TAKING BURNUP CREDIT FOR INTERIM STORAGE AND TRANSPORTATION SYSTEM FOR BWR FUELS

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ABSTRACT

In order to establish a realistic burnup credit design system, a calculation system has been developed for determining isotope compositions, burnup, and criticality. The calculation system consists of several modules such as TGBLA, ORIGEN, CITATION, MCNP, and KENO. The TGBLA code is a fuel design code for LWR fuels developed in TOSHIBA Corporation. A compact measurement system for a fuel assembly has been being developed to meet requirements for the burnup determination, the neutron emission-rate evaluation, and the nuclear materials management. For a spent MOX fuel, a neutron emission rate measurement method has been being developed. The system consists of Cd-Te detectors and / or fission chambers. Some model calculations were carried out for the latest design BWR fuels. The effect of taking burnup credit for a transportation cask is shown.

INTRODUCTION

Burnup credit will assure criticality safety and increase accommodation capacity in the interim storage and transportation of spent nuclear fuels.

This paper proposes an advanced and practical burnup monitoring and evaluation system. The system features the followings:

1. Burnup-credited storage racks and transportation casks.
2. A compact burnup monitoring system with Cd-Te detectors and fission chambers.
3. A neutron emission-rate evaluation methodology, especially important for high burnup MOX fuels.
4. A safeguarding system of nuclear materials.

Current storage systems and transportation casks are designed on the fresh-fuel assumption that is too conservative. Taking burnup credit gives a realistic design while keeping reasonable conservatism [1].

Mimura et al suggests that burnup credit increases storage capacity and reduces the cost by 7 .30% [2].

In order to establish a realistic design system for taking burnup credit was developed. The system evaluates isotope compositions, burnup rate, and criticality, and consists of such modules as TGBLA [3], ORIGEN [4], CITATION [5], MCNP [6], and KENO [7]. TGBLA is an LWR fuel design code developed in Toshiba Corporation, ORIGEN is a one-point isotope-depletion code, CITATION is a three dimensional diffusion code, MCNP is a continuous energy Monte Carlo code, and KENO is a discrete energy Monte Carlo code.

A compact fuel-assembly measurement system has been being developed to meet the features 2 to 4 described above. The system is similar to the well-known FORK system [8]and consists of small-sized Cd-Te detectors and fission chambers.

Neutron measurement methods are utilized for burnup and neutron emission-rate (NER) measure-ments [9]. The NER values are especially important for spent MOX fuels because these are significantly higher than those of spent UO2 fuels [10].

This behavior may cause a serious problem in the neutron shielding design of a transportation cask.

Nuclear materials management is carried out through a burnup management.

The amount of nuclear materials is estimated based on the burnup values, initial enrichments, and/or operational histories. The management is especially important for spent MOX fuels.

CALCULATION SYSTEM

Figure 1 shows a flow of design for burnup credited storage system.

First of all, among actinides and fission products (FPs) with negative reactivity, nuclides considered for taking burnup credit are selected. For keeping conservatism, long-live and fixed nuclides are selected. The 12 actinides and 14 FPs shown in Table 1 are selected, considering DOE [11] and OECD/NEA studies [12].

Total fraction of the absorption by these nuclides is more than 99% of all actinides and more than 70% of all fission products in a spent BWR fuel.

