JAEA-Review
2017-010
DOI:10.11484/jaea-review-2017-010

JAEA-Review

A Guide to Introducing Burnup Credit, Preliminary Version (English Translation)

(Translated by) Hiroshi OKUNO, Kenya SUYAMA and Susumu RYUFUKU

Fuel Cycle Safety Research Division
Nuclear Safety Research Center
Sector of Nuclear Safety Research and Emergency Preparedness

June 2017

Japan Atomic Energy Agency

日本原子力研究開発機構

本レポートは国立研究開発法人日本原子力研究開発機構が不定期に発行する成果報告書です。
本レポートの入手並びに著作権利用に関するお問い合わせは、下記あてにお問い合わせ下さい。
なお、本レポートの全文は日本原子力研究開発機構ホームページ（http://www.jaea.go.jp）
より発信されています。

国立研究開発法人日本原子力研究開発機構 研究連携成果展開部 研究成果管理課
〒319-1195 茨城県那珂郡東海村大字白方 2 番地4
電話 029-282-6387, Fax 029-282-5920, E-mail:ird-support@jaea.go.jp

This report is issued irregularly by Japan Atomic Energy Agency.
Inquiries about availability and/or copyright of this report should be addressed to
Institutional Repository Section,
Intellectual Resources Management and R&D Collaboration Department,
Japan Atomic Energy Agency.
2-4 Shirakata, Tokai-mura, Naka-gun, Ibaraki-ken 319-1195 Japan
Tel +81-29-282-6387, Fax +81-29-282-5920, E-mail:ird-support@jaea.go.jp

JAEA-Review 2017-010

A Guide to Introducing Burnup Credit, Preliminary Version (English Translation)

(Translated by) Hiroshi OKUNO+, Kenya SUYAMA and Susumu RYUFUKU*

Fuel Cycle Safety Research Division
Nuclear Safety Research Center
Sector of Nuclear Safety Research and Emergency Preparedness
Japan Atomic Energy Agency
Tokai-mura, Naka-gun, Ibaraki-ken

(Received April 18, 2017)

There is an ongoing discussion on the application of burnup credit to the criticality safety controls of facilities that treat spent fuels. With regard to such application of burnup credit in Japan, this document summarizes the current technical status of the prediction of the isotopic composition and criticality of spent fuels, as well as safety evaluation concerns and the current status of legal affairs. This report is an English translation of *A Guide to Introducing Burnup Credit, Preliminary Version*, originally published in Japanese as JAERI-Tech 2001-055 by the Nuclear Fuel Cycle Facility Safety Research Committee.

Keywords: Burnup Credit, Criticality Safety, Spent Fuel, Fuel Cycle Facility

This work was carried out by the former Japan Atomic Energy Research Institute (JAERI) under entrustment by the former Science and Technology Agency (STA) of Japan.

+ Nuclear Emergency Assistance and Training Center

* Visible Information Center, Inc.

燃焼度クレジット導入ガイド原案
（英訳版）

日本原子力研究開発機構 安全研究・防災支援部門
安全研究センター 燃料サイクル安全研究ディビジョン
（訳）奥野　浩[+]・須山　賢也・龍福　進[*]

(2017年4月18日受理)

使用済燃料を取扱う施設の臨界安全管理に対して、燃焼度クレジットを導入することが検討されている。本資料は、今後国内の使用済燃料を取扱う施設において燃焼度クレジットを採用することを目的として、使用済燃料の同位体組成と臨界性の予測に関する技術的現状、安全評価上考慮すべき点、そして規制に関する現状をまとめたものである。この報告書は、燃料サイクル安全研究委員会がJAERI-Tech 2001-055として日本語で刊行した「燃焼度クレジット導入ガイド原案」の英訳である。

本報告書は、旧日本原子力研究所が電源開発促進対策特別会計法に基づく旧科学技術庁からの受託として行った研究成果である。

原子力科学研究所：〒319-1195　茨城県那珂郡東海村大字白方2-4
+　原子力緊急時支援・研修センター
*　株式会社 ヴィジブルインフォメーションセンター

Contents

目次

1. Introduction

Recently, the transportation and storage of high burnup spent fuels (SFs) have become a real problem in Japan because of the life extension of nuclear power plants with light water reactors (LWRs), i.e., pressurized water reactors (PWRs) and boiling water reactors (BWRs), and the longer life of their fuels. In addition, a massive accumulation of SFs is predicted; thus, the construction of an interim SF storage facility is planned to solve this problem. Furthermore, a desire to improve economic efficiency means streamlining the receiving process and storage of SFs, as well as the operation of dissolvers at the Rokkasho reprocessing facility. Studies are now in progress on the economic efficiency, safety, and various other aspects of these systems (e.g., transportation packages, storage containers, dissolvers) and facilities (i.e., onsite SF storage pools, interim SF storage facilities) used for storage, transportation, and reprocessing.

So far, an assumption of fresh fuel (i.e., the fuel that is assumed to be not burnt) has been used in the nuclear criticality safety evaluation of equipment and facilities for SF transportation/storage. Because of this assumption, excessive safety margins are involved in SF transportation/storage systems. If the actual fuel burnup that reflects the recent advancement in analytical methods for criticality safety evaluations could be taken into consideration, reasonable safety margins would be applicable in the design of these systems. Thus, adopting burnup credit to criticality safety design leads to three distinct economic benefits, compared with the conventional design, on the basis of the fresh fuel assumption: (1) it addresses the accumulation of SFs associated with the life extension of nuclear power plants with LWRs (regardless of onsite or offsite storage), such that the storage capacity of existing SF storage systems can be increased by decreasing the gap between fuel assemblies; (2) for a new SF storage facility, excessive neutron absorbers in the structural material become unnecessary; and (3) the capacity of SF transportation packages can be increased by introducing a new design, even without using expensive materials, and consequently, the number of transportation shipments can be reduced. In addition to the economic benefits, reducing the number of shipments enhances safety by reducing the exposure risk to workers and the public, on the basis of the As Low as Reasonably Achievable (ALARA) principle. Thus, adopting the latest safety evaluation methods that reflect recent advances in science and technology to the design of transportation/storage systems makes it possible to not only pursue economic efficiency but also enhance safety.

Because of the limited potential resources and available land in Japan, it is essential to adopt burnup credit in the design of SF transportation/storage systems.

Unlike in a fresh fuel evaluation, to introduce burnup credit into a nuclear criticality safety evaluation, an accurate isotopic composition of the fuel must be obtained via a burnup analysis, and new parameters resulted from the burnup analysis and their effects on the reactivity should be considered. In particular, the criticality of the fuel systems should be evaluated using methods that consider factors such as the fuel burnup, their axial and radial distributions, as well as their operating history. Furthermore, the operational controls should properly maintain the conditions that were assumed when evaluating the design of the system/facility.

For a transportation/storage system/facility to adopt burnup credit in a criticality safety evaluation, it must be assured that the actual burnup is in accordance with the loading criteria before the transportation/storage of SFs (see note*). Burnup credit accounting only for uranium and plutonium was adopted for the first time in SF receiving and storage pools at the Rokkasho reprocessing facility in Japan, and there is a desire to improve the method in order to verify the actual burnup (currently, a method using a burnup monitor is adopted for this purpose).

This guide is a systematic collection of concepts, methods, and data necessary for the application of burnup credit to the design and operational control of systems/facilities used for the storage, transportation, and reprocessing of SFs. However, fission products (FPs) are excluded from this current introduction to burnup credit because of the inadequacy of the isotopic composition data necessary for the validation of a burnup calculation code. In chapter 2, basic methods for criticality safety control and procedures for criticality safety evaluation when adopting burnup credit are described. Details of the methods, code validation, and factors to be considered in isotopic composition evaluation and criticality evaluation are described in chapters 3 and 4, respectively, both of which are important. In chapter 5, operational issues caused by the introduction of burnup credit are described, with a focus on the verification of burnup.

Basic methods for criticality safety are cited from the *Nuclear Criticality Safety Handbook* for nuclear fuel facilities in general, and some data on SF transportation/storage and reprocessing are also extracted from the *Handbook*. The isotopic composition data in the *Handbook* are reverified. Furthermore, the latest findings presented at international conferences and in working group meetings, as well as information cited from published documents on burnup credit, are included. Investigations of the contents were performed by the Special Committee on Nuclear Criticality Safety (Chairperson: Kojiro Nishina, a Professor at Aichi Shukutoku University) and the affiliated Working Group on Nuclear Criticality Safety Experiment Data (Leader: Yoshihiro Yamane, Professor of the Graduate School of Engineering, Nagoya University). Members of the Working Group who reviewed the contents of this guide (which was originally written in Japanese) are shown in **Table 1.1.**

Note that this guide was published in Japanese in 2001 as a JAERI report, as a supplement to the *Nuclear Criticality Safety Handbook, Version 2* (published in 1999). Similar to the *Nuclear Criticality Safety Handbook*, this *guide* should be revised in the future to reflect the advance of technology and available data.

Note*The verification method using a subcriticality monitor (now under development) is beyond the scope of this guide.

Table 1.1 Members of the Working Group on Nuclear Criticality Safety Data (FY 1998–2000)

Name	Affiliation	Remarks
Anekawa, Shouji	Tokyo Electric Power Company, Incorporated	Since FY 1999
Itoh, Daiichiro	Mitsui Engineering & Shipbuilding Co., Ltd	
Onishi, Kazunari	Nuclear Fuel Transport Company, Ltd.	FY 2000
Okuno, Hiroshi	Japan Atomic Energy Research Institute	
Kaneko, Toshiyuki	The Japan Research Institute, Ltd.	
Kitagawa, Jun	Nuclear Fuel Transport Company, Ltd.	FY 1999
Kurosawa, Masayoshi	Japan Atomic Energy Research Institute	Until FY 1999
Sakurai, Shungo	Nuclear Power Engineering Corporation	Since Nov. 1999
Suzuki, Kenichi	Mitsubishi Materials Corporation	Until FY 1998
Suda, Kenji	Japan Nuclear Fuel Limited	Until FY 1999
Suyama, Kenya	Japan Atomic Energy Research Institute	
Nakajima, Ken	Japan Atomic Energy Research Institute	
Natsume, Tomohiro	Mitsubishi Heavy Industries, Ltd.	
Nojiri, Ichiro	Japan Nuclear Cycle Development Institute	
Nomura, Yasushi	Japan Atomic Energy Research Institute	
Hazama, Taira	Japan Nuclear Cycle Development Institute	
Hamada, Tadashi	Japan Nuclear Fuel Limited	FY 2000
Hiyama, Nobuyuki	Nuclear Fuel Industries, Ltd.	
Makiguchi, Hirofumi	Japan Nuclear Fuel Company, Ltd.	
Masukawa, Fumihiro	Nuclear Power Engineering Corporation	Until Sep. 1999
Matsumura, Tetsuo	Central Research Institute of Electric Power Industry	Since FY 1999
Misawa, Tsuyoshi	Kyoto University	
Mitsuhashi, Ishi	Toshiba Corporation	
Miyoshi, Yoshinori	Japan Atomic Energy Research Institute	
Morioka, Nobuo	Mitsubishi Materials Corporation	Since FY 1999
Yamaguchi, Masao	Hitachi Engineering Co., Ltd.	
Yamate, Kouichi	Kansai Electric Power Company, Incorporated	Since FY 1999
Yamane, Yoshihiro	Nagoya University	
Yamamoto, Toshihiro	Japan Atomic Energy Research Institute	FY 2000

(The names are arranged in order of the Japanese syllabary.)

2. Criticality Safety Controls Adopting Burnup Credit

2.1 Basic Method

Following is a quote from the *Nuclear Criticality Safety Handbook* [1]:

> "The basic principle of criticality safety control is to prevent criticality for all situations that are technically conceivable. To achieve criticality safety, sufficient margins are considered in each stage of the design, fabrication, construction, and operation of applicable facilities. In the design stage, criticality safety is demonstrated to be achievable, in the fabrication and construction stages, criticality safety is assured by verifying that the design conditions are satisfied, and in the operating stage, all operations are controlled so that criticality safety can be maintained according to the design. Nuclear safety for the conditions of the specified shape and dimensions of a single unit, in normal and abnormal states, is verified by scrutinizing the kinds, quantities, and the physical and chemical forms of the nuclear fuels and other materials contained in the unit. To verify the criticality safety of multiple units, the neutron interaction effects in both normal and abnormal states are evaluated, taking the distance between units or the thickness of the shielding material, or both, into consideration.
>
> When verifying subcriticality by calculation, the evaluation must provide sufficient safety margins by considering the reliability of the data and the calculation method. The reliability of the data and the calculation method used should be fully verified by comparison with experimental data from a system physically similar to the object system. When experimental data can be obtained for a system that is the same as the object system, the criticality safety may be evaluated based on this data."

The nuclear criticality safety evaluations adopting burnup credit for SF transportation/storage systems and reprocessing facilities covered by this guide adhere to the above-mentioned principles.

However, there are two different points in the design and operational control:

1. When adopting burnup credit, determination of the isotopic composition of each SF is required. Most commonly, the isotopic composition is determined using a burnup calculation based on the irradiation history. Thus, the validation of burnup calculation codes is required. In addition, the validation of the criticality calculation code for the SF is also required. However, for the burnup credit accounting only for the actinides, the validation of a criticality calculation for MOX fuels might substitute for the SFs.

2. For operation control with the introduction of burnup credit, verification of the burnup may be required depending on the method used to determine the isotopic composition (see note*).

2.2 Terminology

The terms used in this guide [1] include the following:

1. Criticality safety controls (in the design, construction, and operational stages)

Criticality safety controls shall be implemented at each stage of the design, construction, and operation of any relevant facility; the relationship between the criticality safety controls in these stages is shown in **Fig. 2.1**. The basic principle of criticality safety is to prevent criticality for all technically conceivable situations. In the design stage, criticality safety is demonstrated to be achievable. In the fabrication and construction stages, criticality safety is assured by verifying that the design conditions are satisfied. In the operating stage, all operations are controlled such that criticality safety can be maintained according to the design. In this guide on burnup credit, criticality safety evaluation is subcategorized into two evaluations: isotopic composition evaluation and criticality evaluation.

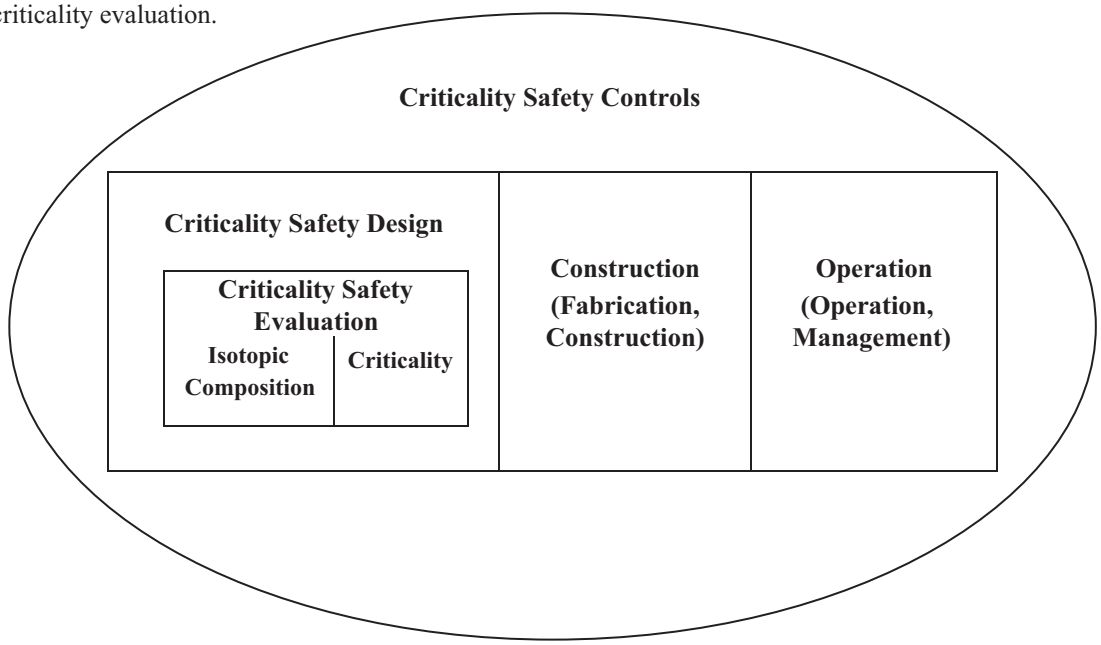

Fig. 2.1 Criticality safety controls

2. Burnup credit

Burnup credit accounts for the decrease in the reactivity associated with fuel burnup. In some cases, "burnup credit" is specifically intended to account for the decrease in reactivity associated with fuel burnup in the criticality evaluation.

---

note* Although there are methods for measuring subcriticality, these methods are still under development; therefore, this guide deals with methods for measuring fuel burnup.

The Fresh fuel contains more fissile material than burned fuel, and does not include FPs. Therefore, a conservative burnup is derived by ignoring the presence of burnable poisons, e.g., gadolinium, and assuming the fuel to be fresh instead of spent. However, when the burnup is estimated properly and the nuclide composition of the SF is obtained adequately from the viewpoint of criticality safety, then the criticality safety evaluation will be reasonably accurate.

3. Fuel burnup

Fuel burnup means the amount of energy released from nuclear fuel per unit mass, or the fraction of numbers of atomic nuclei reacted with neutrons. In general, this value is represented by the unit MWd/t (or GWd/t), which equates to the constant output of 1 MW (or 1 GW) per metric ton of the initial heavy elements (actinides such as uranium and plutonium) throughout a day. In the case of UO_2 fuel, because the initial heavy element is limited to uranium, this value is often represented as MWd/tU instead of MWd/t. Note that this value is sometimes represented as MWd/tIHM (ton Initial Heavy Metal) or MWd/MTU (Metric Tons of initial Uranium). Fission per Initial Metal Atom (FIMA) is a similar unit, i.e., 1% FIMA is the energy released by fission from 1% of the initial heavy element, and it is nearly equivalent to 9600 MWd/t.

4. Isotopic composition

The isotopic composition is the ratio of elements or isotopes in the nuclear fuel and is represented by the number of atoms or mass in each isotope for a unit volume or unit mass.

5. Actinide

Actinide (Actinides) is a generic term used to refer to the elements from thorium (Th: atomic number 90) to lawrencium (Lr: atomic number 103), and includes Th, Pa, U, Np, Pu, Am, Cm, Bk, Cf, Es, Fm, Md, No, and Lr. Actinoid refers to both actinides and actinium (Ac: atomic number 89).

6. Fission product

Fission product (FP) is a generic term for nuclides generated by nuclear fission or following radioactive decay or neutron absorbing reactions.

7. Irradiation history

The irradiation history provides information regarding the changes in the size and pattern of operational parameters that the SF has experienced; that is, the specific power/operating history, fuel temperature, coolant temperature distribution, void fraction distribution in the coolant, cooling time, boron concentration, Gd concentration distribution, the presence of burnable neutron absorbers, and the fraction of control rod insertion. These parameters are categorized as irradiation conditions, and affect the ratio of the generation/decay of the isotopic composition.

8. Operating history

Operating history indicates the changes in the amplitude and pattern of the power output during irradiation of a fuel during reactor operation and control.

9. Gadolinium credit (Gd credit)

In instances where the fuel contains gadolinium, because of the decrease in the atomic number density during fuel irradiation, the reactivity worth increases with fuel burnup and reaches a peak value during the initial stage of fuel burnup. In the safety analysis of SF transportation/storage systems, accounting for this peak value translates to a gadolinium credit.

10. Single failure (see note†)

Single failure indicates the condition where a piece of equipment loses its specified functions because of a single phenomenon. In addition, it includes multiple failures that could occur because of a single phenomenon.

11. Single failure, etc.

Single failure, etc., indicates a single failure or malfunction of a system or piece of equipment, or a single mistake in operation, where the frequency of its occurrence is estimated to be one or more times during the service life of the concerned facility.

12. Changes in peripheral conditions caused by external factors

Changes in peripheral conditions caused by external factors indicate changes in peripheral conditions, including reflection, absorption, moderation, and mutual interaction of neutrons, that are attributable to external factors such as earthquakes, floods, or fires. All external factors whose occurrence cannot be neglected shall be included.

13. All cases conceivable from a technological standpoint

All cases conceivable from a technological standpoint refer to all conceivable single failure, etc., or changes in peripheral conditions caused by external factors. The severest conditions for criticality safety for the assumed events shall apply.

14. Criticality factor

Criticality factors include all factors whose change influences the criticality of the system. Such factors include the shape and dimensions of the equipment containing the nuclear fuel material, the concentration of the nuclear fuel material in solution, the mass and isotopic composition of the nuclear fuel material, and the physical and chemical properties of the nuclear fuel material. In addition, the shape and dimensions, concentration, and material quality of the neutron absorbers and the neutron reflection conditions are also included.

---

note† In the Nuclear Criticality Safety Handbook (Versions 1 and 2), single failure was defined as a "Single failure shall mean the condition where one system or piece of equipment loses its specified functions because of a single phenomenon. Single failure also includes multiple failures that could occur because of a single phenomenon," while the Regulatory Guide for the Licensing Safety Review of Reprocessing Facilities does not include the word "system" in its definition. This guide concurs with the latter, and "system" is omitted in the definition of single failure.

15. Criticality control factor

The criticality control factor applies to the criticality factor used to set the criticality limits for design or operational control in order to secure the criticality safety of nuclear fuel facilities.

16. Criticality limits

A criticality limit is set for each criticality control factor to secure the subcriticality of the system.

17. Estimated criticality value and estimated criticality multiplication factor (k_c in Fig. 2.4)

The estimated criticality value is the value of the criticality control factor where the system is judged to be critical and the estimated criticality multiplication factor is a corresponding neutron multiplication factor. This multiplication factor is estimated for the criticality state of the subject system on the basis of the benchmark calculation results using experimental data measured in a system physically similar to the subject system.

18. Estimated criticality lower limit value and estimated criticality lower limit multiplication factor (k_L in Fig. 2.4)

The estimated lower limit criticality value is the value where a system is judged as subcritical if each of its criticality control factors is less than this value. This value shall be determined considering the estimated criticality value and its calculational error where the system is believed to be critical. When a criticality value is to be calculated, the calculation accuracy of the calculation code or codes used must be evaluated in advance. This accuracy is to be determined according to the benchmark calculation using the criticality test data of systems similar to the object system. The upper limit of the neutron multiplication factor where the system may be judged as subcritical is obtained by this accuracy evaluation and is called the estimated criticality lower limit multiplication factor. The criticality control factor value corresponding to the estimated criticality lower limit multiplication factor is the estimated lower limit criticality value.

19. Single unit

A single unit is defined as a unit for handling nuclear fuel material for which a set of criticality limits can be determined and criticality control can be performed within the limits. Single units for which criticality limits can be determined shall be those units for which peripheral conditions can be imposed to produce the severest conditions for criticality control and for which the kinds, compositions, physical and chemical properties, and quantities of the contained nuclear fuel and other materials can be specified.

20. Multiple unit

A system consisting of two or more single units, reflectors, shielding materials, etc., is defined as a multiple unit. A multiple unit must be a system in which each of the single units contained therein, and the arrangement of each of the single units, can be specified. To ensure criticality safety, the configuration of the single units and that of the neutron shielding materials must be determined such that no criticality can occur in any situation conceivable from a technical standpoint by considering the mutual interaction of the neutrons among the single units.

21. Maximum permissible multiplication factor (k_a in Fig. 2.4)

The maximum permissible multiplication factor is used to judge if a system is subcritical by calculating its multiplication factor. This value shall be set below the estimated criticality lower limit multiplication factor.

22. Maximum permissible limit

In the criticality safety evaluation, a criticality control factor corresponding to the maximum permissible multiplication factor is called the maximum permissible limit.

23. Reactivity, reactivity worth

Reactivity is an indicator of how far a system is from criticality. The reactivity ρ is given by the following equation, where k is the neutron multiplication factor of the subject system:

$$\rho = \frac{k-1}{k}. \tag{2.1}$$

When the state of a system changes, that is, when a neutron absorber is added or the amount of a moderator decreases, the neutron multiplication factor changes accordingly. The difference in the reactivity before and after the state change is called the reactivity worth of the material for the state change. The reactivity worth associated with the state change is given by the equation below, where k_1 and k_2 are the neutron multiplication factors of the system before and after the change, respectively:

$$\rho_2 - \rho_1 = \frac{k_2 - 1}{k_2} - \frac{k_1 - 1}{k_1} = \frac{1}{k_1} - \frac{1}{k_2}. \tag{2.2}$$

2.3 Levels of Adoption of Burnup Credit

The isotopic composition change of nuclear fuel during burnup in a reactor decreases the reactivity of the fuel. The reactivity change is affected by the net decrease of fissile nuclides, the creation and accumulation of nonfissile actinide nuclides, the build-up of FPs, and a decrease in the burnable neutron absorbing material. To ensure the reliability of safety margins in a burnup credit application, the nuclear properties and quantities of fissile nuclides and neutron absorbing nuclides must be estimated with sufficient accuracy. Considering the accuracy limits of the above-mentioned properties and amounts, as well as the consistency between the current criticality safety control methodology and the concept of burnup credits, the following evaluation levels are considered to be appropriate for the adoption of a burnup credit.

Level l Accounting for actinides only: Only changes in the actinide nuclides are considered. There are fissile and non-fissile actinides. The actinides to be considered depend upon the accuracy of the burnup calculation and the safety evaluation method.

Level 2 Accounting for both actinides and FPs: The effect of changes in the actinide nuclides and the neutron absorption by the FPs are considered. The adopted FP nuclides are determined according to the required accuracy of the burnup calculation and the adopted safety evaluation method.

The level of burnup credit in the criticality safety evaluation of an SF transportation/storage system and

reprocessing facility should be determined considering the effect of the accuracy of the isotopic composition evaluation (burnup calculation) and/or the criticality evaluation on the safety margins.

This guide is mainly intended to introduce level 1 burnup credit because the available isotopic composition data for FPs are limited and no critical experiments including FPs have been performed to date. It also puts level 2 in perspective. Nevertheless, the basic principles of a criticality safety evaluation (evaluation of the isotopic composition and criticality) are the same for both levels.

Besides these two levels, there is an additional instance that involves application of a gadolinium credit. If gadolinium is initially contained in the nuclear fuel as a burnable neutron absorbing material, this credit takes into account the reactivity peak in the nuclear criticality safety evaluation.

2.4 Example Calculation of the Capacity for Handling Spent Fuel in a Storage Pool

To demonstrate that the economics for SF transportation/storage are improved by introducing burnup credit, criticality calculations were performed. The subject system was an infinite array of SF assemblies in an SF storage pool neglecting the rack material, and the gaps between the assemblies at which the infinite neutron multiplication factor of the system corresponded to 0.95 and 0.98 were calculated, and then an increase in the storage capacity due to the application of the burnup credit was confirmed [2]. For the calculating subjects, 17 × 17 PWR fuel and 9 × 9 BWR fuel were selected, and initial ^{235}U enrichments of 4.7 wt% and 4.0 wt% were assumed in light of the development trend toward a higher burnup. On the basis of the initial compositions, the ORIGEN2.1 [3] code was used to calculate the isotopic compositions of the SFs. Burnups of 5, 15, 20, 30, 40, and 50 GWd/t and cooling times of 0, 5, 10, 20, 30, and 50 years were selected as parameter values. In addition, for the ORIGEN2.1 calculation, the attached PWR-UE and BWR-UE libraries, as well as new libraries for ORIGEN2, i.e., PWR47J32 and BS340J32 based on JENDL-3.2 [5], were used. Then, on the basis of the aforementioned calculated isotopic compositions, criticality calculations were performed using the continuous-energy Monte Carlo code MVP [6] with different gaps between the fuel assemblies. Finally, the gaps for both infinite neutron multiplication factors of 0.95 and 0.98 were derived by linear interpolation.

The results were compiled into a ratio of an allowable storage number of SF assemblies per unit area based on the burnup credit to that based on the fresh fuel. In the case of the PWR fuel under the conditions with a burnup of 50 GWd/t, a cooling time 10 years, and accounting for the FPs, a ratio of approximately 2 was obtained, as seen in **Fig. 2.2**. On the other hand, in the case of the BWR fuel under the same conditions as in the case of PWR fuel aforementioned, the infinite neutron multiplication factor was derived as less than 0.95 for any value of the gap between the fuel assemblies. In other words, for BWR fuel, the gap between the fuel assemblies is no longer determined according to the criticality safety aspect, but by the need to ensure thermal or structural safety. These results are not strictly correct because the analysis does not account for the axial burnup distribution; however, they indicate that the application of burnup credit to an SF transportation/storage system is an effective way to increase the storage capacity.

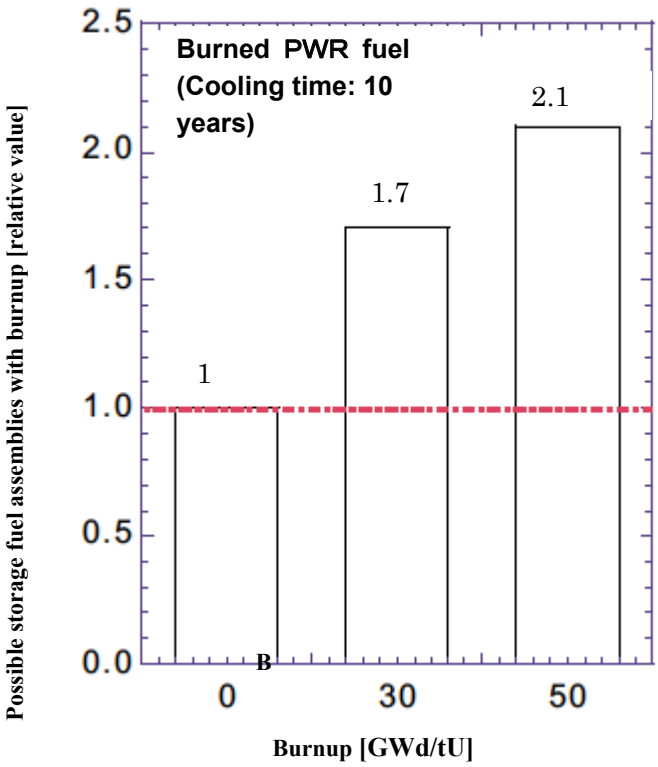

Fig. 2.2 Variation in the possible storage amount of fuel assemblies with burnup
(PWR 17 × 17, initial enrichment 4.7 wt%, infinite array (k_{inf} = 0.95), ORIGEN2.1 burnup calculation, MVP criticality calculation)

2.5 Procedure for Criticality Safety Controls Adopting Burnup Credit

2.5.1 Outline of Nuclear Criticality Safety Evaluation

Criticality safety evaluations adopting burnup credit are divided into two steps. One is the evaluation of the isotopic composition of the SF based on a given burnup, and the other is the criticality evaluation of a single unit or multi-unit system of the SF. The criticality safety of a subject system is determined by comparing the calculated neutron multiplication factor with the reference maximum permissible multiplication factor. In this section, procedures for the isotopic composition evaluation and the criticality safety evaluation described in the following chapters are outlined. The procedure for the criticality safety evaluation is shown in **Fig. 2.3**. The equivalent initial enrichment and equivalent uniform burnup are used as simplified methods in the figure; therefore, it is necessary to confirm in advance if these methods are applicable to the subject system.

When applying burnup credit to a criticality safety evaluation, the level of the burnup credit should be determined first. For this reason, the merit of applying each level of burnup credit to the design should be considered. In the selection of codes and libraries associated with the isotopic composition calculation (burnup calculation) and criticality calculation, the current situation of their calculation accuracies and/or the reliability of the correction methods must be considered. For the criticality safety evaluation, the burnup calculation code and the criticality calculation code should only be used when the reliability has been validated through comparison with sufficient experimental data for systems similar to the subject system.

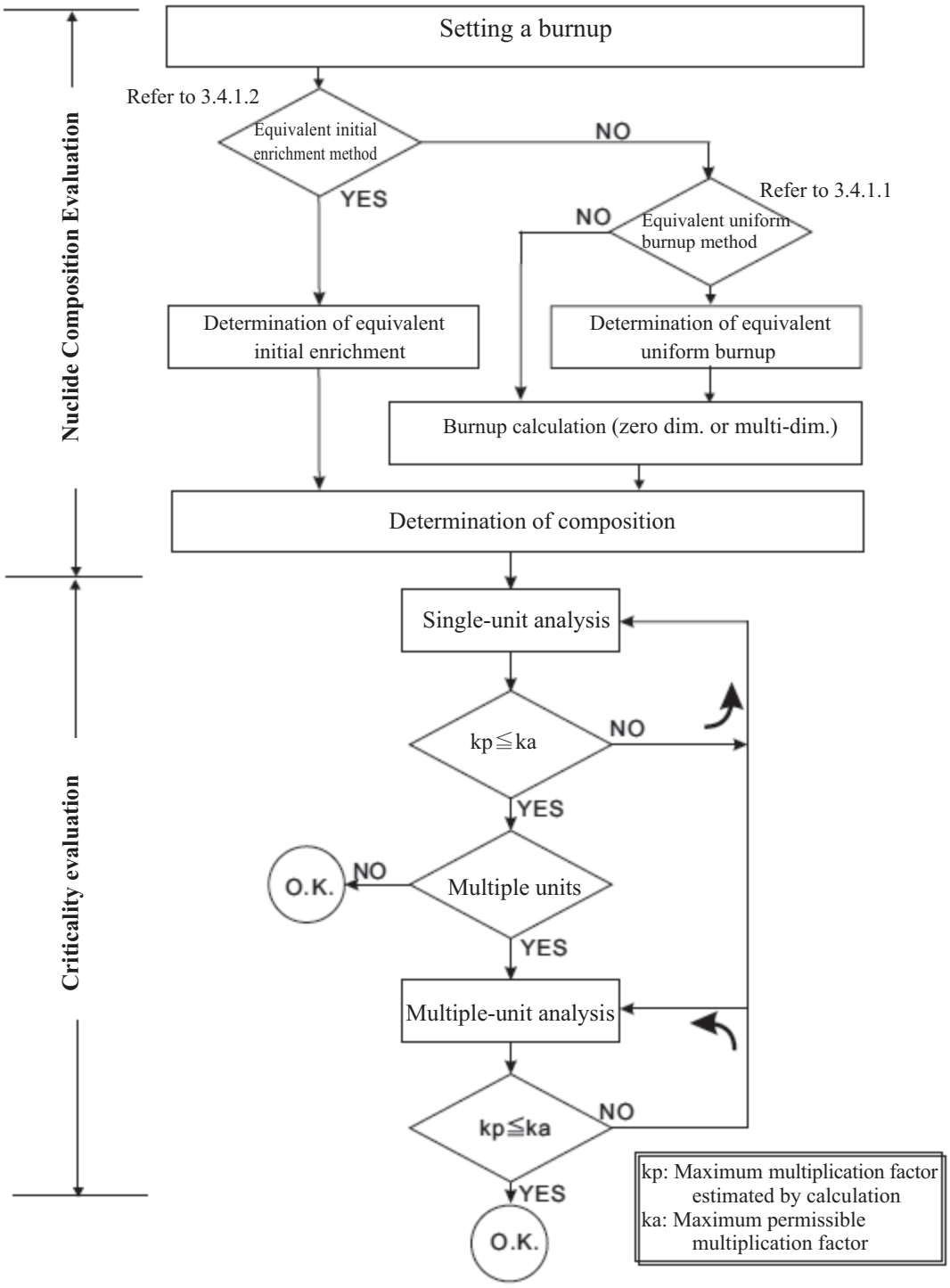

Fig. 2.3 Procedure for a criticality safety assessment applying for burnup credit

2.5.2 Isotopic Composition Evaluation of Spent Fuel

Evaluating the isotopic composition accurately is the most important step in determining the reactivity margins for systems used in SF transportation/storage or reprocessing facilities. The initial composition of the fuel is used in the "fresh fuel assumption," and the isotopic composition of the SF, if it is needed, is calculated using burnup calculation codes. The initial composition value of the fuel has a high reliability because the value is confirmed by isotopic composition analysis performed in the fuel fabrication process. On the other hand, for the SFs, the calculation methods and the isotopic composition results must be validated sufficiently because the isotopic composition results for the SFs vary depending on the conditions, including the initial composition and the burnup conditions (e.g., the specific power and void fraction), even in fuels with the same burnup.

