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BURNUP CREDIT CRITICALITY BENCHMARK
— RESULT OF PHASE-1A —

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日 本 原 子 力 研 究 所
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OECD/NEA
Burnup Credit Criticality Benchmark
- Result of Phase-1A -

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The report describes the final result of the Phase-1A of the Burnup Credit Criticality Benchmark conducted by OECD/NEA. The Phase-1A benchmark problem is an infinite array of a simple PWR spent fuel rod. The analysis has been performed for the PWR spent fuels of 30 and 40 GWd/t after 1 and 5 years of cooling time. In total, 25 results from 19 institutes of 11 countries have been submitted.

For the nuclides in spent fuel, 7 major actinides and 15 major fission products (FP) are selected for the benchmark calculation. In the case of 30 GWd/t burnup, it is found that the major actinides and the major FPs contribute more than 50% and 30% of the total reactivity loss due to burnup, respectively. Therefore, more than 80% of the reactivity loss can be covered by 22 nuclides. However, the larger deviation among the reactivity losses by participants has been found for cases including FPs than the cases with only actinides, indicating the existence of relatively large uncertainties in FP cross sections. The large deviation seen also in the case of the fresh fuel has been found to reduce sufficiently by replacing the cross section library from ENDF-B/IV with that from ENDF-B/V and taking the known bias of MONK6 into account.

Keywords: Burnup Credit, Benchmark, Criticality, OECD, NEA, Reactivity Loss, Actinide, Fission Product, Calculation, Comparison, PWR Fuel

ＯＥＣＤ／ＮＥＡ
燃焼度クレジット臨界計算ベンチマーク
—フェーズ１Ａの結果—

日本原子力研究所東海研究所燃料サイクル安全工学部
高野　誠

（1994年１月５日受理）

本報は，ＯＥＣＤ／ＮＥＡで行われた燃焼度クレジット臨界ベンチマーク計算フェーズ１Ａに対する各国の最終結果を取りまとめたものである。フェーズ１Ａの問題は，簡単なＰＷＲ使用済燃料棒の無限配列格子計算であり，25の結果が11ヶ国の19機関から提出された。ベンチマーク問題では，燃焼度30ＧＷｄ／ｔ，40ＧＷｄ／ｔ，冷却期間１年と５年のものを解析対象とした。

使用済燃料棒中の核種として，主要アクチニド７核種，主要核分裂生成物（ＦＰ）15核種を使用した。燃焼度が30ＧＷｄ／ｔのとき，燃焼による反応度損失の約50％以上を主要アクチニドが，さらに30％以上を主要ＦＰが分担しており，主要アクチニドと主要ＦＰの計22核種で80％以上の反応度損失を考慮できることがわかった。また，主要アクチニドに比べ主要ＦＰによる反応度損失の評価に対する参加者間の偏差が大きく，これはＦＰ断面積の不確実性が比較的大きいことを示唆するものと考えられる。また，新燃料の固有値に対する大きな偏差は，ＥＮＤＦ－Ｂ／ⅣからＢ／Ⅴへの改訂と，ＭＯＮＫ６の有する既知のバイアスを考慮することで十分小さくなることがわかった。

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1. Introduction

In many countries, the criticality evaluation of spent fuel storage or transport casks assumes that fresh fuel is loaded instead of spent fuel. For criticality safety analysis, this assures a safety margin by simple calculations. However, this results in an excessively large criticality safety margin for typical burned light water reactor (LWR) fuel. There are indications that the nuclear power industry is moving to increased fuel enrichments in order to attain even higher burnups in LWRs. The resulting spent fuel can no longer be treated as if it were fresh. Current interest in plutonium recycle also provides an incentive to modify the fresh fuel transport assumption in that it inherently requires that spent LWR fuel be transported to and from reprocessing facilities on a timely and efficient basis. A balance needs to be sought between the two requirements: reduction of cost and optimal safety.

Conventional reactor codes and data used for in-core physics calculations can be used to evaluate the criticality state of burned LWR fuel. However, these codes involve complicated models and have large computational and data requirements. The objective of this benchmark is to verify that simple models using established away-from-reactor codes (KENO, MCNP, etc.) can be used to evaluate the criticality safety margin for spent fuel systems. The benchmark program includes PWR, BWR and MOX fuels and is designed to study the various effects on safety margins, including the effect of various fission products and axial burnup distributions taking fuel enrichment, burnup and cooling time as parameters.

This report describes the results of part 1 of the benchmark series which can be classified as a preliminary survey of Burnup Credit where burnup, cooling time and group of nuclides are taken as parameters. In total, 25 different results have been submitted from 19 institutes of 11 countries. These results have been compared with each other in graphical form to identify sources of discrepancies. The identification of the origin of the differences in this phase (Part 1) is very important so that one can proceed with

confidence into the next phase where a more complex problem will be tackled. Part 2 of the Benchmark will be the analysis of a PWR fuel assembly and a transport cask taking axial burnup distribution into account.

2. Benchmark Specification

The Burnup Credit Benchmark consists of 13 cases. Each case is an eigen-value calculation of a simple infinite cell of a PWR fuel rod. The investigated parameters are burnup, cooling time and combination of nuclides in fuel region. These parameters are shown in Table 2.1 together with case numbers. Table 2.2 shows the relation between nuclides and their groupings, i.e., major actinides, minor actinides, major FPs and minor FPs. The specification of cases 1 to 9 was finalized in January 1992 and was published as NEACRP-L337 Revision 1 (see Appendix 1). Further, during the first meeting of the Benchmark held in June 1992, Cases 10 to 13 were introduced additionally, in order to investigate the reactivity effects by minor actinides and minor FPs.

From the result of the benchmark, various effects on the reactivity loss due to burnup can be shown.

3. Nuclear Data and Codes

In this benchmark, the most probable cause of the differences observed among participants would be the nuclear data since the geometry model is quite simple and in addition, atomic number densities are given. The followings are the brief description of nuclear data and analysis code employed by each participant. Additional comments from participants are also given here. Table 3.1 summrizes the data and the codes of participants.

(1) Belgonuclear, Belgium

a) General Data

*Institute : Belgonuclear SA, Anenue Ariane 4,

1200 Bruxelles, Belgium.
*Participants : Y. Vanerborck, Th. Maldague
*Computer Code : LWR-WIMS, from Reactor Systems Analysis Division AEE, Winfrith, Dorchester, Dorset.
*Data Library : LWR-WIMS-VERSION 1986
*No. of Groups : 69 Energy Groups

(2) Ontario Hydro, Canada

a) General Data

*Institute : Ontario Hydro
*Participant : H. J. Smith
*Computer Code : WIMS-AECL
*Data Library : Winfrith Library Version 90-06-09
*No. of Groups : 69 Energy Groups

b) Comments

The composition of " Zircalloy " was not specified. For these calculations 100% Zr-91 in the Winfrith Library was used. Zr-91 represents Zr-natural. Pu-238 is not available in our version of the library. The reactivity effect of adding materials to make Zr-4 is -0.3 mk.

(3) Technical Research Center of Finland, Finland (see also Appendix 2.1)

a) General Data

*Institute: Technical Research Center of Finland , Nuclear Engineering Laboratory.
*Participant : Markku Anttila
*Computer Code : CASMO-3 (version 4.4, 90/11/20)
*Data Library : CASMO Library, update J, (E4LTJ40, 880818)
*No. of Groups : 40 Energy Groups

b) Comments

The nuclides, Mo-95, Tc-99 and Ru-101 are not included in the library. The nuclides, Gd-155 and O-16 are included in the calculation but the current CASMO code does not print out their reaction rates.

The nuclides, U-235, U-236, U-238 and Pu-239 are treated as resonance absorbers (self-shielding taken into account).

The composition of zircalloy used for cross section generation is not known.

(4) CEA, France

a) General Data

Institute : CEA/DRN and CEA/IPSN**, France

participants : A. Santamarina, P. Albarede*
L. Maubert**, G. Poullot**

*Computer Code : APOLLO1

*Data Library : CEA 86

*No. of Groups : 99 Energy Groups

b) Comments

The resonance shielding of U-235, U-236, U-238,Pu-239, Pu-240, Pu-241 and Pu-242 are considered by the LIVOLANT formula (effective cross-section tabulation, rigorous equivalence heterogeneous/homogeneous)

We give absorption also for zirconium, water and oxygen. The normalization is : one source neutron. For each case, you can check that :

$$\sum_{\text{isotopes}} \text{Production} = k_{\text{eff}}, \quad \text{(multiplication factor)}$$

$$\sum_{\text{isotopes}} \text{Absorption} = P \geq 1 \quad (\text{P neutron is absorbed, and } P > 1 \text{ because of (n,2n)}).$$

Concerning the Cases 10 to 13, we have used the APOLLO-KAFKA package in order to predict the inventory of high burn-up PWR assemblies. Two hundred fission products are accounted for the depletion calculation of a 17 x 17 PWR fuel assembly. We used the 99-group library, so called CEA-86 library, which is based on the JEF-1 file, for F.P. cross sections. The resulting decrease in reactivity confirms the validity of the CEA/AEA choice of the 15 F.P.s to account for burn-up credit in criticality studies; these seleced 15 F.P. nuclides, involved in the CEA/AEA CERES experimental program, represent 80% of the total F.P. poisoning.

(5) IKE/BFS, Germany

a) General Data

*Institute : Institute for Kernenergetik and Energy Systems, University of Stuttgart, Germany (IKE), Bundesamt für Strahlenschutz, Salzgitter, Germany (BFS).

*Participants : D, Lutz, W. Bernnat(IKE). H. -H Schweer(BFS).

*Computer Code : CGM (IKE-development)

*Data Library : JEF-1, Multi-group Library generated by NJOY.

*No. of Groups : 242 Energy Groups

*Spectrum Data revised on 22 June 1993

(6) GRS, Germany

a) General Data

*Institute : Gesellschaft für Anlagen- und Reaktor- sicherheit (GRS) mbH.

*participants : W. Weber, H. Krug, B. Gmal, W. Heinicke, E. F. Moser.

*Computer Code : SCALE-4 (CSAS1X)

*Data Library : 27 BURNUPLIB

*No. of Groups : 27 Energy Groups

(7) ENEA, Italy

a) General Data

*Institute : ENEA, Viale Regina Margherita 125, 00198 Roma, Italy

*Participant : Francesco Siciliano

*Computer Code : MCNP-V3

*Data Library :

Following nuclides are processed by NJOY code employing JEF-1 library :

H-1,	O-16,	Cr-24,	Fe-26,	Gd-155,
U-234,	U-235,	U-236,	U-238,	Pu-238,
Pu-239,	Pu-240,	Pu-241,	Pu-242.	

Following nuclides are processed by THEMIS code employing JEF-1 library :

Tc-99, Rh-103, Cs-133, Nd-143, Nd-145,
Sm-149, Sm-150, Sm-151, Sm-152, Eu-153.

The ENDL-85 of the original MCNP library is employed for the following nuclides:

Zr-40, Ag-109, Sn-50, Np-237, Am-241,
Am-243.

*No. of Groups :
(Continuous)

b) Comments

The absorption reaction data of the nuclides Mo-95 and Ru-101 and Sm-147 are not taken into account since the corresponding neutron data sets are not available.

(8) Hokkaido Univ., Japan

a) General Data

*Institute : Hokkaido University, Sapporo, Japan.
*Participant : M. Narita
*Computer Code : SRAC
*Data Library : ENDF-B/4, JENDL-2
*No. of Groups : 45 Energy Groups

b) Comments

The nuclides, Am-243 and Np-237, are not included in the calculation.

(9) JAERI, Japan

a) General Data

*Institute : Japan Atomic Energy Research Institute, Tokai, Ibaraki, Japan.
*Participant : Makoto Takano
*Computer Code : JACS Code System (ANISN,KENO-IV), MCNP-4
*Data Library :

The multi-group cross section library for ANISN and KENO-IV codes are generated from JENDL-3, processed by MGCL-ACE code, a part of JACS Code System.

The continuous energy library for MCNP-4 is generated from JENDL-3, by NJOY code.

*No. of Groups : 137 Energy Groups (ANISN, KENO)
Continuous Energy (MCNP)

b) Comments

In the multi-group cross section library, all resonance absorbers are self-schielded by employing f-tables.

(10) JINS, JAPAN (see also Appendix 2.2)

a) General Data

*Institute : Japan Institute of Nuclear Safety, Tokyo, Japan.
*Participants : Susumu Mitake
*Computer Code : SCALE-4 (CSAS1X)
BONAMI + NITAWL + XSDRNPM-S
1-D discrete oridinate calculation
*Data Library : 27 BURNUPLIB from ENDF/B- IV and V
*No. of Groups : 27 Energy Groups

(11) PNC (O-Arai), Japan

a) General Data

*Institute : PNC O-arai, Japan
*Participants : Nobuo O-tani
*Computer Code : SCALE-4 (CSAS25)
BONAMI + NAITAWL + KENO-V.a
*Data Library : 27 BURNUPLIB from ENDF/B-IV and V
*No. of Groups : 27 Energy groups

(12) PNC (Tokai), Japan

a) General Data

*Institute : PNC Tokai, Japan
*Participant : Ichiro Nojiri
*Computer Code : SCALE-4 (CSAS1X)
*Data Library : 27 BURNUPLIB
*No. of Groups : 27 Energy Groups

(13) Toshiba, Japan

a) General Data

*Institute : Toshiba/NEL(Nuclear Engineering Laboratory)
*Participant : Munenari Yamamoto

*Computer Code : TGBLA

*Data Library :

U-235, 238, Pu-239, 240, Th-232	----- ENDF/B-V
U-233, 236, Pu-241, 242	----- ENDF/B-IV
Other fissiles and fertiles	----- JENDL-3
Fission Products	----- mainly from ENDF/B-IV and JENDL-2
Other	----- mainly from ENDF/B-IV

*No. of Groups : 95 Energy Groups

b) Comments

The nuclide Mo-95 is not explicitly treated in our FP model. "NU" (neutrons/fission) is defined as the ratio of production to fission rate, not like the definition given in p.6 of NEACRP-L-337.

(14) CSN, Spain

a) General Data

*Institute : CONSEJO DE SEGRIDAD NUCLEAR, MADRID SPAIN

*Participant : A.I. ALVARED, JM. CONDE, M. RECIO

*Computer Code : CASMO-3, VERSION 4.7

*Data Library : E4LTJB7, BASED ON ENDF/B-4 AND OTHER SOURCES. 93 MATERIALS, SAME GROUP STRUCTURE THAN IN WIMS.

*No. of Groups : 70 Energy Groups

b) Comments

The nuclides Mo-95, Tc-99 and Ru-101 are not included in the library.

Likewise, the O-16 absorption reaction rate is not given by the code. The neutron data library is based primarily on data from ENDF/B-4, although some data come from other sources.

Microscopic cross-sections are tabulated in 70 energy groups, with the structure indicated in the file. This structure was taken from the WIMS code with the addition of one boundary at 1.855 eV.

Four nuclides, U-235, 236, 238 and Pu-239 are tabulated as resonance absorbers. The 1 eV resonance in Pu-240 and the 0.3 eV resonance in Pu-239 are considered to be adequately covered by the concentration of thermal groups around them and consequently excluded from the special resonance treatment. The basic principles for the resonance treatment are similar to those in the WIMS code.

Validation and QA of the CASMO code has been extensive.

(15) E. M. Systems Sweden (see also Appendix 2.3)

a) General Data

*Institute : E Mennerdahl Systems Vallentuna, Sweden.

*Participant : Dennis Mennerdahl

*Computer Code : SCALE-4 (XSDRPM-S and CSAS1X)

*Data Library : 27 BURNUPLIB

*No. of Groups : 27 Energy Groups

(16) Studsvik, Sweden (see also Appendix 2.4)

a) General Data

*Institute : Studsvik Core Analysis AB.,Nyköping, Sweden

*Participant : Kim Ekberg

*Computer Code : CASMO-3, version 4.7, creation date 92-03-02

*Data Library :

E4LBJB40. This is the standard CASMO neutron data library in 40 energy groups. The data are mainly from ENDF/B-4, but there are some data from ENDF/B-5 and some from JEF-2. The data have been processed with NJOY to the basic CASMO 70 group format. The condensation from 70 to 40 groups has been done with representative LWR spectra, different for different materials. A calculation with 70 groups could eqally well have been done, and it would have given results very close to the present results.

