modern PWR fuel design with no Gd in the end zones (up to 40 cm long) could be the worst case,

5. Removal of all fuel rods with Gd from a large number of fuel assemblies (a complete reload) was made at a Swedish reactor at Barsebäck about 20 years ago. The Gd rods were suspected to have some deficiency. This was at a time when Gd credit was considered but not yet accepted. During discussions on licensing of the spent fuel pool about a year earlier, removal of a large number of Gd rods had been considered very unlikely.

10. CRITICAL EXPERIMENTS IN SWEDEN

Sweden has benefited from the publication of experimental specifications from many other countries. As far as I am aware, Sweden has not contributed with such specifications. Many critical experiments have been carried out in Studsvik, Sweden. Some of them are quite unique and involve specifications that are of interest in validation of irradiated fuel storage. The experiments were made with fresh fuel, but the temperature and void reactivity coefficient measurements at the KRITZ facility could be useful to validate both static and transient methods. It is now 25 years since the facility was closed and it is a good time to release the commercial protection of the specifications. Hopefully, the specifications are clear enough to allow them to be used as benchmark problems for nuclear criticality safety even though that was not the major purpose of the experiments.

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STATUS OF BURNUP CREDIT IMPLEMENTATION AND RESEARCH IN SWITZERLAND

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Abstract

Burnup credit has recently been approved by the Swiss licensing authority for the spent-fuel storage pool of a PWR plant for fuel exceeding the originally licensed initial enrichment. The criticality safety assessment is based on a configuration consisting of a small number (approximately a reload batch) of fresh assemblies surrounded by assemblies having a burnup corresponding to the minimum value in the top 1 m section after one cycle of irradiation. The allowable initial enrichment in this configuration is about 0.5% higher than for all fresh fuel. A central storage facility for all types of radioactive wastes from Switzerland, including cask storage of spent fuel assemblies is being commissioned presently. The first applications for licenses for casks to be used in this facility have been submitted. Credit for burnup has not been requested in these applications (conforming to the original licenses of the casks in their countries of origin), but utilities are interested in burnup credit for fuel with higher initial enrichments. Reactivity worth measurements as well as chemical assays of spent fuel samples in the LWR-PROTEUS facility at PSI are in detailed planning currently. The experiments, scheduled to start in 2001, will be performed in cooperation with the Swiss utilities and their fuel vendors. Although the focus of interest of these partners is on validation of in-core fuel management tools, the same experiments are also applicable to burnup credit, and contacts with further potential partners interested in this field are underway.

1. GENERAL SITUATION OF SPENT FUEL MANAGEMENT IN SWITZERLAND

Spent fuel from all the Swiss nuclear power plants (3 PWR's, 2 BWR's) has been stored in pools at the reactors. The PWR's have been refurbished with high-density racks employing absorber sleeves with boron-containing materials. A substantial part of the spent fuel, though varying from plant to plant, has been shipped to France and the United Kingdom for reprocessing. A central storage facility for all types of radioactive wastes, called “Zwilag” (from the German word Zwischenlager, i.e. intermediate storage), which also provides room for dry storage of spent fuel in casks, has recently been commissioned. The first spent fuel shipments to this facility are planned for the near future. Utilities are considering an increase of long-term storage of spent fuel, particularly if this alternative proves economically more attractive than reprocessing (after fulfilling the existing contracts). Planning and geological investigations for the disposal of radioactive wastes in Switzerland are performed by NAGRA, a co-operative formed by the nuclear utilities and the federal government (responsible for wastes from medicine, industry and research). A repository for high-level wastes is not expected to become operational before the year 2040.

All transports of spent fuel from Swiss reactors to date have been to foreign reprocessing plants. Moreover, no transport containers have been originally designed and licensed in Switzerland. For these reasons, licensing of transports is not done completely independently in Switzerland. Rather, licensing of spent fuel shipments from the Swiss nuclear power plants is based on the licences for the containers obtained in their countries of origin, and the regulations for transport and reprocessing in the receiving countries must be taken into account.

2. REGULATORY STATUS

The standards and guidelines applied for licensing in Switzerland allow the use of burnup credit. For each fuel assembly to be loaded into a spent fuel management system applying burnup credit, it must be proved, both from the reactor operating records and by a burnup measurement, that it exceeds the minimum burnup on which the burnup credit licence is based.

The licensing of storage pools is based on the US NRC Regulatory Guide 1.13, Appendix A. The containers used for away-from-reactor storage and for spent fuel transports to foreign reprocessing plants are licensed in their countries of origin. The original licences are validated for the use in Switzerland based on an independent review of the original safety assessment by the Swiss authorities, in which amongst other considerations the conformity with international standards (IAEA transport regulations) is verified.

3. CURRENT AND NEW USES OF BURNUP CREDIT, FUTURE INTERESTS

3.1. Pool Storage

Burnup credit has recently been approved by the Swiss licensing authority for the spent-fuel storage pool of a PWR plant for fuel exceeding the originally licensed initial enrichment. The criticality safety assessment is based on the fact that even in the case the core must be unloaded at the beginning of a cycle, only a small number of assemblies in the pool are fresh. Criticality calculations were performed for a small contiguous array (approximately a reload batch) of fresh assemblies surrounded by elements burnt for at least one cycle. The minimum burnup value in the top 1 m section after one cycle is assumed for these latter assemblies. Credit is taken both for actinides and fission products (except for short-lived nuclides), whereas no credit is taken for soluble boron. The allowable initial enrichment of UO_2 fuel, which was 4.40% for all fresh fuel, can be increased to 4.89% in this mixed configuration. Burnup credit is not taken for MOX fuel, because the maximum envisaged initial Pu content is less than the allowable value for fresh fuel.

No credit for burnup is taken for the storage of spent fuel in the pools of the other reactors. The existing storage pools have sufficient margin to accommodate fuel with higher initial enrichment than originally used. Credit is taken, however, for integral burnable absorbers in BWR pools, i.e., the storage pools are designed and licensed for the peak reactivity of the fuel in its lifetime under consideration of burnable absorbers.

3.2. Cask Storage

The first applications for licenses for casks to be used for spent fuel storage in the Zwilag facility have been submitted. Credit for burnup has not been requested in these applications, because they are based on the original licences of the casks in their countries of origin, and because the first assemblies to be stored are old ones with relatively low initial enrichments. However, burnup credit may become interesting in the future, when fuel with higher initial enrichments will be shipped to Zwilag. This will be the case in a few years. Negotiations with cask vendors are underway, but burnup credit has not yet been requested concretely.

3.3. Transportation

The shipments of spent fuel from the Swiss reactors to foreign reprocessing plants have to fulfil not only the Swiss licensing criteria, but also those of the receiving countries and those of the original licences of the transport casks. To date, burnup credit has only been taken by one PWR utility for shipments to France. In this case, the same criteria as in the French regulations are applied for fuel whose initial enrichment exceeds the originally licensed limit of the casks and of the reprocessing plant.

3.4. Disposal

For disposal, burnup credit is regarded as desirable for an economic loading of the fuel assembly canisters (for the criticality safety in the case of water ingress, but intact fuel geometry). Scoping studies on possible scenarios leading to criticality after dissolution and relocation of fissile nuclides (taking into account also the moderation by the clay backfill around the canisters) as well as on the probability of occurrence of such configurations have been started by NAGRA. Burnup credit is intended also for this part of the safety assessment, but given the long time frame until the possible realisation of a repository in Switzerland, a concrete licence application is still far away.

4. RESEARCH: THE LWR-PROTEUS PROGRAMME

A programme of LWR integral experiments is currently being carried out in the PROTEUS facility at PSI. PROTEUS is a driven, zero-power facility, in which the central test zone, which contains the lattice to be investigated, is subcritical. This test zone is surrounded by driver regions containing 5% enriched fuel moderated by heavy water and graphite. The test zone and the driver are separated by a buffer consisting of tightly-packed natural uranium metal rods in air (i.e. without moderator) which helps to spectrally decouple the two regions.

The LWR-PROTEUS experiments are performed in co-operation with the Swiss utilities and their fuel vendors with 50% utility funding. In the first phase, which is about 2/3 complete, the test zone consists of 9 real, full-length SVEA-96 BWR fuel assemblies. The major part of the measurements in this phase deals with pin power distributions and pin removal reactivity worths in these assemblies. The aim of these measurements is to validate design codes for the calculation of modern assembly types which have much stronger heterogeneities (e.g. internal water regions, high number and absorber content of burnable poison rods) than most of the experiments against which the computational methods were originally tested.

In a second phase of the programme, which is in detailed planning currently and scheduled to start in 2001, reactivity worth measurements of PWR spent fuel samples will be performed. For these experiments, the central BWR assembly will be replaced by a small array of actual PWR fuel rods. Samples of burnt rods (~40 cm long) and specially prepared samples containing individual actinides or fission products or standard absorbers (for calibration) in a UO_2 matrix will be oscillated in the centre of the test zone. The burnt samples will be held in a special cask placed on top of the reactor incorporating a remotely controlled oscillator. UO_2 rods from a Swiss PWR with burnups up to 82 GWd/t are available at PSI. MOX samples (up to ~50 GWd/t) will also be measured. In addition to the reactivity experiments, chemical assays of these samples will be performed in the PSI Hotlab. Although the focus of interest of the utility and industry partners is on validation of in-core fuel management tools for high burnups, the same experiments are also applicable to burnup credit, and contacts with further potential partners interested in this field are underway.

BURNUP CREDIT DEMANDS FOR SPENT FUEL MANAGEMENT IN UKRAINE

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Abstract

In fact, till now burnup credit has not be applied in Ukrainian nuclear power for spent fuel management systems (storage and transport). However, application of advanced fuel at VVER reactors, arising spent fuel amounts represent burnup credit as important resource to decrease spent fuel management costs. The paper describes spent fuel management status in Ukraine from viewpoint of subcriticality assurance under spent fuel storage and transport. It is considered: 1. Regulation basis concerning subcriticality assurance, 2. Basic spent fuel and transport casks characteristics, 3. Possibilities and demands for burnup credit application at spent fuel management systems in Ukraine.

1. INTRODUCTION

The VVER reactors represent a basis of Ukrainian nuclear power engineering. There are 11 VVER-1000 reactors and 2 VVER-440 reactors in Ukraine. The reactors were mainly designed in 70th and respective nuclear power units were put into operation in 80th. For last a few years the VVER nuclear power units were covering more 40% electricity production in the country. The VVER reactors are located at 4 NPPs (Zaporizhia, Sourh-Ukraine, Rivne and Khmelnytsky sites).

By the original NPP projects developed at the former USSR, all VVER spent fuel was planned to transport for reprocessing to the radiochemical enterprises located at the South Ural and Siberia. Spent fuel has to be transported after 3-year cooling period since its unloading. According to this concept:

1. At-reactor cooling pools were designed with minor capacities, sufficient to provide 3-4 refuelling,
2. Spent fuel transport casks and means were developed,
3. No spent fuel storage facilities were constructed to store VVER spent fuel for a long time.

After year 1992 as to Ukraine NPPs, the spent fuel management concept concerning VVER spent fuel was broken because of well-known political events (USSR disintegration).

In due course, since beginning of 90th the another kind of nuclear fuel has been begun to apply at VVER reactors. It has had higher enrichment (4,4%U235 against 3,3%U235 prior fuel), more burnup limit (49 MW*d/kgU against 38 MW*d/kgU prior fuel) and it can be applied for 3-year fuel campaign against 2-year fuel campaign for prior fuel. Present-day, advanced fuel assemblies with integrated burnable absorber (Gd2O3) and zirconium alloy structural elements (spacing grids, guide tubes) are in experimental-industrial operation in Ukrainian VVER nuclear power units. It is further expected that the various improvements of VVER both nuclear fuel and core operation will allow to apply fuel with 4,4 %U235 enrichment for 4–5-year fuel campaign, reaching 50-55 MWd/kgU burnup (the mean for discharged VVER fuel assemblies).

2. BACKGROUND

In the designs for spent fuel both storage and transport systems, which were developed in 60-70th years on the basis of the simplified (conservative) analysis of multiplying properties, subcriticality to be required was provided by a choice of the large centre-to-centre distance of placed assemblies (uncondensed placement). In later designs introduced for some NPPs, it is accepted condensed spent fuel storage with implantation of a neutron absorber in structural materials.

Now, when for safety analysis of nuclear fuel management systems it is required to apply marginal assessment methods (account of a complete range of parameter changes), the safety analysis results, in some cases, in unsatisfactory results. Besides, to improve nuclear fuel use economy the fuel with more effective multiplying properties (advanced fuel) has been began to apply for all Ukrainian VVER reactors.

The safety analysis based on marginal assessments, "fresh" fuel assumption and taking into account fuel improved multiplying properties results, in some cases, in conditions, when the nuclear safety criteria are broken. To ensure performance of these criteria it is necessary to worsen the economic characteristics of spent fuel management systems. To avoid economic parameters deterioration of the system the burnup credit can be used.

3. SPENT FUEL TRANSPORT AND STORAGE STATUS

3.1. Regulatory basis and licensing criteria

The requirements to nuclear safety assurance under transport and storage of spent fuel are put in a series of regulations developed as in former Soviet Union as in Ukraine. The main ones, concerning nuclear subcriticality assurance are:

1. Basic regulations for safety and physical protection during nuclear material transport, issued in 1984,
2. Safety regulations for storage and transport of nuclear fuel within nuclear power facilities, issued in 1992.

For both regulations an effective neutron multiplication factor (k_{eff}) has been accepted as main nuclear safety criterion. It is required:

1. The factor has to be less 0.95 ($k_{eff} < 0.95$) under conditions of normal operation and/or in design-basis accidents,
2. Safety analysis has to be based on "fresh fuel" assumption.

It should be noted the following: in contrast to the OPBZ-83 regulations, the PNAE G-14-029-91 regulations permit to use burnup credit as a nuclear safety parameter if burnup level is measured by specified devices and before to place spent fuel into storage facility. No detailed explanations exist concerning this safety norm implementation.

3.2. Spent fuel characterization

Present and prospective (till 2010) spent fuel amounts. As of the beginning of 1992 there was 2205 metric tons of spent fuel (HM) in total in Ukraine. Since 1992 till 1999 about 1626 MTU of VVER spent fuel was generated by Ukrainian NPPs, including:

1. VVER-1000 spent fuel –1458 MTU,
2. VVER-440 spent fuel –168 MTU.

Because of the NPPs' very limited at-reactor storage capacities a share of VVER spent fuel has been transported to the reprocessing plants in Russia. 993 MTU of spent fuel were transported in Russia for its reprocessing during the same period (since 1992 till 1999). Thus, 2838 MTU of VVER spent fuel stayed in Ukraine as of the beginning of 1999.

Nuclear spent fuel amount growth in Ukraine, as a prognosis one so in fact one, is depicted in FIG. 1 . The prognosis growth was estimated provided that the annual design refuelling is as follows: 54 SFAs per a VVER-1000 unit, 120 SFAs per a VVER-440 unit.

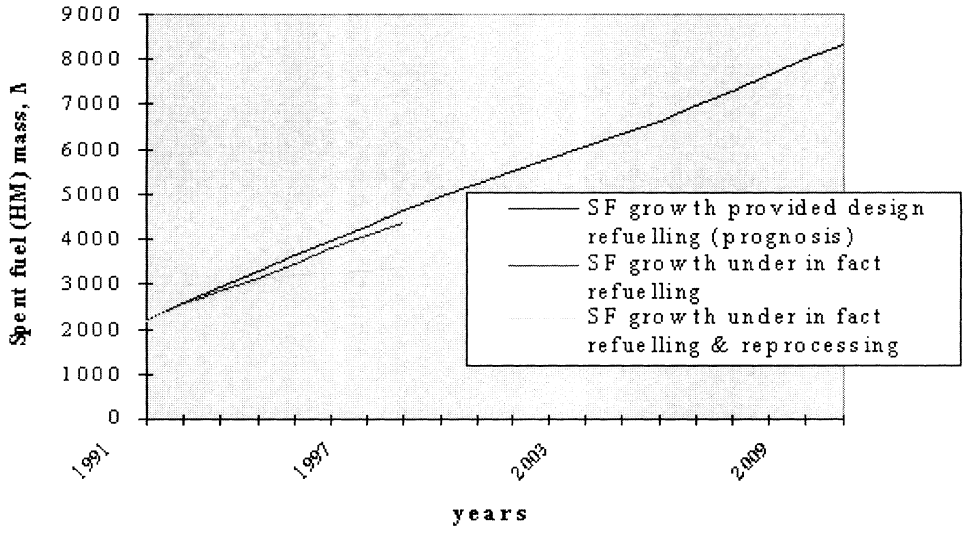


FIG. 1. Spent fuel growth in Ukraine: prognosis and in fact

Some VVER spent fuel characteristics are depicted in the table I.

Table I. The Comparative VVER-1000 Spent Fuel Characterization.

Species of operational cycle	Enrich-ment of make-up fuel, %	Uranium mass per FA, kg	Design-ba-sis burnup (average/ limit), MWd/kg U	Reached burnup (average/ limit), MWd/kg U	Specific natural uranium expenditure, kg/ MWd	Back-end fuel cycle cost/ Total fuel cycle cost, mills/kWh
two-year fuel campaign	3,3%	429,5	28,5/38	25/35	0,263	2,87/7,23
three-year fuel campaign	4,31%	401,6	40,1/49	39/43	0,256	2,11/6,27
three-year fuel campaign subject to a portion of FAs being used in four-year fuel campaign	4,31%	401,6	– /49	41,5/48,8	0,240	1,93/5,76

3.3 Transport systems

Originally, a few cask types were designed to transport VVER spent fuel. They are TK-6 for VVER-440 spent fuel, TK-10 and TK-13 for VVER-1000 spent fuel. Some features of casks accordingly theirs certificates of approval are shown in table II.

Table II. Features of casks accordingly theirs certificates of approval.

Fuel feature		Cask type		
		TK-6	TK-10	TK-13
Reactor type		VVER-440	VVER-1000	
Max. initial enrichment	wt.%	3.6	4.4	4.4
Number of Assemblies		30	6	12
Minimum cooling time	years	3	3	3
Burnup limit	MW*d/kgU	40/24	50	50
Maximum decay heat	kW per cask	15(water)/8(gas)	13	20
Multiplication factor			0.853±0.008	0.943±0.0023
Neutron absorbers			boron carbide backfilling	borated stainless steel as fuel basket material

Each one of cask types assures “fresh fuel” subcriticality ($k_{eff} < 0.95$) for normal operation and design-basis accident conditions.

These casks were originally licensed in former USSR, now Russian regulatory body renews periodically the certificates of approval for TK casks and Ukrainian regulatory body justifies the certificates of approval.

There are the specified rail-road aggregates to ship the TK casks. A specified rail-road aggregate and a TK cask are together a inseparable transport-package assembly (TUK).

There isn't a program in Ukraine to implement burnup credit to the TK transport casks.

3.3. Storage systems

Till nowadays in Ukraine the spent fuel storage capacities are restricted with at-reactor pools only. To provide additional capacities to store spent fuel the possibilities are considered to put the "dry" spent fuel storage technologies in practice in Ukraine.

3.3.1. Wet storage

3.3.1.1. Design-basis capacities for spent fuel storage

Following at-reactor pools capacities for spent fuel storage are defined at original NPP designs:

1. A VVER-440 pool - 729 cells. Annual design-basis spent fuel discharge is 120 SFAs (3-year fuel campaign),
2. A VVER-1000 pool - 392 cells. Annual design-basis spent fuel discharge is 54-55 SFAs (3-year fuel campaign).

As it has been noted above the initial design at-reactor pool capacities were intended for spent fuel storage of 3-4 annual design refuelling, not more.

Pool subcriticality ($k_{eff} < 0.95$) were ensured by large centre-to-centre pitch between fuel assemblies.

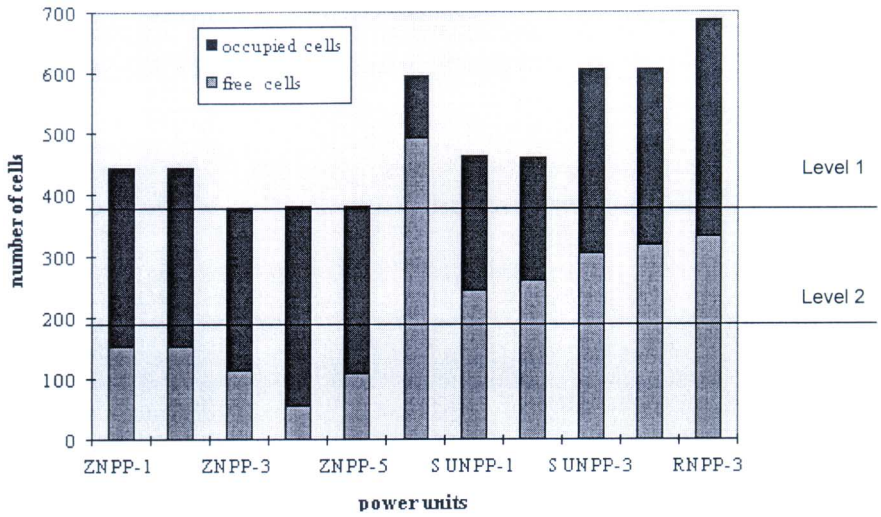
3.3.1.2. Pool re-racking

In the first half of the 90th there were re-racked the VVER at-reactor pools in Ukrainian nuclear power units, except of a few Zaporizhia and Rivne NPPs' units, and pool capacities have been enlarged from 1.4 to 1.7 times in that way.

Subcriticality for the pools re-racked was ensured by a centre-to-centre pitch and neutron absorber in rack materials (2-3% borated stainless steel).

In spite of the pool capacities enlarging the free storage capacities in the at-reactor pools have being diminished a year by year as spent fuel has been shipped for reprocessing in more few amounts then it has been generated.

As example, the VVER-1000 at-reactor pools fullness is depicted in FIG. 2.



Remarks: 1. Level 1 – design-basis pool capacity (392 cells),
2. Level 2 – constantly free pool capacity according to safety standards (163 cells),
3. Column tops – total pool capacity.

FIG. 2. VVER-1000 at-reactorpools fullness with SFAs.

3.3.2. Dry storage

The "dry" storage facility project for Zaporizhia NPP based on the Sierra Nuclear Corp.'s VSC-24 cask design has been begun in 1994. This project (VSC VVER-1000) is aimed to store the whole spent fuel to be generated during Zaporizhia NPP lifetime. Now the facility's "start" stage (3 casks) is ready to be put into operation. It is expected a regulatory body's licence to put the facility's "start" stage into operation will be shortly issued.

3.3.2.1. Cask subcriticality

According to VSC VVER-1000 safety analysis report the cask subcriticality can be ensured by burnup credit implementation or embedding of control rods into spent fuel assemblies.

In case of burnup credit implementation the cask subcriticality can be ensured for any initial fuel enrichment, but in case of control rods embedding the cask subcriticality can be ensured for "fresh" fuel with 3,6% maximum enrichment.

Criticality calculations have been carried out with SCALE-4.3 software code including BONAMI, NITAWL and KENO-VI code modules.

VSC VVER-1000 safety assessment has displayed exceeding of limiting keff under conditions, if fuel is fresh and its initial enrichment is 4,4%, and provided that all 24 spent fuel assemblies (design-basis cask capacity) are loaded into cask.. To provide cask subcriticality with keff < 0.95 under mentioned "conservative" conditions, one should be loaded into cask only 10 spent fuel assemblies, if initial fuel enrichment is 4.4%. Cask loading with spent fuel can be enlarged to 22 assemblies, if the spent control rods are used as a neutron absorber in a cask. In any case, cask subcriticality status ensuring involves an anticipatory economic damages associated with incomplete cask loading.

To avoid these economic damages burnup credit can be used to ensure cask subcriticality under normal operation and design-basis accident conditions, and if initial fuel enrichment is 4.4%.

Zaporizhia NPP operator has begun R&D activity to developed the equipment and technique for burnup credit implementation, however, the results weren't yet obtained.

4. SUMMARY

Burnup credit is presently not used in Ukrainian NPPs neither transport no storage of VVER spent fuel. Prospects to use burnup credit for spent fuel transport and wet storage aren't well-defined sufficiently now. However, there are clear needs to implement burnup credit for dry storage facility with VSC VVER-1000 casks on Zaporizhia NPP site.

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BURNUP CREDIT ACTIVITIES IN THE UNITED STATES

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Abstract

This report covers progress in burnup credit activities that have occurred in the United States of America (USA) since the International Atomic Energy Agency's (IAEA's) Advisory Group Meeting (AGM) on Burnup Credit was convened in October 1997. The Proceeding of the AGM were issued in April 1998 (IAEA-TECDOC-1013, April 1998). The three applications of the use of burnup credit that are discussed in this report are spent fuel storage, spent fuel transportation, and spent fuel disposal.

1. INTRODUCTION

Light water reactor (LWR) systems which are used by the commercial nuclear power industry in the USA use fuels with low concentrations of fissile uranium (less than 5.5% initial concentration of U-235 by weight). The LWR systems used in the USA are boiling water reactor (BWR) and pressurized water reactor (PWR) systems. The fission process consumes the fissile U-235 and produces new isotopes, which include various actinides and fission products. The actinides produced include fissile materials (e.g., Pu-239 and Pu-241) and neutron absorbers (e.g., Pu-240 and Pu-242). Hundreds of fission products are also produced; however, only a small number of them are significant neutron absorbers. Consideration of the reduced reactivity from the changes in isotopes for the purpose of determining reactivity of spent fuel is known as burnup credit.

The USA does not reprocess its spent nuclear fuel, and the fuel from about 100 reactors is stored at the reactor sites in pools or in dry storage units. New reactors are not expected to be built in the USA, and a geological repository is planned for the ultimate disposal of the spent fuel from these reactors. The USA expects to receive about 85,000 tU of commercial spent nuclear fuel after the end-of-life for all current reactors. After a start-up period, the fuel will be accepted into the U.S. Department of Energy's (DOE's) Civilian Radioactive Waste Management System (CRWMS) at a rate of about 3,000 tU per year. Spent fuel and high-level radioactive waste will be accepted from 78 facilities, which include 73 commercial sites and five DOE sites. Some of the reactors share common sites. The proposed disposal site at Yucca Mountain is located in southern Nevada (NV). It is being characterized and studied to determine its suitability as a repository. If suitable and licensed by the U.S. Nuclear Regulatory Commission (NRC), acceptance of spent fuel and high-level nuclear waste will begin in 2010. The period for receiving the total 70,000 tU capacity of the repository is estimated to be 30 years.

Only spent fuel that has been discharged from a reactor and cooled five years or more will be accepted into the CRWMS. This older fuel has undergone significant and rapid decay of its gamma and heat emitting radioactive contents, and the neutron absorbers have begun to stabilize. The reactivity potential of the spent fuel continually decreases for a few hundred years after discharge from the reactor. Then slight, but continued, increases in reactivity occur, peaking at between 10,000 and 30,000 years, the cycle repeats, peaking again at about 300,000 years and decreasing thereafter. However, these peaks in reactivity do not exceed the value at five years after discharge.

In the USA, the use of burnup credit for spent fuel management has been pursued by the private sector and by the Federal government. The private utility companies and their member sponsored research and development organization, the Electric Power Research Institute (EPRI), have pursued burnup credit for various storage applications.

Burnup credit development activities conducted by the Federal government in the USA have been performed primarily by DOE. DOE began these efforts in the mid-1980's to support its CRWMS activities, which include storage, transportation, and disposal of spent nuclear fuel from light water reactors. In the mid-1990s, DOE submitted an initial version of a topical report to the NRC for the use of actinide-only burnup credit for transport of PWR fuel. Thus began a series of formal exchanges that led to NRC issuing guidance documents on actinide only transportation burnup credit in 1999.

The private sector has generally been interested in burnup credit, and has been successful in its efforts to gain approval for burnup credit in wet (pool) storage applications. The industry has been actively involved in the transportation and storage aspects of burnup credit conducted by the DOE. EPRI has been a focal point for industry involvement in burnup credit. EPRI has worked with DOE on burnup credit, and has conducted a number of activities that have been beneficial to the transportation burnup credit efforts.

In 1998, DOE relinquished its role in transportation burnup credit, passing the torch to the private sector. Since then, EPRI has increased its involvement, and cask vendors, instrumentation and transport service organizations, and utilities have become involved in burnup credit activities. In addition, the Nuclear Energy Institute (NEI) has initiated interactions with NRC to assist in defining industry needs in the area of burnup credit for transportation and dry storage of spent fuel. NEI is an organization that is supported by the nuclear industry. NEI represents the nuclear industry in dealing with the federal government on broad issues that face the entire industry.

The DOE also plans to use burnup credit for disposal of spent nuclear fuel. The waste package (WP) which is being designed for disposal of spent nuclear fuel in a deep geological repository is expected to provide criticality control during some time period after disposal. The regulatory period for the repository will be 10,000 or more years. Over this time, degradation of conventional criticality control features cannot be completely precluded. Consideration of the actual reactivity of spent nuclear fuel over the time of disposal is therefore needed. The consideration of the actual reactivity of commercial spent nuclear fuel (burnup credit) is part of the risk-informed, performance based methodology being used to analyze disposal systems.

2. PROJECTED SPENT FUEL INVENTORY

Current projections for spent fuel discharges from commercial nuclear power plants in the USA are based on an assumption that there will be no new reactors built. This assumption leads to an estimate of about 85,000 tU of spent fuel when all current reactors are retired. The mixture of PWR and BWR reactors for the USA is 67% and 33%, respectively. This estimate of spent fuel discharge exceeds the 70,000 tU capacity of the repository site being studied. The repository limit includes about 63,000 tU of commercial spent fuel and 7,000 tU of other radioactive waste. The fate of this excess commercial spent fuel (~22,000 tU) is still to be decided.

An important factor in making decisions regarding the use of burnup credit for storage, transport, and disposal of spent fuel is the number and types of spent fuel assemblies that must be dealt with. For either type of spent fuel (BWR or PWR), two important properties needed for such decisions are their burnup and initial U-235 enrichment values. The projected inventories of 124,761 PWR and 166,942 BWR spent fuel assemblies for the USA are arranged by burnup and initial U-235 enrichment in Tables I and II [1]. Although there is twice the projected discharge mass of PWR spent fuel, the smaller mass of BWR assemblies results in their larger number.

The data in Tables I and II are projections that could be affected by several factors. The projected cumulative spent fuel discharge of about 85,000 tU, for the case of no new reactors in the USA, is expected to be realized in 2042. The accumulation of discharged spent fuel in 1996 was about 40 tU, and about 48 tU in 2000. Factors that could increase the inventory of discharged spent fuel include renewing existing reactor licenses to extend reactor operating lives an additional 20 years, and construction of new reactors. The former is happening now, the latter is not considered likely. Factors that could decrease the inventory of discharged spent fuel include early shut-down of reactors (before their 40 year licenses expire), and increasing operating cycles by using fuel with higher initial enrichment and burning the fuel longer. Both situations are expected. Some reactors have been shut-down prematurely, and fuels are being designed to accommodate longer operating cycles.

3. SPENT FUEL STORAGE

The NRC issued guidance on criticality analysis of spent fuel storage in August 1998 [2]. Anyone interested in the NRC guidance is advised to refer to the report directly.

The guidance is intended to help NRC licensees conduct criticality safety analyses of fuel storage pools for unirradiated and irradiated light water reactor fuel. The guidance is limited to applications for pool storage reviewed by the NRC's Reactor Systems Branch. That is, the guidance does not apply to other areas that the NRC regulates, such as transportation, dry fuel storage, or disposal. The guidance is intended to clarify current NRC practice regarding review of licensee safety analyses for pool storage. The guidance also gives NRC staff positions on storage approaches being proposed. The guidance addresses criticality analysis methods for pool storage of LWR fuel, including BWR fuel and PWR fuel. The document provides general guidance on criticality analysis, and specific guidance for treating abnormal conditions and the double contingency principle, new fuel storage, spent fuel storage, and additional considerations that the NRC staff felt needed clarification.

Table I. PWR Spent Fuel Inventory.

Burnup	Initial U-235 Enrichment (%)									
GWd/t	0-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	4.0-4.5	4.5-5.0	5.0-5.5
70-75							1		18	35
65-70							2	65	1080	195
60-65							5	873	4704	533
55-60					1	2	57	3512	6026	361
50-55					1	9	933	8547	5444	183
45-50					48	274	5066	11440	2460	25
40-45		1	1	1	110	2433	10367	7484	476	
35-40	4			48	544	8569	8135	1470	12	
30-35			1	33	3329	7584	1550	873	8	
25-30			36	1027	3708	2410	453	117	36	
20-25		12	190	1758	880	508	59	162	28	
15-20	4	32	1821	3896	137	184	20	111	4	
10-15			1393	302	47	113	11	74		
5-10		25	9	3	35	92	76	14		
0-5	8		3		1		1	28		

Table II. BWR Spent Fuel Inventory.

Burnup	Initial U-235 Enrichment (%)							
GWd/t	0-1.0	1.0-1.5	1.5-2.0	2.0-2.5	2.5-3.0	3.0-3.5	3.5-4.0	4.0-4.5
70-75								
65-70							10	240
60-65							104	900
55-60						20	1122	2354
50-55						270	5373	3734
45-50						2907	11548	2277
40-45					155	10948	10573	542
35-40			111	24	1019	22216	3199	
30-35			3	242	11220	13748	33	
25-30		28		3251	17094	2227	26	
20-25		27	176	7594	6792	217	49	
15-20	52	53	3000	5636	917	10	229	
10-15	140	435	6997	1875	547	203	173	
5-10	572	588	286	461	307		54	
0-5	1116	2	181	666	58	1	7	

The NRC guidance provides useful information for those wishing to use burnup credit to store spent fuel in pools. The NRC guidance provides several specific considerations when "credit for the reactivity depletion due to burnup" is used. An additional consideration, which applies to cases of burnup credit or storage of fuel with different enrichments, is the possibility of misloading. The guidance further states that assuming a single loading error is usually sufficient, and "under the double contingency principle, credit for soluble boron, if present, is acceptable for these postulated accident conditions. The guidance addressing partial credit for

boron gives two conditions that must be satisfied: 1) for full density unborated water it must be shown that keff < 1.0, and 2) for water borated to a licensee specified concentration, it must be shown that keff is no greater than 0.95. Although the guidance does not preclude burnup credit for BWR spent fuel, only utilities with PWR pools have used burnup credit.

The use of burnup credit for wet storage of spent fuel is generally a final step in maximizing pool capacities. Utilities have used various methods of reracking pools prior to pursuing burnup credit. Furthermore, burnup credit for wet storage has only been used for PWR systems in the USA. About two-thirds of the reactors in the USA are PWR systems and about half of those reactors use burnup credit in their spent fuel storage pools. The goal of reracking and using burnup credit is to assure sufficient spent fuel storage capability without having to expand or build new fuel storage pools. Another storage approach being pursued for spent fuel is the use of dry storage systems. The method avoids the need to find space in existing pools or build new pools. However, the fuel must be sufficiently cool to be stored dry, requiring initial storage in a pool.

4. SPENT FUEL TRANSPORT

4.1 Introduction

To support development of advanced technology spent fuel transportation casks DOE began to pursue the use of burnup credit in the mid-1980's. The approach that DOE planned to follow in its pursuit of burnup credit was first presented to the NRC at a DOE sponsored workshop held in February 1988. The workshop was first of many meetings, and a great deal of correspondence between NRC and DOE on the subject of burnup credit. In 1988, the DOE strategy was to seek NRC approval of "full" burnup credit that would cover the range of initial enrichments and burnup values of all spent fuel in the anticipated inventory. Although the full burnup credit approach was not actually intended to take credit for all possible negative reactivity changes that could be attributed to burnup, it would take credit for an amount considered practical. That is, it would account for all fissile actinides, most neutron absorbing actinides, and a small number of fission products that accounted for about 80% of the available credit for all fission products.

Based on its interactions with NRC, which began in 1988, DOE submitted its first proposed burnup credit methodology to NRC in May 1995 [3]. The Topical Report was an actinide only approach, that is, it did not seek credit for fission products. Although DOE ceased work on advanced technology casks by 1996, the transportation burnup credit activities continued, and the NRC review and comment of the 1995 report led to a revised version of the report submitted in May 1997 [4]. This version of the report provided additional support for the proposed methodology for actinide only burnup credit for spent fuel up to 5% initial enrichment and 50 GWd/t burnup. The proposal was not accepted by NRC.

Following feedback from NRC, DOE decided to develop a final version of the actinide only approach. During development of the second revision of the actinide only burnup credit topical report, DOE concluded that achieving NRC acceptance of the use of burnup credit for transport of spent fuel would be more effectively pursued if requests were made a part of specific cask certification applications. Therefore, submittal of the revised report marked the conclusion of these activities by DOE and the start of efforts to transfer the technology to the private sector.

The final version of the DOE topical report was submitted to NRC in October 1998 [5]. DOE proposed an approach that limited initial enrichment and burnup to 4% and 40 GWd/t, respectively. At the completion of their review, NRC issued interim guidance on transportation burnup credit (ISG-8) on 17 May 1999, and a revision to the guidance in August 1999 [6, 7]. The initial guidance endorsed the use of a limited version of DOE's final proposal for actinide only burnup credit for transport of spent fuel. The revised guidance was more generic. It rescinded the apparent endorsement of DOE's proposed approach, but permitted more flexibility to prospective applicants for approval of burnup credit for spent fuel transport.

NRC offered its initial guidance on 17 May 1999, at a public meeting between NRC and NEI. Participants at the 17 May 1999 meeting were informed by NRC's Spent Fuel Project Office (SFPO) that the guidance represented a first step in a process, intended to expand the credit for burnup. SFPO is the NRC organization responsible for licensing of spent fuel transport and storage systems. SFPO will be supported by NRC's Office of Research (RES) in their efforts to expand the credit given for burnup. The NRC described a two phased research activity to be conducted by RES to expand credit beyond initial guidance issued by SFPO in ISG-8. The phases the NRC's research effort are described as follows:

1. In the first phase, RES will evaluate available information. This effort is expected to result in a modest increase in Actinide-Only burnup credit within six months. The NRC surpassed their promise by issuing revised guidance (ISG-8, Rev 1) in August 1999.
2. A second phase effort, expected to take two to three years, will define data and experimental needs required to further expand the credit given for burnup as a result of the first phase activities. This expansion is expected to include some credit for fission products. The first part of this second phase is well underway. NRC issued a first report on priorities of burnup credit issues in February 2000 [8].

At the May meeting, NRC advised that their ability to fund burnup credit research efforts was limited. Therefore, research and data acquisition needs identified to support any expansion of the burnup credit would be industry's responsibility.

The NRC guidance is in the form of a recommendation to those wishing to use burnup credit for transportation of spent fuel in an NRC approved cask. The guidance indicates that NRC has suggested a basic methodology. The applicant must adapt the basic methodology to the specific cask being proposed for burnup credit, acquire the data required, and provide all analyses to show that regulatory criticality criteria are satisfied.

4.2 Benefits of Using Burnup Credit

The motivation for pursuing burnup credit for transport of spent fuel in the USA has been the opportunity to reduce the number of projected shipments needed to move spent fuel from the reactors located across the country to a repository. For PWR casks, burnup credit has been used to eliminate the need for flux traps, which results in closer packing of spent fuel and significantly increased capacities. These increased capacities result in fewer shipments. The consequences of fewer shipments include reduced exposure to public and workers, and lower costs of shipping. The cost benefit, which has been addressed for the CRWMS is considerable [9]. As an example, if all transport to a repository is done using truck casks, the estimated cost of $US 3 billion without burnup credit can be reduced to $US 1.5 billion, a saving of $US 1.5 billion. For the truck transport case, burnup credit was assumed to can increase cask capacities

from two to four PWR assemblies. For rail, capacities are typically 24 PWR assemblies without burnup credit, and 32 with burnup credit. The corresponding costs for rail transport were estimated to be $US 535 million and $US 401 million, a savings of $US 134 million. Rail transport costs are lower than truck transport costs with or without burnup credit, and savings are less dramatic.

4.3 Truck Casks

In the USA, one can categorize general highway freight systems into two groups. These are called legal weight truck (LWT) shipments and over-weight truck (OWT) shipments. The weight limit for an LWT system, which includes tractor, trailer, and cargo, is 80,000 pounds (36,280 kg or 36.3 t). The significance of this designation is the fact that shipments meeting this weight limit can travel without restriction on interstate highways. Systems exceeding the LWT limit, are designated as OWT shipments. The OWT shipments must meet specific vehicle and weight distribution requirements, and be permitted by each state they transit.

Although OWT restrictions are not onerous, many truck cask systems in the USA are designed to meet LWT limits. Generally, this means that the cask is limited to about 22,680 kg (50,000 pounds), the remainder allocated to the weight of the tractor and trailer. The LWT weight restriction has resulted in numerous truck casks with capacity limits of one PWR or two BWR assemblies. In 1988, DOE contracted with several cask vendors to develop advanced technology casks to support its CRWMS efforts. One of the casks developed under this program was the GA-4, which could carry up to four PWR assemblies. This achievement was accomplished by optimizing the design and using innovative technical approaches. One of those approaches was to use burnup credit. Although it is not the only factor, it is significant.

In addition to the GA-4, DOE contracted for the development of the GA-9 LWT cask with a capacity of nine BWR assemblies. Like the GA-4 cask, the high capacity of the GA-9 was achieved by optimizing the design and using innovative technical approaches. However, burnup credit was not one of the innovations needed for this BWR cask.

4.4 Dual-Purpose Rail Systems

The use of rail casks for spent fuel transport for disposal in the USA is generally preferred over truck transport. This preference is strongly influenced by the number of shipments required and the travel distances expected. The capacities of rail casks are at least a factor of five greater than truck cask capacities. The average distance from reactors in the USA to the proposed site at Yucca Mountain is about 4,000 km (2500 miles). Furthermore, the rail system in the USA, and the locations of the reactor sites and proposed repository are compatible with rail transport. It should be noted that the general advantages and preference for rail transport does not exclude the need for LWT casks in the system. There are several reactors in the USA with facility or near-site constraints that make truck transport the option of choice.

The recent emergence of dual-purpose rail cask systems in the USA is another factor that affects the choice of transport mode. While awaiting the start of operation of the federal repository, many utilities have started to use dry storage at their reactor sites to supplement pool storage. Initially, single purpose storage systems were used, but recently utilities have been opting for canister based systems that can be used to store spent fuel now and be transported off-site later with a minimum amount of fuel handling. Burnup credit has been identified as a means of increasing capacities of these systems.

PWR storage only systems achieved capacities that were comparable to those attainable from using burnup credit by an administrative control approach. Basically, a PWR storage cask was loaded in a pool, which contained some minimum concentration of boron. In addition to the normal internal cask moderator control, the loaded cask was then required to be located in an area absent of water. The approach is not compatible with dual-purpose systems. For these systems, approval for transport burnup credit is a necessity to achieve the desired capacities.

Several dual-purpose cask vendors anticipate seeking NRC approval for cask system designs that use burnup credit. The typical design is a modification of a non-burnup credit configuration of a cask. For these upgraded systems, capacities are expected to increase from 24 PWR assemblies without burnup credit to 32 with burnup credit. Although burnup credit for BWR systems has not been broached yet, at least one cask vendor is considering the option.

4.5 Private Sector Activities

The DOE technology transfer initiative, which began upon submittal to the NRC of the second revision of the Actinide Only Topical Report, appears to be moving forward successfully. The letter from DOE that transmitted Revision 2 of the Actinide Only Topical Report stated that DOE would no longer continue its transportation burnup credit efforts [8]. The letter recognized that progress in transportation burnup credit would be most effective if applied to specific cask design applications. For that reason, the technology would be transferred to the private sector. At least three cask vendors who are likely candidates for seeking burnup credit are expected to do so prior to 2001.

General Atomics (GA) is the developer of the GA-4 truck cask. GA has not expressed any immediate intentions of seeking burnup credit for their cask. However, it is included because it was specifically designed with burnup credit in mind, and it is the only current LWT cask design that is a candidate for burnup credit. The GA-4 cask was approved by the NRC on 27 October 1998. The approval allows shipment of four PWR assemblies with initial enrichment of up to 3.2% U-235 without burnup credit. To achieve the 3.2% initial U-235 enrichment, GA uses enriched boron in the fuel support structure (basket). Although the allowable initial U-235 enrichment is about 3% with natural boron, the small increase in initial enrichment allows a significant increase in fuel that could be shipped at full capacity (see Table I). The demand for LWT cask systems is expected once a repository begins operations. GA will likely see an emerging market for the GA-4 cask, and seek burnup credit.

GNB, a German based company with operations in the USA, is developing the CASTOR X/32 S. The cask is a dual-purpose system similar to the traditional CASTOR designs, except that the containment structure is made of forged carbon steel instead of the ductile iron castings that have come to characterize the CASTOR casks. It is designed to hold 32 PWR assemblies in a single dual-purpose unit that is used for storage and transport. This is not a canister-based system, which makes fuel configuration changes more difficult. For canister-based systems, the same cask body can be used with separate canister designs with and without burnup credit. This makes it relatively easy to convert a design once burnup credit is approved. GNB seems to have overcome this obstacle by using an innovative licensing approach. The cask is designed for 32 PWR assemblies with or without burnup credit. When burnup credit is not used poison rods (e.g., new control rods) will be inserted into the fuel assemblies for criticality control. When burnup credit is approved, the poisons rods will not be needed. GNB is expected to submit an application for a CASTOR X/32 S with burnup credit to NRC in August 2000.

Holtec International currently has an NRC Certificate of Compliance for their Hi-Storm dual-purpose system. The current design has a 24 assembly PWR canister. Holtec International has had discussions with NRC about a modification to the design, which uses burnup credit to allow the use of a 32 PWR assembly canister. They are expected to submit an application for this burnup credit version of the Hi-Storm to NRC in September 2000.

NAC International currently has an NRC Certificate of Compliance for their NAC-UMS dual-purpose system. That system is approved for 24 PWR assemblies or 56 BWR assemblies. NAC, who has recently decided to seek burnup credit for the NAC-UMS system, met with NRC in June 2000 to discuss their plans. They expect to submit an application for the PWR configuration at the end of the year 2000. The burnup credit design will increase the PWR capacity to 32 assemblies. Interestingly, NAC is also looking at burnup credit for their BWR design. They anticipate increasing its capacity to 69 BWR assemblies by using burnup credit. However, they have given no firm date for their BWR design change.

Although there are other vendors developing dual-purpose systems in the USA, they seem to be continuing with non-burnup credit designs. There are several reasons for such a decision, which are noted here. Although NRC has issued guidance for burnup credit, the certainty and difficulty of getting an approval for its use is not yet known. Companies that focus their business on BWR plants find no compelling reason to seek burnup credit. The current restriction in the NRC burnup credit guidance related to burnable poison rods is discouraging. Finally, the industry trend toward the use of higher enrichments and burnup levels may introduce a thermal constraint on cask capacities that will make burnup credit less attractive.

5. SPENT FUEL DISPOSAL

DOE has the responsibility of managing the geologic disposal of high-level radioactive waste (HLW) and spent nuclear fuel (SNF) in the USA. This work includes consideration of criticality. Pursuant to this, the DOE has developed a risk-informed performance based disposal criticality analysis methodology [11]. The methodology provides a systematic approach for evaluating a combined system of a waste form, waste package, engineered barrier, and repository for limiting the potential for criticality through the entire postclosure period of the repository. The methodology includes the building of hypothetical scenarios that lead to intact, partical degraded, and degraded configurations, defining parameters for each configuration, evaluating criticality potential for the range and specific values of parameters, and the process for estimating the probability of critical configurations and their consequences. The methodology also includes the process for combining probability and consequence estimates with total system performance assessment (TSPA) radionuclide transport modeling to obtain an estimate of criticality risk, which is measured by the expected increment in dose rate at the accessible environment due to all potential criticalities. The portion of the methodology concerned with evaluating criticality potential includes consideration of burnup credit for commercial SNF. This criticality analysis methodology includes consideration of burnup credit for commercial SNF.

The NRC does not have specific regulations or guidance for burnup credit applications for disposal. Recently, NRC has issued a proposed rule for the federal repository. It will be included in the Code of Federal Regulations (CFR) Title 10, Part 63 [12]. The proposed regulation for disposal does not include any specific design criteria for post-closure criticality control. The proposed regulation considers criticality as one of the processes or events that

must be considered for the overall system performance. The NRC has issued a *Safety Evaluation Report for the Disposal Criticality Analysis Methodology* [13], which addresses burnup credit issues for disposal in general.

The use of burnup credit in disposal applications has an additional importance compared to transportation and short-term storage applications. Over the long time period considered for disposal, conventional criticality control features such as moderator exclusion barriers, poison (neutron absorbing) plates, and geometry features (e.g., flux traps) will degrade and change. The reduced reactivity associated with irradiated fuel is the only feature that may last. It is of key importance to large capacity waste packages that must provide criticality control for 10,000 or more years.

The disposal burnup credit methodology takes credit for the reduced reactivity associated with the build-up of the primary principal actinides and fission products in irradiated fuel. The disposal burnup credit is referred to as Principal Isotope (PI) Burnup Credit. DOE plans to use PI burnup credit for the criticality evaluations of waste packages loaded with irradiated BWR and PWR fuels from commercial reactors.

6. SUMMARY AND CONCLUSIONS

Burnup credit is being used in the USA for spent fuel storage in pools. The approach, approved by NRC for about half the PWR systems, has been used to avoid the need to expand existing pools or build new ones where additional storage is needed. The use of dry storage systems at reactors is a recent development that also offers a way of avoiding the high costs and licensing difficulties anticipated if pool expansion or new construction is needed.

Burnup credit has been pursued for transport of spent fuel as means of reducing the numbers of shipments required to move a given fuel inventory. The benefits of fewer shipments include those of health and safety, and cost. For the inventory of spent fuel expected to be discharged by all reactors in the USA, which is about 85,000 tU, savings potentials for all truck and all rail transport, using burnup credit, have been estimated to be as much as $US 1.5 billion and $US 134 million, respectively. The advent of dual-purpose casks that can both store and transport fuel has introduced an interesting dilemma in the USA. Single-purpose storage systems have been permitted for burnup credit-like configurations without the need for burnup credit. However, for dual-purpose, storage-transport systems, the option is lost for storage unless burnup credit is approved for transport. Recent developments in the USA suggest that NRC is preparing the necessary technical basis to approve burnup credit for spent fuel transport, and that a number of cask vendors are about to submit applications for such approvals.

The principal isotope burnup credit methodology being used for disposal in the USA takes credit for the reduced reactivity associated with the build-up of the primary principal actinides and fission products in irradiated fuel. Burnup credit is needed in disposal applications because the intrinsic reduced reactivity of irradiated fuel may provide the only criticality control over long time periods, when conventional features fail. Burnup credit will be used for BWR and PWR fuels from commercial reactors.

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REGULATORY ASPECTS

REGULATORY STATUS OF BURNUP CREDIT FOR STORAGE AND TRANSPORT OF SPENT FUEL IN GERMANY

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Abstract

This paper describes the regulatory status of burnup credit applications to pond storage and dry-cask transport and storage of spent fuel in Germany. Burnup credit for wet storage of LWR fuel at nuclear power plants has to comply with the newly developed safety standard DIN 25471. This standard establishes the safety requirements for burnup credit criticality safety analysis of LWR fuel storage ponds and gives guidance on meeting these requirements. Licensing evaluations of dry transport systems are based on the application of the IAEA Safety Standards Series No.ST-1. However, because of the fact, that burnup credit for dry-cask transport becomes more and more inevitable due to increasing initial enrichment of the fuel, and because of the increasing importance of dry-cask storage in Germany, the necessity of giving regulatory guidance on applying burnup credit to dry-cask transport and storage is seen.

1. INTRODUCTION

In Germany criticality safety design criteria are laid down in Regulatory Guides of the German nuclear technology committee KTA (”Kerntechnischer Ausschuß”) and Safety Standards of the German society of standardization DIN (”Deutsches Institut für Normung).

Burnup credit for wet storage of LWR fuel at nuclear power plants has to comply with the newly developed safety standard DIN 25471 [1] passed in November 1999 and to be published shortly. This standard establishes the safety requirements for burnup credit criticality safety analysis of LWR fuel storage ponds and gives guidance on meeting these requirements. In particular, methods acceptable to validation of fuel depletion and criticality calculation codes are specified, parameters affecting the burnup credit are characterized (e.g., axial and horizontal burnup profiles), and methods acceptable to the verification of the fuel loading procedure are specified.

DIN standards are industry codes. Even though not laid down by the law the KTA regulations are commonly regarded as superior to the DIN standards. The basic criticality safety requirements for wet storage of LWR fuel are laid down in the standard KTA 3602 [2]. This standard does not prohibit burnup credit, but requires to give reasons for deviating from the fresh fuel assumption if burnup credit is employed. Therefore, a working group was set up which has the task to revise the standard KTA 3602 in such a way that this standard endorses the new safety code DIN 25471. This working group will probably have finished its work at the end of the year.

Licensing evaluations of dry transport systems are based on the application of the IAEA Safety Standards Series No.ST-1 [3].

2. BURNUP CREDIT FOR WET STORAGE OF LWR FUEL

2.1. The Safety Standard DIN 25471

The safety code DIN 25471 consists of two parts. In the first part the regulatory standards are given, the second part comprises explanatory and advisory remarks which are nonobligatory.

In its first part the code establishes:

1. The criticality safety acceptance criteria,
2. The requirements for validation of the depletion analysis,
3. The requirements for validation of the criticality calculation codes,
4. The requirements for the evaluation of the reactivity effects of axial and horizontal burnup profiles,
5. The requirements for the verification of the loading procedure,
6. The methods acceptable to fuel burnup determination.

No restrictions are placed on the fuel type. Full burnup credit is allowed for PWR and BWR (as well as VVER) UO_2 and MOX fuel according to the state of the art. In the advisory part of the safety code it is stated, however, that at the present time state of the art means:

1. Application of the actinide plus fission burnup credit level [4] to PWR wet storage pools,
2. Application of the integral burnable absorber burnup credit level [4] to BWR wet storage pools.

The integral burnable absorber burnup credit level is allowed to be used as actinide plus fission burnup credit level and may be applied of course to PWR fuel too. Application of the actinide plus fission burnup credit level is not restricted to PWR fuel which had not used (integral) burnable absorbers.

The isotopes to be used in a criticality safety analysis of a wet storage system designed for burnup credit are not prescribed in the safety code DIN 25471. But as stated in the second part of this safety code the isotopes to be used are selected on the basis of their reactivity worth and their nuclear and chemical stability. In order that the selected set of isotopes is enveloping with respect to the spent fuel reactivity the following essentials are observed:

1. Isotopes which have a significant positive reactivity worth (U-235, Pu-239, and Pu-241) are always included in the burnup credit safety analysis,
2. Stable nuclides or radionuclides with half-lives much greater than the maximum possible cooling time of the spent-fuel are included in the burnup credit safety analysis,
3. Utilization of radionuclides with half-lives not much greater than the maximum possible cooling time is justified, either by the neutron absorption properties of the daughter products (e.g., decay of Eu-155 to Gd-155) or by examining the spent fuel reactivity as a function of cooling time (e.g., decay of Pm-147 to Sm-147).

2.1.1. Criticality Safety Acceptance Criteria

The loading curve of storage racks designed for burnup credit has to be based on the reactivity equivalence condition

$$k(e,B) + \lambda\sigma(e,B) = k_{limit} \qquad (2.1)$$

where *k(e,B)* denotes the mean value of the estimated neutron multiplication factor at initial enrichment e and burnup B, and $k(e,B) + \lambda\sigma(e,B)$ is the 95%/95% one-sided upper tolerance limit of the estimate *k(e,B)* (and in fact the correct 95%/95% tolerance limit is meant [5]). k_{limit} is the maximum neutron multiplication factor permissible including all uncertainties to be considered – and all these uncertainties have to be expressed at the 95%/95% tolerance limit.

A loading curve usually refers to normal operation conditions of the storage racks. The analysis of abnormal or accidental events has to comply with the double contingency principle which requires at least two unlikely, independent, concurrent incidents to produce a criticality accident.

For the case of the misloading event (misplacement of a fuel assembly which does not comply with the loading curve into a storage cell of a storage rack designed for burnup), however, the double contingency principle has to be applied directly to this event, i.e., at least two unlikely, independent, and concurrent incidents are required to cause a misloading event.

Application of the double contingency principle directly to the misloading event is required for the following reasons:

1. The safety code DIN 25471 allows burnup credit for all fuel types, i.e., allowance for burnup credit is not restricted to storage ponds with borated water,
2. Even in case of borated storage ponds it cannot be excluded for any cases that a multiple misloading event may result in an unacceptable neutron multiplication factor.

2.1.2. Validation of the Depletion Analysis

As laid down in the safety code DIN 25471, the determination of the isotopic inventory of the irradiated fuel has to be based on fuel operating conditions characteristic of the plant of interest. The depletion calculation code used for this purpose has to comply with the requirements laid down in the standard KTA 3101.2 [6]. These requirements are met if the depletion calculation code used is validated:

1. Against experimental data from:
   - 1.1. observation and evaluation of normal power operation (stationary and non-stationary activation rate distributions, excess reactivity as a function of burnup measured in terms of soluble boron concentration or control rod position),
   - 1.2. special measurement programmes (reactivity coefficients and equivalents describing the dynamic behavior of the reactor, short-term – e.g., rod drop – and long-term – e.g., xenon – transients),
   - 1.3. chemical assay of the isotopic inventory of irradiated fuel;
2. As well as critical or subcritical configurations.

2.1.3. Validation of Criticality Calculation Codes

As laid down in the safety code DIN 25471, criticality calculation codes used for burnup credit have to comply with the requirements laid down in the standard DIN 25478 [7]. These requirements are met if the criticality calculation code is validated against experimental mock-ups or configurations which represent important features of the spent fuel storage racks and of the spent fuel content.

It is accepted in Germany that – in addition to the evaluation of all the critical experiments on different uranium and MOX systems, integral burnable absorbers, and experimental mock-ups simulating storage racks [8] – the evaluation of reactor critical configurations [9-10] is a cornerstone for the use of burnup credit, but there is also agreement that there is a lack of open experimental data for higher initial enrichments and higher burnups. Therefore, German nuclear power plants represented by VGB ("Technische Vereinigung der Großkraftwerksbetreiber") are participating in the REBUS programme [11]. For the same reason Siemens has participated or is participating in different programmes like ARIANE [12] or PROTEUS [13] for instance.

2.1.4. Evaluation of Reactivity Effects of Axial and Horizontal Burnup Profiles

No method is prescribed for the evaluation of reactivity effects of axial and horizontal burnup profiles. It has to be shown, however, that the methods used for this purpose comply with the following requirements:

1. The reactivity effect of axial burnup profiles (end effect) has to be evaluated on the basis of a sufficient number of axial profiles which are characteristic of the plant of interest and covering all the loading and operating strategies used,
2. Application of axial profiles from different plants is not allowed, unless the use of such profiles can be justified,
3. It has to be shown that the model functions (e.g., step functions) used to represent the axial profiles in the analysis does not result in an underestimation of the end effect,
4. The reactivity effect of horizontal profiles may be analyzed by using simple models proven to be enveloping with respect to the reactivity effect of horizontal profiles.

2.1.5. Verification of the Loading Procedure

As laid down in the safety code DIN 25471, in accordance with the quality assurance (QA) requirements established in the standards KTA 1401 [14] and KTA 3602 [2] written procedures shall be employed for the selection of fuel assemblies to be stored in the storage racks designed for burnup credit including:

1. Determination by an authorized person of the burnup of each candidate for storage in these racks,
2. Checking of the determination by an empowered person not involved in the initial determination.

For the determination of the burnup of the fuel assemblies intended for storage in the racks designed for burnup credit the following methods are acceptable:

1. Maximum permissible fuel assembly neutron multiplication factors corresponding to the minimum burnups required according the loading curve should be established, and a measurement of the neutron multiplication factor of the fuel assemblies should be performed to ensure that each assembly meets the loading curve,
2. Measurement of the burnup of the fuel assemblies should be performed to ensure that each assembly meets the loading curve,
3. Determination of the burnup of the fuel assemblies based on the reactor records should be performed to ensure that each assembly meets the loading curve.

For handling of fuel in borated spent fuel ponds burnup determination based on reactor records without additional requirements is generally accepted. In any case, as stated already in section 2.1.1 the misloading event has to be excluded by virtue of the double contingency principle: At least two unlikely, independent, and concurrent incidents are required to cause a misloading event.

2.2. Revision of the Safety Standard KTA 3602

The working group set up to revise the safety standard KTA 3602 [2] has already reached fully agreement in application of burnup credit and in which way the safety code DIN 25471 [1] should be indorsed by the standard KTA 3602. Links redirecting the user of the standard KTA 3602 to the safety code DIN 25471 have been incorporated into the text of the standard KTA 3602. In addition the working group has updated the standard KTA 3602 in numerous issues (e.g., introduction of a maximum allowable neutron multiplication factor of 0.98 for unlikely or postulated accidental events with low risk) and discussed the acceptability of partial boron credit for normal operation conditions in borated storage ponds.

3. BURNUP CREDIT FOR DRY-CASK TRANSPORT AND STORAGE OF LWR FUEL

Licensing evaluations of dry transport systems are based on the application of the IAEA Safety Standards Series No.ST-1 [3].

There are no national regulations that prohibit application of burnup credit to dry-cask transport and storage. However, because of the fact, that burnup credit for dry-cask transport becomes more and more inevitable due to increasing initial enrichment of the fuel, and because of the increasing importance of dry-cask storage in Germany, the necessity of giving regulatory guidance on applying burnup credit to dry-cask transport and storage is seen. It is planned, therefore, to work out criticality safety standards for burnup credit in dry-cask transport and storage on the analogy of the safety standard DIN 25471.

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REGULATORY ASPECTS OF BURNUP CREDIT IMPLEMENTATION

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Abstract

The paper reviews the regulatory aspects of the burnup credit implementation world-wide. Burnup credit methodologies have been accepted for the criticality safety analysis for wet storage in Spain. The depletion calculations, and determination of fuel discharge burnup, and burnup credit methodologies are described.

1. INTRODUCTION

Burnup credit methodologies have been accepted for the criticality safety analysis of all the on-site wet storage of spent fuel at all the plants in operation in Spain. Some applications were licensed more than 10 years ago, and were considered acceptable at that time on the basis on the overall conservatism of the full analysis process. While some specific steps of the methodology raised doubts about their accuracy or conservatism, those doubts were duly compensated by a quantifiable high conservatism in other parts of the analysis.

The progress made in the knowledge and understanding of the phenomena involved in burnup credit, and the evolution of the codes and libraries used for the analysis have helped to solve some of the concerns present in the old analysis. However, some aspects remain to be further clarified and solved. A brief review of some of those issues is given in the following paragraphs.

2. THE VALIDATION ISSUE

Burnup credit analyses involve two main calculations: fuel depletion and management system's reactivity. The first is aimed to determine the isotopic concentration of the different nuclides in the spent fuel as a function of burnup, given the fuel design characteristics and the depletion conditions. The criticality calculation determines the reactivity of the specific system based on the isotopic composition of the spent fuel determined in the prior step.

All the rules and regulations that apply to criticality safety demonstration ask for a detailed validation of the codes used in the process. Therefore, validation of both the depletion and the criticality code should be performed in burnup credit analysis, using experimental data covering a range coherent with the conditions to be analysed.

Validation of the depletion code should in principle be based on chemical assay data of spent fuel with the adequate burnup. The first problem found is that there is little chemical assay data available. Many interesting data is of a proprietary nature. The public data pertains mainly to major actinides, while data on fission product nuclides is very scarce.

Moreover, the majority of the public data can be considered "old". Most of it was obtained from old fuel designs, and the burnup range is in general less than 40 GWd/MTU. Also, there is little data for enrichments above 3.5 wt%. It can be concluded that the data available is not representative enough of present-day fuel and operating strategies. An additional problem is that the lack of data impedes the determination of trends on the code behaviour when important parameters are varied (burnup, enrichment,…).

Likewise, the criticality code should be validated by comparison with critical experiments performed with spent fuel of adequate burnup values. The same problem is again found here: no critical experiments with spent fuel are available. Validation against fresh fuel critical experiments allows the user to get confidence in the correct behaviour of the code, but it is clear that the use of the code for spent fuel analysis is not fully covered. The lack of spent fuel critical data is explained by several facts:

1. A critical state is difficult to obtain with spent fuel only. If fresh fuel is included, there will always be the doubt on whether the code bias obtained can be applied to systems containing spent fuel only,
2. The difficulties of such an experiment from a radiation protection standpoint are high,
3. The correct modelling of a spent fuel critical experiment requires either a fully validated depletion code, what is difficult to achieve as already discussed above, or sufficient chemical assay data of the fuel used for the experiment.

Given these facts, it is not expected that the number of spent fuel critical experiments will be increased in the near future. However, some reactivity worth experiments, in which the known reactivity of a fuel system is modified by including a certain amount of a fission product, are being carried out or planned.

Other validation option is that known as “integral validation”, the best example of which is the use of reactor core critical states as reference for validation. The reactor core is modelled with the depletion code, and the core reactivity is determined with the criticality code. The comparison with the measured critical boron concentrations, control rod patterns and core power distributions provides an overall quantification of the accuracy of the complete calculation system (depletion and criticality). It is felt that this kind of work gives confidence in the adequacy and quality of the calculation system. However, it is difficult to accept this approach as a validation if not complemented with additional validation efforts, due to the following reasons:

1. The uncertainty of the full calculation system is determined, and not that of the individual codes. Undetected error compensation between the different intervening codes is possible,
2. Fuel with different burnup levels is present in the core. The dependence of the uncertainty with the calculated burnup level cannot be determined, as only core average values can be obtained. Error compensation may again occur regarding this aspect,
3. Plant measurements can have a high uncertainty, and are difficult to use as benchmark quality data.

In addition, reactor critical conditions are different from those of spent fuel management systems, either storage of transport.

3. THE DEPLETION CALCULATION

The fuel assembly design has progressively increased its complexity during the years. The design is more heterogeneous both in the radial and axial directions, due to the inclusion of axial zones with different enrichments, axial blankets, “vanished” zones, axially varying burnable poison loading in selected rods only, annular pellet zones, water rods, etc. The trend is more remarkable in BWR fuel bundles, but the complexity of PWR fuel assemblies is also increasing.

These improved fuel designs are duly handled by commercial 2-D lattice codes, although some corrections and refinements have been needed even in the vendor codes. The problem with those codes is that their nuclear data libraries do not include all the nuclides relevant for burnup credit applications.

Codes handling enough nuclides are restricted in their geometric modelling capabilities, in the sense that some fuel characteristics relevant to the depletion calculation cannot be properly modelled. Enhancements are needed in those codes for burnup credit applications.

Other field that might need some improvement is that of the fuel operating conditions used for the depletion calculation. Usually, conservative values are considered for the different parameters (pressure, moderator and fuel temperature, void fractions, and others). These values are bounding with respect to nominal operating conditions. However, some phenomena detected in high duty fuel might indicate that the local conditions in some core zones can be rather different. These phenomena include excessive crud deposition, Boron accumulation in the crud, reduced (or increased) flow in some subchannels due to fuel distortion, very high exit temperatures in high-energy assemblies, and other. Some of these effects are not reproduced by the codes used for core analysis, or at least not completely, and they can have an effect on the calculated isotopic composition.

Likewise, the transition to higher discharge burnup values may deserve some careful consideration in burnup credit applications. Some phenomena have been demonstrated to exist in high burnup fuel, but the rod design codes are not yet prepared to model them. Therefore, the potential impact on the spent fuel reactivity cannot be assessed. As an example, the radial distribution of the isotopes within the pellet has been demonstrated to have a negligible effect on reactivity. However, as burnup is increased, the radial distribution of Plutonium is more and more skewed towards the pellet surface, and the balance between generation and consumption is modified. Whether this effect, as well as other internal to the pellet, can have an effect on the reactivity remains to be analysed.

4. THE OLD FUEL DATABASES

Most NPPs in Spain are now in the process of uprating power, once the operating cycle duration extension process that took place during the past years is ended. Much alike in most countries, the optimisation of the fuel cycle economics is leading to frequent changes in the core loading strategies and to extensions of the core operating domain, as well as to a gradual increase of the fuel discharge burnup. These changes may have two effects on the criticality safety analysis of the spent fuel storage at PWR plants:

1. The fuel operating conditions are changed (higher temperatures, harder spectrum, increased power peaking factors), so that they may lie outside the range originally considered for the depletion calculation of the burnup credit criticality safety analysis. This effect can obviously affect also the BWR applications, and is becoming relevant due to the extensive changes in the operation strategy that these plants are making in a continuous process,
2. Some of these changes, like power uprates and the extension of the operating cycle duration, call for the need of increased fuel enrichments. A new analysis is needed in those cases.

The problem that arises here is that burnup credit analyses rely heavily on the information obtained from the spent fuel databases. It has already been said that the data available has mostly been obtained from low-enrichment and low-burnup old design fuel, which was operated under core conditions very different from those used at present. On top of that, there are some second-order effects on reactivity, such as that of the axial burnup distribution, that are specific to the plant involved. Burnup credit applications make use of the specific plant operating experience accumulated in the past, relying on information that has been gathered from fuel burnt in previous cycles. As those higher enrichments and burnups have never been used before at that plant, there is no plant data to use to evaluate these effects. The only way out from a regulatory perspective is to include a penalty in the analysis to account for this lack of knowledge, and periodically verify that the fuel behaviour under the new conditions is in agreement with the assumptions made.

5. DETERMINATION OF THE FUEL DISCHARGE BURNUP

Burnup credit analyses submittals for spent PWR fuel storage have been licensed without the need for a direct fuel burnup measurement. The fuel discharge burnup value used to verify compliance with the loading curve is determined based on the reactor records. Some uncertainties, such as that of the reactor power calculation, have been included in the plant procedures used to perform this verification.

The coherence of the criticality analysis with the plant procedures and practices has to be emphasised. Core following methods do differ widely from plant to plant, and are still rather crud in some cases. While BWR have traditionally performed a more detailed core operation follow, only some PWR plants have adopted on-line monitoring systems of a similar quality. As a consequence, the discharge burnup calculation uncertainty is plant specific in general terms, and it is better known for BWR plants. It has to be recognised that a high uncertainty in the discharge burnup determination may invalidate a burnup credit analysis if not taken into account properly (either in the analysis or in the plant procedure), or if the reactivity margins available are not wide enough.

Given the variety of plant specific uncertainties, this problem is more relevant for the generic criticality safety analysis of spent fuel casks, which needs to be valid for fuel coming from different plants. Efforts to determine bounding burnup calculation uncertainties, at least for groups of similar plants, are needed, such as that started by EPRI at the USA.

6. BURNUP CREDIT METHODOLOGIES

Many burnup credit applications were licensed early in the 1980-decade. As already discussed above, changes in the operating conditions may affect the continued validity of the analysis. Any departure from the operation practices and core conditions maintained in the past should be evaluated to determine if it has such an impact. These changes are sometimes difficult to identify, because of several reasons:

1. They might be introduced gradually along the cycles, instead of being a step change,
2. From the plant perspective, they are not considered a change, but only a better fuel utilisation within the current licensed limits. As an example, it is frequent to reach burnup levels that are higher than those included in the reactor records, but still within the approved limits,

3. Some changes are not identified as affecting the criticality safety analysis, because they in fact do not affect any other plant analysis. In the BWR case, strategies such as spectral shift at the end of the cycle or feedwater temperature reduction during the cycle are usually covered in the reload safety analysis, and are applied each cycle at the discretion of the plant operator.

Continued validity of the old analysis methodology originally used can be a more important issue in some cases. It is a fact that neither the regulations, nor the calculation tools and data libraries, and for that matter, the general knowledge and understanding of burnup credit insights, were at today's level in the early eighties. Examples can be forwarded of approved analyses that are not as conservative as we all thought they were, or even in some cases, of analyses that could be criticised from the technical point of view.

The root cause of this situation has to be tracked back to the kind of applications that were first approved. They all were storage applications, and those always have additional safety margins. In the case of PWR, because there is Boron in the pool water, and its reactivity reduction effect was not credited at the time. In the BWR case, because it's not full burnup credit, only credit to the burnable absorber effect and depletion. The existence of those additional margins was taken into account in the licensing process.

As a consequence, a detailed development of burnup credit analysis techniques was neither performed nor required, and some sensitivities and second order effects were not really considered, because it was felt that they were duly covered by the existing margins, although it was not always explicitly stated. In the PWR case, the situation is equivalent to a taking a non-quantified credit for Boron, and something similar can be said for BWR. It has to be concluded that full burnup credit (actinides and fission products) has never been accepted on its own basis.

The problem begins when trying to apply burnup credit techniques to systems where those margins do not exist, as in the case of spent fuel transport casks for spent PWR fuel (no diluted Boron available). That is the reason why, nearly 20 years after of the original storage submittals, the international community is devoting efforts to adequately describe and quantify effects and parameters relevant to burnup credit. Those effects are of course already present in the approved applications, but they were never addressed in a detailed and correct manner. The contradiction becomes evident. The possibility of the outcome of the current efforts showing a lack of conservatism of the approved storage analysis cannot be discarded, although little safety impact should be expected.

Caution should be exercised to avoid the danger of this situation repeating itself. As an example, some actinide-only credit methodologies are covering the impact on reactivity of some second order effects claiming that the fission products provide an ample safety margin which is not credited in the analysis. As before, this is like taking a non-quantified credit for fission products, while avoiding a detailed treatment of the minor effects involved. Therefore, additional work would again be needed when trying to implement full burnup credit in the future. It could be avoided if a realistic ("best estimate") full burnup credit reactivity was taken as a basis, determining the uncertainties and covering them conservatively in the licensing submittals. These penalties could be reduced in the future as burnup credit knowledge increases and calculation tools evolve.

REGULATORY STATUS OF BURNUP CREDIT FOR DRY STORAGE AND TRANSPORT OF SPENT NUCLEAR FUEL IN THE UNITED STATES

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Abstract

During 1999, the Spent Fuel Project Office of the U.S. Nuclear Regulatory Commission (NRC) introduced technical guidance for allowing burnup credit in the criticality safety analysis of casks for transporting or storing spent fuel from pressurized water reactors. This paper presents the recommendations embodied by the current NRC guidance, discusses associated technical issues, and reviews information needs and industry priorities for expanding the scope and content of the guidance. Allowable analysis approaches for burnup credit must account for the fuel irradiation variables that affect spent fuel reactivity, including the axial and horizontal variation of burnup within fuel assemblies. Consistent with international transport regulations, the burnup of each fuel assembly must be verified by pre-loading measurements. The current guidance limits the credited burnup to no more than 40 GWd/MTU and the credited cooling time to five years, imposes a burnup offset for fuels with initial enrichments between 4 and 5 wt% 235U, does not include credit for fission products, and excludes burnup credit for damaged fuels and fuels that have used burnable absorbers. Burnup credit outside these limits may be considered when adequately supported by technical information beyond that reviewed to-date by the NRC staff. The guidance further recommends that residual subcritical margins from the neglect of fission products, and any other nuclides not credited in the licensing-basis analysis, be estimated for each cask design and compared against estimates of the maximum reactivity effects associated with remaining computational uncertainties and potentially nonconservative modeling assumptions. The NRC's Office of Nuclear Regulatory Research is conducting a research program to help develop the technical information needed for refining and expanding the evolving guidance. Cask vendors have announced plans to submit the first NRC license applications for burnup credit later this year.

1. INTRODUCTION

Historically, the NRC's approval of criticality safety evaluations for commercial spent fuel in casks has been based on analyzing the irradiated fuel as though it were unirradiated and without burnable poisons. This "fresh-fuel" assumption has provided a straightforward and bounding approach for showing that spent fuel packages will remain subcritical under analyzed normal and accident conditions with assumed water ingress. However, the extreme conservatism inherent in the fresh-fuel assumption can lead to excessive and costly design requirements for neutron absorbers and/or spacing of the spent fuel. The term burnup credit refers to allowing the criticality safety of spent fuel systems to be evaluated using analyses that consider the reduced reactivity of irradiated fuel. In commercial power-reactor fuels that have achieved most of their intended burnup, the major actinides (i.e., isotopes of uranium, plutonium, and americium) generally account for well over half of the change in reactivity relative to the fresh fuel assumption, with fission products accounting for most of the remainder.

In the U.S., interest in burnup credit for spent fuel casks has focused mainly on fuel from pressurized water reactors (PWRs) rather than from boiling water reactors (BWRs). The primary reason for this is that the smaller size and correspondingly lower reactivity of individual BWR assemblies, in relation to PWR assemblies, leads to relatively small economic penalties in cask design and capacity when analyzed under the fresh-fuel assumption. Burnup credit for PWR spent fuel, on the other hand, is expected to significantly increase the allowed capacity of large casks. For example, a rail cask design approved to hold

24 PWR fuel assemblies under the fresh-fuel assumption may eventually be modified to hold 32 assemblies when analyzed with burnup credit.

The NRC and the U.S. nuclear industry have been discussing the issues of applying burnup credit to single-purpose transport casks and dual-purpose storage-and-transport casks for over a decade. The U.S. Department of Energy (DOE) directed considerable resources toward the study of burnup credit and produced a topical report that proposed a method for crediting actinide burnup effects in the analysis of casks for PWR spent fuel [1]. During the 1995-98 time frame, the DOE topical report was revised twice [2, 3] in response to the NRC's review and comments.

Based in part on technical information provided in the DOE topical report, and supplemented by information available from other sources, the NRC's Spent Fuel Project Office (SFPO) issued in May 1999 the initial version of its interim staff guidance document, ISG-8 [4], which described an interim basis for allowing partial burnup credit in PWR spent fuel casks. At about the same time, the NRC's Office of Nuclear Regulatory Research initiated a burnup credit research program to support the staff's phased efforts in this area. In July 1999, early results from the NRC research program enabled SFPO's issuance of Revision 1 of ISG-8, which introduced the currently accepted recommendations for cask-specific approval of limited burnup credit for PWR fuel [5]. These recommendations were discussed at public meetings with stakeholders [6, 7] and have been recently incorporated into the updated standard review plans for spent fuel dry storage and transport [8, 9]. SFPO expects to issue further technical guidance on burnup credit as more information becomes available from ongoing research efforts and the application review process. Cask vendors have announced plans to submit the first NRC license applications for burnup credit later this year.

The following sections present the recommendations embodied by the current NRC guidance, discuss associated technical issues, and review information needs and industry priorities for expanding the scope and content of the evolving guidance.

2. RECOMMENDATIONS FOR BURNUP CREDIT IN PWR SPENT FUEL CASKS

The NRC technical guidance introduced in Revision 1 of ISG-8, and subsequently incorporated into the applicable standard review plans, provides recommendations under the following six headings: (1) Limits for the Licensing Basis, (2) Code Validation, (3) Licensing-Basis Model Assumptions, (4) Loading Curve, (5) Assigned Burnup Loading Value, and (6) Estimate of Additional Reactivity Margin. Except as specified in these recommendations, the application of burnup credit does not alter the current guidance and recommendations provided by the NRC staff for the criticality safety analysis of transport and storage casks. Each recommendation is cited below in italics and is followed by comments on associated technical issues.

1. *Limits for the Licensing Basis*: The licensing-basis analysis performed to demonstrate criticality safety should limit the amount of burnup credit to that available from actinide compositions associated with PWR irradiation of UO2 fuel to an assembly-average burnup value of 40 GWd/MTU or less. This licensing-basis analysis should assume an out-of-reactor cooling time of five years and should be restricted to intact assemblies that have not used burnable absorbers. The initial enrichment of the fuel assumed for the licensing-basis analysis should be no more than 4.0 wt% 235U unless a loading offset is

applied. The loading offset is defined as the minimum amount by which the assigned burnup loading value (see Recommendation 5) must exceed the burnup value used in the licensing safety basis analysis. The loading offset should be at least 1 GWd/MTU for every 0.1 wt% increase in initial enrichment above 4.0 wt%. In any case, the initial enrichment shall not exceed 5.0 wt%. For example, if the applicant performs a safety analysis that demonstrates an appropriate subcritical margin for 4.5 wt% fuel burned to the limit of 40 GWd/MTU, then the loading curve (see Recommendation 4) should be developed to ensure that the assigned burnup loading value is at least 45 GWd/MTU (i.e., a 5 GWd/MTU loading offset resulting from the 0.5 wt% excess enrichment over 4.0 wt%). Applicants requesting use of actinide compositions associated with fuel assemblies, burnup values, or cooling times outside these specifications, or applicants requesting a relaxation of the loading offset for initial enrichments between 4.0 and 5.0 wt%, should provide the measurement data and/or justify extrapolation techniques necessary to adequately extend the isotopic validation and quantify or bound the bias and uncertainty.

Comments

Credit for fission product effects is not included in the current guidance because of large uncertainties arising from the lack of readily available radiochemical assay data and measured reactivity data for neutron absorbing fission products. In addition, the neglect of fission products provides additional reactivity margin that is used in compensating for the remaining uncertainty and modeling issues in actinide-only burnup credit (see Recommendation 6). The restriction of credited burnup levels to no more than 40 GWd/MTU reflects the lack of readily available actinide assay data for fuels burned beyond that level. The loading offset permits limited burnup credit in the absence of actinide assay data from spent fuels with initial enrichments beyond 4 wt%. The loading offset is an example of how conservative modeling adjustments can be judiciously used to compensate for validation uncertainties that arise from moderate extrapolations beyond the measured data. NRC calculations show that applying the offset to fuel with an initial enrichment of 4.5 wt% and an assigned burnup of 45 GWd/MTU (i.e., credited as only 40 GWd/MTU) would typically correspond to a k-effective penalty on the order of 1.5%.

In justifying the loading offset approach, it is noted that, with other factors being equal, an increase in initial enrichment lowers the contribution from actinides to the reduced reactivity of spent fuel, thereby increasing the relative contribution from fission products. Thus, the neglect of fission products in actinide-only burnup credit is especially helpful in further offsetting the uncertainties from this limited extrapolation to initial enrichments above 4 wt%. The neglect of fission products is less helpful for extrapolating to burnups beyond 40 GWd/MTU because the actinide contribution to reducing the reactivity of irradiated fuel increases much more rapidly with burnup than does the contribution from fission products. Extrapolation of the isotopic validation to burnups beyond 40 GWd/MTU is further hindered by the expectation of greater computational challenges in modeling the increased neutronic heterogeneity of high-burnup fuel designs and core loadings.

The exclusion of fuels that have used burnable absorbers is due in part to the lack of readily available, comprehensive information on the range of burnable absorber designs and their past and present uses. It is well known that the use of burnable absorbers

generally leads to more fissile plutonium production per burnup increment. Extending burnup credit to include fuels that have used burnable absorbers will therefore necessitate a compilation of basic information on burnable absorber designs, supplemented by technical studies to establish appropriate modeling practices for the various design categories. Recommended modeling practices will need to incorporate added conservatism to account for the computational uncertainties arising from the present lack of isotopic validation data for fuels with burnable absorbers.

2. *Code Validation:* 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. Bias and uncertainties associated with the calculation of k-effective should be derived from benchmark experiments that represent important features of the cask design and spent fuel contents. The particular set of nuclides used to determine the k-effective value should be limited to that established in the validation process. The bias and uncertainties should be applied in a way that ensures conservatism in the licensing safety analysis. Particular consideration should be given to bias uncertainties arising from the lack of critical experiments that are highly prototypical of spent fuel in a cask.

Comments

The data presented to the NRC for validating actinide-only criticality calculations are based on a series of laboratory critical experiments with unirradiated fuel rods containing either low-enriched UO2 or PuO2 mixed with UO2 (MOX). These benchmark experiments differ from spent fuel in casks with regard to material compositions and geometries and the resulting neutronic competition among non-fuel components and the major actinides present in the fuel rods. The experiments do not represent the effects of axial fuel composition gradients and the typical local peaking of the neutron importance near the ends of spent fuel assemblies. Furthermore, the reactivity worth of poison plates or other absorber components is typically much lower in the benchmark experiments (e.g., *)k/k* < 0.04) than in a cask analysis (e.g.,*)k/k* > 0.20).

To the extent practical, important physical differences between the cask analysis and the validation benchmarks should be considered explicitly in deriving a conservative adjustment for the computational bias and uncertainty. The potential uncertainties associated with any remaining validation issues not explicitly factored into the applied bias adjustment should be estimated and evaluated against estimates of additional reactivity margins (see Recommendation 6). Additional isotopic and criticality validation data are of key importance to (a) extending burnup credit beyond 40 GWd/MTU, (b) reducing or removing the loading offset for initial fuel enrichments between 4 and 5 wt%, and (c) providing credit for fission products.

3. *Licensing-Basis Model Assumptions:* The applicant should ensure that the actinide compositions used in analyzing the licensing safety basis (as described in Recommendation 1) are calculated using fuel design and in-reactor operating parameters selected to provide conservative estimates of the k-effective value under cask conditions. The calculation of the k-effective value should be performed using cask models, appropriate analysis assumptions, and code inputs that allow adequate

representation of the physics. Of particular concern should be the need to account for the axial and horizontal variation of the burnup within a spent fuel assembly (e.g., the assumed axial burnup profiles), the need to consider the more reactive actinide compositions of fuels burned with fixed absorbers or with control rods fully or partly inserted, and the need for a k-effective model that accurately accounts for local reactivity effects at the less-burned axial ends of the fuel region.

Comments

In-core conditions during fuel depletion can strongly affect how the actinide composition changes with burnup. In particular, the production of fissile plutonium per burnup increment is enhanced by in-core conditions that harden the neutron energy spectrum seen by the fuel (e.g., high soluble boron concentration, use of solid absorbers, low moderator density) or that otherwise increase resonant neutron capture (i.e., high fuel temperature). The specific-power history further influences the actinide mix by affecting the competition between radioactive decay and neutron absorption in the intermediate transition actinides.

The in-core neutron energy spectrum can also be affected by cladding creep-down and hydrogen absorption into the cladding. Both of these phenomena lead to increased moderation and can therefore be safely neglected in the fuel depletion models used for actinide-only burnup credit. However, in modeling the isotopic validation benchmarks, these and related effects on in-core fuel geometry (e.g., thermal expansion, pellet swelling, cladding oxidation) may warrant further consideration in order to avoid the masking of any tendency to underpredict the fissile plutonium production. For cask criticality models, the increased moderation caused by such changes in the fuel rod can be safely approximated by assuming unirradiated fuel rod dimensions with water in the pellet-clad gap.

The first and second revisions of the DOE topical report have described a set of modeling assumptions that adequately bounds the effects of horizontal burnup gradients within spent fuel assemblies. Significant uncertainties remain, however, regarding the analysis of reactivity effects associated with axial burnup profiles, often referred to as "end effects." Some of the largest axial effects occur when the top of the fuel assembly is underburned as a result of partial insertion of control rods at power. In these cases, the increased k-effective is governed by two important phenomena: (1) the lower burnup at the top of the fuel, and (2) the increased production of fissile plutonium caused by the in-core spectral hardening effects of the control rods (i.e., thermal neutron absorption and moderator displacement). In many instances, the second phenomenon can be more important than the first. Studies reviewed to-date have considered only the first phenomenon, reduced local burnup, in the identification and recommended noding of bounding axial burnup profiles.

