DEVELOPMENT OF BURNUP CREDIT EVALUATION METHODS AT JAERI

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ABSTRACT

A guide material to introduce the burnup credit in Japan has been prepared and published in the name of *a Guide Introducing Burnup Credit, Preliminary Version*. In this *Guide*, correction factors expressed by C/E (ratio of calculated over measured isotopic composition) are prepared for applying to the results calculated by the ORIGEN 2.1 depletion code. A total of 38 sets of PIE data from PWR such as Obrigheim, Genkai-1, Mihama-3 and Takaham-3 have been used for this calculation. In addition, two simplified evaluation methods, "Equivalent Uniform Burnup" and "Equivalent Initial Enrichment", have been developed and prepared to perform criticality safety analysis by the ORIGEN2.1 depletion code and the KENO-Va criticality code without any detailed assumption.

INTRODUCTION

Recently, it becomes evident that light water rectors (PWR and BWR) are continually operated to generate electric power in the long run in Japan, and the exhausted spent fuel is more and more accumulated. Most spent fuel is now stored in on-site wet storage pools, and partly at on-site dry storage facilities such as that for the Fukushima Daiichi NPP. Some spent fuel has been transported to overseas reprocessing plants to extract valuable plutonium as new fuel material. In anticipation of lack of storage capacity in the next few decades, an off-site intermediate spent fuel storage facility has been decided for operation from around 2010. In addition, the first domestic commercial reprocessing plant is now under construction and is expected to begin operation from 2005.

In Japan, it is traditional to assume the use of fresh fuel in criticality safety assessments for spent fuel transport and storage, resulting in an excessive safety margin taken in the facility design. As an exception, burnup credit for uranium and plutonium composition is only taken in the design and management of the Spent Fuel Receiving and Storage Building of Rokkasho Reprocessing Plant (RRP).

In consideration of the above situation, new challenges have emerged to incorporate burnup credit into criticality safety assessments among utilities and related industries for pursuit of efficient facility design and management of spent fuel storage and transportation, while ensuring adequate safety margin. This is all the more evident when we note that the 1996 IAEA regulation shall be introduced into the national transport regulations early next year (2002).

In order to implement burnup credit in the design and management of spent fuel storage and transportation systems, *A Guide Introducing Burnup Credit, Preliminary Version*, has been successfully prepared and published following discussions of the Working Group on Nuclear Criticality Safety Data (Chairman: Professor Dr. Yoshihiro Yamane of Nagoya University) under the Special Committee on Nuclear Criticality Safety, at JAERI. A procedure chart and corresponding instructions with relevant methods and data are provided in the Guide for analysis and evaluation of criticality safety of spent fuel systems, taking burnup into account. In this connection, correction factors for the calculated results have been prepared by the widely used spent fuel depletion code ORIGEN2.1 together with a variety of nuclear data library. Simplified evaluation methods such as " Equivalent Uniform Burnup" and " Equivalent Initial Enrichment" have also been prepared to obtain the neutron multiplication factors for spent fuel systems.

A GUIDE FOR INTRODUCING BURNUP CREDIT, PRELIMINARY VERSION

This Guide states that it is published as a supplement to the *Nuclear Criticality Safety Handbook Version 2* in chapter 1. Basic principles and procedures for criticality safety assessment introducing burnup credit are depicted in Fig.1 in chapter 2. In chapters 3 and 4 are described methods for evaluation of spent fuel composition and criticality safety, such as methods to derive correction factors applied to nuclide composition data calculated by ORIGEN2.1 and simplified methods called "Equivalent Uniform Burnup" and "Equivalent Initial Enrichment," together with relevant data and information. The last chapter 5 contains important remarks for implementing burnup credit, together with methods to ensure burnup of spent fuel during handling operation.