2.5.2.1 Burnup Analysis Setup

The burnup of the SF is usually calculated on the basis of the reactor operation data (mainly coolant thermohydraulics data such as the flow rate, temperature, and pressure, as well as neutron data measured with nuclear instrumentation) and the in-core power output data calculated using core management codes, and the assembly-averaged value is given in units of MWd/t. There are two ways for determining the burnup, i.e., assuming a uniform average burnup distribution or assuming an actual burnup distribution. Whether the actual burnup distribution is considered may depend on the functional capability of the burnup calculation codes, and it directly affects the criticality evaluation as described below. In cases where a uniform burnup is assumed, the corrections should be made for the disregarded conditions as described in section 2.5.3.3.

There is an approach for obtaining a conservative envelope for the isotopic composition. For example, for BWR fuels, the conservative envelope of the isotopic composition is proposed by setting the conditions to include a severe void fraction instead of assuming a uniform enrichment or a uniform burnup distribution [4].

There are also other approaches in which the given burnup value is somewhat modified. One is a method called "Equivalent Uniform Burnup" (Refer to 3.4.1.1). This method conservatively assumes the effect of various factors (for example, the accuracy of the burnup calculation codes with or without burnup distributions, and the variation in the irradiation conditions) affecting the criticality analysis results for SF transportation/storage systems as the equivalent reduction of the burnup, and the reduced burnup is then used in the evaluation. By using this equivalent uniform burnup method, the isotopic composition of the SF is calculated with the burnup calculation codes, and the critical analysis is performed on the basis of the assumption of a uniform burnup. With this approach, a conservative neutron multiplication factor can be readily obtained. Another approach is called the "Equivalent Initial Enrichment" method (Refer to 3.4.1.2). This method assumes a reactivity decrease due to the actual burnup, as well as the effect of the various above-mentioned factors; these factors are included in the equivalent initial enrichment determination, and then the reactivity of the SF transportation/storage system is evaluated easily with the criticality calculation using only the equivalent initial enrichment method. Both methods mentioned above should only be applied to SF transportation/storage systems after their applicability has been confirmed.

In addition, note that when the burnup is obtained from the core management data, each fuel

experiences different irradiation conditions according to its arrangement in the core. Therefore, when applying the burnup credit, the irradiation history data that give the isotopic composition with the highest reactivity should be adopted.

2.5.2.2 Burnup Calculation

When calculating the isotopic composition of the SF, the following three methods are available: i) zero-dimensional burnup calculation codes (ORIGEN2 [7], etc.), ii) lattice burnup calculation codes (SWAT [8], etc.), and iii) multidimensional burnup calculation codes that can deal with the space and time dependence of the neutron spectrum (CASMO [9], etc.) In any of these codes, the isotopic composition is calculated on the basis of the input data of the burnup, as predetermined in section 2.5.2.1 above, while also considering parameters such as the initial fuel enrichment, the shape and dimensions of the fuel, the in-core arrangement; and the irradiation environment, void fraction, boron concentration, and cooling time after the fuel is discharged from the reactor.

The errors in the burnup calculation codes for calculating the isotopic composition should be adequately confirmed through comparison with experimental data. The calculation code should be selected from the various burnup codes as well as nuclide production and decay library data listed in section 3.2, and a reliable one that has been verified by experimentally measured data should be used. The accuracy of the code may be determined by comparison with other codes that have been verified according to the experimentally measured data.

Where correction factors for the calculated isotopic composition have been identified by comparison with experimental data, the results of the burnup calculation should be rectified by these correction factors. There is, however, another method that involves proceeding to the criticality evaluation without the aforementioned correction and rectifying the resulting neutron multiplication factor obtained at the end of the criticality evaluation.

2.5.3 Criticality Evaluation of a Spent Fuel System

On the basis of the isotopic composition and/or isotopic composition distributions of the SF, as obtained using the methods described above, the criticality of the SF transportation/storage systems or reprocessing facilities may be evaluated.

The procedure for criticality evaluation is the same as in the case with the “fresh fuel assumption” described in the *Nuclear Criticality Safety Handbook* [1]. In particular, the most severe, technologically conceivable event under normal and abnormal conditions is determined while considering the variations in the subject nuclear fuel materials and the arrangement of the various systems and components. After these scenarios are identified, the criticality evaluations for the single and multiple units are performed.

When the isotopic compositions of the SF are obtained by a consideration of burnup distribution, the resulting isotopic composition distribution should also be used in the criticality evaluation.

2.5.3.1 Criticality Evaluation of a Single Unit

For a single unit criticality safety evaluation, a single unit is set up for an object, such as a piece of equipment. The isotopic composition of the single unit is estimated on the basis of the parameters described in section 2.5.2 above, and the most severe reflection condition is assumed considering the boundary conditions such as submersion in water, rainfall, and snowfall, as well as the reflection effect of the structural materials, in the storage building. When the assumption of a submersion condition is

required, a 20 cm or thicker water reflector should be considered.

Next, geometrical modeling should be conducted under the assumed conditions. To simplify the modeling, assumptions giving a more severe reactivity are allowable, such as substituting water for structural materials that exist axially above and below the fuel region, and substituting water for the lattice of a control rod guide thimble.

On the basis of the above analysis conditions and modeling, the criticality of a single unit is analyzed to obtain the neutron multiplication factor. Typically, a multi-group or a continuous energy Monte Carlo calculation code that can deal with complex geometries is used.

2.5.3.2 Criticality Evaluation of Multiple Units

The criticality analysis of multiple units is performed assuming that the arrangement of the single units is based on a planned arrangement. For a criticality evaluation, the arrangement of the single units, the neutron moderation conditions among the single units, and the reflection conditions around the arrangement should be considered, and the most severe nuclear conditions within a technically conceivable range, including both normal and abnormal conditions, should be defined. On the basis of the analysis conditions and the geometrical and alignment models of the multiple units described above, the neutron multiplication factor for the system is calculated with a multi-group or a continuous energy Monte Carlo calculation code that can deal with the input data for complex alignments and geometries.

2.5.3.3 Modeling for Conservative Side and Correction of Criticality Analysis Result

The neutron multiplication factor for each system in SF transportation/storage or reprocessing facilities should be calculated for the most severe evaluation conditions in both normal and abnormal states considering all error factors, including specification conditions such as the axial distributions of the SF burnup and the isotopic composition, the fabrication tolerances, and the variations in the boundary conditions during fuel handling. In cases where it is difficult to determine these conditions, initial calculations of the reactivity worth should be completed for a variation in each parameter that is disregarded in the multiplication factor calculation, and after the neutron multiplication factor has been calculated, its value must be corrected for these variations in the parameters. When the reactivity worth is negative, for example, as is the case when disregarding the cooling time after discharge from the reactor, it can be counted as a margin. To account for this margin, a preliminary sensitivity analysis is necessary to determine the range for each parameter that provides severe reactivity results. In other words, a preliminary sensitivity analysis is required that considers, for both normal and abnormal operating conditions, the calculation accuracy of the code and the parameters that affect the criticality calculation (e.g., the chemical composition of the nuclear fuel material, the isotopic compositions and their distributions, the fuel rod pitch, the fuel assembly arrangement, the facilities and containers for transportation/storage, the shape, dimensions and material of the dissolvers, and the neutron moderation conditions in the containers). After these corrections, the “maximum neutron multiplication factor by calculation, k_p” is obtained.

2.5.3.4 Determination of Nuclear Criticality Safety [1]

In the evaluation process explained in sections 2.5.3.1 and 2.5.3.2, the calculation errors in the calculation processes (for Monte Carlo calculation codes, three times the standard deviation in statistical calculations is taken into account), as well as the predetermined reactivity worth for disregarded

conditions and for modeling are added, and the "maximum neutron multiplication factor by calculation, k_p" is obtained. When k_p is equal to or smaller than the "maximum permissible multiplication factor, k_a," which is set as equal to or less than the "the estimated lower limit multiplication factor, k_L," as stated in the *Nuclear Criticality Safety Handbook* [1], it is determined that the nuclear criticality safety of the SF system is assured. In cases where k_p is larger than k_a, the design is modified and the evaluation conditions and modeling are reset so as to make k_p equal to or less than k_a. The relationships between these types of neutron multiplication factors are shown in **Fig. 2.4**.

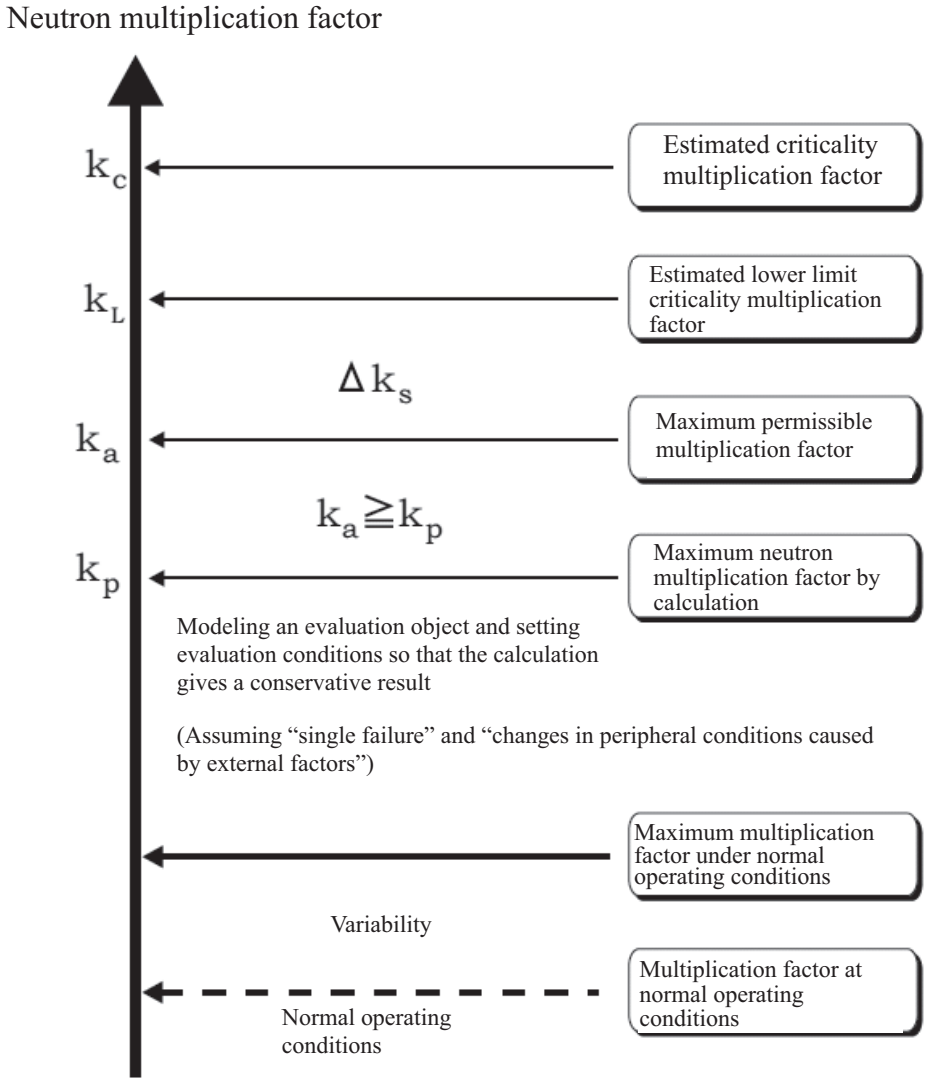

Fig. 2.4 Relationships among the various neutron multiplication factors

3. Method of Isotopic Composition Evaluation

For evaluating the criticality safety of an SF system adopting the burnup credit, the isotopic composition of the SF must be known. The composition is obtained using a burnup calculation; therefore, the codes used for the calculation must be verified based on the measured data.

3.1 Summary of the Burnup Calculation

The isotopic composition and the amount of generated activity of the SF are obtained by solving many simultaneous time-differential equations with a numerical calculation method. These simultaneous equations are a combination of the following basic equations related to the generation of nuclides from neutron absorption and the radioactive decay in a reactor.

$$\frac{dX_i}{dt} = \sum_{j=1}^{N} L_{ij}\lambda_j X_j + \phi\sum_{k=1}^{N} f_{ik}\sigma_k X_k - (\lambda_i + \phi\sigma_i)X_i + \sum_{l=1;(i:FP)} \phi\sigma_l^f Y_{il} \quad , \qquad (3.1)$$

where

X_i	Atomic number density of the nuclide i
λ_i	Radioactive decay constant of the nuclide i
σ_i	Neutron absorption cross section of the nuclide i
σ_l^f	Fission cross section of the nuclide l
Y_{il}	Ratio of the nuclide i generated from fission of the nuclide l
L_{ij}	Ratio of the nuclide i generated from fission of the nuclide j
f_{ik}	Ratio of the nuclide i generated from neutron absorption of the nuclide k
ϕ	Neutron flux assumed to be constant in a short time step

The method to solve the burnup equation is summarized below with an example using ORIGEN [10], the most well-known burnup calculation code [8, 10–13]. This summary becomes necessary for evaluating the accuracy of the calculation at an algorithmic level. Proceed to section 3.2 when this evaluation is unnecessary.

In addition, ORIGEN2 [7] is a revised version of ORIGEN, and the same solution method is adopted by both codes, excluding the input method for the cross sections described below. Therefore, ORIGEN may be replaced with ORIGEN2.

3.1.1 Matrix Exponential Method

In any burnup calculation, solving equation (3.1) is required. The matrix exponential method can be used for solving large-scale simultaneous differential equations. This method is adopted in ORIGEN (also in ORIGEN2), and was studied extensively at the Oak Ridge National Laboratory (ORNL) in the 1960s.

First, in the matrix exponential method, equation (3.1) is written as follows:

$$\frac{d\boldsymbol{X}}{dt} = \boldsymbol{A}\boldsymbol{X} \quad .$$

Here A is called the transition matrix and X is a vector that represents the amount of the isotope (atomic number density). (3.2)

Then, equation (3.2) is solved as

$$\boldsymbol{X} = \exp(\boldsymbol{A}t)\,\boldsymbol{X}(0). \tag{3.3}$$

In equation (3.3), the exponential term is expanded with $\boldsymbol{I}$ as the unit matrix, as follows:

$$\exp(\boldsymbol{A}t) = I + \boldsymbol{A}t + \frac{1}{2!}(\boldsymbol{A}t)^2 + \cdots = \sum_{n=0}^{\infty} \frac{(\boldsymbol{A}t)^n}{n!}. \tag{3.4}$$

How the exponent in equation (3.3) is expanded contributes to the efficiency of the matrix exponential method. When the matrix exponential method was developed, the main storage capacity of a computer was too small to deal with the transition matrix directly. Therefore, a technique was adopted to solve the matrix by selecting only the nonzero elements in the transition matrix. Moreover, ORIGEN utilizes the fact that the quantity of short half-life isotopes promptly reaches an equilibrium state (the values of the corresponding elements in the transition matrix are large.) Thus, only the isotopes for which the diagonal elements in the transition matrix are not larger than the predecided value are calculated by the matrix exponential method, and the other isotopes are evaluated using an analytical method.

3.1.2 Bateman Equation

The isotopes determined but not addressed with the matrix exponential method are dealt with in the following Bateman equation:

$$X_i = X_i(0)\exp(-d_i t) + \sum_{k=1}^{i-1} X_k(0)\left(\sum_{j=k}^{i-1} \frac{\exp(-d_j t - \exp(-d_i t)}{d_i - d_j} a_{j+1,j} \prod_{\substack{n=k \\ n\neq j}}^{i-1} \frac{a_{n+1,n}}{d_n - d_j} \right). \quad (3.5)$$

Here $a_{i,j}$ is an element in the transition matrix $\boldsymbol{A}$ represented by equation (3.2). The first-order rate of change, d_j, corresponds to the diagonal element $a_{j,j}$ in the transition matrix. Equation (3.5) is transformed by multiplying it by $\prod_{n=k}^{i-1} d_n$ and then dividing it by $\prod_{n=k}^{i-1} d_n$:

$$X_i = X_i(0)\exp(-d_i t) + \sum_{k=1}^{i-1} X_k(0) \prod_{n=k}^{i-1} \frac{a_{n+1,n}}{d_n} \left(\sum_{j=1}^{i-1} d_j \frac{\exp(-d_j t) - \exp(-d_i t)}{d_i - d_j} \prod_{\substack{n=k \\ n\neq j}}^{i-1} \frac{d_n}{d_n - d_j} \right). \quad (3.6)$$

The factor multiplied by $X_k(0)$ in the second term in equation (3.6) is the ratio of the number of atoms in nuclide k that exists after a specified decay and the neutron capture reaction. When the product in the second term becomes 10^{-6} or less, the contribution to nuclide i from both nuclide k and its precursor is disregarded. A small $d_i t$ does not make the Bateman equation difficult because the nuclide composition for this case is solved by the matrix exponential method described in the previous section. Therefore, the matrix exponential method and the Bateman equation are complementary to one another in ORIGEN2.

3.1.3 Variation in the Cross-Section Data Associated with Burnup

The cross-section data, including the nuclear fission and neutron absorption, are used as the input data in the burnup calculation described above. These values are varied along with the changes in the neutron spectrum. The neutron spectrum differs according to the ratio H/U in the fuel and the fuel geometry (system). Moreover, the neutron spectrum varies according to the change in the nuclide composition in the fuel that occurs with burnup. Therefore, it is important for the improvement of the burnup calculation accuracy that the type of fuel and the change in the neutron spectrum that occurs with burnup are accurately portrayed in the calculation code. Each burnup calculation code has its own strategy to deal with the neutron spectrum change.

One of the strategies is to allow users to select an appropriate data file from the preserved data files that contain the cross-section changes occurring with burnup. This technique was typically applied to ORIGEN2 [7]. Although it is easy to use such codes, the calculation accuracy is compromised when there is a difference between the burnup calculation and library creation conditions.

There are other codes that do not rely on the aforementioned process; instead, the neutron spectrum calculations are performed successively, as burnup continues, using a divided operation history. In this method, the burnup calculation switches the reaction cross sections according to a successive neutron spectrum calculation. Therefore, errors due to condition differences between the calculation and library creation can be lessened. This type of code is advantageous for introducing the effect of a detailed irradiation history. Examples of this type of code include SRAC95 [14], SWAT [8], and the reactor core calculation code.

3.2 Burnup Calculation Codes

A burnup calculation code is characterized by the calculation method and library, as described in the previous section. However, when a short half-life nuclide is built into the burnup chain in ORIGEN, the following methods are adopted in order to overcome the difficulty of the numerical calculation. First, the burnup equation is solved with the Bateman equation, and then, the saturation value of the short half-life nuclide is obtained. Next, the entire burnup chain is solved by the matrix exponential method. Therefore, note that the solution method cannot be divided completely into two parts with ORIGEN when a short half-life nuclide is considered in the burnup chain.

Table 3.1 shows a list of the burnup calculation codes related to the present document. The outline and features of the burnup calculation code quoted in the table are described below. The outlines of other burnup calculation codes that are considered for use in Japan are included in the appendix. The nuclear calculation codes of the fuel assembly, which generate a group constant on the basis of the nuclide composition obtained from the burnup calculation and apply it to a critical calculation code of the system, are also included in **Table 3.1.** When such a code is used, the result generated from the nuclide composition calculation and utilized for the critical calculation in **Fig. 2.3** is not exactly the nuclide composition, but group constants, in which the nuclide composition, geometry, and the moderator conditions, etc., have been considered.

3.2.1 ORIGEN2.1

ORIGEN2 [7, 15] is the most commonly used burnup calculation code throughout the world. The first version of ORIGEN [10], which is the basis of ORIGEN2, became public in 1963. ORIGEN79 was then released to the public in 1979, with improvements in both the increment in the FP nuclides handled in the code and the renewal of the library. Next, ORIGEN79 became the basis of ORIGEN-S [11] of SAS2H, which is known as a module of the safety evaluation calculation code system SCALE [16] in the United States. The burnup solution in ORIGEN is based on the matrix exponential method, which enables the calculation of burnup for many nuclides.

ORIGEN2 is one of the codes in the ORIGEN family derived from different development flows from ORIGEN-S, and was released in 1980 as ORIGEN2 (80) [7]. In the best-known ORIGEN2 (82) [15], an update of the (α,n) yield, addition of the cross-section data in the library, and an update of the nuclear data were performed, and the code was opened to the public. ORIGEN2 (86) was issued in 1986, and a part of its decay data was revised. Finally, ORIGEN2.1 [3], which is the latest version, was released in 1991. In this version, the cross-section library data were developed to be applicable to the new types of fuels and cores. This version was created with the aim to enable calculations to be performed using a personal computer with an Intel 80386 or better processor and Windows or MS-DOS as the operating system.

ORIGEN2 has a simplified command, called the ORIGEN command that allows an easy input of data. In addition, for the cross-section data, effective one-group cross sections are used in the code. This cross-section data were prepared for each reactor core. The user selects the data on the basis of the available information on the analysis subject.

The ORLIBJ32 [5] library for ORIGEN2 was developed in 1999 by the Nuclide Generation Amount Evaluation Working Group of the Sigma Committee. The library used JENDL-3.2 [17] for the nuclear data and adopted the second edition of the JNDC library [18] for the decay constants and nuclear fission yields. With this library, the calculation accuracy was improved compared to that of the existing library built into ORIGEN2.

Table 3.1 List of Burnup Calculation Codes Utilized in Japan

No.	Code Name	Development Organization	Year	Cross section		Spectrum calculation	Generation and depletion chain*	
				Library	Number of groups		Actinide	Fission product (FP)
1	ORIGEN2.1	ORNL	1991	B4[(1)],B5[(2)]	Effective 1	0 D	B4,B5	B4,B5
2	UNITBURN	JAERI	1990	J32[(3)]	26	1 D	JDDL	JNDCV2
3	SRAC95	JAERI	1995	J32	107	1, 2 D	—	—
4	MKENO-BURN	JAERI	1996	J32	26	1, 2, 3 D	JDDL	JNDCV2
5	FPGS90	JAERI	1995	J32	Effective 1	0 D	—	JNDCV2
6	COMRAD	JAERI	1996	J32	Effective 1	0 D	JDDL	JDDL
7	SWAT	UTOHOKU,JAERI	1991	J32	107	1, 2 D	JNDC	JNDCV2
8	MVP-BURN	JAERI	1996	J32	Continuous Energy	3 D	—	—
9	SCALE4.3	ORNL	1996	B4,5	Multiple	1 D	—	B5,B6
10	FLEXBURN	CRIEPI	1993	B4,5/J32	67/26	2 D	—	—
11	CASMO-4	Studsvik Scandpower	---	B4	70	2 D		
12	PHOENIX-P	Westinghouse	1987	B5 etc	42 etc	2 D	B5 etc	B5 etc
13	NEUPHYS	NFI	1985	B4,5	98	2 D	—	—
14	NULIF	NFI	1988	B5	94	1 D	B5	B5
15	TGBLA	Toshiba, JNF, GE	1982	B4,5	98	2 D	—	B5
16	HINES	Hitachi, JNF	1982	B4,5	98	2 D	—	—
17	VMONT	Hitachi, JNF	1987	J32,J2	190	3 D	—	—

(1): ENDF/B-IV [21] (2): ENDF/B-V [22] 3: JENDL-3.2 [17]

*: — Unknown (Refer to Appendix II)

3.2.2 SWAT

SWAT [19, 20] is a burnup calculation code developed at Tohoku University and JAERI. In this code, the change in the neutron spectrum can be considered as a function of the burnup. SWAT transforms the effective one-group cross sections, which are calculated at each burnup step by SRAC with the stepwise burnup calculation method, into library data with an ORIGEN2 format, and performs the burnup calculation using ORIGEN2. With this approach, a burnup calculation that considers the change in the neutron spectrum with burnup becomes possible for about 1000 nuclides. The features of SWAT are as follows:

1. SWAT is a completely automated program, and all of its functions are controlled by a single input.
2. SWAT has a 147-group library, in which 107-group cross-section data, calculated with SRAC, have been expanded to a neutron energy of 20 MeV. It also has a threshold reaction cross section such as (n, 3n). All of the nuclear data evaluated with JENDL-3.2 and the decay data in the latest edition of the JNDC library were built into the SWAT library. Therefore, the SWAT burnup calculation code can perform the burnup calculation with the most recent data.
3. SWAT can deal with two or more burnup regions.
4. The coolant water density is automatically calculated within the code on the basis of the temperature. Because the coolant density must be calculated when the burnup analysis of a light-water reactor is completed, this function is practically useful.
5. A file in the same format as the library for ORIGEN2 is generated by SWAT and thus is usable in ORIGEN2. SWAT is a burnup calculation code that also generates the library for ORIGEN2.
6. Tools to create a subroutine for the Variable Actinide Cross Section are available. These tools can be used to create not only libraries for ORIGEN2 but also subroutines to install ORIGEN2.

3.2.3 SCALE4 [16]

SCALE is used in the United States for the safety evaluation of nuclear facilities. The code system is also used in some of the nuclear facilities in Japan. The development and maintenance of this code system are performed by ORNL, and can be used by combining the necessary codes for the safety analysis with a program called the SCALE driver. The distinctive feature of this code is that it becomes a systematized package. The codes for the neutron spectrum calculation, the cross-section generation, the burnup calculation, and the Monte Carlo calculation for the critical analysis are built into a systematized package. Most of the safety evaluations can be performed by utilizing this package. The origin of the data in each safety evaluation is common, and consistent analyses can be performed.

In SCALE, the ORIGEN-S code of the SAS2H sequence is used for the burnup calculation. This code has inherited the development resources of the original ORIGEN. Users can specify neutron energy groups via three parameters, called THERM, RES, and FAST, in the code. Then, effective one-group cross-section data are generated, and the burnup calculation is performed. Of course, the analysis can also be performed with the neutron spectrum obtained by other SCALE modules.

3.3 Validation of Burnup Calculation Codes

When the nuclide composition data for the critical safety evaluation are obtained using a burnup calculation code, an evaluation of the errors included in the calculated nuclide composition is required.

3.3.1 Data for Validation

3.3.1.1 SFCOMPO

Both reactor operation data and core power distribution data are useful for the validation of a burnup calculation code. A comparison between the experimental and calculated results concerning the nuclide composition data is especially important to the burnup credit calculation. In this respect, the database SFCOMPO [23] from JAERI is useful because the validation data, including documents open to the public, have been widely investigated, and the nuclide composition data for SFs are stored in the database. The data stored in SFCOMPO are listed in **Table 3.2**, which shows that most of the data were obtained from measurements performed in the 1960s and 70s. It also shows the results of the postirradiation examinations (PIEs) performed recently in Japan, which were collected, released to the public, and then stored in SFCOMPO. Unfortunately, the available data related to the composition of the FPs are limited. In order to apply the level 2 or higher-level burnup credit with the reactivity effect of FPs, data including those obtained in the recent ARIANE program are necessary for validation.

Table 3.2 List of Post Irradiation Examinations [23]

Reactor	Country	Type	Burnup (GWd/MTU)	Irradiation term
Yankee	USA	PWR	6–33	1960–1964
Trino Vercellese	ITA	PWR	3–27	1964–1971
Obrigheim	FRG	PWR	16–38	1970–1974
Garigliano	ITA	BWR	8–13	1964–1967
Gundermmingen	FRG	BWR	14–27	1969–1973
Monticello	USA	BWR	39–59	1974–1982
Mihama-3	JPN	PWR	6–34	1978–1982
Genkai-1	JPN	PWR	38	1975–1979
Tsuruga-1	JPN	BWR	8–26	1972–1978
JPDR	JPN	BWR	0.1–6	1963–1969
Calvert Cliffs-1	USA	PWR	27–44	1977–1982
TMI-2 (accident)	USA	PWR	3	1979
H.B.Robinson-2	USA	PWR	24–32	~1975
Fukushima 1-3	JPN	BWR	6–32	1977–1982

The isotopic composition of the SF varies according to the reactor type, the initial fuel enrichment, the geometry, and the water-to-fuel volume ratio for various fuels with the same burnup. Moreover, it is necessary to consider the experimental errors for verification of the burnup calculation code. Therefore, in order to verify a burnup calculation code, the measured nuclide composition data in many PWR and BWR SFs built into SFCOMPO were surveyed, and the following nuclide composition ratios were arranged as functions of burnup only (here the atomic number densities are represented with atomic symbols).

(a)	$(^{235}U/U)/(^{235}U/U)_{initial}$	Ratio of the ^{235}U enrichment of the fuel directly after discharge from the reactor to the initial enrichment
(b)	Pu/U	Ratio of the plutonium total to the uranium total
(c)	$^{239}Pu/Pu$	Ratio of ^{239}Pu to the plutonium total
(d)	$^{240}Pu/Pu$	Ratio of ^{240}Pu to the plutonium total
(e)	$^{241}Pu/Pu$	Ratio of ^{241}Pu to the plutonium total
(f)	$^{242}Pu/Pu$	Ratio of ^{242}Pu to the plutonium total

In each group of the nuclide composition ratio from (a) to (f) above, the measured data dispersion was considered as a fluctuation around a central value; the measured data were fitted to the second or third multinomial expression of the burnup by the least-squares method. The multinomial expression of the smaller deviation from the measured data was adopted as the regression expression. An example is shown below.

Postulating f (B_i) as the most probable estimated value in burnup B_i of the composition ratio, f (B_i) is written as the following equation:

$$f(B_i) = C_0 + C_1 * B_i + C_2 * B_i^2 + C_3 * B_i^3, \tag{3.7}$$

where B_i is the burnup (GWd/t), and C_0, C_1, C_2, and C_3 are the coefficients obtained by the least-squares method.

The standard deviation of the measurements based on the regression curve of the adopted polynomial expression is obtained by the next equation:

$$\sigma = \sqrt{\left[\frac{\sum (y_i - f(B_i))^2}{(n-1)}\right]}, \tag{3.8}$$

where y_i represents the measurements of the nuclide composition at burnup B_i, and n is the number of samples. The regression curves of the measurements described above and the resulting ±3σ deviation curve for the burnup are plotted in **Figs. 3.1**–**3.6**. (The data for a population of 99% or more are included within ±3σ deviation curve.) These figures show that the majority of the data for BWR fuel assemblies are also within the ±3σ deviation curve, despite the fact that the measured data for the BWR are minor in

these figures. Note that the data for JPDR and the Fukushima 1-unit 3 are not used in the curve; they are added later to these figures.