*No. of Groups : 40 Energy Groups

b) Comments

The number densities given in the Appendix to NEACRP-L-337 have NOT been used. It is not meaningful to introduce into CASMO number densities from other sources, and therefore number densities at all exposures have been calculated with CASMO. For number densities after 1 and 5 years of decay an option of CASMO, Shut Down Cooling, has been used.

(17) AEA (Culcheth), U.K.

a) General Data

*Institute :	Safety & Reliability Directorate, AEA Technology, Culcheth, Warrington WA3 4NE, U.K. .
*Participant :	A. J. Rudge
*Computer Codes :	Part-1 LWRWIMS2a (Cases 1-13.) Part-2 MONK6B (Cases 1, 4, 5, 8, 9, 10, 11)
*Data Library :	Part-1 the 1986 WIMS 69 Group library (identification WEM3N/35) Part-2 MONK6 Point Nuclear Data Library
*No. of Groups :	Part-1 69 Energy Groups Part-2 8220 cross-section energy groups from 15 MeV to 0 MeV.

b) Comments

The MONK-6 data library does not contain fission product data. Calculations were only performed for cases with no fission product selection. Zircaloy was represented as zirconium in all calculations with a density of 6.5 g/cm^3. Water was taken to have a dentisy of 0.9965 g/cm^3.

Neutronics data is appended on a case basis. It has only been provided for the LWRWIMS calculations since these comprise the complete study. The MONK-6 calculations have been included for completeness but only covered part of the study.

The computer code LWRWIMS2a was used throughout the study in default mode. Hence all relevant information can be obtained from the following reference - LWRWIMS User Guide, ANSWERS (LWRWIMS) 3, AEEW-R2444, July 1991.

Part1 - LWRWIMS2a Data

Referring to the neutron flux data, details are provided for all 69 groups. To obtain more accurate sampling two mesh regions of equal volume were modelled for the fuel only.

Neutrons per fission can be simply derived by dividing the relevant values in the 'neutron yields by nuclide' table by the relevant values in the 'fission by nuclide' table.

Note that new data has not been provided for the rerun Cases 2&6. The data supplied previously is considered adequate.

(18) AEA (Winfrith), UK

a) General Data

*Institute : AEA Technology, Winfrith,UK
*Participant : N. T. Gulliford
*Computer Code : LWRWIMS (PERSEUS) run on SUN Workstation
*Data Library : WIMS 1986 Nuclear Data Library
*No. of Groups : 69 Energy Groups

(19) BNF, UK (see also Appendix 2.5)

a) General Data

*Institute : British Nuclear Fuels plc, Risley, Warring ton, Eng/and.
*Participant : P. R. Thorne, P. E. Broome
*Computer Code : MONK6B
*Data Library : MONK6B 8220 Point Energy Library, derived from UKNDL and JEF-2.
*No. of Groups : (Continuous)

(20) Department of Transport, UK

a) General Data

*Institute : UK department of Transport
*Participant : J.T.Stewart
*Computer Code : MONK6B (CESIUS VERSION - CES-3)
*Data Library : MONK6B 8220 Point Energy Library

*No. of Groups : (Continuous)

b) Comments

The missing nuclides in the calculations are Mo-95, Tc-99, Ru-101, Rh-103, Cs-133, Sm-147, Sm-149, Sm-150, Sm-151, Sm-152, Nd-143, Nd-145 and Gd-155.

(21) ORNL, USA (see also Appendix 2.6)

a) General Data

*Institute : Oak Ridge National Laboratory
Participant : Michaele C. Brady , Mark D. DeHart**
Computer Code : SCALE-4 (CSAS1X) , SCALE-4.2**
Data Library : 27 BURNUPLIB ,
No. of Groups : 27 Energy Groups .
44 group, ENDF-5,collapsed from 238 group**
The result by SCALE-4.2 was supplied in September 1993.

4. Results and Discussions

In the report, results are shown up to 4 digits after the decimal point since most of the participants have supplied the data with more than 4 digits after the decimal point. However, readers should keep in mind that significant digits are usually 3 or so in this type of calculation.

4.1 Multiplication factors

All of the multiplication factors from participants are listed in Table 4.1 together with the values of average and 2*σ standard deviation, where the value σ is obtained by;

$$\sigma = \sqrt{\frac{\sum_{i=1}^{n} (x_i - \bar{x})^2}{n}}$$

n : Number of data, $\bar{x}$: Average, x_i : i-th data

In the table, there are three sets of groups, i.e., groups A, B and C. The group A consists of the data from all

participants and the group B (23 Participants) excludes the data of Studsvik and Dept. of Transport from the group A since Studsvik has employed the different atomic number densities and Dept. of Transport has exculded 5 important FPs in the calculation. The group C (17 Participants) consists of the participants whose input data fully meet the benchmark specification. However, most of the participants in the group B meet the benchmark specification for no-FP cases. In the report, the data from the group C is used as a reference but the group B data is also useful for comparison among no-FP cases.

Table 4.2 and Fig. 4.1 show the relative differences from the average of Group C (17 Participants). Figure 4.1 indicates that the most of the cases are within ± 1% of relative difference from the average and the large relative differences are seen for the cases with FPs. In Fig. 4.2, the average and the 2σ values of 3 groups of participants are plotted for all cases. From Fig. 4.2(B), 2σ of the 17 participants (Group C) is about 0.01 for no-FP cases, i.e., cases 4, 5, 8, 9, 10 and 11. For cases 2, 3, 6, 7, 12 and 13 (with FPs), the 2σ value exceeds 0.015. At this point, components of the 2σ deviation are not identified, but it can be said that the differences among F.P. cross sections used by the participants are relatively larger than those of actinides. The component will be analyed and discussed in the next section.

From Fig. 4.2(B), the 2σ deviation of case 1 (Fresh Fuel) reaches more than 0.015. For the case 1, major nuclides are only U-235 and U-238 in the fuel region but rather large deviation has been observed. In order to see in more detail, the relative difference of only case 1 is extracted from Fig. 4.1 and is shown as Fig. 4.3. When we look at the figure by code or data, the differences from the average are about +1.4, +0.5, +0.5 and -0.8 % for the results by MONK6B, WIMS, JENDL3 and SCALE4 respectively. However, ORNL supplied us the result by SCALE4.2, the latest version of SCALE with the cross section library from ENDF-B/V, just after the meeting at ORNL in September 1993. The result by SCALE4.2 has become very close to the average as seen in Fig. 4.3 and is favorable for a better agreement among participants. Assuming that SCALE4.2

were used instead of SCALE4 by all SCALE users in the participants, the 2σ deviation of case 1 would reduce from 0.0168 to 0.0119. Further, the MONK6 is know to have about 1 % positve bias for the uranium system as described in the Reference (1). Taking this bias into consideration, the 2σ deviation will be further improved to 0.0061.

For the other cases, the assumption, to replace the SCALE-4 result with the SCALE-4.2 result, will also reduce the deviation, since the results of cases 2 to 13 by SCALE-4.2 have become closer to the average than those by SCALE-4 as seen in Fig. 4.1. For more detail on the difference between SCALE-4 and SCALE-4.2, please see the Appendix 2.6.

4.2 Reactivity Decrease

The reactivity decrease due to burnup is shown in Table 4.3. The values in the table are Δk, namely,

$$\Delta k_N = k(\text{Case N}) - k(\text{Case 1}) \ .$$

The relative difference, $(\Delta k/\Delta k_{avg}-1.0)X100(\%)$, of each participant from the average of Δk values of Group C (17 Participants) are shown in Fig. 4.4. In Fig. 4.5, the average Δk and its 2σ deviation are shown. It can be seen that the reactivity decreases with burnup and cooling time. The reactivity loss caused by both actinides and major FPs becomes larger with burnup and cooling time as shown in Table 4.4. When the burnup increases from 30 GWd/t to 40 GWd/t, the reactivity decreases 0.0764 Δk and 0.0883 Δk after 1 and 5 years of cooling time respectively. The effects of cooling time from 1 year to 5 years, on the other hand, are 0.0279Δk and 0.0398Δk for 30 GWd/t and 40 GWd/t burnups respectively.

From the average Δk decrease in Table 4.3, the reactivity losses caused by major actinides, minor actinides, major FPs and minor FPs are summarized as shown in Table 4.5 and Fig. 4.6. From the table, it can be said that the major actinides (7 nuclides) contribute more than 50 % of the total Δk decrease but the minor actinides (5 nuclides) less than 10 %. In the case of fission products, the major FPs (15 nuclides)

contribute more than 30 % of total Δk decrease but the rest of FPs only about 10 % . Further, the major actinides cover more than 80 % of the Δk caused by all actinides. The major FPs also cover nearly 80 % of the Δk by all FPs. In the practical application of the burnup credit, it is important for analysis to select the minimum set of nuclides out of more than 200 nuclides existing in a spent fuel. The above result shows the selection of nuclides in the report are quite adequate for practical applications. For example, when we take only 7 nuclides (Major Actinides) as the minimum set of nuclides, they contribute more than 50 % of reactivity loss due to burnup. In this case, each major actinide has the equivalent reactivity contribution of 7 % to 8 % in average. Adding 15 major FPs to the major actinides, resulting 22 nuclides are able to represent more than 80 % of the reactivity loss. The average of equivalent reactivity contribution by one major FP is more than 2 %.

In order to obtain the 2*σ deviations in Table 4.5, the Δk values of each participant are calculated for all sets of nuclides, i.e., the major and the minor actinides, and the major and the minor FPs. From the 2σ and the 2σ/Δk values of the major FPs, it seems that there exists relatively large discrepancy among participants in the neutron data of the major FPs. It is considered that the discrepancy is mainly originates from the uncertainty of FP neutron data. In the table, large 2*σ deviations are also seen for the cases of the minor FPs. This is, however, not caused by the discrepancy of FP cross section data, but simply originates from the variation of nuclides included as the minor FPs among participants in the calculation. Since there are so many FP nuclides, participants were encouraged to include the minor FP nuclides as much as possible for the calculation.

From Fig. 4.4, it is seen that the SCALE-4.2 agrees with the average Δk better than the SCALE-4. Therefore, when all SCALE-4 data are replaced by those of SCALE-4.2, it is expected that the 2*σ deviation of Δk caused by the actinides will improve from 0.0104 to 0.0071 (average of cases 4, 5, 8 and 9), and by the actinides plus the major FPs from 0.146 to 0.136 (average of cases 2, 3, 6 and 7).

4.3 Neutron Spectrum

In the multi-group calculation, the number of energy groups varies from 27 to 247 among participants. Also the maximum upper energy boundaries vary from 20 MeV to 8.2 MeV. The energy group structures are shown in Fig. 4.7. The data sent from some participants were not spectrum but group-wise flux, flux divided by energy width ,flux multiplied by region volume, and so on. They are ,however, converted into spectrum for mutual comparison.

The neutron spectrum in fuel and moderator regions for Cases 1, 7 and 9 are plotted in Figs. 4.8 to 4.13. The spectra of the other cases are almost the same as shown in these figures. In these figures, all legends are placed at the mid-point of each energy group for mutual comparison. The mid-point is calculated to be at the center of the energy group in the lethargy term. In other words, the upper and lower energy boundaries of each group are converted into the unit of lethargy. Then, the mid-point in lethargy is obtained and this mid-point is again reconverted into the unit of eV. From the figures, the effects of Pu resonances are clearly seen at around 0.3 eV and 1.0 eV in the fuel region. In the moderator region, the effects of Pu resonances are also seen but becomes smaller than those of fuel region.

The spectra in these figures agree with each other quite well. The spectra preliminary shown in the draft report have been revised by replacing the data of IKE/BFS with their corrections. From the figures, however, the spectra of CEA seem to be shifted to lower energy side. Their profiles are quite similar to others. This may happen when the supplied energy data from CEA does not correspond to the upper energy boundary of the group as specified in the specification. The shift of the CEA spectrum is clearly observed also in Fig. 4.14 where the dip casued by the resonace of Pu is seen.

Another discrepancy appears from 5 MeV as shown in Fig. 4.15. Above 5 MeV, there seems to exist two curves of spectrum when we exclude the CEA data. The upper curve shows the data of JAERI(Anisn) and JAERI(Keno). The middle curve is

formed by the rest of participants including JAERI(Mcnp). Here, the cross section libraries of JAERI(A), (K) and (M) are produced from the same neutron data file, JENDL-3. Both JAERI(A) and JAERI(K) use the 137 group library (named as MGCL) as a part of JACS code system, but JAERI(M) uses the continuous energy library edited by NJOY. Therefore, the cause of the discrepancy may exist in MGCL. A similar discrepancy caused by MGCL has been reported by Mr. Mitake of JINS as described in the Appendix 2.2. The investigation of the JACS code system has revealed that the fission spectrum used in the system is slightly harder than expected and the corrective work has been initiated.

4.4 Reaction Rates

The reaction rates are shown in graphical form in order to help the participants in identifying possible causes of discrepacy involved in their data and code.

In order to compare reaction rates supplied from participants, we have employed the data which fully meet the benchmark specification. Namely, the reaction rate data produced by the calculation employing all of the specified nuclides (Group C or 17 Participants) and supplied by the computer readable media. Therefore, the data recorded on sheets of paper was not considered except the values of multiplication factors. Further, the reaction rate data from different participants, but produced by the same code system, such as SCALE-4, have been represented by one data. Under the above mentioned condition, the reaction rates produced by SCALE-4, MONK6B, MCNP-4(JENDL-3), LWR-WIMS, JACS(K), CGM and APOLLO1 have been compared.

In the following comparison, both absorption and production reaction rates are normalized to unity. It may be convienient for comparison without the normalization to identify the absolute difference of reaction rates among participants, however, it will somtimes draw misleading result. When we compare reaction rates of two participants, for example, in order to identify the nuclide causing the discrepancy of two multiplication factors, the difference of the production

reaction rates may conveniently tell you nuclides which cause the eigen-value discrepancy. But, the identified nuclides may not be responsible for the discrepancy, rather, the difference between the production reaction rates of the nuclides may just reflect the differences of absorption reaction rates of various nuclides since the sum of the absorption rates is always adjusted to unity.

It is not an easy task to identify the major nuclides which cause the discrepancy in the multiplication factors, however, it can be said that only the large differences observed among participants should be focused. But it may be difficult to deduce any result by looking at small differences of reaction rates, since they are probably induced just by adjusting the large differences due to normalization.

(a) Absorption Reaction Rates

For each nuclide, the absorption reaction rates evaluated by 7 different codes are shown in one figure. When the sum of absorption reaction rates calculated by a code does not equal unity, the normalization has been done to make the sum unity. The re-normalization was also necessary for the cases where the reaction rates were evaluated over the region containing both fuel and moderator.

The differences from the average reaction rate are shown in the Appendix 3.1 for each nuclide. The difference in positive value induces the multiplication factor lower and vice versa.

From these figures, relatively large differences of around 0.004 to 0.007 are observed for the nuclides, U-238, U-235 and Pu-239. Around 0.001 to 0.002 of differences are seen for Pu-240, Pu-241, Gd-155, Nd-143, Rh-103, Sm-149, Sm-151 and Tc-99. Since the FP nuclides are all absorbers, the differences of Gd-155, Nd-143, Rh-103, Sm-149, Sm-151 and Tc-99 can be considered as the main reason of larger 2σ values of multiplication factors as already seen in Section 4.1.

(b) Production Reaction Rates

In the same manner, the differences of production reaction rates are shown in the Appendix 3.2. Rather large differences around 0.006 to 0.008 are seen for Pu-239, U-235 and U-238. But for the remaining nuclides, the differences are less than 0.001.

(c) ν values

The definition of ν values in the original benchmark specification was not adequate, however, most of the participants were aware of it and supplied the data by using appropriate definition as given in the revised benchmark specification (see Appendix 1). In the figures, the data by the MONK-6 has been omitted since the statistical error for ν values are relatively large. Figures in the Appendix 3.3 show the relative differences of ν values from the average. It should be noted here that the differences shown in these figures may include the differences originated from the participants' definitions of ν values.