CORRECTION FACTORS FOR CALCULATED NUCLIDE COMPOSITION

Correction factors have been derived to apply to the results obtained by ORIGEN2.1, which is used world wide to calculate nuclide composition in spent fuel. In this attempt, 38 experimental sample data of spent fuel from PWRs, such as Obrigheim, Genkai-1, Mihama-3 and Takahama-3, are analyzed in combination with various nuclear data libraries to give (C/E), namely the ratio of calculated result divided by the experimental one. The correction factor can be chosen as the minimum value among (C/E)s given for fissile actinides, and as the maximum value among (C/E)s given for absorptive nuclides in respect of reaction with neutrons. In addition, the value of 1.0 is given to the correction factor whenever the minimum value goes below 1.0 and the maximum value goes above 1.0 in order to assure further conservatism. Table 1 shows a list of the correction factors derived by the method described above. These are listed in *A Guide Introducing Burnup Credit, Preliminary Version.*

EQUIVALENT UNIFORM BURNUP AND *EQUIVALENT INITIAL ENRICHMENT*

Based on the PWR spent fuel composition chemical analysis data measured at JAERI, two kinds of simplified methods, "Equivalent Uniform Burnup" and "Equivalent Initial Enrichment", have been introduced for criticality safety evaluation of spent fuel storage pool and transport casks, taking burnup of spent fuel into consideration.

These simplified methods can be used to obtain an effective neutron multiplication factor for a spent fuel storage/transportation system by using the ORIGEN2.1 burnup code and the KENO-Va criticality code without considering axial burnup profile in spent fuel and various other factors introducing calculated errors. "Equivalent Uniform Burnup" is set up so that its criticality analysis will be reactivity equivalent with detailed analysis, in which the experimentally obtained isotopic composition together with a typical axial burnup profile and various factors such as irradiation history are considered on the conservative side. On the other hand, “Equivalent Initial Enrichment” is set up so that its criticality analysis will be reactivity equivalent with detailed analysis such as above when used in the so called fresh fuel assumption. The methods to derive “Equivalent Uniform Burnup” and “Equivalent Initial Enrichment” are shown in Fig.2 and Fig.3. Examples of “Equivalent Uniform Burnup” and “Equivalent Initial Enrichment”, applied to a spent fuel infinite array submerged in water simulating a storage pool and a spent fuel transport cask, are illustrated in Fig.4 through Fig.7.

SUMMARY

A Guide for Introducing Burnup Credit, Preliminary Version has been successfully prepared and published. Materials containing relevant data and detailed consideration will be published soon to give theoretical and practical background to this Guide. Presently, actinide only burnup credit (Level 1) has been introduced and advancement will be made with fission products data available in the near future. Correction factors have been prepared to apply to ORIGEN2.1 calculated results for PWR spent fuel nuclide composition. It is anticipated that those for BWR spent fuel should be obtained hereafter. Criticality experiments with fresh MOX fuel have been conducted so far and the obtained data have been utilized to validate criticality codes for Level 1 burnup credit application. To promote to Level 2 burnup credit application, criticality experiments with spent fuel will be needed. Finally, burnup monitor is mainly used at present to determine burnup of spent fuel prior to loading to storage pool and transport casks. In consideration of its complexity and tedious maintenance work, monitoring device should be replaced by the methods referring only to reactor management data.

REFERENCES

[1] Working Group on Nuclear Criticality Safety Data, Nuclear Fuel Cycle Facility Safety Research Committee, “A Guide Introducing Burnup Credit, Preliminary Version”, Japan Atomic Energy Research Institute, July 2001.

[2] Working Group on Nuclear Criticality Safety Data, Nuclear Fuel Facility Safety Research Committee, “Nuclear Criticality Safety Handbook Version 2”, Japan Atomic Energy Research Institute, March 1999.

