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Implementation of burnup credit in spent fuel management systems

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FOREWORD

Spent fuel management has always been one of the important stages in the nuclear fuel cycle and it is still one of the most vital problems common to all countries with nuclear reactors. It begins with the discharge of spent fuel from a power or a research reactor and ends with its ultimate disposition, either by direct disposal or by reprocessing of the spent fuel.

Continuous attention is being given by the IAEA to the collection, analysis and exchange of information on spent fuel management. Its role in this area is to provide a forum for exchanging information and to co-ordinate and to encourage closer co-operation among Member States in certain research and development activities that are of common interest. Spent fuel management is recognized as a high priority IAEA activity.

As one of the tasks in its 1997 subprogramme on nuclear fuel and materials, the IAEA has monitored the progress on burnup credit implementation in spent fuel management systems. A consultancy was held and an Advisory Group meeting was organized (20–24 October 1997) in order to gather information on the status of national practices of burnup credit implementation and on current and future aspects. This TECDOC provides an overview of these practices in a number of countries and the status of burnup credit implementation.

The IAEA wishes to thank all participants of the Advisory Group meeting for their fruitful contributions and especially the Chairman of the meeting, W. Lake, and the Chairmen of the Working Groups, J.C. Neuber and J.M. Conde Lopez. Special thanks are due to M.J. Crijns who collaborated in preparing and editing this report. The Scientific Secretary of the IAEA, responsible for the organization of the meeting was H.P. Dyck of the Division of Nuclear Fuel Cycle and Waste Technology.

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CONTENTS

SUMMARY

1. INTRODUCTION

The criticality safety analysis of spent fuel 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 allows one to take credit for the reactivity reduction associated with fuel burnup, hence reducing the analysis conservatism while maintaining an adequate criticality safety margin.

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 may be realized. Many of the additional benefits of burnup credit that are not strictly economic, may be considered to contribute to public health and safety, and resource conservation and environmental quality. Interest in the implementation of burnup credit has been shown by many countries. 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 I (mainly from countries participating in IAEA meetings on burnup credit implementation in 1997).

Burnup credit implementation introduces new parameters and effects that should be addressed in the criticality analysis (e.g., axial and radial burnup shapes, fuel irradiation history, and others). Analysis of these parameters introduces new variations as well as the uncertainties, that should be considered in the safety assessment of the system. Also, the need arises to validate the isotopic composition that results from a depletion calculation, as well as to extend the current validation range of criticality codes to cover spent fuel.

The use of burnup credit implies a verification of the fuel burnup before loading for transport, storage, disposal, or reprocessing each assembly, to make sure that the burnup level achieved complies with the criteria established. Methods and procedures used in different countries are described in this report.

2. DEFINITION OF BURNUP CREDIT

Burnup credit is defined in this report as the consideration of the reduction in reactivity associated with the use of the fuel in power reactors. Changes in the isotopic composition during fuel burnup which result in a reduced reactivity can be conveniently characterized by the reduction of the net fissile content, the build-up of actinides, the increase of the concentration of fission products, and the reduction of burnable absorber concentration where applicable. In practice, the conservative use of burnup credit requires consideration of all fissile isotopes, and allows consideration of any neutron absorbing isotopes for which properties and quantities are known with sufficient certainty. The different levels of burnup credit which are commonly used are described as follows:

- Credit for the net decrease of the fuel fissile content, taking into account both burnup and buildup of the different fissile nuclides (*net fissile content level*).
- Credit for the net fissile content and for the absorption effect of actinides (*actinide only level*).
- Credit for the actinides and the neutron absorption in fission products (*actinide plus fission product level*).
- Credit for the presence of integral burnable absorbers in the fuel design (*integral burnable absorber level*). This credit uses the maximum reactivity of the fuel, which is often not the initial reactivity. Although not really consistent with the definition of burnup credit, it is generally considered to be a level of burnup credit because fuel depletion calculations are needed to determine the reactivity state as a function of burnup.

TABLE I. WORLDWIDE USES OF BURNUP CREDIT: NATIONAL PRACTICES 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. of	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	-	-
SWITZERLAND	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 in 1997.
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.

3. MOTIVATION FOR USING BURNUP CREDIT

The implementation of burnup credit has been dictated in many countries by different needs. Some examples are as follows:

- Introduction of higher enriched fuel in the existing storage, reprocessing or transport systems. In these cases, the use of burnup credit may avoid the need for new installations or for extensive changes to existing facilities or equipment for spent fuel management activities.
- Burnup credit can increase the storage capacity (both on-site and independent) by allowing smaller center-to-center distances in the fuel storage systems.
- Burnup credit can be used for new casks to increase cask capacities over current design capacities to reduce the number of shipments needed.

In the short term, the motivation for using burnup credit is of an economic nature. However, one should keep in mind that part of the work performed for this purpose (e.g., qualification of fuel depletion codes) will be useful to better characterize the spent fuel. In the long term, that knowledge will lead to a better understanding of the physics of the reactor and spent fuel management practices, and therefore, to enhanced safety. This knowledge will also be key to technically sound development of fuel disposal systems and alternatives.

In the area of spent fuel storage, burnup credit and its resultant capacity improvements can avoid or minimize the environmental impacts associated with expanding or building new storage pools or dry storage facilities. For dry cask storage and transportation, higher capacity casks result in fewer shipments, less worker and public exposure, and lower risk both radiological and non-radiological. Burnup credit can be used to maintain production rates at existing reprocessing facilities even while fuel enrichments increase, thus avoiding the environmental impacts of constructing new facilities, or expanding old ones. For disposal of spent fuel, burnup credit is considered, by many, to be a necessity for any viable disposal scheme. Ignoring the reduced reactivity from burnup credit could lead to larger disposal sites and unnecessary use of land.

4. REGULATORY STATUS, CURRENT PRACTICE AND ACTIVITIES PLANNED

4.1. Regulatory status

Criticality safety regulations for fuel storage and transport do not prohibit the use of burnup credit in the criticality safety analysis of spent fuel systems. In particular, IAEA Safety Series 116 [1] explicitly describes the implementation of burnup credit in storage facilities, and IAEA Safety Standards Series No. ST-1 [2] contains the requirements for transport. Likewise, national regulations do not prohibit the use of burnup credit for criticality safety analysis of spent fuel systems.

However, given the difficulty of validating the analysis codes used to demonstrate criticality safety and the challenge of demonstrating that all the relevant effects have been considered, no unrestricted burnup credit methodology has yet been approved. The reactivity effects of a limited number of nuclides have been approved by the regulatory authorities of different countries. Which of the burnup credit levels, as defined in Section 2, have been approved depends on the function and the characteristics of the given spent fuel management system and is specified in the following section.

4.2. National status and current practice by burnup level

4.2.1 Burnup credit for net fissile content

Generally burnup credit for net fissile content is not used alone.

4.2.2. Actinide only burnup credit

Applications of actinide only burnup credit are found in France, Germany, Russian Federation and Switzerland.

France

In France this burnup credit level is approved for:

- wet storage
- wet and dry transport and
- reprocessing of spent PWR fuel.

This burnup credit level refers to the average burnup of the first 50 cm of the fuel zone end, which is burned least, and it is only allowed for the isotopes U-235, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241 and Pu-242.

Germany

In Germany the actinide only burnup credit level is approved for:

- dry transport of spent BWR fuel in the CASTOR V/52 cask.
  The licensing is based on:
  - the use of uranium and plutonium isotopes only, and
  - a minimum average discharge burnup of 5 MW·d/kg U.
- dry transport of spent BWR fuel in the TN13/2 cask.
  The licensing is based on:
  - the validation of the certificate issued by the French competent authority (see France above).

Russian Federation

In Russian Federation the actinide only burnup credit level is approved for:

- wet transport of spent WWER-440 fuel from the Kola Nuclear Power Plant (NPP).

In this case the licensing is based on:

- the use of uranium and plutonium isotopes only and
- a minimum average burnup of 25 MW·d/kg U.

Switzerland

In Switzerland this burnup credit level is approved for:

- dry transport of spent PWR fuel.

This burnup credit level refers to the average burnup of the first 50 cm of the fuel zone end, as is the case for France.

4.2.3. Actinide plus fission product burnup credit

Actinide plus fission product burnup credit is approved for:

- PWR wet storage in the USA, Korea and Spain,
- RBMK wet storage in Russian Federation (Smolenskaja NPP) and Lithuania (Ignalina NPP),
- Dry storage cask loading in the USA.

The PWR and RBMK wet storage systems have a certain amount of soluble boron in the pool water. In the USA and Spain credit is taken in some of these systems for a fraction of this boron content in addition to burnup credit. In all other countries it is not allowed to take credit for this boron content in the criticality safety analysis of the normal operation conditions of the wet storage systems.

4.2.4. Credit for the presence of integral burnable absorbers in the fuel design

Credit for the presence of integral burnable absorbers is approved for:

- BWR wet storage systems in Germany, Spain, Sweden and Switzerland.

4.3. International fuel transfer

Some countries have performed limited independent regulatory review for approval of burnup credit for spent fuel transport, and have accepted the approval of casks designs licensed in the country of origin. This practice is consistent with international agreements to accept approval for designs designated as type B(U)F packaging (see IAEA Safety Series Standard No. ST-1 [2]). For example, Switzerland has accepted the burnup credit approach for PWR fuel transport approved by the French (see Section 4.2.2).

4.4. National activities planned

In addition to the ongoing activities presented earlier, many countries are planning or extending their activities on burnup credit. Information from some countries has been obtained through organizations assisting those countries in spent fuel management activities.

Brazil

Actinide plus fission product burnup credit:

- It is planned to apply this burnup credit level to PWR wet storage (Angra NPP).

Bulgaria

In Bulgaria, it is intended to apply the SCALE modular code system (PC version 4.3) for burnup credit analyses of WWER spent fuel facilities.

Czech Republic

In the Czech Republic, it is under discussion in which way burnup credit might be implemented in the wet and interim storage spent fuel management systems at the NPP Dukovany.

France

Actinide only burnup credit:

- It is planned to apply the concept described in Section 4.2.2. to the reprocessing of spent BWR fuel in France. This plan is presently under regulatory review.

Actinide plus fission product burnup credit:

- It is planned to apply this burnup credit level to the wet and dry transport, wet storage and reprocessing. In a first step, 6 fission products will be taken into account.

Germany

Actinide only burnup credit:

- It is intended to apply the concept described in Section 4.2.2. to other CASTOR casks (e.g., the CASTOR V/19 cask used for spent PWR shipping) to achieve an increase of the respective maximum permissible initial enrichments by about 0.5 wt.% U-235.

Actinide plus fission product burnup credit:

- It is planned to apply this burnup credit level to the wet storage systems of the Konvoi plants. Therefore, application of this burnup credit level to wet storage systems (with soluble boron in the coolant) is presently under regulatory review.

Credit for the presence of integral burnable absorbers in the fuel design:

- It is planned to apply this burnup credit level to the wet storage systems of Gundremmingen NPPs.

Hungary

It is intended to investigate the possible use of the MULTICEL, MCNP and SCALE codes for burnup credit applications.

Russian Federation

Actinide only burnup credit:

It is intended to apply the concept described in Section 4.2.2. to wet transport of the fuel from all the WWER-440 NPPs.

South Africa

Actinide plus fission product burnup credit:

Implementation of this burnup credit level applied to PWR wet storage (Koeberg NPPs) is under way.

Slovak Republic

Verification of the SCALE modular code system for burnup credit analyses of WWER-440 spent fuel facilities is underway for future application of burnup credit.

Sweden

Actinide plus fission product burnup credit:

In Sweden actinide plus fission product burnup credit for BWR and PWR is planned to be used in the final repository of spent nuclear fuel.

United Kingdom

Actinide only burnup credit:

It is intended to apply actinide only burnup credit for:

- the wet transport of PWR and BWR uranium oxide fuels in a number of different cask designs;
- the reprocessing of PWR, BWR and CAGR uranium oxide fuels;
- the wet transport and reprocessing of MOX fuels.

Actinide plus fission product burnup credit:

- Implementation of this burnup credit level applied to PWR wet storage (Sizewell B NPP) is underway. Therefore, application of this burnup credit level to wet storage systems is presently under regulatory review.

USA

Actinide only burnup credit:

- In the USA, the use of the actinide only burnup credit for transport of spent PWR fuel is under regulatory review.
- The methods used for PWR transport will be used for dry storage of spent PWR fuel.

Actinide plus fission product burnup credit:

- Use of actinide plus fission product burnup credit for BWR and PWR spent fuel is planned for disposal activities in the USA.
- The approval of actinide only burnup credit by the regulatory authorities may lead to pursuit of actinide plus fission product burnup credit for spent fuel transport.

5. DEPLETION CODES

Application of burnup credit requires knowledge of the reactivity state of the irradiated fuel for which burnup credit is taken. To obtain this knowledge, the isotopic inventory of the irradiated fuel has to be calculated with the aid of depletion codes and will depend on the initial state and the irradiation history of the fuel. The results of the depletion analysis are a necessary input to the criticality analysis of the system for which burnup credit is taken. Three sources of uncertainties have to be taken into account:

- the knowledge of the fuel's initial isotopic state;
- its irradiation history; and
- the calculation of its final isotopic state.

Generally there is high confidence in defining the initial state of the isotopic inventory of the fuel. Irradiation histories are obtained from reactor records. However, the wide variety of fuel irradiation histories makes it necessary to look for a bounding history contemplating the most limiting fuel operating conditions, meaning those that lead to the highest spent fuel reactivity. The uncertainty of a depletion code is controlled and established through verification of that code, usually, by comparison with suitable and appropriate experiments.

The following sections give an overview of the depletion codes and describe the general methods applied to perform verification of these codes. Guidelines for achieving the appropriate level of precision and accuracy, or conservatism to satisfy the safety assurance required for criticality safety analysis are given in Sections 7 and 8.

5.1. Depletion codes and verification methods

The list of depletion codes provided here is not comprehensive. Additional codes, short descriptions thereof and the conditions under which they can be acquired can be found by accessing the following World Wide Web sites: OECD/NEA Data Bank: http://www.nea.fr/html/dbprog, Radiation Shielding Information Computational Center (RSICC): http://www-rsicc.ornl.gov/. Experimental data for code and data validation can also be found in these sites, but in particular the ICSBEP at: http://wastenot.inel.gov/icsbep/.

BOXER

The BOXER code, developed by the Paul Scherrer Institute in Switzerland, is used both for depletion and criticality calculations [3]. BOXER performs cell and two-dimensional transport and depletion calculations. Resonance self-shielding is calculated by a two-region collision-probability method in typically 7,000-8,000 energy points. The fine-group cell calculation and the broad-group transport calculation in rectangular geometry are performed using integral transport methods. Depletion calculations are performed using one-group cross-sections collapsed with the spectrum from the two-dimensional calculation. 34 actinides (from Th-232 through Cm-248) 55 explicit fission products and two pseudo fission products are calculated.

BOXER was validated against a large number of critical experiments including both uniform lattices and configurations representative of storage pools and transport casks (neutron absorbers, metal reflectors). Depletion calculations were verified against Yankee-Rowe assay and international benchmark problems (OECD/NEA benchmarks on burnup credit and on recycling of reprocessed uranium).

CASMO

CASMO, developed by Studsvik, is a 2-D integral transport cross section generating code for LWR fuel lattices. It calculates the flux, burnup and the isotopic composition of the fuel assembly. The nuclear data file is based on data from ENDF/B and JEF-2, processed with NJOY, and with a group structure similar to that of the WIMS code. CASMO allows for an exact description of the fuel geometry, and the calculations can be made using 40 or 70 energy groups. Earlier versions of CASMO (prior to CASMO-4) made use of MICBURN to independently perform the integral burnable absorber burnup calculation. The libraries generated by MICBURN were used by CASMO to perform the lattice burnup calculation [4,5].

CASMO is a widely used code that has received extensive verification work, both against critical experiments, chemical assay data, international benchmark programmes and reactor core measurements.

CESAR

CESAR is a depletion code dedicated to the characterization of spent fuel UOX (PWR and BWR) and MOX [6]. 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. It enables to evaluate 40 heavy nuclides, 204 fission products and 2 activation products. It uses a Runge-Kutta method for calculations during irradiation and a matrix type method for calculations between cycles and during cooling time.

FISPIN

The FISPIN inventory prediction code [7,8] uses a point model for the fuel depletion calculation, which means that the fuel is treated as a single lump, with no fuel or core geometry modeled. The geometry of the problem is accounted for within the nuclear data library, which provides the in-core spectrum and flux weighted cross section data constructed from a detailed core lattice calculation. The analyst can use nuclear data taken from a 'standard' data library, or else construct a problem specific data library via the WIMS-BUDLIA calculational route. The FISPIN calculation is performed under a three energy group scheme which encompasses thermal, epithermal and fast neutron energies.

The FISPIN code has been extensively validated for a range of applications, principally by comparison of calculated fuel inventories with chemical analysis of spent fuel samples. For burnup credit applications, FISPIN has been validated for PWR, BWR and CAGR uranium oxide fuels, as well as for MOX fuels [9].

KASSETA

KASSETA (Kurchatov Institute, Russian Federation) is a few-group two-dimensional diffusion code that treats a WWER assembly as a regular hexagonal cell containing fuel pins, absorber rods, metal tubes, etc. [10]. The code was verified and validated against numerous experimental data resulting from dissolution experiments and in-core reactor measurements. Recently, Kurchatov Institute issued an advanced version of the code named TVS [11].

MULTICEL

MULTICEL is a two dimensional depletion code used in Hungary. By the approximate solution of the integral transport equation, the pin by pin power distribution and isotopic composition can be calculated for a WWER-440 fuel assembly. The calculation is performed in 35 epithermal and 35 thermal groups. The concentrations of the 18 actinides and 147 fission products are followed.

The code has been verified with the aid of critical experiments, mathematical benchmarks (for example vs. WIMS7 and HELIOS) and isotopic compositions of the Yankee-Rowe reactor.

NESSEL-4, PYTHIA/TRAPEZ, DERAB

This code system applied to WWER fuel was developed by KAB Berlin in Germany. It is used for spectrum and nodal reactor analyses. Among other data and features this code system provides the isotopic inventory as a function of burnup and axial power and burnup profiles.

The verification of the code system is mainly based on analysis of normal power operation measurement data, test track measurement data and actinide densities in spent WWER-440 fuel.

ORIGEN

ORIGEN is part of the SAS2H package, see below. As a stand-alone module, it has many times been compared against chemical assay data; however, since the code must use collapsed cross sections, the code use should be restricted to applications close to the validation data.

PHOENIX

PHOENIX is ABB Atom's standard burnup program for pin cell calculations for both BWR and PWR assemblies. PHOENIX works with the 25-group PHOENIX Nuclear Data Library, based on ENDF/B-III, IV and V.

The code has been verified against critical experiments, which consisted of low enriched UO_2 fuel pins in a water-moderated lattice, simulating a variety of close packed LWR fuel storage configurations.

SAS2H

SAS2H is a sequence controller in SCALE that takes simplified input and runs a series of computer codes for depletion analysis [12]. The sequence uses BONAMI-S that corrects the group cross sections for resonance self shielding in the unresolved region. It then uses NITAWL-II to correct the group cross sections in the resolved resonance region. These corrections use the problem dependent geometry and nuclide concentrations. The next step is the creation of cell averaged cross sections done by analysis with XSDRNPM-S, a one dimensional transport code. XSDRNPM-S calculates the flux in the moderator, the cladding, and the fuel to allow the flux volume weighting for the cell homogenization. After resonance treatment of the materials not present in the fuel cell, XSDRNPM-S is run again to model the complete fuel assembly. The flux from this model is used to collapse the energy group cross sections to allow depletion analysis. ORIGEN-S is then run to deplete the fuel for a segment of the fuel cycle. After completing the depletion of a segment, the code sequence repeats with the updated nuclide concentrations. The user selects the appropriate number of segments in modeling the fuel cycle so that the desired accuracy is obtained.

The SAS2H sequence of the SCALE package version was verified against numerous experimental data resulting from dissolution experiments and in-core reactor measurements [13,14].

SAV90

SAV90 is the Siemens standard core design procedure applied to PWR UO_2 and MOX fuels. This system is used for spectrum and nodal reactor analysis. Among other data and features, this code system provides the isotopic inventory as a function of burnup and axial power and burnup profiles [15].

The verification of this system is based on 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), special measurement programs (reactivity coefficients and equivalents describing the dynamic behavior of the reactor, short-term and long-term transients) and analysis of the isotopic inventory of spent fuel (chemical assay data).

TGBLA

TGBLA is the lattice physics/depletion code developed and used by Toshiba and General Electric. It starts with a 30 group THERMOS-type and a 68 group GAM-type library. The major isotopes come from ENDF/B-V with lessor isotopes from ENDF/B-IV, JENDL-1,-2 and RCN-2. The first step of the TGBLA solution is to generate few-group effective cross sections, which include a thermal neutron spectrum calculation, resonance integral calculation for each resonant nuclide, fast and epithermal neutron spectrum calculation, and generation of diffusion parameters to be used in the succeeding diffusion calculation. This is followed by calculating the infinite multiplication factor and rod-by-rod power distribution by use of a coarse-mesh two dimensional few-group (3 or 4 group) diffusion method based on a volume center finite difference formalism. Finally, the burnup calculation is performed using 25 fissile and fertile nuclides and 45 explicit fission products and one lumped fission product. Care is taken to correctly treat Gd rods which exhibit an "onion-skin" depletion.

TGBLA has been validated with both Monte-Carlo calculations, analysis of over 150 relevant critical experiments and finally operating plant data analysis [16].

WIMS

The WIMS suite of codes [17] is a general purpose reactor physics package developed by AEA. The individual WIMSD, WIMSE and LWRWIMS codes has been widely used over more than 30 years for a wide range of applications, including core modeling, depletion and criticality safety assessment problems. The latest version of the code, WIMS7 [18], combines all of the functionality of the previous codes within a single code format, and uses the 172-group WIMS '1996' nuclear data library which is based upon JEF2.2 data evaluations. The WIMS7 code has powerful geometry modeling capability, which allows a detailed representation, where needed, of the core and fuel assembly geometry. This can be of particular importance for the depletion of integral burnable poisons in BWR and PWR fuels, and for MOX fuels which are irradiated alongside uranium oxide fuels in the core.

The WIMS7 code has been validated for burnup credit applications by comparison with chemical analysis results for UO_2 and MOX spent fuel samples from PWR, BWR and CAGR [9].

5.2. Experimental programmes for verification of depletion codes

In-core reactor measurement data are very well suited to verifications of depletion codes. However, as for burnup credit in criticality safety analysis, particular significance should be attached to direct comparisons of calculated to measured isotopic concentrations.

A large number of chemical assay data available in the open literature have been compiled in the SFCOMPO data base by JAERI and made available to the international community. Experimental programmes are in progress or planned in several countries. Part of them, however, will be proprietary to the participating organizations and results will not be generally available.

Programmes comprising measurements of reactivity worths of spent fuel samples in critical facilities and chemical analysis of the samples for actinides and fission products include:

- CERES (UK, France, USA) in progress since 1991;
- REBUS (international co-operation headed by Belgonucléaire) to start in near future, including MOX fuel;
- PROTEUS (Paul Scherrer Institute, Switzerland, in co-operation with Swiss utilities and fuel vendors) in early stage of planning.

Programmes to determine the isotopic composition of spent fuel from commercial PWRs and BWRs are in progress in Europe (ARIANE) and Japan. Experiments including chemical analyses will be carried out by BNFL (UK) when dissolving MOX at the reprocessing plant. The European programmes will have commercial restrictions at first, but may be generally available in the future. The results of the Japanese programme will be made available in the SFCOMPCO data base.

5.3. Isotopic correction factors

To achieve conservatism in criticality calculations, isotopic correction factors can be derived from comparisons of measured and calculated isotopic concentrations with the aid of statistical methods (see the national contributions of the Republic of Korea and the USA). Conservatism in criticality calculations can also be achieved by deriving isotopic correction factors through other methods (see Section 6.2.4 and the German national contribution).

6. CRITICALITY CALCULATION CODES

Criticality safety is demonstrated by calculational methods applied to a system of known configuration and isotopic content. The codes used for these criticality calculations must be verified by comparison to some acceptable standard, of known quality. The standards for comparison may be

experiments, other accepted codes, or recognized standard problems. These standards or benchmarks must be chosen appropriately to establish the quality of the criticality calculation of the system of interest. The following sections give an overview of the general methods used and describe the general methods applied to perform verification of these codes. Guidelines for achieving the appropriate level of precision and accuracy or conservatism to satisfy the safety assurance required for criticality safety analysis will be given in Sections 7 and 8.

6.1. Criticality calculation codes and verification methods applied

As far as criticality codes are concerned, there are codes that are used worldwide which are similar to those found in the SCALE package and MCNP (both have been developed in the USA), and codes which are primarily used in the country where they were developed. The list of criticality codes provided here is not comprehensive (see Section 5.1. for acquisition of other codes):

APOLLO-MORET

APOLLO 1 (France) is a 99 group deterministic criticality code with CEA 86 library [19]. MORET III is a Monte Carlo code with 16 groups [20]. For burnup credit with actinides only these codes are qualified with 102 benchmark experiments. The experiments used fuel compositions equivalent to 4.5 % enrichment and 3.75 MW·d/kg burnup without fission products. These experiments simulate transport, storage, and dissolution configurations.

BOXER, CASMO

Two dimensional depletion codes such as BOXER (PSI, Switzerland) and CASMO are often used for criticality safety analyses of simple geometric systems (see Section 5.1).

KENO, MCNP

The KENO module of the SCALE package is a three dimensional multigroup Monte Carlo criticality code which can be used on a stand alone basis, or as a module within the SCALE code system. MCNP is a three-dimensional general purpose coupled neutron-photon-electron Monte Carlo code with an option to perform criticality calculations. It uses point-wise cross-section representation. Both code systems are verified against a large number of critical experiments (see the OECD/NEA International Handbook of Evaluated Criticality Safety Benchmark Experiments for instance[21]) and they are widely used in the nuclear industry.

MCU

MCU (Kurchatov Institute, Moscow, Russian Federation) is a three dimensional Monte Carlo code with multigroup or continuous energy treatment in resonance end epithermal energy range plus forty thermal group cross-sections [22]. The code is verified against critical experiments.

MMKFK-2

MMKFK-2 (IPPE, Obninsk, Russian Federation) is a three dimensional Monte-Carlo code with 24 group cross-sections in fast and resonance energy range plus 40 group cross-sections in thermal energy range [23]. It is possible to perform calculations using a few-group approach with effective cross-sections calculated by WIMS-D4 or KASETTA. The code is verified against critical experiments.

MONK

The Monte Carlo neutronics code MONK is the principal criticality safety assessment tool used in the UK, the latest version of which is MONK7B [24]. The MONK code uses a 8220 point nuclear data library based on UKNDL and JEF2.2 data evaluations, and is supported by an extensive validation database covering a wide range of fissile systems and moderation and reflection conditions. Specific validation has been obtained from experimental programmes such as CERES for burnup credit reactivity calculations involving the major actinides and fission products [9].

WIMS

As with other two-dimensional depletion codes, WIMS7 [18] is commonly used in the UK for criticality safety analyses of simple geometric systems, and for sensitivity analyses supporting the primary criticality calculations performed with the MONK code.

6.2. Benchmarks

6.2.1. Benchmarks for net fissile content or actinide only burnup credit

Critical experiments on mixed uranium-plutonium systems are well suited to verifications of net fissile content or actinide only burnup credit applications [25]. Such experiments can be found, e.g., in Volume VI of the OECD/NEA International Handbook of Evaluated Criticality Safety Benchmark Experiments [21].

6.2.2. Burnup Credit Criticality Safety Benchmark Group sponsored by OECD/NEA

The objective of the criticality safety benchmark group sponsored by the OECD/NEA 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. Towards this end, the suite of burnup credit criticality benchmarks was established by the OECD/NEA Burnup Credit Criticality Benchmark Group [26]. The benchmark problems were selected to investigate code performance over a variety of physics issues associated with burnup credit: relative performance of fission products and actinides with respect to the multiplication factor (k) for PWRs [27]; trends in k and isotopic composition with burnup and enrichment for PWRs [28]; effect of axially distributed burnup in PWRs [29,30]; effects for BWRs [31,32]; and effects for MOX fuels [33]. A similar benchmark specification has been made for spent WWER-440 fuel [34-36].

6.2.3. Reactor critical configuration calculations

The developers of the SCALE package have made extensive contributions to the burnup credit validation of the SCALE package by analyzing PWR reactor critical configurations [25].

6.2.4. Worth experiments

Worth experiments can be used for added validation for isotopes. The nuclide density of an isotope can be corrected based on the ratio of measured to calculated worth.

7. PARAMETERS AFFECTING BURNUP CREDIT

The reactivity of a spent fuel management system depends on:

- the reactivity of the fuel assumed for this system; and
- the characteristics of this system.

The proof of sufficient subcriticality of a spent fuel management system is usually given for:

- the most reactive fuel type to be considered; and
- the most reactive system characteristics.

The reactivity of irradiated fuel depends on:

- the initial fissile fuel loading and its distribution within the fuel assemblies;
- the initial amount of burnable absorber and its distribution within the fuel assemblies;
- the fuel assembly's irradiation history determining:
  - the rates of plutonium and fission product buildup;
  - the rates of fissile, fission product and burnable absorber depletion; and
  - the distribution of the burnup and, therefore, the distribution of the fissile, fissionable and fission product content within the fuel assemblies.

In addition, the reactivity of irradiated fuel is affected by the decay of the radioisotopes contained in the fuel after reactor shutdown (in-core and ex-core), number and duration of cycle outages, missing cycles and might be affected by the distribution - in particular, the axial distribution - of the burnup within the fuel assemblies.

7.1. Effects of the spent fuel reactivity on the spent fuel management system's reactivity

If burnup credit is used for the design of a spent fuel management system the following statements should be noted:

- If a fuel assembly does not have any initial burnable absorber content the reactivity of this assembly decreases monotonically with increasing average discharge burnup. However, for a MOX fuel assembly this decrease in reactivity is significantly less than it is for an uranium dioxide assembly.
- The initial presence of burnable absorbers within a fuel assembly affects the rates of plutonium and fission product buildup due to spectral effects. The reactivity of the fuel assembly therefore depends significantly on the absorber content and the number and the locations of the burnable absorber bearing fuel rods.
- The reactivity of a BWR fuel assembly which has attained a certain average burnup depends significantly on the void history of the fuel assembly. Due to the fact that in an operating BWR the moderator density rapidly varies axially it is essential to take different void histories into consideration.

In fact, as already stated in Section 5, what matters is to find a bounding irradiation history which leads to the highest spent fuel reactivity. Due to the wide variety of void histories taking place during irradiation this is usually difficult for BWR fuel. Therefore, for BWR fuel the criticality safety analysis is usually restricted to application of credit for the presence of integral burnable absorbers to the most reactive void history confirmed by analysis.

The effects of initial enrichment distributions can be analyzed in sensitivity studies. Up to now it was never found that, for a given average enrichment, a mixed enrichment distribution results in a higher reactivity than the averaged (uniform) enrichment distribution. Effects of initial enrichment distributions are included in the Phase III-A of the benchmark programme described in Section 6.2.2.

7.2. Effects of the system's characteristics on the system's reactivity

The determination of the most reactive fuel type assumed for the system of interest should include evaluation of the characteristics of this system. The following statements should be noted:

- The geometry and temperature of the system are usually quite different from the geometry and the operating temperature of the reactor core. Therefore, the fuel assembly type which is the most reactive type under core operating conditions is not necessarily the most reactive type under the characteristics of the system of interest.
- Likewise, the void history which is found to be the most reactive one for a given BWR design under core operating conditions is not necessarily the most reactive one under the characteristics of the system of interest.
- The presence of (fixed or soluble) neutron absorbers in the system of interest may be one of the parameters determining the most reactive fuel type (e.g., the reactivity of a wet PWR storage system decreases monotonically with increasing boron content in the water. However, for a wet MOX fuel assembly storage system this decrease in reactivity is significantly less than it is observed for a wet uranium dioxide assembly storage system.)
- Vice versa, spectral effects due to plutonium and burnable fission product buildup may reduce - compared to the fresh fuel case - the efficiency of neutron absorbers used in the system of interest.

7.3. Effects of the radioactive decay of spent fuel isotopes on the system's reactivity

The effects of the radioactive decay of the radioisotopes contained in the spent fuel which are observed depend on the burnup credit level applied to the system of interest.

Promptly after shutdown, Xe-135 builds-up thereby reducing the reactivity in the core, but due to the rapid decay of I-135 and Xe-135, the reactivity of spent fuel will increase within the first four days after reactor shutdown. In the same time, there is an accumulation of Pu-239 due to the decay of Np-239. The maximum Pu-239 inventory is at about 20 days to allow Np-239 to decay (half-life is about 2.5 days).

Four days after shutdown, the production of strongly absorbing nuclei like Sm-149 through β-decay of Pm-149 leads (on the actinide plus fission product burnup credit level) to a slight decrease in reactivity. After about 100 days, the production of Gd-155 and Am-241 through β-decay of Eu-155 and Pu-241 respectively, becomes predominant and causes a stronger reactivity decrease for time periods covering all spent fuel management systems with the exception of disposal. For periods of times which are to be considered for disposal, it has to be taken into account that the half-life times of Pu-238 and Pu-240 are significantly less than the half-life time of Pu-239, and that Pu-239 decays via α-decay to U-235.

7.4. Effects of the burnup distribution within the fuel assemblies on the system's reactivity

7.4.1. Effect of radial burnup distribution

The effect of radial burnup distributions can be analyzed in sensitivity studies. It should be noted that the radial (or horizontal) tilt in assembly burnup might be dependent on the assembly average burnup (see national contribution of the USA). It seems that, for a given radially averaged burnup, a radially varying burnup distribution usually results in a lower reactivity than the averaged burnup distribution. However, the axial distribution of the assembly burnup has to be taken into account in all cases.

7.4.2. Axial burnup profiles of PWR fuel

For PWR fuel it is an established fact that axial burnup shapes may have a strong effect on burnup credit, i.e., on

- the loading curve (reactivity equivalence curve) providing the minimum required average discharge burnup as a function of the initial enrichment of the fuel for the spent PWR fuel management system of interest; and/or on
- the design characteristics of this system.

How strong this effect is depends not only on the management system of interest, but also on

- the specific core geometry of the PWR nuclear power plant of interest; and
- the specific reactor steering strategy pursued in that plant.

It is essential, therefore, to analyze axial burnup shapes specific of the nuclear power plant of interest. For transport problems this means that axial shapes from all plants to be considered have to be taken into account.

7.4.3. Axial burnup profiles of BWR fuel

For BWR fuel, the present difficulty is to determine bounding burnup shapes. This has been a key factor to impede highly detailed burnup credit studies. The difficulties arise from the wide variety of irradiation histories and axial fuel zones present in the same core. Also, the operating strategy of each plant heavily impacts both the spent fuel isotopic composition and burnup shape of the individual spent fuel assemblies.

8. CRITICALITY SAFETY ACCEPTANCE CRITERIA

The criticality safety acceptance criteria to be applied to burnup credit analysis are the same as those used for analysis based on the fresh fuel assumption. No regulation, whether national or international, has been found that prescribe different criteria for burnup credit cases.

As a consequence, the basis for criticality safety acceptance and approval for burnup credit analysis is the demonstration of compliance with a subcritical limit taking into account all the calculational uncertainties, manufacturing tolerances and other uncertainties related to burnup (declared burnup, axial burnup profile and others). The safety margins taken for defining the subcritical limit to be applied are usually the same as those applied to fresh fuel analysis. It should also be realized that for final disposal of spent fuel assemblies the long-term behavior of reactivity (1,000 – 1,000,000 years) has to be addressed. To fulfill these criteria in a general manner, the main tasks to be performed in a burnup credit analysis are as follows:

- Define a calculational scheme, including depletion codes, cross-section libraries, criticality codes, etc.;
- Define the fuel characteristics, including physical description, depletion conditions (moderator temperature and void fraction, power density, etc.) and cooling time;
- Define the system characteristics and tolerances (storage or reprocessing device, transport cask, etc.);
- Evaluate the sensitivity of both the depletion code and the criticality calculation code to the appropriate parameters, in order to determine their bounding values which are to be used for the analysis;
- Define the code validation scope needed;
- Perform benchmarks for the codes (depletion and criticality, see Sections 5 and 6) by comparison with experimental data or with validated codes, to quantify the biases and uncertainties;
- Define the acceptance criteria in terms of a system's reactivity, including all the uncertainties and tolerances;
- Determine the burnup level required for the fuel to fulfill the acceptance criteria;
- Prepare the operational procedures to assure that the analysis assumptions adequately represent the system and its operation.

The accuracy and detail with which each of these steps has to be studied depends on the overall conservatism retained in the analysis methodology. It is clear that actinide plus fission product burnup credit methodologies would need further investigations to be sure that all the uncertainties, sensitivities and lower order effects have been accurately taken into account. On the other hand, the embedded conservatism in net fissile or actinide only methods (though not quantified) may allow for simplified evaluations of some aspects.

9. VERIFICATION OF FUEL SELECTION AND CASK/UNIT LOADING PROCEDURES

The fresh fuel assumption for criticality safety analysis has always required verification prior to loading to confirm that the enrichment (and thermal limits where applicable) are met. Traditionally, verification of fuel selection has consisted of double verification of the assembly identification.

Likewise, burnup credit implementation requires a verification of the burnup level of each individual fuel assembly to make sure it satisfies the burnup acceptance criteria derived from the analysis. This verification can be performed by a measurement (qualitative or quantitative), reactor records, or some combination of both. Verification obviously should be performed before loading, storing or reprocessing the spent fuel.

The requirements for verification are often related to the amount of margin retained in the criticality analysis (i.e. to the analysis conservatism). Traditionally, for handling of fuel in borated

spent fuel pools, assembly burnup determination based on reactor records are generally accepted without any additional requirements, since a considerable margin generally exists due to not taking full credit for the boron concentration in the pool. However, it is often decided that some burnup verification measurement is required for spent fuel transport and reprocessing.

This verification measurement also may vary, depending on the amount of margin to criticality. The following cases may be distinguished:

- if the burnup required to fulfill the reactivity limit is less than the minimum that the reactor can guarantee after one cycle of irradiation, a qualitative measurement such as gamma scanning may be used to confirm that the fuel is irradiated;
- when the burnup level required is higher than that achieved in one cycle, quantitative measurements may be performed using qualified devices, generally based on passive measurement of neutrons emitted by the heavy nuclides or gamma radiation emitted by fission products that are present in the irradiated fuel. In this case, the measurement uncertainty has to be considered in the verification.

In some cases the fuel burnup is calculated from the reactor's records. In these cases, the data for the operation cycle which are obtained through periodic core surveillance (power level, radial and axial power distributions) are used to determine the burnup level of each individual fuel assembly. The uncertainties inherent in measurements and the uncertainties in the calculations have to be conservatively taken into account.

Due to the added complexity of burnup credit, it is more likely that a significant fraction of the fuel at a site would not be allowed for loading in the cask/unit. The procedure used to control the verification process should describe the actions to be taken to assure that criticality safety is maintained. The procedures should include methods to segregate individual spent fuel assembly that do not comply with the minimum burnup level required.

Some examples of measurement practices and equipment used in different countries are included below. More detailed information can be found in the contributions of the individual countries.

France

Measurements at the reactor storage facility are performed with two different devices:

- The PYTHON device evaluates the burnup of the fuel assembly using a passive neutron measurement and a total gamma irradiation profile measurement. The whole active length of the fuel is measured.
- The Fork device described below, in the subsection dedicated to the USA, has been adapted by the CEN-MOL laboratory (Belgium) and validated by the French competent authority.

In the reprocessing facility, the burnup of the fuel assembly is determined by a passive neutron measurement, and by gamma spectrometry which examines the irradiation profile over the whole active length of the assembly.

Germany

For dry transport of spent BWR fuel in the CASTOR V/52 cask a gamma spectrum approach, that measures Cs-137, is used, and the exposure history of each fuel assembly is analyzed.

Russian Federation

Burnup measurements are required for the transport of some WWER fuel, and the measurement is performed using the FAMOS-III detector. At this time, only the Kola plant is doing this measurement. The Kola plant has a FAMOS-III detector permanently installed in the spent fuel pool. The detector measures the Cm-244 neutron emissions and corrects for boron concentration in the spent fuel pool by use of shielded and unshielded neutron detectors. The FAMOS-III system also uses a Geiger-Mueller tube for total gamma to aid in the determination

of the burnup distribution. The FAMOS-III system contains detectors on three of the six faces of the assembly to provide for a correct horizontal averaging. To obtain a correct value for the assembly average burnup, the system makes measurements at 10 axial heights. The detector is calibrated against the plant's declared burnup for a large number of assemblies, as well as against a Cf source. The error of the measurement system is about 12% in burnup.

United Kingdom

Currently, no burnup measurements are required for burnup credit fuel verification purposes. The THORP reprocessing plant possesses a fuel monitor for the assaying of fuel prior to dissolution, which employs passive and active neutron measurements and gamma spectrometry for the accurate determination of fuel burnup, cooling time and initial enrichment. BNFL has used the THORP device as the basis for the portable BNFL Spent Fuel Monitor for external use, such as in the USA. The Spent Fuel Monitor is described below.

USA

Burnup verification has been demonstrated using the Fork detector and the BNFL Spent Fuel Monitor. Burnup verification will be required for the actinide only burnup credit method currently under review. The current program requires only the measurement of the assembly average burnup. The Fork detector [37] uses the Cm-244 neutron emissions to measure the fluence observed by the fuel. By use of the known enrichment this measurement can be converted to burnup. The reactor declared burnup for all assemblies measured is used to determine the burnup calibration. This detector has been used at three reactors with a demonstrated accuracy of about 3% [38,39].

The BNFL Spent Fuel Monitor uses a gamma spectrometry approach that measures the Cs-137 as well as other isotopes [40]. It uses the ratio of the Cs-137 peak to those of isotopes having shorter half-lives to predict cooling time. Since the fission yield of Cs-137 is close to the same for both U-235 and Pu-239, the content of Cs-137 is proportional to burnup. The BNFL device has been used at Arkansas Nuclear One. The calibration for that set of measurements was similar to that done for the Fork detector, i.e.,: the operator declared burnup for the assemblies measured was used as a basis. BNFL reports that this detector can be calibrated by use of a known Cs-137 source.