(d) Comparison

The absorption reaction rate of Case 7 (40 GWd/t, 5 years, with FP), is plotted in Fig. 4.16 as an example. The nuclides shown in the figure have differences in absorption rates more than 0.001 among participants, namely, each nuclide can be the source of more than 0.1 % Δk discrepancy. From the figure, the LWR-WIMS has large negative differences for FP nuclides, especially Sm-149 and Sm-151. The absorption cross section of these nuclides might be one of the possible reasons for higher multiplication factors seen in the cases with FPs, as already observed in Fig. 4.1. The LWR-WIMS also shows the large positive differences for actinides. However, they seem simply compensating the negative differences in order to make the total absorption rate by all nuclides to unity, since the multiplication factors of no-FP cases (namely only actinides) by LWR-WIMS are very close to the average.

In the same manner, the differences of production rates and ν values of case 7 are shown in Fig. 4.17. It is seen that

the JACS(K) and the MCNP(JENDL) do not show the same tendency despite the fact that both cross section libraries are generated from JENDL-3. For example, the large positive difference of U-238 production rate may correspond to the higher multiplication factor than MCNP(JENDL3) for fresh fuel, and the large negative difference of Pu-239 absorption rate in Fig. 4.18 may add another cause of discrepancy leading the lager difference for spent fuel cases. From the figure of ν values, both JACS(K) and MCNP(JENDL3) show the same tendency but always disagree with other codes. This may lead to check the ν data in JENDL-3 neutron data file.

It may be possible to obtain other important findings from these data, but this will be left to readers. Because the parameters are too many, i.e., 7 codes, 13 cases, 27 nuclides for absorption rate and 12 nuclides both for production rate and for ν value, resulting 4641 data to be compared. Since most of the readers will focus on one particular code, the number of data to be compared will be greatly reduced. In Figs. 4.19 to 4.21, the reaction rates of all nuclides for cases 1 to 9 are drawn as stacked graphs in order to show the relative importace of each nuclide to the total reaction rate. The data is taken, as an example, from the result of the JAERI(Anisn) calculation.

5. Concluding Remarks

As the first step of the burnup credit benchmark exercises, a simple PWR pin cell has been selected and in total 13 cases have been analyzed by taking the burnup, cooling time and group of nuclides as parameters. The reactivity loss due to burnup has been calculated and its average varies from 0.298 Δk for 30 GWd/t after 1 year cooling to 0.414 Δk for 40 GWd/t after 5 years cooling. The reactivity loss has been divided into four contributions, namely, by 7 major actinides, 5 minor actinides, 15 major fission products (FPs) and many minor FPs. The nuclides for burnup calculation need to be selected so that they cover the reactivity loss as much as possible by a minimum number of nuclides. The nuclides for the major actinides and major

FPs, employed in the benchmark exercise, are considered to be an adequate selection since it has been found that these nuclides cover more than 80 % of the reactivity loss due to burnup. In the case of 30 GWd/t burnup, the major actinides and the major FPs represent about 80 % of reactivity losses due to all actinides and all FPs respectively. The reactivity loss obtained here, however, will decrease in more realistic situations due to the neutron leakage.

In the report, the 2σ standard deviations are also calculated to analyze reactivity loss. From 2σ values, the error components in the reactivity loss have been identified. It has been found that the largest component originates from the major FPs indicating large uncertainties of FP cross sections among participants.

The 2σ value of the fresh fuel case (case 1) is 0.168 which is larger than those of the cases without FPs and in some cases, even with FPs. Why the fresh fuel case has large deviation, can be explained by the following two facts. One is the release of SCALE4.2 whose library is generated from ENDF-B/V. It is estimated that the new results by SCALE-4.2 from ORNL will reduce the 2σ value to 0.119. The other fact is that the MONK6 is known to have about 1 % positive bias for uranium systems. Taking these two facts into account, the 2σ will further reduce to 0.0061.

It is important to grasp the deviation of results among participants by using a simple problem before tackling with more complicated ones, that is one of the purposes of the Phase IA benchmark. The 2σ deviations for the multiplication factors are in the ranges of 0.0099 to 0.0110, and 0.0156 to 0.0170 for the cases of the actinides and the actinides plus FPs respectively. Similarly, in the case of the reactivity loss, the 2σ deviations are also in the ranges of 0.0097 to 0.0114, and 0.0134 to 0.0154 for the cases of the actinides and the actinides plus FPs respectively. The inclusion of SCALE4.2 results will improve the above values.

While processing the data from participants, I found that the modern PCs with spread-sheet software are capable of analyzing substantial amount of data very quickly. We can analyze the data visually by drawing various graphs

interactively. All data from the participants and data used to produce tables and figures in the report are available for your further analysis (stored in WINGZ format on Macintosh). It will be beneficial to make a small database for this type of benchmark for various code users to identify potential problems involved in generating the cross section, calculating the spectrum and so on.

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References

(1) R. J. Brissenden, "Increasing the Physical Realism of the MONK Criticality Code", Proc. Topical Meeting on Physics and Methods in Criticality Safety, ISBN 0-89448-183-5, pp. 51-57, Nashville, TN, USA, 19-23, Sep, 1993.

Table 2.1 Parameters and Case Numbers

Case	1	2	3	4	5	6	7	8	9	10	11	12	13
Burnup (GWd/t)	Fresh	30	40	30	40	30	40	30	40	30	30	30	30
Cooling time (years)	N/A	1	1	1	1	5	5	5	5	1	5	1	5
Actinides	N/A	All	All	All	All	All	All	All	All	Major	Major	All	All
Fission Products	N/A	Major	Major	No	No	Major	Major	No	No	No	No	All	All

Table 2.2 Grouping of Nuclides

Nuclides	Major Actinides	Minor Actinides	All Actinides
U-234	○		○
U-235	○		○
U-236	○		○
U-238	○		○
Pu-238		○	○
Pu-239	○		○
Pu-240	○		○
Pu-241	○		○
Pu-242		○	○
Am-241		○	○
Am-243		○	○
Np-237		○	○

Nuclides	Major FPs	Minor FPs	All FPs
Mo-95	○		○
Tc-99	○		○
Ru-101	○		○
Rh-103	○		○
Ag-109	○		○
Cs-133	○		○
Sm-147	○		○
Sm-149	○		○
Sm-150	○		○
Sm-151	○		○
Sm-152	○		○
Nd-143	○		○
Nd-145	○		○
Eu-153	○		○
Gd-155	○		○
Other FPs		○	○

Table 3.1 Summary of Participants, Data and Codes

No.	COUNTRY	INSTITUTE	PARTICIPANT	NUCLEAR DATA			CODE	REMARKS
				Origin	Library	No of Groups		
1	Belgium	Belgonuclear	Vanderborck		WIMS-1986	69	LWR-WIMS	
2	Canada	Ontario Hydro	Smith		Winfrith-90/06/09	69	WIMS-AECL	
3	Finland	Tech. Res. Center	Anttila	(ENDF-B/4)	E4LTJ70	70	CASMO-3	Excl. Mo95,Tc99,Ru101
4	Finland	Tech. Res. Center	Anttila	(ENDF-B/4)	E4LTJ40	40	CASMO-3	Excl. Mo95,Tc99,Ru101
5	France	CEA	Santamarina	JEF-1	CEA-86	99	APOLLO 1	
6	Germany	IKE/BFS	Lutz	JEF-1	(NJOY)	242	CGM(IKE)	
7	Germany	GRS	Weber	ENDF-B/4,B/5	27BURNUPLIB	27	SCALE-4(1X)	
8	Italy	ENEA	Siciliano	JEF-1+ENDL-85	(NJOY)	(Continuous)	MCNP-3	Excl. Mo-95,Ru-101,Sm-147
9	Japan	Hokkaido Univ.	Narita	JENDL-2/ENDF-B/4	SRAC	45	SRAC	Excl. Am243,Np237
10	Japan	JAERI	Takano	JENDL-3	MGCL	137	JACS(ANISN)	
11	Japan	JAERI	Takano	JENDL-3	MGCL	137	JACS(KENO)	
12	Japan	JAERI	Takano	JENDL-3	(NJOY)	(Continuous)	MCNP-4	
13	Japan	JINS	Mitake	ENDF-B/4,B/5	27BURNUPLIB	27	SCALE-4(1X)	
14	Japan	PNC(O-Arai)	Ohtani	ENDF-B/4,B/5	27BURNUPLIB	27	SCALE-4(25)	
15	Japan	PNC(Tokai)	Nojiri	ENDF-B/4,B/5	27BURNUPLIB	27	SCALE-4(1X)	
16	Japan	Toshiba	Yamamoto	JENDL-2,3,ENDF-B/4,B/5	B-4,5,J-3	95	TGBLA	Excl. Mo-95
17	Spain	CSN	Conde	(ENDF-B/4)	E4LTJB7	70	CASMO-3	
18	Sweden	E.M. Systems	Mennerdahl	ENDF-B/4,B/5	27BURNUPLIB	27	SCALE-4	
19	Sweden	Studsvik	Ekberg	ENDF-B/4,B/5,JEF-2	E4LBJB40	40	CASMO-3	Number densities by CASMO
20	U.K.	AEA (Culcheth)	Rugde		WIMS-1986	69	LWR-WIMS2a	Mo96 instead of Mo95
21	U.K.	AEA (Winfrith)	Gulliford		WIMS-1986	69	LWR-WIMS	
22	U.K.	BNF	Thorne	UKNDL+JEF2	MONK6B	(Continuous)	MONK6B	
23	U.K.	Dept. of Transport	Stewart	UKNDL+JEF2	MONK6B	(Continuous)	MONK6B	Excl. Mo,Tc,Ru,Rh,Cs,Sm,Nd,Gd
24	U.S.A.	ORNL	Brady	ENDF-B/4,B/5	27BURNUPLIB	27	SCALE-4(1X)	
25	U.S.A.	ORNL	DeHart	ENDF-B/5	(From 238 group)	44	SCALE-4.2	

Table 4.1 Multiplication Factors of All Participants

			Group				Case	1	2	3	4	5	6	7	8	9	10	11	12	13
							Burnup (GWd/t)	Fresh	30	40	30	40	30	40	30	40	30	30	30	30
							Cooling time (years)	N/A	1	1	1	1	5	5	5	5	1	5	1	5
							Actinides	N/A	All	All	All	All	All	All	All	All	Major	Major	All	All
							Fission Products	N/A	Major	Major	No	No	Major	Major	No	No	No	No	All	All
No.	COUNTRY	CODE	A	B	C		INSTITUTE													
1	Belgium	LWR-WIMS	○	△	□		Belgonuclear	1.4456	1.1582	1.0822	1.2481	1.1891	1.1296	1.0408	1.2311	1.1657				
2	Canada	WIMS-AECL	○	△	□		Ontario Hydro	1.4419	1.1454	1.0696	1.2502	1.1943	1.1161	1.0274	1.2322	1.1696				
3	Finland	CASMO-3(70G)	○	△			Tech. Res. Center	1.4370	1.1492	1.0744	1.2483	1.1909	1.1212	1.0340	1.2312	1.1675	1.2639	1.2552		
4	Finland	CASMO-3(40G)	○	△			Tech. Res. Center	1.4366	1.1488	1.0740	1.2478	1.1903	1.1209	1.0337	1.2307	1.1669				
5	France	APOLLO-1	○	△	□		CEA	1.4413	1.1335	1.0561	1.2440	1.1859	1.1058	1.0161	1.2275	1.1631	1.2623	1.2556	1.1039	1.0824
6	Germany	CGM	○	△	□		IKE/BFS	1.4410	1.1388	1.0619	1.2473	1.1894	1.1107	1.0214	1.2305	1.1662				
7	Germany	SCALE-4(1X)	○	△	□		GRS	1.4273	1.1321	1.0567	1.2394	1.1833	1.1049	1.0177	1.2225	1.1601				
8	Italy	MCNP-3	○	△			ENEA	1.4359	1.1494	1.0515	1.2404	1.1781	1.1176	1.0195	1.2232	1.1673				
9	Japan	SRAC	○	△			Hokkaido Univ.	1.4375	1.1376	1.0538	1.2498	1.1887	1.1074	1.0363	1.2306	1.1664				
10	Japan	JACS(ANISN)	○	△	□		JAERI(A)	1.4447	1.1429	1.0663	1.2529	1.1955	1.1154	1.0265	1.2366	1.1730	1.2708	1.2640	1.1282	1.0618
11	Japan	JACS(KENO-IV)	○	△	□		JAERI(K)	1.4464	1.1453	1.0684	1.2550	1.2013	1.1202	1.0289	1.2380	1.1768				
12	Japan	MCNP-4	○	△	□		JAERI(M)	1.4427	1.1382	1.0591	1.2426	1.1930	1.1042	1.0216	1.2231	1.1668				
13	Japan	SCALE-4(1X)	○	△	□		JINS	1.4273	1.1321	1.0568	1.2395	1.1833	1.1049	1.0177	1.2225	1.1601	1.2571	1.2502		
14	Japan	SCALE-4(25)	○	△	□		PNC(O-Arai)	1.4272	1.1327	1.0548	1.2429	1.1806	1.1085	1.0153	1.2233	1.1657				
15	Japan	SCALE-4(1X)	○	△	□		PNC(Tokai)	1.4270	1.1320	1.0570	1.2390	1.1830	1.1050	1.0180	1.2230	1.1600	1.2570	1.2500	1.1040	1.0670
16	Japan	TGBLA	○	△			Toshiba	1.4373	1.1382	1.0617	1.2436	1.1855	1.1108	1.0223	1.2271	1.1628				
17	Spain	CASMO-3	○	△	□		CSN	1.4366	1.1489	1.0741	1.2480	1.1906	1.1209	1.0337	1.2309	1.1671	1.2668	1.2597	1.1132	1.0914
18	Sweden	SCALE-4(1X)	○	△	□		E.M.Systems	1.4276	1.1324	1.0570	1.2397	1.1836	1.1051	1.0180	1.2228	1.1604				
19	Sweden	CASMO-3	○				Studsvik	1.4366			1.2365	1.1719			1.2202	1.1491				
20	U.K.	LWR-WIMS2a	○	△			AEA(Culcheth/W)	1.4458	1.1596	1.0839	1.2482	1.1893	1.1310	1.0425	1.2312	1.1659	1.2667	1.2597	1.0963	1.0727
21	U.K.	LWR-WIMS	○	△	□		AEA(Winfrith/W)	1.4449	1.1575	1.0815	1.2473	1.1884	1.1289	1.0402	1.2303	1.1650	1.2658	1.2588	1.0953	1.0720
22	U.K.	MONK6B	○	△	□		BNF	1.4564	1.1445	1.0652	1.2553	1.1945	1.1149	1.0257	1.2382	1.1733	1.2719	1.2653		
23	U.K.	MONK6B	○				Dept. of Transport	1.4581	1.2474	1.1887	1.2531	1.1932	1.2324	1.1607	1.2397	1.1725				
24	U.S.A.	SCALE-4(1X)	○	△	□		ORNL V4	1.4276	1.1323	1.0570	1.2397	1.1836	1.1051	1.0180	1.2228	1.1604	1.2576	1.2505	1.1026	1.0809
25	U.S.A.	SCALE-4.2	○	△	□		ORNL V4.2	1.4369	1.1366	1.0604	1.2436	1.1859	1.1093	1.0209	1.2273	1.1634	1.2619	1.2552	1.1088	1.0753
								Case 1	Case 2	Case 3	Case 4	Case 5	Case 6	Case 7	Case 8	Case 9	Case 10	Case 11	Case 12	Case 13
		All Participants	○				Average	1.4387	1.1464	1.0697	1.2457	1.1877	1.1188	1.0315	1.2287	1.1654	1.2638	1.2567	1.1065	1.0754
			○				2*sigma	0.0168	0.0455	0.0530	0.0105	0.0122	0.0502	0.0564	0.0110	0.0110	0.0099	0.0101	0.0197	0.0175
		23 Participants		△			Average	1.4379	1.1420	1.0645	1.2458	1.1882	1.1138	1.0259	1.2285	1.1658	1.2638	1.2567	1.1065	1.0754
				△			2*sigma	0.0154	0.0175	0.0191	0.0098	0.0106	0.0169	0.0170	0.0099	0.0087	0.0099	0.0101	0.0197	0.0175
		17 Participants			□		Average	1.4378	1.1402	1.0638	1.2456	1.1885	1.1123	1.0240	1.2284	1.1657	1.2635	1.2566	1.1080	1.0758
					□		2*sigma	0.0175	0.0169	0.0170	0.0107	0.0110	0.0164	0.0156	0.0109	0.0099	0.0108	0.0109	0.0194	0.0185