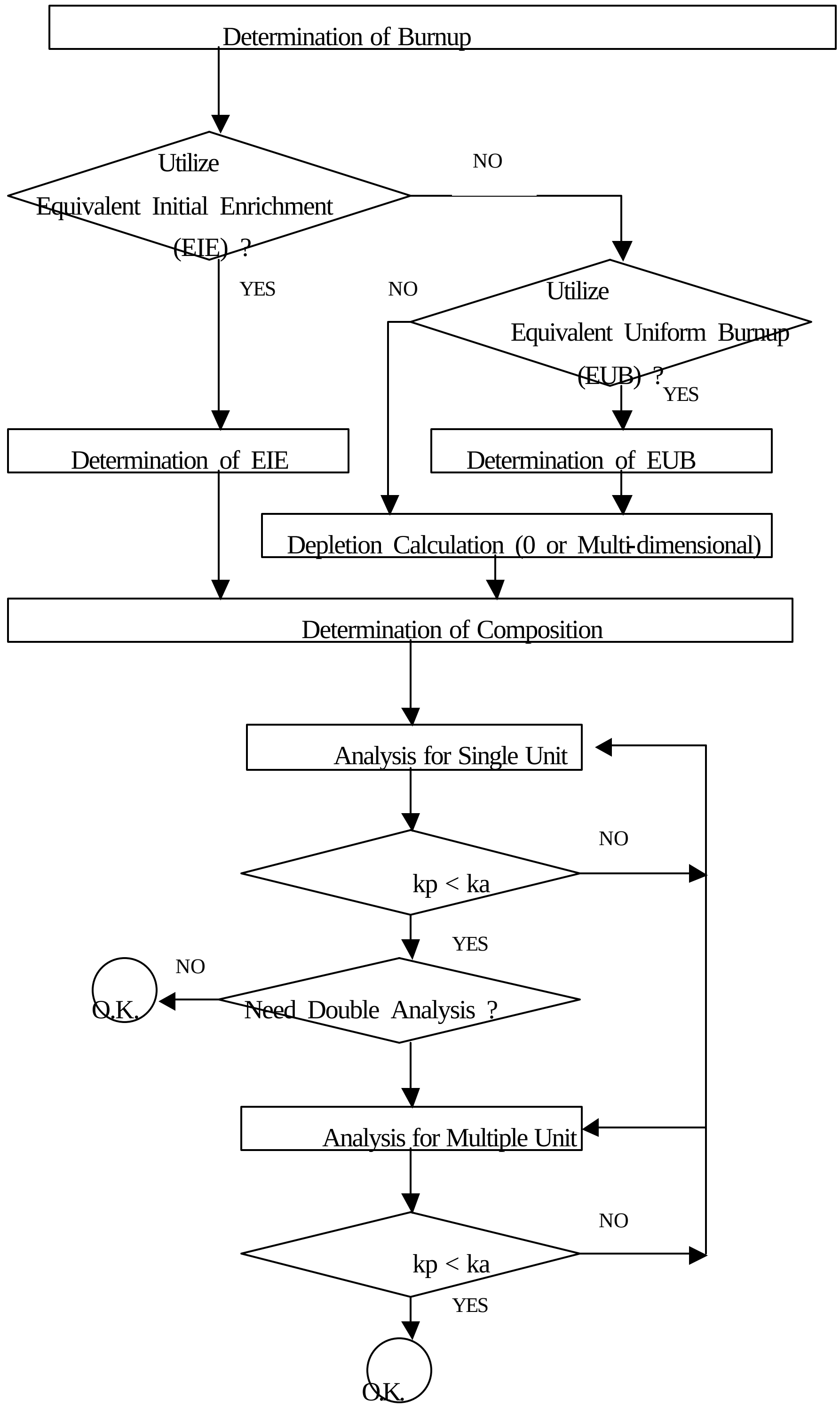

Fig.1 Procedure for criticality safety evaluation of spent fuel transport and storage system taking burnup credit into consideration.

Table 1 Correction Factors to apply to ORIGEN2.1 calculated results

Isotope	Obrigheim		Mihama-3		Genkai-1		Takahama-3	
	PWR-U	PWR-US	PWR-U	PWR-US	PWR-U	PWR-US	PWR-UE	PWR41J32
U-234	-	-	1.15	1.15	1.00	1.00	1.30	1.29
U-235	0.73	0.69	0.91	0.87	0.88	0.82	0.89	0.99
U-236	1.09	1.10	1.06	1.07	1.00	1.00	1.00	1.00
U-238	1.01	1.01	1.01	1.01	1.01	1.01	1.01	1.01
Pu-238	1.49	1.59	1.00	1.01	1.00	1.00	1.10	1.00
Pu-239	0.94	0.95	0.85	0.83	0.86	0.94	0.91	0.97
Pu-240	1.36	1.23	1.08	1.01	1.00	1.00	1.16	1.07
Pu-241	0.94	0.99	0.78	0.84	0.86	0.81	0.86	0.92
Pu-242	1.85	1.96	0.94	1.08	0.87	0.86	0.93	0.99
Am-241	2.41	2.62	1.06	1.18	0.82	0.78	1.51	1.62