10. DATA NEEDS AND COUNTRY SPECIFIC NEEDS

Data (e.g. on reactivity and results of chemical assay of spent fuel) is needed in various degrees to support burnup credit activities for the four types of fuel (BWR, PWR, MOX, and WWER) and applications (storage, transport, reprocessing, and disposal) considered in this report. The activities may vary from investigations in modest burnup credit levels to applications of higher burnup credit levels. Some data development activities are already underway, and should be continued. The data is necessary to support benchmarking activities. Because OECD/NEA has taken the lead in this area, they should be requested to continue this important work, and to expand their efforts to cover additional needs as they develop.

Countries and organizations needing specific categories of data should be encouraged to co-operate to the extent possible to share costs and results of experimental work. Countries that conduct their own data development programmes, should consider making such data available to others through contractual or co-operative arrangements. Finally, whenever possible, data developed for burnup credit should be published in the open literature.

Efforts should also be made to determine the applicability of data exchange between fuel types and applications. There may be some regions of applicability where overlaps may be present. Data collection activities should be designed to identify and assess the feasibility of such interchanges.

10.1. Data needs

10.1.1. PWR data needs

It is believed that a significant amount of data is available in the open literature to support actinide only approaches to burnup credit. Even more data, that is proprietary, exists. Indeed, approvals for various applications of actinide only burnup credit have been granted for PWR fuel. Actinide plus fission product burnup credit has been approved in a number of countries for storage applications; however, similar approvals have not been issued for transport applications. For transport applications, additional benchmarking data for validation of depletion and criticality codes are needed. The depletion code validations require chemical assays to determine concentrations of the fission products of interest as a function of reactor histories. For criticality code validation worth experiments, or critical or approach to critical experiments are needed. Treatment of axial burnup variations may require collection of data on such distributions to determine bounding values.

10.1.2. BWR data needs

The activity on burnup credit for BWR fuel has been less than for PWR. The reasons for this seem to be the greater difficulty associated with treatment of BWR burnup credit issues, and the fact that application of the integral burnable absorber level is sufficient enough in most cases. Programmes for BWR fuel, which are still to be developed, will follow similar paths as those done for PWR. Because of the unique fuel designs and methods of criticality control for operating BWR reactors, validations of depletion and criticality codes for the actinide plus fission product burnup level are expected to be more challenging than has been the case for PWR systems. Additionally, axial effects are expected to be more difficult to deal with.

10.1.3. MOX data needs

Compared to uranium dioxide fuel, employment of MOX fuel leads to a neutron spectrum hardening which affects the rates of fissile depletion and plutonium and fission product buildup, and may affect the distribution of burnup within a fuel assembly. As stated in Section 7.1, the reactivity decrease with increasing burnup for MOX fuel is significantly less than that observed for uranium dioxide fuel. In addition, the reactivity change of MOX fuel with increasing burnup may be dependent on the initial Pu quality isotopic vector of the MOX fuel of interest. Accordingly, improvements in depletion code validations require chemical assay data to determine concentrations of actinides and fission products of interest as a function of initial Pu quality isotopic vectors and irradiation histories. Evaluation of the reactivity effect of MOX PWR axial burnup shapes on burnup credit requires collection of data on such shapes. As it has been the case for uranium dioxide systems, treatment of MOX BWR fuel is more complex than examination of MOX PWR fuel.

10.1.4. WWER data needs

Most of the Eastern and Central European Countries discontinued the use of their own or Russian methodologies used in the past and started using codes and libraries developed in western countries. However, these codes and libraries were not developed for the WWER environment. Not only the hexagonal lattice algorithms should be properly applied in criticality codes (e.g. see several recent corrections of the RSIC CCC-545/SCALE 4.3 package connected with the latest version of KENO VI for the WWER applications) but also the specific WWER spectrum could be used for processing a SCALE neutron cross section library with a smaller number of groups.

The countries that use WWER reactors have a definite interest in pursuing burnup credit for storage and transport applications, and for one there may be interest in using burnup credit for reprocessing. According to Atomnaja Energija T-55 [41], some experiments have been carried out to determine burnup and isotopic composition of WWER-440 spent fuel. The irradiated samples are from the 3rd and 4th units of Novovoronez NPP. For code validation purposes there is a need for

additional information on the irradiation history, fuel and moderator temperatures and even some design information. However, more data is needed (e.g. on WWER spent fuel on isotopic composition and criticality, see Section 10.2.).

The Czech Republic could possibly contribute to an international experimental data base from critical experiments. In 1995, a unique critical experiment on LR-0 research facility of NRI at Rez was performed. The lattice of the CASTOR 440/84 cask, which is used for WWER-440 spent fuel storage at NNP Dukovany (interim storage), was modeled in part of the LR-0 core and various measurements were carried out in addition to an evaluation of its subcriticality for the most accidental - fully flooded - case. Several MCNP calculations were performed to support the experiment in a calculational way. The CASTOR producer, as a collaborating partner in the experiment, has been asked to release the experimental data.

10.2. Country specific needs

Bulgaria

- Experimental data (isotopic concentrations) for selected WWER-440 and WWER-1000 spent fuel samples with well defined irradiation history and initial isotopic state are needed for comparison with the calculational data, obtained by depletion codes;
- Support for an additional verification and validation of the SCALE modular code system (KENO, ORIGEN-S) with correct cross section libraries, suitable for WWER applications would be needed. This could be done on the basis of an international collaboration between countries operating WWERs. This would provide a suitable and reliable tool for burnup credit analysis of WWER spent fuel management systems.

Czech Republic

- Samples from selected WWER 440 spent fuel with a well-defined irradiation history should be taken and analyzed, preferably in Russian Federation. The results should be made available for a comparison with results of calculations, such as the proposed burnup credit benchmark for WWER 440 [34-36] specified according to the pattern agreed within the OECD/NEA/NSC burnup credit Criticality Benchmarks Working Group [26]. The experiments should provide isotopic concentrations for the major (U-235, -236, -238, Pu-239, -240, -241)and minor (Pu-238, -242, Am-241, -243, Np-237) actinides and the major fission products (Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Nd-143, Nd-145, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Eu-153, Gd-155).
- Support for an additional validation and verification of the currently used codes and libraries focused for the WWER environment (SCALE (KENO, ORIGEN-S, etc.), MCNP, etc.) is required. The activity could be made under an international collaboration of the countries operating the WWERs. It should start for away-from-reactor codes (MCNP, KENO, etc.) and connected libraries as basic tools for the burnup credit computation.
- The validation of western codes for the WWER environment is necessary, because it has not been done and because of changed regulatory requirements. (In the Czech Republic a strict new directive of the Czech State Office on Nuclear Safety on the validation procedures and criteria for the acceptance of codes and libraries being used for nuclear safety licensing analyses has recently been issued. According to the directive, the codes/libraries developed/processed abroad must undergo an additional validation/verification for the WWER environment.)

Germany

- Improvements in depletion code validation require enlargements of the chemical assay data basis on actinides and fission products of interest. The fuel types to be considered are PWR uranium dioxide and MOX fuel as well as BWR uranium dioxide and MOX fuel. Accordingly, in addition to the "top fifteen" fission products already specified above (see data needs Czech Republic), data on the following isotopes are required: Cd-113, Cs-135, Nd-144, Nd-146, Nd-148, Nd-150, Sm-154, Gd-154, Gd-156, Gd-157, Gd-158, Gd-160.

- Improvements in criticality code validation require worth experiments, or critical or approach to critical experiments on spent PWR and BWR uranium dioxide and MOX fuel including variations in initial enrichment and initial burnable absorber distributions.
- Having no own national programme dealing with burnup credit Germany has to rely on international co-operations and data published in the open literature.

Hungary

In order to investigate the possible application of burnup credit in Hungary, there is a great need of reliable measured data on the isotopic composition of WWER-440 spent fuel with well-documented irradiation history of the assemblies. A data base containing the compositions together with the axial and radial position of the samples used in the measurements would be particularly useful.

Russian Federation

Russian Federation has common needs in critical experiments with depleted fuel and results of chemical and mass spectrometric analyses for actinides and fission products in materials concerned.

Slovak Republic

Development of a well characterized experimental database of WWER-440 spent fuel isotopic composition for validation purposes. It should contain the U, Pu and actinide isotopes.

Spain

There is no current experimental programme aimed to burnup credit application for criticality safety in Spain. The needs that can be identified refer to the few data available to validate the codes, both depletion and criticality, used for burnup credit applications. Hence, experimental work on the field of spent fuel assay and spent fuel (or individual nuclides) reactivity worth experiments are identified as a need. More accurate validation of the codes would allow for the deletion of some conservative margins that have to be currently retained.

USA

The USA is currently pursuing actinide-only burnup credit for transportation, for which current experimental data is assumed to be sufficient Similarly, pursuit of burnup credit for disposal criticality is relying on reactor critical data, and no new experiments are believed to be necessary. If burnup credit for transportation beyond actinide-only is pursued, then additional chemical assays of specific fission products are anticipated along with experimental data for benchmarking criticality codes. For critical benchmark data acquisition, worth experiments for specific fission products of interest are anticipated.

11. CONCLUSIONS AND RECOMMENDATIONS

11.1. Conclusions

The use of burnup credit to demonstrate criticality safety for spent fuel management activities has gained worldwide interest. A generalized interest in the different countries has been noticed and demonstrated by the high number of ongoing activities and by the country's perspectives given in the report. Burnup credit is recognized as a means of increasing efficiency for storage, transportation, reprocessing, and disposal of spent fuel. By accounting for the decrease in reactivity of spent nuclear fuel criticality safety can be assured without relying on the excessive conservatism that is necessary under current practices which assume the fuel to be fresh, unused or at its maximum reactivity. The use of burnup credit holds the promise of increased system efficiencies by allowing closer packing of fuel to increase capacities of storage pools and casks, transport casks and waste packages. It can be used to reduce reliance on external criticality controls (e.g., neutron absorber plates). In the case of reprocessing, burnup credit can be used to accommodate increased initial fissile concentrations in current and future generations of fuel without modifying processing facilities.

A basic requirement when using burnup credit is that the safety margin against criticality should be sufficiently large to avoid the risk of a criticality accident. To assure the margin against criticality the reliability of the different code's ability to predict isotopic compositions and reactivity is most important. It is also necessary to have control of the uncertainties introduced by the fact that burnup is used. Such uncertainties are besides from the calculational uncertainties also uncertainties in declared burnup of the fuel and uncertainties that are introduced by variations in operational history of the fuel assemblies.

The regulatory status of burnup credit varies from country to country. However, the best estimate character of burnup credit cannot be completely claimed due to the difficulty to quantify the specific intervening uncertainties.

11.1.1. Spent fuel storage

Regulatory authorities in a number of countries have approved burnup credit for wet storage of spent BWR and PWR fuel. Although some restrictions apply to these approvals, they have generally been for actinide plus fission product methods. PWR fuel is the use where a higher level of burnup credit has been accepted, while much reduced levels are being used for BWR fuel due to several reasons. The few countries that use these reactor types, but haven't gotten burnup credit for wet storage, are engaged in development of regulatory review. Burnup credit for wet storage of BWR and PWR spent fuel appears to be the most successful application at this time.

The only country developing application of burnup credit to wet storage of spent MOX fuel at this time is the UK. For spent WWER fuel there is strong interest by Eastern European countries that have WWER reactors, but progress has been constrained by limited funds.

The activity and progress for dry storage applications trails that of wet storage. There has been limited success in this area. Since the use of dry storage systems is generally pursued when pool space is exhausted, its slower progress is not surprising.

11.1.2. Spent fuel transportation

For spent PWR fuel, an actinide only method has been approved in France for wet and dry transport. The use of burnup credit for wet and dry transport of spent PWR fuel is under regulatory review in the UK and US respectively. Although the US does not use wet transport of spent fuel its dry cask method could be applied to wet transport. Germany has an approval for dry transport of spent BWR fuel, and Swiss have accepted the French approval for dry transport of spent PWR fuel in Switzerland.

There appears to be considerable interest in using burnup credit for transport of spent fuel. Many countries are looking beyond the actinide only approaches and plan to pursue actinide plus fission product methodologies. The two main reasons for using burnup credit for transport are continued use of existing casks under an environment of increasing initial fissile concentration for fuel designs, and the desire to design new high capacity cask systems.

11.1.3. Spent fuel reprocessing

Reprocessing activities are currently underway in France and the UK for spent BWR and PWR fuel. France already has approval for burnup credit for PWR activities and regulatory review is being conducted for BWR applications. The UK has development efforts underway for both BWRs and PWRs, and is also developing a program for MOX fuel. Russian Federation is considering initiation of burnup credit for WWER reprocessing.

11.1.4. Spent fuel disposal

Although most countries will eventually need some type of disposal programme, only the US's is in an advanced stage. The US has determined that burnup credit should be used for both BWR and PWR spent fuel for a successful repository design. The Department of Energy has the responsibility for developing the US repository. The Department has initiated preliminary discussions with the US regulatory authorities (NRC) on methods of applying burnup credit as part of criticality control for long term disposal. A Topical Report on the subject will be submitted to the NRC during 1998.

11.2. Recommendations

The Advisory Group is convinced that due to economic considerations actinide plus fission product burnup credit methodologies, which are presently restricted to PWR and RBMK wet storage systems with soluble neutron absorber in the coolant, will be applied more and more to other spent fuel management systems and will gain, therefore, increasing importance in future. The Advisory Group makes the following comments and recommendations, therefore:

- The Advisory Group recognizes that data is needed in various degrees to support burnup credit activities for the four types of fuel (BWR, PWR, MOX, and WWER) and applications (storage, transport, reprocessing, and disposal) considered in this report. Some data development activities are already underway, and should be continued. The data is necessary to support benchmarking activities.
- The Advisory Group recommends that countries and organizations needing specific categories of data should be encouraged to co-operate to the extent possible to share experiences, costs and results of experimental work. Countries that conduct their own data development programmes, should consider making such data available to others through contractual or co-operative arrangements. Whenever possible, data developed for burnup credit should be published in the open literature.
- The Advisory Group recommends that efforts should also be made to determine the applicability of data exchange between fuel types and applications. There may be some regions of applicability where overlaps may be present. Data collection activities should be designed to identify and assess the feasibility of such interchanges.
- The Advisory Group recommends that the efforts should be focused on attaining to well-established standards for verification of actinide plus fission product burnup credit methodologies. These standards should compare for quality with the standards given in the OECD/NEA International Handbook of Evaluated Criticality Safety Benchmark Experiments [21] and should ultimately include the following fission products: Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cd-113, Cs-133, Cs-135, Nd-143, Nd-144, Nd-145, Nd-146, Nd-148, Nd-150, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Sm-154, Eu-153, Gd-154, Gd-155, Gd-156, Gd-157, Gd-158, Gd-160.
- The Advisory Group recommends to assess the precision of different calculational tools and to develop methods for managing the uncertainties introduced when burnup is used as reactivity control. In this context, the Advisory Group recommends to initiate an activity for comparing different eastern and western calculational tools (for depletion and or criticality) and for developing interfaces where and when required.
- The Advisory Group has observed that the uncertainties inherent to the quantitative burnup verification measurement methods presently available should be reduced by significant amounts. The Group recommends, therefore, to draw up programmes for improving the burnup verification measurement methods presently used or developing new measurement methods if required.
- The Advisory Group has observed that axial burnup shapes may strongly effect burnup credit for spent fuel management systems. If a quantitative burnup verification measurement is required then the reactivity effect of the axial burnup shape on the management system of interest has to be evaluated unless this effect is either included a priori in the loading curve / criterion of the system or covered by applying a conservatively estimated safety factor

to the measured burnup. Accordingly, the Advisory Group recommends to draw up a programme for developing a method / device that combines on-line the measurement of the axial shape and the estimation of the reactivity effect of this shape on the system of interest and makes the decision whether this reactivity effect is compatible with the loading curve / criterion of the system or not.

- Radial burnup distributions are of minor importance. The Advisory Group recognizes however that it is not possible at the present time to come to a final decision whether radial burnup distributions are always enveloped - with respect to reactivity - by radially averaged (uniform) burnup distributions or not. The Group recommends, therefore, to make a systematic study of the reactivity effect of radial burnup distributions.

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BURNUP CREDIT IMPLEMENTATION IN WWER SPENT FUEL MANAGEMENT SYSTEMS: STATUS AND FUTURE ASPECTS

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Abstract

This paper describes the motivation for possible burnup credit implementation in WWER spent fuel management systems in Bulgaria. The activities being done are described, namely: the development and verification of a 3D few-group diffusion burnup model; the application of the KORIGEN code for evaluation of WWER fuel nuclear inventory during reactor core lifetime and after spent fuel discharge; using the SCALE modular system (PC Version 4.1) for criticality safety analyses of spent fuel storage facilities. Future plans, involving such important tasks as validation and verification of computer systems and libraries for WWER burnup credit analysis are shown.

1. INTRODUCTION

The storage of the spent nuclear fuel, irradiated during the operation of the WWER-440 and WWER-1000 reactors is a problem of primary importance. The solution can provide further successful and safe operation of the NPP "Kozloduy", as well as the development of nuclear engineering in Bulgaria [1]. Since 1988, spent fuel has been stored in our country. The away-from reactor basin (AFR) for intermediate wet storage is designed to store WWER-440 and WWER-1000 spent fuel, but there are some problems with the storage licensing because of enhanced safety demands. At the moment (1994), there are about 2000 spent fuel assemblies stored in a special water pool (AFR) at the NPP site and about 1000 fuel assemblies in water pools in the reactor hall [1]. The Bulgarian authorities attend to apply the dry long-term storage technology and for that purpose organize marketing competition between different firms. As a result of this situation an implementation of burnup credit in both wet and dry storage facilities will be necessary and useful.

2. BURNUP CREDIT IMPLEMENTATION: STATUS OF THE PROGRAMME DEVELOPMENT

Recently the critically safety studies of spent fuel storage systems are based mainly on calculations with fresh fuel assemblies. This results in unnecessary large criticality safety margins for WWER spent fuel with typical burnup. The upgrading of safety and economy usage of nuclear fuel needs an evaluation of the criticality safety margins for spent fuel management systems on the basis of burnup credit implementation.

For these reasons we assess the importance and necessity to develop Bulgarian national programme on the implementation of burnup credit in spent fuel management systems. But at the moment we have not got the full scope of necessary knowledge and tools (such as WWER oriented computer codes and data libraries) and need some support and exchange of information.

2.1. Activities being done

2.1.1. Development of a 3D few-group diffusion burnup model

An advanced 3D few-group diffusion burnup model in hexagonal geometry has been developed for fuel depletion analysis as well as for investigations of reactor transmutation characteristics. The model involves the determination of the three dimensional neutron flux and power distribution in reactor core by a coarse mesh diffusion code (HEXAB-3D code) and the solution of the nuclide kinetic equations by an analytical approach (BURNUP code). The model is verified on different fast breeder reactor benchmarks (SNR-300, BN-1600). Results from the verification, as well as the application of the model for neutronic design studies, are presented in Ref.[2].

2.1.2. Application of the KORIGEN code

With the aim to evaluate the fuel nuclear inventory during reactor core lifetime and after spent fuel discharge, the KORIGEN code has been implemented and applied [3]. Using this code, the fuel (U and Pu) and waste (minor actinides and fission products) characteristics, such as isotope and/or element concentrations, radioactivity, radiotoxicity, decay heat and radiation including (α,n)-neutrons in vitrified high-active waste for given initial composition, irradiation history and decay times can be estimate.

The KORIGEN code is an extension of the ORIGEN code [4], developed at the Research Center Karlsruhe (FZK) in Germany. Improvements are made in the calculational method and in the input nuclear data. The KORIGEN cross section libraries for Light Water Reactors (LWR) include the LWR libraries of ORIGEN-2. The burnup-dependent actinide cross sections and burnup averaged cross sections for typical German PWRs are generated with the Karlsruhe cell-burnup code system KARBUS [4], which includes the UK spectral code WIMS and a part of ORIGEN. The KORIGEN code is extensively validated against experimental data for European PWRs.

Using this code, some investigations are carried out [5] for nuclear inventory changes of WWER-440 fuel assemblies with a three-year core lifetime, an initial enrichment of 3.6% and a batch average burnup of about 33 MWd/tU. In the calculations use is made of: two one-group cross section sets for PWR neutron spectrum, reactor spectrum dependent burnup averaged ORNL cross sections for UO_2 fuel irradiated up to 33 GWd/tU, as well as burnup dependent FZK actinide cross sections for European PWRs.

Some important preliminary results for fuel nuclide concentrations (^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu), minor actinide concentrations (^{237}Np, ^{241}Am, ^{243}Am, ^{244}Cm etc.), fission product concentrations (^{99}Tc, ^{129}I, ^{135}Cs etc.) , as well as for their post-irradiated characteristics such as radioactivity, decay heat etc. have been obtained and shown in Ref.[5].

2.1.3. Using the SCALE modular system for criticality safety analyses

Criticality safety is fundamental to the design and operation of nuclear fuel storage facilities. The full range of normal operational conditions together with potential accident scenarios must be analyzed by a mixture of qualitative arguments, reference to safety guides and handbooks, and by detailed computer modeling in order to establish safe and efficient operating conditions.

For criticality safety analyses of spent fuel facilities the modular code system SCALE (PC Version 4.1) [6] has been implemented and applied. This version performs only problem dependent multigroup cross section processing, as well as criticality safety analyses. It involves the following functional modules:

- BONAMI performs resonance self-shielding calculation based on the Bondarenko method;
- NITAWAL-II is an updated version of the NITAWAL code that performs resonance self-shielding calculations using the Nordheim Integral Treatment. In NITAWAL-II, the self-shielding is made with reference to infinite dilution values. In addition NITAWAL-II has been modified to add an annulus model to allow processing of resonance nuclides in an annular region;
- XSDRNPM is a general-purpose one dimensional discrete-ordinates code, used in the SCALE system for several purposes: k-effective calculation, cross section collapsing, and shielding analysis;
- ICE is a cross section mixing code which will accept cross sections from an AMPX working library and produce mixed cross sections;
- KENO Va. is a three-dimensional Monte Carlo criticality program, developed for use in the SCALE system. KENO Va. offers versatile geometry capabilities including the array of arrays option, the holes option, and variable chords for hemicylinders and hemispheres;

- SCALE PC Version 4.1 includes the following data libraries: the 16 group Hansen-Roach cross section library and the 27-group ENDF/B-IV library. It is seen, that with this version of the SCALE we can not carry out the burnup credit calculations and implement it in criticality safety evaluation of spent fuel management system. For this purpose we need the improved PC Version of SCALE, or other code system, which can perform also spent fuel and high-level waste characteristics calculations.

With the aim to acquire a knowledge on the modular code system SCALE a series of test problems has been solved, involving different possibilities of the system, namely: criticality safety of test fuel casks with fuel located in triangular lattice, PWR fuel cask typical configuration, etc. Criticality safety studies of different type of PWR spent fuel cask models, as well as the simple spent fuel cask with hexagonal geometry have been performed using this version of the SCALE modular system [7].

2.2. Future work

The plans for the future involve the following important issues :

- implementation and application of the improved SCALE Version or other computer code systems with hexagonal geometry and spent fuel characteristics, suitable for spent fuel burnup credit calculations;
- criticality safety analysis of WWER spent fuel facilities models, using depleted fuel assemblies;
- verification of the chosen code system on the basis of WWER benchmark experiments carried out on the critical assemblies ;
- extensive problem validation of the code system applied for solving the WWER spent fuel problems, involving both using correct cross section libraries, suitable for WWER reactor type and code models validation.

The Bulgarian regulatory body (Committee for Use of the Atomic Energy for Peaceful Purposes) principally supports such investigations.

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STATUS OF THE DEVELOPMENT OF BURNUP CREDIT IN THE CZECH REPUBLIC

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Abstract

In the paper a possible intention of implementing burnup credit (BUC) in Czech spent fuel management systems is discussed. As a part of preparedness for BUC calculations the Nuclear Research Institute (NRI) at Řež has started studying burnup credit for WWER spent fuel. After the Czech Republic had joined the OECD/NEA/NSC Working Group on BUC Criticality Benchmarks a calculational BUC benchmark focused on spent WWER-440 fuel was specified in NRI and proposed the Eastern and Central European research community for the calculation.

1. INTRODUCTION

In the Czech Republic, no burnup credit has been taken into consideration in practice so far, but there are some indications of intention to use burnup credit:

- the leading Czech company (CEZ), which produces and distributes electricity, has admitted the possibility of using burnup credit for its interim storage container technology and therefore started an introductory discussion on BUC with the Czech State Office on Nuclear Safety;
- the NPP Dukovany would need to use BUC for part of its spent fuel pool at the reactor.

2. STATUS OF THE DEVELOPMENT OF BURNUP CREDIT STUDIES

The Nuclear Research Institute at Řež has started studying burnup credit for WWER spent fuel as part of preparedness the BUC calculations.

- A calculational burnup credit criticality benchmark for WWER 440 spent fuel has been formulated [1] (see Section 3), following the activities of the OECD/NEA Burnup Credit Criticality Benchmark Group. This Group attempts to produce state-of-the-art reports on evaluation procedures for PWR, BWR and MOX spent fuel system reactivities, taking into account the isotopic composition and non-uniform burnup of the fuel ([2 to 5]). The WWER benchmark mentioned above was proposed to the nuclear safety research community from Eastern and Central European Countries at the 6th Atomic Energy Research (AER) Symposium in 1996 in Finland and its calculation is in progress.
- An experimental burnup credit criticality benchmark (the REBUS experiment, the proposal of which was presented by T. Maldague of Belgonucleaire S.A., Belgium, at the 5th Meeting of the Burnup Credit Criticality Benchmarks Working Group, OECD/NEA 1996) is being prepared at the LR-0 research facility at the NRI at Řež.

Calculating BUC correctly depends mainly on the model, method and data used for the criticality calculation. While the two first sources of uncertainties can be quite fairly evaluated (and possibly decreased), the last one describes a 'quality' of

- the criticality calculation input data for the spent fuel isotopic concentrations (resulted from an isotopic inventory calculation, e.g. using the ORIGEN code, either under the SCALE code system or as a stand - alone module);
- the cross sections libraries for fission products (FP) and actinides used in the criticality calculations.

To find the uncertainty connected with the data, one needs to validate and verify the methods (used for inventory calculations) and libraries (used either for inventory calculations or criticality calculation) for the given application field (e.g. the WWER environment). The NRI BUC study to

date concludes that, with respect to the validation and verification of the methodology, there is a need for chemical assay data of WWER spent fuel and for an international collaboration:

- to take and analyze samples from selected WWER 440 spent fuel of a well defined irradiation history for a comparison with results of calculations of the proposed benchmarks, but relating to the 'well-defined' spent fuel (should be done in Russia);
- additional validation and verification of the currently used codes and libraries focused on the WWER environment (SCALE (KENO, ORIGEN-S) [6], MCNP) should be made under an international collaboration of the countries operating WWERs. It should start for away-from-reactor codes (MCNP, KENO) and libraries as basic tools for the BUC computation.

A research proposal on codes and libraries validation and verification for the WWER environment was submitted to the Czech State Office on Nuclear Safety in 1997 (also because a strict new directive of the regulatory body has recently been issued with respect to the validation procedures and criteria for the acceptance of codes and libraries used for its nuclear safety licensing analyses). Unfortunately, no funding sources have been found in the Czech Republic so far to support the project.

3. CALCULATIONAL BURNUP CREDIT BENCHMARK for WWER-440

3.1. Summary of a calculational burnup credit benchmark proposal [1]

The benchmark has been designed as a series of calculational benchmarks for WWER-440 spent fuel, similar to the benchmarks as developed by the OECD/NEA Burnup Credit Criticality Benchmark Group, Phases 1 and 2 ([2 to 5]) and calculated by its members. The benchmarks for WWER-440 spent fuel, entitled CB1, CB2, CB3, and CB4, should deal step by step with the main issues related to WWER-440 spent fuel analyses and their results should also be utilized for intercomparison of codes and libraries used for WWER applications. The benchmark data for CB1 and CB2 as well as characteristics of the fuel during the fuel cycles for CB2 were compiled from [7] taking into consideration the licensing documents for CASTOR 440/84 submitted to the Czech State Office on Nuclear Safety.

3.2. Calculational benchmark no. 1 (CB1)

As a first step of the proposed series of benchmark problem exercises, an infinite array of a simple WWER-440 pin cells with fresh and spent fuel rods has been described and analyzed (criticality calculation) with the burnup, cooling time and groups of selected nuclides as parameters. The nuclides selected in the benchmark were 6 major and 5 minor actinides and 15 major fission products (FP). This selection is based on the fact that these major actinides and FPs represent about 80% of the reactivity losses from all actinides and Fps. CB1 is similar to NEA/NSC Burnup Credit Criticality Benchmark-Phase I-A [2] studying effects of 7 major actinides and 15 major fission products upon criticality of a simple PWR spent fuel cell with fuel burnup of 30 and 40 GWd/t after 1 and 5 year(s) cooling time. Obtaining the CB1 results, calculations of the benchmarks CB2, CB3 and CB3 should be carried out. Up to now the CB1 (an eigenvalue calculation of a simple infinite hexagonal lattice of spent and fresh WWER-440 fuel rods) has been calculated by VUJE Trnava (Slovakia) and VTT Energy (Finland) both using KENO VI under SCALE 4.3, by VTT Energy (Finland) using CASMO4 and by NRI using MCNP4A. The resulting multiplication factors were compared [8] and the discussion of the results is still under way.

3.3. Calculational benchmark no. 2 (CB2)

This part of the benchmark should result in finding and a comparison of the ability of various code systems and data libraries to predict spent fuel isotopic concentrations using depletion analysis. The specification is similar to the previous case. Unfortunately, only an intercomparison of codes and data libraries is possible as there is no possibility of using a chemical assay - measurements of

isotopic concentrations of some concrete WWER spent fuel sample of a well-documented irradiation history. The explicit specification of this part of the whole benchmark was presented at the 7th AER Symposium on WWER Reactor Physics and Reactor Safety, Hornitz near Zittau, Germany, Sept 23-26,1997 [9] in spite of the fact that VTT Energy Finland had calculated and KKAB Berlin even presented its results (based on the whole benchmark specification [1]) already at Zittau. Results of the calculations coming from other workplaces are expected and will be summarized and compared one with another. Conclusions are likely to be discussed at the 8th AER Symposium in 1998.

3.4. Calculational benchmark no. 3 (CB3)

The effect of an axial burnup profile of WWER spent fuel on criticality ('end effect') will be studied in this benchmark problem of a laterally infinite array of WWER spent fuel rods. Cases will have various burnup and cooling time as parameters. CB3 will be specified by NRI and presented to the WWER research community in the near future. The specification requires to perform many calculations, the funding of which has not been resolved yet.

3.5. Calculational benchmark no. 4 (CB4)

Using some data of CB3 (composition), a realistic spent fuel cask configuration (probably CASTOR 440/84, the finite WWER spent fuel array of 84 spent WWER-440 fuel assemblies) will be calculated and BUC will be found. Various cases will be specified with parameters such as burnup and cooling time. CB4 will be specified by NRI when the CB3 results are obtained.

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COGEMA/TRANSNUCLEAIRE's EXPERIENCE WITH BURNUP CREDIT

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Abstract

Facing a continuous increase in the fuel enrichments, COGEMA and TRANSNUCLEAIRE have implemented step by step a burnup credit programme to improve the capacity of their equipment without major physical modification. Many authorizations have been granted by the French competent authority in wet storage, reprocessing and transport since 1981. As concerns transport, numerous authorizations have been validated by foreign competent authorities. Up to now, those authorizations are restricted to PWR Fuel type assemblies made of enriched uranium. The characterization of the irradiated fuel and the reactivity of the systems are evaluated by calculations performed with well qualified French codes developed by the CEA (French Atomic Energy Commission) : CESAR as a depletion code and APPOLO-MORET as a criticality code. The authorizations are based on the assurance that the burnup considered is met on the least irradiated part of the fuel assemblies. Besides, the most reactive configuration is calculated and the burnup credit is restricted to major actinides only. This conservative approach allows not to take credit for any axial profile.

On the operational side, the procedures have been reevaluated to avoid misloadings and a burnup verification is made before transport, storage and reprocessing. Depending on the level of burnup credit, it consists of a qualitative (go/no-go) verification or of a quantitative measurement. Thus the use of burnup credit is now a common practice in France and Germany and new improvements are still in progress: extended qualifications of the codes are made to enable the use of six selected fission products in the criticality evaluations.

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, TRANSNUCLEAIRE is in charge, on behalf of COGEMA, of most of the transports of irradiated fuels to the La Hague reprocessing plant.

TRANSNUCLEAIRE operates a wide range of transport packages with capacities from 3 PWR 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. Figure 1 shows the TN 12/2 cask utilized for the transport of 12 PWR or 32 BWR fuel assemblies with a maximum allowed heat content of 76 kW. For longer cooling times, capacities of 37 PWR and 97 BWR fuel assemblies have been achieved for transport/interim storage casks.

Today, TRANSNUCLEAIRE carries out roughly 300 transports of spent fuel casks per year, to the La Hague reprocessing plants. The casks are then unloaded and the fuel assemblies stored into the pools prior to reprocessing. The total reprocessing capability of the two facilities located in La Hague is 1,600 t of heavy metal per year. The two facilities operated are: UP2 and UP3. They have today reprocessed numerous type of fuel assemblies including PWR, BWR, UOX and MOX, and fuels from fast breeder reactors.

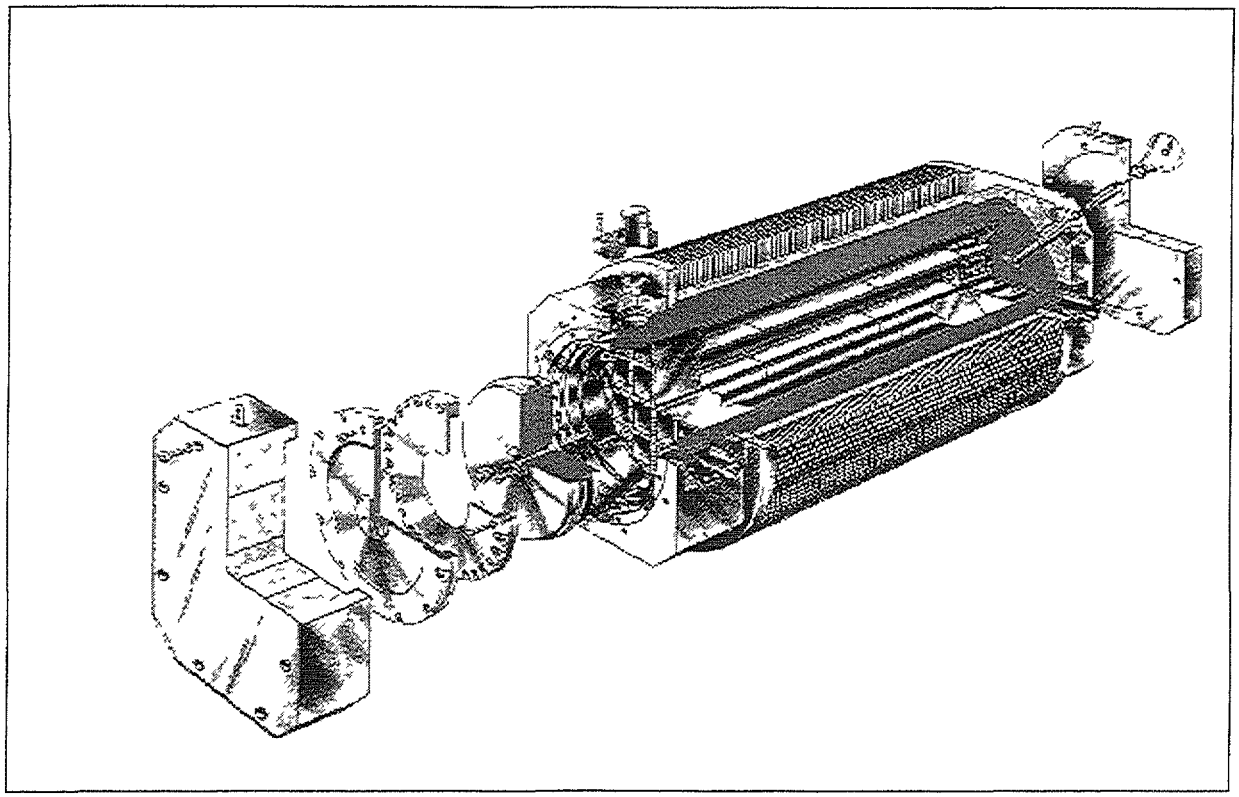

FIG. 1 TN 12/2 Cask

During transport, subcriticality is controlled by the fuel support frame also know as " basket". It is made of materials containing neutron absorbers, usually boronated aluminum or stainless steel and ensures a well defined geometry. The unloading in La Hague takes place in dry or wet conditions. During the unloading operation and the storage in the pool, the fuel assemblies are placed into a special designed basket with boronated stainless steel shrouds. The water of the pool is not boronated.

UTILIZATION OF BURNUP CREDIT

Facing a continuous increase in the enrichment of the fuel assemblies, COGEMA and TRANSNUCLEAIRE has been interested in the use of burnup credit. The experience acquired now is more than 15 years. Shown in the Table I are the different authorizations that have been granted by the French competent authority to COGEMA, for using burnup credit in wet storage and reprocessing For transport casks, numerous authorizations have been granted in France and validated in foreign countries since the first approval which was obtained in 1987 for 16x16 PWR fuel assemblies loaded in TN 13/1 or TN 13/2 casks.

TABLE I. CRITICALITY SAFETY AND BURNUP CREDIT

Year	Facility	Initial enrichment E_i of the fuel assemblies
1981	Reprocessing UP2 400 .	$E_i < 4\ 20\ \%\ ^{235}U$
1987	Storage UP2	$E_i < 3.55\ \%\ ^{235}U$
1990	Reprocessing UP3 .	$E_i < 3.70\ \%\ ^{235}U$
1993	Storage UP2	$E_i < 4.00\ \%\ ^{235}U$

The different authorizations granted in France, for wet storage (in La Hague), reprocessing and transportation, are up to now limited to PWR fuel assemblies consisting of enriched uranium oxide. 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 analyses, 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 Table II illustrates the different authorizations that have been granted to transport 16x16 PWR fuel type in the TN 13/2 cask.

TABLE II. GRANTED AUTHORIZATIONS

TRANSPORT CASK	FUEL ASSEMBLY TYPE	FUEL ENRICHMENT	BURNUP CREDIT	BURNUP VERIFICATION
TN 13/2	UO2 16x16	3.30 %	0 MW·d/tU	NONE
TN 13/2	UO2 16x16	3.3 < E < 3.55 %	3,200 MW·d/tU (<1 cycle)	QUALITATIVE (GO/NO-GO) BURNUP VERIFICATION
TN 13/2	UO2 16x16	3.5 < E < 4 %	12,000 MW·d/tU	QUANTITATIVE BURNUP MEASUREMENT

For storage the same policy applies: enrichment limits are defined for three fuel assembly types, characterized by the dimensions of the cross section (BWR, PWR 214 x 214, PWR 230 x 230).

For reprocessing, only one enrichment limit is given. If the fuel assembly's enrichment is less than this value, no mass limit is required in the dissolver. Else, two cases are possible: either a mass limit is required, depending on the burnup and of the fuel enrichment, or no mass limit is required but then a burnup verification has to be made.

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 (French Atomic Energy Commission). No burnup profile is taken into account to calculate the composition of the fuel after irradiation. The conservatism of this assumption relies on the fact that the burnup is measured on the least irradiated part of the fuel assembly, i.e. the last 50 cm. The depletion code used is CESAR, while criticality calculations are performed with APOLLO-MORET.

CESAR [1]
CESAR is a depletion code dedicated to the characterization of spent fuels UOX (PWR and BWR) and MOX. 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. It enables to evaluate 40 heavy nuclides, 204 fission products and 2 activation products. It uses a Runge-

Kutta method for calculations during irradiation and a matrix type method for calculations between cycles and during cooling time.

APOLLO-MORET [2,3]

APOLLO 1 is a 99 group deterministic criticality code with CEA 86 library. MORET III is a Monte Carlo code with 16 groups. For actinide only burnup credit, these codes are qualified with 102 benchmark experiments. These experiments use fuel composition equivalent to a fuel enriched to 4.5 % and 37.5 GWd/t burnup without fission products. These experiments simulate transport, storage and dissolution configurations.

For the criticality calculation, only major actinides are considered (^{235}U, ^{236}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu). The fission products are neglected. The most reactive configuration is described and considers fuel assemblies with no or a few missing pins. 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.

FIRST STEP: QUALITATIVE (GO/NO-GO) BURNUP CHECK

The approval has been obtained with the following requirements:

- 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.
- 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 on cycle.
- The irradiation status of each fuel assembly must be checked by a qualitative go/no go physical measurement in the reactor pool just before cask loading. These measurement have to be in accordance with the plant quality assurance policy.
- In addition, safety authorities require COGEMA and TRANSNUCLEAIRE to demonstrate the reliability of their fuel identification procedures.

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 of the two following devices :

- The PYTHON device, developed by CEA-CADARACHE, evaluates the burnup of the fuel assembly on the basis of a passive neutron measurement and of a total gamma irradiation profile measurement. The whole active length of the fuel is measured. This device has been validated by the French competent authority.
- The FORK detector, developed by Los Alamos, uses the Cm 244 neutron emissions to measure the fluence observed by the fuel. By use of the enrichment, this measure can be converted to burnup. This detector has been adapted by the CEN-MOL laboratory (Belgium), and validated by the French competent authority.

For storage and reprocessing the same principle applies. Besides, a verification of the data sent by the reactor is performed.

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:

- Segregation of fresh or low irradiated fuel assemblies in the pool.
- 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.
- Training of the TRANSNUCLEAIRE's operators in charge of the fuel identification on behalf of COGEMA.
- Increasing the awareness of the reactor fuel handling staff to fuel identification prior to transport.
- Introduction of redundancy on fuel identification with low dependence level between operators.
- Written records of the checks carried out on each action.

NEW DEVELOPMENT

Thus, the use of burnup credit has become a common practice well mastered by the different organizations involved, from the reactor to the reprocessing in La Hague. 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 composition 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 2001, 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, ^{243}Nd, ^{149}Sm, ^{152}Sm, ^{155}Gd. This will enable COGEMA and TRANSNUCLEAIRE to take credit for these fission products in their criticality evaluations and thus improve the evaluation of the reactivity of the spent fuels.

CONCLUSION

Facing a continuous increase in the fuel enrichments, COGEMA and TRANSNUCLEAIRE have implemented step by step a burnup credit programme to improve the capacity of their equipment without major physical modification. Many authorizations have been granted by the French competent authority in wet storage, reprocessing and transport since 1981. 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.