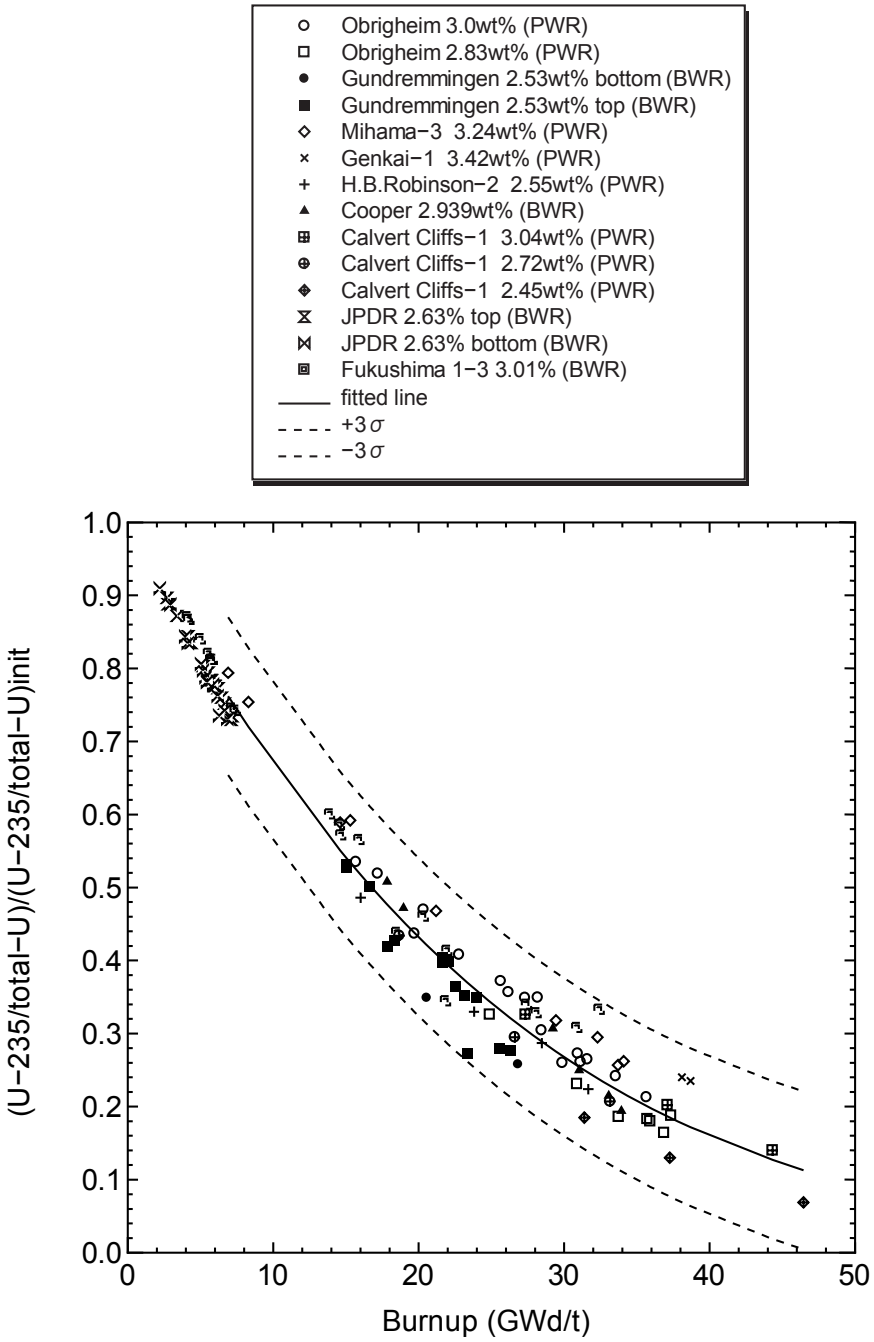

Fig. 3.1 Summary of post irradiation examination data: (U-235/Total-U)/(U-235/Total-U)$_{init}$

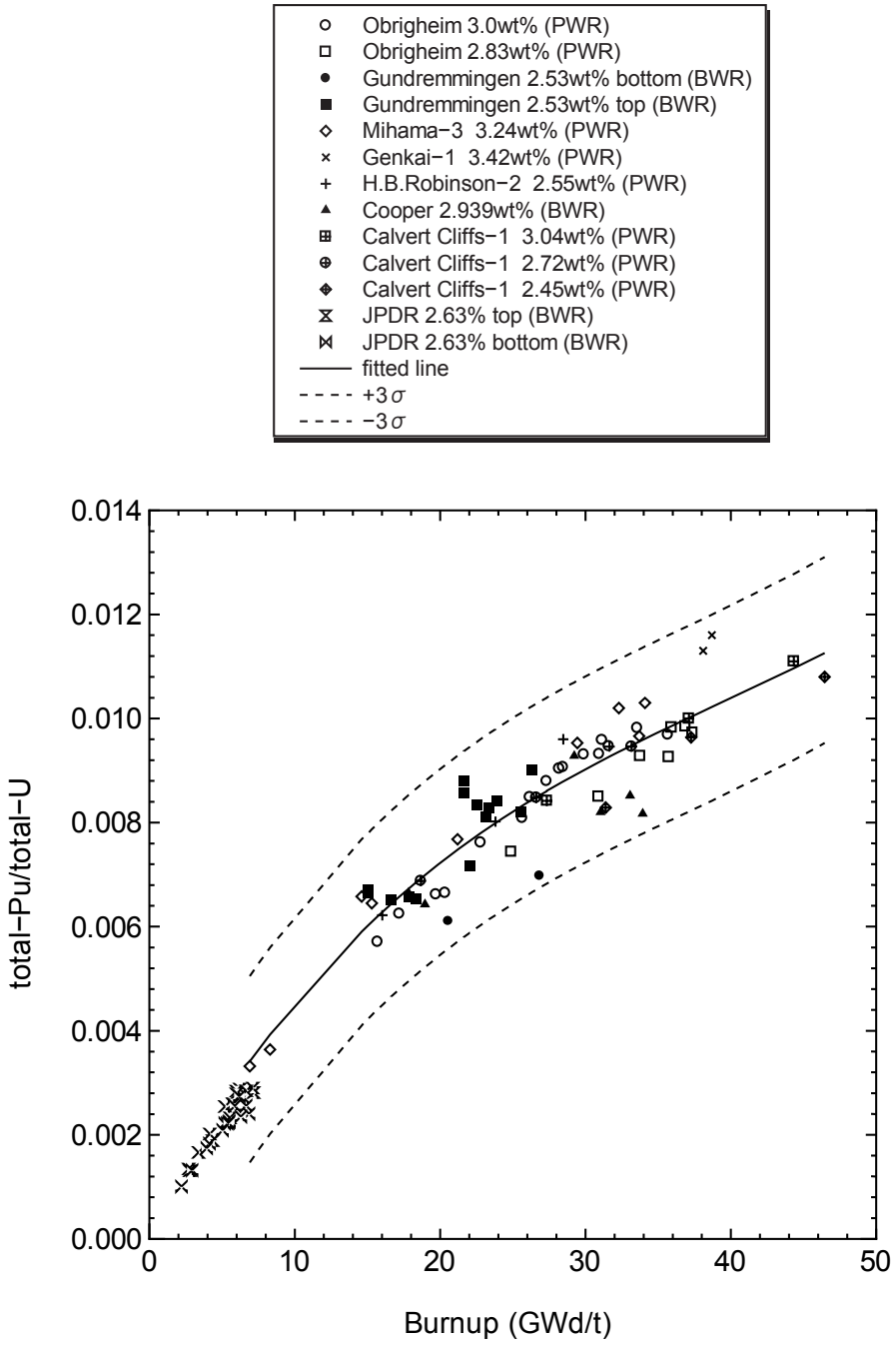

Fig. 3.2 Summary of post irradiation examination data: Total Pu/Total U

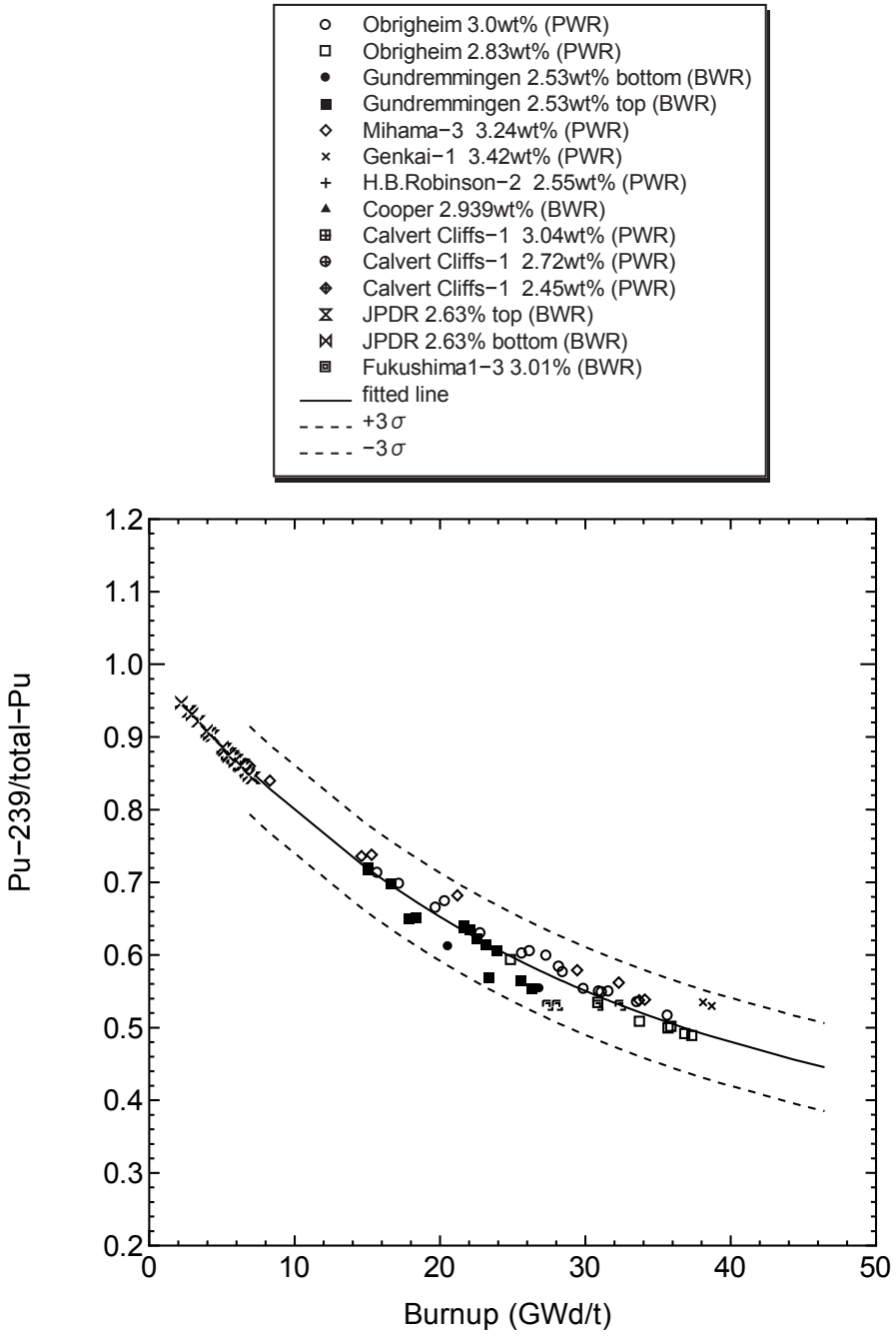

Fig. 3.3 Summary of post irradiation examination data: Pu-239/Total Pu

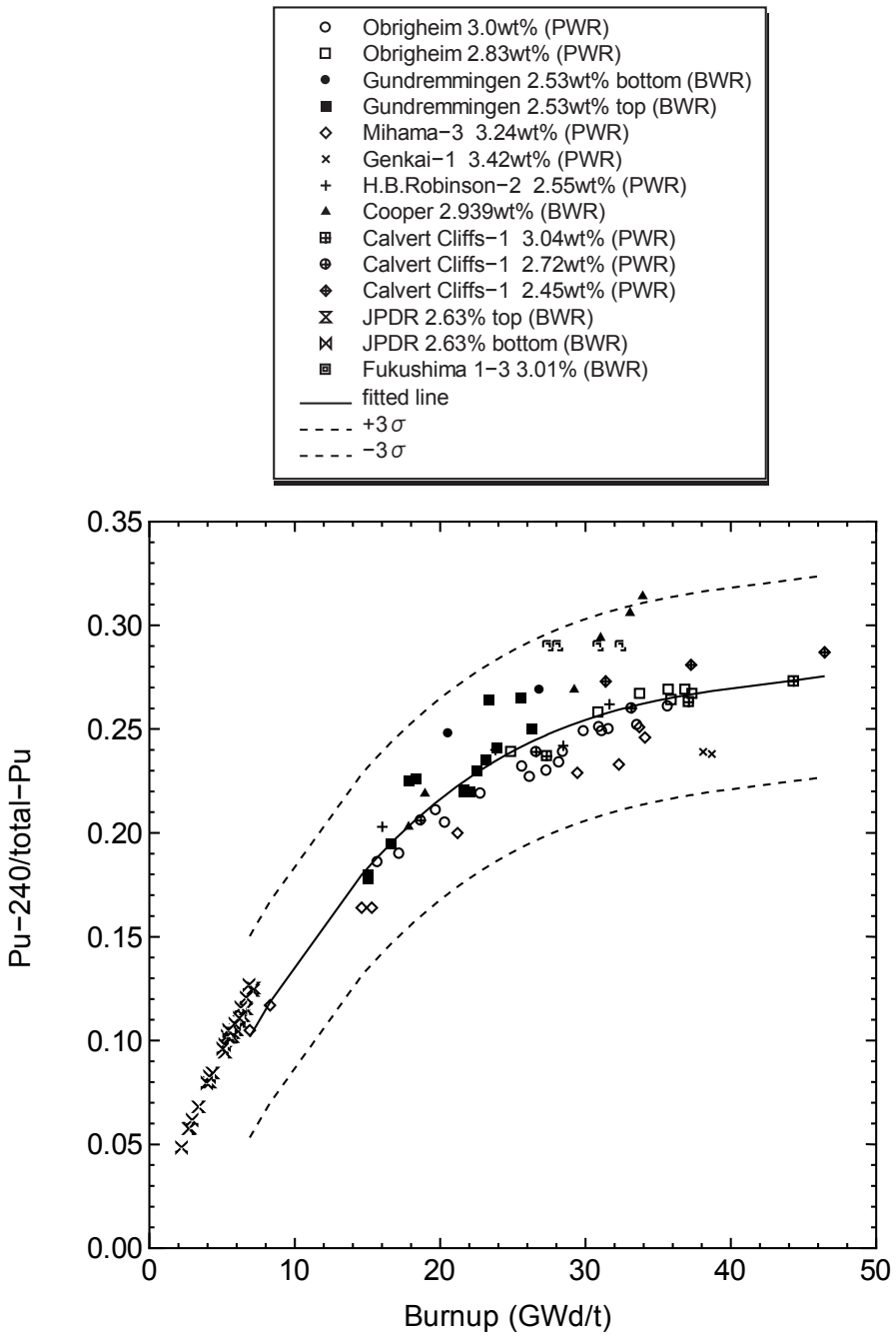

Fig. 3.4 Summary of post irradiation examination data: Pu-240/Total Pu

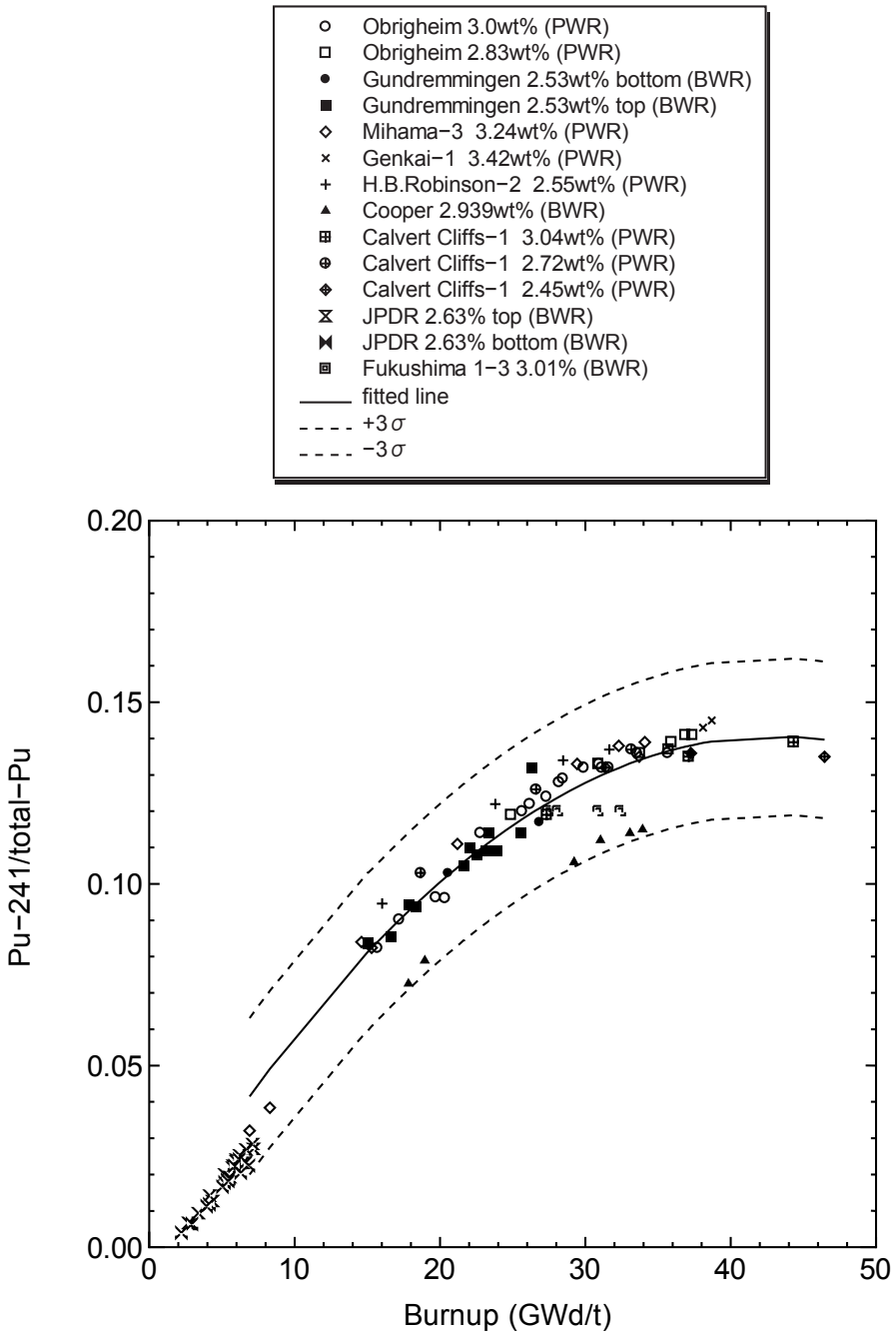

Fig. 3.5 Summary of post irradiation examination data: Pu-241/Total Pu

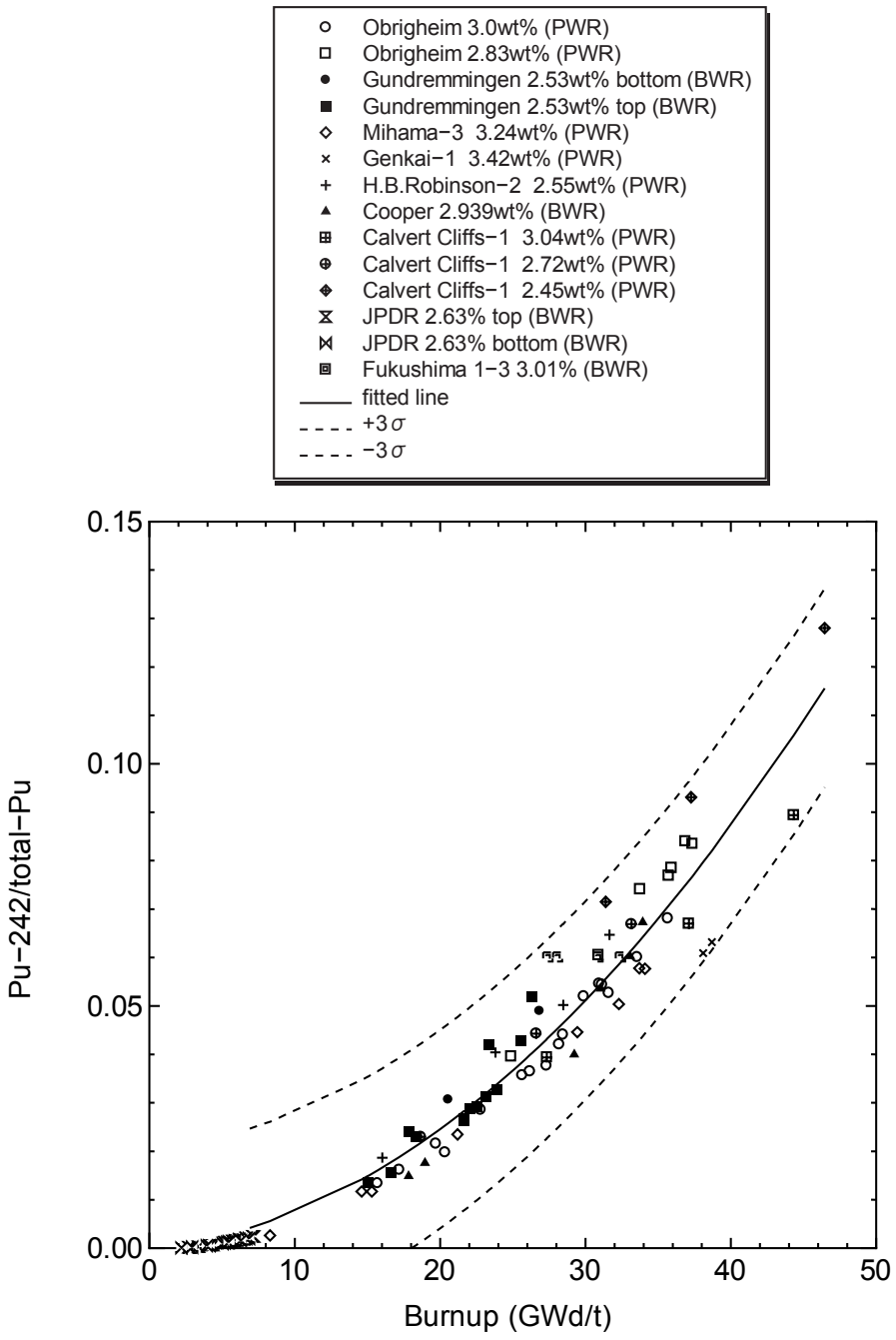

Fig. 3.6 Summary of post irradiation examination data: Pu-242/Total Pu

3.3.2 Examples of Validation

In the validation of the burnup calculation code, it is important to evaluate any errors in the nuclide composition calculation for the measured data by analyzing the obtained data from the PIE. In order to apply the calculated nuclide composition to a critical calculation, a correction factor that takes the errors of the calculated nuclide composition into account must be introduced.

3.3.2.1 Calculation of the Correction Factors

The correction factors are obtained with the prediction interval method, which is used for a normal population with a limited degree of freedom. When the measured data do not show clear tendencies, the correction factor F_i for the nuclide *i* is calculated using the following equation, with the number of samples denoted as N_i [24]:

$$F_i = \bar{x}_i \pm k_{N_i}^{\alpha,\beta} . \sigma_i , \tag{3.9}$$

where

$\bar{x}_i$ is the average of the sample values in the population C/E for nuclide *i*;

$k_{N_i}^{\alpha,\beta}$ is a coefficient to determine β, the probability where the next, newer measurement point exists with the reliability α from the normal population with a degree of freedom $N_i - 1$; and

σ_i is the standard deviation.

In equation (3.9), F_i with a plus sign represents the upper boundary, while F_i with a minus sign represents the lower boundary of the measured data range. The choice is determined depending on the nuclear characteristics of the subject nuclide in order for the critical safety evaluation to give a conservative result. In other words, for a nuclide that gives a positive reactivity worth, such as a fissile nuclide, the minus sign is selected so that F_i becomes small, and for a nuclide that gives a negative reactivity worth, such as a thermal absorptive nuclide, the plus sign is selected so that F_i becomes large.

In addition, there is a simple calculation method for the correction factor when the burnup calculation is performed with PIE data. For a fissile nuclide, the minimum value for C/E, or the ratio of the calculated value to the obtained measured value, is assumed to be the correction factor. For a nuclide to act as a neutron absorber rather than a fissile nuclide, the maximum value of C/E is assumed as the correction factor.

The correction factor obtained by these methods is used as follows: the nuclide composition of the subject SF is obtained via a burnup calculation. The resulting nuclide composition is divided by the correction factor prepared beforehand to obtain the corrected nuclide composition. Finally, the corrected nuclide composition is applied to the critical calculation for conservative results.

3.3.2.2 Validation of ORIGEN2.1

An example derivation of the correction factor for ORIGEN2.1 can be seen in reference [25]. In this reference, the correction factor simply applies the maximum and minimum values of the C/E obtained using the burnup calculation. In order to make a conservative evaluation for criticality safety, a correction factor of 1.0 is given when the correction factor of a fissile nuclide exceeds 1.0 and when the correction factor of a neutron absorptive nuclide becomes less than 1.0.

The following PIE data for SFs in **Table 3.2** were selected, which were used to perform a burnup calculation: PWR PIE data from Obriheim, Genkai-1, and Mihama-3, which were recently measured and stored in the light-water reactor SF nuclide composition measured database (SFCOMPO), and the PWR PIE data of Takahama-3, which were measured by JAERI (The number of samples are 23, 2, 8, and 5, respectively, for a total of 38).

The calculation used the latest version of ORIGEN2.1 [3], along with the data from the library attached to ORIGEN2.1 and the library ORLIBJ32 [5] for ORIGEN2, which is based on JENDL-3.2. In the PIE data for the PWR SF assemblies, values were obtained for various nuclides, but the data commonly given are only for the 10 main actinides (^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{241}Am), and the measured and calculated results were compared for these 10 nuclides. The burnup calculation results were summarized in the form of C/E, and the correction factor F was then calculated for each previously identified actinide. The applicable scope of the burnup is from 0 GWd/t to 40 GWd/t. The calculated correction factors in reference [25] are shown in **Table 3.3.**

For the actual burnup credit evaluation, a conservative result will be obtained by applying the corrected nuclide composition to the criticality calculation. The corrected nuclide composition is the calculated nuclide composition from ORIGEN2.1 divided by the correction factor. The correction factor can be chosen from **Table 3.3** for each nuclide and the similar irradiation reactor type/irradiation conditions of the subject SF.

Table 3.3 Examples of Correction Factors for ORIGEN2.1 Calculation Results

Nuclide	Correction factors							
	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	1.00	1.08	1.00	1.00	1.00	1.00
Am-241	2.41	2.62	1.06	1.18	1.00	1.00	1.51	1.62

3.3.2.3 Validation of Other Codes

In addition, the PIE data were analyzed for SWAT and SCALE, and the correction factors were obtained using the data stored in the LWR SF composition database SFCOMPO, as shown in **Table 3.2** [26]*. The PIE data were analyzed for 26 samples from the Obrigheim reactor and 55 samples from seven other reactors [27, 28]; in particular, Yankee, Genkai-1, Mihama-3, Trino Vercellese, Turuga-1, Calvert Cliffs-1, and the Obrigheim reactor. The 55 samples were used to analyze the PWR SF by SAS2H, as described in the report by ORNL and DOE.

For the code verification, 71 samples were selected considering the difference in the burnup measurement method. In determining the correction factors, statistical data processing methods (the acceptable limitation value method and the linear regression method) were applied to the data fixed by a 95% authorization to the obtained C/E, and the correction factors were then calculated for all nuclides, including the fission products (FPs) obtained by PIE.

3.3.2.4 Analysis of PIE Data with SCALE

The PIE data for SFs that are also described in SFCOMPO were analyzed with SCALE4.2 [29]. The US DOE proposed a design method with the burnup credit [30] only accounting for actinides in the transportation of PWR SF, and the NRC completed the regulatory review of this method. In the proposal from the DOE, the following verification concerning the nuclide composition evaluation was implemented using the SAS2H sequence in SCALE4.2 (radiation source calculation sequence with ORIGEN-S) as a representative example of the calculation code.

The verification used 54 measured data quantities collected by the Materials Characterization Center of the Pacific Northwest Laboratories (PNL) from 18 types of fuel assemblies in seven reactors, namely Yankee, Mihama-3, Trino Vercellose, Turkey Point, Calvert Cliffs-1, H.B.Robinson-2, and Obrigheim. The compositions were calculated using the SAS2H sequence with 27-group cross-section data built into SCALE4.2 that accounts for the neutron spectrum of the burnup dependence for nine actinides (^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{241}Am), and the results were compared with the measured data. Correction factors were obtained using the same statistical processes mentioned in section 3.3.2.1. In addition, a correction term was added for the nuclides for which a burnup and an irradiation spectrum dependency exist. These correction factors are applicable to PWR SFs with an initial ^{235}U enrichment range of 1.5–4.05 wt% and a discharged burnup of 0–40 GWd/t.

3.4 Factors to be Considered in Isotopic Composition Evaluations and Examples of Evaluation

3.4.1 Burnup

3.4.1.1 Equivalent Uniform Burnup

When evaluating the subcriticality of the SF transportation/storage system while accounting for the burnup, it is necessary that the isotopic composition of the actinides and FPs in the SF should be derived accurately based on burnup, and the reactivity of the subject system should be derived conservatively in considering variation in parameters, including the irradiation history.

Note*Here E/C is addressed with statistical processing, and the correction factor is calculated. In this case, the correction factor is applied—multiplying it by the calculated value of the nuclide composition.

Therefore, not only the accuracy of the burnup calculation codes, but also the influence of the burnup conditions including the irradiation conditions, and the axial and horizontal distribution of the burnup in the SF should be evaluated, and the "maximum neutron multiplication factor by calculation" can be obtained. Usually, a sensitivity analysis of the criticality calculation results (neutron multiplication factor) on the items below should be performed to obtain a conservative neutron multiplication factor.

1. Analytical accuracy of the burnup calculation codes for the isotopic composition based on comparisons with experimental measurements
2. Burnup distribution/Void distribution
3. Effect of irradiation history

The "equivalent uniform burnup" is equivalent to an actual SF storage system in terms of reactivity, which enables a simple and easy evaluation when considering the effect of each factor in the above-described criticality calculation, and assuming a more conservative factor than the actual burnup. By performing a nuclide composition evaluation and the criticality safety analysis of the SF transportation/storage system with this "equivalent uniform burnup," the "maximum neutron multiplication factor estimated by the calculation" can be obtained simply and easily without directly considering the axial burnup distribution of the fuel rod, the calculation accuracy of the burnup calculation code, or the irradiation history of the fuel. Examples in which the "equivalent uniform burnup" obtained for an SF dry storage cask, wet storage pool, and wet transportation cask are described below.

1. Examples for Dry Storage Casks

For dry storage casks with 52 BWR SF assemblies and with 21 PWR SF assemblies, the effect of the calculation accuracy on the cask reactivity was evaluated using the ORIGEN2.1 [3] and the KENO V.a code in the SCALE system [16], along with the attached 27-group neutron cross-section library [31]. (For this criticality analysis, the dry storage cask was assumed to be filled with water). By evaluating with ORIGEN2.1 based on the uranium and plutonium nuclide composition measured data at 13 burnup points within the burnup range of approximately 10–50 GWd/t, a calculation accuracy of approximately 10% in the mean-square error was obtained for important nuclides such as ^{235}U and ^{239}Pu in the reactivity evaluation. The effect of the calculation accuracy in ORIGEN2.1 on the cask reactivity was then evaluated with the following procedure. First, the effect (sensitivity) of an increase and decrease of each actinide in the SF on the reactivity of the cask system was analyzed. Second, the absolute figure of the products of the resulting sensitivities and ORIGEN2.1 calculation accuracies was summed. The evaluation resulted in an approximately 3% Δk/k for the PWRs and BWRs at a burnup of 40 GWd/t.

Because the burnup is distributed axially in the SF, it is necessary to evaluate the effect of this distribution on the cask reactivity. In addition in BWRs, an axial void distribution exists in the core, and the history of this void distribution affects reactivity. Because the nuclide composition of the SF is usually evaluated for the average burnup of the fuel, the difference in the cask reactivity between two cases assuming the same burnup and uniform void fraction along the entire length of the fuel assembly and the modeled axial distributions was evaluated. In addition, the nuclide composition of the SF is affected by the irradiation histories of the boron concentration in the coolant (PWR), the fuel rating (BWR, PWR), the control rod insertion (BWR), and the BP rod insertion (PWR) in the core. Therefore, the effects of the irradiation history on the cask reactivity were evaluated.

When the same burnup and uniform void fraction were assumed along the entire length of the fuel assembly, the cask reactivity calculated using KENO V.a based on ORIGEN2.1 had to be adjusted to compensate for a reactivity worth of approximately 8% Δk/k in the PWRs and 12% Δk/k in the BWRs for a discharged burnup of 45 GWd/t. Similarly, the cask reactivity worth was calculated at burnups

other than 45 GWd/t. Converting the reactivity worth into a decrease in the fuel burnup, the "equivalent uniform burnup" was obtained and is shown in **Fig. 3.7.** The eight subject nuclides (which were obtained by removing ^{241}Am from the nine nuclides recommended by NRC, as described in section 4.4.1.1) were considered, and the cooling time was set at 15 years. This figure presents the evaluation curve for the "equivalent uniform burnup," which is integrally applicable to evaluating the reactivity of dry cask systems bearing PWR and/or BWR SF assemblies. In case of a fuel-assembly-average burnup of 45 GWd/t, for example, various calculations with ORIGEN2.1 assuming an input of a uniform burnup of 28 GWd/t give a conservative evaluation that allows for the compensation of the reactivity worth of the axial burnup effects, the irradiation history, and the errors associated with the ORIGEN2.1 analyses.