Table 4.2 Relative Difference from Average Value of Group C [Calculation/Average-1.0]*100

						Case	1	2	3	4	5	6	7	8	9	10	11	12	13
						Burnup (GWd/t)	Fresh	30	40	30	40	30	40	30	40	30	30	30	30
						Cooling time (years)	N/A	1	1	1	1	5	5	5	5	1	5	1	5
						Actinides	N/A	All	All	All	All	All	All	All	All	Major	Major	All	All
						Fission Products	N/A	Major	Major	No	No	Major	Major	No	No	No	No	All	All
			Group																
No.	COUNTRY	CODE	A	B	C	INSTITUTE													
1	Belgium	LWR-WIMS	○	△	□	Belgonuclear	0.5436	1.5794	1.7298	0.2006	0.0464	1.5533	1.6417	0.2213	0.0010				
2	Canada	WIMS-AECL	○	△	□	Ontario Hydro	0.2834	0.4585	0.5472	0.3700	0.4839	0.3360	0.3331	0.3084	0.3321				
3	Finland	CASMO-3(70G)	○	△		Tech. Res. Center	-0.0546	0.7901	1.0003	0.2207	0.1979	0.7981	0.9776	0.2294	0.1554	0.0339	-0.1106		
4	Finland	CASMO-3(40G)	○	△		Tech. Res. Center	-0.0824	0.7550	0.9627	0.1805	0.1474	0.7711	0.9483	0.1887	0.1039				
5	France	APOLLO-1	○	△	□	CEA	0.2445	-0.5869	-0.7200	-0.1245	-0.2228	-0.5864	-0.7705	-0.0718	-0.2221	-0.0927	-0.0788	-0.3700	0.6108
6	Germany	CGM	○	△	□	IKE/BFS	0.2236	-0.1221	-0.1748	0.1404	0.0717	-0.1459	-0.2529	0.1724	0.0439				
7	Germany	SCALE-4(1X)	○	△	□	GRS	-0.7292	-0.7097	-0.6636	-0.4939	-0.4416	-0.6673	-0.6142	-0.4788	-0.4794				
8	Italy	MCNP-3	○	△		ENEA	-0.1311	0.8076	-1.1496	-0.4160	-0.8799	0.4772	-0.4384	-0.4194	0.1399				
9	Japan	SRAC	○	△		Hokkaido Univ.	-0.0226	-0.2273	-0.9325	0.3443	0.0111	-0.4461	1.2051	0.1814	0.0610				
10	Japan	JACS(ANISN)	○	△	□	JAERI(A)	0.4810	0.2375	0.2341	0.5900	0.5849	0.2767	0.2452	0.6690	0.6272	0.5832	0.5905	1.8231	-1.3040
11	Japan	JACS(KENO-IV)	○	△	□	JAERI(K)	0.5992	0.4480	0.4362	0.7586	1.0729	0.7082	0.4796	0.7789	0.9532				
12	Japan	MCNP-4	○	△	□	JAERI(M)	0.3419	-0.1782	-0.4380	-0.2369	0.3746	0.7302	-0.2333	-0.4300	0.0953				
13	Japan	SCALE-4(1X)	○	△	□	JINS	-0.7292	-0.7097	-0.6542	-0.4858	-0.4416	-0.6673	-0.6142	-0.4788	-0.4794	-0.5043	-0.5085		
14	Japan	SCALE-4(25)	○	△	□	PNC(O-Arai)	-0.7383	-0.6579	-0.8469	-0.2120	-0.6662	-0.3409	-0.8496	-0.4137	0.0044				
15	Japan	SCALE-4(1X)	○	△	□	PNC(Tokai)	-0.7501	-0.7185	-0.6354	-0.5260	-0.4668	-0.6583	-0.5849	-0.4381	-0.4880	-0.5122	-0.5244	-0.3610	-0.8206
16	Japan	TGBLA	○	△		Toshiba	-0.0372	-0.1756	-0.1945	-0.1550	-0.2590	-0.1351	-0.1669	-0.1011	-0.2495				
17	Spain	CASMO-3	○	△	□	CSN	-0.0817	0.7620	0.9702	0.1926	0.1718	0.7702	0.9454	0.2026	0.1219	0.2635	0.2475	0.4693	1.4474
18	Sweden	SCALE-4(1X)	○	△	□	E.M.Systems	-0.7084	-0.6869	-0.6364	-0.4698	-0.4163	-0.6493	-0.5888	-0.4544	-0.4537				
19	Sweden	CASMO-3	○			Studsvik	-0.0824			-0.7267	-1.4007			-0.6661	-1.4231				
20	U.K.	LWR-WIMS2a	○	△		AEA(Culcheth/W)	0.5575	1.7022	1.8933	0.2127	0.0633	1.6792	1.8077	0.2294	0.0181	0.2556	0.2475	-1.0560	-0.2908
21	U.K.	LWR-WIMS	○	△	□	AEA(Winfrith/W)	0.4949	1.5136	1.6677	0.1404	-0.0125	1.4904	1.5831	0.1562	-0.0591	0.1843	0.1759	-1.1462	-0.3559
22	U.K.	MONK6B	○	△	□	BNF	1.2947	0.3779	0.1354	0.7827	0.5008	0.2317	0.1671	0.7993	0.6529	0.6671	0.6931		
23	U.K.	MONK6B	○			Dept. of Transport	1.4130	9.4026	11.7452	0.6061	0.3914	10.7952	13.3508	0.9214	0.5843				
24	U.S.A.	SCALE-4(1X)	○	△	□	ORNL V4	-0.7084	-0.6921	-0.6354	-0.4698	-0.4163	-0.6493	-0.5849	-0.4544	-0.4537	-0.4647	-0.4846	-0.4874	0.4714
25	U.S.A.	SCALE-4.2	○	△	□	ORNL V4.2	-0.0615	-0.3150	-0.3158	-0.1567	-0.2228	-0.2717	-0.3017	-0.0881	-0.1963	-0.1243	-0.1106	0.0722	-0.0491

Table 4.3 Reactivity Decrease (Δk)

			Group				Case	1	2	3	4	5	6	7	8	9	10	11	12	13
							Burnup (GWd/t)	Fresh	30	40	30	40	30	40	30	40	30	30	30	30
							Cooling time (years)	N/A	1	1	1	1	5	5	5	5	1	5	1	5
							Actinides	N/A	All	All	All	All	All	All	All	All	Major	Major	All	All
							Fission Products	N/A	Major	Major	No	No	Major	Major	No	No	No	No	All	All
No.	COUNTRY	CODE	A	B	C		INSTITUTE													
1	Belgium	LWR-WIMS	○	△	□		Belgonuclear	0.0000	-0.2874	-0.3634	-0.1975	-0.2565	-0.3160	-0.4048	-0.2145	-0.2799				
2	Canada	WIMS-AECL	○	△	□		Ontario Hydro	0.0000	-0.2964	-0.3723	-0.1917	-0.2476	-0.3258	-0.4145	-0.2097	-0.2723				
3	Finland	CASMO-3(70G)	○	△			Tech. Res. Center	0.0000	-0.2878	-0.3626	-0.1887	-0.2461	-0.3158	-0.4030	-0.2058	-0.2695	-0.1731	-0.1818		
4	Finland	CASMO-3(40G)	○	△			Tech. Res. Center	0.0000	-0.2878	-0.3626	-0.1888	-0.2463	-0.3157	-0.4029	-0.2059	-0.2697				
5	France	APOLLO-1	○	△	□		CEA	0.0000	-0.3078	-0.3852	-0.1973	-0.2554	-0.3355	-0.4252	-0.2138	-0.2782	-0.1790	-0.1857	-0.3374	-0.3589
6	Germany	CGM	○	△	□		IKE/BFS	0.0000	-0.3022	-0.3791	-0.1937	-0.2516	-0.3303	-0.4196	-0.2105	-0.2748				
7	Germany	SCALE-4(1X)	○	△	□		GRS	0.0000	-0.2952	-0.3706	-0.1879	-0.2440	-0.3224	-0.4096	-0.2048	-0.2672				
8	Italy	MCNP-3	○	△			ENEA	0.0000	-0.2865	-0.3844	-0.1955	-0.2578	-0.3183	-0.4164	-0.2127	-0.2686				
9	Japan	SRAC	○	△			Hokkaido Univ.	0.0000	-0.2999	-0.3836	-0.1876	-0.2488	-0.3301	-0.4011	-0.2069	-0.2711				
10	Japan	JACS(ANISN)	○	△	□		JAERI(A)	0.0000	-0.3018	-0.3785	-0.1918	-0.2492	-0.3293	-0.4182	-0.2081	-0.2717	-0.1739	-0.1807	-0.3165	-0.3829
11	Japan	JACS(KENO-IV)	○	△	□		JAERI(K)	0.0000	-0.3011	-0.3780	-0.1914	-0.2451	-0.3262	-0.4175	-0.2084	-0.2696				
12	Japan	MCNP-4	○	△	□		JAERI(M)	0.0000	-0.3045	-0.3836	-0.2001	-0.2497	-0.3385	-0.4211	-0.2196	-0.2759				
13	Japan	SCALE-4(1X)	○	△	□		JINS	0.0000	-0.2952	-0.3705	-0.1878	-0.2440	-0.3224	-0.4096	-0.2048	-0.2672	-0.1702	-0.1771		
14	Japan	SCALE-4(25)	○	△	□		PNC(O-Arai)	0.0000	-0.2945	-0.3724	-0.1843	-0.2465	-0.3186	-0.4119	-0.2039	-0.2614				
15	Japan	SCALE-4(1X)	○	△	□		PNC(Tokai)	0.0000	-0.2950	-0.3700	-0.1880	-0.2440	-0.3220	-0.4090	-0.2040	-0.2670	-0.1700	-0.1770	-0.3230	-0.3600
16	Japan	TGBLA	○	△			Toshiba	0.0000	-0.2991	-0.3756	-0.1936	-0.2518	-0.3264	-0.4150	-0.2101	-0.2745				
17	Spain	CASMO-3	○	△	□		CSN	0.0000	-0.2877	-0.3625	-0.1887	-0.2460	-0.3157	-0.4029	-0.2057	-0.2695	-0.1698	-0.1769	-0.3234	-0.3452
18	Sweden	SCALE-4(1X)	○	△	□		E.M.Systems	0.0000	-0.2952	-0.3706	-0.1879	-0.2440	-0.3225	-0.4096	-0.2048	-0.2672				
19	Sweden	CASMO-3	○				Studsvik	0.0000			-0.2001	-0.2647			-0.2164	-0.2875				
20	U.K.	LWR-WIMS2a	○	△			AEA(Culcheth/W)	0.0000	-0.2862	-0.3619	-0.1976	-0.2565	-0.3148	-0.4033	-0.2146	-0.2799	-0.1791	-0.1861	-0.3495	-0.3731
21	U.K.	LWR-WIMS	○	△	□		AEA(Winfrith/W)	0.0000	-0.2875	-0.3634	-0.1976	-0.2565	-0.3160	-0.4047	-0.2146	-0.2799	-0.1791	-0.1861	-0.3496	-0.3729
22	U.K.	MONK6B	○	△	□		BNF	0.0000	-0.3119	-0.3912	-0.2011	-0.2619	-0.3415	-0.4307	-0.2182	-0.2831	-0.1845	-0.1911		
23	U.K.	MONK6B	○				Dept. of Transport	0.0000	-0.2107	-0.2694	-0.2050	-0.2649	-0.2257	-0.2974	-0.2184	-0.2856				
24	U.S.A.	SCALE-4(1X)	○	△	□		ORNL V4	0.0000	-0.2953	-0.3706	-0.1879	-0.2440	-0.3225	-0.4096	-0.2048	-0.2672	-0.1700	-0.1771	-0.3250	-0.3467
25	U.S.A.	SCALE-4.2	○	△	□		ORNL V4.2	0.0000	-0.3003	-0.3765	-0.1933	-0.2510	-0.3276	-0.4160	-0.2096	-0.2735	-0.1750	-0.1817	-0.3281	-0.3616
								Case 1	Case 2	Case 3	Case 4	Case 5	Case 6	Case 7	Case 8	Case 9	Case 10	Case 11	Case 12	Case 13
		All Participants	○				Average	0.0000	-0.2924	-0.3691	-0.1930	-0.2510	-0.3200	-0.4072	-0.2100	-0.2733	-0.1749	-0.1819	-0.3316	-0.3627
			○				2*sigma	0.0000	0.0368	0.0447	0.0105	0.0130	0.0419	0.0482	0.0098	0.0128	0.0094	0.0092	0.0235	0.0245
		23 Participants		△			Average	0.0000	-0.2959	-0.3734	-0.1922	-0.2498	-0.3241	-0.4120	-0.2094	-0.2721	-0.1749	-0.1819	-0.3316	-0.3627
				△			2*sigma	0.0000	0.0141	0.0167	0.0092	0.0105	0.0149	0.0154	0.0092	0.0106	0.0094	0.0092	0.0235	0.0245
		17 Participants			□		Average	0.0000	-0.2976	-0.3740	-0.1922	-0.2492	-0.3255	-0.4138	-0.2094	-0.2721	-0.1746	-0.1815	-0.3290	-0.3612
					□		2*sigma	0.0000	0.0134	0.0154	0.0097	0.0107	0.0149	0.0147	0.0099	0.0114	0.0100	0.0097	0.0205	0.0248

Table 4.4 Reactivity Loss by Actinides and Major FPs in delta k (17 Participants)

		30GWd/t	40GWd/t
1 Year	Actinides	0.1922	0.2492
	Major FP	0.1054	0.1248
	Total	0.2976	0.3740
5 Years	Actinides	0.2094	0.2721
	Major FP	0.1161	0.1417
	Total	0.3255	0.4138

Table 4.5 Reactivity Loss by Each Set of Nuclides

	30 GWd/t, 1 Year				30 GWd/t, 5 Years			
	Δ κ	Ratio(%)	2σ	2σ/Δκ (%)	Δ κ	Ratio(%)	2σ	2σ/Δκ (%)
Major Actinides	0.1746	53.2	0.0094	5.4	0.1815	50.2	0.0092	5.1
Minor Actinides	0.0176	5.3	0.0018	10.2	0.0279	7.7	0.0025	9.0
Major FPs	0.1054	32.0	0.0126	12.0	0.1161	32.2	0.0121	10.4
Minor FPs	0.0314	9.5	0.0269	85.7	0.0357	9.9	0.0249	69.7
Total	0.3290	100.0	0.0205	6.2	0.3612	100.0	0.0248	6.9

The values of delta k are the differences of average values in Table 4.3

The values of 2*sigma are calculated from reactivity decrease obtained from Table 4.1

The large deviation of Minor FPs comes from the difference of FP nuclides included as Minor FPs.

