ANALYSIS OF BURNUP CREDIT ON SPENT FUEL TRANSPORT / STORAGE CASKS - ESTIMATION OF REACTIVITY BIAS

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SUMMARY

Chemical analyses of high burnup UO_2 (65 GWd/t) and MOX (45 GWd/t) spent fuel pins were carried out. Measured data of nuclides' composition from U234 to Pu242 were used for evaluation of ORIGEN-2/82 code and a nuclear fuel design code (NULIF).

Criticality calculations were executed for transport and storage casks for 52 BWR or 21 PWR spent fuel assemblies. The reactivity biases were evaluated for (1) axial and horizontal profiles of burnup, and historical void fraction (BWR), (2) operational histories such as control rod insertion history, BPR insertion history and others, and (3) calculational accuracy of ORIGEN-2/82 on nuclides' composition. This study shows that introduction of burnup credit has a large merit in criticality safety analysis of casks, even if these reactivity biases are considered.

The concept of equivalent uniform burnup was adapted for the present reactivity bias evaluation and showed the possibility of simplifying the reactivity bias evaluation in burnup credit.

INTRODUCTION

An introduction of burnup credit can reduce costs of spent fuel transport / storage casks, especially for high burnup fuels and MOX fuels, by rational reactivity analysis of spent fuel and reducing redundancy in the cask design. As actinide and fission product compositions of spent fuel are evaluated by burnup codes such as ORIGEN-2 code and nuclear fuel design codes, the verification of these burnup codes is important for introducing burnup credit, using chemical analysis data of spent fuel compositions. Additionally, burnup / historical void fraction profile in the fuel rod axial direction, operating histories and other parameters affect nuclides' composition of spent fuel and cask reactivity. This study evaluated effects of these parameters on spent fuel transport / storage cask reactivity.

CHEMICAL ANALYSIS AND EVALUATION OF BURNUP CODES

The chemical analyses of high burnup UO_2 and MOX spent fuel pins were carried out by the European Commission JRC Institute for Transuranium Elements (ITU). Characteristics of fuel pins are shown in Table 1. Four samples of a high burnup UO_2 fuel rod were used for chemical analysis considering axial burnup profile. Burnups of UO_2 fuel samples are 54, 61, 65, and 65 GWd/t. As major actinides are used for burnup credit evaluation in this study, measured results of nuclides' composition from U234 to Pu242 were used for evaluation of ORIGEN-2 code and a nuclear fuel design code.

Fig. 1 and Fig. 2 show representative previous data of UO_2 fuel obtained by chemical analyses and the conservative boundary curves shown in JAERI-Tech 95-048 for U235/total-U and total-Pu/total-U, respectively. These Figures show that previous studies can be completed with present chemical analysis data for the high burnup region and that the conservative boundary curve of JAERI-Tech 95-048 is also valid for the estimation of U235/total-U and total-Pu/total-U in higher burnup.

To estimate the accuracy of nuclides' composition from computational analyses, the calculated results of ORIGEN-2/82 code were compared with Nakahara's, Adachi's and the present data obtained from UO_2 fuel. NULIF code (NFI's nuclear fuel design code) was also evaluated by the present data. Fig. 3 shows the average square errors of these codes for uranium and plutonium nuclides as representative nuclides. NULIF code has smaller errors in the evaluation of uranium and plutonium nuclides in UO_2 spent fuel than ORIGEN-2/82 code, mainly due to considering fuel assembly and reactor operational data.

EVALUATION OF REACTIVITY BIAS

The criticality calculations were executed for transport and storage casks for 52 BWR or 21 PWR spent fuel assemblies. Three dimensional models are adopted for the present calculations using the KENO-Va Monte Calro code of SCALE-4 code system with the neutron cross section library of SCALE 27-neutron-groups ENDF/B-IV library. In criticality analyses, only uranium and plutonium dioxides were considered as major actinides of spent fuel.

As evaluation errors of uranium and plutonium nuclides cause an error in criticality calculations, the reactivity bias of ORIGEN-2/82 calculation should be evaluated. The sensitivity of each major actinide nuclide on cask reactivity was calculated and combined with ORIGEN-2/82 evaluation error of each nuclide (Fig. 3). The reactivity biases of ORIGEN-2/82 code shown in Fig. 4 are calculated by summation of absolute values of the sensitivity coefficient multiplied with the evaluation error for each nuclide.