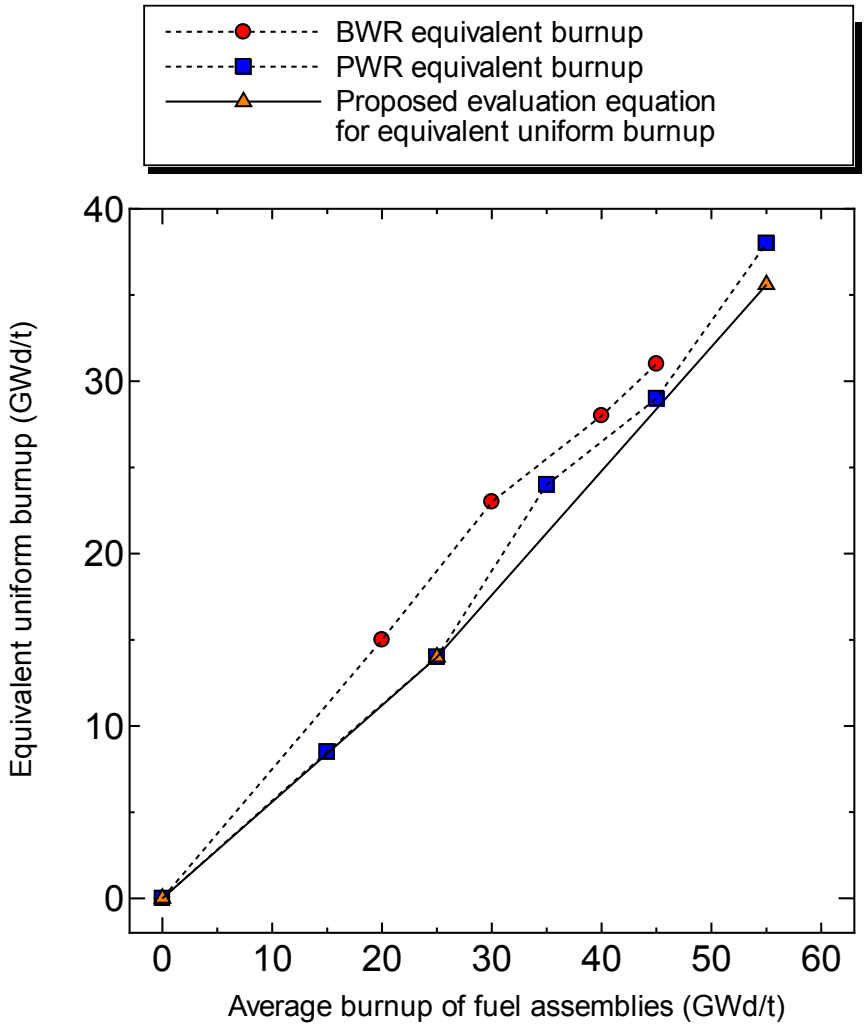

Fig. 3.7 Equivalent uniform burnup in a dry storage cask considering the axial burnup distribution, irradiation histories, and nuclide composition calculation accuracy of ORIGEN2 (U + Pu only) [31]

2. Examples for Storage Pools [32]

Recently, the PIE test of a PWR (15 × 15) SF was performed at JAERI, and the chemical analysis data of the actinide and FP nuclides were obtained [33]. The accuracy of the ORIGEN2.1 burnup calculation

code was validated using these data. In addition, a criticality analysis was performed for the system modeled for an SF assembly alignment in a storage pool with an infinite array geometry of a single-fuel pin cell neglecting some features including the basket, using the KENO V.a code. The neutron multiplication factor was obtained for the system parameters with and without an axial burnup distribution, FPs, the effect of the irradiation history, and the fuel cooling time. These calculation results were arranged, and the "equivalent uniform burnup" was obtained such that its reactivity was equivalent to the reactivity when the neutron multiplication factor was calculated to give a conservative result using the average burnup of the actual fuel assembly, as well as the axial burnup distribution and the irradiation history variation. As a result, by using the "equivalent uniform burnup," it was simple and easy to calculate the neutron multiplication factor with a nuclide composition that was calculated assuming a uniform axial burnup distribution. The "equivalent uniform burnup" evaluation curves obtained in this way are shown in **Fig. 3.8** for the following four cases: **(case1)** actinides only (18 nuclides including the nine nuclides recommended by the NRC, as shown in section 4.4.1.1) with a cooling time of 0 years, **(case2)** actinides only, same as in case1, but with a cooling time of 30 years, **(case3)** actinides and FPs (15 nuclides including the 12 nuclides recommended in the Criticality Safety Handbook, as shown in section 4.4.1.2) with a cooling time of 0 years, and **(case4)** actinides and FPs, same as in case3, but with a cooling time of 30 years. The applicable scope of this burnup conversion chart is (15–50 GWd/t) according to the sample burnup range of the PIE data.

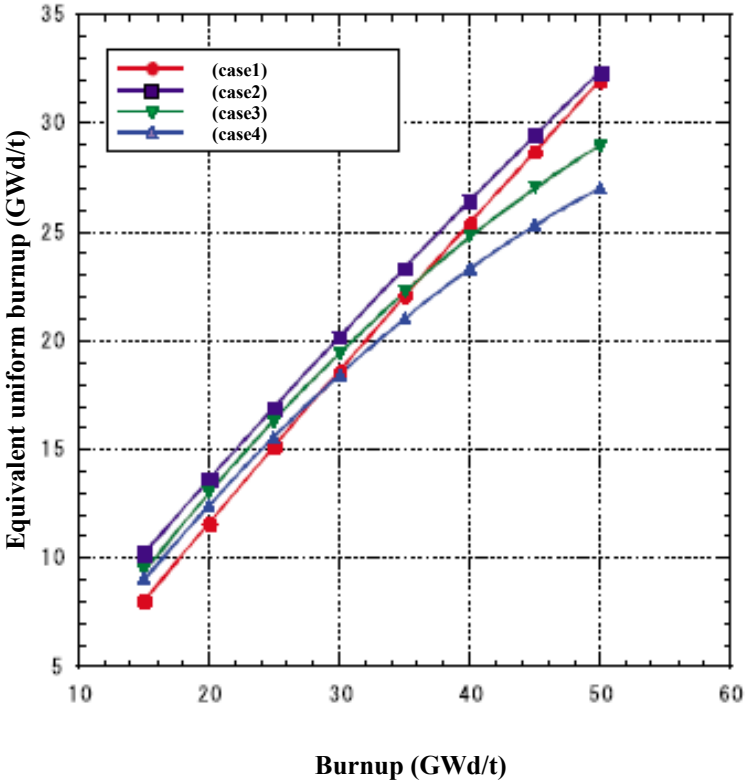

Fig. 3.8 Equivalent uniform burnup in an infinite arrangement geometry system of a single-fuel pin cell

3. Examples for Wet Transportation Packages [32]

Burnup and criticality calculations were performed for a wet transportation cask system containing 14 PWR SF assemblies, and the "equivalent uniform burnup" evaluation curve was obtained using a procedure similar to the one described previously. The result is shown in **Fig. 3.9**. Given the procedures for deriving the "equivalent uniform burnup," the obtained data can be applied to other types of SF transportation/storage systems without being affected considerably by the specific differences in the fuels. In fact, when the evaluation curves of **Fig. 3.9** are compared with the "equivalent uniform burnup" evaluation curves obtained for the cask in which more SF assemblies were stored (**Fig. 3.7**) (see *Examples for Dry Storage casks*, above), the data in **Fig. 3.9** provide an evaluation result at approximately the same level (or more conservatively) for the BWR SF system. The applicable scope of this burnup conversion chart is (15–50 GWd/t) according to the sample burnup range of the PIE data.

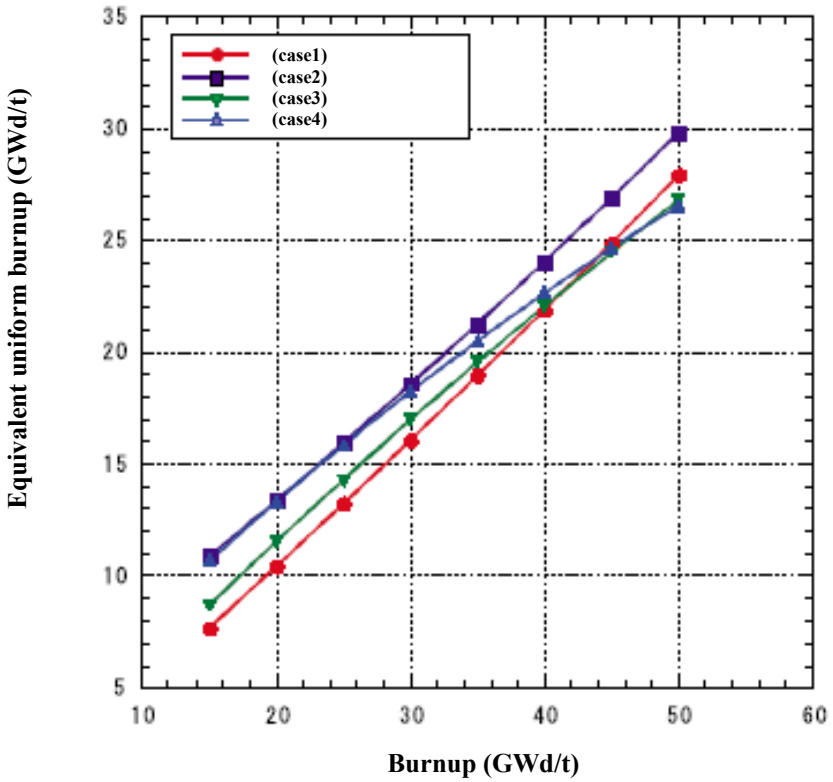

Fig. 3.9 Equivalent uniform burnup in a transportation package system for PWR SF

3.4.1.2 Equivalent Initial Fuel Enrichment

The "equivalent initial fuel enrichment" is the initial fuel enrichment that is determined to be conservative with regard to the effects of various error factors on the criticality calculation described in section 3.4.1.1, and is also defined as the initial fuel enrichment with a reactivity worth equivalent to a decrease in the reactivity worth due to the burnup of actual fuel assemblies. By performing a criticality safety analysis of the SF transportation/storage system using this "equivalent initial fuel enrichment," the "maximum neutron multiplication factor estimated by the calculation" can be obtained simply and easily without directly accounting for the axial burnup distribution of the fuel rods, the calculation accuracy of the burnup calculation code, and the irradiation history of the fuel.

The procedure for this method is as follows: i) the relationship between the burnup and reactivity worth,

as well as the corresponding initial fuel enrichment and neutron multiplication factor, is evaluated at burnup points with the effect of the various error factors when the type of PWR or BWR SFs and their transportation/storage systems are known in advance; ii) a conversion diagram is then prepared using these results; iii) finally, the "equivalent initial fuel enrichment" at a given burnup can be obtained for the SF storage system.

Evaluation curves of the "equivalent initial fuel enrichment" and conversion charts were obtained for both a single-fuel pin cell infinity arrangement system with a conservatively modeled storage pool of PWR (15 × 15) SF assemblies and a wet transportation cask system using the above-described procedure. The results are shown in **Figs. 3.10** and **3.11** [32], respectively. These curves were obtained on the basis of the PIE data of sampled PWR SF with an initial fuel enrichment of 4.2 wt%; however, they are also applicable to SF transportation cask systems with an initial fuel enrichment lower than 4.2 wt%. According to the sample burnup range of the PIE data, the applicable scope of this burnup conversion chart is 15–50 GWd/t. This "equivalent initial fuel enrichment" is a quantity basically dependent on the characteristics of the individual SF system, which is different from the "equivalent uniform burnup." Therefore, particular attention should be paid to its applicability range.

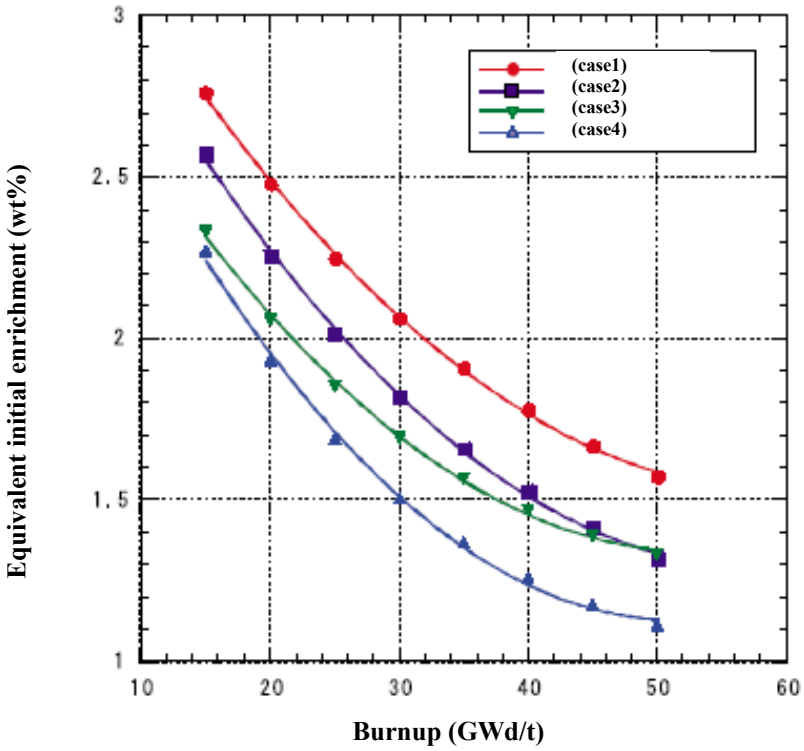

Fig. 3.10 Equivalent initial enrichment in an infinite arrangement geometry of a single-fuel pin cell

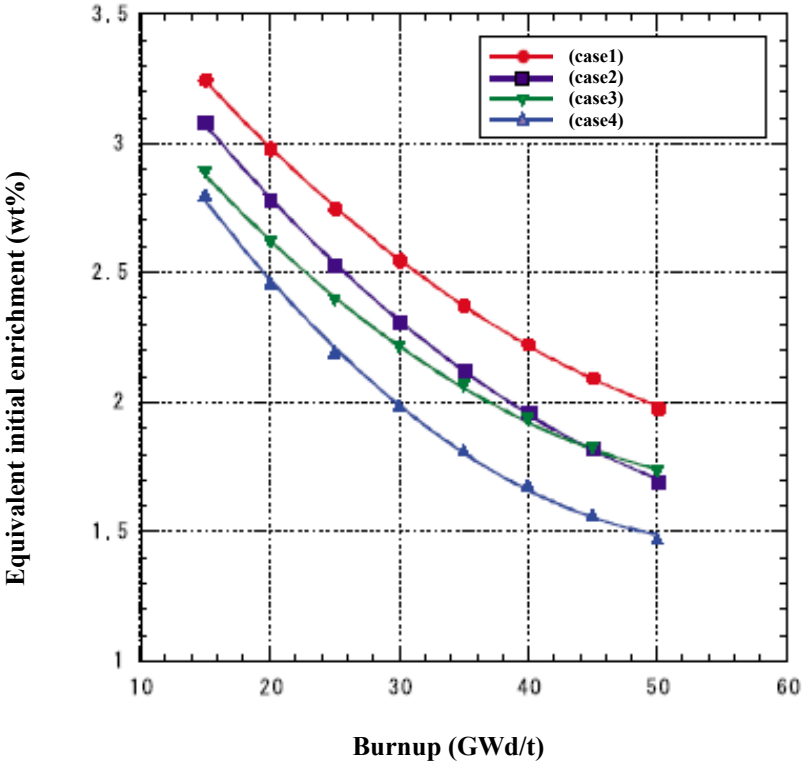

Fig. 3.11 Equivalent initial enrichment in a system of spent fuel transportation packages

3.4.2 Initial Composition, etc.

3.4.2.1 Initial Burnable Neutron-Absorbing Material

The reactivity worth of the fuel assembly without a burnable neutron absorber decreases monotonically with an increase in the assembly average burnup. In addition, the percentage of the decrease in the reactivity worth for a mixed oxide fuel assembly is less than that for UO_2 fuel.

Where a burnable neutron absorber such as Gd is contained in the fuel, there is a period of increasing reactivity worth along with the burnup. Therefore, when a credit is taken to account for the effect of the burnable neutron absorber, a criticality safety evaluation accounting for the maximum reactivity during the burnup is required. **Fig. 3.12** shows an example calculation for the change in the infinite multiplication factor k_∞ as a function of the burnup for a BWR fuel assembly with a constant moderator volumetric void fraction of 40% [34]. As shown in this example, k_∞ increases as the burnup increases up to about 8 GWd/t because of the decreasing effect of the Gd. This change might be called Gd credit in cases when the criticality safety evaluation accounts for this effect.

In addition, during the early irradiation stage when the burnable neutron absorber remains in the fuel, the accumulation ratio of the plutonium and the FPs is affected by the spectrum effect during irradiation. The reactivity of the fuel assembly is affected by the initial amount of the burnable neutron absorber, the number of fuel rods with absorbers, and their alignment in the assembly.

3.4.3 Irradiation History

3.4.3.1 Specific Power and Operating History

The power level of the fuel assembly during the operation influences the generation ratio of the actinides and FPs. Along with a specific power increase, the following occurs:

1. The neutron flux increases, and consequently, the neutron-nuclear reactions are enhanced.
2. The irradiation time decreases when assuming a constant burnup.

These two effects influence nuclear transformation processes with radioactive decay and neutron capture, therefore causing changes in the isotopic composition and reactivity worth. Most of the nuclides are not affected by a shortening of the exposure time (influence from item 2 above) because of their long half-life periods. However, fissile ^{241}Pu is affected by the exposure time because of its short half-life period of 14.4 years. That is, with a larger specific power, more of the ^{241}Pu nuclide will remain in the SF. On the other hand, according to the effect of item 1 above, the radioactive equilibrium atomic number density of ^{135}Xe, with its short half-life period, increases and consequently, the neutron spectrum is hardened in the relevant fuel, and the generation of ^{239}Pu is promoted by increasing the neutron resonance absorption in ^{238}U. Therefore, the burnup ratio of fissile ^{235}U decreases, and instead, the burnup ratio of fissile Pu increases. Considering all these effects together (in cases with the same discharged burnup conditions when the specific power increases during irradiation in the core), in general, the reactivity worth of the discharged fuel increases. Therefore, when evaluating the nuclide composition, the highest specific power during the operation must be postulated [30].

Moreover, there remains a pattern associated with the operation history as a parameter during reactor operation. The fuel is generally irradiated in three or four cycles, each for more than one year, until the SF is discharged from the core.

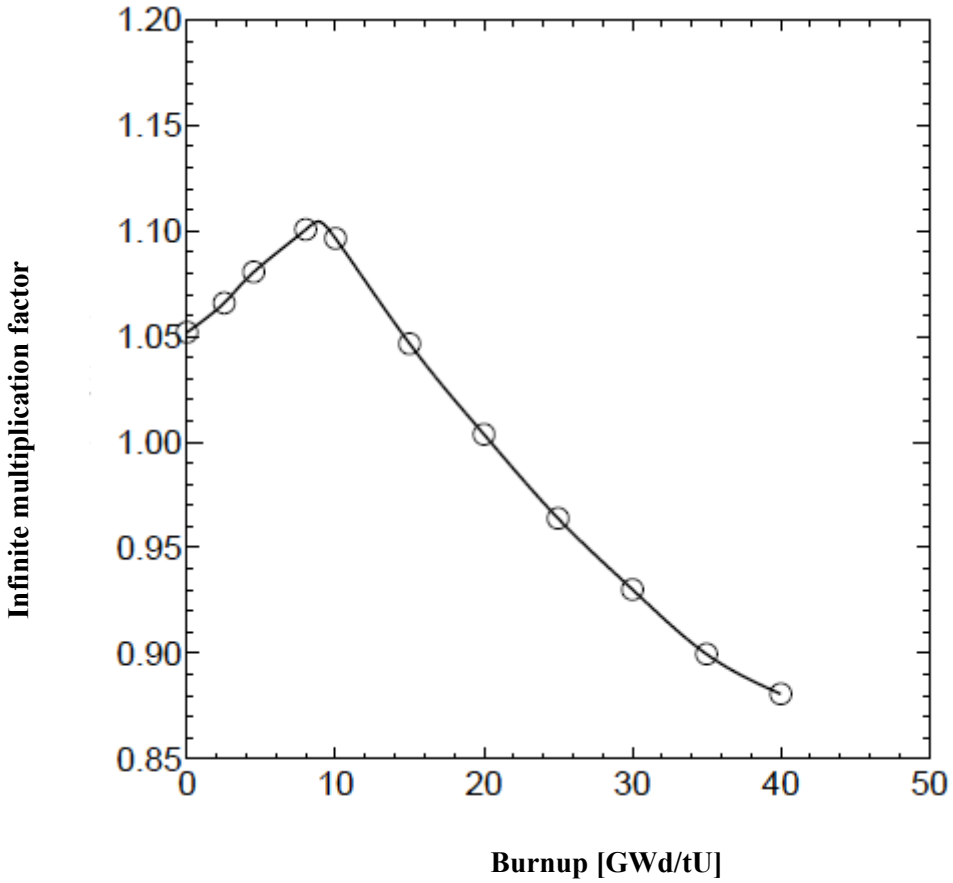

Fig. 3.12 Variation in the infinite neutron multiplication factor for the burnup of a BWR fuel assembly containing a burnable poison (moderator void fraction: 40%)

The fuel position in the core is relocated with shuffling during the irradiation period, and a shut-down period of 30–60 days is scheduled between each operation cycle according to the inspection and maintenance schedules for the reactor. This shutdown period is considered as a cooling period for the fuel, and this cooling period must be neglected or shortened for nuclide composition evaluation so that the most conservative results can be obtained for the criticality safety analysis. In cases where the discharged burnup and the irradiation time are assumed to be constant, the following results are confirmed in analysis calculation; concerning the fuel relocation by shuffling that means a specific power change for the fuel, the discharged fuel reactivity worth increases as the irradiation pattern with a high specific power shifts to the latter half of the cycle [35].

3.4.3.2 Fuel Temperature

The fuel pellet temperature increases by the nuclear heat generation with the reactor start-up, and the resonance absorption cross section of ^{238}U increases because of the Doppler effect. As a result, the number of fissile plutonium isotopes such as ^{239}Pu and ^{241}Pu increases because of the neutron resonance absorption of ^{238}U, and this increase contributes to the nuclear heating. In cases where the discharged burnup is the same, the amount of ^{235}U depletion (decrease) is suppressed by the increase in ^{239}Pu and

^{241}Pu, and the reactivity worth of the discharged fuel assembly is increased. Therefore, it is necessary to set a higher fuel temperature for the evaluation of the nuclide composition.

To calculate the average fuel pellet temperature, it is necessary to multiply the radial power peaking factor with the linear power rating of the fuel pellets. In this case, the fuel temperature is calculated for a pellet in the upper part of the fuel assembly, assuming a constant linear power rating. In order to determine the gap conductance and the thermal conductivity of the fuel pellet, it is necessary to assume a burnup that results in the maximum fuel temperature [30].

3.4.3.3 Coolant Temperature Distribution

The neutron spectrum of the core and fuel assembly is affected by the coolant density during the operation. In LWRs, when the reactor coolant pressure is kept constant during the operation, the coolant (moderator) density decreases with a temperature increase in the subcooled condition. As the density of the moderator decreases, the hydrogen atomic number density of the fuel rods decreases, the moderating effect of the coolant on the neutrons decreases, and the neutron spectrum then shifts to the higher energy side. Next, the generation of fissile plutonium increases because of an increase in the resonance absorption of ^{238}U. Consequently, the number of the nuclear fission by plutonium, instead of nuclear fission of ^{235}U, increases, and the reactivity worth of the discharged fuel assembly increases. Therefore, when evaluating the nuclide composition, an assumption that the coolant temperature has a higher value during the operation must be made [30].

It is well known that the coolant temperature of PWRs is 287.2°C–289.0°C at the coolant inlet and 321.0°C–325.0°C at the coolant outlet. The coolant temperature increases monotonically along with the coolant flow from the inlet to the outlet; therefore, in the evaluation of the nuclide composition, the core average outlet coolant temperature must be applied. In fact, a maximum core average outlet temperature that accounts for the coolant temperature fluctuation with a variation in the cycles is required [35].

3.4.3.4 Coolant Void Fraction Distribution

The reactivity worth of a BWR SF assembly that has reached a constant burnup level depends on variations in the void fraction during irradiation. BWRs exhibit certain axial void distribution during the operation. Because the void fraction differs largely between the upper and lower parts of the fuel assembly, considering the history of the void fraction is necessary for the nuclide composition evaluation. In fact, when applying the burnup credit, it is important to determine the limiting value of the void fraction history during irradiation in which the reactivity worth of the SF is estimated to be the maximum. Thus, for the criticality evaluation of a typical BWR SF system, the following procedures are generally performed: the burnable neutron absorber, which is loaded with the fresh fuel, is considered, and the void fraction history that provides the most severe results of the criticality analysis is adopted [1].

The distribution of uranium enrichment in a BWR fuel assembly varies in three dimensions, and it is much more complicated compared to that of PWR fuel assembly. In addition, the irradiation conditions vary because of the effect of the moderator void fraction and control rod insertion positions in the core. Therefore, it is necessary to perform a careful, individual evaluation of this effect.

3.4.3.5 Cooling Time

An example of a calculation that demonstrates the effect of the cooling time after irradiation on the infinite multiplication factor of an SF rod system in a water pool is shown in **Fig. 3.13** [36]. This example was obtained using the following procedure. First, the nuclide composition of the PWR fuel with an initial ^{235}U enrichment of 4.7 wt% and a burnup of 50 GWd/t was calculated using ORIGEN2.1 [3], with the PWR-UE library built into the code. Then, the infinite neutron multiplication factor was calculated with the SRAC code system [37]. The actinides included in the calculation consisted of U, Pu, Am, and Cm, and the FPs were comprised of 45 nuclides in addition to noble gases corresponding to a neutron absorption ratio of 95% or more of all FPs.

When the decay of the FPs with short half-life periods is complete, the transformation of ^{241}Pu to ^{241}Am and ^{155}Gd with a large neutron absorption cross section proceeds as the fuel cools. Therefore, except for a period during which the reactivity attains a peak because of the attenuation of the initial FP nuclides with short half-lives, the reactivity of the system decreases monotonically along with the cooling time. Thus, to evaluate the nuclide composition, it is necessary to consider the shortest cooling time according to the operation of the transportation and storage facilities.

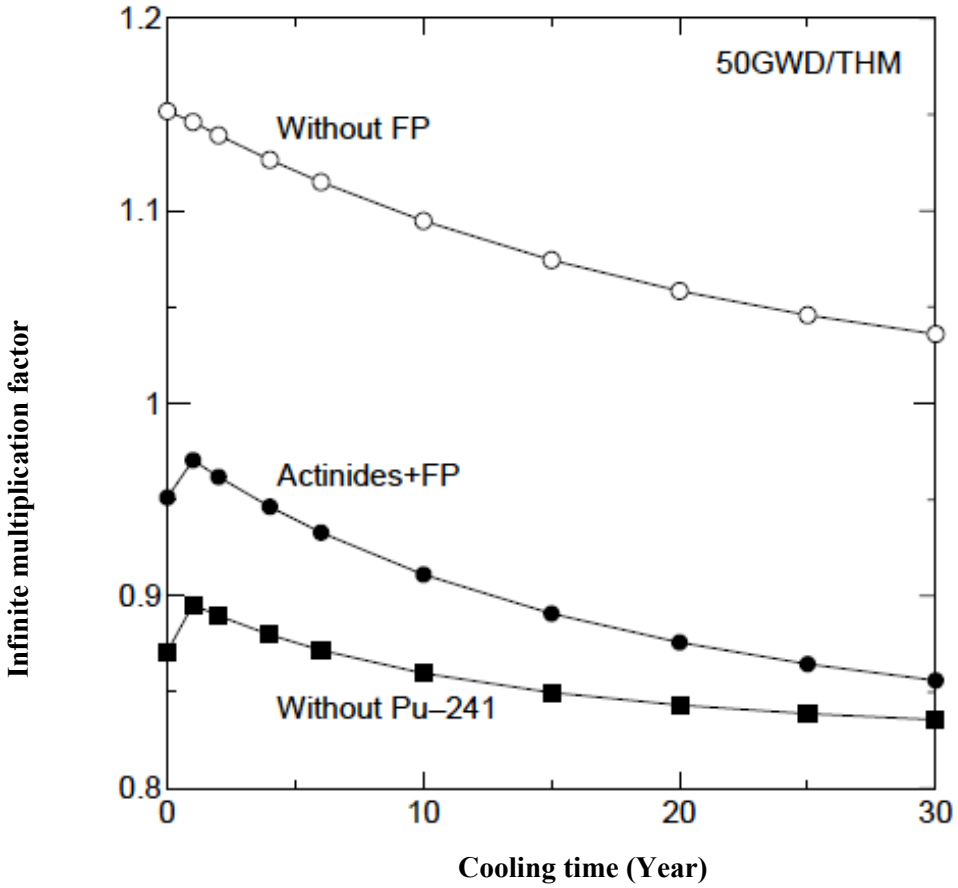

Fig. 3.13 Variation in infinite multiplication factor with cooling time [36]

3.4.3.6 Boron Concentration

Boron is added to the primary cooling system of PWRs to suppress the excess reactivity effect. The addition of boron, including the ^{10}B isotope, which has a large thermal neutron absorption cross section, is reduced along with the burnup so as to cancel the reactivity-decrease effect during the burnup. Consequently, the reactor is operated such that the boron concentration is the highest at the beginning of the cycle, and is then gradually lowered along with the burnup.

The thermal neutron absorption of boron in the coolant causes a hardening of the neutron spectrum in the core and fuel assembly, and the generation of fissile plutonium is promoted because of an increase in the ^{238}U resonance absorption. As a result, the boron induces the same energy with a decrease in ^{235}U consumption, and the discharged fuel has a higher reactivity worth when compared with cases where boron is not used under the same discharged burnup conditions. Therefore, when evaluating the nuclide composition, it is necessary to set a higher boron concentration in the coolant during reactor operation [30].

It is generally known that the boron concentration is approximately 1200 ppm during the initial stages of the cycle, and falls to less than approximately 50 ppm at the end of the cycle. The boron concentration is continuously adjusted during the irradiation period of the cycle along with the burnup. Therefore, when evaluating the nuclide composition, a calculated cycle-averaged boron concentration may be applied. In fact, because the boron concentration varies to some extent at each phase of the cycle according to variations in the cycle, the maximum average boron concentration accounting for these variations is required [35].

4. Criticality Evaluation Method

When the shape, dimensions, and material composition, including the isotopic composition of the SF, are given, the nuclear criticality safety is generally determined with a criticality calculation. The calculation methodology, outlines of the codes and verifications, factors to be considered for the critical calculation, and other examples are introduced. In addition, as explained in section 3.2, in burnup calculation codes, there are codes that evaluate group constants on the basis of the isotopic composition from the burnup calculation. These group constants, accounting for any variations in conditions such as the isotopic composition, geometrical conditions, and moderator conditions, instead of the isotopic composition, are transferred to the criticality calculation process for the subject system.

4.1 Summary of the Criticality Calculation Methods

In criticality calculations, the neutron multiplication factor, k_{eff}, an eigenvalue of the system, is obtained by solving a neutron transport equation. In the calculation of the neutron multiplication factor of a system having simple shapes, the one- and two-dimensional diffusion approximation methods, as well as the discrete ordinate (S_N) method, are widely used. In analyzing complex systems such as SF transportation packages and concrete storage containers with complex shapes, the three-dimensional Monte Carlo method is used. When a system is simple, a calculation code using two- or three-dimensional diffusion approximations may be used to minimize the computation time.

4.1.1 Diffusion Approximation Method

The diffusion approximation method describes the average behavior of the neutrons in a system; that is, it describes the macroscopic behavior of the neutrons. In this method, neutron density is treated as a statistically averaged value. When a calculation code utilizing the diffusion approximation method is to be used, the shape of the system must be normally represented by a regular, orthogonal coordinate; therefore, modeling must be performed carefully. With the diffusion approximation method, it is impossible to strictly handle the neutron movement orientation. The orientation variable in the transport equation is eliminated. Instead, the current J of the neutrons in the orientation of the coordinate axis is introduced according to Fick's Diffusion Law, as follows:

$$\boldsymbol{J} = -\boldsymbol{D} \cdot \boldsymbol{grad}\phi \quad , \tag{4.1}$$

where D is the diffusion constant and ϕ is the neutron flux.

Fick's Diffusion Law is valid when the neutron absorption in the system is low and most of the neutrons are scattered due to collisions with atomic nuclei. Namely, $\Sigma_a << \Sigma_s$ holds. Furthermore, the region in which Fick's Diffusion Law is valid is more than several times that of the neutron mean free path away from a studied system boundary or a neutron source. Therefore, this method must be carefully applied to a system of small dimensions, a system containing voids, a system containing locally strong neutron sources, or one with few neutron scatterers. For this reason, diffusion approximation codes are generally used for analyzing a large system. Where the diffusion approximation is not valid (i.e., the

periphery of the fuel elements), a correction given by the transport theory is required. This adjustment is called the transport correction.

In general, the diffusion approximation method is simpler than other calculation methods. Even for systems with two- or three-dimensional complex shapes, the calculation can be performed rapidly and with high accuracy, if a calculation code adopting the finite element method or the finite difference method is used. To obtain the neutron multiplication factor with high accuracy, a calculation code based on the strict theory may be used. However, diffusion approximation calculation codes are widely used for rough estimates of neutron flux. With a diffusion code, the neutron production distribution corresponding to the maximum eigenvalue (neutron multiplication factor) of the diffusion equation is obtained by an iteration method until its convergence. The determination of convergence should particularly be noted. Convergence is determined by comparing the relative error between the previous and present values in the iterative calculations. However, this error is not the difference from the true solution. Thus, if the required error range of the eigenvalues is 10^{-3}, the reference value for determining convergence in the calculation is preferably 10^{-4}, one order of magnitude smaller. In addition, when using a code in which the variables are discretized, as in the diffusion equation, the obtained neutron multiplication factor varies according to the numbers chosen for the spatial mesh points and the neutron energy groups; corrections for these variations may be made. These variations are often called the mesh effect and the group effect, respectively.

4.1.2 Discrete Ordinate (S_N) Method

With the S_N method, the numerical solution of the transport equation may be obtained directly. As with the diffusion approximation method, the S_N method handles neutron density as a statistical average value.

The S_N method can manage the movement orientation of neutrons as variables and in this method, the space domain is partitioned into meshes, and the domain of the orientation variables (angle) is divided into several orientations. The N in S_N represents the number of scattered azimuths. As N is set at a large value, the directional division is very fine; therefore, the calculation accuracy is improved, but the computation time increases significantly and the convergence rate diminishes. Similar findings have been stated for the number of mesh divisions of the space domain. The merit of the S_N method is that it applies to systems that the diffusion approximation method cannot handle, including systems having small dimensions and volumes, systems having voids, and systems having locally strong neutron absorption or generating sources. A one-dimensional S_N code is frequently used for surveying the trend of criticality or surveying parameters because of its short computation time.

With both S_N code and diffusion code, the distribution of the neutron production corresponding to the maximum eigenvalue (the effective neutron multiplication factor) of the transport equation is obtained by an iterative method until it converges. The determination of convergence should particularly be noted, which requires the same careful management as the diffusion approximation method.