A closely related aspect of the axial profile issue concerns the use of part-length absorber rods for axial power shaping. Some of the axial burnup profiles in the data base evaluated for the DOE topical report featured strong local burnup depressions caused by the use of part-length absorbers near the fuel midplane. The neutron importance in fuels with this type of burnup profile and absorber-rod history is highest near the middle of the fuel rather than near the top. Such "saddle shaped" burnup profiles therefore represent a potential departure from the more widely studied end-effect profiles. To help

resolve the remaining issues and uncertainties of axial profile effects, additional analytical studies are needed on both types of profiles to correctly evaluate the combined absorber-rod effects of depressed local burnup and increased fissile plutonium production and to determine which, if any, of the two profile types is generally more reactive in representative cask designs. Further work is also needed to expand the data base of calculated or measured axial burnup profiles, with emphasis on the bounding shapes arising from the historical uses of full-length and part-length control rods.

It is noted that the higher k-effective of fuel burned in the presence of absorber rods is a strong function of initial fuel enrichment, with lower enrichments showing larger effects from the absorber-rod-induced spectral hardening. This is partially explained by noting that, with less 235U initially present, the depletion of 235U provides less offset for the increase in fissile plutonium production. The current data base of readily available isotopic assay benchmarks has only limited applicability for validating the computed effects of absorber rods on the actinide composition of spent fuel.

4. *Loading Curve:* The applicant should prepare one or more loading curves that plot, as a function of initial enrichment, the assigned burnup loading value above which fuel assemblies may be loaded in the cask. Loading curves should be established based on a 5-year cooling time and only fuel cooled at least five years should be loaded in a cask approved for burnup credit.

Comments

A burnup credit loading curve is derived from a series of k-effective calculations performed on a licensing-basis cask model. The resulting points on the loading curve give, for each value of initial enrichment, the fuel burnup value at which the computed k-effective equals the upper subcritical limit, i.e., where the bias-adjusted k-effective equals the recommended acceptance criterion of 0.95. Each calculation generally models a cask loaded with identical fuel assemblies (i.e., assemblies identical in design, initial enrichment, average burnup, assumed burnup profiles, and assumed in-core depletion parameters). To assess the effects of mixed fuel loadings, supplemental calculations may needed on cask models containing fuels from two or more points on the loading curve. Any increase in k-effective resulting from mixed loadings may necessitate an adjustment to the derived loading curve.

The 5-year cooling time has been chosen in large part because the burnup credit modeling studies in the U.S. have been based largely on fuel cooled for five years. Another consideration has been that the use of a single cooling time helps limit the added complexity of fuel loading specifications, which may be further governed by thermal and radiation shielding criteria. There is currently little need in the U.S. to load fuels cooled less than five years. It is well known that the reactivity of spent fuel decreases with time for all cooling times between 100 hours and 100 years. This effect is governed mainly by the decay of fissile 241Pu to nonfissile 241Am but, as discussed below, is reduced by the amplification of axial profile effects with cooling time. Basing the loading curves on a 5-year cooling time provides added conservatism for fuel with longer cooling times.

5. *Assigned Burnup Loading Value:* The applicant should describe administrative procedures that should be used by licensees to ensure that the cask will be loaded with

fuel that is within the specifications of the approved contents. The administrative procedures should include an assembly measurement that confirms the reactor record assembly burnup. The measurement technique may be calibrated to the reactor records for a representative set of assemblies. For an assembly reactor burnup record to be confirmed, the measurement should provide agreement within a 95 percent confidence interval based on the measurement uncertainty. The assembly burnup value to be used for loading acceptance (termed the assigned burnup loading value) should be the confirmed reactor record value as adjusted by reducing the record value by the combined uncertainties in the records and the measurement.

Comments

The NRC considered whether to accept the burnup values of record solely as determined by in-core physics calculations. However, reactor records have been known to contain errors and criticality safety is usually based on measured values rather than estimated values. In addition, the history of reported operating events in the fuel storage pools at reactors suggests that administrative and operational errors can be expected in the selection and handling of fuel assemblies for cask loading. Thus, it desirable to have some measurement check of the record burnup values. The recommended use of pre-loading measurements is consistent with requirements in the international transport regulations [10] and with the applicable guidance in NRC Regulatory Guide 3.71 [11].

The measurement strategy used here will provide protection against internal inconsistencies in the records data. Because of energy balance checks and the shuffling of fuel assemblies between burn cycles, the uncertainty in the absolute record values is expected to be small but potentially variable from plant to plant. Reducing the record value by the uncertainties in the records and measurements encourages the operators to improve their core calculation methods and employ high quality measurement techniques. Initially, the measurement of all fuel assemblies is planned. A sampling plan for the measurements may be justified after positive experience is gained with administrative controls, loading operations, and the quality of records data.

Burnup verification techniques may be based on gamma-ray measurements or a combination of gamma-ray and neutron measurements and may include axial scans. It is noted that passive neutron measurements are sensitive to the greatly increased production of 242Cm and 244Cm caused by the spectral hardening effects of absorber rods and, because of this sensitivity, may find eventual use in addressing the effects of absorber-rodded burnup histories. (See also related comments under Recommendations 3 and 6 and in the subsequent discussion of information needs.)

6. *Estimate of Additional Reactivity Margin:* The applicant should provide design-specific analyses that estimate the additional reactivity margins available from fission product and actinide nuclides not included in the licensing safety basis (as described in Recommendation 1). The analysis methods used for determining these estimated reactivity margins should be verified using available experimental data (e.g., isotopic assay data) and computational benchmarks that demonstrate the performance of the applicant's methods in comparison with independent methods and analyses. The Organization for Economic Cooperation and Development Nuclear Energy Agency's Working Group on Burnup Credit provides a source of computational benchmarks that may be considered. The design-specific margins should be evaluated over the full range

of initial enrichments and burnups on the burnup credit loading curve(s). The resulting estimated margins should then be assessed against estimates of: (a) any uncertainties not directly evaluated in the modeling or validation processes for actinide-only burnup credit (e.g., k-effective validation uncertainties caused by a lack of critical experiment benchmarks with either actinide compositions that match those in spent fuel or material geometries that represent the most reactive ends of spent fuel in casks); and (b) any potential nonconservatisms in the models for calculating the licensing-basis actinide inventories (e.g., any outlier assemblies with higher-than-modeled reactivity caused by the use of control rod insertion during burnup).

Comments

This recommendation arises from the NRC staff's efforts at addressing the following question: Can the combined effects of uncertainties and approximations in actinide-only burnup credit outweigh the margins from the neglect of fission products? Table 1 summarizes the results of DOE's and NRC's initial analyses toward answering this question [12]. At three places in the second revision of DOE's topical report (Sections 3.2, 4.1.5, and 4.2.3.3), a portion of the large reactivity margin arising from the neglect of fission products and 236U was used in attempting to bring closure to an issue. In response to requests from the NRC staff, the final revision of the topical report provided in its Table 7-4 a tally of the estimated effects of uncertainties in the proposed burnup credit methodology and how well they are offset by reactivity margins resulting from the neglect of fission products and 236U. Specifically, for selected values of initial enrichment and burnup, the DOE tabulation (included in Table 1) subtracted three reactivity allowances from the estimated fission-product margins. The three reactivity allowances were to account for (a) the unmodeled higher reactivity of fuel assemblies in which control rods were inserted during part of the burnup and the uncertainties associated with (b) criticality validation issues (i.e., physical differences between the benchmarks and cask analyses) and (c) computer code adequacy issues (e.g., source sampling and convergence). The DOE results showed a net residual margin for all evaluated combinations of initial enrichment and burnup.

NRC calculations on representative cask models have demonstrated that the estimated fission-product margins can vary substantially between cask designs. For example, higher poison loadings in the basket reduce the margins by capturing neutrons otherwise absorbed by fission products. Estimated fission-product margins can be further reduced by the effects of nonuniform burnup within fuel assemblies. All fuel-in-cask models analyzed by the NRC yielded calculated fission-product margins significantly smaller than those given in DOE's topical report, which were based on a poison-free pin-cell model. As shown in Table 1, subtracting the topical report's three reactivity allowances from the NRC-calculated margins for fission products and 236U leaves negative residual margins at certain values of low initial enrichment and low burnup. These results can be explained in part by noting that DOE's assumed reactivity allowances for the reactivity effects of burnup in the presence of control rods are greatest at low initial enrichments and constant beyond burnups of 15 GWD/MTU. It is possible, however, that such combinations of low burnup and low initial enrichment will fall below the burnup credit loading curve for a given cask design.

Table I. Results from NRC's Independent Analysis of Table 7-4 in DOE Topical Report, DOE/RW-0472 Rev.2, Tally of the Use of Fission-Product and 236U Margin for Addressing Uncertainties of Actinide-Only Burnup Credit.

Enrichment (wt% ^{235}U) and Burnup (GWD/MTU)		EFPM = Estimated Fission Product and ^{236}U Margin (%)Δk_{eff})		DOE's Reactivity Allowances for Uncertainty Issues and Approximations in Actinide-Only Burnup Credit (%)Δ_{eff})			Estimated Remaining Margin (%) Δk_{eff}) with EFPM from:	
		DOE TR Rev.2 (Pin Cell)	NRC Case A (OECD GBC-32 Cask)	Criticality Validation Issues	Effect if Control Rods were Inserted During Depletion	Computer Code Adequacy Issues	DOE TR Rev.2 (Pin Cell)	NRC Case A (OECD GBC-32 Cask)
3.0	15	8.4	4.4	2.0	3.3	1.0	2.1	**-1.9**
	30	13.0	5.9	2.0	3.3	1.0	6.7	**-0.4**
	45	16.0	6.9	2.0	3.3	1.0	9.7	0.6
3.6	15	8.2	4.3	2.0	2.1	1.0	3.1	**-0.8**
	30	12.8	5.6	2.0	2.1	1.0	7.7	0.5
	45	16.2	6.7	2.0	2.1	1.0	11.1	1.6
4.5	15	7.9	4.2	2.0	1.0	1.0	3.9	0.2
	30	12.4	5.6	2.0	1.0	1.0	8.4	1.6
	45	16.1	6.5	2.0	1.0	1.0	12.1	2.5

In response to NRC questions, section 7.4 of the final DOE topical report discussed several smaller margins, in addition to those resulting from the neglect of fission products and 236U, that are associated with apparent modeling conservatisms in the proposed actinide-only methodology for burnup credit. Such additional margins would generally tend to offset some or all of the negative residual margins in Table 1. However, most of the additional margins are based on comparisons against the typical or mean case and therefore do not cover the full range of possible or credible fuel loadings that would be allowed under the proposed burnup credit methods. The NRC staff therefore concludes that it is not possible, based on information considered to-date, to ensure categorically that the aggregate of such additional margins is large enough to offset actinide-only uncertainties in all cask designs. The staff expects that further insights into the existence and magnitude of residual margins will emerge from NRC research efforts and the application review process.

3. INFORMATION NEEDS AND INDUSTRY PRIORITIES FOR EXTENDED BURNUP CREDIT

Industry stakeholders have expressed interests in (1) applying burnup credit to PWR fuels that have been exposed to burnable absorbers, (2) crediting cooling times beyond five years, (3) crediting burnups beyond 40 GWd/MTU, (4) reducing the burnup offset penalty for fuels with initial enrichments between 4 and 5 wt% 235U, (5) seeking credit for fission products, and (6) establishing limited burnup credit for BWR spent fuel in casks. The NRC staff has in turn requested industry assistance in acquiring the technical information needed for developing NRC technical review guidance addressing each of these areas [6, 7]. These areas are the focus of ongoing and planned activities within the NRC's burnup credit research program [13, 14]. Observations on relevant technical issues and information needs are provided below.

Current NRC research activities include analytical studies toward establishing guidance on acceptable methods and modeling assumptions for computing the effects of burnable poisons on spent fuel isotopics. Consistent with industry priorities, the initial emphasis has been on two early design categories of removable burnable poison rods. However, progress to-date has been limited by difficulties in gathering comprehensive information on the configurations and uses of these and other categories of burnable absorber designs. As more design information becomes available, the analytical studies will be expanded to address the remaining categories of removable, fixed, and integral burnable poison designs. In general, the evolving NRC guidance will seek to identify appropriately conservative modeling assumptions to compensate for any uncertainties associated with incomplete design documentation and for shortages of isotopic validation data pertaining to burnable absorbers.

Research studies on the crediting of cooling times beyond five years are also in progress. An important phenomenon in this context is that, as cooling time increases, the computed k-effective of a loaded cask becomes more sensitive to variations in the assumed axial burnup profiles. Accordingly, the NRC's ongoing analytical studies seek to address (1) how the amplification of the axial profile effects (e.g., end effects) with cooling time slows the net decrease in k-effective and (2) whether and how the existence of stronger axial effects at cooling times beyond five years may necessitate reassessment of the bounding axial burnup profiles and the axial noding schemes used in modeling them. It is noted that the importance of axial-effect uncertainty grows with cooling time in proportion to the axial effect itself.

From the preceding comments on the estimation of additional reactivity margin (see Recommendation 6 and Table 1), it is clear that future credit for fission product effects will be limited by the uncertainties and potential nonconservatisms remaining in the analysis of actinide effects. In particular, fission product credit will necessitate a direct accounting for the potentially strong effects that absorber-rodded burnup histories can have on the reactivity of PWR spent fuel assemblies. It has been noted that at-power insertion of full-length or part-length control rods has seen only limited practice in the recent operating histories of U.S. PWRs. For example, present-day reactor operations generally restrict at-power control rod insertions to the "bite position," a position near the top of the active fuel that may vary from plant to plant and cycle to cycle. However, because the NRC licenses cask designs to accept the spent fuel from many or all plants, the safety analyses for casks must account for the worst-case rodded-burnup histories in the worst-case cycles at the worst-case plants. The NRC staff has therefore solicited industry assistance in compiling and summarizing comprehensive information on worst-case rodded burnup histories from all past and present operations at U.S. PWRs [7, 12].

The NRC research program is now engaging in international collaborations to acquire the additional isotopic and criticality validation data needed for extending burnup credit beyond 40 GWd/MTU, for reducing or eliminating the loading offset for initial enrichments between 4 and 5 wt% 235U, and for adding credit for fission products. These and related NRC research efforts are described in another paper presented at this meeting [15]. The experimental data emerging from these international efforts will help in further reducing uncertainties within the current and evolving guidance limits for PWR burnup credit in casks and will find eventual use in establishing limited burnup credit in casks for BWR spent fuel.

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DEPLETION AND CRITICALITY CALCULATION AND CODE VALIDATION

REBUS: A BURNUP CREDIT EXPERIMENTAL PROGRAMME

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Abstract

An international programme called REBUS (REactivity tests for a direct evaluation of the Burn-Up credit on Selected irradiated LWR fuel bundles) for the investigation of the burn-up credit has been initiated by the Belgian Nuclear Research Center SCK•CEN and Belgonucléaire. At present it is sponsored by USNRC, EdF from France and VGB, representing German nuclear utilities. The programme aims to establish a neutronic benchmark for reactor physics codes. This benchmark would qualify the codes to perform calculations of the burn-up credit. The benchmark exercise will investigate the following fuel types with associated burn-up. 1. Reference absorber test bundle, 2. Fresh commercial PWR UO_2 fuel, 3. Irradiated commercial PWR UO_2 fuel (50 GWd/tM), 4. Fresh PWR UO_2 fuel, 5. Irradiated PWR UO_2 fuel (30 GWd/tM). Reactivity effects will be measured in the critical facility VENUS. The accumulated burn-up of all rods will be measured non-destructively by gamma-spectrometry. Some rods will be analyzed destructively with respect to accumulated burn-up, actinides content and TOP-18 fission products (i.e. those non-gaseous fission products that have most implications on the reactivity). The experimental implementation of the programme will start in 2000.

1. INTRODUCTION

Present criticality safety calculations of irradiated fuel still have to model the fuel as fresh fuel, since no precise experimental confirmation exists of the decrease of reactivity due to accumulated burn-up. The fact that this so-called "burn-up credit" cannot be taken into account has serious economical implications for transport and storage of irradiated fuel. In this paper the REBUS programme will be described, that aims to establish an experimental benchmark for validating reactorphysics codes for burn-up credit issues.

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 burn-up.

The reactivity effect of PWR UO2 spent fuel will be measured in the VENUS critical facility for two different burn-up values, 30 and 50 GWd/tM. Other fuel types can be investigated in future extensions of the programme, like BWR fuel and MOX fuel.

Measuring the reactivity effect is not sufficient for a good experimental benchmark. The spent fuel used in the experiment should be very well-characterized. The characterization of the fuel is performed in two steps:

The first step is non-destructive and is performed before the reactivity measurements are performed. 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 137Cs content and in this way the burn-up of the rod. Via the gross gamma-scans this will give a good picture of the burn-up of all rods. This first step is necessary to verify that the selected rods have a similar burn-up.

The second step is destructive and aims at determining both the actinides content, some burn-up indicators and the most important fission products with respect to neutron absorption. A sample of the same rod, on which gamma-spectrometry has been performed, will be investigated by radiochemistry in order to obtain an accurate and precise measurement of the actinides, burn-up indicators and fission products content.

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. (see figure 1).

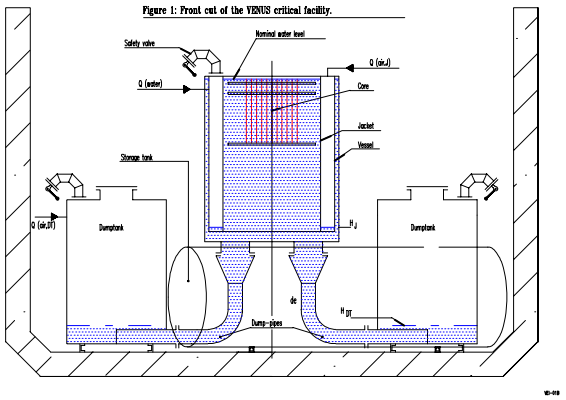

Figure 1. Vertical cross-section of the VENUS critical facility.

Parameters that are measured with the VENUS reactor are the critical water level hc, the reactivity coefficient δρ/δh, 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 βeff.

3.1. The critical water level

The critical water level is measured in order to determine the critical mass of the loaded configuration. Due to adjustments of the outer feeding zone, the critical level can be arranged to a limited extent. Recent configurations had a critical height of about 40 to 45 cm. For the REBUS programme the VENUS core will be adjusted to fuel with a maximum length of 1 m. The critical level is measured with a random uncertainty of 0.02 cm and a systematic uncertainty of about 0.07 cm.

The comparison of the same or similar configurations will only be affected by the random uncertainty. This is in particular important when reactivity effects of small changes of the configuration are measured.

With help of the critical level measurements also changes in reactivity can be measured. These changes can be due to e.g. Am effect, burn-up, changes in moderator density, replacements of absorber rods (Gd, B4C, AgInCd), etc.

3.2. Reactivity effect of the water level

The reactivity coefficient $\delta\rho/\delta h$ is measured to make a link between the uncertainty of the critical level measurement and the uncertainty of the reactivity. Normally a value for $\delta\rho/\delta h$ is measured of about 0.30 to 0.35 %/cm for a core with 40-45 cm height. This value will decrease in case the critical level increases. Given the random uncertainty of the critical water level of 0.02 cm, the uncertainty of the reactivity is about 0.007% or 7 pcm. In case the systematic uncertainty plays a role, too, the uncertainty of the reactivity is about 24.5 pcm.

The reactivity coefficient is determined by the measurement of two successive periods, created by making the reactor supercritical with an increment of the water level. It is calculated on basis of the Nordheim equation with the measured periods T together with some calculated parameters like the delayed neutron fraction βeff, the prompt neutron lifetime l, the relative fraction of each delayed neutron group ai and the corresponding decay constant λi.

The uncertainty on the value of the reactivity coefficient due to the experimental uncertainties (period, water level increment) is estimated to be between 3 and 5%.

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

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

1. Reference absorber test bundle,
2. Fresh commercial PWR UO_2 fuel, constructed by SIEMENS,
3. Irradiated commercial PWR UO_2 fuel (50 GWd/tM), originating from Neckarwestheim NPP, Germany,
4. Fresh PWR UO_2 fuel, original BR3 fuel,
5. Irradiated PWR UO_2 fuel (30 GWd/tM), BR3 reactor, Belgium.

All test bundles will be loaded as a 7x7 fuel assembly in the center of a 4% enriched UO_2 fuel driver zone. The 7x7 assembly is chosen because the VENUS reactor has removable grids where this assembly fits in. The outer row of the assembly will always contain fresh fuel rods, so the real test bundle will be 5x5 fuel rods.

The fresh BR3 UO_2 fuel bundle (figure 2a) consists of 21 fuel rods (a 5x5 assembly minus the four corner rods). The reason is simply that no more rods were available. The fuel is 5% enriched UO_2 fuel, fabricated at FBFC, Belgium, and provided by Belgonucléaire. It is the original composition of the irradiated BR3 fuel.

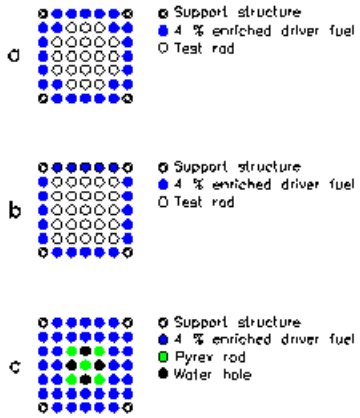

Figure 2. Fuel bundle design for the REBUS PWR UO_2 case.

The irradiated PWR UO_2 fuel of intermediate burn-up 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 2a).

The fresh commercial fuel bundle (figure 2b) consists of a 5x5 fuel assembly. The fuel is 3.8% enriched UO_2 fuel, fabricated at 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.

The reference absorber test bundle (figure 2c) consists partly of the same 4% enriched UO_2 fuel as there are in the driver zone and partly of so-called "pyrex" rods, these are glass tubes containing boron. The glass tubes are inserted in a zircaloy cladding. The number of absorber rods is chosen in order to have the same order of reactivity effect as with the introduction of the spent fuel assembly with a burn-up of 50 GWd/tM (about 1500 pcm, or 1.5%).

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 at the main axes will be measured. Due to the impossibility of measuring this parameter in the spent fuel assembly, Co wire activation measurements will be performed in this assembly to measure the thermal and epithermal neutron flux.

5. REFABRICATION AND MEASUREMENTS PERFORMED AT HOT CELL LABORATORY

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 have to be assembled in the experimental 7x7 bundle. This is also performed at the hot cell laboratory.

5.1. Gross gamma-scan

A gross gamma-scan is performed by moving the fuel rod slowly in front of the collimator of a high-purity Ge gamma-detector. The total number of counts of the detector for subsequent fixed time periods is stored in 4096 channels of an ADC. In this way a readout is obtained of the total gamma emission rate in function of the axial position on the rod. This so-called gross gamma-scan is a good measure of the gamma activity distribution in the fuel and the distribution of 137Cs, since most other gamma-emitting fission products have already decayed after a long cooling time.

5.2. Burn-up measurement by gamma-spectrometry

The burn-up is determined by gamma-spectrometry by measuring the long-living fission product 137Cs.
The method consists in performing comparative measurements of the 137Cs content of the fuel to be investigated and some well-known reference fuel samples (the so-called burnothèque or SENA-samples) and a well-calibrated Eu-source [1, 2].

The results of the comparative measurements have to be corrected for the different history and geometry conditions of the burnothèque/Eu-source and the examined fuel rod. These corrections include:

1. Self-absorption in the fuel,
2. Absorption by materials between fuel and detector,
3. Decay of 137Cs during the irradiation,
4. Decay of 137Cs during cooling time,
5. Fission yield,
6. Energy released per fission.

The last two corrections are important if the measurement concerns different fissile isotopes.
The uncertainty of the burn-up value is approximately 4 to 5% (1ó), mainly due to the uncertainties of the burn-up of the reference samples and the determination of the peak area of the 137Cs gamma peaks.

6. MEASUREMENTS OF ACTINIDES, BURN-UP INDICATORS AND FISSION PRODUCTS AT RADIOCHEMISTRY LAB

6.1. Main solution processing

The element content and isotopic composition are generally obtained by Thermal Ionisation Mass Spectrometry (TIMS) coupled to the Isotopic Dilution (ID) technique.

This technique allows to determine the concentration of an element by measuring the change in the ratio of (at least) two isotopes of the element of interest, induced by the addition of a spike, i.e. a known amount of the same element with a different isotope ratio of the same isotopes, to a weighed aliquot of a sample. With the known isotopic composition of the sample, the spike and blend, and the known amounts of sample and spike, the concentration of the element of interest can be calculated. Since only isotope ratios in the different materials are measured, its follows that sample treatment or recoveries of chemical separations need not to be quantitative once isotopic equilibration after spiking has been achieved. Also, variability of chemical separation recovery is generally not important.

Before performing MS, numerous separations are needed in order to obtain "clean" solutions containing the relevant elements only and from which interfering isobars have been removed.

After a Redox cycle, U and Pu are first separated on a Dowex column from the other elements (fission products (FP) + Rare earth elements (REE) + Am + Cm). Ce is then separated.

Elements not measurable by TIMS, or measured in complement to TIMS, are determined by alpha-spectrometry (238Pu, Cm), gamma-spectrometry (Np, 241Am, Cm, 129I) and scintillation (Prn). These measurements allow also to verify the absence of interference in TIMS measurements (e.g. absence of 242Cm (measured by alpha-spectrometry) when measuring 242mAm by TIMS).

Low concentration elements like 99Tc (base programme) and 232U, 95Mo, 102Ru, 103Rh and 109Ag (TOP 25) are analyzed by ICP-MS technique.

6.2. Residue solution processing

The elements contained (partly) in the residue are Mo, Tc, Ru, Rh, Ag and Sb.

They are separated and measured by ICP-MS technique, except for Ru-106 and Sb-125, for which gamma-spectrometry is used.

6.3. Thermal Ionisation Mass Spectrometry

The Thermal Ionisation Mass Spectrometry (TIMS) is the reference technique for the determination of the isotopic composition of an element. For a TIMS analysis, first a µg amount of sample is evaporated on a source filament. Mounted in the source housing of the instrument, this filament is electrically heated under vacuum. The sample evaporates and electrons generated by another filament ionise the atoms. The ions are extracted by an extraction lens and a collimator focuses the ion beam to the entrance of the MS. The MS consists of an electric sector (to filter the ion energies) and a magnetic sector (to separate different mass-to-charge ratios). At the end of the MS, the separated ions are detected by Faraday cups or a Daly detector.

To avoid possible isobaric interference during measurement, the elements of interest are separated from interfering elements prior to analysis.

On a regular basis, certified isotopic standards are measured, allowing control of isotopic fractionation and mass bias corrections.

7. 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 k_{eff}), it is considered necessary to create an experimental reactivity difference of about 1500 pcm.

Preliminary calculations showed for the pyrex configuration a difference in reactivity of 1806 pcm with respect to a configuration where the pyrex rods are replaced by water holes.

Similar calculations showed a reactivity difference of 2135 pcm for the high burn-up commercial fuel with respect to fresh fuel. No estimate has been done so far with respect to the intermediate burn-up case.

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 25 pcm (the reactivity effect of the present core is about 350 pcm/cm for a critical level of 45 cm). The random uncertainty is smaller, about 7 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 a 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 $\Delta\rho$ is shown in the following equation:

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

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

When due to the large difference in critical height the values of $\partial\rho/\partial h$ will be too different for the two configurations, in principle an integration should be used for the determination of the total reactivity effect $\Delta\rho$:

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

This integration can be approximated by measuring eventually an intermediate configuration with a critical level in between the two measured configurations. This can be performed by adaptation of the driver zone.

The uncertainty related to the measurement of the reactivity effect $\Delta\rho$ 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 $\partial\rho/\partial h$ in function of the critical height.

The uncertainty of the non-destructive ^{137}Cs burn-up measurement is estimated to be 4 to 5%.

The uncertainties of the determination of the actinides, the burn-up indicators and the fission products by radiochemical analysis are shown in the next tables:

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%				

Uncertainties by radiochemical analysis for the different burn-up 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%

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 burn-up 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 reactorphysics codes with respect to burn-up credit of PWR UO_2 fuel with intermediate and high burn-up 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 burn-up 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 MOX fuel.

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STUDY OF MULTIPLICATION FACTOR SENSITIVITY TO THE SPREAD OF WWER SPENT FUEL ISOTOPICS CALCULATED BY DIFFERENT CODES

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Abstract

As a sensitivity study the impact on the system reactivity was studied in the case that different calculational methodologies of spent fuel isotopic concentrations were used for WWER spent fuel inventory computations. The sets of isotopic concentrations obtained by calculations with different codes and libraries as a result of the CB2 international benchmark focused on WWER-440 burnup credit were used to show the spread of the calculated spent fuel system reactivity. Using the MCNP 4B code and changing the isotopics input data, the multiplication factor of an infinite array of the WWER-440 fuel pin cells was calculated. The evaluation of the results shows the sensitivity of the calculated reactivity to different calculational methodologies used for the spent fuel inventory computation. In the studied cases of the CB2 benchmark, the spread of the reference k-results relative to the mean was found less or about ±1% in spite of the fact that the data of isotopic concentrations were spread much more.

1. INTRODUCTION

According to the burnup credit methodology, two calculational steps must be performed because of the criticality safety analysis of spent fuel systems.

Spent fuel consists of many nuclides resulting from reactions in progress during the fuel irradiation in nuclear reactors. Therefore, the first phase of any spent fuel system analysis should be finding the spent fuel inventory relating to the irradiation history of the specified fuel. In spite of the fact that the spent fuel contains more than 200 isotopes, analysts have agreed on a relatively small number of isotopes which are the most valuable contributors to the reactivity of spent fuel systems. The chosen set of the recommended isotopes (actinides and fission products) was proved to be a conservative one. The description of the spent fuel using concentrations only of the selected isotopes makes the criticality analysis of the spent fuel systems less time consuming.

Due to spent fuel system licensing, the maximum k_{eff} must be evaluated taking into account all uncertainties and the methodology bias and compared with the k regulatory safety limit. Connected with this, an impact on the system reactivity was studied in the case that different calculational methodologies of spent fuel isotopic concentrations were used for the spent fuel inventory computation. Following the sensitivity study which has recently been performed for PWRs by members of OECD/NEA/NSC Burnup Credit Criticality Benchmark Working Group [1], a similar study has just been made also for WWERs. Up to twelve sets of isotopic concentrations obtained by calculations with different codes and libraries as a result of an international benchmark focused on WWER-440 burnup credit [2] were used to show the spread of the calculated spent fuel system reactivity. Using the MCNP 4B code and changing the isotopics input data, the multiplication factors of an infinite array of the WWER-440 fuel pin cells were calculated. The evaluation of the results shows the sensitivity of the calculated reactivity to different calculational methodologies used for the spent fuel inventory computation.

2. SENSITIVITY STUDY

The four-piece international benchmark focused on burnup credit issues was specified for WWER spent fuel in 1996 [2] in collaboration with OECD/NEA/NSC Burnup Credit Criticality Benchmark Working Group. The benchmark was designed and proposed to be calculated by analyst in Eastern and Central European countries and possibly members of the Burnup Credit Criticality Benchmark Working Group interested in the WWER applications. The second portion of the benchmark, CB2 [3], consisted of the nuclide concentration computation for depletion in a simple WWER-440 pin cell. Fresh fuel of 3.6 wt % ^{235}U enrichment was supposed to be burnt to 30 and 40 GWd/t_U and cooled 0, 1 and 5 years after discharge. Since no supporting data obtained from sample measurements were available, the benchmark evaluation resulted in the methodology intercomparison only.

As the CB2 results, 12, 10, 11 and 10 sets of 26 nuclide concentrations [4] were obtained from the benchmark participants for the cases of 30 GWd/t_U and 1 year of cooling time, 30 GWd/t_U and 5 years of cooling time, 40 GWd/t_U and 1 year of cooling time, 40 GWd/t_U and 5 years of cooling time, respectively. Each set of the nuclide concentrations resulted from a calculation using a calculational methodology specific to the given participant. The selected nuclides consist of 11 actinides and 15 major fission products, main contributors to the spent fuel reactivity [3] recommended for the burnup credit calculations: U-235, 236, 238, Pu-238, 239, 240, 241, 242, Am-241, 243, Np-237, 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.

The CB2 evaluation [4] showed the spread of nuclide concentrations calculated by the participants. If the estimated standard deviation of 10% was chosen as a point of a change between ‘good’ and ‘poor’ agreement in the calculations of the nuclide concentration for a given isotope by the participants, there were several more spread isotopes : Am-241 (11%), Am-243 (12%), Ag-109 (13%), Sm-149 (12%), Sm-151 (15%) and even Gd-155 (52%). It should be noted that a similar spread resulted from the evaluated results of the similar OECD benchmark (Pu-238 (14.78%), Am-243 (10.71%), Ag-109 (10.62%), Sm-149 (14.92%), Sm-151 (22.15%) and Gd-155 (33.23%)) [1].

Each set of the CB2 results were used as input data for a reference criticality calculation by MCNP 4B + DLC189. Thus, in total, 12, 10, 11 and 10 sets of multiplication factors of the infinite array of the WWER-440 fuel pin cells were computed. It was believed that the k-results should be normally distributed for each separate case of a given burnup and cooling time if the calculational methodology was good and used in a right manner. To confirm or reject this hypothesis, all the sets were statistically explored and tested (α=0.05). The statistical evaluations found two outliers (see Fig. 1, 3, 5, 7 below). One marked calculational methodology (the Cuban participant) was the ORIGEN II code with PWRU libraries [5], which is an obsolete approach not using any flux and library recalculation, the other one (the Finnish participant) used the SCALE 4.3 code [6], but, unfortunately, some specific features of the calculational sequence SAS2 were not fully utilized so the correct result could not be obtained. It can be seen in the pictures, that SCALE 4.3 was used by three other participant successfully.

After removing outliers the sets of k-results were explored once again and found normally distributed. As figures 2, 4, 6 and 8 show, the spread of the k-results relative to the mean is a little more than ± 0.5 % for both cases with the burnup of 30 GWd/t_U (30_1 and 30_5 cases) and about ± 1 % for 40 GWd/t_U and 5 years of cooling time (40_5 case). In the case of 40

GWd/t_U and 1 year of cooling time (40_1 case), an outlier was not found at the chosen confidence level. The spread of the relative deviations in this case - except of the 'potential' outlier (which was found in all the other cases) - is less than ± 1 %.

3. CONCLUSIONS

In the field of WWER spent fuel applications, there is a total lack of open experimental data which could help to find out the methodology bias for the isotopic concentration calculations. The bias should correct nuclide concentration data prior to the criticality calculation of a given spent fuel facility. The only publicly mentioned experimental data coming from Russian assays in seventies ([7], [8]) involve only actinides (except ^{237}Np) and do not yield any results for fission products which would be important for the validation of the burnup credit calculational methodologies. The detailed data for operational codes were published only in 1998 [9] to offer the measured isotopics eventually for an open recalculation and comparison. The proposed calculational case was entitled as a benchmark but rather it is a large calculational exercise relating to the whole reactor core and needs several codes to be involved to deal with the detailed operational history.

For benchmarking methodologies used for the isotopic concentration calculations due to the spent fuel applications, it would be more useful to take a benchmark preparation after [1]. The calculational benchmark [1] was specified in context with existing chemical assay measurements, however, in contrast to the reality of the sample depletion, the problem specification was simplified somewhat to provide an approximate representation of the fuel configuration during the depletion. The simplification would allow a more straightforward and consistent comparison between the codes used by various benchmark participants.

An urgent need of benchmarks for WWER spent fuel isotopic concentration supported by 'well documented' samples persists. Measured data only would serve as a quantitative measure of a code ability to predict the spent fuel composition.

For the time being, only-calculational benchmarks can only show spread of results which - projected into a spread of the multiplication factor - yields a certain relative piece of information about a selected methodology. In the studied cases of the CB2 benchmark it resulted in finding two outlying methodologies (and, of course, reasons for it) and the spread of the reference k-results relative to the mean less or about ±1% beside the estimated standard deviations less or about 1%. At the same time, the spread of the nuclide concentrations data described by the estimated standard deviation was of the order of one and ten % (see above).

FIG.1. Multiplication factors resulted from the MCNP calculation for WWER fuel of 3.6% enrichment, 30MWd/kg burnup, 1y cooling tome

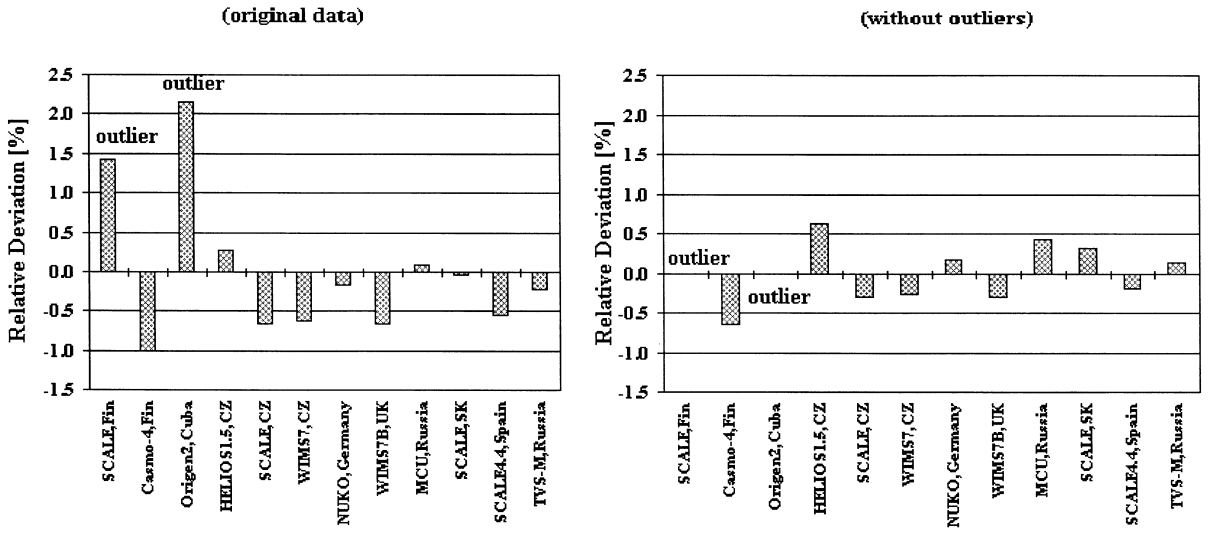

FIG.2. 30-1 k Results Relative to the Mean

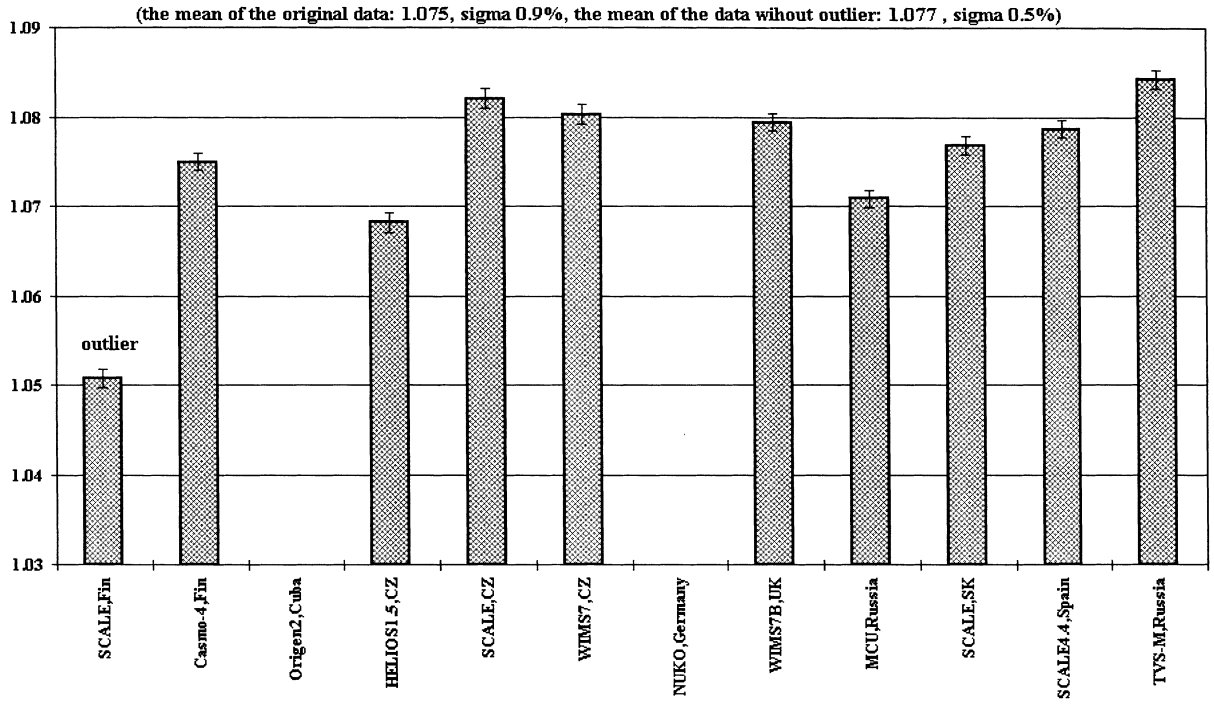

FIG.3. Multiplication factors resulted from the MCNP calculation for WWER fuel of 3.6% enrichment, 30MWd/kg burnup, 5y cooling time

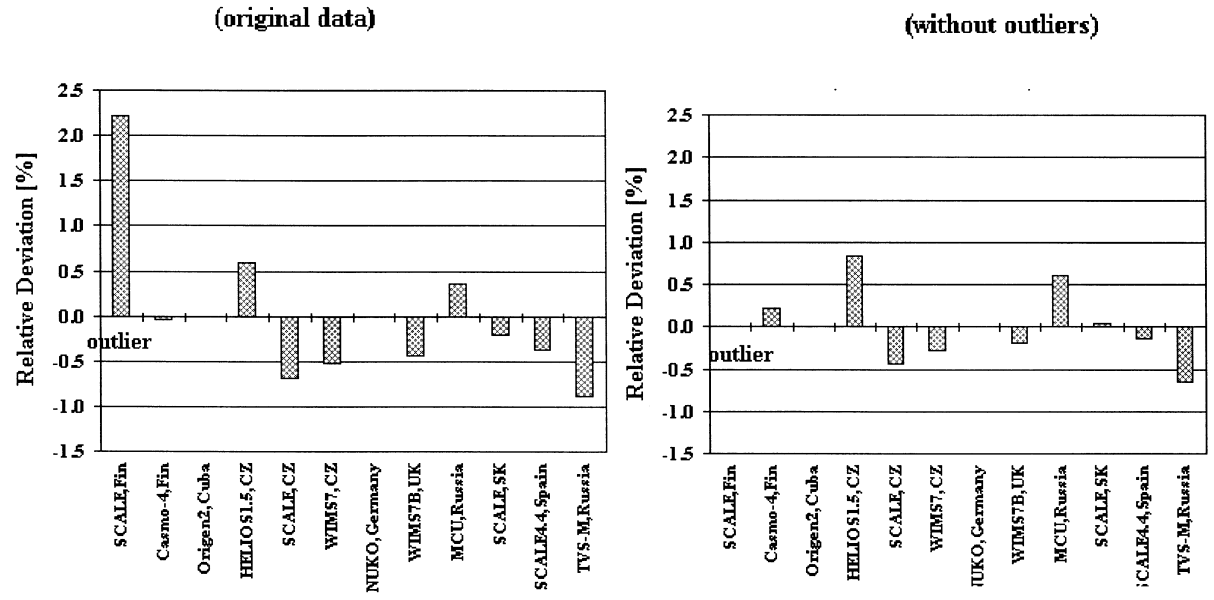

FIG.4. 30-5 k Results Relative to the Mean

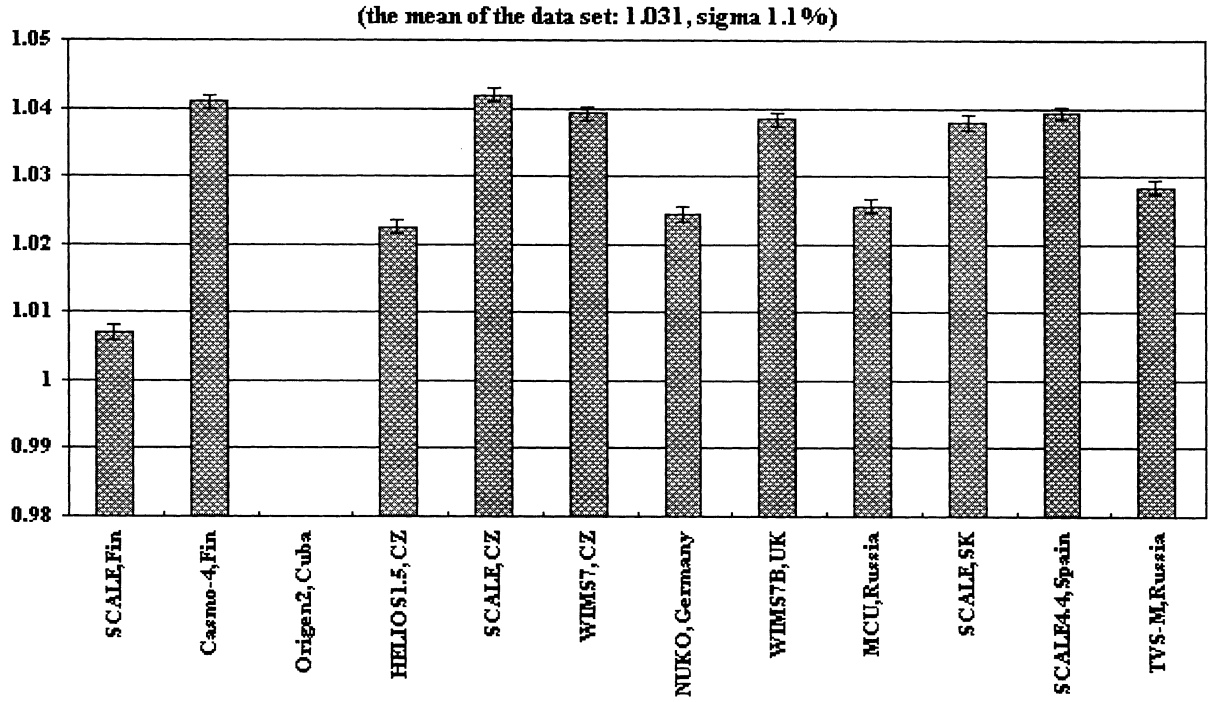

FIG.5. Multiplication factors resulted from the MCNP calculation for WWER fuel of 3.6% enrichment, 40MWd/kg burnup, 1y cooling time

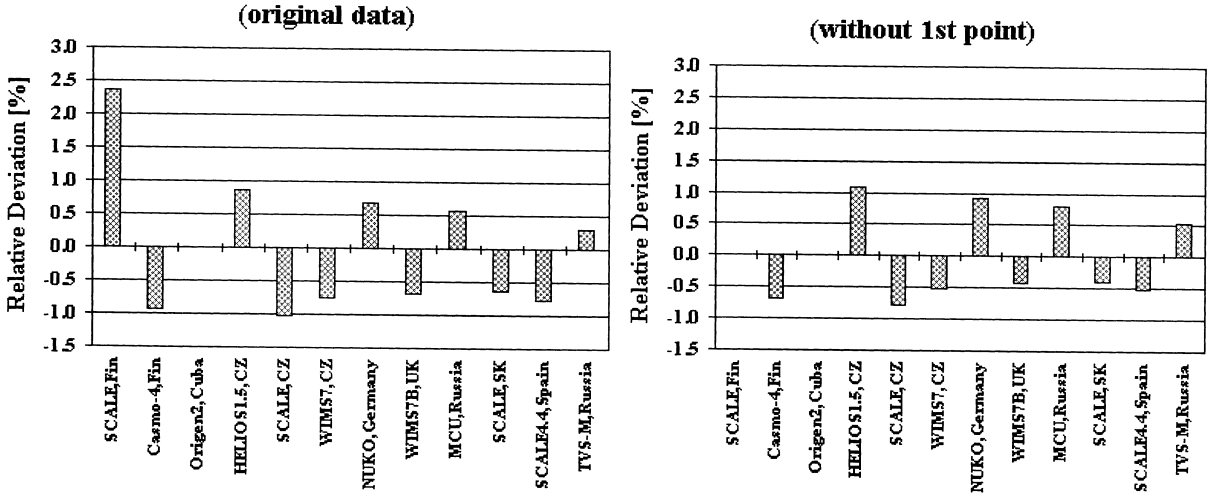

FIG.6. 40-1 k Results Relative to the Mean

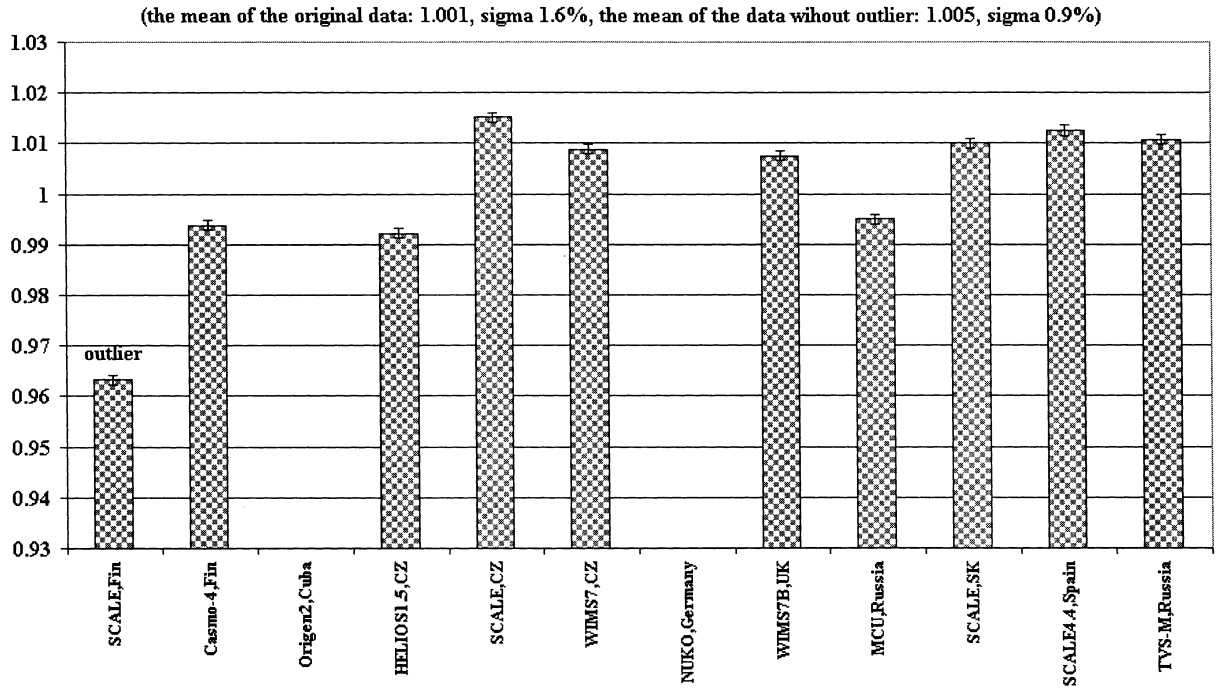

FIG.7. Multiplication factors resulted from the MCNP calculation for WWER fuel of 3.6% enrichment, 40 MW d/kg U burnup, 5 yr. Cooling time.

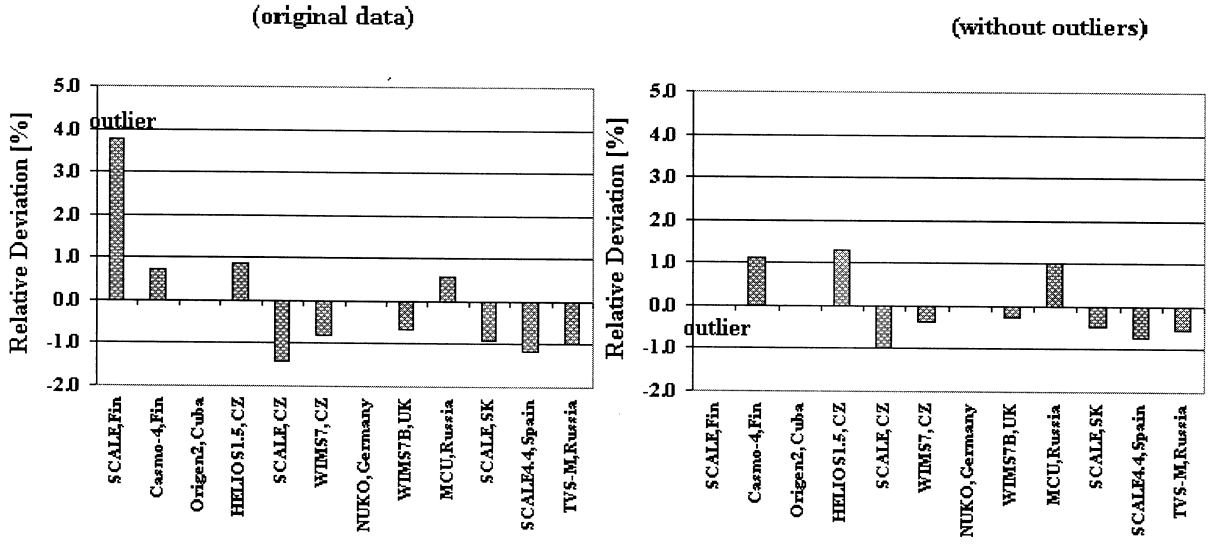

FIG.8. 40-5 k results relative to the mean.

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KENOREST — A NEW COUPLED CODE SYSTEM BASED ON KENO AND OREST FOR CRITICALITY AND BURNUP INVENTORY CALCULATIONS

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Abstract

The program system KENOREST version 1998 will be presented, which is a useful tool for burnup and reactivity calculations for LWR fuel. The three-dimensional Monte Carlo code KENO-V.a is coupled with the one-dimensional GRS burnup program system OREST-98. The objective is to achieve a better modelling of plutonium and actinide build-up or burnout for advanced heterogeneous fuel assembly designs. Further objectives are directed to reliable calculations of the pin power distributions and of reactor safety parameters including axial and radial rod temperatures for fuel assemblies of modern design. The stand-alone-code KENO-V.a version is used without any changes in the program source. The OREST-98 system was developed to handle multirod problems and additional burnup dependent moderator conditions which can be applied to stretch-out simulations in the reactor. A new interface module RESPEFF between KENO and OREST transforms the 2-d or 3-d KENO flux results to the one-dimensional lattice code OREST in a fully automated manner to maintain reaction rate balance between the codes. First results for assembly multiplication factors, isotope inventories are compared with OECD results.

1. INTRODUCTION

Reliable prediction of the characteristics of irradiated light water reactor fuels is needed for many aspects of the nuclear fuel cycle. In burnup credit investigations, the data of final uranium, produced plutonium contents and the fission product build-up should be known in the axial and radial distribution inside a fuel assembly to account for criticality codes. If criticality aspects and the inventory distributions are strongly coupled together in three dimensions, two main problems must be solved: the simulation of all isotopic nuclear reactions in the fuel assembly and the simulation of more-dimensional neutron fluxes setting the reactions in motion. In state-of-the-art computer techniques, a combination of specialised codes for reactivity search, for cross section generation and for burnup calculations is preferred to solve these cross-linked problems.

2. KENOREST-98

In the OREST system of 1983 [1] the HAMMER lattice code [2] (Savanna River Laboratory, 1967) used for cross section generation and the depletion code ORIGEN [3] (Oak Ridge National Laboratory, 1973) are directly coupled to an one-dimensional burnup system. Now the KENO-V.a code [4] (ORNL, 1983) and OREST were directly linked in a time step approximation to a new three-dimensional reactivity and inventory system KENOREST-98 [5]. A simplified diagram of program loops and of data transfers is shown in Figure 1. The cross section libraries of KENO and of OREST, based on JEF-2.2 [6], coincide for the ‘infinite dilution’ starting point and the energy group structure, to avoid possible inconsistencies between different libraries. Resonance shielding and cross section calculations are treated by the Flux Equivalent Cell method (FEC) at each local position of the rods. The method comprises the standard Nordheim resonance treatment and a search for effective one-dimensional cell moderator conditions. This step is performed fully automatically in a separate interface module.

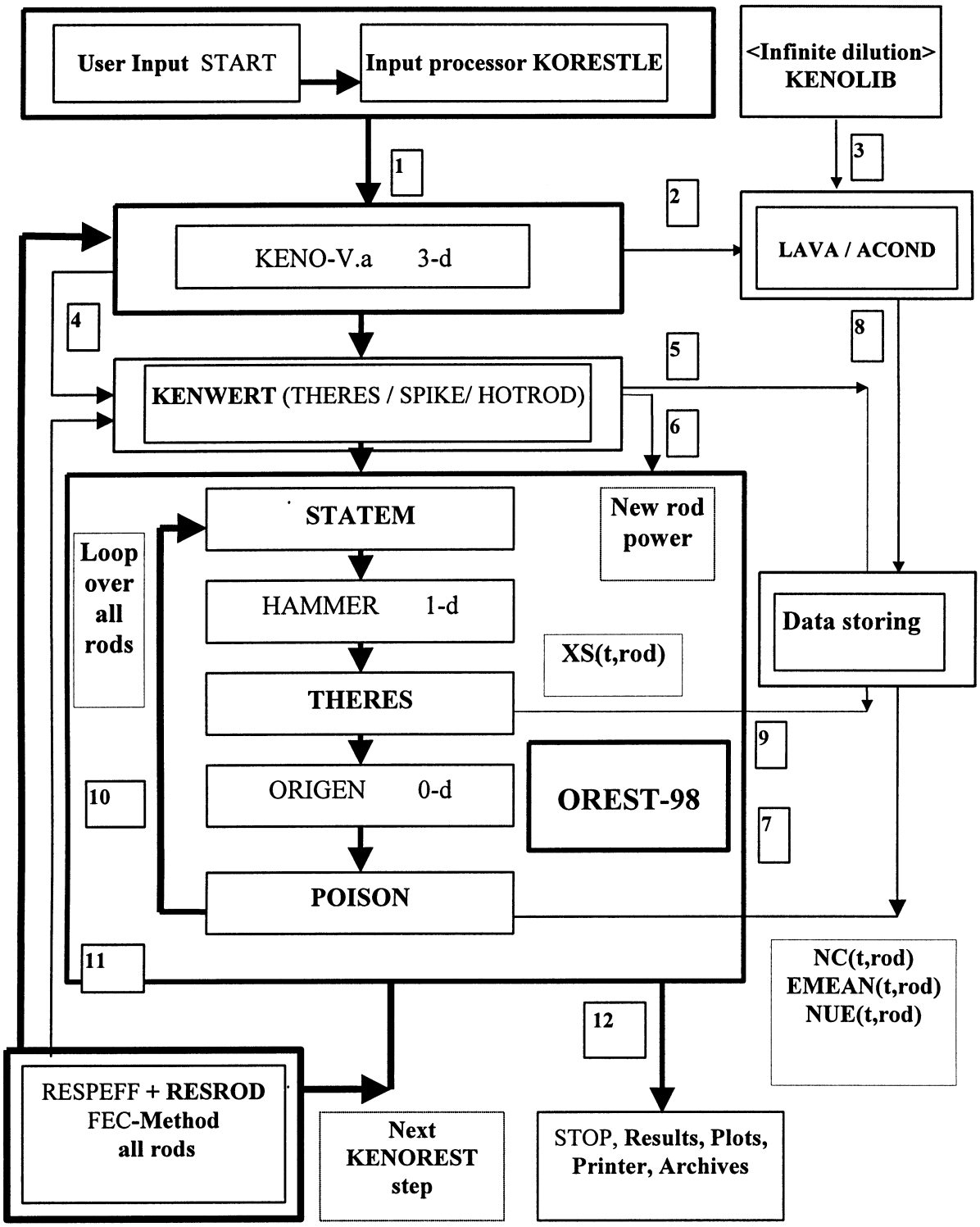

FIG. 1. KENOREST Program Loops (thick lines) and Data Transfers (thin lines).

2.1. Coupling Features

The neutron-physical coupling of KENO and OREST is obtained by transferring the three-dimensional KENO results (neutron fluxes and fission densities, line 4) for each fuel rod position to the one-dimensional HAMMER code (fuel lattice simulation) and to the zero-dimensional ORIGEN depletion code by the basic relation of reaction rate balance. After a short OREST burnup step, the cross sections and inventories (lines 7, 8, 9) for up to 1000 local positions are reloaded to KENO (line 2). The coupling is done using well defined data files during the calculation loops.

The stand-alone-code KENO-V.a is used without any changes in the program source. So the coupling between KENO and the remainder code system is simply done by the KENO printer channel and the burnup dependent KENO cross section library which had been generated by OREST in the previous burnup time step. The KENO output, multiplication factors, fluxes and neutron generation rates for all rods are used by the module KENWERT (Figure 1). The neutron generation rates are divided by <NUE(t,rod)> to get KENO fission densities and a fission density distribution is calculated in three dimensions. These data are multiplied by <EMEAN(t,rod)> and the user defined assembly power to get the rod power distributions in three dimensions. The new rod powers (line 6) are used to calculate the next new rod temperatures (STATEM) and the next burnup step (ORIGEN) for each rod is performed. A more detailed explanation of the coupling model is given in Reference [14].

2.2. The FEC Method

The transformation of the three-dimensional KENO results to the one-dimensional OREST system including resonance shielding and cross section calculation, is treated by our Flux Equivalent Cell (FEC) method. This method, described in detail in Ref. [7], includes a search for the effective pitch or for the effective steam content for the one-dimensional lattice cell calculation and solves the problem of searching the neutron physically equivalent infinite lattice in OREST for a certain single rod in its environment. This procedure, done for each rod and for each KENOREST loop, is necessary to achieve reaction rate balance between KENO and OREST. With this calculation tool a treatment of neutron fluxes and reactions for both simple and complex fuel rod lattices is possible. This calculation method runs fully automated as an interface module RESPEFF in KENOREST and is repeated at each rod position for each KENO start-up and burnup loop.

2.3. Improved Burnup Calculations with OREST-98

After several start-up iterations of KENOREST, the irradiation history, defined by the user, will be started following the program loops as shown in Figure 1. During burnup calculation the lattice cell calculation is performed by the HAMMER code, using refined resonance parameters [10] for the Nordheim resonance treatment at each burnup step. The burnout of the fissionable nuclides, the build-up of the fission products and the depletion of the gadolinium isotopes including all decay and transmutation processes are calculated by the ORIGEN code with stepwise feedback to HAMMER and KENO. Special user input options are directed to handle single gadolinium rods.

For the burnup calculations in KENOREST all build-up and decay chains of the ORIGEN code with ENDF/B-5 updated decay libraries for 800 structural isotopes, 144 actinides and over 800 fission product isotopes according to the OREST-96 version are used [11]. The

ORIGEN cross section data for the most important isotopes are replaced by updated 83 group results from HAMMER calculations at each OREST step. The remainder isotopes are handled with the known ORIGEN three group procedure using problem dependent spectral indices. All capture and fission cross sections of all isotopes are sampled by the module POISON with feedback to HAMMER and KENO.

Based on the released version of OREST-96 [11], an improved burnup system OREST-98 was developed and installed in the coupled KENO-OREST system. Additionally to the time dependent input options for specific fuel power and the boron poisoning inside the standard version, the new OREST-98 code has an extended simulation range of reactor operation parameters:

1. time dependent input data of coolant pressure are taken into account (moderator density effect during burnup, essential for reactor safety aspects),
2. time dependent input data of coolant temperature (moderator density and temperature effects during burnup, essential for stretch-out operations and for reactor safety aspects),
3. time dependent input data of steam fractions (moderator density effects during burnup, essential for BWR reactors, for reactor safety aspects and for the FEC method),
4. time dependent input data of lattice cell specification (moderator ratio variations during burnup).

Furthermore OREST-98 can handle multiple burnup problems from one serial input file, used for all rods. With these additional features inside OREST-98 a burnup dependent Wigner cell can be defined according to the FEC results, which allows an improved calculation of multirod problems, of stretch-out operations and in generally the evaluation of the neutron fluxes and reaction rates for complex fuel rod lattice configurations in their environment, better than standard pincell calculations with constant properties can do it. OREST-98 is also available as a stand-alone-module.

2.4. Nuclear Data Libraries

The development of a new KENO library started with a 292 group library based on JEF-2.2 [6], which was collapsed to the HAMMER 83 group structure. At each new KENOREST application this library is available in the ANISN format at the problem independent <infinite dilution> condition where no resonance shielding or Doppler broadening is included. During the iterations of KENOREST this library is overwritten by problem dependent resonance shielded and Doppler broadened OREST data for each rod position. Additionally we developed a set of new HAMMER (THERMOS-HAMLET) libraries called <99STANDARD> starting from the libraries of [11] based on [10]. The updating procedure of these regions was completed by data of JEF-2.2 [6] in such a manner, that <99STANDARD> corresponds in the ‘infinite dilution’ condition for fission and capture cross sections and the values for the number of neutrons per fission with the KENO library KORLIB-V2. The THERMOS upscattering cross section data were extended for finer temperature steps to allow a better description of stretch-out operations, of moderator temperature coefficients and of other reactivity and burnup effects. The application range is now extended from room temperature up to hot reactor conditions in constant steps of 10 K.

3. INVENTORY CALCULATIONS

The reliable prediction of important actinide and fission product concentrations is shown in comparison with the III-B burnup credit benchmark [12], performed by the OECD working party on nuclear criticality safety (WPNCS) (additional inventory calculations with OREST-98 had been shown in [9]). In this benchmark important actinides (U234, U235, U236, U238, Np237, Pu238, Pu239, Pu240, Pu241, Am241 and Am243) and fission products (Mo95, Tc99, Ru101, Rh103, Ag109, Cs133, Sm147 up to 152, Nd143, Nd145, Eu153, Eu155, Gd155 up to 158 and Xe131) are compared. The inventories had been calculated by 14 participants for three void conditions of a BWR fuel assembly. The results are preliminary because the final report is not yet issued.

The Table I. shows averaged results from the participants for the 40 % void case (col. 2), compared with GRS results (col. 3). The deviations from <OECD> are given in column 4. The one sigma distribution of the <OECD> results are listed in column 5. Some extreme diverging solutions in Ref. [12] were omitted for this comparison to get the relatively precise data of Table I. So the scattering of the results in column 5 became quite small, with exceptions of Eu155 (56 % deviation of KENOREST, 38 % scattering of OECD results) and Gd155 (59 % deviation, 32 % scattering). These only remaining problems for KENOREST-98 could be analysed as a problem of the large uncertainty in the libraries for the capture resonance integral of the fission product Europium-155 [13] in the same mass decay chain of Gadolinium-155. These deviations should probably be eliminated by using JEF2.2-data for that isotope [9]. On the other hand relatively great mean deviations of 5 % up to 10 % were found between the participants for all plutonium isotopes. In KENOREST-98 the Pu240 concentrations are underestimated and the higher plutonium isotopes are overestimated.

With KENOREST three-dimensional inventory calculation are possible with respect to the finite length of the fuel assemblies, the axial variation of the coolant temperature and the steam contents in axial core direction. An axial control rod model is in preparation. Axial profile calculations are important for burnup credit with high burnt fuel, were the assumption of a flat axial burnup may be not conservative.

At each burnup step the KENO Monte-Carlo code calculates the assembly reactivity, the three-dimensional neutron fluxes and the three-dimensional fission rates, which are transformed into 3-d power density distributions for the following burnup steps in OREST. Up to 20 different axial sections of a lattice grid of a maximum of 99 x 99 fuel rods can be calculated at one run. Each fuel rod portion is handled as a separate fuel-mixture with its own burnup power history and with its own cross section sets, which are generated by OREST and fed back into the KENO code. So it is possible to achieve pin by pin and in the axial direction a three dimensional build-up of actinide inventories and fission products. A good simulation of ‘end-effects’ and corner positions of fuel rods is important for the resonance shielding procedures and for the build-up of plutonium.

After the KENOREST calculation all inventories are stored and can be used for other purposes, e.g. for additional decay steps or criticality calculations. Additionally all data are available for realistic decay heat predictions and for neutron and gamma source terms for three dimensional shielding calculations.

Table I: Preliminary Inventory Results BWRUO2 Phase 3b [12], 40 % void, 5 years cooling time, Outliers excluded.

Nuclide	<OECD> preliminary	Kenorest-98 preliminary	Deviation % Kenorest-98	Mean Deviation OECD %
U-234	4.68E-06	4.54E-06	-3	+/- 4
U-235	1.85E-04	1.90E-04	3	+/- 4
U-236	1.16E-04	1.13E-04	-2	+/- 4
U-238	2.11E-02	2.11E-02	0	+/- 0
PU-238	4.21E-06	4.68E-06	**11**	+/- 7
PU-239	1.04E-04	1.11E-04	7	+/- **11**
PU-240	5.43E-05	5.08E-05	-6	+/- 5
PU-241	2.05E-05	2.31E-05	**13**	+/- 8
PU-242	1.29E-05	1.45E-05	**12**	+/- 6
AM-241	6.57E-06	7.27E-06	**11**	+/- 6
AM-243	2.36E-06	2.66E-06	**13**	+/- **11**
NP-237	1.13E-05	1.18E-05	4	+/- **19**
MO-95	5.15E-05	5.20E-05	1	+/- 3
TC-99	5.22E-05	5.29E-05	1	+/- 5
RU-101	5.00E-05	4.90E-05	-2	+/- 4
RH-103	2.72E-05	2.92E-05	7	+/- 6
AG-109	4.53E-06	5.08E-06	**12**	+/- 7
CS-133	5.48E-05	5.49E-05	0	+/- 2
SM-147	9.58E-06	9.29E-06	-3	+/- 3
SM-149	9.82E-08	1.07E-07	9	+/- 6
SM-150	1.20E-05	1.20E-05	0	+/- 6
SM-151	4.04E-07	4.23E-07	5	+/- 8
SM-152	5.22E-06	5.41E-06	4	+/- 8
ND-143	3.43E-05	3.44E-05	1	+/- 3
ND-145	3.01E-05	2.99E-05	-1	+/- 2
EU-153	4.69E-06	4.99E-06	6	+/- 6
EU-155	1.74E-07	2.71E-07	**56**	+/- **38**
GD-155	1.85E-07	2.95E-07	**59**	+/- **32**
GD-156	6.61E-05	7.43E-05	**12**	+/- **18**
GD-157	1.49E-08	1.84E-08	**23**	+/- **28**
GD-158	7.89E-05	8.63E-05	9	+/- **17**
XE-131	2.07E-05	2.03E-05	-2	+/- 4

Until now only a few experimental data of 3-d isotopic analysis are available, which can be recalculated for validation and verification of the 3-d-option of the code system. Those calculations for the italian TRINO reactor and the german OBRIGHEIM reactor are in work.

4. CONCLUSIONS

The code system KENOREST-98 is available for first applications. The combination of KENO and OREST works in a neutron physically satisfying way for infinite fuel lattices and for heterogeneous fuel assemblies with different fuel types, gadolinium contents and moderator conditions. Further works and developments on a future KENOREST-2000 system will be directed to a speed-up of the data transfers and to a further refinement of the libraries. Additionally it could be of interest to extend the application of KENOREST for other reactor systems like WWER or heavy water moderated and graphite moderated reactor systems. For such systems OREST is already prepared.

ACKNOWLEDGEMENT

The program development of KENOREST is supported by the Federal Minister of Education, Science, Research and Technology (BMBF) of Germany.

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SIEMENS PWR BURNUP CREDIT CRITICALITY ANALYSIS METHODOLOGY: DEPLETION CODE AND VERIFICATION METHODS

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Abstract

Application of burnup credit requires knowledge of the reactivity state of the irradiated fuel for which burnup credit is taken. The isotopic inventory of the irradiated fuel has to be calculated, therefore, by means of depletion codes. Siemens performs depletion calculations for PWR fuel burnup credit applications with the aid of the code package SAV. This code package is based on the first principles approach, i.e., avoids cycle or reactor specific fitting or adjustment parameters. This approach requires a general and comprehensive qualification of SAV by comparing experimental with calculational results. In the paper on hand the attention is focused mainly on the evaluation of chemical assay data received from different experimental programmes.

1. INTRODUCTION

Siemens performs depletion calculations for PWR fuel burnup credit applications with the aid of its core design package SAV [1-3]. SAV, an acronym for ”Standard-Auslegungs-Verfahren” (standard design procedure) includes the depletion codes FASER [4], CASMO [5-7], and KORIGEN [8]. The codes FASER and CASMO are applied to determine the isotopic densities at the time of shutdown. To get cooling time dependent isotopic densities the depletion code KORIGEN is used with the cross-sections generated for the main actinides by FASER or CASMO.

SAV is based on broad and comprehensive verification and validation. The experience with the version SAV90 and the preceding version SAV79A has been accumulated to about 300 first core and reload designs including Siemens, Framatome and Westinghouse PWR plants with power levels ranging from 357 MWe to 1430 MWe [1-3]. In 1993 Siemens started with the development of a new version called SAV95 [3]. Except for the code FASER all advanced components of SAV95 have been developed on the basis of the proven methodology applied in SAV90. In SAV95 the code FASER is replaced by CASMO which is a multi-group two-dimensional transport theory code for burnup calculations on BWR and PWR fuel assemblies [5-7].

2. DEPLETION CODE VERIFICATION

2.1. Overview of the Verification Methods Applied

SAV is based on the first principles approach, i.e., avoids cycle or reactor specific fitting or adjustment parameters. This approach requires, accordingly, a general and comprehensive qualification of SAV by comparing experimental with calculational results. With the experimental data currently received from 22 PWR plants (14 Siemens and 8 non-Siemens

plants) and added annually to the data pool the quality of the prediction of the SAV system relies on broad and growing statistics on the differences between measurement and calculation.

The experimental data are obtained from:

1. startup and core follow measurements (detector signal distributions, boron let down, control rod worth),
2. special measurement programmes (e.g., reactivity coefficients, xenon transients) as well as,
3. post irradiation examinations, in particular chemical assay of the isotopic inventory of irradiated fuel.

Figure 1. shows as an example the frequency distribution of the differences between predicted and measured critical boron concentrations at hot full power conditions at BOC, MOC, and EOC for 207 cycles calculated with SAV90.

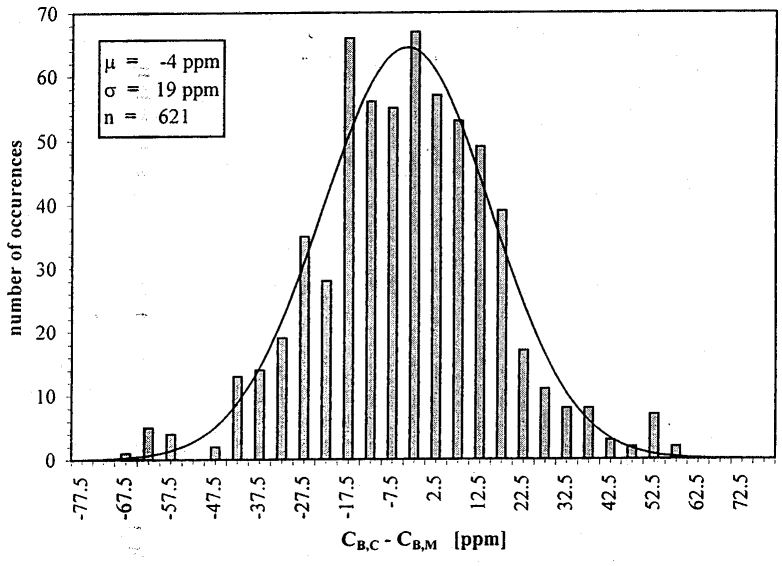


FIG. 1. Hot Full Power Critical Boron Concentrations at BOC, MOC and EOC: Histogram Showing the Differences of the Concentrations $C_{B,C}$ Predicted by SAV90 and the Concentrations $C_{B,M}$ Measured

In addition to the observation of the standard data measured at startup and periodically during each cycle, pre- and recalculation of special core measurement campaigns are an important tool for the validation of the code system. An impressive example for the prediction quality of SAV for this type of data is the confirmation of predicted values through experimental determination of the EOC reactivity balance performed at Nuclear Power Plant Neckarwestheim II (NPP GKN II) two years ago. The main objective of this experiment was to measure the critical boron concentrations (cold zero power, xenon free reactor) for the unrodded core as well as for the condition net-bank inserted (all rods in minus stuck rod) in order to get - in addition to [1] – information about the accuracy of SAV in the prediction of the EOC shutdown reactivity balance. Comparison of the values pre-calculated with SAV90 to the measurement results shows good agreement for the absolute critical boron concentration level and excellent agreement in the boron equivalents associated with temperature decrease and net-bank insertion.

2.2. Evaluation of Chemical Assay Data

In burnup credit applications to spent fuel management systems the isotopic inventory of the irradiated fuel is required as necessary and direct input to the criticality analysis of these systems. Therefore, as regards depletion code validation with respect to burnup credit applications, the attention is mainly focused on comparisons between measured and calculated isotopic concentrations.

2.2.1.Actinides and Burnup Indicators

The SAV system is verified against numerous chemical assay data. Most of these data refer, however, to actinides and burnup indicators (Nd-148 mostly). Figures 2 through 10 show a few examples obtained from assays of UO_2, reprocessed UO_2, MOX, and reprocessed MOX fuel assemblies. There are lots of more verification results including evaluations of assays on burnable absorber bearing fuel rods as well as evaluations of assays on nuclide concentrations as a function of the pellet radius [1]. All these results show:

1. that the measured and calculated isotopic ratios or concentrations are mostly in agreement within the experimental error bounds,
2. that SAV does not underestimate the concentrations of the fissile isotopes, and
3. that SAV tends to slightly underestimate the fertile actinides U-236, Pu-242 and sometimes also Pu-240.

An underestimation of the fertile isotopes U-236, Pu-242 and Pu-240 in particular results in an overestimation of the neutron multiplication factor of a thermal system. Usually the systems which require burnup credit are thermal systems.

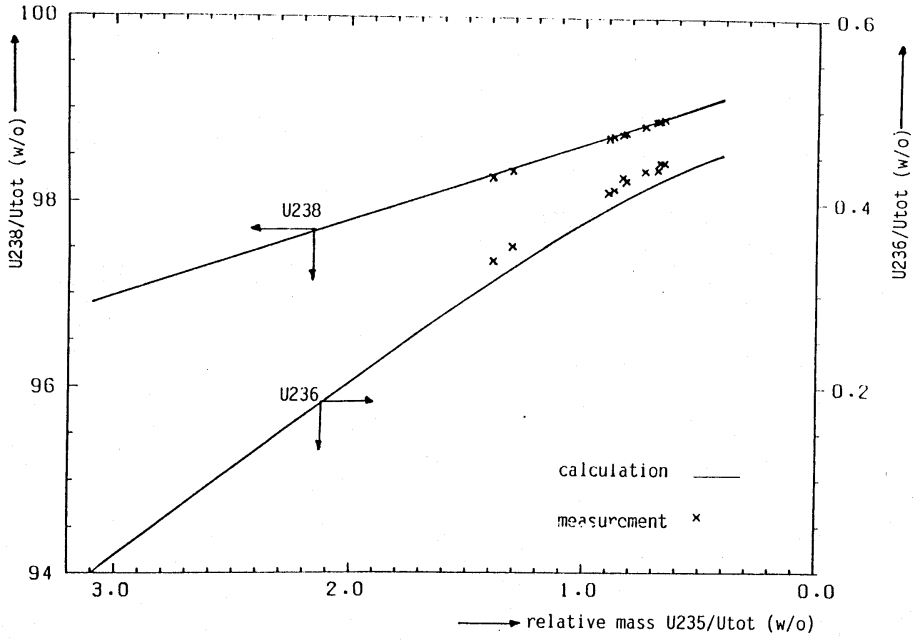


FIG. 2. Fuel Nuclide Concentrations in UO_2 Rods: Verification of the Siemens Standard Design Procedure SAV: Comparison of Measured and Calculated Isotopic Fractions.

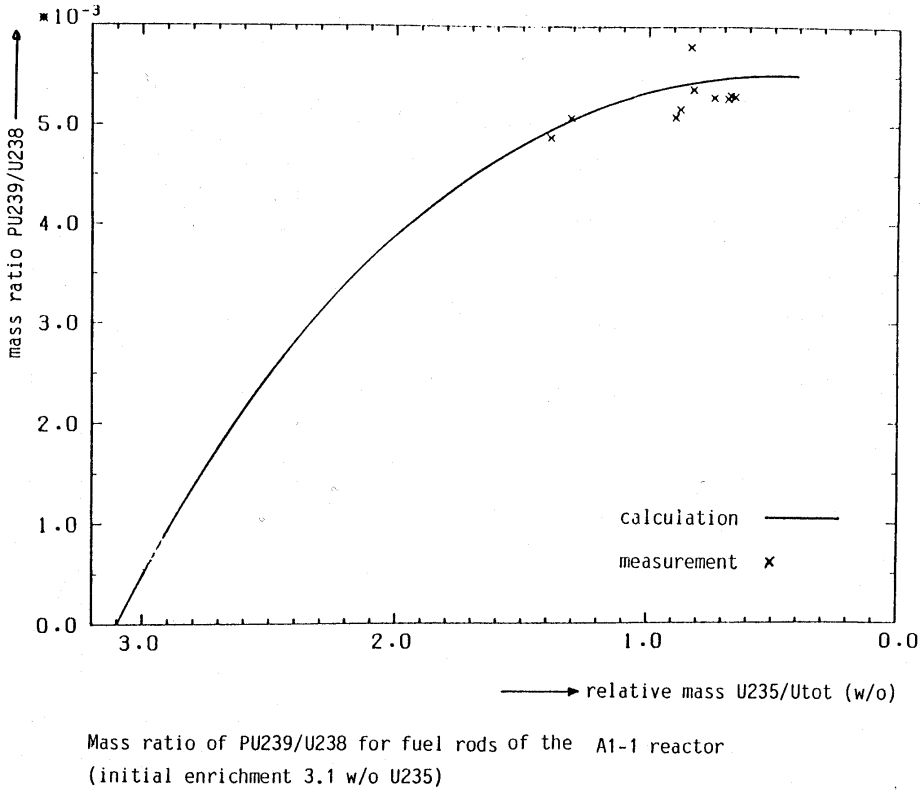


FIG. 3. Fuel Nuclide Concentrations in UO_2 Rods: Verification of the Siemens Standard Design Procedure SAV: Comparison of Measured and Calculated Isotopic Fractions.

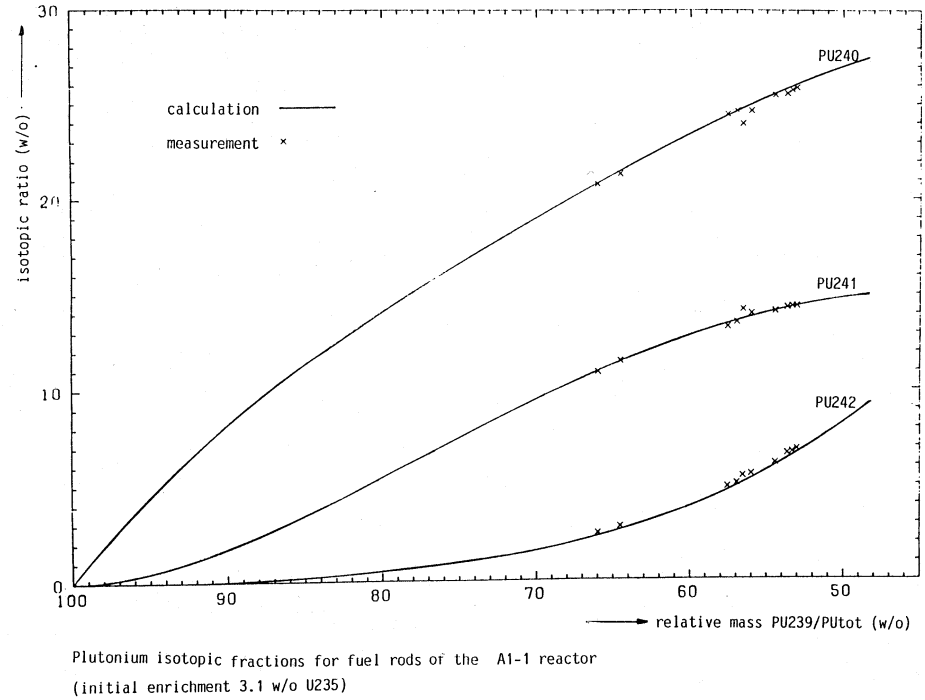
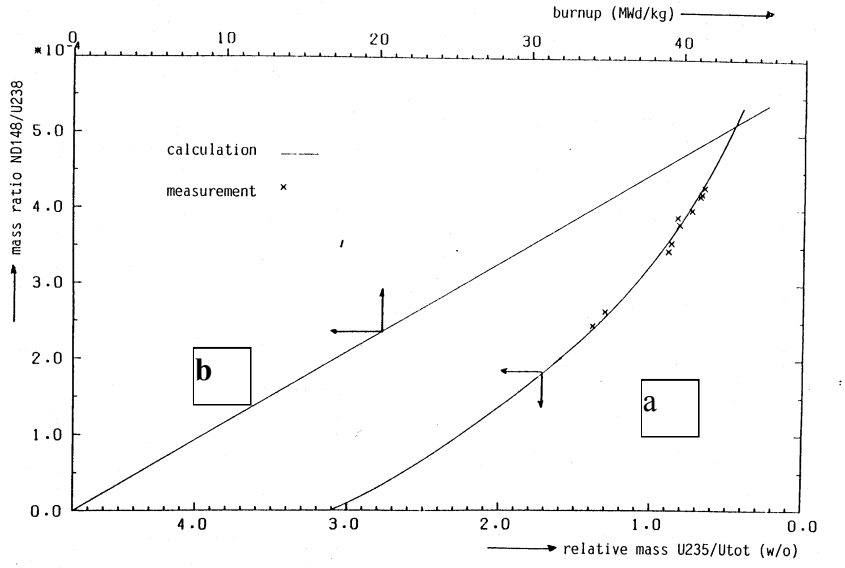
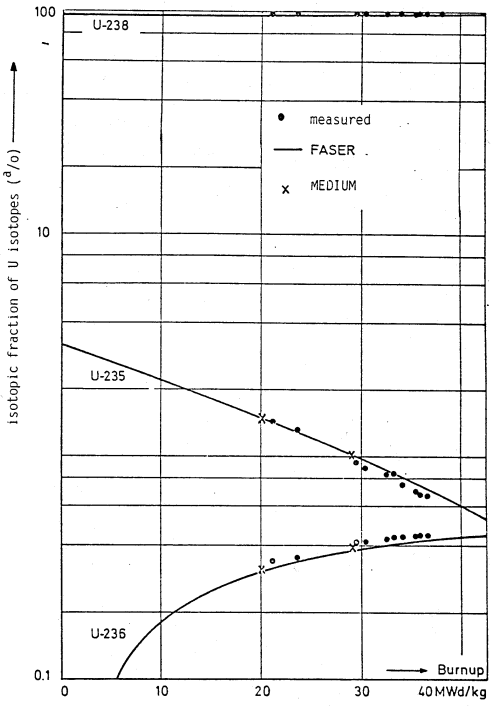


FIG. 4. Fuel Nuclide Concentrations in UO_2 Rods: Verification of the Siemens Standard Design Procedure SAV: Comparison of Measured and Calculated Isotopic Fractions.