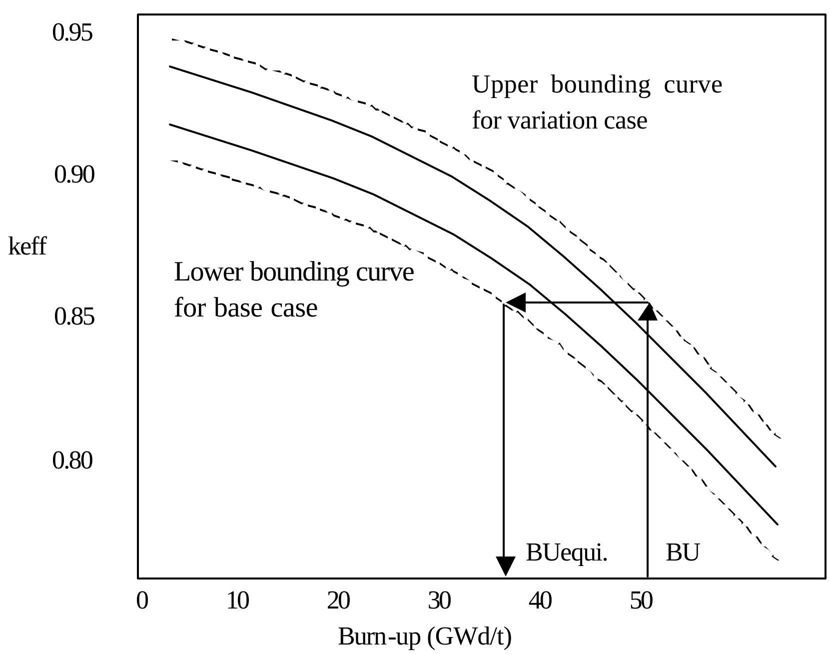

Fig.2 Diagram to obtain "Equivalent Uniform Burnup", where variation case means criticality calculation considering burnup distribution with nuclide composition measured by chemical analyses, and base case means criticality calculation considering uniform burnup distribution with nuclide composition calculated by ORIGEN2.1 code.