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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

The paper describes the experience gained in Germany in applying burnup credit methodologies to wet storage and dry transport systems of spent LWR fuel. It gives a survey of the levels of burnup credit presently used or intended to be used, the regulatory status and future developments planned, the codes used for performing depletion and criticality calculations, the methods applied to verification of these codes, and the methods used to treat parameters specific of burnup credit. In particular it is shown that the effect of axial burnup profiles on wet PWR storage designs based on burnup credit varies from fuel type to fuel type. For wet BWR storage systems the method of estimating a loading curve is described which provides for a given BWR fuel assembly design the minimum required initial burnable absorber content as a function of the initial enrichment of the fuel.

1. INTRODUCTION

Implementation of burnup credit in spent fuel management systems operated in Germany is required for two reasons. It is intended to:

- Increase initial enrichments by considerable amounts without scrapping existing systems;
- Reduce the frequency of spent fuel shipments to the minimum unavoidable.

The spent fuel management systems concerned are:

- Wet storage of PWR and BWR fuel (and, possibly, of MOX fuel);
- Dry transport of PWR and BWR fuel (and, possibly, of MOX and WWER fuel);
- Disposal (final storage) of PWR and BWR fuel (and, possibly, of MOX and WWER fuel).

Burnup credit for disposal might be required if burnup credit is taken for dry transport, but at the present moment there is no thinking about applying burnup credit to final storage. Applying burnup credit for WWER fuel might be inefficient because of the number of irradiated WWER fuel assemblies presently stored in Germany and because of the fact that there might be serious difficulties in describing axial burnup distributions in criticality safety analysis of spent WWER fuel assemblies. Applying burnup credit for MOX fuel might be inefficient because of the physics of the plutonium isotopes:

- The reactivity of a dry and, therefore, fast MOX system is determined by the total plutonium content of the system;
- As illustrated in Figure 1, in a wet storage system the reactivity changes of MOX and UO_2 fuels with increasing burnup are significantly different.

The attention is, therefore, mainly focused on wet storage and dry transport of PWR and BWR fuel in the following.

2. REGULATORY STATUS AND ACTIVITIES

Higher initial enrichments and lower frequencies of spent fuel shipments are in compliance with the objectives of the relevant German laws and regulations establishing that spent fuel management systems have to be designed and operated in such a way that worker and public exposure as well as off-normal or accident risks are reduced to a minimum, and that the design and the safety evaluation of these systems have to be consistent with established developments in science and technology.

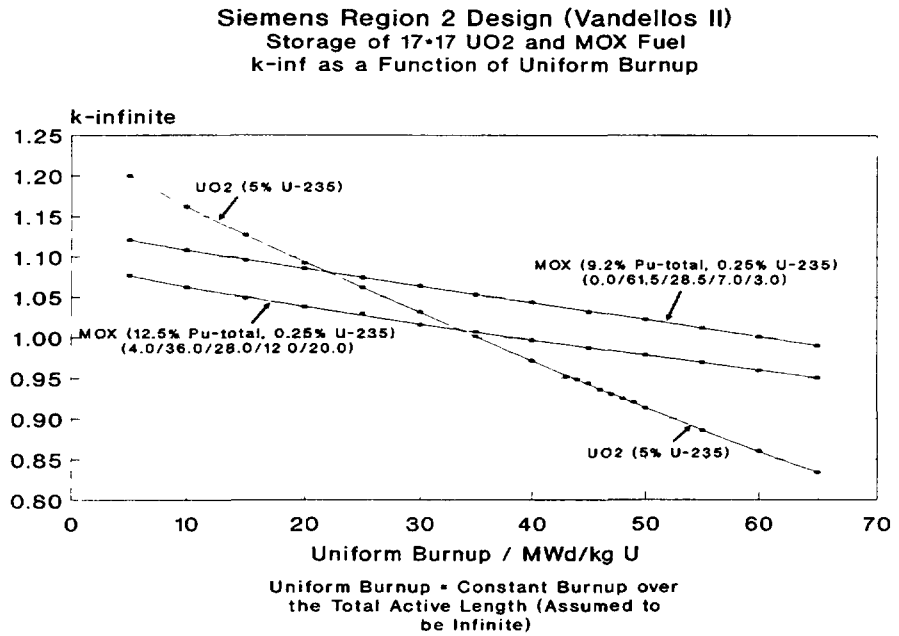

FIG. 1. Neutron multiplication of a wet storage system for loadings with spent PWR UO_2 fuel assemblies having an initial enrichment of 5 wt.% ^{235}U and loadings with spent PWR MOX fuel assemblies having different total plutonium contents and different isotopic vectors.

2.1. Wet storage of LWR fuel

The criticality safety requirements for wet storage of LWR fuel are laid down in the standard KTA 3602 [1]. This code gives permission to take credit for the initial presence of burnable poisons within the fuel, prohibits to consider the presence of soluble boron in PWR storage pools in the evaluation of the normal operation conditions, does not prohibit to take credit for burnup, but requires to give reasons for deviating from the fresh fuel assumption if burnup credit is employed.

To provide obligatory guidelines for justifying the use of burnup credit the Criticality Safety Committee of the German society of standardization - *Deutsches Institut für Normung (DIN)* - is presently working out a criticality safety code for burnup credit in wet storage. This code will probably establish:

- criticality safety criteria being applicable to burnup credit;
- requirements for evaluating parameters characteristic of burnup credit (e.g., axial profiles, irradiation histories);
- methods acceptable for verification of fuel selection and loading.

2.2. Dry transport of LWR fuel

Licensing evaluations of dry transport systems are based on the application of the IAEA regulations Safety Series No. 6.

3. CURRENT AND INTENDED LEVELS OF BURNUP CREDIT

3.1. Wet Storage of PWR Fuel

PWR spent fuel storage racks developed by *Siemens KWU* on the basis of burnup credit were - or will be - delivered to foreign countries, namely to Spain (all PWR plants), Republic of Korea (Kori 3), South Africa (Koeberg 1+2) and Brazil (Angra 2). The design of all of these storage racks is based on the use of net fissile content plus actinide absorbers plus fission products (cf. Table I). This level of burnup credit is required for PWR storage racks for economic reasons.

TABLE I: LEVEL OF BURNUP CREDIT IN WET STORAGE OF PWR FUEL
(SEE REF. [2] FOR COMPARISON)

Actinides	Fission Products		
^{235}U	^{95}Mo	^{144}Ce	^{149}Sm
^{236}U	^{99}Tc	^{143}Nd	^{150}Sm
^{238}U	^{103}Rh	^{144}Nd	^{151}Sm
^{237}Np	^{113}Cd	^{145}Nd	^{152}Sm
^{239}Pu	^{131}Xe	^{146}Nd	^{154}Sm
^{240}Pu	^{133}Xe	^{148}Nd	^{153}Eu
^{241}Pu	^{133}Cs	^{150}Nd	^{154}Eu
^{242}Pu	^{134}Cs	^{147}Pm	^{155}Eu
^{243}Am	^{135}Cs	^{155}Gd	
	^{148m}Pm	^{156}Gd	
	^{148}Pm	^{157}Gd	
	^{149}Pm		

3.2. Wet storage of BWR fuel

The spent BWR fuel storage racks developed by *Siemens KWU* for German BWR plants and for the Spanish plant Santa Maria de Garoña are based on a reactivity equivalence concept which provides for the maximum reactivity point in the fuel assembly's lifetime the minimum initial burnable absorber content required for a given fuel assembly design at given average initial enrichment, cf. Figure 2.

To get the reactivity equivalence curve (loading curve) shown in Figure 2 the neutron multiplication of the storage design has to be calculated - for the maximum reactivity point in the fuel assembly's lifetime at given average initial enrichment - as a function of the initial burnable absorber content (cf. Figure 3), and to obtain the maximum reactivity point at given average initial enrichment and given initial burnable absorber content the neutron multiplication of the storage design has to be calculated as a function of fuel burnup (cf. Figure 4). In all of these calculations one has to take account of the parameters (e.g., void history) affecting the maximum reactivity point.

Accordingly, the BWR storage racks based on the reactivity equivalence concept Figure 2 are designed to accommodate BWR fuel at its maximum reactivity point. In other words, taking credit for burnup of BWR fuel is actually taking credit for the initial presence of burnable absorber. This concept is based on the burnup credit level given in Table II.

Due to the fact that the problem of describing BWR axial burnup shapes in criticality safety analysis is unsolved at the present (under economic aspects), it is not intended to take credit for burnups higher than the one which refers to the maximum reactivity point.

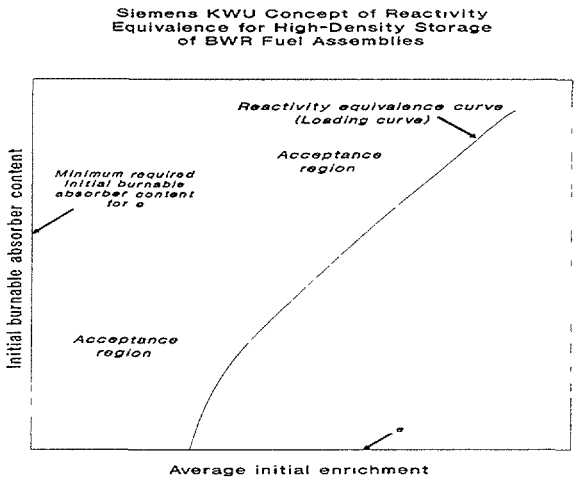

FIG. 2. Reactivity equivalence concept applied to wet storage of BWR fuel assemblies having attained their maximum reactivity.

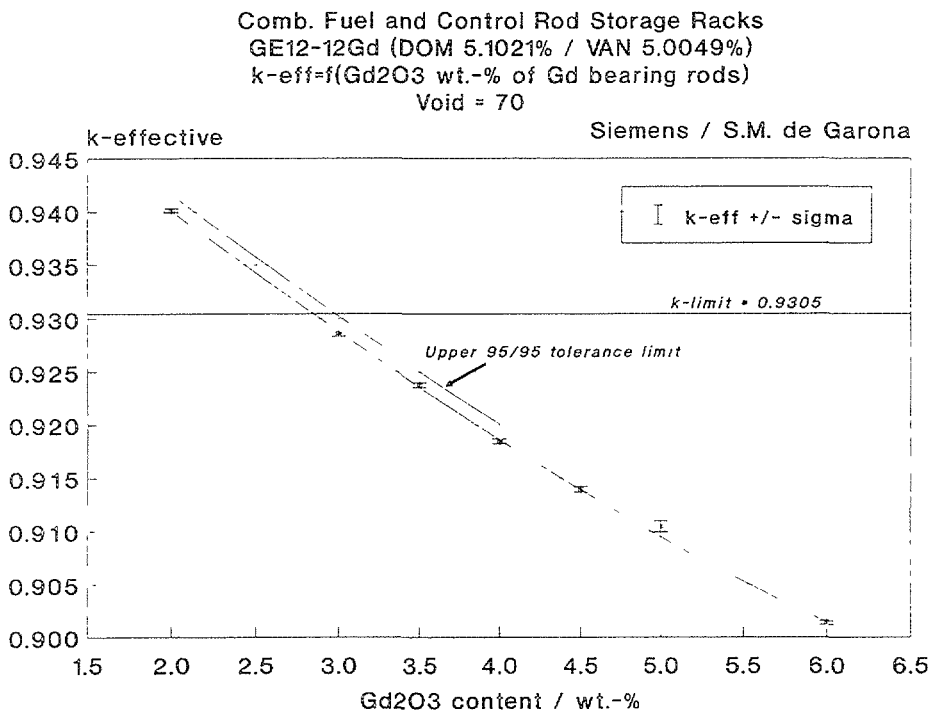

FIG. 3. Illustration of estimating the loading curve shown in Figure 2: The intersection of the upper 95%/95% tolerance limit of the neutron multiplication as a function of the initial burnable absorber content with the maximum permissible neutron multiplication k_{limit} provides one point of the loading curve Fig. 2.

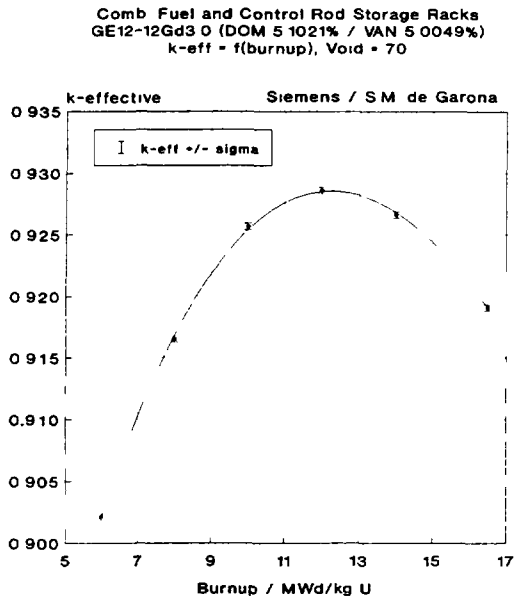

FIG. 4. Illustration of estimating the neutron multiplication as a function of the initial burnable absorber content: The maximum of the curve representing the neutron multiplication as a function of burnup provides one of the $k_{eff} \pm \sigma$ bars shown in Fig. 3 .

TABLE II. LEVEL OF BURNUP CREDIT IN WET STORAGE OF BWR FUEL

Actinides	Fission Products	Initial Burnable Absorber
^{234}U	^{133}Cs	^{154}Gd
^{235}U	^{135}Cs	^{155}Gd
^{236}U	^{143}Nd	^{156}Gd
^{238}U	^{145}Nd	^{157}Gd
^{237}Np	^{149}Sm	^{158}Gd
^{238}Pu	^{150}Sm	
^{239}Pu	^{152}Sm	
^{240}Pu	^{153}Eu	
^{241}Pu		
^{242}Pu		
^{241}Am		
^{242}Am		
^{243}Am		

3.3. Dry transport of LWR fuel

The standard casks used for shipping spent LWR fuel are the CASTOR casks developed by the *Gesellschaft fur Nuklear-Behälter (GNB)*, Essen. 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:

- the fresh fuel approach for initial enrichments up to 4.2 wt.% ^{235}U;
- "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); and that
- this fuel has a minimum average discharge burnup of 5 MW·d/kg U (this is ensured through the analysis of each fuel assembly's exposure history).

GNB intends to apply this "uranium plus plutonium isotopes only" burnup credit concept to other casks (e.g., the CASTOR V/19 cask used for spent PWR shipping) in order to:
- achieve firstly an increase of the respective maximum permissible initial enrichments by about 0.5 wt.% ^{235}U; and to
- get finally the license to consider the actual discharge burnups of the fuel to be loaded.

4. CALCULATION CODES

4.1. Depletion codes

The standards to be applied to depletion codes and to verifications of such codes are laid down in the safety code KTA 3101.2 [3].

4.1.1. PWR UO2 and MOX fuels

Depletion calculations for PWR UO2 and MOX fuels are performed with the aid of the *Siemens KWU* standard core design procedure SAV90 [4]. This procedure is used for spectrum and nodal reactor calculations as well as for pinwise reactor analysis. Among other data and features this procedure provides:
- the isotopic inventory as a function of burnup;
- axial power and burnup profiles.

The SAV90 procedure is based on broad empirical verification and validation. The experience with this procedure has been accumulated to about 200 first core and reload designs including KWU, Westinghouse and Framatome PWRs. The quality of prediction relies on statistics on the differences between measurement and calculation. To obtain these statistics the following sources of experimental information were exploited:
- 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
- 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

  In addition to these physics measurements conducted at nuclear power reactors, the nuclide densities of irradiated fuel were analyzed:
- Isotopic inventory of spent fuel.

4.1.2. BWR UO2 and MOX fuels

The isotopic inventories of irradiated BWR UO2 and MOX fuels are calculated with the aid of the code system MICBURN/CASMO [5-6]. The quality of prediction based on the published benchmarks can be supplemented by statistics derived from:

- comparisons to other BWR spectrum depletion codes (e.g. TGBLA, [7]); and from
- experience gained with the aid of off-line core simulation code systems such as CASMO/MICROBURN used by *Siemens KWU* and with the aid of on-line core simulator code systems such as the *Siemens KWU* code FNR-K:
  Comparisons of measured and calculated tip-signals result in an uncertainty of the calculated burnup. This uncertainty can be used to correct the calculated isotopic densities in an enveloping manner.

4.1.3. WWER fuels

Depletion calculations for WWER fuels can be performed with the aid of a code system developed by the *Kraftwerks- und Anlagenbau (K.A.B.) AG*, Berlin-Marzahn. This code system consisting of the modules NESSEL-4, NUKO, PYTHIA/TRAPEZ, DERAB is used for spectrum and nodal reactor calculations as well as for pinwise reactor analysis. Among other data and features this code system provides:

- the isotopic inventory as a function of burnup;
- axial power and burnup profiles.

The verification of the code system is mainly based on:

- Analysis of normal power operation measurement data which have been accumulated to 80 reactor years of German and foreign WWER plants;
- Analysis of test track measurement data;
- Analysis of actinide densities in spent WWER-440 fuel.

4.2. Criticality codes

The standards to be applied to criticality codes and to verifications of such codes are laid down in the safety standard DIN 25478 [8]. The criticality codes mainly employed in Germany are:

- the criticality portion of the SCALE package [9]; and
- the MCNP code [10].

Germany is represented in the Criticality Safety Benchmark Group at OECD NEA by the following institutions:

- Institut für Kernenergetik und Energiesysteme (IKE), University of Stuttgart;
- Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Garching,; and
- Bundesamt für Strahlenschutz (BfS), Salzgitter.

These institutions participate in the analytic activities of the benchmark group. It should be noted that the developers of the SCALE package have made extensive contributions to the burnup credit validation of the SCALE package by analyzing PWR reactor critical configurations [2].

5. PWR AXIAL BURNUP PROFILES

The *Siemens KWU* method used in the recent years for modeling PWR axial burnup shapes is illustrated in Figure 5. The real distribution is modeled by a step distribution. The number of steps is a free parameter because neighboring steps with differences smaller than a given threshold are combined to larger steps.

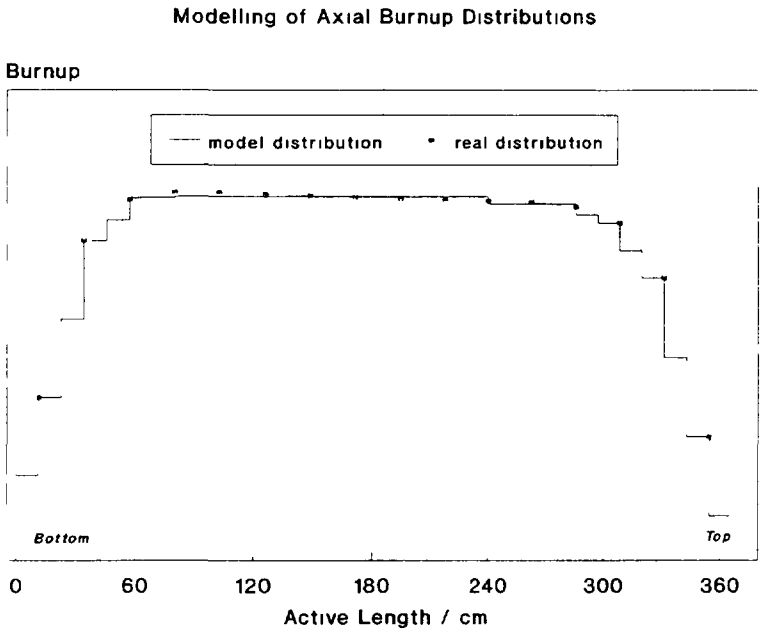

FIG 5. Typical PWR axial burnup distribution

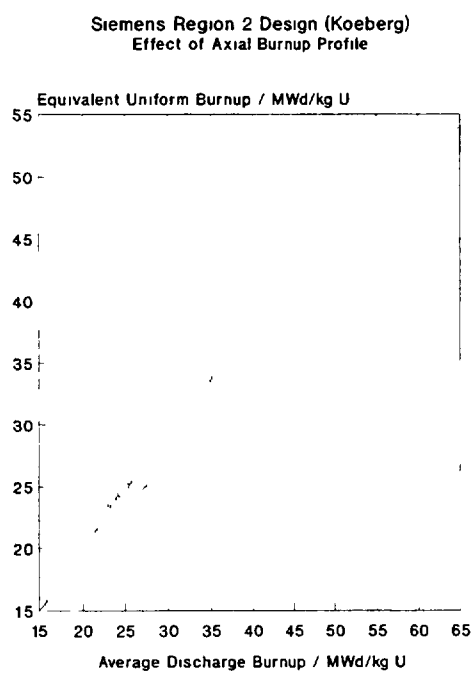

*FIG 6 Wet storage of spent 17*17 PWR fuel assemblies: Effect of axial burnup shapes on burnup credit.*

Results obtained with the aid of this step distribution model for a PWR wet storage case are shown in Figure 6. Each of the small bars shown in that figure represent an analyzed axial shape based on measured data delivered from the nuclear power plant under examination. "Uniform burnup" means constant burnup over the full active length of the fuel assemblies. The "equivalent uniform burnup" is the uniform burnup which has the same neutron multiplication as the analyzed axial shape characterized in Fig. 6 by its average discharge burnup. If the equivalent uniform burnup obtained for an axial shape is less than the average discharge burnup referring to this shape, then this shape has a neutron multiplication higher than that one which would be obtained with a uniform distribution of the average discharge burnup. So, in the case shown in Figure 6 the effect of axial burnup shapes on the burnup credit is significant.

However, due to different reactor steering strategies and core geometries, this effect might differ from plant to plant. Figure 7 shows an other PWR wet storage case. In that case the axial shapes have no effect on the burnup credit. Therefore, the only conclusion which can be drawn is that it is essential to analyze axial burnup shapes specific of the plant under examination.

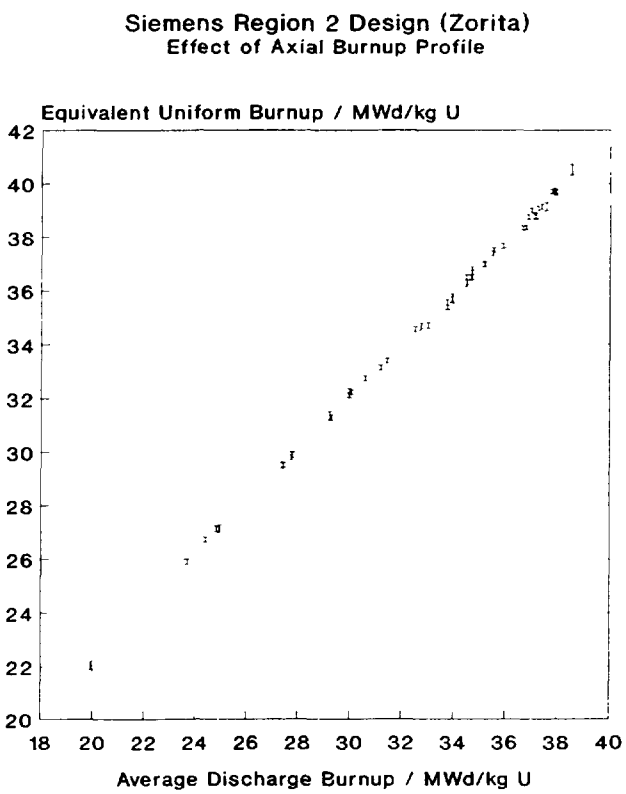

*FIG. 7. Wet storage of spent 14*14 PWR fuel assemblies of the José Cabrera (Zorita) type: In this case the axial burnup shapes analyzed show no effect on burnup credit.*

6. CRITICALITY SAFETY CRITERION

As already told in Section 2.1, a criticality safety standard for burnup credit in wet storage is presently worked out in Germany. This standard will probably establish the following safety criterion:

$$k + \lambda\sigma \leq (1 - \Delta kS) - \Delta kI - \Delta kB - \Delta kM - \Delta kT$$

$k + \lambda\sigma$	:=	upper 95%/95% tolerance limit of the evaluated neutron multiplication
ΔkS	:=	margin of subcriticality:
ΔkS	$\geq$	0.02 for accident cases which are radiological not relevant and which have very small probabilities of occurrence, otherwise $\Delta kS = 0.05$.
ΔkI	:=	uncertainty related to the depletion calculations applied
ΔkB	:=	bias of the criticality code applied
ΔkM	:=	uncertainty related to the effect of axial burnup profiles
ΔkT	:=	uncertainties arising from manufacturing tolerances of the fuel management system.

All uncertainties have to be expressed at the 95%/95% tolerance limit because they are statistics, i.e. random variables defined by probability distributions.

Let's take the term ΔkI for example. It was already stated (cp. Section 4.1.1, e.g.) that the quality of the depletion calculation predictions is based on statistics on differences between measurement and calculation. Figure 8 shows as an example one of the SAV90 statistics on critical boron concentrations. (1 ppm corresponds to about 10-4 in k_∞. The amount of the lower 95%/95% tolerance limit is about 40 ppm).

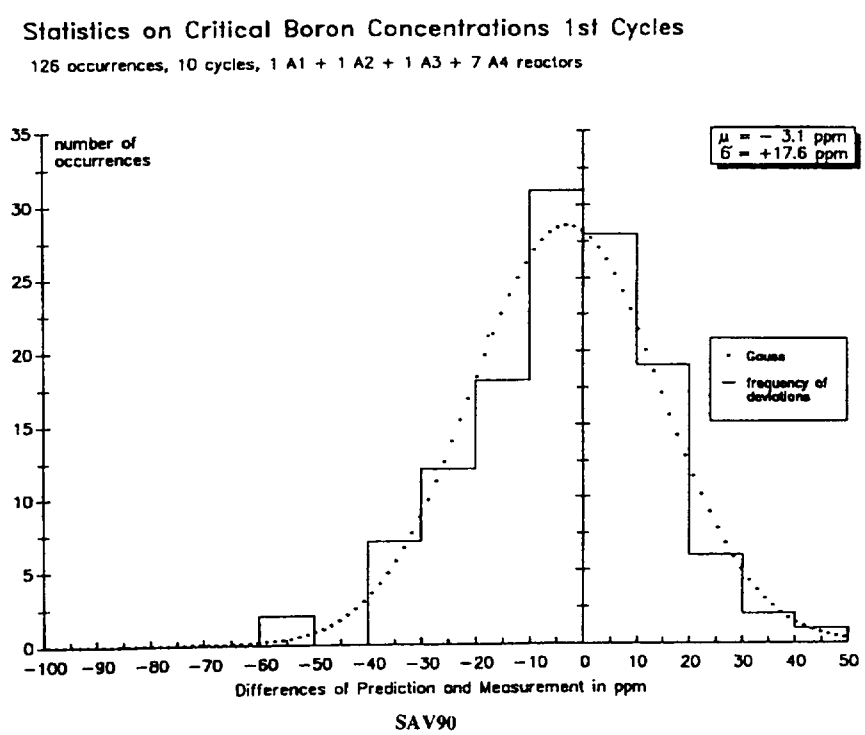

FIG. 8. SAV90 statistics on critical boron concentrations.

7. VERIFICATION OF FUEL SELECTION AND LOADING

According to the above-mentioned draft of the criticality safety standard for burnup credit in wet storage the following methods will be available:

- Measurement of each fuel assembly's reactivity and comparison of the results to the fuel assembly's reactivity referring to the minimum required burnup;
- Measurement of each fuel assembly's burnup or other correlative parameters;
- Analysis of each fuel assembly's exposure history or other correlative parameters to determine its burnup.

All uncertainties inherent to these methods have to be taken into account by deriving respective adequate decision criteria based on a 5% significance level.

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BURNUP CREDIT ASSESSMENT IN HUNGARY

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Abstract

At present, there is no immediate need of using burnup credit in Hungary. However, because of possible future developments in spent fuel storage, some calculational tools are planned to be used for investigating whether they are appropriate for such purpose. Besides the widely known codes SCALE and MCNP, the code MULTICEL developed in AERI is considered for this purpose. The testing of MULTICEL against critical experiments, depletion measurements and mathematical benchmarks is briefly reviewed.

1. INTRODUCTION

Until the recent past, the applicability of burnup credit has not been investigated in Hungary. Based on the corresponding agreements between the governments, the spent fuel has been sent back to the Soviet Union, later to the Russia for storage and disposal. These agreements are still effective at present, however, an off-site middle-term dry storage pool was build near to the Paks NPP and the possible applicability of the burnup credit in spent fuel storage was considered.

2. PRESENT STATUS OF THE STORAGE AND REGULATION

Hungary has four units of WWER-440 at Paks NPP, the Budapest Research Reactor at KFKI Atomic Energy Research Institute and a training reactor at Technical University of Budapest. The NPP in Paks presently uses fuel with 3.6 % maximal enrichment, but the introduction of new, profiled assemblies with average 3.82 % enrichment is under consideration. The Budapest Research now uses VVRSM type fuel with 36 % enrichment. Formerly EK-10 fuel was used with 10 % enrichment. The training reactor of the Technical University of Budapest uses EK-10 fuel with enrichment of 10 %.
The storage and transportation of fissile material in Hungary are regulated by the orders of the government and by the guidelines of the nuclear authorities. Exceptions from the guidelines are possible in such cases when the nuclear authority gives special permission based on the study of a team of experts. The orders don't prescribe the use of the fresh fuel assumption neither for power nor for research reactor fuel. The guidelines for power reactors requires the fresh fuel assumption. The guidelines for research reactors are now under revision.

The reactor-site short-term wet storage pool would also be used for emergency core unloads in the case of Paks NPP as well as the Budapest Research Reactor. Because emergency unload might happen with almost fresh fuel, the use of burnup credit should not be applied for these pools. In the case of the new dry storage pool near to Paks NPP and the wet middle-term storage pool at the Budapest Research Reactor, the use of burnup credit is not forbidden.

3. ACTIVITIES AND PLANS

Due to the above circumstances only general considerations and some preliminary calculations were done until now. The aim of these investigations was to identify those criticality and depletion codes which could be used in burnup credit criticality analysis. Until now the use of the SCALE code package, the MCNP and the MULTICEL codes were considered. These codes were planned to apply to the benchmark calculations proposed by L. Marková for WWER spent fuel [1, 2]. This benchmark corresponds to the OECD burnup credit benchmark.

The SCALE code package is well-known and widely used. It contains the criticality calculation and the depletion calculation codes as well [3]. It is validated for burnup credit applications in the case of PWR reactors [4]. However, in our opinion, further testing is necessary to apply it to WWER

fuel. Another possibility is connecting the well-known MCNP code [5] for criticality calculations and the MULTICEL code for depletion calculations. The MULTICEL code was developed in AERI and will be described in more detail in the next section.

To analyze the capability of the criticality codes in the case of hexagonal lattice, a series of calculations were made by MCNP4A and SCALE KENO V.a for a critical assembly consisting of hexagonal array of fuel pins [6]. The calculations were performed for different values of the boron concentration and of the lattice pitch. The best agreement with the experimental results was found with ENDF-B/V based libraries both with MCNP and SCALE KENO. The results of the two codes were the same within the statistical uncertainties. Also, the difference between the calculated and measured results was found independent of boron concentration in both cases. This work was done in collaboration with the Polytechnical University of Valencia [7].

4. THE MULTICEL CODE

The MULTICEL code was developed as a module of the three-dimensional coupled neutron physical - thermohydraulical core design code system KARATE [8]. It is a two-dimensional code, which can be used for the calculation of the pin-by-pin multigroup flux distribution in a fuel assembly. The approximate solution of the integral transport equations is calculated by the multicell collision probability method. That means, that the collision probabilities are obtained as a combination of the cell transmission, the escape probabilities and the collision probabilities inside the cylindrical cells. The calculation is performed in 35 epithermal and 35 thermal groups. Using the calculated reaction rates, the concentrations of 18 actinides and 148 fission products are calculated. The code was validated against critical experiments, the compositions measured on the Yankee-Row reactor and mathematical benchmarks. Here, only the most interesting points of the validation are described briefly.

The capability of MULTICEL for two-dimensional calculations was tested by the measurements made on the ZR-6 critical assembly and by mathematical benchmarks. ZR-6 is a critical assembly which was used for several years in our institute. It was a large hexagonal array of fuel pins. Several measurements were made for its homogeneous and perturbed states [9]. The perturbations were introduced into the system by removing fuel pins according to different schemes. To simulate a fuel assembly, lines of the pins were removed so that the resulting structure imitate the WWER fuel assembly in a lattice. The measured and calculated (by MULTICEL) power distributions are shown in Fig. 1 [10]. The agreement is quite good. Note, that the perturbation in this case is stronger than in the case of the real assemblies because the water gap contains more water than that of a real geometry.

A mathematical benchmark investigated by such widely used codes as the WIMS7 or HELIOS is the calculation of the k_{inf} of WWER assembly with 3.6 % enrichment as a function of burnup. The comparison of the result calculated by MULTICEL, WIMS7, HELIOS and KASETTA is shown in Fig. 2 [11].

The most interesting test which is really a test of the depletion calculation is the comparison with the measurements made on a Yankee reactor and published in [12]. The calculations and comparison were made for pins which were located approximately in asymptotic positions. To illustrate the agreement for the most important isotopes, the measured and calculated concentrations of U-235, U-238, Pu-239 and Pu-241 are shown in Figs. 3-6 as a function of burnup.

5. SUMMARY

The possibility of implementation of burnup credit in spent fuel management systems in Hungary is investigated. The present aim of the research is to identify or develop the computational tools which can qualify as appropriate for such purpose.

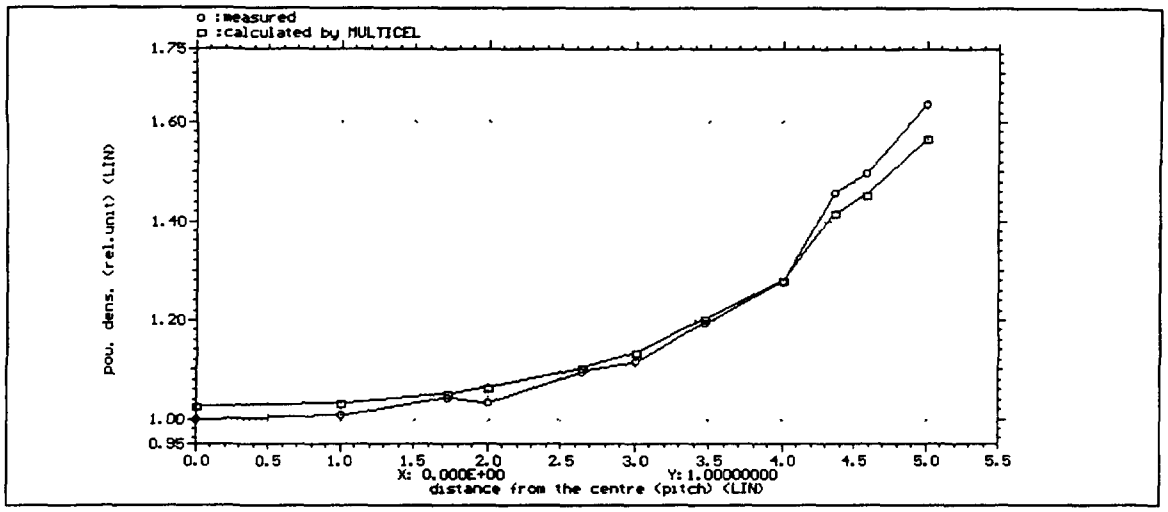

FIG. 1. Measured and calculated power density

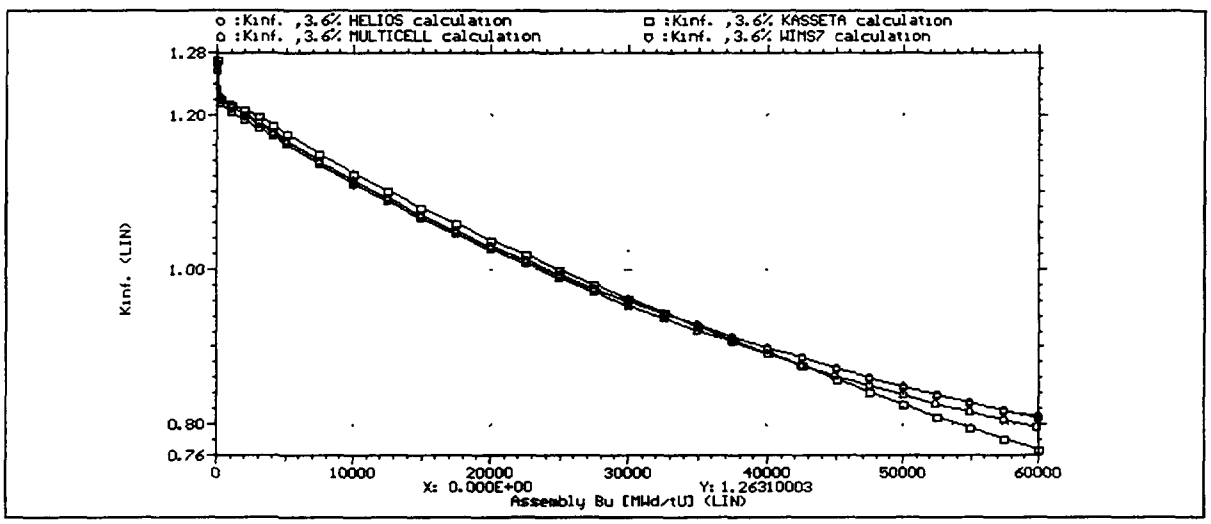

FIG. 2. k_{inf} vs. burnup

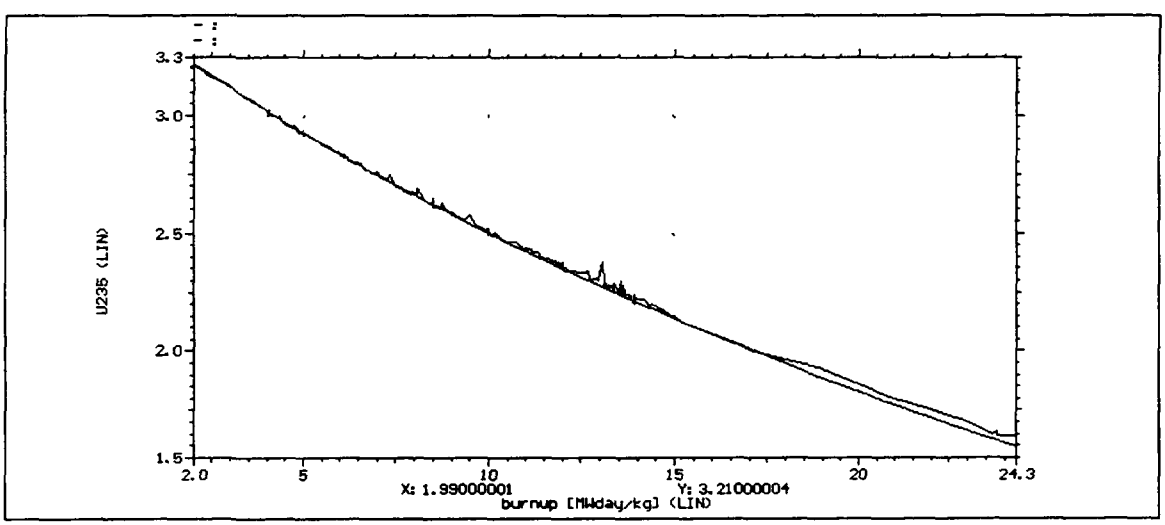

FIG. 3. Concentration of U-235

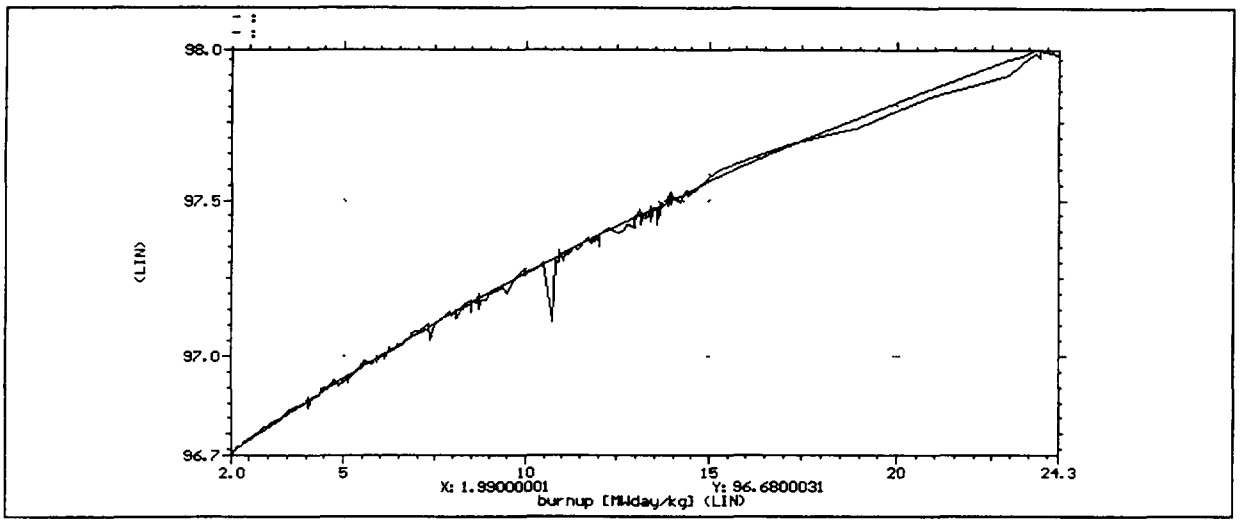

FIG. 4. Concentration of U-238

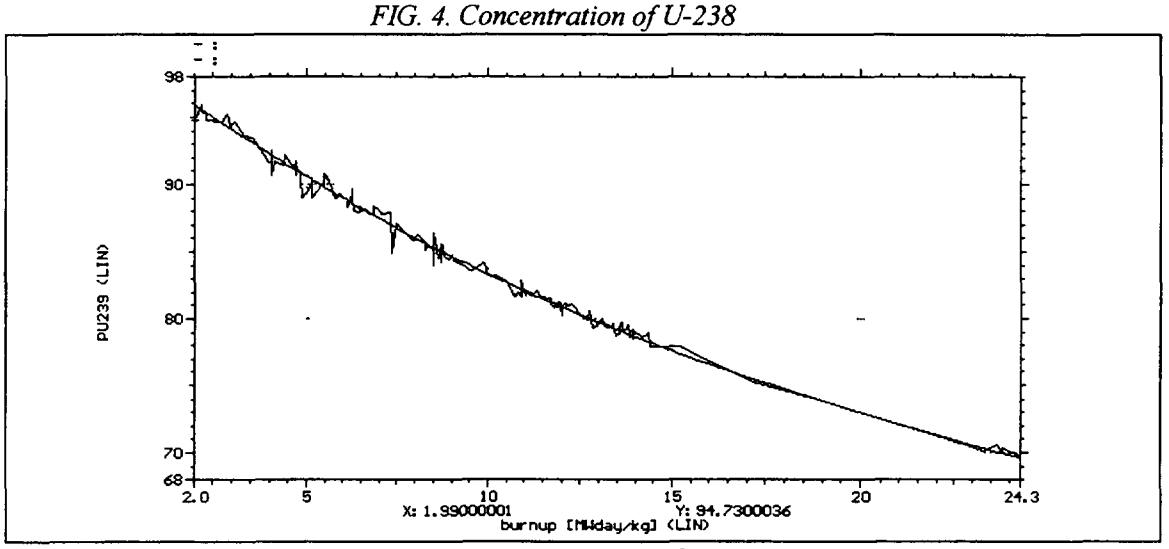

FIG. 5. Concentration of Pu-239

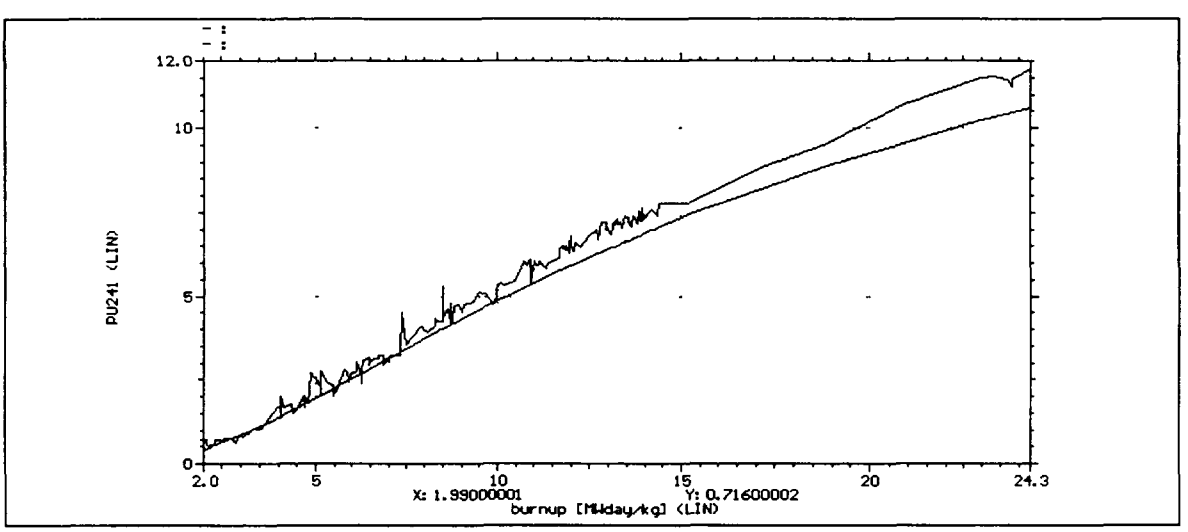

FIG. 6. Concentration of Pu-241

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STUDY ON BURNUP CREDIT EVALUATION METHOD AT JAERI TOWARDS SECURING CRITICALITY SAFETY RATIONALE FOR MANAGEMENT OF SPENT FUEL

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Presented by M.J. Crijns

Abstract

Lately, due to massive accumulation of spent fuel discharged from light water reactors in Japan, it is gradually demanded to introduce the so-called burnup credit methodology into criticality safety design for nuclear fuel cycle facilities, such as spent fuel storage pools and transport casks. In order to save space in the spent fuel storage pool of the Rokkasho Reprocessing Plant, the burnup credit design has been firstly implemented for its criticality safety evaluation. Here, its design conditions and operational control procedures are briefly shown and research using burned fuel at JAERI is explained to support its licensing safety review, focusing on the relevant content of the Nuclear Criticality Safety Handbook of Japan, which has been prepared so far and planned in the near future. Finally, international co-operation for study on burnup credit issues practiced by JAERI is addressed.