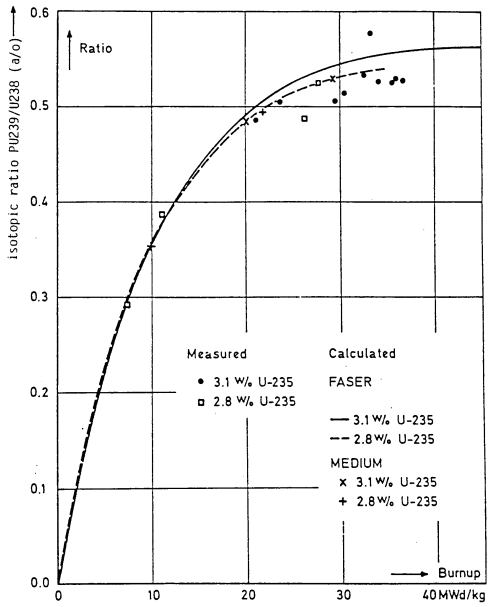
Mass ratio of ND148/U238 for fuel rods of the A1-1 reactor
(initial enrichment 3.1 w/o U235) for the generation of a burnup scale

FIG. 5. Fuel Nuclide Concentrations in UO_2 Rods: Verification of the Siemens Standard Design Procedure SAV: : Mass Ratio Nd-148/U-238 as a Function of the Ratio U-235/U (Curve a) (for the Transformation of the Nd-148 Scale into the More Convenient Burnup Scale the Mass Ratio Nd-148/U-238 is Calculated as a Function of Burnup as Shown by the Curve b).



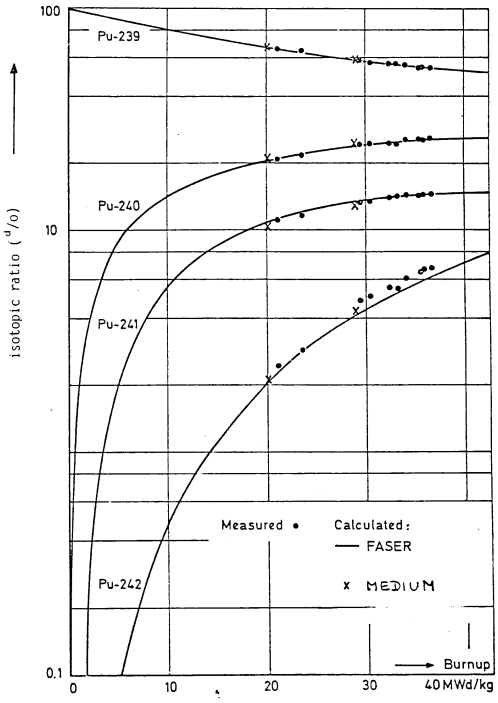
Isotopic composition of Uranium for reprocessed fuel
assemblies of the A1-1 reactor
(initial enrichment of the FAs 3.1 w/o)

FIG. 6.Fuel Nuclide Concentrations in Reprocessed UO_2: Verification of the Siemens Standard Design Procedure SAV: Comparison of Measured and Calculated Isotopic Fractions.



Isotopic ratio PU239/U238 for reprocessed fuel assemblies of the A1-1 reactor (initial enrichment of the FAS 2.8 and 3.1 w/o U235)

FIG. 7. Fuel Nuclide Concentrations in Reprocessed UO_2: Verification of the Siemens Standard Design Procedure SAV: Comparison of Measured and Calculated Isotopic Fractions.



Isotopic composition of Plutonium for reprocessed fuel assemblies of the A1-1 reactor (initial enrichment of the FAs 3.1 w/o)

FIG. 8. Fuel Nuclide Concentrations in Reprocessed UO_2: Verification of the Siemens Standard Design Procedure SAV: Comparison of Measured and Calculated Isotopic Fractions.

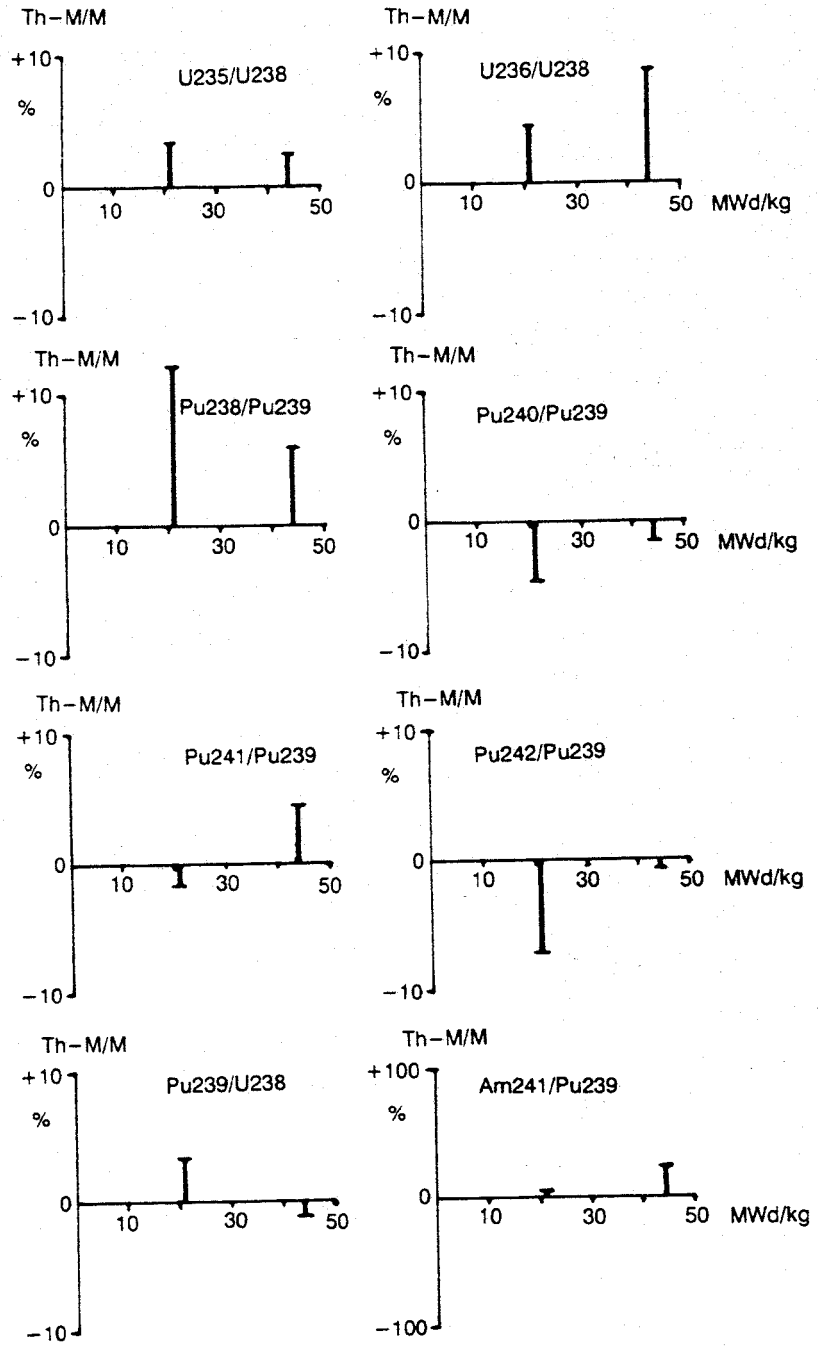


FIG. 9. Fuel Nuclide Concentrations in First Generation Recycle MOX fuel (NPP Obrigheim): Verification of the Siemens Standard Design Procedure SAV: Comparison of Measured and Calculated Concentrations (Th:= Calculated, M:= Measured)

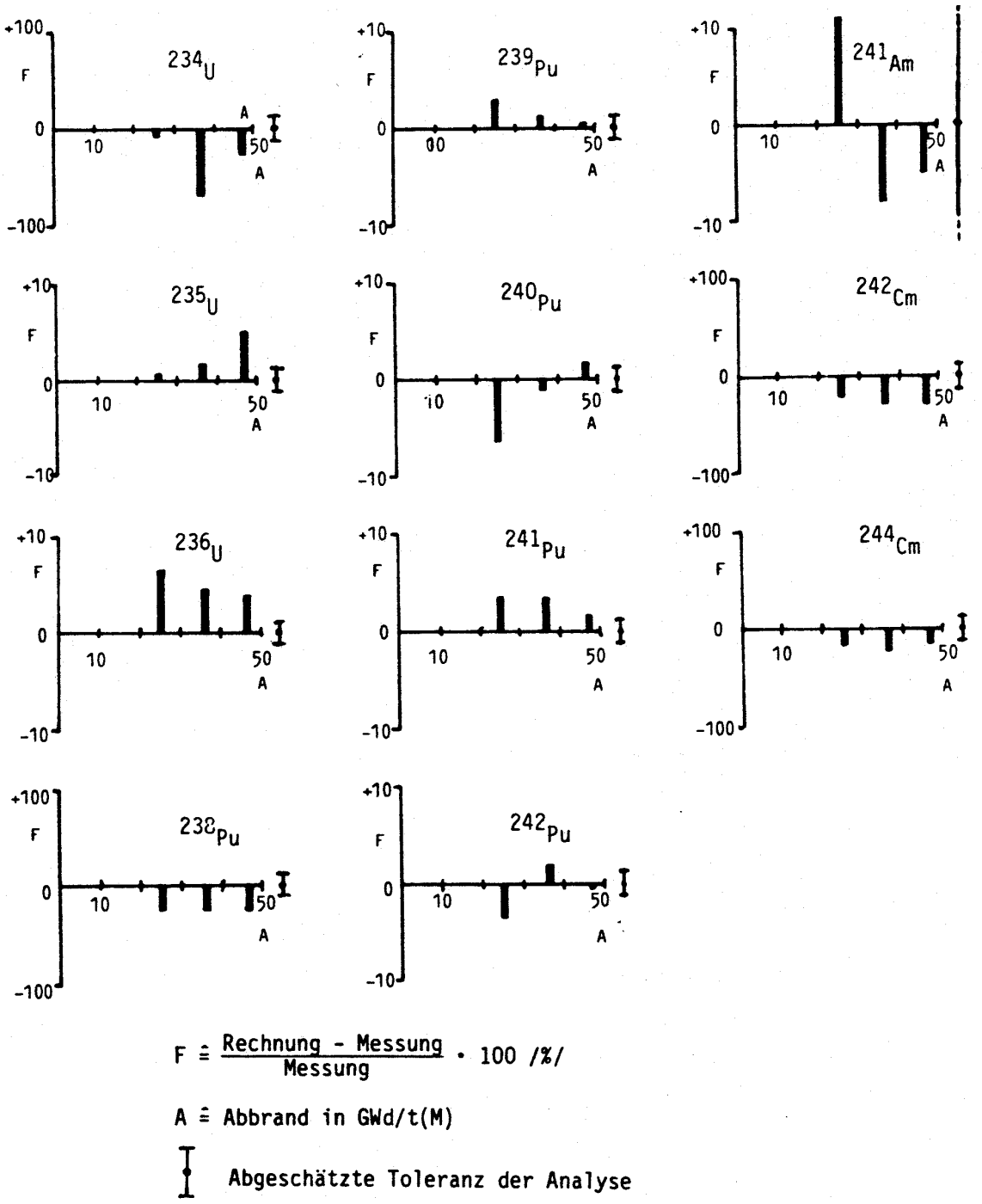


FIG. 10. Fuel Nuclide Concentrations in Second Generation Recycle MOX fuel (NPP Obrigheim): Verification of the Siemens Standard Design Procedure SAV: Comparison of Measured and Calculated Concentrations (Rechnung:= Calculation, Messung:= Measurement).

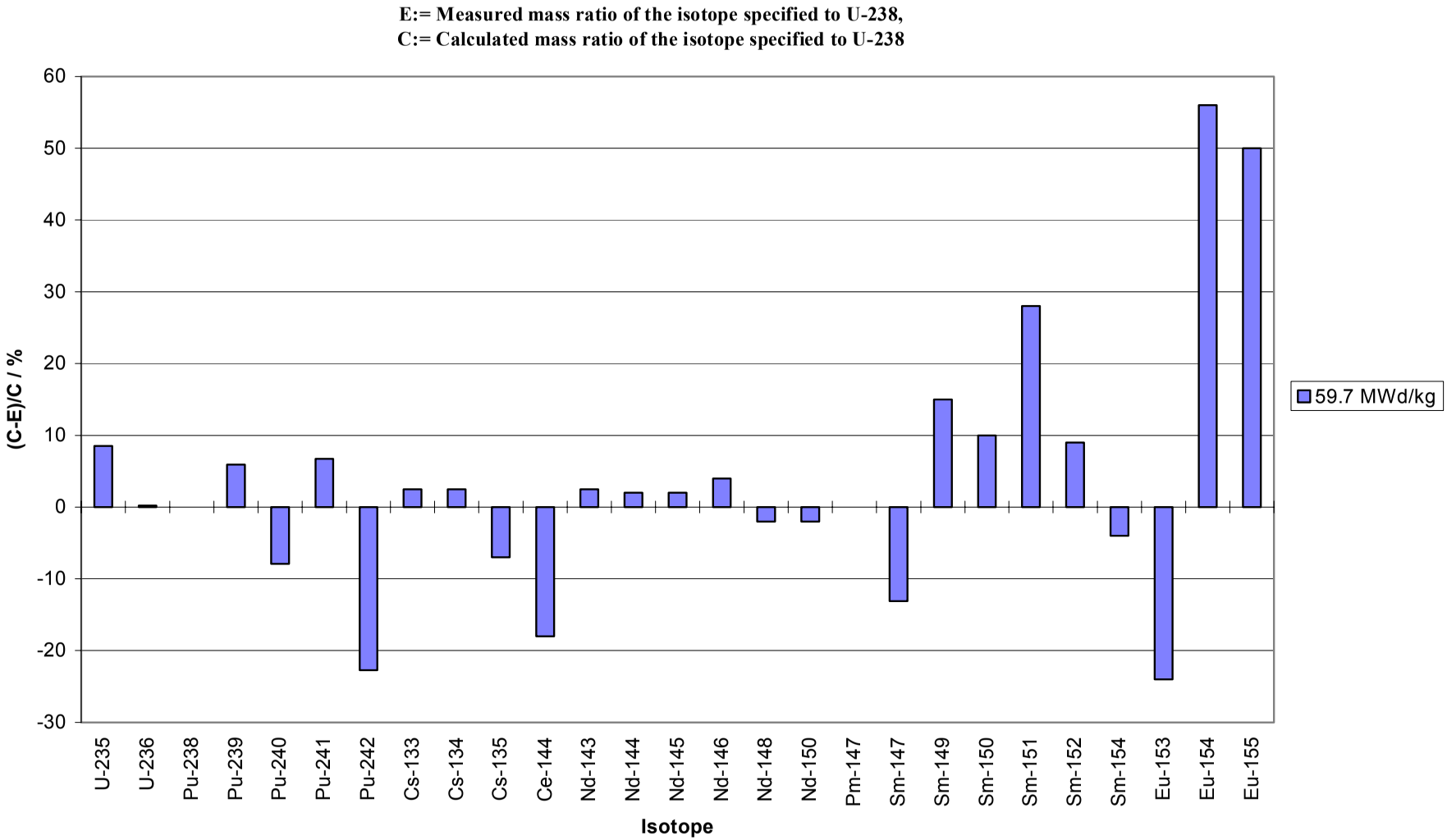


FIG. 11. Results from the ARIANE International Programme [9]: Differences between Measured (E) and Calculated (C) Isotopic Mass Ratios.

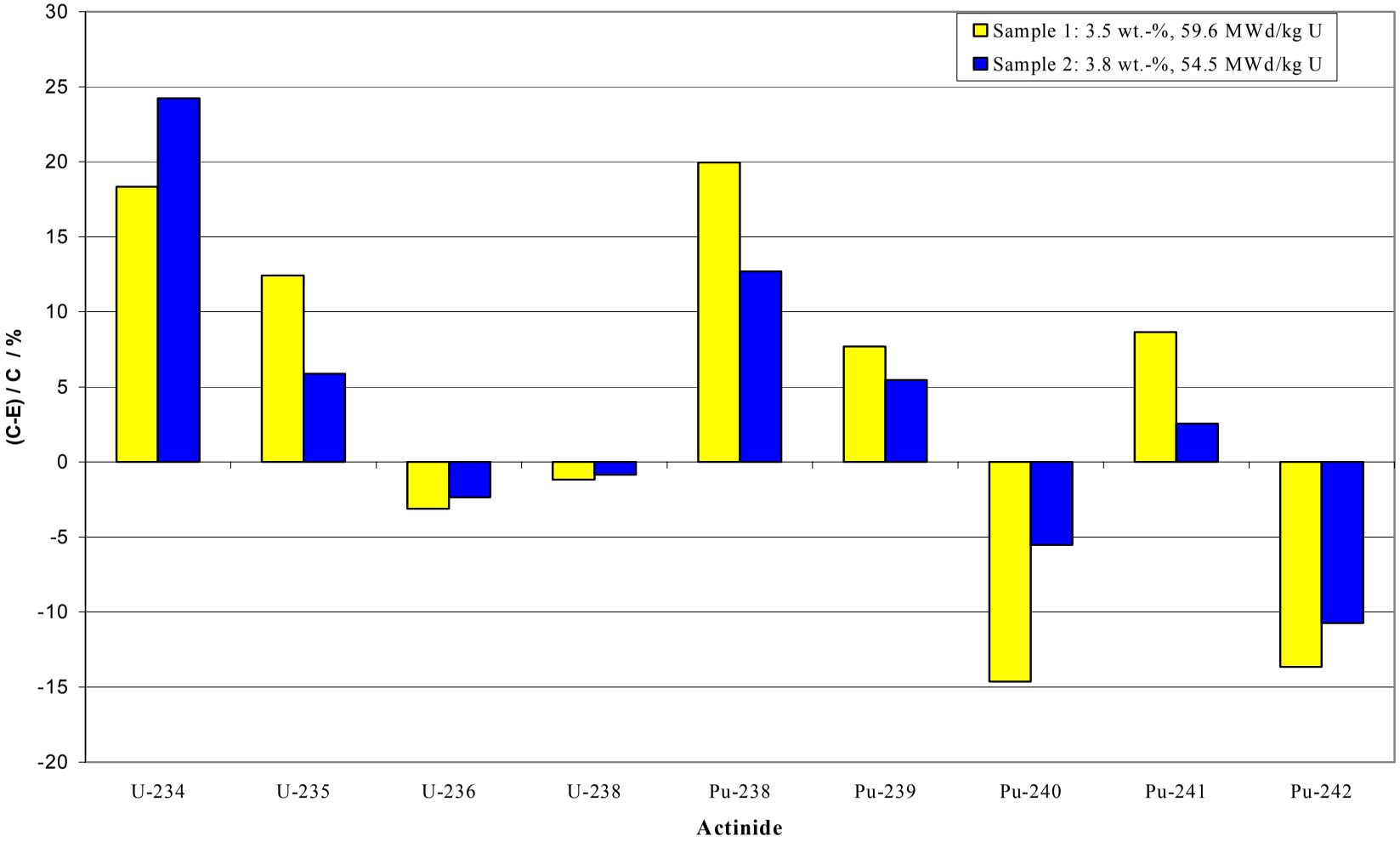


FIG. 12. Evaluation of Chemical Assay on Fuel Samples from NPP GKN II : Differences between Measured (E) and Calculated (C) Isotopic Mass Ratios.

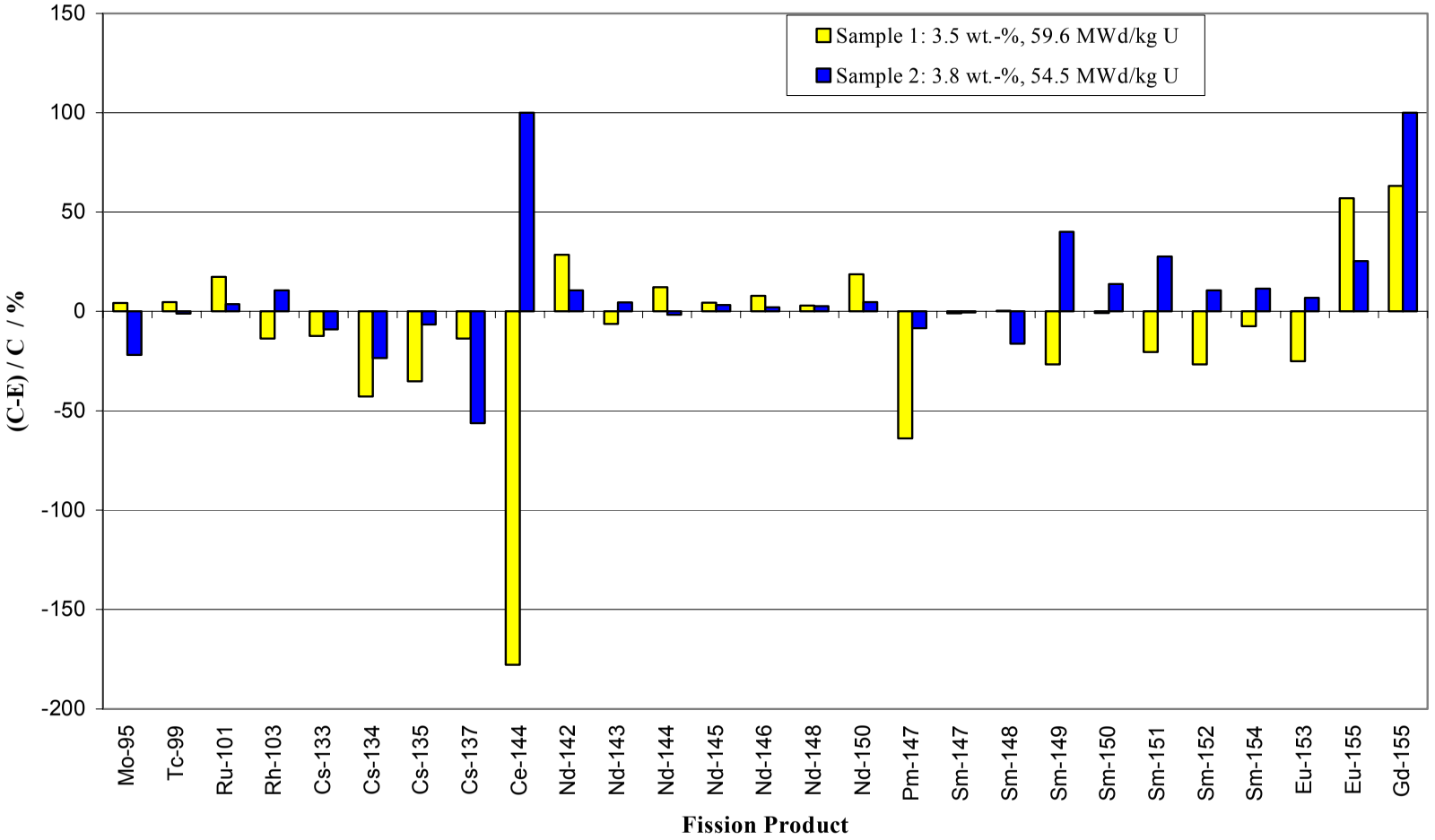


FIG. 13. Evaluation of Chemical Assay on Fuel Samples from NPP GKN II : Differences between Measured (E) and Calculated (C) Isotopic Mass Ratios.

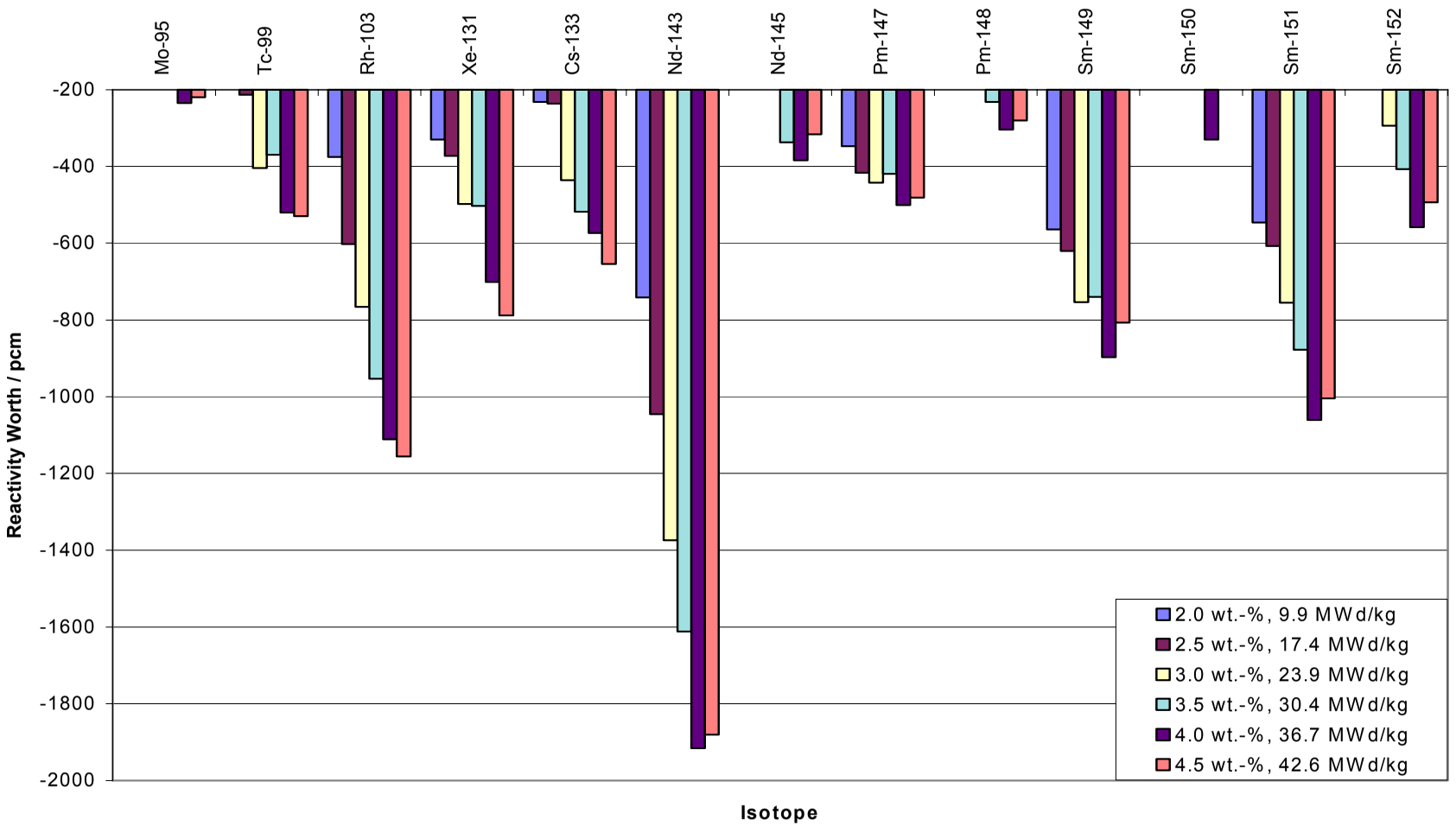


FIG. 14. Storage of Spent KONVOI UO_2 Fuel in Storage Racks of the Region II Type: Reactivity Worth of Fission Products at the Time of Reactor Shut Down (Uncertainty: ± 50 pcm.) (Initial Enrichments and Burnups Specified Correspond of the Loading Curve of the Region II Racks).

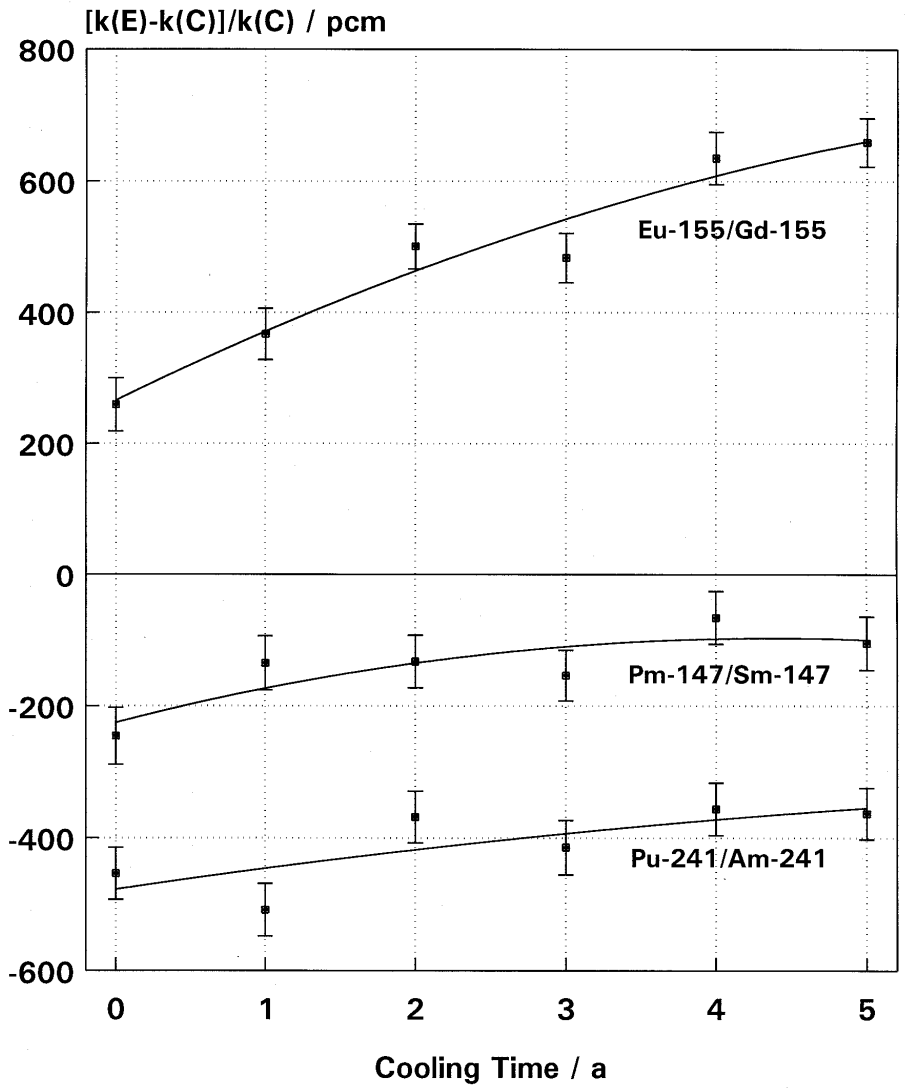


FIG. 15. Storage of Spent KONVOI UO_2 Fuel in Storage Racks of the Region II Type: Relative Changes of the Bias in the Neutron Multiplication Factor as a Function of Cooling Time, cp. Equation (3.3). (Initial Enrichment 4.0 wt.-%, Burnup 36.7 MWd/kg U, cp. Figure 14).

1. Experimentalists should check their results on consistency as far as possible at all. The differences between measured and calculated data obtained for Eu-155 and Gd-155 from the GKN II samples for instance (cp. Figure 13) are not fully consistent. With these differences one gets negative number densities for Gd-155 at time of shut down.
2. One must admit, however, that chemical assay is hard to do, and so it is not surprising that one observes fluctuations in the differences between measured and calculated concentrations. For instance, even though the two GKN II samples are close together in initial enrichment and burnup the differences between measured and calculated mass ratios observed for the Sm isotopes are different in sign, cf. Figure 13. From the view of physics this means that there is no reason for revising the depletion code applied.

In addition, what matters with respect to burnup credit applications is the reactivity worth of the nuclides, since the actual point of interest is to estimate the uncertainty in the neutron

multiplication factor of a spent fuel management system of interest which is due to the uncertainties in the calculated isotopic concentrations. The result of a burnup credit criticality safety analysis is the determination of a loading curve for the spent fuel management system of interest. This curve specifies the loading criterion by indicating the minimum burnup necessary for the fuel assembly with a specific initial enrichment to be placed in the spent fuel management system designed for burnup credit. Reasonable loading curves provide for initial enrichments of 3.5 wt.-% or 3.8 wt.-% U-235 of course minimum burnup values which are significantly smaller than the burnup values of the GKN II samples. That means that the reactivity worth of the fission product concentrations referring to a loading curve is significantly smaller than the reactivity worth of the fission products in the GKN II samples. It follows, therefore, that the uncertainty in the neutron multiplication factor referring to the loading curve is smaller than that one which follows from the GKN II sample data.

3. FISSION PRODUCT REACTIVITY WORTH AND UNCERTAINTY IN THE NEUTRON MULTIPLICATIN FACTOR OF SPENT FUEL MANAGEMENT SYSTEMS

3.1 Without Cooling Time Credit

The concentration and hence the contribution of the isotopes to neutron absorption, resulting in either fission or simple neutron capture reactions are dependent on cooling time. In burnup credit methodologies applied to spent PWR fuel storage pools the fission product isotopes are frozen at the concentrations existing at the time of shut down, except for I-135 and Xe-135 which are not considered because of their small half-lives. Consequently, as exemplified in Figure 14 by the case of irradiated KONVOI UO_2 fuel assemblies stored in region II type storage racks typical of KONVOI plants, among the fission products usually considered in burnup credit applications [10-11] only the isotopes Rh-103, Xe-131, Cs-133, Nd-143, Sm-149, and Sm-151 show a considerable reactivity worth for all the burnups specified in Figure 14, whereas the isotopes Tc-99, Nd-145, Pm-147, and Sm-152 have a significant reactivity worth only in the range of the higher values of these burnups. As indicated in Figure 14, these burnups correspond to the loading curve of the region II KONVOI fuel storage racks mentioned above. As can be seen from Figure 14, the reactivity worth of the isotopes Mo-95, Pm-148, and Sm-150 is of minor importance, and the reactivity worth of all the remaining fission products is more or less negligible. So therefore, as regards depletion code validation for burnup credit applications to spent fuel storage pools the attention can mainly be focused on the verification of the calculated inventory of actinides plus a few fission products.

As can be seen from a comparison of Figure 14 to Figures 11 and 13, the fission products with a considerable reactivity worth for all or, at least, some of the burnups specified in Figure 14 are those ones for which mostly fairly good agreement between measured and calculated mass ratios were found. This is not surprising. With the exception of the Sm isotopes Sm-149 and Sm-151 which have very high thermal cross sections for neutron capture [10] the considerable reactivity worth of all the other fission products specified in Figure 14 is mainly due to the isobaric yields for fission of U-235 and hence to the number densities of these fission products. Accordingly, a higher accuracy can be achieved in the chemical assay of these fission products.

It is not surprising, therefore, that the differences between the measured and calculated fission product mass ratios as obtained from the ARIANE programme (cp. Figure 11) result in a

relatively small bias of the neutron multiplication factor of the region II KONVOI fuel storage racks mentioned above of

$$\Delta k \pm \sigma(\Delta k) = 0.0052 \pm 0.0005\,, \qquad (3.1)$$

with

$$\Delta k = k(E) - k(C) \qquad (3.2)$$

where *k(C)* refers to the calculated isotope number densities and *k(E)* refers to the corrected isotope number densities obtained with the aid of the differences shown in Figure 11. $\sigma(\Delta k)$ denotes the one standard deviation of Δk.

The result (3.1) refers to the fission products only and gives the underestimation of the neutron multiplication factor of the region II KONVOI fuel storage racks due to the observed differences between the measured and calculated fission product mass ratios. The isotope number densities of the actinides were not corrected. Due to the slight underestimation of the fertile isotopes U-236, Pu-240, and Pu-242 and due to the slight overestimation in the fissile isotopes U-235, Pu-239, and Pu-241 a correction of the actinide number densities would result in a decrease of the calculated neutron multiplication factor of a spent fuel storage pool.

As shown in [10], the result (3.1) is confirmed by evaluations of experimental results of the French programme on burnup credit [11-12].

3.2 With Cooling Time Credit

The reactivity worth of the fission products referring to a loading curve of a PWR transport or storage cask system designed for burnup credit should not be very different from Figure 14, because the loading curve of that system is based on the case of a fully flooded cask, i.e., on a thermal system. However, in burnup credit methodologies applied to spent fuel transport and storage casks often credit for cooling time is taken. One has to take into account, therefore, that the bias in the neutron multiplication factor of the cask may increase with the cooling time due to radioactive decay of some isotopes.

Figure 15. shows for example the relative changes

$$\frac{\Delta k}{k(C)} = \frac{k(E) - k(C)}{k(C)} = \frac{k(E)}{k(C)} - 1 \qquad (3.3)$$

of the bias in the neutron multiplication factor of the region II KONVOI fuel storage racks used already for Figure 14. due to the radioactive decays of:

(a) Pu-241 to Am-241 (half life 14.4 a),
(b) Pm-147 to Sm-147 (half-life 2.62 a),
(c) Eu-155 to Gd-155 (half-life 4.96 a).

The outcomes obtained for the relative changes (3.3) at an initial enrichment of 4.0 wt.-% and a burnup of 36.7 MWd/kg U (cp. Figure 14) are derived from the GKN II results shown in Figures 12 and 13 indicating:

(a) an overestimation of Pu-241 by 9%,
(b) an underestimation of Pm-147 and Sm-147 by 64% and 1% respectively,
(c) an overestimation of Eu-155 and Gd-155 by 57% and 57.45% (the value for Gd-155 is corrected in order to prevent negative number densities for Gd-155 at time of shut down).

As shown in Figure 15.:

1. Due to the fact that the fissile isotope Pu-241 is overestimated in the depletion calculation and due to the fact that no result for Am-241 is available from the chemical assay (cp. Figure 12) expression (3.3) is negative for the Pu-241 decay and increases with increasing cooling time (converges towards zero due to the fact that no correction was made for Am-241),
2. Because the fission products Pm-147 and Sm-147 are underestimated in the depletion calculation and the underestimation of Pm-147 is significantly higher than the underestimation of Sm-147 expression (3.3) is negative for the Pm-147 decay and increases with increasing cooling time (converges towards zero nearly due to the fact that the underestimation of Sm-147 is nearly negligible),
3. Due to the overestimation of Eu-155 and Gd-155 in the depletion calculation the absolute amount of the uncertainty in the number density of Gd-155 increases with increasing cooling time and hence the bias in the neutron multiplication factor increases with the cooling time.

So therefore, as follows from Figure 15, if credit for cooling time is taken one has to be aware of the possibility that the bias in the neutron multiplication factor of the spent fuel management system of interest may increase. To which extent the bias is changing depends on the outcomes of the depletion calculation in comparison to chemical assay data. As follows from Figures 11 and 13 as well as from the results from the French programme on burnup credit [10-12] the isotopes Eu-155 and Gd-155 seem to be overestimated always in depletion calculations. There are grounds for the assumption that there are some difficulties either in the calculational or in the experimental methods, therefore. It is a good thing, therefore, that in the REBUS programme [13] it is planned not only to analyze the isotopic concentrations through chemical assay but also to measure the reactivity worth of the spent fuel. The reactivity worth measurement provides an additional tool to check the chemical assay data on consistency, and in addition to that, the reactivity worth measurement makes it possible to recalculate the experimental mock-up with criticality calculation codes, thus benchmarking these codes.

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THE IMPLEMENTATION OF BURNUP CREDIT IN VVER-440 SPENT FUEL

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Abstract

The countries used russian reactors VVER-440 cooperate in reactor physics in Atomic Energy Research (AER). One of topic areas is "Physical Problems of Spent Fuel, Radwaste and Decommissioning" (Working Group E). In this article in the first part is overview about our activity for numerical and experimental verification of codes which participants use to calculation of criticality, isotopic concentration, activity, neutron and gamma sources and shielding shown. The set of numerical benchmarks (CB1, CB2, CB3 and CB4) is very similar (the same idea, the VVER-440) to the OECD/NEA/NSC Burnup Credit Criticality Benchmarks, Phases 1 and 2. In the second part is verification of the SCALE 4.4 system (only criticality and nuclide concentrations) for VVER-440 fuel shown. In the third part is dependence of criticality on burnup (only actinides and actinides + fission products) for transport cask C30 with VVER-440 fuel by optimal moderation shown. In last part is current status in implementation burnup credit in Slovakia.

1. INTRODUCTION

In Slovakia are in operation 6 units VVER-440 with hexagonal assemblies and triangular lattice of fuel pins. Countries with reactors VVER-440 cooperate in Atomic Energy Research (AER). One of topic areas is "Physical Problems of Spent Fuel, Radwaste and Decommissioning" (Working Group E). To verify code used by spent fuel calculations we prepared numerical and experimental benchmarks.

In Slovakia we use very well known the SCALE system for calculation. We started with version SCALE 4.1, now we use SCALE 4.4. A new version we always verify on actual set of benchmarks.

2. BENCHMARKS

We (Working Group E) prepared numerical and experimental benchmarks for criticality, nuclide composition, activity, neutron and gamma sources and shielding. This set is for VVER-440 fuel.

The basic is numerical benchmark. The benchmark was designed as a series of calculational benchmarks for VVER-440 fuel similar to the OECD/NEA/NSC Burnup Credit Criticality Benchmarks, Phases 1 and 2, which are solved by the members of an international working group working under the OECD/NEA/NSC at present. The benchmarks mentioned above entitled CB1, CB2, CB3 and CB4 should deal step by step with the main issues related to VVER-440 spent fuel analyses and their results could be used also for the codes and libraries intercomparison as for the VVER applications.

CB1 an eigenvalue calculation of a single infinite hexagonal lattice of fresh and spent fuel rods
CB2 nuclide compositions calculation
CB2-S activity, neutron and gamma sources, decay heat
CB3 as CB1, but with axial profile of burnup
CB4 real cask

CB1 and CB2 were evaluated and finished, CB2-S and CB3 are now in progress and CB4 will be in future (may be in year 2001).

Criticality experiments were on reactors ZR-6 (in Budapest) and LR-0 (in Øe²) made. From a lot of critical experiments we chose only several (approximately 25).

A nuclide composition was in Kurchatov Institute (in Moscow) measured in 1975, now we re-calculate it again.

Dose rate (neutron, gamma) on surface of a real loaded transport cask C30 were in NPP Jaslovské Bohunice (Slovakia) measured. The benchmark was evaluated and finished.

The number of participants is between 4 (shielding) and 12 (CB1 and CB2), some benchmarks were finished, some are in progress.

3. VERIFICATION OF THE SCALE 4.4 SYSTEM

In Slovakia we use the SCALE family code, the latest version (which we use) is the SCALE 4.4 system, we use the 44GROUPNDF5 library. This system was verified on calculational and experimental benchmarks ([8], [9]). The accuracy is acceptable. For criticality calculations are deviations less then 0.5 % for numerical benchmarks and less then 1 % for experimental benchmarks. For nuclide composition calculation are deviations usually less then 10 %, only for several isotopes up to 50 %. Relative deviations are in Fig.1 ÷ 4 shown.

4. THE IMPLEMENTATION OF BURNUP CREDIT IN VVER-440 SPENT FUEL

To demonstrate effect of burning on reactivity I calculated very simply case: infinitive lattice of assemblies with flat axial burnup. In our transport cask C30 are assemblies only in water. By normal conditions for fresh fuel is $K_{ef} < 0.95$, but by optimal moderation (water density is 0.2 g/kg) $K_{ef} > 1$. In Fig.5 is dependence of K_{ef} on burnup shown. The list of nuclides (actinides and fission products) is according benchmark CB1 (the same as in [7]). The burnup is after irradiation 1, 2, 3 and 4 years. Only fuel after irradiation 4 years (average burnup 40 MWd/kgU) and both groups of nuclides satisfies limit $K_{ef} < 0.98$.

Text cont. on page 159

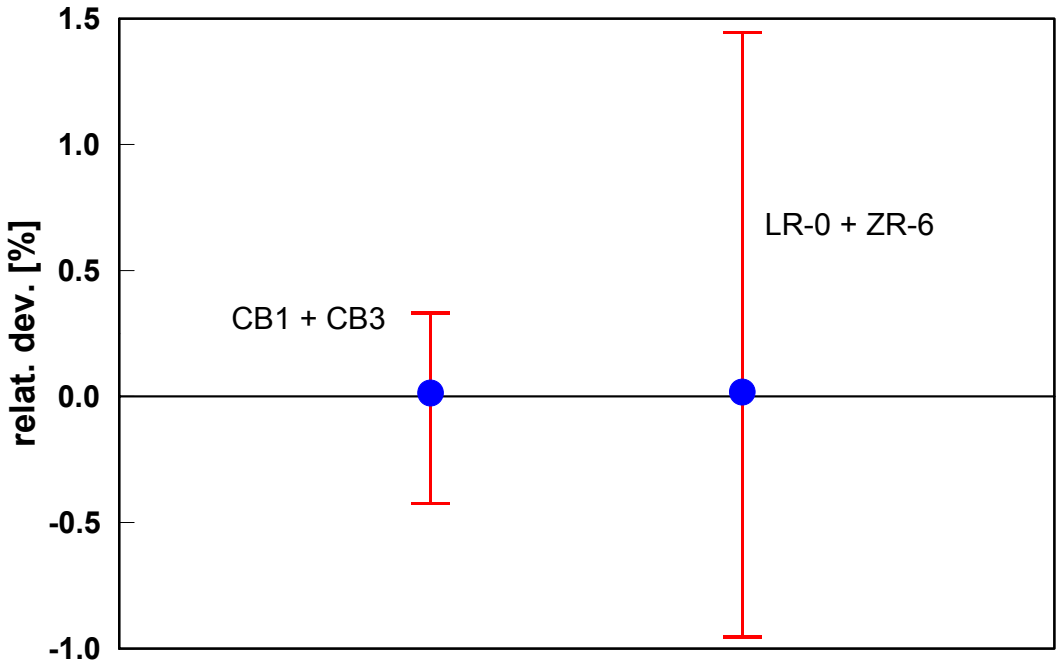

FIG. 1. Deviation of Keff.

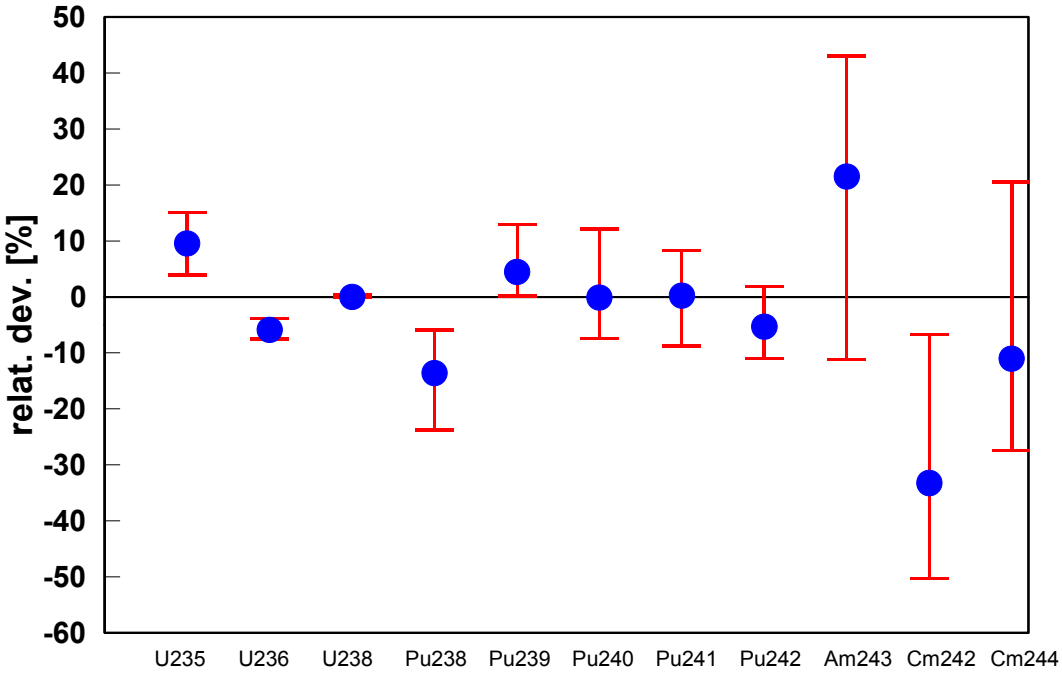

FIG. 2. Deviation of nuclide composition for experiment.

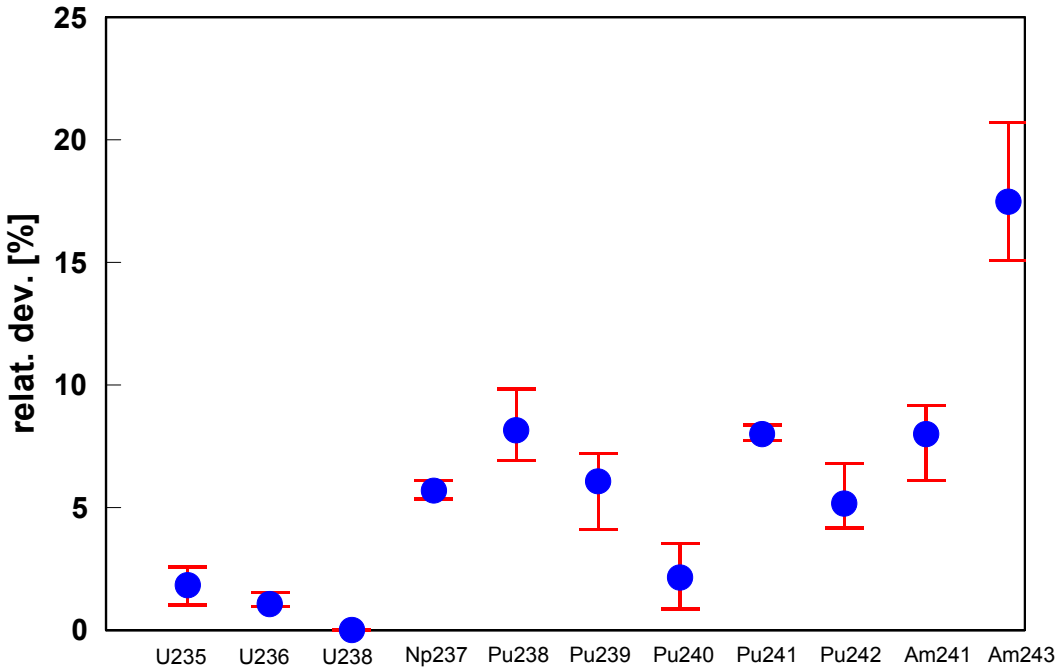

FIG. 3. Deviation of nuclide composition for CB2 (actinides).

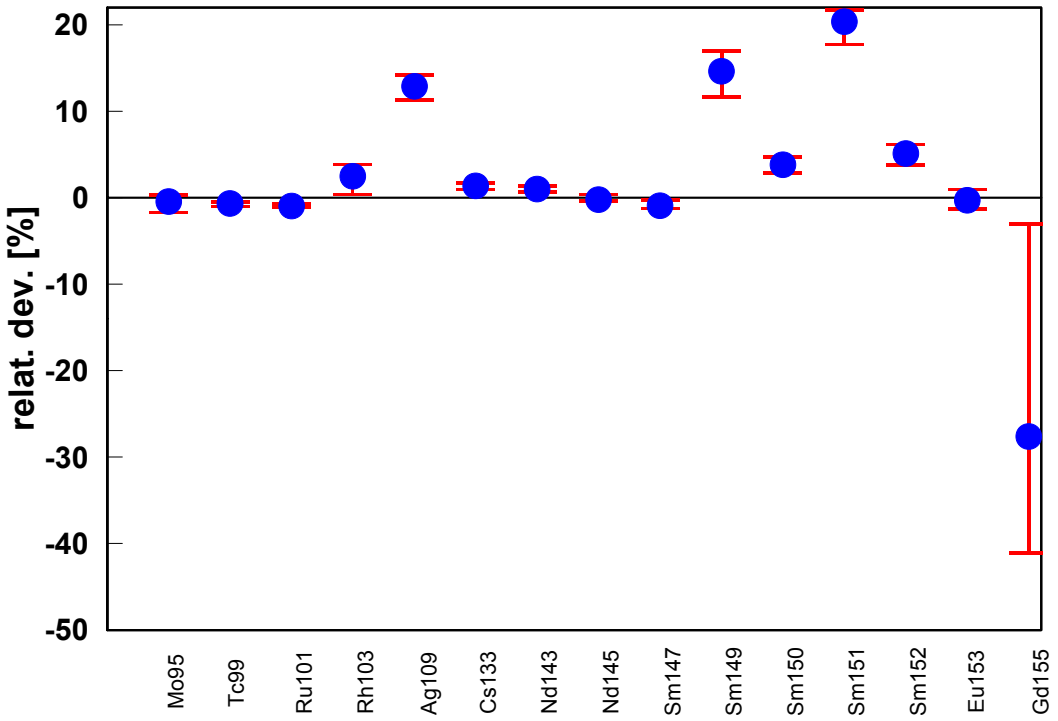

FIG. 4. Deviation of nuclide composition for CB2 (fission products).

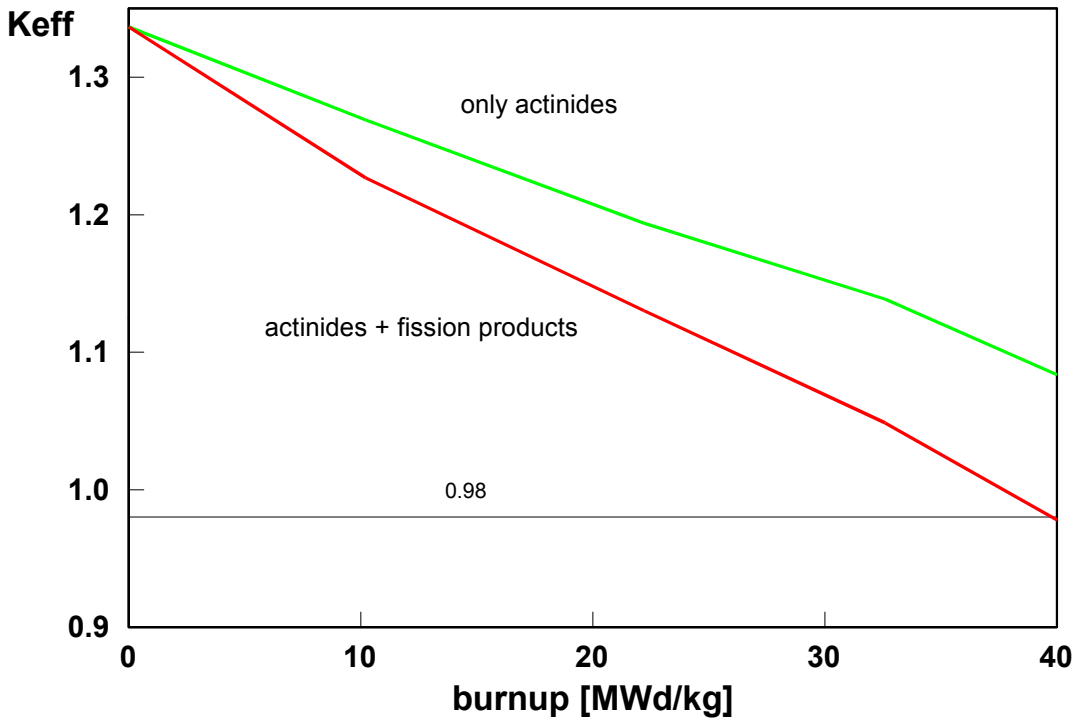

FIG. 5. Keff by optimal moderation.

5. CURRENT STATUS IN THE IMPLEMENTATION OF BURNUP CREDIT IN SLOVAKIA

In Slovakia two kinds of basket are in operations – old C30 and new compact with tubes from boron steel. The on-site pool in NPP Jaslovské Bohunice has similar construction as basket C30 (only assemblies in water). The compact basket meets subcriticality condition by both normal and abnormal conditions. The old cask C30 has license. The on-site pool by abnormal conditions (lower water density, but not optimal) with fresh fuel has Kef>1. For safety analyses we use partialy burnup credit (actinides + fission products, burnup 3 years), this way decreses Kef below 0.98.

Now is not necessary to aplicate the burnup credit. In Slovakia we are able to aplicate the burnup credit, our code and library are validated.

6. CONCLUSION

The using of burnup credit allows proving safety of cask without strong conservativity. In Slovakia we are able to use the burnup credit. In future (when will be in operation new advanced fuel with higher enrichment) will be possible to implement burnup credit.

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DISPOSAL CRITICALITY ANALYSIS METHODOLOGY'S PRINCIPAL ISOTOPE BURNUP CREDIT

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Abstract

This paper presents the burnup credit aspects of the United States Department of Energy Yucca Mountain Project's methodology for performing criticality analyses for commercial light-water-reactor fuel. The disposal burnup credit methodology uses a "principal isotope" model, which takes credit for the reduced reactivity associated with the build-up of the primary principal actinides and fission products in irradiated fuel. Burnup credit is important to the disposal criticality analysis methodology and to the design of commercial fuel waste packages.

The burnup credit methodology developed for disposal of irradiated commercial nuclear fuel can also be applied to storage and transportation of irradiated commercial nuclear fuel. For all applications a series of loading curves are developed using a best estimate methodology and depending on the application, an additional administrative safety margin may be applied. The burnup credit methodology better represents the "true" reactivity of the irradiated fuel configuration, and hence the real safety margin, than do evaluations using the "fresh fuel" assumption.

1. INTRODUCTION

The United States Department of Energy (DOE) Yucca Mountain Project's (YMP) methodology for performing disposal criticality analyses includes the use of principal isotope burnup credit for commercial light-water-reactor fuel [5]. Burnup credit involves taking credit for the reduced reactivity associated with irradiated fuel, compared to unirradiated fuel, in criticality safety evaluations. Principal isotope burnup credit takes credit for the reduced reactivity associated with the presence of the principal actinides and fission products in the irradiated fuel. DOE plans to use principal isotope burnup credit for the criticality evaluations of waste packages loaded with irradiated fuels from commercial boiling water reactors and pressurized water reactors.

Burnup credit does not eliminate safety margin but provides a better representation of the "true" reactivity of the spent nuclear fuel. Development of a burnup credit program based on best-estimate evaluations allows the applicant, and regulator, to understand the real margin of safety of the configuration, and to make an informed decision regarding the application of an additional administrative margin.

The implementation and demonstration of burnup credit methods require an understanding of reactor core physics, reactor operations, and traditional out-of-reactor criticality safety. Unlike the traditional out-of-reactor criticality approach of a "fresh fuel" assumption, burnup credit

requires an understanding of a fuel assemblies irradiation conditions and history. Knowledge of reactor physics and reactor operations is needed to simulate the way the fuel is irradiated and understand how this affects the isotopic composition, and hence the reactivity, of the irradiated fuel. Knowledge of traditional out-of-reactor criticality safety is needed for applying the information on irradiated fuel's reduced reactivity for the out-of-reactor environment.

In addition to a better understanding of the reactivity of the irradiated nuclear fuel configurations, burnup credit for transport, storage, or disposal of irradiated commercial fuel provides design flexibility that may provide economic and ALARA benefits. Burnup credit facilitates increased assembly loading in casks, which leads to reduced cumulative radiological risks (less packages) and associated cost savings. For existing systems, burnup credit would allow the loading, transport, or storage of irradiated fuel that could not be readily handled with the fresh fuel assumption.

The use of burnup credit in disposal applications has an additional value compared to transportation and short-term storage applications. Over the long time period considered for disposal, active criticality control features such as moderator exclusion barriers, neutron absorbing (poison) plates, and geometry features will degrade and change. The reduced reactivity associated with the presence of actinide and fission product absorbers in irradiated fuel is the only feature that may last.

2. METHOD

The disposal criticality analysis methodology is a risk-informed, performance-based methodology. One part of the methodology is concerned with identifying potentially critical configurations. The principal isotope burnup credit method is part of this portion of the overall methodology.

Two system models are used in the burnup credit method, an isotopic model and a criticality model. The isotopic model is used to calculate the concentrations of actinides and fission products in irradiated fuel. The criticality model is used to calculate the keff for the various configurations of irradiated fuel, using the isotopic concentrations from the isotopic model. The YMP isotopic model uses the SAS2H sequence of the SCALE computer code system [6]. The SAS2H sequence is used with a 44-energy group cross-section library. The YMP criticality model uses the MCNP computer code [7]. The MCNP code is used with its point-wise/continuous energy cross-sections. Both cross-section sets are based on ENDF/B libraries.

The isotopic model uses reactor operating parameters and fuel assembly information to model the fuel irradiation and depletion. The reactor operating parameters modeled include fuel temperature, moderator temperature/density, void histories (for boiling water reactors), soluble boron concentrations (for pressurized water reactors), specific power histories, and the presence of control mechanisms (control rods, control blades, axial power shaping rods, etc.). The fuel assembly information includes items such as the initial enrichment, the assembly burnup, the time since discharge from the reactor (cooling time), dimensional and mass information, and information about burnable poisons (removable and integral) that may have been present during irradiation cycles. The specific values used for these depletion parameters are conservative for waste package loading calculations for the specific fuel and reactor type being modeled. Sensitivity evaluations have been performed to determine how each of the depletion parameters affects the reactivity of irradiated fuel in a waste package or cask system.

For risk-informed disposal applications, the isotopic model is used to generate isotopic concentrations over large ranges of cooling times, not just conservative ones. Also, in long-term disposal applications, the isotopic concentrations in the irradiated fuel are modified to account for geochemical processes that can remove, transport, and deposit isotopes under certain conditions (i.e., fuel cladding breached).

In addition to the concentrations of actinides and fission products from the isotopic model, the criticality model uses detailed geometry and material information for the fuel assembly and waste package. The criticality model accounts for the spatial variations in burnup that occur in fuel assemblies (e.g., axial burnup profiles).

The isotopic and criticality models must be validated prior to use in evaluations. Benchmark calculations of measured data are an important part of the validation process. The validation of the models used for principal isotope burnup credit relies on three types of benchmarks: Commercial Reactor Criticals (CRCs), Radiochemical Assays (RCAs), and Laboratory Critical Experiments (LCEs). Each of these types of benchmarks is discussed below.

CRCs are measured critical configurations of commercial light-water-reactor fuel. The critical conditions for CRCs are measured under zero power conditions (isothermal) to minimize thermal gradients and to eliminate the concern over xenon in the benchmark. The CRC cases (state points) analyzed include all fresh fuel (initial cores), mixed fresh and irradiated fuel (beginning of cycle cores), and all irradiated (middle/end of cycle cores). To date, approximately 45 PWR and 20 BWR CRC state points have been calculated, each state point consists of many the detailed reactor power histories. The CRC state points include fuel with a large range of expected fuel characteristics (axial blankets, multiple axial and radial enrichments, burnable poison rods, integral fuel burnable absorbers, etc.). At present the assembly average enrichments from 1.93 weight percent (wt %) U-235 to 4.02 wt % U-235 and core average burnups from 0 GWd/tU to 33 GWd/tU have been evaluated [2]. Additional cases are currently being evaluated that will extend the range well above 4.5 wt % U-235 initial enrichment.

Evaluations of the neutronic conditions in CRCs and the neutronic conditions in waste packages have confirmed the applicability of CRC as benchmarks for waste packages containing irradiated commercial nuclear fuel [4]. The evaluations of neutronic conditions included consideration of neutron spectrum, reflection, and leakage as well as the materials present in the fuel.

Radiochemical assays are isotopic concentration measurements made with samples of irradiated fuel. The majority of the measurements are for small (fraction of a fuel pellet) measurements, but a few measurements were made on samples from half assemblies. For the principal isotope burnup credit work, approximately 85 samples (BWR and PWR) have been measured for fuel assemblies with enrichments from 2.45 wt % U-235 to 3.87 wt % U-235 and burnups from 2.16 GWd/tU to 46.46 GWd/[1]. Additional samples (26) with enrichments up to 4.64 wt % U-235 and burnups up to 70.4 GWd/tU are currently being evaluated.

LCEs are the standard critical experiments performed in laboratories and designed to replicate particular geometry or material combinations of interest to real-world applications. Nearly 500 LCEs have been analyzed so far for various disposal applications [3]. Only about 91 apply to irradiated commercial nuclear fuel in intact configurations, and 66 of these are fresh UO2. The LCEs are generally much smaller than waste packages and do not contain true representations of irradiated commercial nuclear fuel.

CRCs and LCEs are used for estimating bias and uncertainty for the criticality model. In addition, CRC data are used, with RCAs, to estimate bias and uncertainty for the isotopic model for intact configurations of irradiated commercial nuclear fuel. RCAs will also be used for confirming the adequacy of the isotopic model used for the waste package design analyses.

CRCs represent irradiated commercial fuel in known critical configurations. Although the CRC evaluations provide excellent criticality benchmarks for irradiated fuel in a reactor, they do not provide benchmarks for isotopic concentration of individual isotopes and they do not have some of the characteristics of a waste package (e.g., leakage, fuel temperature, moderator temperature, fixed absorbers). However, CRC evaluations provide valuable information on the integral capability of the models (SAS2H and MCNP) to predict the keff of a measured system containing similar fuel geometry. The neutronic characteristics that differ between CRCs and waste packages have been evaluated and the effect can be accounted for. Evaluations of both LCEs and RCAs will be performed to supplement the CRC evaluations and complete the model validations for principal isotope burnup credit applications.

Principal isotope burnup credit only includes a subset of the isotopes present in irradiated commercial fuel. The process for choosing this subset considers the nuclear, physical, and chemical properties of the irradiated commercial fuel isotopes. The nuclear properties are cross sections and half-lives of the isotopes; the physical properties are concentration (amount present in the irradiated fuel) and state (solid, liquid, or gas) of the isotopes; and the chemical properties are the volatility and solubility of the isotopes. Isotopic decay and build-up, as well as relative importance of isotopes for criticality (combination of cross sections and concentrations), are also considered in this selection process. No isotopes with significant positive reactivity effects (fissile isotopes with significant concentrations) are removed from consideration. Thus, the selection process is conservative. The process results in selecting 14 actinides and 15 fission products. Table I lists these isotopes.

Table I. Principal Isotopes.

^{95}Mo	^{145}Nd	^{151}Eu	^{236}U	^{241}Pu
^{99}Tc	^{147}Sm	^{153}Eu	^{238}U	^{242}Pu
^{101}Ru	^{149}Sm	^{155}Gd	^{237}Np	^{241}Am
^{103}Rh	^{150}Sm	^{233}U	^{238}Pu	^{242m}Am[a]
^{109}Ag	^{151}Sm	^{234}U	^{239}Pu	^{243}Am
^{143}Nd	^{152}Sm	^{235}U	^{240}Pu	

[a] Half Life = 152 years

For design applications, two aspects of the isotopic model for irradiated commercial fuel must be addressed. First, values for the discharged isotopic concentrations must be conservative with respect to their contribution to criticality. Second, changes to the discharged isotopic concentration values as a function of time must also be conservative with respect to their contribution to criticality. Three requirements have been developed as part of the methodology for principal isotope burnup credit to ensure these conservatisms:

1. Reactor operating histories and conditions must be selected together with axial burnup profiles such that the isotopic concentrations used to represent irradiated commercial fuel assemblies shall produce values for keff that are conservative in comparison to any other expected combination of reactor history, conditions, or profiles.

2. These bounding reactor parameters will be used to predict isotopic concentrations that, when compared to best-estimate isotopic predictions of the measured RCA data and the measured radiochemical data itself, must produce values for keff that are conservative.
3. The values for the isotopic concentrations representing irradiated commercial fuel must produce conservative values for keff for all time periods for which criticality analyses are performed.

3. SUMMARY

The use of principal isotope burnup credit provides for a more realistic prediction of the potential for criticality within the long repository time frames over which the fuel remains intact. Thereby allowing the applicant and regulator a more realistic understanding of the potential criticality scenarios and consequences. In addition, this methodology can be applied to storage or transportation system. A methodology for implementing principal isotope burnup credit has been developed by the YMP for use in disposal of all irradiated commercial light-water-reactor fuel. The methodology addresses operating history effects and includes validation requirements that will ensure that the results are conservative.

The presentation will explain the methodology and summarize the work performed to validate the models via use of CRCs, LCEs and RCAs. The lessons-learned and data obtained through implementation of this methodology to disposal should have benefit to the understanding and implementation of burnup credit in transportation and storage of irradiated commercial nuclear fuel. This process has been reviewed and approved by the US Nuclear Regulatory Commission for disposal.

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VALIDATION ISSUES FOR DEPLETION AND CRITICALITY ANALYSIS IN BURNUP CREDIT*

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Abstract

This paper reviews validation issues associated with implementation of burnup credit in transport, dry storage, and disposal. The issues discussed are ones that have been identified by one or more constituents of the United States technical community (national laboratories, licensees, and regulators) that have been exploring the use of burnup credit. There is not necessarily agreement on the importance of the various issues, which sometimes is what creates the issue. The broad issues relate to the paucity of available experimental data (radiochemical assays and critical experiments) covering the full range and characteristics of spent nuclear fuel in away-from-reactor systems. The paper will also introduce recent efforts initiated at Oak Ridge National Laboratory (ORNL) to provide technical information that can help better assess the value of different experiments. The focus of the paper is on experience with validation issues related to use of burnup credit for transport and dry storage applications.

1. INTRODUCTION

Requirements applied within the United States, for validation of codes and data used for criticality safety outside reactors, are provided by ANSI/ANS-8.1 [1]. This standard requires that the calculational method be validated by comparison with "the results of critical and exponential experiments." Such a comparison yields information on biases and uncertainties in the calculational methods and model. The area of applicability for the calculational method is established by the characteristics of the measured critical experiments that are considered in the validation. The standard gives no guidance on how to establish the area of applicability (e.g., which parameters, characteristics, etc., and how similar they should be to the application).

The process of performing criticality calculations for spent fuel in a burnup credit model for transport or dry cask storage requires two distinct sets of calculations - the first to estimate the isotopic contents of spent fuel based on depletion calculations; the second to perform a criticality calculation based on the predicted isotopic contents from the first set of calculations. Thus, application of ANSI/ANS-8.1 to burnup credit validation becomes somewhat complicated by (1) the need to consider both the depletion analysis methodology and the criticality analysis methodology and (2) the lack of spent fuel critical experiments.

The objective of a validation effort per ANSI/ANS-8.1 guidance is to establish a limit for the calculated neutron multiplication factor (keff) below which the system of concern would be considered subcritical. The "fresh fuel" assumption has provided a simple, bounding approach which allows less scrutiny of the validation needs relative to fuel composition. Under burnup

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credit, applicability of experiments are not as obvious and validation efforts may be more closely scrutinized to ensure adequate definition and understanding of the subcritical margin.

The nature of experimental data appropriate for use in validation of burnup credit analysis methodologies and the value and applicability of such data have been debated topics for over a decade. Available (albeit some are proprietary) experimental data include chemical assays of spent nuclear fuel (SNF) inventories, critical experiments performed with fresh fuel (unirradiated fissile material) in cask-like geometries, reactivity-worth measurements, subcritical experiments, and reactor critical configurations. The following subsections discuss each of these sources of measured information and their potential value to the validation process.

2. CHEMICAL ASSAY MEASUREMENTS

2.1. Review of Available Data

Radiochemical assay measurements in the United States have been made for select spent fuel nuclides, for both PWR [2 to 8] and BWR fuels [9,10]. In addition, Ref. [11] is a compilation of sources of radiochemical assay data from these and other sources. Reference [12] describes sources for additional isotopic assays and assesses the completeness of available data describing each set of measurements.

Within the United States, chemical assay data have historically focused on the major actinides within PWR spent fuel. The actinides of importance in burnup credit have been measured for approximately 50 PWR fuel samples that provide the basis for performing code validation. Of these 50 samples, only seven had burnable poison rods available during irradiation – an indication of the age of the fuel designs and the data. For most fission product nuclides important in burnup credit, very few assay measurements (typically three samples) have been made in the United States. The enrichment and burnup ranges of the PWR spent fuel samples readily available in the United States are shown in Fig. 1 as the “existing database.” The majority of these measurements have been used to determine the biases and uncertainties of computational methods [13 to 15]. With the current trend towards higher enrichment and burnup values, the acquisition of additional assay data to support code validation in this regime is considered a high priority in the United States. Additional PWR and BWR spent fuel assays with the desired characteristics are currently being performed to support U.S. Department of Energy (DOE) programs, but will not be available until sometime in 2001. Other sources of chemical assay data currently exist and/or are planned, largely within programs organized by other countries and held proprietary by those who procured the data. The French program on burnup credit [16], the REBUS program organized by Belgonucleaire [17], the LWR-PROTEUS program organized in Switzerland [18], and referenced Japanese data [19] are all potential sources of additional chemical assay data for use in validation. Figure 1 highlights the characteristics of the known sources of assay data identified by ORNL for potential use in burnup credit validation.

2.2. Validation Approaches

The use of the chemical assays in the validation process involves a comparison of predicted nuclide concentrations to the measured concentrations. The depletion model is based on the known in-core history for the fuel sample that was characterized. Given a significant number of comparisons, it becomes possible to statistically estimate the bias and uncertainty in the

calculated prediction of each individual nuclide concentration. The bias is obtained by finding the average difference between computed and measured concentrations for each individual nuclide; the uncertainty characterizes in a statistical manner the variation of individual comparisons around the mean [20]. The total uncertainty should also include statistical uncertainty based on a limited sample size. Reference [20] describes an approach for calculating bias and uncertainties such that one has a reasonable confidence that one can conservatively predict the concentration of a nuclide. Conservatism is defined in terms of correcting a nuclide concentration in such a way that has the effect of maximizing k_{eff} for a system. A second statistical approach is presented in Ref. 21. In both of these procedures, total calculated biases and uncertainties include any biases and uncertainties inherent in the experimental measurements. Thus, there is potential for offsetting errors in the bias, and the uncertainty may not be properly characterized. However, this is a random process, and non-offsetting errors would also be present.

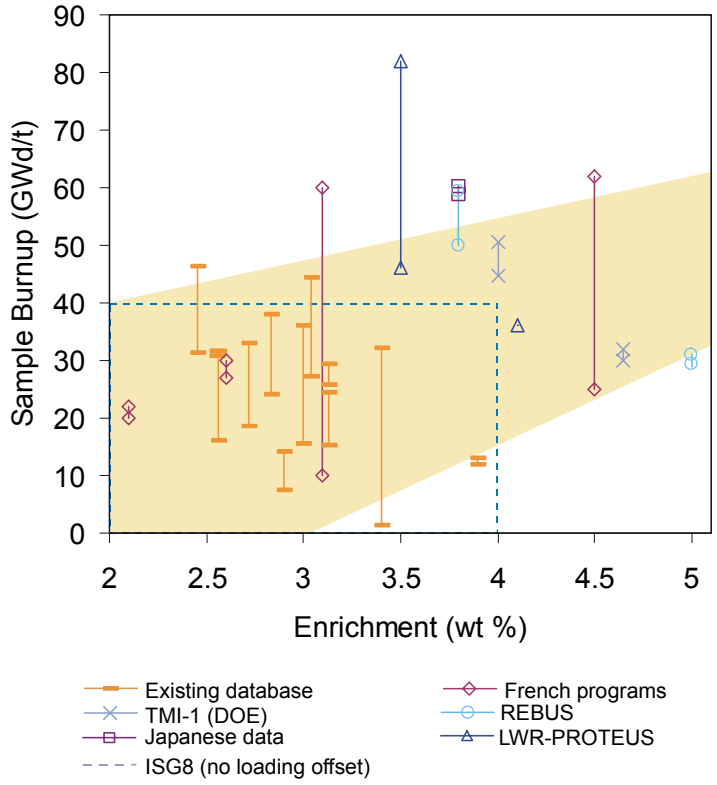

FIG. 1. Available and Potential PWR Chemical Assay Data Overlaid on a Shaded Range of High Applicability With Respect to the Enrichment and Burnup Regime of Existing Assay Data.

Note that the procedure described above determines the calculational biases and uncertainties for each individual isotope evaluated. Simultaneous application of conservative corrections to individual nuclides within a predicted SNF inventory is a bounding, but unrealistic approach.

Another approach that could be used to obtain uncertainties in the SNF inventory would be to assess the integral effect on k_{eff} due to random variations to the SNF nuclide set within the characterized uncertainty bounds defined for each nuclide. This random variation of the inventory may provide a more realistic distribution of k_{eff} values for the application that can be directly tied to nuclide uncertainties and prevent simultaneous conservative correction of each

nuclide. A conservative margin can be assigned based on the expected statistical distribution of k_{eff} values. Perceived advantages (better estimates of the impact of the uncertainty in the spent fuel inventory) and disadvantages (increased complexity and computational time) of this approach are being studied at ORNL [22].

2.3.Extending the Use of the Data

Given the limited number of chemical assays available, and the range of enrichments and burnups represented by these data, it has not been possible to clearly establish trends in biases and uncertainties as a function of the governing parameters. Although some chemical assay data exist for a moderate range of burnups, other factors also vary (i.e., assembly design, operating history, poison concentrations, etc.). Insufficient data prevent application of a reliable multivariate evaluation. Although additional measurements should be pursued where essential (high-enrichment and high-burnup conditions), the lack of experimental facilities available in the United States to handle and process commercial spent fuel, combined with the cost of the procedure itself, will limit the number of samples available for validation in the near future.

Thus, other options that can provide technical justification for extending the range of the validation and/or interpolating on the range are being considered in conjunction with new experimental data. The dashed lines in Fig. 1 represent the limits beyond which an additional reactivity margin is recommended by the current burnup credit regulatory guidance [23] on transport and dry storage. The recommended reactivity margin is equal to 1 GWd/t for every 0.1 wt % initial enrichment above 4.0 wt %. Developing validation techniques and/or data that enables extension beyond these limits is of keen interest in the United States. However, extending the area of applicability by making use of trends in the bias and uncertainty has proven to be challenging due to a relatively large variability in the existing data and the many factors that may influence the overall bias and uncertainty: fuel enrichment, burnup, assembly design complexity, calculational methods, nuclear data, and uncertainties in reactor operating conditions, irradiation history, and sample burnup. A reliable trending assessment is challenged by the limited amount of experimental data and the large number of different parameters that can affect the bias.

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. These studies indicate that there is a strong correlation between spent fuel systems with a constant enrichment-to-burnup ratio. An example of the band of “high applicability” with respect to the enrichment and burnup regime of existing assay data found by the sensitivity-based methods is shown by the shaded area in Figure 1. The results suggest that existing isotopic assay data may be highly applicable to regimes well beyond that of the measured data. However, there is currently insufficient experimental data to validate these findings. It is anticipated that, as new assay data become available, it will be possible to combine the limited amount of experimental data with the sensitivity-based methods to provide additional evidence to support predictions on bias and uncertainty beyond the range where the majority of experimental data exist.

Reactor critical configurations and some planned reactivity-worth experiments with spent fuel are integral experiments that require prediction of the nuclide composition and k_{eff} analysis. Thus, these experiments are also potential sources of experimental data that may be used to supplement, or potentially replace, the use of assay data. These other data sources will

continue to be considered as work continues to expand the range of data available for validation of the SNF nuclide compositions.

3. CRITICAL EXPERIMENTS WITH FRESH FUEL

The validation of criticality safety analysis codes using critical experiments with unirradiated fissile material is a procedure that has been applied for years to meet the requirement of ANSI/ANS-8.1. Experiments exist for a wide variety of conditions representative of pin lattices within cask environments [24]. The value of fresh fuel critical experiments to validation of spent fuel in casks relates to the fact that these experiments provide validation of the particle transport models and cross-section data within cask-like conditions (e.g., similar geometry, reflectors, intermediate absorbers). However, these experiments do not contain the same relative compositions or even all of the nuclides that are present in spent fuel. Thus, there is a need to limit the applicability of the experiments to validation for nuclide compositions present in the critical experiments (typically plutonium and uranium isotopes only) and/or demonstrate their applicability to a spent fuel inventory.

The current U.S. regulatory guidance for burnup credit application to transport and dry storage [23] recommends that burnup credit be limited to the actinide-only nuclide compositions in the SNF. Thus, initial work [21] has placed a strong reliance on mixed-oxide (MOX) critical experiments. However, MOX critical experiments available in the U.S. do not have the same relative composition of uranium and plutonium nuclides as found in typical spent fuel. Proprietary fresh fuel experiments using pin lattices with uranium and plutonium compositions similar to that for typical spent fuel (37.5 GWd/t) have been performed in France [25]. Approximately 150 experiments with these "HTC" pins simulating SNF compositions have been performed including arrays in pure water, arrays in poisoned water, pool storage configurations, consolidated pool storage configurations, shipping cask configurations, and mixed arrays of HTC and UO_2 pins. Also, proprietary fresh fuel experiments with pin lattices surrounding cans of fission product solutions have been performed in France [26] and would be excellent experiments to use if credit for fission products is sought.

Sensitivity and uncertainty (S/U) methods recently developed [27] at ORNL are being used to provide information on the strengths and potential limitations of various types of experiments relative to validation needs for burnup credit. The S/U methodology utilizes two different parameters as measures of applicability; one is a global measure for system-to-system applicability (c_k value), the other is a nuclide-specific measure of applicability (T value). This S/U methodology has been used to study the applicability of the available U.S. MOX experiments, the French HTC experiments, and the fission product solution experiments. A representative set of approximately 450 UO_2 and MOX fresh fuel criticals have been used in the study together with a representative set of five of the HTC experiments (lattices in pure water) and a single fission product solution (performed with ^{149}Sm) experiment which is publicly available [25].

A set of 36 U.S. MOX experiments proposed [21] for validation of actinide-only burnup credit was analyzed for applicability to a series of pin-cell spent fuel calculations covering burnups of 10-60 GWd/t. The global applicability parameter indicated that 15 of the 36 systems were of value in validating systems with enrichments near 60 GWd/t. For the lower-burnup cases, the number of systems predicted to be applicable fell to 1 and 0 for 30

and 10 GWd/t, respectively. The reduced applicability for lower-burnup values is not surprising since these MOX systems are designed to mockup systems with substantial amounts of plutonium. The standard UO_2 experiments are expected to fill the gap for lower enrichments. The nuclide-specific values indicated the same applicability trends for the primary plutonium isotopes with 0, 2–7, and 7–21 systems meeting the criterion for 10, 30, and 60 GWd/t systems. Thus, these systems are useful for validation of burnup credit studies; however, they must be supplemented with additional systems at low burnups.

The c_k values for the HTC experiments indicate a high degree of applicability to a series of infinite pin-cell calculations for burnups ranging from 10–60 GWd/t. The *T* values also indicate a high degree of applicability for the primary plutonium isotopes for burnups less than 60 GWd/t. Thus, these experiments are beneficial to burnup credit validation efforts.

The fission product solution experiments have been evaluated using only the nuclide-specific *T* parameters. This is because the system-to-system parameters are not currently appropriate for fission products due to the lack of uncertainty data on the fission product cross sections. Also an examination of the *T* values is performed only for ^{149}Sm, since this is the only experiment in the open literature. The *T* values obtained for this experiment indicated that it is highly applicable to ^{149}Sm capture in the series of pin-cell applications for 10–60 GWd/t. This indicates that the fission product solution experiments should be good experiments for validation of the fission products in a cask environment. These fission product solution experiments are valuable in that they allow for the effect of individual fission product cross-section uncertainties on the system k_{eff} to be evaluated separately.

Work is continuing at ORNL to develop a sound basis for the S/U methodology such that it can be used to enable the maximum benefit to be obtained from validation efforts with well-defined fresh-fuel critical experiments.

4. REACTIVITY WORTH EXPERIMENTS

To bypass the difficulties associated with using spent fuel assemblies in critical experiments, spent fuel samples (pellets) and samples doped with individual fission products have been inserted within a fresh fuel system to obtain reactivity worths [16,28]. Sufficient system perturbation to enable an accurate measure of reactivity worth typically requires isotope concentrations much greater than those present in a spent fuel sample. Unless the sample is large enough to provide a significant perturbation to the reference fresh fuel system, the reactivity worth cannot be easily calculated with conventional Monte Carlo codes that are typically used for criticality safety analyses. These experiments can thus provide a means to obtain validation of the reference cross sections used in the criticality analysis, but may be limited for use in typical validation approaches used by storage and transport cask vendors in the United States. The French program for burnup credit relies heavily on the use of reactivity worth measurements in conjunction with chemical assay data to demonstrate that the predicted fission product worths are conservative for their specific design codes and that the uncertainty in the fission product cross sections is encompassed by the uncertainty in the prediction of the fission product inventory.

Reactivity worth measurements [16,28], using small individual fission product samples and oscillation techniques, are more sensitive to the fission product cross-section uncertainties than the fission product solution criticals discussed in the previous section. The oscillation-

type reactivity worth measurements provide more detailed information on the uncertainties in the individual fission product cross-section via evaluation of the uncertainties in the measured worth of each individual fission product. Unfortunately, this type of data can not be readily utilized within the typical code-to-experiment comparison approach to validation historically used in the U.S. for criticality safety analysis. This situation should change if, and when, the S/U methods of Ref. 26 are made available within publicly available production codes.

The U.S. DOE is currently exploring the potential and the benefits of obtaining fission product samples that can be used to perform similar reactivity worth measurements in a critical experiments facility or a research reactor. A DOE Nuclear Energy Research Initiative (NERI) project is also funded to investigate performance of worth experiments in the facilities at Sandia National Laboratories [29]. Current activities are directed on obtaining safety approvals; ideas on the specific type of worth measurements have not been formulated.

Reactivity worth measurements using portions of spent fuel assemblies are being planned as part of the REBUS experimental program [17] and the U.S. Nuclear Regulatory Commission (NRC) is participating in the program. These reactivity worth experiments provide Δk worths that will hopefully be large enough for evaluation with the actual Monte Carlo codes used in the safety analysis. When doing reactivity worth measurements with spent fuel segments, the experiment must either have accompanying destructive assays performed or the fuel design and reactor operation needs to be sufficiently characterized such that an integral-type benchmark (isotopic prediction and reactivity worth prediction) can be performed to provide a combined validation of both the depletion and criticality methodology. Such chemical assay measurements are planned as part of the REBUS program.

5. SUBCRITICAL EXPERIMENTS

The ideal experimental method for assessing the ability of a model to predict the multiplication factor of a system would be to place spent fuel in a cask or cask-like configuration and perform critical experiments. Such experiments are extremely challenging because it is extremely difficult to make even low-burnup spent fuel go critical in a controlled manner without first adding some fresh fuel. This is particularly true under cask conditions where external absorbers (basket material) are present. Spent fuel critical experiments are also complicated by the fact that the fuel samples are highly radioactive, and not as easily manipulated as unirradiated fuel. The expense and complexity of a spent fuel critical is further exacerbated by the need to determine the spent fuel composition by chemical assay (very expensive due to the potentially large number of measurements required) or perform predictive analysis validated against other chemical assay information. At this time, no critical experiment using commercial spent fuel in a cask configuration is known to have been performed, although they have been studied [30].

An alternative to spent fuel critical experiments are subcritical multiplication measurements using spent fuel. Calculations could then be performed to show the capability to match the predicted multiplication factor to the measured value. As with a spent fuel critical experiment, this validation process would require predictions of spent fuel contents prior to the criticality calculations, and would therefore be an integral approach for validation. However, the spectrum should be very similar to that seen in a cask environment and the use of subcritical methods should allow increased flexibility in measuring different configurations. Besides the practical difficulty of handling spent fuel, the performance of subcritical measurements using

spent fuel are made difficult by the practical difficulties with such measurements in a strong radiation field and the need to interpret k_{eff} from the actual measured quantities [31]. The accuracy of subcritical measurements in providing a k_{eff} value for validation is not as good as that provided by a critical experiment, but the advantage of having an actual spent fuel measurement and its potential to validate SNF cross sections (actual measured quantities are very sensitive to cross-section errors) means that such an experiment should be explored if additional measured data are deemed necessary.