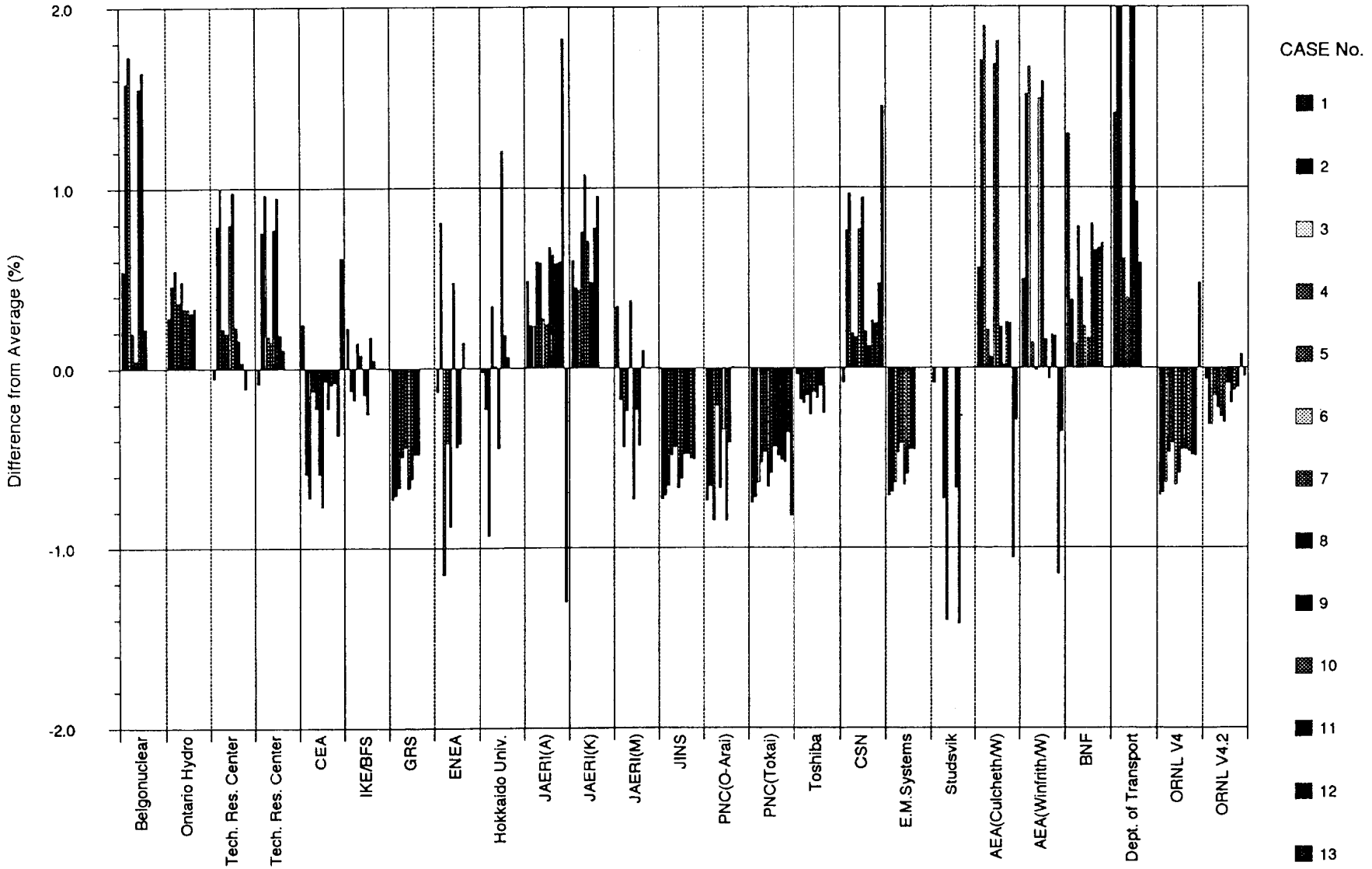
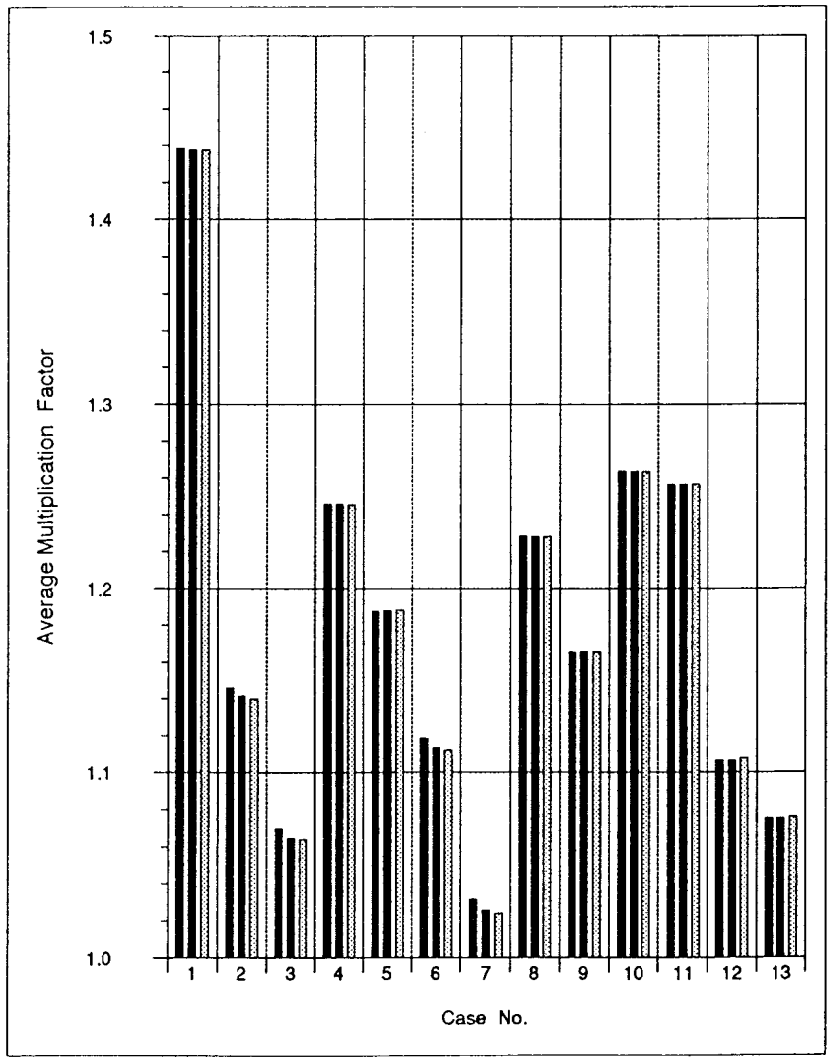
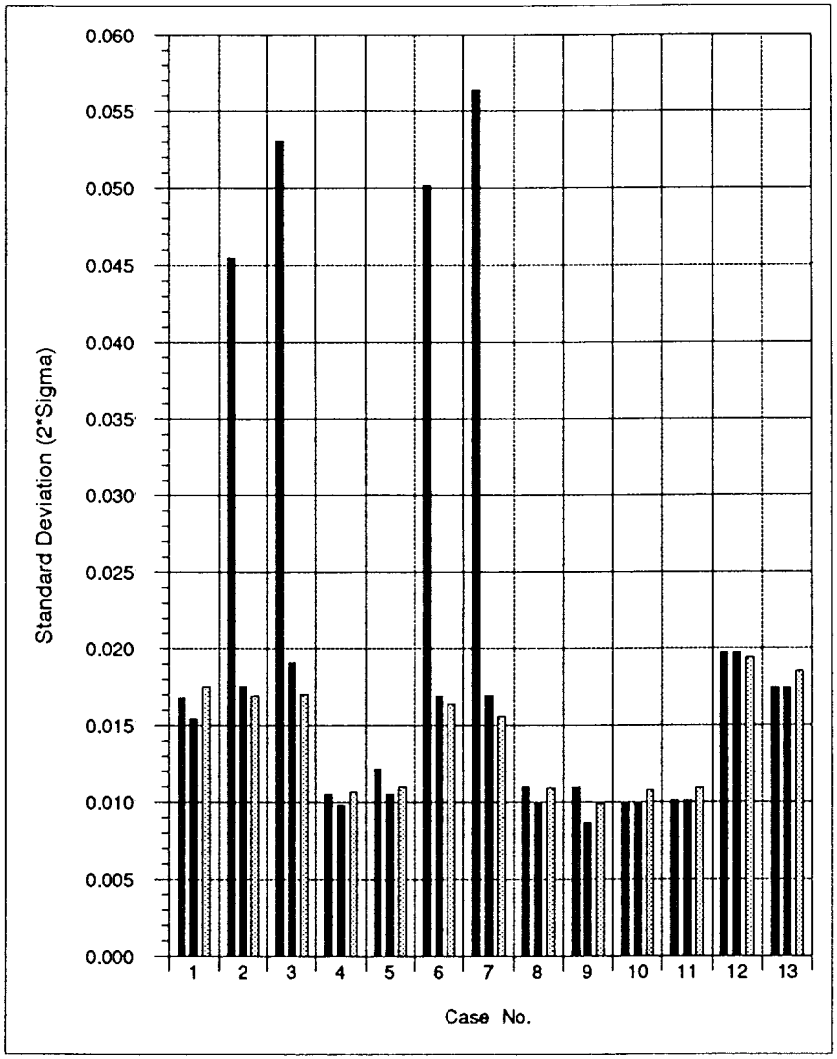