1. INTRODUCTION

Since light water reactors (LWR) now provide an important share of the electric power supply and will continue to be utilized in the future in Japan, the nuclear fuel is expected to be used for long duration basically from an economical point of view. As a consequence, a higher initial ^{235}U enrichment is currently required in the nuclear fuel fabrication specification to realize higher fuel burnup.

Traditionally, in the criticality safety design of spent fuel (SF) storage and transportation (S/T) casks or facilities, the fuel is usually assumed to be at its full initial enrichment (so called fresh fuel assumption) to provide a large safety allowance, which is sometimes excessively given, for example, requiring unnecessarily large space between fuel assemblies. The burnup credit taken for criticality safety design is firstly implemented to the *SF Storage Rack of Rokkasho Reprocessing Facility*, which is completed and expected for operation soon. Except for that, no burnup credit has been taken in criticality safety design for SF S/T casks or intermediate storage facilities in Japan.

Since in the near future it is considered inevitable to handle spent fuel massively, it is desired to implement the rational S/T design saving safety and economy by taking into account the fuel burnup in the criticality safety control. Computer codes and data which are vital to assess criticality safety in the design stage of nuclear fuel cycle facility have been developed and prepared to constitute a Japanese criticality safety handbook at JAERI.

2. PRESENT STATUS OF BURNUP CREDIT IMPLEMENTED

According to the report written by the *Electric Utility Industry Council*, a government advisory organization, nuclear power generation capacity is about 45 GWe and the cumulative amount of generated SF is some 13,600 tU (12,300 tU from LWRs and 1,200 tU from Gas Cooled Reactors (GCR) as of the end of August 1997. Some 7,800 tU of the SF have been shipped to reprocessing plants (some 6,800 tU to overseas and the rest to Tokai Reprocessing Plant). Therefore the total amount of spent fuel stored in nuclear power station pools is some 5,800 tU as of the end of March 1997.

The available capacity for spent fuel storage at LWR sites is some 9,920 tU, and its expansion at the reactor sites has been carried out by alternation of racks in existing storage pools (re-racking), common use of pools at the reactor site, building additional pools or dry-cask storage facilities without virtue of the burnup credit concept. As for SF transport casks, the design incorporating neutron absorbers as basket materials has been practiced with the fresh-fuel assumption to fill up SF assemblies inside the cask.

The criticality safety design of the *Rokkasho Reprocessing Plant Spent Fuel Storage Racks* is the first case to incorporate the burnup credit concept. The practices implemented for the plant design are as follow:

- Burnup Control for the Spent Fuel Storage Racks
  In general, the residual uranium enrichment could be less than 2.0 wt.% for SF discharged from light water reactors, the main racks have been designed so as to accept SF after ensuring that the enrichment is less than 2.0 wt % with the assembly average fuel burnup measured by a burnup monitor. The capacity of the racks is for 8,500 BWR and 3,500 PWR SF assemblies (about 3,000 tU).
- Assumptions Made for the Criticality Safety Assessment for the Rack Design
  - Residual uranium enrichment: 2 wt.%
  - Plutonium content: calculated for the maximum assumed burnup
  - Axial distribution of uranium enrichment: uniform
  - Fission products: not considered

3. RESEARCH USING BURNED FUEL

Under the auspices of *Science and Technology Agency (STA) of Japan*, JAERI has conducted a new research project since 1994, in which some SF assemblies discharged from Japanese LWRs have been measured by nondestructive and destructive methods so that their burnup and nuclide composition data are obtained and correlated. Subsequently, burnup analysis computer codes are validated with these experiment data. A combination of standard codes for criticality, shielding and thermal analyses have been arranged and the analyzed results are studied in comparison with corresponding safety criteria in applying to the pragmatic S/T cask design by taking into account the burnup credit concept.

3.1. Nondestructive and Destructive Measurements of Spent Fuel

In order to conduct a rational assessment for criticality, shielding and thermal analyses for SF S/T casks, it is required to use accurate burnup data with corresponding nuclide composition including transuranium (TRU) isotopes and fission products (FP). These data are especially important for verification of burnup analysis codes. The following procedures are established to obtain the measured data:

- Some 100 points along the longitudinal axis of a spent fuel rod are selected and gamma scanned nondestructively to assess burnup distribution.
- The spent fuel rod is sliced in five positions along the whole length and destructively analyzed for their nuclide composition. After dissolution and chemical separation of each specimen, TRU isotopes (uranium, plutonium, neptunium, americium and curium) and FP (neodium, gadolinium, samarium and other gamma radioactive nuclides) are measured with respect to their mass content and isotopic composition. The burnup evaluation is done by mass spectrometry and alpha/gamma radiochemical analyses.

Subsequently, a correlation between burnup and a particular FP gamma radioactivity ratio such as $^{134}Cs/^{137}Cs$ is obtained as shown in Fig. 1. This correlation is useful for the development of a burnup monitor as well as for computer code validation.

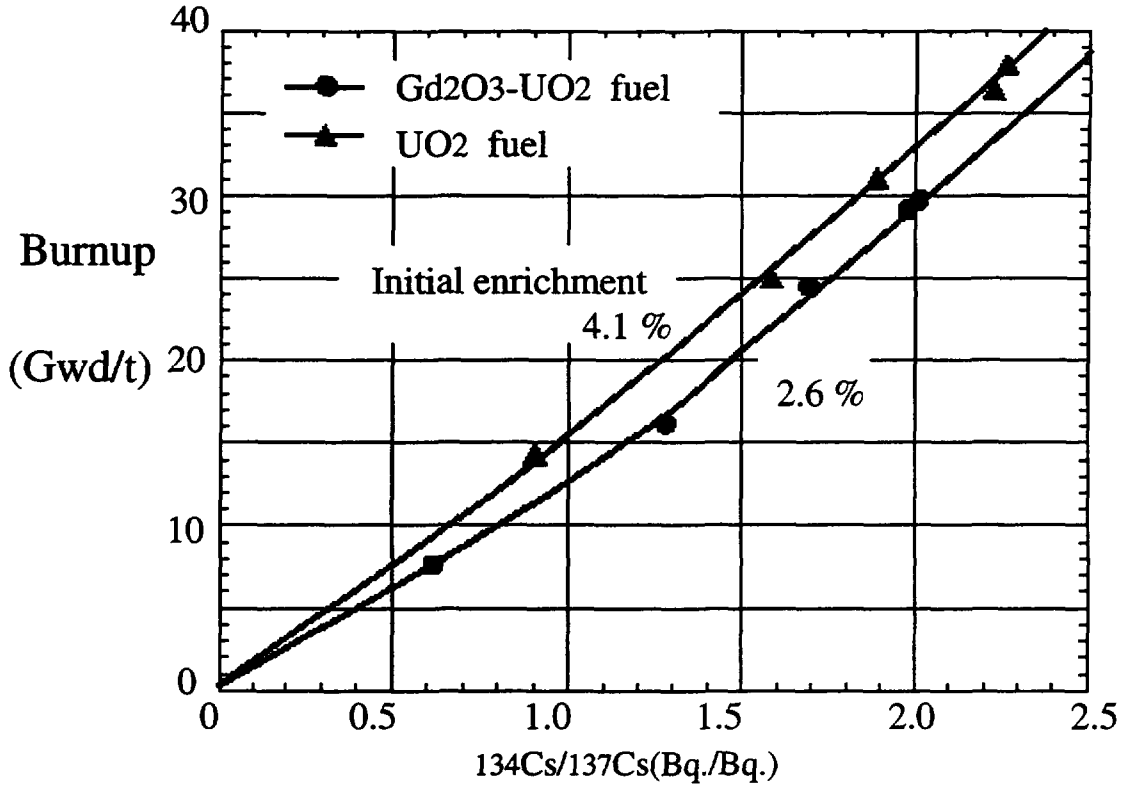

FIG.1. Calibration curve for burnup and γ radioactivity ratio

3.2. Validation of Burnup Analysis Codes

Standard problems have been established based on the aforementioned Post Irradiation Examination (PIE) data and with the reactor operation history data given. These standard problems are used to validate the burnup codes utilized in Japan such as the ORIGEN2.1 code and the SWAT code system developed at JAERI.

3.3. Comprehensive Safety Evaluation System for Cask Design

As the burnup of nuclear fuel increases, both the heat generation and the radioactivity of the spent fuel assembly increase as well, which is due to the accumulation of minor actinides such as ^{232}U, ^{237}Np, ^{238}Pu, ^{242}Cm. Accordingly, not only criticality but also thermal and shielding considerations have to be addressed for safety designs of SF S/T. In this context, a comprehensive safety evaluation system consisting of standard computer codes for criticality, thermal and shielding analyses is developed at JAERI to find an optimum point of design conditions with sufficient and balanced margins to ensure various safety aspects.

4. PREPARATION OF THE *NUCLEAR CRITICALITY SAFETY HANDBOOK OF JAPAN*

The *Nuclear Criticality Safety Handbook of Japan* was edited and published by STA in 1988, and its English translation was issued in 1995. Supplemental work were conducted since its first publication and the results were published as a JAERI report titled the *Supplement Report to the Nuclear Criticality Safety Handbook of Japan*. In this supplement, actinide nuclide compositions are taken from the open literature reporting PIE of spent fuel from light water reactors and arranged in a

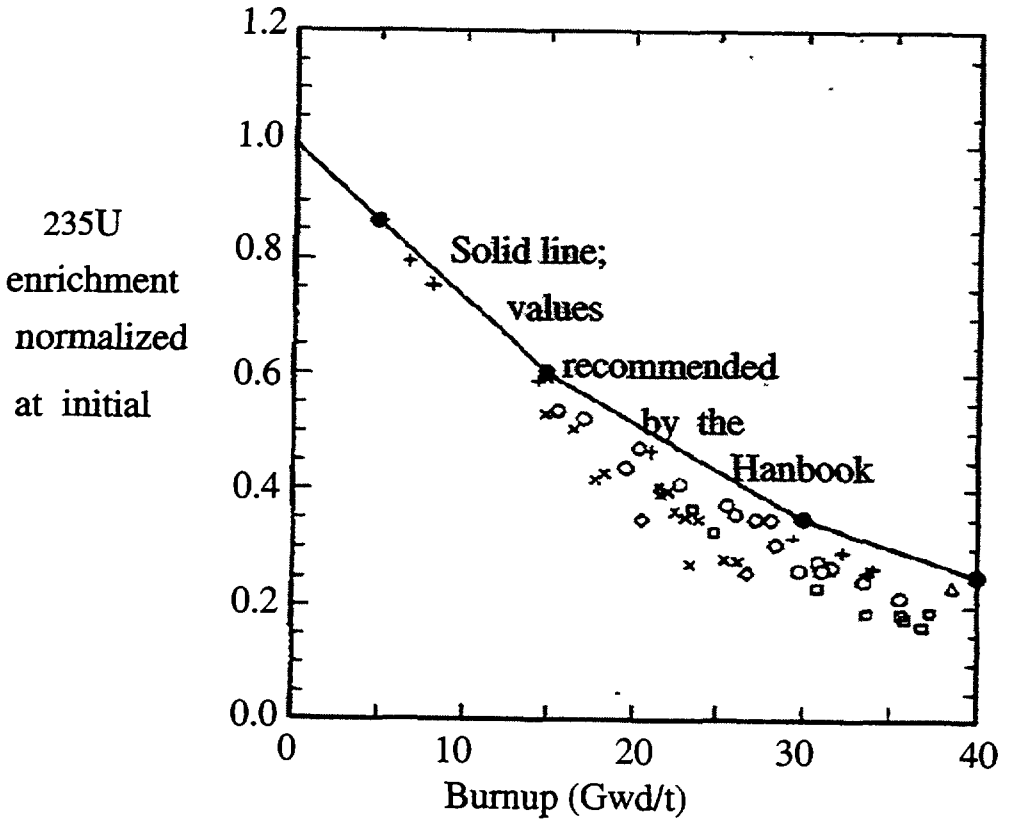

FIG. 2. 235U enrichment change with burnup of LWR fuel

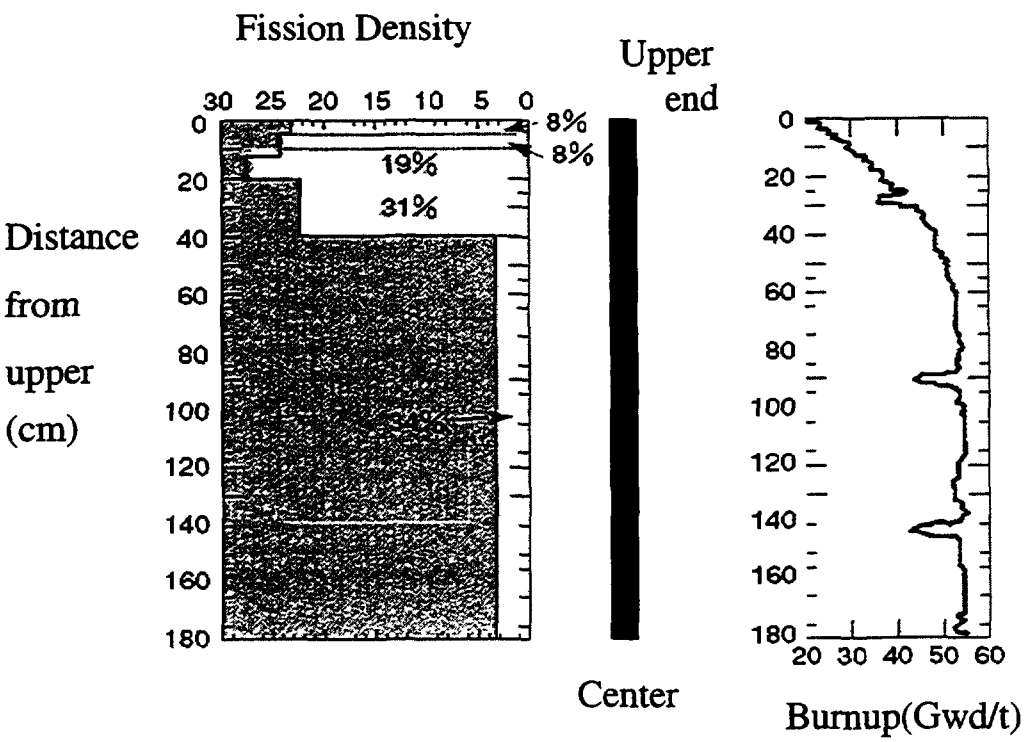

FIG 3 Axially distributed burnup and fission density along a PWR fuel pin

simple diagram such as Fig. 2 in order to give recommended values for criticality safety evaluation of high burnup fuel systems. The second version of the handbook will be published at the end of 1997 fiscal year with useful data and information on burnup credit such as the selected FP isotopes to be considered. Axial burnup profile effect is important for criticality safety evaluation of high burnup fuel systems as shown in Fig. 3.

In the near future, a national policy will be implemented for utilizing mixed oxide fuel (MOX) consisting of uranium dioxide and plutonium dioxide in many conventional LWRs. The nuclide composition data for the spent MOX fuel are no less important than those for the spent UO_2 fuel, and have been so far accumulated by an international cooperation with the *Actinides Research In A Nuclear Element (ARIANE)* project. These MOX spent fuel nuclide composition data will be used for validation of the burnup codes. These codes will be used in the burnup credit design for S/T systems of MOX spent fuel assemblies irradiated in LWRs.

5. INTERNATIONAL COOPERATION FOR STUDY ON BURNUP CREDIT ISSUES

There are several working groups active in the *Organization for Economic Co-operation and Development / Nuclear Energy Agency (OECD/NEA)* which promotes international understanding in the field of nuclear criticality safety. The *ICSBEP (International Criticality Safety Benchmark Evaluation Project)*, is carried out by one working group. Its purpose is to provide the nuclear industry with qualified benchmark data-sets by collecting criticality experiment data mainly from US-DOE national laboratories, rigorously reviewing the data and information, adding supplemental data, and collating in a consistent format. The *Criticality Safety Benchmark Working Group* is another working group, which focuses on an effort to establish the validity of computational methods applicable to the so-called burnup credit design for SF S/T. Currently, examination of PWR fuel benchmark calculation results contributed by many member organizations is almost complete and thus the focus is now shifting toward the analysis of BWR fuel benchmark results. Recently, a *Working Party for Nuclear Criticality Safety* has been set up to review aforementioned activities of technical issues concerning away-from-reactor criticality safety and to provide guidance and overall coordination for these activities.

JAERI has ardently considered to contribute to these international activities with the accumulated useful information and data. Reciprocally, the results obtained by the international cooperation have been incorporated into the *Criticality Safety Handbook of Japan* to be utilized for criticality safety design of SF management for S/T.

APPLICATION OF BURNUP CREDIT FOR SPENT FUEL MANAGEMENT IN THE REPUBLIC OF KOREA

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Abstract

Burnup credit implementation for the storage capacity expansion in the spent fuel pools of Kori units 3 and 4, Yonggwang units 1 and 2 and Ulchin units 1 and 2, which are operated by Korea Electric Power Cooperation (KEPCO), is described. The burnup credit application for the Korean spent fuel management programme in the Nuclear Environment TEChnology Institute (NETEC) is also introduced. Finally, the determination of correction factors for predicting conservative isotopic compositions of PWR spent fuel performed in the Korea Atomic Energy Research Institute (KAERI) is presented.

1. STORAGE CAPACITY EXPANSION OF SPENT FUEL POOL IN POWER PLANTS

1.1. Status of Spent Fuel Pool Re-Racking in Power Plants

Burnup credit implementation for the storage capacity expansion in the spent fuel pools of Kori units 3 and 4, Yonggwang units 1 and 2 and Ulchin units 1 and 2 was performed by Korea Electric Power Cooperation (KEPCO). The re-racking status is shown in the Table 1. The design basis of spent fuel racks was 35.56 cm in cent-to-center distance when the reactors were constructed. It was thus intended to reduce the center-to-center distance as much as possible by adopting neutron poisonous materials and burnup credit implementation. As a general rule for PWR, high density racks feature the so-called discrete two region storage arrangement to reduce the center-to-center distance. Region 1 racks are used to store 5 wt.% enriched-uranium fuel while maintaining $k_{eff} < 0.95$ without any burnup credit. On the other hand, region 2 racks have an enrichment/burnup credit on it.

1.2. Criticality Safety Evaluation

1.2.1. Fuel Burnup Calculations and Uncertainties

CASMO-3 was used for burnup calculations during core operations. Since critical experiment data with spent fuel is not available for determining the uncertainty in burnup dependent reactivity calculations, an allowance for uncertainty in reactivity was assigned based upon other considerations. Assuming the uncertainty in depletion calculation is less than 5% of the total reactivity decrement, a burnup dependent uncertainty in reactivity may be assigned. For the Kori 4 and Yonggwang 1 and 2 storage racks at the design basis burnup, the total reactivity decrease from beginning of life to 4.49 MW·d/kgU is 0.034 Δk and the reactivity uncertainty therefore is $\pm$ 0.0017 Δk. Allowances for uncertainties at lower burnups are smaller. The method of estimating an allowance for uncertainty in burnup calculations has been previously accepted by the US-NRC and is believed to be a conservative estimate, particularly in view of the continuing reactivity decrease with time in storage of aged fuel.

1.2.2. Axial Distribution in Burnup

Generic analytic results of the axial burnup effect have been published, based upon calculated and measured axial burnup distributions. These analyses confirm the generally negative reactivity effect of the axially burnup distribution less than about 30 MW·d/kgU [1]. Therefore, for the very low burnups required for Kori 4 and Yonggwang 1 and 2 fuel (4.49 GW·d/tU for 5 wt.% enrichment) the effect of the axial distribution is negative, and no correction is required.

TABLE 1. STATUS OF SPENT FUEL POOL RE-RACKING IN POWER PLANTS

Plant	Region	Maximum Enrichment (wt.%)	Minimum Burnup (GW·d/tU)	Boral or Boraflex	Burnup Credit	Assembly Pitch (mm)		Storage Capacity (Assembly #)		Saturation Year
						Original Rack	Re-Rack	Original Rack	Re-Rack	
Kori 3	I	4.2				356 (14``)	356 (14``)	746	1193	2006
	II	4.2	42.0	O	O	356 (14``)	278 (9.25``)			
Kori 4	I	5.0		O		356 (14``)	278 (10.96``)	746	1152	2006
	II	5.0	4.49	X	O	356 (14``)	356 (14``)			
Yonggwang 1, 2	I	5.0		O		356 (14``)	278 (10.96``)	746	1152	2006
	II	5.0	4.49	X	O	356 (14``)	356 (14``)			
Ulchin 1	I	5.0		O		276 x 267	276 x 267	472	1114	2007
	II	5.0	45.0	O	O	276 x 267	227 x 227			
Ulchin 2	I	5.0		O		276 x 267	276 x 267	472	875	
	II	5.0	7.56	O	O	276 x 267	264 x 259			

1.2.3. Benchmark Calculations

The objective of the benchmarking study was to verify the four code used: NITAWL-KENO-5a (with both the 27-group and 218-group SCALE cross-section libraries), MCNP, and the CASMO-3 code for use in criticality safety calculations of high density spent fuel storage racks. These calculational methods are based upon transport theory and have been benchmarked against critical experiments that simulate typical spent fuel storage rack designs as realistically as possible.

Results of the benchmark calculations are consistent with corresponding calculations reported in the literature and show that the 27-group (SCALE) NITAWL-KENO-5a calculations consistently underpredict the critical eigenvalue by 0.0103±0.0018 Δk (with a 95% probability at a 95% confidence level) for critical experiments that are as representative as possible of realistic spent fuel storage rack configurations and poison worths. Similarly, the 218-group library with KENO-5a resulted in a bias of 0.0128±0.0020, and the benchmark calculations with MCNP resulted in a bias of 0.0032±0.0020. Extensive benchmarking calculations of critical experiments with CASMO-3 have also been reported, giving a mean k_{eff} of 1.0004±0.0011 for 37 cases.

2. SPENT FUEL MANAGEMENT TECHNOLOGY DEVELOPMENT (IN NETEC)

By taking burnup credit, the high capacity of the storage and transportation can be more fully utilized, reducing the space of storage and the number of shipments. Larger storage and fewer shipments for a given inventory of spent fuel result in remarkable cost savings and more importantly reduce the risks to the public and occupational workers for the Korean Spent Fuel Management Programme. Recently, the incentives for pursuing burnup credit over the current fresh fuel approach are widely recognized in Korea. As NETEC is responsible for the Korean Spent Fuel Management Programme, several approaches can extend enrichment limitations for existing storage and transport containers, and may contribute to the development of higher capacity storage and transport systems that would result in fewer fuel shipments and therefore decreased risk to the public.

However, before such an approach can be approved by licensing agencies, it would be necessary to demonstrate that the available criticality safety calculational tools are appropriate for application to spent fuel analyses system and a reasonable safety margin can be established. For further studies of the Korean Radwaste Management programme, NETEC will assess the optimal radionuclide generation, depletion and nuclear criticality codes as to their capabilities and possibilities in addressing the requirements associated with the transportation, storage, and geologic disposal of spent fuel and HLW.

3. CORRECTION FACTOR (IN KAERI)

3.1. Experimental Data

Reactor types, initial enrichments, burnup and cooling times in the measuring isotopic compositions are shown in table 2 [2-4]. Table 2 shows that most ORIGEN2-calculated compositions are consistent to the experiments no less than those of ORIGEN-S. But the differences are very large in the Calvert Cliff Unit 1 with 2.453 wt.% initial enrichment. Considering the fact that the initial enrichment of the spent fuel is much lower than ordinary PWR fuel, ORIGEN2 code seem to be inappropriate to the depletion calculation on the lower initial enrichment fuel. Therefore, the spent fuel is not included in the statistical process. Then the approach to determine the conservative isotopic compositions are carried out only on the PWR spent fuel with initial enrichment range from 2.561 wt.% to 3.415 wt.% and burnup range from 6.9 GW·d/tU to 44.43 GW·d/tU.

TABLE 2. SPECIFICATIONS OF PWR SPENT FUEL USED IN THE EXPERIMENTS AND RELATIVE ERRORS OF THE CALCULATED U-235 AND PU-239 COMPOSITIONS TO EXPERIMENTS

Reactor	Enrichment (wt.%)	Burnup (GW-d/tU)	Cooling Time	^{235}U , (C-E)/E* (%)		^{239}Pu , (C-E)/E* (%)	
				ORIGEN2	ORIGEN-S	ORIGEN2	ORIGEN-S
Calvert Cliff	3.038	27.35	1870 d	-0.06	-1.5	1.58	-8.3
Unit 1	3.038	37.12	1870 d	-1.15	-7.7	3.23	-4.6
	3.038	44.34	1870 d	-3.37	-8.7	4.00	-1.5
	2.72	18.68	2374 d	4.05	-9.0	2.55	-6.6
	2.72	26.62	2374 d	4.30	-3.4	2.44	-6.4
	2.72	33.17	2374 d	7.38	-1.4	7.07	-0.5
	2.453	31.40	2447 d	20.84	-0.2	17.49	-0.5
	2.453	37.27	2447 d	22.39	-3.1	18.27	0.1
	2.453	46.46	2447 d	37.54	2.5	20.95	5.1
H.B. Robinson	2.561	16.02	3936 d	3.21	-1.8	7.24	1.6
Unit 2	2.561	23.81	3936 d	4.75	-2.4	6.95	1.8
	2.561	28.47	3631 d	-4.32	2.1	0.67	-0.5
	2.561	31.66	3631 d	2.27	3.3	6.57	-6.5
Obrigheim	3.13	25.93	0	-1.00	-2.5	-0.52	0.5
	3.13	26.54	0	-0.47	-0.2	1.85	0.29
	3.13	27.99	0	-0.37	-2.6	-1.66	-0.7
	3.13	28.40	0	-0.60	-1.7	-3.19	-2.1
	3.13	29.04	0	-2.67	-3.8	-1.77	-0.6
	3.13	29.52	0	-0.83	-2.0	-1.25	0.0
Mihama-3	3.208	6.90	5 y	-1.49	-	-4.05	-
	3.208	8.30	5 y	-1.40	-	0.76	-
	3.203	14.60	5 y	-0.43	-	-14.50	-
	3.203	15.30	5 y	-3.78	-	-11.12	-
	3.203	21.20	5 y	-4.28	-	-8.97	-
	3.21	29.44	5 y	-2.19	-	-6.62	-
	3.21	32.30	5 y	-8.34	-	-8.31	-
	3.21	33.70	5 y	-1.83	-	1.48	-
	3.21	34.10	5 y	-5.62	-	-5.03	-
Genkai-1	3.415	38.10	5 y	-10.40	-	-11.78	-
	3.415	38.70	5 y	-11.36	-	-13.26	-

* (C-E)/E(%) : Calculated composition - Experimental composition) / Experimental compositions x 100 (%)

3.2. Determination of correction factor [5]

The relative error of the calculated isotopic composition to the measured composition for each isotope over the range of all experimental measurement can be defined as

$$X_{ij} = \left[\frac{C_{ij} - E_{ij}}{E_{ij}} \right] \tag{1}$$

where E_{ij} and C_{ij} are the experimental and calculated value respectively for the nuclide i and sample j of the different initial enrichment, burnup, and/or cooling.

$$\bar{x}_i = \sum_{j=0}^{N_i} \left[\frac{X_{ij}}{N_i} \right], \quad s_i^2 = \sum_{j=0}^{N_i} \left[\frac{(x_{ij} - \bar{x}_i)^2}{N_i - 1} \right] \tag{2}$$

The means and standard deviations of 43 isotopes are provided in Table 3. Under a tolerance interval approach to assign a tolerance factor $k_{N_i}^{\alpha,\gamma}$ for N_i, for a given α and γ, there is a probability γ that a future value of x_{ij} will lie within the range $\bar{x}_i \pm k_{N_i}^{\alpha,\gamma} s_i$, with an α confidence [6,7].

TABLE 3. COMPARISON OF THE CALCULATED ISOTOPE COMPOSITIONS WITH EXPERIMENTS

Nuclide	(C -E) / E			95/95 confidence	Correction
	Data #	Average(X)	Std. dev.(σ)	(X ± k* σ)	factor(*f*)**
^{232}U	9	-0.112	0.246	-0.112 ± 0.868	0.57
^{234}U	9	0.097	0.057	0.097 ± 0.201	0.77
^{235}U	25	-0.007	0.035	-0.007 ± 0.091	1.11
^{236}U	25	0.003	0.029	0.003 ± 0.076	0.93
^{238}U	19	-0.003	0.005	-0.003 ± 0.015	0.99
^{237}Np	15	0.094	0.091	0.094 ± 0.268	0.73
^{236}Pu	3	-0.181	0.522	-0.181 ± 5.180	0.00
^{238}Pu	25	-0.072	0.086	-0.072 ± 0.225	0.87
^{239}Pu	25	-0.008	0.057	-0.008 ± 0.150	1.19
^{240}Pu	25	0.009	0.035	0.009 ± 0.091	0.91
^{241}Pu	25	-0.014	0.072	-0.014 ± 0.191	1.26
^{242}Pu	21	-0.127	0.057	-0.127 ± 0.156	0.97
^{241}Am	15	-0.091	0.077	-0.091 ± 0.228	0.88
^{243}Am	9	-0.252	0.161	-0.252 ± 0.569	0.76
^{242}Cm	14	0.145	0.671	0.145 ± 2.022	0.00
^{243}Cm	3	0.260	0.091	0.260 ± 2.906	2.82
^{244}Cm	15	-0.292	0.143	-0.292 ± 0.424	0.88
^{243A}Cm[1)]	6	-0.056	0.046	-0.056 ± 0.202	0.87
^{79}Se	6	5.567	0.348	5.567 ± 1.536	0.12
^{90}Sr	6	0.014	0.027	0.014 ± 0.119	0.88
^{99}Tc	10	0.045	0.044	0.045 ± 0.148	0.84
^{106}Ru	9	-0.063	0.084	-0.063 ± 0.298	0.81
^{126}Sb	3	3.303	0.094	3.303 ± 0.931	0.19
^{125}Sb	9	1.064	0.161	1.064 ± 0.567	0.38
^{129}I	3	-0.098	0.037	-0.098 ± 0.371	0.79
^{133}Cs	3	-0.004	0.003	-0.004 ± 0.032	0.97
^{134}Cs	12	0.002	0.141	0.002 ± 0.445	0.69
^{135}Cs	6	-0.229	0.090	-0.229 ± 0.397	0.86
^{137}Cs	19	-0.018	0.032	-0.018 ± 0.089	0.93
^{144}Ce	9	-0.083	0.088	-0.083 ± 0.310	0.81
^{144}Nd	12	0.024	0.022	0.024 ± 0.068	0.92
^{145}Nd	13	0.021	0.021	0.021 ± 0.065	0.92
^{146}Nd	11	0.028	0.013	0.028 ± 0.044	0.93
^{148}Nd	12	0.033	0.009	0.033 ± 0.029	0.94
^{150}Nd	12	0.039	0.009	0.039 ± 0.030	0.94
^{147A}Pm[2)]	3	-0.275	0.046	-0.275 ± 0.454	0.85
^{148}Sm	3	0.022	0.052	0.022 ± 0.513	0.65
^{149}Sm	3	0.002	0.132	0.002 ± 1.312	0.00
^{150}Sm	3.	0.096	0.046	0.096 ± 0.460	0.64
^{152}Sm	3	0.204	0.038	0.204 ± 0.372	0.63
^{151A}Sm[3)]	3	0.332	0.124	0.332 ± 1.233	0.39
^{154A}Sm[4)]	3	0.350	0.070	0.350 ± 0.695	0.49
^{155A}Eu[5)]	3	1.094	0.139	1.094 ± 0.374	0.29

* k : The coefficient of 95 % tolerance interval with 95 % confidence [8].

** *f* is obtained from taking the upper bound of X ± k* σ for fissile material and the lower bound for neutron poison material (non-fissile material).

1) $^{243}Cm+^{244}Cm$

2) $^{147}Pm+^{147}Sm$

3) $^{151}Sm+^{151}Eu$

4) $^{154}Sm+^{154}Eu+^{154}Gd$

5) $^{155}Eu + ^{155}Gd$.

$$\bar{X}_i - k_{N_i}^{\alpha,\gamma} s_i \le \left[\frac{C_i' - E_i'}{E_i'}\right] \le \bar{X}_i + k_{N_i}^{\alpha,\gamma} s_i \qquad (3)$$

where C_i' and E_i' are calculated and true composition of an isotope i, respectively. transforming eq.(3), the upper and lower bounds of the true value are expressed by the following.

$$C_i' \cdot \left[\frac{1}{1+\bar{X}_i + k_{N_i}^{\alpha,\gamma} s_i}\right] \le E_i' \le \left[\frac{1}{1+\bar{X}_i - k_{N_i}^{\alpha,\gamma} s_i}\right] \cdot C_i' \qquad (4)$$

In criticality calculation, a conservative approach in the prediction of a neutron multiplication factor, k_{eff}, is to assume the maximum composition of fissile isotopes combined with the minimum composition of non-fissile isotopes to set an uppermost expected limit on k_{eff}. These correction factors can be written for each isotope i as

$$f_i(fissile) = \frac{1}{1+\bar{X}_i - k_{N_i}^{\alpha,\gamma} s_i}, \quad f_i(non-fissile) = \frac{1}{1+\bar{X}_i + k_{N_i}^{\alpha,\gamma} s_i} \qquad (5)$$

The correction factors are provided in Table 3 for 43 isotopes at 95/95 confidence level. The correction factors in Table 3 indicate that the ORIGEN2-prediction compositions of actinides except ^{232}U, ^{236}Pu, ^{242}Am and ^{243}Cm are good as the difference of correction factors from 1.0 are at least less than 0.27. ORIGEN2-prediction compositions of fission products except ^{79}Se, ^{126}Sn ^{125}Sb, ^{149}Sm, (^{154}Sm + ^{154}Eu + ^{154}Gd) and (^{155}Eu + ^{155}Gd) seem to be relatively good as the deviations of the correction factors are less than 0.36.

Use of the correction factor, determined at 95/95 confidence level for each isotope, in a criticality calculation for a given spent fuel configuration will result in a conservative upper estimate of k_{eff} for the configuration.

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APPLICATION OF BURNUP CREDIT IN SPENT FUEL MANAGEMENT AT RUSSIAN NPPs

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Abstract

The article concerns implementation of burnup credit in spent fuel storage and transportation. some of the problems with increased enrichment fuel can be resolved by use of modified transport methodology. such as shipping in gas-filled casks only, reduced number of assemblies in casks, etc. However, the use of modified schemes of transportation results in essential financial losses. An actinide-only burnup credit is taken into account in most part of criticality calculations, and a parameter limiting loading of spent fuel in the cask or the repository is the avenge value of burnup on an assembly. The main method of burnup depth definition is its defect measurement. A short description of devices for measurement as well as some technical results of suing burnup credit approach in storage and transport are given.

1. INTRODUCTION

The development of the nuclear industry in most countries of the world demands:

- an increase of the initial fuel enrichment for operating and projected NPPs;
- an increase in spent nuclear fuel (SNF) storage and cask capacity;
- a stricter approach in the safety analysis of casks and storage facilities.

The criticality problem for such demand can be solved by using solid neutron absorbers. Absorbers, such as boron, gadolinium, etc. can be mixed with the steel of the baskets. This practice is rather widely used, though it involves considerable materials and financial costs. Recently, the possibility of solving the problem through application of burnup credit is discussed in Russia and other countries.

2. CASK DESCRIPTION

There are four types of casks frequently used for NPP spent fuel transportation in Russia. These are specified as TK-6, TK-10, TK-11 and TK-13.

The TK-6 cask is designed for transporting thirty assemblies from the WWER-440 by rail. It is a thick walled forged flask weighting more then 78 tones. The cask walls, bottom and lid protect against ionizing radiation and no special neutron shielding is provided. Fuel decay heat is removed by natural convection and radiation from the cask outer surface which contains many ribs. The inner cavity of the cask may be filled with either gas or water.

The TK-10 is intended to transport six assemblies from the WWER-1000, WWER-500 or ACT-500 by rail. It is manufactured as a thick-wall forged steel flask, on the outer side of which there is a liquid neutron shielding of ethylene glycol (67%) mixed with water. Residual decay heat is removed by natural convection and radiation from the cask smooth outer surface.

The TK-11 cask may be used for transportation of fifty-one assemblies from the RBMK-1000 or thirty-five assemblies from the BN reactor by rail. It also can be used for transporting fuel assemblies from the WWER-440 and RBMK-1500. This cask is similar to the TK-6 by design and safety parameters.

The TK-13 cask is designed for transporting twelve assemblies from the WWER-1000 or twelve assemblies from the ACT-500 by railway. The cask basic construction is a forged steel flask

with external neutron shielding of ethylene glycol (67%) mixed with water. Decay heat is dissipated from a smooth outer surface of the cask by natural convection and radiation.

Some technical characteristics of the casks are given in Table I.

3. OPPORTUNITIES TAKEN FROM THE MODIFIED TRANSPORT METHODOLOGY

In accordance with the advanced programme for safe operation of RBMK power plants, the maximum fuel enrichment was increased from 2.0% to 2.4% which resulted in fuel burnup increase to about 28 GW·d/tU. Transportation of such fuel in the TK-11 casks necessitates a change in the transport procedure:

- For assuring nuclear safety and a permissible pressure in the cask cavity under emergency conditions, transportation should be made in gas-filled casks only;
- Meanwhile, radiation shielding becomes the principal limiting factor. Spent fuel with an initial enrichment of 2.0% and burnup of 22 GW·d/tU can be transported only after being cooled for not less than 8 years. As for spent fuel initially enriched to 2.4% and with a burnup of 28.5 GW·d/tU, the dose rate outside the shielding exceeds the standard value, even though the fuel is cooled for 15 years. In this case, the number of fuel assemblies loaded into TK-11 casks has to be reduced; for example, it is possible to load 15 assemblies (being cooled for 8 years) in the cask, without violation of the safety requirements, instead of the planned 51 assemblies. For higher burnup fuel a new cask should be developed or a reactor storage facility built.

To improve the economics, some nuclear power plants with WWERs-440 intend to use fuel with 4.4% enrichment. In this case, as seen from the experience of Kolskaya NPP (KOLA NPP), the fuel burnup can reach 50 GW·d/tU. Such fuel can be transported in TK-6 casks, only in the water-filled cask, and its full loading is not permissible according to both criticality and radiation protection requirements. The problem can be solved by replacing the outer row of fuel assemblies with dummy assemblies, which will serve as an additional shield and supersede excess water. The cask loading, in this case, reduces to 18 fuel assemblies instead of 30. In some instances, transportation of fuel, cooled for less than the design period, is required. Such conditions can arise when the schedule for fuel dispatch to the reprocessing plant is changed or when a NPP is decommissioned ahead of schedule, as the Armenian NPP for example.