6. REACTOR CRITICAL CONFIGURATIONS

A broad database of critical experiments with partially burned and spent fuel exists in the form of critical configurations within an operating reactor environment. During a commercial reactor startup, a controlled approach to criticality is always taken as part of the startup testing. The conditions at the point of criticality are well defined. Startups at the beginning of a fuel cycle contain a mixture of fresh and burned fuel, and often burnable poisons are present; startups occurring late in a fuel cycle are based on a combination of burned and spent fuel, and burnable poisons have typically been depleted.

Like the spent fuel experiments described earlier, the calculational model of a reactor critical configuration will require the prediction of spent fuel inventory for each assembly. Given the size of a commercial reactor combined with the variation in operating conditions during a fuel cycle, the task of estimating spent fuel contents at the time of a startup critical can be rather formidable. However, several reactor critical models have been developed with codes and data typically used for transport and storage cask safety analyses. The results [32 to 33] demonstrate the abilities of such codes to closely predict criticality under reactor conditions.

The advantage, of using reactor criticals in some fashion as part of the validation process, is that they provide measured critical values for systems actually containing SNF. The worth of the spent fuel, with respect to the fresh fuel and the degree to which fission products and boron poison concentrations impact the k_{eff} value, are issues that need to be considered when selecting critical configurations for validation. Just as with all of the other types of experiments discussed to this point, the reactor critical configurations do not provide the same neutronic environment as found in a flooded transport or storage cask. For example, the presence of fresh or partially burned fuel in the reactor and the physical differences between a cask and reactor conditions (e.g., soluble boron versus boron plates) can cause changes in the governing spectrum. Thus, the S/U methodology is also being used to further explore the value of the reactor critical configurations for code and data validation. Three PWR commercial reactor critical state points have been analyzed using the S/U methodology and comparisons made with SNF cask environments. The results indicate that the reactor critical state points have adequate similarity to cask environments. Reactor critical configurations are the only measured information where significant quantities of SNF are used and, from an integral perspective, provide a viable source of validation information for both actinides and fission products.

7. SUMMARY AND DISCUSSION

The purpose of a validation activity is to assess the capability of the codes and data to predict reality. As used in criticality safety, the validation process should be able to demonstrate the bias and the uncertainties associated with the analysis code(s) and data. The overall

uncertainties can arise from uncertainties in the experiments, uncertainties inherent in the code models and data, and uncertainties specific to the user (model approximations, selection of code options, etc.). Currently, approaches used in the United States for criticality safety validation often apply statistical techniques to derive "bounding" estimates of the bias and uncertainty based on the differences between critical ($k_{eff} = 1.0$) and the computed result, together with the spread in the computed results [24]. As demonstrated from the previous subsection, no one set of critical experiments can provide adequate validation for burnup credit applications using this approach alone. Thus, the necessary approach involves utilization of all applicable experimental information in a manner that reasonably ensures that bounding estimates have been determined for the bias and uncertainty. Work at ORNL is focused on quantifying the value of each type of experiment and investigating validation approaches that effectively combine analysis correlations with the types of experimental information.

The DOE Topical report [21] used only fresh fuel critical experiments to validate the analysis of k_{eff} for spent fuel casks and incorporated the nuclide bias and uncertainty by separate adjustment of the predicted SNF isotopics based on comparison with chemical assay data. The limited database of chemical assays and the difficulty and/or uncertainty inherent in the measurement of many of the individual nuclides (most fission products and minor actinides) pose significant obstacles to this phase of the validation process. Even with additional measurements, relatively few data points will ever be available relative to the variety of fuel designs and operating histories to be considered. Thus, given a database with such a limited sample size, it is difficult to obtain meaningful statistics relating predictions as a function of spent fuel characteristics (enrichment, burnup, fuel design, etc.). The critical experiments proposed in Ref. 21 for actinide-only validation include all the nuclides of the actinide-only inventory used in the safety assessment, although the concentrations and combinations are not that observed in spent fuel. To overcome the limitations of the fresh fuel critical experiments relative to their material compositions, use of reactor-critical configurations have been proposed [34] and studied [32,33].

Integral validation involves the use of depletion methods coupled with criticality calculations to determine k_{eff} for a measured system containing SNF (e.g., a spent fuel critical or reactor critical configurations). In practice, this procedure would be applied in spent fuel cask calculations. The perceived limitation with integral validation is that the biases and uncertainties for the depletion approach cannot be separated from those associated with the criticality calculation, and only the net biases and uncertainties in the entire procedure are obtained. Integral validation allows for compensating errors in the depletion approach (i.e., underprediction of a given nuclide's concentration coupled with simultaneous overprediction of a different nuclide's inventory). Thus, it is desirable to ensure the uncertainty estimated for individual nuclides is understood and properly considered in the safety analysis.

Arguments have been made that reactor critical configurations are not appropriate even for integral validation because of differences between reactor conditions and cask conditions. However, other arguments can be made in favor of reactor critical configurations as integral benchmarks, primarily because the design and material composition of the fuel to be placed in a cask is identical to that present in commercial reactors. Thus, the issue with reactor critical configurations is their relevance to cask geometries, whereas the issue with fresh fuel critical experiments is their relevance to the inventory of SNF. Both issues are appropriate when discussing the relevance of reactivity worth measurements.

Initial work with S/U methods [27] have indicated there is some benefit in utilizing each of the types of experimental data; the challenge will be appropriate combination of the data into an adequate validation process. The S/U methods have been used in a preliminary fashion to address the applicability of the fresh fuel critical experiments and reactor critical configurations to cask designs with SNF. Besides defining the applicability of these classes of experiments, S/U analysis can potentially identify deficiencies in the current database and provide a quantitative basis for extension beyond the existing database. The S/U methods also have the potential to assess the benefit of reactivity worth experiments and subcritical experiments and identify potential reductions in the margin of subcriticality allowed by their use. Reactivity worth experiments and subcritical experiments are prime candidates to support additional validation of fission product cross sections, which have typically not had the scrutiny or intense evaluation dedicated to the primary actinides.

Sensitivity/uncertainty methods may also be an approach that can be used to support expansion of the area of applicability for the chemical assays beyond their current limits in terms of burnup and initial enrichment (see Fig. 1). In other words, does 3.6 wt % fuel burned to 40 GWd/t have similar irradiation characteristics as 4.5 wt % fuel burned to 55 GWd/t? If sensitivity methods can be used to quantify the similarity between different SNF characteristics, then S/U methods may be able to establish and justify trends such that interpolation and/or extrapolation techniques can be used to estimate the bias and uncertainty associated with SNF for which there is no chemical assay data.

For actinide-only burnup credit in transport and storage cask applications, it appears there is sufficient experimental data available to enable reasonable validation of the codes and data - although the best process for combining and using the various data may be debated and the bias and uncertainties may not be as low as desired. However, for burnup credit cask applications with fission products, there is a significant paucity of readily available measured data and the validation process acceptable for using the measured data is still in its infancy. A prudent approach to burnup credit validation should involve assay data validation, followed by cross-section validation for the actinides and fission products. The existing mixed-oxide fuel criticals combined with French HTC experiments [25] are believed to be sufficient for actinide-only cross-section validation purposes. Additionally, applications that take credit for fission products need to consider experiments that validate individual fission-product cross sections. Validation may be best accomplished by a combined approach of large-sample, individual fission product worth measurements, such as the French fission product solution criticals [26] or the DOE/NERI experiments [27], and the small-sample, individual doped-fission product worth measurements like those of Refs. 16 and 28. Although more complex to model, commercial reactor critical data provide a valuable source of experimental information for integral validation of the SNF compositions and cross sections and the effect of neutronic interaction between assemblies. Work needs to continue to best determine how to incorporate this information into the validation process for use in transport and storage cask licensing.

Utilizing the negative reactivity credit from fission products continues to be a goal for optimum use of burnup credit in the United States. One approach that has been offered [35] is to quantify two independent factors to account for the effects of isotopic prediction inaccuracies and isotope cross-section inaccuracies (derived from reactivity worth measurements). The product of these two factors and the predicted worth values in the cask configuration gives an estimate of the “guaranteed” fission-product worth in the cask application of interest. Efforts are underway to quantify these effects for an example application and investigate the methodology for use in transport and storage applications.

Similarly the U.S. DOE is seeking burnup credit that includes fission product credit in their efforts to license the permanent SNF repository [36]. The DOE efforts rely heavily on the use of reactor critical configurations with fresh fuel criticals and chemical assay data being used as appropriate.

ACKNOWLEDGEMENTS

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PARAMETERS AFFECTING BURNUP CREDIT

EVALUATION OF AXIAL AND HORIZONTAL BURNUP PROFILES

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Abstract

This paper describes methods used to evaluate the impact of axial and horizontal burnup shapes on the loading criterion of a spent fuel management system of interest. Different modeling assumptions all being made to ensure that the burnup credit criticality analysis of the spent fuel management system is based on a fuel irradiation history which leads to a bounding loading criterion are compared to each other.

1. INTRODUCTION

The difference between the neutron multiplication factor of an axial burnup shape and the neutron multiplication factor obtained by assuming a uniform distribution of the averaged burnup of the shape is known as the ”end effect”. It is well known that this difference can be positive for averaged burnups greater than 15 MWd/kg U, cf. [1–3] for instance. If positive, this difference has to be enveloped by the fuel loading criterion being applicable to the spent fuel management system of interest. To attain a sufficiently high confidence level, that the highest end effect that may occur under normal core operating conditions is covered by the loading curve, a sufficiently large ensemble of axial shapes covering the parameters affecting the form of the shapes has to be evaluated [3]. Parameters affecting the form of the shapes are for instance partial control rod insertion during operating cycle, presence of burnable poisons, extended low power operation, and fuel assembly position inside core.

The fuel position inside core in particular can result in significant horizontal variation in burnup [2]. As shown in [3] this variation might affect the loading criterion of the spent fuel management system of interest.

In the paper on hand impacts of variations in the form of axial and horizontal burnup shapes on the loading criterion are analyzed. In addition different modeling assumptions all being made to ensure that a burnup credit criticality analysis is based on a fuel irradiation history which leads to a bounding loading criterion are compared to each other.

2. PARAMETERS AFFECTING AXIAL BURNUP SHAPES

Figure 1 shows the result of the evaluation of a sample of 708 axial burnup shapes which Siemens AG Power Generation Group (KWU) received from Nuclear Power Plant Neckarwestheim I (NPP GKN I), Germany, on February 09, 1998. These shapes were passed on to OECD NEA Data Bank, and NEA Data Bank has distributed these shapes on CD on June 24, 1999, [4].

NPP GKN I is a 2500 MW_{th} plant with 177 fuel assemblies of the 15x15-20 type having an active length of 298.5 cm.

All the shapes analyzed refer to EOCs. They are derived from in-core 3D power density distribution measurements based on flux measurements performed every fourteenth day.

These measurements are performed with high spatial resolution: At 24 fuel assembly positions the flux data are monitored at 30 equidistant axial nodes. Thus, one has a total of 720 measuring points. The measurements are performed with the aid of the Siemens/KWU's Aeroball System [5] which has the advantage of monitoring simultaneously all the axial nodes as indicated in Figure 2. The high spatial resolution and the high frequency of the measurement campaigns as well as the accuracy of the measurement device [5] result in shapes of outstanding quality. For instance, the spatial resolution suffices to discriminate the flux dips caused by the presence of the spacer grids. What regards the end effect, the presence of spacer grids in the ends of the fuel zone should attract one's attention.

In Figure 1 all the 708 shapes are collapsed to a normalized shape: The solid line gives the sample mean

$$\hat{\alpha}_i = \frac{1}{N}\sum_{j=1}^{N}\alpha_{ij} = \frac{1}{N}\sum_{j=1,}^{N}\frac{B_{ij}}{\hat{B}_j}\,,\ \mathrm{i} = 1, ..., 30\,,\ \mathrm{N} = 708\,, \qquad (1.1)$$

of the ratios α_{ij} of the burnup B_{ij} at node i of the j-th axial shape to the average burnup

$$\hat{B}_j = \frac{1}{30}\sum_{i=1}^{30} B_{ij} \qquad (1.2)$$

of this axial shape, and the dashed line gives the square root of the sample variance, i.e.,

$$\hat{\sigma}_i \equiv \sqrt{\frac{1}{N-1}\sum_{j=1}^{N}\left(\alpha_{ij} - \hat{\alpha}_i\right)^2}\ ,\ \mathrm{i} = 1, ..., 30\,,\ \mathrm{N} = 708\,. \qquad (1.3)$$

The results obtained for the expressions (1.1) and (1.3) reflect all the effects determining axial burnup shapes:

The mean values (1.1) (solid line in Figure 1) represent the asymmetry of the axial shapes due to the higher moderator density in the lower half of the core (node 30 is close to the bottom end of the fuel zone, node 1 is at the top end of the fuel zone).

The asymmetric component of axial shapes is strongly dependent on the average burnup of the fuel assembly and strongly affected by control rod movements as well as (re-)load strategies used. All these history effects are reflected by the sample variance (1.3):

1. In a fresh fuel assembly the axial power distribution is more cosine-shaped thus resulting in high ratios α_{ij} in the central section of the active zone and low ratios α_{ij} at the ends of the active zone (Figures 3 and 5 show for example axial power distributions at the begin of the first and the second cycle, and Figure 7 shows axial power distributions of the NPP GKN I at the begin of the nineteenth cycle).
2. With increasing average burnup of the fuel assembly the axial power distribution flattens out thus resulting in an increase of the ratios α_{ij} at the ends of the fuel zone and leading to a decrease of the ratios α_{ij} in the central section of the active zone zone (Figures 4 and 6 show axial power distributions at the end of the first and the second cycle, and Figure 8 shows axial power distributions of NPP GKN I at 196 EFPD (effective full power days) of the nineteenth cycle).

So therefore, the ratios α_{ij} for the nodes at the fuel zone ends increase with increasing average burnup whereas the ratios α_{ij} for the nodes of the central section of the fuel zone decrease with increasing average burnup. That's exactly the way how the analyzed shapes behave as can be seen from Figures 10 through 24:

1. Figures 10 through 14 and Figures 15 through 19 show the increase of the ratios α_{ij} with increasing burnup for the fuel zone end nodes 1 and 30 respectively, whereas
2. Figures 20 through 24 show the decrease of the ratios α_{ij} with increasing burnup for the more centrally located node 20.

The burnup groups used in these figures are derived from Figure 9.

Due to the fact that the asymmetric component of the axial shapes is strongly affected by control rod movements as well as reload patterns used (determining the interactions between fresh fuel assemblies and fuel assemblies with different burnup shapes at BOC) the variance (1.3) is considerably greater at the top end of the fuel zone than at the bottom end of this zone (i.e., the distribution of the ratios α_{ij} is broader in the range of the top end zone than in the range of the bottom zone, cp. Figure 10 with Figure 15).

The higher variance (1.3) at the top nodes causes, however, also a higher variance at the nodes between 17 and 25: Low ratios α_{ij} at the top nodes mean strong underpeaking of the axial power distribution and hence higher ratios α_{ij} at the lower half of the active zone. On the other hand, higher ratios α_{ij} at the top nodes mean less asymmetry, i.e., lower ratios α_{ij} at the lower half of the fuel zone. For these reasons the distribution of the ratios α_{ij} obtained for node 20 for example (cp. Figure 20) is broader than for the node 9 for example (cp. Figure 25).

As can be seen from Figures 11 and 16, the distributions of the ratios α_{ij} at the top node (node 1) and the bottom node (node 30) are very broad for the burnup group No. 1 in particular (average burnups between 8 and 22 MWd/kg U), and consequently the variance (1.3) is greater for the bottom node than for the more centrally located nodes (cp. Figure 1).

In summary, due to the measurement methods applied the evaluated axial shapes are of outstanding quality (compared to shapes one usually gets from core-following measurement data). It is not surprising, therefore, that the results (1.1) and (1.3) as well as Figures 10 through 25 are consistent with what is to be expected due to reactor physics.

2. EVALUATION OF AXIAL BURNUP PROFILES

2.1. Necessity of Formulating the Evaluation of Axial Burnup Profiles as a Decision Problem

Due to all the history effects affecting the axial shapes one gets for each axial height z of the fuel zone a burnup dependent statistic on the ratio $\alpha(z,B_{av})$ of the normalized burnup (B_{av} denotes the averaged burnup). The distributions shown in Figures 10, 15, 20, and 25 for example are samples on this statistic, but the true distribution of the ratio $\alpha(z,B_{av})$ remains unknown until the plant of interest is finally shut down – and this goes for all the plants of interest. So therefore, having a statistic on the ratios $\alpha(z,B_{av})$ which may be biased with respect to the true distribution of $\alpha(z,B_{av})$ one has to decide whether or not the highest end

effect that may occur under normal core operating conditions is covered by the loading curve of the spent fuel management system of interest. One needs a decision theory, therefore, which is closely tied to the methods of statistics. This means that one has to analyze a sufficiently large number of observed axial shapes in order to construct a sufficiently high confidence level that the highest end effect that may occur under normal core operating conditions is covered by the loading curve of the spent fuel management system of interest.

In order to be able to analyze a large number of axial shapes curves representing the neutron multiplication factor of the spent fuel management system of interest as a function of the axially uniformly distributed burnup at given initial enrichments are used as ”calibration curves”, as exemplified in Figure 26. As illustrated in this figure, the difference Δk between the neutron multiplication factor obtained with an axial burnup shape and the neutron multiplication factor obtained by assuming the average burnup of the shape uniformly distributed is represented by the difference between the average burnup and the so-called ”equivalent uniform burnup” which is the uniformly distributed burnup that leads to the same neutron multiplication factor as obtained with the axial shape. As indicated in Figure 26, this equivalent uniform burnup is obtained by comparing the neutron multiplication factor obtained with the axial shape to the calibration curve taking account of course of all the statistical uncertainties that might be involved in such a comparison [3]. Due to these statistical uncertainties each axial shape analyzed in this way is represented by a bar in a diagram which shows the equivalent uniform burnup of a shape in correlation with the average burnup of this shape. Figure 27 shows such a diagram, and – as can be seen from this figure – a big lot of analyzed axial shapes is represented in that diagram. From this big lot of results a correlation between equivalent uniform and average burnup can be derived which represents the end effect (i.e., the reactivity effect due to the fact that the axial distribution of burnup is non-uniform) in an enveloping manner. This correlation is represented by the solid line in Figure 27 (the dashed line corresponds to zero end effect – in that case is the equivalent uniform burnup equal to the average burnup). As can be seen from Figure 27, this correlation is defined in fact in an enveloping manner: All the bars representing the axial shapes analyzed are above the solid line representing the correlation derived.

The correlation of equivalent uniform burnup to average burnup can be used as set forth below:

1. First, with this correlation one is able to derive a loading curve from the calibration curves. As indicated in Figure 28, once a minimum required uniform burnup is determined, the average burnup which covers the end effect can be calculated with the aid of the correlation - and the minimum required uniform burnup is obtained in fact by comparing the upper 95%/95% tolerance limit of the calibration curve with the maximum neutron multiplication factor allowed (cp. Figure 29).
2. Secondly, with the aid of the correlation one is able to made a decision whether or not axial burnup profiles obtained later meet the loading curve. An axial profile is acceptable only then if the related equivalent uniform burnup is not beneath the correlation curve.

Text cont. on page 206.

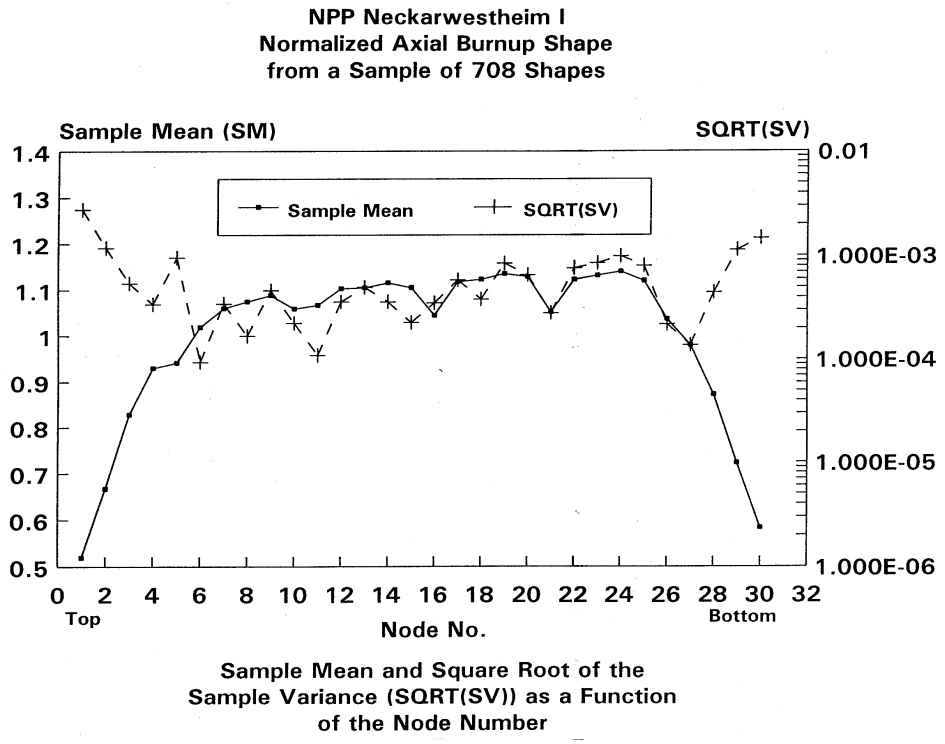

FIG. 1: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4].

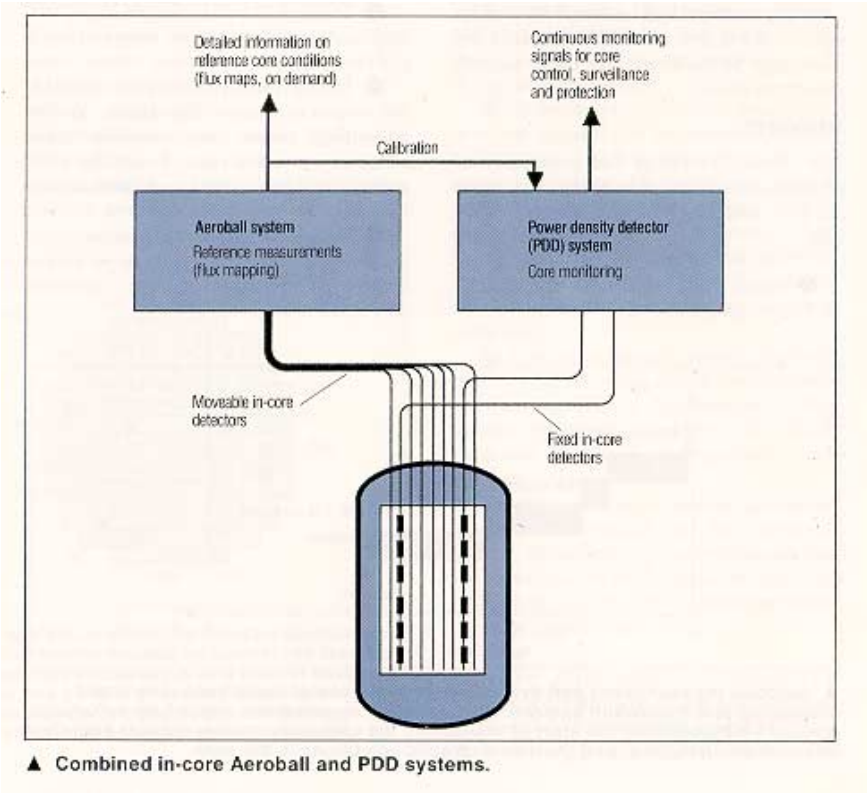

FIG. 2: Illustration of the Siemens/KWU's Aeroball System [5].

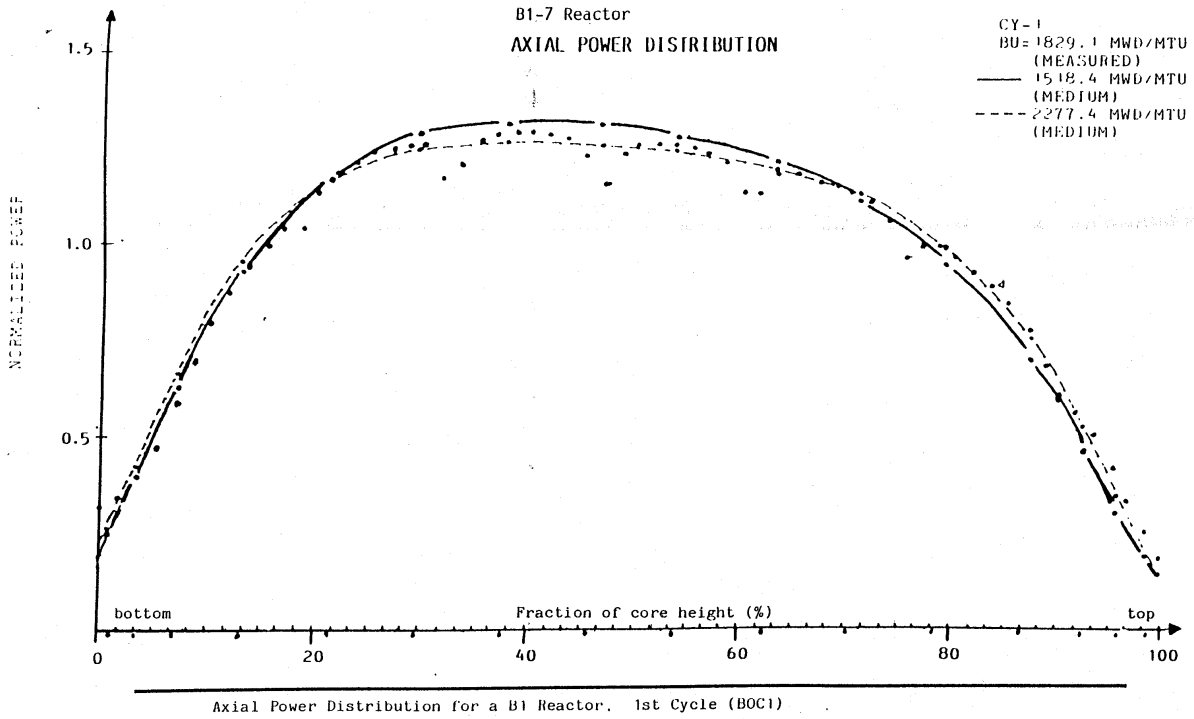

FIG. 3: Axial Power Distribution Typical of the Begin of the First Cycle [9].

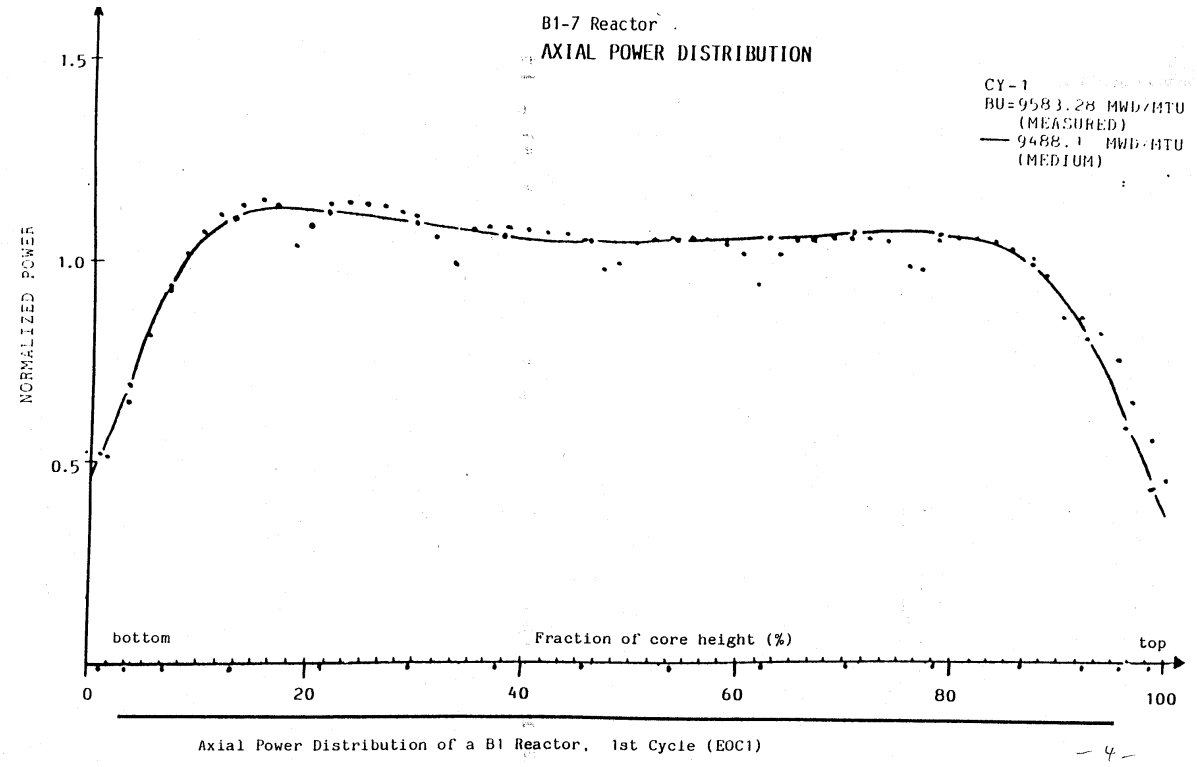

FIG. 4: Axial Power Distribution Typical of the End of the First Cycle [9].

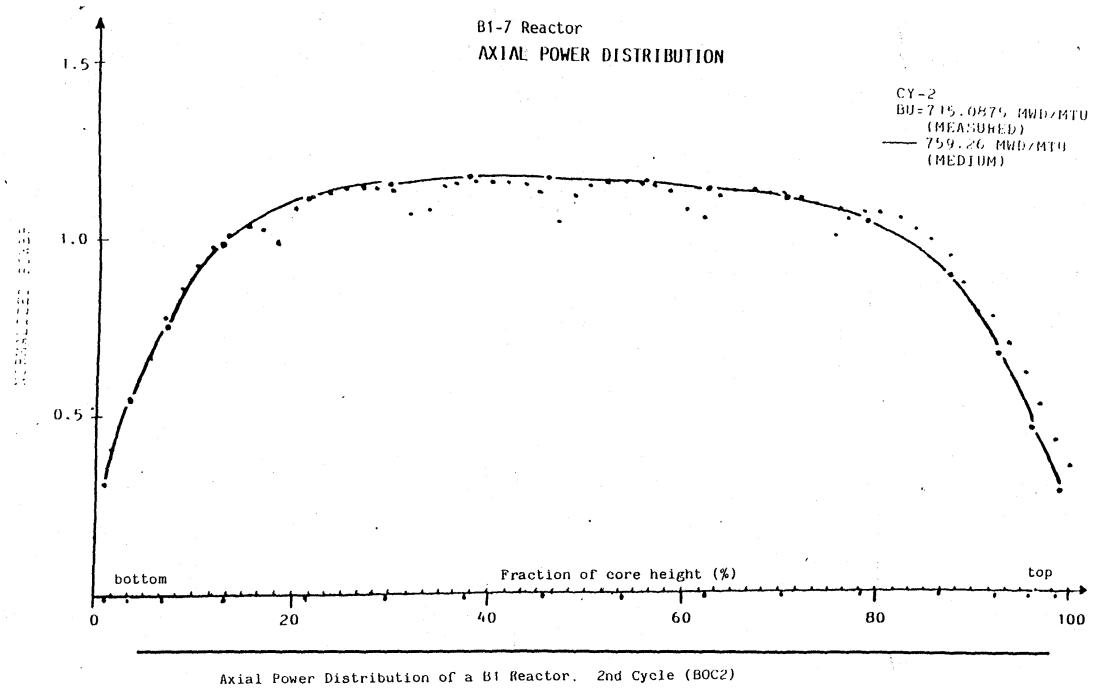

FIG. 5: Axial Power Distribution Typical of the Begin of the Second Cycle [9].

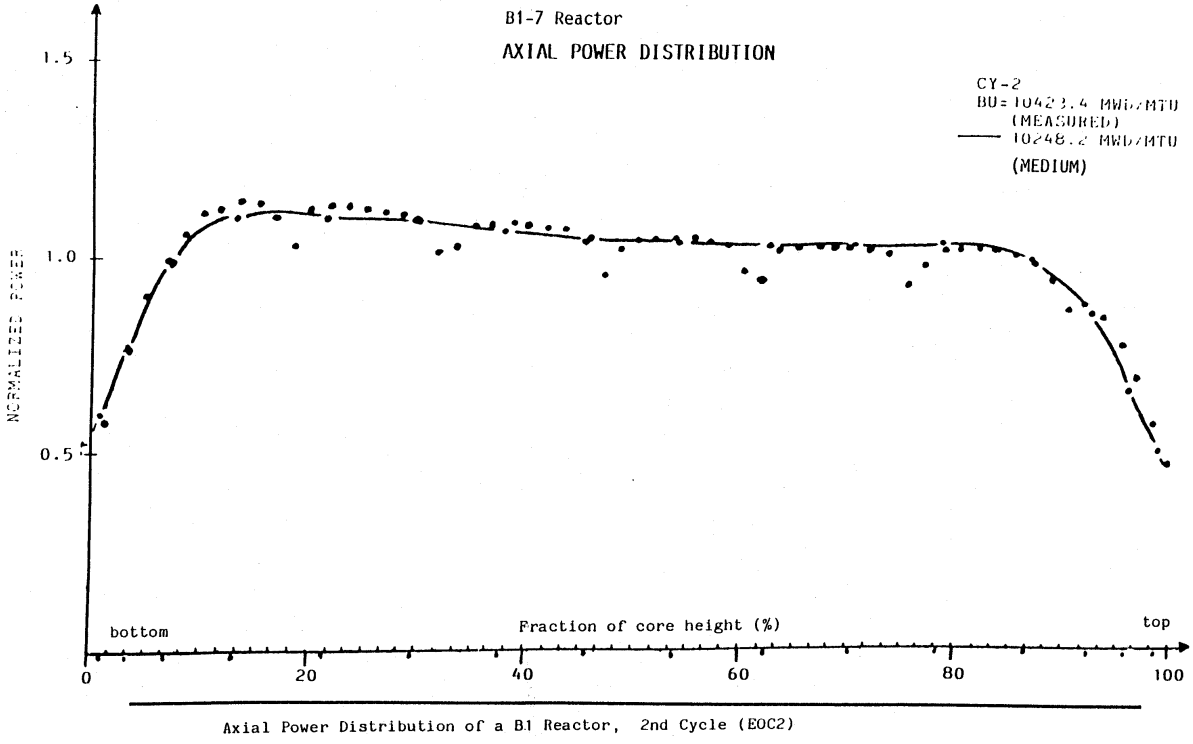

FIG. 6: Axial Power Distribution Typical of the End of the Second Cycle [9].

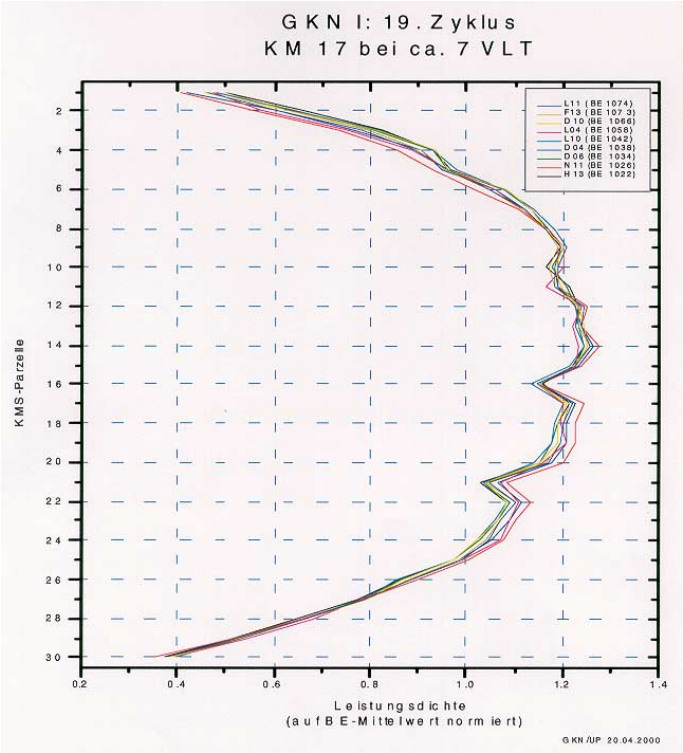

FIG. 7: NPP Neckarwestheim I: Axial Power Distributions at the Begin of the Nineteenth Cycle (at 7 EFPD).

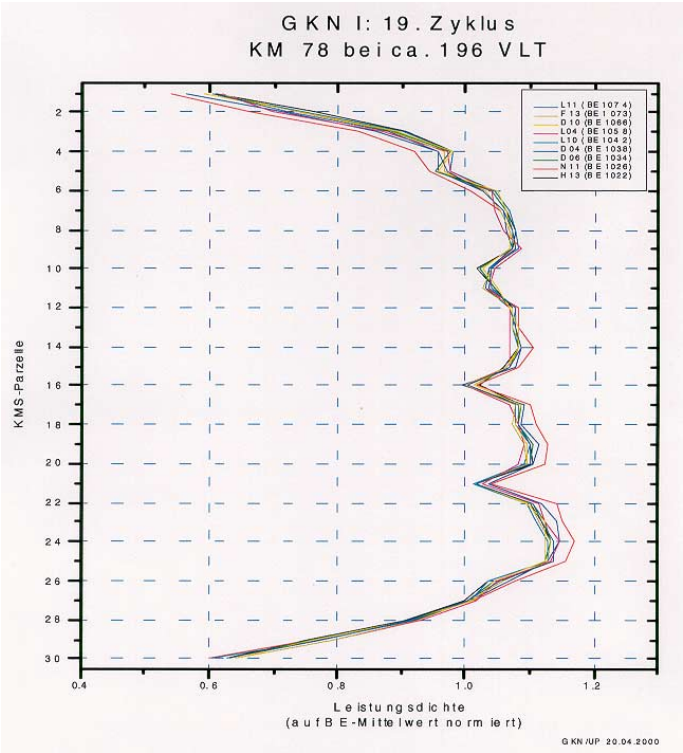

FIG. 8: NPP Neckarwestheim I: Axial Power Distributions at 196 EFPD of the Nineteenth Cycle.

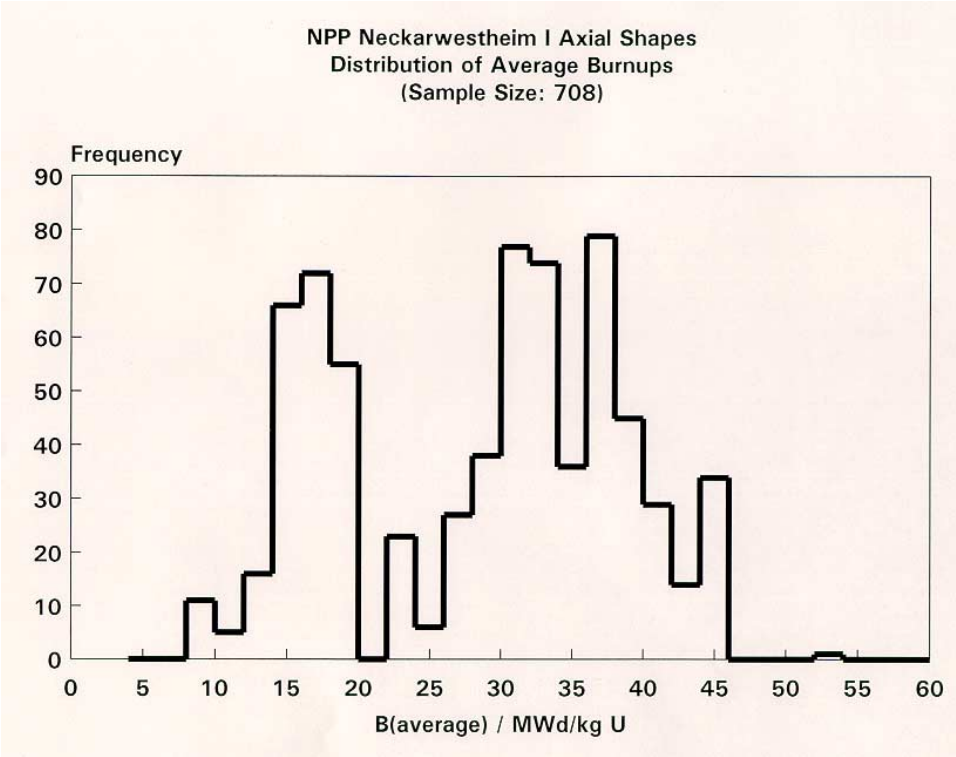

FIG. 9: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Average Burnups of the Shapes.

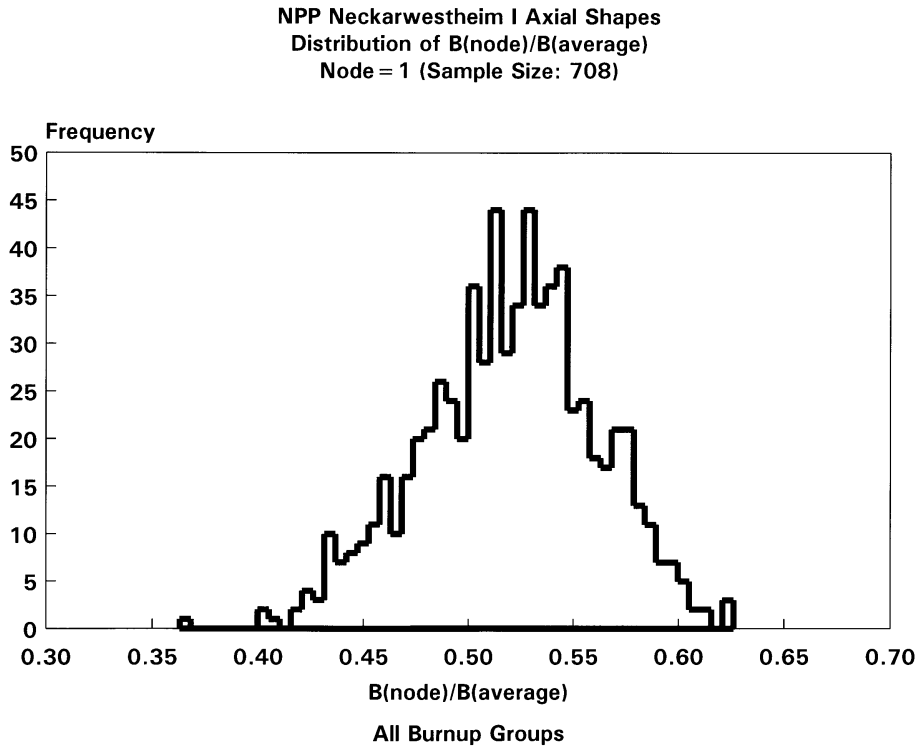

FIG. 10: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Ratios (1.1) at Node 1.

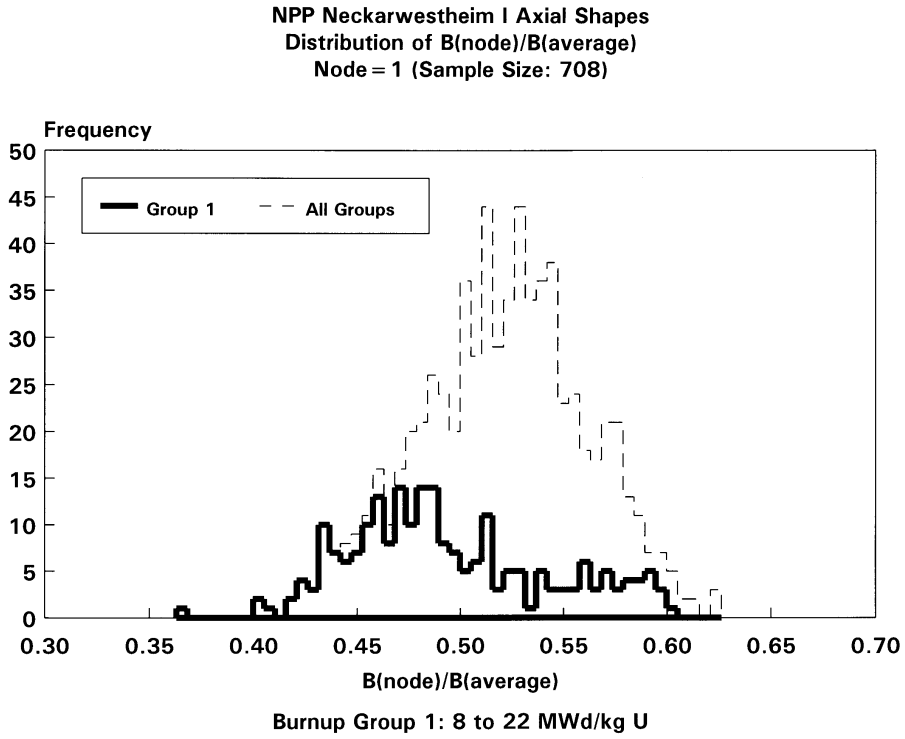

FIG. 11: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Ratios (1.1) at Node 1.

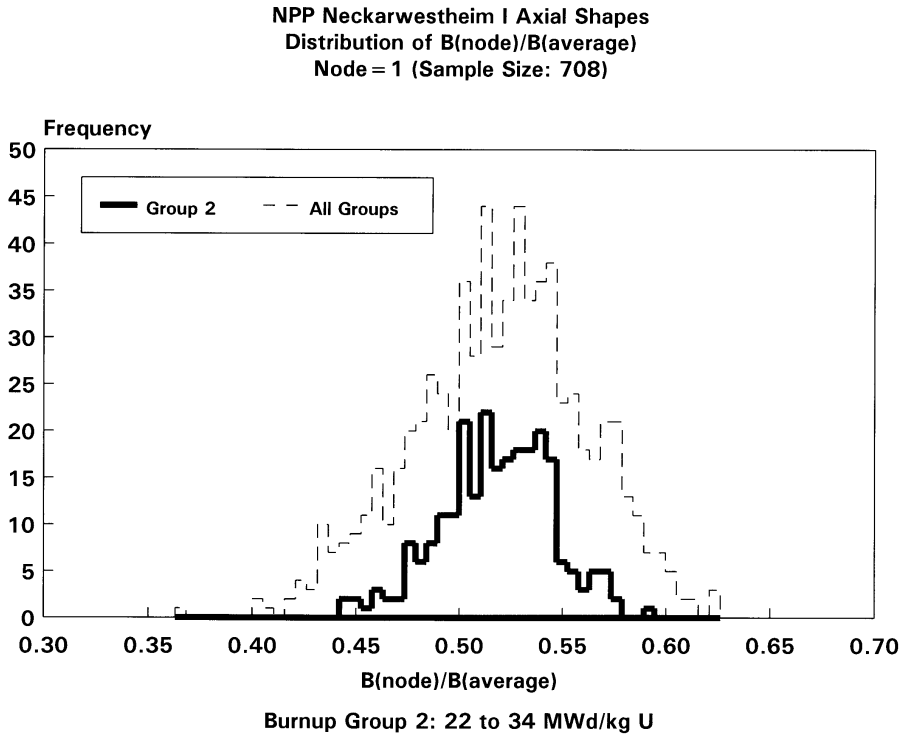

FIG. 12: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Ratios (1.1) at Node 1.

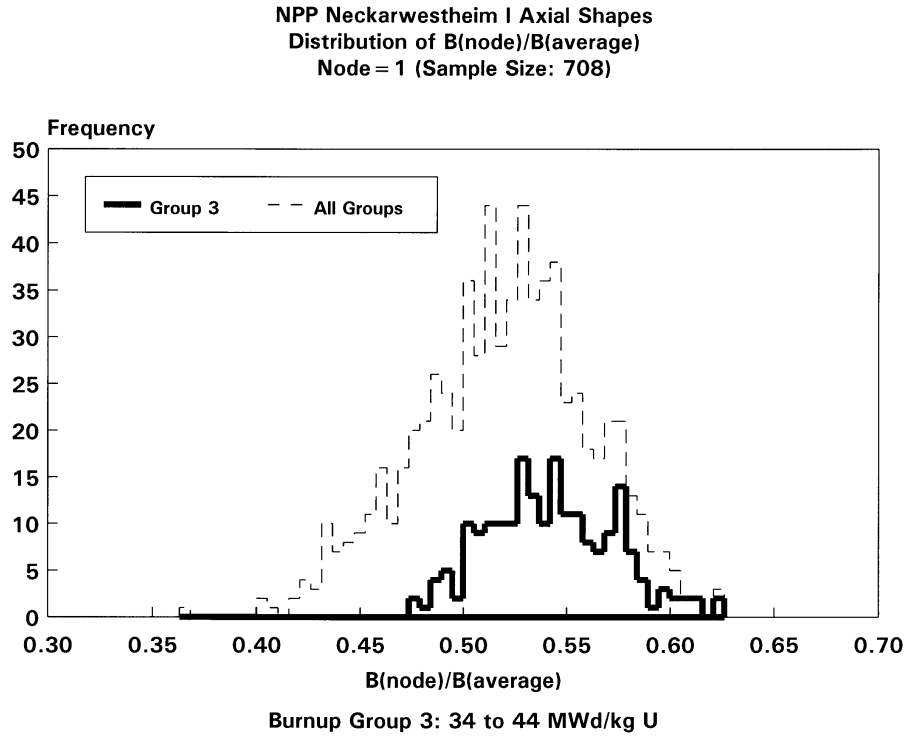

FIG. 13: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Ratios (1.1) at Node 1.

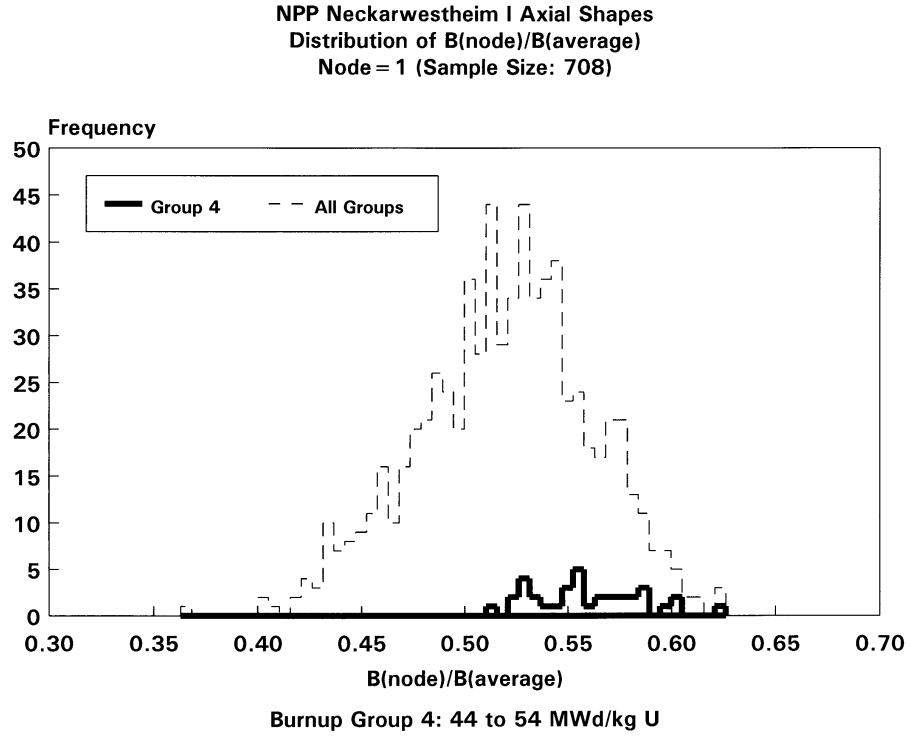

FIG. 14: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Ratios (1.1) at Node 1.

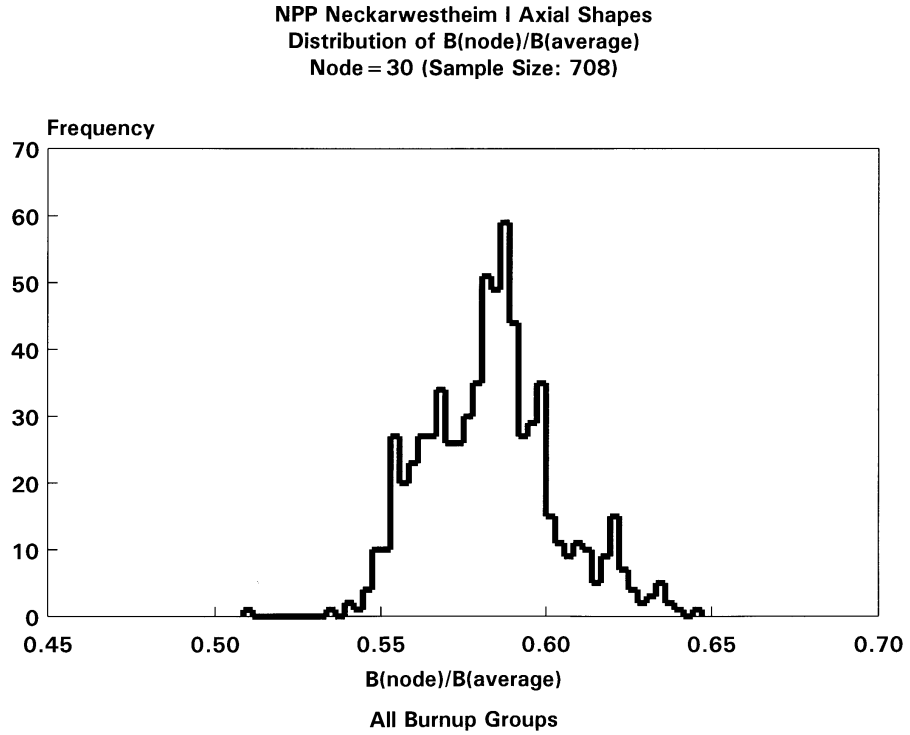

FIG. 15: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Ratios (1.1) at Node 30.

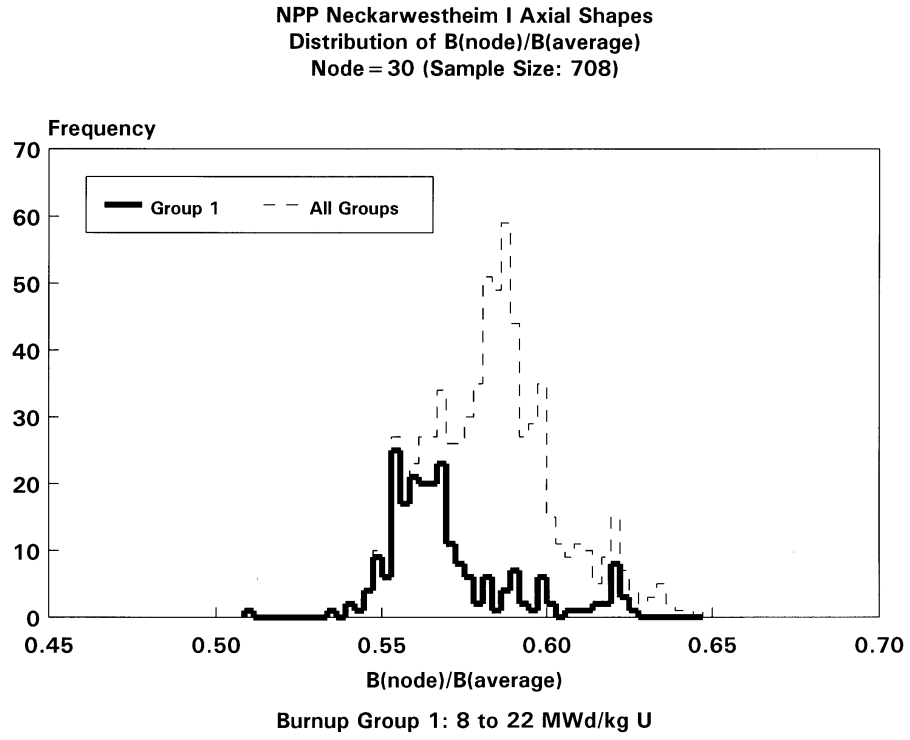

FIG. 16: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Ratios (1.1) at Node 30.

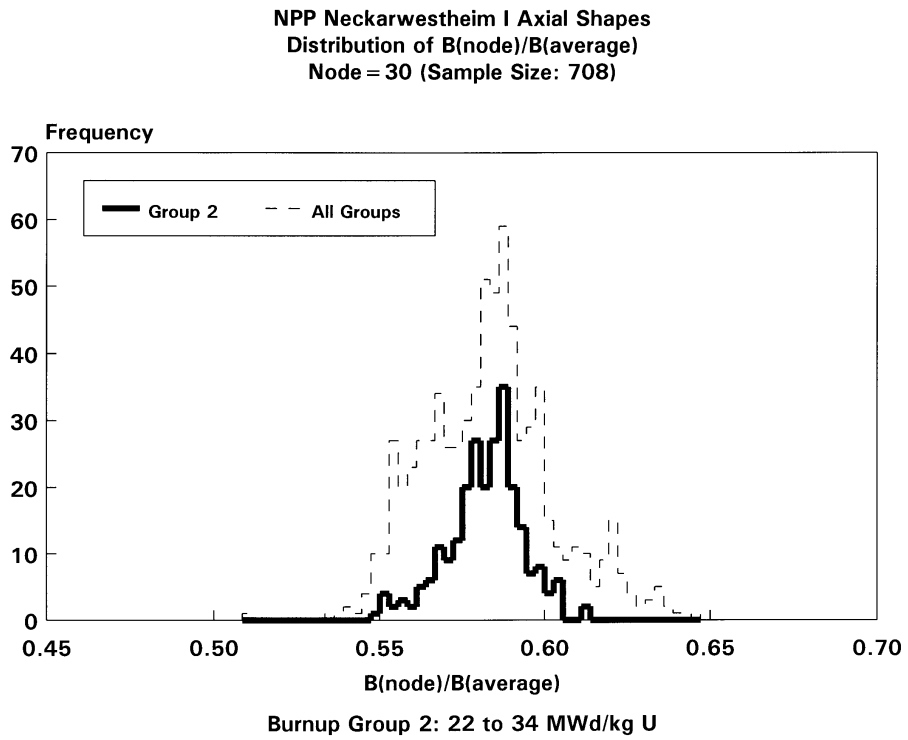

FIG. 17: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Ratios (1.1) at Node 30.

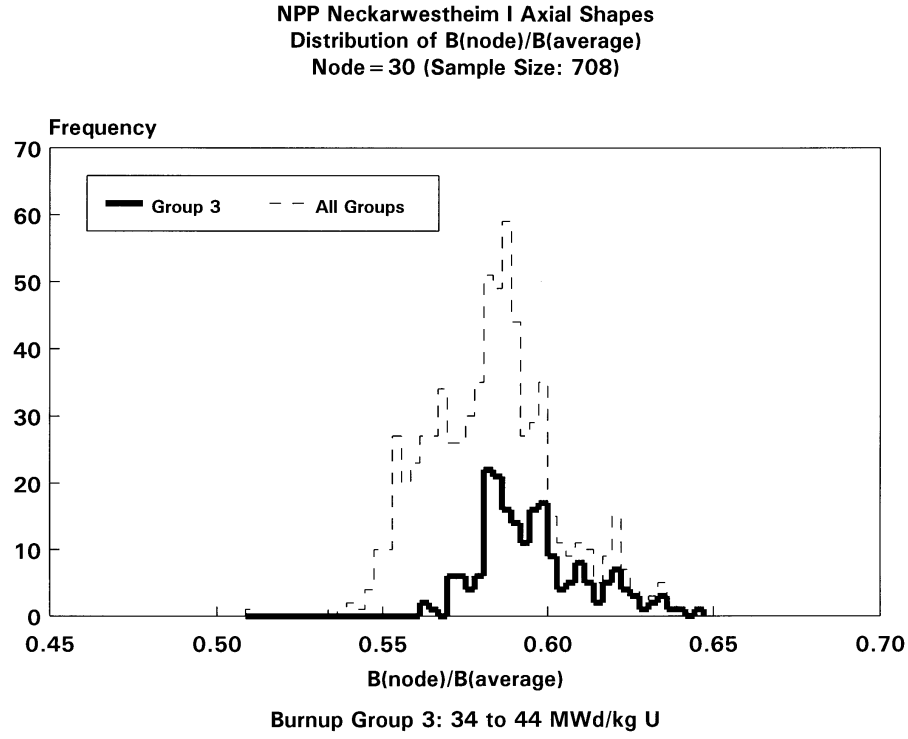

FIG. 18: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Ratios (1.1) at Node 30.

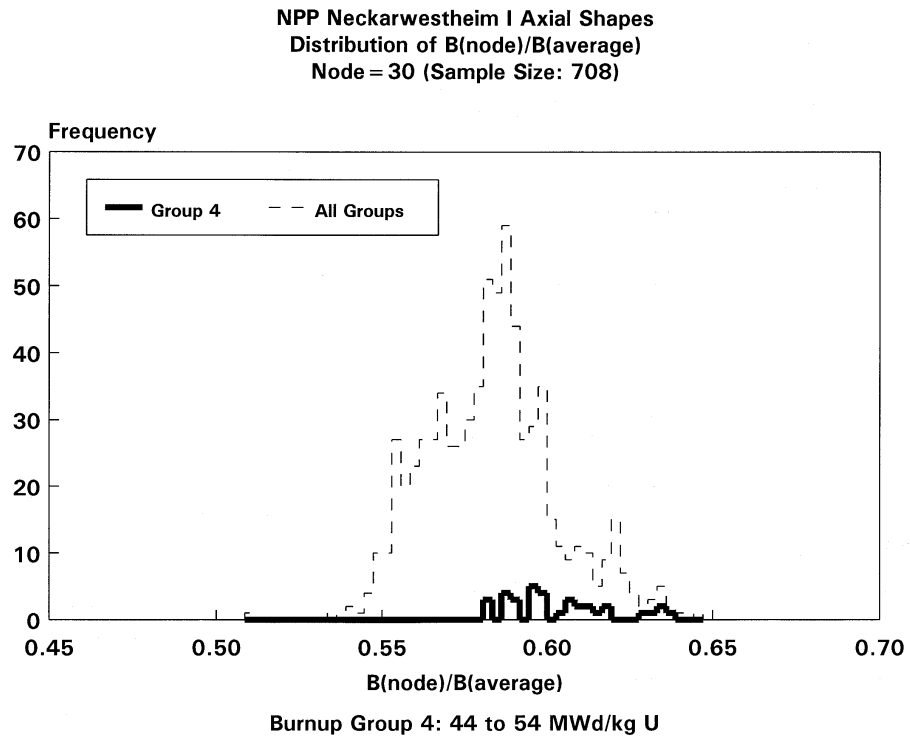

FIG. 19: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Ratios (1.1) at Node 30.

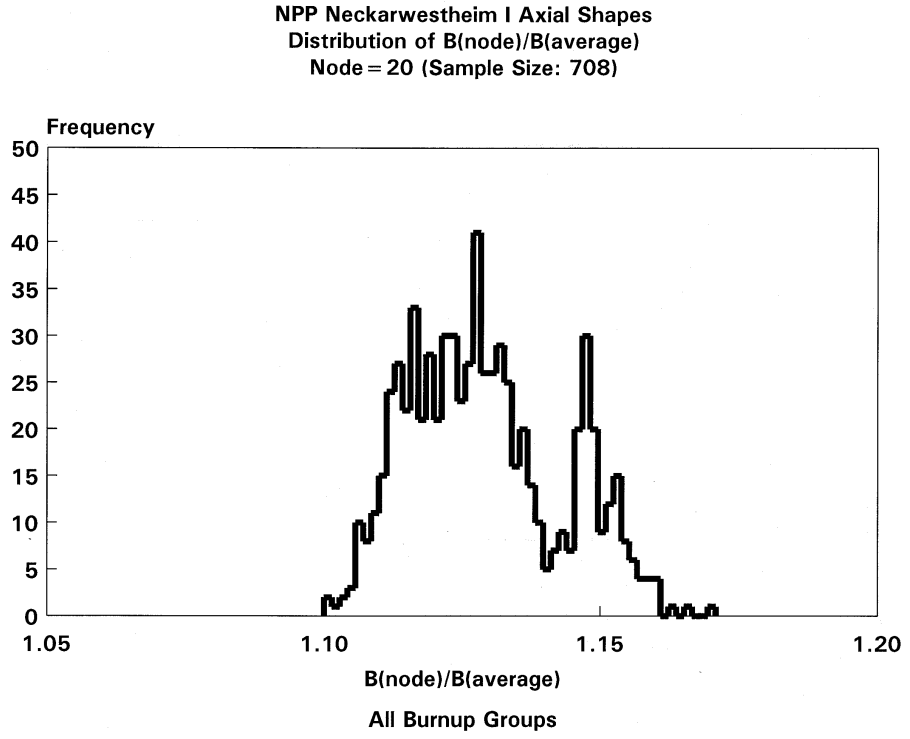

FIG. 20: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Ratios (1.1) at Node 20.

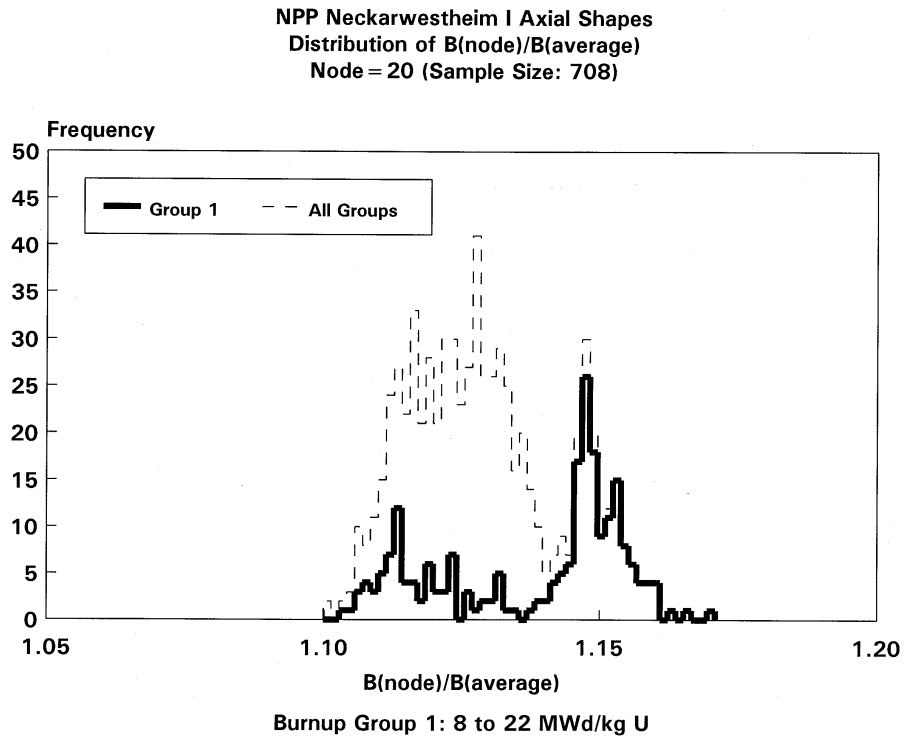

FIG. 21: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Ratios (1.1) at Node 20.

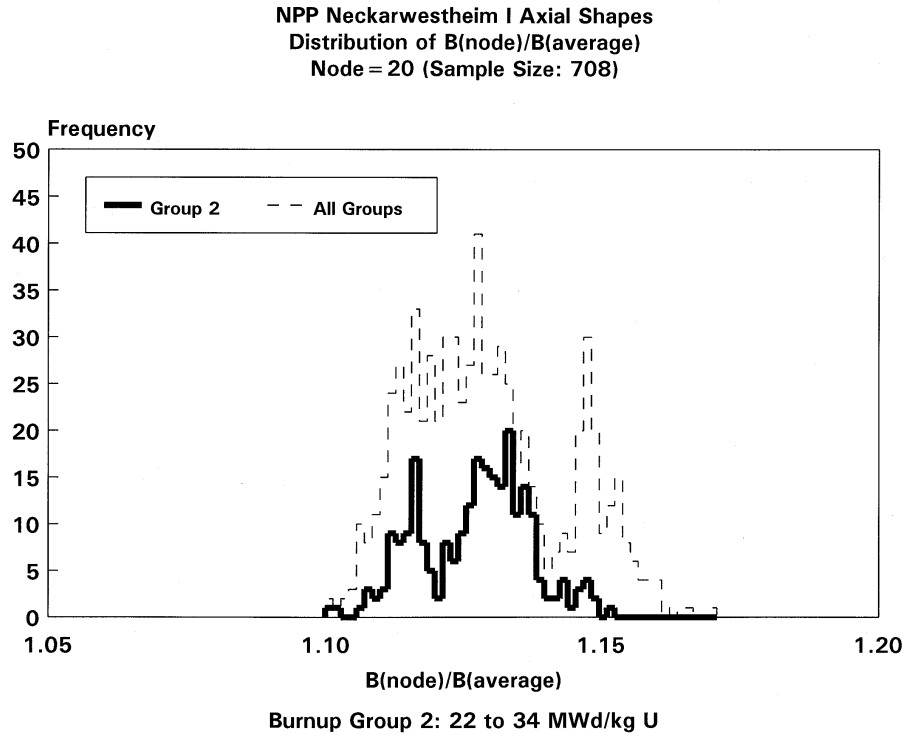

FIG. 22: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Ratios (1.1) at Node 20.

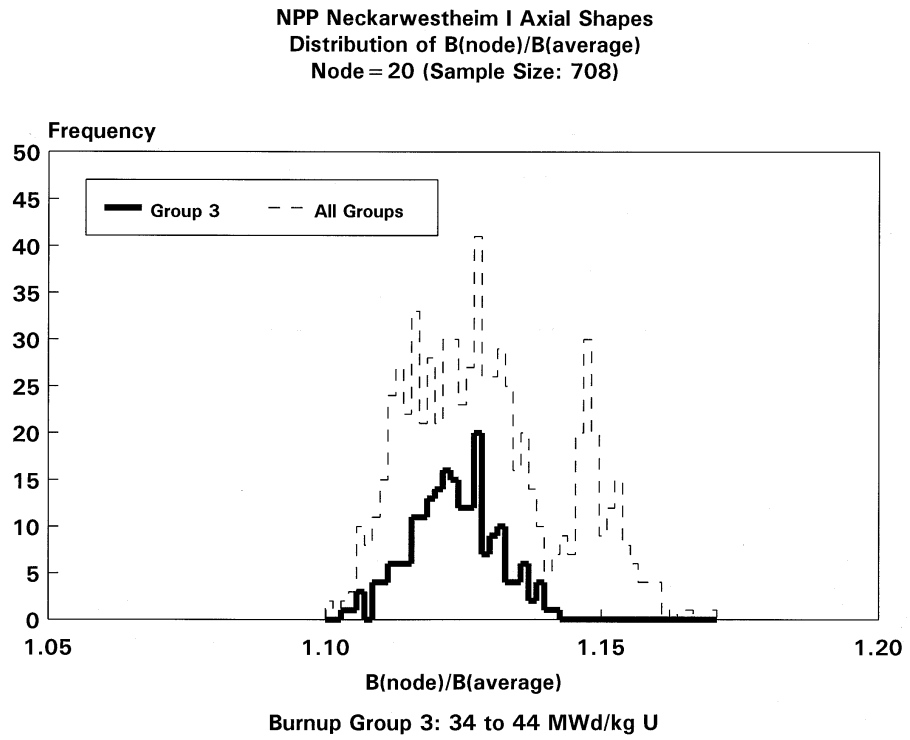

FIG. 23: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Ratios (1.1) at Node 20.

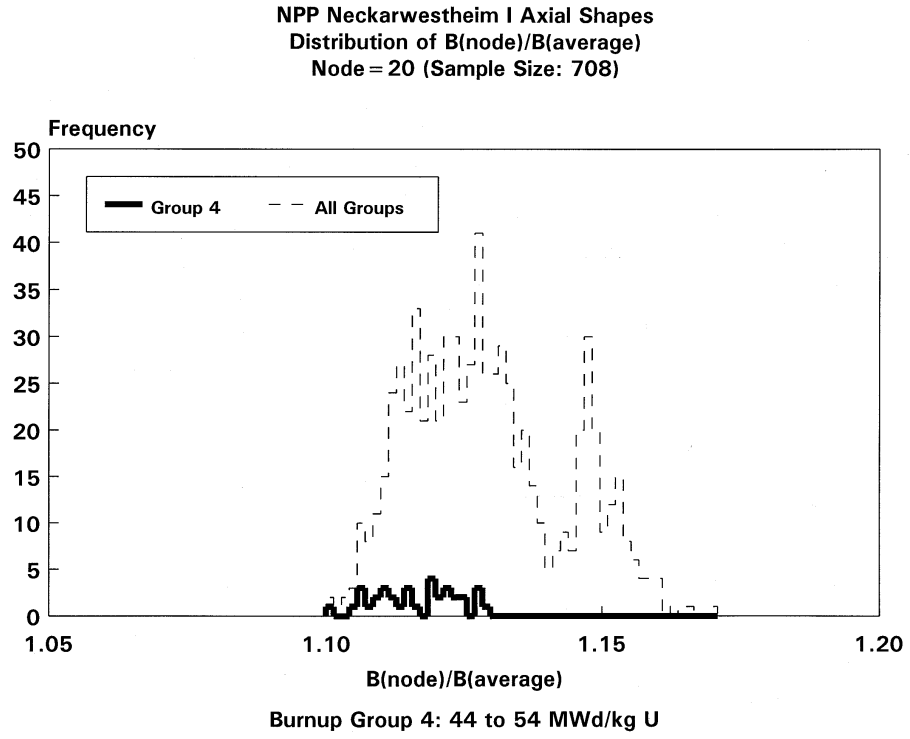

FIG. 24: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Ratios (1.1) at Node 20.

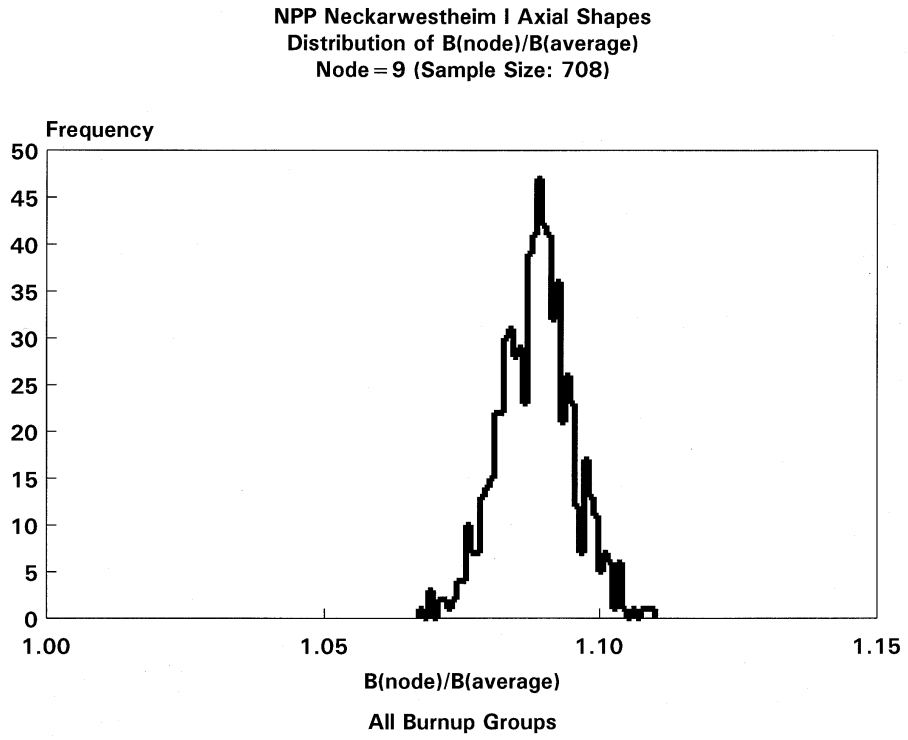

FIG. 25: Evaluation of a Sample of 708 Axial Burnup Shapes given in Reference [4]: Distribution of the Ratios (1.1) at Node 9.

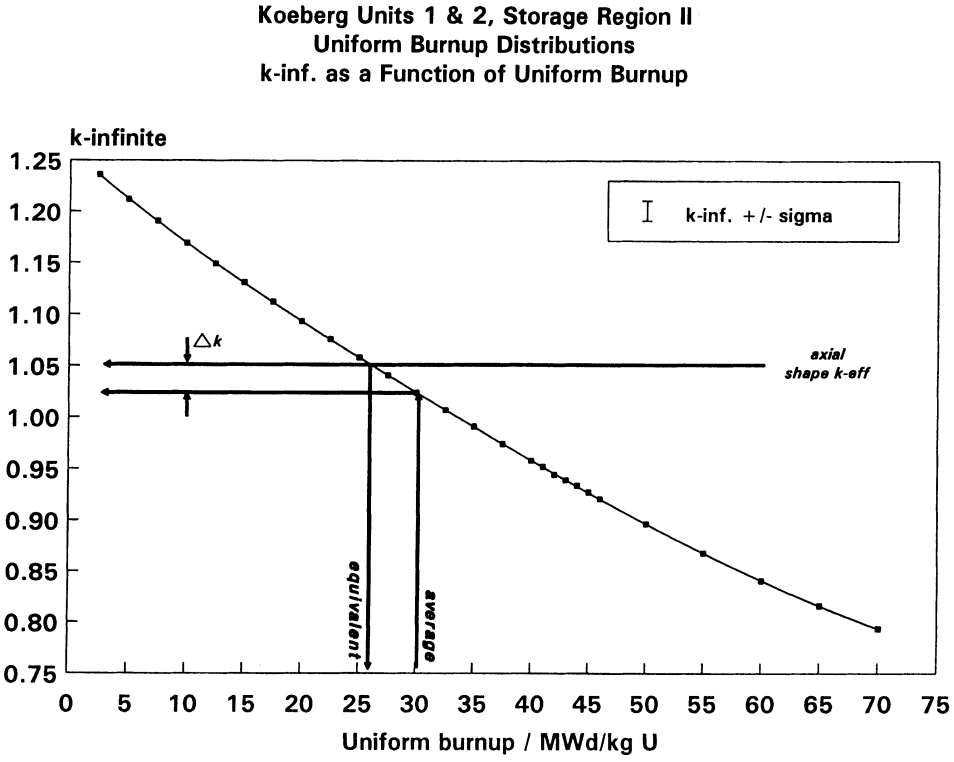

FIG. 26: Evaluation of Axial Burnup Profiles: Calibration Curve.

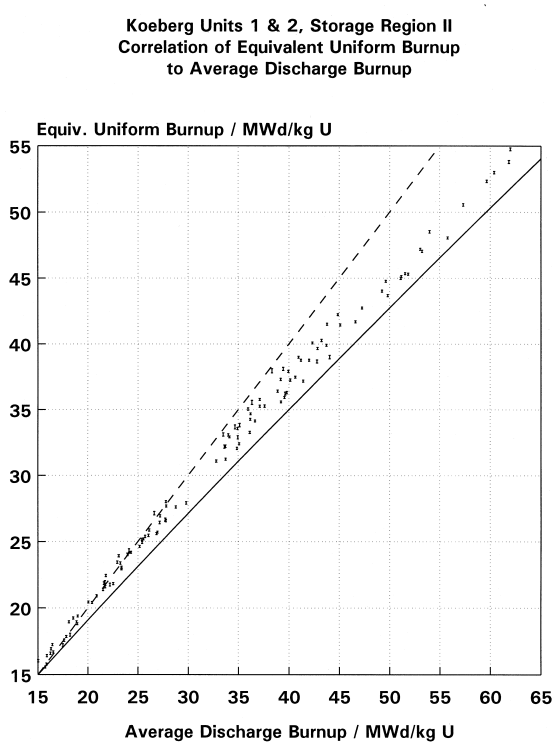

FIG. 27: Evaluation of Axial Burnup Profiles: Determination of the Correlation of Equivalent Uniform to Average Burnup.

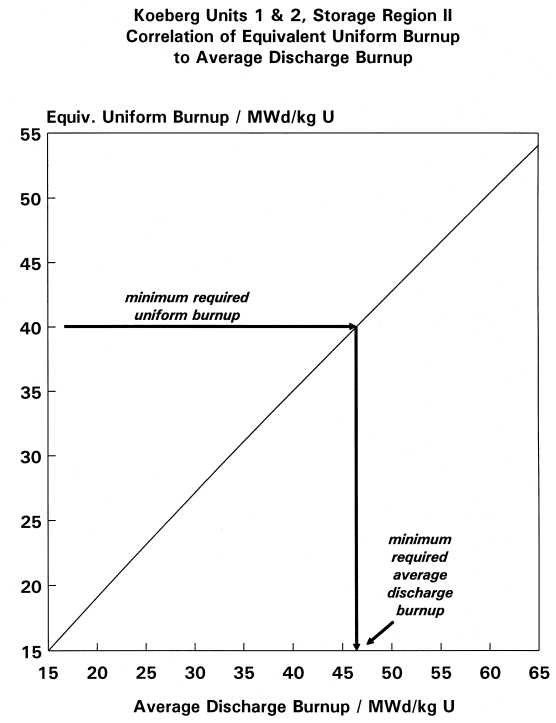

FIG. 28: Evaluation of Axial Burnup Profiles: Application of the Correlation of Equivalent Uniform to Average Burnup.

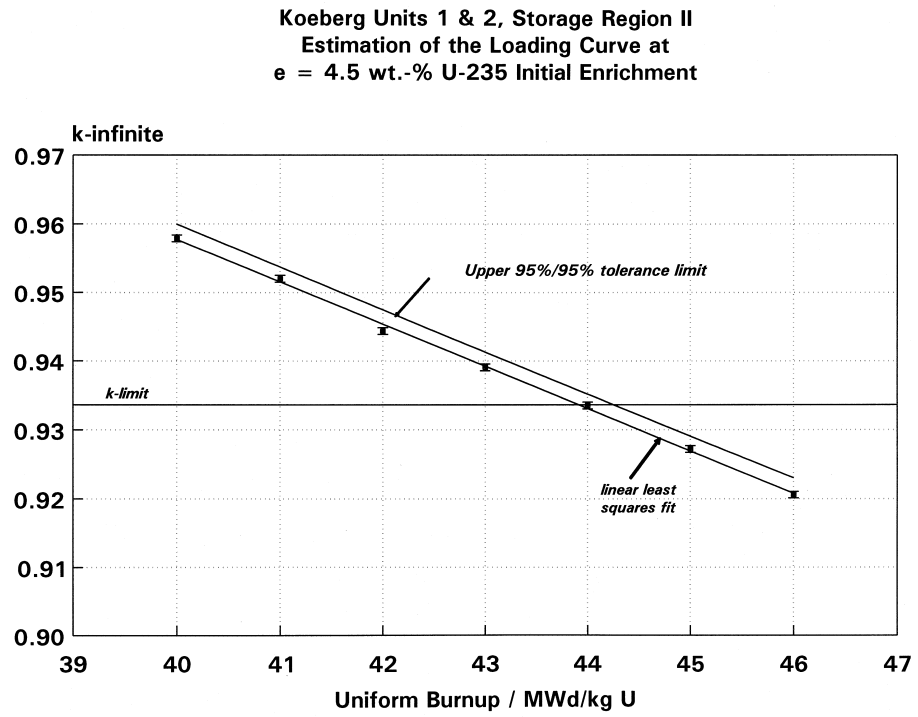

FIG. 29: Determination of a Loading Curve: Estimation of the Minimum Required Uniform Burnup at given Initial Enrichment of the Fuel.

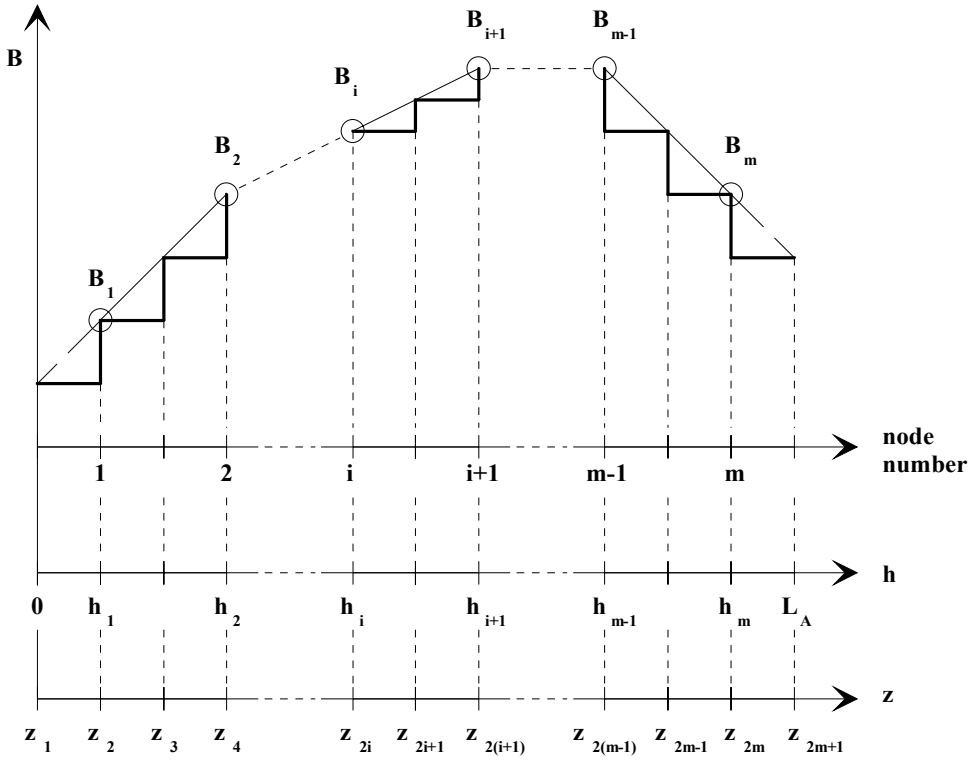

FIG. 30: Illustration of the First Step in Constructing an Axial Burnup Model Distribution (Example n = 1)

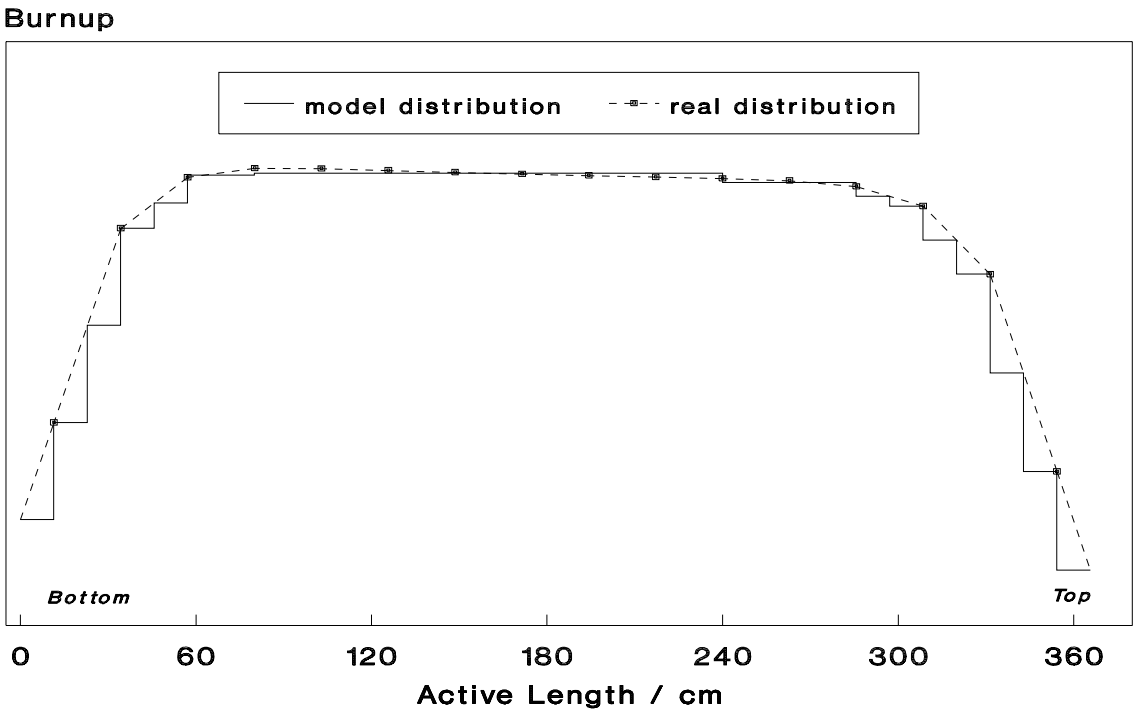

FIG. 31: Example for an Axial Burnup Model Distribution (Obtained after Collapsing Neighboring Steps together which are Similar in Height, cp. Figure 30)

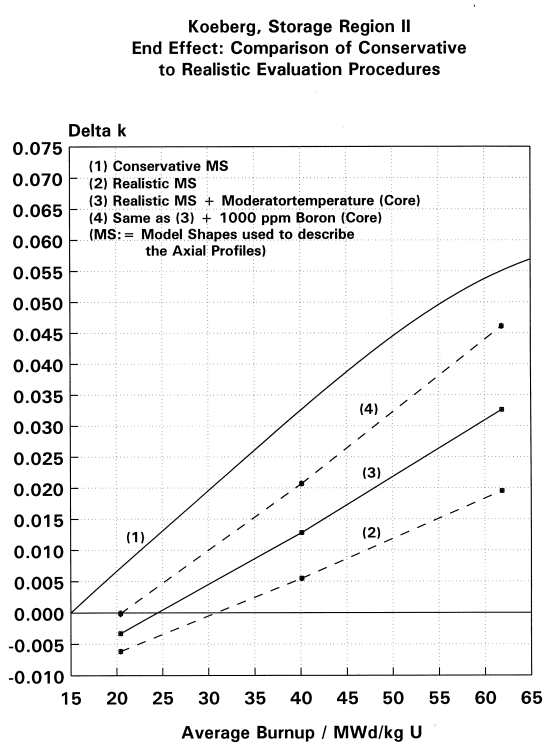

FIG. 32: Proof of Conservatism.