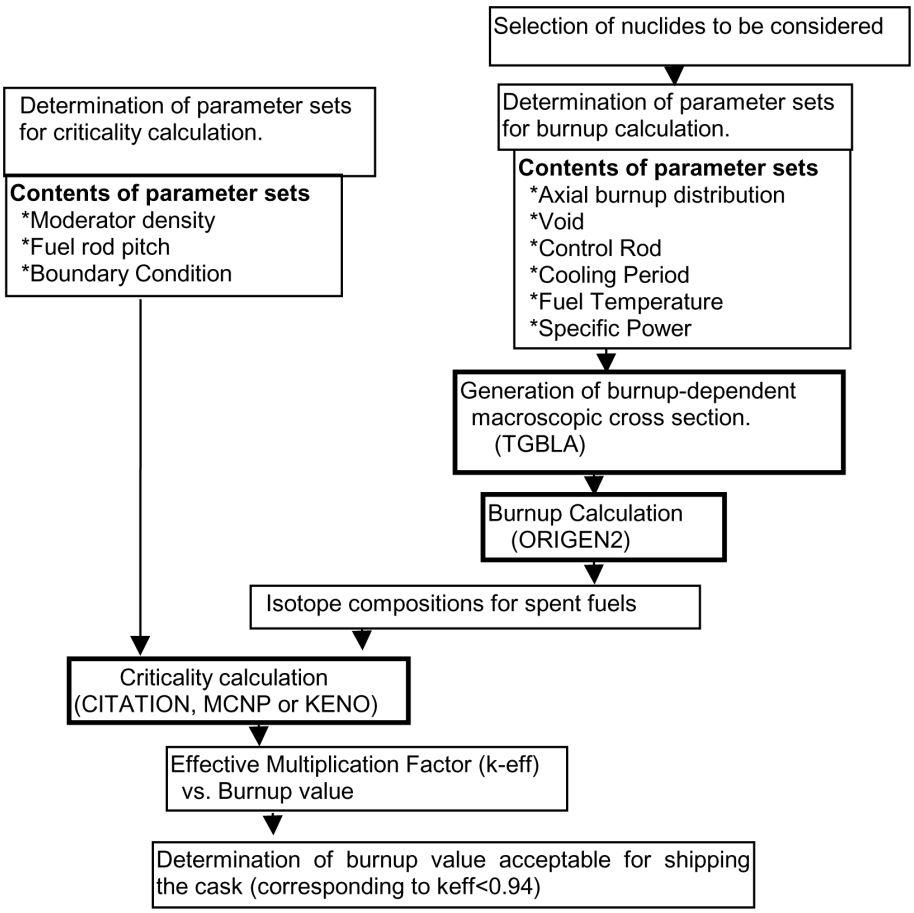


FIGURE 1. Flow of Design for Burnup Credited Storage System

TABLE 1. Nuclides Considered in the Criticality Safety Calculation

Actinide	U-234,235,236,238, Np-237, Pu-238,239,240,241,242, Am-241,243 (12 nuclides)
Fission Products	Mo-95, Tc-99, Ru-101, Rh-103, Cs-133, Nd-143,145, Sm-147,149,150,151,152, Eu-153,Gd-155 (14 nuclides)

Next, parameter sets for burnup calculations are determined for keeping conservatism.

Using the selected nuclides and the parameter set, burnup-dependent macroscopic cross sections for the ORIGEN2 code are generated with the TGBLA code. The code takes into account of operational history such as power density, void fraction, control blade insertions, and so on. This code is applied to the back-end issues for a more accurate design of a storage and a transportation system.

Burnup calculations with the ORIGEN2 code are carried out for the determination of isotope compositions in a spent fuel.

With using the above isotope compositions and parameter sets for criticality calculations, critical calculations are carried out.

As the result of these calculations, the correlation of multiplication factors (k-eff) and burnup values can be determined.

In this study, The upper limit of k-eff is assumed to be 0.94. Based on the above correlation, the lower limit of burnup value acceptable for shipping a spent fuel to a cask is deduced.

MEASUREMENT SYSTEM

Figure 2 is an example of a measurement system.

Cd-Te detectors and fission chambers are employed in the measurement system.

The formers determine the Cs134-to-Cs137 gamma-intensity ratios which are not so sensitive to change in the measurement position. This performance is advantageous for an under-water measurement in a fuel storage pool where precise positioning is not easy for a measurement system to the fuel assembly.

The superiority of a Cd-Te gamma detector is in the followings:

a. Energy resolution is acceptable in the burnup rate measurement, e.g., the Schottky type Cd-Te detector achieves the resolution of about 2% at Cs137 photo-peak of 662 keV.

b. A compact system can be designed. The Cd-Te detector has high gamma-ray detecting efficiency due to the large atomic number. This detector works at room temperature without a liquid nitrogen tank which a Hp-Ge detector requires.