When using an S_N code, care must be paid to the ray effect. The ray effect is a phenomenon resulting in the calculations of physically impossible oscillations of the neutron flux in a two or three-dimensional code caused by neutrons moving only along the lines defined by discretized angles. Care is required because this ray effect is more significant when the scattering cross section is small and the absorption cross section is large. In addition, attention must be given to the mesh and group effects, as described in the diffusion approximation method.

4.1.3 Collision Probability Method

In the diffusion approximation method and the S_N method described previously, differential equations with discretized variables are solved to obtain neutron flux distribution. These methods can give accurate solutions with a small number of mesh points when the spatial change in the neutron flux is gradual. However, when the mean free path of the neutrons is shorter than the size of the object system, the spatial changes in the neutron flux within the system are more complicated, and a greater number of mesh points are required to obtain accurate solutions using differential equations. In contrast, the integral transport equation is known to give an accurate solution using a small number of mesh points, because it does not rely on solving discretized differential equations. Because the kernel of the integral neutron transport equation is related to the probability of a neutron collision with an atomic nucleus after traveling a known distance, the integral diffusion equation is solved by the collision probability method. Because of the above-described feature, the solution to the integral transport equation is based on the collision probability method, and is often used to obtain an average effective cross section of a small region of large heterogeneity, such as fuel rods in a light water fuel assembly. A widely adopted practice is that, based on the average effective cross section of the small region obtained in this manner, the neutron flux distribution or eigenvalue of a larger system is obtained using a diffusion approximation, S_N method, or multigroup Monte Carlo method, which is described below.

4.1.4 Monte Carlo Method

Calculations with the Monte Carlo method have two advantages over other methods. One is this method's superior ability to handle fuel shapes, and the other is its ability to handle nuclear cross-section data. While the calculation of complex shapes that contain fuel assemblies (for example, transport packages) with one- or two-dimensional codes have some limitations, a Monte Carlo code consists of packaged input options for various geometries and combinations of these options. In addition, general geometry options can be applied to more complex shapes. Therefore, this method has high adaptability to fuel geometry input. Specific to Monte Carlo code, the behavior of the neutrons in the system is traced one by one, and thus, the value of the nuclear cross section data is used as the probability of a reaction when the neutrons collide with atomic nuclei. In a typical multi-group Monte Carlo code, such as KENO-V.a [45], the neutron energy spectrum is divided into multi-groups considering the limited computer memory capacity. However, the present computers with a larger memory capacity and faster processors allow some continuous-energy Monte Carlo codes to use a continuous neutron energy spectrum instead of a multi-group one. Monte Carlo code calculations are free of errors caused by the use of the difference approximation method, because they dispense with the mesh divisions for space and orientation that are used by the S_N code. However, this is a statistical calculation process that employs a random number sampling method, and therefore, with a Monte Carlo code, the calculation result lies within the range of the statistical error. To minimize this statistical error, it is necessary to increase the number of neutron histories (neutrons per cycle × number of cycles), and thus the calculation time. With a Monte Carlo code, the reaction process for each neutron is traced according to the given probability data: F, which represents the number of neutrons that result from fission; A, which represents the number of neutrons that are absorbed and annihilated; and L, which represents the number of neutrons lost as they leak from the system. These data are used to calculate the effective multiplication factor, k_{eff}, using the following equation:

$$k_{eff} = \frac{F}{A + L} . \tag{4.2}$$

In calculations using a Monte Carlo code, the spatial distribution of neutron production requires special attention. Because the fundamental mode of neutron production is not known at the start of the calculation, a flat or cosine distribution is arbitrarily assumed, and several hundred to several thousand

neutrons per cycle are normally generated according to the distribution. From the second cycle onward, the place where fission occurred in the previous cycle is saved in memory (which is assumed as the distribution of the neutron cycle for this time calculation), and this iteration is normally repeated for more than 100 cycles. As the cycles advance, the distribution of neutron production approaches the fundamental mode (i.e., a stable mode where the distribution changes little from one cycle to the next), and the distributions for the first several cycles are very different from those in the fundamental mode. Therefore, when the statistically averaged k_{eff} of the system is to be calculated, the k_{eff} values for the first several cycles are skipped. The procedure for selecting the neutron number, the number of calculated cycles, the number of skipped cycles, and the initial distribution of neutron production must be examined for each calculation, and the required accuracy and computing time must be considered. However, the final average k_{eff} must agree with the distribution of neutron production in the fundamental mode, and the standard deviation must be limited to an appropriate value. Most of the errors caused by the use of a Monte Carlo code are attributable to the calculation of the final k_{eff} of the system by summing the calculation results obtained before the neutron production distribution reaches the fundamental mode. This mistake will cause errors in the value of the cycle average for the k_{eff} that is calculated for each cycle, and cause errors in the estimate of the standard deviation of the calculated value of k_{eff}. There is no definite standard established for determining the initial number of skipped cycles; as the neutron production distribution approaches the fundamental mode, the fluctuation in the distribution for each cycle decreases, and thus, the fluctuation in the calculated value of k_{eff} for each cycle should also decrease. Accordingly, if the fluctuation in the calculated value of k_{eff} for each cycle lies within the error range, the distribution of neutron production may be determined to have approached the fundamental mode, and the calculation result for that cycle onward may be used for the calculation of the final statistical value.

The above situation applies when the number of neutrons produced per cycle is large enough; however, if this number is small, another problem may occur. For example, an unstable distribution of neutrons at each cycle may approach the fundamental mode, but eventually move away. In such an instance, a new calculation should be implemented by increasing the number of neutrons produced per cycle. Furthermore, when the number of neutrons per cycle is small, a large statistical error in the effective multiplication factor for each cycle will result. This problem is compounded with the necessity to increase the number of skipped cycles, and will increase the standard deviation of the effective multiplication factor obtained by averaging for the remaining cycles. When using a Monte Carlo code, if the number of histories or the calculation time is limited, the user should reduce the number of cycles and select a larger number of neutrons per cycle. To make the number of cycles as small as possible, the distribution of the initial neutron production should be made as close to the fundamental mode as possible. Theoretically, when the calculation involves a simple shape, the fundamental mode distribution can be anticipated, and even if a flat distribution is an input, it will converge into the fundamental mode after a few cycle times; thus, few problems will occur. In systems with complex shapes, however, many cycles are required to arrive at the fundamental mode. In this situation, a method may be used where the distribution is estimated in advance using diffusion approximation with a simplified model, and this estimated distribution is input into the Monte Carlo code calculation.

4.2 Criticality Calculation Codes

Table 4.1 shows a list of criticality calculation codes [6, 14, 16, 38–42, 44–47, 49]. To use these calculation codes, nuclear constants libraries are required. These libraries are in the form of multi-group and continuous neutron energy spectra, which are an aggregate of many energy points and are prepared by processing nuclear data files such as ENDF/B [50] and JENDL [17] for a specific code's format requirements.

Multi-group libraries are processed with a cross-section processing program that incorporates

resonance self-shielding and heterogeneous corrections to tailor the format of the cross-section data to the criticality calculation code. In this way, a cross-section data file for inputting into each criticality calculation code is prepared. A typical method for preparing an effective cross-section file that accounts for resonance self-shielding uses an f-table. A general method for correcting for the heterogeneous effect is to apply the Dancoff factor. Detailed, specific descriptions of the cross-section processing are omitted here, because they are beyond the scope of this document. However, such methods characterize the library processing code. One advantage of continuous energy libraries is that they do not require additional library processing for resonance self-shielding, nor heterogeneous effect corrections. Before calculating the effective multiplication factor using a criticality calculation code, the accuracy of the libraries that will be used must be fully assessed. The library accuracy has a direct influence on the accuracy of the neutron multiplication factor.

Table 4.1 Criticality Calculation Codes

Solution Method	Code Name	System Covered	Energy Group
Diffusion	CITATION [38]	3 D	Multigroup
S_N	ANISN [39]	1 D	Multigroup
	XSDRN [40]	1 D	Multigroup
	DOT [41]	2 D	Multigroup
	TWOTRAN [42]	2 D	Multigroup
	DANTSYS [43]	1-3 D	Multigroup
Collision Probability Method	SRAC95 [14]	2 D	Multigroup
Monte Carlo Method	KENO IV [44]	3 D	Multigroup
	KENO V.a [45]	3 D	Multigroup
	KENO VI [46]	3 D	Multigroup
	MULTI-KENO [47]	3 D	Multigroup
	MCNP4C [48]	3 D	Continuous
	MVP [6]	3 D	Continuous

With the development of faster computers, the Monte Carlo method has become a major method for the criticality evaluation code. Outlines of the typical Monte Carlo codes are described below.

4.2.1 MVP

MVP was developed on the basis of the continuous-energy Monte Carlo code VIM at JAERI. MVP has the advantage of precisely calculating the neutron spectrum in undivided resonance regions. Because MVP can deal with neutron and gamma transportation, it is available for both criticality and shielding calculations.

Development of this code was initiated with the aim of completing high-speed and high-accuracy computations using a vector computer in the early stages. At present, a higher computational capability

has been achieved compared to MCNP using a vector parallel computing method and workstation, and not only vector computers.

A multigrid representing method was introduced into MVP for achieving high-speed computations. Rectangular or regular hexagonal lattice domains can be defined, with a lattice comprised of cells. The interior portion of a cell is represented with an inherent coordination system, and additionally, the lattice domain can be further defined. To improve the calculation accuracy of the neutron multiplication factor, the most reliable k_{eff} is obtained using the maximum-likelihood method, which is obtained using six independent methods that are combinations of the two definitions of the neutron multiplication factor (weighting of the neutrons generated in nuclear fission/weighting of the total neutrons in a source and neutron production/neutron annihilation), as well as the three calculation methods for the reaction rate (collision, absorption, and trace length evaluation). In addition, the following characteristics are provided for user convenience.

1. For modeling of the geometrical configuration, a shape combination method is applied.
2. The multi-group energy calculation code GMVP is usable without modifying its algorithm.
3. Libraries for JENDL-3 and JENDL-3.2 have been prepared for nuclear data and provide measures for adding data for additional nuclides.
4. Users can request clarification from the developers and other users by accessing the mailing list.
5. A free format with the data name is prepared for the input data description. That is, the data are written in the format of "data name (value of data)," which gives freedom for input description and makes the input data easily viewed.

4.2.2 MCNP

MCNP is a continuous-energy Monte Carlo transportation calculation code developed by Los Alamos National Laboratory in the United States. It is characterized by the ability to continuously deal with energy, and is capable of transportation calculations for both neutrons and photons. Although a newer version of this code, MCNP4C, has been created, the features described below are from MCNP4B [49] because it is still widely used.

For criticality calculations with the MCNP code, the following points outline the adopted methodology: i) the neutrons are traced one by one; ii) the physical phenomena, including absorption and scattering, are analyzed; iii) the information, including the multiplication factor and prompt neutron lifetime, are accumulated; and iv) to carry out this methodology, a so-called non-analog Monte Carlo calculation is adopted, in which the neutrons are given a statistical weight, and the weighting is then decreased with a capture in the atomic nucleus, and finally, the life and death of the neutrons is determined with the Russian Roulette technique. MCNP can deal with arbitrary three-dimensional geometrical configurations, which is one of the specialities of the Monte Carlo method, and the domain surrounded by plain, spherical, and cylindrical faces are dealt with as cells. The calculation flow of the MCNP is as follows:

Nominal numbers of neutron histories (N) per cycle are specified, and different numbers (M) of neutrons are isotropically ejected, one by one, from the source at the beginning of the neutron history in each cycle. This approach allows for weighting of the source neutron, N/M, which is normalized as the

neutron histories of N, instead of M. The spatial production distribution of the fission neutrons in the initial cycle is specified with one of three methods that the user selects with the input data.

1. A three-dimensional coordinate system of x, y, z is specified (KSRC card),
2. A uniform distribution of the volume is specified (SDEF card), or
3. Stored fission neutron distributions in previous MCNP criticality calculations are utilized (SRCTP file).

The accumulated distribution is used in the subsequent cycle; this information is obtained by assuming the fission neutron production point as the collision point with atomic nuclei in the neutron transportation of the previous cycle. The initial I_p cycle (skipped number of cycles) is used for converging the spatial neutron production distribution with the correct distribution, and is not used for obtaining the averaged k_{eff} value of a given problem. Information related to k_{eff} and other reactor physics parameters is accumulated with neutron tracing after the next cycle.

MCNP transports the neutron by a standard random walk process within a geometrical configuration system, in addition to handling fission as a capture. The following four-step calculation is carried out at each collision point:

1. The evaluated value of the prompt neutron lifetime is accumulated using an estimated operator of the absorption and collision processes.
2. The evaluated value of k_{eff} for a fissionable medium is accumulated using three estimated operators (collision, absorption, and trace length).
3. η ($\geq$0) fission points for the fissionable medium are accumulated, where

$$\eta = W\nu_{bar}(\sigma_f/\sigma_t)(1/k_{eff}) + \text{random number} \quad (4.3)$$

   - W Weighting of the neutron
   - ν_{bar} Average number of neutrons generated by fission
   - σ_f Microscopic fission cross section
   - σ_t Microscopic total cross section
   - k_{eff} k_{eff} obtained in the previous cycle using an estimated operator of collision
4. The collided nuclides and the type of nuclear reaction are sampled after the above three steps. Fission is excluded from this sampling because fission is handled as a capture.

The averaged k_{eff} value at the end of each cycle is obtained by processing the three operators of the k_{eff} accumulation information, as explained in step 2, with the maximum likelihood estimation method. The fission neutron production distribution and the number of fissions, M, for the next cycle are calculated with the accumulated information in step 3. Consequently, M differs in each cycle, while the number of neutron histories per cycle N is constant. The weighting of the entire neutron is retained.

4.2.3 KENO V.a [45]

KENO V.a is a modification of the multi-group energy Monte Carlo criticality calculation program KENO IV [44], which was developed by Oak Ridge National Laboratory in the United States and is widely used throughout the world. Modified characteristics of this code are as follows.

1. Geometry input

   KENO V.a can specify the center of a cylinder, partial cylinder, sphere, or partial sphere for the available specialized shape in KENO IV. This ability allows for the application of the program to nonconcentric cylinders and spheres.

In addition, partial cylinders or partial spheres are not limited to semi-cylinders or hemispheres, respectively, and can be specified by input with the distance (ρ) from the center to a cut surface. This value, ρ, is given for the range from the positive radius amplitude (full cylinder or sphere) to the negative radius amplitude (zero volume of a cylinder or sphere, not existing). In addition, an array-of-arrays and "hole" options are newly provided in KENO V.a. The array-of-arrays option makes it possible to compose an array from other arrays, and the "hole" option allows for the arrangement of some UNIT (BOX TYPE) within another UNIT.

2. Supergrouping

KENO V.a has a super-grouping function for energy-dependent information, such as the cross section, neutron flux, and albedo. This function starts automatically when the memory capacity of the computer is not sufficient to retain all problems. KENO V.a determines the necessary numbers of a super-group for executing the problem. The numbers of a super-group are determined, and the energy dependence data are arranged in the super-group and written in the direct access super-group file. The merit of super-grouping is that an enormous problem is executable with a small computer.

3. Differential albedo

The differential albedo was developed to exclude the neutron tracing in a reflector, and the history begins again at the point where the neutron enters the reflector. The emitted energy and polar angle are selected with the junction density function depending upon the incident energy and polar angle. Weighting of the history is adjusted with the return of functions from the reflector on the basis of the incident energy and the polar angle. The differential albedo data corresponding to the Hansen-Roach 16-group energy structure are the only ones available in KENO V.a. However, they are applicable to problems for other energy groups in the new version by converting the energy structures of the albedo data.

In addition, the input data format for setting the parameter data has been modified from a plain card input format to a flexible input order.

4.3 Validation of Criticality Calculation Codes

The most essential requirement for criticality calculation codes is to calculate the neutron multiplication factor of a system accurately. The accuracy of the calculation results using a code is validated by comparison with the calculation results for the experimental data obtained for systems analogous to the subject system, or with the calculation results for certified standard problems that were obtained using a calculation code whose reliability has been verified with experimental data and calculation results.

Benchmark critical experiments using SF for the validation of criticality calculation codes for SF systems have rarely been implemented because they are difficult to complete. Therefore, critical experimental data for MOX fresh fuel, which is a mixture of uranium oxide and plutonium oxide, must be used as a substitute for the verification of a burnup credit at level 1 (actinide-only).

Currently, critical experimental data for fuel systems containing FPs are difficult to obtain; therefore, the available option is to use data generated from the reactivity measurements of simulated fuel containing natural FPs [51, 52]. Implementation of the same sort of experiments related to FPs is expected in the future, which will enable the verification of the criticality calculation codes for FPs. Moreover, obtaining benchmark critical experimental data for an SF is desirable in order to be able, sometime in the future, to prepare burnup credit applications on the level that accounts for the FPs contained in an SF. Data applicable for validation and an example of a validation for MOX are explained below.

4.3.1 Data for Validation

4.3.1.1 ICSBEP

The experimental data for the uranium–plutonium mixed fuel systems included in vol.6 of the OECD/NEA International Criticality Safety Benchmark Experiment Project (ICSBEP) are usable as the benchmarks for a system containing actinides [53]. Nine benchmark evaluations for a fuel rod array system are presented in **Table 4.2**.

Table 4.2 List of critical experiments for MOX evaluated in ICSBEP

Name	Title	Number of Data
MIX-COMP-THERM-001	Water-reflected mixed plutonium–uranium oxide (20 wt.% Pu) pins	4
MIX-COMP-THERM-002	Rectangular arrays of water-moderated UO_2-2wt% PuO_2 (8% ^{240}Pu) fuel rods	6
MIX-COMP-THERM-003	Rectangular arrays of water-moderated UO_2-6.6 wt.% PuO_2 fuel rods	6
MIX-COMP-THERM-004	Critical arrays of plutonium–uranium fuel rods with water-to-fuel volume ratios ranging from 2.4 to 5.6	11
MIX-COMP-THERM-005	Water-moderated mixed plutonium–uranium oxide pins, 4.0wt. % PuO_2 18% ^{240}Pu, natural uranium	7
MIX-COMP-THERM-007	Hexagonal arrays of water-moderated UO_2-2 wt.% PuO_2 (16% ^{240}Pu) fuel rods	27
MIX-COMP-THERM-008	Mixed oxide fuel pin lattice-2.0wt .% PuO_2, 24% ^{240}Pu, natural uranium	6
MIX-COMP-THERM-009	Mixed oxide fuel pin lattice-1.5wt. % PuO_2, 8% ^{240}Pu, depleted uranium	6
MIX-COMP-THERM-010	Mixed oxide fuel pin lattices in plutonium–uranium nitrate solution	11
	Total:	84

MIX-COMP-THERM-010 is an array system with arranged MOX fuel rods that has approximately 20% plutonium in a uranium–plutonium mixed solution. Because there are a few disclosed experimental systems that arrange fuel rods in the fuel solution, these experimental data are considered as the only data with adequate reliability.

A benchmark evaluation from 001 to 005 for a uranium–plutonium mixed solution system is shown in **Table 4.3.**

All isotopic compositions of these plutonium solutions have concentrations exceeding 90 wt% ^{239}Pu. All uranium is depleted uranium. In addition, experimental data for MIX-SOL-THERM-006 with six cases are planned for introduction. The data are characterized as having an isotopic composition of 20 wt% ^{240}Pu and containing gadolinium.

Table 4.3 List of Critical Experiments for MIX Solutions Evaluated in the ICSBEP

Name	Title	Number of Data
MIX-SOL-THERM-001	Critical experiments with mixed plutonium and uranium nitrate solution at a plutonium fraction of 0.2 and 1.0 in annular geometry	13
MIX-SOL-THERM-002	Critical experiments with mixed plutonium and uranium nitrate solution at a plutonium fraction of 0.2 and 0.5 in large cylindrical geometry	3
MIX-SOL-THERM-003	Water/polyethylene reflected cylinders of plutonium/natural uranium nitrate	10
MIX-SOL-THERM-004	Critical experiments with mixed plutonium and uranium nitrate solution at a plutonium fraction of 0.4 in a small cylindrical geometry	9
MIX-SOL-THERM-005	Critical experiments with mixed plutonium and uranium nitrate solution at a plutonium fraction of 0.4 in slab geometry	7
MIX-SOL-THERM-006 (Not Included Now)	Mixed uranium-plutonium (29.87%) nitrate solutions poisoned with gadolinium	(6)
	Total:	42 (+6)

4.3.1.2 Data Adopted in the Nuclear Criticality Safety Handbook

The following benchmark experiments related to MOX are used for validation of criticality calculation codes, in addition to the aforementioned experiments registered for the ICSBEP.

- UK AEW, Critical experiment with a homogeneous mixture of plutonium and natural uranium oxides with polystyrene [54]
- USA PNL, Critical experiment with a homogeneous mixture of plutonium and natural uranium oxides with polystyrene [55–57]
- USA PNL, Criticality Experiments with Fast Test Reactor Fuel Pins [58]

4.3.1.3 Data Adopted for the PWR Burnup Credit Design Accounting for Actinides Only

The following benchmark experiments related to MOX are used for validation of criticality calculation codes, in addition to the experiments registered for the ICSBEP.

- PNL Mixed Fuel Pin (MOX,UO_2) Critical Experiments [59]
- Plutonium Utilization Program (PUP) Critical Experiment [60]

4.3.2 Validation Examples

4.3.2.1 Validation Examples Described in the Nuclear Criticality Safety Handbook

The estimated criticality multiplication factors and the estimated criticality lower limit multiplication factors were obtained using the nuclear criticality safety evaluation system JACS (combination of the 137 Multi-group Constants Library MGCL and the Monte Carlo method neutron transportation calculation code KENO IV) by analyzing the criticality benchmark experiments for each fuel system, a simple shape/normal shape, a homogeneous/heterogeneous system, and error assessment and statistical processing, as described in the Nuclear Criticality Safety Handbook [1].

The estimated criticality multiplication factor (analysis value of the multiplication factor using critical experimental data makes it possible to determine that the system is considered to be critical) for the homogeneous and heterogeneous MOX system groups, both with reflectors, equal nearly 1.0, and the overestimation of the results is ca. 1%. The estimated criticality lower limit multiplication factor (obtained after statistical processing of the benchmark calculation results and determination of the subject system as subcritical when the criticality calculation results are below this value) results as 0.98 in every case.

4.3.2.2 Validation Example from the United States for Burnup Credit Considering Actinides only

The US DOE proposed that a design method with a burnup credit [30] that only considers actinides in the transportation of PWR SF, and the NRC completed the regulatory review of this method. In the proposal from the DOE, the following verification concerning the criticality evaluation was implemented using a CSAS/CSASXH sequence in SCALE 4.2 (criticality calculation sequence mainly with KENO V.a), as the representative example of the calculation code.

The validation method involves obtaining the criticality lower limit multiplication factor by analyzing many critical experiments and statistical processing results using a Lower-Tolerance-Band Technique (Meanwhile, the maximum permissible multiplication factor is set by subtracting a control margin of 5 % $\Delta k/k$ from the criticality lower limit multiplication factor). The validation is performed independently for the UO_2 and MOX systems, and the severest result is adopted.

Followings are the critical experiments used for validation of the MOX systems:

- EPRI Clean Critical Experiment
- Saxton MOX Critical Experiment
- PNL Mixed Fuel Pin (MOX,UO2) Critical Experiment
- Plutonium Utilization Program (PUP) Critical Experiment
- Tank-Type Critical Assembly (TCA) MOX Critical Experiment

The number of cases for each experiment is 6, 6, 1, 23 and 11 respectively, for a total of 47.

4.3.2.3 International Benchmark for Burnup Credit Criticality Safety

The Criticality Safety International Benchmark for validation of burnup/criticality calculation codes is carried out for burned fuel systems hosted by OECD/NEA. Specialists at the organizations are delegates from member states that implement the nuclide production and depletion calculation of PWRs and BWRs, as well as the neutron multiplication factor calculations of SF assemblies and transportation/storage systems, using their own analysis methods for standard problem sets corresponding to several phases. The delegates provide their calculation results and compare and discuss them [61–66]. On the basis of the results, the effect of minor actinides and FPs on the calculated neutron multiplication factor, as well as the sensitivities related to burnup and burnup distribution, is evaluated with international cooperation. **Table 4.4** shows the progress of the benchmark study [65]. The order of the calculation errors for criticality and burnup calculation codes applied to a criticality safety evaluation of burned fuel is estimated with reference to these results.

4.3.2.4 Additional Examples of Evaluations

Recently, for validating a criticality calculation code system combined with the continuous-energy Monte Carlo calculation code MCNP4A and the JENDL-3.2 or ENDF/B-V library, data for a critical experiment on light water-moderated UO_2-6.6 wt% PuO_2 have been analyzed. The experiment was performed in 1965 using a Critical Reactor Experiment (CRX) facility at the Westinghouse Reactor Evaluation Center (WREC) in the United States [67–70]. Although these critical experimental data are dated, they are of high quality and were re-evaluated recently in the OECD/NEA International Criticality Safety Benchmark Evaluation Project (ICSBEP). The neutron multiplication factor was calculated as 0.996–1.007 when using each library [71]. The critical experiment for light water-moderated UO_2-3 wt% PuO_2 using the TCA of JAERI has also been performed, and the data were analyzed with a combination of MCNP4A and JENDL-3.2 [72–74]. An average value of 1.000 ± 0.005 for four cases was obtained with the distance betwen the fuel rods and the size of the lattice as the parameters [75]. This experiment was performed in 1971 and was recently re-evaluated and quality-assured by the ICSBEP, as the aforementioned WREC experimental data.

Table 4.4 List of Benchmark Subjects Presented by the Criticality Safety Benchmark Group of the OECD/NEA [65]

Benchmark	Primary objective	Status
Phase I-A	Examine effects of seven major actinides and 15 major fission products for an infinite array of PWR rods. Isotopic composition specified at 3.6 wt.% ^{235}U at 0, 30, 40 GWd/MTU and at 1 and 5 yearscooled.	Completed 13 cases. Ref. [61]
Phase I-B	Compare computed nuclide concentrations for depletion in a simple PWR pin cell model, comparison to actual measurements at three burn-ups (27.34, 37.12, and 44.34 GWd/MTU). Comparisons made for 12 major actinides and 15 fission products for each burn-up case.	Completed 3 cases. Ref. [62]
Phase II-A	Examine effect of axially distributed burn-up in an array of PWR pins as a function of initial enrichment, burn-up and cooling time. Effects of fission products independently examined. Isotopic compositions specified.	Completed 26 cases. Ref. [66]
Phase II-B	Repeat study of Phase II-A in a 3-D geometry representative of a conceptual burn-up credit transportation container. Isotopic compositions specified.	Completed 9 basic cases and 2 accident configurations. Ref. [63]
Phase II-C	A complementary study on the sensitivities due to different burnup axial profiles across the full range of burnups.	Proposed.
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.	Report for 22 cases being finalised. Ref. [64]
Phase III-B	Compare computed nuclide concentrations for depletion in a BWR pin-cell model.	Draft results in review. Ref. [64]
Phase IV	Investigate burn-up credit for MOX spent fuel.	In progress.

4.4 Factors to be Considered in Criticality Calculations and Examples of Evaluation

4.4.1 Nuclides Considered for Criticality Safety Evaluation

When a burnup calculation code is used, the SF isotopic composition of dozens to hundreds of nuclides is obtained depending on the performance of the code used. Which nuclides should be included (or disregarded) for the calculation is an important point for criticality evaluation.

In order to maintain a conservative estimate for the neutron multiplication factor, influential nuclides are generally selected on the basis of the following criteria.

1. All fissile material should be included.
2. Nuclides absorbing neutrons may be omitted.

Concerning item 2, any nuclide whose existence is doubtful must not be considered to have conservative reactivity. The following sections describe actinides and FPs that can be considered in the criticality calculations.

4.4.1.1 Actinide Nuclides

The Burnup Credit Criticality Safety International Benchmark calculation explained in section 4.3.2.3 selected the actinides ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, and ^{237}Np. The burnup credit application is limited to actinides approved by the NRC, which include ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{241}Am. The reason for the exclusion of ^{236}U and ^{237}N in the latter case is that the reactivity decrease by ^{236}U is not proven, and that a significant difference exists between the calculated and experimental values for ^{237}Np.

The selected actinides in this guide are ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{241}Am, the same as those selected by the United States.

4.4.1.2 Fission Product (FP) Nuclides

In this guide, the working group assumed that an actinide-only burnup credit will be adopted in Japan; however, the additional adoption of FPs as burnup credit will be considered in the future. The consideration of FPs in a burnup credit calculation induces lower reactivity than that in actinide-only calculations. Therefore, the selection of FP nuclides should be carefully determined for criticality evaluations.

The amount of neutron absorption due to FP nuclides in the SF with a 30 GWd/t burnup discharged from a PWR was calculated with the ORIGEN2 Code [7, 15], and the results are listed in descending order of neutron absorption in **Table 4.5**.

However, it is not beneficial to consider all the nuclides in the table in a criticality safety evaluation for

an SF transportation/storage system or reprocessing facility, because of the calculation accuracy of the FP formation and the cumbersome compilation of the cross section and its accuracy evaluation. Therefore, 10 or so FP nuclides were selected to be considered in the criticality safety evaluation. The FP nuclides in the following categories were excluded from the evaluation.

1. Nuclides with small neutron absorption

FP nuclides whose contribution to the total neutron absorption is small are excluded. As a guideline, FP nuclides to be excluded are selected such that the sum of the absorption of the selected nuclides amounts to not more than approximately 30% of the total neutron absorption of all FP nuclides.

2. Nuclides with short half-lives

FP nuclides that reduce their masses significantly by decaying after the SF are discharged from a reactor (FP nuclides having half-lives of less than 30,000 years) are excluded. For nuclides with half-lives of more than 30,000 years, 99.9% of those nuclides are still present 40 years after the fuel was discharged from a reactor. Therefore, an increase in the reactivity because of their decay is negligible. For SFs that have been cooled outside of a reactor for several years, this half-life period criterion is applicable to only Pm-147 (half-life period of 2.6 years) among the top 10 nuclides of neutron absorption; however, Pm-147 is neglected because its impact is small.

3. Gaseous or volatile nuclides

Gaseous FP nuclides include Kr and Xe, and volatile FP nuclides include Br and I. These nuclides (elements) are released from the pellets of a fuel and distributed in the gas plenum (or released to the outside if the fuel rod is damaged); therefore, it is unlikely that these nuclides would stay in the fuel.

4. Semi-volatile nuclides

Nuclides such as Rb, Te, Se, Cd, Sb, and Ag form semi-volatile elements or compounds. These nuclides (elements) are deposited on the inside surface of the cladding; however, they may be released from the inside surface because of the impact when handling the fuel, and therefore, it is unlikely that these nuclides would stay in the fuel.

Thus, the selected FP nuclides are a total of 12 (^{149}Sm, ^{103}Rh, ^{143}Nd, ^{133}Cs, ^{99}Tc, ^{152}Sm, ^{155}Gd, ^{145}Nd, ^{147}Sm, ^{95}Mo, ^{153}Eu, and ^{150}Sm). **Fig. 4.1** shows the ratio of the absorbed neutron amounts for each FP nuclide above to all FP nuclides as a function of cooling time, for the abovementioned system of PWR SF assemblies arranged in water. The selected 12 nuclides account for more than 65% of the amount of the absorption for the FPs

Because FP nuclides such as Rh, Tc, and Mo have the tendency to form an insoluble residue in solution, it is expected that the neutron absorbing effects of the nine FP nuclides, excluding ^{103}Rh, ^{99}Tc, and ^{95}Mo from the above-described 12 nuclides, can be considered in subsequent solution processes in a reprocessing facility.

4.4.2 Burnup Distribution in Fuel Assemblies

As the fuel burnup proceeds, the central region is more easily burnt than the end regions, which causes an axial burnup distribution with a peak in the center. Criticality safety evaluations for SF transportation/storage systems are sometimes analyzed with the assumption of a uniform axial burnup distribution for simplicity and convenience.