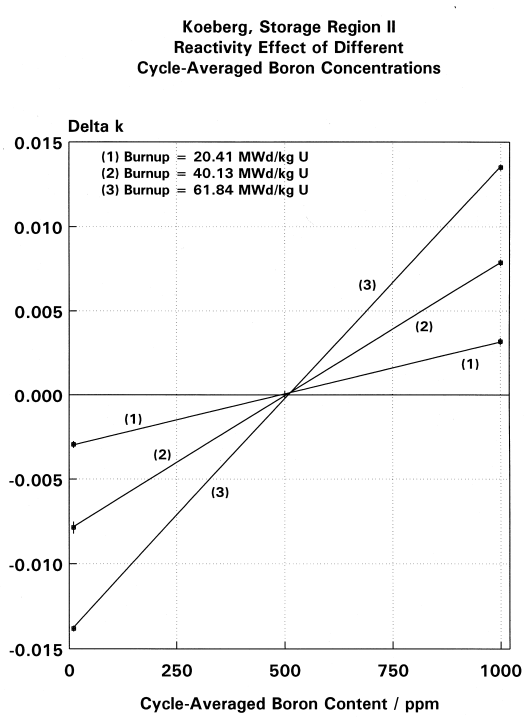

FIG. 33: Reactivity Effect of the Cycle-Averaged Boron Concentration used in the Depletion Calculation. (The Curves No. (1), (2), and (3) in Figure 32 are based on the Use of the Cycle-Averaged Boron Content of 500 ppm in the Depletion Calculations.)

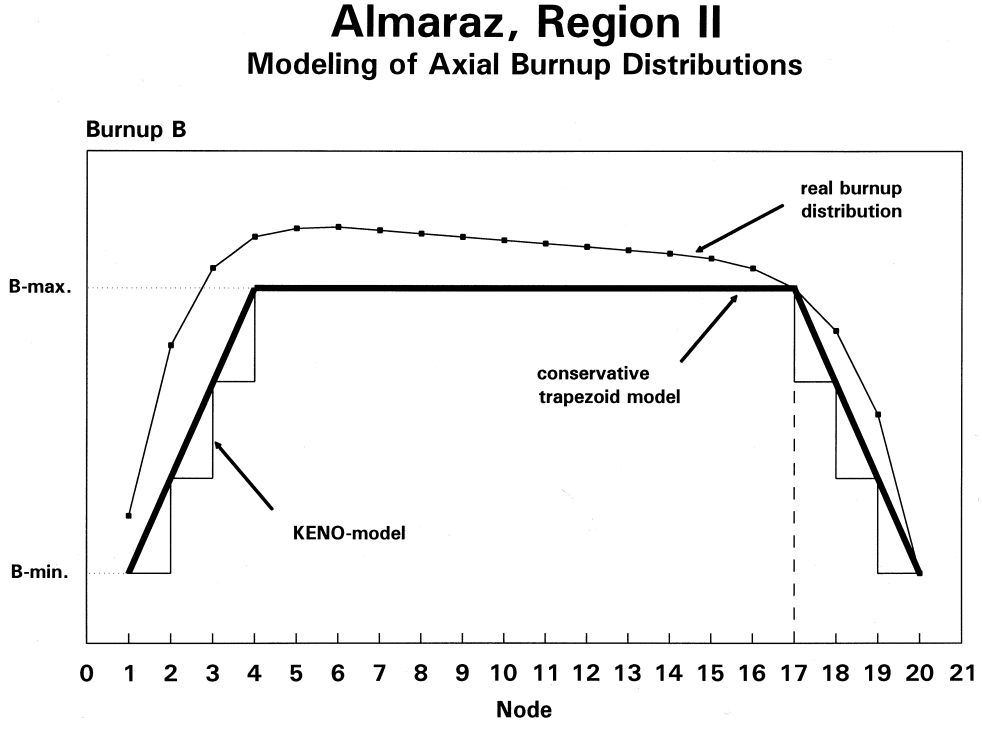

FIG. 34: Determination of a Conservative Trapezoidal Model Distribution [7].

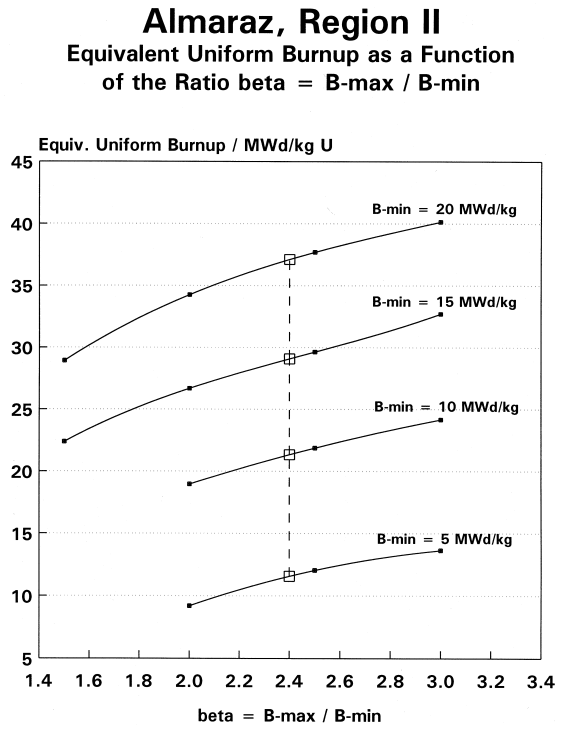

FIG. 35: Use of a Trapezoid Model.

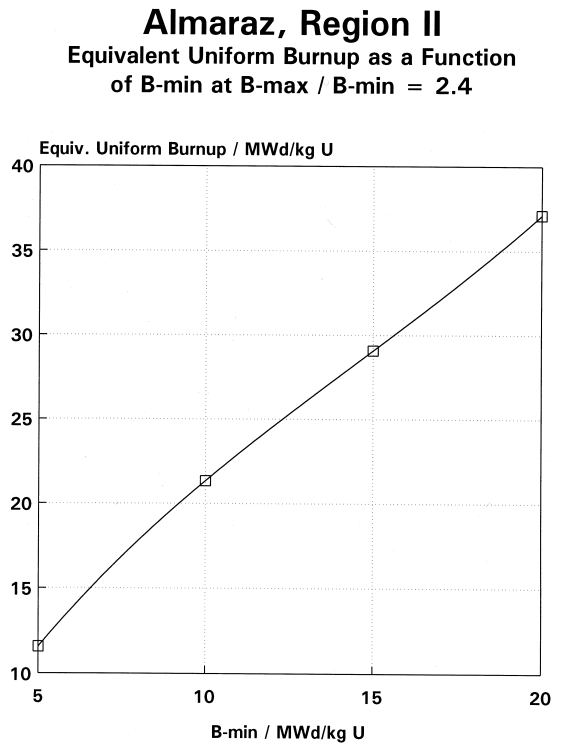

FIG. 36: Use of a Trapezoid Model.

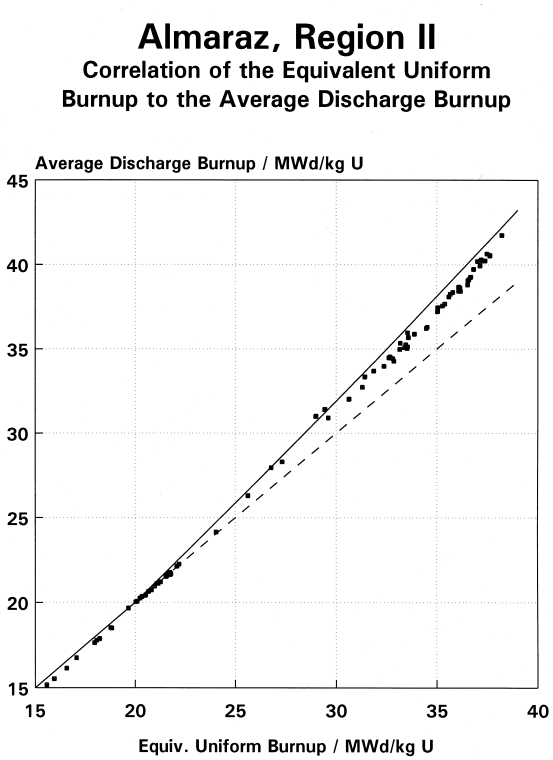

FIG. 37: Use of a Trapezoid Model.

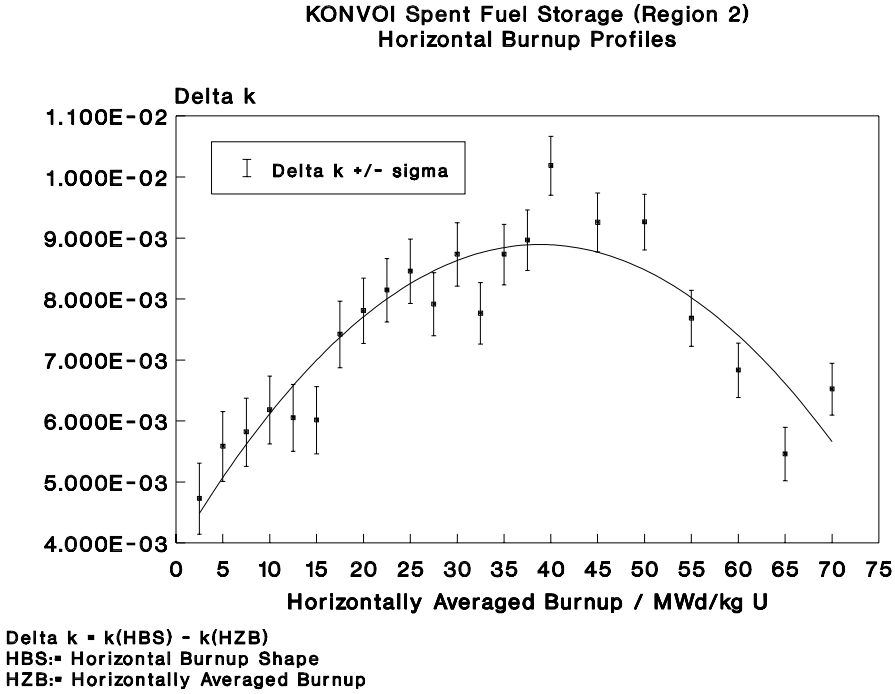

FIG. 38. KONVOI Storage Region II [8]: Effect of Horizontal Burnup Profiles defined by Equations (3.1) and (3.2) [3].

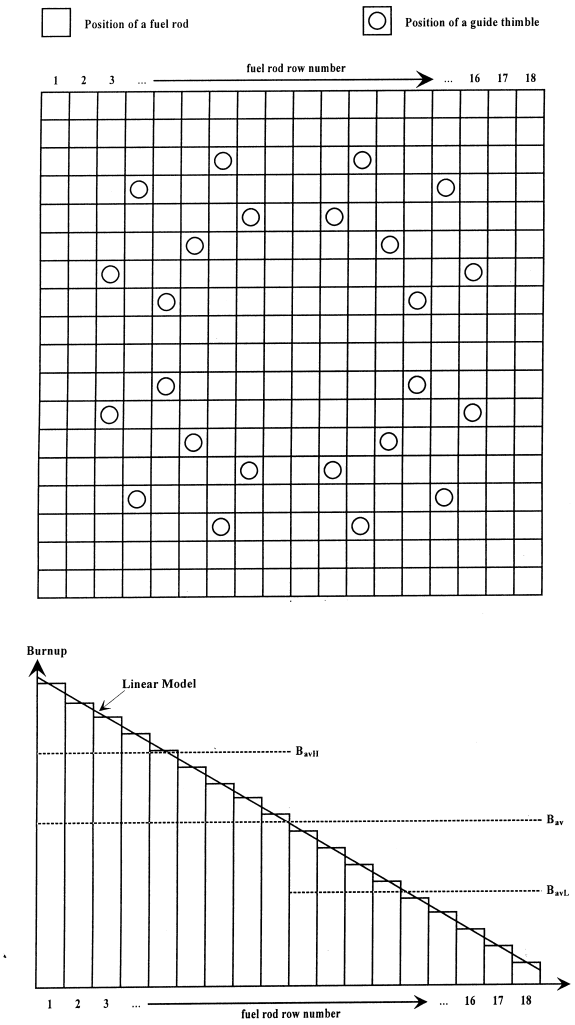

FIG. 39. Illustration to Equation (3.2) for N = 18 (Fuel Assembly of the KONVOI Type)

2.2. Modeling of Axial Profiles

The method described in the preceding section is a method to evaluate axial shapes in order to be able to determine the end effect. This method does not depend on the shapes analyzed, and hence this method does not depend on procedures employed to extract model distributions from the real shapes - which has to be done in order to be able to use axial burnup shapes in the calculation of the neutron multiplication factor of a spent fuel management system of interest.

Usually step functions are used to represent axial burnup shapes in the calculation of the neutron multiplication factors. The method Siemens usually employs is based on application of step functions which are inscribed in the shapes to be analyzed. As illustrated in Figure 30 for the case n = 1, the first step in deriving an axial model distribution from a real shape is to inscribe a step distribution which has $2 \cdot n \cdot m$ steps in the real shape. The second step is to collapse those neighboring steps together which are similar in height. Figure 31 gives an example for a resultant model distribution representing an real axial shape.

Siemens usually applies this method of inscribing step functions in the observed axial shapes to be analyzed in order to take into account that the spatial resolution of the axial shapes observed may not be sufficient. For example, the determination of the shape shown in Figure 31 is based on only sixteen nodes. Except for Germany (as far as known to the author) one usually has a coarse mesh of axial nodes only (which is not sensitive for instance to the burnup dips due to the presence of spacer grids - but the fact that in many fuel designs there is a spacer grid in the range of the upper end of the fuel zone is of importance for the end effect, cf. Figure 1).

In addition, the inscribing of the step functions used to describe the axial shapes observed allows to apply a well-defined set *N(z, B(z))* of isotopic number densities for the burnup steps *B(z)* of the step function. In the criticality safety analysis of a spent fuel management system of interest one has to anticipate the highest end effect of each of the axial profiles analyzed. An axial profile does not tell us, however, whether its shape is mainly due to the locations of the core positions of the fuel assembly during its lifetime, to the interaction with the adjacent fuel assemblies and, perhaps, the rim, or to the use of control rods or burnup shaping rods etc.. All these different history effects result, due to different spectral effects, in variations of the isotopic number densities in each of the burnup steps of the step function used to describe the axial shape. It is not obvious which combination of these variations result in an ensemble of bounding isotopic sets for the whole shape (remember that an actinide-only set is obviously conservative with respect to local reactivity – and this goes for each axial height of the fuel zone -, but results in a significant underestimation of the end effect [6]), but the inscribing of the step functions used to describe the axial shapes covers such variations. So, in summary the inscribing of the step functions is

(a) consistent with the requirement to ensure that the neutron multiplication factor of a spent fuel management system of interest is not underestimated and
(b) obviously conservative and a priori acceptable, therefore.

That this is really true is demonstrated in the following section.

2.3. Proof of Conservatism

The method of inscribing step functions in the shapes to be analyzed was used to generate the diagram Figure 27. The correlation of equivalent uniform to average burnup thus obtained can be expressed (with the aid of the relations shown in Figure 26) in terms of the difference Δk between the neutron multiplication factor obtained in the axially burnup-dependent analysis (inscribing step functions in the shapes) and the neutron multiplication factor obtained in the uniform analysis (assuming a uniform distribution of the averaged burnup). The difference Δk related to the correlation curve Figure 27 is given by the curve No.(1) presented in Figure 32.

The method of inscribing step functions in the shapes to be analyzed is different from the methodologies commonly used. Usually the axial shapes are modeled ”realistically” by putting the measured (or calculated) nodal burnups in the middle of the steps of the step functions used. The results obtained by doing it this way for the Koeberg Region II storage racks are given by curve No.(2) of Figure 32.

However, the methodologies commonly used have to take into account that the moderator density in the core decreases from the bottom to the top of the active zone of the fuel assemblies. This decrease affects the isotopic content of the irradiated fuel. A decrease in the

moderator density results in spectrum hardening and hence in more plutonium production. Usually, this results in a higher neutron multiplication factor of the spent fuel management system of interest. Describing the decrease of the moderator density in the core in an enveloping manner by using the difference between outlet and inlet temperature for the depletion calculations one gets curve No.(3) instead of curve No.(2) in Figure 32. As can be seen, curve No.(3) is covered by far by curve No.(1) representing Siemens' methodology.

By the way, no difference is observed when instead of the decrease in the core moderator density referring to the difference between outlet and inlet temperature only the core moderator density referring to the outlet temperature is used in the depletion calculations. This confirms that the end effect is usually determined by the top end of the active zone of the fuel assemblies because this end is usually less burned than the bottom end of the fuel zone. (In the sample of the 708 axial shapes evaluated in Figure 1 only 6 shapes were found where the ratio α_{ij} of the top node i = 1 is greater than the ratio α_{ij} of the bottom node i = 30.)

The depletion calculations on which the curves No.(1), (2) and (3) of Figure 32 are based are performed by assuming a fixed, cycle-averaged boron content of 500 ppm in the core coolant. Using of a different boron content results of course in a spectrum change and hence in a different isotopic inventory of the irradiated fuel. The higher the assumed boron content is, the harder is the neutron spectrum, the higher is the plutonium production, the higher is usually the neutron multiplication factor of the spent fuel management system of interest. Assuming a cycle-averaged boron content of 1000 ppm, taking account of the changing moderator density in the core and modeling the axial burnup shapes in a realistic manner, all these steps result in curve No.(4) of Figure 32. Also this curve is sufficiently covered by curve No. (1) representing Siemens' methodology.

However, using of a fixed value of 1000 ppm boron in the core coolant is a much too conservative assumption. This can be seen from Figure 33. This figure shows the change Δk in the neutron multiplication factor of the system analyzed in Figure 32 with the change of the core boron content relative to the boron content of 500 ppm used in Figure 32 as a basis for the depletion calculations. The Δk values shown in Figure 33 include the end effect (i.e., are based on axially burnup-dependent analyses). As can be seen from Figure 33, the change Δk in the neutron multiplication factor depends - at given burnup of the fuel - linearly on the boron content, but - at given boron content - non-linearly on the burnup of the fuel. The higher the burnup is, the greater is - at given boron content - the increase in the amount of Δk. From this it follows that even the use of the cycle-averaged boron content of 500 ppm for the depletion calculations is conservative. At the beginning of a cycle the burnup (increment) is low and hence the change Δk remains low. With increasing burnup the boron content in the core coolant is reduced. Thus the increase of Δk with increasing burnup is significantly delayed. For higher burnups the boron content falls below the cycle-averaged boron content and Δk decreases, therefore. Due to the fact that the decrease in Δk takes place at higher burnups whereas the increase in Δk occurs at lower burnups the decrease in Δk predominates over the increase in Δk. The use of the cycle-averaged boron content of 500 ppm for the depletion calculations is conservative, therefore. From that it follows that there is no need to apply a higher boron content than the cycle-averaged boron content to the depletion calculations. So therefore, curve No.(4) of Figure 32 may be left out of account.

However, curve No.(3) covers shapes that might be affected by the use of control rods during operation, but does not cover the change in the isotopic content due to the use of control rods.

Insertion of control rods in fuel assemblies results in spectrum hardening and hence in an increase in the neutron multiplication factor. Thus, the effect of insertion of control rods is the same as that of soluble boron. However, control rod effects are more localized, resulting in localized spectral hardening and non-uniform burnup across the assembly at given axial height. Curve No.(4) indicates therefore, that the use of control rods is a priori covered by the Siemens' methodology represented by curve No.(1) of Figure 32.

In addition, as follows from the gap between the curves No.(4) and (1), the reactivity effect of horizontal burnup profiles – if positive at all – is also covered by the methodology represented by curve No.(1). (In section 3 the reactivity effect of horizontal profiles is conservatively estimated at less than 1% Δk.)

2.4. Standardization of the Evaluation Procedure by Use of Simplified Model Distributions

In the preceding section it was shown that the method of inscribing step functions in the axial burnup shapes to be analyzed is conservative and a priori acceptable, therefore. From that it is obvious that the use of trapezoidal model distributions determined as shown in Figure 34 is conservative [7].

The use of such model distributions results in a very efficient standardization of the evaluation of axial burnup profiles and a significant reduction of the amount of effort necessary in the analysis of the end effect. This is due to the fact that the trapezoidal model, as shown in Figure 34, is completely described by two parameters, the burnup B_{min} at the top node (node 20 in Figure 34) of the fuel zone and the burnup B_{max} of the plateau of the trapezoid determined by choosing a given node (node 17 in Figure 34) which is used in the analysis of all the real axial profiles to be evaluated. The analysis of these profiles consists then only in a registration of the values B_{min} and B_{max} of these profiles.

Whatever the shapes of these profiles are, due to the fact that the trapezoidal model is completely described by the two parameters B_{min} and B_{max} it is possible to determine a priori the equivalent uniform burnup of a trapezoidal distribution as a function of the ratio B_{max}/B_{min} at given values of B_{min} and B_{max}. This can be done, as illustrated in Figure 35, on the basis of a few criticality calculations: The outcomes of these calculations are evaluated by means of the method described in section 2.1, and the lower bounds of the resultant equivalent uniform burnup intervals [3] are fitted with the aid of a linear fit procedure (as follows from Figure 27, these lower bounds are determining for the correlation between equivalent uniform burnup and average burnup). One obtains thus curves such as shown in Figure 35, and in fact only a few criticality calculations are needed to obtain these curves. (In addition, due to the symmetry of a trapezoidal distribution only one half of such a distribution has to be represented in these calculations, the other half can be simulated by applying an appropriately chosen mirror reflection boundary condition.)

The correlation between equivalent uniform burnup and average burnup is determined by applying the curves shown in Figure 35 to the evaluation of the real axial burnup profiles. As stated above, the burnup at the top node of a profile (node 20 in case of Figure 34) is taken as B_{min} and the burnup at a appropriately fixed node (node 17 in case of Figure 34) is taken as B_{max}. The values of the curves Figure 35 at the ratio B_{max}/B_{min} of these values of B_{min} and B_{max} are fitted then with the aid of a linear fit procedure. One obtains thus the equivalent uniform burnup as a function of B_{min} at the ratio B_{max}/B_{min} of the profile (as exemplified in Figures 35

and 36 for the ratio $B_{max}/B_{min} = 2.4$), and from this function one obtains the equivalent uniform burnup of the profile. This equivalent uniform burnup is correlated to the average burnup of the profile, as described in section 2.1.

In this way one can evaluate any number of profiles, as shown for instance in Figure 37. Each of the small squares shown in this figure represents an evaluated profile, and the solid line in this figure represents the resultant correlation between equivalent uniform and average burnup.

In summary, the use of trapezoidal model distributions determined as exemplified in Figure 34 reduces the amount of criticality calculations necessary in the analysis of the end effect to the amount of neutron multiplication factors necessary for the determination of curves such as shown in Figure 35. The use of these curves results in a very efficient determination of the correlation between equivalent uniform and average burnup.

3. EVALUATION OF HORIZONTAL BURNUP PROFILES

Horizontal burnup profiles are covered by the linear model

$$\frac{B_{avH} - B_{av}}{B_{av}} = 0.33 - \frac{0.08}{15} \cdot \left(\frac{B_{av}}{MWd / kg\,U} - 10 \right) \qquad (3.1.)$$

which gives the difference ΔB between the horizontally averaged burnup of one half of the fuel assembly and the horizontally averaged burnup B_{av} of the entire fuel assembly as a function of B_{av}. The averaged burnup B_{avH} of the higher burned half and the averaged burnup B_{avL} of the lower burned half of the fuel assembly are bounded by the equation $\Delta B = B_{avH} - B_{av} = B_{av} - B_{avL}$. The model (3.1.) covers the horizontal profiles presented in [2].

Results obtained for the difference Δk between the neutron multiplication factor obtained with the model (3.1) and the neutron multiplication factor obtained for the averaged burnup B_{av} are shown in Figure 38. These results refer to the KONVOI storage region II represented by the loading curve shown in Figure 3 of reference [8]. These results are based on the following assumptions [3]:

1. It is assumed that all the fuel rods belonging to one and the same row (as defined in Figure 39) have one and the same burnup.
2. The horizontal profiles are assumed to be given by step distributions (cp Figure 39), the steps of which decrease with increasing fuel rod row number according to

$$B(n) = B_{avH} + \frac{4}{N} \cdot (B_{avH} - B_{av}) \cdot \left(n - \frac{N+2}{4} \right) \qquad (3.2.)$$

where n denotes the fuel rod row number, N denotes the total number of rows and B_{avH} is given by the model (3.1.).

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STUDIES ON FUTURE APPLICATION OF BURNUP CREDIT IN HUNGARY

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Abstract

This paper describes the present status of the fuel storage and the possible future applications of burnup credit in wet and dry storage systems in Hungary. It gives a survey of the activities planned in AERI concerning the burnup credit. Some part of these investigations dealing with the influence of the axial changing of the assembly burnup are given in more details.

1. INTRODUCTION

Paks Nuclear Power Plant is the only NPP in Hungary. Approximately 40-50 % of the annual electricity production of the country has been supplied by this plant. It has four units of VVER-440/213 type. These units at present utilise fuel assemblies with 1.6 %, 2.4 % and 3.6 % enrichment. After removing from the reactor core, the assemblies are placed into water in the at-reactor pools. These pools have two levels or "shelves". The pitch of the upper shelf is 22.5 cm (which is quite typical value in VVER wet storage facilities without absorber). The lower compact shelf has a 16 cm pitch and it contains boron steel absorber plates. In the case of emergency unload of the reactor core this wet storage pool is used for storing the fuel assemblies.

After the sufficient cooling time the assemblies from the wet storage pool can be placed into the Modular Vault Dry Storage system built in the area of the NPP. It was built by GEC-ALSTHOM (UK) and came into operation in 1997. The anticipated length of operation of this dry storage facility is about 50 years. The storage facilities listed above meet the criticality safety requirements using the fresh fuel assumption if they contain the fuel types described previously.

2. PLANNED ACTIVITIES

Since the early 90's Paks NPP has been considering the possibility to introduce new fuel types with higher enrichment into the fuel cycles. The fuel types which are now considered are BNFL designed profiled assembly and a new Russian designed profiled assembly.

Until now, the NPP has not been applied to the Authority for licensing burnup credit in storage or transportation. However, the Hungarian Atomic Energy Authority has recognised that the general tendency of using fuel with higher enrichment can result the need of application of burnup credit sooner or later. For this reason, the Authority supports the burnup credit research and development activities performed in the KFKI Atomic Energy Research Institute (AERI). Until now a study on burnup credit has been prepared and preliminary investigation of the end effect was performed for the Authority. At present, a negotiation is in progress between the Authority and AERI to make more systematic studies on the issues relevant in the application of burnup credit in VVER fuel storage/transport systems. The aim of this work would be to ensure the necessary background to the Authority for questions related to burnup credit.

The main points which we plan to study are the next:

1. Validation of depletion calculations. The lack of experimental data of spent VVER fuel composition is a major difficulty in depletion code validation, but there is one detailed experimental benchmark in which the operational history/reload patterns of three reactor cycles are given and the concentration of the actinides in some selected samples after one/two/three cycles are measured. This benchmark is to be investigated by the core design code KARATE developed in KFKI AERI. There are other such measurements in Russia, but they are available on commercial bases only.
2. Determination of proposed parameters for depletion modelling. While the fuel reach a given value of burnup (i.e. a given amount of fission in U235) it can be irradiated under a variety of conditions. The values of fuel and moderator temperature, boron concentration, etc. influence the amount of Pu produced via spectral hardening. The importance of this point is increased by the fact, that a typical VVER spectrum is harder than a PWR spectrum.
3. Influence of the spatial variation of the assembly burnup. The axial variation is expected to play an essential role because of the regular use of the control rods. This point will be discussed later in more details in this paper.
4. Investigation of the importance of the different nuclides and the influence of cooling time. Essential difference from the PWR results are not expected at this point.
5. Validation of the code MCNP for such type of criticality calculations by MOX critical experiments. The validation by reactor critical configurations is challenging in spite of the huge amount of the work required. It should be noted that the value of this way of validation is sometimes questioned.
6. Sensitivity and uncertainty analysis of the methods applied in the burnup credit calculations. Determination of the conservatism of the particular burnup credit approaches.

The calculations for the analysis outlined above are planned to be performed by the MCNP Monte Carlo code and the KARATE core design code system. MCNP is a well known code and it is widely used for criticality calculations [1]. KARATE-440 is a three dimensional coupled neutron physical – thermohydraulical program system which was developed in AERI. It is made up from calculations in cell, assembly and global level. There is a consistent bi-directional connection between the levels via the parameterization of the libraries. It was validated against a number of reactor measurements and numerical benchmarks [2] [3] [4]. It is routinely used at the Paks NPP.

The composition of the spent fuel will be calculated by the MULTICELL module of the system. It is a two-dimensional code, which can be used for the calculation of the pin-by-pin flux multigroup distribution in a fuel assembly. The calculation is made in 35 epithermal and 35 thermal groups. Using the calculated reaction rates, the concentration of several hundred isotopes are calculated. This module was also carefully validated against measurements and mathematical benchmarks [5][6] [7].

In the former version of KARATE the three dimensional calculations were made in 10 axial layers. In its present version the number of layers is arbitrary (in practice is 40). The criticality calculations described in the next point were based on a former calculation performed using 10 layers.

3. INFLUENCE OF THE AXIAL BURNUP CHANGE

In this point the impact of the axial burnup profile of the spent fuel assembly on the multiplication factor will be studied on a sample case. The aim of this study is just to get a general view about the importance of this effect in VVER storage and not to derive or apply a general procedure to take into account it in a particular burnup credit application. Two approaches to this question are the derivation of conservative equivalent uniform burnup by statistical investigation of a number of distribution [8], or construction of enveloping burnup profiles [9] [10].

This criticality concern, stemming from the fuel ends having low burnup, is usually referred as the end effect. Its measure can be defined as ρ = (keff (axial distribution)- keff (uniform distribution))/ keff (uniform distribution). Here keff (axial distribution) means the multiplication factor calculated by a particular axial burnup distribution of the assembly and keff (uniform distribution) means the multiplication factor calculated by using constant burnup all along the active part of the assembly which is equal with the average of the actual distribution.

The reason which makes this point specially interesting in the case of VVER reactors is the use of the absorber rods during normal operation. These rods are partially inserted in the core during most part of a cycle and each of them are connected to a fuel assembly called follower. A follower moves together with the control rod during the cycle and a part of its length is generally outside of the core, so it has larger axial burnup variation than a regular fuel assembly. Beside, the fuel assemblies surrounding a partially inserted absorber rod also may have quite essential burnup variation. The schematic cross section of a VVER-440 core with absorber rods used during normal operation is shown on Fig. 1

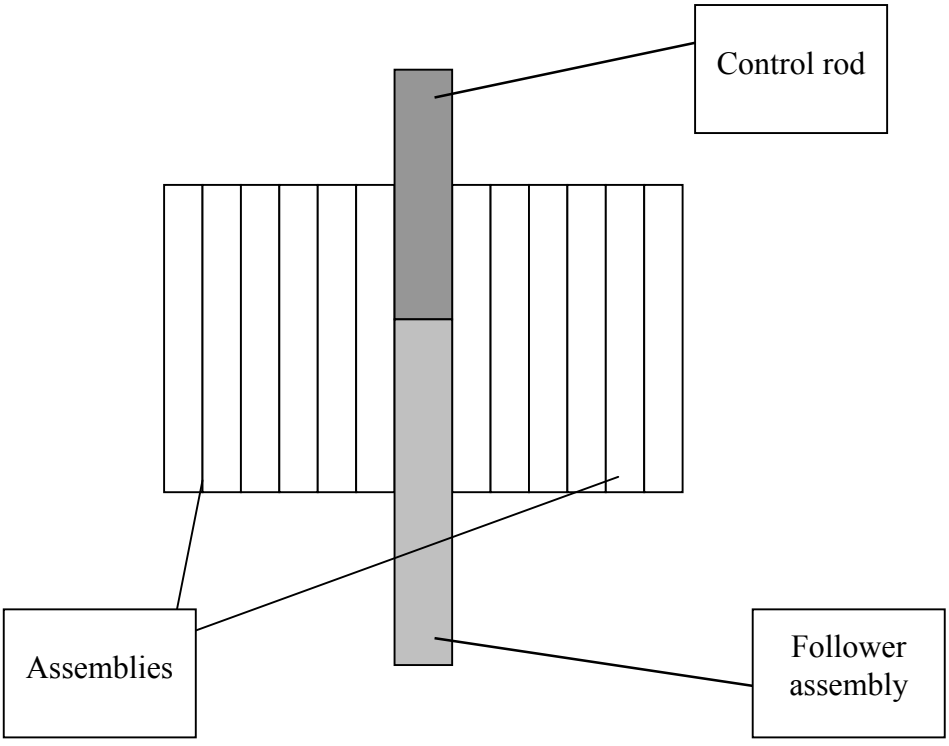

FIG. 1. Schematic cross section of a VVER-440 core with absorber rod and follower.

The multiplication factor of the storage facility is influenced by the characteristics of the storage facility and of the spent fuel assembly. To get a (hopefully) typical view of the end effect in VVER storage systems, we constructed a simple model of a wet storage pool with characteristics close to a usual VVER storage pool, but with reduced pitch such that subcriticality could be ensured only with taking into account the burnup. The KOLA benchmark [11] was chosen as the core history from which the fuel characteristics could be derived. This choice was motivated by the use of assemblies with 4.4 % enrichment and achieving burnup unusually high in VVER reactors. The criticality analysis was performed using actinides plus fission products as well as actinides only burnup credit methodology. The criticality calculations were performed by the MCNP4B Monte Carlo code, while the core and depletion calculations were done by the KARATE-440 core design code system described above.

The first results on using higher enriched fuel in Kolskaya Nuclear Power Plant (KOLA NPP) Unit 3. were published in 1991 [12] [13]. Later a detailed load follow benchmark was defined in the framework of the Atomic Energy Research collaboration of the VVER user countries [11]. This benchmark has already been carefully analyzed in the previous years from reactor core physical point of view in AERI and in other institutes of VVER user community. The benchmark specification contains the reload patterns and detailed operational histories of the first 12 cycles of the KOLA NPP Unit 3. From the 5th cycle the core contained assemblies with 4.4 % enrichment. Several assemblies achieved average burnup level up to 50 MWd/kgU and a few assemblies even higher. Apart from the usual VVER case, assemblies with enrichment 3.6 % also were used as follower. These features makes this benchmark particularly interesting for an end effect study.

In this model calculations the spent fuel assemblies were located into water in a horizontally infinite hexagonal array having 20 cm lattice pitch. VVER assemblies are made up from 126 fuel pins with 242 cm axial height and with 1.22 cm pitch in hexagonal lattice. The pins are surrounded by 2 mm zirconium shroud. The outer diameter of the assembly is 14.4 cm. The top and bottom passive parts of the assembly were modeled approximately but keeping the steel/water/zirconium volumes at their correct values. 30 cm water layer was assumed under the lower end and above to upper end of the assembly. Black boundary condition was described on the top/bottom surfaces of the water layers. (Test calculations shown that the influence of the boundary condition on keff in this system is negligible.)

A series of core calculations were performed by KARATE for the 12 cycles of KOLA according to the specification of the benchmark. The calculations were made in 60 degree symmetry sector. At the end of each cycle, the axial burnup distribution of assemblies calculated in 10 layers was collected from the corresponding core calculations. The isotopic compositions of the fuel pins in these layers were then calculated by MULTICELL using the additional assumption, that the pins have identical burnup distribution in a particular assembly. The concentration of U-235, U-236, U-238, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-243 and O-16 as well as 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 was determined after 1 year cooling time. (This set of isotopes is identical of that used in the benchmark of Markova [14].)

Having these compositions two sets of keff calculations were performed for a spent fuel assembly in the model pool configuration described above, one with axially dependent concentration of actinides and an other with uniform concentrations corresponding to the

average values of assembly burnup. The calculations were performed by MCNP4B. The statistical error of each calculation was less than 1.7 x 10-3. Both series was repeated including the fission products. The keff values determined from uniform distributions were fitted by a smooth function of the burnup. These curves together with the multiplication factors calculated using the axially dependent burnup distributions are plotted on Figs. 2.-4. against the average burnup of the assemblies . 326 different distributions due to 4.4 % enrichment, 242 due to 3.6 % enrichment and 77 due to 2.4 % enrichment were examined.

3.1. Results

The multiplication factors from these calculations are shown on Fig. 2.- Fig. 4 for assemblies with 4.4 %, 3.6 % and 2.4 % enrichment. Four curves shown on each figures. The upper two are due to the actinides only, the lower two are due to the actinides plus fission products methodology, as it is expected. Results from the calculations with axial distributions are shown by solid line, the keff values due to uniform distributions are shown by dashed line.

3.1.1. Assemblies with 4.4 % enrichment

In the actinides plus fission products case an apparent dependence on average burnup can be seen on Fig. 2 . The end effect is negative bellow about 28 MWd/kgU and mostly positive above this value. A closer look of the data shows, that the maximal value of ρ = (keff (axial distribution)- keff (uniform distribution))/ keff (uniform distribution) is 0.028, while the minimal is -0.1. All of the 24 values of ρ which are higher than 0.01 are due to burnup higher than 30 MWd/kgU. From these 24 cases there are only 7 when the burnup less than 40 MWd/kgU and 6 of them is the case of an assembly adjacent to an absorber. Without fission products the absolute value of the end effect is less and the tendency is not apparent. The maximal value of ρ is 0.005, not more than 3σ, where σ is the standard deviation of the corresponding Monte Carlo calculation.

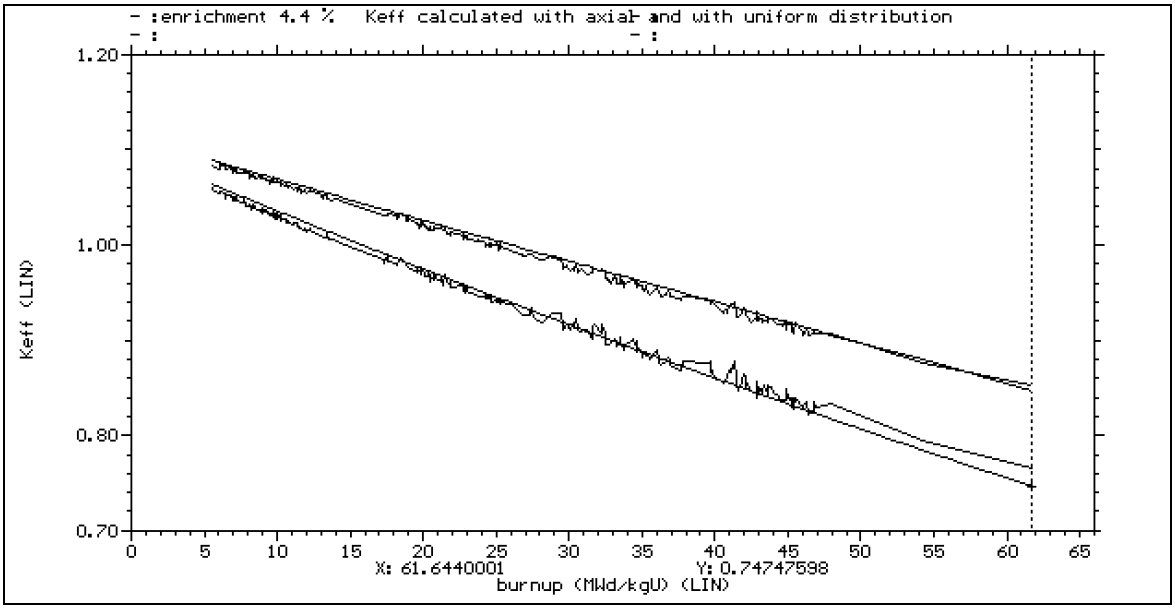

FIG. 2. keff as function of burnup for 4.4 % assemblies with uniform and with axially changing burnup distribution. The lower two curves due to the actinides + fission products, the upper two due to the actinides only cases.

3.1.2. Assemblies with 3.6 % enrichment

In this case the end effect is essentially stronger, the maximal ρ is 0.056 with inclusion of the fission products. This can be explained by the presence of the followers, which mostly have highly asymmetric burnup profile and their lower end may have a very low burnup. (The spike on Fig. 3 at 7.8 MWd/kgU is due to a follower having unburned part with length of 75 cm). The behavior of ρ now is determined by the increasing of the burnup and by the presence of followers. The influence of the burnup is similar to the previous case. There are 36 ρ values higher than 0.01. 27 of them are due to burnup above 30 MWd/kgU, and 6 from the other 9 are followers. The highest values of ρ are due to followers with high burnup. The trend is quite similar without fission products. In that case the maximal ρ value is 0.034. The exclusion of the followers from the investigation reduce essentially the end effect: in the actinides + fission products case the maximal value of ρ decreases to 0.032 from 0.056.

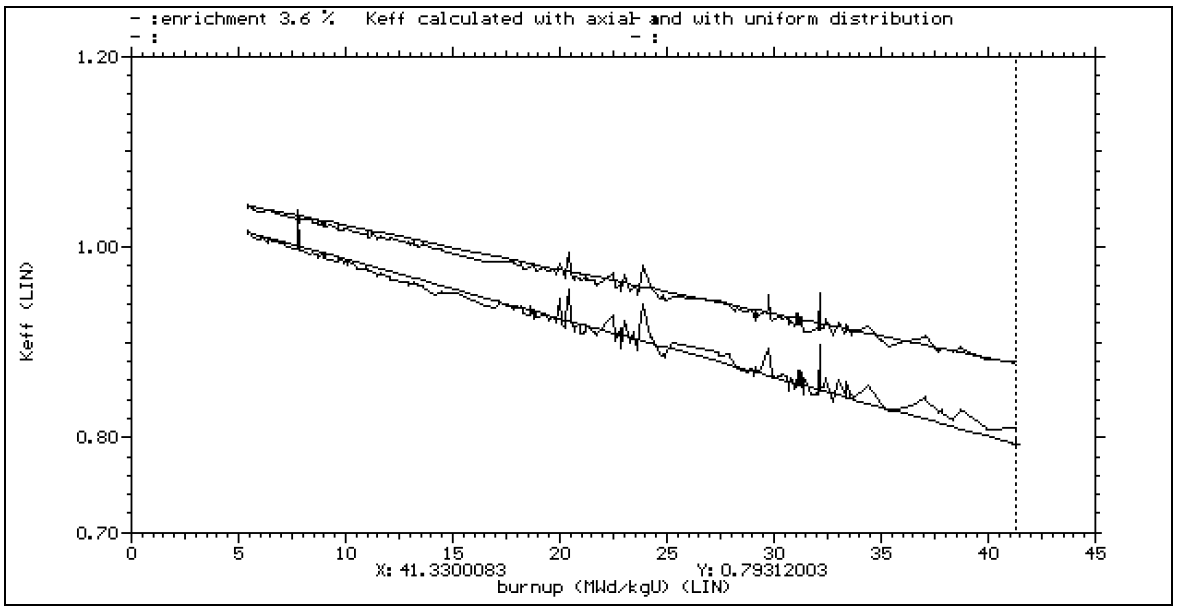

FIG. 3. keff as function of burnup for 3.6 % assemblies with uniform and with axially changing burnup distribution. The lower two curves due to the actinides + fission products, the upper two due to the actinides only cases.

3.1.3. Assemblies with 2.4 % enrichment

The results now show a different picture. The effect of the burnup increase is not apparent on Fig. 4 and the higher end effects mainly due to the followers. This might be caused by the less number of the analyzed axial distributions. (We could not perform the criticality calculations for all of the distributions in this case). Including the fission products it was found that the maximal ρ value is 0.042. 12 values of ρ were found to be higher than 0.01 and 8 of them now are due to average burnup lower than 10 MWd/kgU. All of this 8 distributions are due to a follower. In 3 from the other 4 cases the average burnup of the assembly is above 20 MWd/kgU. Exclusion of the fission products reduces the maximal ρ to 0.025.

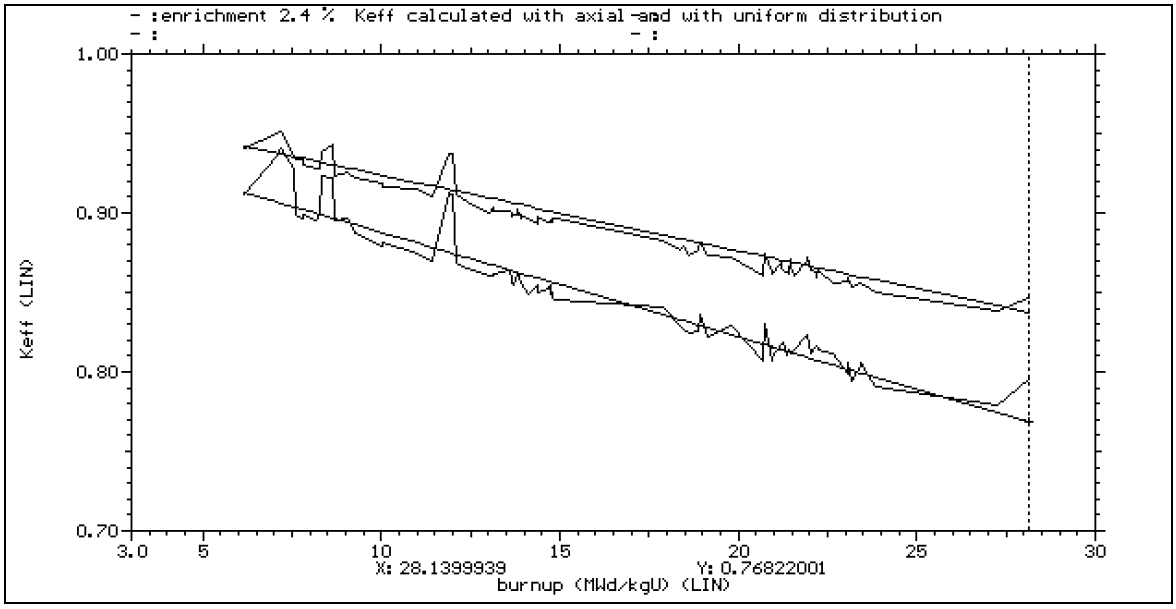

FIG. 4. keff as function of burnup for 2.4 % assemblies with uniform and with axially changing burnup distribution. The lower two curves due to the actinides + fission products, the upper two due to the actinides only cases.

4. CONCLUSIONS

The end effect may have an essential influence on VVER storage facility design if burnup credit is used. In the cases of 4.4 % and 3.6 % enrichments its significance increases with the burnup. This behavior is similar to that of find in the case of PWR fuel. This tendency can not be observed when the enrichment is 2.4 %. The highest value of ρ was found to be a little below 0.06 at 3.6 % enrichment. The presence of the followers increases strongly the end effect, the highest ρ values are due to followers. This shows that the accurate modeling of the burnup distribution has particularly high importance at the ends of an assembly. In an implementation for VVER fuel might be worthy to consider the separate handling of the followers. For any implementation the application of a general approach (conservative equivalent uniform burnup or enveloping burnup profile) is necessary. It seems worthy to perform this study using higher number of axial divisions of the core and of the assemblies in the KARATE as well as in the MCNP calculations.

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DETAILS ON AN ACTINIDE-ONLY BURNUP CREDIT APPLICATION IN THE USA

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Abstract

Details on the Actinide-Only burnup credit assumptions that will be used for the CASTOR® X/32 S cask are presented. Preliminary results show that using a conservative set of assumptions the cask will allow most fuel to be loaded without the addition of any additional reactivity control. With the addition of 8 control rod elements it is possible to load most of the rest of the fuel.

1. INTRODUCTION

Burnup Credit has been activity pursued in the United States since the mid 1980's. [1] In 1995 the US Department of Energy (DOE) submitted a Topical Report on Actinide-Only Burnup Credit to the US Nuclear Regulatory Commission (NRC) [2] and after four years revisions and discussion; the NRC ultimately issued the Interim Staff Guidance 8 (ISG-8). [3] ISG-8 provides guidelines but leaves a number of items up to the applicant. This paper will describe how one applicant is resolving the issues left open by ISG-8 and also describes how one applicant will take some exceptions to ISG-8.

ISG-8 is as the title suggests an interim position of the NRC. The NRC is currently reviewing burnup credit through their research division via Phenomena Identification and Ranking Tables (PIRT). The PIRT process has already led to further understanding of some of the outstanding issues. A number of the positions taken in this paper come directly from this NRC work in progress.

The application described in this paper is for GNB's CASTOR® X/32 S storage and transportation cask. A non-burnup credit version of the cask is under review by the NRC for its storage license, DOCKET 72-1028. The burnup credit application will be included in the transport application and then the storage application will be amended to include burnup credit. The transport application is expected before the end of 2000. Although several cask vendors are also attempting burnup credit in a similar time frame, this cask will most likely be the first actinide-only burnup credit application in the US for a transport cask.

This paper will begin by describing the cask under consideration followed by section on the six major efforts in burnup credit:

1. Isotopic Depletion Model Validation,
2. Criticality Model Validation,
3. Conservative Depletion Modeling,
4. Generation of Loading Limits,
5. Verification of Fuel Characteristics,
6. Verification of Criticality Margin.

2. THE CASTOR® X/32 S STORAGE AND TRANSPORT CASK

The CASTOR® X/32 S storage and transport cask is typical of large burnup credit casks. This cask weighs about the same as most flux trap designs holding 24 PWR assemblies but it is able to hold 32 assemblies plus burnable absorber inserts due to burnup credit. The current non-burnup credit application can also hold 32 assemblies by utilizing Absorber Rod Modules (ARMs) which are fresh control rod placed in the fuel assembly guide tubes. However, these ARMs prevent the shipment of burnable absorber inserts so burnup credit is desirable.

The cask uses 7.85 mm (minimum) aluminum plates containing a minimum Boron-10 content of 0.09615 gm/cc (prior to the NRC mandated 75% reduction in the analytical models) sandwiched between stainless steel receptacle walls 5.71 mm thick. The receptacle inside width is 220.8 mm. Figure 1 shows a cross sectional view of the cask.

The advantage of burnup credit is the ability to remove the ARMs and to allow the insertion of burnable absorber inserts. Table I. show the fresh fuel assumption limits for the CASTOR® X/32 S cask.

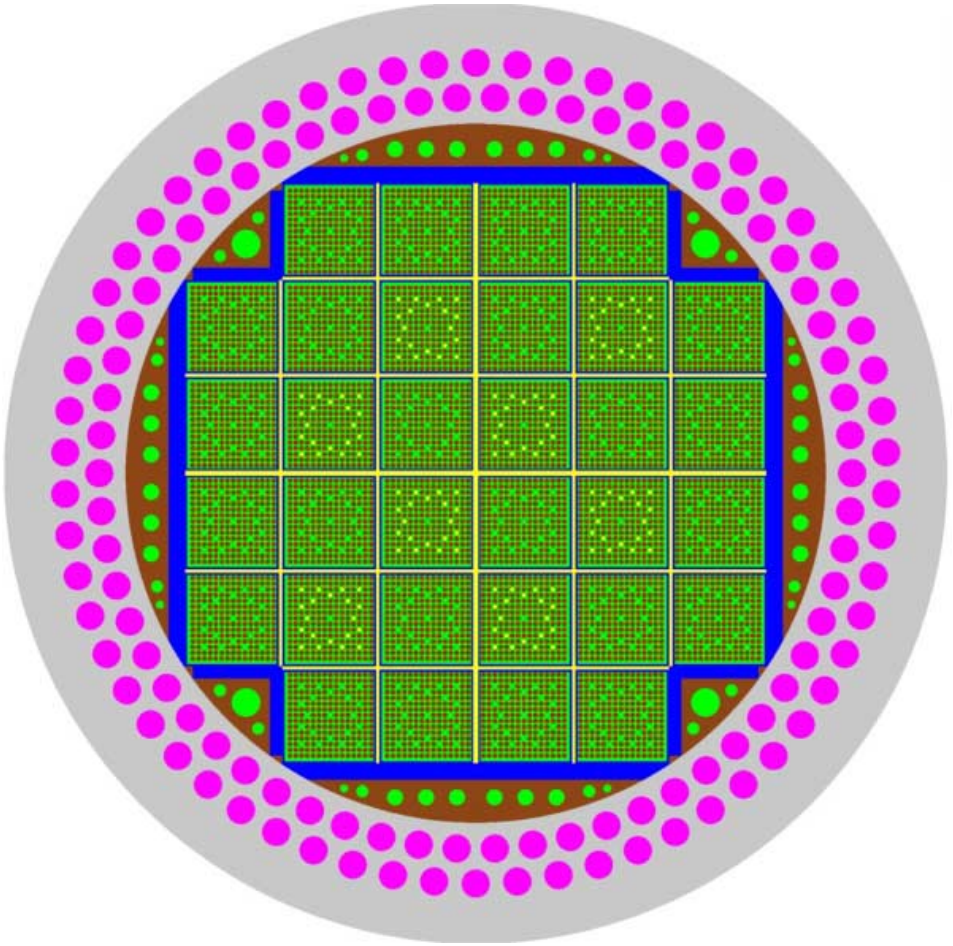

FIG. 1. The CASTOR® X/32 S Storage and Transportation Cask.

Table I. Enrichment Limits for Fuel From Various Plant Types as a Function of the Number of Absorber Rod Modules (ARMs) Inserted.

ARMs Inserted	8	16	20	24	28	32
Plant Type	Maximum Enrichment					
CE 14 x 14	3.01	3.60	4.06	4.33	5.	-
St. Lucie 2	3.00	3.51	3.94	4.20	4.74	5.
W 14 x 14	3.36	4.22	4.83	5.	-	-
W 15 x 15	2.70	3.55	4.01	4.48	5.	-
W 17 x 17	2.73	3.56	4.03	4.50	5.	-

3. ISOTOPIC DEPLETION MODEL VALIDATION

The isotopic depletion analysis is being performed with SCALE 4.4a (SAS2H) and the 44 group library. The chemical assays presented in the DOE topical report [2] consisting of 54 separate chemical assays are used for the validation. The statistical treatment presented in DOE's Topical Report [2] is also used. The NRC has reviewed this data and has taken the position that it is only sufficient for validation up to 40 GWD/MTU. The have also taken the position that this data requires additional margin to be added when going above 4 wt%. This additional margin is described in their ISG-8. [3]

Table II: Isotopic Correction Factors Using ENDF/B-IV and ENDF/B-V Libraries.

Isotope	ENDF/B-V (44 group)			ENDF/B-IV (27 Group)			Expected % Difference
	Mean M/C	Correction Factor (CF)	CF/(M/C)	Mean M/C	Correction Factor (CF)	CF/(M/C)	
U-234	0.997	0.903	0.905	0.973	0.818	0.841	6.5
U-235	1.023	1.080	1.056	1.025	1.084	1.058	-0.1
U-238	1.001	0.991	0.991	1.001	0.991	0.991	0.0
Pu-238	1.089	0.919	0.844	1.005	0.868	0.864	-2.0
Pu-239	1.008	1.080	1.071	0.979	1.052	1.075	-0.3
Pu-240	0.995	0.961	0.966	1.063	1.000	0.941	2.5
Pu-241	1.044	1.126	1.079	0.960	1.028	1.071	0.8
Pu-242	1.006	0.909	0.903	1.092	0.969	0.887	1.6
Am-241	1.044	0.964	0.924	0.960	0.886	0.923	0.1

Table II. shows the isotopic Correction Factors (CF) for the nine isotopes used in Actinide-Only burnup credit. These correction factors are derived by a statistical treatment of the Measured isotopic concentration (M) over the Calculated isotopic concentration (C). The calculated concentration of each isotope is multiplied by the CF before use in the criticality analysis. If the calculated isotopic concentrations were multiplied by the mean M/C then on average they would produce the experimental results. Table II. shows the ratio of the CF to the mean M/C to show the required margin to cover the uncertainty in the validation. Table II. shows the same analysis performed with both the ENDF/B-IV and ENDF/B-V libraries. As can be seen by the M/C values, the ENDF/B-V results show better agreement with the experiments than ENDF/B-IV. (M/C's are closer to one.) On average the CF/(M/C) ratio gives

the deviation from the experimental values after correction. Note that this ratio is quite similar between ENDF/B-IV and ENDF/B-V. The last column gives the ratio of the CF/(M/C) values minus one for ENDF/B-IV and ENDF/B-V. This value is the expected difference in the corrected calculated values input to the criticality analysis. Since the mean M/C improved due to the library change but the corrected calculated values remain about the same, one can surmise that the dominant uncertainty in covered by the CF is experimental uncertainty rather than calculational uncertainty.

4. CRITICALITY MODEL VALIDATION

The burnup credit criticality validation comes from appropriate fresh UO_2 critical experiments supplemented by MOX critical experiments. For the CASTOR® X/32 S cask, 123 UO_2 critical experiments along with 64 MOX critical experiments were used. The analysis was performed using the KENOVa module of SCALE 4.4a using the 44 group library.

The criticality safety limit is determined by the use of the Upper Subcriticality Limit (USL) based on work done at ORNL. [4] The DOE Topical Report [2] also uses this approach. The USL for the UO_2 experiments were determined by selecting the minimum USL after testing against ten possible trending parameters. Table III. show the results of this trending analysis. The minimum USL from the UO_2 analysis was determined to be 0.9413. The analysis of the MOX experiments produced a minimum USL of 0.9414 so the USL of 0.9413 was utilized in the criticality analysis.

Table III: Determination of the Upper Subcriticality Limit for the UO2 Experiments.

Trending Parameter	Minimum USL Ignoring Significance	Significant Trend?	Range in Criticals	Expected Value in Cask
Enrichment	0.9414	Yes	2.35-5.74	2-5
Fuel Pin Pitch	0.9414	Yes	1.2-2.5	1.2
Average Group Causing Fission (AEG)	0.9422	No	27.6-36.6	33-35
Average Energy of Fission	0.9413	Yes	0.0827-3.505	0.2-0.4
Water to Fuel Ratio	0.9426	No	0.383-5.067	1.6
H/X Ratio	0.9419	No	34-504	100-250
Dancoff Factor	0.9415	No	0.039-0.615	0.1
Separator Plate Areal Boron Density	0.9420	No	0.0003-0.067	0.05
Assembly Separation	0.9421	Yes	0-15.87	2
Boron Content	0.9419	No	0.0001-0.006	0.012

It is important in the criticality validation that all the important isotopes in the safety analysis are covered in the critical experiments. For that reason the MOX experiments used cover a wide range of plutonium vectors (the ratios of the plutonium isotopes). Credit will be taken for the isotopic change with cooling. The minimum cooling will be 9.5 years (which matches shielding limits). Pu-241 decays to Am-241 with a half life of 14.4 years. Since the critical experiments often used old MOX pins there is sufficient Am-241 for validation of up to about

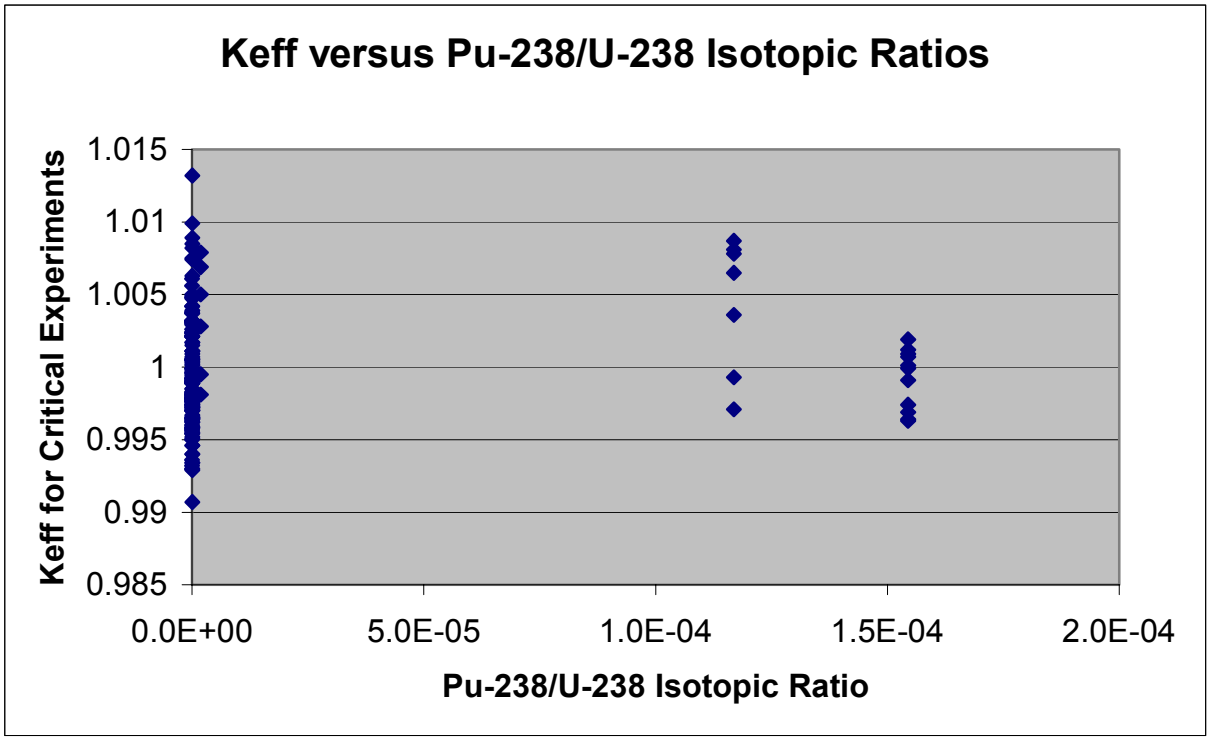

FIG. 2. k_{eff} as a Function of Pu-238 Isotopic Ratio (Pu-238 Isotopic Ratio at 40 GWD/MTU is 0.0002).

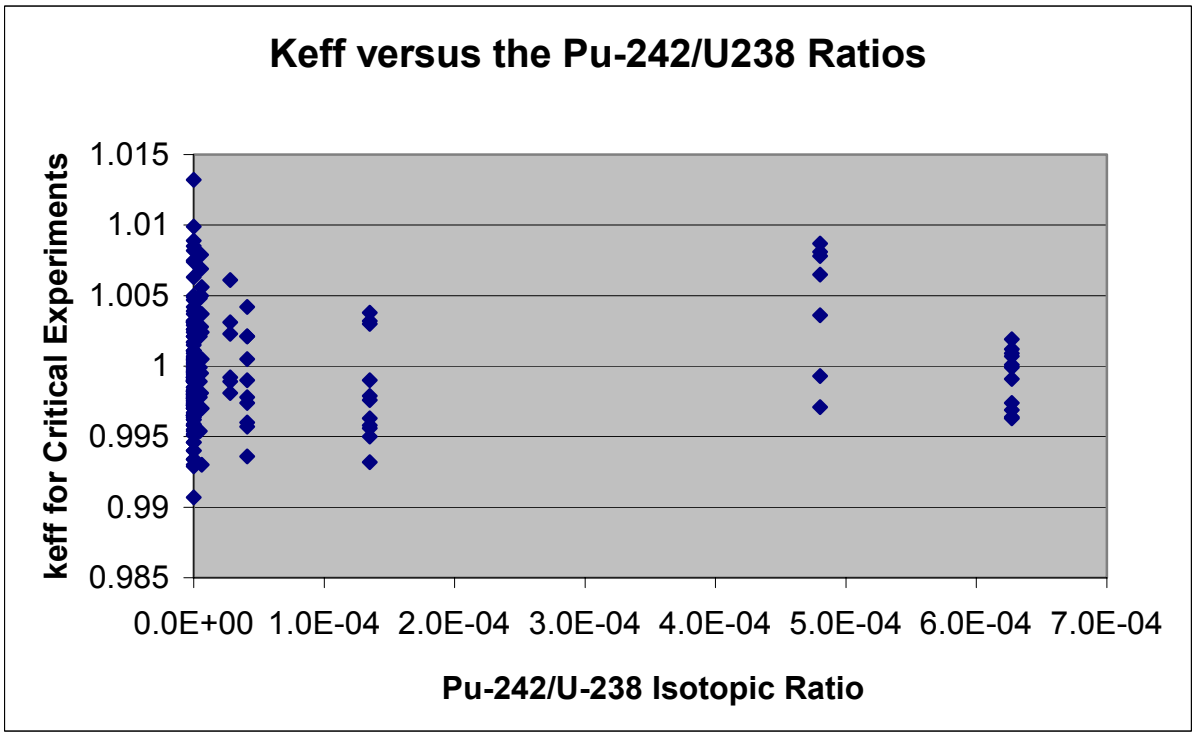

FIG. 3. k_{eff} as a Function of Pu-242 Isotopic Ratio (Pu-242 Isotopic Ratio at 40 GWD/MTU is 0.0006).

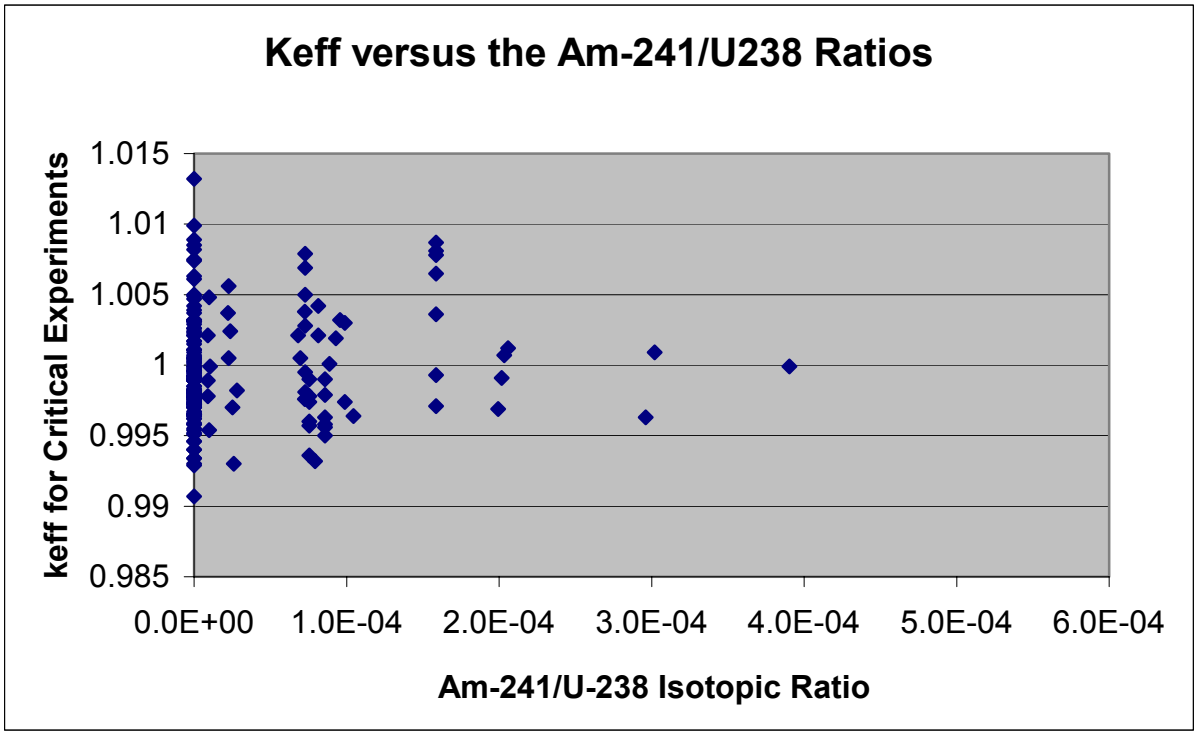

FIG. 4. k_{eff} as a Function of Am-241 Isotopic Ratio (Am-241 Isotopic Ratio at 40 GWD/MTU and 10 years cooling is 0.0006).

10 years cooling but some extrapolation is required. In order to demonstrate that the isotopic content of the fuel in the cask is well represented by the critical experiments plots were generated for the calculated k_{eff} of the critical experiments as a function of the isotope concentration over the U-238 concentration. Only Pu-238 and Am-241 require extrapolation of the isotopic ratios. Pu-242 does not require extrapolation but the data is getting sparse at the isotopic ratio needed. Figures 2 through 4 show the k_{eff} as a function of isotopic ratio for these three isotopes. No trend on any of these isotopic ratios exists so the critical experiments are sufficient.

5. CONSERVATIVE DEPLETION MODELING

Depletion modeling must address a number of issues. They are:

1. Fuel Temperature,
2. Soluble Boron,
3. Moderator Temperature,
4. Burnable Absorbers
5. Control Rods,
6. Axial and Horizontal burnup distribution.

Each of this issues is addressed in a subsections below.

5.1. Fuel Temperature

Higher fuel temperature during depletion produces more reactive depleted fuel. The estimated sensitivity of this effect is 4-5 pcm/K. [5] The value selected for the fuel temperature was 1040 degrees K. This comes from analysis of the highest rated PWR and using a 1.5 radial peaking factor. The highest rated plant and a 1.5 radial peaking factor results in a linear power rating of 9 kw/ft. The temperature analysis accounts for the gap closure and the absorption reaction weighting of the temperature. Since the fuel temperature mainly depends on the kw/ft limit caution is expected if greater than 9 kw/ft is determined for a plant. However, since it is not expected that an average fuel linear heat rate will ever exceed 9 kw/ft there is no verification requirements.
Note that a radial peaking factor was used rather than a total peaking factor since the top portion of the fuel assembly dominates the reactivity effects where the axial peaking factor is less than one.

5.2. Soluble Boron Concentration

Higher soluble boron concentration during depletion increases k_{eff} of the depleted fuel. The estimated sensitivity of this effect is 3-3.5 pcm/ppm. [5] PWRs are limited in their soluble boron concentration by the moderator temperature coefficient (MTC). The MTC becomes more positive with increasing soluble boron. PWRs are required to have negative MTCs at full power. Historically they were required to have negative MTCs at hot zero power. Some plants have received license amendments to allow up to a +7 pcm/degreeF MTC at hot zero power. To estimate the maximum ppm to expect at a PWR a review of some PWR cycles was performed. The highest ppm found was for Farley Unit 1 cycle 7. Its hot zero power MTC was +4.0 pcm/F. Its full power soluble boron concentration after reaching Xe equilibrium was 1540 ppm. PWRs are generally run to about 10 ppm at end of cycle so the average ppm for this cycle was 775 ppm. Since this plant did not run at a plus 7 pcm/F the value selected for the average ppm during depletion was raised to 900 ppm. Since this is expected to be limiting for all plants, no verification of this parameter prior to loading the cask will be required.

It is possible to for an assembly to experience a higher average ppm during its burnup if the plant shuts down well before it reaches the targeted 10 ppm end of cycle. This is rare and not expected to occur multiple times for the same fuel.

5.3. Moderator Temperature

The moderator temperature effect is dominated by the moderator density effect. The less dense the water the harder the spectrum and hence the more reactive the fuel is for any given burnup. The estimated sensitivity of this effect is 35-90 pcm/K. [5] A review of the outlet temperatures of PWRs determined that the highest outlet temperature is 604 K. This corresponds to 0.6489 g/cc. No verification of this temperature is required since it is not expected to be exceeded.

5.4. Burnable Absorbers

Burnable absorbers harden the neutron spectrum and hence the fuel for any given burnup is more reactive. Due to limited information available to the NRC at the time of issuing ISG-8 [6], ISG-8 prohibits burnup credit for fuel assemblies that contained burnable absorbers. Work as part of the PIRT process has resulted in greater NRC knowledge on burnable absorbers and

it is generally believed that the NRC will accept applications with fuel that contained burnable absorbers.

Since a large fraction of the fuel did contain burnable absorbers, it will be conservatively assumed that all fuel contained burnable absorbers with the most rodlets possible and with the highest B-10 loading of any design. Further, it will be conservatively assumed that throughout the entire burnup of the assembly the burnable absorber remains displacing water. (The B-10 content is depleted.) The penalty due to this is about 2% in k. Since burnable absorbers are generally removed after one cycle the real effect would be less than 1% in k.

The DOE topical report [2] had excluded integral burnable absorbers due to lack of chemical assay data. The assumption that all assemblies contain removable absorbers is sufficiently conservative to cover all integral burnable absorbers. The Westinghouse ZrB_2 coating produces slightly more reactive fuel for burnups greater than 15 GWD/MTU than fuel without the coating but effect is far less than 1% in k. Gd and Er integral burnable absorbers leave sufficient even isotopes that the reactivity of fuel without these elements is greater at all burnup than the fuel that contained the burnable absorbers.

5.5. Control Rods

Control Rods harden the spectrum and produce more reactive fuel. Generally, the control rods are fully withdrawn and have no effect. At power only the lead control bank is allowed to be inserted. Thus less than 7% of all the fuel assemblies would be affected in any cycle by control rods even if they were inserted to their Tech Spec. limit. An exception to the general position on control rods is the part length control rods. These control rods are designed to flatten the power distribution which compensates for the spectral hardening effect.

The depletion analysis will not take into account control rods. Analysis will be performed to assure that there is sufficient margin due to ignoring fission products to cover this effect. The effect of control rods is reduced from previously reported since the effect is the reactivity difference between being depleted with control rods versus burnable absorbers. Previously, the reactivity difference was between being burned with control rods versus being burned with no inserts.

Note that the effect of control rods on the axial burnup distribution is handled conservatively by the axial burnup distribution assumptions that are discussed in the next subsection.

5.6. Axial and Horizontal Burnup Distribution

DOE commissioned a study of the axial burnup distribution in actual PWRs. The work produced a database of 3169 axial burnup shapes. [7] This database contains shapes from cases with control rods inserted, part length control rods, and transition cores between full length cores to axial blanketed cores. The database uses 18 axial nodes. The work for DOE's Topical Report [2] identified three limiting axial shapes from the 3169 shapes. Those limiting shapes are used for the burnup analysis. Figure 5 reproduced from a report by Parish and Chen shows the reactivity effect of various shapes. [8] As can be seen in this figure the outlying shapes are far off from the mean. Figure 6 shows the three profiles selected compared to the mean profile for the same burnup range. The three burnup ranges are 0 to 18 GWD/MTU, 18 to 30 GWD/MTU and 30 and up GWD/MTU.

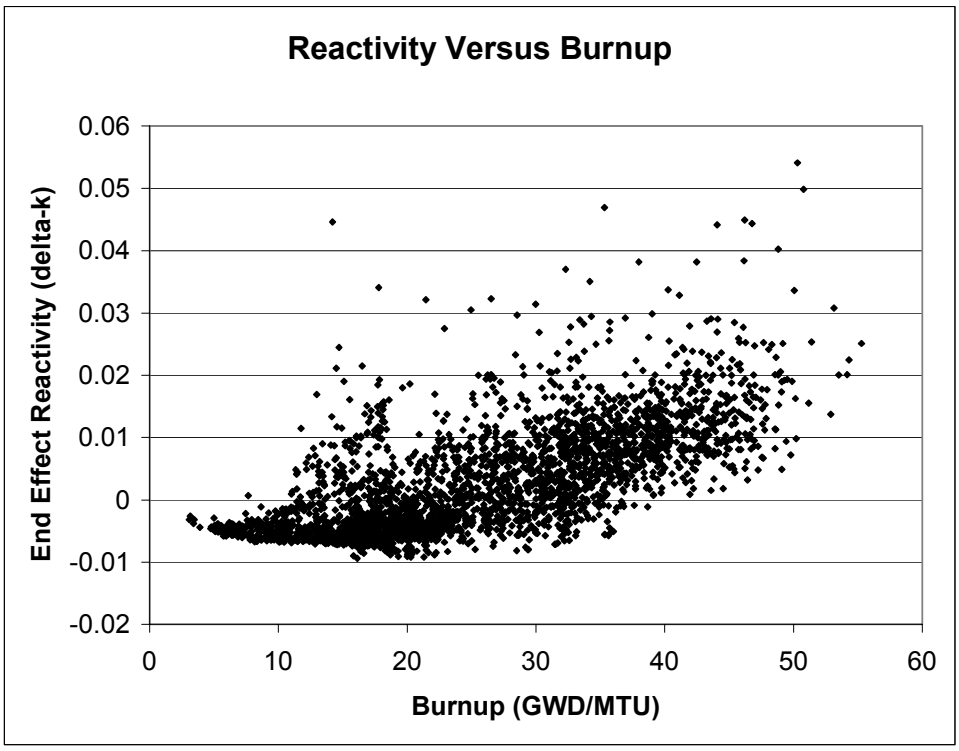

FIG. 5. End Effect Reactivity for the 3169 Axial Shapes [8].

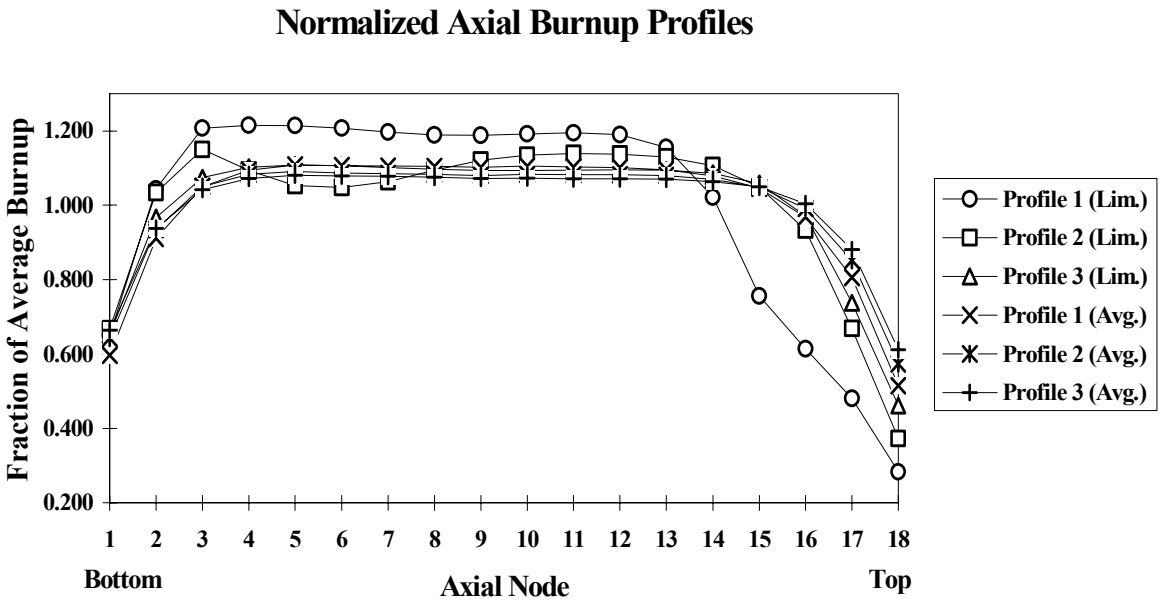

FIG. 6. Limiting Axial Burnup Profiles and Average Axial Profiles

The horizontal tilt from DOE's Topical Report will be used. This tilt has a less than 0.003 effect on k_{eff} for large casks. The tilt is 33%, 25%, and 20% for the same three burnup ranges used for the axial shape.

6. GENERATION OF LOADING LIMITS

Using SCALE 4.4a SAS2H for depletion with the conservative depletion assumptions and then correcting the isotopics with the correction factors and finally using KENOVa for criticality analysis utilizing the bounding axial and radial tilts it is possible to generate a curve of the minimum burnup required for a given initial enrichment that will meet the USL. The only assumption not discussed to this point is the cooling time. The NRC stated in ISG-8 that the cooling time must be 5 years. This application will utilize 9.5 years to match other constraints from the shielding analysis. Since it has already been demonstrated that the criticality validation covers cooling times up to 10 years this should be acceptable.

In exchanges with the NRC, it is clear that the NRC does not wish to have a large number of loading curves for a given cask. In DOE's topical report [2] it was allowable to have a series of loading curves as a function of the number of burnable absorber rodlets or even cooling time. For the CASTOR® X/32 S cask the loading curves will be restricted to one cooling time and one burnable absorber loading. There will, however, be loading curves for the most limiting Westinghouse fuel 15X15 and the most common fuel Westinghouse 17X17. It will be shown that the Westinghouse 17X17 loading curves are conservative for the Westinghouse 14X14, CE 14X14, and the St. Lucie fuel. Loading curves will be generated with no ARMS, 8 ARMS and 16 ARMS.

7. VERIFICATION OF FUEL CHARACTERISTICS

ISG-8 requires that each assembly undergo a burnup verification measurement prior to loading into the cask. The ISG-8 requirements will be followed. In addition to measuring the assembly average burnup the CASTOR® X/32 S application will require verification of the minimum average burnup in the top two feet (61 cm). This minimum burnup corresponds to the axial burnup distribution used in the criticality calculations. For each assembly there is a loading curve burnup requirement associated with its initial enrichment. For assemblies where this burnup requirement is less than 18 GWD/MTU, it will be required to show that the minimum average burnup in the top 2 feet is greater than 46% of the loading curve burnup. For assemblies where this burnup requirement is greater than 18 GWD/MTU but less than 30 GWD/MTU, it will be required to show that the minimum average burnup in the top 2 feet is greater than 66% of the loading curve burnup. For assemblies where this burnup requirement is greater than 30 GWD/MTU, it will be required to show that the minimum average burnup in the top 2 feet is greater than 72% of the loading curve burnup. This requirement can be met by use of the incore monitoring equipment or by an axial scan associated with the assembly burnup verification measurement.

8. VERIFICATION OF CRITICALITY MARGIN

The NRC in ISG-8 requires verification of criticality margin which comes from the fission products. This verification comes from analysis of OECD benchmarks and by calculation of the fission product worth via an alternative code. For the CASTOR® X/32 S cask MCNP will be utilized for verification by an alternative code. The OECD cases that will be analyzed are the Phase II-B benchmarks documented in reference 9.

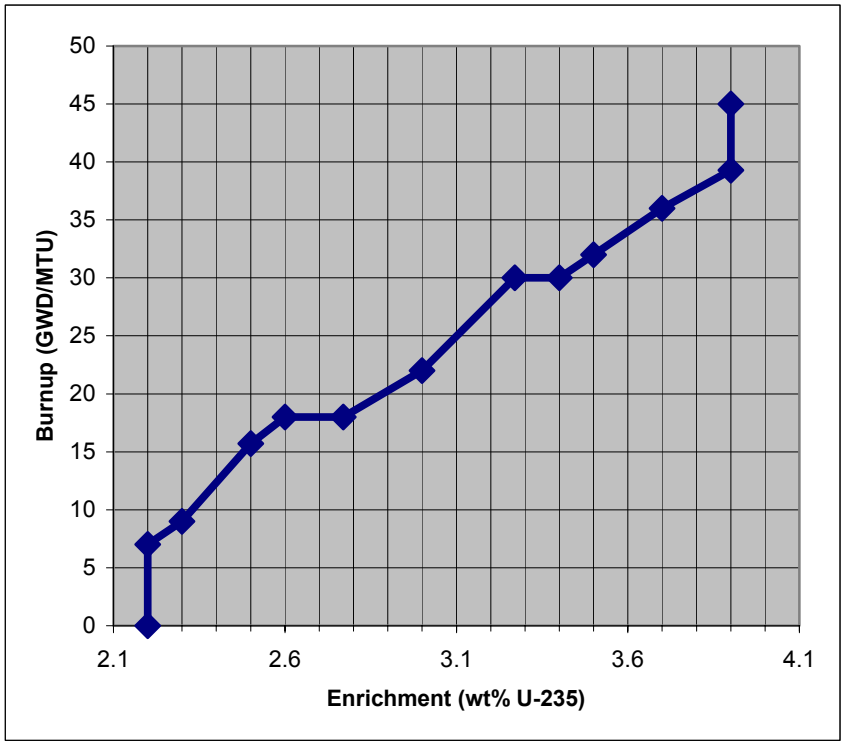

FIG. 7. Minimum Burnup Requirements for Westinghouse 15X15 Fuel in the CASTOR® X/32 S Cask with No ARMs.