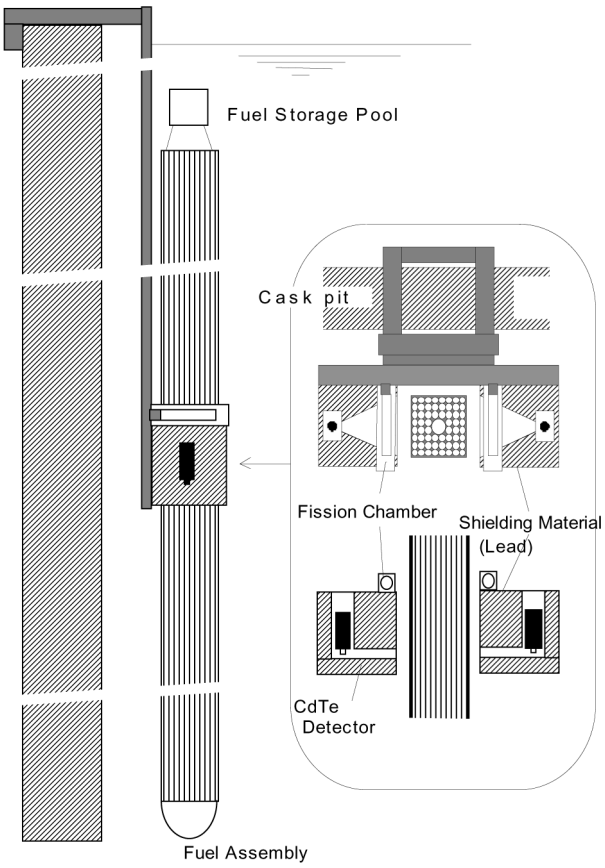


FIGURE 2. Example of a burnup measurement system

Neutron measurements are carried out with a couple of fission chambers placed on the both sides of the assembly, and the neutron counting data from both chambers are summed up. These arrangement and summing up procedure can reduce the measurement uncertainties resulted from detector positioning uncertainty.

A fission chamber does not require gamma-ray shielding, which enables the measurement system compact.

As can be understood from the above, the employment of Cd-Te detectors and fission chambers makes the measurement system compact.

A Neutron measurement method for UO2-fuel burnup determination has already been employed in the FORK system, the PHYTON system [13], the FAMOS system [14], and the ROKKASHO burnup monitoring system [15] in the Japanese reprocessing plant.

Since a correlation of neutron emission rate and burnup rate for the MOX fuel is quite different from that of UO2 fuel, the correlation for the MOX fuel has been developed.

Figures 3 and 4 are examples of experimental results of burnup-dependent neutron emission rates. UO2 and MOX fuel rods were removed from an island type MOX assembly exposed in a BWR core [10, and they were measured.

Neutron emission rates are approximately proportional to the 3rd to 4th power of burnup rate in UO2 fuels. On the other hand, neutron emission rates are approximately proportional to the square of burnup rate in MOX fuels.

Since neutron emission rates from a BWR exposed fuel are void-fraction dependent, the data are scattered axially in Figs.3. and 4.

Neutron measurement methods are also utilized for criticality-safety control and for safeguarding certification of the amount of nuclear materials.

Burnup rates, neutron emission rates, and the amount of nuclear materials are determined with the combination of measured values and calculated values.

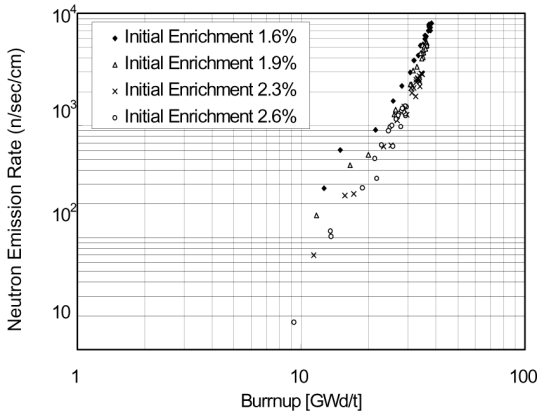


FIGURE 3. Burnup-dependent neutron emission rates for UO2 fuels

rates for UO2 fuels .