The working group on the Burnup Credit Criticality Safety International Benchmark of the OECD/NEA has demonstrated the significant difference in the analyzed results for the neutron multiplication factor with and without accounting for burnup distribution for PWR SF systems modeled with a single pin cell. The dependency on the axial burnup distribution is confirmable in **Fig. 4.2.**

Table 4.5 Neutron Absorption Contribution (%)

Order	Time discharged		0.5 years later		4 years later		10 years later		30 years later	
1	Xe135	18.78	Sm149	12.94	Sm149	12.96	Sm149	12.8	Sm149	12.58
2	Rh103	9.02	Rh103	12.09	Rh103	12.16	Rh103	12.01	Rh103	11.8
3	Nd143	8.53	Nd143	10.64	Nd143	10.66	Nd143	10.53	Gd155	11.6
4	Sm149	7.02	Cs133	7.17	Cs133	7.18	Gd155	9.04	Nd143	10.34
5	Pm149	6.13	Xe131	7.07	Xe131	7.08	Cs133	7.09	Cs133	6.97
6	Cs133	5.9	Pm147	6.81	Gd155	5.25	Xe131	6.99	Xe131	6.87
7	Xe131	5.77	Tc99	5.17	Tc99	5.18	Tc99	5.11	Tc99	5.02
8	Tc99	4.28	Sm151	4.71	Sm151	4.59	Sm152	4.49	Sm152	4.42
9	Sm151	3.87	Sm152	4.54	Sm152	4.55	Sm151	4.33	Sm151	3.65
10	Sm152	3.78	Eu153	3.01	Eu153	3.02	Eu153	2.98	Eu153	2.93
11	Nd145	2.49	Nd145	2.99	Nd145	3	Nd145	2.96	Nd145	2.91
12	Eu153	2.48	Mo95	2.31	Pm147	2.7	Sm147	2.44	Sm147	2.53
13	Sm150	1.7	Sm150	2.04	Mo95	2.4	Mo95	2.37	Mo95	2.32
14	Pm148m	1.65	Ag109	1.63	Sm150	2.04	Sm150	2.02	Sm150	1.98
15	Mo95	1.64	Eu154	1.58	Sm147	1.93	Ag109	1.61	Ag109	1.58
16	Eu155	1.39	Eu155	1.56	Ag109	1.63	Ru101	1.54	Ru101	1.51
17	Eu154	1.37	Ru101	1.55	Ru101	1.56	Pr141	0.95	Eu151	1.39
18	Ag109	1.35	Pr141	0.96	Eu154	1.2	Pd105	0.79	Pr141	0.94
19	Ru101	1.29	Cs134	0.93	Pr141	0.97	Eu154	0.73	Pd105	0.77
20	Cs134	0.92	Gd155	0.9	Eu155	0.95	Kr83	0.73	Kr83	0.71
21	Rh105	0.87	Sm147	0.89	Pd105	0.8	La139	0.58	La139	0.57
22	Pm147	0.84	Pd105	0.8	Kr83	0.74	Zr93	0.57	Zr93	0.56
23	Pr141	0.76	Kr83	0.73	La139	0.59	Pm147	0.55	Pd108	0.52
24	Pd105	0.66	La139	0.58	Zr93	0.58	Pd108	0.53	Cs135	0.42
25	Kr83	0.61	Zr93	0.58	Pd108	0.54	Eu151	0.51	I129	0.42
26	Sm147	0.54	Pd108	0.54	Cs135	0.43	Cs135	0.43	Gd157	0.41
27	La139	0.49	Cs135	0.43	I129	0.43	I129	0.43	Cd113	0.38
28	Zr93	0.48	I129	0.43	Gd157	0.43	Gd157	0.42	Mo97	0.38
29	Pd108	0.45	Gd157	0.43	Cd113	0.4	Eu155	0.41	Pd107	0.3
30	Cs135	0.36	Cd113	0.4	Mo97	0.39	Cd113	0.39	Nd144	0.26

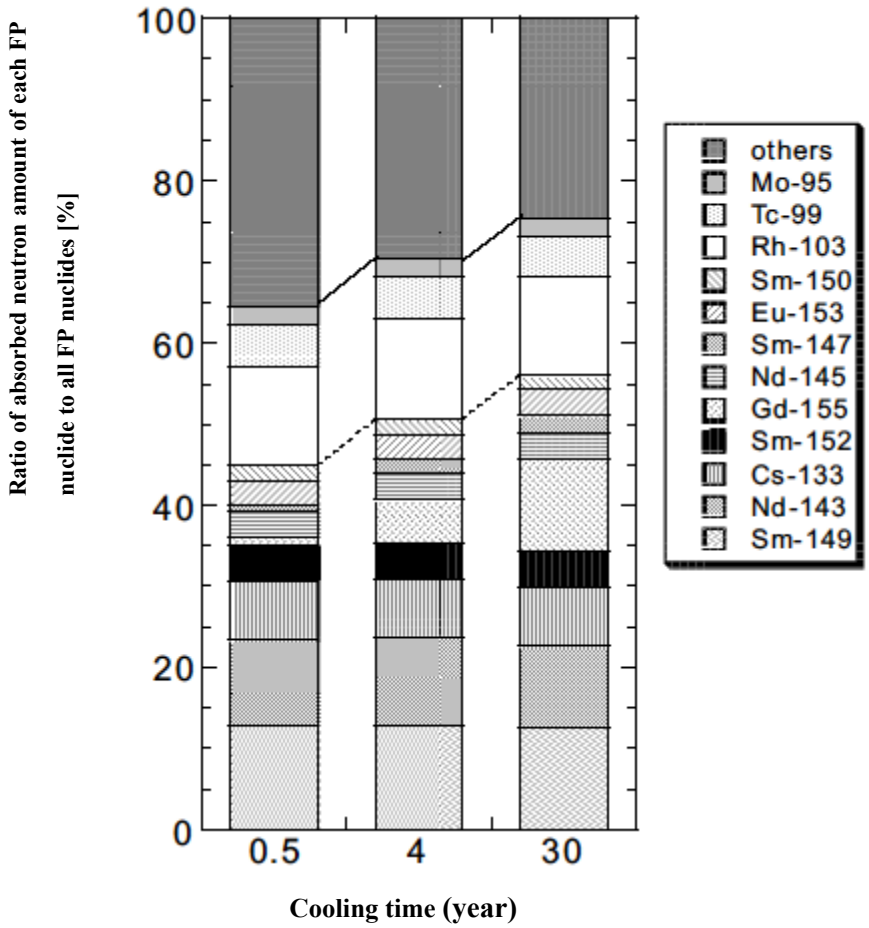

Fig. 4.1 Ratio of a number of absorbed neutrons for each FP nuclide to that of all FP nuclides (contributions of the 12 nuclides are on the continuous line, and those of the 9 nuclides excluding ^{103}Rh, ^{99}Tc, and ^{95}Mo are on the dotted line)

Thus, in evaluating the criticality safety of PWR SF with uranium enrichment below 5 wt%, postulating a uniform distribution with an average burnup is adequate for a burnup below 20 GWd/t, but the effect of the burnup distribution must be considered for a burnup greater than 20 GWd/t. The above statement is valid when the effect of FP production is considered. When evaluating the criticality safety without this effect, however, the difference caused with or without the burnup distribution would be relatively small, so a uniform distribution may be assumed. Because this example assumes a symmetric axial profile, a treatment based on an asymmetric analysis is required if an asymmetric burnup profile is suspected.

The isotopic composition considering the burnup distribution can be calculated directly with multidimensional burnup calculation codes; however, using a code that lacks multidimensional handling requires the calculation of a composition evaluation for various burnup regions beforehand.

A database concerning PWR SF axial burnup distribution, based on substantial data from the Yankee reactor in the United States [76], is useful when referring to the burnup distribution. A database concerning quadrant burnup distribution in the horizontal direction of SF assemblies has also been developed and provides deviation values for each burnup region [77].

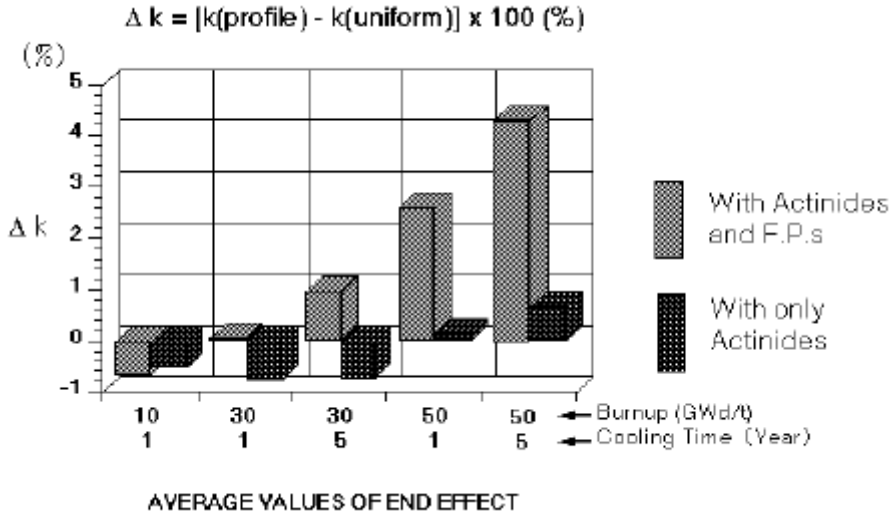

Fig. 4.2 Effect of axial burnup distribution (End effect of infinite array system with a single pin cell) [66]

4.4.3 Management Conditions [78]

4.4.3.1 Arrangement, Shape, Temperature, and Moderator Density

The arrangement, shape, temperature, and moderator density during SF transportation/storage usually differ significantly from those during irradiation in the reactor. The criticality safety evaluation requires the most extreme conditions for the reactivity, for normal as well as abnormal operation. In particular, the most appropriate moderator condition in SF assemblies (including abnormal operation) should be selected, much like the assumption of fresh fuel.

4.4.3.2 Neutron Absorbers

The capability of a fixed neutron absorber used for any component can be accounted for when the absorption effect is preserved during its in-service period, including normal and abnormal reactor conditions. The effect of a neutron absorber dissolved in a storage pool (e.g., boron) is admissible when the effect is confirmed periodically during the service term of a facility.

4.4.3.3 Reflection Conditions

Reflection conditions should be determined to show the highest reactivity considering normal and abnormal states. Geometrical and configuration models with a 20 cm water reflector surrounding a container are applicable for the criticality safety analysis of transportation/storage containers. Other models that have been assured representing the actual geometry with an appropriate experimental verification and rational explanation are also applicable for setting up the reflection conditions.

5. Burnup Credit Implementation

Initially, the criticality safety of an SF transportation/storage facility is guaranteed at the design stage with an evaluation procedure, as explained in previous chapters. Then, each facility or system is fabricated and constructed in order to conform to the design conditions, and as a result, the facility is operated such that criticality safety is maintained during actual operation.

Burnup of each fuel assembly and conservative estimates of the conditions used in the evaluation of the isotopic composition are required for implementation of burnup credit for the operation of the facility. Proof of the validity of the fuel selection must be confirmed in independent and redundant procedures. Confirmation of the burnup is carried out either by actual measurement with a burnup monitor or by identification with reactor management data whose accuracy has been verified.

5.1 Verification of Burnup before Spent Fuel Loading

The application of a burnup credit requires burnup verification when accepting SFs. This verification is performed either using a qualitative/quantitative measurement, confirmation of the reactor operational record, or a combination of both.

5.1.1 Verification with a Burnup Monitor

The specific setup of the burnup monitors used for SF acceptance depends on the safety margin selected for the criticality safety evaluation and/or the degree of conservative measures adopted for the evaluation. Accordingly, the accuracy of the burnup measurement depends on the degree of reactivity gain expected by introducing a burnup credit.

The following measurements with a burnup monitor are performed depending on the degree of conservativeness taken for the evaluation [79, 80].

1. When the burnup required to meet the reference value of reactivity is estimated at less than the minimum burnup assured by a one-cycle irradiation, then qualitative burnup monitors (such as gamma scanners) are enough to show that the subject fuel is actually irradiated. Burnup verification using a simple burnup monitor is appropriate in this case, and fills the provision against an error in fuel selection. Verification via a visual observation of the cladding tarnish or Cerenkov emission is also permissible for distinguishing between the burned and fresh fuel.
2. When the required burnup is larger than the burnup attained in a one-cycle irradiation, the measurement is generally performed using a passive quantitative burnup monitor (detailed burnup monitor) with high accuracy in order to measure the neutrons emitted from the actinides or gamma rays emitted from the FPs. At the same time, the measurement accuracy must be demonstrated for the burnup determination.

5.1.2 Verification of Burnup by Surveying the Operating Record

The core management code calculates and records the fuel assembly burnup during fuel burning in a reactor, and the method that involves identification of the record by a third party is "Verification of Burnup by Surveying the Operating Record." The verification of burnup using the reactor operational record has been shown to confirm the burnup design within the shielding and heat removal design of SF storage pools and transportation packages (this fact does not automatically assure the application of a burnup credit to the design).

Burnup verification using a monitor is the premise for a burnup credit application accepted in France and the United States. However, some examples exist where the burnup was confirmed with reactor operational records when the safety margin was predicted far enough in advance to assure the criticality safety evaluation in cases that did not apply a boron credit in the pool water [70, 80]. Burnup verification solely with reactor operational records requires verification of the method and the accuracy of determining the burnup of each fuel assembly.

5.1.3 Outline of Burnup Monitors

5.1.3.1 FORK System

The FORK system [81], developed at Los Alamos National Laboratory in the United States, measures the neutron flux in the peripheral area of the SF assemblies with a passive neutron detection method that utilizes a neutron source, for example, ^{244}Cm, in the SF, and then determines the fuel burnup by referring to the measured neutron flux and the initial enrichment data in the fuel specifications.

5.1.3.2 PYTHON System

The PYTHON system [82], developed at CEA-Cadarache in France, obtains the burnup distribution extending the entire length of the fuel assembly by measuring the neutron and gross gamma distributions with a passive neutron detection method utilizing the neutron source in the SF.

5.1.4 Burnup Monitor Adopted at the Rokkasho Reprocessing Plant

The burnup measurement device installed in the SF acceptance system of the Rokkasho reprocessing facility in Japan determines the burnup by measuring the values of neutron and gamma rays emitted from the SF. On the basis of this burnup, the decreased amount of ^{235}U by fission is evaluated, and the residual enrichment of the SF is obtained [83].

The burnup measurement device in the Rokkasho reprocessing facility has 40 ionization chambers arranged vertically on both sides of the fuel assembly (20 chambers per each side), which detect gamma ray doses, two germanium semiconductor detectors for gamma ray detection at a specified energy, and four fission counters arranged in the center, which detect neutrons. The germanium semiconductor detectors measure the gamma ray emission rates from ^{137}Cs, and the fission counters measure neutron emission rates from ^{244}Cm. The burnup in the central part of the SF is then calculated using the

relationship between these radiation emission rates and SF burnup. The vertical relative burnup distribution of the SF is obtained from the gamma-ray distribution measured by the many ionization chambers arranged in the vertical direction, and the average burnup of the SF assemblies is obtained on the basis of the relative burnup distribution and the burnup in the central part of the SF. The average enrichment is calculated from the average burnup.

The calibration of burnup measurement devices is performed by comparing the burnup obtained when measuring the SF for calibration and the burnup data of the relevant SF in a nuclear power station. The calibration is performed by collecting five measurements per SF assembly with the burnup measurement device and comparing the average value of the five measurements with the burnup data for the nuclear power station. Adjustments to the calibration are made with this procedure for multiple SFs; the burnup of the SFs is then measured with the burnup measurement device after the adjustment, and the accuracy of the device is finally confirmed.

5.2 Spent Fuel Management

Caution is needed when handling fuels in order to prevent identification errors for the said fuels, particularly for transactions after confirming the burnup of the SF, such as when loading the specified SF transportation/storage system. It is suggested to take precautions against the fallibility of procedures in order to ensure the effective prevention of identification errors. For example, adopting the following precautions is recommended [78].

1. The separate arrangement of fresh fuel and low burned fuel assemblies in the storage pool; this arrangement is made in view of their administration prior to application of a burnup credit.
2. Arrangement and mutual separation of SFs in storage and SFs waiting for transportation in the storage pool at the reactor site.
3. Periodic staff training and confirmation of storage.
4. Adoption of independent and redundant processes for fuel identification.
5. Recording of handling operations.

In addition, the handling procedures for fuel selection and loading in the SF acceptance operation must be designed such that they coincide with the terms and conditions used in the analysis at the design stage.

5.3 Example Criteria for Acceptance of Spent Fuel in the United States

Examples of criteria for the acceptance of SF in the United States will be explained below [84]. The estimated minimum burnup f_e is obtained by first estimating the burnup and considering the errors in the measurements and operational records when the SF is loaded in the transportation or storage system that applied burnup credit. Proof of acceptance and loading of the relevant SF assemblies are determined by comparing the f_e with the reference burnup (specified ordinarily using the initial enrichment and a loading reference curve) for acceptance. This reference curve is generally drawn on the basis of the initial enrichment in the horizontal axis and the burnup in the vertical axis. Acceptance is allowed above this curve, i.e., for a higher burnup.

For example, reference curves for PWR SF assemblies are obtained using the following procedure:

1. The maximum initial enrichment of ^{235}U (E4) for possible loading of fresh fuel is determined in the design specification of a relevant transportation storage system. For this determination, several neutron multiplication factors are obtained using a verified criticality calculation code for the system loading fresh fuel enriched in the specified range, and the initial enrichment is set equal to the maximum permissible multiplication factor with the interpolation method.
2. Next, (E4, 0) is taken on the reference curve, and a line is drawn from this point vertically upward, as shown in **Fig. 5.1**. That is, SFs that have an initial ^{235}U enrichment less than or equal to the maximum fresh fuel enrichment, E4, are acceptable regardless of the burnup.
3. The criticality calculation using a validated code with the initial enrichment as a parameter for the given burning range considering the accepted SF burnup range is carried out, and the burnup resulting in the maximum permissible multiplication factor is obtained for each initial enrichment. This procedure, as shown in **Fig. 5.2**, first determines the maximum initial enrichment (E0) among the postulated SF acceptance levels and the minimum required burnup (C0) corresponding to the maximum permissible multiplication factor. The minimum required burnup (C4) corresponding to the maximum fresh fuel permissible enrichment E4 is then obtained. The minimum required burnup is obtained sequentially with the initial enrichment as a parameter from (E0, C0) to (E4, C4), and the loading reference curve is obtained by plotting the minimum required burnup. In this case, as (E6, 18) and (E8, 30) shown in **Fig. 5.1**, the minimum required burnup differs with the setup method of the axial burnup distribution in the burnup range, and a step occurs on the reference curve, and these steps are kept as options.
4. Obtaining the above reference values requires a setup of moderate conditions so as to result in a conservative reactivity, even when including abnormal states. The cooling time is set to the minimum value for the SF accepted in the relevant system.

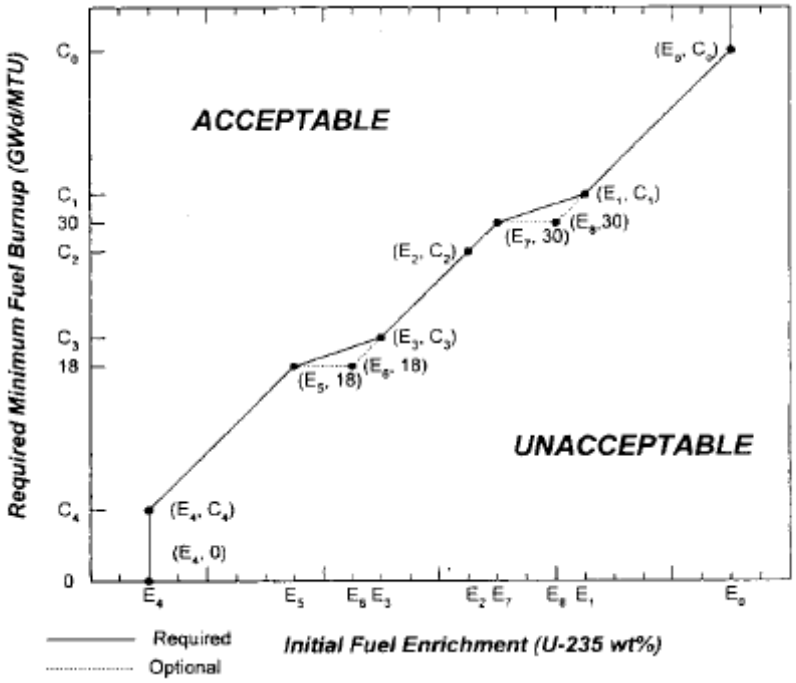

Fig. 5.1 Spent fuel loading curve [84]

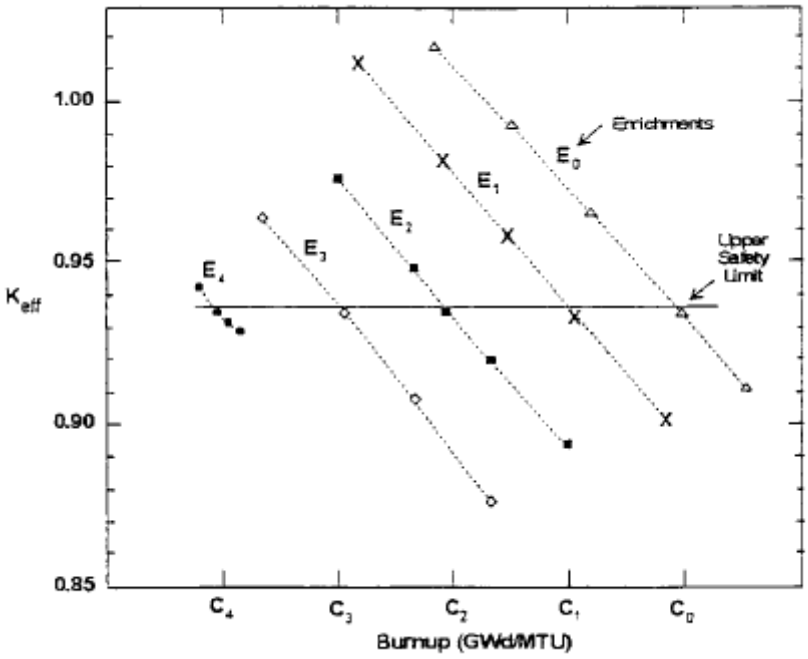

Fig. 5.2 Setup method for the initial enrichment reference burnup [84]

5.4 Example of Adopting Burnup Credit in Japan

A burnup credit considering the uranium burnup and plutonium generation has been applied to the criticality safety design of the Rokkasho reprocessing facility. In this section, as an example of facilities with applied burnup credit, the criticality safety design of the SF storage system and the dissolution system of the Rokkasho reprocessing facility is explained below [85, 86].

Criticality safety controls for the fuel storage pools in the Rokkasho reprocessing facility are implemented by storing the SFs in racks with rigid structures and assuring proper spacing among the SFs. SFs with higher enrichments have a tendency to initiate fission reactions more easily in comparison to lower-enriched SFs, and therefore require larger storage spacing in general. In fact, as the fuel enrichment decreases along with the progress of the fission reactions in the reactor, criticality is prevented by controls that implement smaller storage spacing for SFs with residual enrichment rather than fresh fuel values. In the SF acceptance and storage systems, before transferring to the fuel storage pools, the average enrichment of the SF is measured with a burnup measurement device; on the basis of the result, the average enrichment of the SF is confirmed below 3.5 wt%; at the same time, SFs with its average enrichment value larger than 2.0 wt% are classified as high residual enrichment fuels, and others as low residual enrichment fuels. SFs classified as high residual enrichment fuels are loaded in exclusive cans and stored in the racks with larger spacing than those for lower residual enrichment fuels.

In the dissolution system at the Rokkasho reprocessing facility, the SFs are resolved with hot nitric acid in the dissolvers after chopping, and then separated with nitric acid into a solution containing the nuclear fuel material and the sheared chops from the fuel cladding. Criticality safety control in these dissolvers is implemented as a combination of shape and dimension control, concentration control, and mass control; moreover, when the calculated k_{eff} is larger than 0.95 for the resolving SF with a lower burnup and a higher initial enrichment, as determined by measuring the SF with a measurement burnup device and considering the initial enrichment, a soluble neutron-absorber control is also combined. In this case, gadolinium nitrate is used as the soluble neutron absorber.

The Rokkasho reprocessing facility utilizes detailed burnup monitors (section 5.1.4) in the fuel acceptance and storage systems. However, current problems have been identified, including the adoption of a safety margin that is too large. While burnup credit is not adopted for the SF transportation/storage at the reactors, there has been a movement toward the use of only the reactor management data for the burnup confirmation concerning the application of burnup credits, and a study of the reliability of this approach is planned.

6. Concluding Remarks

It is well known that the reactivity of LWR fuel decreases as the burnup proceeds. However, this reduction in reactivity has not been reflected in the criticality safety design of systems in Japan, except in the case of the Rokkasho reprocessing facility. The reasons are summarized as follows:

1. The economic benefit from the application of burnup credit has been small.
2. Compared to fresh fuel, various factors are involved in burned fuel, and the setup conditions are complicated; therefore, strategies for introducing a burnup credit application have been uncertain.
3. Measured data confirming the reliability of burnup calculation codes or of criticality calculation codes for burned fuel have been scarce, and arrangement of the data for confirming the reliability of burnup or criticality calculation codes for burned fuel has not been implemented.
4. Uncertainties have been introduced into the confirmation of the burnup identification.
5. Implementation experiences have generally been scarce in the world.

Concerning item 1, recently, circumstances have been changing gradually and affected by the tendency toward higher burnup. One solution for the uncertainties in item 2 is this guide. Benchmark calculation activities of the Expert Group on Burnup Credit of the OECD/NEA have been solving the issue of item 3. Data acquisition efforts through the destructive analyses of SFs at JAERI and participation in international cooperation programs such as ARIANE have progressed. With respect to item 4, the progress of burnup monitor development was explained in section 5.1. For item 5, the Nuclear Regulatory Committee (NRC) of the United States recently (May 1999) approved the application of burnup credit with restrictions. In addition, burnup credit has been applied in France for many years. Therefore, it is clear that these issues have more or less been resolved sequentially.

As explained in chapter 2, this guide is primarily intended for level 1 burnup credit, and describes the evaluation methodology and compiled data. The accumulation and upgradation of FP isotopic composition data are intended for the future, and will eventually allow revision of this guide to describe the application of level 2 burnup credit, accounting for FPs in addition to actinides. First, it is necessary for Japan to utilize the chemical analysis data of the batch-processed dissolver solution at the Tokai reprocessing facility, which can be realized with the cooperation of the Japan Nuclear Cycle Development Institute. To ultimately achieve this goal, further acquisition of FP isotopic composition data by agencies around the world and international cooperation through participation in the REBUS program is necessary.

As explained in chapter 3, correction factors are obtained for PWR SFs through verification of the burnup calculation codes. At present, verification is limited to PWR fuel assemblies because it is complicated to establish analysis objectives and conditions (such as the three-dimensional initial enrichment distribution, existence of poisons, and variation in the axial distribution of a coolant void fraction during irradiation in the reactor) for BWR fuel assemblies.

For the SF storage pools at BWRs and their transportation package systems, the advantage of a burnup credit application is very small, if it is limited to level 1, as reviewed in chapter 2; however, expanding the subject to dissolvers for BWR SFs in a reprocessing facility, an economical advantage, such as enhancement of the processing amounts, is expected to be as sufficient as that for dissolvers for PWR SFs. Appendix III introduces examples of isotopic composition analyses for typical specifications and irradiation conditions of not only PWR fuels but also BWR fuels, as references, which are considered to be helpful to users for checking their calculations. There is a desire to be able, in the future, to identify the degree of error in isotopic composition calculations of spent BWR fuels through verification with experimental data.

As explained in chapter 4, open experimental data are scarce for verification of criticality calculation codes applied to SF transportation/storage systems because conducting such experiments is very difficult. The main target of this guide is the application of level 1 burnup credit regarding only actinide changes; for this level of implementation, critical experiments on MOX fuel systems and uranium and plutonium mixed solution (MIX) systems are considered to be sufficient for code validation. At present, the application of a burnup credit to the dissolvers in the Rokkasho reprocessing facility is limited to considering the variation in uranium and plutonium contents (level 1). However, for further rationalization of safety margins, level 2 burnup credit application pertaining to FPs is expected. To fulfil this expectation, critical experiments with actual SFs are required. Under the present circumstances, data collection is required through international cooperation (such as the REBUS program) in order to share the cost and/or acquisition of data on the experimental reactivity effects of FPs using the critical experimental apparatus at JAERI.

In addition, as explained in chapter 5, in applying burnup credit to the criticality safety control of the Rokkasho reprocessing facility and the SF transportation/storage at the reactor site, there are excessive safety margins included in the measured values of the burnup monitors for identification, and cumbersome maintenance work is required. Considering these issues, it is also conceivable that the adoption of a fuel burnup procedure based only on the reactor operational management data without measurements, for the verification of fuel burnup required in the IAEA international transport regulations, may occur. Accumulation of confirmational data is desired before implementation of such practices.

This guide is published as a supplement to the *Nuclear Criticality Safety Handbook, Version 2*, as described in chapter 1 (Introduction). This guide should be reviewed as advances in technology, and the progressive accumulation of new knowledge requires enhancement of the content and timely revision in the future.

Acknowledgements

Many people concerned with nuclear education, research, and industry provided valuable comments and suggestions for the publication of this document. Although this document was issued under the name of the Working Group on Nuclear Criticality Safety Data, many members in the Special Committee on Nuclear Criticality Safety Nuclear Fuel Facility Safety Research Committee at the Japan Atomic Energy Research Institute provided both direct and indirect assistance in reviewing this document. We express our deepest gratitude to them. Furthermore, Mr. Akira Tsubosaka (formerly, Nuclear Fuel Industries, Ltd., currently Nuclear Safety Technology Center) and Mr. Hiroki Mochizuki (currently, Japan Research Institute, Ltd.) made great contributions in the editing of this document. We express our most heartfelt thanks to them.

References

[1] Working Group on Nuclear Criticality Safety Data, “Nuclear Criticality Safety Handbook Version 2”, JAERI 1340, 1999, 189p.

[2] Kenya Suyama, Akira Tsubosaka, Liem Peng Hong “Quantitative Evaluation of Possible Storage Amount of Fuel Assemblies by Introducing Burnup Credit” 1997 Fall Meeting of the Atomic Energy Society of Japan, September 1999.

[3] K. J. Notz, “ORIGEN2, Version 2.1 Release Notes”, CCC-371, pp. 200–208, August 1991.

[4] Y.Ando, T.Sasaki, I.Mitsuhashi, K.Yoshioka, F.Yamamoto, K.Matsumura, T.Kitano, D.Ito, A.Terada, Y.Momma, “Burnup Credit Application to Transportation and Storage of Spent BWR Fuel Assemblies (1) - Axial Zoning Model” Proceedings of the sixth International Conference on Nuclear Criticality Safety (ICNC’99) CCC-371, Vol. IV, p.1935, Versailles, France, September 1999.

[5] Kenya Suyama, Jun-ichi Katakura, Yasushi Ohkawachi, Makoto Ishikawa, “Libraries Based on JENDL-3.2 for ORIGEN2 code: ORLIBJ32”, JAERI-Data/Code 99-003, 1999, 83p.

[6] Takamasa Mori, Masayuki Nakagawa, “MVP/GMVP: General Purpose Monte Carlo Codes for Neutron and Photon Transport Calculation based on Continuous Energy and Multigroup Methods”, JAERI-Data/Code 94-007, 1994, 152p.

[7] A. G. Croff, “ORIGEN2 – A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code”, ORNL-5621, July 1980.

[8] Kenya Suyama, Tomohiko Iwasaki, Naohiro Hirakawa, “Integrated Burnup Calculation Code System SWAT”, JAERI-Data/Code 97-047, 1997, 128p.

[9] Toden Software Incorporated,“On Steady-State BWR Core Analysis Methods”, TCMO-96001, August 1996.

[10] M. J. Bell, “ORIGEN - The ORNL Isotope Generation and Depletion Code”, ORNL-4628, May 1973.

[11] O. W. Hermann and R. M. Westfall, “ORIGEN-S: Scale System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms”, NUREG/CR-0200Rev.5 (ORNL/NUREG/CSD-2/R5) as CCC-545, Vol. 2, Section F7, 1995.

[12] S. J. Ball and R. K. Adams, “MATEXP”, A General Purpose Digital Computer Program for Solving Ordinary Differential Equations by the Matrix Exponential Method”, ORNL/TM-1933, 1967.

[13] D. R. Vondy, “Development of a General Method of Explicit Solution to the Nuclide Chain Equations for Digital Machine Calculations”, ORNL/TM-361, 1962.

[14] Keisuke Okumura, Kunio Kaneko, Keiichiro Tsuchihashi, “SRAC95; General Purpose Neutronics Code System”, JAERI-Data/Code 96-015, 1996, 445p.

[15] A. G. Croff, “ORIGEN2 - A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials”, Nucl. Technol., Vol. 62, pp. 335–352, September 1983.

[16] C. V. Parks, “SCALE-4.3: Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation for Workstations and Personal Computers”, NUREG/CR-0200Rev.5 (ORNL/NUREG/CSD-2/R5) as CCC-545, 1995.

[17] Tsuneo Nakagawa, Keiichi Shibata, Satoshi Chiba, Tokio Fukahori, Yutaka Nakajima, Yasuyuki Kikuchi, Toshihiko Kawano, Yukinori Kanda, Takaaki Ohsawa, Hiroyuki Matsunobu, Masayoshi Kawai, Atsushi Zukeran, Takashi Watanabe, Sin-iti Igarasi, Kazuaki Kosako, and

Tetsuo Asami, “Japanese Evaluated Nuclear Data Library Version 3 Revision-2: JENDL-3.2”, J. Nucl. Sci. Technol., Vol. 32, pp. 1259–1271, December 1995.

[18] Kanji Tasaka, Jun-ichi Katakura, Hitoshi Ihara, Tadashi Yoshida, Shungo Iijima, Ryuzo Nakasima, Tsuneo Nakagawa, Hideki Takano, “JNDC Nuclear Data Library of Fission Products, - Second Version -”, JAERI 1320, 1990.

[19] Kenya Suyama, Tomohiko Iwasaki, and Naohiro Hirakawa, “Analysis of Post Irradiation Experiments in PWR Using New Nuclear Data Libraries”, J. Nucl. Sci. Technol., Vol. 31, No. 6, pp. 596–608, June 1994.

[20] Kenya Suyama, Tomohiko Iwasaki, and Naohiro Hirakawa, “Improvement of Burnup Code System SWAT for Use in Burnup Credit Problem”, Proceedings of International Conference on the Physics of Reactors (PHYSOR96), Vol. 4, pp. L–53–L–62, Mito, Ibaraki, Japan, Japan Atomic Energy Research Institute, September 1996.

[21] “ENDF/B Summary Documentation”, BNL-NCS-17541 2nd ed., 1975.

[22] R. Kinsey, “Data Formats and Procedures for the Evaluated Nuclear Data File, ENDF”, BNLNCS-50496 2nd ed., 1979.

[23] Masayoshi Kurosawa, Yoshitaka Naito, and Toshiyuki Kaneko, “Isotopic Composition of Spent Fuels for Criticality Safety Evaluation and Isotopic Composition Data Base (SFCOMPO)”, Proceedings of the Fifth International Conference on Nuclear Criticality Safety (ICNC’95), Vol. 1, p. 2.11, Albuquerque, New Mexico, USA, September 1995.

[24] D.B Owen, “Factors for One-Sided Tolerance Limits and for Variable Sampling Plans”, SCR-607, 1963.

[25] Kenya Suyama, Minoru Murazaki, Hiroki Mochizuki, Yasushi Nomura, “Correction of Calculation Result for PWR Spent Fuel Isotopic Composition with ORIGEN2 Burnup Calculation Code and Validation with MVP Criticality Calculation Code”, to be published.[1]

[26] Hee Sung Shin, Kenya Suyama, Hiroki Mochizuki, Hiroshi Okuno, Yasushi Nomura, “Analyses of PWR Spent Fuel Composition Using SCALE and SWAT Code Systems to Find Correction Factors for Criticality Safety Applications Adopting Burnup Credit”, JAERI-Research 2000-066, 2001, 131p.