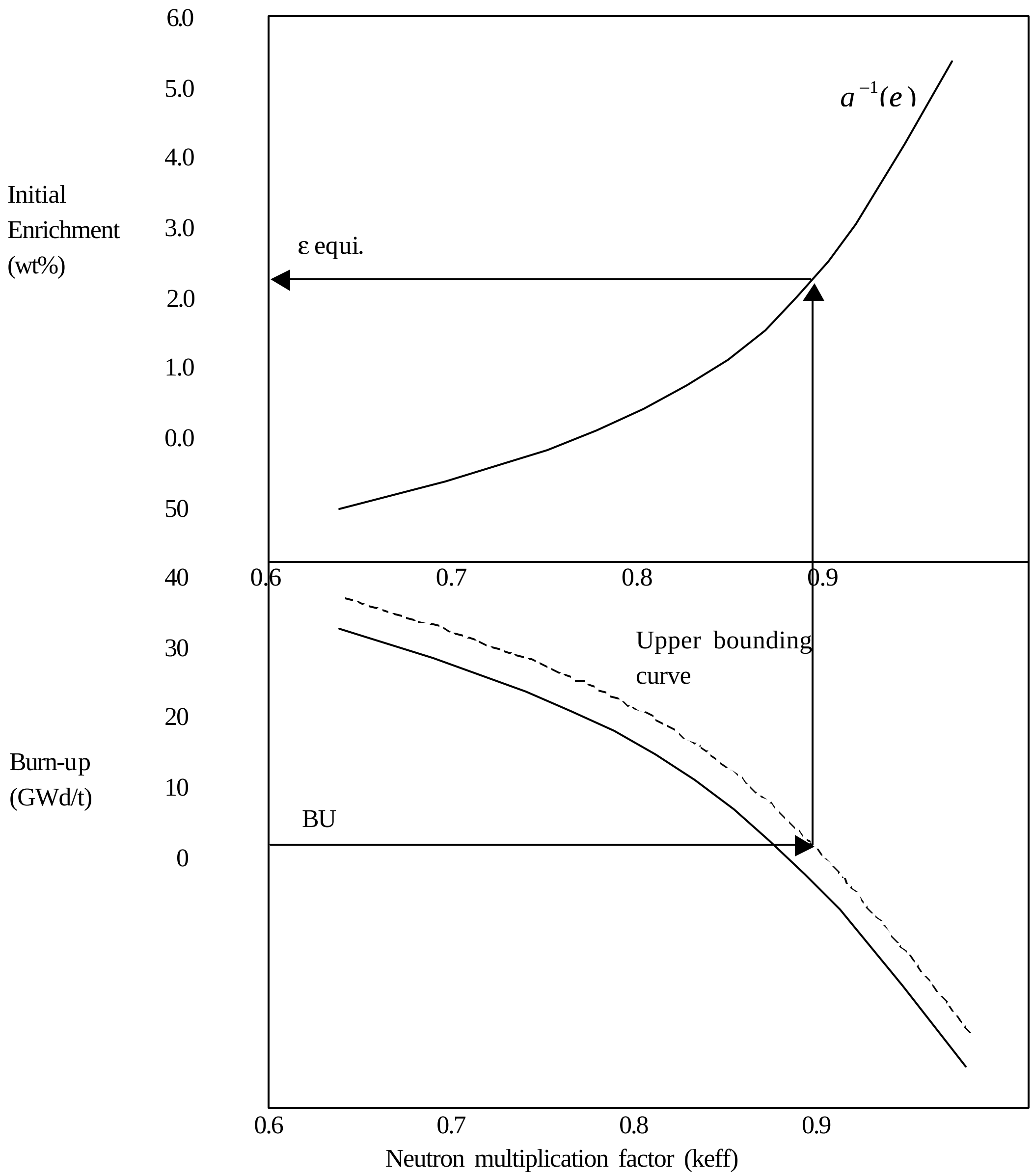

Fig.3 Diagram to obtain "Equivalent Uniform Burnup", where variation case means criticality calculation considering burnup distribution with nuclide composition measured by chemical analyses.

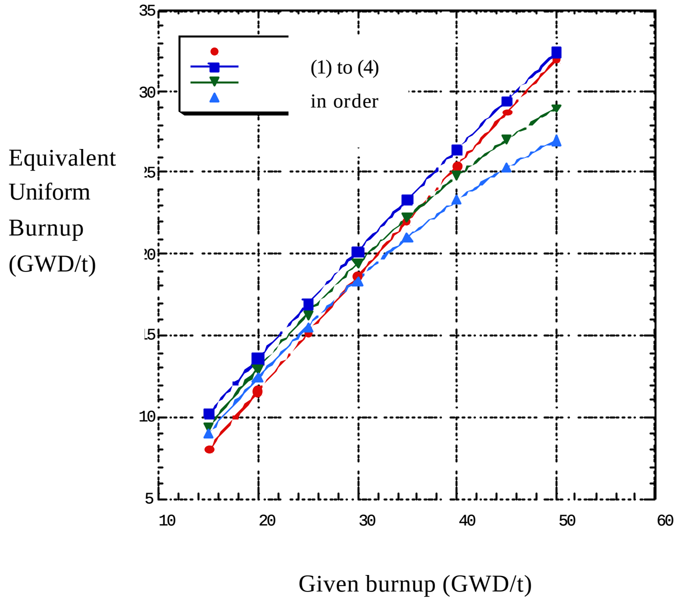

Fig.4 Correlation diagram of "Equivalent Uniform Burnup" for an unit cell infinite array system simulating spent fuel storage pool, showing the cases (1) for actinides only with 0 cooling, (2) for actinides only with 30 years cooling, (3) for actinides plus fission products with 0 cooling and (4) for actinides plus fission products with 0 cooling.