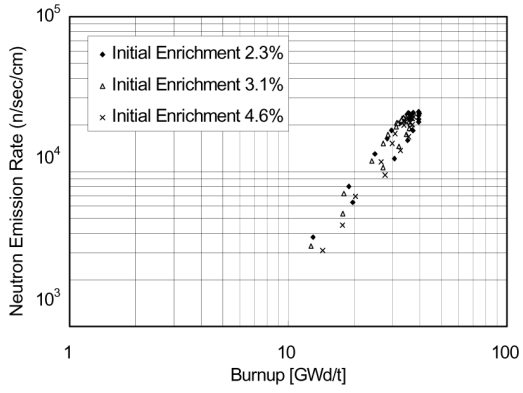


FIGURE 4. Burnup-dependent neutron emission rates for MOX fuels

DISCUSSION

Effect of taking burnup credit

Calculations were carried out for the BWR fuels of the latest design, named STEP-3. The assembly-averaged

initial enrichment was 3.5wt-% and assembly-averaged burnup was 45GWd/t. Criticality safety analysis was done for standard racks in a storage pool.

Figure 5 shows a model of a standard storage rack. In this case, the structural material is stainless steel, axial enrichment distribution and burnup distribution are not considered, void fraction is uniform by 70%. Only actinide nuclide are considered.

Criticality calculation was carried out with CITATION. The boundary condition was reflective, that implies an infinite system.

Figure 6 shows an example of calculation results. The figure shows the relation between acceptable burnup value and initial enrichments. A acceptable burnnup value is determined as the minimum burnup at which the effective multiplication factor is equal to the critical criterion (k-eff =0.94). As shown in Fig.6, in the case that a spent fuel has initial enrichment of 3.5wt-%, a more than 37Gwd/t fuel is acceptable for loading in a rack. It was found that burnup credit enabled a storage rack design without borated stainless steel.

Calculations for a transportation cask were also done. As a result, in a current transportation cask which accommodates 38 assemblies, it was found that taking burnup credit firstly made it possible to ship all assemblies to a cask.

For a simple case, a procedure taking credit for only one cycle burnup has been also developed. In this case, measurement system consists of a single ion chamber for a fuel storage pool in a power plant. Since gamma ray intensities are quite different between fresh and irradiated fuels, only total gamma-ray measurement with ion chamber is carried out instead of gamma-ray spectrum measurement. The one cycle burnup credit design gives us a simpler system although storage capacity decreases than that of multi-cycle burnup credit design. The design is considered to be the first step to introduce a burnup credit to a storage system.

Burnup Identification

Burnup value is identified following the flow diagram shown in Fig.7.

Measurements of burnup and neutron emission rates are carried out on the way of shipping to a transportation cask in a power plant. Measurement system will be installed on the wall of the fuel storage pool. If measured burnup rate is less than that of a given upper safety limit, such a fuel assembly is returned to the storage pool.

Nuclear materials management are periodically carried out in interim storage facilities by the measurement system. The system is hung from the fuel transportation machine as in the case of the FORK system. A fuel assembly is raised partially in the storage cell, neutron measurement is made, and then settled at the storage position. Nuclear material inventory is evaluated through the burnup value deduced from the measurement.

Burnup and Multiplication Factor

Determinations from Neutron Emission Rate

Measured thermal neutron flux is correlated to the neutron emission rate . and multiplication factor. by Eq.(1), where Ż is a constant determined for the measurement geometry [16].

$$\phi = \frac{\alpha \cdot S}{1-k} \tag{1}$$

ϕ: Constant determined for measurement geometry

k: Multiplication factor

S: Neutron emission rate

Figure 8 gives a simplified Cm244(18years of half life) and Cm242(163days) production chains obtained through numerical studies. After one to two years cooling, almost all the neutrons originate from Cm244.