Fig. 4.1 Relative Difference from Average Value of Group C



(A) Average Multiplication Factor



(B) Standard Deviation

Fig. 4.2 Average and Standard Deviation

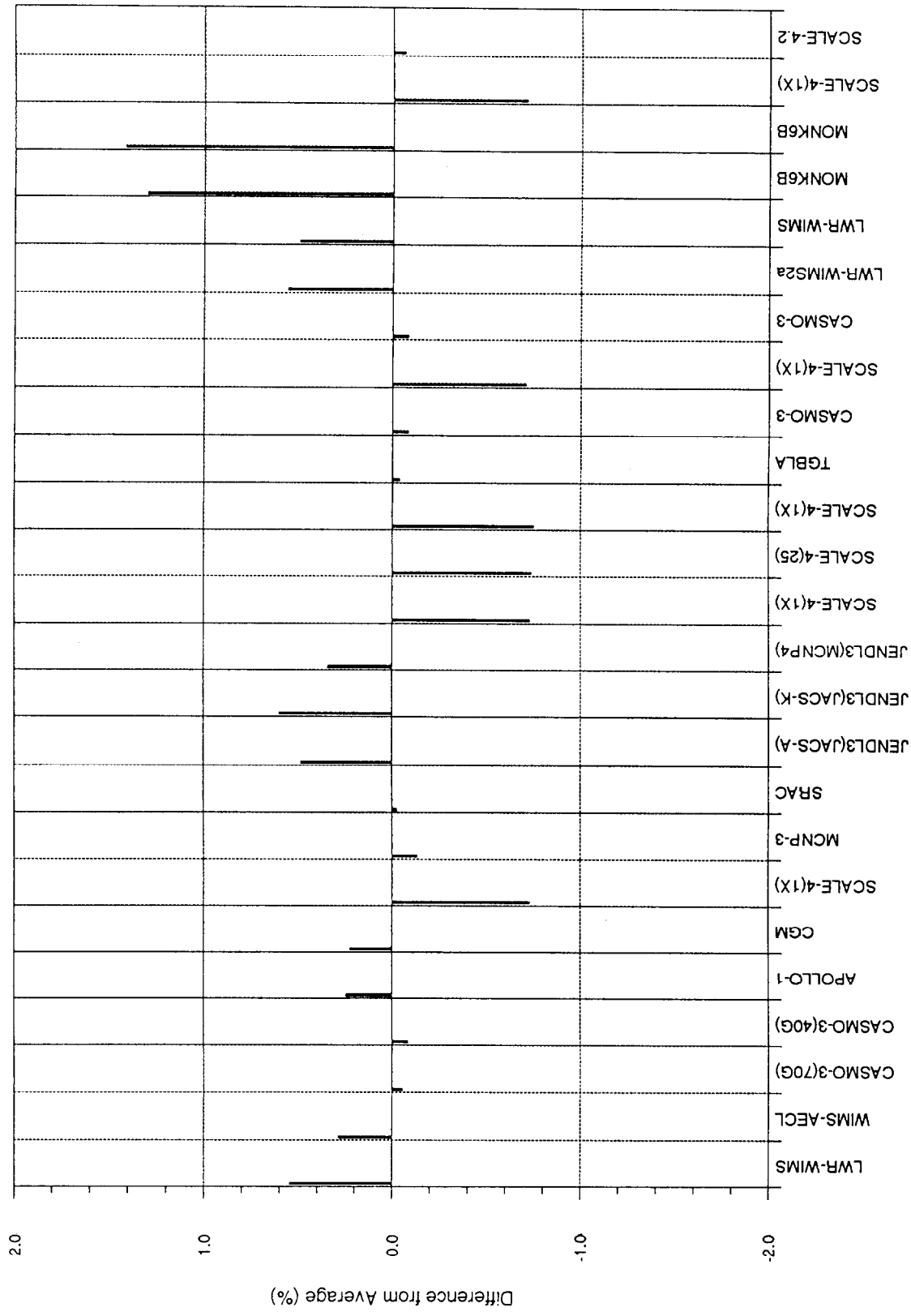


Fig. 4.3 Relative Difference of Case 1

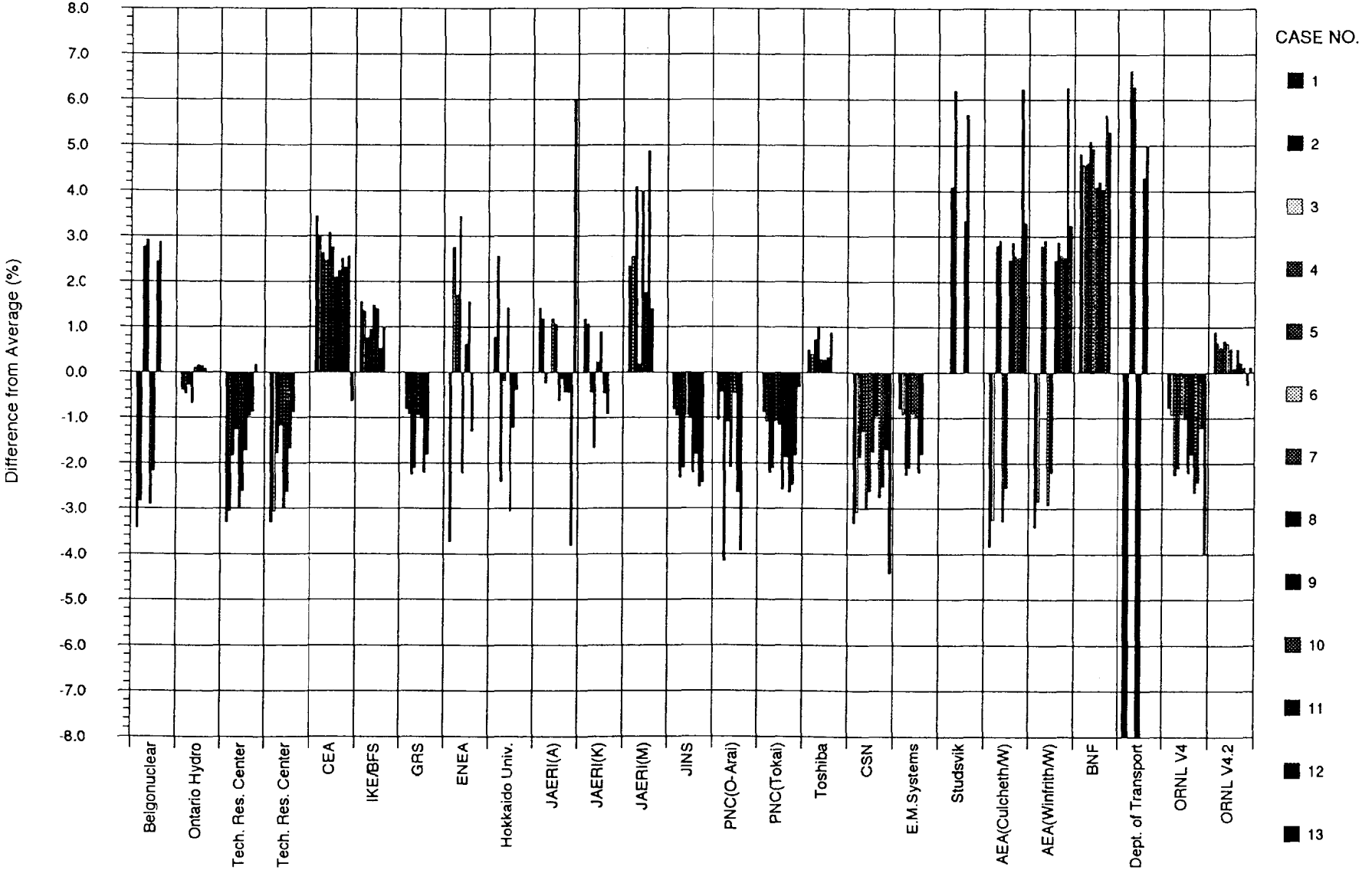
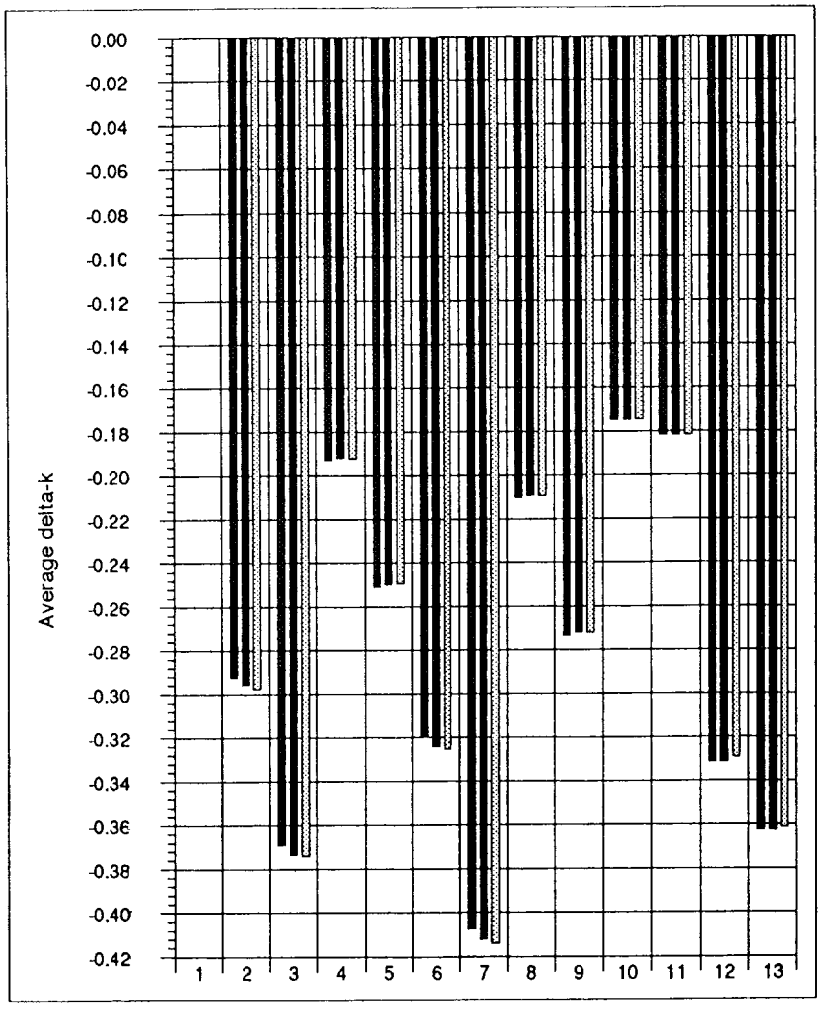
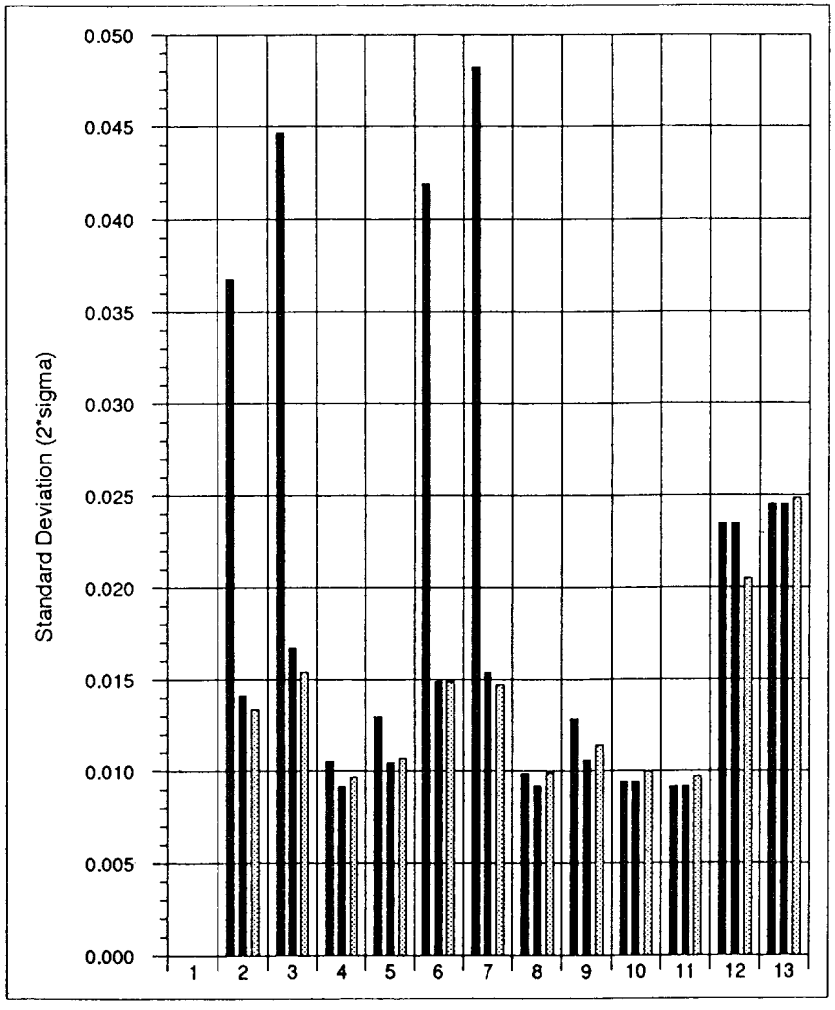


Fig. 4.4 Relative Difference from Average delta-k Value



(A) Average delta-k



(B) Standard Deviation (2*sigma)

Fig. 4.5 Average and Standard Deviation of delta-k Values

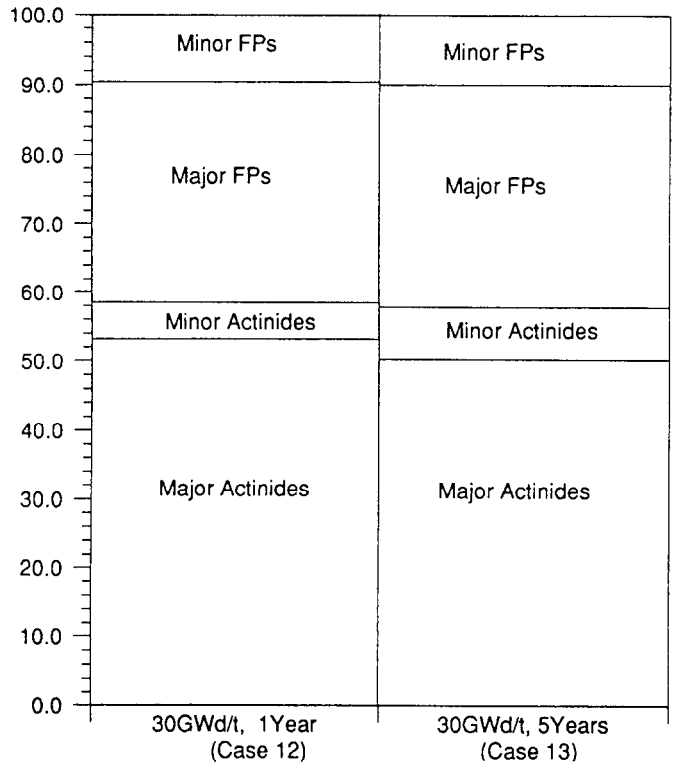


Fig. 4.6 Contributions to Reactivity Loss

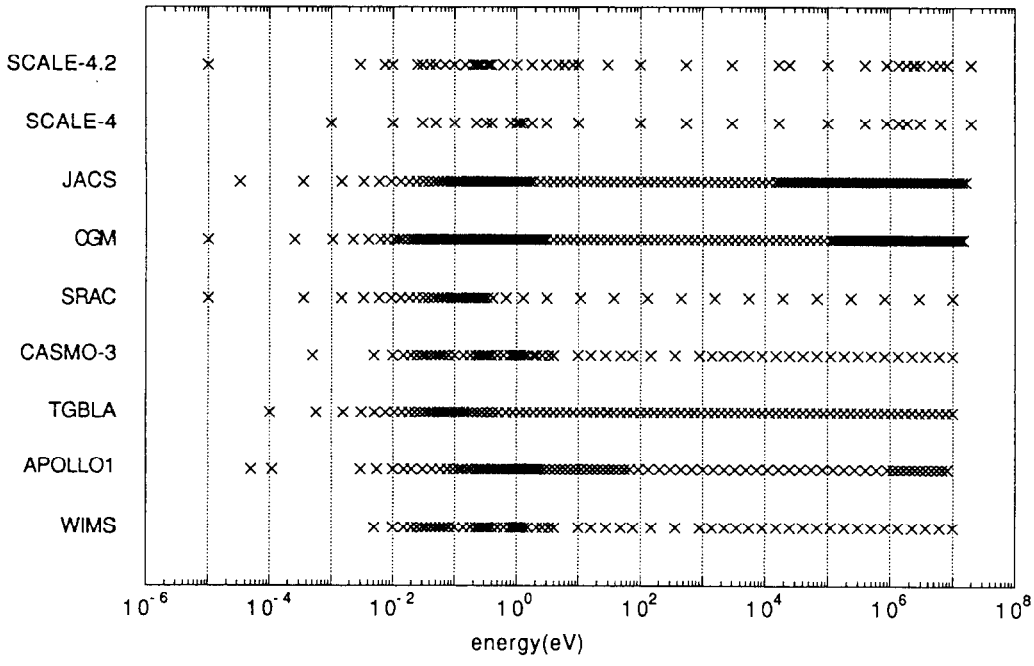


Fig. 4.7 Upper Energy Boundaries

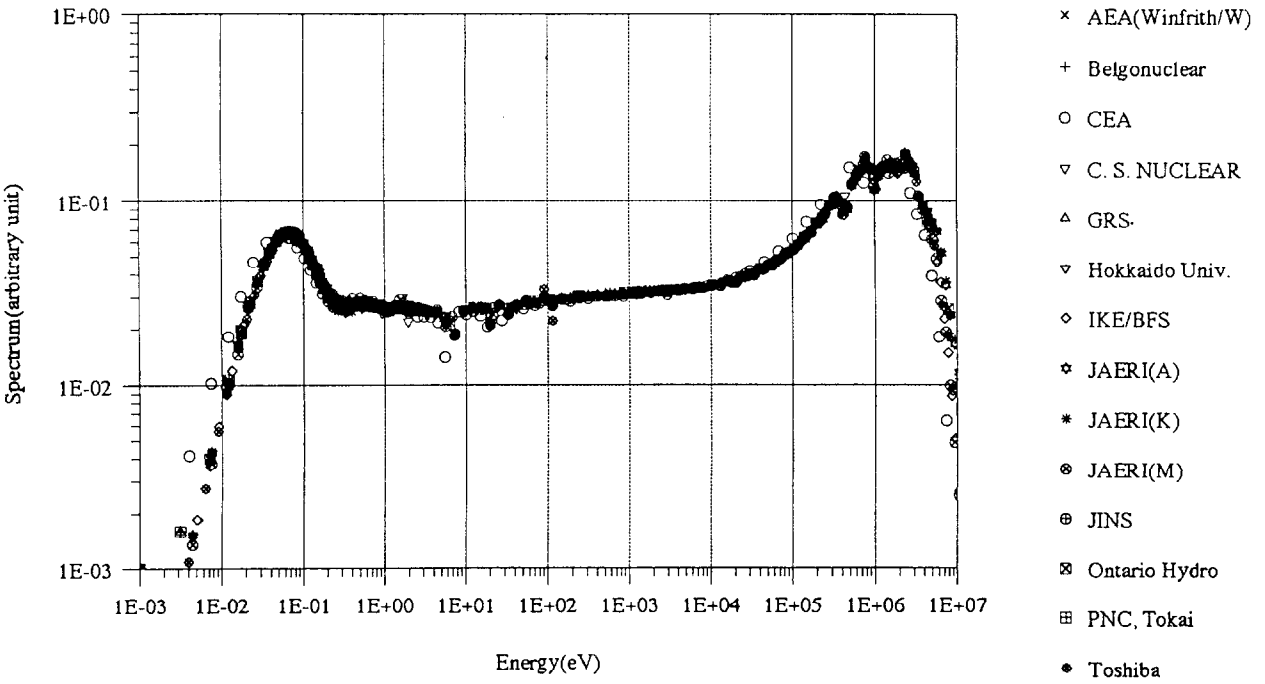


Fig. 4.8 CASE 1 Spectrum in Fuel (Fresh Fuel)

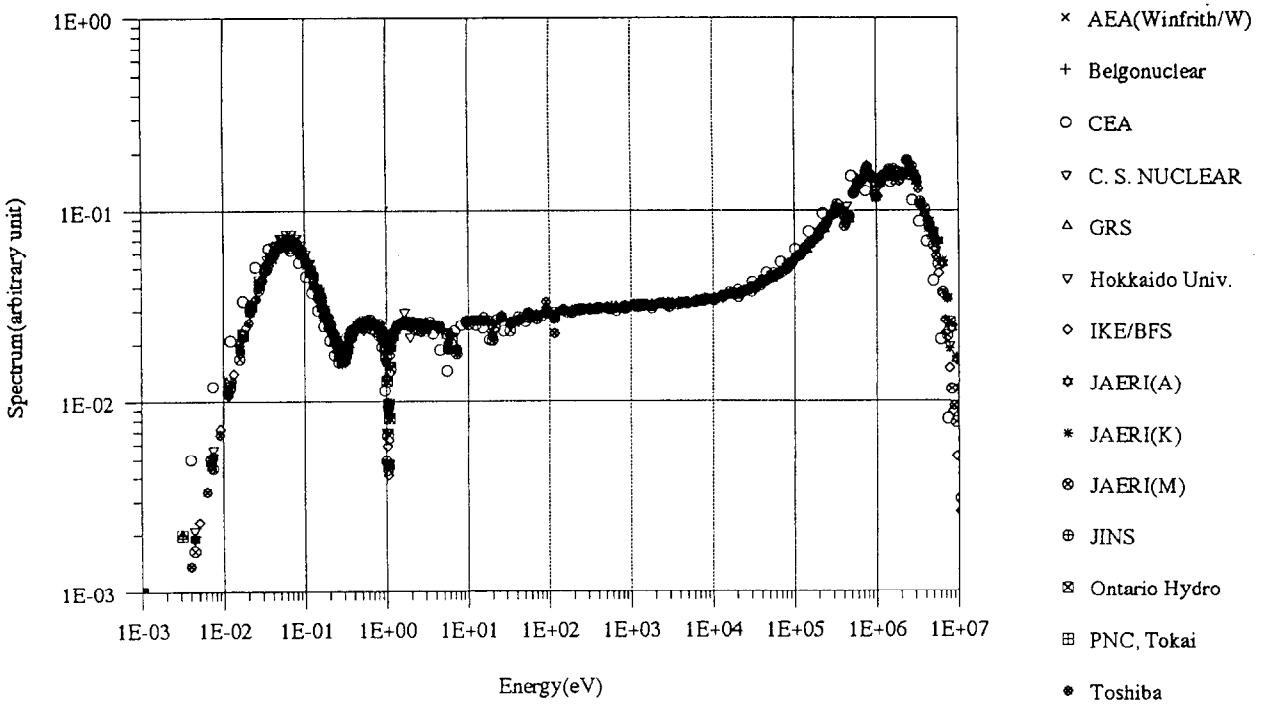


Fig. 4.9 CASE 7 Spectrum in Fuel (40GWd/t, 5yr, selected F.P.)