[27] M.D.Dehart and O.W.Hermann, “An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel”, ORNL/TM-13317, 1996.

[28] Meraj Rahimi, “Isotopic and Criticality Validation for PWR Actinide-only Burnup Credit”, DOE/RW-0497, U.S. Department of Energy, 1997.

[29] O. W. Hermann, S. M. Bowman, M. C. Brady, and C. V. Parks, “Validation of the Scale System for PWR Spent Fuel Isotopic Composition Analyses”, ORNL/TM-12667, March 1995.

[30] Office of Civilian Radioactive Waste Management U.S. Department of Energy, editor, “Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages”, DOE/RW-0472Rev.2, September 1998.

[31] Tetsuo Matsumura, Akihiro Kasahara, Masanobu Takei, Tetsuya Takegawa, Katsushi Kagehira, “Strategy Investigation on Burnup Credit Introduction to Spent Fuel Storage(3) - Evaluation Methodology of Subcriticality and Evaluation Equation for Equivalent Uniform Burnup”, 1997 Fall Meeting of the Atomic Energy Society of Japan, October 1997.

---

[1] **Translators' Note:** Reference [25] was published as follows:
Kenya Suyama, Minoru Murazaki, Hiroki Mochizuki, Yasushi Nomura, “Derivation of Correction Factor to be Applied for Calculated Results of PWR Fuel Isotopic Composition by ORIGEN2 Code”, JAERI-Tech 2001-074, 2001, 119p.

[32] Yasushi Nomura, Minoru Murazaki, Hiroshi Okuno, “Derivation of Equations evaluating Equivalent Uniform Burnup and Equivalent Initial Enrichment for Isotopic Composition of PWR Spent Fuel”, to be published, JAERI-Data/Code 2001.[2]

[33] (Edited by)Yoshinori Nakahara, Kenya Suyama, Takenori Suzaki, “Technical Development on Burn-up Credit for Spent LWR Fuels”, JAERI-Tech 2000-071, 2000, 381p.

[34] Yoshitaka Naito, Makoto Takano, Masayoshi Kurosawa, and Takenori Suzaki, “Study on the Criticality Safety Evaluation Method for Burnup Credit in Japan”, Nucl. Technol., Vol. 110, p. 40, April 1995.

[35] Takafumi Hayashi, Kenya Suyama, Hiroki Mochizuki, “Effects of burnup history on Spent Fuel Reactivity”, to be published.[3]

[36] Masayoshi Kurosawa, Yoshitaka Naito, Kenya Suyama, Kuniyuki Itahara, Katuo Suzuki, Kouji Hamada, “Effects of LWR Spent Fuel Long Term Storage on Fuel Cycle of Plutonium for Thermal Use”, Journal of the Atomic Energy Society of Japan, Vol. 40, No. 6, pp. 486–494, June 1998.

[37] Keiichiro Tsuchihashi, Yukio Ishiguro, Kunio Kaneko, Masaru Ido, “Revised SRAC Code System”, JAERI 1302, 1986, 281p.

[38] T. B. Flower, D. R. Vondy, and G. W. Cunningham, “Nuclear Reactor Core Analysis Code: CITATION”, ORNL/TM-2496 Rev.2, July 1971.

[39] W. W. Engle Jr., “A Users Manual for ANISN, a One Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering”, Union Carbide Corporation Report K-1693, March 1967.

[40] N.M. Greene, “XSDRN: A Discrete Ordinates Spectral Averaging Code”, ORNL/TM-2500, 1969.

[41] W. A. Rhoades and R. L. Childs, “An Updated Version of DOT4: One- and Two-dimensional Neutron/Photon Transport Code”, ORNL-5851, 1982.

[42] K. D. Lathrop and F. W. Brinkley, “TWOTRAN-II: An Interfaced, Exportable Version of the TWOTRAN Code for Two-dimensional Transport”, LA-4848-MS, July 1973.

[43] Ray E. Alcouffe, et al., “DANTSYS: A Diffusion Accelerated Neutral Particle Transport Code System”, LA-12969-M, June 1995.

[44] L. M. Petrie and N. F. Cross, “KENO IV An Improved Monte Carlo Criticality Program”, ORNL-4938, 1975.

[45] L. M. Petrie and N. F. Landers, “KENO V.a: An Improved Monte Carlo Criticality Program with Supergrouping”, NUREG/CR-0200Rev.2 (ORNL/NUREG/CSD-2/V1/R2), Vol. 2, 1984, Section F11.

[46] D.F. Hollenbach, L. M. Petrie, and N. F. Landers, “KENO VI: A General Quadratic Version of the KENO Program”, NUREG/CR-0200Rev.5 (ORNL/NUREG/CSD-2/V2/R5), Vol. 2, 1995, Section F17.

[47] Y. Naito, et al, “MULTI-KENO: A Monte Carlo Code for Criticality Safety Analysis”, JAERI-M 83-049, 1983, 232p.

---

[2] **Translators' Note:** Reference [32] was published as follows:
Yasushi Nomura, Minoru Murazaki, Hiroshi Okuno, “Preparation of Data Relevant to "Equivalent Uniform Burnup" and "Equivalent Initial Enrichment" for Burnup Credit Evaluation”, JAERI-Data/Code 2001-029, 2001, 120p.

[3] **Translators' Note:** Reference [35] was published as follows:
Kenya Suyama, Minoru Murazaki, Hiroki Mochizuki, Yasushi Nomura, “Derivation of Correction Factor to be Applied for Calculated Results of PWR Fuel Isotopic Composition by ORIGEN2 Code”, JAERI-Tech 2001-074, 2001, 119p.

[48] Judith F. Briesmeister, Editor, “MCNPTM – A General Monte Carlo N-Particle Transport Code Version 4C”, LA-13709-M, April 2000.
[49] J. F. Briesmeister, editor, “MCNP – A General Monte Carlo N-Particle Transport Code Version 4B”, LA-12625-M, March 1997.
[50] P. Rose and C. Dunford, “Data Formats and Procedures for the Evaluated Nuclear Data FileENDF-6”, ENDF-102, 1991.
[51] Yuichi Komuro, Takenori Suzaki, Shoichi Ootomo, Kiyoshi Sakurai, Oichiro Horiki, “Measurement of Reactivity Worths of Sm, Cs, Gd, Nd, Rh, Eu, B and Er Aqueous Solution Samples”, JAERI-Research 97-088, 1997, 19p.
[52] J. Anno, G. Poullot and P. Fouillaud, P. Grivot and E. Lejeune, “Planned Experimental Program Qualifying the Safety Margins Given by 6 selected Fission Products in Spent Fuels”, Proceedings of the Topical Meeting on the Criticality Safety Challenges in the Next Decade, 271–278, Chelan, Washington, USA, 1997.
[53] J. B. Briggs, editor, “International Handbook of Evaluated Criticality Safety Benchmark Experiments”, NEA/NSC/DOC(95), September 2000 Edition, September 2000.
[54] R.C Lane and O.J.E. Perkins, “The Measurement of the Critical Size of a Homogeneous Mixture of Plutonium and Natural Uranium Oxides with Polythene”, AWRE O32/68, 1968.
[55] S.R. Bierman, E.D. Clayton, and L.E. Hansen, “Critical Experiments with Homogeneous Mixtures of Plutonium and Uranium Oxides Containing 8, 15, and 30wt% Plutonium”, Nucl. Sci. Eng., Vol. 50, pp. 115–126, 1973.
[56] S.R. Bierman and E.D. Clayton, “Critical Experiments with Low-Moderated Homogeneous Mixtures of Plutonium and Uranium Oxides Containing 8, 15, and 30wt% Plutonium”, Nucl. Sci. Eng., Vol. 61, pp. 370–376, 1976.
[57] S.R. Bierman, B.M. Durst, and E.D. Clayton, “Critical Experiments Measuring the Reactivity Worths of Materials Commonly Encountered as Fixed Neutron Poisons”, BNWL-2129, 1976.
[58] B.M. Durst, S.R. Bierman, E.D. Clayton, and R.I. Scherpelz, “Critical Experiments with Fast Test Reactor Fuel Pins in Water”, PNL-SA-7064S, 1978.
[59] S.R. Bierman, “Criticality Experiments with Low Enriched UO_2 Fuel Rods in Water Containing Dissolved Gadolinium”, PNL-4976, 1984.
[60] V.O. Uotine, “Lattices of Plutonium-Enriched Rods in Light Water - Part1: Experimental Results”, Nucl. Technol., Vol. 15, p. 257, 1972.
[61] Makoto Takano, “OECD/NEA Burnup Credit Criticality Benchmark — Result of Phase-1A —”, JAERI-M 94-003, 1994 (as NEA/NSC/DOC(93)22), 145p.
[62] M. D. DeHart, M. C. Brady, and C. V. Parks, “OECD/NEA Burnup Credit Calculational Criticality benchmark Phase I-B Results”, ORNL-6901, June 1996, (as NEA/NSC/DOC(96)06).
[63] A. Nouri, “OECD/NEA Burnup Credit Criticality Benchmark — Analysis of Phase II-B Results: Conceptual PWR Spent Fuel Transportation Cask”, Rapport IPSN/98-05, 1998, (as NEA/NSC/DOC(98)1).
[64] Hiroshi Okuno, Yoshitaka Naito, Kenya Suyama, and Yoshihira Ando, “International Studies on burnup Credit Criticality Safety by an OECD/NEA Working Group (2) Calculation Benchmarks or BWR Spent Fuels”, Proceedings of Sixth International Conference on Nuclear Criticality Safety, Vol. II, pp. 566–575, Versailles, France, September 1999, Institute of Protection and Nuclear Safety.
[65] M.C. Brady, H. Okuno, M.D. DeHart, A. Nouri, and E. Sartori, “International Studies on Burnup credit Criticality Safety by an OECD/NEA Working Group”, Proceedings of International Conference on the Physics of Nuclear Science and Technology, Vol. 1, pp. 624–630, Long Island,

USA, October 1998, American Nuclear Society.

[66] Makoto Takano and Hiroshi Okuno, “OECD/NEA Burnup Credit Criticality Benchmark—Result of Phase-IIA —”, JAERI-Research 96-003, 1996 (as NEA/NSC/DOC(96)01), 170p.

[67] E. G. Taylor, SAXTON Plutonium Program, “Critical Experiments for the SAXTON Partial Plutonium Core”, WCAP-3385-54 (EURAEC-1493), 1965.

[68] E. A. Bassler, D. C. Fischer, N. J. Georges, E. A. McCabe, SAXTON Plutonium Program, Mechanical, “Thermal and Hydraulic Design of SAXTON Partial Plutonium Core”, WCAP-3385-52 (EURAEC-1491), 1965.

[69] A. Biancheria, R. N. Stanutz, R. J. Allio, M. D. Houston, W. E. Ray, R. G. Rose, SAXTON Plutonium Program, “Material Design and Fabrication of the SAXTON Partial Plutonium Core”, WCAP-3385-53 (EURAEC-1492), 1965.

[70] R. C. LIIKALA, V. O. UOTINEN, and U. P. JENQUIN, ”Lattices of Plutonium-Enriched Rods n Light Water - Part II, “Theoretical Analysis of Plutonium-Fueled Systems”, Nucl. Technol., 15, 272, 1972.

[71] J. B. Briggs, editor, “International Handbook of Evaluated Criticality Safety Benchmark Experiments IX-COMP-THERM-003”, NEA/NSC/DOC(95), September 1998 Edition, September 1998.

[72] H. Tsuruta et al., “Critical Sizes of Light Water Moderated UO_2 and PuO_2-UO_2 Lattice”, JAERI 1254, 1978, 38p.

[73] H. Tsuruta et al., “Experiment and Calculation of β_{eff}/l for the Light-Water Moderated Two-Region Core of UO_2 and PuO_2-UO_2 Lattices”, JAERI-M 4696, 1972, 56p.

[74] H. Tsuruta et al., “Critical Experiments and Analyses on 7×7 PuO_2-UO_2 Lattices in Light-Water Moderated UO_2 Core”, JAERI 1234, 1973, 76p.

[75] J. B. Briggs, editor, “International Handbook of Evaluated Criticality Safety Benchmark Experiments IX-COMP-THERM-004”, NEA/NSC/DOC(95), September 1998 Edition, September 1998.

[76] Chi H. Kang and Dale B. Lancaster, “Depletion and package Modeling Assumptions for Actinide-only Burnup Credit”, DOE/RW-0495, U.S. Department of Energy, May 1997.

[77] Chi H. Kang, et al, “Horizontal Burnup Gradient Datafile for PWR Assemblies”, DOE/RW-0496, U.S. Department of Energy, May 1997.

[78] International Atomic Energy Agency, “Implementation of Burnup Credit in Spent Fuel Management systems”, IAEA-TECDOC-1013, April 1998, (Proceedings of an Advisory Group Meeting held in Vienna, 20–24 October 1997).

[79] Y. Chanzy and E. Guillou, “COGEMA/TRANSNUCLEAIRE’s Experience with Burnup Credit”, Proceedings of an Advisory Group Meeting, Implementation of Burnup Credit in Spent Fuel Management Systems, p. 33, Vienna, April 1998, International Atomic Energy Agency, (as IAEATECDOC-1013).

[80] W. Lake, “Burnup Credit Activity being Conducted in the United States”, Proceedings of an Advisory Group Meeting, Implementation of Burnup Credit in Spent Fuel Management Systems, p. 109, Vienna, April 1998, International Atomic Energy Agency, (as IAEA-TECDOC-1013).

[81] Christopher E. Olson, D.R. Bronowski, Wayne McMurtry, Ronald Ewing, Robert Jordan, and David Rivard, “The FORK+ Burnup Measurement System: Design and First Measurement Campaign”, Trans. Am. Nucl. Soc., Vol. 79, pp. 282–283, 1998, (1998 Winter Meeting at Washington, DC November 15-19,1998).

[82] Alain Lebrun, Gilles Bignan, Herve Recroix, and Martial Huver, “Characterization of Spent Fuel ssemblies for Storage Facilities Using Non Destructive Assay”, Proceedings of International

Symposium on Storage of Spent Fuel from Power Reactors, p. 67, Vienna, November 1998, International Atomic Energy Agency, (as IAEA-SM-352/33).

[83] Keiji Kanda, et al, " Nuclear Criticality Safety and Burnup Credit", Nuclear Industry, Vol. 40, No. 3, 1994.

[84] D. B. Lancaster, E. Fuentes, and C. Kang, "Actinide-only Burnup Credit Methodology for PWR Spent Nuclear Fuel", Proceedings of an Advisory Group Meeting, Implementation of Burnup Credit in Spent Fuel Management Systems, p. 117, Vienna, April 1998, International Atomic Energy Agency, (as IAEA-TECDOC-1013).

[85] Keiji Kanda, Osamu Kawakatsu, "Criticality safety and Burnup Credit(2) Introduction of Burnup Credit Concept to Reprocessing Facility", Nuclear Industry, Vol. 40, No. 3, pp. 11–19, 1994.

[86] Japan Nuclear Fuel Limited, "Approval Application for Installation of Reprocessing Facility, Annex 6", August 1992.

Appendix I

Current Status of Burnup Credit in Various Countries

I.1 Criticality Safety Standards and Regulations on Transportation and Storage Systems

Descriptions of the standards and regulations on transportation and storage systems for SFs and the application of burnup credit in various countries are shown below. Every regulatory authority does not prohibit the application of a burnup credit because it improves safety for workers and the public.

I.1.1 Japan

For the transportation of fissile material, Order 57 of the General Administrative Agency of the Cabinet states that, “Regulation on transportation of nuclear fuel material etc. in a factory or outside of nuclear facilities is established in response to the Laws for Regulation of Nuclear Source Materials, Nuclear Fuel Materials and Reactors (LRNR).” Article 11, which is related to the criticality safety of this regulation, specifies that criticality should not be reached, and it requires that subcriticality be maintained for isolated or arrayed systems (single transportation containers containing nuclear fuel material) in a damaged or undamaged state under special test conditions, including a 9 m drop, and that subcriticality must be attained in the case of loading conditions that are two or five times that of the limited number for transportation (defined as the limited number of packages containing fissile material capable of being loaded at one location) corresponding to the damaged state. In the above description, isolated and arrayed systems correspond to single and multiple unit systems, respectively. There is no provision related to a burnup credit implementation. In safety reviews relating to transportation, criticality safety is evaluated with fresh fuel instead of SF, which means that criticality safety is evaluated without considering the burnup effect. The order allows the introduction of a criticality safety evaluation with the burnup effect when such methodology is approved in the future.

“Concerning Spent Fuel Cask Storage at a Reactor Site,” issued by the Nuclear Safety Commision (NSC) of Japan, describes the methodology for criticality prevention. However, it does not mention burnup credit, as is the case with transportation. The LRNR were revised in 1999, and the related laws and regulations for introducing interim SF storage facilities are now in preparation.

Meanwhile, the regulatory review of the Rokkasho reprocessing facility, which is the only facility that uses an applicable burnup credit, has been carried out according to the “Regulatory Guide to Nuclear Fuel Reprocessing Facilities.” This guide (articles 10 and 11) specifies that single and multiple units shall not reach criticality; however, it does not include a description of burnup credit.

I.1.2 IAEA

For transportation of radioactive (fissile) material, the International Atomic Energy Agency (IAEA) issued "Regulations for the Safe Transport of Radioactive Material" (hereinafter "Transport Regulations"), and the member states are implementing their own safety regulations according to these regulations. In Japan, a revised "Safety Standard for the Transportation of Radioactive Materials" (based on a 1985 version of the Transport Regulations) has been in force since 1991. This 1985 version clearly states that the evaluation of nuclear criticality safety may depend on the actual irradiation history [1]. The IAEA Transport Regulations, revised in 1996 (IAEA Safety Standards Series No.ST-1), are now under consideration for integration into Japan's legal system. The following criticality safety requirements for transportation packages are described in paragraph 671(a).

> Maintain subcriticality during normal and accident conditions of transport; in particular, the following contingencies shall be considered:
> 1. water leaking into or out of packages;
> 2. the loss of efficiency of built-in neutron absorbers or moderators;
> 3. rearrangement of the contents either within the package or as a result of loss from the package;
> 4. reduction of spaces within or between packages;
> 5. packages becoming immersed in water or buried in snow; and
> 6. temperature changes.

Detailed, concrete requirements for the application of burnup credit are based on isotopic compositions demonstrated to provide [2]:

> 674(a) the maximum neutron multiplication during the irradiation history, or
>
> 674(b) a conservative estimate of the neutron multiplication for the package assessments. After irradiation but prior to shipment, a measurement shall be performed to confirm the conservatism of the isotopic composition.

The measurement mentioned here includes a burnup measurement and a direct measurement of subcriticality. These measurements are performed to demonstrate the conservativeness of the subcriticality evaluation. For an SF storage facility, the IAEA Safety Series No.116 [3] specifies that calculating the neutron multiplication factor of an infinite system as the estimated value is permissible for a conservative application of burnup credit; and the isotopic composition value of fissile material used in the calculation, along with the decreased amount of burnt up material, must be confirmed with a direct or indirect measurement. In addition, IAEA Safety Series No.11 specifies that, for a soluble neutron absorber in pool water to be applied to the design of an SF storage pool when boiling of the pool water in an incident or accident condition is postulated, and a criticality safety analysis should be performed considering the density decrease of the neutron moderator subject to this boiling situation. The shapes and arrangement of the fuels supported by baskets and canisters should ensure subcriticality at the steps of loading, storage, and retrieval. When designing an SF dry storage or silo, the criticality safety analysis should be performed assuming that 1) subcriticality cannot be maintained by a disturbance of the internal/external events, and 2) introduction of a moderator is inevitable.

I.1.3 Others

The US DOE proposed a specific method for burn up calculations that considers actinides only for the application of burnup credit (i.e., Level 1, See section 2.3) to the transportation/storage of PWR fuel. The NRC approved the proposal in May 1999 for the transport of PWR SF [4]. In addition, in France, the application of burnup credit (Level 1) has been accepted previously. In Germany, DIN25471 was established in July 1999 as the standard for applying burnup credit to SF storage pools.

I.2 Current Status of Burnup Credit Adoption

I.2.1 Current Status in the World

The selection of the level of burnup credit (See section 2.3) to the design of SF transportation/storage systems should be determined to assure a safety margin consistent with the current status of the accuracy of burnup calculations and/or criticality calculations. The current status of the application in each country differs depending on its regulatory system. Burnup criteria that only account for the decrease in the fissile material have not been applied in general.

The regulatory authorities of each country approve the effect of the reactivity of several limited nuclides to the burnup credit according to their technology level. The level of burnup credit should be chosen on the basis of the accuracy of the burnup credit evaluation applied and the characteristics of the subject SF management system. The current situation (in the year 1998) in various countries is described below [5].

I.2.1.1 USA

In the United States, SFs are stored in wet or dry storage facilities at each reactor site until their reception into the Civilian Radioactive Waste Management System (CRWMS) of the final geological repository. As commercial reprocessing is not planned, if a decision for a reception at the repository is made, SF is transported from each site to the final repository, and this shipping is conducted using the dry method. The NRC approved a license for the application of burnup credit to PWR SF storage pools at two sites—Lesko and Newmyer—in 1997. In these cases, the boron content in the pool water is the condition for assuring criticality safety. BWR SF storage pools do not apply burnup credit because they do not contain boron. The PWR SF dry storage burnup credit utilizes the concept of moderator exclusion, i.e., the credit is approved after loading the SF into a storage container in the boron-containing pool water (application of the burnup credit is approved at this step), then draining and storing the container in an area with no possibility of water intrusion. For this situation, in 1998, the DOE reviewed the current status report for the application of burnup credit, adopting a calculation that considers only the absorbing effect of the actinides. The NRC approved the report in May 1999. Usually, once a PWR SF dry transportation package is approved, it is applied to the dry storage system [6]. The license requirement for the confirmation of the burnup is the utilization of a burnup monitor.

I.2.1.2 France

At present, licensing stages for wet storage (LaHague), transportation, and reprocessing of PWR SF assemblies with low enriched uranium oxide are reviewed separately. The utilization of a burnup monitor is a requirement for burnup verification at each stage. The following two requirements are imposed on the basis of the results of the criticality safety analysis: the first case is when the required burnup is lower than the minimum burnup after one cycle of irradiation, which is assured by reactor operation; a submission of the quantitative measured result is enough to show that the fuel was actually irradiated in the reactor. The second case requires the confirmation of burnup with a quantitative measurement. The quantitative measurement is performed over the bottom 50 cm of the effective fuel length with minimum irradiation. Only the prime actinides (^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu) are considered for the nuclear criticality safety evaluation without accounting for the effects of FP. The same criteria are also applied to the storage of BWR SF [7].

In cases of international transportation, as specified in the IAEA Transport Regulation ST-1, after obtaining design approval for a B(U)F-type transportation package from the regulatory authority in the shipping country, the design is automatically approved in the receiving country, according to international agreements [8].

I.2.1.3 Germany

SF from LWRs (PWR and BWR) is stored wet, and the effect of the burnable poison dispersed in the fuel matrix is accepted in the safety evaluation of the nuclear criticality. However, the effect of soluble boron in PWR SF storage pools is disregarded in safety evaluations for normal conditions. The application of burnup credit is accepted; however, the reason for a denial of the fresh fuel assumption must be submitted. PWR SF wet storage racks manufactured in the Siemens KWU have been exported to countries such as Spain, the Republic of Korea, South Africa, and Brazil. The neutron absorbing effects on the net reduction of fissile nuclides and actinides (U, Np, Pu, Am), as well as FP nuclides, are considered in the criticality safety evaluation of the design. At the same time, PWR and BWR SFs are being transported using dry CASTOR casks, and the nuclear criticality safety is evaluated with an assumption of fresh fuel in cases where the initial enrichment is below 4.2 wt%. For SFs above that initial enrichment, an application of burnup credit is accepted only when accounting for the absorbing effect of the actinides in a minimum average discharged burnup of 5 MWd/kgU. At present, the application of burnup credit to the final disposal of the SF is not considered [9].

Typically, burnup monitors are applied to identify the burnup, and identification with reactor data is accepted for wet storage at the reactor site.

I.2.1.4 TheUnited Kingdom

Burnup credit is presently applied to the criticality safety controls for LWR (PWR, BWR) SF at the following facilities:

1. SF wet storage in Sizewell
2. SF wet transportation, mainly from overseas to the Sellafield reprocessing plant

3. SF wet storage in Sellafield
4. SF reprocessing in the Sellafield reprocessing plant

In addition, research and evaluation for the application of burnup credit are advancing, particularly for MOX SFs in the United Kingdom [10].

I.2.1.5 Other Countries

In Belgium, the application of burnup credit to wet and dry transportation/storage of PWR UO_2 and MOX SFs is accepted.

In Russia, regarding level-one credits, a burnup credit for uranium and plutonium nuclides at a minimum average discharged burnup of 25 MWd/kgU is accepted for wet transportation discharge of VVER-440 SF from the Kola Nuclear Power Plant. In Switzerland, as well as in France, the average burnup over 50cm from the bottom of the fuel assembly is guaranteed for dry storage of PWR spent fuels.

Burnup credit calculation levels considering actinides and the neutron absorbing effect of FPs are accepted in the Republic of Korea, the PWR SF wet storage facility in Spain, Smolensk NPP of Russia, and the RBMK SF wet storage facility at Ignalina NPP in Lithuania. Certain amounts of soluble boron are contained in the PWR and BWR wet storage pool water; in Spain, aspects of this boron were considered for the criticality safety analysis in the past. Currently, in Spain, burnup identification before SF transportation/storage relating to the application of burnup credit has been implemented, only through reliance on reactor management data.

A credit level of burnable absorbing material was approved in the licensing procedures for BWR wet storage facilities in Spain and Sweden.

I.2.2 Current Status in Japan

In Japan, accumulation of SFs from NPPs is increasing, and most of these SFs are stored in storage pools at the reactor site. A portion of the fuels is stored within the metal-cask system in a dry storage building at the NPP site (Fukushima Daiichi nuclear power station). Criticality safety design is performed on the basis of the assumption of fresh fuel, including packages used for transportation to the reprocessing facility. However, burnup credit accounting for the burnup of uranium and the generation of plutonium are applied to the nuclear criticality safety design of the SF storage systems of the Rokkasho reprocessing facility.

Appendix II

Burnup Calculation Codes (not described in the main text)

II.1 Burnup Calculation Codes

II.1.1 UNITBURN

UNITBURN [11] is a burnup calculation code developed by the Japan Atomic Energy Research Institute (JAERI) that utilizes the cross-section data of MGCL [12].

For the decay chains, the data of JDDL [13] and FPGS are used. The neutron spectrum, depending on the geometry and nuclide composition, is calculated with ANISN [14], which is built into the code, and the effective one-group cross sections are then generated with UNITBURN using the said spectrum. The UNITBURN burnup calculation is performed using the Bateman method; ca. 100 nuclides are calculated.

II.1.2 SRAC95

SRAC [15] is a typical neutron transport calculation code for thermal reactors developed by JAERI. In 1995, the latest version was released as SRAC95 [16]. The key routine of the code is PIJ, in which the neutron transport calculation is performed using the collision probability method. The code also exhibits characteristics such that it is able to perform the resonance absorption calculation for ultrafine energy groups. In addition, because SRAC has built-in codes, such as CITATION [17] and ANISN, users can utilize them from SRAC. The burnup calculation is performed using the DCHAIN burnup routine [18], which is also built into the FPGS90 code explained below. SRAC has been utilized by many users, not only within JAERI but also at many universities. In other words, it has a history of continuous improvement with the support of many users. Therefore, the code's characteristics are well known and user-friendly.

From the standpoint of the burnup calculation, ASMBURN for the fuel assembly as well as COREBN and HIST for the core are provided as supplementary codes. In these codes, the cross sections are generated with SRAC95 in advance, and the burnup calculation is performed with these cross sections.

II.1.3 MKENO-BURN

MKENO-BURN [19] is a burnup calculation code that consists of combinations of the burnup calculation routine of the UNITBURN code and the neutron transport calculation code MULTI-KENO [21], developed on the basis of the multi-group Monte Carlo calculation code KENO IV [20], which enables the evaluation of the neutron spectrum for a three-dimensional geometry. MGCL is utilized for the cross–section data. As is clear from using UNITBURN, the burnup equation is solved with the Bateman equation. This code was developed in the 1980s, and its early development, lack of a continuous development effort, and relatively few users make it difficult to understand the characteristics of this code in detail, particularly the statistical errors, which cannot be avoided in the Monte Carlo code. As a result, there is a lack of clarity in confirming this effect to the burnup calculation. Moreover, the necessary number of histories required for sufficient accuracy is unknown. Consequently, adequate care is required when using this code.

II.1.4 FPGS90

FPGS90 [22] is a burnup calculation code developed by JAERI. As is clear from the name, FPGS is a code that was originally developed to calculate the decay heat of FPs, and is characterized by the fact that the calculation/display of the γ ray spectrum is available. Because FPGS90 was developed on the basis of a burnup calculation code named DCHAIN [18], the burnup equation is solved with the Bateman method. While the Bateman method cannot consider a so-called feedback reaction that daughter nuclides contribute to the generation of parent nuclides, DCHAIN is peculiar in being able to deal with the feedback. In addition, FPGS90 is editable for the library built into the code, and consequently, it is a considerably large system. FPGS90 is peculiar in that it imports the cross-section data for solving the burnup calculation. That is, instead of using the effective one-group cross section, it uses a cross-section value calculated with the following equation, which consists of coefficient A, the g-factor showing the deviance from the 1/v law of the cross section, and the resonance integration RI:

$$\sigma = A \cdot g \cdot \sigma_{2200} + r \cdot RI, \tag{II.1}$$

where $A = \sqrt{\frac{\pi T_0}{4T}}$, $T_0 = 293.16$, K, $r = \frac{\phi_{epi}}{\phi_{th}\Delta u}$, $\Delta u = 9.09$, and the g-factor is given by the following equation, where M(E) is the Maxwell distribution:

$$g = 1 + \sqrt{\frac{\pi T_0}{4T}}\frac{1}{\sigma_0}\int_{10^{-5}ev}^{0.625ev}\left(\sigma(E) - \frac{\sigma_0 v_0}{v}\right)M(E)dE\,. \tag{II.2}$$

For the burnup calculation of a thermal reactor, the g-factor and RI are built into the FPGS90 library, and therefore, users must calculate r and A for each reactor in order to render the input data for the code. That is, the change in the cross section along with burnup is not considered in the burnup calculation.

In addition, because the 25-group cross-section library is built into FPGS, users can utilize the code as an option in such a way that the burnup calculation can be performed using the input data of the neutron spectrum. Moreover, it is possible to input the cross-section data directly into the code. When using this option, the cross-section data that is dependent on the burnup can be utilized. However, in order to obtain the cross-section data depending on the burnup, the use of another program is ultimately required.

II.1.5 COMRAD

COMRAD [13] is a burnup calculation code developed by JAERI. A feature of COMRAD is that the burnup equation is solved using either the Bateman equation or the matrix exponential method, and is selected in the same way as is done by ORIGEN. Therefore, this code can also deal with short time irradiation problems like instantaneous irradiation. In addition, the matrix exponential method routine of the code was developed using the source program of ORIGEN. Therefore, COMRAD is a derivative work of ORIGEN. The decay chain data of COMRAD uses the JDDL library data edited on the basis of ENSDF. This JDDL library has the advantage of numerous built-in nuclides and many types of data. The calculation procedures in COMRAD are as follows: the cross sections depending on the burnup are calculated with UNITBURN, and those data are stored in the file. COMRAD reads the data in the file and then performs the burnup calculations. Therefore, the calculation procedure tends to become complex. In order to improve this issue, a UNIX version was developed in 1996.

II.1.6 MVP-BURN

MVP-BURN [23] is a burnup calculation code in which the burnup calculation routine and MVP are combined together. Here MVP is a continuous-energy transportation Monte Carlo code for neutrons and photons developed by JAERI. In MVP-BURN, the burnup calculation routine uses the routine of DCHAIN, as does SRAC95. Therefore, the burnup chains used in MVP-BURN are the same as those used in SRAC95.

Recently, the continuous-energy Monte Carlo code has become more common because of both faster computers and popularization of parallel computers. When using a continuous-energy Monte Carlo code, the following advantages are obtained in calculating the neutron spectrum: “The approximation of the geometry becomes unnecessary” and “In the handling of the resonance integral, introducing conventional assumption becomes of no use.” Thus, it is expected that such a code will become the main code for nuclear reactor calculations if the processing speed of computers increases in future years.

However, this code must perform numerous neutron spectrum calculations during the burnup analysis. Therefore, even if computers become faster, it is certain that the burnup calculations with this code will still be costly (machine time). Of course, it is very likely that it will be used commonly in a few years.

II.1.7 FLEXBURN

FLEXBURN [24] is a burnup calculation code developed by the Central Research Institute of the Electric Power Industry (CRIEPI) in Japan. Both the two-dimensional neutron transportation calculation and the burnup calculation are performed with FLEXBURN. The fuel rod, control rod, water region, fuel assemblies, and water channels are represented for the simulation by using a calculation mesh with an arbitrary convex quadrangle. In the neutron transport calculation, the neutron transport equation is solved numerically for the probabilities of neutrons in each direction of the S_N angle that penetrate each calculation mesh and leak from it, on the basis of the discrete ordinates method (S_N method). The neutron flux distributions in a heterogeneous system can be calculated with a detailed calculation mesh, and then by calculating the direction of the angle of the neutron flux. A multi-group cross section library (CRIEPI version MGCL) [25, 26] is used for the neutron transportation calculation. The burnup calculations are performed in each detailed region using the calculated neutron flux distributions and effective cross sections. In the burnup calculation, both the calculated multi-group neutron flux and the effective cross sections are reduced to one group. Then, the power density, local burnup, and atomic densities in each region are calculated with the matrix exponential method, similar to the ORIGEN2 code [27], with the reduced, one-group data.