Neutron emission rate observed is found to be approximated by the following

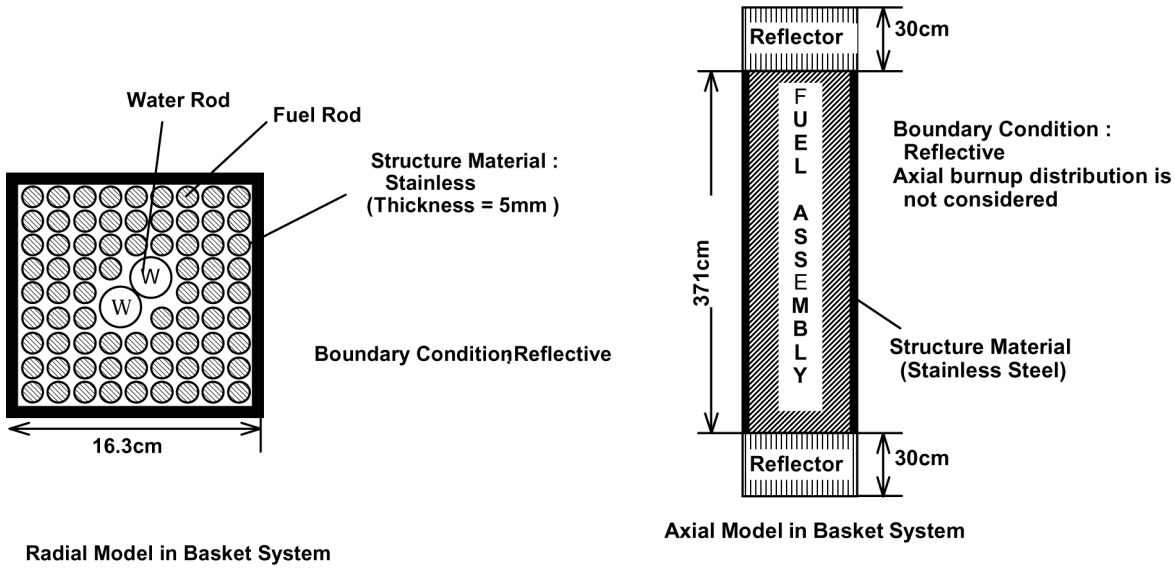


FIGURE 5. Calculation Model of standard rack system

$$S = S40 \cdot \left(1 + \frac{S2}{S40}\right) \cdot V \cdot T$$
$$S40 = S4 + S0 \qquad (2)$$
$$S4 = B \cdot (Bu)^{A}$$

S4: Neutron emission rate from Cm244
S2: Neutron emission rate from Cm242
S0: Neutron emission rate from the other nuclides
S: Total Neutron emission rate
V: Correction factor for void fraction variation
T: Correction factor for cooling time
A,B: Constants, which are function of initial enrichment for UO2, function of initial enrichment and Pu isotope composition for MOX
Bu: Burnup value