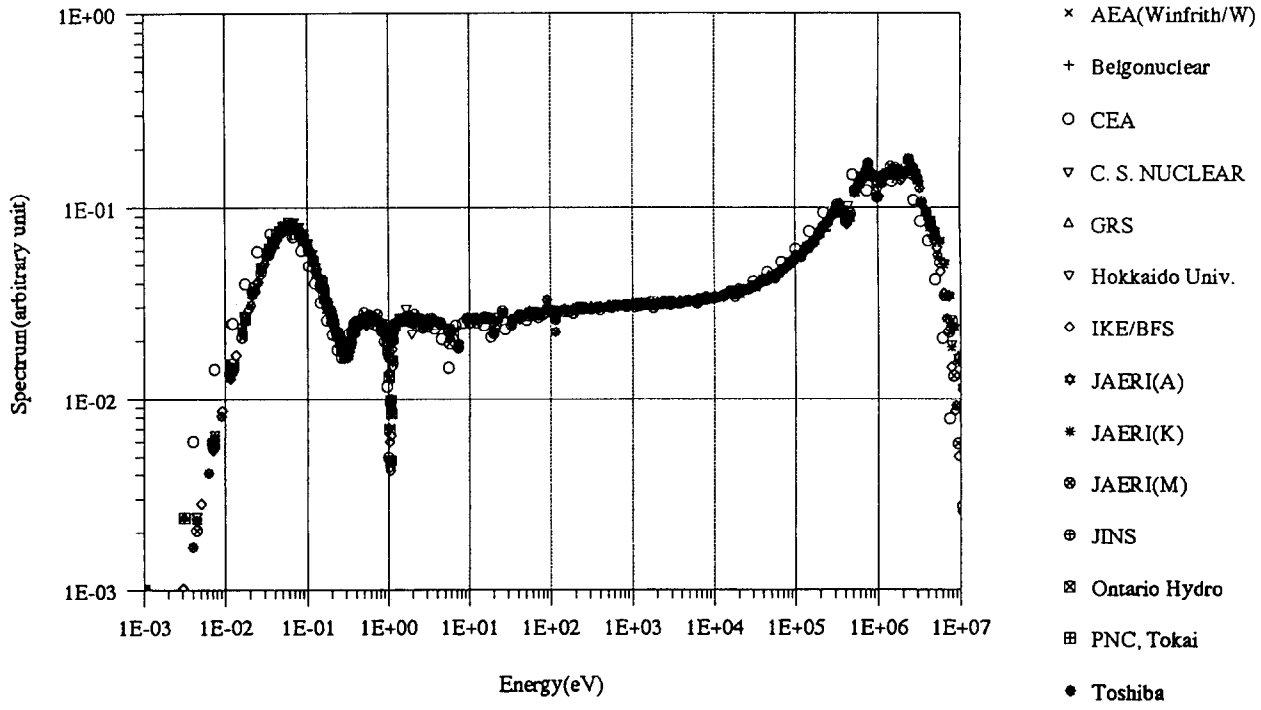


Fig. 4.10 CASE 9 Spectrum in Fuel (40GWd/t, 5yr, no F.P.)

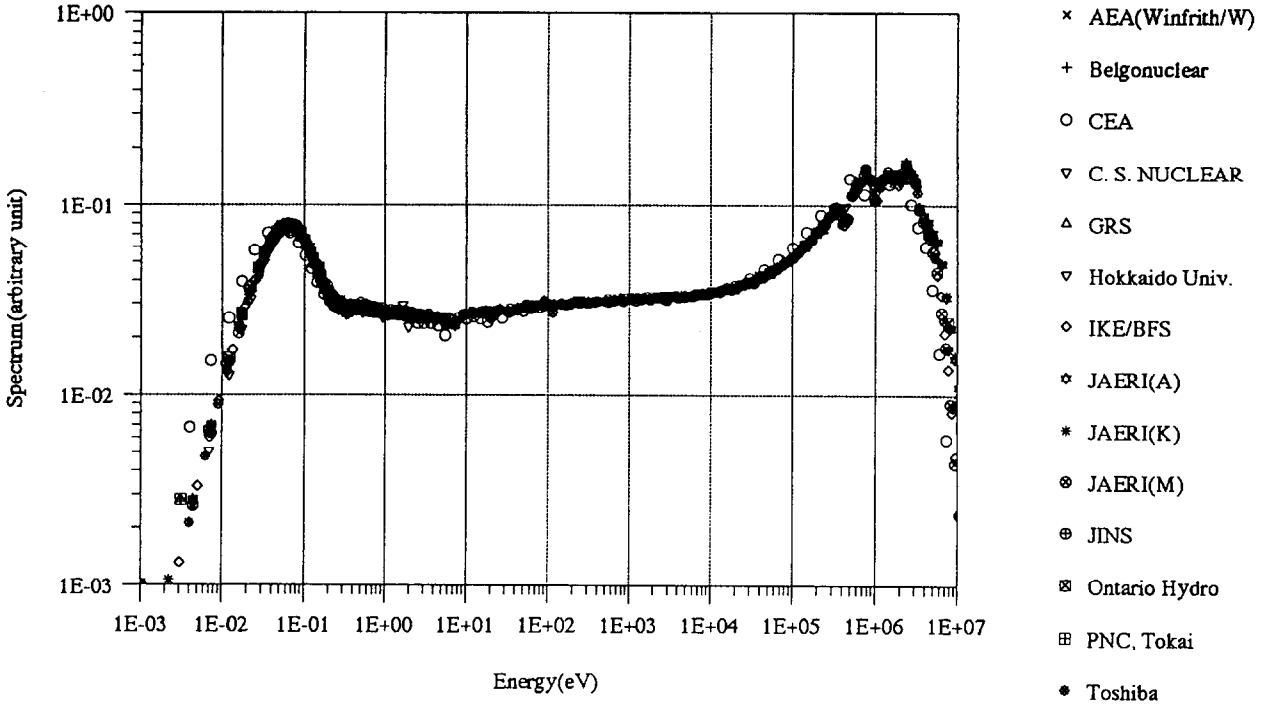


Fig. 4.11 CASE 1 Spectrum in Reflector (Fresh Fuel)

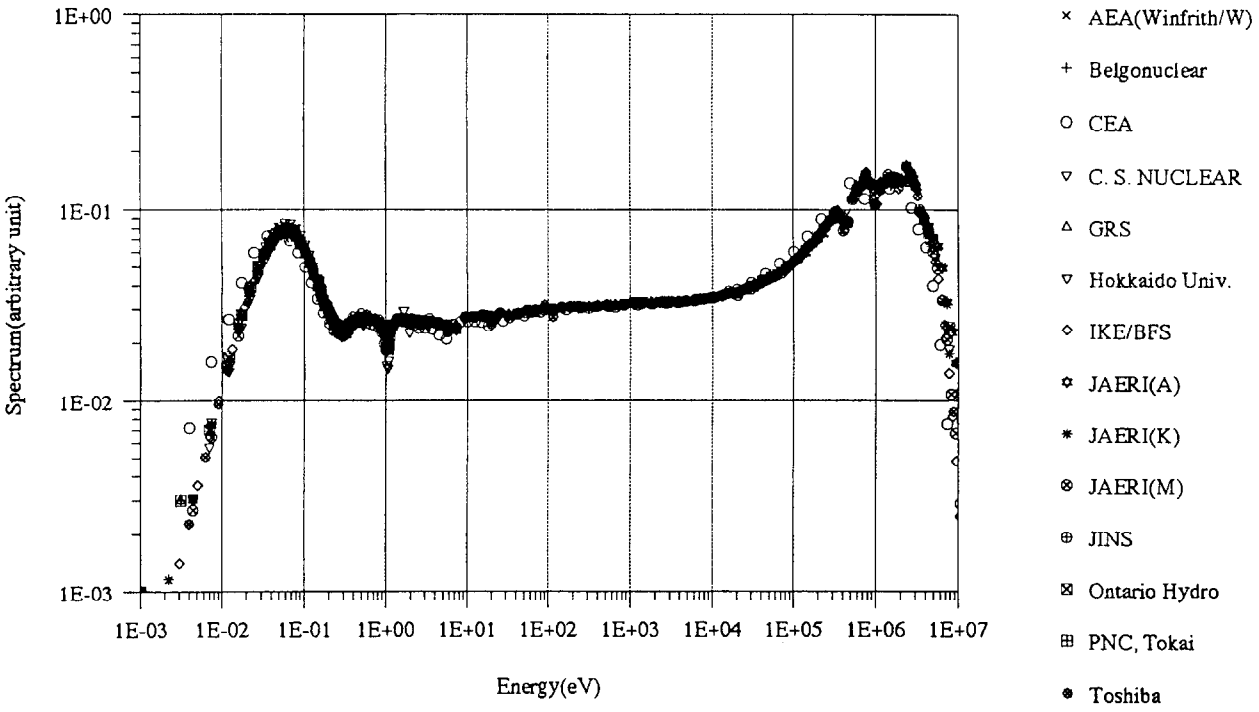


Fig. 4.12 CASE 7 Spectrum in Reflector (40GWd/t, 5yr, selected F.P.)

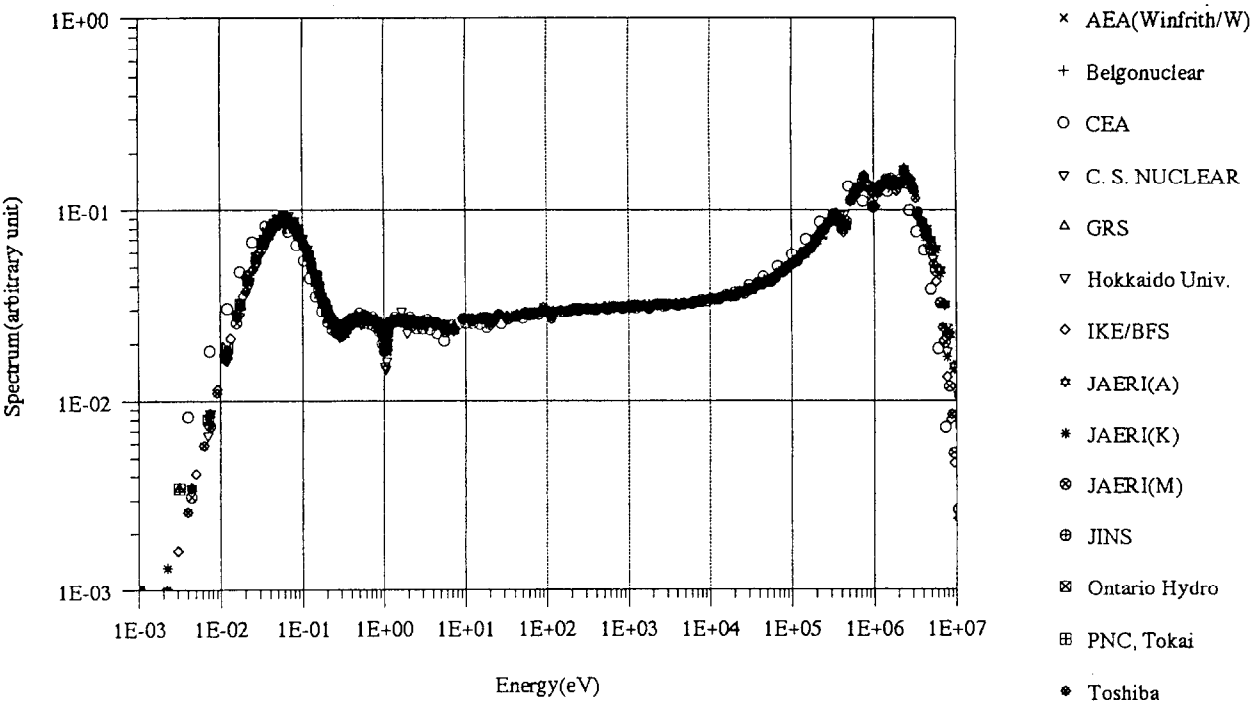


Fig. 4.13 CASE 9 Spectrum in Reflector (40GWd/t, 5yr, no F.P.)

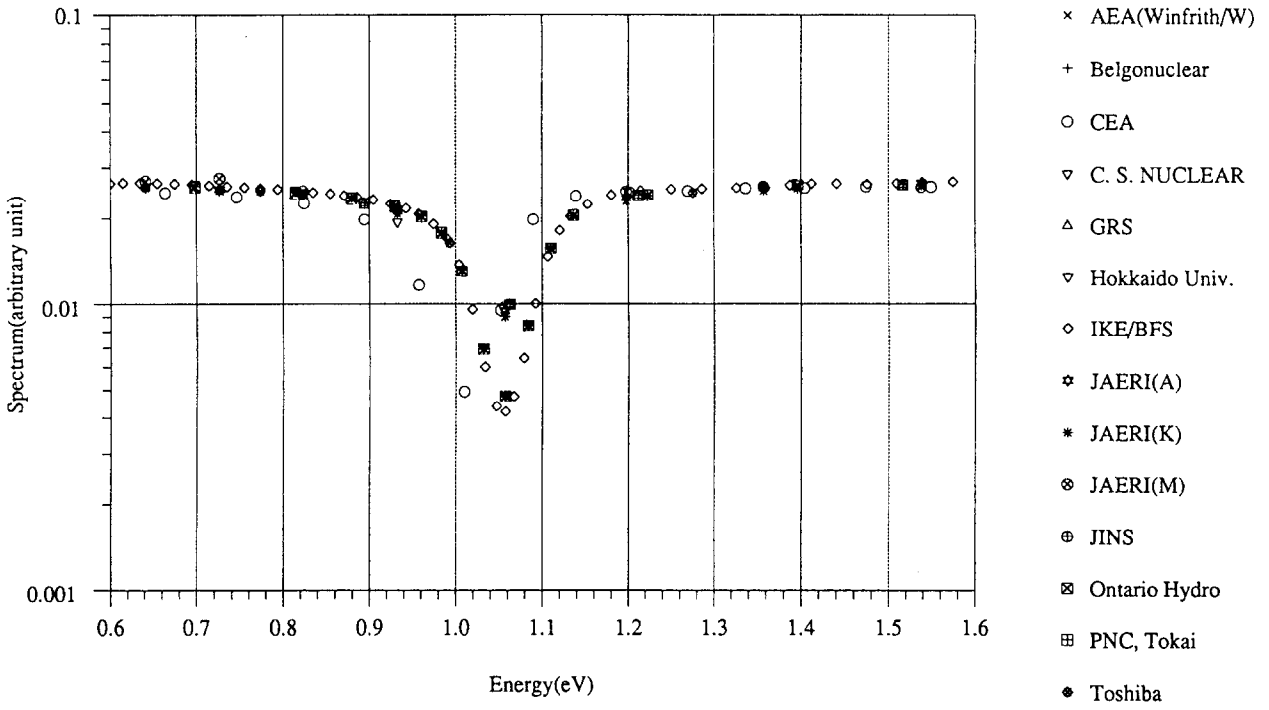


Fig. 4.14 CASE 9 Spectrum in Fuel (From 0.6 to 1.4 eV.)