LWR spent fuel assemblies have axial burnup profiles with higher burnups in the axial central regions and lower burnups in both axial ends of the rod. In the case of BWR, it is further necessary to consider the axial void distribution in the coolant. Examples of axial profiles of burnup and historical void fraction are shown in Fig. 5. Fig.6 shows the reactivity change due to burnup and historical void fraction with axial profile (i.e. Axial Profile) and without axial profile (i.e. Uniform Axial Profile). From Fig.6, the cask reactivity with consideration of the axial profile is higher than that of uniform profile of burnup and historical void fraction. Fig.7 shows the axial profile of fission density at 20GWd/t and 45GWd/t. The peak of fission density in the upper fuel region is increased as fuel burnup extends. Higher fission density of the upper fuel region in the case of axial profile consideration introduces relatively higher cask reactivity.

Horizontal burnup profiles within a fuel assembly were also analyzed for BWR and PWR fuel assemblies. These analyses showed that horizontal burnup profiles have very small reactivity effect less than 1% $\Delta k/k$ on cask reactivities.

Spent fuel operational history including control rod insertion or BPR insertion affects actinides' composition and reactivity. In this study, control rod insertion history (BWR), BPR insertion history (PWR), boron concentration history (PWR), power density and cooling time after burnup were considered.

The effects of burnup credit were -27% $\Delta k/k$ (BWR) and -19% $\Delta k/k$ (PWR) at the bundle average burnup of 45 GWd/t in case of uniform burnup profile and 40% historical void fraction (BWR). In this case, the effects of actinide burning were introduced into burnup credit. Consequentially, Table 2 shows the various reactivity biases in comparison with reactivity of uniform axial profile. Reactivity biases of nuclide composition calculation error of ORIGEN-2/82 in addition to the above reactivity biases of operational history are summarized in Fig. 8 and Fig. 9.

INTRODUCTION OF EQUIVALENT UNIFORM BURNUP

The reactivity of the spent fuel is affected by the burnup and void profile, reactor operational history and cooling time as shown in previous sections even if the fuels have the same average burnup. If the cask reactivity can be evaluated adequately and conservatively by simple calculation, introduction procedure of burnup credit will be more generalized and simplified. There is a concept of equivalent uniform burnup. In this concept, an average burnup of the real spent fuel can be converted to an equivalent uniform burnup with the same reactivity. The BWR bundle average burnup of 45 GWd/t including reactivity biases discussed in previous sections was converted into the 30 GWd/t of equivalent uniform burnup profile with 40% historical void fraction in this sense. A concept of equivalent uniform burnup was proposed by Conde, et al. in 1995 to generalize reactivity biases of axial burnup profile. In this study, we extended this concept of equivalent uniform burnup to including other reactivity biases such as operational histories and composition calculation errors. Fig. 10 shows that equivalent uniform burnup curves for BWR and PWR have a good similarity considering different cask designs, different fuel compositions and so on. Therefore a generalized conservative curve could be recommended.

CONCLUSION

Chemical analyses of high burnup UO_2 and MOX spent fuel pins were carried out for evaluation of burnup codes such as ORIGEN-2 code and NULIF code. Calculational accuracy of nuclides' composition was presented for these codes in comparison with the chemical analyses.

The reactivity biases for introducing burnup credit were evaluated for spent fuel transport and storage casks. Axial and horizontal profiles of burnup, historical void fraction (BWR only), operational histories and above-mentioned calculational accuracy of nuclides' composition were considered in this study. The introduction of burnup credit has a large merit in criticality safety analysis of casks even if the reactivity biases are considered.

Additionally, the concept of equivalent uniform burnup generalizes and simplifies the reactivity bias evaluation in burnup credit.

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Table 1 Characteristics of Fuel Pin

Fuel Type	Pellet Max. Burnup(GWd/t)	Enrichment	Num. of Sample
High Burnup UO_2 Fuel (PWR 15×15)	~66	3.8 wt%U235	4
MOX Fuel (1) (PWR 14×14)	44.5	3.5 wt%Puf	1
MOX Fuel (2) (PWR 14×14)	45.7	3.5 wt%Puf	1

Table 2 Reactivity Bias

Effect	Reactivity Bias at Fuel Average Burnup of 45 GWd/t (% Δ k/k)	
	BWR	PWR
Burnup Credit	-27	-19
Axial Profile	+8	+4
— Burnup	(+3)	(+4)
— Historical Void Fraction	(+5)	—
Horizontal Profile (Burnup)	~0	~0
Burnup History	+2	+2
— Control Rod Insertion	(+2)	—
— BPR Insertion	—	(+1)
— Boron Concentration	—	(+1)
— Power Density	(~0)	(~0)
Cooling Time (15 years)	-4	-3