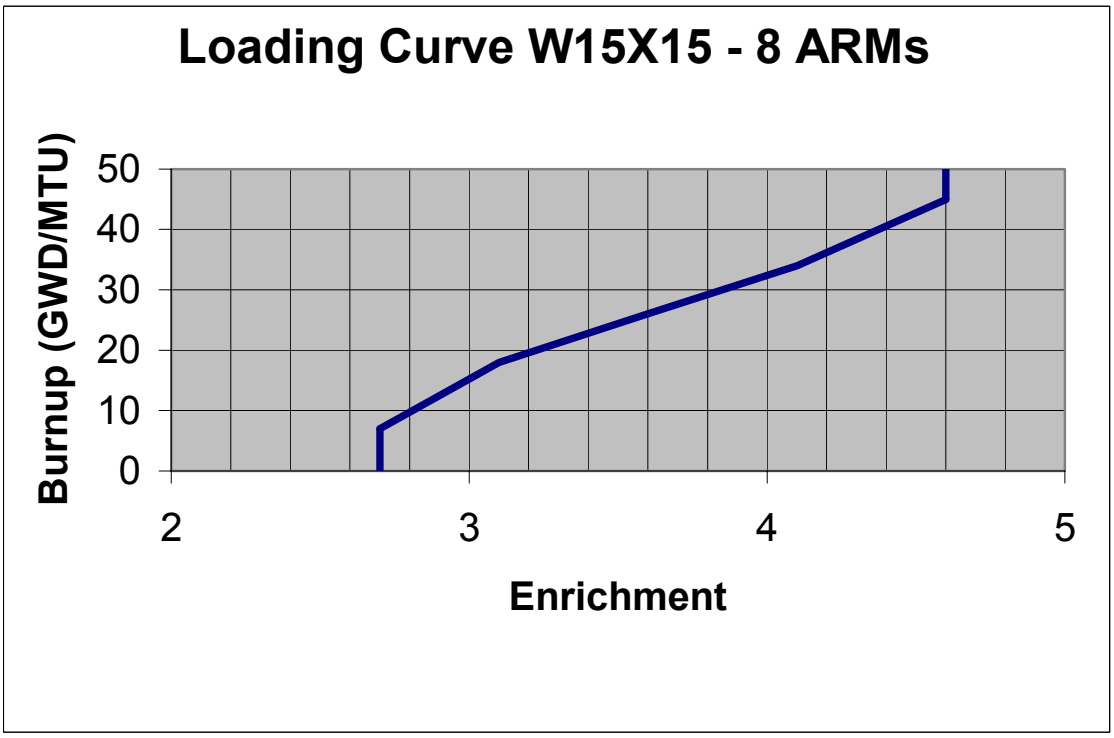

FIG. 8. Minimum Burnup Requirements for Westinghouse 15X15 Fuel in the CASTOR® X/32 S Cask with 8 ARMs

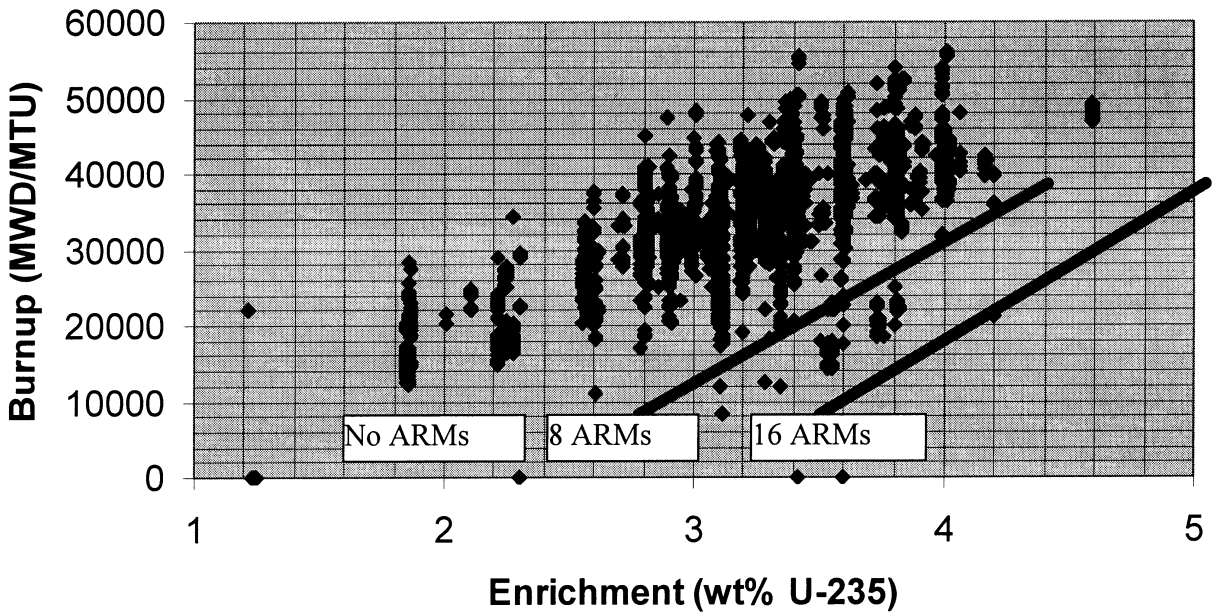

FIG. 9: Discharged Westinghouse 15X15 Assemblies Shown By Enrichment and Burnup Along with Approximate Loading Curves.

9. SUMMARY OF RESULTS

The analysis is still continuing, however, some results can be presented. Figure 7 shows the minimum burnup requirements for Westinghouse 15X15 fuel in the CASTOR X/32 S Cask. Figure 8 shows more preliminary results for the same fuel and cask but using 8 ARMS for additional reactivity control. Although the burnup requirements may seem large they appear to be small enough to allow loading must existing fuel with only a minimum number of ARMS. Figure 9 shows the burnup and enrichment of all the currently discharged Westinghouse 15X15 fuel in the US. The approximate loading curves are drawn on the same figure to show that most of the fuel will require no ARMs and most of the remaining fuel would only require 8 ARMS. A 16 ARM loading curve is drawn to show that with 16 ARMs all the fuel would be able to be loaded.

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PHENOMENA AND PARAMETERS IMPORTANT TO BURNUP CREDIT

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Abstract

Since the mid-1980s, a significant number of studies have been directed at understanding the phenomena and parameters important to implementation of burnup credit in out-of-reactor applications involving pressurized-water-reactor (PWR) spent fuel. The efforts directed at burnup credit involving boiling-water-reactor (BWR) spent fuel have been more limited. This paper reviews the knowledge and experience gained from work performed in the United States and other countries in the study of burnup credit. Relevant physics and analysis phenomenon are identified, and an assessment of their importance to burnup credit implementation for transport and dry cask storage is given.

1. INTRODUCTION

In contrast to criticality safety analyses that employ the fresh-fuel assumption, credit for fuel burnup necessitates careful consideration of the fuel operating history, additional validation of calculational methods (due to prediction and use of nuclide compositions for spent fuel), consideration of new conditions or configurations for the licensing basis, and additional measures to ensure proper cask loading. For pressurized-water-reactor (PWR) fuel, each of these four areas have been studied in some detail over the last decade and considerable progress has been made in understanding the issues and developing the information needed for an effective safety evaluation that applies burnup credit. More recently, studies to expand the understanding needed to use burnup credit with spent nuclear fuel (SNF) from boiling-water reactors (BWRs) have been performed in the United States. The purpose of this paper is to identify the characteristic parameters and physics phenomena that are important to understanding burnup credit and review the current knowledge as gleaned from the studies performed in the United States, in other countries, and within international organizations. The following sections discuss the parameters and physics associated with the nuclides important to burnup credit, depletion and decay phenomena, and modeling of a SNF cask.

2. NUCLIDES IMPORTANT TO BURNUP CREDIT

Spent nuclear fuel contains hundreds of unique nuclides. The actual reactivity worth of the fuel is a function of the net neutron production and absorption by all nuclides present. However, if criticality calculations are performed based on all fissile nuclides and a limited subset of absorbers, the calculated value of the effective neutron multiplication factor (keff) is conservative (i.e., keff is overestimated). To date, the approach proposed in the United States for burnup credit in storage and transport casks has involved the qualification of calculated isotopic predictions via validation against destructive assay measurements from SNF samples. Thus, utilization of nuclides in the safety analysis process has been limited based on the availability of measured assay data and chemical characteristics (e.g., volatility) that might cause the nuclide to escape the fuel matrix [1, 2].

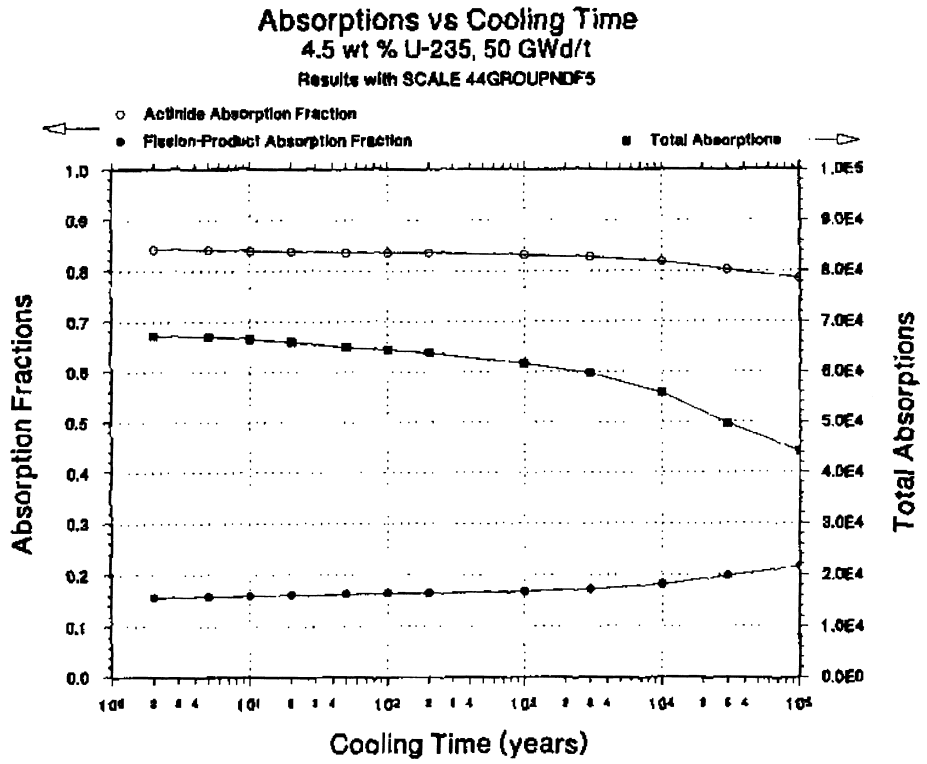

FIG. 1. Fraction of Neutron Absorptions versus Cooling Time for 4.5-wt %-Enriched PWR Fuel Burned to 50 GWd/t.

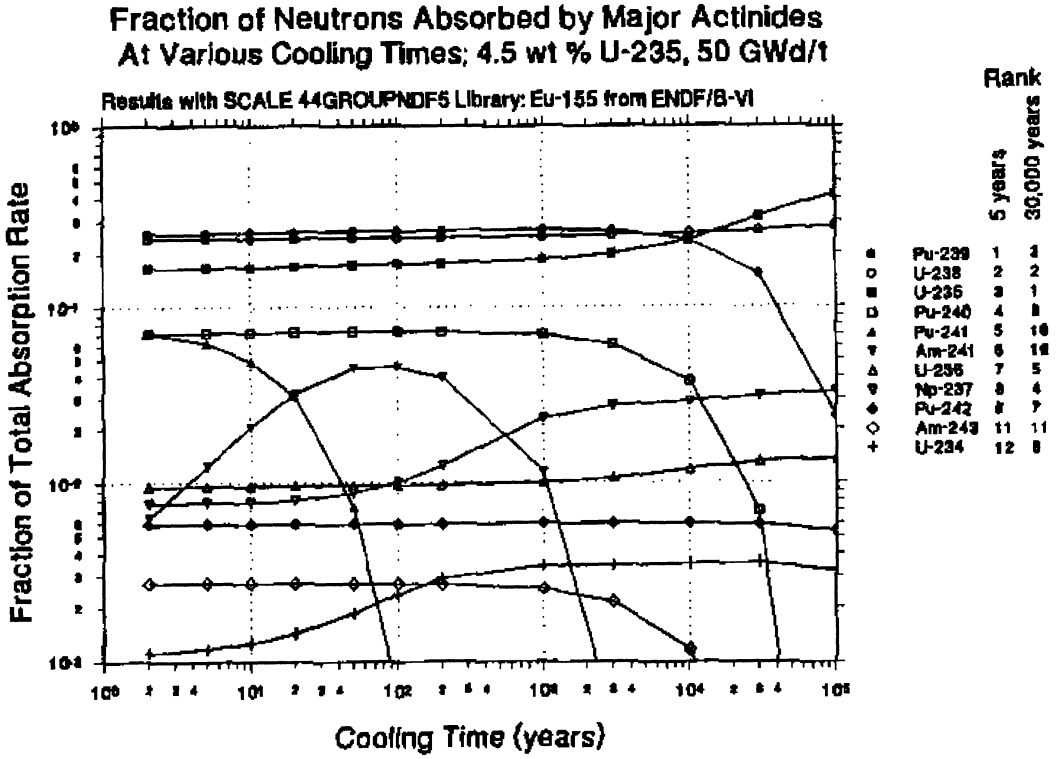

FIG. 2. Fraction of Neutron Absorbed by Major Actinides at Various Cooling Times for 4.5-wt %-Enriched PWR Fuel Burned to 50 GWd/t.

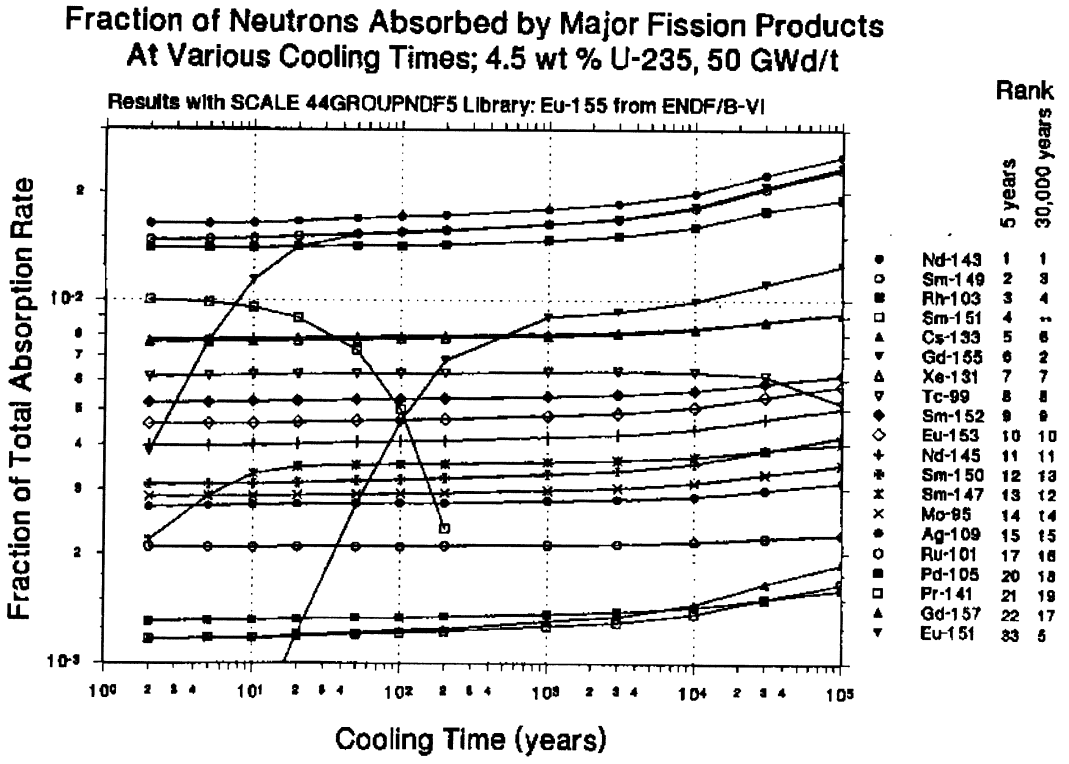

FIG. 3. Fraction of Neutrons Absorbed for Major Fission Products at Various Cooling Times for 4.5-wt %-Enriched PWR Fuel Burned to 50 GWd/t

Table I. Prime Candidate Nuclides for Burnup Credit Criticality Calculations.

PRIME CANDIDATE NUCLIDES FOR BURNUP CREDIT CRITICALITY CALCULATIONS					
^{234}U	^{235}U	^{236}U	^{238}U	^{238}Pu	^{239}Pu
^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am	^{243}Am*	^{237}Np
^{95}Mo*	^{99}Tc	^{101}Ru*	^{103}Rh*	^{109}Ag*	^{133}Cs
^{143}Nd	^{145}Nd	^{147}Sm	^{149}Sm	^{150}Sm	^{151}Sm
^{151}Eu*	^{152}Sm	^{153}Eu	^{155}Gd		

*Nuclides for which measured chemical assay data is not currently available in the United States (note a very limited amount of data is available for 243Am).

Several studies have been performed to identify the nuclides that have the most significant effect on the calculated value of keff as a function of burnup and cooling time [2, 3]. Figures 1–3 provide the results of one study [3] which performed a relative ranking based on the fraction of total absorptions for each nuclide. The adequacy of this simple ranking approach has been confirmed with more rigorous approaches that obtained the actual change in keff for an infinite lattice of rods based on a change in each nuclide [2]. The relative worth of the nuclides will vary somewhat with fuel design, initial enrichment, and cooling time, but the important nuclides remain the same. A recent study for BWR spent fuel also indicates the ranking of important nuclides changes only slightly in going from PWR to BWR operating conditions [4], and that the important nuclides are the same.

Figures 1–3 indicate that the majority of neutron absorption is caused by only a few actinide isotopes and, individually, the fission products contribute much less to neutron absorption. Within the cooling time range of interest to transport and dry cask storage (approximately 2 to 100 years), Figures 2–3 indicate that the relative importance of only a few nuclides change significantly. The buildup of 155Gd and 147Sm from the decay of other essentially non-absorbing fission products and the decay of 241Pu (14.4 y-half-life) to 241Am contribute to the decrease in keff as cooling time increases. The effect of the decay of 151Sm appears to be compensated by the commensurate buildup of 151Eu. Based on these and other studies, the nuclides listed in Table I are considered to be the prime candidates for inclusion in burnup credit analyses related to storage and transport casks. Obviously, 151Sm (90-y half-life) and 151Eu are a pair, and 151Eu only needs to be considered if the absorption credit for 151Sm must be maintained. Note, 135Cs is a relatively minor absorber that has a negligible effect on cask reactivity; however, it has been included in many previous studies because measured isotopic data currently exist.

As indicated earlier, validation of calculated isotopic predictions against experimental measurements is desirable for any nuclide upon which burnup credit criticality calculations are based. For BWR fuel, the number of nuclides for which there are measured data is significantly reduced and is limited primarily to the actinides of Table I [5]. For the most part, the fission product measurements available in the United States for PWR fuel is limited to 3–6 measurements, and prediction methods for these nuclides may not be considered to be fully validated [6]. This situation is a major reason that only partial or "actinide-only" burnup credit was considered in the U.S. Department of Energy (DOE) Topical Report on burnup credit [1] and the current U.S. regulatory guidance on burnup credit for transport and storage casks [7]. The fission product margin is still present, but since sufficient measured data for isotopic validation do not exist, credit for its negative worth has not been recommended for inclusion in safety analyses.

Table II. OECD Phase IA ΔK values(Actinides are relative to fresh fuel).

OECD PHASE IA Δ*K* VALUES (ACTINIDES ARE RELATIVE TO FRESH FUEL)			
		30 GWd/t	40 GWd/t
1-year cooled	Actinides	0.1922	0.2492
	Fission products	0.1054	0.1248
	Total	0.2976	0.3740
5-year cooled	Actinides	0.2094	0.2721
	Fission products	0.1161	0.1417
	Total	0.3255	0.4138

Table II. shows the participant-averaged incremental worth of actinides and fission products as determined by the Working Group on Burnup Credit, an international group of experts organized by the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) [8]. This particular study, one of many performed by the Working Group, involved an infinite lattice of fuel pins with an initial enrichment of 3.6 wt % ^{235}U and

nuclides nearly identical to those of Table I. The results of Table II indicate that, for these burnup values, the reactivity decrease is roughly 2/3 due to actinides, another 1/3 due to fission products. This finding is consistent with earlier work [9] for infinite lattices. However, it is important to remember that the competing effect of external absorbers in cask designs will change this ratio for finite cask analysis resulting in the fission products with less relative worth. This reduced effect is seen in Figure 4, which is based on a generic rail cask design with 5-year cooled fuel. This figure shows the reactivity worth of the eleven actinides, with measured assay data as identified in Table I, in comparison to the additional worth that can be obtained from: fission products with measured assay data as identified in Table I, all the nuclides of Table I, and all nuclides (approximately 230) for which cross-section data are available in Version 5 of the U.S. Evaluated Nuclear Data Files (ENDF/B-V). The fission products provide approximately 1/4 of the total reactivity decrease for this particular cask design, somewhat lower than the 1/3 value seen for infinite lattices.

3. PARAMETERS FOR DEPLETION ANALYSIS

It is anticipated that burnup credit will be applied for a wide variety of fuel types, each irradiated under a variety of reactor operating conditions (temperature, PWR boron concentration, BWR blade/fixed poison usage, etc.). If a cask design is intended to accept such a variety of fuel, assumptions that encompass the known variations must be employed in depletion calculations to ensure that the nuclide content of the fuel is conservatively represented. Several studies [2, 10 to 13] have been performed to assess the effect of depletion modeling assumptions on SNF nuclide predictions. In these parametric analyses, calculated nuclide concentrations were used to calculate k_{eff} for infinite SNF pin lattices and generic casks loaded with SNF. Trends in the neutron multiplication were then examined as a function of each parameter to determine the conservative direction (e.g., high temperature vs. low temperature) for that parameter, and the magnitude of the effect over a realistic operating range.

For each parameter studied in Refs. [2, 10 to 13], the sensitivities of the neutron multiplication to changes in the parameter increases with increasing burnup. Furthermore, with the exception of specific power/operating history effects, all of the trends discussed below are related to spectral hardening. Spectral hardening results in an increased production rate of plutonium from increased epithermal neutron capture in ^{238}U. Enhanced plutonium production and the concurrent diminished fission of ^{235}U due to increased plutonium fission have the effect of increasing the reactivity of the fuel at discharge and beyond. The exact mechanisms that result in spectral hardening for various operating conditions are discussed below. Unlike the other parameters, specific power and operating history effects are driven by the balance of the various equilibrium states of the nuclides present, as a function of power.

3.1. Fuel Temperature

Studies [2, 10 to 13] of both BWR and PWR depletion models show a clear trend for increased conservatism (i.e., increase in k_{eff} value) as the assumed fuel temperature during operation is increased. It is believed that at higher fuel temperatures, resonance absorption in ^{238}U is increased due to Doppler broadening, resulting in spectral hardening and increased plutonium production. The effect is burnup dependent, increasing linearly with increasing burnup. Thus conservative SNF nuclide inventories are predicted by assuming an upper estimate of fuel temperature during depletion calculations. The bounding case is for high-

burnup fuel and Ref. [2] shows that the reactivity worth of temperature change is on the order of 5 pcm/K (pcm = percent mill = 10^{-5} $\Delta k/k$) for an infinite lattice of PWR fuel pins and 4 pcm/K for a generic cask [10]. Ref. [12] shows similar behavior for an infinite lattice of BWR fuel. Thus use of the maximum pellet-averaged temperature in the depletion analysis should be acceptable for PWR and BWR depletion analyses. A value of 1000 K would seem appropriately conservative to cover normal PWR reactor operations.

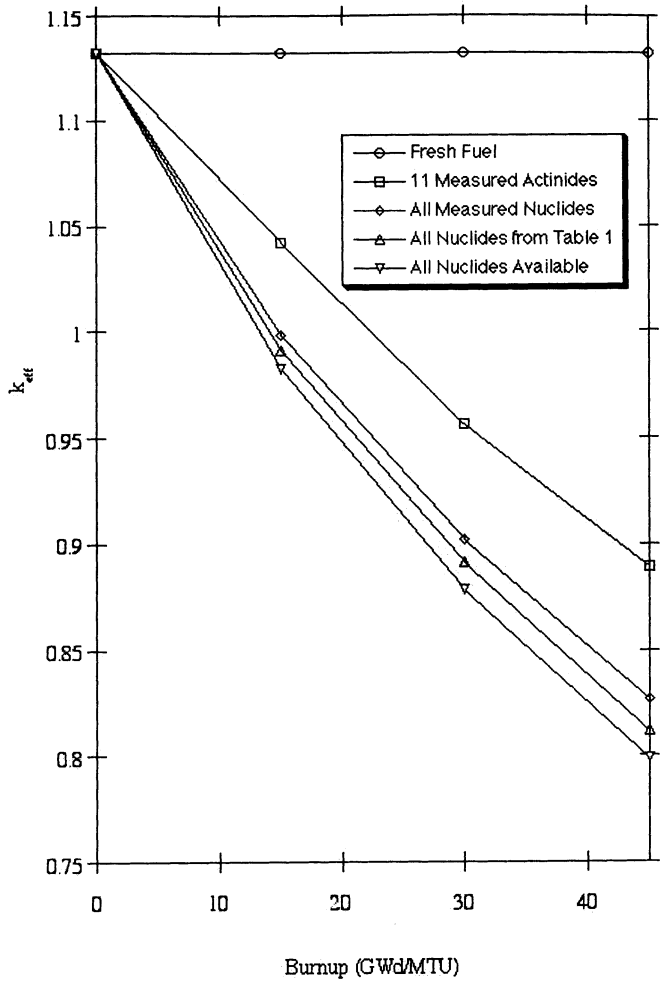

FIG. 4. Values of k_{eff} for a Generic Rail Cask as a Function of Burnup Using Different Sets of Isotopic Assumptions and 5-year Cooling Time.

3.2. Moderator Temperature/Density

As with fuel temperatures, calculations performed with varying moderator temperatures show that nuclide compositions are most conservative with respect to neutron multiplication when calculated assuming an upper bound on moderator temperature (e.g., core outlet temperature) [2, 10 to 13]. Although the mechanisms are different, the net effect is the same. In a PWR, as the moderator temperature increases, the moderator density decreases. Decreased density results in reduced moderation, which results in spectral hardening. The response is close to linear, but has a slight exponential shape with increasing moderator density (due to the fact that water density is not linear with respect to temperature). The reactivity effect also increases with increasing burnup. For the bounding case of high-burnup fuel, Ref. [2] shows a reactivity worth of about 90 pcm/K for an infinite lattice of PWR fuel pins and Ref. [10] indicates 35 pcm/K in a cask environment. In general, however, the variation in temperature

and corresponding density is relatively small in a PWR design. Thus, use of the maximum core outlet temperature (e.g., 600 K) is recommended.

Spectral hardening resulting from decreased moderator density is intentionally applied in the control of a BWR. However, the net effect is unchanged from the effect discussed for PWR designs. In BWR systems, moderator temperatures change very little axially once boiling begins in the flow channel. However, reactor operation allows significant variation in axial moderator density as the void fraction increases with increasing height. The void fraction can change significantly both axially and as a function of time. Hence, it is more instructive to study depletion effects as a function of moderator density rather than moderator temperature. Reference [12] demonstrates that for an infinite lattice of BWR assemblies, k_{inf} increases linearly with decreasing moderator density and the trend is more pronounced as the SNF burnup increases. The magnitude of the effect is on the order of 10^3 pcm/(g/cm^3) for high burnup fuel. Thus, the highest average void fraction (minimum average moderator density) would appear to be the simple bounding value to use for depletion analysis of BWR fuel. Since the reactivity of BWR fuel in a cask is driven by the fuel at the top of the assembly, it is anticipated that using the highest average void fraction should be a prudent, yet practical assumption for the safety analysis. However, additional work in this area may be warranted to substantiate the initial findings.

3.3. Soluble Boron

Soluble boron is used to control excess reactivity in PWRs. Soluble boron concentrations of 1000−1500 ppm boron are typical at beginning-of-cycle and decrease to 0−200 ppm at end-of-cycle. Depletion calculations may model the boron change in steps, or assume an average boron concentration for a full cycle. Studies have been performed to assess the effect of the soluble boron concentration used during depletion [2, 10, 13, 14]. The results of these bounding high-burnup calculations show a clear linear increase in reactivity with increased boron concentration at a rate of approximately 3 pcm/ppm for an infinite lattice of pins and 3.5 pcm/ppm in a cask configuration. Again, although the mechanism is different from that which occurs in fuel and moderator temperature variations, the end result is the same. Spectral hardening results from the absorption of thermal neutrons in the moderator by the soluble poison. As with temperatures, the effect of higher boron concentrations is more significant with higher burnup values, since more conversion occurs over the fuel cycle. Use of an average cycle boron value of 750 ppm should be adequately bounding based on the studies performed; however, the establishment of a bounding value for the maximum average boron per cycle based on boron let-down curves would be informative.

3.4. Specific Power and Operating History

The effect of various operating histories (variations in specific power with time) on the reactivity of spent fuel has been studied for a limited set of hypothetical power histograms [2, 10, 12]. Rather than attempt to determine a real operating history that would bound all other operating histories, histograms were developed to represent the key aspects of operating histories (e.g., extended downtime early in life, extended downtime late in cycle, high- power operation early in life, short intercycle downtimes, long intercycle downtimes, etc.). Results showed a wide variability in response due to the significantly different decay rates and equilibrium concentrations for the nuclides studied. In general, low-power operation near the end of cycle produces the highest reactivity due to decreased fission product inventory. However, the opposite is true when only actinides are considered for burnup credit - high-

power operation is more conservative at end of life. Fission product worth is more sensitive to specific power than that of actinides; thus, when both are present, the net effect is driven by fission product behavior. Hence low-power operation toward end of life yields the most conservative estimate of reactivity. The net effect is rather small, up to 200 pcm for the operating histories considered. It appears that the optimum approach would be to assume a simple continuous-power operating history, and add in a margin to account for operating-history-induced effects.

The effect of specific power assumed during depletion calculations has also been studied independently of operating history for PWRs [2,10]. Although an operating history is simply a time-varying specific-power profile, it is important to understand the effect of the magnitude of specific power when held constant with time. Calculations with both actinide and fission product credit show a trend for conservative prediction of fuel reactivity worth when fuel is burned at lower specific power for a longer period of time for a given burnup. The magnitude of the conservatism increases with increasing burnup. However, the opposite is true for calculations in which only actinides are considered in criticality calculations. For actinide-only credit, higher specific powers result in the most conservative set of isotopics. The magnitude of the conservatism also increases with increasing burnup. The difference in behavior between actinides and fission products is due to the relatively short decay times of fission product precursors relative to actinides.

Recent work [12] has shown that for high-burnup fuel with fission products present, the behavior of the SNF neutron multiplication as a function of specific power departs from a linear response. For high-burnup fuel, the neutron multiplication initially increases with increasing specific power, before turning (e.g., in the range of 10−20 MW/t) and decreasing as specific power continues to increase. Thus, there is a specific power that maximizes the neutron multiplication for high-burnup fuel with actinides and fission products assumed. The phenomenon will require further study to understand and quantify.

3.5. Burnable Absorbers

Burnable absorbers may be classified into two distinct categories: (1) Burnable Poison Rods (BPRs) and (2) Integral Burnable Absorbers. BPRs are rods containing neutron absorbing material that are inserted into the guide tubes of a PWR assembly during normal operation and are commonly used for reactivity control and enhanced fuel utilization. Due to the depletion of the neutron absorbing material, BPRs are often (but not always) withdrawn after one-cycle residence in the core. In contrast to BPRs, integral burnable absorbers are burnable poisons that are a non-removable or integral part of the fuel assembly once it is manufactured. An example of an integral burnable absorber is the Westinghouse Integral Fuel Burnable Absorber (IFBA) rod, which has a coating of zirconium diboride (ZrB_2) on the fuel pellets.

The net effect of poison rods is the same as that of soluble boron, since the same mechanism applies: preferential removal of thermal neutrons. However, rod effects are more localized, resulting in localized spectral hardening, non-uniform burnup across the assembly at a given axial height, and atypical axial burnup profiles. Recently completed studies have demonstrated that assemblies exposed to BPRs will show an increased k_{eff} in the range of 0.5% to 3% Δk depending on the number and poison loading of BPRs present, the length of the exposure to BPRs, the initial fuel enrichment and the burnup of the assembly. Inclusion of the axial burnup distribution reduces the reactivity increase associated with the BPRs. This is due to the fact that the lower burnup regions near the ends, which control the reactivity of the

fuel when the axial burnup distribution is included, have less burnup, and thus less than average burnup exposure to the BPRs.

Assuming maximum BPR exposure during depletion would be a simple, conservative approach to bound the reactivity effect of BPRs; where maximum BPR exposure may be defined as the maximum possible number of BPRs with the most bounding BPR design (i.e., most bounding geometric design and maximum possible poison loading) for the entire depletion. Other less conservative approaches, incorporating risk-informed approaches regarding the percentage of assemblies exposed to mutiple cycles, will be explored in the future.

A study has recently been completed that investigated the impact of integral burnable absorbers on the k_{eff} values in cask environments. Depending on the design and loading of neutron poison, the presence of integral burnable absorbers can slightly lower or raise the k_{eff} values of SNF assemblies, in comparison to assemblies without the integral burnable absorber. Integral burnable absorber analyses for multiple designs have been studied, and the maximum increase in k_{eff} is less than that identified for assemblies depleted with BPRs present.

The impact of control rods used during power operations can also have the effect of increasing fissile plutonium production at the ends of the fuel. Parametric studies to understand the potential magnitude of these effects are planned for the near future.

Table III: Summary of information on depletion modelling parameters.

Parameter	Bounding condition	Estimated sensitivity	Recommended conservative value/model
Fuel temperature	Highest temperature	4−5 pcm/K	Max. pellet-average temperature
Moderator temperature (PWR)	Highest temperature	35−90 pcm/K	Maximum core outlet temperature
Moderator density (BWR)	Lowest density	10^3 pcm/(g/cm^3)	Minimum channel outlet density
Soluble boron concentration	Highest concentration	3−3.5 pcm/ppm	Maximum cycle-averaged concentration
Operating history	High power late in life (actinide-only)	N/A	Assume simple operating history, with margin of 200 pcm or more
Specific power	High specific power (actinide-only)	N/A	High but credible specific power
Fixed/Integral burnable absorbers	Burnable absorbers present during depletion	0.5–3% Δk over full range	Maximum absorber loading used for full irradiation history.

3.6. Summary of Depletion Modeling Parameters

Table III. summarizes the discussion in the preceding paragraphs, including specific power and operating history effects. No specific recommendations for bounding parameters are given. Although expected values are listed in the preceding subsections, these values should be confirmed or revised by a survey of operational data before firm recommendations are made. Simultaneous use of realistic bounding parameter values in a depletion model provides a simple, prudent approach to the modeling process since it is unlikely that any fuel would be depleted under all such conditions simultaneously. However, the use of bounding values and/or models may not be the most appropriate for a risk-informed approach to implementing burnup credit. Work to investigate more realistic approaches based on actual ranges of operating conditions and the statistical probability of "outlier" bounding conditions will be explored in the future. However, a key to the success of such approaches is development of a database that provides information on the range of actual operating conditions with sufficient data points that "typical" conditions can be established. A reference industry report establishing a defensible value for PWR and BWR operations would be beneficial to facilitate future safety analyses.

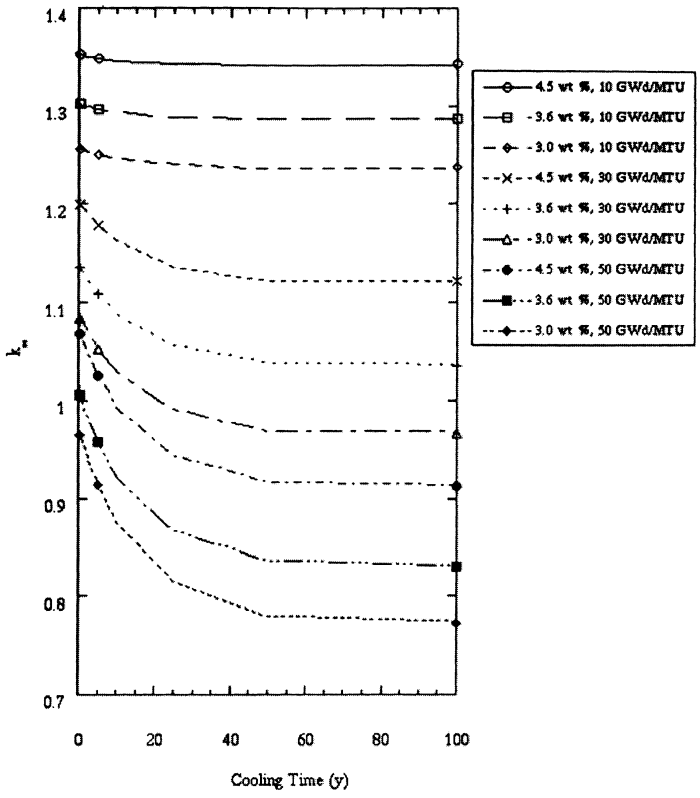

FIG. 5. Plot of k_{eff} versus Cooling Time for Various Enrichments and Burnup Values

4. COOLING TIME

Fuel discharged from a reactor increases in reactivity for several days due to the decay of short-lived poisons. After this point, reactivity decreases continuously with time out to about 100 years, at which time it begins to increase. The reactivity continues to increase until a second peak at around 30,000 years, after which time it begins decreasing again [3]. The reactivity of the second peak is always less than that occurring at 5 years when actinide and

fission products are used in the criticality analysis. This means that an assumed cooling time for 5 years is conservative for any cooling time beyond 5 years. The magnitude of the conservatism depends on the initial enrichment and burnup of the fuel [2,3].

The effect of cooling time on k_{eff} for an infinite PWR pin-cell lattice is shown in Fig. 5 for various burnup and initial enrichment values. Note that as burnup increases, the effect of cooling time is more pronounced due to the increased quantity of ^{241}Pu and fission products relative to the remaining inventory. Reference [3] provides a comparison of absorption fraction versus burnup and further illustrates this increase in the negative reactivity worth from ^{241}Pu decay and fission product absorption. Since the reactivity of low-burnup fuel at the ends of the SNF is rather insensitive to cooling time and the reactivity of higher burned fuel decreases significantly with cooling time, the relative reactivity worth of the ends will increase with cooling time.

5. AXIAL BURNUP PROFILES

5.1. Phenomena Associated with Axial Effects

The dynamics of reactor operation result in non-uniform axial-burnup profiles in fuel with any significant burnup. At beginning of life in a PWR, a near-cosine axial flux shape will begin depleting fuel near the axial center of a fuel assembly at a faster rate than at the ends. As the reactor continues to operate, the cosine flux shape will flatten because of the fuel depletion and fission product poisoning that occurs near the center. However, because of the relatively high leakage near the end of the fuel, burnup will drop off rapidly near the ends. Partial length absorbers or non-uniform axial enrichment loadings can further complicate the burnup profile. In a BWR, the same phenomena occur [12], but the burnup profile is also influenced by the significantly varying moderator density profile and by non-uniform axial loadings of burnable poison rods and uranium enrichment.

The most reactive region of spent fuel is toward 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[2]. A fairly extensive review of axial burnup distribution issues that are important to burnup credit criticality safety analyses is presented in Ref. [15]. The fact that there is a difference between the k_{eff} value calculated assuming an axially varying burnup profile and that calculated assuming a uniform (flat) burnup profile (associated with the average assembly burnup value) has become known as the “end effect”.

Participants in the OECD/NEA Working Group on Burnup Credit performed criticality calculations for a 3-D infinite lattice of axially finite PWR pin cells [16]. The following items were noted in the results with respect to the end effect: (1) the end effect increases with increasing burnup and cooling time; (2) it is most pronounced when fission products are present; (3) the end effect is negative for low-burnup and short cooling times, but becomes positive and of greater magnitude at high-burnup and cooling time; (4) the cross-over from negative to positive occurs around 25 GWd/t when fission products are modeled, and near 30 GWd/t when fission products are not modeled; and (5) the crossover from negative to positive occurs at slightly higher burnup when fuel enrichment increases. In general, the same trends noted here for the infinite array model were also noted in the cask model analyzed by the participants [17].

In a BWR, the burnup profile is further complicated by several factors, including: (1) axially and time varying moderator density, (2) axially and radially varying fuel enrichments, (3) axially varying poison rod enrichments, and (4) partial control rod insertion. The reactivity of BWR fuel increases with burnup to a maximum or peak reactivity where the integral absorber (Gd) is nearly depleted. When considering the axial-burnup profile, it becomes apparent that local heights will not reach their peak reactivity simultaneously. Rather, the integral absorber will be depleted earlier near the axial center, and thus the reactivity will peak at the center while significant integral absorber is still present at the ends. Similar to PWR fuel, the axial burnup distribution results in an increasing positive end effect with increasing burnup. However, early work [12, 18] has shown the magnitude of the reactivity increase associated with the axial burnup distribution in BWRs may be larger than that which is typically observed for PWR fuel.

5.2. Profile Database

The true axial-burnup distribution is not known for the majority of spent fuel assemblies that will be loaded in a cask. In general, only the average burnup is known and documented in plant records associated with each SNF assembly. Thus to be conservative, one must identify and assume an axial-burnup profile that is realistic but is limiting in terms of the value of k_{eff} associated with the axially varying SNF nuclide compositions. To date, attempts to bound PWR profiles [2, 10, 11] in the United States have been based on a database of 3169 axial-burnup profiles for PWR assemblies [19]. The database of Ref. 19 consists of burnups calculated by utilities or vendors for a discrete number (18−24) of axial heights based on core-follow calculations and in-core measurement data. Although the profiles in the database are not measured directly, the use of the same analysis procedures for reactor core-following analyses inspires confidence that the profiles are representative of the actual fuel burnup.

If it is desirable to continue to base limiting axial profiles on profiles found to be limiting from a database, then it may be necessary to expand the existing database to include a broader variety of fuel designs, especially some of the more recent fuel designs. Furthermore, since control rods and partial-length absorbers can have a significant effect on axial profiles, a decision must be made whether to include or exclude such conditions in a database. Information on the use of control-rod insertion during normal reactor operations would be beneficial to better study and understand the potential impact on the axial profile and/or the SNF nuclide composition.

No attempt has been made to define a bounding profile for BWR fuel assemblies due to the lack of a database of burnup profiles. The fact that BWR fuel assemblies are manufactured with variable enrichments both radially and axially, are exposed to time-varying void distributions, contain fixed burnable poison rods, and are subject to partial control blade insertion during operation means that BWR profiles are likely to have more variation than that observed for PWR fuels. Thus a large database may be necessary to capture all of the important characteristics. Again, no such database exists for BWR profiles, and an industry activity to develop such a database would surely have value in implementation of burnup credit in cask storage and transport for BWR fuels.

5.3. Axial Modeling Approximations

In any spent fuel assembly, fuel burnup is a continuous function of axial location. However, in a numerical model, a depletion calculation must be performed for each finite burnup region in

the model to estimate the contents of the spent fuel at that burnup state. Therefore, in practical application, spent fuel models must apply a set of discrete burnup intervals in which a constant burnup over each interval is assumed. As with any differencing approach, care must be taken to ensure that the spatial discretization is fine enough to capture physical phenomena. Sensitivity studies [2, 10, 12, 15] have shown that a relatively coarse axial discretization, typically consisting of 7−11 axial regions, is sufficient to converge on the predicted eigenvalue for a spent fuel system. However, the axial discretization used in these studies and elsewhere [16, 17] is non-uniform and tailored to the shape of the burnup profile. All known spent fuel profiles tend to be fairly uniform over most of the central region, but with significantly decreasing burnup near the axial ends of the active fuel. Thus, discrete models of burnup can use one to three burnup zones to represent the majority of the length of the fuel (central region), but more discrete zones are necessary to capture the more rapid change in burnup with position near the ends of the fuel. It would be valuable to safety analysts if there were criteria for determining the number and length of zones required in the model based on the axial profile being considered. An example of such criteria would be a zone for each 10% change in burnup. Such criteria need to be developed and tested.

As noted above, the spent fuel reactivity is a function of both the burnup distribution and axial leakage; thus the boundary conditions (i.e., assembly or cask conditions at the end of the fuel) may play a role in the strategy for determining appropriate axial modeling approximations. Calculations reported to date have been based on simple axial models with a fixed set of boundary conditions. Additional work may be needed to better evaluate potential limiting boundary conditions that should be used for normal and potential accident conditions.

6. HORIZONTAL BURNUP PROFILES

Radial variations in the neutron flux, which are primarily due to leakage at the core periphery, result in a non-uniform horizontal burnup distribution over the radial extent of the reactor core. As the reactor operates, the radial flux shape flattens due to fuel depletion and fission product poisoning near the core center. However, because of the high leakage near the core periphery, burnup drops off rapidly near the periphery. Ultimately, at the end of a cycle, the individual assemblies located near the center of the core will have a relatively uniform horizontal burnup distribution, while the assemblies near the core periphery may have a significant horizontal variation in burnup [20]. Fuel shuffling schemes designed to enhance fuel utilization typically relocate assemblies within the reactor core between cycle operations. These fuel management practices tend to effectively reduce the horizontal burnup gradient in normal discharged fuel. However, a periphery assembly discharged after a single irradiation cycle may exhibit a significant horizontal burnup gradient [20].

A database containing quadrant-wise horizontal burnup gradients for 808 PWR assemblies (Westinghouse 17 – 17 and Babcock and Wilcox 15 – 15) has been prepared [20], and the database has been examined for trends with the number of operating cycles, accumulated burnup, and initial enrichment. No trend with initial enrichment was observed. However, the horizontal gradient was shown to be inversely proportional to accumulated burnup and number of cycles, which are obviously closely related. In other words, the horizontal variation in burnup decreases with increasing burnup. Axial variation of the horizontal burnup distribution has not been addressed.

The horizontal variation in burnup is a criticality safety concern in the event that two or more assemblies are placed in a configuration such that their low-burnup regions are adjacent, thus resulting in an increase in reactivity [1]. This reactivity increase will be greatest in small cask designs -such as truck casks.

7. EPILOGUE

The basic information of this paper was derived from a report [21] prepared for the U.S. Nuclear Regulatory Commission (NRC) Office of Regulatory Research to help initiate a process called Phenomena Identification and Ranking Tables (PIRT), which has been used by the NRC to identify phenomena and prioritize their importance in helping to resolve a broad technical issue. This PIRT process involves the efforts of an international panel of experts. The final report from the PIRT process (due in 2001) will build on the foundation of Ref. [21] to provide an identification and ranking of phenomena and technology issues deemed important to effective burnup credit implementation and propose a table that prioritizes the areas where technical resolution is needed. The phenomena and technology issues, as well as the ranking table, will be updated by the NRC as additional input and feedback is obtained. Information on the PIRT process can be obtained at the following web site: http://www.nrc.gov/RES/pirt/BUC/index.html. Work continues at Oak Ridge National Laboratory to improve understanding and investigate analysis approaches that will facilitate safe implementation of burnup credit in transport and storage casks.

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IMPLEMENTATION ISSUES

NON DESTRUCTIVE ASSAY OF NUCLEAR LEU SPENT FUELS FOR BURNUP CREDIT APPLICATION

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Abstract

Criticality safety analysis devoted to spent fuel storage and transportation has to be conservative in order to be sure no accident will ever happen. In the spent fuel storage field, the assumption of freshness has been used to achieve the conservative aspect of criticality safety procedures. Nevertheless, after being irradiated in a reactor core, the fuel elements have obviously lost part of their original reactivity. The concept of taking into account this reactivity loss in criticality safety analysis is known as Burnup credit. To be used, Burnup credit involves obtaining evidence of the reactivity loss with a Burnup measurement. Many non destructive assays (NDA) based on neutron as well as on gamma ray emissions are devoted to spent fuel characterization. Heavy nuclei that compose the fuels are modified during irradiation and cooling. Some of them emit neutrons spontaneously and the link to Burnup is a power link. As a result, burn-up determination with passive neutron measurement is extremely accurate. Some gamma emitters also have interesting properties in order to characterize spent fuels but the convenience of the gamma spectrometric methods is very dependent on characteristics of spent fuel. In addition, contrary to the neutron emission, the gamma signal is mostly representative of the peripheral rods of the fuels. Two devices based on neutron methods but combining different NDA methods which have been studied in the past are described in detail:1. The PYTHON ^{TM} device is a combination of a passive neutron measurement, a collimated total gamma measurement, and an online depletion code. This device, which has been used in several Nuclear Power Plants in western Europe, gives the average Burnup within a 5% uncertainty and also the extremity Burnup, 2. The NAJA device is an automatic device that involves three nuclear methods and an online depletion code. It is designed to cover the whole fuel assembly panel (Active Neutron Interrogation, Passive Neutron Counting, and Gamma Spectrometry).

1. INTRODUCTION

Criticality safety analysis devoted to spent fuel storage and transportation has to be conservative in order to make sure no accident will ever happen. In the spent fuel storage field, the assumption of freshness has been used to achieve the conservative aspect of criticality safety procedures. Nevertheless, after being irradiated in a reactor core, the fuel elements have obviously lost part of their original reactivity. The concept of taking into account this reactivity loss for criticality safety analysis is known as Burnup credit.

Burnup credit implementation in a facility for transportation or storage requires, to be able to predict, using calculation, the actual reactivity of the spent fuels in order to design the transportation or storage facilities. Moreover, it requires to be able to confirm the item traceability with a representative measurement.

The first topic – criticality calculations using results of validated depletion codes- is not the aim of this paper. Many related papers cover this first topic as those given in references [1], [2] and [3]. It is noticeable that all authors of these papers point out the measurement requirements but give very few details about existing measurement techniques and devices. To supply this need, this paper deals with the Low Enriched Uranium (LEU) spent fuel measurement problem. First, we will try to understand what is to be measured and give a description of the perfect measurement technique. Then a list of the existing techniques will be given including their respective qualities and drawbacks. Finally some devices developed by the French Atomic Energy Commission (CEA) are described in detail.

2. BURNUP CREDIT AND SPENT FUEL MEASUREMENT

2.1. Criticality and relevant physical parameters of spent fuel

For Burnup credit use, the relevant physical property of the spent fuel is its reactivity in the storage condition. In other words, how much the fuel element is able to induce neutron multiplying when in storage conditions. Unfortunately this property is not an intrinsic one, as said above ; it depends on the environment of the item. This means that, whatever technique could directly measure the reactivity, its use would require the knowledge of a transfer function linking the reactivity in the measuring conditions to the reactivity in the storage conditions.

Another way can be used ; it consists in establishing an intrinsic link between the actual components of the irradiated fuel - which are the input data of the criticality codes - and the total amount of energy extracted from it. This quantity known as the Burnup has no physical signification in the criticality field but, for a given fuel assembly and considering a particular irradiation history, it is unique and does not depend on the fuel environment after irradiation. Most of the criticality studies are done using this method [2], [3].

2.2. Burnup measurement and Burnup credit

For a given facility, criticality studies produce a value of the maximum reactivity acceptable for each fuel assembly to be authorized to be stored in the facility. This individual reactivity can be linked with an isotopic composition taking into account neutron absorption by fission products or not. Use of validated depletion codes, like CESAR [5], allows us to establish the link with Burnup considering a given initial enrichment. As a result, a conservative lower limit of Burnup can be calculated which will guarantee safety of the storage or transportation [3].

In order to complete Safety Authority requirements, the problem which has to be addressed, is then: “how to check that spent fuels are actually burned enough?”

In the past, initial reactivity was low due to low initial enrichments (< 3,25 % ^{235}U), so a simple evidence of irradiation was enough to make sure the actual reactivity was correct. This evidence can easily be gained through a simple total gamma measurement.

At the present time, in order to achieve high Burnup, the uranium 235 initial enrichments are more than 4%. Depending on the design of facilities and whether fission product effect is taken into account or not, Burnup credit use requires that actual Burnup has to be higher than achievable during one single irradiation cycle. (e.g extremity Burnup (BU) higher than 9 GWd/t_{HM} for 4 % ^{235}U enriched[1] which is an extremity Burnup reached after at least 2 irradiation cycles). In such conditions, accurate and reliable non-destructive measurement techniques have to be used.

[1] Given as an example, the $9GWd/t_{HM}$ value for lower limit of extremity Burnup is related to 16x16 KWU fuels ^{235}U enriched from 3.55 to 4 % in TN 13-2 transportation casks with fuel basket n° 928 (agreement n° F/274/B(U)F 85 G m (valid till 01/10/99)).

Characteristics of the hypothetical perfect measurement technique are summarized below :

1. It should be an absolute measurement with no calibration curve and no detector yield to be known,
2. The measurement should be representative of the whole fuel element (not only the external rods),
3. It should be insensitive to cooling time and irradiation history knowledge,
4. It should determine itself the initial enrichment,
5. And last but not least, the device should be inexpensive and easy to use.

Unfortunately, this perfect method does not exist, but many different assays have been studied and some of them developed into instruments. We will now detail the measurement technique characteristics and describe the physical behavior of the spent fuel emissions in order to understand better their advantages and drawbacks.

2.3. Sorting the Measurement Methods

Many NDA methods devoted to spent fuel characterization have been studied [4]. Most of them are based on neutron and gamma emissions. Depending on their particular features and interests, they are used in the laboratories, in the industrial reprocessing plants or for safeguard purposes. In order to make their description clear ; let us build a classification related to some relevant characteristics of the measurement methods.

Firstly, we can divide the different techniques into 2 kinds whether they give a result with irradiation history dependency or not. Irradiation history dependency means that, for a given Burnup, the physical parameter measured can take different values. It also means that for a measured value of the assayed physical parameter (neutron or gamma emission), Burnup can take different values depending on the irradiation history. Inside the reactor core, the neutron flux is described by a time function making allowance for the reactor's power history, hence the expression "irradiation history". By way of example, Figure 1. shows typical irradiation histories of a pressurized water reactor. Figure 1. shows two histories resulting in the same Burnup. Assembly No. 2 began its irradiation before assembly No. 1 but was withdrawn from the reactor during cycle No. 4. In addition, the specific power was different ; fuel No. 1 kept always the same specific power, fuel No. 1 did not.

Secondly, another sorting can be made checking to see whether the measurement requires the detector yield knowledge or not. If required, the detector yield has to be determined using either a representative standard or a calculated yield using simulation codes.

Thirdly, we will divide the methods depending whether the result they give is representative of the whole fuel or only a peripheral part of it.

To summarize, we will call:

1. *Dependent* or *independent*, the methods that require or not irradiation history knowledge,
2. *Relative* or *absolute*, the methods that require or not the detector yield knowledge,
3. *Global* or *peripheral*, the methods that provide a result for the whole fuel or only its peripheral part.

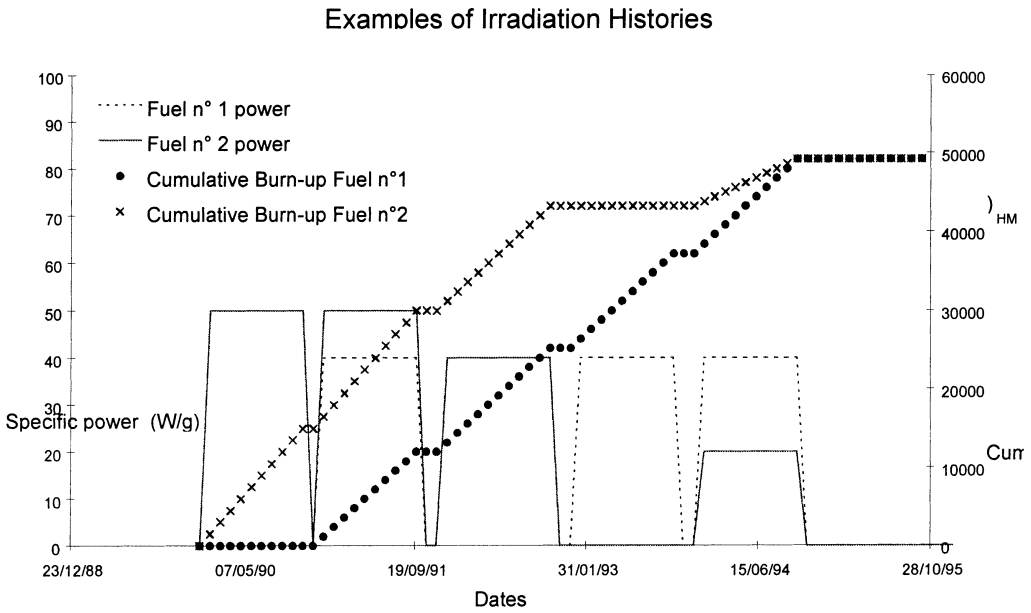

FIG. 1. Examples of irradiation histories. Fuel n°1 and n° 2 reach the same final Burnup at the same date but following two different irradiation histories which lead to different gamma and neutron emissions.

In addition, extra considerations have to be taken into account for selecting a measurement method such as: easiness of use, cost or device transportability. Since, these considerations are case dependent, we will mention them in the application section.

3. EVOLUTION OF FUEL DURING AND AFTER IRRADIATION.

Heavy nuclei that compose the fuels are modified during irradiation and cooling. At the same time, light isotopes appear as the result of fission and capture. Different nuclear interactions and decays lead to production and disappearance of heavy nuclei as well as fission and activation products.

The evolution of the fuel components result from fission, neutron capture, (n,2n) reactions and (α, β) radioactive decay. The following differential equation known as the Bateman equation [5] describes this complex process:

$$\begin{aligned}
&\frac{dN(t)}{dt}[A,Z] = \\
&\left[\Phi \quad \sigma_{c(El,BU)} \quad N(t)\right]_{[A-1,Z]} + \left[\Phi \quad \sigma_{n,2n(El,BU)} \quad N(t)\right]_{[A+1,Z]} \\
&+ \left[\lambda_{\beta+} \quad N(t)\right]_{[A,Z+1]} + \left[\lambda_{\beta-} \quad N(t)\right]_{[A,Z-1]} \\
&+ \left[\lambda_{\alpha} \quad N(t)\right]_{[A+4,Z+2]} + \left[\lambda_{TI} \quad N(t)\right]_{[A_{metastable},Z]} \qquad , (1) \\
&- \left[\Phi\left(\sigma_{c(El,BU)} + \sigma_{f(El,BU)} + \sigma_{n,2n(El,BU)}\right) \quad N(t)\right]_{[A,Z]} \\
&- \left[\left(\lambda_{\beta+} + \lambda_{\beta-} + \lambda_{\alpha} + \lambda_{TI} + \lambda_{SF}\right)N(t)\right]_{[A,Z]}
\end{aligned}$$

In this relation $N(t)_{[A,Z]}$ is the atomic abundance of the isotope with charge number Z and mass number A.Φ is the neutron flux,σ_c,σ_f,$\sigma_{(n,2n)}$ are the cross sections respectively for capture, fission and (n,2n) reactions, λ_i the decay constants for β^+, β^-, α, spontaneous fission and isomeric transition reactions.

Contrary to decay constants, which are intrinsic values, neutron cross sections depend on irradiation conditions. In the relation (1), neutron cross sections are summed into one energy group representative of the neutron energy spectrum and auto-protection effects.

$$(\text{i. e. } \sigma_i = \int_E \sigma_i(E) \cdot \Phi(E) \cdot dE)$$

During fuel life these parameters change. As a result, for each kind of fuel, neutron cross-sections have to be tabulated regards to initial enrichment and Burnup. This important work has been carried out to provide the French reprocessing company COGEMA using a calculation tool CESAR [5]. This code is able to calculate the components and emission of fuels taking into account their particular geometry, initial enrichment and irradiation history. An on line version of this code has been developed for spent fuel NDA characterization purposes.

4. PHYSICAL PRINCIPLES OF SPENT FUEL CHARACTERIZATION

All known NDA techniques for spent fuel characterization use spontaneous or induced nuclear emissions. The gamma rays as well as neutron emissions are used [4]. We will describe the properties of each emission, point out their link with the Burnup and finally describe the measurement based on each emission.

4.1. Neutrons

4.1.1. Passive Neutron Emission

Irradiation produces new heavy isotopes in the fuel due to neutron captures. Some of them emit neutrons from spontaneous fissions or alpha neutron reactions. Since spontaneous neutron emission (NE) is linked to Burnup with (BU) a power law (with exponent >1), Burnup determination with passive neutron measurement is extremely accurate.

The correlation law is :

$$BU = aNE^b, \quad (2)$$

in which “a” is a constant slightly dependent on initial enrichment and “b” another constant close to 0.2 for usual irradiation histories which, however, has only a small influence on neutron emission. For a PWR fuel assembly (3,8 % initial ^{235}U enrichment, cooled 1 year after irradiation, 60GWd/t_{HM} Burnup), the neutron emission varies less than 10% considering the two histories displayed in Figure 1. Due to the power law (2), this 10% discrepancy on neutron emission leads to a 2% uncertainty on the Burnup. Assuming that cooling time is known, we can consider that passive neutron emissions are only slightly dependent on irradiation history due to the period of the main neutron emitter (^{244}Cm).

In addition, as shown on Figure 2., this particularity leads to a slow decrease of the total passive emission which is less than half a decade over 30 years.

4.1.2. Induced Neutron Emission.

In nuclear fuels that contain fissile materials, neutron emission can be induced by an external neutron flux. It could also be created by a high-energy gamma intense flux (photofission) but, at the present time, this technique has never been applied to spent fuel because of its cost and complexity.

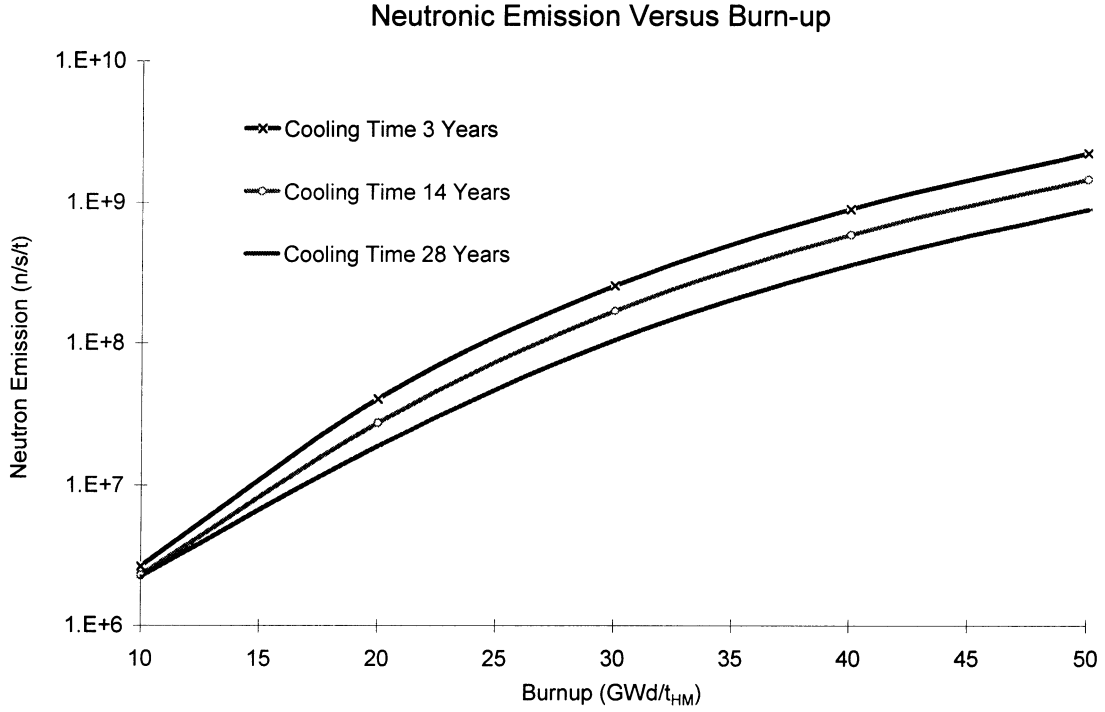

FIG. 2. Passive neutron emission dependence on Burnup for various cooling times. Passive neutron emission is decreasing slowly during cooling : less than half a decade over 28 years.

Neutron Induced Emission NIE is linked to the multiplying factor K_{eff} by the relation :

$$NIE = a\frac{Keff}{(1-Keff)}, \quad (3)$$

Using pulsed neutron irradiation, the neutron detector measures on the one hand the passive emission and on the other hand both induced and passive contributions. This NDA method is used for initial ^{235}U enrichment determination or for MOX fuel characterization. Considering only LEU fuels, and since initial ^{235}U enrichments are assumed to be known, this technique is not often applied to spent fuels because it induces higher complexity and costs.

4.2. Neutron measurements

4.2.1. Neutron emission and neutron count rate

Because of the multiplying effect of the fuel elements, the measured count rates are not directly proportional to the neutron emissions. The relation between the Neutron Emission (NE) and the Count Rate (CR) is :

$$CR = Ka \bullet \frac{NE}{\left(1 - K_{eff}(IE,BU)\right)}, \quad (4)$$

In this relation, Ka is the detector yield and K_{eff} (IE,BU) the multiplying factor of the fuel element. The K_{eff}(IE,BU) value depends on the measuring conditions (pure or borated water, air ...) and on the Burnup (BU) and the ^{235}U initial enrichment (IE). As a result, neutron emission cannot be determined without knowledge of the multiplying factor whose parametrical calculation versus Burnup and initial enrichment is required for each measuring condition (i. e. each boron concentration for underwater measurement).

Figure 3..displays the variation of the multiplying effect versus Burnup considering different initial enrichments for a 2400 ppm borated water measurement. Variation of K_{eff} during fuel life can reach 30% and cannot be neglected.

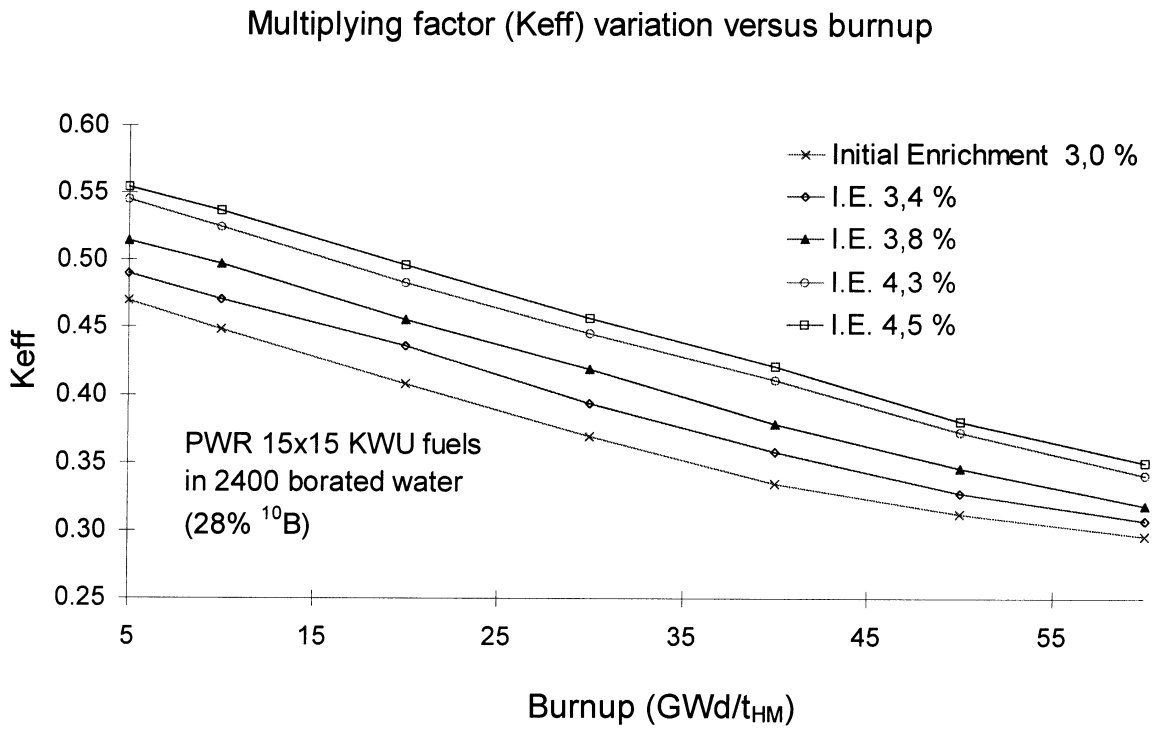

FIG. 3. Multiplying factor versus Burnup in borated water. During its life, the fuel reactivity (in borated water) decreases. The multiplying factor (K_{eff}) decreases more than 30% between 5 and 60 GWd/t_{HM} .

4.2.2. Measured calibration curve

The total passive neutron measurements provide count rates, which are above all, related to curium amounts in the fuels. Considering LEU spent fuels, the link between curium amounts and Burnup is a correlation that depends only on fuel type and initial enrichment. Consequently, to be used for Burnup assay, the neutron count rates have to be compared with a calibration curve previously built. This method gives good results but a set of fuel with homogeneous (and known) irradiation histories has to be assayed before any measurement. Moreover, irradiation histories of calibration fuels and measured fuels should be similar in order to achieve accurate results.

Regards to the classification given in section 2.3 (Sorting the Measurement Methods), this method can be qualified as dependent, relative and global.

4.2.3. Calculated correlation curve

As described in section 4.2.2, the measured calibration curve establishes the link between the Burnup and the neutron count rate considering a set of known fuel assemblies assumed to be similar to the ones to be assayed. On the contrary, the calculated correlation curve is related to the fuel element to be measured. Using the appropriate and validated cross-section library, the online depletion code calculates the isotope amounts resulting from irradiation. Using spontaneous fission and (a,n) reaction data, the online code calculates the correlation curve linking the Burnup and the actual neutron emission. So, for any fuel to be assayed, the constants (*a,b*) of the relation (2) are calculated using a validated depletion code taking into account initial enrichment and actual irradiation history of the assembly to be measured.

Knowing *a* and *b*, the Burnup is directly determined by the relation (5) obtained from relations (2) and (4):

$$BU = a \cdot \left(\frac{CR \cdot \left(1 - K_{eff}(IE,BU)\right)}{Ka} \right)^{b} , (5)$$

The order of magnitude of the detector yield (Ka) is determined with a simulation code (e.g. MCNP [9]). The yield has to be precisely determined during a setup measurement on a known item. In addition, it is confirmed with a second known item. This item is chosen to be as completely different as possible from the first one (irradiation history, initial enrichment, Burnup).

Since the multiplying factor is a function of the Burnup, an iterative process has to be used to determine the actual Burnup. Its consists in establishing a first guess of the Burnup assuming K_{eff} equals 0. This gives a first value for BU. Then K_{eff} is recalculated using the calculated multiplying effect curves (K_{eff}=f(IE,BU))(Cf . 4.2.1.). A new value of BU is then obtained from relation (5). This process is repeated until K_{eff} reaches a stable value.

With this method, relative error on neutron emission $^{\Delta NE}/_{NE}$ from the measurement leads to small Burnup relative error because of the power correlation law that links Burnup and neutron emission. The relative Burnup uncertainty can be expressed from the following relation:

$$\frac{\Delta BU}{BU} \cong 0.2 \frac{\Delta NE}{NE} , (6)$$

To summarize, using a correlation curve (BU versus NE) calculated with an on-line depletion code, the passive neutron method gives accurate Burnup results for any fuel for which validated cross-section library is available. It requires only one setup measurement to cover the whole fuel range (initial enrichment, burn-up). This method is relative, (slightly) dependent and global.

4.3. Gamma emission

4.3.1. Total gamma emission

Gamma emissions are mainly due to fission products and activation products. Just after reactor core unloading, short lived fission and activation products are responsible for most of the total gamma emission that decreases very fast. After several months, Total Gamma Emission (*TGE)* can be linked to the cooling time with the following correlation law :

$$CT = a(BU) \cdot \left(\frac{TGE}{BU} \right)^{b(BU)} , (7)$$

$a_{(BU)}$ and $b_{(BU)}$ are quadratic functions of BU determined by fitting of parametrical depletion calculation results. Since total gamma emission can be measured with simple and inexpensive detectors (e. g. ionization chambers), cooling time can be simply estimated.

Contrary to neutron emission, gamma emission can be collimated. This property is used to determine a relative irradiation profile along the assembly, in order to measure the Burnup of the fuel extremities for criticality safety purpose.

Average BU being known (using a neutron method for example), the total gamma profile is measured along the assembly (from 0 to z_{max}) with a collimated detector. The result of the scanning measurement is an array TGE(z).

Extremity Burnup $EBU_{(0,z0)}$ in the section from z = 0 to z = z_0 is defined by the relation:

$$EBU_{(0,z0)} = BU \cdot \frac{z_{Max}}{z_0} \cdot \frac{\int_0^{z_0} TGE(z) \cdot dz}{\int_0^{z_{Max}} TGE(z) \cdot dz} \cdot F(CT) \quad ,(8)$$

This relation assumes that TGE is mainly composed of burn-up proportionally produced gamma emitters. Actually, a correction factor F(CT) has to be used for short cooling times CT, because short lived fission products induce non proportionality. The following relation is convenient for PWR reactors:

$$F(CT) = 0.5 \cdot CT^{0.1}, (9)$$

4.3.2. Gamma spectrometry

Many gamma emitters have interesting properties in order to characterize spent fuels. Table I. and Table II. give an overview of several relevant isotope abundance and isotopic ratios to determine Burnup and cooling time. Convenience of the different isotopes is displayed, with regard to the range of cooling time.

Convenience of the gamma spectrometric methods is dependent on characteristics of spent fuel. They have to be carefully selected to produce proper results. Figure 4.:displays the irradiation history dependence of the most used cesium isotopes. Figure 5. displays the resulting cesium ratios. It can be seen, that irradiation history can change the cesium ratio up to 30%.

Table I. Overview of gamma spectrometric Burnup determination.

Isotope	Type of method	Correlation law : *(example value given for a PWR 17x17 fuel, IE 3% CT 3 years)*	Appropriateness over the Cooling time ranges		
			0 to 90 days	90 to 5000 days	Over 5000 days
^{134}Cs	Relative Dependent Peripheral	$^{134}Cs = a.BU^2$ *(example a :50)*	+	-	--
^{137}Cs	Relative **Independent** Peripheral	$^{137}Cs = a.BU$ *(example a :3000)*	+	+	+
^{154}Eu	Relative Dependent Peripheral	$^{154}Eu = a.BU^b$ *(example a : 5, b : 2)*	--	-	+
$^{134}Cs/^{137}Cs$	Absolute Dependent Peripheral	$^{134}Cs/^{137}Cs = a.BU^b$ *(example a : 10^{-2}, b : 1)*	++	+	-
$^{154}Eu_{/}{}^{137}Cs$	Absolute Dependent Peripheral	$^{154}Eu_{/}{}^{137}Cs = a.BU^b$ *(example a : 10^{-3}, b : 1)*	--	+	++

Type of method : describe properties of the gamma spectrometric measurement with regards to the isotope or isotopic ratio it uses (Cf. §2.2).

Correlation Law : indicates the type of law that links isotope activity or isotopic ratio to Burnup. Values of the constants are given for a typical spent fuel. In the correlation examples, Activities are given in Curie/g, Burnup in GWd/t_{HM} and Cooling Times in days.

Appropriateness over the cooling time ranges : indicates wether the gamma spectrometric measurement is relevant for Burnup measurement regards to the cooling time. (-- means "really not appropriate", - means "not appropriate" + means "can be successfully used", ++ means "recommended").

Table II. Overview of gamma spectrometric cooling time determination.

Isotope	Correlation law *(example PWR 17x17 IE 3%)*	Appropriateness over the Cooling time ranges		
		0 to 90 days	90 to 5000 days	Over 5000 days
$^{144}Ce/^{137}Cs$	$^{144}Ce/^{137}Cs = a.e^{b.CT}$ *(example a : 10, b : -0.002)*	+	++	-
$^{106}Ru/^{137}Cs$	$(^{106}Ru/^{137}Cs)/TC^{0.5} = a.e^{b.CT}$ *(example a : 1, b : -0.02)*	++	+	-

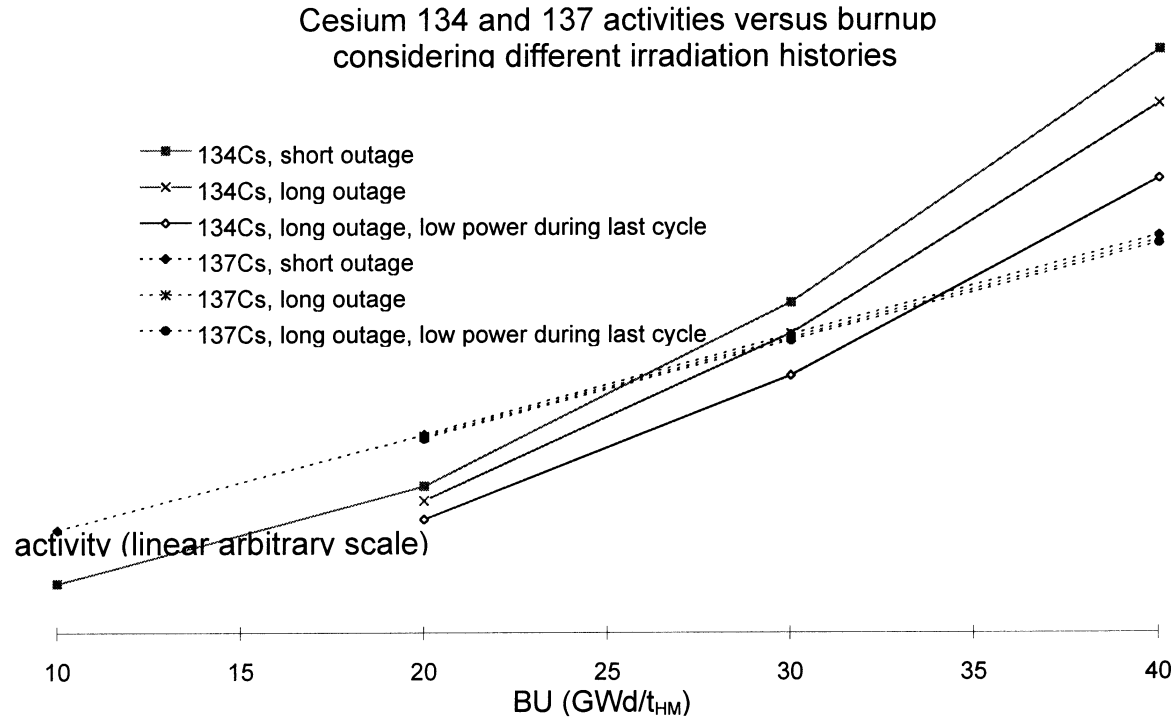

FIG. 4. Irradiation history dependency of the ^{134}Cs and ^{137}Cs isotopes. Contrary to ^{137}Cs, ^{134}Cs activity depends on irradiation history.

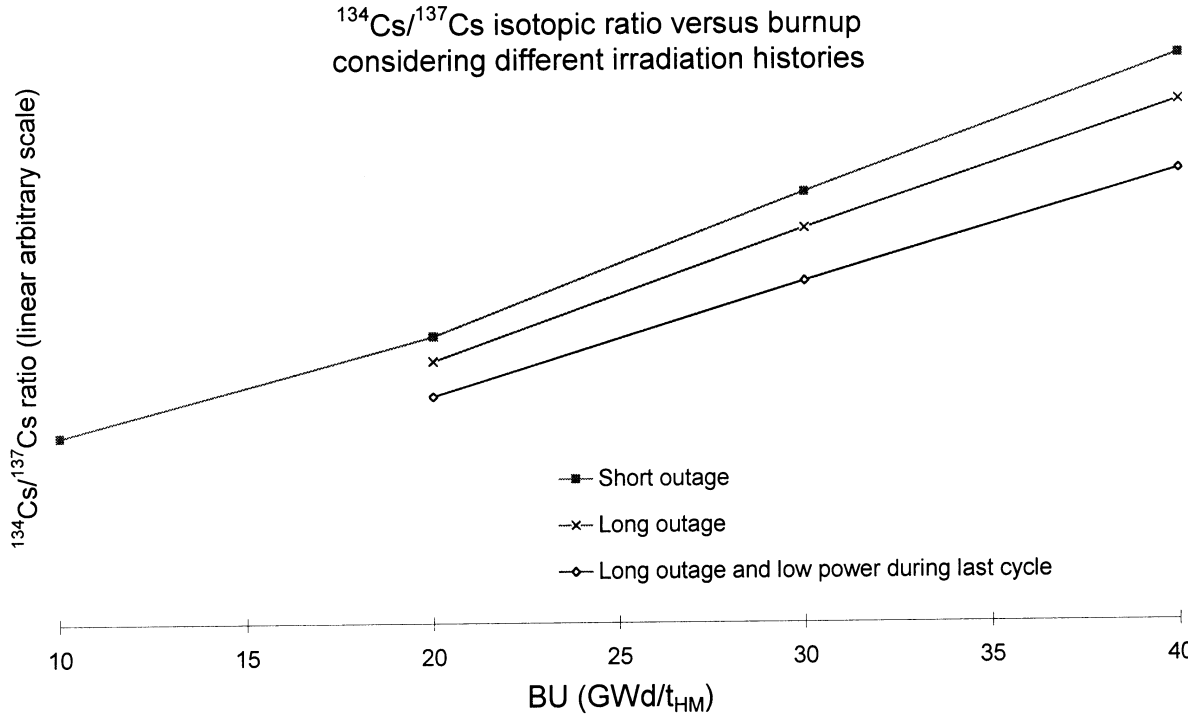

FIG. 5. Irradiation history dependency of the $^{134}Cs/^{137}Cs$ ratio. Depending on different irradiation history but considering the same cooling time, a 40 GWd/t_{HM} burned fuel could be confused with a 30GWd/t_{HM} one

The measurement of isotopic ratios through high resolution gamma spectrometry is an absolute method but it is very dependent on irradiation history. In addition, due to the high absorption rate in the fuel pins, it gives only a peripheral assay of the fuel elements.

Anyway, because the slope of the correlation laws is always smaller for gamma emitters than for neutron emission, Burnup determination using passive neutron counting leads to more accurate results than gamma measurements.

4.4. Gamma spectrometric measurements

Gamma spectrometric measurement of one single isotope whose amount is proportional to the Burnup [e.g. ^{137}Cs] can be used with a calibration curve as for the neutron methods. The calibration curve gives a direct estimation of the Burnup, through knowledge of the detector yield. This simple method requires the detector yield to be very stable, so the geometry has to be strictly reproducible.

It is shown in table I that only the gamma spectrometric ratios give an absolute Burnup because the detector yield has not to be known. Unfortunately, no gamma ratio relevant to Burnup with irradiation history independence exists.

In addition, all the gamma rays detected out of the fuel element are emitted in the peripheral rods. As a result, however absolute and accurate the gamma methods are, they cannot give a result representative of the entire fuel which is typically the requested property for safety criticality purposes.

Concerning the measurement devices, room temperature gamma spectrometry devices have been dramatically improved during last years [6]. Now, cadmium (zinc) telluride (CdZnTe) gamma probes can operate under water and provide spent fuel gamma spectrum with a resolution better than 8 keV (at 662 keV). Many instruments devoted to safeguards have already been developed, but a the moment studies are still under way in order to transfer this technology to criticality safety devoted devices.

At the present time and for the short cooling times, because of the non proportionality of the total gamma emission, measurement of total gamma probes (ionization chambers) requires the use of a correction factor using the relation No. 9. So, CdZnTe probes could provide improvements as a combination with neutron methods to achieve a true Burnup proportionality of the Burnup profile along the assembly.

5. APPLICATIONS OF NDA METHODS.

NDA methods have been applied by CEA in several plants. They have to be shared in two families depending whether they are applied under water or in air.

5.1. In air measurements

In air, gamma spectrometric methods are easier to carry out and give better results than under water. As an example, we can describe the spectrometric analysis of the spent fuels in the head-end workshop of the COGEMA La Hague reprocessing plant [4].

This device is composed of two High Purity Germanium detectors with collimators. Burnup and cooling time are determined using respectively $^{134}Cs/^{137}Cs$ and $^{144}Ce/^{137}Cs$ ratios. Scanning the fuel between the detectors leads to a very accurate Burnup profile measurement.

Neutron passive counting combined with an on line evolution code, is also used and gives very accurate Burnup determination (within 2 % considering PWR assemblies).

5.2. Underwater measurements

Various techniques have been used for under water measurements but neutron measurements are particularly convenient. Two major R&D projects have been carried out by CEA resulting in the PYTHON ™ device [7] and NAJA device [8].

5.2.1. The PYTHON ™ device

The PYTHON device has been developed in collaboration between EDF and CEA. Its main objectives are to measure the average and extremity burn-up for criticality safety purposes. The PYTHON device is a combination of:

- a passive neutron measurement,
- a collimated total gamma measurement,
- an on line evolution code.

Figure 6. is a graphic output of a MCNP [9] model of the measurement heads. It shows a schematic view of the two measurement heads that operate on top of the storage racks in the NPP ponds. This is used to optimize the head's design, to pre-calculate the measurement yields and to parametrically calculate the multiplying factor K_{eff} of the fuel taking into account boron concentration in the water and Burnup.

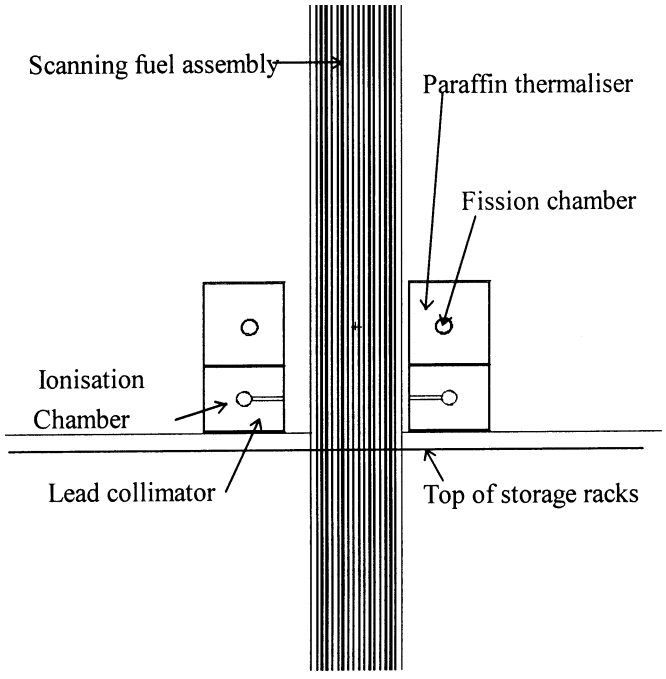

FIG. 6. The 2 detection heads of the PYTHON™ system. (View of a MCNP model used for multiplying factor and detector yield calculations)

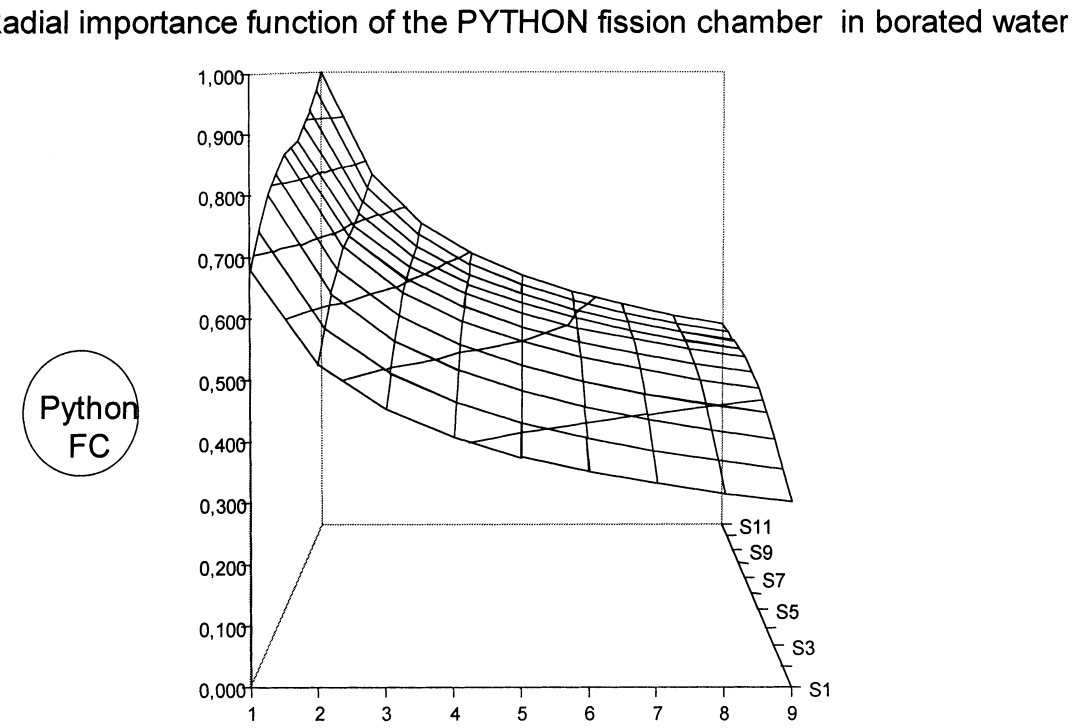

FIG. 7. Radial importance function (y axis) of a Python fission chamber (FC) measured with a 99 pin mock-up fuel (9 rows (x axis) x 11 rows (z axis)). Contrast between the front side and the opposite side of the mock-up is less than 2.