As for the TK-6 cask, a system for optimizing the loading of baskets was developed. The system is based on sorting out the fuel assemblies to be loaded in the cask with burnup below 40 GW·d/tU, or by reducing their number for loading. An increase in total heat release for full loaded casks up to 15 kW instead of 12 kW, proves to be a possibility for transportation in winter-time.

Similar recommendations for transportation of WWER spent fuel with a relatively shorter cooling period as compared to the designed one, are developed to fit the TK-10 and TK-13 casks. According to experimental data, the cask shield is well within the safety margins, and the limiting factor is the cask design capacity for decay heat to be dissipated. For example, assemblies with initial fuel enrichment below 4.4%, which are used under reactor start-up conditions without extended burnups, can be transported when the fuel cooling time is shorter than designed. Furthermore, the radiation safety requirements are met, if the decay heat from all assemblies do not exceed 13 kW for a TK-10 cask and 20 kW for a TK-13 cask. However , use of the modified scheme of transportation results in essential financial losses. From this point of view the application of the scheme based on the use of burnup credit is more favorable.

TABLE I. SPENT FUEL PACKAGE CHARACTERISTICS

CASK		TK-6	TK-10	TK-11		TK-13	
FUEL							
Type of fuel		WWER-440	WWER-1000	RBMK-1000	BN-600	WWER-1000	ACT-500
Initial enrichment	wt %	3.6	4.4	2	33	4.4	2
Number of sssemblies		30	6	51	35	12	12
Average burnup	GW·d/tU	20/28	42	20	80	42	15
Cooling time	years	3	3	3	3	3	3
Maximum decay heat	kW/pack	8/12	13	10.3	10.7	20	20
Loaded fuel	tU/pack	3.6	2.6	5.9	1.0	5.2	4.8
CAVITY							
Diameter	cm	147.5	100	148.5		132	
Length in cask	cm	348	503	379		495.5	
Coolant in cask		gas/water	water	gas		gas	
Pressure in cask	atm.	1.66/2.5	2.5	2		1.66	
SHIELDING							
γ shield mat.		steel	steel	steel		steel	
Thickness of γ s. mat.	cm	36	38	36		36	
γ dose rate 2 m from side surface	mrem/h	4 / 3	2	4	3	5.6	1.6
Neutron shield material		water	water + ethylene glycol			water + ethylene glycol	
Thickness of shielding	cm		12			12.7	
Neutron dose rate 2 m from side surface	mrem/h	4 / 2.6	0.4	3	1	1	0.02
DIMENSIONS							
Outside diameter	cm	219.5	200	219.5		229.5	
Length of cask	cm	410.5	613	445.5		600.0	
Mode of surface		steel fins	smooth surface	steel fins		smooth surface	
Cask shape		vertical cylinder	horizontal cylinder	vertical cylinder		horizontal cylinder	
Transport means		railway, truck	railway, truck	railway, truck		railway, truck	
Empty weight	t	76.5	84	86.5		106	

4. APPLICATION OF BURNUP CREDIT FOR SPENT FUEL MANAGEMENT

4.1. Criticality calculations

The majority of the codes for isotope composition calculations of the spent fuel provide a low accuracy of the fission products concentration. Therefore in criticality calculations an actinide only burnup credit is taken into account. Actinide only burnup credit is a form of partial burnup credit that accounts for the changes in actinide isotopes due to exposure in the reactor, but ignores the reduction in reactivity caused by fission products that are neutron absorbers. The following isotopes are

considered in actinide only burnup credit: uranium-235, -238 and plutonium-239, -240, -241. The concentrations of fissile isotopes (U-235, Pu-239, Pu-241) are increased by an estimated error, and the values of thermal neutron absorbers (U-238 and Pu-240) concentrations are decreased by a similar error.

The parameter limiting the loading of spent fuel casks, is the average value of the burnup of the assemblies. The reactivity, caused by the non-uniform burnup over the length of the assembly, can be both positive and negative. Therefore, in criticality calculations of the cask all possible burnup distributions over the length of the fuel assembly are considered.

4.2. Definition of spent fuel burnup

The main problem in cask loading and in the design of storage facilities with allowance for burnup credit is the definition of excess of the allowable k_{eff} value, because of uncertainties in the spent fuel burnup. Careful documentation of fuel histories, calculational methods and data of in-core reactor measurements cannot give an accurate value of the burnup and its axial distribution. Therefore, the main method of measurement of burnup is its direct measurement. Here preference is given to the methods based on measurement of the neutron and photon radiation of the spent fuel (a passive method).

Criticality safety analyses for transportation of WWER-440 assemblies with initial enrichment of 4,4 % in the TK-6 cask have established, that the burnup should not be less than 25 GW·d/tU. The device for burnup measurement FAMOS-III was developed, made and delivered to the Kola NPP by NUKEM (GERMANY). The device had passed tests under real conditions, was certified by a commission of the Ministry for Atomic Energy and is recommended for application.

In Russia, a similar device is developed by " Khlopkin Radium Institute " and IPPE (Obninsk). The device was tested at the NPPs of New Voronezh, Kozloduy (Bulgaria) and Leningrad. The error in the measurement of the concentrations of the isotopes Cm-244, Ru-106 and Rh-106 was about 5% and of the average burnup 10 % for spent fuel with a cooling time not less than 3 years. There is an experimental device for burnup checking under development by a method of multiplication factor measurement in an assembly (active method).

4.3. Organizational measures

Research conducted in various countries has shown, that the largest mistakes in using burnup credit for transport casks are the mistakes made by staff during unloading of the spent fuel. The loading of not specified assemblies can be the result of two independent events: a mistake in the preparation of the loading documentation and an error during the loading of the cask.

To avoid the indicated mistakes the following measures are accepted:

- the quality of the documentation system is checked periodically;
- the records of each check are carried out;
- the records about the history of an assembly site and its burnup are kept during the whole storage period;
- the selection of fuel assemblies for transportation is established separately from main bulk of assemblies;
- a periodic training of responsible staff for fuel identification is carried out;
- allocation of places for an arrangement of fuel assemblies with low burnup.

4.4 Increasing Spent Fuel Storage Capacity

This problem has become more urgent in connection with the decision taken not to reprocess RBMK spent fuel and, consequently, the long-term spent fuel storage in quantities exceeding the storage capacities available. At present, densification of interim storage is suggested to be increased 1.5-2.0 times. The principal limiting factor for the design and densification rate to be chosen is a nuclear safety requirement.

Design studies conducted at the Physics-Power Institute (IPPE), Obninsk, showed that even though the 1986 decision on increasing initial fuel enrichment up to 2.4% being considered, burnup taken as a nuclear safety parameter allows to overcome the problem of densified RBMK fuel storage without using solid neutron absorbers. Consideration was given to various fuel assembly arrangement pitches adopted in storage projects and planned for their being densified: 250x160 mm, 230x110 mm, 250x80 mm (canister storage), 125x110 mm, 115x110 mm (storage without canisters). The analysis shows that, with any fuel arrangement pitches planned and even under emergency situations, the multiplication factor for an infinite lattice of assemblies does not exceed the value of 0.95, under the condition that the fuel is stored in canisters and its burnup exceeds 10 GW·d/tU (which corresponds to the average burnup of the first-year fuel campaign). If the fuel is stored without canisters, the burnup should be not less than 21 GW·d/tU.

During development of an additional storage project for the Ignalina NPP, the justification of criticality safety was fulfilled for the densified spent fuel storage without canisters and with a lattice of 136x136x112 mm, with initial enrichment of 2 % and with the assumption of absence of burnup. Under normal cooling conditions of this storage project, the multiplication factor is equal to 0.788 for such lattice. As design failure raising reactivity in the pool, the failure of system of cooling downturn of water density is considered. k_{eff} for such situation is equal to 0.96. This result does not meet the requirements of criticality safety. Taking into account a minimal burnup (2.5 GW·d/tU) the result is a $k_{eff} < 0.93$, which meets the criticality safety requirements. The project of the densified spent fuel storage with burnup credit was agreed by the supervision body of Lithuania and is implemented for the Ignalina NPP.

Assemblies, which have burnup less than 2.5 GW·d/tU, are as a rule discharged from the reactor by results of leak test and are stored in tight canisters within a lattice 250x160 mm. The planning for reloading the fuel of the Ignalia NPP (a RBMK-1500) is carried out with the help of the code "PLANER". The code "STEPAN-R" is introduced for selection of "candidate" assemblies for reloading. The average burnup for normally discharged assemblies is about 14.4 GW·d/tU.

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BURNUP CREDIT IMPLEMENTATION FOR SPENT FUEL MANAGEMENT OPTIONS IN SLOVAK REPUBLIC

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Abstract

In the mid 80's, an industrial project devoted to the improvement of the WWER fuel cycle was conducted in East European Countries and Soviet Union. The utilization of 4.4% enriched fuel for WWER-440 reactors was also part of the project. Taking into account burnup credit, this high enriched fuel was considered to meet the subcriticality requirement of the spent fuel storage facilities licensed for 3.6% enriched fuel.

1. INTRODUCTION

At present, there are four operating WWER-440 reactors in Slovakia at the Bohunice site. Four WWER-440 reactors are in different stages of construction at the Mochovce site. The spent fuel assemblies (SFAs), after discharge from the reactor, are stored in the at Reactor Storage (ARS) facility during 3 years at the NPP Bohunice (363 SFAs) and 6 years at the NPP Mochovce with compact racks (662 SFAs). After 3 years storage in ARS in Bohunice NPP, the spent fuel is transported to "Away From Reactor Storage facility" (AFRS) at the Jaslovské Bohunice site. The SFAs in AFRS are stored under pure water in open baskets placed in stainless steel lined concrete pools. The storage baskets also serve as a C-30 [1] transport cask internals used for SFAs transport.

The spent fuel storage facilities and transport casks are licensed for a maximum initial enrichment of 3.6%. The storage facilities and transport casks meet the criterion of subcriticality ≤ 0.95 filled with pure water and fresh fuel with enrichment 3.6%. The required subcriticality is maintained by distance between SFAs a in triangular lattice of 225 mm or 162 mm with boron steel hexagonal tube with 1% boron content in the case of compact racks. The subcriticality during the transportation of the SFAs and storage in the AFRS with higher enrichment up to 4.4% is maintained by utilization of burnup credit, which was evaluated with the WIMS-D4 code [2]. Rough estimates show, that a burnup > 9 MW·d/kgU is satisfactory for maintaining the required subcriticality for facilities licensed up to 3.6% enriched fuel.

The capacity of the AFRS will be expanded by the end of 1998 [1]. In order to ensure the storage capacity for all spent fuel produced in Slovakia, for a storing period of approximately 50 years, the Slovak Electricity Company (SE) initiated plans to build a new interim spent fuel storage facility. For that purpose tenders were invited by SE at the end of 1993. The capacity extension of the present AFRS was also part of the bidding procedure. The requirement for subcriticality in the bidding process was $k_{eff} \leq 0.95$ for fresh fuel with enrichment 4.4%.

2. CRITICALITY STUDIES OF THE AFRS AND TRANSPORT CASKS

As was mentioned in the previous part, the subcriticality of spent fuel storage facilities in Slovakia is $k_{eff} \leq 0.95$ for fresh fuel assemblies with an enrichment of 3.6%. In accordance to the original Soviet project, this is assured for an infinite triangular mesh of fuel assemblies with a lattice pitch of 225 mm in pure water. In the framework of the industrial project devoted to fuel cycle economy improvement of the WWER-440 reactors by introduction of higher enriched fuel (4.4%), the subcriticality of the storage facilities was evaluated. The objectives of the studies were to include

minor modifications of the transport casks and storage facilities and to fulfill the requirements determined by Regulation No 2/1978 and No 6/1980 [3,4] of the former Czechoslovak Atomic Energy Commission (CSKAE). These requirements are:

- to prevent criticality, through geometrically safe configuration or other physical means and methods even in the optimum moderation case;
- during the transportation and storage of the spent fuel the subcriticality of 0.05 must be ensured even in the case of postulated accidents.

The subcriticality calculation was done by WIMS-D4 code [5] for an infinite lattice of SFAs with initial enrichment of 4.4% in pure water moderator with a temperature of 40°C and a lattice pitch of 225 mm. No burnup profile for the fuel assemblies was taken into the account. Some results of the calculation are presented in [2] . The main conclusion of the calculation was that burnup exceeding 9 MW·d/kgU for 4.4% enriched fuel is enough for maintaining the required subcriticality after a 3 years cooling period. This rough estimation shows that SFAs with an initial enrichment of 4.4% are acceptable for transportation and storage at AFRS after the first burnup cycle.

3. THE PRESENT STATUS OF SPENT FUEL MANAGEMENT

3.1. AFRS extension [1]

In order to ensure the storage capacity for all spent fuel produced in Slovakia, for a storing period of approximately 50 years, SE initiated plans to build a new interim spent fuel storage facility. For that purpose tenders were invited by SE at the end of the year 1993. SE received 15 proposals from 15 different vendors. Most proposals have been dry storage solutions. The bid evaluation resulted in elaboration of a proposed short-list of bidders comprising four vendors. At that moment, it was decided by the SE top management to suspend the bid evaluation process for a certain period of time and to complete the evaluation of possibilities for capacity and lifetime extension of the existing AFR storage facility on the Bohunice NPP site. The capacity extension was achieved by compacting the array of fuel assemblies and by minor modification of the shape of storage baskets. The subcriticality calculations were done by SCALE 4.3 code [6]. The lattice pitch for infinite triangular mesh of 4.4% enriched fuel assemblies surrounded by hexagonal tubes of boron steel with 1% boron content 168 mm was determined. This arrangement doubles the present capacity (600 t of heavy metal) of AFRS to about 1200 t.

At present, the safety report for the extension of AFRS is in the stage of a review process at the NRA SR. The bid evaluation process will continue in a way and to an extent which depends on the results of the authorities reviewing the safety report, the approval of the design and other conditions for capacity extension of that storage facility.

3.2. Burnup credit criticality benchmark

The introduction of burnup credit in the fuel storage pool introduces some new points to be considered, in addition to the traditional criticality issues. Examples of such new points are:

- the effect of burnup profile;
- the effect of operating history;
- the accuracy of calculation methods with regard to burnup, fission product, plutonium buildup, etc.;
- the accuracy of calculation methods with regard to predicting criticality of spent fuel.

To address the above mentioned problems for WWER reactors a benchmark proposal [7] was formulated in 1996 in the framework of a research agreement "Atomic Energy Research" signed by

Eastern and Central European research utilities. The main goal of the benchmark proposal is to develop qualified tools for supporting license evaluations of spent fuel storage and transport facilities. Such a research activities are strongly supported by OECD/NEA. The Nuclear Power Plant Research Institute, Trnava from Slovak Republic participates in this benchmark problem with the SCALE 4.3 code.

The benchmark proposal is divided to four steps:

CE1 – to exercise and infinite array of a simple WWER-440 pin cell with fresh and spent fuel rod (criticality calculations) taking the burnup, cooling time and groups of nuclides as parameters;

CB2 – is devoted to comparison of the ability of various code systems and data libraries to predict spent fuel isotope concentrations using depletion analysis[8];

CB3 – study of axial burnup profile (end effect) in infinite array of WWER spent fuel rods;

CB4 – realistic spent fuel configuration (finite WWER spent fuel array) calculations for determining the burnup credit calculations are presented in [9]. Slight differences were found in the multiplication factors determined by different codes and also between calculations performed with the same code by different utilities.

4. CONCLUSIONS

Utilization of the burnup credit for spent fuel storage and transport facilities is foreseen in the case of slight change of the fuel enrichment in spent fuel management activities in Slovak Republic. Research activities are conducted for development qualified tools for supporting license evaluations of spent fuel storage and transport facilities.

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CREDIT TO FUEL BURNUP FOR CRITICALITY SAFETY EVALUATIONS 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. Analysis characteristics specific to burnup credit are described, namely the need to increase the experimental data to allow for a more detailed validation of the depletion codes, and of the criticality codes when applied to spent fuel. Reactivity effects that arise in burnup credit analysis, such as axial and radial effects, fuel irradiation history and others are revised. The methods used to address them in the approved methodologies are outlined. Finally, the regulatory approach used to accept these new analytical methodologies is described.

1. INTRODUCTION

Burnup credit has been taken 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 use of burnup credit. At present, credit to burnup is included in the criticality safety analysis of six out of the seven PWR units in operation, as well as in the two BWR units. The analyses refer to wet on-site storage in all cases, and cover a wide variety of fuel types for each plant.

There is an ongoing dual-purpose cask design programme 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 future designs of dry casks is currently under consideration. 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.

2. BURNUP CREDIT RELATED ACTIVITIES IN SPAIN

There is no experimental programme 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. Two Spanish engineering companies have developed burnup credit analysis methodologies for spent fuel storage systems. Both methodologies have been licensed, and the regulatory authority has accepted their applications at least once.

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 modeled 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.

3. BURNUP CREDIT ANALYSIS OUTLINE

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 maximizing the spent fuel reactivity. Second, a criticality calculation is performed using the specific characteristics and tolerances of the spent fuel storage system.

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 Gadolinium. 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 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 maximum (due to the decrease of the Gadolinium contents with burnup) is calculated, and the maximum lattice average enrichment that fulfills 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.

4. REACTIVITY PARAMETERS SPECIFIC TO BURNUP CREDIT APPLICATIONS

Fuel depletion calculations necessary for burnup credit criticality safety analysis introduce new parameters and effects that should be taken into account to obtain a correct 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 characterized by an average burnup value, there is an axial burnup distribution in the fuel assemblies, the burnup level being higher in the center 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 transport should take into account the specific burnup distributions of fuel coming from the different plants involved in the transport process.

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.

An equivalent process for BWR fuel is very difficult to formalize. The axial burnup shape of BWR fuel assemblies depends on too many factors, so that fuel with the same burnup level can have very different axial burnup shapes. 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.

The effect on reactivity of the radial distribution of burnup seems to be more limited. The limiting case would be that of a fuel assembly that has been loaded in the core periphery and has acquired there a big fraction of its final burnup. Although this situation is not very common, its possible impact on the final reactivity should be taken into account.

Another aspect that effects the calculated spent fuel reactivity is the irradiation history of the fuel. The depletion conditions assumed have to be chosen so that the spent fuel reactivity is maximized, in order to conservatively cover all the individual histories the fuel assemblies undergo. This issue is especially important for BWR fuel, where the spent fuel composition is heavily affected by the void fraction and control history assumed for the depletion. Especially the void fraction history and its related quantities (power density, fuel temperature) should be carefully studied. The same effects are present for PWR fuel. However, the impact on the spent fuel' reactivity is much reduced than in the case of BWR.

The spent fuel undergoes some reactivity changes after it is discharged from the core. The build-up and decay of the fissile and absorber nuclides make the reactivity change with time during the storage period. A bounding isotopic composition is usually considered, by eliminating the short lived absorbers for the reactivity calculation.

5. REGULATORY STATUS OF BURNUP CREDIT IN SPAIN

As already said above, burnup credit has been licensed in Spain for wet storage of PWR fuel, as well as credit to the integral burnable absorbers for wet storage of BWR fuel. The subcriticality criteria applied for these analyses have been the same than those used for analysis based on fresh fuel.

Burnup credit is a step forward towards making criticality safety analysis more realistic. When a safety analysis follows a realistic approach, whatever partially, the calculation uncertainties have to be determined with enough accuracy. However, uncertainties inherent to burnup credit methodologies are not well known.

The first problem found is that the validation of the depletion codes is quiet reduced, due to the few experimental data available in this field. Hence the uncertainty of each isotope's concentration is not well known. In some cases, comparisons of design calculations with plant data (critical boron concentrations, power distributions) have been used to support the quality of the depletion calculation performed. However, this approach cannot be accepted without a thorough revision for the following reasons:

- A core calculation includes a 3-D calculation after the depletion. When comparing with plant data, the uncertainty of the full calculation system is determined, and not only that of the depletion code. Undetected error compensation between the different intervening codes is possible.

- Fuel with different burnup levels is loaded in the core. Dependence of the uncertainty with the calculated burnup level cannot be determined, as only core average values can be obtained.

- Plant measurements can have a high uncertainty, and are difficult to use as benchmark quality data.

The same problem can be found regarding the validation of the criticality codes: the number of spent fuel criticals and fission product reactivity worth measurement is reduced, and does not allow for a complete validation of the codes.

As a result of these uncertainty quantification problems, the regulatory approach followed in Spain has been to make sure that the burnup credit methodology submitted is conservative as a whole, i.e. that there are reactivity margins embedded to compensate for the "weak" parts of the methodology. The burnup credit level accepted is hence case dependent. The specific nuclides credited depend on the conservatism of the methodology. In practice, as all the submittals received up to now for PWR have been directed to wet storage, with soluble Boron in the pool water, credit to actinides and fission products has been accepted. Even in those cases, short-lived fission products have been discarded, as well as the absorber nuclides not individually modeled in the depletion code (lumped fission products).

For BWR fuel storage burnable poison credit analyses, a similar credit level has been accepted. The conservative margin comes in this case from at least two sources: first, the highest reactivity in the fuel's life is considered; second, all the fuel is assumed to be axially uniform with the highest reactivity.

Another aspect that should be considered when accepting burnup credit submittals is the coherence of the analysis with the plant procedures and practices. Each plant has different tools to determine the burnup value of each individual assembly, with different accuracy. A high uncertainty in the burnup determination of the fuel assemblies may invalidate a burnup credit analysis if not taken into account properly or again if the reactivity margins available are not wide enough.

For PWR fuel storage submittals, no direct burnup measurement has been required for the same reasons already stated. However, the burnup determination procedures have been revised. It is envisaged that submittals for dry fuel transport would need to be supported by fuel burnup measurement prior to loading.

Finally, it has to be stated that burnup credit applications make use of the specific plant operating experience accumulated in the past. The analyses do rely on information that has been gathered from fuel burnt in previous cycles, i.e. burnup shapes, power densities, void fraction histories, and operating conditions in general. Any departure from the operation practices and conditions maintained in the past should be evaluated to determine if it has an impact on the analysis continued validity. It is frequent to reach burnup levels that are higher than those included in the fuel database available (PWR), or to introduce changes in the fuel's void fraction history (BWR). Extrapolation of the analysis results to include the new fuel conditions should be carefully performed.

BURNUP CREDIT IN SWEDEN

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Abstract

Burnup credit is not used at present in the Swedish nuclear programme. A central facility for storage of spent nuclear fuel credit for integral burnable absorbers is approved for BWR-fuel. Actinide plus fission product burnup credit for BWR and PWR is planned to be used in the deep repository of spent nuclear fuel.

1. INTRODUCTION

The Swedish nuclear programme consists of twelve nuclear power plants (NPPs), a transport system for spent nuclear fuel and radioactive waste, an interim storage facility for spent nuclear fuel and a final storage facility for high and medium level radioactive waste. The twelve NPPs consist of three plants of the PWR type and nine of BWR type. The total power generating capacity is 10040 MW(e). The plants are located at the Swedish coast: Three BWR-plants at Forsmark, three BWR-plants at Oskarshamn, two BWR-plants at Barsebäck, one BWR-plant and three PWR-plants at Ringhals.

All spent nuclear fuel from the power plants is transported to the central interim storage facility (CLAB) which is located at Oskarshamn close to the three Oskarshamn reactors. The spent nuclear fuel transport system consists of a specially designed transport ship, M/S Sigyn, ten spent fuel transport casks of the TN17/2 type and a number of cask transport vehicles for land transportation.

After the interim storage period the fuel will be prepared for final disposal. For that purpose an encapsulation plant is planned in connection to CLAB. In the encapsulation plant the fuel will be placed in canisters of steel and copper. These canisters will later be placed in the planned underground final deep repository in the bedrock. The location of this repository is not yet selected.

2. STORAGE AND TRANSPORTATION OF SPENT NUCLEAR FUEL

2.1. Licensing criteria

The design criteria for spent fuel storage are based on the US NRC Regulatory Guide 1.13 - Spent Fuel Storage Facility Design Basis. A draft of the proposed revision 2, published in 1981, addresses burnup credit. Revision 2, still in draft, has become accepted as the design basis for spent fuel storage facilities.

The basic design criteria for spent fuel storage used in Sweden are:

- Assumption that the fuel in the storage facility is fresh fuel;
- The multiplication factor k_{eff} should be less then 0.95 during normal operation;
- For unlikely accident events k_{eff} should be less than 0.98.

2.2. Storage at nuclear power plants

The spent nuclear fuel storage capacity at the NPPs varies from one year up to several years of spent fuel production. In some plants poisoned storage racks are used. The enrichment limit for the storage facilities varies from plant to the plant, but can be summarized as follows:

- for PWRs maximum 4% U-235 in the fresh fuel;

- for BWRs maximum 5% U-235 in the fresh fuel.

Burnup credit is not used and there are no plans to introduce burnup credit at the plants.

2.3. Transport system

The spent fuel is transported to CLAB in transport casks of the type TN17/2. These casks are licensed in France where the licensing procedure is based on the maximum enrichment in the fuel rod. The enrichment limits are different for different fuel types. Burnup credit is not used presently.

However, during the last year the transport capacity of the casks has been reduced due to revised calculation procedures in France. In order to restore full transport capability, work is in progress to change the licensing procedure from maximum rod enrichment to assembly average enrichment. This will give a considerable advantage, in particular for BWR fuel, where for example an assembly with average enrichment of 3.3% U-235 could have fuel rods with an enrichment of 4.8% U-235.

Burnup credit is not considered at present but it could be an option for the future and should not be ruled out for the transport system.

2.4. Interim Storage (CLAB)

The present storage capacity in CLAB is 5,000 t uranium and the licensed maximum assembly average enrichment is:

- for PWR: 4.2 % U-235 in the fresh fuel;
- for BWR: 4.2 % U-235, with credit for burnable poison (integral absorbers in the fuel).

For PWRs, the licensing basis is the average assembly enrichment and fresh fuel. For BWRs, credit for burnable absorbers is taken into account, which means that the maximum reactivity of the fuel will occur at a burnup between 4,000 - 6,000 MW·d/tU which has to be considered in the design calculations.

Full burnup credit was studied for CLAB during 1990 - 1991. Based on the results of the study, it was decided that burnup credit was not the preferable method to increase the capacity in CLAB. Instead, canisters with borated steel are used. The study is summarized in the next paper.

2.5. Encapsulation plant and deep repository

Different concepts of final storage canisters are being studied. Presently, the plan is to use a storage canister (capsule) which could contain 12 BWR or 4 PWR assemblies. The assemblies will be stored in a close pattern. For long-term storage, credit for neutron absorbing materials in the capsule can not be taken into account since it has to be assumed that the capsule could be filled with water. This means that, in order to control reactivity, credit for burnup may have to be used. Based on the calculations performed for CLAB it has been concluded that burnup credit could provide sufficient reactivity margin for safe long-term storage in the suggested storage capsules.

3. SUMMARY

Burnup credit is presently not used in Sweden, but could be an option for the transport system in the future. Credit for burnable absorbers (integral burnable absorbers in the fuel) has been licensed for wet storage. Burnup credit will be considered for the final repository. Feasibility studies have so far shown that burnup credit is a possible way to control the reactivity in the final repository geometry.

BURNUP CREDIT IN THE CENTRAL STORAGE FACILITY FOR SPENT NUCLEAR FUEL IN SWEDEN

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Abstract

During 1990 - 1991, SKB AB evaluated if burnup credit could be an acceptable method to increase the storage capacity of spent nuclear fuel in the Central Storage Facility for Spent Nuclear Fuel in Sweden (CLAB). The result of the evaluation was, that burnup credit would not give a sufficient reactivity margin in the required storage pattern. The conclusion of the study was then, that burnup credit alone was not a suitable way to control the reactivity in CLAB. The burnup credit activities are summarized in this paper.

1. BACKGROUND

The original storage capacity in CLAB was 3,000 t spent uranium fuel. During 1990 - 1991, the Swedish nuclear programme was expected to generate about 7800 t of spent uranium fuel up to year 2010. An extension of the storage capacity was needed to be completed before 1996. In CLAB the fuel assemblies originally were placed in storage canisters containing either 16 BWR or 5 PWR assemblies, see Figure 1. Within the measurements of the canister it is shown, that it was possible to store 25 BWR assemblies instead of 16 in the BWR canister. In a PWR canister, it was possible to store 9 instead of 5 assemblies. This change would increase the storage capacity in CLAB from 3,000 to 5,000 t in the existing pools. SKB decided to evaluate whether burnup credit was a suitable way to control reactivity in a closer packed storage pattern in CLAB.

2. PROBLEM DEFINITION

In the original CLAB design, the nuclear safety analysis is based on the assumption that the fuel from a reactivity standpoint is fresh fuel. Since the fuel assemblies in CLAB normally have reached a considerable burnup, this assumption means that in reality a substantial concealed reactivity margin exists. Burnup credit means that this concealed margin could be used to store the fuel in a denser and more reactive storage pattern. Initial average enrichment and assembly average burnup are the key parameters that control the reactivity of a fuel assembly, but also other parameters, such as burnup profile and void history profile, are important, in particular for BWR-fuel. The purpose of the study was to determine what combinations of enrichment and burnup could be accepted in the proposed storage pattern without exceeding the limiting reactivity value.

3. DESIGN CRITERIA

Design criteria for spent fuel storage could be found in the US NRC Regulatory Guide 1.13 - Spent Fuel Storage Facility Design Basis. A draft of the proposed revision 2, published in 1981, addresses burnup credit. Revision 2, still in draft form, has become accepted as the design basis for spent fuel storage facilities. The design criteria of the original storage pattern in CLAB were based on the Regulatory Guide 1.13:

- The multiplication factor k_{eff} should be less than 0.95 during normal operation;
- In unlikely accident events k_{eff} should be less than 0.98;
- Unirradiated fuel is assumed to be stored;
- No credit is given to burnable absorbers in the fuel;
- No soluble neutron absorbers in the pool.

In the original design, the k_{eff} in CLAB is close to 0.95 both during normal and accident conditions.

4. CALCULATIONS

4.1. Assumptions

The design target for CLAB, at the time of the study, was to be able to store PWR fuel of an average initial enrichment of 4.2% U-235 and BWR fuel of 4.0 % U-235 (Later this was revised to 4.2% U-235 for both PWR and BWR, which is the current limit). Earlier calculations have shown that the fuel type SVEA-64 of the ABB Atom design is the most reactive BWR-fuel within the present Swedish programme. Therefore, this fuel type was chosen as the reference BWR-fuel in the calculations. Later, checks were made for other fuel types. For the PWR-case a standard 17x17 fuel assembly basically of the Westinghouse design was chosen. Figure 1 shows the canister geometry for BWR and PWR fuel. The new BWR-canister contains 25 assemblies and the PWR-canister 9 assemblies.

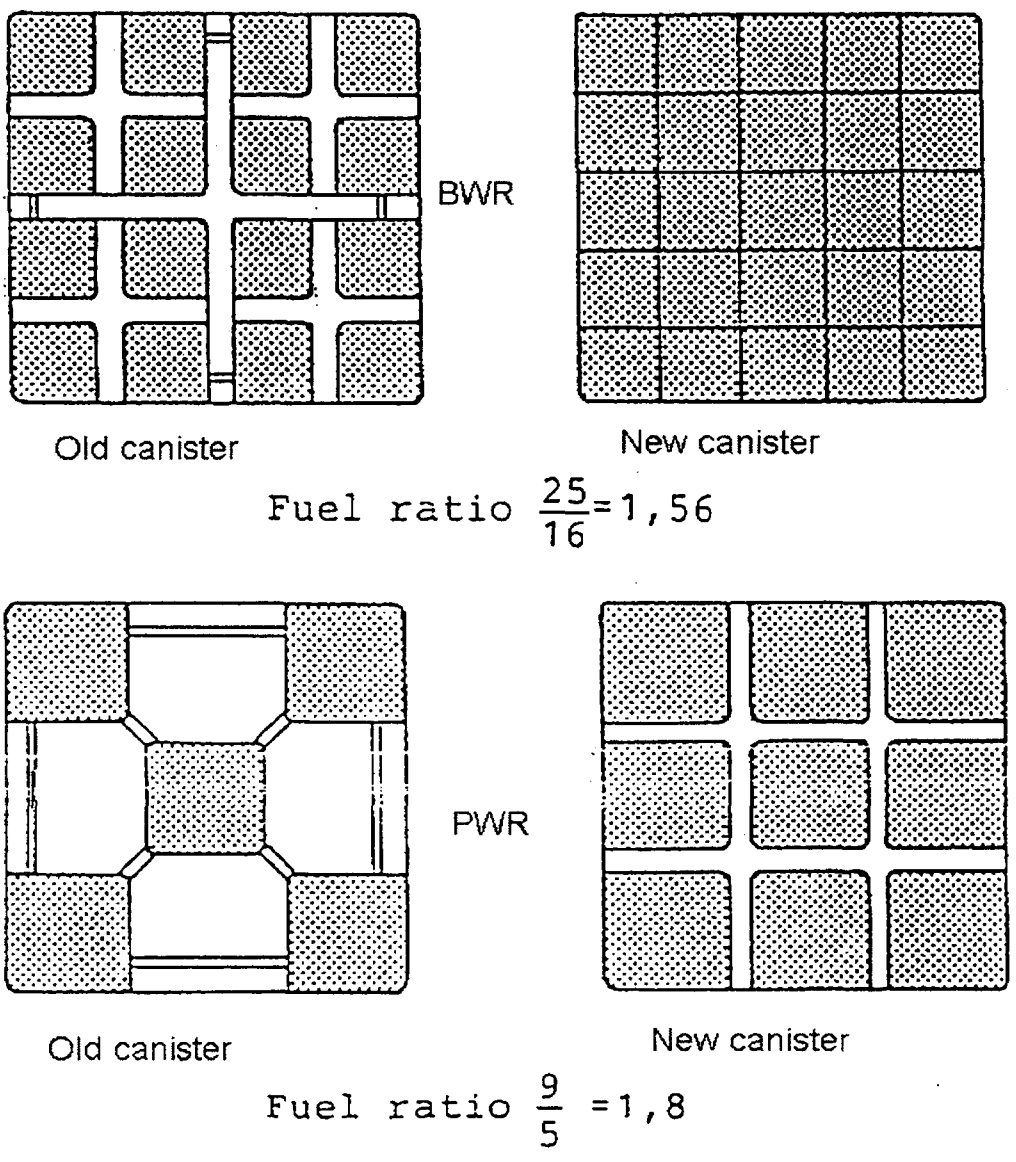

FIG. 1. Geometry of canister.

4.2. Methods and tools for calculations

The calculations have been performed by ABB Atom. PHOENIX was used for pin cell depletion calculations for both BWR and PWR assemblies. KENO IV was used to represent fuel, canister geometry, water gaps and surrounding water in the storage geometry. The three-dimensional static code POLCA was used to calculate fuel burnup in standard core tracking mode and to calculate the 3D effects of burnup in storage geometry.

4.3. Calculations performed

4.3.1. Homogenous burnup

Preliminary calculations were performed to assess the feasibility of burnup credit in CLAB. Calculations were performed for BWR fuel of SVEA-64 type and for PWR fuel of 17*17 type in the geometry of the new canister design under storage conditions. A 3-D KENO IV-model with a flat axial and radial burnup distribution was set up to calculate combinations of average burnup and initial enrichment to find the limit curve k_{eff} = 0.95. Based on the PHOENIX results at different burnups, initial enrichments and the buckling from KENO, k_{eff} values were calculated for fuel at different burnups and for different initial enrichments. It showed that the required burnup was around 33.6 MW·d/kgU and 33.3 MW·d/kgU, respectively for 4.0% enriched BWR and 4.2% enriched PWR fuel.

Preliminary limiting curves based on these calculations are shown in Figure 2. The expected average discharge burnup is also shown. The results show a reasonable large margin to the expected discharge burnup. The conclusion is that a fuel assembly burnup of 60% of the design burnup could be sufficient for safe storage. It should be noted that these results do not include any uncertainties.

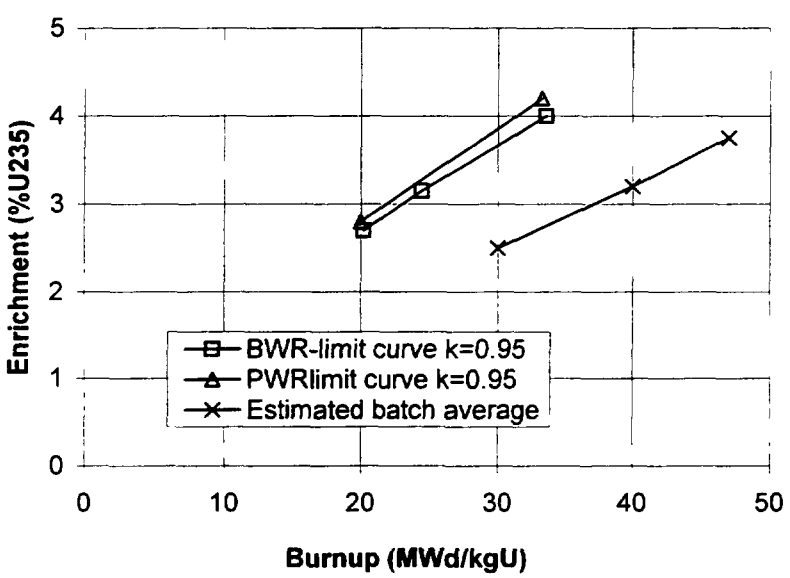

FIG 2. Preliminary burnup requirements.

4.3.2. Axial burnup variation

The next step was to assess the reactivity effect of the axial distribution of the burnup. The 3D KENO IV model of BWR fuel was used and the burnup distribution was modeled axially. The strategy was to keep the average burnup constant and try to model the shape of burnup distribution at the top of the core in a simplified way. The following results were obtained:

- If a flat axial burnup profile was changed to model the decrease of the burnup at the top in the fuel with one step, the reactivity increased 11%;
- With a more detailed model of the burnup distribution, the reactivity increase was 8 %.

The results appeared to be very dependent on the degree of detail in the model. KENO showed not to be a suitable tool to evaluate the effect of burnup distribution. There are two reasons for this. Cross section generation is cumbersome and the statistical uncertainty makes it difficult to conclude about distribution effects. Based on these results it was concluded that an alternate way to assess the reactivity effect of the burnup distribution had to be developed.

Two important factors shift the reactivity upwards in burned fuel: the downward skewed burnup and the high void content in the upper part of BWR cores during operation. The high void content results in a harder neutron spectrum which results in a higher conversion factor with a higher degree of Pu build up and lower U-235 burnup. This effect is particularly strong in BWR. A similar but smaller effect is observed in PWR fuel due to the higher temperature at the top of PWR cores.

A method to account for axial effects was developed. The basis for the method is the fact that the CLAB geometry is similar to the reactor geometry in cold conditions. This makes POLCA an excellent tool to evaluate the 3D conditions. First, it had to be shown that the axial burnup distribution can be separated from the radial distribution. Axial burnup distributions from a number of batches of discharged fuel from the reactor Oskarshamn 2 were studied. The study showed that the axial burnup distribution has about the same shape in all assemblies.

It is known from shutdown margin evaluations that the flux distributions in cold BWR cores are heavily tilted upwards, see Figure 3. This fact has been used earlier to develop a weighting scheme to find the worst point for shut down margin calculations. The local k value could be calculated using a weighting scheme: $k_{inf} = \Sigma w_i k_i / \Sigma w_i$, where the neutron flux weighting factor w_i is taken from Figure 3. The same weighting scheme was applied to CLAB. The consequence is that only the upper part of the fuel has any significance for the criticality safety. The weighting scheme can also be applied to burnup, as reactivity and burnup have an almost linear relationship:

$$E_{eff} = \Sigma w_i E_i / \Sigma w_i$$

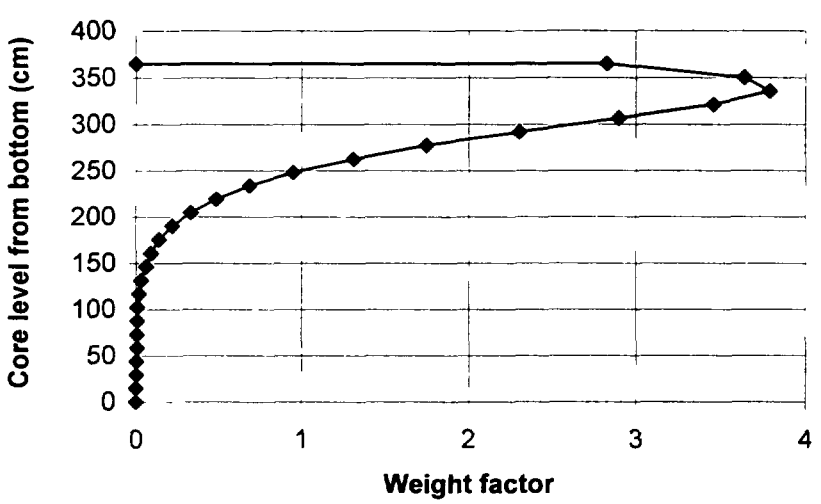

FIG. 3. Axial weight factors for BWR-fuel.

The axial burnup effect can then be expressed as a burnup penalty, e.g. weighted axial burnup minus average burnup (E_{eff}-E). This weighting scheme was applied to predict end of life burnup distributions on around 120 discharged BWR fuel assemblies from Oskarshamn 2. The result was an average burnup penalty of 3.86 MW·d/kgU with a standard deviation of 0.82 MW·d/kgU.

In order to convert the burnup values to reactivity a 3D core model of POLCA was used to calculate the reactivity for several cases. The k_{eff} difference between a core with flat burnup distribution and a core with a realistic axial burnup distribution was calculated. The average conversion constant was found to be 1.22 ± 0.099 k_{eff}/E. Combining the burnup penalty with the conversion constant an axial burnup penalty of 0.048 ± 0.011Δk was obtained. Including two standard deviations the final burnup penalty was determined to 0.07 or 7% Δk.

4.3.3. Design margin

A design margin has to be applied to cover effects that are not considered at this stage. The study has only investigated tentative designs of the canister. All dimensional and material composition variations have to be covered. Accidents with reduced criticality safety have not been identified. Some cases of handling of the canisters could be found to cause reactivity increases. Fuel with empty rod positions causes sometimes a slight reactivity increase. Fuel without channels have not been analyzed as most fuel will be stored with its belonging channel, etc.