In the burnup calculations of the fuel assembly, the FLEXBURN code can calculate the neutron flux distributions in a heterogeneous geometry with the method described above, where the fuel rod is expressed in detail with calculation meshes. Thus, it is not necessary to deal with the fuel rod as a fuel cell in the calculation. Even burnup calculations for a heterogeneous system comprised of a mix of UO_2 fuel, (U.Gd) O_2 fuel, and MOX fuel can be calculated accurately. In addition, by finely dividing both the calculation mesh and burnup regions, even asymmetric burnup distributions in the fuel pellet can be calculated. At present, the two-dimensional burnup calculation code has been applied for the geometries of a single fuel rod to multifuel assemblies. FLEXBURN is available on UNIX computer systems. In accordance with the upgrading of computers, a three-dimensional version of the code is under development.

II.2 Core Analysis Codes

II.2.1 CASMO

CASMO [28] is a nuclear characteristics analysis code for the light-water reactor fuel assembly developed by Studsvik of America Co. (at present, Studsvik Scandpower Co.).

A cross-section library of 70 groups based on ENDF/B-IV was prepared in the code. The cross-section data of 18 main higher-order-actinide nuclides (up to ^{246}Cm), and those of 29 FP nuclides (other FP nuclides are treated as a lump), which exhibited a significant influence on the reactivity calculation, are included in the library. The nuclear characteristics analysis of the fuel assembly is as follows: the geometry of a fuel assembly is properly approximated to a simple shape, and then, the transportation calculation is performed for the approximated geometry, and the 70-group neutron spectrum is obtained. The 70-group cross sections are reduced to small groups (usually 7 groups) using the 70-group neutron spectrum. A two-dimensional transportation calculation for the fuel assembly is performed using the small-group cross section. The analysis is performed with a penetration probability method [29] based on an integral transportation equation or the characteristics method [30]. In this way, the neutron flux and each nuclide reaction rate are calculated for all regions of the fuel assembly. The burnup calculation of the fuel assembly is performed using the reaction rate of each nuclide calculated at a certain point in time. The analysis is performed by numerically solving time-dependent differential equations in which the chain between the nuclides is considered. The burnup equation is solved with the Matrix Exponential method similar to ORIGEN. It is also possible to get detailed burnup distributions in the fuel pellet by dividing it into radial and circumferential directions.

CASMO is widely used as a nuclear-constant calculation code mainly in the reactor core design and core management of Japanese and foreign LWRs. The calculation accuracy for the reactivity change with burnup is verified through past results in reactor power plants.

II.2.2 PHOENIX-P

PHOENIX-P is a two-dimensional multi-group transportation calculation program for PWR fuel assemblies developed by Westinghouse Co. The nuclear constants that are necessary for the reactor core calculation are obtained with this code. A cross-section set of 42 groups based on ENDF/B-V is used for the nuclear data. The neutron behavior in the fuel assembly is solved with a node coupling calculation method based on a collision probability method or the S_4 transportation calculation method.

In order to obtain the fuel composition change with burnup, the burnup calculation is performed on the basis of the calculated results for the fuel assembly. The burnup calculation is performed as follows: the differential equation, relating to the change in the atomic number density due to the generation and annihilation of nuclides in each fuel of the assembly, is solved using the Laplace transform. In a burnup calculation, the neutron flux calculation and the burnup calculation are repeated as many times as the number of burnup steps. At this time, the predictor–corrector method is used.

II.2.3 TGBLA

TGBLA [31] is a nuclear design code for BWR fuel assemblies, jointly developed by the Toshiba Corporation and General Electric. The first version of TGBLA was developed in 1982. Since then, it has been used as a nuclear design code by both the aforementioned companies. Various improvements, such as an addition of the hexagonal lattice calculation function and model strengthening for resonance calculation, have also been performed in order to correspond to the design of a higher burnup core, the MOX fuel design, and the lattice pitch survey of a high conversion core [32–34]. The chain of 25 actinide nuclides from ^{228}Th to ^{242}Cm, and the chain of 55 FP nuclides (one pseudo FP is included), as well as several transition nuclides that were added to the Iijima model [35], are handled in the burnup calculation. For the integration method of the burnup equation, a methodology for combining a coarse time step method with the fourth-order Runge-Kutta-Gill method is applied. Here the coarse time step method is a technique developed originally and is called the KLY and YQA algorisms [36]. Using this technique, the burnup calculation can be performed with the coarse time step even for fuel assemblies containing gadolinium fuel rods, in which the change in the burnup characteristics with time is very large and the nonlinearity is strong.

II.2.4 HINES

HINES is a nuclear design code for BWR fuel assemblies developed by Hitachi Ltd. For the fast and resonance energy regions, a 68-group neutron spectrum is obtained by solving the neutron transportation equations with the collision probability method for the configuration of a modeled fuel assembly system with a cluster geometry. Here the resonance cross sections of the prime resonance nuclides are calculated with IR approximation [37]. For the thermal energy region, the thermal neutron spectrum is obtained using a neutron transportation calculation of the THERMOS [38] type. For a fuel with a strong Gd_2O_3 absorption, the effect on the adjacent fuel can be considered in the cluster configuration. The neutron flux distributions in the fuel assembly are obtained by a multi-group neutron diffusion calculation with corrected transportation.

The fourth-order Runge-Kutta-Gill method is used for the burnup calculation, and the neutron spectrum of the fuel containing Gd_2O_3 is calculated iteratively in each burnup step. The calculation deals with 25 heavy nuclides from Th to Cm and 30 FP nuclides.

II.2.5 VMONT

VMONT [39] is a Monte Carlo neutron transportation calculation code for the fuel assembly developed by Hitachi Ltd. Because the Monte Carlo method can be applied to various geometries and a detailed neutron energy group consisting of 190 is adopted, this method has potential applicability to square and hexagonal lattices for assemblies, as well as to both thermal and fast reactors, with respect to the neutron spectrum.

Because VMONT can calculate the entire fuel assembly with the detailed energy group mentioned above, the effects of the fuel rod on the neutron spectrum due to a water gap, burnable poison rod, and neutron moderation rod are calculated accurately. Such calculations can be performed in a short computing time with the Monte Carlo calculation method using virtual scattering and the high-speed region identification method.

Moreover, in order to correspond to the high burnup core, the burnup calculations are performed with the generation and decay chain consisting of 32 actinide nuclides and 83 FP nuclides. Therefore, the calculation accuracy of the accumulated FPs in this high burnup state is assured.

II.2.6 NEUPHYS

NEUPHYS is a nuclear characteristic calculation code for BWR fuel assemblies used by Nuclear Fuel Industry Ltd. in Japan. The nuclear characteristics calculation for the fuel assembly is divided into two parts. In the first part, the small energy group nuclear constants in each region of the fuel assembly are calculated with a fuel lattice analysis model. In the second part, the nuclear characteristics of the fuel assembly are calculated using its nuclear constants. In the fuel lattice analysis model, by using the nuclear cross-section library data prepared in advance, and by using input data from the fuel assembly design specifications and the calculation conditions, the calculation of the neutron spectrum in each region of the fuel assembly is performed. Then, the nuclear constants of the small energy group are calculated using this spectrum. The data of approximately 100 isotopes based on ENDF/B are built into the nuclear cross-section library. For calculations in fast and intermediate energy groups, a one-dimensional multi-group calculation using the collision probability method is carried out. For the calculation of resonance integration, the IR approximation method [37] is used. For the thermal energy group calculation, the thermal cross sections for the use of the neutron diffusion calculation are calculated according to the THERMOS format [38]. The self-shielding effect of the resonance absorption, taking into account the fuel lattice geometry and the spectrum interference effect of the strong neutron absorbers, such as the control rod and the fuel rod with Gd_2O_3, is appropriately considered. For the nuclear calculation of the unit fuel assembly, the power distribution in the unit fuel assembly, the infinite multiplication factor, and the average nuclear constants are calculated using a two-dimensional neutron diffusion model with a small number of energy groups concerning the burnup calculation. The equation that represents the burnup process of the fuel elements is solved. Here a necessary neutron flux distribution is calculated using the fuel lattice analysis model and the two-dimensional neutron diffusion model with few energy groups. These calculations are performed for each appropriate burnup step using the number density of each fuel element at the time of each step. The depletion calculation of the heavy nuclides has been performed for 25 nuclides, including U, Pu, Np, Am, and Cm. The production and depletion model for FPs considers 37 main FP nuclides.

II.2.7 NULIF

The NULIF system and the improved NULIF system are the nuclear characteristic calculation code systems for PWRs used by Nuclear Fuel Industrial Ltd. The improved NULIF system is a modification of the NULIF system corresponding to the design of a MOX-loaded core. Here the improved NULIF system is explained. The improved NULIF system consists of the NULIF and GDLUX codes (nuclear calculation for the cell geometry of a fuel rod), ASY5G (nuclear calculation for the fuel assembly geometry), and SHARP (nuclear calculation for the core geometry). In the NULIF code, using the nuclear cross-section library data prepared beforehand, and the fuel assembly design specifications and calculation conditions given as input data, the calculation of the neutron spectrum and burnup in each region of the fuel assembly are performed. Then, the calculation of the nuclear constants of a small energy group is carried out using its spectrum. The data for approximately 99 isotopes based on ENDF/B-V are built into the nuclear cross-section library. For the burnup calculation of heavy nuclides, nuclides such as U, Pu, Np, Am, and Cm are considered. For the generation and annihilation model for the FPs, the prime FPs such as Xe and Sm are considered. The calculation of the multi-group neutron spectrum is performed with the neutron transport equation method for fast and thermal energy groups, respectively. For the calculation of resonance integration, the NR approximation [40] or the NRIM approximation [40] is used. Furthermore, the GDLUX code is used for the calculation of strong neutron absorbers such as fuel rods with Gd_2O_3. The calculation flow of the GDLUX code is nearly the same as that of the NULIF code, except that the inside of a fuel pellet is divided into more detail in order to appropriately take into account the space dependency of the thermal flux distribution. In the ASY5G code, by using the nuclear constant of few energy groups obtained with the fuel rod calculation code, the calculations of the neutron spectrum and the burnup in the fuel assembly are performed on the basis of the neutron diffusion theory, and the two-group nuclear constants for the average fuel assembly or each fuel rod are then obtained. The ASY5G code can calculate adjacent fuels of different types, and it considers the spectrum interference effect between the fuel assemblies.

Appendix III

Burnup Calculation Examples for PWR and BWR Fuel Assemblies

When a burnup calculation has been performed, it is necessary to validate the calculated results. It is useful for this determination that the burnup calculations are performed beforehand, thereby changing the calculation parameters and making them specific to the various fuel assemblies. It is now possible to compare the calculated results with the previous results.

Examples of burnup calculations performed with varying parameters are shown in open documents [41]. These documents describe the following:

(1) Calculation parameters: 12 cases consisting of

- Core type (LWRs): PWRs and BWRs
- Discharged burnup (high burnup core): 33, 45, and 60 GWd/tHM
- Fuel types: UO_2 and MOX

(2) Calculation code: Integral Burnup Calculation Code SWAT

(3) Calculated result: Nuclide composition of the discharged fuel assembly

For PWR fuels, a cell model was prepared on the basis of the geometry of the current 17 × 17 fuel assembly. For BWR fuels, the cell models used STEP-I, STEP-II, and STEP-III fuel assembly geometries, which were improved in series according to the burnup increase in the discharged fuel assembly from the high burnup core.

For the PWRs, the enrichment of the UO_2 fuel was used at the value described in reference [42]. For the BWRs, the values for the enrichment of the UO_2 fuel for the STEP-I (discharged burnup 33GWd/tHM) and STEP-II (discharged burnup 45GWd/tHM) geometries were 3.0 wt% and 3.7 wt%, respectively, as described in reference [43]. The UO_2 enrichment of the BWR STEP-III fuel (discharged burnup of 60 GWd/tHM) was determined by examining the neutron multiplication factor after the burnup. These values are summarized in **Table III.1** below.

Table III.1 Initial Composition of the UO_2 Fuels

Discharged burnup	PWR fuel	BWR fuel
33 GWd/tHM	3.2 wt%	3.0 wt%
45 GWd/tHM	3.9 wt%	3.7 wt%
60 GWd/tHM	5.0 wt%	4.7 wt%

For the MOX fuel enrichment, the initial Pu weight fraction values described in reference [44] were used for both PWRs and BWRs as shown in **Table III.2** below. For both UO_2 and MOX fuels, the difference in the initial enrichment of the PWRs and BWRs depends on the difference in the averaged operating cycle number in the core.

Table III.2 Pu Enrichment of the MOX Fuels

Discharged burnup	PWR fuel	BWR fuel
33 GWd/tHM	5.3 wt%	4.7 wt%
45 GWd/tHM	6.3 wt%	5.7 wt%
60 GWd/tHM	7.9 wt%	7.2 wt%

The following 17 actinide nuclides and 18 FP nuclides are evaluated in Appendix III.

Actinide nuclides ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{242}mAm, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm

FP nuclides ^{79}Se, ^{85}Kr, ^{87}Rb, ^{90}Sr, ^{79}Se, ^{93}Zr, ^{99}Tc, ^{107}Pd, ^{115}In, ^{126}Sn, ^{129}I, ^{135}Cs, ^{137}Cs, ^{142}Ce, ^{144}Nd, ^{147}Sm, ^{148}Sm, ^{149}Sm, ^{151}Sm

References for the Appendices

[1] International Atomic Energy Agency, “Regulations for the Safe Transport of Radioactive Material 1985 Edition”, Safety Series No.6, 1985.

[2] International Atomic Energy Agency, “Regulations for the Safe Transport of Radioactive Material 1996 Edition”, Safety Standards Series No.1, 1996.

[3] International Atomic Energy Agency, “Design of Spent Fuel Storage Facilities”, Safety Series No.116, 1994.

[4] Office of Civilian Radioactive Waste Management, U.S. Department of Energy, Editor, “Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages”, DOE/RW-0472 Rev.2. September 1998.

[5] International Atomic Energy Agency, “Implementation of Burnup Credit in Spent Fuel Management Systems”, IAEA-TECDOC-1013, April 1998, (Proceedings of an Advisory Group Meeting held in Vienna, 20–24 October 1997).

[6] W. Lake, “Burnup Credit Activity being Conducted in the United States”, Proceedings of an Advisory Group Meeting, Implementation of Burnup Credit in Spent Fuel Management Systems, p. 109, Vienna, April 1998, International Atomic Energy Agency, (as IAEA-TECDOC-1013).

[7] Y. Chanzy and E. Guillou, “COGEMA/TRANSNUCLEAIRE’s Experience with Burnup Credit”, Proceedings of an Advisory Group Meeting, Implementation of Burnup Credit in Spent Fuel Management Systems, p. 33, Vienna, April 1998, International Atomic Energy Agency, (as IAEA-TECDOC-1013).

[8] P. Grimm, “Status of Burnup Credit Implementation in Switzerland”, Proceedings of an Advisory Group Meeting, Implementation of Burnup Credit in Spent Fuel Management Systems, p. 93, Vienna, April 1998, International Atomic Energy Agency, (as IAEA-TECDOC-1013).

[9] J. C. Neuber, “Present Status and Future Developments of the Implementation of Burnup Credit in Spent Fuel Management Systems in Germany”, Proceedings of an Advisory Group Meeting, Implementation of Burnup Credit in Spent Fuel Management Systems, p. 39, Vienna, April 1998, International Atomic Energy Agency, (as IAEA-TECDOC-1013).

[10] Russsel Bowden, “The Application of Burnup Credit for Spent Fuel Operations in the United Kingdom”, Proceedings of an Advisory Group Meeting, Implementation of Burnup Credit in Spent Fuel Management Systems, p. 97, Vienna, April 1998, International Atomic Energy Agency, (as IAEA-TECDOC-1013).

[11] Yoshitaka Naito, Minoru Inamura, Fumihiro Masukawa, Yasuhisa Okuda, “UNITBURN: A Computer Code for Burnup Calculation of a Unit Fuel Cell” JAERI-M 90-019, 1990, 62p.

[12] Yuichi Komuro, Hirokazu Tsunoda, and Jun Sakurai, “Neutron Cross Section Data in Multigroup Constant Library MGCL for Criticality Safety Analysis”, JAERI-M 87-092, 1987, 115p.

[13] Yoshitaka Naito, Shunji Hara, Hitoshi Ihara, Jun-ichi Katakura, “Development of Nuclide Decay Data Library JDDL, and Nuclide Generation and Decay Calculation Code COMRAD”, JAERI-M 86-121, 1986, 159p.

[14] W. W. Engle Jr., “A Users Manual for ANISN, a One Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering”, Union Carbide Corporation Report K-1693, March 1967.

[15] Keichiro Tsuchihashi, Yukio Ishiguro, Kunio Kaneko, and Masaru Ido, “Revised SRAC Code System”, JAERI 1302, 1986, 281p.

[16] Keisuke Okumura, Kunio Kaneko, Keiichiro Tsuchihashi, “SRAC95; General Purpose Neutronics Code System” JAERI-Data/Code 96-015, 1996, 445p.

[17] T. B. Flower, D. R. Vondy, and G. W. Cunningham, “Nuclear Reactor Core Analysis Code:

CITATION", ORNL/TM-2496 Rev.2, July 1971.

[18] Kanji Tasaka, "DCHAIN Code for Analysis of Build-up and Decay of Nuclides", JAERI 1250, 1977, 25p.

[19] Yoshitaka Naito, Kenya Suyama, Fumihiro Masukawa, Kiyoshi Matsumoto, Masayoshi Kurosawa, Toshiyuki Kaneko, "Burnup Code for Fuel Assembly by Monte Carlo Code – MKENO-BURN –", JAERI-Data/Code 96-037, 1996, 70p.

[20] L. M. Petrie and N. F. Cross, "KENO IV An Improved Monte Carlo Criticality Program", ORNL-4938, 1975.

[21] Yoshitaka Naito, et al., "MULTI-KENO: A Monte Carlo Code for Criticality Safety Analysis", JAERI-M 83-049, 1983, 232p.

[22] Hitoshi Ihara, Jun-ichi Katakura, Tsuneo Nakagawa "A Computer Code for Calculation of Radioactive Nuclide Generation and Depletion, Decay Heat and g Ray Spectrum – FPGS90 –", JAERI-Data/Code 95-014, 1995, 216p.

[23] Keisuke Okumura, Takamasa Mori, Masayuki Nakagawa, and Kunio Kaneko, "Validation of a Continuous-Energy Monte Carlo Burn-up Code MVP-BURN and its Application to Analysis of Post Irradiation Experiment", J. Nucl. Sci. Technol., Vol. 37, February 2000.

[24] Takanori Kameyama and Tetsuo Matsumura, "Neutron Transport Code FLEXBURN by SN Method with Transmission Probabilities in Arbitrary Square Meshes for Light Water Reactor Fuel Assemblies", Nucl. Sci. Eng., Vol. 123, pp. 86-95, May 1996.

[25] Tetsuo Matsumura, Masahiro Mimura, Yoshitaka Naito, "Development of Multi-group Constants Library for Light Water Reactors", Research Report 279065, Central Research Institute of Electric Power Industry, August 1980.

[26] Yuichi Komuro, Hiroshi Okuno, Yoshitaka Naito, Hisako Oda, Masakatsu Nagai, Yasuhisa Okuda, Tomohiro Sakai, Masyuki Shiota, "Generation and Verification of the Multigroup Constants Library MGCL-J3 for Nuclear Criticality Calculations" JAERI-M 93-190, 1993, 94p.

[27] A. G. Croff, "ORIGEN2: A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials", Nuclear Technology, Vol.62, pp. 335–352, 1983.

[28] Toden Software Incorporated, "On Steady-State BWR Core Analysis Methods", TCMO-96001, August 1996.

[29] H. Haggblom, et al., "Transmission Probability Method of Integral Neutron Transport Calculation for Two-Dimensional Rectangular Cells", Nucl. Sci. Eng., Vol. 56, p. 441, 1975.

[30] J.R Askew, "A Characteristics Formulation of the Neutron Transport Equation in Complicated Geometries", AEEW-M1108, 1972.

[31] M. Yamamoto, et al., "Development and Validation of TGBLA Lattice Physics Methods", Proceedings of the Topical Meeting on Reactor Physics and Shielding, Chicago, USA, September 17–19 1984.

[32] M. Yamamoto, et al., "Recent Developments in TGBLA Lattice Physics Code", Proceedings of the International Topical Meeting on Advances in Reactor Physics, Mathematics and Computation, Paris, France, April 27–30 1987.

[33] M. Yamamoto, et al., "New Physics Models Recently Incorporated in TGBLA", Proceedings of the International Topical Meeting on Advances in Mathematics, Computations, and Reactor Physics, Pittsburgh, USA, April 28 – May 2 1991.

[34] M. Yamamoto, et al., "Validation of the HELIOS.HX Code for High Conversion Light Water Reactor Lattice Analysis", Nucl. Technol., Vol. 80, pp. 240–249, 1988.

[35] S. Iijima, et al., "Fission Product Model for BWR Lattice Calculation Code", J. Nucl. Sci. Technol., Vol. 19, pp. 96–106, 1982.

[36] M. Yamamoto, "Coarse Time-Step Integration Method for Burnup Calculation of LWR Lattice

Containing Gadolinium-Poisoned Rods", J. Nucl. Sci. Technol., Vol. 22, pp. 1–15, 1985.
[37] R. Goldstein and E.R. Cohen, "Theory of resonance absorption of neutrons", Nuclear Science and Engineering, Vol. 13, pp. 132–140, 1962.
[38] H.C. Honeck, "THERMOS: A Thermalization Transport Theory Code for Reactor Lattice Calculations", BNL-5826, 1961.
[39] Y. Morimoto, H. Maruyama, K. Ishii, and M. Aoyama, "Neutronic Analysis Code for Fuel Assembly Using a Vectorized Monte Carlo Method", Nuclear Science and Engineering, 103, 351, 1989.
[40] Rudi J.J. Stamm'ler and M´aximo J. Abbate, "Methods of Steady-State Reactor Physics in Nuclear Design", Academic Press, London, 1983.
[41] Yoshihira Ando, Hideki Takano, "Estimation of LWR Spent Fuel Composition", JAERI-Research 99-004, 1999, 270p.
[42] Yutaka Shirai, et al., "Current Status of Light Water Reactor Fuel Upgrading", Plutonium Fuel engineering, Chap.5, Atomic Energy Society of Japan, Tokyo, 1998.
[43] Kenya Suyama, Jun-ichi Katakura, Yasushi Ohkawachi, Makoto Ishikawa, "Libraries Based on JENDL-3.2 for ORIGEN2 code: ORLIBJ32", JAERI-Data/Code 99-003, 1999, 83p.
[44] "Plutonium Fuel An Assessment", OECD, Paris, 1989.

This is a blank page.

国際単位系（SI）

表１．SI 基本単位

基本量	SI 基本単位	
	名称	記号
長さ	メートル	m
質量	キログラム	kg
時間	秒	s
電流	アンペア	A
熱力学温度	ケルビン	K
物質量	モル	mol
光度	カンデラ	cd

表２．基本単位を用いて表されるSI組立単位の例

組立量	SI 組立単位	
	名称	記号
面積	平方メートル	m^2
体積	立方メートル	m^3
速さ，速度	メートル毎秒	m/s
加速度	メートル毎秒毎秒	m/s^2
波数	毎メートル	m^{-1}
密度，質量密度	キログラム毎立方メートル	kg/m^3
面積密度	キログラム毎平方メートル	kg/m^2
比体積	立方メートル毎キログラム	m^3/kg
電流密度	アンペア毎平方メートル	A/m^2
磁界の強さ	アンペア毎メートル	A/m
量濃度(a)，濃度	モル毎立方メートル	mol/m^3
質量濃度	キログラム毎立方メートル	kg/m^3
輝度	カンデラ毎平方メートル	cd/m^2
屈折率(b)	（数字の） 1	1
比透磁率(b)	（数字の） 1	1

(a) 量濃度（amount concentration）は臨床化学の分野では物質濃度（substance concentration）ともよばれる。
(b) これらは無次元量あるいは次元１をもつ量であるが、そのことを表す単位記号である数字の１は通常は表記しない。

表３．固有の名称と記号で表されるSI組立単位

組立量	SI 組立単位			
	名称	記号	他のSI単位による表し方	SI基本単位による表し方
平面角	ラジアン(b)	rad	1 (b)	m/m
立体角	ステラジアン(b)	sr(c)	1 (b)	m^2/m^2
周波数	ヘルツ(d)	Hz		s^{-1}
力	ニュートン	N		$m\ kg\ s^{-2}$
圧力，応力	パスカル	Pa	N/m^2	$m^{-1}\ kg\ s^{-2}$
エネルギー，仕事，熱量	ジュール	J	N m	$m^2\ kg\ s^{-2}$
仕事率，工率，放射束	ワット	W	J/s	$m^2\ kg\ s^{-3}$
電荷，電気量	クーロン	C		s A
電位差（電圧），起電力	ボルト	V	W/A	$m^2\ kg\ s^{-3}\ A^{-1}$
静電容量	ファラド	F	C/V	$m^{-2}\ kg^{-1}\ s^4\ A^2$
電気抵抗	オーム	Ω	V/A	$m^2\ kg\ s^{-3}\ A^{-2}$
コンダクタンス	ジーメンス	S	A/V	$m^{-2}\ kg^{-1}\ s^3\ A^2$
磁束	ウエーバ	Wb	Vs	$m^2\ kg\ s^{-2}\ A^{-1}$
磁束密度	テスラ	T	Wb/m^2	$kg\ s^{-2}\ A^{-1}$
インダクタンス	ヘンリー	H	Wb/A	$m^2\ kg\ s^{-2}\ A^{-2}$
セルシウス温度	セルシウス度(e)	℃		K
光束	ルーメン	lm	cd sr(c)	cd
照度	ルクス	lx	lm/m^2	m^{-2} cd
放射性核種の放射能(f)	ベクレル(d)	Bq		s^{-1}
吸収線量，比エネルギー分与，カーマ	グレイ	Gy	J/kg	$m^2\ s^{-2}$
線量当量，周辺線量当量，方向性線量当量，個人線量当量	シーベルト(g)	Sv	J/kg	$m^2\ s^{-2}$
酸素活性	カタール	kat		s^{-1} mol

(a)SI接頭語は固有の名称と記号を持つ組立単位と組み合わせても使用できる。しかし接頭語を付した単位はもはやコヒーレントではない。
(b)ラジアンとステラジアンは数字の１に対する単位の特別な名称で、量についての情報をつたえるために使われる。実際には、使用する時には記号radおよびsrが用いられるが、習慣として組立単位としての記号である数字の１は明示されない。
(c)測光学ではステラジアンという名称と記号srを単位の表し方の中に、そのまま維持している。
(d)ヘルツは周期現象についてのみ、ベクレルは放射性核種の統計的過程についてのみ使用される。
(e)セルシウス度はケルビンの特別な名称で、セルシウス温度を表すために使用される。セルシウス度とケルビンの単位の大きさは同一である。したがって、温度差や温度間隔を表す数値はどちらの単位で表しても同じである。
(f)放射性核種の放射能（activity referred to a radionuclide）は、しばしば誤った用語で"radioactivity"と記される。
(g)単位シーベルト（PV,2002,70,205）についてはCIPM勧告2（CI-2002）を参照。

表４．単位の中に固有の名称と記号を含むSI組立単位の例

組立量	SI 組立単位		
	名称	記号	SI 基本単位による表し方
粘度	パスカル秒	Pa s	$m^{-1}\ kg\ s^{-1}$
力のモーメント	ニュートンメートル	N m	$m^2\ kg\ s^{-2}$
表面張力	ニュートン毎メートル	N/m	$kg\ s^{-2}$
角速度	ラジアン毎秒	rad/s	$m\ m^{-1}\ s^{-1}=s^{-1}$
角加速度	ラジアン毎秒毎秒	rad/s^2	$m\ m^{-1}\ s^{-2}=s^{-2}$
熱流密度，放射照度	ワット毎平方メートル	W/m^2	$kg\ s^{-3}$
熱容量，エントロピー	ジュール毎ケルビン	J/K	$m^2\ kg\ s^{-2}\ K^{-1}$
比熱容量，比エントロピー	ジュール毎キログラム毎ケルビン	J/(kg K)	$m^2\ s^{-2}\ K^{-1}$
比エネルギー	ジュール毎キログラム	J/kg	$m^2\ s^{-2}$
熱伝導率	ワット毎メートル毎ケルビン	W/(m K)	$m\ kg\ s^{-3}\ K^{-1}$
体積エネルギー	ジュール毎立方メートル	J/m^3	$m^{-1}\ kg\ s^{-2}$
電界の強さ	ボルト毎メートル	V/m	$m\ kg\ s^{-3}\ A^{-1}$
電荷密度	クーロン毎立方メートル	C/m^3	m^{-3} s A
表面電荷	クーロン毎平方メートル	C/m^2	m^{-2} s A
電束密度，電気変位	クーロン毎平方メートル	C/m^2	m^{-2} s A
誘電率	ファラド毎メートル	F/m	$m^{-3}\ kg^{-1}\ s^4\ A^2$
透磁率	ヘンリー毎メートル	H/m	$m\ kg\ s^{-2}\ A^{-2}$
モルエネルギー	ジュール毎モル	J/mol	$m^2\ kg\ s^{-2}\ mol^{-1}$
モルエントロピー，モル熱容量	ジュール毎モル毎ケルビン	J/(mol K)	$m^2\ kg\ s^{-2}\ K^{-1}\ mol^{-1}$
照射線量（X線及びγ線）	クーロン毎キログラム	C/kg	kg^{-1} s A
吸収線量率	グレイ毎秒	Gy/s	$m^2\ s^{-3}$
放射強度	ワット毎ステラジアン	W/sr	$m^4\ m^{-2}\ kg\ s^{-3}=m^2\ kg\ s^{-3}$
放射輝度	ワット毎平方メートル毎ステラジアン	$W/(m^2\ sr)$	$m^2\ m^{-2}\ kg\ s^{-3}=kg\ s^{-3}$
酵素活性濃度	カタール毎立方メートル	kat/m^3	$m^{-3}\ s^{-1}$ mol

表５．SI 接頭語

乗数	名称	記号	乗数	名称	記号
10^{24}	ヨタ	Y	10^{-1}	デシ	d
10^{21}	ゼタ	Z	10^{-2}	センチ	c
10^{18}	エクサ	E	10^{-3}	ミリ	m
10^{15}	ペタ	P	10^{-6}	マイクロ	μ
10^{12}	テラ	T	10^{-9}	ナノ	n
10^9	ギガ	G	10^{-12}	ピコ	p
10^6	メガ	M	10^{-15}	フェムト	f
10^3	キロ	k	10^{-18}	アト	a
10^2	ヘクト	h	10^{-21}	ゼプト	z
10^1	デカ	da	10^{-24}	ヨクト	y

表６．SIに属さないが、SIと併用される単位

名称	記号	SI 単位による値
分	min	1 min=60 s
時	h	1 h =60 min=3600 s
日	d	1 d=24 h=86 400 s
度	°	1°=(π/180) rad
分	’	1’=(1/60)°=(π/10 800) rad
秒	”	1”=(1/60)’=(π/648 000) rad
ヘクタール	ha	1 ha=1 $hm^2=10^4m^2$
リットル	L, l	1 L=1 l=1 $dm^3=10^3cm^3=10^{-3}m^3$
トン	t	1 t=10^3 kg

表７．SIに属さないが、SIと併用される単位で、SI単位で表される数値が実験的に得られるもの

名称	記号	SI 単位で表される数値
電子ボルト	eV	1 eV=1.602 176 53(14)×10^{-19}J
ダルトン	Da	1 Da=1.660 538 86(28)×10^{-27}kg
統一原子質量単位	u	1 u=1 Da
天文単位	ua	1 ua=1.495 978 706 91(6)×10^{11}m

表８．SIに属さないが、SIと併用されるその他の単位

名称	記号	SI 単位で表される数値
バール	bar	1 bar=0.1MPa=100 kPa=10^5Pa
水銀柱ミリメートル	mmHg	1 mmHg≈133.322Pa
オングストローム	Å	1 Å=0.1nm=100pm=10^{-10}m
海里	M	1 M=1852m
バーン	b	1 b=100fm^2=$(10^{-12}cm)^2$=$10^{-28}m^2$
ノット	kn	1 kn=(1852/3600)m/s
ネーパ	Np	SI単位との数値的な関係は、対数量の定義に依存。
ベル	B	
デシベル	dB	

表９．固有の名称をもつCGS組立単位

名称	記号	SI 単位で表される数値
エルグ	erg	1 erg=10^{-7} J
ダイン	dyn	1 dyn=10^{-5}N
ポアズ	P	1 P=1 dyn s cm^{-2}=0.1Pa s
ストークス	St	1 St =1$cm^2\ s^{-1}$=$10^{-4}m^2\ s^{-1}$
スチルブ	sb	1 sb =1cd cm^{-2}=10^4cd m^{-2}
フォト	ph	1 ph=1cd sr cm^{-2} =10^4lx
ガル	Gal	1 Gal =1cm s^{-2}=$10^{-2}ms^{-2}$
マクスウエル	Mx	1 Mx = 1G cm^2=10^{-8}Wb
ガウス	G	1 G =1Mx cm^{-2} =10^{-4}T
エルステッド(a)	Oe	1 Oe ≙ $(10^3/4\pi)$A m^{-1}

(a) ３元系のCGS単位系とSIでは直接比較できないため、等号「 ≙ 」は対応関係を示すものである。

表10．SIに属さないその他の単位の例

名称	記号	SI 単位で表される数値
キュリー	Ci	1 Ci=3.7×10^{10}Bq
レントゲン	R	1 R = 2.58×10^{-4}C/kg
ラド	rad	1 rad=1cGy=10^{-2}Gy
レム	rem	1 rem=1 cSv=10^{-2}Sv
ガンマ	γ	1 γ=1 nT=10^{-9}T
フェルミ		1 フェルミ=1 fm=10^{-15}m
メートル系カラット		1 メートル系カラット= 0.2 g = 2×10^{-4}kg
トル	Torr	1 Torr = (101 325/760) Pa
標準大気圧	atm	1 atm = 101 325 Pa
カロリー	cal	1 cal=4.1858J（「15℃」カロリー），4.1868J（「IT」カロリー），4.184J（「熱化学」カロリー）
ミクロン	μ	1 μ =1μm=10^{-6}m

（第8版，2006年）