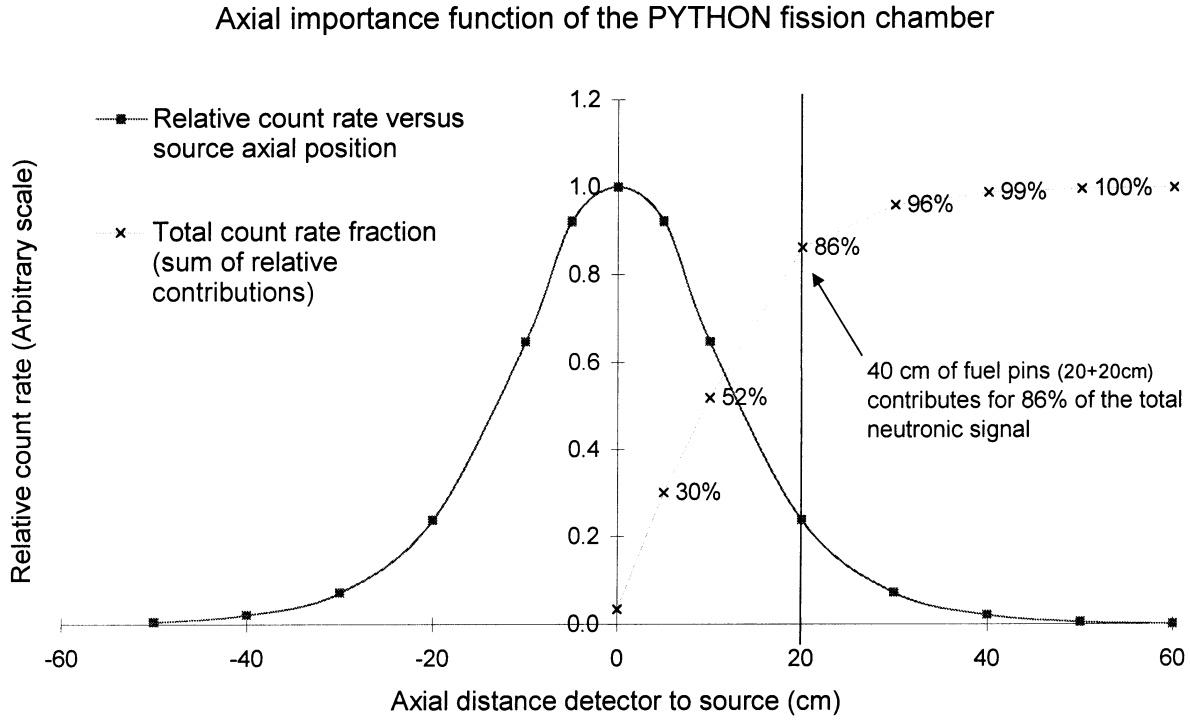

FIG. 8. Axial importance function of Python fission chamber. Calculated on a 17X17 PWR fuel. Since the fission chamber signal originates from a 50 cm height fuel column, no accurate Burnup profile can be given with a neutron method.

Table III. Python qualification range.

NPP Location	Tricastin (France)	Gröhnde (Germany)	Brökdorf (Germany)
Number of assemblies assayed	35	50	35
Fuel types	UOX PWR 17 x 17	UOX PWR 16 x 16	UOX PWR 16 x 16
Initial enrichments	1.8% to 3.7%	2.1% to 4%	1.8% to 3.7%
Cooling Time	80 days to 7.5 years	60 days to 6 years	500 days to 3.5 years
Reactivity (K_{eff})	0.4 to 0.7	No reactivity measurement (Neutron Passive mode only)	

In order to achieve good statistical accuracy with low Burnup, the detection heads are equipped with high efficiency fission chambers (1 c/n.cm^{-2}). As a result, for standard PWR fuel assemblies, the neutron yield is about 10^{-5} c/neutron.

The gamma detectors are simple ionization chambers with 10^{-9} A/Gy.h^{-1} efficiency.

The measurement method of the average BU is described in section 4.2.3. The relation (8) is used to calculate a conservative extremity Burnup using the total gamma signal given by two ionization chambers during the scanning along the assembly.

Since the PYTHON device is intended to measure the average Burnup and since this Burnup value is intended to be used for Burnup credit, the measured signal has to be representative of the entire fuel assembly. This means that contributions to the average signal have to originate from the whole fuel. Figure 7. shows an example of the radial importance function of the fission chamber in the fuel section measured on a mock-up fuel in borated water [10]. Contrary to the gamma emissions that are absorbed when crossing fuel pins, the neutrons detected by the fission chamber originate from almost all the fuel section. In addition, the results from two measurement heads are averaged and therefore, the sensitiveness to radial gradient for the Burnup measurement is very low.

To take into account the Burnup axial profile, the fuel is scanned between the two heads and signals are averaged. However, despite neutron signals being acquired along the fuel, it is not possible to get a Burnup profile from them. Figure 8. shows the axial importance function of the detectors. It is clear that almost a length of several tenths of centimeters contributes significantly to the signal. As a result, Burnup profile has to be measured with the collimated total gamma detector and neutron signals are used only to evaluate the average Burnup. The extremity Burnup is calculated using both gamma profile and average Burnup using the relation 8.

For safety criticality purposes, the irradiation histories are considered to be known as well as the initial components of the fuels. They are used as input data for the on-line depletion code that determines a correlation law (BU = f(NE)) for each fuel assembly. As a result no calibration curve is required to determine the relevant correlation law.

Only one setup measurement is required to measure the detector yields. In order to avoid mistakes the measured yield values are confirmed with MCNP calculations.

The PYTHON system has intensively been qualified using a prototype device with active neutron mode capability in Tricastin NPP (France° and with industrial systems in Gröhnde and Brökdorf NPPs (Germany). Table III summarizes the qualification range.

The accuracy evaluated by a comparison with declared values for burn-up and cooling time and with calculations for Keff is as follows : on average burn-up within 2%, on cooling time within 15% and on the multiplying factor Keff within 3%.

Other industrial PYTHON systems have been delivered to customers in western Europe[2]. In the mean time, further R&D projects have been focused on extensions of the PYTHON capabilities in order to use Burnup and reactivity measurements for automatic core and pond loading checks.

5.2.2. The NAJA device

The main objectives of the NAJA device consist in developing a measurement device which combines nuclear and non-nuclear methods in order to evaluate the physical characteristics of each fuel assembly (Identification, Burnup, Reactivity, Initial Enrichment, ...) and to automatically validate the final core loading. It would be placed on the passage of the fuel assembly between the storage pond and the reactor building. In addition to core loading conformity control and on-line core monitoring, it should be useful for Burnup credit implementation in the NPP ponds and the spent fuel transportation.

The NAJA device is able to automatically determine for each assembly:

1. the nature of the fuel element (fresh or irradiated, UOX or MOX),
2. the presence and the kind of neutron absorber,
3. the initial enrichment in ^{235}U for fresh UOX assembly,
4. the identification number.

These information allow us to characterize the fuel assemblies accurately and to be sure, without human factor hazard, of the core and pond loading conformity.

The nuclear methods involved in the NAJA device are a passive and active neutron measurement combined to on-line depletion code and a room-temperature gamma spectrometry. In addition two non-nuclear measurements are used. An ultra-sonic probe locates the fuel assembly with regard to the nuclear detectors and a video system reads each fuel identification number. The automatic fuel identification allows to associate each fuel element placed in the core or in the pond to its physical characteristics without any information coming from the operator.

[2] The supplier of PYTHON devices is a COGEMA subsidiary : EURISYS MESURES

The feasibility study of the device has been performed using experimental results from the PYTHON device and theoretical calculations for its optimization. The panel of the fuel assembly characteristics which have been taken into account is large and representative of the French Fuel Cycle :

1. average burn-up of the spent fuel from 6000 MWd/tHM to 48000 MWd/tHM,
2. cooling-time varying from 1 to 90 days[3],
3. initial enrichment in ^{235}U for UOX assembly varying between 3 and 4 %,
4. nature of neutron absorbers : pins containing Silver - Indium - Cadmium and/or pins containing Silver - Indium - Cadmium and Boron Carbide (B_4C).

Detailed results of this feasibility study are given in reference [8].

Potential uses of the NAJA device derive directly from its main functions:

1. The core conformity control which allows us to increase the safety level of the plant significantly,
2. The accurate Burnup measurements of the irradiated fuel assemblies which allow us to implement Burnup credit for storage and transportation of the spent fuels.

6. CONCLUSION

The use of reactivity loss due to fuel irradiation is known as Burnup credit. Its implementation for transportation and fuel storage could have large economic consequences since it induces subsequent investment savings. The Burnup credit should obviously not have any consequences on the safety level of the facilities. So, a measurement of the actual Burnup of each item could be required by the Safety Authority to make sure that any fuel assembly considered in the Burnup credit procedure has reached the Burnup threshold determined by the criticality studies.

Considering the LEU fuels, we have shown that, despite the fact that the Burnup has no intrinsic physical link to the reactivity in storage condition, this parameter is the relevant one for measurements related to Burnup credit. To determine the Burnup, neutron as well as gamma emissions are used in many non destructive measurement techniques. The only absolute NDA method is based on isotopic ratios assessed by gamma spectrometry. Unfortunately, these ratios have an important dependence on the irradiation history and only the peripheral rods of the fuels are actually assessed.

No passive neutron method is absolute. Calibration curves or detector yields are always required, but dependence on irradiation history is quite small and almost the totality of the fuel section has an influence on the measured signal. Combining passive neutron measurement with online depletion codes allows us to enlarge the range of measurement since it is only limited by the availability of the relevant and validated cross-section libraries.

Devices combining neutron methods and room temperature gamma spectrometry are being studied at the moment in order to combine their respective qualities and advantages.
For MOX fuels, the problem is more complex due to the various parameters which interfere in

3 The NAJA specific feasibility study was limited to 90 days since the range from 90 days to 15 years was previously done for the PYTHON device.

the relation “Burnup versus neutron emission ”. At the moment only the induced neutron assay is considered as relevant for MOX assay. Some R&D programs are under way to address the MOX fuel measurement problem with other NDA methods.

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TRANSNUCLÉAIRE'S EXERIENCE WITH BURNUP CREDIT IN TRANSPORT OPERATIONS

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Abstract

Facing a continued increase in fuel enrichment values, Transnucléaire has progressively implemented a burnup credit programme in order to maintain or, where possible, to improve the capacity of its transport packagings without physical modification. Many package design approvals, based on a notion of burnup credit, have been granted by the French competent authority for transport since the early eighties, and many of these approvals have been validated by foreign competent authorities. Up to now, these approvals are restricted to fuel assemblies made of enriched uranium and irradiated in pressurized water reactors (PWR). The characterization of the irradiated fuel and the reactivity of the package are evaluated by calculation, performed using qualified French codes developed by the CEA (Commisariat à l'Energie Atomique / French Atomic Energy Commission): CESAR as a depletion code and APOLO-MORET as a criticality code. The approvals are based on the hypothesis that the burnup considered is that applied on the least irradiated region of the fuel assemblies, the conservative approach being not to take credit for any axial profile of burnup along the fuel assembly. The most reactive configuration is calculated and the burnup credit is also restricted to major actinides only. On the operational side and in compliance with regulatory requirements, verification is made before transport, in order to meet safety objectives as required by the transport regulations. Besides a review of documentation related to the irradiation history of each fuel assembly, it consists of either a qualitative (go/no-go) verification or of a quantitative measurement, depending on the level of burnup credit. Thus the use of burnup credit is now a common practice with Transnucléaire's packages, particularly in France and Germany. New improvements are still in progress and qualifications of the calculation code are now well advanced, which will allow in the near future the use of six selected fission products in the criticality assessment.

1. BACKGROUND

For years, France has developed a policy of reprocessing its spent fuels from nuclear power reactors. Today, the facilities located in La Hague (North-West of France), and operated by COGEMA, reprocess the fuel assemblies from the French light water reactors operated by EDF (Electricité de France) and also fuel assemblies from numerous European and Japanese reactors. To support this policy, Transnucléaire is in charge of the transports of spent fuel assemblies to the La Hague reprocessing plant.

For this purpose, Transnucléaire operates a wide range of transport packages with capacities up to 12 PWR or 32 BWR fuel assemblies. These capacities are achieved for short cooled fuels with high heat load. The cooling time depends on the burnup but, most often, is close to one year.

For longer cooling times, capacities of 37 PWR and 97 BWR fuel assemblies have been achieved for transport / interim storage casks.

During transport, subcriticality is controlled by the fuel support frame also known as "basket". It is made of materials containing neutron absorbers, usually boronated aluminum or stainless steel sleeves, and ensures a well defined geometry.

Spacers are designed to allow the active part of the fuel assemblies to be in front of the poisoned length of the basket in any circumstances: normal and accident conditions of

transport. If the ends of one fuel assembly can reach regions where there is no neutron absorber, the criticality assessment considers this configuration.

2. UTILIZATION OF BURNUP CREDIT IN TRANSPORT OPERATION

Facing a continuous increase in the enrichment of the fuel assemblies, Transnucléaire has been interested in the use of burnup credit for a long time. The experience acquired now is about 15 years. This has allowed to maintain or, where possible, to improve the capacity of its transport packagings without physical modification.

Numerous package design approvals have been granted in France and validated in foreign countries (particularly in Germany) since the first approval which was obtained in 1987 for 16x16 PWR fuel assemblies loaded in TN 13/1 or TN 13/2 casks.

The different approval granted in France, for transportation, are up to now limited to PWR fuel assemblies consisting of enriched uranium oxyde. The acceptance of burnup credit is subject to the verification by the reactor operator that the burnup assumed is effectively met.

Depending on the results of the criticality analysis, two levels of requirements have been accepted. 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 which is the least irradiated. As an example, the following table illustrates different approvals that are currently valid to transport 18x18 and 17x17 PWR fuel type in the TN 13/2 cask.

Table I. Package design approval F/274/B(U)F-85 Hn

BASKET 928 Array 18 x 18		BASKET 925 Array 17 x 17	
Burnup (MW.d/tU)	Enrichment (%)	Burnup (MW.d/tU)	Enrichment (%)
0	3.31	0	3.25
3200	3.55	3200	3.45
10000	4.10		

In any case, the same methodology is used to determine the burnup limits depending on the enrichment. The calculations are performed with qualified codes developed in France by the CEA (Commissariat à l'Energie Atomique / French Atomic Energy Commission). The assessment is based on the hypothesis that the burnup considered is that applied on the least irradiated part of the fuel assembly, i.e. the last 50 cm. The conservative approach is not to take credit for any axial profile of burnup along the fuel assembly. The depletion code used is CESAR, while criticality calculations are performed with APOLO-MORET.

2.1. CESAR

CESAR is a depletion code dedicated to the characterization of spent uranium oxyde and mixed oxyde (MOX) fuel (PWR and BWR). For a given fuel assembly and based on its initial composition, the reactor type and the history of irradiation, it provides the masses of isotopes, activities, heat power and neutrons sources.

2.2. APOLO-MORET

APOLO-MORET is a package which allows the calculations of the cross sections in a single cell and the calculations of the neutron multiplication factor – k_{eff} – of the package, using a Monte-Carlo method.

For major actinides only burnup credit, these codes are qualified with numerous benchmark experiments. These experiments use fuel composition equivalent to a fuel enriched to 4.5 % with a 37.5 GWd/t burnup without fission products. These experiments simulate transport configurations, among others.

For the criticality calculation, only major actinides are considered (^{235}U, ^{236}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am). No cooling time is considered which is an additional conservative hypothesis. The fission products are neglected. Based on the results of calculation performed in accordance with these principles, the use of burnup has been implemented in two steps, as concerns transport.

3. FIRST STEP: QUALITATIVE (GO / NO-GO) BURNUP CHECK

The approval has been obtained with the following requirements:

1. In order to ensure a safety margin on the irradiation, the allowed burnup credit must be reached on the least irradiated part (i.e. the last 50 cm) of the active fuel length instead of an average value over the total active length,
2. On the basis of the fuel management and in-core measurement, the operator of the power plant must guarantee that after one cycle the minimum burnup on the least irradiated part (i.e. the last 50 cm) of the active fuel length exceeds the allowed burnup credit, and that each fuel to be loaded in the cask has been irradiated during at least one cycle,
3. The irradiation status of each fuel assembly must be checked by a qualitative go / no-go physical measurement in the reactor pool before cask loading. These measurements have to be in accordance with the plant quality assurance policy,
4. In addition, safety authorities require the demonstration of the reliability of the fuel identification procedures.

4. SECOND STEP: QUANTITATIVE BURNUP MEASUREMENT

For increased burnup credit, a new requirement has been introduced, which consists of replacing the qualitative irradiation go / no-go checking as previously performed, by an independent quantitative measurement at the reactor storage pools performed with one qualified device.

Besides, a verification of the data sent by the reactor is performed.

5. OPERATIONAL PROCEDURES

On the operational side it appears that the safety depends on the reliability of the fuel identification procedures. Thus, study of the most likely failure scenario enables an optimization in the procedures by focusing on the prevention of misloading and increasing the possibility of recovering errors prior to shipment. This has led to the definition of the following principles to set up the operational procedures:

1. Segregation of fresh or low irradiated fuel assemblies in the pool,
2. Fuel element pre-loading positions. The fuel assemblies selected for a given transport are set apart from the bulk of the stored fuel in the reactor pool,
3. Training of the operators in charge of the fuel identification,
4. Increasing the awareness of the reactor fuel handling staff to fuel identification prior to transport,
5. Introduction of redundancy on fuel identification with low dependence level between operators,
6. Written records of the checks carried out on each action.

6. REGULATORY CONTEXT

Transport activities are regulated by the "Regulations for the safe transport of radioactive material" set forth by the International Atomic Energy Agency. The edition currently applicable is the 1985 Edition (as amended 1990). 1996 Edition will be applicable in 2001.

With regards to burnup credit, 1985 Edition (as amended 1990) stipulates, in paragraph 568, that "in evaluating the subcriticality of fissile material in its transport configuration, the following shall apply:

(a) the determination of subcriticality for irradiated fissile material may be based on the actual irradiation experience, taking into account significant variations in composition".

This edition of the regulations does not include neither incentive, not objection to the burnup credit techniques.

In the 1996 Edition, it is stipulated in paragraph 674 that "for irradiated nuclear fuel the assessments … shall be based on an isotopic composition demonstrated to provide:

(a) the maximum neutron multiplication during the irradiation history, or
(b) a conservative estimate of the neutron multiplication for the package assessments. After irradiation but prior to shipment, a measurement shall be performed to confirm the conservatism of the isotopic composition."

The burnup checks, qualitative and quantitative, as implemented (and described here above), allow to fulfill the regulatory requirements.

7. NEW DEVELOPMENT

The use of burnup credit has become a common practice well mastered by the different organizations involved. However, there is still a need to improve the scope of burnup credit.

One need is to take account of the fission products in the criticality calculations. That is why IPSN and COGEMA are currently implementing an experimental programme on actinides (U, Pu) and fission products. The aim is to perform criticality experiments in order to extend the scope of qualification of the French criticality codes used.

The HTC (HTC means high burnup) Programme, which started in 1988 and ended in 1992, was the first step in the experimental programme. The fuel compositin considered was equivalent to 4.5 % ^{235}U enrichment with a 37.5 GWd/tU burnup without fission products. The actinides involved where U, Pu, Am.

The second step, which started in 1996 and is due to end in 2002, is the FPs (Fission Products) Programme. The intend is to qualify, in the criticality codes, the use of six selected fission products: ^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{152}Sm, ^{155}Gd. This will enable to take credit for these fission products in the criticality evaluations and thus improve the evaluation of the reactivity of the spent fuels.

8. CONCLUSION

Facing a continuous increase in the fuel enrichments, Transnucléaire has implemented step by step a burnup credit programme to improve the capacity of the equipments without major physical modification. Many authorizations have been granted by the French competent authority in charge of transport since the early eighties. Thus the use of burnup credit is now a common practice in France and new improvements are still in progress: extended qualification of the codes is made to enable, in a first step, the use of six selected fission products in the criticality evaluations.

THE NECKARWESTHEIM FUEL HANDLING PROCEDURE

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Abstract

This paper describes the procedure applied in Nuclear Power Plant Neckarwestheim II to prevent fuel handling errors at the planning and the operation stage. In this procedure an interlock logic protected against malfunction is used for the loading machine hindering this machine from handling operations which are not laid down in a ”handling sequence plan” established by an authorized person and checked by an empowered person according to the quality assurance requirements. The ”handling sequence plan” is generated with the computer code ALFA which uses appropriate interlock logic schemes to prevent fuel handling errors already at the planning stage. Fuel handling operations cannot be executed until the ”handling sequence plan” is installed in the control unit of the fuel handling machine by an authorized person.

1. INTRODUCTION

1.1. Background

The PWR Nuclear Power Plant NECKARWESTHEIM II (NPP GKN II) is increasing the initial enrichment of its UO_2 fuel from 4.0 wt.-% to 4.4 wt.-% U-235. The NPP GKN II is one of the KONVOI type plants in Germany. Plants of this type use a fuel assembly type which has a 18x18 lattice with 24 guide thimbles in case of UO_2 fuel and 4 additional water rods in case of MOX fuel. An isolated, water-flooded, unirradiated and unpoisoned fuel assembly of the UO_2 type attains the neutron multiplication factor k_{eff} of 0.95 at an initial enrichment of 4.4 wt.-%.

The existing spent fuel storage racks at NPP GKN II are designed to accommodate fresh and spent fuel with a maximum enrichment of 4 wt.-% U-235. The increase of the initial enrichment requires a reracking, therefore.

In order to minimize the costs of the reracking as well as the amount of waste to be managed (decontamination and disposal of the rack materials replaced) the spent fuel storage pool is divided into two regions, as indicated in Figure 1. Only 5 storage racks, called storage region I racks in the following, are equipped with new absorber channels suitable for accommodating fresh 4.4 wt.-% U-235 enriched fuel. These 5 storage racks with 320 storage positions in total suffice to accommodate one full core (193 fuel assemblies) plus one reload batch plus all the fuel assemblies which haven’t attained the end of their life time. The remaining 7 racks (448 storage cells) plus 18 additional storage cells are left unchanged and are used as storage region II racks, i.e., as racks designed to accommodate spent fuel only with a minimum burnup depending on the initial enrichment of the fuel. Accordingly, in the criticality safety analysis of this region the actinide plus fission product burnup credit methodology [1] was applied. Due to the fact that the storage positions of this region are designed to accommodate fresh and spent fuel with a maximum enrichment of 4 wt.-% U-235 only a low burnup credit of 5 MWd/kg U is required for fuel assemblies with initial enrichments between 4.0 wt.-%

and 4.4 wt.-%. However, the criticality safety analysis includes already plans for a future increase of the storage capacity of this region from 466 positions to 732 positions, cp. Figure 2. The burnup credit required then is given by the loading curve shown in Figure 3.

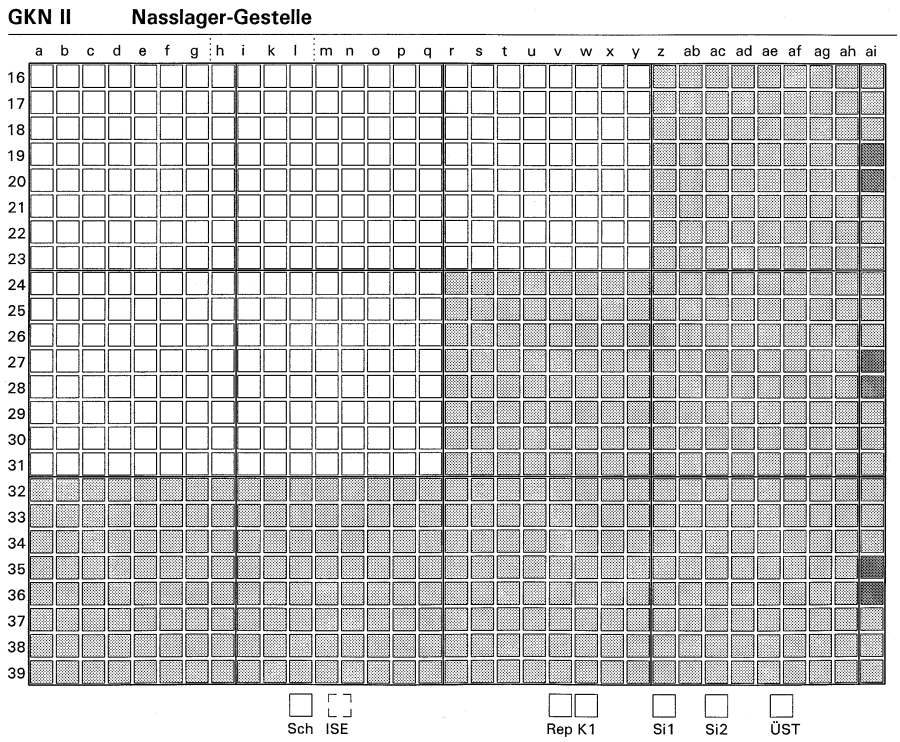

FIG. 1. Storage Pond at NPP GKN2: Reracking due to the Increase of the Initial Enrichment

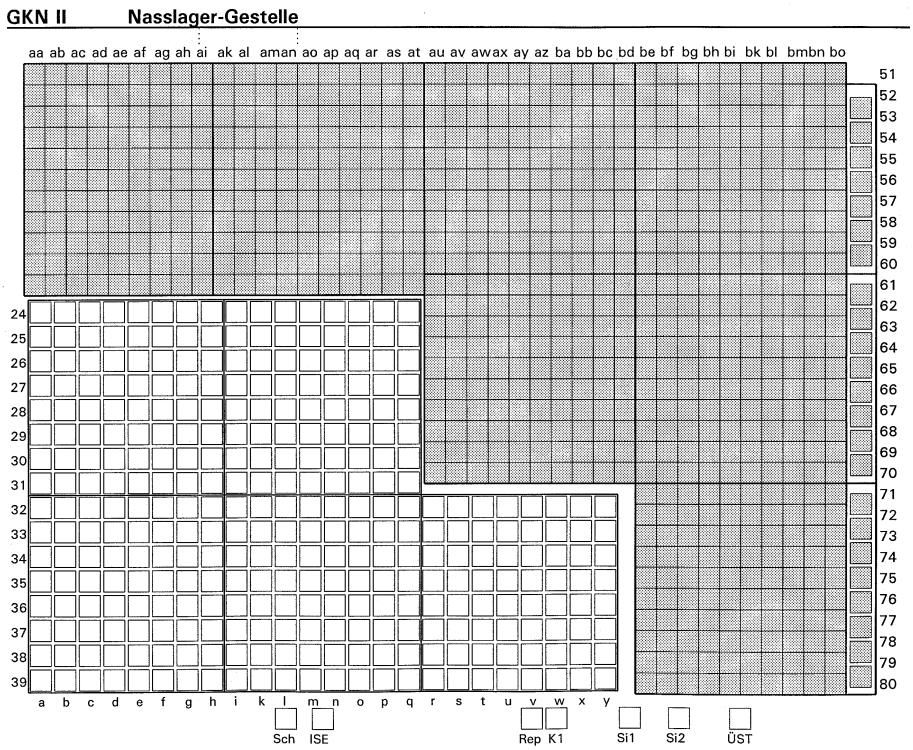

FIG. 2. Storage Pond at NPP GKN2: Plan for Future Increase of the Storage Capacity of the Region II

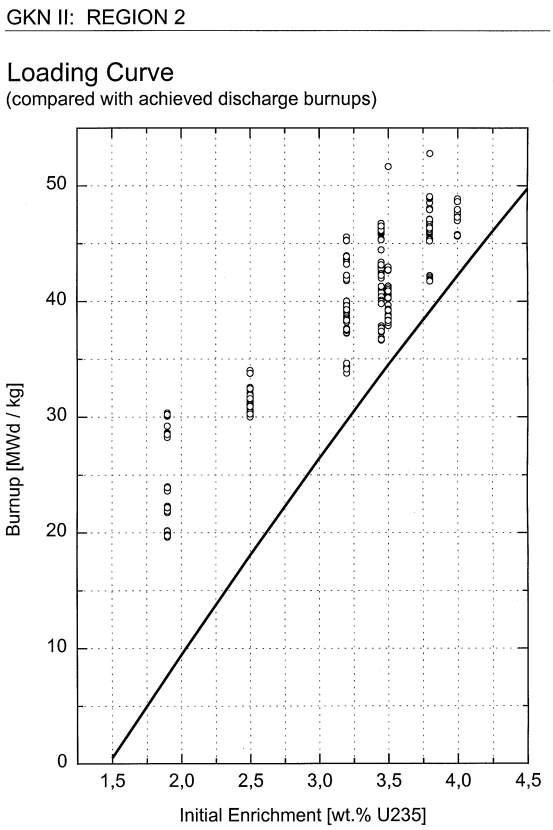

FIG. 3. Storage Pond at NPP GKN2: Loading Curve Referring to the Planned Future Increase of the Region II Storage Capacity.

1.2. Regulatory Requirements

Irrespective of the height of the burnup credit required the use of burnup credit requires, as laid down in the German criticality safety standard DIN 25471 [2],

1. determination of the burnup of each of the fuel assemblies intended for storage in a region II storage cell,
2. establishment of a control procedure that ensures
   1.1. that the region II loading criterion given by the burnup credit required is met,
   1.2. that individual fuel assemblies which do not comply with the loading criterion are segregated.

As laid down in the safety standard DIN 25471,

1. the control procedure has to comply with the double contingency principle, i.e., at least two unlikely, independent and concurrent incidents must occur before a misplacement of a fuel assembly that does not meet the region II loading criterion into a region II storage cell can occur;
2. fuel assembly burnup determination based on reactor records without any additional measurements is acceptable.

In the following sections the technical, administrative and operational measures are described which were developed by NPP GKN II to prevent a fuel handling error in compliance with the regulatory requirements.

2. DESCRIPTION OF THE PROCEDURE

2.1. Principles of the Procedure

The fuel handling procedure used in NPP GKN II ensures that errors in fuel handling operations are prevented at the planning and the operation stage.

To prevent errors at the operation stage an interlock logic protected against malfunction is used for the fuel loading machine hindering this machine from handling operations which are not laid down in a ”handling sequence plan” established by an authorized person and checked by an empowered person according to the quality assurance (QA) requirements. Hindering the loading machine means blocking the functions ”lifting” and ”lowering” of the main hoist of the loading machine.

The handling sequence plan comprises the sequence of all the handling operations which are part of one and the same well defined ”action” such as, e.g.:

1. Transfer of unirradiated fuel from the new fuel store to region I of the storage pond (called action ”A1” in the following),
2. Reshuffling of the core (named action ”A2” in the following),
3. Transfer of spent fuel from region I to region II of the storage pond (action ”A3”),
4. Loading of a spent fuel transport cask (action ”A4”).

Fuel handling operations cannot be executed until the handling sequence plan is installed in the control unit of the fuel loading machine by an authorized person.

In order to prevent misplacement of a fuel assembly that does not meet the region II loading criterion into a region II storage cell the fuel handling procedure is based on the following principles:

1. During action A2 (reshuffling of the core) only the storage region I is available to fuel handling operations.
2. Except for action A3 the storage region II is always closed to relocating a fuel assembly to a region II storage cell. To meet this requirement the following measures are taken:
   - 2.1. In addition to the principle that fuel handling operations which are not laid down in the ”handling sequence plan” to be applied cannot be executed the loading machine is blocked with the aid of a key switch hindering the machine from transferring fuel to a region II storage cell.
   - 2.2. The key is maintained under positive administrative control in the safety control room of the NPP.
   - 2.3. Issue of the key is effected only on presentation of that work order in written form which is required for an action A3 and approved by an authorized person.
   - 2.4. The blocking of the loading machine is raised only for the duration of an action A3. Immediately after completion of such an action the blocking of the loading machine is switched on again.
   - 2.5. The blocking of the loading machine is ensured through the operational and the fail-safe control of the machine.
3. In a ”handling sequence plan” for an action A3 only fuel assemblies can be included which comply with the region II loading criterion. To meet this requirement, generation of the ”handling sequence plan” includes, as described in more detail in section 2.2,

3.1. evaluation of the reactor records and
3.2. application of an appropriate interlock logic scheme that discriminates the fuel assemblies which do not comply with the loading criterion.

In addition, unirradiated fuel is stored in only one of the five region I storage racks. It is impossible to include calls of the loading machine at storage positions of this particular rack in a ”handling sequence plan” for an action A3.

2.2. Generation of a ”Handling Sequence Plan”

Each handling sequence plan is generated with the aid of the computer code system ALFA. This code system developed under the responsibility of NPP GKN II serves for proper identification, management and documentation of locations and relocations of fuel assemblies and internals within the plant. ALFA has access to all pertinent data files such as the reactor records for instance to get all the data required such as names, initial fuel enrichments and topical burnups of the fuel assemblies. With ALFA any handling of fuel assemblies or internals can be simulated and hence planned in compliance with the QA requirements.

ALFA distinguishes between the different actions by applying appropriate interlock logic schemes. For example, when the action A1 (transfer of unirradiated fuel from the new fuel store to region I) or the action A2 (reshuffling of the core) are called in ALFA it is impossible to call the action A3 (transfer of spent fuel from region I to region II) in ALFA. So therefore, no fuel handling operation belonging to an action A3 can be included in a ”handling sequence plan” for an action A1 or an action A2.

If the action A3 is called in ALFA the particular region I storage rack which is designed for accommodating unirradiated fuel (cp. section 2.1) is closed to fuel handling operations by blocking the loading machine for this rack. So therefore, no transfer of unirradiated fuel can be included in a ”handling sequence plan” for an action A3.

The blocking of the loading machine for the region II storage positions is raised then and only then if the action A3 is called in ALFA. After this action is called ALFA applies the region II loading criterion to all the fuel assemblies stored in region I. That fuel assemblies, which meet the loading criterion, are visually differentiated by ALFA from those ones, which do not comply with the loading criterion. It is impossible for ALFA to accept a transfer of a fuel assembly to region II which does not comply with the region II loading criterion.

In fact, for each of the actions that can be simulated ALFA has interlock logic schemes appropriate to prevent handling errors. A fuel handling operation which is rejected by ALFA is not included in the ”handling sequence plan” for the action of interest. Fuel handling operations which are not included in the ”handling sequence plan” cannot be executed because this plan is installed in the control unit of the fuel loading machine.

A ”handling sequence plan” can be generated and authorized by empowered persons only. An authorized ”handling sequence plan” can be installed in the control unit of the fuel loading machine by an empowered person only. The ”handling sequence plan” cannot be executed until:

1. it is printed out and signed by the persons who generated and authorized this plan and
2. all the other working orders and permits required are given.

3. IMPLEMENTATION OF THE PROCEDURE

In case of an action A3 (transfer of spent fuel from region I to region II) first of all the following steps are to be taken:

1. Update of the burnup data of the fuel assemblies stored in region I and
2. QA of the updated data.

Then, and this goes for all actions, the following steps are to be taken:

1. Generation of the ”handling sequence plan” with the aid of the code ALFA by an authorized person.
2. Checking of the ”handling sequence plan” by an authorized person not involved in the generation of the plan. (Without an authorized ”handling sequence plan” any fuel handling operation is forbidden.)
3. Installation of the authorized ”handling sequence plan” in the control unit of the fuel loading machine by an empowered person. (Without installation in the control unit of the loading machine no fuel handling operation is possible.)
4. Issue of authorized working orders and permits for performing the fuel handling operations. (Without these orders and permits no fuel handling operation is allowed.)
5. In case of an action A3: Issue of the key necessary to raise the blocking of the loading machine for region II.
6. Written confirmation of the execution of the handling operations. In case of an action A3: Return of the key and blocking of the loading machine for region II.
7. Update of the data files documenting the actual loading of the stores and the reactor core.

Due to its principles and implementation this procedure ensures that errors in fuel handling operations are prevented at the planning and the operation stage. Except for the action A3 and the blocking of the loading machine this procedure has been already used since many years, not only in the NPP GKN II but also in the neighboring NPP GKN I as well as in seven different PWR plants. Since the introduction of this procedure no fuel handling error occurred in all these plants. The action A3 now added due to the application of burnup credit is realized in NPP GKN II.

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BURNUP CREDIT IMPLEMENTATION PLAN AND PREPARATION WORK AT JAERI

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Abstract

Application of the burnup credit concept is considered to be very effective to the design of spent fuel transport and storage facilities. This technology is all the more important when considering construction of the intermediate spent fuel storage facility, which is to be commissioned by 2010 due to increasing amount of accumulated spent fuel in Japan. Until reprocessing and recycling all the spent fuel arising, they will be stored as an energy stockpile until such time as they can be reprocessed. On the other hand, the burnup credit has been partly taken into account for the spent fuel management at Rokkasho Reprocessing Plant, which is to be commissioned in 2005. They have just finished the calibration tests for their burnup monitor with initially accepted several spent fuel assemblies. Because this monitoring system is employed with highly conservative safety margin, it is considered necessary to develop the more rational and simplified method to confirm burnup of spent fuel. A research program has been instituted to improve the present method employed at the spent fuel management system for the Spent Fuel Receiving and Storage Pool of Rokkasho Reprocessing Plant. This program is jointly performed by Japan Nuclear Fuel Limited (JNFL) and JAERI.This presentation describes the current status of spent fuel accumulation discharged from PWR and BWR in Japan and the recent incentive to introduce burnup credit into design of spent fuel storage and transport facilities. This also includes the content of the joint research program initiated by JNFL and JAERI. The relevant study has been continued at JAERI. The results by these research programs will be included in the Burnup Credit Guide Original Version compiled by JAERI

1. INTRODUCTION

Recently, it becomes evident that light water rectors (PWR and BWR) are continually operated to generate electric power in the long run in Japan, and the exhausted spent fuel is more and more accumulated. Most of spent fuel is now stored at on-site wet storage pools, and partly at on-site dry storage facilities such as that for the Fukushima Daiichi NPP. Some of spent fuel has been transported to overseas reprocessing plants to extract valuable plutonium as a new fuel material. In anticipation of lack of storage capacity in the next few decades, an off-site intermediate spent fuel storage facility has been decided in operation from around 2010. In addition, the first domestic commercial reprocessing plant is now under construction and it is expected to begin operation from 2005.

In Japan, it is traditional to assume a fresh fuel assumption in criticality safety assessments for spent fuel transport and storage, resulting in an excessive safety margin taken in the facility design. As an exception, burnup credit for uranium and plutonium composition is only taken in the design of the Spent Fuel Receiving and Storage Building of Rokkasho Reprocessing Plant (RRP).

In consideration of the above situation, new challenges have emerged to take burnup credit into criticality safety assessments among utilities and related industries for pursuit of economical design and handling of spent fuel storage and transportation with adequate safety margin ensured. It is all the more evident to note that the 1996 IAEA regulation shall be

introduced into the national transport regulations early next year (2001), in which the implementation of burnup credit has been clearly related.

2. PREPARATION WORK FOR BURNUP CREDIT GUIDE PUBLICATION AT JAERI

At Japan Atomic Energy Research Institute (JAERI), studies have been made to develop and prepare criticality safety methods and data together with useful knowledge and information for use in design and licensing of nuclear fuel cycle facilities. Since the first version of Nuclear Criticality Safety Handbook in Japan was published in 1988 [1], works had been continued to prepare the second version of the Handbook, which was recently completed and published in 1999 as a style of JAERI report [2]. In this work scope, study on burnup credit analyses and preparation of spent fuel isotopic composition data for validation of depletion calculation codes have been made to provide useful data and methods for burnup credit implementation in Japan. For example, a more detailed depletion analysis code has been developed [3], and a lot of spent fuel composition data have been measured and collated from open literatures to make an internationally available database [4]. Fig.1 shows the results of criticality safety assessments applied for PWR spent fuel assemblies stored in water pool to illustrate how much of spent fuel can be stored and increased with increased burnup. Moreover, destructive measurements have been performed to obtain uranium and plutonium compositions of LWR spent fuel rods for use in validation of depletion calculation code.

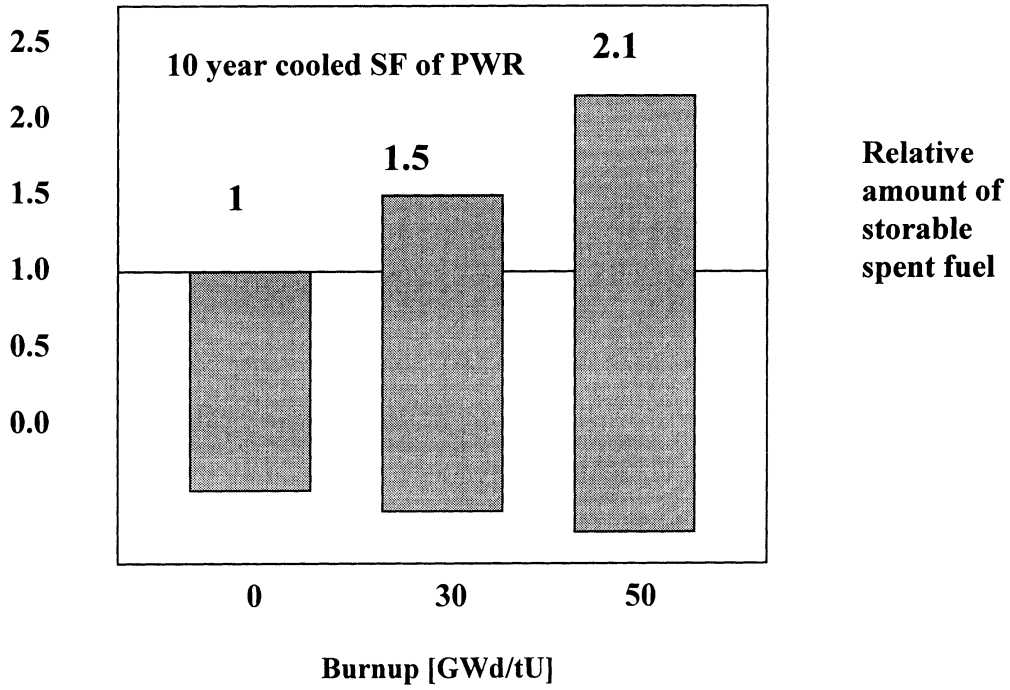

FIG. 1. Relative increase of storable spent fuel amount.

A draft of Japanese Burnup Credit Guide Original Version is now under preparation and submitted for strenuous and ardent discussions of Japanese specialists and experts in the meetings of the Criticality Safety Experiment Data Review Working Group organized by JAERI. This Original Version is expected for use in preparation of documents for licensing safety review and in safety evaluation not only for spent fuel transport and storage but also for spent fuel dissolution at reprocessing facilities. Studies for burnup credit implementation in Japan are to be performed at JAERI by reciprocal exchange of relevant information and co-

operation with national and international industries and research organizations as shown in Fig.2. Since measurement data on spent fuel isotopic composition especially for fission products are generally scarce and difficult to obtain, these cooperation scheme is considered to be vital to achieve the task successfully.

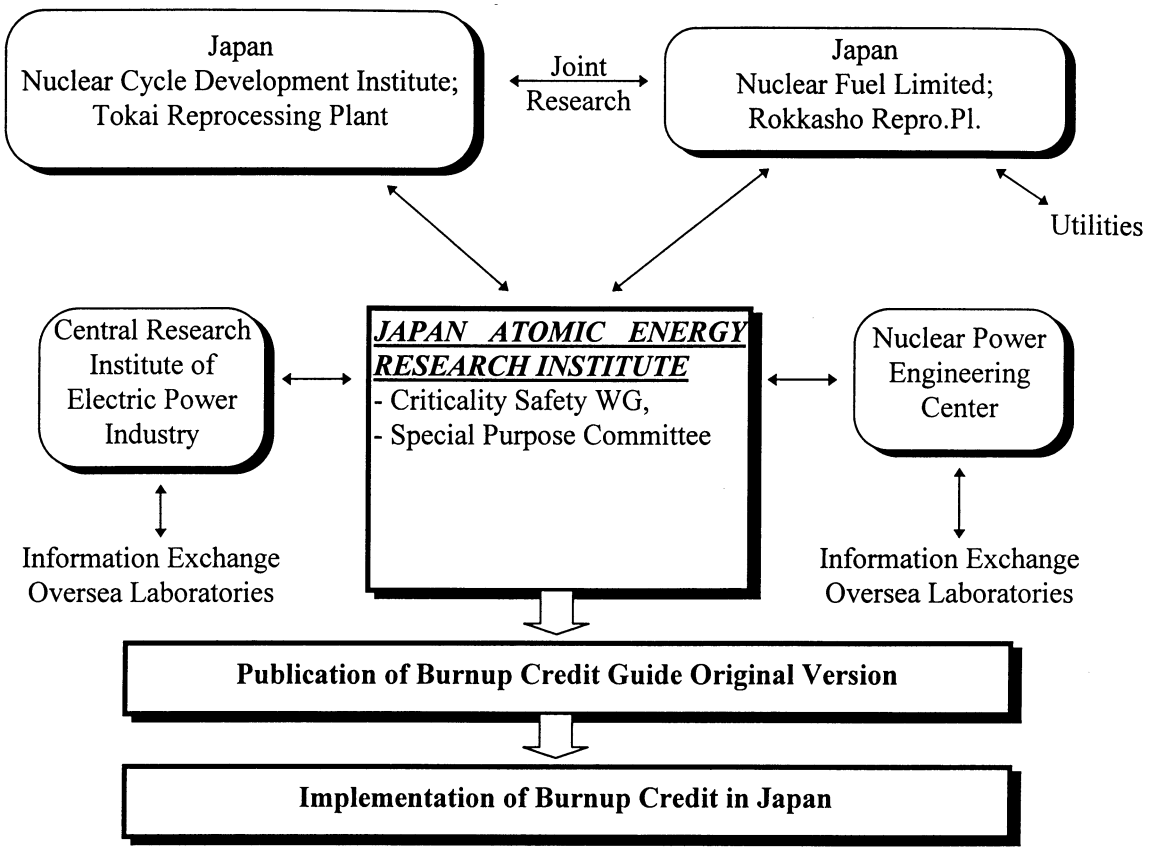

FIG. 2. Scheme for burnup credit implementation in Japan.

3. RESEARCH COOPERATION WITH JNFL ON BURNUP CREDIT IMPLEMENTATION

3.1. Burnup Credit Design Adopted in RRP

In the Spent Fuel Receiving and Storage Building of RRP operated by JNFL (Japan Nuclear Fuel Limited), spent fuel as carried in is firstly stored in the Spent Fuel Temporary Storage Rack designed to accept spent fuel with the maximum uranium enrichment of 5wt%. This specification is universal for the initial UO2 fuel composition of PWR or BWR in Japan. Then, the spent fuel is classified by its residual uranium enrichment determined by burnup measurement into the low (less than 2 wt% 235U) and the high (less than 3.5 wt% 235U) categories. Because the fact shows that most of the spent fuel assemblies exhausted from Japanese LWR have residual uranium enrichment less than 2 wt% 235U. Subsequently, each spent fuel assembly is stored separately in the differently spaced storage racks corresponding to the classification. The correlation of residual uranium enrichment of spent fuel with its averaged burnup is shown in Fig.3. The storage racks have been constructed by robust structure maintaining an appropriately designed spacing between spent fuel assemblies so that criticality accident is not considered to happen due to deformation of storage racks.

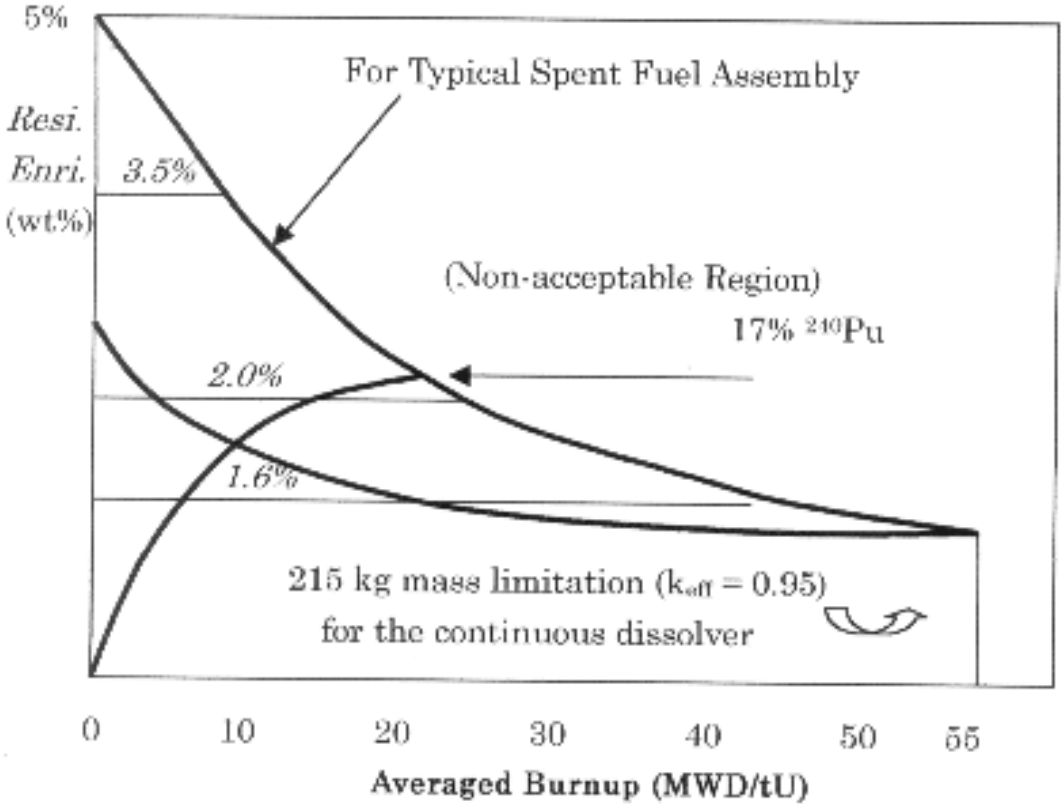

FIG. 3. Correlation of uranium residual enrichment with assembly averaged burnup for spent fuel.

On the other hand, criticality safety design of the continuous dissolver adopts burnup credit so as to keep the multiplication factor less than 0.95 by a combination of burnup and initial uranium fuel enrichment. Consequently, any spent fuel burned up to less than that corresponding to keff = 0.95 curve such as shown in Fig.3 has to be treated with soluble neutron absorber (gadolinium) mixed in the dissolution process. If the effects of fission products could be taken into account for the burnup credit design of the dissolver, the use of extraneous neutron absorber would be avoided to bring an economic and safety merit to the design and operation. Studies have been continued to realize such a technical breakthrough.

As for the processes for spent fuel receiving, storage, sheering and dissolution, criticality safety control is based on specification of one spent fuel assembly. On the other hand, criticality safety control for the following process such as separation and purification is based on whole amount of solution held in the Accountability and Adjustment vessel which should be processed in one day or so. In the latter process region, vessel designs and their operation are based on liquid isotopic composition to secure criticality safety. The operation control is made by keeping the uranium enrichment to be less than 1.6 wt% 235U and the plutonium ratio to be 240Pu more than 17 wt% which is assured by sampling liquid and its measurement made in the accounting vessel. By this isotopic composition control, vessel geometry can be made favorable from the criticality safety point of view in combination with the restriction of fissile fuel concentration.

3.2. Implementation of Burnup Credit at RRP

Presently, calibration tests have been finished for the burnup monitor in anticipation of full-fledged receiving of spent fuel assembly hereafter. However, in consideration of long termed

operation of spent fuel receiving and storage, a lot of problems have been pointed out concerning operation and maintenance of the burnup monitor from effective and economical operation point of view. Namely, the current usage of the burnup monitor is based on an excessive safety margin assigned to the measurement, and needs significant maintenance work to keep its reliability. These situations would be surely improved by employing alternative methods such as resorting to burnup data supplied by reactor core management.

A cooperative research program between JAERI and JNFL has been instituted to develop procedures to take spent fuel burnup data obtained from reactor core management instead of performing measurement for spent fuel receiving in the storage racks. The following research items are now to be addressed for the 3-year cooperative work.

1. Assessment of precision of burnup data obtained from reactor core management:
   - 1.1. Assess precision of average burnup and axial burnup profile data obtained from reactor core management on PWR or BWR spent fuel assemblies,
   - 1.2. Determine the most reliable method to assess the precision in consideration of the existent data of approximately 4 % obtained by several available methods,
   - 1.3. Investigate reliability and representation of the axial burnup profile data of spent fuel assemblies.
2. Assessment of reliability of procedure to evaluate the fuel burnup data:
   - 2.1. Assess human error probability conceived in the evaluation procedure for burnup data from the reactor core management,
   - 2.2. Survey any evaluation procedure available in each utility company, and obtain data on its relevant human error probability,
   - 2.3. Check the procedure adopted by both utility and fuel vender to investigate the causes for discrepancies with burnup data evaluation, if any, and to rectify the error thus to reduce the human error probability. If the error of the burnup data be not rectified and reflected to the next cycle core management, a difference could be observed between the predicted core power distribution and the real one. If the difference could be detected by any means, one can assess the relevant human error probability in this process.
3. Work out a data transfer management system taken both by utility and consignee:
   - 3.1. Work out a data transfer procedure concerning spent fuel released from a power reactor,
   - 3.2. Work out a data confirmation procedure at the power station for spent fuel shipment and at the Rokkasho Reprocessing Plant for spent fuel receiving,
   - 3.3. Determine necessary conditions to confirm burnup data without using burnup monitor, such as those for the fuel rack address and the fuel ID number management at the on-site spent fuel storage rack and also for the independent double check by the operator for ID confirmation, etc,
   - 3.4. Survey any methods to confirm the spent fuel ID implemented in overseas facilities, and make comparison with the proposed one.
4. Assessment of reliability of the whole through data transfer management system:
   - 4.1. Assess human error probability conceived in the procedure concerning treatment of burnup data released from the power station, spent fuel transport, and spent fuel receiving at the RRP Storage Rack,
   - 4.2. Clarify the difference of reliability in determining burnup of spent fuel by the methods with and without a burnup monitor.
5. Assessment of conservatism for burnup data averaged over 50cm from the end of a fuel assembly:

5.1. Evaluate burnup data averaged over 50 cm from the end of a fuel assembly by multiplying assembly average burnup with a certain comprehensive factor,
5.2. Investigate conservatism for the above evaluation method by applying the axial burnup profile data obtained with the burnup monitor at the Rokkasho Reprocessing Plant.

6. Working out whole through logic applied to burnup credit licensing:
6.1. Work out whole through logic for burnup data acceptance procedure without using burnup monitor, which is applicable to the licensing process.

4. CONCLUDING REMARKS

It is now evident for Japan to introduce burnup credit into design and operation of spent fuel storage and transport facilities due to continual accumulation of spent fuel discharged from PWR and BWR power reactors. In the near future, more economical design and operation will be realized for spent fuel on-site storage and transportation from power reactors to RRP by implementing burnup credit methods.

These burnup credits would be firstly at the actinide only level, and partially and already introduced into the design of the Spent Fuel Receiving and Storage Building at RRP. Studies have been continued to take fission products into burnup credit evaluation to improve further economical and safety merits to the dissolver design and operation.

Recently, a cooperative research program has been instituted to study adopting burnup data supplied from reactor core management without burnup monitoring to accept spent fuel into the storage pool. Utilities also seek to the methodology to confirm burnup data without burnup monitoring for loading spent fuel to transport casks.

At JAERI, studies have been continued to develop burnup credit evaluation method including isotopic composition databases and the burnup Credit Guide Original Version is now under preparation for its publication for use in licensing reference documents and design safety evaluation. These studies have been executed under a cooperative work scheme throughout Japan to enhance reciprocal information exchange and common usage of necessary data for burnup credit implementation.

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STATUS OF THE MULTI-DETECTOR ANALYSIS SYSTEM (MDAS) AND THE FORK DETECTOR RESEARCH PROGRAMS

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Abstract

During the past 10 years, the Electric Power Research Institute (EPRI) has supported several projects aimed at obtaining regulatory acceptance of the burnup credit concept in the design of commercial spent-fuel storage and transportation systems. Among others, these projects have included documentation of spent-fuel burnup measurements with the Fork system and development of the Fork+ system, a more advanced instrument capable of measuring burnup independently of utility records. Another EPRI project is a study estimating the maximum uncertainties in utility burnup records. Based on the latter study, it is evident that in many cases reactor records are sufficiently reliable to eliminate the need for at-reactor burnup measurements. Independent of the EPRI research, the U. S. Department of Energy (DOE) is funding the Idaho National Engineering and Environmental Laboratory (INEEL) to research and prototype a nondestructive assay system for characterization of DOE spent nuclear fuel (SNF) and remote-handled transuranic waste (RH-TRU) in preparation for transportation and long-term storage of the fuel and waste. The Multi-Detector Analysis System (MDAS) is intended to provide some understanding of the fissile mass, radiation source term, and fissile isotopic content information from recorded energy spectra, without reliance on initial fuel composition or historic plant operating records. It can verify burnup calculations or be used directly for DOE fuel and waste characterization. The work is conducted under DOE Contract No. DE-AC07-94ID 13223. Patent pending.

1. INTRODUCTION

With over 50 years of commercial and close to 60 years of defense spent nuclear fuel that must be dispositioned, the need for a system that verifies fuel burnup and/or characterizes the spent nuclear fuel is now being mandated by the regulatory agenesis and the need to understand the fissile material in spent nuclear fuel.

The MDAS and the Fork detector systems stem from two different needs. The MDAS system is being developed to provide direct characterization for the many DOE fuel types, over 250 types, with limited qualified documentation. The Fork detector system was developed to confirm the exposure of commercial spent nuclear fuel, with detailed reactor records.

As with many devices, the Fork detector has evolved over the years. The Fork detector was first developed to support IAEA by simply verifying that the fuel that had gone through the reactor was truly exposed, thereby relying on the reactor records for the detailed understanding. As the technology improved and the demands of the industry changed so has the Fork detector. Presently the Fork detector measures the thermal, epi-thermal, gross gamma emissions and uses a Cadium-Zinc-Tellurium (CZT) spectrometer to identify cesium 137 (137Cs) from spent nuclear fuel. The hardware and software modifications have allowed a more direct measurement of the burnup of the fuel and provide a real confirmation of the reactor records or, in some cases where the reactor records are not complete, a substitute method to the reactor records.

The MDAS system is being prototyped at this time. The design also is being modified as the research progresses, with a research goal of demonstrating that the spent nuclear fuel can be characterized for total fissile mass, radiation source term, and fissile isotopic content, through active interrogation of the spent fuel assembly, without special calibration standards or a priori knowledge. After this demonstration is complete, the system will be engineered with new hardware and software to make the system more compact, with greater throughput, automatic data analysis, and lower error. A customized version will be developed to characterize the contents of remotely handled transuranic waste containers.

As noted above there are three basic approaches in verification and characterization devices. The first category involves the indirect determination of these characteristics through the use of plant records and simulation computer software. The second category performs a direct measurement of the spent fuel characteristics. The third category then would be some combination of the first two approaches, depending on the quality of reactor records, operational issues, and costs of measurement.

2. THE FORK DETECTOR

A need for verification of assembly bum up arises from concerns of nuclear criticality safety in the design of storage and transport systems for spent reactor fuel. A verification measurement can provide a check of reactor records that could affect nuclear criticality safety. Studies have concluded that the utility-supplied data on burnup are of greater accuracy and reliability than could be provided by additional radiation measurements on spent-fuel assemblies. The primary reason for this conclusion is the nature of the measurements from which the burnup is derived. The thermal power output of a reactor is measured very accurately (less than one- percent uncertainty) using thermocouples and flow meters at the inlet and outlet of the cooling water circulating in the reactor. The time integral of the thermal power is the thermal energy (gigawatt days, GWd) produced by the reactor and is the basis of the burnup assignment to individual assemblies. In-core radiation measurements located throughout the reactor are used to distribute the GWd to each assembly through a distribution function provided by the manufacturer of the reactor. The distribution function assigns a fraction of the total GWd to each assembly based on the in-core measurements and the history of the assembly. Since the sum of the GWd of all assemblies must equal the total GWd of the reactor over any time span, if the burnup for one assembly is “high,” another must be “low.” The distribution function could possibly generate errors that can be characterized as random, because of the “zero-sum” aspect of the errors. The most likely source of a possible systematic error, one that applies to all assemblies, is the-integral of the power output of the reactor, since the bum up for each assembly is determined by multiplying the distribution fraction by that number. Since the integral of the power output of the reactor is measured very accurately, a relative burnup measurement among the assemblies can determine the extent of the random error among the assemblies and indicate any assemblies whose radiation output is inconsistent with the record for bum up. The Fork system is designed to determine the extent of the random variation among assembly burnups, and to identify any anomalous values.

2.1. Radiation

In the application of nuclear criticality safety to the transport and storage of spent fuel from commercial nuclear reactors, the fuel assemblies of interest have been cooled for over five years, which greatly simplifies the analysis of the emitted radiation. For shorter cooling times,

many isotopes are significant emitters, but most have decayed to insignificance after five years because of the predominance of short half-lives in the fission and activation products. After five years of cooling time, the predominant neutron emitter is curium-244, which is formed by successive neutron capture beginning with uranium-238. The production of curium-244 is found to increase with about the fourth power of the burnup. The neutron emission is therefore very sensitive to variations in burnup. Cesium-137, the major gamma emitter after five years of cooling, is produced as a fission product. Its production is essentially a linear function of burnup. The attenuation factor in the assembly is greater for gamma rays than it is for neutrons, so that neutrons emitted from the assembly can originate from rods deeper inside the assembly than can gamma rays. In the case of neutrons from curium-244 and gamma rays from cesium-137, the neutrons can sample about 80% of the rods of an assembly, while the gamma rays can sample the outer 30% of the rods.

2.2. The Fork Detector Design

The Fork system measures the passive neutron and gamma ray emissions from individual spent-fuel assemblies while the assembly is in the storage pool. The effectiveness of the Fork system for verification of reactor records is due to the sensitivity of the neutron yield to burnup, the self-calibration generated by a series of measurements, and the redundancy provided by three independent detection systems. The assembled Fork detector and its associated electronics are shown in FIG. 1.

FIG. 1. Fork Detector Configuration.

FIG. 2. shows the Fork instrument head. Each of the two arms of the Fork detector contains two fission chambers (outer steel cylinders) to measure the yield of neutrons, and one ion chamber (inner brass cylinders positioned between the fission chambers) to measure gross gamma-ray emission. One fission chamber (the epithermal detector) in each arm is embedded

in a polyethylene cylinder that is surrounded by a thin sheet of cadmium, which serves to absorb thermal energy neutrons. The other fission chamber, outside the cadmium cover, is sensitive to thermal neutrons and the boron content of the water in the spent-fuel pool. The entire assembly is inserted into the polyethylene outer cover as shown in FIG. 2.

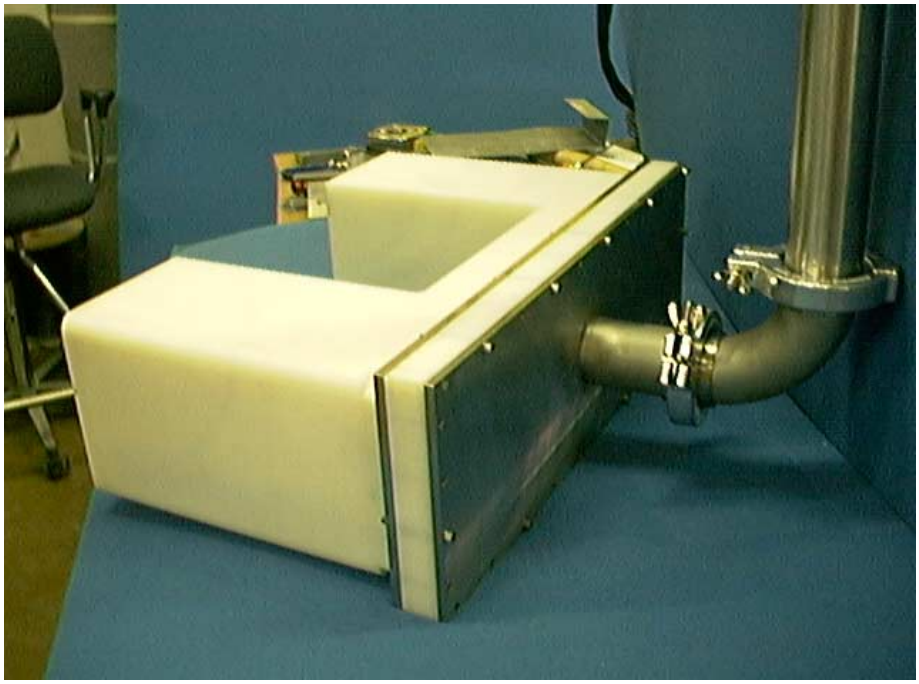

FIG. 2. Fork+ Instrumentation Head.

The epithermal detectors provide the primary data used in the Fork technique. In the original (IAEA) application, the thermal neutron detectors were used to check the variation of the boron content among the spent-fuel pools at different sites. In the present use, the thermal detectors serve as a backup measurement to the epithermal data. In the Fork + configuration, a cadmium-zinc-telluride (CZT) detector was incorporated into the back of the device. The CZT addition provides a third measurement and another level of understanding of the radionuclides present in the spent nuclear fuel. The CZT technique aims to analyze the 662keV peak of cesium-137 (137Cs). Separation of the gamma rays produced by 137Cs from those produced by other isotopes found in the spent nuclear fuel requires that the gamma ray sensor signal processing resolve gamma ray energies to a difference of about 40keV or better. (1)

The CZT crystals produce an electrical output in response to gamma rays. By processing the electrical output of the CZT, it is possible to measure the number of gamma rays striking the crystal and bin the gamma rays according to their energy over a range of energies. The CZT is cooled thermionically, without liquid nitrogen or other means that would be needed if the CZT crystal were larger. Thermionic cooling is accomplished by the addition of a microelectronic thermionic cooling device to the CZT subassembly. By allowing the system to equalibrate to about 275 degrees Kelvin, the system was able to perform the measurement quicker without a lose of accuracy.

FIG. 3. Fork Detector Instrumentation.

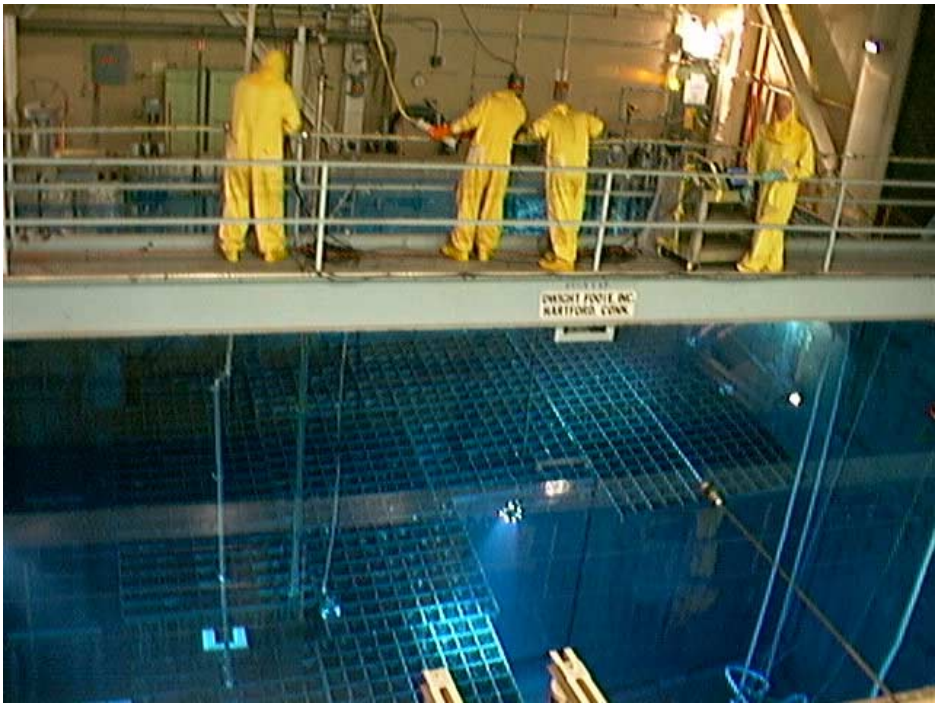

FIG. 4. Fork Detector Configuration in Spent Fuel Pool Area.

2.3. Operation of the Fork and Fork +

The basic equipment for the Fork and the Fork + are the same: instrumentation head, piping that holds the neutron and gamma sensors, and the data gathering equipment, a lap top computer. The detector instrumentation is shown in FIG. 3.

A number of measurement campaigns have been performed at different reactor sites to demonstrate that the Fork system can be used on all commercial fuel types, Oconee Nuclear Station(2), Arkansas Nuclear One(3,4), and Maine Yankee(5). In each of the cases, the Fork detector and its instrumentation was located next to the spent nuclear fuel pool. One of the Fork system's strengths is the compact nature of the equipment as is seen in FIG. 1. The detector head is suspended from the bridge crane while the instrumentation can travel with the bridge crane or can be placed on the pool side depending on the spent fuel pool configuration. The Fork detector can be handled by hand or can be attached to the bridge crane as is seen in FIG. 4. Either the fuel is moved over to the detector or the detector is moved to the fuel position, and the fuel assembly is withdrawn through the detector head. The detector is moved in the storage pool to the location of the spent-fuel assembly that is to be examined. The assembly is raised until the measuring point is located at the detector head, the detector is moved into contact with the assembly, and the neutron and gamma ray data are collected for approximately 60 seconds. A battery-powered electronics unit and microprocessor are used to supply all power to the detectors, collect and analyze the detector outputs, and perform necessary calculations and documentation.

2.4. Fork Results

The Fork measurements correlated well with the records of both reactors at Arkansas Nuclear One, Oconee, and Maine Yankee. The correlation of the reactor records with the emitted neutrons indicated random variations of less than 3% in the records of assembly bum up. No obvious anomalous assemblies were detected, and the maximum deviation for a single assembly was less than 10%. The Fork system proved to be compatible with utility operations and equipment at all of the reactor sites. The effect of the different assembly designs in the two reactors is shown by the functional dependence of the neutron emission on burnup. For the B&W assemblies the neutron signal increased as the 3.83 power of the burnup, in agreement with earlier measurements on the B&W assemblies at Oconee Nuclear Station. For the CE assemblies, the neutron signal increased as the 4.35 power of the burnup. This result suggests that the burnup dependence of the neutron signal may be specific to each assembly design. A variation in the burnup exponent has been noted in IAEA measurements with the Fork system. This variation is probably due to the varying reactor operating parameters among the different reactor designs. The axial scan measurements indicate the resolution that can be obtained with uncollimated detectors, about 15 inches along the axis of the assembly.

The resolution is adequate to locate the neutron sources in the two non-standard assemblies, which produced a rise in the neutron count rate of 25 to 40%. The two non-standard assemblies had been cooling for about 2 years. The relative increase in neutron count rate would be expected to increase with increased cooling time, since the curium-244 signal will decay with an 18-year half-life, while the neutron output of the plutonium-beryllium sources will remain essentially constant. This effect may account for the anomalous results observed in two assemblies at Oconee Nuclear Station. Those two assemblies had been cooling for about 15 years, and are known to have contained neutron sources at one time.

The Maine Yankee, CE unit, campaign used the Fork + system(6) in which the CTZ sensor was added to the system. As before, the neutron sensor data was consistent and well within the same level of accuracy as before. Through the testing program it was learned that additional shielding and collimation of the beam will improve the accuracy of the cesium-137 results. Even though the CZT gamma scan data limited the measurements to relative low assembly burnups, due to signal strength, simple modifications can be made to the Fork+ to ensure proper measurement accuracy at any burnup levels.

2.5. Conclusion

As the need has arisen to verify the reactor records concerning the exposure, burnup, of commercial spent nuclear fuel the development of the Fork and Fork + system has demonstrated that a compact and simple system can perform the needed measurements. It has also been demonstrated that with a small modification to the detector an independent measurement can be performed. In each case, it has been demonstrated that the Fork systems are compatible with spent fuel pool operations. In addition the Fork systems have been demonstrated to be geometry independent when it comes to fuel design.

3. THE MULTI-DETECTOR ANALYSIS SYSTEM (MDAS)

The U.S. Department of Energy (DOE) has a large variety and quantity of legacy spent nuclear fuel destined for permanent disposal at a geologic repository. This fuel often lacks complete and adequate characterization documentation necessary for shipment of the fuel and for its permanent disposal. This lack is caused, in part, by changing spent nuclear fuel characterization requirements over the decades since the first DOE (then Atomic Energy Commission) reactors were built. Also, some of the spent nuclear fuel has deteriorated during extended (usually wet) storage. The lack of adequate spent nuclear fuel documentation may be a major impediment to geologic repository disposal.

Therefore, the Idaho National Engineering and Environmental Laboratory (INEEL) is prototyping a system called the Multi-Detector Analysis System (MDAS) to characterize DOE SNF for fissile mass, radiation source term, and fissile isotopic content(7). A U.S. patent is pending on the system. Funding for the project comes from the DOE Office of Environmental Management.

The MDAS research is being conducted by the INEEL at the TREAT facility on the Argonne National Laboratory(ANL)-West site in Idaho, in close proximity to stored spent nuclear fuel. The work is being done with the collaboration of ANL scientists and operations staff.

3.1. Physics Basis of the MDAS Design

The combination of multiple detectors and coincidence methods is not new, having been used by the physics community for the past 15 years to study the fission process. These studies have revealed new information on the fission and decay processes(8). The MDAS design takes advantage of this new information.

The fission of an atom produces zero to 10 simultaneous evaporation neutrons and, except in very rare cases, two fission fragments. Each fissile isotope has a unique suite of possible fission fragment pairs. The fission fragments immediately lose energy by emitting strong

signature de-excitation gamma rays of the order of 1 to 5 Mev. These gamma rays are energetic enough to be detected outside the spent fuel assembly.

Specially designed fast detectors and fast coincidence methods are used to recognize the fission event. Multiple detector signals are compared by the trigger electronics to evaluate whether a coincidence event has occurred. Neutrons and gamma rays in the very short time window of 50-100 nano-seconds associated with the coincident event are recorded. Background radiations not associated in time with a fission event are ignored by the electronics, thereby maximizing the fission event information in the acquired data and excluding much of the background radiation. The data is permanently stored on 19mm tape. At any later time, the data can be read from the tape and compiled into spectra for analysis. The total fissile mass is computed from the total count of coincident neutrons, using the fact that fission is the only non-accelerator source of coincident neutrons. The identity of the element fissioning, and even the fissile isotopes, are recognized by the signature gamma ray energies that are specific to the fission fragments.

3.2. MDAS Components

The MDAS is shown in its initial configuration in FIG. 5, and FIG. 6 depicts the graphite shield ring and the neutron generator that have been added to the system in recent months.

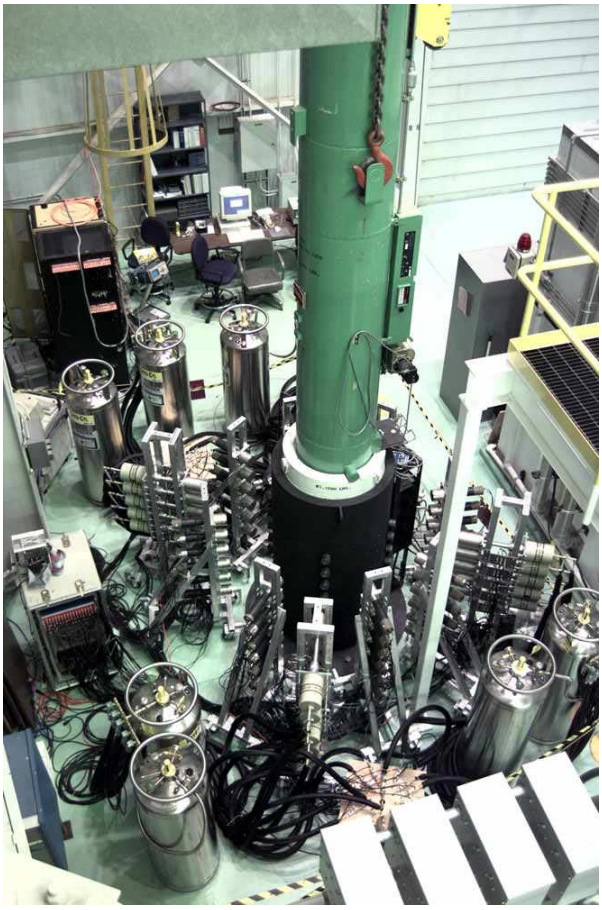

FIG. 5. MDAS Experiment Phase 2, Full Setup With Cask.

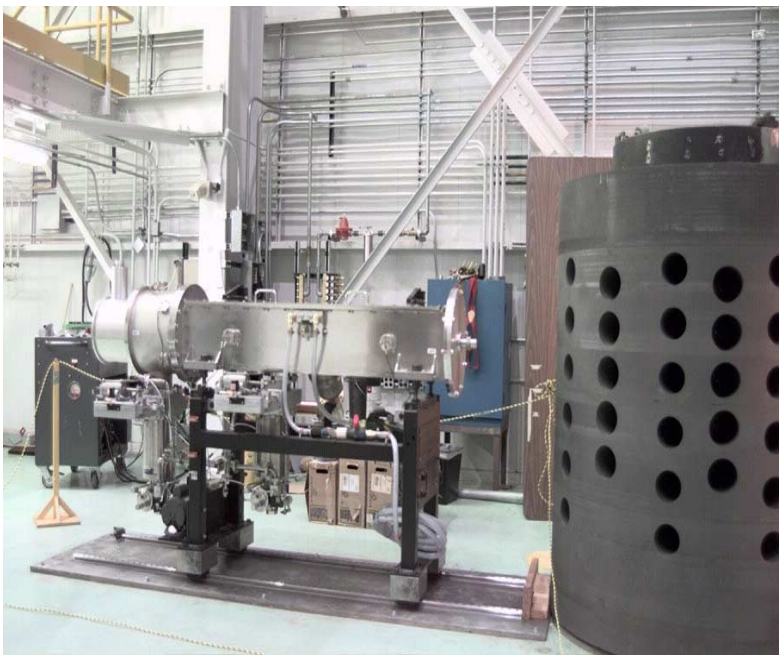

FIG. 6. MDAS Experiment Phase 2, Neutron Generator and Shield Ring.

At the center of the MDAS system is a shield ring which actually is constructed of two concentric rings. The original ring of stainless steel is now inside the new one of graphite. The combined ring stands 8-feet high with a 26 inch inside diameter. There are 68 detectors mounted around the shield ring: 20 solid-state, high purity germanium crystals to detect gamma rays and 48 xylene-filled scintillators to detect neutrons. Each detector is aligned with a penetration hole through the combined ring.

The germanium detectors were specifically designed and constructed to operate in the radiation fields associated with spent nuclear fuel. This is accomplished by having a small, low-efficiency crystal with energy resolution appropriate for spectroscopy of fission fragments. The crystal is housed together with an on-board pre-amplifier and a liquid nitrogen dewar, used for cooling.

The xylene-filled scintillators consist simply of the scintillator and a photo-multiplier tube for light collection. They have a high efficiency for fast neutrons but also respond to gamma rays. However, the incident gamma rays and neutrons interact differently with the detectors, with the detector signal having a much sharper drop off time for gamma rays. The system can distinguish the two radiation types through electronic pulse shape discrimination.

The neutron source now used with the prototype is a 1.5 MeV deuterium on beryllium neutron generator that will provide up to 1011 neutrons per second in an energy range of up to 4 MeV. The majority of neutron energies are around 200 keV range to take advantage of the higher cross sections at that energy.

3.3. MDAS Operation

In operation, a cask loaded with one spent fuel assembly in a container is brought by overhead crane to the top of the shield ring. The fuel assembly, still in the container, is lowered into the ring. The neutron source bombards the already highly radioactive assembly with neutrons to

induce fission in the fissile mass. A single fission event is recognized when coincident radiations are detected, and a 50-100-nanosecond signal-processing window is opened. In this time window, gamma-ray energies, neutron multiplicity, and their time relations are stored by the data acquisition system on an event-by-event basis. This data includes the background data as well as the fission data. The fast electronics are capable of recording up to about 50,000 coincident events per second. To obtain good statistics, it is expected that it will take the MDAS prototype 1-4 hours to gather data on one spent fuel assembly. Increasing the number of detectors would increase the number of recorded events per second, thus reducing this measurement time.

The MDAS operation depends on recent developments in fast electronics and the custom software written by the research team for the process of coincidence detection, data rejection/data acceptance, and data recording.

3.4. MDAS Progress to Date and Future Plans

During the initial Phase 1 testing stage, it was demonstrated that the research design accurately detected the coincident neutrons from spontaneous fission and recorded the associated gamma energies. The final evaluation of the prototype will be determined in Phase 2 by measurement of characterization data for two ANL-W spent fuel assemblies, analysis of the data, and comparison of the MDAS characterization with destructive chemical analysis previously performed by ANL-W.

Engineering improvements to the MDAS system in Phase 3 are expected to make the system more compact, with greater throughput, automatic data analysis, and lower error. Future work will also construct a modification of the MDAS system to characterize the contents of remotely handled transuranic waste containers.

4. CHARACTERIZATION AND CERTIFICATION REQUIREMENTS

The DOE National Spent Nuclear Fuel Program has conducted a study of the spent nuclear fuel properties that might be required for the characterization and certification of the fuel for transportation, storage, and geologic disposal. At the time of the study, the MDAS system and the Fork detector were already under development. The study combined and extended several lists of parameters that already existed within the DOE. The study report was released in September 1998(9). It identifies the following 9 key properties and 33 non-key properties that nondestructive analysis/nondestructive evaluation (NDA/NDE) systems potentially should be able to characterize.

Key properties: (1) total fissile mass, (2) enrichment, (3) Pu-239 content, (4) Pu-241 content, (5) Np-237 content, (6) U-233 content, (7) U-235 content, (8) I-129 content, and (9) Tc-99 content.

Non-key properties: (1) activation products, (2) burnup (MWD/MTHM), (3) canister thermal output, (4) clad material, (5) concentration and masses of absorbers, (6) condition of fuel (intact, degraded, failed), (7) disposable canister leak rates, (8) free liquids in canister, (9) fuel compound (UO2, U-metal, etc.), (10) fuel dimensions (pellet and clad ID and OD), (11) fuel rod pitch or plate spacing, (12) gross gamma flux, (13) gross neutron flux, (14) identification of control components, (15) k-eff, (16) last date of irradiation (YYYYMMDD), (17) list of

fission products and actinides, (18) moisture content of fuel item, (19) moisture content of loaded canister, (20) MTHM (Pu+Th+U), (21) number of SNF assemblies/items, (22) organics, (23) Pu-238 and Pu-240, (24) requirements that canister be sealed, (25) Th-232, (26) thermal output for disposable canisters, (27) total Pu, (28) total Th, (29) total U, (30) type of SNF assembly – name, (31)U-232, (32) U-234 and U-236, (33) U-238.

A review of 27 available NDA/NDE processes was performed, with the results documented in the same report(9). The evaluation indicated that the MDAS system could be developed to directly measure seven of the nine key spent nuclear fuel properties. The other two key isotopes, 129I and 99Tc content, may be computationally determined, e.g., using the masses of other fission products that can be measured with MDAS plus the calculated ratio of masses of these radionuclides to the masses of 129I and 99Tc. In addition, the MDAS is expected to measure 14 of the 33 non-key properties directly or indirectly. The evaluation further indicated that the Fork and Fork+ detector already could address one of the nine key parameters and seven of the 33 non-key properties.

5. SUMMARY

In summary, both the Fork and MDAS technologies promise to provide long term benefits to the U.S. commercial nuclear power industry, the U.S. Department of Energy, and similar international organizations.

Table I. Comparison of Fork and MDAS Systems.

	Fork	MDAS
1. Purpose	Determine burnup indirectly, & directly neutron and gamma counts	Characterize SNF directly
2. Status	IN USE	Under development
3. Interrogation mode	Passive	Active
4. Burnup codes, records required	No/Supportive	No
5. Portability	Yes	Not at this time
6. Operating environment	Underwater, dry cell	Dry cell
7. Time for one measurement	1 minutes	<Two hours (goal)
8. Time for data analysis	1 minutes	<Two hours (goal)
9. Axial profiling	Yes	Yes
10. Applications	Commercial SNF	DOE SNF, Commercial SNF, & Transuranic waste

The two technologies meet two diverse sets of requirements. The Fork detector is easily deployed and rapidly completes its measurements, providing minimal but perfectly adequate information for determination of burnup credit at the 50% limit currently accepted by the U.S. Nuclear Regulatory Commission. The MDAS prototype, on the other hand, is not at all portable at this time. The MDAS technology will provide the detailed information needed to characterize the incompletely documented spent nuclear fuel and waste forms within the DOE complex and will also provide more complete characterization of spent nuclear fuel and transuranic waste should it be required in the future.

Table 1. summarizes the comparison of the two technologies. The progress of the MDAS research and the continued improvements in the Fork detector will be presented at future meetings.

ACKNOWLEDGEMENTS

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ISSUES FOR EFFECTIVE IMPLEMENTATION OF BURNUP CREDIT

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Abstract

In the United States, burnup credit has been used in the criticality safety evaluation for storage pools at pressurized water reactors (PWRs) and considerable work has been performed to lay the foundation for use of burnup credit in dry storage and transport cask applications and permanent disposal applications. Many of the technical issues related to the basic physics phenomena and parameters of importance are similar in each of these applications. However, the nuclear fuel cycle in the United States has never been fully integrated and the implementation of burnup credit to each of these applications is dependent somewhat on the specific safety bases developed over the history of each operational area. This paper will briefly review the implementation status of burnup credit for each application area and explore some of the remaining issues associated with effective implementation of burnup credit.

1. INTRODUCTION

Since the mid-1980s the domestic utility industry, the U.S. Department of Energy (DOE), and the U.S. Nuclear Regulatory Commission (NRC) have actively considered the incentives, benefits, and obstacles associated with implementing burnup credit in the criticality safety evaluation for storage, transport, and disposal of spent nuclear fuel (SNF). The incentives first emerged with spent fuel storage pools. Lack of off-site alternatives (i.e., reprocessing, permanent disposal, or interim storage) provided significant incentives for utilities to obtain optimum use of the fixed pool storage capacity currently in place. Exacerbating the demand to optimize pool storage space was the trend towards increased initial enrichments, a trend, which continues to the present. Thus the simple, yet conservative, assumption of using unirradiated "fresh fuel" isotopics for the criticality safety analysis became a significant economic barrier to continued operation of reactor power plants.