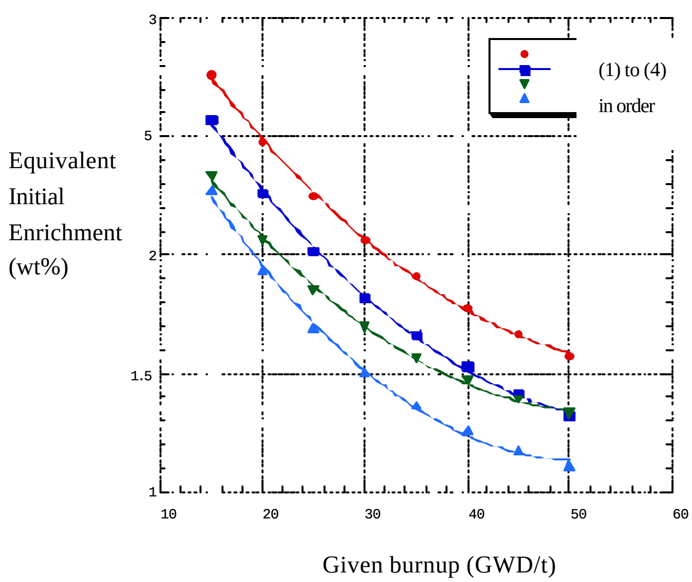

Fig.5 Correlation diagram of "Equivalent Initial Enrichment" for an unit cell infinite array system simulating spent fuel storage pool, showing the same cases as in Fig.4.

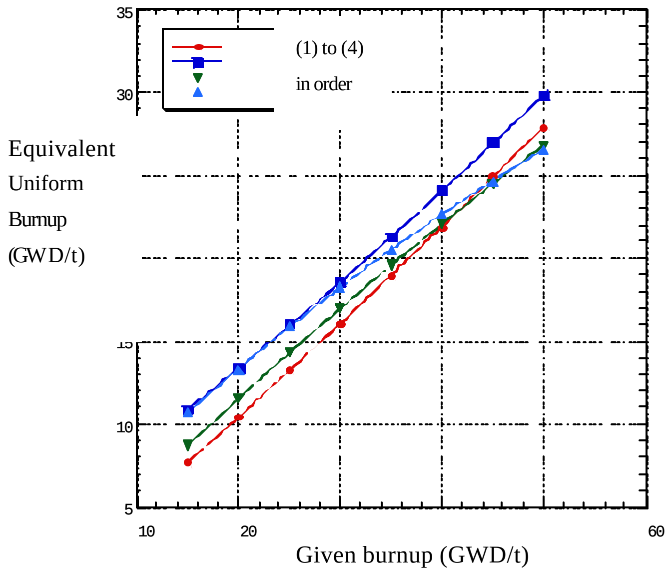

Fig.6 Correlation diagram of "Equivalent Uniform Burnup" for a spent fuel transport cask system simulating spent fuel storage pool, showing the same cases as in Fig.4 and Fig.5.

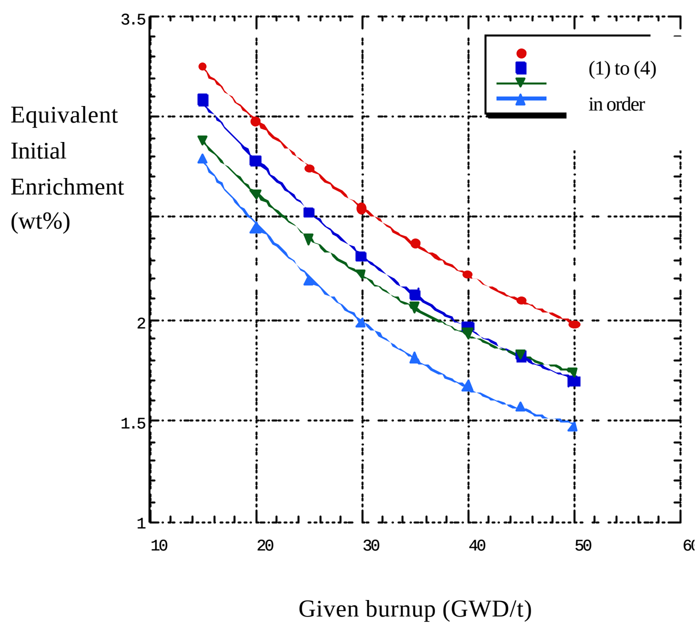

Fig.7 Correlation diagram of "Equivalent Initial Enrichment" for a spent fuel transport cask system simulating spent fuel storage pool, showing the same cases as in Fig.4, Fig.5 and Fig.6.