For the time being all the above uncertainties are covered by a design margin of 0.02 in reactivity. This value is based on experience from previous design work. For example the accident analysis in the original design of CLAB revealed accident sequences that required an additional margin of around 2 % in reactivity.

4.3.4. Uncertainty margin

The uncertainty margin includes code verification and calculational uncertainties as well as material and fabrication tolerances, model uncertainties and uncertainties in burnup. In a final licensing a thorough uncertainty analysis is required. Burnup credit makes this more complicated but the uncertainties are not necessarily larger, since reactor operation has inherent healing effects on systematic errors and variations in reactivity, i.e. manufacturing errors. Throughout this study a total uncertainty of 0.03 has been used.

4.4. Resulting burnup requirements

The necessary margins are then as follows for BWR- fuel:

- Axial penalty 0.07;
- Design margin 0.02;
- Uncertainty margin 0.03;
- Total required margin 0.12.

For PWR the axial penalty was estimated to 0.02 and the total required margin is 0.07.

The required value is converted into burnup and then added to the calculated enrichment - burnup curve in Figure 2. The resulting limit curve gives the burnup required for each initial enrichment for k_{eff} = 0.95 in CLAB storage conditions, which can be seen in Figure 4. All assemblies on the right side of the limit curve are acceptable for storage in the new canisters. It can be seen from Figure 4 that the margin from the required burnup to the expected batch average is small.

From this it is evident that already the normal variation of burnup will cause some assemblies to fall short of the required burnup. This fact indicates that burnup credit alone is not sufficient to control reactivity in CLAB. Actual values on average burnup and initial enrichment for 7,103 BWR

fuel assemblies in CLAB were compared to the calculated burnup requirement. Figure 5 shows the result of this analysis for BWR fuel. It can be seen that a large number of assemblies fall outside the burnup requirement, in fact around 40 % of the fuel assemblies could not be placed in new canisters. The reason for this is that the burnup values show a large spread around the discharge target value.

For PWR, however, almost all of the 665 fuel assemblies in CLAB fall on the right side of the limit curve, Figure 6. It is therefore possible, to store all PWR fuel in CLAB in the new canisters if credit is given to burnup.

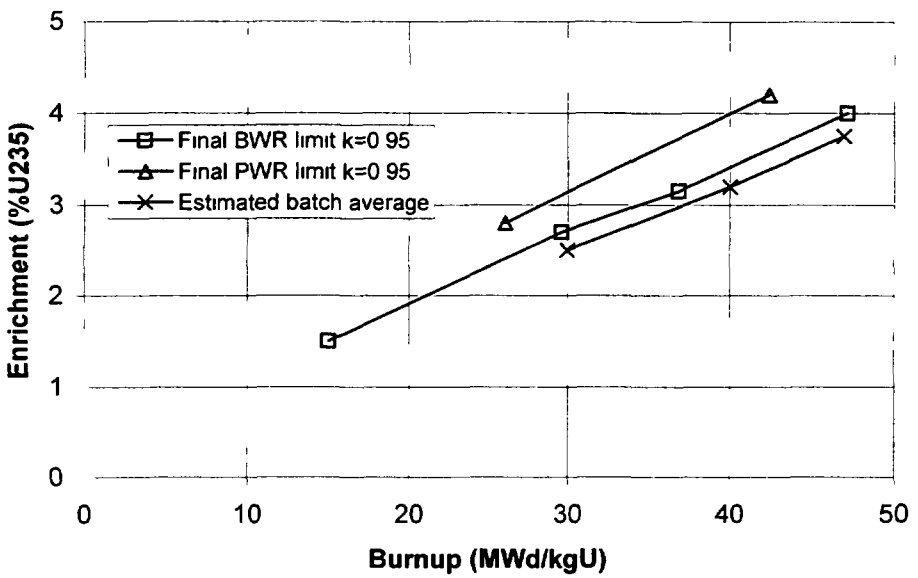

FIG. 4. Final burnup requirements.

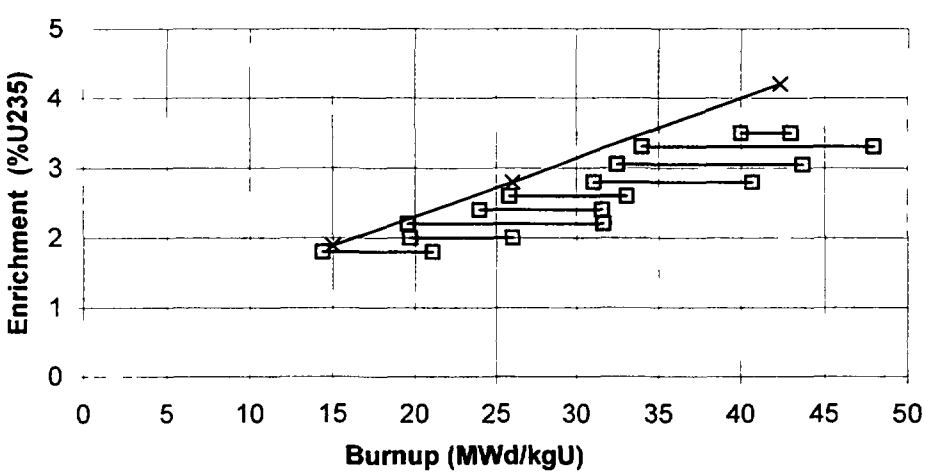

FIG. 5 Final limit curve (k=0.95) compared with CLAB PWR-fuel inventory.

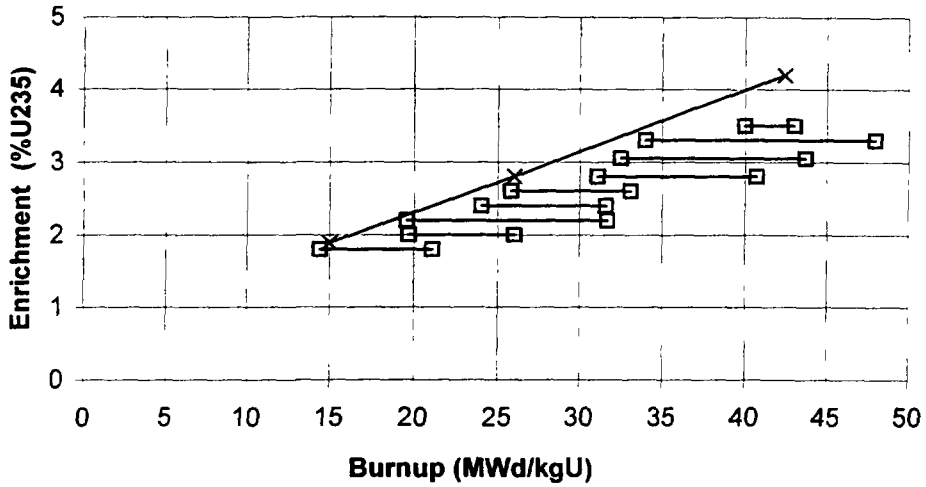

FIG. 6. Final limit curve (k=0.95) compared with CLAB PWR-fuel inventory.

5. CONCLUSIONS

Based on the results of the study, it was decided that burnup credit was not the preferable method to increase the capacity in CLAB. Instead canisters with borated steel was used.

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STATUS OF BURNUP CREDIT IMPLEMENTATION IN SWITZERLAND

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Abstract

Burnup credit is currently not used for the storage of spent fuel in the reactor pools in Switzerland, but credit is taken for integral burnable absorbers. Interest exists to take credit of burnup in future for the storage in a central away-from-reactor facility presently under construction. For spent fuel transports to foreign reprocessing plants the regulations of the receiving countries must be applied in addition to the Swiss licensing criteria. Burnup credit has been applied by one Swiss PWR utility for such transports in a consistent manner with the licensing practice in the receiving countries. Measurements of reactivity worths of small spent fuel samples in a Swiss zero-power research reactor are at an early stage of planning.

1. GENERAL SITUATION OF SPENT FUEL MANAGEMENT IN SWITZERLAND

Spent fuel from all the Swiss nuclear power plants (3 PWRs, 2 BWRs) is stored in pools at the reactors. The PWRs 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. For the future, utilities are also considering long-term storage of spent fuel, particularly if this alternative should prove economically more attractive than reprocessing. A central storage facility for all types of radioactive wastes, which will also provide room for dry storage of spent fuel in casks, is currently under construction.

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 licenses 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 license is based.

The licensing of storage systems is based on the US NRC Regulatory Guide 1.13, Appendix A. The containers used for spent fuel transports to foreign reprocessing plants are licensed in their countries of origin. The original licenses 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 INTENDED USES OF BURNUP CREDIT

No credit for burnup is taken for the storage of spent fuel in the reactor pools in Switzerland and there is no intention currently to do so in the future. 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, i.e., the storage pools are designed and licensed for the peak reactivity of the fuel in its lifetime under consideration of burnable absorbers. The licensing of BWR pools is based on the maximum k_{inf} of the fuel assemblies at cold conditions in reactor core geometry.

There is no dry storage of spent fuel in Switzerland at present. A central away-from-reactor storage facility is under construction, and the first proposals for storage casks have been submitted to the licensing authorities. Utilities are interested in taking credit of burnup for storage in this facility, but no concrete action has yet been taken, and the level of burnup credit to be applied is not yet defined.

All the shipments of spent fuel from the Swiss reactors have been to foreign reprocessing plants. Therefore, these transports have to fulfill not only the Swiss licensing criteria, but also those of the receiving countries and those of the original licenses of the transport casks.

To date, burnup credit has only been taken by one PWR utility for fuel whose initial enrichment exceeds the licensed limits of the reprocessing plants. For reprocessing in France, the following conditions, which are consistent with the French regulations, are applied:

- For initial enrichments up to 3.5%, no credit is taken of burnup, because the reprocessing plant and the transport casks are licensed for fresh fuel with 3.5% enrichment.
- For initial enrichments between 3.5% and 3.75%, a qualitative burnup check, showing that the assembly has been irradiated, is required.
- For initial enrichments exceeding 3.75%, a quantitative burnup measurement along the entire assembly length has to be performed. The average burnup of the least depleted 50 cm at the top of the active length has to exceed an enrichment-dependent limit. Burnup credit is taken for uranium and plutonium, but not for higher actinides or fission products.

The British reprocessing plant has a higher licensed limit for the initial enrichment. Burnup credit has been accepted for shipment of a few assemblies exceeding this limit based on the study performed for the reprocessing in France.

4. COMPUTATIONAL METHODS USED

A large number of criticality calculations for storage pools have been performed since the 1970's by Paul Scherrer Institute (PSI) as a small part of its activities in the field of light water reactor neutronics. Calculations have been done for the Swiss utilities, manufacturers of storage racks, and the licensing authorities. For all these calculations the BOXER [1] code developed at PSI was used. BOXER is a cell and two-dimensional transport and depletion code primarily intended for the generation of assembly-averaged few-group cross sections for core simulations. It has sufficient flexibility and calculational accuracy to allow also the modeling of more complex configurations, with the limitation that the two-dimensional transport calculations are performed in Cartesian x-y geometry and with homogenized cells.

The BOXER cross section library is based on JEF-1 nuclear data (except Gd-155 from JENDL-2 and Zircalloy-2 from ENDF/B-4). It contains cross sections in 70 energy groups (69 group WIMS structure plus one group between 10 and 15 MeV). Resonance cross sections are given in point wise lists between 1.3 eV and 907 eV and tabulated as a function of temperature and dilution cross section at higher energies. The burnup chains comprise 34 actinides (from Th-232 through Cm-248), 55 explicit fission products, and two pseudo fission products.

In BOXER, self-shielded resonance cross sections are determined by a point wise two-region collision probability calculation between 1.3 and 907 eV, and by tabular interpolation above this range. The group wise cell calculation is performed by means of an integral transport method in cylindrical geometry. The two-dimensional transport calculations are performed using a transmission probability integral transport method for homogenized cells. Depletion calculations are performed using reaction rates collapsed to one group by weighting with the fluxes from the two-dimensional calculation for each material in the configuration.

BOXER has been validated successfully against a large number of critical experiments and international benchmark problems, including both uniform lattices and configurations containing features representative of storage pools and transport casks (such as neutron absorbers and metal reflectors). The depletion calculations were verified against Yankee-Rowe assay results and benchmark problems such as the OECD/NEA benchmarks on burnup credit and on recycling of reprocessed uranium.

The MCNP continuous energy Monte Carlo code is also available at PSI, but it has not been used for criticality calculations to date.

5. RESEARCH AND DEVELOPMENT ACTIVITIES

5.1. Analytical

PSI has participated in some of the OECD/NEA criticality safety benchmarks since the benchmark group was founded. The current series of benchmarks dealing with burnup credit have also been calculated with BOXER, and the results compare well with those of the other participants.

5.2. Experimental

A programme of LWR integral experiments is currently in preparation 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 will be carried out in cooperation with the Swiss utilities and their fuel vendors and also be partly funded by them. In the first phase, scheduled to start in early 1998, the test zone will consist of 9 real, full-length BWR fuel assemblies. The major part of the measurements in this phase will deal with pin power distributions 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.

A second phase of the LWR-PROTEUS experiments, dealing with PWR lattices, is presently at an early stage of planning. The idea is to construct a test zone of real, full-length PWR fuel rods which, after completion of the measurements, will be reconditioned to actual fuel assemblies and loaded in the power reactor. As part of this PWR programme, it is planned (and project partners are very interested) to measure reactivity worths of small samples of spent PWR fuel with a range of burnups in this test zone. The configuration of LWR-PROTEUS with a long test zone (4 m) and a shorter driver (approx. 1 m) is very suitable for sample oscillation experiments, which allow the measurement of small reactivity changes (a few cents) with high accuracy. After the reactivity measurements, the samples will be analyzed for actinides and fission products in the PSI hot laboratory.

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THE APPLICATION OF BURNUP CREDIT FOR SPENT FUEL OPERATIONS IN THE UNITED KINGDOM

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Presented by B. Lake

Abstract

This paper begins by outlining the structure of the nuclear industry in the United Kingdom. It then sets out the methodology of burnup credit, and provides a brief discussion of the validation and robustness of the calculational route. This leads to a description of both the current and intended applications of burnup credit in the United Kingdom.

1. INTRODUCTION

Criticality safety assessments for irradiated fuel operations have historically assumed the fuel to be unirradiated. In the vast majority of cases, this is a conservative assumption, as it neglects the reduction in fuel reactivity that is known to occur due to the overall depletion of fissile material and the build-up of neutron absorbers in the fuel. This 'fresh fuel' methodology has largely been adopted for the sake of calculational and operational convenience, since it provides a simple and pessimistic demonstration of criticality safety whilst not constraining the plant operations.

Although this approach has often provided operational flexibility, it has always been recognised its application may impose a considerable economic burden upon the design and operation of a nuclear plant. This might be through the imposition of restrictive criticality controls or the implementation of neutron poisons to maintain criticality safety. With the current trend towards the use of high burnup (HBU) and mixed oxide (MOX) fuel in thermal reactors world-wide, it is clear that the magnitude of this financial penalty will increase. In some cases, such as spent fuel transport casks, it may become impossible to demonstrate criticality safety for the latest fuel types. This therefore leads to a choice of one or more of the following options:

i. to redesign the plant, flask or storage system;

ii. to operate with more restrictive limits, such as a reduction in the flask fuel load;

iii. to undertake calculations of spent fuel reactivity that more closely represent the real nature of the spent fuel, i.e. remove some of the pessimism in the criticality safety assessment by claiming credit for the fuel irradiation.

It is clear that significant cost savings can be realised if credit is taken in criticality safety assessments for the reduction in fuel reactivity that occurs with irradiation. Potentially, the application of burnup credit in the criticality safety assessments for spent fuel operations may have other benefits, since it could result in extended operational lifetimes for existing plant and flask designs. Moreover, for new storage and transport systems, burnup credit offers massive benefit in design optimisation, either through enhanced fuel payloads or the elimination of fixed neutron poisons from the design.

2. THE UK NUCLEAR INDUSTRY

2.1. Regulation

Regulation of the UK nuclear industry is undertaken by two government bodies, these being:

i. HM Nuclear Installations Inspectorate (HM NII), which regulates the design and operation of nuclear plants;

ii. the Department of Environment, Transport and the Regions (DETR), which is the competent authority in the UK for the licensing of nuclear transport packages against the requirements of the IAEA transport regulations.

2.2. Nuclear Plant Operators

In the context of this paper, concerning the application of burnup credit to spent fuel operations in the UK, the major operators of nuclear plant are:

i. Nuclear Electric, which operates Advanced Gas-Cooled Reactor (AGR) stations in England and Wales, and the Pressurised Water Reactor (PWR) at Sizewell;

ii. Scottish Nuclear, which operates Advanced Gas-Cooled Reactor (AGR) stations in Scotland,

iii. British Nuclear Fuels (BNFL), which operates the Magnox stations in the UK, in addition to the manufacture, transport, storage and reprocessing of Magnox, AGR and LWR fuel at their various sites.

3. THE BURNUP CREDIT METHODOLOGY

The generally accepted methodology for the implementation of burnup credit is to perform reactivity calculations for irradiated fuel, using fuel material compositions generated by an inventory prediction code, usually in order to determine the limiting burnup required for that fuel in a particular environment. In addition, it has always been recognised that the regulators may require a confirmatory measurement of burnup to be made prior to certain operations, such as the loading of fuel into a transport flask or the shearing of fuel into a dissolver, particularly where large burnup credit claims are being made. This requirement is being encompassed by the latest edition of the IAEA Transport Regulations [1], which forms the basis of transport approvals world-wide.

The burnup credit strategy therefore relies upon three key components of:

i. the inventory prediction;

ii. the reactivity calculation;

iii. the quantification and verification of the fuel burnup.

These components will be discussed in turn within this paper, with particular attention being given to the methods involved, the current validation status, and their robustness for successfully attaining regulator approval for burnup credit.

4. INVENTORY PREDICTION CALCULATIONS

4.1. The Inventory Prediction Problem

During irradiation of the fuel in the reactor, the reactivity of the fuel is reduced by a net loss of fissile isotopes and a build-up of neutron poisons in the form of fission products and non-fissile actinides. In certain fuel types which incorporate integral burnable poisons, the variation in reactivity with irradiation is complicated by the depletion of the neutron poison over the initial irradiation of the fuel. In typical commercial reactor UO_2 fuel, the most important fissile isotopes are U-235, which was present at the start of irradiation, and Pu-239, which is produced in the core by neutron captures in U-238. Most neutron absorptions in Pu-239 result in fission, but some neutrons will undergo capture to

form other actinides, some of which act as neutron poisons (notably Pu-240) or which may themselves be fissile (e.g. Pu-241).

The fission process yields a range of fission products, with atomic numbers mainly between 70 and 165. In thermal reactor systems, such as PWR, BWR and AGR cores, the fission yield is highest for fission products around mass number 95 and around mass number 140. Neutron capture in fission products during irradiation may also modify the fission product isotopic composition, either through burn-in (from capture in the preceding mass chain) or through burn-out (by capture to the next mass chain). Many fission product inventories will also change with time due to radioactive decay, which is an important consideration if credit is to be taken for post-irradiation cooling of the fuel, particularly for Gd-155 and Sm-149, which grow-in during cooling from the decay of Eu-155 and Pm-149 respectively.

The prediction of a spent fuel composition at any time during its irradiation or cooling requires the calculation of a vast number of nuclide inventories. Over a given time period, the rate of change of inventory for any nuclide is influenced by the rates of various production and removal modes covering radioactive decays and neutron induced reactions (see box).

The rate of change of inventory for any given nuclide can be expressed simply by:

Rate of Change = - *Loss by Decay*
- *Loss by Neutron Capture*
- *Loss by Fission (actinides)*
+ *Production by Fission (fission products)*
+ *Production by Neutron Capture*
+ *Production by Decay*

The complexity of the inventory changes during burnup is such that, in principle, an accurate burnup calculation requires consideration of the following parameters which influence the composition of the irradiated fuel:

i. start-of-life fuel and core material (e.g. structures, control rods) composition;

ii. fission rate or power rating of fuel in the core;

iii. fuel, cladding and coolant temperature and density;

iv. irradiation time;

v. cooling time;

vi. the presence of burnable neutron poisons within the fuel assembly or the core coolant during irradiation of the fuel;

vii. neutron flux and spectrum;

viii. various neutron data:

a) neutron fission, absorption and capture cross sections, and the variation of effective cross section during fuel irradiation;

b) fission yield data;

c) radioactive decay data.

In common with traditional criticality safety assessments, the assessor can make a number of approximations to simplify the calculation of the spent fuel composition. In general, the intention is for these approximations to conservatively bound all of the possible fuel conditions, in order that the criticality safety case is not overly restrictive with regard to the intended fuel handling operations. In

some cases, where non-conservative assumptions are made or where there are uncertainties in the fuel parameters, an allowance for potential optimism in the calculation can be made, in terms of a calculational bias.

These approximations and simplifications to the inventory modelling have been extensively reported, such as [2, 3 and 4]. Where appropriate, this guidance needs to distinguish between the extent of burnup credit being claimed, either as:

i. an 'actinide-only' claim for fissile depletion and the presence of actinide absorbers only; or

ii. the use of 'fission product credit', taking account of the presence of the major fission product absorbers in addition to the fissile depletion and actinide absorbers of the 'actinide-only' burnup credit case.

4.2. Inventory Prediction Methods

The major inventory prediction codes utilised within the UK for the application of burnup credit are the point reactor code FISPIN [5 and 6], the reactor lattice code WIMS [7] and the use of the WIMS-PANTHER core management system [8]. Brief descriptions of the FISPIN and WIMS methods are given in the main text of this technical document.

4.3. Inventory Prediction Validation

Validation of inventory predictions is typically achieved by comparison with destructive chemical analysis data (often termed Post-Irradiation Examination, PIE) for irradiated fuel samples. Sound validation requires the samples to be well characterised in terms of their irradiation history, initial fuel composition and reactor operating parameters, in order to perform an inventory calculation that can be considered as being truly representative if the irradiated fuel.

Inventory prediction has historically been available for the WIMS and FISPIN codes, but was limited for HBU and MOX fuel. These shortcomings have been addressed by UK industry involvement in the CERES international experimental programme [9]. Beyond the CERES programme, validation has been secured by individual organisations investing in other experimental programmes. An example of this is BNFL's involvement in the ARIANE programme [10] to further enhance the validation of inventory prediction codes. Chemical analysis data for the irradiated fuel samples examined across these programmes has been used in BNFL to extend the scope of validation to cover:

i. PWR UO_2 fuel for burnup of up to 60GW·d/te and initial enrichments of up to 4.5w/o U-235/U;

ii. BWR UO_2 fuel for burnup of up to 55GW·d/te and initial enrichments of up to 5w/o U-235/U;

iii. PWR MOX fuel for burnup of up to 50GW·d/te, with initial plutonium contents of up to 6w/o Pu_{total}/[U+Pu];

iv. BWR MOX fuel for burnup of up to 55GW·d/te, with initial plutonium contents of up to 6.5w/o Pu_{total}/[U+Pu].

Validation has also been attained by BNFL by comparison of inventory calculations against data from the SFCOMPO computer database developed by the Japan Atomic Energy Research Institute (JAERI), which contains PIE data for 13 LWRs from Europe, Japan and the United States [11].

Considerable confidence has also been gained by UK involvement in the studies of the criticality working group of the OECD Nuclear Energy Agency Nuclear Science Committee (NEANSC), which has been investigating the use of burnup credit since 1991. Whilst the group is not intending to define any policy relating to the implementation of burnup credit, an extensive benchmarking programme is being undertaken to investigate the ability of the group members codes, methods and nuclear data to calculate spent fuel reactivity [12].

5. REACTIVITY CALCULATIONS

5.1. Calculation of Spent Fuel Reactivity

Having calculated the irradiated fuel composition, the criticality assessor then has to make a choice as to which nuclides are represented in the reactivity calculation. The assessor clearly needs to be selective, since an inventory prediction code such as FISPIN will generate a spent fuel composition consisting of something like 1000 nuclides. The majority of these nuclides will have negligible effect upon the reactivity of the spent fuel, because the total reaction rates (i.e. the product of the nuclide inventory, the neutron flux and the total neutron cross section) for the nuclides are insignificant. This arises for nuclides which either:

i. have neutron reaction cross sections which are negligible, such that they have little impact regardless of the quantity present;

ii. are present in small quantities, so that even with large neutron cross sections, they also have little impact upon fuel reactivity.

Previous studies have demonstrated that the actinides can be pessimistically represented using just the major isotopes of uranium and plutonium [13]. Fortunately for burnup credit, around 90% of the total fission product absorption arises from 15 nuclides [1], the majority of which also have the advantage of being stable and being soluble in nitric acid, thereby aiding the use of burnup credit for a fuel dissolver application. As a result, the reactivity calculations are manageable in terms of the number of nuclides represented, and hence the validation of the cross section data for reactivity calculations can be directed at a relatively small number of nuclides:

i. the major actinides of U-234, U-235, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241 and Pu-242;

ii. the major fission products of Mo-95, Tc-99, Ru-101, Rh-103, Ag-109, Cs-133, Nd-143, Nd-145, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Eu-153 and Gd-155.

It is known that the reactor power varies throughout the reactor core, with axial power distributions resulting in an axial burnup profile along the length of the assembly. Burnup credit reactivity calculations must therefore take account of this variation in fuel burnup.

5.2. Reactivity Calculation Methods

The principal tool used in the UK for reactivity calculations is the Monte Carlo neutronics code MONK [14]. The point nuclear database used by the MONK code is derived principally from UKNDL data, although this has been extended to include the latest JEF cross section data for the major fission product absorbers. Sensitivity analyses in support of the main MONK reactivity calculations have made extensive use of the deterministic code WIMS [7]. Brief descriptions of the MONK and WIMS codes are given in the main text of this technical document.

5.3. Reactivity Calculation Validation

The main source of validation evidence for reactivity calculations is by direct comparison with high quality critical experiments. The validation database for the MONK code and its associated point

nuclear data set consists of a number of calculations for critical configurations of uranium, plutonium and mixed fissile systems over a range of moderation and reflection conditions. None of the critical experiments featured in this database have involved irradiated fuel.

The validation database has been established over many years and has focused upon the nuclides most commonly encountered in criticality safety assessments. As a result, it is generally accepted that the neutron cross section data in the MONK nuclear data set is well established for the major uranium and plutonium isotopes. Although cross section data is provided for many of the major fission product absorbers, little validation was in existence for the fission products of interest to burnup credit.

The use of the MONK code and nuclear data therefore required supporting validation data for:

i. the major fission product absorbers, as listed above;

ii. the calculation of reactivity for irradiated UO_2 and MOX fuel;

iii. the calculation of fuel reactivity for realistic fresh MOX fuel, manufactured from plutonium recovered from the reprocessing of commercial uranium fuel, since the majority of the MOX and plutonium system validation at that time was for high quality plutonium, containing little Pu-241 and higher isotopes of plutonium.

The main source of validation for burnup credit reactivity calculations was the CERES experimental programme [9]. Reactivity worth measurements were undertaken in the DIMPLE reactor at AEA, Winfrith and the MINERVE reactor at CEA, Cadarache. The results of these measurements are reported in [9 and 15].

As with the inventory prediction calculations, comparison of codes and data via the OECD/NEANSC burnup credit benchmarking programme adds confidence in the validity of the proposed reactivity calculation methods in the burnup credit methodology.

5.4. Criticality Safety Criteria

Traditionally, criticality safety assessments have been performed using a criticality safety criterion of the form $k_{effective} + 3s$ £ X - E, where s is the statistical uncertainty on the calculated value of the effective neutron multiplication factor, $k_{effective}$. X is the subcritical limit and E represents an allowance for errors in the program, nuclear data, system modelling and non-optimisation of the system, i.e. the calculational bias.

The criticality safety criterion used in burnup credit criticality safety assessments must account for any additional errors arising from the burnup credit calculational method, in terms of their effect on the reactivity of the system. This could arise from any calculational uncertainties in the inventory prediction calculation, or in the modelling of the system in the reactivity calculation, either due to the neutron cross section data or any fuel modelling assumptions.

The assignation of calculational biases arising from the burnup credit calculation needs to account for:

i. the reactivity errors associated with uncertainties in the inventory calculations, as observed from the validation of the inventory code for burnup credit;

ii. the reactivity errors associated with the nuclear data itself;

iii. errors potentially arising from the assumptions made in the reactivity calculation, such as the sensitivity of reactivity to the axial burnup profile.

6. QUANTIFICATION AND VERIFICATION OF FUEL BURNUP

It has been assumed in the development of burnup credit within BNFL that the regulatory bodies may require a confirmatory measurement of burnup to be made prior to certain operations, such as the loading of a transport flask. This clearly has some foundation in terms of demonstrating a robust safety case, since it provides a level of protection against the handling of fuel assemblies with less than the minimum burnup required to ensure criticality safety. As the magnitude of the burnup credit claim increases, more reliance is being placed upon the correct identification of irradiated fuel assemblies, and in some cases it may be that the loading of fresh fuel would result in a criticality incident within the flask.

One approach being proposed in the UK is for the requirements of fuel burnup quantification and verification to be addressed on a case-by-case basis, since it is recognised that the quality of core and fuel management, together with the accuracy of burnup estimation and identification of individual assemblies may vary considerably between reactor sites. The issue of quantifying and verifying fuel burnup relies on three key components:

i. the method by which the burnup of each fuel assembly is determined for the purpose of the core management and the station records, and the accuracy of that method;

ii. the integrity of the fuel management procedures in place at the reactor site, linking the estimated fuel burnup with the assembly identifier and its location in the reactor core or fuel storage pond;

iii. the use of burnup monitoring equipment at the reactor site, and the accuracy of such devices in the determination of fuel assembly burnup.

Moreover, it is felt that the quality of the burnup measurement should reflect the consequences of maloperation. An example of this might be that the loading of fresh fuel into a burnup credit transport flask would exceed the safety criterion, but would not be critical. In this case, it might be adequate for a measurement to be made confirming that the fuel was irradiated, but which did not provide an accurate measure of the fuel burnup.

Sophisticated burnup monitoring technology has been under development for some time, as discussed in the main text of this technical document. No further consideration is given to burnup monitors here.

7. APPLICATIONS FOR BURNUP CREDIT IN THE UK

At present, the burnup credit activities in the UK are limited to LWR fuel operations, although it is recognised that AGR fuel operations could also benefit from the application of a burnup credit methodology. The current interest in burnup credit in the UK is related to:

i. the pond storage of irradiated PWR fuel at Sizewell, operated by Nuclear Electric;

ii. the transport of spent fuel from mainly non-UK reactor utilities to BNFL's Sellafield site;

iii. the pond storage of spent fuel at Sellafield;

iv. the reprocessing of spent fuel in the Thorp reprocessing plant at Sellafield.

As with most other nations, the principle interest in burnup credit centres on the application to uranium oxide (UO_2) fuel, although significant effort is also being directed at the development of a methodology for mixed oxide (MOX) fuel. The development of a burnup credit methodology for MOX fuel is discussed later in this paper.

7.1. Pond Storage at the Sizewell PWR

Nuclear Electric have developed a burnup credit criticality safety case for the core pond of the Sizewell PWR. This approach utilises actinide and fission product credit for the storage of irradiated UO_2 fuel in compact storage racks in the pond, with other racks being used for the storage of fresh fuel or invalidated irradiated fuel in the same pond. This criticality safety case is currently under review by HM NII [16].

7.2. Transport Flasks

The shipment of spent PWR and BWR UO_2 fuel to BNFL's Sellafield site is undertaken in a wide range of 'wet' transport flask designs, with the fuel typically contained in multi-element bottles (MEBs). Many of these flasks are cleared for the shipment of fuel with initial enrichments of the order of 4 w/o U-235/U. The incentive for burnup credit for transport is therefore to extend the operational lifetime of the flask and MEB designs, through stepped increases in the maximum fuel enrichment using burnup credit. At the current time, there is little commercial necessity to utilise burnup credit for BNFL's transport operations, although it is recognised that the need for burnup credit is not far away. The design of new transport packages using burnup credit offers a number of advantages, such as increased payload or the removal of neutron poisons from the fuel basket or MEB.

7.3. Pond Storage at Sellafield

Irradiated fuel is currently stored underwater in MEBs prior to reprocessing, and so the potential benefits of burnup credit are the same as those discussed above for spent fuel transport operations

7.4. Reprocessing in THORP

The criticality safety case for the Thorp fuel dissolvers currently assume a maximum initial UO_2 fuel enrichment of 4w/o U-235/U, taking no credit for fuel burnup and requiring the use of gadolinium as a neutron poison in the dissolver acid. BNFL is exploring the potential for burnup credit to reduce the gadolinium loading in the fuel dissolver.

8. APPLICATION OF BURNUP CREDIT TO MIXED OXIDE (MOX) FUEL

A key area of interest to BNFL is the application of burnup credit to mixed oxide (MOX) fuel. Some of the early MOX fuel types are likely to require burnup credit for the underwater removal of the fuel assembly from the MEB, since the pond environment is more onerous that the conditions of limited interaction and fixed local neutron poisons within the MEB.

A comprehensive approach to MOX burnup credit would address all of the issues discussed previously for uranium oxide fuel, but with the added difficulties associated with the non-unique specification of MOX fuel and the manner in which it would be utilised within existing thermal reactor designs. Thermal reactor fuel is traditionally made from uranium oxide fuel where the uranium is enriched in the fissile U-235 isotope up to the order of 5w/o U-235/U. The concept of MOX fuel is to mix natural or depleted uranium oxide with plutonium oxide in order to produce a fuel that is interchangeable with uranium oxide fuel in the reactor core, i.e. equivalent in terms of its lifetime performance.

The principal source of plutonium for use in MOX fuel arises from the reprocessing of discharged UO_2 fuel assemblies. The discharged fuel may come from reactors of various types, e.g. PWR, BWR and the UK AGR and Magnox reactors, and will invariably have been subjected to differing irradiation histories and neutron spectra. This is also true to a lesser extent of the fuel assemblies discharged from the same reactor. These variations can have a significant impact upon the

isotopic composition of the plutonium component of the discharged fuel, which feeds through to isotopic variation in the plutonium recovered from batched fuel reprocessing operations. In addition, the plutonium isotopic composition will be affected by the age of the material since reprocessing, principally due to the decay of Pu-241 to Am-241. These effects mean that the manufacture of MOX fuel to a given specification must accommodate a wide range of plutonium vectors, i.e. varying relative proportions of the key plutonium isotopes. In the future, the range of plutonium vectors in MOX fuel will undoubtedly be extended by the reprocessing of irradiated MOX and the potential disposition of weapons plutonium in MOX fuel.

The isotopic variations in the plutonium used for the manufacture of MOX fuel can affect two important performance characteristics of the fuel assembly produced, these being:

i. the reactivity of assemblies over their lifetime in the reactor (referred to as the lifetime average reactivity);

ii. the within-assembly power peaking factors.

A number of approaches to MOX manufacture can therefore be envisaged, whereby the fuel supplier can either:

i. homogenise all of the available plutonium before making the fuel;

ii. mix plutonium oxide batches of known isotopic composition to meet some uniform required isotopic composition; or

iii. process individual batches of plutonium oxide and vary the total plutonium content of the MOX to compensate for the varying isotopic composition of the plutonium being used.

In any case, it can be seen that a given MOX specification, in terms of uranium fuel equivalence, can be met by different plutonium vectors, i.e. that the specification is not unique.

One possible equivalence method is based upon matching the lifetime average reactivity of the MOX and uranium oxide fuel assemblies. The variation of neutron multiplication with irradiation is the key consideration in the management of MOX fuel in the reactor. The infinite neutron multiplication factor, k_{inf}, is largely determined by the ratio of the fission and absorption cross sections of the fuel. The high fission cross section of MOX fuel relative to uranium oxide fuel is more than offset by the high absorption cross section, such that the start-of-life multiplication factor for MOX fuel tends to be lower than that for uranium oxide fuel with the same lifetime average reactivity. However, the MOX fuel will typically have a lower reactivity loss with burnup, resulting in greater reactivity at discharge.

The higher absorption in MOX fuel also has the effect of reducing the effectiveness of control rods and other means of controlling excess core reactivity. The reduced control rod worth leads directly to a limitation on the MOX loading fraction within existing LWR designs, e.g. of the order of 40% for a typical PWR, such that MOX fuel must be irradiated alongside traditional UO_2 fuel assemblies. Appropriate representation of the fuel in the core is another consideration that needs to be made in the calculation of spent fuel inventory for MOX fuel burnup credit.

The higher fission cross section in MOX fuel causes the neutron flux to be relatively smaller for a given power rating. This tends to cause power peaking at the interface between MOX and UO_2 fuel assemblies, due to the increased neutron flux in the outermost MOX fuel pins. The effect of power peaking can be mitigated by varying the total plutonium content of the fuel pins across the fuel assembly. Typically, this would be achieved using three regions of varying plutonium content, with the lowest fissile content occurring in the outermost pins.

For a batch loaded reactor, where the core is divided into a number of refuelling batches, the lifetime average reactivity (LAR) can be determined by averaging the infinite neutron multiplication

factor for the fuel at the end of each irradiation cycle. In an idealised reactor with no neutron leakage and an LAR of 1.0, the core would be just capable of sustaining a chain reaction in the absence of control rods or neutron poisons at the end of each operating cycle. In practice, there is always some neutron leakage and the fuel LAR must be designed to be greater than unity.

In designing a core containing MOX fuel, the fuel equivalence could be determined by matching the LAR of the MOX fuel assemblies to that of the UO_2 fuel assemblies. However, a number of issues still remain in the comparison of LARs for MOX and UO_2 fuel:

i. although MOX and UO_2 fuel assemblies can be designed with the same LAR, the burnup characteristics of the two fuel types will be different;

ii. the LAR is only meaningful if it relates to an irradiation cycle length and a particular batch loading scheme - the specified MOX and UO_2 fuel will not, in general, be equivalent for other core management schemes;

iii. there will be a dependence of the LAR on the plutonium isotopic composition, since MOX fuel with different plutonium vectors will exhibit different characteristic k_{inf} versus burnup curves;

iv. a rigorous determination of the LAR might use k_{inf} values weighted by the assembly contribution to the overall core power.

It can be seen from the above discussion for MOX fuel that the development of a robust burnup credit approach requires a number of issues to be taken into account. These are in addition to the complexities of burnup credit for certain core designs, such as BWRs.

The application of burnup credit to MOX fuel is being addressed by the OECD/NEANSC criticality working group [17], whose current benchmark exercise covers burnup credit for MOX fuel using a range of initial plutonium isotopic compositions.

9. CONCLUSIONS

This paper outlines the application of burnup credit in the UK, covering the methodology, methods and data. A brief overview is given of the current applications, as well as a discussion of the intended application to mixed oxide (MOX) fuel.

Burnup credit is seen to be of significant benefit to spent fuel operations in the UK, and the industry has been fully involved in international experimental collaborations and working groups to secure appropriate validation and benchmarking of the burnup credit methodology.

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BURNUP CREDIT ACTIVITIES BEING CONDUCTED IN THE UNITED STATES

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Abstract

The paper describes burnup credit activities being conducted in the U.S. where burnup credit is either being used or being planned to be used for storage, transport, and disposal of spent nuclear fuel. Currently approved uses of burnup credit are for wet storage of PWR fuel. For dry storage of spent PWR fuel, burnup credit is used to supplement a principle of moderator exclusion. These storage applications have been pursued by the private sector. The Department of Energy (DOE) which is an organization of the U.S. Federal government is seeking approval for burnup credit for transport and disposal applications. For transport of spent fuel, regulatory review of an actinide-only PWR burnup credit method is now being conducted. A request by DOE for regulatory review of actinide and fission product burnup credit for disposal of spent BWR and PWR fuel is scheduled to occur in 1998.

1. INTRODUCTION

The US does not reprocess its spent nuclear fuel, and the fuel is stored at about 100 reactor sites in pools or in dry storage units. Furthermore, there is little expectation that new reactors will be built. A geological repository is planned for the ultimate disposal of this material as a nuclear waste. The US expects to receive about 84,000 tU of spent nuclear fuel after the end-of-life for all current reactors. After a start-up period, the fuel will be accepted into the Civilian Radioactive Waste Management System (CRWMS) at a rate of about 3,000 tU per year. Figure 1 shows the location of U.S. commercial reactor sites and other sources of radioactive materials that will be disposed of as waste. The proposed disposal site at Yucca Mountain is located in the southeast corner of Nevada.

Light water reactor systems which are used by the US commercial nuclear power industry use fuels with low concentrations of fissile uranium (typically less than 5% initial concentration of U-235 by weight). 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.

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 U. S., 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 US Federal government have been performed primarily by DOE. DOE began these efforts in the mid-1980's to support its CRWMS activities which included storage, transportation, and disposal of spent nuclear fuel from light water reactors. In 1995, the DOE submitted a topical report to the Nuclear Regulatory Commission (NRC) for the use of actinide-only burnup credit for transport of PWR fuel (DOE, 1995). The NRC has reviewed the report and given comments to DOE, and a revised report was submitted to them in May 1997 (DOE, 1997).