By the end of the 1980s several utilities had begun to use burnup credit in the safety analysis for their storage pools at PWRs. Efforts were initiated to evaluate the incentives and seek resolution of technical issues associated with the use of burnup credit in SNF storage and transport casks. In contrast to many countries where burnup credit is desired primarily to increase the allowable enrichment within existing cask designs, the United States nuclear industry is seeking to develop a new fleet of storage and transport casks that are optimized for the anticipated SNF contents. The long cooling times, on the order of 5 years or more, provide considerable flexibility for capacity increase in comparison to the shorter cooling times used in countries that reprocess. Rail casks with capacities of 32 PWR assemblies are being designed—an ~30% increase over existing storage cask concepts. These increased cask capacities can enable a reduction in the number of casks and shipments, and thus have notable economic benefits while providing a risk-based approach to improving safety. Arguments for improvement in safety have noted that the fewer shipments required with burnup credit cask designs will reduce the radiation exposure to both workers and the public as well as reducing the potential for a transport accident involving a cask. Arguments have also been made that the increased capacity per cask increases the potential consequence from any hypothetical transport accident. In either case, from the perspective of criticality safety, it is clear that the

use of burnup credit should enable an adequate margin of subcriticality to be maintained while increasing cask capacity.

Incentives for use of burnup credit in boiling water reactor (BWR) applications have not been as significant as for PWR applications. The reason for this reduced incentive is that BWR fuels have less reactivity than PWR fuels and increased use of neutron poisons in intervening regions between assemblies have proven effective for maximizing capacities and allowing fairly high initial enrichments [1]. Thus, the incentives are largely limited to reducing the cost of neutron poison plates and allowance for higher initial enrichment fuel (up to 5.0 wt% U-235).

However, the incentives for implementing burnup credit have really not been a debated issue in the United States. Rather the debated issues have been associated with the ability to demonstrate the technical basis commensurate with the existing expectations of each application area. This paper will briefly review the implementation status of burnup credit for each application area and explore some of the remaining issues associated with effective implementation of burnup credit.

2. APPLICATION AREAS

2.1. Reactor Operations

Accurate prediction and understanding of the changing nuclide inventory as a function of burnup is a necessity to safe and efficient operation of a nuclear reactor. Major efforts have been expended by the nuclear industry to ensure that the changing isotopic compositions of fuel assemblies in an operating reactor are properly accounted for and that effective analysis methods are available to "follow" and predict operating conditions for the reactor. Of primary interest is the integral effect (i.e., neutron multiplication) of the changing SNF inventory. The analytic methods used in reactor operations have traditionally been based on geometric and physics approximations (primarily applicability of neutron diffusion theory) to the Boltzmann radiation transport equation, but have been made increasingly reliable with continuous feedback experience (i.e., integral validation) gleaned from a 40-year period of operating commercial light water reactors (LWRs) in a controlled facility. However, the analysis methods used for calculation of the effective neutron multiplication factor (keff) in commercial LWR operations are typically not applicable for out-of-reactor situations where their geometric and physics approximations are not valid. In addition, the nuclide inventory provided by the reactor core-following codes has historically not included many of the nuclides that are important to the prediction of keff in out-of-reactor operations because of the build-up of absorbers in the absence of a significant neutron fluence.

2.2. Pool Storage

Storage of spent fuel in underwater racks at reactors has been standard practice in the United States since the start of the nuclear industry. Spent fuel pools (SFPs) at reactors are licensed in the United States under 10 CFR 50 [2]. They represent controlled facilities operated in conjunction with the reactor operations. In lieu of credit for boron in the water, the NRC Office of Nuclear Reactor Regulation has licensed use of burnup credit for many years in borated SFPs at PWR plants. In establishing the safety basis, the general approach used in the United States involves blending the experience and reliability from the reactor core-following codes with the double contingency principle typically applied for out-of-reactor criticality

safety. The SNF inventory subsequent to decay of the short-lived 135Xe isotope is typically used within a storage pool geometry to determine a fresh fuel enrichment that provides the same reactivity as the SNF inventory. This "equivalent" fresh fuel enrichment is then used within a criticality safety analysis code to perform the actual safety analysis for the pool. Little or no validation of the isotopic inventory prediction via comparison with SNF chemical assays is performed; instead, the reliability of the analysis approach in performing core-following calculations is considered to be adequate. Similarly, validation of the cross-section data, as typically provided by critical experiments, is limited to the fresh fuel nuclide inventory.

The current burnup credit approach for SFPs hinges on the adequacy of the process to determine the SNF-equivalent fresh fuel assembly enrichment as well as the proper use of the equivalence information within environments that provide similar neutronic characteristics. Until recently, this general process had been used to obtain burnup credit in PWR SFPs where credit for the soluble boron is taken only for postulated accident events. Recently, however, credit for soluble boron up to 5% in reactivity has been allowed by the NRC [3]. Credit for reactivity decreases associated with fissile depletion and absorber nuclide increases (i.e., burnup credit) has not been allowed for BWR storage pools (where there is no soluble boron); instead, the approach has been to obtain an equivalent fresh fuel enrichment associated with the peak reactivity anticipated for the BWR fuel during the depletion process (reactivity initially increases early in life due to depletion of the gadolinium absorber in the assembly).

2.3. Transport and Storage Casks

The U.S. regulatory requirements for transport and dry storage (as opposed to wet storage in a pool) of SNF are included in 10 CFR 71 and 72, respectively [4,5]. Both regulations are the responsibility of the NRC Office of Nuclear Material Safety and Safeguards. Neither regulation has any specific requirement that would prevent burnup credit from being implemented in the safety analysis. In the case of dry spent fuel storage, water in-leakage to the cask during storage is not considered credible; thus, burnup credit for PWR fuel is not typically necessary since the only flooded condition corresponds to fuel loading and unloading, where soluble boron in the water may be used for reactivity control. Soluble boron is not present in BWR SFPs, and thus for fuel loading or unloading at a BWR, negative reactivity associated with soluble boron is not available.

The domestic and international practice of assumed upset conditions for transport is that water in-leakage be considered in the evaluation of a single cask. Consequently, spent fuel canisters planned for use in transport must be shown to maintain an adequate subcriticality margin when flooded with fresh water. It is not desirable to have separate spent fuel canisters for storage and transport; thus, canisters designed for use with both storage and transport casks (or overpacks) have become the standard industry practice in the United States. As a result, the regulatory requirements for transport directly impact storage practice. For example, it is not desirable to load spent fuel into a canister and seal-weld the canister for storage if the contents are not allowable for transport. Therefore, the need for burnup credit in casks is driven by the regulatory requirements for transport.

Since 1985 significant effort has been devoted to investigating the operational merits and technical issues associated with burnup credit for cask transport and storage of LWR spent fuel. The efforts have focused on PWR fuel with only scoping studies performed for BWR fuel. To date, there is no regulatory experience in the United States with licensing an LWR cask with burnup credit. However, the NRC has issued interim staff guidance (ISG8) [6] that

provides recommendations for implementing burnup credit in the safety analysis of PWR casks. The recommendations within ISG8 limit the burnup credit to that available from actinide-only nuclides for SNF with assembly-average burnup of 40 GWd/MTU or less and a cooling time of 5 years. The ISG8 recommendations allow spent fuel with burnup values greater than 40 GWd/MTU to be loaded in a cask, but burnup to only 40 GWd/MTU can be credited. Initial enrichments up to 5.0 wt % U-235 are allowed (special provisions/penalties are required for enrichments beyond 4.0 wt % U-235). However, assemblies with burnable absorbers are not allowed. The approach to implementation of burnup credit in safety analysis for transport packages will involve predicting the nuclide inventory with a code that will provide adequate individual isotopic information for SNF and subsequent use of that inventory to determine the keff value.

The ISG8 recommends that the analysis methods used to predict the SNF isotopics and keff value be validated against measured data and that efforts be made to identify and/or bound potential uncertainties caused by variation in reactor operating histories, lack of measured data for validation, and the spatial variation of the burnup within the assembly (axial and horizontal). Further, the ISG recommends the use of a measurement prior to or during the loading procedure to ensure that each assembly is within the loading specifications for the approved contents (e.g., a burnup measurement). The recommendations for a bounding approach and pre-shipment measurements are consistent with the international regulations for transport of fissile material [7], which directly address transport of irradiated nuclear fuel.

2.4. Permanent Disposal

Licensing requirements for permanent disposal of SNF at a proposed repository in the United States are continuing to evolve as the NRC Office of Nuclear Material Safety and Safeguards considers realistic requirements appropriate for demonstrating protection of the public health and safety. Proposed changes to the regulations allow the potential for criticality in the post-closure phase of the repository to be considered in light of the probability of occurrence and the consequences to the total system performance. The quantity of fissile material being considered for disposal together with the uncertainties associated with degradation and movement of the material over geological time frames makes this a practical approach that will provide safety to the public. Thus, the licensing approach [8] being considered seeks to identify credible (above a certain probability of occurrence) configurations with a potential for criticality and explore the consequences that might result from such critical events. For intact fuel, the licensee is seeking to evaluate the configurations using SNF isotopic compositions that include both actinides and stable fission products. Additionally, burnup credit for both PWR and BWR fuel is being considered. The analysis and validation approach for disposal waste packages is more similar to that considered for storage and transport casks than the approach used for SFPs. Excessive conservatism is often used for criticality safety analyses outside reactors as a means to simplify development of the safety basis and the review process. However, recognizing the impact that such excess conservatism will have on the facility design, significant effort is being expended to mitigate any undue conservatism and provide realistic estimates of the potential critical configurations needed for the risk-based approach used in the repository licensing.

3. DISCUSSION OF APPROACHES

The approaches used to resolve a technical problem are typically based on historical precedence and experience in the subject area. The need to consider burnup credit came initially to the SFPs, when the absence of disposal and reprocessing options caused the capacity requirements to progressively exceed initial design expectations. Credit for burnup or soluble boron was needed to extend the pool capacity. At the time, the Advisory Committee on Reactor Safety considered potential loss of soluble boron to be of greater concern than any uncertainties associated with implementing burnup credit. Thus burnup credit was implemented in a fashion consistent with the analysis and operations experience within the reactor industry and NRC Office of Nuclear Reactor Regulation. However, the presence of boron in the pool remains an important component of the safety basis in that it provides support for satisfying the double contingency principle of out-of-reactor criticality safety [3]. Licensing analysis for burnup credit is based on site-specific conditions and assumptions relative to plant operations and fuel inventory.

The second application area to address burnup credit was transport and dry storage. The use of transport casks in the public domain means the operational environment is more unpredictable and the controls less reliable—a fact considered in the existing U.S. [4] and international [7] regulations for transport, which are considerably more prescriptive relative to the assumptions for normal conditions of transport and hypothetical accident conditions. The approach that was used immediately sought to meet the requirements of national consensus standards for criticality safety outside reactors while extending the safety analysis to use a “bounding” spent fuel inventory. Also, since transport casks have been historically licensed based on specified contents and independent of a specific facility, the need to assume reactor conditions and assumptions that bound all potential plant operations had to be considered. The composite result of all of these constraints was that the technical complexity for using burnup credit increased. In addition, the original applicant seeking a viable approach to burnup credit in transportation (the U.S. DOE) did not have initial success convincing the regulatory office (the NRC Office of Nuclear Material Safety and Safeguards) that there was ample short-term need to focus resources on the issue. This situation changed as the SFP storage availability continued to decrease, and the reactor industry reliance on storage casks increased. Recently, the need for more efficient storage capacity coupled with the potential for dual utilization in the transport mode has made burnup credit a near-term issue that has demanded increased attention from both the U.S. NRC and the domestic nuclear industry.

The latest application area to consider use of burnup credit has been permanent disposal. Being a first-of-its-kind facility, the regulatory requirements and the licensee safety basis are both evolving as information is gained. The applicant is seeking to use a best estimate approach to predicting keff that considers actinides and fission product nuclides for intact fuel only. The repository is a site-specific application; but the SNF is from all operating plants, and so, conditions and fuel from all reactors must be considered. However, the regulations are far less prescriptive than those for storage and transport and the risk-based approach anticipated for the latest regulatory change allows considerable flexibility in the assumptions and approaches that can be used to assure public safety.

4. COMPARISON OF APPROACHES

The regulatory allowance of burnup credit in SFPs, including credit for fission products, seems to be partly justified [9] by the presence of soluble boron in the spent fuel pool. The reactivity margin associated with the soluble boron is inherently credited in SFP burnup credit analyses to account for uncertainties associated with the utilization of burnup credit. This approach is justified on the basis that there is typically sufficient soluble boron present in PWR SFPs (soluble boron concentrations of ~2000 ppm are common) to maintain subcriticality even if an entire storage rack intended to accommodate burned fuel were misloaded with fresh fuel assemblies of the highest allowable enrichment. Note that recent allowance for partial soluble boron credit (up to 5%) reduces this associated margin. In contrast, guidance for burnup credit criticality safety evaluations for dry storage and transport [6] calls for an assessment of individual sources of uncertainty and consideration of these uncertainties in the safety evaluation—a practice consistent with the national consensus standards for criticality safety outside reactors.

Spent fuel pools provide a protected, controlled environment within the confines of the reactor site and where responsibility for safety resides. This may account for why burnup credit criticality analyses for SFPs do not typically address the numerous issues that have been identified in the context of burnup credit for transportation. The following paragraphs briefly review the three major differences between the requirements for criticality safety analyses for SFPs and cask storage and transport. In the comparison noted below, which highlights the added constraints for burnup credit in transportation, the allowances for SFP analyses are all justified by the presence and control of soluble boron.

The first notable difference between the two NRC guidance documents for pool storage [3] and dry storage/transport [6] is the selection of nuclides used in the implementation of burnup credit. The SFP analyses included credit for all nuclides except Xe-135 without explicit consideration of uncertainties in the calculated nuclide concentrations or assurance of their presence (e.g., fission-product gases). To account for uncertainties in fuel depletion calculations and nuclide presence, an uncertainty equivalent to 5% of the reactivity decrement to the burnup of interest (5% of the reactivity reduction from fresh to the burnup of interest) is suggested as an acceptable assumption [3]. In contrast, proposed burnup credit for dry storage and transport [6] may credit only a subset of the available actinides present and must employ conservative isotopic biases determined from benchmarks of applicable fuel assay measurements. In addition, Ref. 6 limits the safety analyses to a single cooling time of 5 years while Ref. 3 allows consideration for all cooling times. Thus, SFP analyses are allowed 95% credit for the reduction in reactivity associated with all of the calculated isotopics (except Xe-135), but analyses for a transport application currently allow only a limited number of actinides and must substantiate the uncertainty in their prediction via comparison with measurement.

In regard to depletion calculations, no clear guidance or requirements for bounding depletion parameters, similar to those suggested in Refs. 10–11, exist for SFP analyses. Assemblies that used fixed burnable absorber rods (e.g., burnable poison rods and axial power shaping rods) are currently allowed to assume burnup credit in SFPs. In addition, assemblies with integral burnable absorbers (e.g., integral fuel burnable absorber and UO2/Gd2O3 rods) are allowed in SFPs. Allowance of burnup credit for assemblies with burnable absorber rods or integral burnable absorbers is not recommended in the current guidance for dry storage and transport [6]. The U.S. NRC is sponsoring work to provide a basis for removing this restriction.

The second major distinction between the approach used in SFPs and that currently proposed for transport and dry storage is that the safety evaluation for SFPs typically uses fresh fuel with a reactivity determined to be equivalent to spent fuel at a specified burnup. Uncertainties are associated with this approach in terms of the effect on the neutron spectrum (and associated reactivity worth of the poison material) and the geometric conditions under which the equivalency may be valid. For example, the fresh fuel equivalent for SNF in unborated water will be different than that in borated water [12]. Other illustrations, perhaps extreme, of the uncertainties and concerns have been documented [13]. The finite geometry of a cask in comparison to the effectively infinite geometry of an SFP leads to significant differences in reactivity depending on the location of the assembly within the cask, thus making the reactivity equivalence approach inadequate for use in cask analysis. Instead, the criticality safety analyses for transport and dry storage are currently required to use SNF nuclides predicted using codes and data validated against measured isotopic information. Furthermore, the analysis methodologies for calculating keff must be validated for the specific nuclides that are credited.
The recommendations of ISG8 note that the axial and horizontal variation of burnup within an assembly merit special consideration be given to the spatial variation of the SNF nuclide inventory such that conservative estimates of keff are determined in the analysis. Modeling for SFP analyses typically assume uniform axial burnup (modeled as equivalent fresh fuel), and thus are required to determine and include a reactivity penalty associated with the axial burnup distribution [3]. This penalty is determined based on the comparison of a calculation with uniform axial burnup (using equivalent enrichment) and a calculation with axial distributed burnup (using equivalent enrichments for each axial zone). Unlike analyses for transport and dry storage, use of a bounding axial burnup distribution is not required. Further, there are currently no requirements related to horizontal burnup distributions for SFP burnup credit criticality safety assessments.

The third significant distinction between burnup credit applications in SFPs and transport and storage casks is that verification of assembly burnup through measurement is recommended prior to cask loading, but administrative confirmation procedures are acceptable for SFP storage. In both cases, the assembly burnup value used for comparison to the loading criteria is a percentage of the reactor record burnup value. Although variations among utilities are believed to exist, the assembly burnup value used for establishing acceptance for SFP storage is typically between 90 and 95% of the reactor record value. For transport and dry storage, the percentage of the reactor record burnup value will be determined based on comparisons to measurements that can be related to the burnup.

Industry would like to eliminate the regulatory requirement for pre-shipment measurements of each assembly for cask loading or reduce the burden by performing measurements within the SFP to obtain a statistical sampling that demonstrates the accuracy of the utilities administrative records relative to fuel exposure history. This can be a significant economic benefit to the industry, but its implementation must be done in a manner that does not compromise assurance of the characteristics of the fuel assemblies being loaded in a particular cask. The measurement methods and the various proposed methods for their implementation need to be further reviewed to support development of future regulatory guidance. Such regulatory expectations would include specification of proper measurement criteria needed to corroborate of reactor records. An industry report that discusses the variation in the way that utilities obtain and maintain their records on spent fuel burnup together with a discussion of the anticipated uncertainty in the reported burnup would be beneficial to development of loading curves that are independent of the reactor facility.

Table I. Comparison of regulatory requirements for pwr burnup credit criticality safety assessments in pool storage, dry cask storage, and transport.

Issue	Regulatory guidance	
	Spent fuel pools[a]	**Transport and dry storage**[b]
Nuclides credited	All nuclides except ^{135}Xe, with depletion uncertainty equal to 5% of the reactivity decrement	Select actinides-only, with conservative biases applied to the concentrations
Modeling B fuel	Equivalent fresh fuel enrichments	Explicit isotopic content
Modeling B burnup distributions	Consideration of axial burnup distribution	Bounding consideration of axial and horizontal burnup distributions
Validation requirements	Criticality code validation with fresh fuel isotopics	Validation of criticality and depletion methodologies for the specific isotopics credited
Maximum allowable burnup	None specified	No credit for burnup beyond 40 GWd/MTU
Maximum allowable initial enrichment	5.0 wt % ^{235}U	4.0 wt % ^{235}U (5.0 wt % with offset penalty)
Fixed burnable Absorbers	Acceptable	Perhaps unclear from the text of ISG8, but intended to be not acceptable
Integral burnable Absorbers	Acceptable	Not acceptable
Requirement for Burnup Measurements	No	Yes
Cooling time	All cooling times allowed	5-year cooling time

[a]Guidance per Ref. 3.
[b]Guidance per Ref. 6

The three distinctions discussed above are meant to illustrate the disparity that can arise in the implementation of burnup credit even within a single country. A comparison between the regulatory guidance on burnup credit for SFPs and transport or storage casks is summarized in Table I. These differences can be attributed to the different approaches for demonstrating safety that have evolved within each application area prior to the introduction of burnup credit as an option. In the United States, the industry and regulatory components responsible for each application area have historically sought to develop the basis for burnup credit with little consideration towards developing a consistent and viable approach amenable to all areas. To date the only country that has approved transport casks for use with burnup credit has been France. Unlike the United States, the French have used virtually identical approaches for applying burnup credit in storage pools and in transport casks: the minimum burnup as averaged over any contiguous 50-cm segment of the fuel is applied as a uniform burnup over the entire fuel length, and only the uranium and plutonium isotopes are considered. The advantage of using the same technical approach for all applications (SFPs, transport, storage, etc.) is that it allows an effective interface of the safety evaluations between the application areas.

5. CHALLENGES TO IMPLEMENTATION

From the authors' perspective, the major challenge to the implementation of burnup credit for out-of-reactor applications is the added complexity required for the safety evaluation. Figure 1 provides a schematic that highlights the differences between criticality safety evaluations performed assuming burnup credit and those assuming the fresh fuel assumption. The safety analysis report (SAR) will become more complex, thus increasing the time required for thorough preparation and review. In addition, there is a need to establish technical specifications to ensure that loaded contents are consistent with the allowable contents analyzed in the SARs. Consequently, the technical specifications and operating procedures associated with cask loading will be more complicated. The SAR for a burnup credit cask must assure that the restrictions imposed for certifying the cask contents can be readily understood and implemented at any potential facility that has a license to handle SNF.

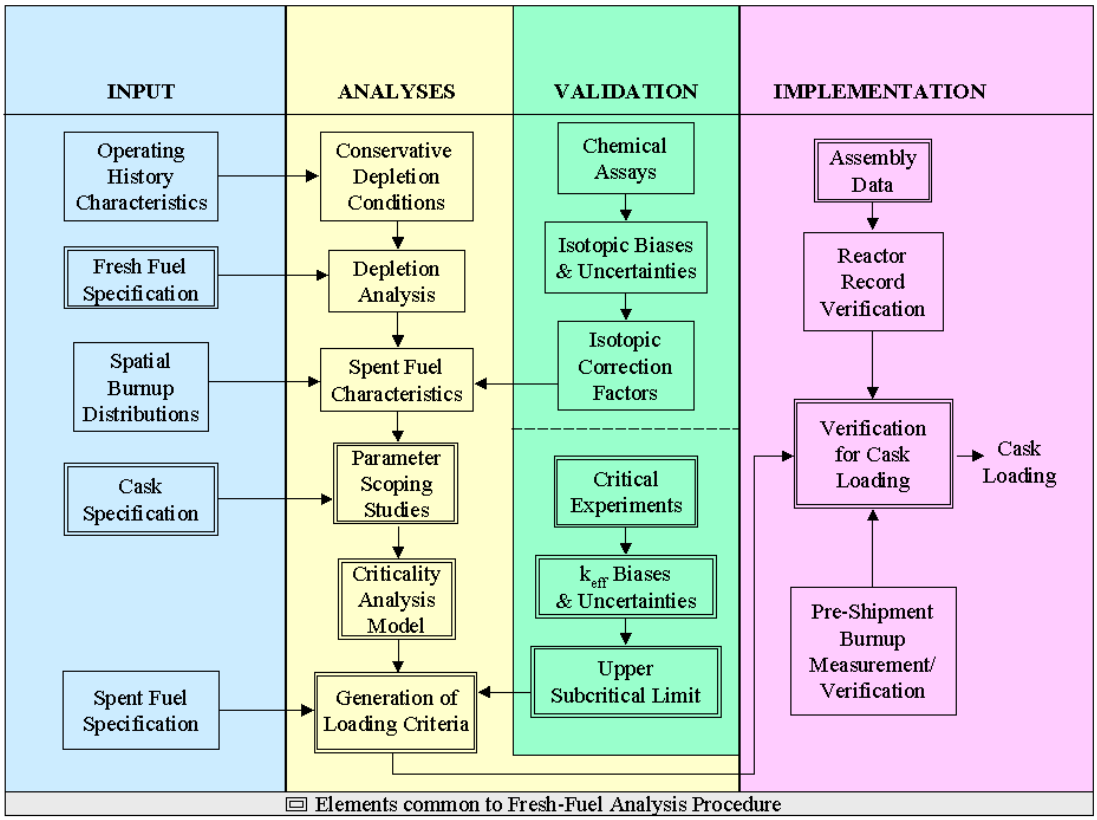

FIG. 1. Procedure for burnup credit criticality safety evaluation and implementation in transport and storage cask applications.

A number of technical issues with regard to burnup credit criticality assessments are not fully resolved, and thus, variations in submitted safety assessments, which will prolong the associated review time, should be expected. Notable among the technical issues for burnup credit implementation in transport and disposal are:

1. Selection of the appropriate reactor operating conditions that should be used in the safety analysis,
2. Selection, acquisition and use of measured data for code and data validation,

3. Clear guidance on requirements and criteria for, or possibly elimination of, pre-shipment burnup measurements, to provide a minimum impact on loading operations.

Because the inclusion of burnup credit in the criticality safety assessment for casks is a new addition to industry and NRC procedures, diligence will be required in both the preparation and review process. Ready access to the technical information of import to burnup credit and computational tools that expedite the analyses should facilitate preparation and review of SARs. A goal of current research has been to develop sound technical guidance and criteria to be considered in preparation and review of the SARs and to ensure that adequate computational tools and data are readily available.

The operational background and historical bases for safety varies between the different application areas discussed in this paper. Burnup credit is a relatively new approach being used within these various application areas and it involves a number of diverse technical topics (e.g., reactor physics and operations, criticality safety principles and analysis methods, and experiment and measurement technology). A wealth of information exists on each of these technical topics, and the key to effective implementation is successful integration of the information to develop an adequate safety basis for the application of interest. As the various approaches used by different application areas to integrate the technical information become better understood, the effort required for preparation and review of SARs for burnup credit should decrease.

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GROUP DISCUSSIONS

WET STORAGE AND DRY STORAGE

1. INTRODUCTION

This group has been assigned the task of reviewing the status of burnup credit (BUC) applications to wet and dry storage of spent fuel, the motivations for BUC implementation and the related benefits. Future plans for BUC activities in each country have also been included. Based on this review, recommendations and conclusions to improve the methodologies used for criticality safety analysis of spent fuel storage have been obtained.

BUC for wet and dry storage systems is needed in many of the member countries to allow for increased fuel initial enrichment, and to increase the storage capacity. In both cases, BUC techniques avoid the need for extensive modifications of the spent fuel management systems involved.

If dry storage takes place in casks, BUC is only needed to demonstrate a sufficient criticality safety margin for both the cask loading/unloading process and for the accidental flooding case. The same is true for dry transport. Therefore, the limiting scenario for dry transport is similar to the normal operation case of a wet storage, except for the Boron effect (if credited) in the PWR case. Hence, from the physics standpoint, wet storage and dry transport are not as different as it is commonly assumed. Except for small size transport casks, the neutron spectrum is similar in both cases.

As a consequence of the above reasoning, the methodologies applied to the criticality safety analysis of wet and dry storage systems are basically the same. However, the BUC level allowed for PWR management systems has traditionally been different. The origin of this difference lies on the presence of soluble Boron in most of the PWR wet storage systems, and on the use of the double contingency principle. Comprehensive descriptions of this issue can be found in references [1 - 3].

2. STATUS OF CURRENT IMPLEMENTATION ACTIVITIES

The current BUC applications status is shown in Table I. The BUC level implemented in each application is given in Table II. The BUC levels quoted correspond to those defined in reference [3].

As can be seen in Tables I and II, the actinide and fission product BUC level is applied to PWR wet storage systems in Brazil, Germany, Korea, Spain and United States. In the case of Germany, a very low minimum burnup is required at the present time (5 GWd/MTU at 4.4 wt.-% enrichment).

Actinide and fission product BUC level is intended to be used for PWR wet storage also in China P.R., at the Lin Ao plant (under construction), and in South Africa, at Koeberg NPP.

In addition to this level of BUC implementation, partial boron credit is allowed in Spain and in the USA, and it is foreseen to be allowed also in South Africa.

As can be seen in Table II, in all the other cases either actinide only BUC level is used, or no decision has been made at the present time.

Table I. Worldwide uses of burnup credit: national practices and status in wet and dry storage

COUNTRIES	WET STORAGE					DRY STORAGE				
	PWR	BWR	RBMK	MOX	WWER	PWR	BWR	RBMK	MOX	WWER
BELGIUM	AP [1]	-	-	-	-	-	-	-	-	-
BRAZIL	AP	-	-	-	-	-	-	-	-	-
BULGARIA	-	-	-	-	IC	-	-	-	-	IC
CHINA P. REP.	IC/U	-	-	-	-	IC	-	-	-	-
CZECH REP.	-	-	-	-	IC	-	-	-	-	UD
FRANCE	AP	-	-	UD	-	-	IC	-	IC	-
GERMANY	AP	AP [2]	-	IC	NO	UD	UD	-	IC	IC
HUNGARY	-	-	-	-	IC	-	-	-	-	IC
JAPAN	IC	IC	-	UD	-	IC	IC	-	UD	-
KOREA, REP.	AP	-	-	-	-	IC	-	-	-	-
LITHUANIA	-	-	AP [3]	-	-	-	-	IC	-	-
RUSSIAN FED.	-	-	AP [4]	-	IC	-	-	-	-	IC
SLOVAKIA	-	-	-	-	IC	-	-	-	-	IC
SOUTH AFRICA	RR	-	-	-	-	-	-	-	-	-
SPAIN	AP	AP [2]	-	-	-	IC	IC	-	-	-
SWEDEN	-	AP [2]	-	IC	-	-	-	-	-	-
SWITZERLAND	NO	AP [2]	-	NO	-	IC	IC	-	IC	-
UKRAINE	-	-	-	-	IC	-	-	-	-	IC
UK	RR	UD	-	UD	-	IC	IC	-	IC	IC
USA	AP	AP [2]	-	-	-	RR [5]	IC	-	-	-

Note: The Table contains information from countries participating in IAEA meetings on burnup credit implementation and from personal communications.

AP = Approved.
UD = Under Development.
NO = Applicable but not intended.
IC = Interest/Considering, or Applicable.
RR = Regulatory Review.
- = Not Applicable.

[1] Burnup credit has been approved on a case-by-case basis using actinides only, no fission products
[2] Credit for the presence of integral burnable absorbers.
[3] For Ignalina.
[4] For Smolenskaja.
[5] Use of burnup credit for loading single purpose PWR casks is implemented.

For wet storage of BWR fuel, burnable absorber BUC level is taken in Germany, Spain, Sweden, Switzerland and the USA.

2.1. Motivations and benefits

The motivations of using burnup credit are mostly increase of fuel enrichment and increase or maximizing of storage capacity:

- Increase of enrichment: Belgium, China P.R., Czech Republic, France (wet storage pond at Le Hague), Germany, Hungary, Lithuania, Russia, South Africa, Spain, Ukraine, and USA.

TABLE II. Worldwide uses of burnup credit: buc level allowed in wet and dry storage

COUNTRIES	WET STORAGE					DRY STORAGE				
	PWR	BWR	RBMK	MOX	WWER	PWR	BWR	RBMK	MOX	WWER
BELGIUM	A	-	-	-	-	-	-	-	-	-
BRAZIL	FP	-	-	-	-	-	-	-	-	-
BULGARIA	-	-	-	-	ND	-	-	-	-	ND
CHINA P. REP.	FP	-	-	-	-	FP	-	-	-	-
CZECH REP.	-	-	-	-	ND	-	-	-	-	ND
FRANCE	A	-	-	-	-	-	-	-	-	-
GERMANY	FP	BA	-	ND	-	A	A	-	ND	ND
HUNGARY	-	-	-	-	A	-	-	-	-	A
JAPAN	ND	ND	-	ND	-	-	-	-	-	-
KOREA, REP.	FP	-	-	-	-	-	-	-	-	-
LITHUANIA	-	-	A	-	-	-	-	A	-	-
RUSSIAN FED.	-	-	A	-	-	-	-	-	-	-
SLOVAKIA	-	-	-	-	-	-	-	-	-	-
SOUTH AFRICA	FP	-	-	-	-	-	-	-	-	-
SPAIN	FP	BA	-	-	-	ND	ND	-	-	-
SWEDEN	-	BA	-	ND	-	-	-	-	-	-
SWITZERLAND	-	BA	-	-	-	-	-	-	-	-
UKRAINE	-	-	-	-	ND	-	-	-	-	ND
UK	ND	ND	-	ND	-	-	-	-	-	-
USA	FP	BA	-	-	-	ND	ND	-	-	-

Note: The Table contains information from countries participating in IAEA meetings on burnup credit implementation and from personal communications.

A = Reduction of the actinides concentration only considered
FP = Reduction of actinides and build-up of fission products considered
BA = Credit for the presence of integral burnable absorbers
ND = Not decided

In Germany and Spain the increase of initial enrichment is implemented or is meant to be implemented in PWR as well as in BWR storage ponds.

- Increase/Maximizing of the storage capacity: Belgium, Brazil, Czech Republic, France (wet storage pond at Le Hague), Russia, South Africa, Spain (PWR and BWR wet storage ponds), Ukraine, and USA.

The benefits gotten from using burnup credit are mostly avoiding design changes at ponds and casks and hence significant cost savings.

In the People's Republic of China the Daya Bay NPP is presently operating in 12-month cycles, using fuel assemblies with an initial enrichment of 3.2 wt %. The average discharge burnup of the assemblies is about 30 GWd/MTU. In order to enhance the economics of nuclear power, 18-month cycles will soon be used with fuel assemblies of 4.45 wt % initial

enrichment. The average discharge burnup of the assemblies will then be around 45 GWd/MTU.

In order to increase the capacity of the spent fuel storage pool, a feasibility study of BUC implementation in the spent fuel storage is underway in the People's Republic of China. The spent fuel pools of the Lin Ao NPP, now in the construction phase, are designed with BUC.

In the Czech Republic the increase of the VVER 440 fuel initial enrichment from 3.6 wt % to 3.82 wt % at NPP Dukovany will force to either modify the license of the existing CASTOR-440/84 storage and transport cask or to make design modifications. It seems obvious that future submittals for approval will implement BUC. The new biding process to enlarge the storage capacity at NPP Dukovany indicates that some vendors have the intention to design large casks (more than 100 fuel assemblies) based on BUC.

In South Africa the exchange of the NPP Koeberg storage racks with Boral neutron absorbing plates is necessary due to technical problems (swelling of Boral).

Ukraine has needs for BUC implementation in dry spent fuel (SF) storage and wet SF storage. Regarding dry storage systems, VSC VVER-1000 dry storage cask design (VSC-24 originally) is under implementation at Zaporizhia NPP dry storage facility. To meet the subcriticality acceptance criterion ($k_{eff} < 0.95$) for each of the casks one can reduce the number of fuel assemblies per cask, e.g. from 24 to 10 at 4.4% initial enrichment. Hence, there is an urgent need to implement BUC for this cask model to avoid a decrease in the number of spent fuel assemblies to be loaded per cask.

Regarding wet storage, the capacity of some of the on-site storage ponds in the Ukraine is to be increased. The subcriticality acceptance criterion is met if neutron absorber integrated in the rack structure material are assumed, but for cost saving reasons it is intended to apply BUC instead of neutron absorber.

In the USA the Boraflex degradation problem has been an additional driving force to apply BUC.

3. FUTURE PLANS

3.1. Czech Republic

Regarding future cask license submittals, the vendors have indicated that dual purpose casks (storage/transport) will be BUC based. Two research programs are ongoing at NRI Rež. One is focused on BUC implementation in the SKODA cask. The other focuses on the methodology to be used for the regulatory review and licensing the storage/transport system that use BUC methodologies.

3.2. Germany

At present, BUC is approved for only one PWR plant. The value of the approved minimum burnup credit is relatively low (5 GWd/MTU). Approval of the same level of burnup credit for a second PWR plant is expected in near future.

It is planned to obtain regulatory approval for application of actinides and fission product BUC level to higher burnups levels (in conjunction with an increase of the storage capacity of wet storage ponds).

In addition to the actinide and fission product BUC level also the burnable absorber credit level is used in Germany, not only for BWR plants but also for one PWR plant due to the increase of the initial enrichment. It is expected that this burnup credit level will be replaced by boron credit in the future.

3.3. Hungary

For the near future there are the following plans:
Performance of calculations for the isotopic composition benchmark defined by the Kurchatov Institute and which at that time the only available experimental information on spent VVER fuel composition.
Investigations of the impact of the spectral effect on the burnup credit in VVER storage.
Investigations of the effect of the cooling time and the relative importance of the different uranium and plutonium isotopes.
Estimation of the conservatism of the actinide only approach due to the fission product.
Investigations of the influence of the spatial variation of the assembly burnup, in particular the axial variation (end effect).

For these purposes Hungary plans to use the MCNP Monte Carlo code and the core design code KARATE, which was developed in KFKI Atomic Energy Research Institute.

3.4. Lithuania

Implementation of the burnup measurement device, which has been developed by French specialists based on the PYTHON method, is scheduled the end of 2000. The discharge burnup and the axial burnup profile of each fuel assembly discharged from the reactor will be measured.

There is plan to ship four spent fuel assemblies with different enrichment and burnup to Russian hot cell facilities in Dimitrovgrad for post- irradiation examination. Chemical assay data will be available for the depletion calculation validation.

3.5.Spain

National regulations governing burnup credit applications are now 10 year old. A revision of the rules to include the accumulated knowledge and the progress in the burnup credit field during this period is scheduled for 2001.

Also, the next storage/transport cask generation will include BUC in the design, credit level to be yet determined. It is intended to publish transport regulation and guidance in the same rule revision mentioned above.

3.6. Sweden

Storage of spent fuel, BWR including MOX and PWR, is done in water pools. This applies to the nuclear power plants and to the intermediate spent fuel storage facility (CLAB). No need for BUC can be anticipated now for PWR fuel.

About 10 years ago a study was made for CLAB to use burnup credit. Due to that the bounding axial burnup shape for BWR fuel assemblies gave no significant benefits, it was considered not efficient from an economical standpoint to use burnup credit. Instead it was decided to use canisters made of boron steel to reduce keff. Gadolinium credit is applied at the BWR nuclear power plants and at CLAB.

3.7. Ukraine

Development of a programme for BUC implementation to promote and coordinate the activities of the bodies in this area it would be helpful for Ukraine.

4. RESEARCH & DEVELOPMENT AND OPERATIONAL NEEDS

There is common agreement among all the participants of the working group that there are lacks of data open to the public needed for validation of depletion and calculation codes as set forth below:

1. There is a lack of available chemical assay data to be taken as a basis for depletion code validation, in particular in the range of higher enrichments (> 4 wt.-%) and higher burnups (> 40 MWd/kg U). Also, the available data was obtained in most cases from fuel designs now discontinued, which were burned under core conditions rather different from those in use now. Therefore, a doubt can be put forward on how representative is this information with regard to present-day fuel.
    1.1. The VVER countries, Hungary and Czech Republic in particular, pointed out that there is almost a total lack of reliable chemical assay data. So the depletion codes cannot be validated, and it is not possible to define bounding/enveloping values for the parameters affecting burnup credit,
    1.2. China P.R. in particular demands more detailed knowledge in fission products concerning the cross-sections (e.g. Gd-155 cross-section for neutron capture) as well as fission yields/number densities. This regards the short-term scale (depletion in the core) as well as to the long-term scale (cooling time credit).
2. All the group participants have detected a similar lack of spent fuel critical/subcritical experiments, in particular in the case of VVER fuel.
3. Correct consideration of the end effect (axial burnup profile effect on the assembly's reactivity) is now obtained in most analysis by considering burnup shapes specific of the plant involved. The definition of bounding axial shapes could be of great help. Several groups are making efforts in this direction. The participants feel that the bounding axial shapes should clearly be demonstrated to be bounding for present-day fuel types and operating conditions, and a mean should provided to verify they are enveloping of an specific case.
4. Assessment of the long-term validity of the criticality safety analysis:
    4.1. It is necessary to periodically control the assumptions relevant to burnup credit made in the BUC criticality safety analysis performed for the storage facility of interest, in particular if the core operation conditions have been modified, e.g., in case of reactor power uprates or changes in the core loading strategy,
    4.2. However, for that purpose decision criteria are needed in order to be able to come to the conclusion whether or not some parts of the analysis, or even the full analysis, have to be revised,

4.3. An example could be given using again the end effect. The axial burnup shapes are affected by changes in the core reload strategy, of the plant operating conditions, or by fuel design changes. Of special importance are the axial variations of the fuel composition, like the use of axial blankets or partial length burnable absorbers. Therefore, a criterion is needed to determine whether the original estimation of the end effect remains bounding after those modifications are introduced.

The participants of the working group agreed that it is necessary to participate in international experimental programmes like REBUS (see appendix 1 and reference [4]) or PROTEUS (see country report Switzerland) for instance. The REBUS programme was taken in the group discussion as a good example for a worthwhile experiment. The advantage of the programme is that it includes both reactivity measurements and chemical and radioactive assay of actinides and fission products. The outcomes of these programme are therefore appropriate to benchmark both depletion and criticality calculation codes. In addition it seems possible to correlate the biases in the calculated isotopic densities to the bias in the calculated reactivity worth. A short description of the REBUS programme is given in appendix 1 to this group report.

Programmes like REBUS and PROTEUS are dealing with PWR and BWR UO2 and MOX fuels. Accordingly, there remain problems to be solved. Hungary for instance pointed out that VVER fuel has a harder spectrum (due to smaller V(H2O)/V(UO2) ratios within the fuel lattices. Accordingly a higher Plutonium build-up is to be expected. In addition, the end effect has been shown to be stronger for the follower fuel assemblies (see country report Hungary). However, the enrichment is in general lower for those followers, so BUC might not really needed for them. There remains an open question about how to consider the end-effect in the analysis: by penalising the neutron multiplication, by using the equivalent uniform burnup, or bounding burnup profiles. In the latter case, the problem is how to get bounding shapes.

There are also special problems due to specific applications as set forth below:

Lithuania reported that the use of Erbium for reactivity control results in a change of the neutron spectrum in the core. There are problems in re-calculating the fuel operation histories now due to the use of Erbium. Therefore, there are problems in defining the bounding irradiation history (that one which leads under storage conditions to the bounding reactivity). Work to overcome the problem is underway.

South Africa reports that Koeberg NPP intends to use actinides and fission products BUC level, and partial boron credit. However, release of slugs of unborated water from the spent fuel pool cooling return line at the bottom of the wet storage pond cannot be excluded under certain conditions. From a regulatory perspective, the only possibility to accept partial boron credit is to reduce the probability of occurrence of the dilution event (the release of the slugs) by technical/administrative measures in order for it to be declared incredible (for more details see country report South Africa).

4.1. Conclusions and recommendations

4.1.1. Regulatory considerations

Present items of interest and regulatory concerns have been resumed in the presentation of J.M. Conde [5], and are not repeated here. The South African regulatory body NNR pointed out in addition the following issues:

Adequate justification for the use of the different actinides and fission products with respect to cross sections and densities of the nuclides involved.

Typically the confidence levels in terms of accuracy of concentration densities as calculated by depletion codes, the confidence in microscopic cross-section data, benchmarking with experimental tests, etc. were investigated.

The bases for the assignment of uncertainties of k_{eff} values have to be well understood.

The significance of the axial shapes and the end effects has to be well understood. This is especially true in the case where localised criticality effects, such as unborated slugs of water – if partial boron credit is taken - could be credible. It is important to have assurances that the combination of axial shapes and isotopic data sets used in these shapes are bounding. A bounding shape without conservatively depleted isotopic sets could lead to non-conservative k_{eff} values. In the same light a conservative isotopic set with a non-bounding axial shape could also lead to an underestimation of k_{eff}.

The modelling of a spent fuel pool system with radial and axial leakage is important to investigate properly local effects such as the case of unborated slugs.

As pointed out by the Swedish participant, whether or not BUC is used in a country, from a regulative perspective it would be interesting to make calculations that take credit for burnup in order to get realistic estimates of the real safety margins. Such calculations are also of value to adequate assessments of postulated incidents and accidents and hence emergency preparedness.

4.1.2. Training course on BUC implementation

For those member states that are going to consider BUC implementation in the spent fuel management (SMF) systems in the near future, a transfer of knowledge and expertise from those states having their own experience in this area has a high importance. Such a transfer may be efficiently organized by the IAEA in the form of a Training Course.

The objectives of a training course should lead to facilitate agreement among regulators, analysts and plant operators on what are - from the view of physics - the basic requirements in using burnup credit. So it should be stressed out:

1. What are the relevant parameters affecting burnup credit,
2. Which are the methods to come to bounding results,
3. What is the difference between bounding and conservative,
4. Where is need to use bounding/conservative methods or estimates and where it is allowed to be as close to reality as possible (best estimate),
5. What are the existing regulatory systems and requirements for BUC implementation (experience with these systems.),
6. What are the possibilities of BUC measurements and their limitations.

In addition there is need to give guidance concerning operational procedures connected with the BUC implementation. It has to be worked out how periodical controls of the loading criteria can be performed. In recent years a lot of changes in core fuel management have taken place (increase in power, changes in reload techniques and cycle lengths) which affect the

burnup credit criticality safety analyses of SMF systems involved. So therefore, one needs decision criteria in order to be able to come to a decision whether or not such a burnup credit safety criticality analysis or parts of this analysis have to be revised.

4.1.3. Participation in international experimental programmes

The participants of the working group recommend participation in the international experimental programmes (like REBUS, PROTEUS, and further programmes) to obtain a database for burn-up credit validation of depletion and criticality codes that is as accurate and extensive as possible.

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[5] CONDE, José M., ”Regulatory Aspects of Burnup Credit Implementation”, presented on the TCM on ”Evaluation and Review of the Implementation of Burnup Credit in Spent Fuel Management Systems, July 10-14 (2000) Vienna.

APPENDIX 1: BRIEF DESCRIPTION OF THE REBUS PROGRAM

1. THE REBUS EXPERIMENTAL BURNUP CREDIT PROGRAM

There is a common agreement among experts that there is a lack of experimental data with respect to chemical assay data and reactivity measurements for validating burn-up credit calculations. An international programme called REBUS (REactivity tests for a direct evaluation of the Burn-Up credit on Selected irradiated LWR fuel bundles) for the investigation of the burn-up credit has been initiated by the Belgian Nuclear Research Center SCK•CEN and Belgonucléaire. At present it is sponsored by USNRC, EdF from France and VGB, representing German nuclear utilities. The programme aims to establish a benchmark for both depletion and criticality codes. This benchmark would qualify the codes to perform calculations of the burn-up credit, taking into account depletion of fissile isotopes and production of fission products.

The present proposed programme will investigate the following fuel types with associated burn-up:

1. Reference absorber test bundle, containing rods of borated glass,
2. Fresh commercial PWR UO_2 fuel, constructed by SIEMENS,
3. Irradiated commercial PWR UO_2 fuel (55 GWd/tM), originating from Neckarwestheim NPP, Germany,
4. Fresh PWR UO_2 fuel, original BR3 fuel,
5. Irradiated PWR UO_2 fuel (30 GWd/tM), BR3 reactor, Belgium.

The bundles will contain approximately 25 rods of 1 meter length in a 5x5 array. Reactivity effects will be measured in the critical facility VENUS by means of measuring differences in critical water level of configurations with fresh and spent fuel test assemblies. The measured effects will be typically in the range of 2000 pcm, which is significantly larger than the uncertainty of criticality calculations (300-500 pcm). The measurement uncertainty of the reactivity difference will be typically around 20-40 pcm, the measurement uncertainty of a critical water level is 10-15 pcm. The accumulated burn-up of all rods will be measured non-destructively by gamma-spectrometry. Some rods will be analyzed destructively by radiochemical assay with respect to accumulated burn-up (Nd, 137Cs), actinides content and TOP-19 fission products (i.e. those non-volatile fission products that have most implications on the reactivity). The used techniques are TIMS, ICP-MS and as a check alpha-spectrometry (238Pu, Cm) and gamma-spectrometry (Np, 241Am, Cm, 129I). The measurement uncertainties (referring to two standard deviations of the mean) of the radiochemical assays are listed in the next tables.

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%				

Uncertainties by radiochemical analysis for the different burn-up 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%

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%		

The experimental implementation of the programme will start in 2000.

The benefits of the programme for validation of depletion and criticality codes are its unique combination of criticality measurements and radiochemical assay data. The criticality measurements are easy to model (well-qualified fuel, clean and homogeneous grid) for validation of criticality calculations. The radiochemical assay data are very complete and accurate for use of depletion code validation.

This data may also be advantageous for burn-up credit applications to dry storage. An additional programme especially for dry storage application is possible.

The present programme is limited to PWR fuel with two different burn-ups. Other fuel types can be investigated in future extensions of the programme and are negotiated for, like BWR fuel and MOX fuel. Other, more future extensions can be criticality measurements on dry bundles (for dry storage applications) and degraded bundles (for final disposal of spent fuel).

TRANSPORT WORKING GROUP

1. INTRODUCTION

The application of burnup credit criticality to transportation cannot be treated in isolation. Transportation is used to facilitate off-site storage (wet or dry interim storage facilities), reprocessing and disposal operations. A systematic technical interface with these end-user applications is essential to the safe and efficient use of burnup credit for transportation. Additionally, transportation of spent fuel can cross international boundaries and therefore requires international coordination of acceptance requirements and safety standards.

2. SPECIFIC APPLICATIONS

The use of burnup credit in the criticality safety analysis of transportation and loading/unloading operations is based on a fundamental set of principles. These principles are based on 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 type and physical characteristics of the transportation cask/flask as well as consideration of operator mistakes and incidents.

2.1. Fuel Types

Existing fuel types that should be evaluated for the potential benefit of applying burnup credit in transportation have been divided into three groups.

High category:

1. PWR - UO2 then MOX,
2. VVER,
3. BWR - less interest than PWR, UO2 then MOX.

Intermediate category:

1. Gas-cooled (UK),
2. RBMK – enrichments are going higher (about 2.6 wt%); however felt to not be that important for transport, perhaps for storage.

Low interest currently (generally not significant contributors to the total MTU of fuel to be transported):

1. Research reactor (typically >20 wt%),
2. Fast reactors, including breeder reactors (France, Russia, Japan),
3. Pebble-bed,
4. PHWR (China has indicated interest in long term as new reactors come on line),
5. Advanced CANDU – trend toward slight fuel enrichment reduces positive cooling void coefficient.

2.2. Cask Types/Characteristics

Existing vs. new design came out as one key issue of importance to BUC strategy.

Recertifying existing casks with burnup credit can permit the newer higher enriched fuels to be transported – current casks could continue to be utilized, particularly important for small casks.

New cask designs that are developed using burnup credit would be larger capacity requiring fewer casks to be manufactured.

The size of the cask (i.e., small capacity vs. large capacity) may impact the sensitivities to various physics parameters and therefore the implementation of BUC. The difference between 'small' and 'large' can be defined using guidelines from fuel management system (8 assemblies are generally the most would expect to have similar worst case characteristics). The size of large casks is further limited by the capacity of the facility fuel handling system.

Other cask characteristics that could potentially impact the details of how burnup credit is analyzed and implemented include:

1. Dual purpose (S/T), multi-purpose (S/T/D) [versus single-purpose casks] require coordination requirements for fuel acceptance and qualification,
2. Wet/dry shipment, not a significant difference because the regulatory assumption for dry shipments generally include the assumption of flooding accident [casks using moderator exclusion may implement burnup credit differently],
3. On-site transfers (determined on-site transfers is not a transportation issue),
4. Domestic vs. international transportation require coordination among different licensing authorities,
5. Transport to reprocessing facilities (generally involve fuels with shorter cooling times),
6. Poison control system - use of distributed poisons internal to cask structure and the use of removeable, rodded poisons placed within fuel assemblies have a need for evaluating accident scenarios that can potentially disturb the relative positions of the fuel and poison controls.

3. MOTIVATION AND BENEFITS

The motivation and benefit of burnup credit for the transportation of spent nuclear fuel is generally the same for all applications cited above.

The principle benefit is the potential to design more cost-effective transportation casks without measurably impacting the probability of a criticality incident. The primary motivations include increasing initial enrichments exceeding the values that can be accommodated in casks designed under the very conservative fresh fuel assumption and also the large inventories of spent fuel to be transported to a geologic waste repository for final disposal in the U.S.

The environmental impact of burn up credit should also be borne in mind. The waste arising due to package flushing and cleaning are directly linked to the number of shipments. Thus reducing the number of shipments by introducing burnup credit will reduce the production of

liquid wastes, which has been a major issue in some parts of the world. In other cases there will be a significant reduction in the solid waste due to a reduction in the number of packages used, or a reduction in the amount of package furniture used. Overall burnup credit in transport will be likely to result in at least a moderate environmental benefit in areas of importance - with a potential for significant benefit.

Other benefits and motivations that were discussed were related to the outcome of fewer operations (fewer shipments). These included the following:

1. Decrease the risk to public - difficult to assess quantitatively, arguments have been put forth that a decrease in the number of shipments presents no change in radiological risk due to the potential for contamination is based on the same quantity of spent fuel regardless of burnup credit; a decrease in non-radiological risk from fewer shipments can be argued on an integral basis – the risk of a traffic/rail accident per shipment may be constant, but the integral over a number of shipments or the per MTU risk may decrease,
2. Decrease in public exposure/shipment – due to normal transportation, cask dose rates are the same as for non-burnup credit (again, difficult to assess quantitatively),
3. Decrease in worker exposure/shipment - decided there was reduction due to fewer cask loading operations, dose/assembly is not reduced,
4. Reduce regulatory burden - reduce over regulation (neutral impact to safety), concluded that this U.S. concept largely captured by other items on list,
5. Reduce of future volume/mass of radioactive waste:
    5.1. continue to utilize existing casks, and permits larger capacity new casks to be designed so that fewer casks will need to be produced to transport existing and future spent fuel inventories,
    5.2. operations associated with fuel loading are reduced, radioactive wastes associated with those activities (decontamination, etc.) will be reduced,
6. Improve emergency preparedness - BUC allows better assessment/response (recovery plan) to potential accidents because it provides tools and ready information for best estimate analysis [this benefit concerns beyond design basis accidents],
7. Reduced environmental impact (e.g., cleaning casks)- similar to waste reduction,
8. Harmonization - long-term international uniformity/consistency, this is generally a benefit, not a motivation.

4. BUC STRATEGIES/CURRENT ACTIVITIES

There are several approaches to evaluating and licensing burnup credit in use or under consideration. In general, burnup credit is considered as anything beyond the fresh fuel assumption. With this definition, there clearly are a number of different levels of burnup credit in practice and being pursued from simple to more complex. There are five primary strategies that are considered with multiple levels within each depending on the specific nuclides utilized in the analysis. These primary strategies are described below with annotations to clarify the general characteristics of the strategy and to identify those currently being used:

1. Limiting burnup “debit” - in practice, quantified by taking credit for burnable absorbers in fuels,
2. Net fissile (U-235 + Pu fissile) - in practice, slight modifications in how net fissile content is determined, no credit for actinide absorbers,

3. Actinides Only (AO)– in practice, different sets of actinides are selected depending on purpose for fuel transfer, some actinide only approaches take advantage of the fission product margin,
4. Actinides + Fission Products (AFP)– there are various levels correlate to the specific actinides and fission products that are considered. It is expected that nuclides will be added as users gain experience and new data become available,
5. Best estimate – needed to accurately quantify margins

Simpler (less credit) strategies may provide benefit to provide experience for regulator and applicant alike. A key benefit of the net fissile and actinide only strategies should be to generate such experience and increase confidence.

There is some controversy on whether the first strategy listed above is a true burnup credit strategy. The confusion is linked to the idea that credit for the presence of burnable absorbers (i.e., gadolinium) in the fuel is not the same as "burnup credit". The Scope and Objectives for the OECD/NEA Expert Group on Burnup Credit Criticality Safety contains a footnote defining burnup credit: "Burn-up credit is a term that applies to the reduction in reactivity of burned nuclear fuel due to the change in composition during irradiation". This definition does no explicitly cover Gd credit. However, in the group discussions, burnup credit was considered anything in excess of the fresh fuel assumption. The issue extends past that of definition and into how the transport regulations would be applied to strategy 1.

Table I. Worldwide uses of burnup credit for spent fuel transportation: national practices and status (as of 14 July 2000).

COUNTRIES	**TRANSPORTATION[1]**							
	WET				DRY			
	PWR	BWR	MOX	WWER	PWR	BWR	MOX	WWER
BELGIUM	AP[2]							
BULGARIA	-	-	-	IC	-	-	-	IC
CZECH REP.	-	-	-	-	-	-	-	IC
FRANCE	AP	IC	IC	IC	AP	IC	IC	IC
GERMANY	-	-	-	-	AP	IC	IC	IC
HUNGARY	-	-	-	-	-	-	-	IC
JAPAN	UD	UD	-	-	-	-	-	-
KOREA, REP. Of	IC	-	-	-	IC	-	-	-
LITHUANIA	-	-	-	-	-	-	-	-
RUSSIAN FED.	-	-	-	AP[3]	-	-	-	IC
SLOVAKIA	-	-	-	IC	-	-	-	IC
SPAIN	-	-	-	-	IC	IC	-	-
SWEDEN	-	-	-	-	IC	IC	-	-
SWITZERLAND	-	-	-	-	AP[4]	IC	IC	-
UK	RR	UD	UD	-	IC	IC	IC	IC
USA	-	-	-	-	RR	UD	-	-

Note: The Table contains information from countries participating in IAEA meetings on burnup credit implementation and from personal communications.

AP=Approved. IC=Interest/Considering, or Applicable. NO=Applicable but not intended.
RR = Regulatory Review. UD = Under Development. - = Not Applicable.

[1] Wet/Dry refers to the assumed condition of the package as transported. Regulatory assumptions for the safety analysis are typically performed wet.
[2] Burnup credit has been approved on a case-by-case basis using actinides only, no fission products.
[3] For Kola.
[4] Approved for one case in connection with reprocessing in foreign plants.

In the new international transport regulations, ST-1 from 1996, irradiated nuclear fuel is treated in paragraph 674. This paragraph states that the safety assessments shall be based on isotopic compositions demonstrated to provide either (a) "the maximum neutron multiplication factor during the irradiation history" or (b) "a conservative estimate of the neutron multiplication for the package assessments". Under (b) you also find the requirement that "a measurement after irradiation but prior to shipment shall be made".

Gadolinium (Gd) credit without burnup credit is usually based on subparagraph (a) using the maximum neutron multiplication factor during the irradiation history. If a measurement were needed, it would be to confirm the presence of the BA rods and not of the burnup. Burnup credit as in the OECD/NEA Expert Group definition is related to subparagraph (b). If both burnup credit and Gd credit are used, it may be necessary to confirm both the presence of Gd rods and of the burnup.

Gadolinium credit is licensed for fresh fuel transport and storage in Sweden. In the Table 1, there is no mention of any country that uses Gd credit (or any other burnable absorber/poison) for irradiated fuel.

As the group discussed strategies 3, 4 and 5 there was concensus that the definition of "Actinide Only" and "Actinide + Fission Products" should not include any specification of nuclides. In fact, there is an opportunity for many levels of burnup credit within these strategies depending on the variation in the number of and how the nuclides considered. For example, fuel cooling time is relevant to the selection of nuclides – the USA application for burnup credit takes Pu-241 at a specific cooling time and Am-241 from decay only and does not consider the component produced in reactor. Whereas, France takes Pu-241 at discharge and Am-241 from reactor inventory and neglects the Am-241 produced from decay. The difference in approach is related to the purpose of the shipments and the existing conditions. In the case of France, shipments are predominantly for short cooled fuel destined for reprocessing facilities. A fleet of casks is already hard at work for this purpose. However, the capacities of these existing casks are threatened by increasing initial enrichments. The burnup credit sought in this situation is just the amount needed to compensate for the increase in enrichment. In the USA, the goal is to develop a fleet of casks with maximum capacity ratings that will accommodate current SNF inventories and those expected in the future. Since many of the USA's reactors have been operating for more than 30 years, much of the SNF inventories have seen significant cooling times. Clearly the strategies 3, 4, and 5 must be flexible and permit a graded approach to burnup credit consistent with the application.

In 1997 the IAEA organized an advisory group meeting (AGM) to examine and report on the status of burnup credit for transport, storage, reprocessing and disposal of PWR, BWR, VVER and MOX spent fuel. The AGM compiled information relating the regulatory status of burnup credit in different countries as of December of 1997. Table 1 is an update of the information related to transportation.

For the most part, the group felt that the presentations in the general sessions of the TCM had been thorough in reviewing the status of current burnup credit activities in their countries. In the interest of time, there was not much additional discussion in the group session. However, some group members did submit written comments for inclusion in the report. These comments are given below to provide a succinct summary of activities. Details of the current burnup credit activities may be found in the country reports included in the proceedings of this TCM.

United States: During 1999, the U.S. Nuclear Regulatory Commission (NRC) introduced technical guidance for allowing burnup credit in the criticality safety analysis of casks for transporting spent fuel from pressurized water reactors. The NRC is conducting a research program to help develop the technical information needed for refining and expanding the evolving guidance. Cask vendors have announced plans to submit the first NRC license applications for burnup credit late in 2000.

4.1. Future plans (for those using or pursuing use)

Philosophically, the future plans are to increase the degree of detail in burnup credit modeling as (1) users gain experience, (2) there is a requirement for additional margin (e.g., increasing fuel enrichments), and/or (3) additional experimental benchmark data become available.

The group felt that the presentations in the general sessions of the TCM had been thorough in reviewing the future plans for burnup credit activities in their countries. In the interest of time, there was not much additional discussion in the group session. We refer you to the proceedings of the TCM for the details.

4.2. Research & Development and operational needs

Table II. is a summary of needs and issues requiring additional investigation as discussed by the group.

The concensus of the group was that the prioritization of needs was driven by the graded approach to the advantage of burnup credit being pursued.

The different levels of advantage (implementation) of burnup credit were as discussed previously:

1. Limited Burnup Debit (sometimes referred to as Gd credit, see earlier discussion),
2. Net fissile Depletion,
3. Actinide Only (AO),
4. Actinide + FPs (AFP),
5. Best Estimate.

The transition from one level to another is accomplished by adding nuclides therefore a continuum from 2) to 5) is possible with the general categories of actinide only (AO) and actinide plus fission products (AFP) being preserved.

The discussion of needed information for adding nuclides was also according to the degree of burnup credit. Generally, it was felt that the areas of chemical stability (assuring the isotope remains in the fuel), chemical assay data (assuring the quantity of the isotope in the fuel), reactivity benchmarking data, i.e., worth measurements and the quality the nuclear data (assuring the effect of the isotope on reactivity) should be addressed in making such determinations.

Nuclear data: co-variance data, uncertainty of nuclear cross section data, fission yield data, decay data. The suggestion to consult with the OECD Working Party on Nuclear Data Evaluation to confirm the general reliability of data important to the calculation of burnup credit was offered. It is recommended that this issue be formally pursued by the IAEA in a manner such that a written response is provided which may be circulated to interested parties.

Table II. Summary of Topics Discussed as R&D and Operation Needs

Depletion Analyses	k_{eff} Calculations	CASK OPERATIONS
Validation:	***Validation:***	- Tie between international regulators
- PIE data	- spent fuel criticals; (overall validation needs based on degree of BUC sought)	- shipment-dependent analyses
- private data	- existing experiments with more information than now	- multiple loading curves (one for each set of parameters)
- reactor criticals, how to use	- private data	***Measurement:***
Modeling:	- reactor criticals (integral experiment data)	- verification requirement
- use of solid absorbers	- differential (nuclide specific) cross section measurements (reactivity worth)	- "measurement" needs to be defined
- control rods	***Modeling:***	- Japan proposing that for some parameters reactor records are measurements and additional measurements at loading are not necessary
- depletion parameters	- axial distribution of burnup	***Reactor Records:***
- improved geometrical modeling, particularly for BWR (improved from point depletion)	- adequacy of point depletion being studied, e.g., Japan, MCNP-BURN (integrates depletion and criticality in one calculation)	- validate reactor records
- temperature distribution	- source convergence (OECD)	- Japan proposing that for some parameters reactor records are measurements and additional measurements at loading are not necessary
PARAMETER LIST:	- absorber materials (particle heterogeneity), not really a BUC issue	
Specific Power	- absorber distribution in basket/cask	
Operating History	- improved modeling for BUC needs (add end fittings to model, etc.)	
Fuel Temperature Distribution	- confirm (qualify or quantify) representativeness of benchmarks and modeling assumptions	
Moderator Temperature Distribution		
Changes in Soluble Boron (i.e., Boron Letdown)		
Location of Burnable Poison Rods (BPR)		
Integral Burnable Poisons (BPA)		
History of Control Rod movement		
Void history and distribution (BWR)		
Fuel Geometry (design)		
Axial blankets		
Assembly Geometry		
Initial Enrichment		
Burnup		

Chemical stability data - felt to be largely known although some debate on cesium solubility.

Benchmarking data:

Critical experiments/spent fuel:
PIE data:

1. Adequate for approved concepts, net fissile and actinide only?
2. What is needed to justify the addition of a fission product nuclide?
3. What is needed to justify the addition of an actinide absorber?

4. What is needed to provide a conservative representation of the net fissile content of fuel?

Reactivity Worth experiments:

1. Lumped source (e.g., actual spent fuel samples including UO2 and MOX fuel types),
2. Specific isotopes important to burnup credit,
3. Specific elements important to burnup credit.

Modeling Assumptions:

1. Depletion Analysis,
2. Calculation of the system k-effective.

Loading measurements:

1. Are measurements at cask loading needed?
2. Purpose, type, strategy?
3. Verifying reactor records?
4. Are reactor records adequate for determining initial enrichment, burnup and cooling (demonstrating compliance with the loading curve)?

Cask Operations/Implementation:

1. Ties between international regulations,
2. Reactor based licensing,
3. Shipment-dependent analyses,
4. Multiple loading curves,
5. Verification requirements:
   5.1. validate reactor records,
   5.2. physical "measurement".

4.3. Regulatory considerations:

1. Confidence in analysis approaches,
2. Safety margin is a major issue. The margin of safety should be justified and consistent with the approval sought,
3. Physical measurements:
   3.1. review type (cask-specific analysis vs. operation-specific analysis),
   3.2. approve methodology,
   3.3. license amendment process: adding fuel types, revised certificate (new calcs with different assumptions) or modification using same methods,
   3.4. generic analyses can be deficient because of poor availability of fuel information.

4.4. What the regulator looks for?

1. Compliance - confidence in the analysis, safety practice (safety margin), measurement,
2. Training,
3. Use of best practice guides such as ST-2,

4. reasonable assurance - qualifications/applicant experience, does the applicant have suitably qualified and experienced personnel (SQEP). Are criticality safety people with reactor operation experience/knowledge involved,
5. detailed computer inputs - sufficient detail to do independent calculations by regulator,
6. Understanding of correlation of effects, particularly if safety margin is "low". For example France has stepped approach. Want to look at and understand approximations well. Problem can arise perhaps if building one approximation upon another without proper understanding,
7. Mixed loading effects - leakage/spectra alter appropriateness of loading curve (mixing of assemblies with different profiles). Missing pins can be an issue particularly for MOX,
8. Safety margin - how quantified/qualified and justified, UK WP looking at this issue. May be reasonable to envision decrease in acceptable margin as understanding/experience is gained. Statistical/experience to support assumptions for "operating" conditions,
9. Horizontal/axial BU effects,
10. Accident effects to intact fuel, e.g., bowing at ends due to drops (claims potential concern may be with bowing near ends - not important in fresh, but COULD be important for BUC),
11. Off-normal conditions must include phenomena that can impact fuel geometry, evaluate impact on k (spectral effects). More important for BUC because of importance of ends (vs. middle for fresh),
12. Bending of fuel under normal irradiation conditions may also be a concern for BUC, again because of interest in better understanding actual fuel geometry effects,
13. Effects of preferential loading for shielding and thermal considerations. This may put problem assemblies together,
14. Cooling time effects and associated uncertainties of added end effects (and associated conservative profiles) and decay data,
15. Credit for existing measurements.

4.5. Physical Measurements – Regulatory Drivers

What to measure and why?

Demonstrate within specifications

Misloading - defined as improper placement of assembly with wrong ID. Thus, one could misload and still be above loading curve BECAUSE measurement may show "wrong" assembly OK. Procedures to assure assembly with proper ID have been in place for fresh casks.

Measurement verifies burnup record. The amount/kind of measurement depend on what is important to assessment. Important to confirm with measurement - cooling time, burnup.

Some guidance (general) within draft advisory material (ST-2) for international transport regulations (ST-1).

UK typically approves loading procedure to confirm initial enrichment used in fresh fuel assessment. Therefore, do not see a need to confirm initial enrichment - have experience with this from fresh fuel assumption.

US approach has largely looked to assure consistency of reactor records and confirm their adequacy in an integral sense. Initial enrichment, cooling time, and burnup from the record assured by the measurement to be consistent.

Input for measurement - include uncertainties as part of measurement uncertainty

Enrichment - Cs-137 has no tie to enrichment, burnup, cooling time – Cs-134/Cs-137 ratio.

Fuel manufacturing tolerance for enrichment is within +/-.05. Record value for enrichment felt to be good.

Administrative techniques including: visual checks, enrichment, assembly type.

Measured to confirm - *b*urnup is within specification.

Measurement technique needs to be consistent with safety assessment. More complex and/or simultaneous use of different types of measurements may be needed as degree of BUC increases. Little BUC, simple measurement. Lots of BUC, more scrutiny on type of measurement. In all cases the uncertainties of the "assumed" record info (input to measurement) needs to be considered.

Discussion on need to measure axial shape of burnup. ST-1 regs and ST-2 guidance indicate that axial measurements (scans) may be appropriate to assure absorbers counted on for crit safety are present. Graded approach: if use bounding profile agreed to within licensing process may need no measurement; if "typical" profile needed, then need for measurement more dramatic. ST-1 wants assurance of the poison distribution - bounding axial profile may be adequate.

4.6. Reactivity measurements

In France, work has been done to perform subcritical measurements in one configuration and using transfer functions relate that measurement to reactivity for a cask. Reactivity measurements are generally more complex, have potential for more error (uncertainty), and more intrusive to operations. However, measurements of this type do provide something closer to the "bottom line" in terms of criticality safety, i.e., relative closeness of system to critical.

Ability, complexity, and accuracy to measure are likely to vary with configuration that is measured: actual cask, individual assembly, or canister (multi-element bottle) that fits inside cask.

4.7. Conclusions and recommendations

Significant data needs include:

1. Nuclear data – cross-section covariance data and perhaps minor fission products and actinide cross sections,
2. Post irradiation examination (PIE) data for VVER, PWR, and BWR (even to extend enrichment/burnup range for AO) fuels,
3. Experimental criticality benchmark data supporting extensions to the range of applicable fuel enrichments, burnups and nuclides used in modeling.

The discussion group felt strongly that the most efficient path forward for obtaining these data are is the continued support of experiments on international collaborative basis; e.g., NERI, REBUS, PROTEUS. Fission product worth experiments such as those performed and being performed at the IPSN and Argonne, and those from the international CERES program were also noted as providing valuable information and encouraged to be continued on an international collaborative basis.

Comprehensive benchmark databases such as that compiled as a result of the ongoing activities of the OECD/NEA coordinated International Criticality Safety Benchmark Experiment Project (ICSBEP) should continue to be supported by the international criticality safety community. Also, the Spent Fuel Composition Database (SFCOMPO) compiled by JAERI continues to be under development and data (including axially distributed burnups) are being added. The data in SFCOMPO include contributions from a number of countries. A new release is planned for future but no date is available. International participation and contributions to SFCOMPO should be encouraged.

Identify sources of private data.

Trends for impact of operating data largely understood. Values selected for assessments should be consistent with utility/reactor practice

Sources of measurement uncertainties from input value should be included in the final measurement uncertainty. Objection to this - final point is that uncertainties from information used in measurement need to be considered in determination of final uncertainty of measurement.

As you reduce conservatism and move to best estimate, various items (deemed insignificant relative to conservatism of FF assumption) may warrant further study to obtain better understanding as to their importance in BUC safety case.

Group concluded that there is no change in safety in going from FF to BUC. The consequence of a criticality does not change, and the categorization (e.g., extremely unlikely, etc.) of a criticality incident does not change because proper steps are included to mitigate probability (risks). As experience is gained the steps may be reduced.

Worker dose and environmental impact can be reduced by use of BUC. Less shipments reduces dose. Reduction in environmental impact due to cask decontamination and less cask material is feasible.

Training is important. ST-2 provides some general guidance for burnup credit transportation. Individual countries and the OECD/NEA Expert Group on Burnup Credit Criticality have provided numerous technical documents that provide a basis for “good practices”. Burnup credit does increase complexity of model assumptions and (at least initially due to measurement) operations. Thus, applicants and regulators need to be properly trained in the issues, the process and elements of importance, and international similarities/differences/regulations. It should also be noted that with burnup credit, more reliance may be placed on administrative controls such as verifying reactor records. A lesson-learned from the Tokaimura incident is that administrative controls are only credible if they are taken seriously. Operator training to understand the significance of administrative controls is an important issue. The future development of international standards is perhaps an area that might be of value.

Quality control of reactor records (FF to loading). The future development of international standards in this area may also be of value.

5. NOTES

"Knowing the real safety margin - Perceived safety margins or risks are used to explain the safety of real operations to management, the public and others. They are also used to prioritize safety controls by authorities and by safety departments. Without burnup credit, the real safety margins cannot be estimated. In the past, crude and non-validated burnup credit methods have been applied informally and non-conservatively to assess the real safety margins. General availability of validated burnup credit methods will reduce the problem with incorrect, informal estimations of the real safety margin or risk."

"Incredibility", last page. I heard Don mentioning this but did not hear what it was all about. It is not wise to refer to something possible as incredible. Even less wise is to require that something possible must be made incredible. Don's remark seemed to indicate that the public in the U.S. requires transport criticality accidents to be incredible so that is what the public will hear from NRC and the industry. I can accept "very small probability" or similar expressions but not incredibility. After the Tokaimura accident, even large parts of the public know that incredible events are not uncommon.

APPLICATION OF BURNUP CREDIT TO REPROCESSING

1. INTRODUCTION

Application of burn-up credit criticality to fuel reprocessing cannot be treated in isolation. Reprocessing is tightly linked to transportation and wet storage applications. Consequently the implementation of Burn-up credit will have to be consistent with these two other application fields.

2. SPECIFIC APPLICATIONS

The criticality-safety analysis during transportation and unloading at the reprocessing facilities is based on the same principle. Therefore this specific aspect is treated in the group dealing with transportation.

The use of burn-up credit as applied to reprocessing can be of great benefit in the following aspects:

1. Wet storage,
2. Dissolution,
3. Tank design,
4. Waste conditioning and storage.

3. MOTIVATION AND BENEFITS

For the previously mentioned fields of application the burn-up credit aims at:

1. Extending the reprocessing in existing facilities to a wider range of fuel assemblies characteristics,
2. Optimizing the design of new facilities,
3. Cost savings.

3.1. Dissolution

The burn-up credit extends the upper mass limit in the dissolver or avoids the use of neutron absorber such as gadolinium.

Due to the burn-up credit the same capacity of reprocessing can be maintained in spite of the increase of initial enrichment of fuel assemblies.

3.2. Tank design

The burn-up credit allows:
1. Higher concentration limits,
2. Less penalizing geometries.

3.3. Waste conditioning and storage

The burn-up credit reduces the criticality constraints in terms of mass limits and geometrical configurations.

4. STATUS OF CURRENT ACTIVITIES

Modifications and remarks concerning activities on actinides-only burn-up credit in France.

4.1. Wet storage

PWR: AP for all types of lattice.

MOX: UD.

4.2. Transport

MOX: UD.

4.3. Reprocessing

BWR:

1. AP for specific types of fuel design,
2. RR for the other designs.

In France:

1. A working group addresses the fission products burn-up credit,
2. Emphasis is placed today on experiments and code validation.

5. FUTURE PLANS

In France, the main fission products will start to be taken into account as of 2003 for transport, wet storage and reprocessing. In the end the same approach will be applied to UOX and MOX fuels.

6. RESEARCH AND DEVELOPMENT AND OPERATIONAL NEEDS

In France the research and development program concerning the main fission products contains the following aspects:

1. Measurements,
    - 1.1. Development of a new burn-up measurement system for MOX fuels,
    - 1.2. Irradiated fuel samples assay,
    - 1.3. Oscillations of isolated fission products and irradiated fuel samples in MINERVE Reactor at CEA Cadarache,
    - 1.4. Critical experiments at IPSN VALDUC site.
2. Codes validation,
    - 2.1. Qualification of depletion codes (APOLLO - DARWIN and CESAR),
    - 2.2. Qualification of the CRISTAL French criticality package (TRIPOLI, APOLLO, MORET).

7. REGULATORY CONSIDERATION

In France, there is no specific regulation relative to burn-up credit, however there is a practice, which has been approved and implemented at the end of the 1980s.

8. CONCLUSIONS AND RECOMMENDATIONS

A large experience has been achieved in France concerning actinides-only burn-up credit. The methodology adopted was proved to be reliable over the last ten years, it has been successfully implemented and no problems were encountered.

In general when moving beyond actinides-only, a detailed analysis has to be introduced. Today there is a global concern about validation of fission products data for criticality applications. Studies are in progress to identify parameters that influence the multiplication factor and how they affect safety margins.

DISPOSAL ISSUES

1. SPECIFIC APPLICATIONS

There are two specific disposal applications for burnup credit corresponding to the phases of operations at a disposal facility. The two phases are the operations or pre-closure phase and the isolation or post-closure phase. The pre-closure phase consists of the handling, packaging, and emplacement/storage of wastes prior to the permanent closure of a repository. The post-closure phase consists of the long-term storage of the waste, after permanent closure of the repository, out to the end of the time period of regulatory concern. The post-closure period may include the waste forms from intact fuel, partially degraded fuel, fully degraded fuel in the waste package, fully degraded fuel in the engineered barrier area (e.g., emplacement drifts, bore-holes, etc.), and fully degraded fuel in the natural rock.

2. MOTIVATIONS, BENEFITS AND NECESSITIES OF BURNUP CREDIT FOR DISPOSAL

For pre-closure applications burnup credit for disposal of irradiated commercial fuel provides design flexibility that may provide economic and ALARA benefits. Burnup credit facilitates increased assembly loading in casks, which leads to reduced cumulative radiological risks (less packages) and associated cost savings. For existing systems, burnup credit would allow the loading of higher initial enrichments of irradiated fuel that could not be readily handled with the fresh fuel assumption.

The use of burnup credit in post-closure disposal applications has the additional value that over the long time period considered for disposal, active criticality control features such as moderator exclusion barriers, neutron absorbing (poison) plates, and geometry features (e.g., flux-traps) will degrade and change. The reduced reactivity associated with the presence of actinide and fission product absorbers in irradiated fuel is the only feature that may last.

3. CURRENT ACTIVITIES AND FUTURE PLANS

This section discusses, by country, the current activities and future plans for using burnup credit for disposal.

3.1. Czech Republic

The Czech State authorities have been supporting a program focused on development of a deep geological repository. Scientific projects for the program are in progress at the Nuclear Research Institute (NRI) at Rez. A burnup credit approach, using actinides and fission products, is being considered as part of these projects. Development of a risk-informed probabilistic approach for post-closure is being recommended as another project. For work on these project tasks, NRI is collaborating with the SKODA JS Company where disposal containers are to be designed. These activities are collaboratively discussed within a Central European context with Slovakia and Germany. Addition information is exchanged with professional contacts from Switzerland and Finland.

At NRI, a project aimed at the development of burnup credit technology for SKODA dual purpose, storage and transport, casks is in progress. It is proposed that the results from this

project will also be used for the disposal container development. In support of this work, there is a clear lack of radiochemical assay data to characterize the isotopic concentration of the VVER spent fuel. The evaluation of radiochemical assays is therefore considered the most urgent issue that needs to be addressed for the burnup credit methodology development for VVER spent fuel. The only VVER operating country capable of performing the measurements is Russia. International support will be needed for VVER depletion validation benchmarks to be specified and carried out.

3.2. Germany

Criticality analysis for final disposal in Germany:

The concept of direct disposal of SNF as an alternative or addition to reprocessing has been considered in Germany since the mid 80s. The salt dome of Gorleben has been selected for investigating its suitability as a disposal site. Up to now, no decision was made. In 1995 the first criticality analyses were performed on behalf of the German Federal Office of Radiation Protection (BfS). The BfS is responsible for planning and operation of the final repository for radioactive material. Criticality analyses are to be distinguished between Operational Phase (pre-closure) and Post-Operations Phase (post-closure). Criticality Safety in Operational Phase has to be ensured by proper design of the disposal canister. In the Post-Operational Phase the demonstration of suitability in a deterministic approach is desired. The time scale to be supposed for the analyses is expected to be greater than 10^6 years. Burnup and post-irradiation decay will be taken into account. The assumption for the analysis should cover the burnup range of SNF to be disposed in a conservative approach. The conservatism of the assumptions used must be demonstrated.

For disposal of SNF, the multi-purpose cask POLLUX has been developed (GNB). The POLLUX will take consolidated rods from 10 PWR fuel assemblies. The POLLUX cask was designed for storage, transport, and final disposal. For standard PWR fuel (up to 4 wt % U-235 initial enrichment), no burnup credit was applied for criticality analysis. A second type of disposal canister, “ELB-3”, is actively being developed to take the consolidated rods from three PWR fuel assemblies.

For the deterministic criticality analyses selected post-closure scenarios will be analyzed. If demonstration of subcriticality can not be achieved in this way, probabilistic risk-informed analyses will need to be considered.

3.3. Slovakia

Slovakia has just started its first considerations of disposal for its spent nuclear fuel. The current poor economic situation has not allowed for resources to perform any detailed evaluations to date. The expectations are that it will be necessary to include burnup credit in the design of casks for final disposal. Near term needs and plans are for developing experimental verification of spent fuel compositions.

3.4. Sweden

Sweden has evaluated the disposal criticality issue since the late 70s. Recently, new studies were conducted to evaluate cask design changes, i.e., the addition of cooling tubes inside the waste package. The Swedish Nuclear Fuel and Waste Management Company (SKB) has

adopted an underground repository concept (KBS-3) for disposal of long-lived radioactive waste including Swedish spent nuclear fuel. SKB has adopted a multi-barrier repository concept that includes the spent nuclear fuel inside a corrosion resistant canister, bentonite buffer (clay), and crystalline host rock. The requirement for the canister design is that no criticality should happen even in the case of water insertion.

The criticality studies performed included: 1) calculations with MONK and WIMS for PWRs, BWRs, and MOX fuels, and 2) review of previous criticality assessments. For the criticality calculations, the disposal configurations included different types of canister and fuel assemblies, initial enrichment, burnup, and amount of burnable poison. The scenarios covered canister failure at some point after disposal with insertion of water in the canister, voids, and cooling tubes. The parameters used in the reference case study for each fuel type where: 40GWd/tHM, 40 years cooling time, and canister failure. All BWR studies showed subcriticality even in the fresh fuel case. The PWR studies in the fresh fuel case showed subcriticality when burnable poison pins were present, however, unirradiated PWR fuel analysed without the presence of burnable poison pins was marginally the most reactive canister under these parameters, showing inadequate margin of subcriticality. Further studies could be performed to investigate this issue.

As conclusion, the study found that there will not be enough fissile material accumulated anywhere in the repository for a criticality event to form. Even if criticality occurs, the consequences on safety would be insignificant. And, although several uncertainties remain from the details of previous analyses (approaches are in general applicable to the current disposal concept in Sweden) it is unlikely that recent modifications to the cask design will have any substantial effect on the findings of the earlier work. More studies are required to confirm this conclusion.

3.5. United States of America

The United States has developed a risk-informed, performance based methodology for disposal criticality analyses. The general methodology is split into two time frames, Pre-Closure and Post-Closure. While Pre-Closure focuses on personnel safety Post closure deals with repository performance issues. Each part of the general methodology use similar analytical methods. The methods are modified to account for their associated time of concern. The methodology includes consideration of actinide and fission product burnup credit, known as Principal Isotope Burnup Credit. Principal Isotope Burnup Credit is to be used for development of the waste packages for commercial boiling water reactor and pressurized water reactor fuels (both uranium and mixed oxide).

The U.S. Nuclear Regulatory Commission, in a Safety Evaluation Report (NRC 2000), has recently approved the disposal criticality analysis methodology for post-closure. Justification of a design using the methodology is still under development.

3.6. Research, development, and implementation needs to support burnup credit

In general the research and development information needed to validate the methods for evaluating burnup credit in irradiated fuel assemblies includes:

1. Radiochemical assays for benchmarking models for calculating fuel isotopic inventory,
2. Critical experiments for benchmarking models for calculating k_{eff} of irradiated fuel,
3. Site specific material data for determining applicable benchmarks.

Table I. Disposal Burnup Credit: National Practices and Status

Country	PWR	BWR	MOX	VVER
Czech Republic	-	-	-	ID
Germany	UD	IC	UD	IC
Slovankia	-	-	-	IC
Sweden	UD	UD	-	-
USA	AP/UD[1]	AP/UD[1]	IC	-

[1] Method approved, designs implementing burnup credit under development

IC = Interest/Considering, or Applicable
ID = Initial Development
UD = Under Development
AP = Approved

In general the information needed for implementing burnup credit includes:

1. Irradiation history and fuel assembly design information for the range of fuel parameters to be evaluated,
2. Verification measurement technology to verify irradiation information records,
3. Details of the specific package design and environment the package will be disposed in.

The specific information needed depends on the specific fuel type(s) being considered, the range of fuel characteristics the models are being validated for, and the country where the disposal site will be licensed. Much of this information is already available for some fuel types, over certain ranges, and for some countries, but it is not available in others. For example there is little radiochemical assay information for disposal burnup credit (or any burnup credit) of VVER fuel, for any range of characteristics, in the Czech Republic, Germany, or Slovakia. In contrast, there is a great deal of information available for PWR fuel, in Germany and the United States.

3.7. Regulatory issues

As each nation moves forward in the development in a disposal site, the regulator will develop specific regulatory issues. It is anticipated that the regulator will require verification of burnup components through the time of regulatory concern, i.e., for the US the time of regulatory concern is 10,000 years. In addition to the Post-Closure concerns, Pre-Closure concerns will most likely include:

1. Verification of the loading procedure,
2. Evaluation of reactor records, measurement of burnup, measurement of axial/horizontal profiles in conjunction with the loading verification.

3.8. Conclusions and recommendations

Burnup credit must be implemented for direct disposal of irradiated nuclear fuel. Without the inclusion of the intrinsic characteristic of irradiated fuel isotopics (i.e., burnup credit), long term criticality control is difficult, if not impossible. Therefore, a burnup methodology should

Table II. Detailed Information Needed.

FUEL TYPE	INFORMATION NEED
BWR	Radiochemical Assays - Burnups > ? GWd/tU - Enrichment > ? wt % U-235 Critical Experiments Design Details - Axial/Radial Enrichments - Integral Absorber Rods
PWR	Radiochemical Assays - Burnups > ? GWd/tU - Enrichment > ? wt % U-235 Critical Experiments Design Details - Axial/Radial Enrichments - Integral Absorber Rods
MOX	Radiochemical Assays - Burnups > ? GWd/tU - Enrichment > ? wt % Pu-239
VVER	Radiochemical Assays - Burnups > ? GWd/tU - Enrichment > ? wt % U-235

be implemented. A burnup credit methodology, which incorporates a best estimate approach including actinide and fission products, is recommended. A risk-informed process for the development and evaluation of probable criticality scenarios should also be considered. Validation data should include discrete and integral data, i.e., radiochemical assay data and reactor critical data. An example of a process that uses such data is referenced in the U.S. NRC's Safety Evaluation Report for the "Disposal Criticality Analysis Methodology Topical Report," YMP-004 Q. It is recommended that a forum be developed for exchanging technical validation information and experience concerning applying the validation data and processes.

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