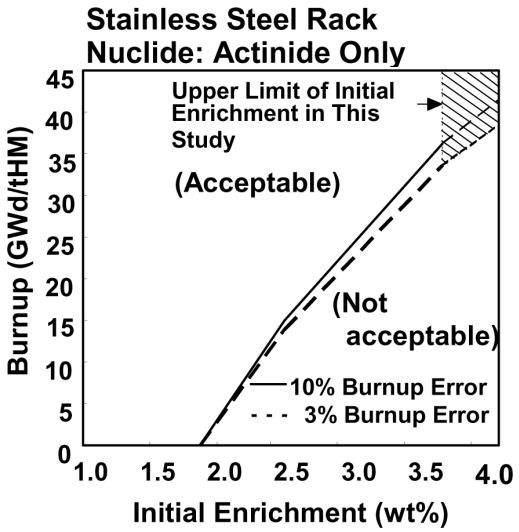


FIGURE 6. Relation between Initial Enrichments and Acceptable Burnup values

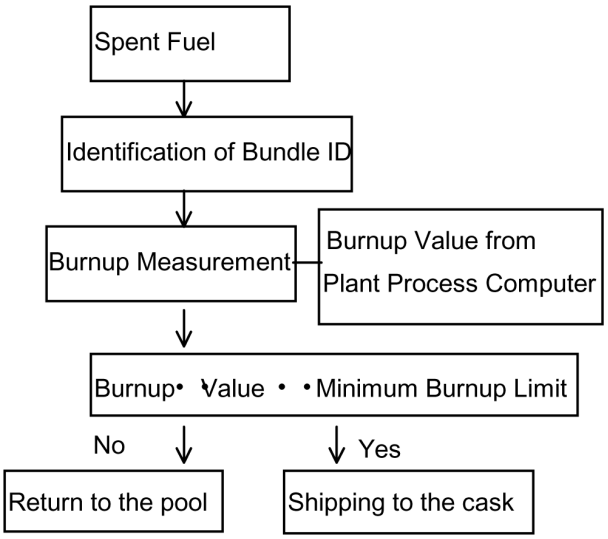


FIGURE 7. Flow of shipping spent fuels to a cask

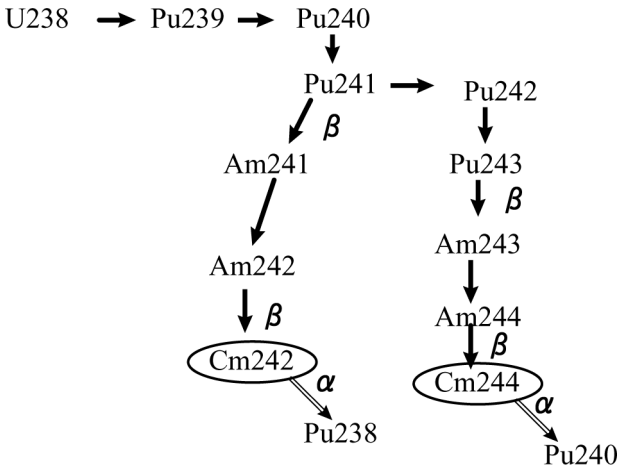
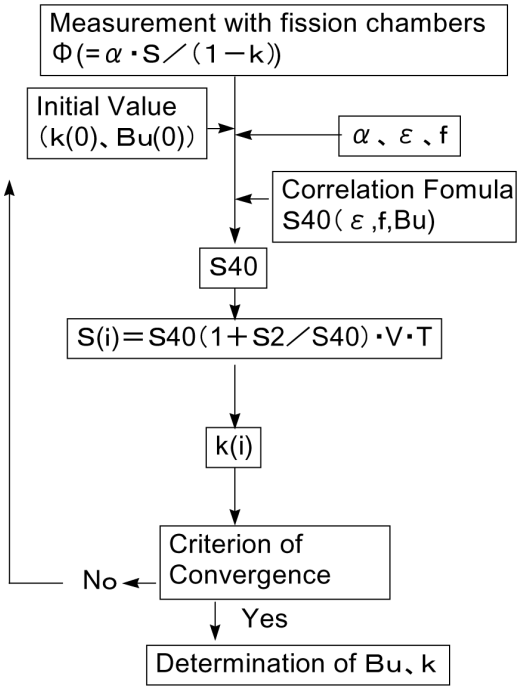


FIGURE 8. Cm244 and Cm242 production chains

Figure 9 is a procedure of determining burnup rate and multiplication factor. At first, Initial value are assumed. Ž is given by calculations or calibration experiments. After few iterations, Bu and k are determined.



: Initial enrichment
f : Pu isotope composition, f=Pu-fissile/Pu-total
For UO2 fuels, f value is not entered.

FIGURE 9. Procedure of burnup and multiplication factor determination

CONCLUSION

An advanced realistic system was proposed for monitoring and identifying burnup and neutron emission rates of spent LWR fuel assembly that takes burnup credit into account.

Calculation system has been developed for a reasonable burnup credit design system. Calculations were carried out for the BWR fuels of the latest design called STEP-3. It is found that burnup credit enables a storage rack design without borated stainless steel.

A burnup determination system using a compact monitor with Cd-Te detectors and fission chambers was proposed and a practical usage is also shown.

On the basis of neutron emission rate method, a method of burnup and multiplication factor determination is proposed.

The above-mentioned advanced storage and transportation system has many merits. For the validation of the calculation system and the measurement system, sufficient post irradiated experiments and benchmark calculations are needed with international-basis to meet material management requirements.

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