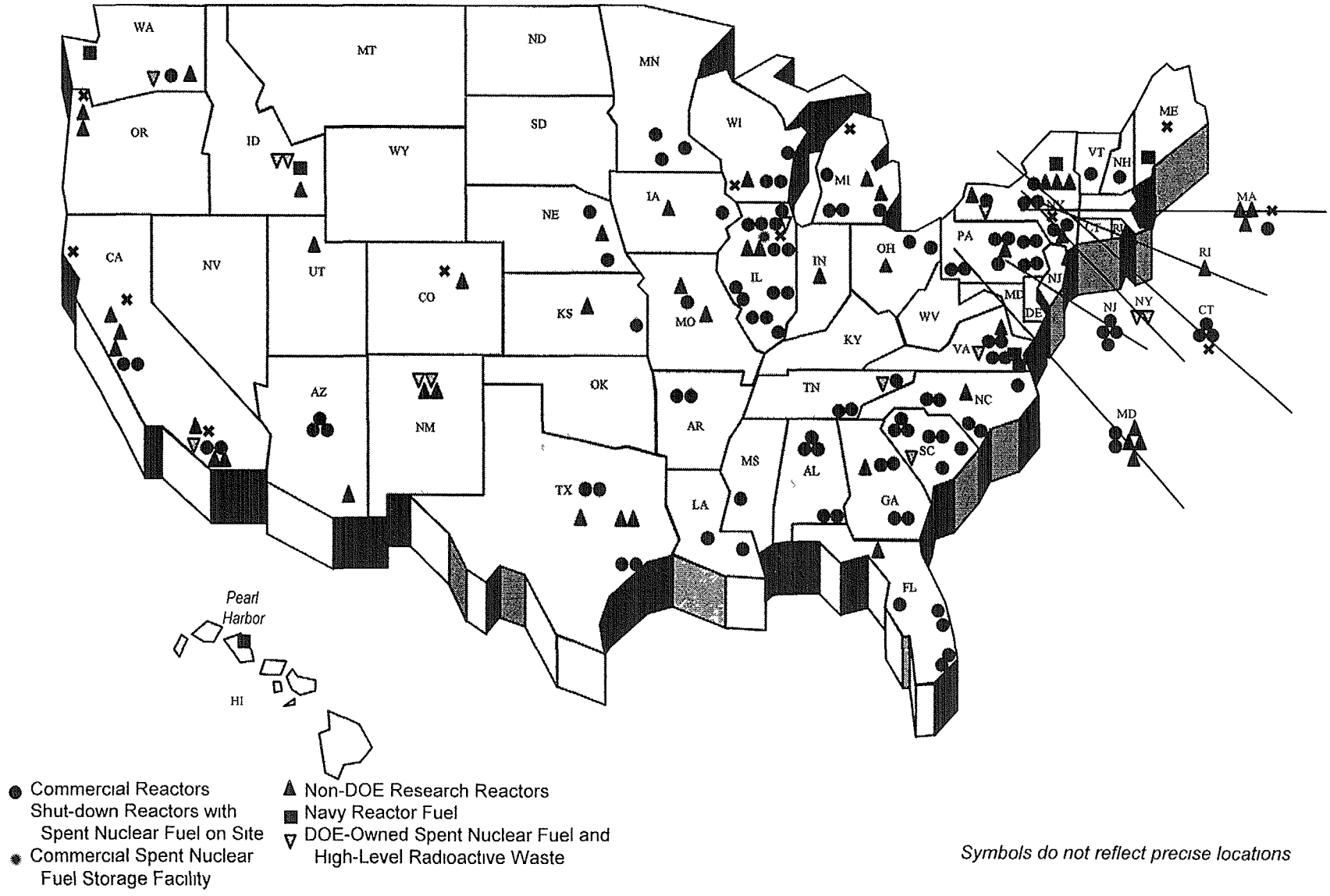

FIG. 1. Nuclear Fuel and High- Level Radioactive Waste in the United States

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. 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 consulted with DOE on burnup credit, and has conducted a number of activities that have been beneficial to the DOE's efforts. EPRI is a research and development organization funded by member utility companies. Recently, EPRI has increased its involvement, and cask vendors, instrumentation and transport service organizations, and utilities have become involved in burnup credit activities.

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. Currently, the design life of the WP is expected to be about 3,000 years, while the licensing period for the repository is 10,000 years or more. Since the licensing period for a repository will exceed the life expectancy for the WP and its engineered criticality control systems, the need to consider the actual reduced reactivity of spent nuclear fuel is evident.

2. STORAGE

2.1. Wet Storage

2.1.1. Wet Storage of PWR Spent Fuel

To accommodate increased inventories of spent fuel which must be stored at US reactor sites, some utility companies have relied on their existing fuel storage pools. Most of these pools were designed for a reprocessing economy and were never intended to accommodate a reactor's lifetime inventory. Some utility companies have used burnup credit to increase storage capacities at their PWR fuel pools. The NRC has approved burnup credit for these pools, but only if boron content of pools is also considered for criticality safety (Lesko and Newmyer, 1997).

For approval, a utility company must first show equivalence between reactivities of burned and unburned systems at different initial enrichments (initial U-235 concentrations). For example, a pool approved for 3 % enriched fuel with no burnup may be used for 4% enriched spent fuel if the burnup of the spent fuel is sufficient so that reactivity of the two systems are shown to be equivalent. The utility company must also show that some minimum boron concentration in the pool has enough negative reactivity to provide for a 5% criticality safety margin, and cover any uncertainties associated with the use of burnup credit.

There are also a number of other conservative assumptions used in this approach. The reactivity equivalence uses a harder than normal depletion spectrum to maximize the predicted production of fissile actinides. Not all fission products are considered in determining reactivity equivalence. Also, boron concentrations are generally three to four times higher than the minimums predicted for the 5% margin and uncertainties. Recently, the utility industry has gotten NRC approval for consideration of actinide decay effects on reduced reactivity.

2.1.2. Wet Storage of BWR Spent Fuel

BWR pools do not contain boron, and burnup credit is not used.

2.1.3. Wet Storage of MOX Spent Fuel

The US does not use MOX reactors.

2.1.4. Wet Storage of WWER Spent Fuel

The US does not use WWER reactors.

2.2. Dry Storage

2.2.1. Dry Storage of PWR Spent Fuel

Once burnup credit is approved for dry transport of spent PWR fuel, it can also be used for dry storage applications. Currently a moderator exclusion concept may be used for dry PWR storage systems. Under this approach, a storage cask is loaded in a borated pool, and later located at a storage site having no possibility of water intrusion. The minimum boron concentration that is determined necessary to achieve a 5% criticality safety margin must be maintained during cask loading. To accommodate possible off-normal and accident conditions, a criticality analysis is also performed for a cask flooded with fresh (non-borated) water. For storage configurations not having a minimum 5% criticality safety margin when flooded with fresh water, burnup credit is used to show that the necessary margin is achieved. The approach works in some cases for storage only systems, but cannot be used for storage transport systems because a claim of moderator exclusion can not be used for transport.

Many of the NRC approved storage only systems were developed with plans of later seeking NRC approval for transportation as well as storage. Systems now under development are typically dual purpose systems for storage and transport rather than single purpose (storage or transport). For many of the existing PWR storage units that use moderator exclusion, burnup credit will have to be used if they are to be transported at sometime after the at-reactor storage period ends. For PWR systems now being developed, moderator exclusion may still be used while the use of burnup credit continues to be reviewed by the U.S. regulatory authorities, allowing to continue a practice that assures maximized cask capacities.

If no new reactors are assumed in the U.S., about 132,000 spent PWR fuel assemblies will have to be stored at reactors or at a Federal storage facility. If a canistered dual purpose system is assumed the cost of storage with and without burnup credit can be estimated. What follows is a simple cost estimate for hardware associated with storage of spent PWR fuel. The storage system consists of a canister and storage overpack. A canister is assumed to cost at least $US 300,000, and an overpack is expected to cost at least $US 150,000. The canister is assumed to hold 24 PWR assemblies without burnup credit, and 32 assemblies with burnup credit. For this cost model a no burnup credit storage system would cost $US 2,475 million, while and a burnup credit storage system would cost $US 1,856 million. The analysis shows a savings from burnup credit of $US 619 million. The results of the storage cost analysis are summarized and presented in Table 1.

2.2.2. Dry Storage of BWR Spent Fuel

There are currently no plans to pursue burnup credit for BWR spent fuel.

2.2.3. Dry Storage of MOX Spent Fuel

The US does not use MOX reactors.

2.2.4. Dry Storage of WWER Spent Fuel

The US does not use WWER reactors.

TABLE 1. STORAGE COST ANALYSIS

	UNITS	NO BURNUP CREDIT	BURNUP CREDIT
PWR INVENTORY (Number of assemblies)	#	132,000	132,000
CANISTER CAPACITY	#	24	32
NUMBER OF CANISTERS	#	5,500	4,125
CANISTER COST	$US 1,000	300	300
OVERPACK COST	$US 1,000	150	150
STORAGE UNIT COST	$US 1,000	450	450
STORAGE COST FOR PWR INVENTORY	$US Million	2,475	1,856
COST AVOIDED FROM USING BURNUP CREDIT	**$US Million**		**619**

3. TRANSPORTATION

3.1. Wet Transportation

Water cooled casks are not used to transport spent fuel in the US

3.1.1. Wet Transportation of PWR Spent Fuel

Not applicable.

3.1.2. Wet Transportation of BWR Spent Fuel

Not applicable.

3.1.3. Wet Transportation of MOX Spent Fuel

The US does not use MOX reactors.

3.1.4. Wet Transportation of WWER Spent Fuel

The US does not use WWER reactors.

3.2. Dry Transportation

3.2.1. Dry Transportation of PWR Spent Fuel

DOE's initial motivation for pursuing burnup credit for transportation was to support its advanced technology cask development program. The use of burnup credit for weight limited transport cask designs can lead to increased capacity, fewer shipments, reduced costs, and reduced risk and exposure to workers and the public. Based on DOE's experience with transport casks developed in the US, burnup credit will be used for PWR fuel where it has enabled the elimination of flux traps which results in increased cask capacity. Cost benefits of using burnup credit are presented here for transportation of spent PWR fuel.

If no new reactor orders are assumed in the US, about 132,000 PWR assemblies will eventually have to be shipped to a repository. Two transport systems are considered, a legal weight truck cask system, and a large canistered rail cask system that weighs about 125 tons (113 tonnes). To accommodate a cost-benefit analysis, a number of assumptions are made about casks and transport systems that may be used. Both cask systems have a 25 year life, and each has an average round trip shipping distance of about 8,050 km (5,000 miles). A truck shipment costs $US 38,000, and a truck cask which costs $US 3.5 million can make 500 shipments in a lifetime which is $US 7,000 a shipment. The resulting total cost is $US 45,000 a shipment. A rail cask shipment costs $US 76,000, and the reusable cask which costs $US 4.25 million can make 200 shipments in its lifetime which is $US 21,250 a shipment. The resulting total cost is $US 97,250 a shipment. The assumptions used in the transport cost analysis are presented in Table 2.

TABLE 2. TRANSPORT COST ASSUMPTIONS

	TRANSPORT MODE	
	TRUCK	RAIL
TRAVEL DISTANCE (ROUND TRIP – km)	8,050	8,050
COST PER CASK TRIP ($US)	38,000	76,000
CASK UNIT COST ($US Million)	3.5	4.25
CASK LIFE (YEARS)	25	25
ROUND TRIPS IN A LIFETIME	500	200
COST PER SHIPMENT ($US)	45,000	97,250

Rail systems currently being developed in the US are canister based storage/transport types. They consist of a non-reusable canister which holds spent fuel for storage and transport, a reusable transport module, and storage module, which is not reusable. The canister, which is not considered in this transport cost analysis is assumed to cost $US 300,000. It should be noted that the canister of a dual purpose, or storage transport cask, serves as a fuel basket for transport. For a single purpose, or transport only cask, a reusable fuel basket would be needed for each reusable transport cask. However, the increased cost of the basket for a single use cask was not included in this analysis.

The truck cask is assumed to carry four PWR assemblies with burnup credit, and two without burnup credit. The use of canisters which can dictate the rail cask capacities are assumed to carry 32 PWR assemblies with burnup credit, and 24 without burnup credit. Although capacities vary depending on burnup credit use, transportation and hardware costs are assumed independent of capacity.

The costs for shipping all 132,000 PWR assemblies by truck, using the above assumptions are $US 2.97 billion without burnup credit and $US 1.485 billion with burnup credit. That is, burnup credit could save $US 1.485 billion for an all truck system. The total costs for a system with all shipments made by rail are $US 535 million without burnup credit and $US 401 million with burnup credit. That is, burnup credit could save $US 134 million for an all rail system. These results indicate that rail transport, because it is less costly, should be maximized, and burnup credit used for either transport mode. The results of the transport cost analysis are summarized and presented in Table 3.

This simple cost model neglects a number of factors that could impact costs and savings. On the savings side, such things as reduced worker exposure and handling at reactors and receipt facilities are neglected. On the cost side of burnup credit, such things as design, development, and implementation costs are neglected. Furthermore, the predicted savings may be reduced somewhat by the degree of burnup credit used, and other cask capacity limitations not considered in this analysis. Finally, it should be noted that the DOE will not handle fuel at reactor sites, and utilities will be responsible for costs associated with cask loading at their reactor sites. The individual utilities will

need to balance savings from using burnup credit against costs of implementation. One such cost will be the cost of measuring burnup.

TABLE 3. TRANSPORT COST ANALYSIS

	TRUCK		RAIL	
	NO BURNUP CREDIT	BURNUP CREDIT	NO BURNUP CREDIT	BURNUP CREDIT
PWR INVENTORY (Number of assemblies)	132,000	132,000	132,000	132,000
CASK CAPACITY	2	4	24	32
NUMBER OF SHIPMENTS	66,000	33,000	5,500	4,125
COST OF SHIPPING ENTIRE INVENTORY ($US Million)	2,970	1,485	535	401
AVOIDED COST FROM USING BURNUP CREDIT ($US Million)	----	**1,485**	----	**134**

In May 1995, the DOE submitted a topical report to the NRC for actinide-only burnup credit for transport of irradiated PWR fuel. The topical report describes a general method for burnup credit design, provides specific data for benchmarking depletion and criticality codes with examples for the SCALE system, and provides a general approach to verify fuel loading by measurement. The NRC reviewed the report and requested additional information on March 22, 1996 (Travers, 1996). In response, the DOE submitted a revised topical report in May 1997. Once the NRC approves the methods and data proposed for actinide-only burnup credit, they may be used by cask designers who apply for NRC approval of burnup credit for specific casks.

The technical issues that have been identified, and are being addressed for actinide-only burnup credit for PWR fuel, are benchmarking for depletion and criticality codes, modeling effects (e.g., axial distribution of burnup, horizontal gradients), and cask loading verification. Approval of the DOE's actinide-only topical report by the NRC is expected sometime in 1998. Once the methods described in the actinide-only report are approved by NRC, the DOE will consider the possibility of expanding its burnup credit efforts by seeking additional burnup credit.

DOE is currently assessing options to pursue burnup credit for PWR spent fuel beyond the limited actinide-only approach. A number of factors impact this assessment, including technical feasibility, cost, and benefits. If DOE pursues the use of fission products it will not be initiated until NRC accepts and approves the current actinide-only approach. Also, only a limited number of fission products would be used for burnup credit.

3.2.2. Dry Transportation of BWR Spent Fuel

In the case of BWR fuel, technical issues are expected to be more difficult than for PWR fuel. Furthermore, increased cask capacities are not expected because current BWR cask designs do not need flux traps for criticality control. Although burnup credit for transport of irradiated BWR fuel could lead to cost savings from decreased poison content in fuel baskets (e.g., boron), it is not being considered at this time.

3.2.3. Dry Transportation of MOX Spent Fuel

The US does not use MOX reactors.

3.2.4. Dry Transportation of WWER Spent Fuel

The US does not use WWER reactors.

4. REPROCESSING

Reprocessing of commercial nuclear fuel is not done in the US

5. DISPOSAL

The use of burnup credit for disposal of irradiated fuel is needed to demonstrate criticality control in the current WP and repository designs. For the longer term, it can provide a means of demonstrating criticality control for a repository beyond the time period that the integrity of engineered criticality control features of a WP can be guaranteed. The methodology used for demonstrating long-term criticality control is both probabilistic and deterministic. For comparing alternative criticality control designs probabilistic methods are being used to weigh the importance of potential hazards and degradation modes. Deterministic criticality analysis is used to predict fuel and waste package behavior for the specific degradation modes and hazards of regulatory importance. For the disposal case, the DOE will consider burnup credit for both BWR and PWR irradiated fuel.

DOE plans to submit a topical report for disposal criticality to the NRC in 1998. In preparation for that submittal the DOE has prepared and issued a technical report on the methods it is developing to address disposal criticality. The DOE has also had a number of formal technical exchange meetings with the NRC on the subject.

6. CONCLUSIONS

In the U.S., the ability to use burnup credit for spent fuel management is being pursued by both the private sector and the Federal government. The use of burnup credit for wet and dry storage of spent PWR fuel has been pursued by the private sector. These applications of burnup credit, for spent fuel storage has been approved by the NRC. The use of burnup credit at reactor storage pools has been motivated by the need to store more fuel at reactor sites until a Federal facility becomes available to receive commercially generated spent fuel. For the same reason, the private sector has also gotten NRC approval for moderator exclusion for dry storage of spent PWR fuel. Approvals of this type are supplemented by reliance on burnup credit, where necessary, to show that a minimum 5% criticality safety margin has been achieved for an assumed fresh water flooding situation. The Federal government is seeking NRC approval for PWR actinide-only burnup credit for transport of spent fuel, and actinide and fission product burnup credit for disposal of PWR and BWR spent fuel. Reducing the number of spent fuel shipments which can result in reduced radiation exposure, reduced risk, and cost reduction is the primary motivating factor for using burnup credit for transport. For disposal, burnup credit is considered necessary to achieve an effective and efficient disposal facility.

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ACTINIDE-ONLY BURNUP CREDIT METHODOLOGY FOR PWR SPENT NUCLEAR FUEL

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Abstract

A conservative methodology is presented that would allow taking credit for burnup in the criticality safety analysis of spent nuclear fuel (SNF) packages. The method is based on the assumption that the isotopic concentration in the SNF and cross sections of each isotope for which credit is taken must be supported by validation experiments. The method allows credit for the changes in the U-234, U-235, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, and Am-241 concentration with burnup. No credit for fission product neutron absorbers is taken. The methodology consists of five major steps:

1. Validate a computer code system to calculate isotopic concentrations of spent nuclear fuel created during burnup in the reactor core and subsequent decay.
2. Validate a computer code system to predict the subcritical multiplication factor, k_{eff}, of a spent nuclear fuel package by use of UO_2 and UO_2/PuO_2 critical experiments.
3. Establish conditions for the SNF (depletion analysis) and package (criticality analysis) which bounds k_{eff}.
4. Use the validated codes and bounding conditions to generate package loading criteria (burnup credit loading curves).
5. Verify by measurement that SNF assemblies meet the package loading criteria and confirm proper assembly selection prior to loading.

1. INTRODUCTION

This paper describes a methodology for validating analytical methods and for applying burnup credit in the design of criticality control systems for pressurized water reactor (PWR) spent fuel packages. The methodology was included in a Topical Report submitted to the NRC in May 1995 [1]. After considering the NRC's request for additional information, a revision was issued in May 1997 [2]. This paper describes the methodology submitted to the NRC for gaining burnup credit and a demonstration of the methodology with SCALE 4.2 [3] using the 27BURNUPLIB.

The burnup credit methodology presented here is expected to be used by commercial SNF storage and transportation package designers. Design-specific burnup credit criticality analyses will be defined, developed, and documented in the Safety Analysis Report (SAR) for each specific storage or transportation package that uses burnup credit. These SARs will then be submitted to the NRC for review and approval. The methodology presented here is expected to be referenced in a number of storage and transportation cask applications to be submitted by commercial cask and canister designers to the NRC.

2. SCOPE

The methodology presented here addresses only the reduced reactivity of SNF due to changes in actinide isotopes; specifically, U-234, U-235, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, and Am-241. The considerable additional negative reactivity effect of fission products is not included. Additionally the scope is limited to PWR spent fuel packages. BWR spent fuel packages could also benefit from burnup credit but since the transportation packages normally do not require flux traps and the designs are more complex, work on BWR burnup credit has been deferred.

Nevertheless, the actinide-only burnup credit methodology presented in this paper has a wide applicability. It applies to all current generation commercial PWR fuel, with the following restrictions due to data limitations:

- Burnup credit benefits can be gained from fuel burned up to 50 GW·d/tU. SNF with an assembly average burnup greater than 50 GW·d/tU shall be treated as having a burnup of 50 GW·d/tU for the purposes of this methodology.
- Enrichments above five weight percent U-235 are not considered.
- Assemblies with integral fuel burnable absorbers (IFBAs) are not considered.
- The methodology applies to SNF with cooling times ranging from 1 to 100 years.
- Mixed oxide (MOX) initial content fuel is not considered.
- Reconstituted or disassembled fuel is not considered. Also not considered are fuel assemblies which have had any of their original rods removed or replaced.

Furthermore, there are analysis and modeling parameters that affect criticality which are not unique to burnup credit. None of those parameters or effects impact the proposed burnup credit methodology; therefore, they are not included in further discussion. A licensee's Safety Analysis Report is required to address those parameters in the usual manner. Examples include:

- Material and fabrication tolerances;
- Uncertainties due to limitations in the geometric or material representations used in the computational method;
- Effects of symmetric or asymmetric fuel assembly clustering within the spent fuel basket.

3. THE ACTINIDE-ONLY BURNUP CREDIT METHODOLOGY

The burnup credit criticality analysis procedure has been developed to be consistent with the "fresh fuel" assumption criticality analysis procedure currently accepted by the NRC. The generic criticality safety analysis procedure using the fresh fuel assumption is illustrated in Figure 1. The purpose of the criticality safety analysis using the fresh fuel assumption is to develop a cask loading criterion that establishes the maximum initial enrichment of an SNF assembly design that can be loaded into a cask. Figure 2 illustrates the generic burnup credit criticality analysis procedure recommended in this paper. The burnup credit criticality analysis procedure builds upon the fresh fuel procedure. The burnup credit procedure results in spent nuclear fuel package loading criteria that specify minimum burnups necessary for a range of initial enrichment values for a specific fuel assembly design. The results are presented as burnup credit loading curves.

The key elements in Figure 2 that distinguish the burnup credit procedure from the fresh fuel procedure are shaded. Descriptions of each of these elements and their relevance to the regulatory requirements are provided in this paper. Figure 2 graphically illustrates where these key elements fit into the overall burnup credit criticality analysis procedure. As can be seen on Figure 2, there are five major steps to implementing burnup credit:

1. Validate a computer code system to calculate isotopic concentrations in SNF created during burnup in the reactor core and subsequent decay;
2. Validate a computer code system to predict the subcritical multiplication factor, k_{eff}, of a spent nuclear fuel package;
3. Establish bounding conditions for the isotopic concentration and criticality calculations;
4. Use the validated codes and bounding assumptions to generate package loading criteria (burnup credit loading curves);
5. Verify that SNF assemblies meet the package loading criteria and confirm proper assembly selection prior to loading.

It should be noted that steps one through four are to be performed by the package designer, while step five is the particular utility's responsibility.

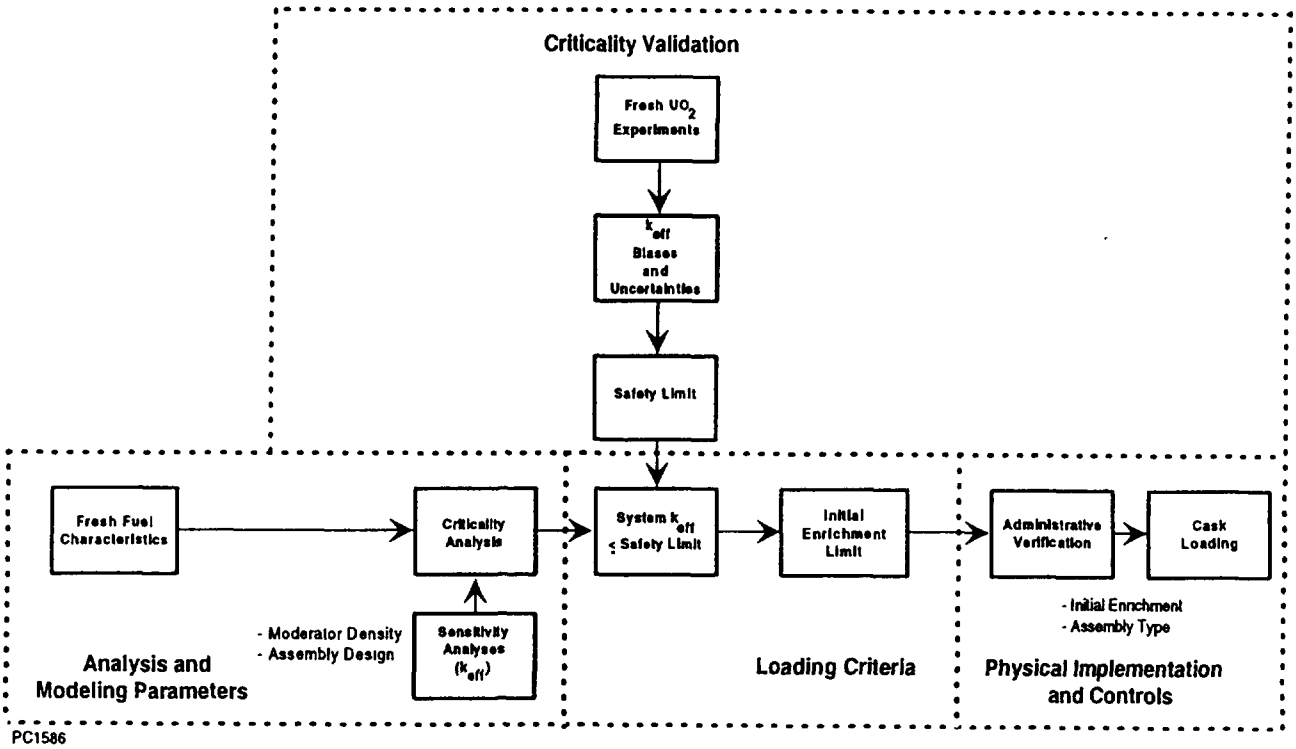

FIG. 1. Criticality Safety Requirements Using the "Fresh Fuel" Assumption

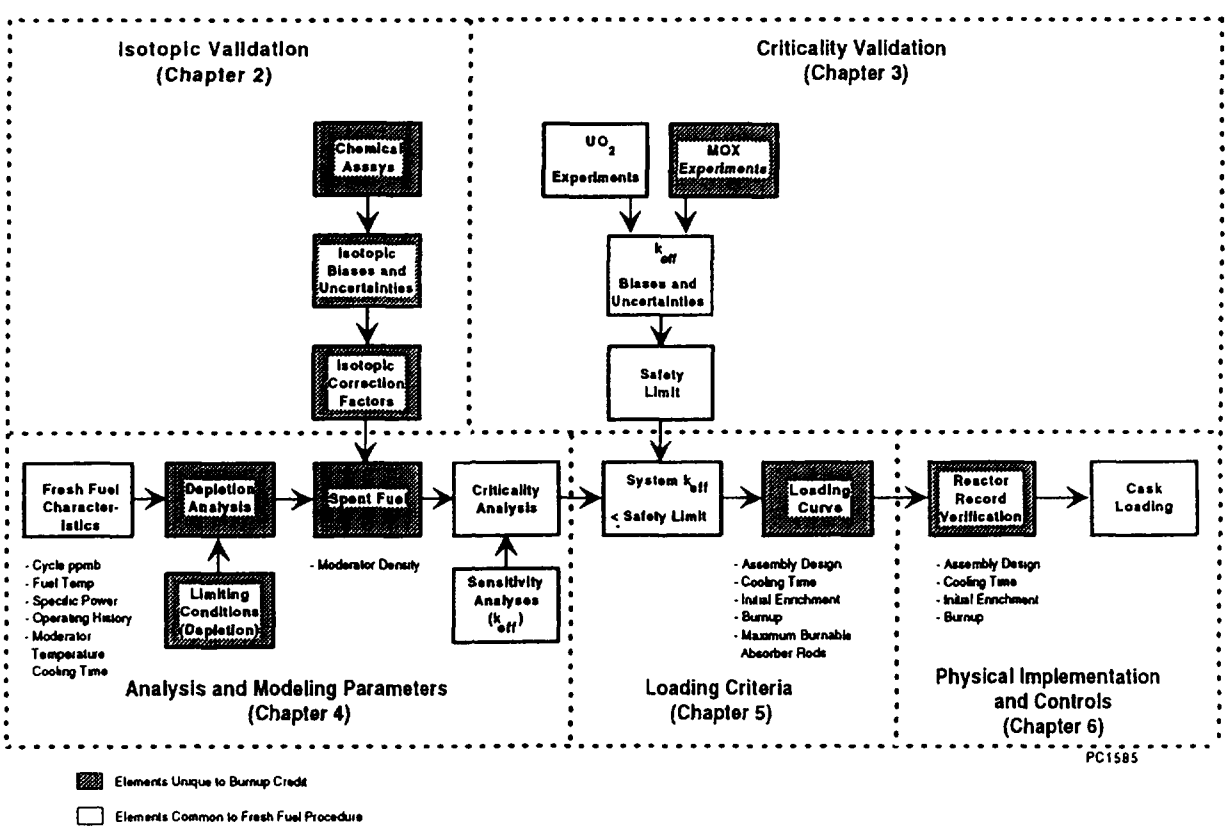

FIG. 2. Criticality Safety Requirements Using Actinide-Only Burnup Credit

3.1 Isotopic Validation

The isotopic composition of fresh fuel is well known through extensive, routine measurements by fuel manufacturers. However, after fuel is irradiated in a reactor, the isotopic composition of the spent fuel is routinely determined through analysis, rather than through measurement. Routine measurement of the isotopic content of discharged fuel using chemical assays is not practical due to radiological health, safety, and cost concerns. Confidence in the analytical capabilities is high due to the good agreement between the analytical predictions used for core reload analyses and the continual measurements of reactivity and power distributions at power plants. Source terms generated for thermal analyses have also shown good agreement with experiments.

For the burnup credit methodology presented in this paper, the computer code system used for predicting isotopic content must be validated using chemical assays of spent fuel. The chemical assays represent measured data for which best estimate predictions are generated with the computer code. Since there is very little data on chemical assays which include fission products, the isotopes for burnup credit are limited to the uranium and plutonium isotopes (and Am-241 as a daughter of Pu-241). Specifically, the isotopes selected are U-234, U-235, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241, and Pu-242. Credit will be taken for Am-241 produced by post irradiation decay of Pu-241. Fifty-four chemical assay experiments have been identified for the validation. They are from:

1) One Yankee Rowe assembly (8 samples from 3 fuel pins) [4];
2) Three Mihama-3 assemblies (8 samples from 5 pins) [4];
3) Three Trino Vercellese assemblies (14 samples from 6 fuel pins) [5
4) Two Turkey Point assemblies (5 samples from 5 fuel pins) [6];
5) Three Calvert Cliffs Unit 1 assemblies (9 samples from 3 fuel pins) [7,8,9];
6) One H. B. Robinson assembly (4 samples from one fuel pin) [10]; and
7) Five Obrigheim assemblies (6 samples from dissolved halves of five assemblies) [11,12].

A compilation of all the measurements along with details of benchmark calculations is provided in References [5, 6, and 13]. Table I summarizes some of the key features of these chemical assays.

The validation procedure begins with the ratio of the measured benchmarks and the computed best estimate predictions, which are used to determine multiplicative biases and uncertainties. The biases and uncertainties for each isotope are combined in a conservative manner into a correction factor for each isotope. The correction factors are calculated and applied conservatively to ensure that criticality safety evaluations employing the burnup credit method result in a neutron multiplication factor that is conservative for the system being evaluated.

For example, if an isotope has a bias of 0.98 and an uncertainty of 5%, the isotopic concentration correction factor would be calculated as 0.98+0.05= 1.03 for a fissile material but 0.98-0.05 = 0.93 for an absorber.

To put the above in terms of an equation, define a bias ($\mathbf{x}$) as:

$$\mathbf{x} = M / C \qquad (1)$$

where M and C are the measured and calculated isotopic content, respectively. Now the correction factor ($\mathbf{f}$) for an absorber would be:

$$\mathbf{f} = \mathbf{x} - \sigma\, t_{95,n-1}\, [(n+1)/n]^{0.5} \qquad (2)$$

where $\mathbf{s}$ is the standard deviation, $t_{95,n-1}$ is the Student's t value for 95% confidence, and $\mathbf{n}$ is the number of samples. The calculated atom density of the isotope of interest should be multiplied by $\mathbf{f}$ before the criticality calculations are performed.

TABLE I. PROPERTIES OF THE CHEMICAL ASSAYS

Sample	Burnups	Average Lethargy of Absorption	Enrichment
	(GW·d/tU)		(wt.% U-235)
Yankee Rowe	15.95, 30.39, 31.33, 20.19 32.03, 31.41, 35.97, 35.26	15.6 - 15.7	3.40
Mihama Assembly:86b	8.30, 6.92	16.2 - 16.4	3.21
Assembly:86g	15.36, 21.29	16.2 - 16.3	3.20
Assembly:87c	29.50, 32.20, 33.71, 34.32	16.1-16.3	3.21
Trino Vercellese Assembly: 104	12.04	15.82	3.90
Assembly: 032	15.38, 15.90, 11.53	15.8 - 16.0	3.13
Assembly: 069	12.86, 20.60, 23.72, 24.30, 23.87, 24.55, 23.93, 24.36, 24.33, 24.31	15.8 - 15.9	3.13
Turkey Point (5 assemblies)	30.72, 30.51, 31.56, 31.26, 31.31	16.3	2.56
Calvert Cliffs Assembly D047	27.35, 37.12, 44.34	16.3 -16.4	3.04
Assembly D101	18.68, 26.62, 33.17	16.3 - 16.5	2.72
Assembly BT03	31.40, 37.27, 46.46	16.2 - 16.3	2.45
H. B. Robinson	16.02, 23.81, 28.47, 31.66	16.2 - 16.5	2.56
Obrigheim Assembly 170	25.93	16.3	3.13
Assembly 172	26.54	16.3	3.13
Assembly 176	27.99, 29.52	16.3	3.13
Assembly 168	28.40	16.3	3.13
Assembly 171	29.04	16.3	3.13

* Multiple burnups are due to taking the samples from different axial heights or different locations in the fuel assembly.

Since it is possible that the bias, **x**, could be a function of key parameters, trend analyses are required. Upon review of the parameters that can affect **x**, the following relationship is assumed:

$$x_{fit} = (M/C)_{fit} = 1.0 + b_1 *B + b_2 *B*S + b_3*B*E + b_4*B*P \quad (3)$$

where:

x_{fit} = bias as a function of input parameters
B = burnup (GW·d/tU)
b_1 = slope for burnup
S = a spectral index (Average Lethargy of Absorption (ALA))
b_2 = slope for product of burnup*spectral index
E = initial enrichment (wt.% initial U-235)
b_3 = slope for product of burnup*initial enrichment
P = specific power (MW/tU)
b_4 = slope for product of burnup* specific power

As seen above, the burnup variable appears in each of the terms on the right side of the equation. This is because the amount of change in $\mathbf{x}_{fit}$ due to spectrum, enrichment and specific power related problems is proportional to burnup. The $\mathbf{x}_{fit}$ value at zero burnup is one because the calculated value becomes the initial condition (measured value) if there is no burnup.

Since non-zero coefficients (i.e., b_1 through b_4) would always be expected, a statistical test to determine the significance of each coefficient is required. The null hypothesis of this test is that the coefficient is zero. For all statistically significant trends a prediction band technique is used to cover the uncertainty in the bias. Finally, since regulatory practice does not allow biasing in a direction that results in less safety margin, the final correction factor is called $\mathbf{f}_{buc}$ and is defined as **f** with the limit of 1.0 (max. or min.) applied.

The isotopic validation method is applicable to any computer code system. For the purpose of demonstrating the method, the SAS2H sequence of SCALE 4.2 and the 27BURNUPLIB was used. The following equations are the correction factors consistent with the use of SCALE 4.2 and the 27BURNUPLIB:

$$\mathbf{f}_{buc}(\text{U-234}) = 0.814 \quad (4)$$

$$\mathbf{f}_{buc}(\text{U-235}) = 1.0 + 0.00105\mathbf{B} + 0.000255 * \text{SQRT}(41{,}100 + \mathbf{B}^2) \quad (5)$$

$$\mathbf{f}_{buc}(\text{U-236}) = 0.936 \quad (6)$$

$$\mathbf{f}_{buc}(\text{U-238}) = 0.991 \quad (7)$$

$$\mathbf{f}_{buc}(\text{Pu-238}) = 0.866 \quad (8)$$

$$\mathbf{f}_{buc}(\text{Pu-239}) = 1.0 - 0.000852\mathbf{B} + 0.000378 * \text{SQRT}(41{,}100 + \mathbf{B}^2) \quad (9)$$

$$\mathbf{f}_{buc}(\text{Pu-240}) = 1.0 + 0.00231\mathbf{B} - 0.000179 * \text{SQRT}(41{,}100 + \mathbf{B}^2) \text{ for } \mathbf{B} < 15.7 \quad (10)$$

$$\mathbf{f}_{buc}(\text{Pu-240}) = 1.0 \text{ for } \mathbf{B} > 15.7 \quad (11)$$

$$\mathbf{f}_{buc}(\text{Pu-241}) = 1.0 - 0.00142\mathbf{B} + 0.000347 * \text{SQRT}(41{,}100 + \mathbf{B}^2) \quad (12)$$

$$\mathbf{f}_{buc}(\text{Pu-242}) = 1.0 + 0.00300\mathbf{B} - 0.000607 * \text{SQRT}(38{,}500 + \mathbf{B}^2) \text{ for } \mathbf{B} < 40.5 \quad (13)$$

$$\mathbf{f}_{buc}(\text{Pu-242}) = 1.0 \text{ for } \mathbf{B} > 40.5 \quad (14)$$

$$\mathbf{f}_{buc}(\text{Am-241}) = 1.0 - 0.00142\mathbf{B} - 0.000347 * \text{SQRT}(41{,}100 + \mathbf{B}^2) \quad (15)$$

where **B** is burnup in GW·d/tU.

For more detail on the method and analysis refer to references [2 and 5].

3.2. Criticality Validation

With the isotopic content conservatively determined, the next step is to conservatively determine the multiplication factor, k_{eff}. Criticality analyses under the fresh fuel assumption use criticality experiments to validate the cross sections and computer codes. The burnup credit criticality analyses are also validated using criticality experiments. Fresh fuel assumption methods are typically benchmarked against low enrichment, unirradiated heterogeneous UO_2 fueled systems. The actinide-only burnup credit method is additionally benchmarked against low enrichment, unirradiated heterogeneous mixed oxide (MOX) fueled systems. MOX experiments provide benchmark data for other transuranic isotopes present in spent fuel and included in the actinide-only burnup credit analysis procedure.

The criticality validation method combines biases, uncertainties, and an administrative safety margin to arrive at an Upper Safety Limit (USL) for k_{eff} [5,14]. First, a lower prediction band is defined by the 95% confidence level for a single future calculation. The prediction band width accounts for the statistical uncertainty in the bias. Then an administrative safety margin of 5% k_m is added to establish the USL, which becomes the bounding value for the criticality safety criterion. The USL technique provides a statistically sound method of establishing the bias as a function of any parameter while incorporating an additional safety margin that is consistent with the current practice.

Since spent nuclear fuel contains less than 2% Pu at discharge, using UO_2 criticals and MOX criticals with 2% or greater Pu bounds the actinide concentrations in spent fuel. A USL can be calculated separately on the UO_2 criticals and the MOX criticals. The most limiting of the USL curves would then be the actinide-only burnup credit USL.

Fifty-seven critical experiments have been selected to establish the bias over the anticipated range of PWR burnup credit package conditions. They span the range of the various parameters associated with a spent fuel shipping package. The experiments consist of 21 UO_2 criticals (including two gadolinium criticals) and 36 mixed oxide configurations. References [3,7] describe the experiments selected.

To demonstrate the criticality validation, the CSAS criticality sequences of SCALE 4.2 using the 27BURNUPLIB cross section set were employed to analyze the 54 experiments. The USL as a function of the Average Lethargy for Absorption (ALA) is presented in Figure 3, along with results for each of the 54 validation experiments.

Further information on criticality validation can be found in the technical report prepared on validation [5].

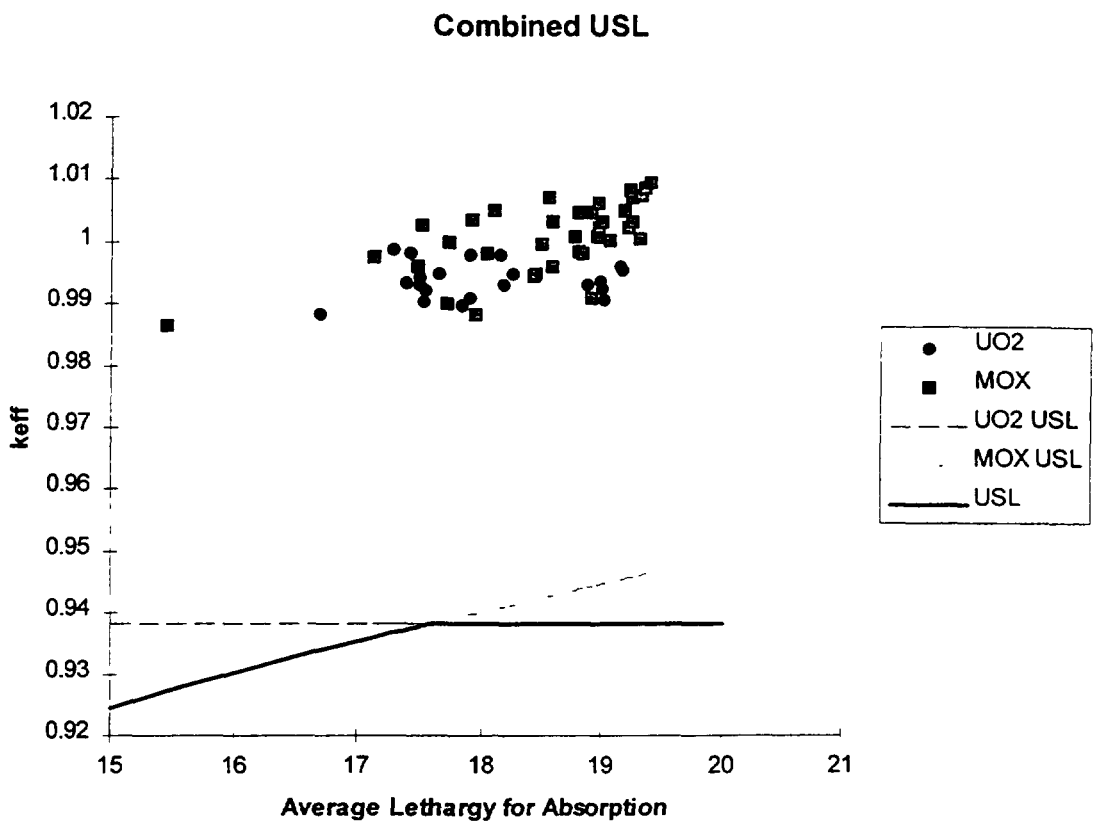

FIG. 3. Upper Safety Limit on k_{eff} for Actinide-Only Burnup Credit Using SCALE 4.2 and the 27BURNUPLIB

3.3. Analysis and Modeling Parameters

The analysis for burnup credit must be performed with validated codes at limiting conditions for the SNF package. These limiting conditions apply to the generation of SNF isotopic compositions as well as the package criticality analysis. The actual values of the limiting conditions depend on the set of assemblies that they are intended to address.