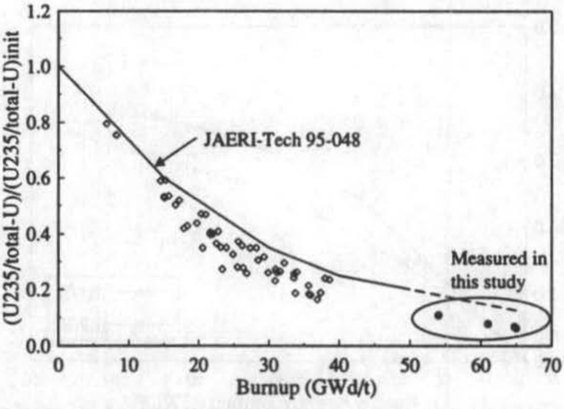

Fig.1 Measured Nuclide Composition (U235/total-U)

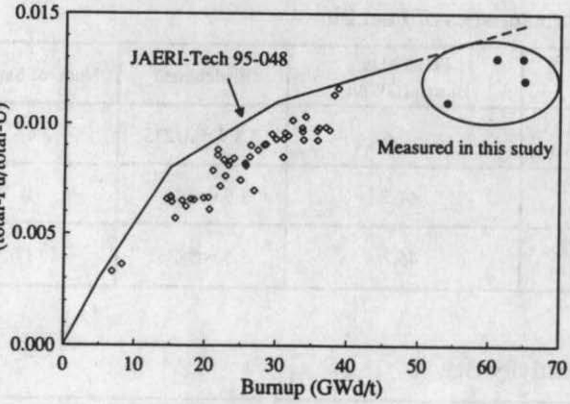

Fig.2 Measured Nuclide Composition (total-Pu/total-U)

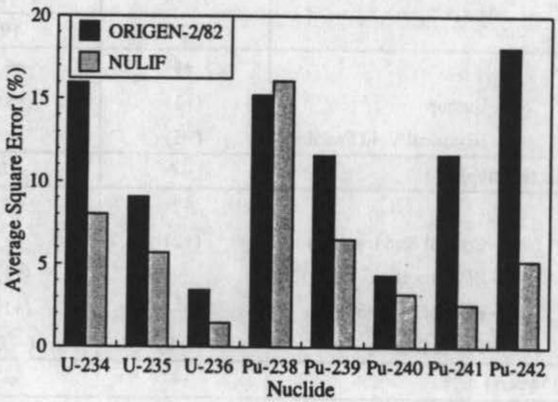

Fig.3 Accuracy of Nuclide Composition Calculation (High Burnup UO_2 Fuel)

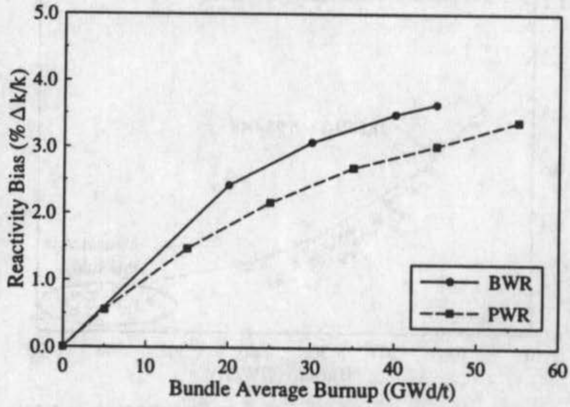

Fig.4 Reactivity Bias of ORIGEN-2/82 Code

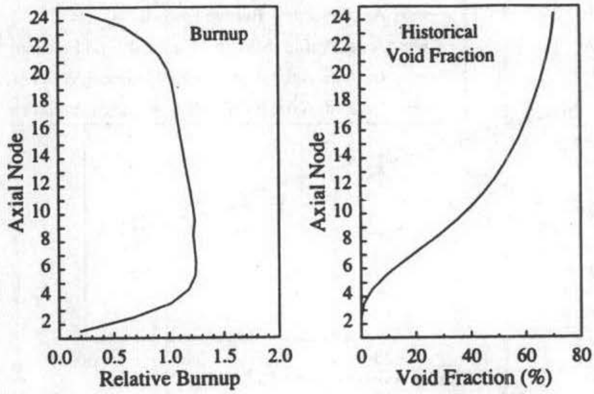

Fig.5 Axial Profile of Burnup and Historical Void Fraction (BWR)