The isotopic analysis depends on the reactor conditions during the burnup. These conditions are specific power, moderator temperature, fuel temperature, soluble boron concentration, and power

versus time for the life of the fuel. The fuel reactivity is maximized by using the highest specific power (MW/tU) and temperatures [15]. A specific power of 60 MW/tU bounds PWR fuel designs and does not overly burden the analysis with conservatism. For the moderator temperature, the core outlet temperature represents a high value that conservatively covers the reactivity of discharged fuel. A high fuel temperature can be determined by calculating the average pellet temperature assuming the outlet moderator temperature and the plant technical specification for the radial peaking factor [15].

The higher the average soluble boron concentration during burnup, the more reactive the fuel assembly would be following the discharge [15]. The highest average boron concentration for any cycle for each fuel design should be used. The less time the reactor is shut down during the burnup, the more reactive the fuel assembly [15]. Therefore, the burnup analysis should be performed as one continuous burn with no down time.

The criticality analysis of the SNF package must also be done at the most limiting conditions. There are three effects that are treated slightly differently for burnup credit. First, the optimum moderator density must be established for each specific package design for at least two burnup-enrichment conditions. The second consideration is the axial burnup modeling. A large database of axial burnup profiles has been developed [16] and the most limiting shapes (as a function of burnup) have been selected [15]. Package criticality analysis is to be performed with 18 axial nodes and the limiting shapes presented in Table II. If the package has large margins, k_{eff} bias curves can be used to account for the "end effects." If k_{eff} bias curves are used, axially uniform analysis is allowed. A sample k_{eff} bias curve for a twelve-foot active fuel length is shown as Figure 4. The final effect is the horizontal burnup gradient modeling. Again, a database of assembly quadrant horizontal burnup gradients has been created [17]. Conservatively assumed burnup tilts as a function of burnup are provided in Figure 5. All package analyses must assume each assembly has two different burnups at each of the 18 axial nodes such that the maximum variation per half conforms to the tilt shown on Figure 5. The tilt can be flat to flat or corner to corner depending on which configuration produces the highest k_{eff}.

3.4. Spent Nuclear Fuel Package Loading Criteria

Burnup credit loading curves are the criteria used to determine whether it is permissible to load an assembly in an SNF package using burnup credit. This section describes the steps required to develop burnup credit loading curves. These curves identify the lowest acceptable burnup as a function of the initial enrichment.

To generate a loading curve, the maximum fresh fuel enrichment meeting the upper safety limit on k_{eff} is determined. Subsequently, a curve of required minimum burnup versus initial enrichment is developed by applying the burnup credit methodology at various initial enrichments. Loading curves may be developed for each assembly type which will be put in the SNF package. Since additional cooling time makes the loading curves less restrictive, the loading curves can also be generated as a function of cooling time.

The maximum fresh fuel U-235 enrichment that may be used in a given SNF package is determined first. The k_{eff} is calculated with a validated code system for a range of initial enrichments to determine the enrichment that produces a k_{eff} (or k + 1.645 for Monte Carlo results) equal to the upper safety limit. This is the maximum fresh fuel enrichment point and is labeled as (E_4, 0) on the loading curve (Figure 6). The loading curve consists of an abscissa that represents initial (fresh) fuel enrichment and an ordinate that represents the required minimum burnup for a given initial enrichment. Next, a vertical line is drawn at the maximum fresh fuel enrichment limit. All assemblies that have initial U-235 enrichments less than or equal to the maximum fresh fuel enrichment limit, E_4, may be stored or transported regardless of burnup.

TABLE II. LIMITING AXIAL BURNUP PROFILES

Axial Position (% of Core Height)	Normalized Burnup (Fraction of Assembly Average)		
	Profile 1 **BU < 18 GW·d/tU**	Profile 2 **18 < BU < 30 GW·d/tU**	Profile 3 **30 GW·d/tU < BU**
2.78	0.649	0.668	0.652
8.33	1.044	1.034	0.967
13.89	1.208	1.150	1.074
19.44	1.215	1.094	1.103
25.00	1.214	1.053	1.108
30.56	1.208	1.048	1.106
36.11	1.197	1.064	1.102
41.67	1.189	1.095	1.097
47.22	1.188	1.121	1.094
52.78	1.192	1.135	1.094
58.33	1.195	1.140	1.095
63.89	1.190	1.138	1.096
69.44	1.156	1.130	1.095
75.00	1.022	1.106	1.086
80.56	0.756	1.049	1.059
86.11	0.614	0.933	0.971
91.67	0.481	0.669	0.738
97.22	0.284	0.373	0.462

The required minimum burnup for a specific initial enrichment value is the burnup at which the calculated k_{eff}, using the burnup credit methodology, is just equal to the upper safety limit. The process for determining a required minimum burnup for a given initial enrichment is illustrated in Figure 7. A series of runs of validated computer codes (i.e., SAS2H and CSAS25) is performed to calculate k_{eff} values for a range of burnups to search for the burnup value that produces the reactivity limit, the upper safety limit. As indicated in Figure 7, the calculated k_{eff} is plotted against the burnup that produced that value of k_{eff}. The curve is then fitted to estimate the burnup that crosses the upper safety limit. The process is repeated for various initial enrichments, as illustrated in Figure 7. A "verifying" calculation is performed near that burnup (for each initial enrichment value) which will be less than or equal to the upper safety limit. This limiting burnup will be used with the corresponding initial enrichment to establish a point on the burnup credit loading curve.

The loading curve may contain discontinuities. These are due to changing the axial and horizontal burnup profiles at certain burnups (18 and 30 GW·d/tU using the current axial and radial burnup profile databases). The process of determining required minimum initial enrichments at the burnup discontinuities is shown in Figure 8. The process is similar to the description in the previous paragraph; however, the initial enrichment is varied this time while the burnup is fixed. Two different

minimum initial enrichments result depending on the axial burnup profiles (or k_{eff} bias values from Figure 4) and horizontal burnup gradient utilized at 18 and 30 GW·d/tU. It is required that the lower of the two minimum initial enrichments be determined. This is achieved by adopting axial burnup profiles 1 and 2 in Table II (or higher k_{eff} bias values) and the horizonal burnup gradients of 33% and 25% for 18 and 30 GW·d/tU, respectively. Determining the other initial enrichment is not required. However, it can be achieved by adopting axial burnup profiles 2 and 3 in Table II (or lower k_{eff} bias values) and the horizontal burnup gradients of 25% and 20% for 18 and 30 GW·d/tU, respectively. The distance between E_5 and E_6 or E_7 and E_8 is expected to be on the order of 0.2 or 0.1 wt. % U-235, respectively.

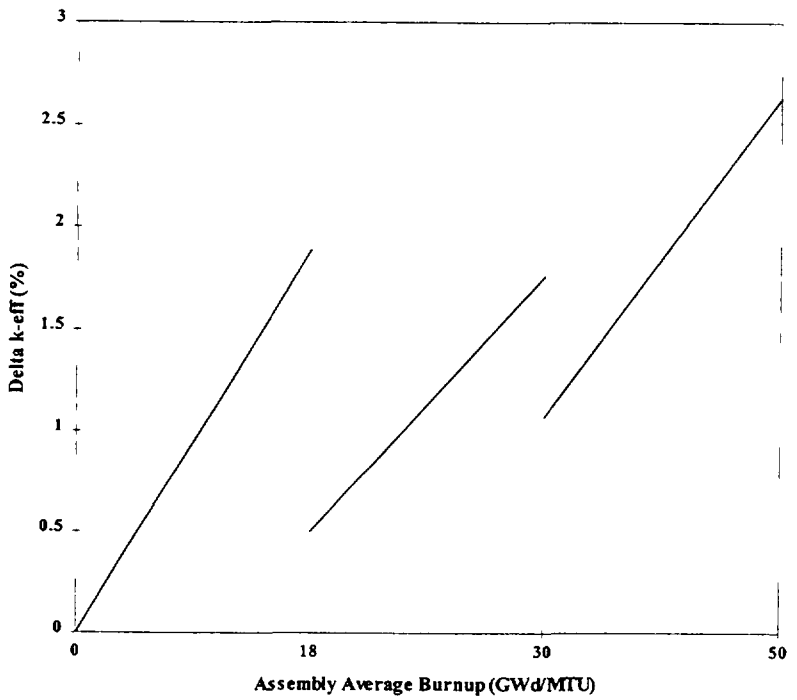

FIG. 4. k_{eff} Bias Curve for End Effects for a 12 Foot Active Core and Five Years Cooling Time

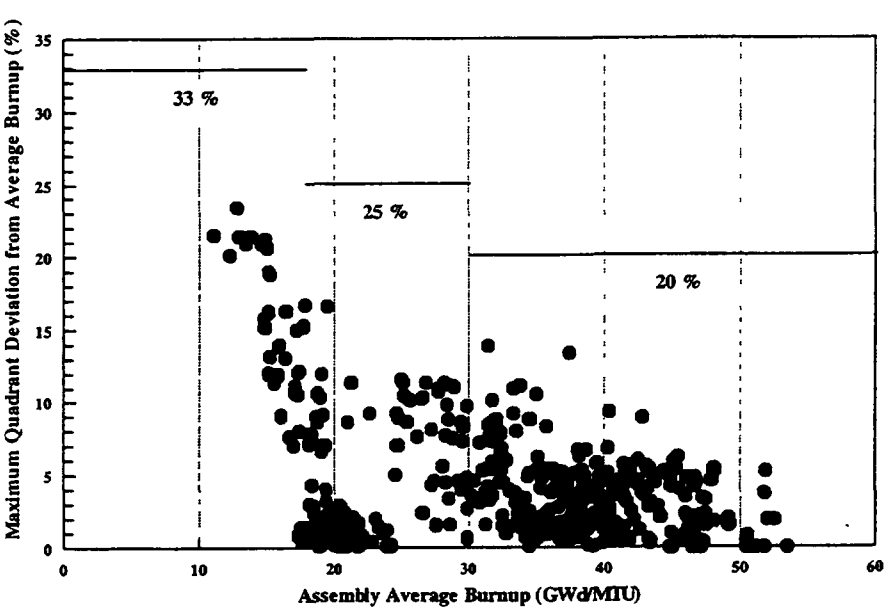

FIG 5. Quadrant Burnup Deviation Requirements and Database as a Function of Burnup

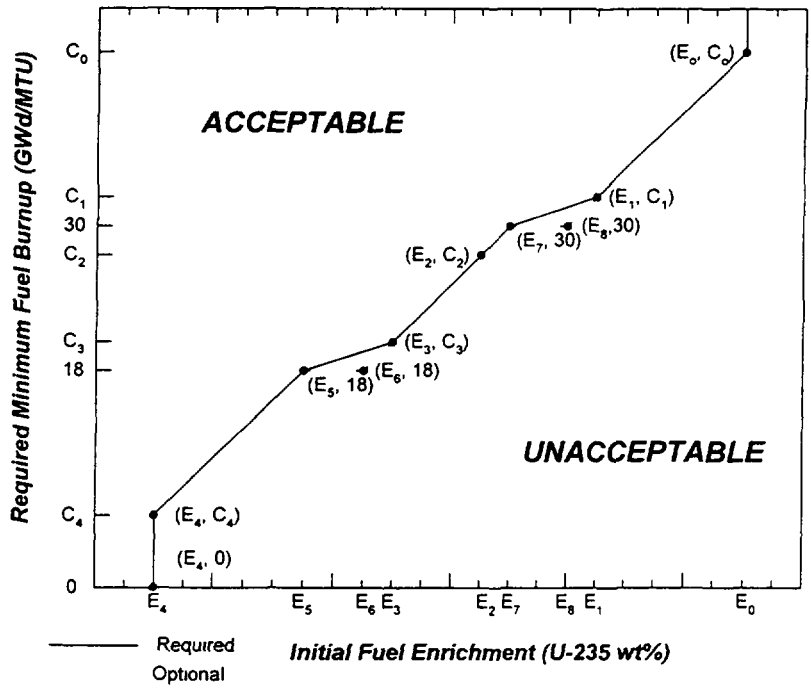

FIG. 6. Development of the Burnup Credit Loading Curve

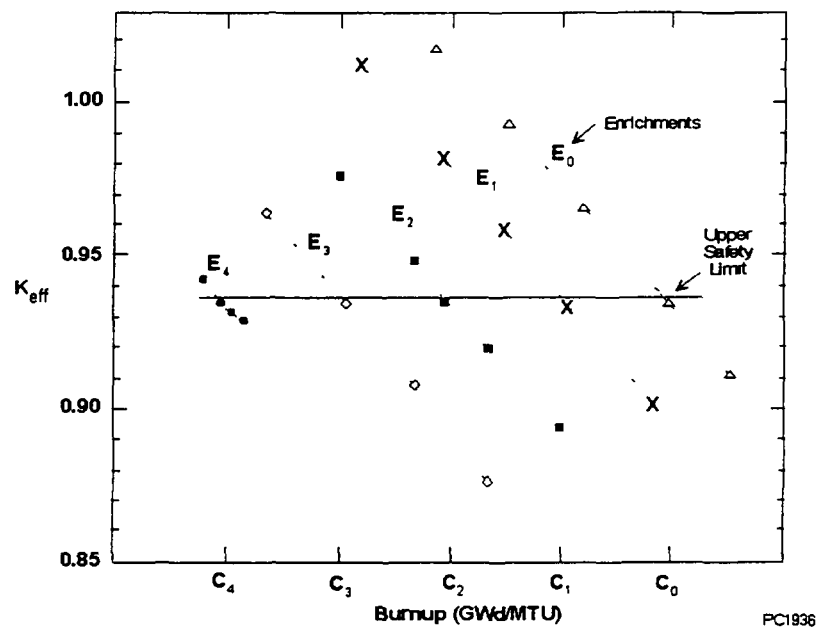

FIG. 7. Determination of the Limiting Burnup for a Given Enrichment

After the above calculations are performed, a curve of minimum burnup as a function of the initial enrichment is generated (see Figure 6). Calculations of the required minimum burnup must be performed at the maximum enrichment for the SNF package (E_0). Calculations of the required minimum burnup must also be performed at the maximum fresh fuel enrichment for the package (E_4). Burnup credit calculations will not show a zero minimum burnup for the maximum fresh fuel limit demonstrated using fresh fuel assumptions. This is because in performing the calculations, the isotopic correction factors on U-238 and U-235 are used that only need to be applied for irradiated fuel. The required minimum burnup for the highest enrichment is indicated as point C_0 on Figure 6. Subsequent values C_1 through C_n are obtained by decreasing the initial enrichment parameter by a value not to exceed 0.5 weight percent U-235 until an initial enrichment equal to the maximum fresh fuel enrichment limit is reached. The optimum moderation must be checked at point (E_4, 0) and the

point (E_0, C_0). The required minimum initial enrichments, E_5 and E_7, must be found at 18 and 30 GW·d/tU. The loading curve is created by a segmented straight line through the data points. As previously mentioned, points (E_6, 18) and (E_8, 30) may be determined and incorporated into the loading curve, but this is optional. If there is significant curvature in the loading curve at burnups other than 18 and 30 GW·d/tU, the enrichment points should be spaced so that the loading curve is smooth, with no abrupt direction changes.

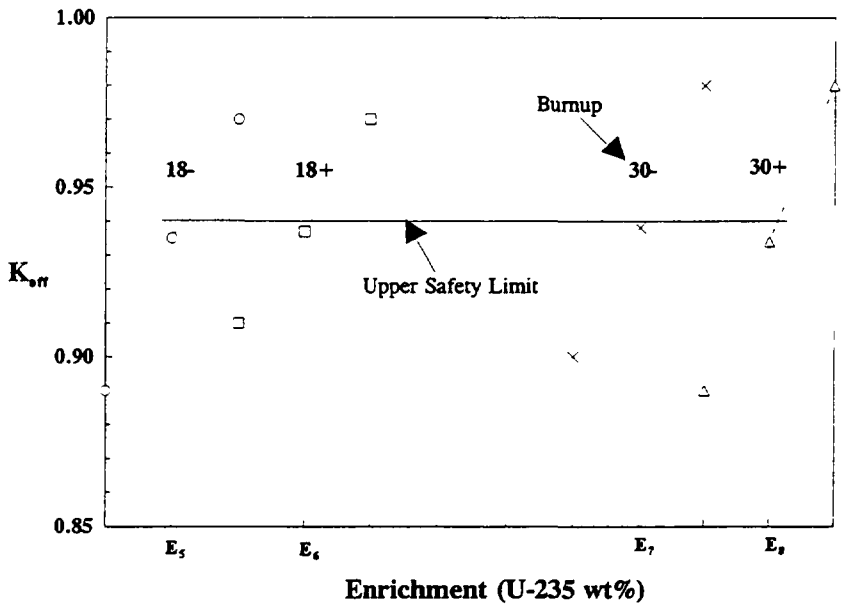

FIG. 8. Determination of the Limiting Enrichment at the Burnup Discontinuities

A spent fuel assembly that has a verified burnup greater than the required minimum burnup on the loading curve at the assembly's initial enrichment may be loaded into the SNF package. Note that an assembly that has an initial enrichment less than the maximum fresh fuel enrichment limit does not require any burnup. Conversely, an assembly that has an initial enrichment that exceeds the highest enrichment on the loading curve may not be loaded into the package regardless of its burnup. If an assembly is initially loaded with fuel of different enrichments, the maximum enrichment value anywhere in the assembly is used to compare against the loading curve. Using the maximum enrichment value conservatively bounds the reactivity of a multi-enrichment assembly.

Burnup credit loading curves should be generated for each assembly design. Separate loading curves may be generated for assemblies with removable burnable absorbers. The burnup credit loading curve will specify the minimum cooling time used in the analysis. Cooling times longer than the minimum specified are conservative for the first 100 years of cooling.

3.5. Physical Implementation and Controls

The loading of spent nuclear fuel transportation packages designed for burnup credit requires the implementation of controls during loading to ensure design basis fuel requirements and licensing conditions are met. These controls are in addition to those that are already being implemented for fresh-fuel based packages. ANSI/ANS-8.17 [18] indicates that credit may be taken for fuel burnup by establishing a maximum spent fuel reactivity and ensuring that each fuel assembly has a reactivity no greater than the maximum established by "analysis and verification of the exposure history of each fuel unit." The previous sections introduced the methodology for determining a conservative reactivity for the SNF assemblies. In addition, the actinide-only methodology [2] requires a physical measurement of each SNF assembly to verify the reactor records before loading burnup credit packages.

The analysis of an SNF package using burnup credit results in loading criteria to identify assemblies that may be placed in a burnup credit package. These criteria provide the relationship between the minimum allowable average burnup and the initial enrichment of an assembly for a given assembly design, burnable absorber (BA) loading, and cooling time. Therefore, the package loading procedure requires knowledge of the burnup, enrichment, BA loading and cooling time for a candidate assembly. This data resides in the reactor operating records. The reactor records associate this data with a storage rack location and the ID of the assembly. Part of the record, the initial enrichment, cooling time, and storage rack location, is used to satisfy the criterion for current spent fuel shipments. Thus, the operational aspects of burnup credit require only an extension of the reliance of reactor records currently used for package loading. However, such an extension increases the reliance on administrative controls to ensure criticality safety. In order to guard against an error in the reactor records, the burnup credit process includes a measurement to verify that the reactor records specified for a specific assembly correlate with the measured neutron or gamma emissions for the assembly. The verification measurement reduces reliance on administrative controls and provides sufficient additional protection against misloading to satisfy ANSI/ANS 8.1 [19].

The burnup on the loading curve is to be compared to the utility's reactor records after accounting for the uncertainty in that record. The independent burnup verification by a physical measurement is to confirm the recorded burnup prior to loading specific fuel assemblies into a burnup credit package. This confirmation approach is taken since it is generally believed that the reactor record burnup is more accurate than that from a measurement. However, the assembly is accepted for loading in a burnup credit package only if a measurement criterion for acceptance is met. This criterion must be established to be consistent with the need for confirmation as well as the technology available to do the verification. The acceptance criterion is that the measured burnup must be within 10% of the nominal reactor record burnup. This is a two-sided requirement since it is desirable to reject any assembly with an unexpected result. Although an assembly with a measured burnup greater than the reactor record by more than 10% may be safe with regard to burnup, the measurement implies a problem with the reactor record. Since no direct measurement of enrichment is required, any indication of an erroneous record is grounds for rejection of the assembly.

The question arises as to whether an unnoticed error of up to 10% would lead to an unsafe condition. First, it is projected that approximately half of this difference is accounted for in the reduction of the assembly burnup due to uncertainty in the reactor records. However, if the assembly was at the low end of the reactor record uncertainty, the maximum error in burnup would be 10%. This would imply a 10% error in the reactivity change due to burnup. Since about 30% of the change in reactivity due to burnup is from fission products [2], this unexpected event is well within the available safety margin.

The measurement acceptance criterion should not be so restrictive that the current state of the art of burnup measurement systems would produce large numbers of spurious rejections. Five percent is an engineering approximation of the uncertainty in both of the reactor records and measurement systems. Using this estimate, it would appear that deviations of greater than 10% between the measurement and reactor records would be unlikely and a reasonable basis for rejection.

Burnup measurement systems fall into two broad classes, herein termed "dependent," and "independent." Dependent systems (e.g., gross neutron detection systems) rely on knowledge of the reactor record burnup values for a set of assemblies for calibration. Therefore, these systems cannot truly "measure" burnup independently. The primary use for such systems is detection of "outlier" assemblies which for some reason have a radiation signature at odds with their reactor record burnup value. On the other hand, independent measurement systems (e.g., gamma spectrum detection systems) are capable of performing a true independent measurement of assembly burnup, without reliance on reactor records, using the gamma emission signatures of fission products (principally cesium isotopes).

For dependent systems, a calibration curve of the following form is used to correlate the neutron counts to the reactor record burnup:

$$y_{counts} = a + bx_{reac} \quad (16)$$

where a and b are constants, y_{counts} is the count rate (or, for neutron detection systems, typically the logarithm of the neutron count rate), and x_{reac} is the reactor record burnup value (or, for neutron detection systems, typically the logarithm of the reactor record value). Constants a and b are determined using standard linear regression techniques, following measurement of a group of assemblies.

The burnup uncertainty of dependent measurement systems is most conveniently stated in terms of a count rate prediction band. (Note that for dependent measurement systems, the count rate prediction band incorporates both reactor records errors and intrinsic measurement system errors.)

For dependent measurement systems, the count rate for a particular assembly should not differ from the calibration line by more than the following amount:

$$\text{Prediction Band Width (count rate)} = t_{0.025,n-2}\{[(n+1)/n+(x_i - x_{avg})^2/S_{XX}]SS_R(n-2)\}^{0.5} \quad (17)$$

where,
- $t_{0.025,n-2}$ is the Student's t-distribution statistic bounding 95% of distribution for n-2, degrees of freedom (two-sided distribution),
- n is the number of assemblies in a calibration run,
- x_i is the x_{reac} (burnup or log of burnup) for assembly i,
- x_{avg} is the average of the x_{reac}'s for all assemblies in a calibration run,
- $S_{XX} = (x_i - x_{avg})^2$,
- $SS_R = (y_i - y_{fit})^2$,
- y_i is the count rate (or log of the neutron count rate) measured for assembly i,
- y_{fit} is the value from equation 16 for assembly i.

Since, for dependent measurement systems, prediction band width on uncertainty depends on the number of assemblies measured, an appropriate bound on the band width is required to ensure an adequate sample size for the calibration curve. Thus, dependent measurement systems must demonstrate, via analysis and confirmatory testing, that the following criterion can be met:

$$\text{Prediction Band Width (converted to burnup units) / Assembly Burnup} < 0.1 \quad (18)$$

The 10% requirement on the prediction band width is consistent with the 10% value used as a acceptance criterion so the measurement system will not cause random rejections or acceptances.

Independent measurement systems should demonstrate, via analysis and confirmatory testing, the uncertainty associated with a single assembly-average burnup measurement. That uncertainty should be 10% or less. This again is consistent with the acceptance criteria.

4. CONSERVATISM IN THE ACTINIDE-ONLY BURNUP CREDIT METHOD

The methodology for utilizing actinide-only burnup credit described in this paper includes substantial conservatism. The conservatisms are included to compensate for the limited knowledge of the fuel isotopic composition (including the spatial distribution), cross sections and burnup profiles, and uncertainties in the measurements and calculational tools. This section will explore some of the issues associated with the methodology's conservatisms.

Analyses have been performed to quantify the reactivity effects due to three of the conservatisms in the methodology: the bounding depletion parameters, the isotopic correction factors,

and the exclusion of the fission products [5]. To assess each of the three effects, criticality calculations are performed using four sets of calculations with different modeling conditions. Each set consists of several combinations of typical burnups and enrichments, using a standard Westinghouse 17x17 assembly with a 5-year cooling time after the final cycle. The initial set represents best-estimate conditions, using nominal modeling parameters for the isotopic calculations, bias corrected isotopics, and fission products. The nominal modeling parameters represent average values for the fuel, clad and moderator temperatures, soluble boron concentration, and specific power. The bias corrected isotopics are computed using the isotopic biases but the concentrations are not corrected for the uncertainties.

The remaining three sets vary the modeling conditions in order to be able to quantify the various effects on the system's reactivity. The second set excludes the fission products; the third set excludes fission products and uses bounding modeling parameters for the isotopic calculations. The fourth set represents the actinide-only burnup credit methodology values, which requires bounding depletion parameters, use of conservative correction factors for isotopic concentrations, and no fission products. Using the various sets, the effects of each of the modeling considerations are computed at different burnups and enrichments, and are presented in Table III. Results shown are differences in k between the corresponding cases.

The fission product conservatism shown on Table III is large. Nevertheless, since strong documentation of individual fission products' worth is not available at this time, credit cannot be taken for fission products, and thus negative reactivity is present that is not taken credit for. Although fission product yields can be measured, the transmutation in the reactor has little experimental verification, and thus fission products' concentrations cannot be easily predicted. Therefore, although obviously present in SNF providing considerable negative reactivity, fission products are not included in the burnup credit methodology and are left as added conservatism.

TABLE III. CONSERVATISMS IN THE ACTINIDE-ONLY BURNUP CREDIT METHODOLOGY

Enrichment (wt.% U-235)	Burnup (GW·d/tU)	Fission Product Conservatism (% k)	Bounding Depletion Parameters Conservatism (% k)	Isotopic Correction Factors Conservatism (% k)	Added Conservatism (% k)
3.0	15	7.8	1.1	1.8	10.7
	30	12.2	3.1	2.4	17.7
	45	15.2	5.2	3.1	23.5
3.6	15	7.5	0.8	1.6	9.9
	30	11.9	2.3	2.2	16.4
	45	15.2	4.4	2.9	22.5
4.5	15	7.1	0.4	1.4	8.9
	30	11.4	1.4	1.9	14.7
	45	15.0	3.0	2.6	20.6

The other conservatisms shown on Table III are due to the modeling parameters and isotopic correction factors. Although not as large as the fission product values, considerable margin is provided by both of these bounding modeling conditions. It may not seem to be appropriate to talk about the correction factors since they account for the uncertainty in the data. This would be logical if these correction factors were being determined for only one isotope but since they are determined for

each isotope, the implication is that each isotope deviates from its expected value in the same direction (in the direction that creates more reactivity). Unfortunately, since the isotopes are all of different worths, it is not clear how to statistically combine the uncertainties. It is anticipated that future work may allow the combination of these errors.

Table IV uses the same analyses results to show the change in reactivity due to burnup. The third column presents the difference in k between the zero burnup case and cases at the various burnup values for the best estimate set (which includes fission products). The fourth column presents analogous results, but the computed difference is between the zero burnup case and the actinide-only burnup credit set. The fifth column gives the ratio of the values in columns four and three to show the reactivity percentage accounted for with actinide-only burnup credit. It is easily noted that credit is taken for only half of the reactivity change.

TABLE IV. CONSERVATISMS IN THE CHANGE IN REACTIVITY AS A FUNCTION OF BURNUP

Enrichment (wt.% U-235)	Burnup (GW·d/tU)	Best Estimate Change in Reactivity with Burnup (% k)	Actinide-Only Change in Reactivity with Burnup (% k)	Percent of Best Estimate
3.0	15	19.4	8.7	45%
	30	34.5	16.9	49%
	45	46.6	23.1	50%
3.6	15	18.2	8.3	46%
	30	32.8	16.4	50%
	45	45.6	23.1	51%
4.5	15	16.5	7.7	46%
	30	29.9	15.2	51%
	45	42.5	21.9	52%

Tables III and IV review only the conservatisms in the isotopic calculations and exclusion of the fission products. In addition to those, conservatism is also present due to the use of the most limiting axial burnup profiles. Again, since these profiles are all possible profiles, it might not be appropriate to consider this as a conservatism, yet most fuel assemblies have burnup profiles that do not produce positive end effects with the actinide-only assumption. In the methodology, it is assumed that the package is full of assemblies with the limiting profile. Clearly, most packages will contain assemblies with a mix of axial profiles. The magnitude of this conservatism can be estimated similar to the k_{eff} bias curves (Figure 4). This results in a few more % k conservatism.

There is also a conservatism due to the horizontal burnup tilt. Although small for large packages, the effect is considerably large for four assembly packages. For this conservatism, it is not only assumed that strong horizontal gradients exist in every assembly, but that they are loaded in the most limiting way.

Other conservatisms are also introduced in the criticality validation and measurement sections. Additionally, the method does not give credit for those assemblies with reactivities below the maximum allowed. The aggregate of these below design basis reactivities provides additional criticality safety margin and conservatism.

The methodology presented in this paper has been developed to meet the regulatory assumption of limiting k_{eff} = 0.95, which has been determined to provide an adequate safety margin. The conservatisms that have been discussed here are in excess of that margin.

5. SUMMARY

A conservative methodology has been presented that can allow higher capacity spent fuel packages. The higher capacity will result in reduced public risk to reduced transportation accident potential as well as an economic benefit. The method treats all aspects of the analysis in a demonstratively conservative way and maintains as added conservatism the fission products. It also maintains the standard assumptions of flooded conditions with a 5% k administrative margin.

In order to implement burnup credit, the method requires validating the computer code's predictions of isotopic content and reactivity worth. It also requires limiting parameters be used in the depletion modeling and package modeling. Finally, the method requires a measurement to confirm the utility reactor records of burnup.

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FINDINGS OF THE OECD/NEA STUDY ON BURNUP CREDIT

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Abstract

Findings from a four year study by an international benchmarking group in the comparison of computational methods for evaluating burnup credit in criticality safety analyses are presented in this paper. Approximately 20 participants from 11 countries have provided results for most problems. Four detailed benchmark problems for Pressurized Water Reactor (PWR) fuel have been completed and are summarized in this paper. Preliminary results from current work addressing burnup credit for Boiling Water Reactor (BWR) fuel will also be discussed as well as planned activities for additional benchmarks including Mixed-Oxide (MOX) fuels, subcritical benchmarks, international databases, and other activities.

1. INTRODUCTION

The Organization for Economic Cooperation and Development's Nuclear Energy Agency (OECD/NEA) has sponsored a criticality safety benchmark group for more than a decade. The group has addressed criticality safety issues of storage, dissolution and transportation of nuclear materials. In 1991, the benchmark group elected to pursue 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. Conventional reactor codes and data used for in-core physics calculations can be used to evaluate the criticality state of burned light water reactor (LWR) fuel. However, these codes involve complicated models and have large computational and data requirements. In reactor applications, these detailed analyses are required for the efficient operation of specific reactors. In away-from-reactor applications such as the design of casks (flasks) for the transportation of spent nuclear fuel, the candidate fuel for use in the cask may come from any reactor and it is desirable that the design bound as much of the existing and expected fuel inventories as safely possible. In other words, for reactor operations the objective is to most effectively use very specific fuel in a specific application and for away-from-reactor applications the objective is a general design for a wide variety of fuel.

Traditionally, established away-from-reactor codes (MCNP, KENO, WIMS, APOLLO, etc.) have been used for applications such as the design of storage and transportation (S/T) casks. In this type of analysis, the fuel is usually assumed to be at its full initial enrichment to provide a large safety margin for criticality safety analyses. The incentives for pursuing burnup credit over the current fresh fuel approach are widely recognized [1,2]. The approach can extend enrichment limitations for existing S/T containers, and may contribute to the development of higher capacity S/T systems that would result in fewer fuel shipments and therefore decreased risk to the public. There is also potential application to criticality safety in dissolvers for fuel reprocessing as well as for timely and efficient transport to and from reprocessing facilities.

However, before such an approach can be approved by licensing agencies, it would be necessary 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 [2,3]. Towards this end, this paper describes the suite of burnup credit criticality benchmarks that was established by the OECD/NEA Burnup Credit Criticality Benchmark Group. The benchmarks have been selected to allow a comparison of results among participants using a wide variety of calculational tools and nuclear data sets. The nature of the burnup credit problem requires that the capability to calculate both spent fuel composition and reactivity be demonstrated. The benchmark problems were selected to investigate code performance over a variety of physics issues associated with burnup credit: relative performance of fission products and actinides with respect to the multiplication factor (k) for pressurized water reactors (PWRs); trends in k and isotopic composition with burnup and enrichment for PWRs; effects of axially distributed burnup in PWRs; effects for boiling water reactors (BWRs); and effects 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. Participants used a wide variety of codes and methods based on transport and diffusion 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 multigroup and continuous energy cross section data were used in the study. Table I is a summary of the benchmark problems addressed by the group noting both the primary objective and current status of each.

The following section provides a brief description of each of the benchmark problems and a summary of results. Since the objective of the benchmark group has been, thus far, to assess code capabilities, the results are most often presented as the standard deviation (σ) among participants. The group has not attempted to make a safety case for licensing nor has there been an attempt to provide bounding values on the observed trends or physical phenomena (e.g., the effect of axially distributed burnup). Specific or suspected sources of discrepancies are discussed. Based on 2σ results, some areas for future study are identified.

2. BENCHMARK PROBLEMS

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 [5].

TABLE I. SUMMARY OF BENCHMARK PROBLEMS ADDRESSED BY OECD/NEA CRITICALITY SAFETY BENCHMARK GROUP

Benchmark	Primary Objective	Status
Phase I-A	Examine effects of 7 major actinides and 15 major fission products for an infinite array of PWR rods. Isotopic composition specified at 3.6 wt. % U-235 at 0, 30, 40 GW·Dd/tU and at 1 and 5 y cooled.	Completed 13 cases [5]
Phase I-B	Compare computed nuclide concentrations for depletion in a simple PWR pin-cell model, comparison to actual measurements at 3 burnups (27.34, 37.12, and 44.34 GW·Dd/tU). Comparisons made for 12 major actinides and 15 fission products for each burnup case.	Completed 3 cases [6]
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. Isotopic compositions specified.	Completed 26 cases [7]
Phase II-B	Repeat study of Phase II-A in a 3-D geometry representative of a conceptual burnup credit transportation container. Isotopic compositions specified.	Completed 5 cases
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. Isotopic compositions specified.	Preliminary results for 22 cases
Phase III-B	Compare computed nuclide concentrations for depletion in a BWR pin-cell model.	Draft specifications
Phase IV	Investigate burnup credit for MOX spent fuel	Proposed
Phase V	Investigate burnup credit in subcritical systems	Draft specifications

Phase I-A is perhaps the most detailed of the benchmark problems in terms of types of data collected and analyzed. Participants were asked to provide the following: 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 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.

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].

TABLE II. RESULTS OF PHASE I-A: AVERAGE MULTIPLICATION FACTOR, k (2σ) [5]

Nuclides Set	Fresh Fuel	30 GW·Dd/tU* 1 y cooled	40 GW·Dd/tU* 1 y cooled	30 GW·Dd/tU* 5 y cooled	40 GW·Dd/tU* 5 y 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

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.

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.[10,11] The detailed problem description and results are given in [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 list of nuclides with a standard deviation greater than 10% 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 $\Delta k/\%\Delta N$ of 0.01%.

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

The configuration considered in this benchmark problem was 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 $V_{mod}/V_{ox} = 2.0$; different burnups were considered (0, 10, 30 or 50 GW·Dd/tU) 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 effect 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. In total, 22 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 [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 [6]

Nuclide	Case A (27.35 GW·Dd/tU)[a]		Case B (37.12 GW·Dd/tU)		Case C (44.34 GW·Dd/tU)	
	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 GW·Dd/tU) and with fission products have some indications of increasing 2σ when axially distributed burnup is considered. At higher burnup (50 GW·Dd/tU with and without fission products) there is a suggestion of a trend in 2σ with cooling time. A comparison 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: 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 [7]

Case	Initial Enrichment	Burnup GW·Dd/tU	Cooling time(y)	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)

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 V_{mod}/V_{ox} = 1.67; as in Phase II-A, the fuel was divided axially into 9 symmetrical zones; burnups of 0, 30 and 50 GW·Dd/tU 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.

This benchmark is in the final stages of study by the working group and the detailed results are expected to be published as an OECD/NEA report in late 1996. Table V is a summary of the current status of this benchmark. Note that there could be additional submissions/corrections before the final results are published and, therefore, Table V should be considered preliminary.

TABLE V. PRELIMINARY OF PHASE II-B RESULTS - AVERAGE MULTIPLICATION FACTORS

Case	Initial Enrichment	Burnup GW·Dd/tU	Cooling time(y)	Fission Products	Burnup Profile	k (2σ)
1	4.5 wt %	Fresh	N/A	N/A	N/A	1.1256 (0.0155)
2	4.5 wt %	30	5	Yes	No	0.8934 (0.0065)
3	4.5 wt %	30	5	No	No	0.9714 (0.0099)
4	4.5 wt %	30	5	Yes	Yes	0.8949 (0.0087)
5	4.5 wt %	30	5	No	Yes	0.9640 (0.0106)
6	4.5 wt %	50	5	Yes	No	0.7641 (0.0042)
7	4.5 wt %	50	5	No	No	0.8735 (0.0065)
8	4.5 wt %	50	5	Yes	Yes	0.7929 (0.0058)
9	4.5 wt %	50	5	No	Yes	0.8781 (0.0077)

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.

Phase III: Proposed BWR Studies

Phase III-A: 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 GW·Dd/tU, 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.

Preliminary 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 Pu production rate is high compared to the 40% cases and the cases with an axially distributed void fraction. The detailed results are not presented in this paper because of their preliminary nature (not yet reviewed by the working group). It is expected that the detailed results for this benchmark problem will be published with an OECD/NEA designation early in 1997.

Phase III-B: 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 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 GW·Dd/tU for each of the three void fraction cases.

The working group is expected to begin evaluating this benchmark problem in late 1996. Normally, the evaluation period for a benchmark problem is 18 to 24 months from acceptance of a specification to publishing results.

3. ADDITIONAL STUDIES

Spent Fuel Isotopic Composition Database

Reference 12 discusses a database of LWR spent fuel assay data that has been compiled. This database system, SFCOMPO [13], contains data collected from 13 LWRs, including 7 PWRs and 6 BWRs in Europe, the USA and Japan. Over the past year, axial burnup profiles from 2 Japanese reactors have been added to the database. The database will be maintained by adding new data as they become available, revising old data as necessary, and providing recommendations for criticality evaluations. This database is unique and provides a valuable resource for the evaluation of burnup credit.

Criticality Benchmark Experiments

There have been several activities that involve experiments that are applicable to burnup credit.

Exponential Experiments in the Tank Typed Critical Assembly (TCA) of JAERI [14]: Reference 15 describes an experimental technique which has been applied to 2 PWR spent fuel assemblies stored in a pool after post irradiation examination. The technique measures the exponential decay factor in the axial direction, which is one of the eigenvalues representing the degree of subcriticality. The measured results are in good agreement with calculations based on a 4-group neutron diffusion model. The effective multiplication factor of the assembly is estimated from the decay factor. The estimated multiplication factors were found to be in good agreement with MCNP analyses. Chemical assay data are also available for these assemblies.

International CERES Experimental Programme: The CERES programme was designed for the validation of cross section data and inventory predictions for actinides and fission products important to the study of burnup credit [2]. The principal participants were from France and Great Britain with some United States involvement in the latter stages of the programme. CERES was part of an extensive experimental programme developed at CEA Cadarache involving oscillation experiments in the MINERVE reactor [16]. The British experiments [17,18] were performed in the DIMPLE reactor at the United Kingdom Atomic Energy Agency (UKAEA) in Winfrith and were also supported by

British Nuclear Fuels Limited (BNFL). These experiments provide data for the validation of major fission product cross sections, reactivity worths of UO_2 samples enriched with separated fission products, spent fuel worths as well as chemical assay data for spent fuel.

Fission Product Experiments by the Institut de Protection et de Sûreté Nucléaire (IPSN): CEA/IPSN has initiated an experimental programme [19] to provide benchmark data for 6 fission products; Rh-103, Cs-133, Nd-143, Sm-149, Sm-152, and Gd-155. The critical experiments use a tank assembly and with the fission products in solution. Results for Sm-149 have been published [19] indicating reactivity worths of Sm in the assembly at 2000-6000 pcm. The benchmarks results indicate the Sm-149 cross section at energies less than 1 < eV is well qualified. The experimental programme is ongoing.

Spent Fuel Safety Experiment (SFSX): The United States has proposed a set of experiments, Spent Fuel Safety Experiment (SFSX) [20], to provide integral benchmarks for validating spent fuel reactivity. The assembly height is approximately 30 cm which should allow independent study of the "end effect." The SFSX critical assembly is a fuel replacement experiment designed to measure the critical array size for three fuel configurations, fresh fuel, spent fuel center region, spent fuel and regions. The spent fuel to be used in the experiment is from a US PWR (CE14x14) that has been carefully analyzed including measured fuel composition and burnup. This experiment is in the proposal stage.

Effect of Nuclide Radial Distribution in LWR Pins

The effect of the nuclide radial distribution inside a LWR spent fuel rod was investigated in an extreme burnup case, 63 GW·Dd/tU. The calculated results were compared with experimental nuclide profiles obtained from a 17x17 PWR assembly irradiated for 5 cycles. Both fission products and actinides were investigated. The reactivity calculations of cooled LWR assemblies indicated negligible reactivity worth due to radial nuclide distributions, about 30 pcm. These results permit the recommendation of burnup credit calculation with averaged nuclide concentrations and one mesh point in the fuel rod [21].

4. FUTURE WORK

The benchmark group is continuing to pursue studies with BWR fuel and to initiate studies with MOX fuel (proposed Phase IV benchmark). Other proposals being evaluated by the working group include pursuing an international criticality handbook, a benchmark calculation for basic minimum critical values, a subcritical benchmark problem (proposed Phase V benchmark) and an international database for axial burnup profiles.

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