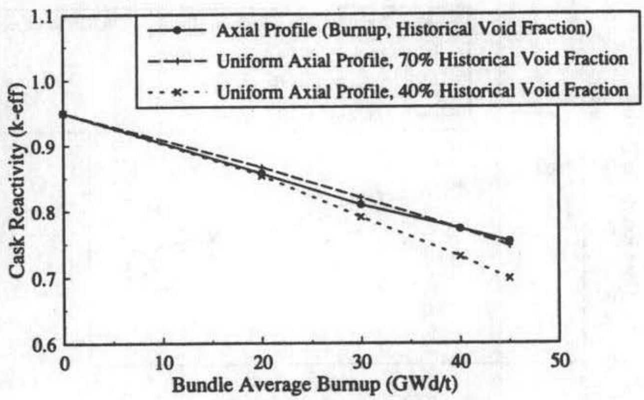

Fig.6 Effect of Axial Profile (BWR)

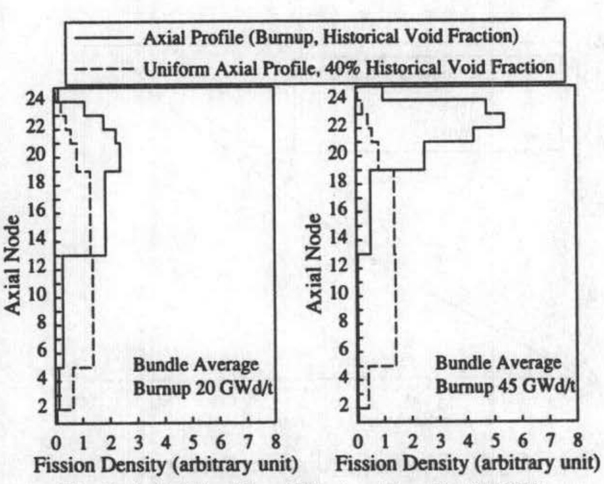

Fig.7 Axial Profile of Fission Density (BWR)

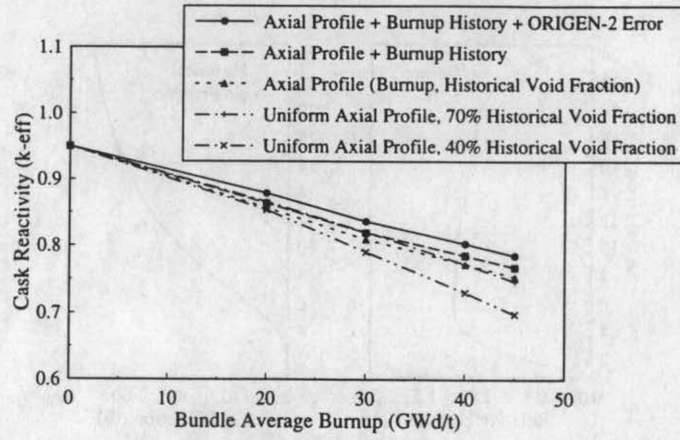

Fig.8 Cask Reactivity (BWR)

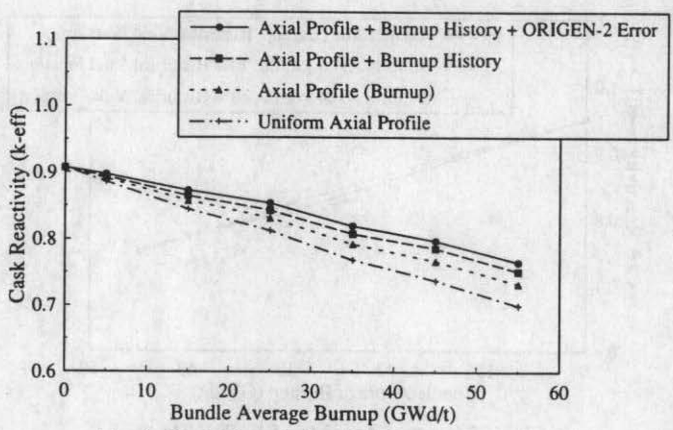

Fig.9 Cask Reactivity (PWR)

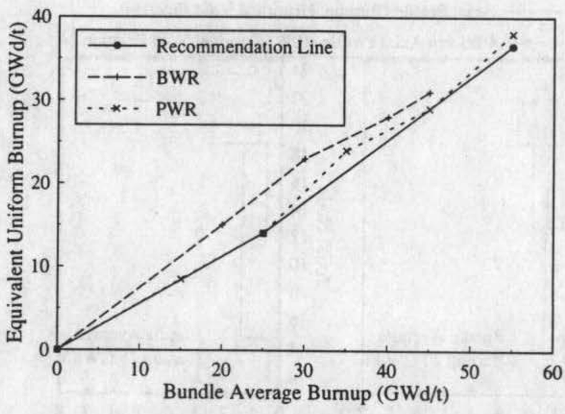

Fig.10 Equivalent Uniform Burnup