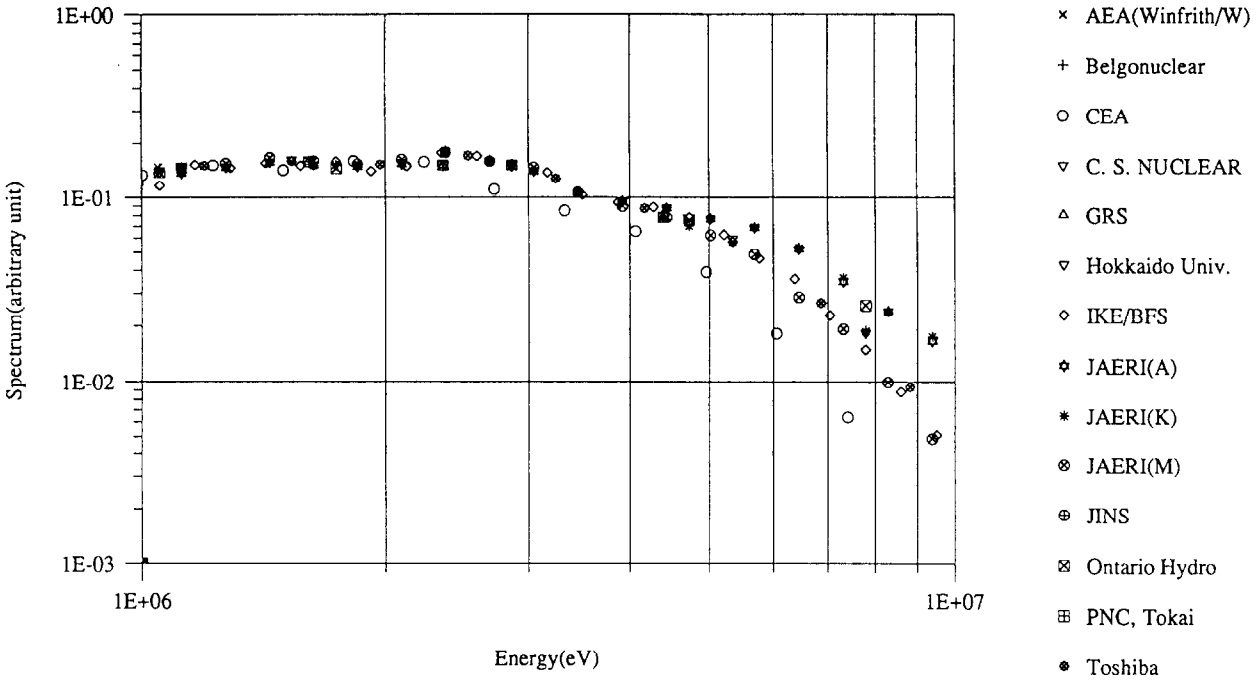


Fig. 4.15 CASE 1 Spectrum in Fuel (From 1 MeV to 10 MeV)

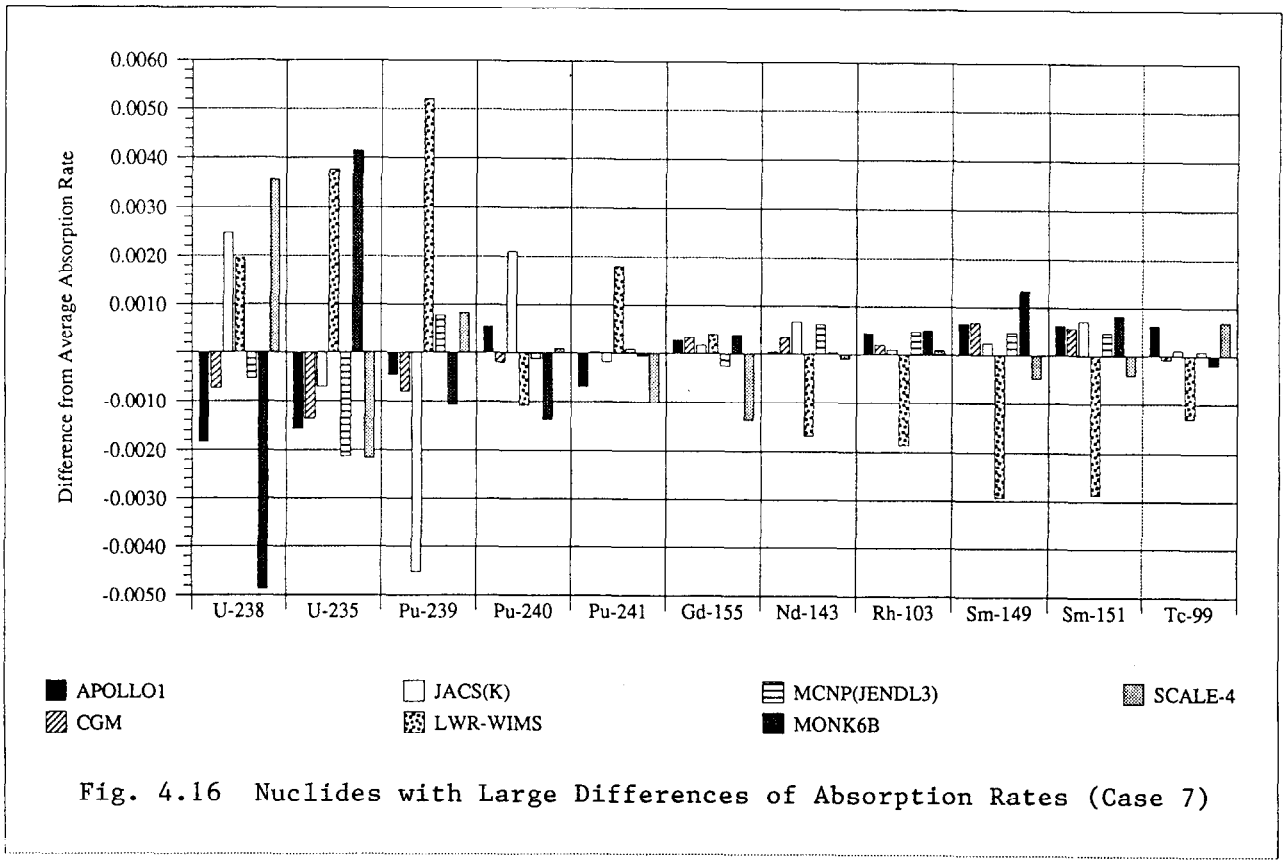


Fig. 4.16 Nuclides with Large Differences of Absorption Rates (Case 7)

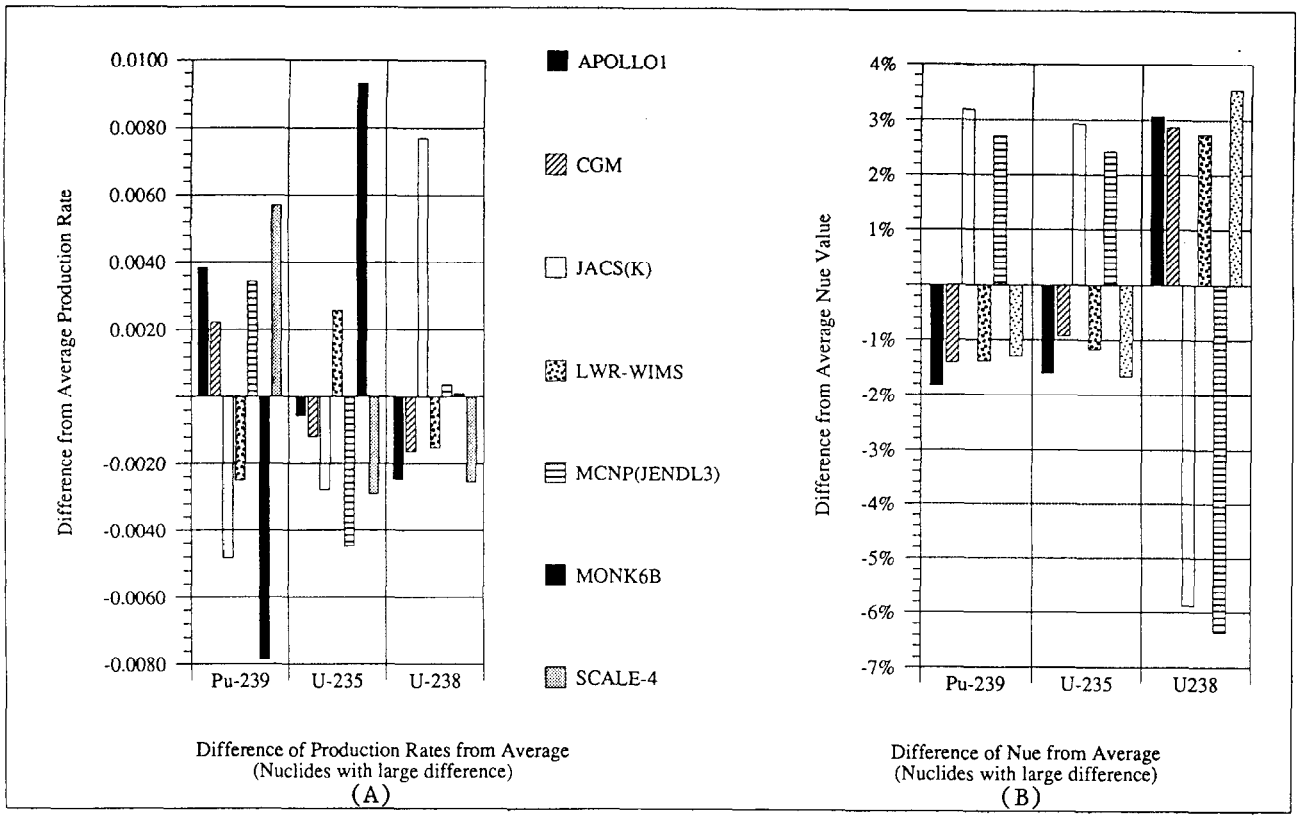


Fig. 4.17 Nuclides with Large Differences of Production Rates and ν Values (Case 7)

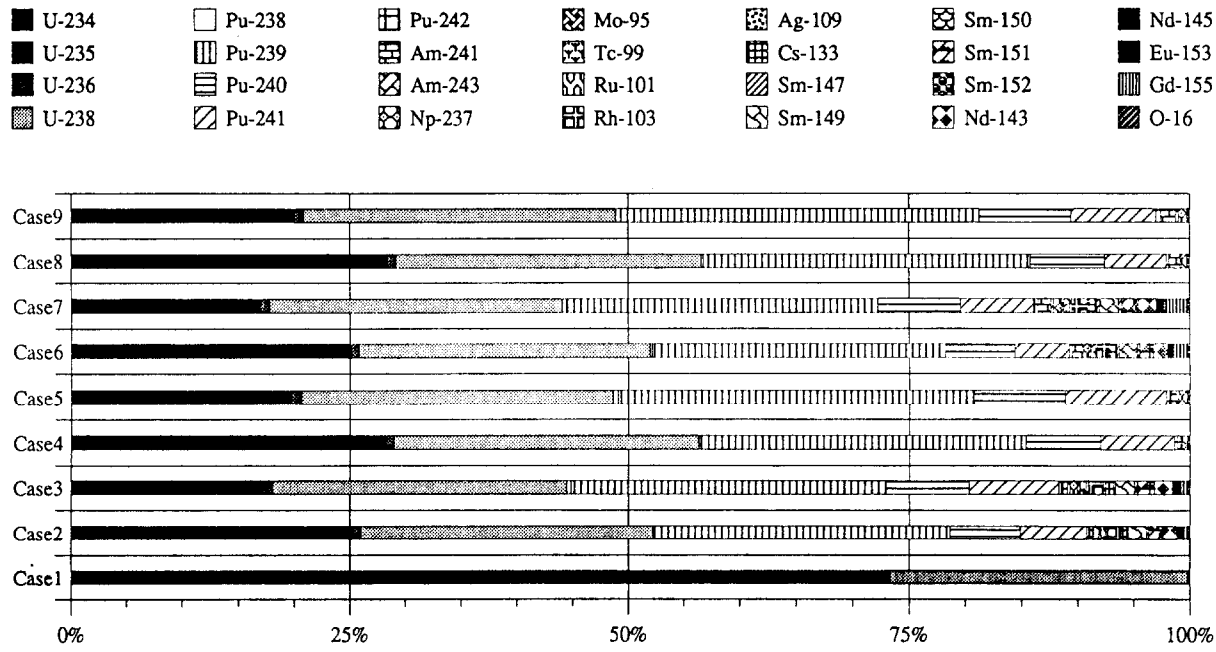


Fig. 4.18 Absorption-Rate of Each Nuclide (Example)

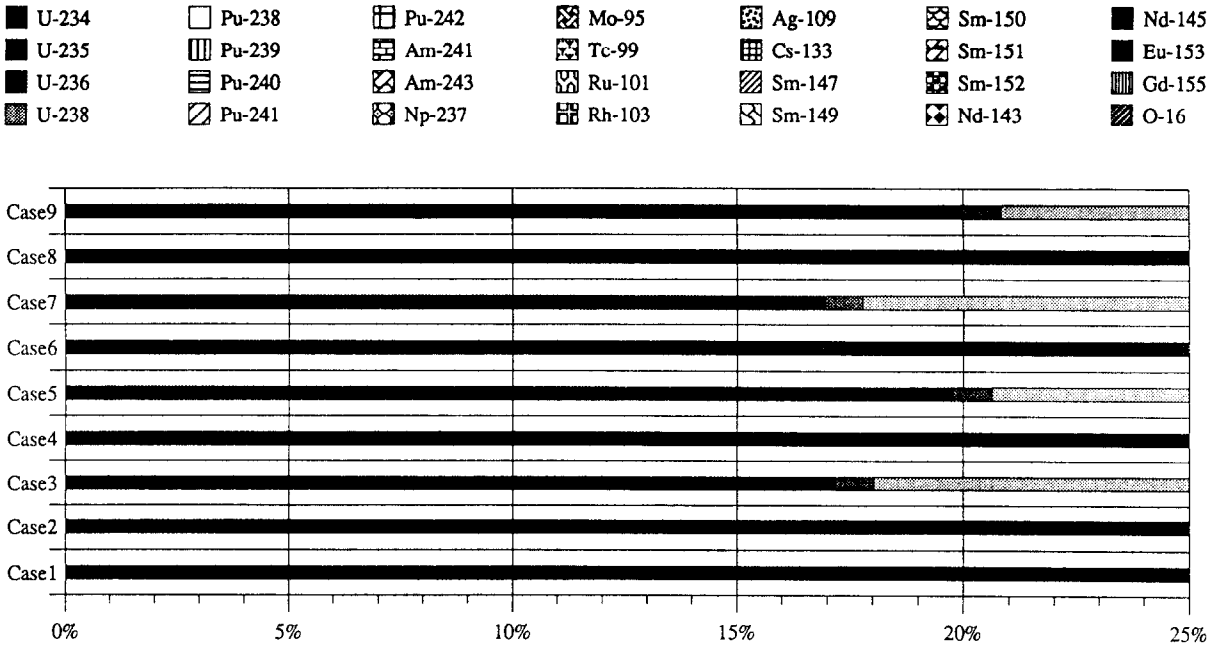


Fig. 4.19 Absorption-Rate of Each Nuclide (Enlarged)

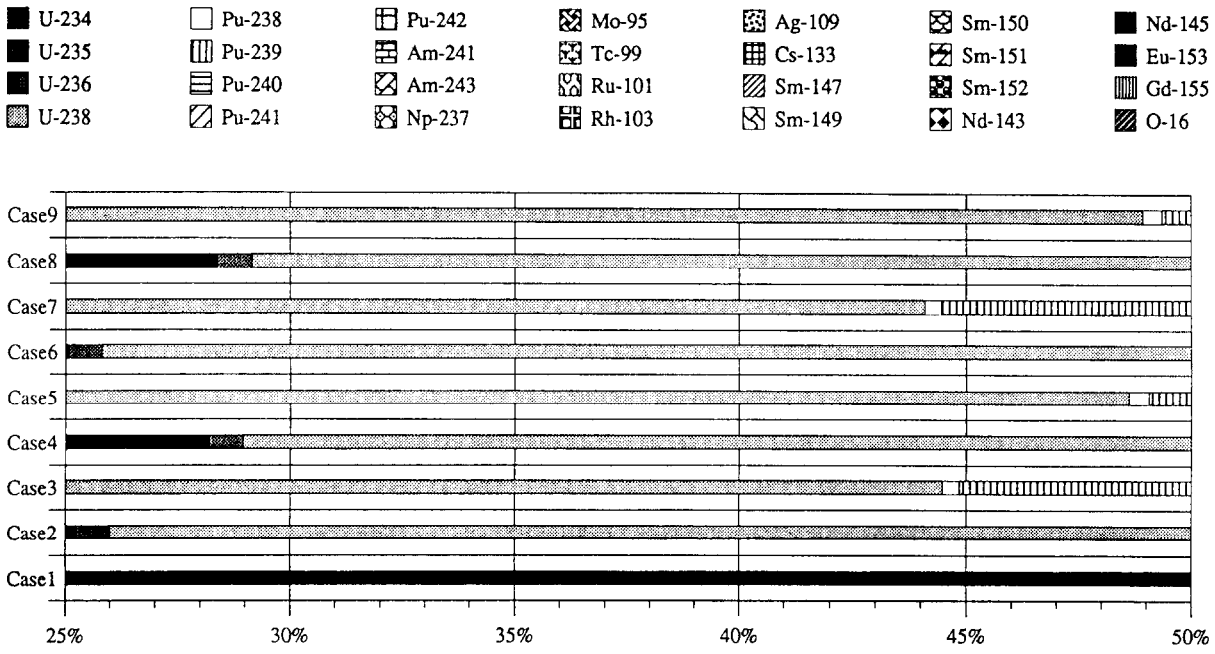


Fig. 4.19 (Continued)

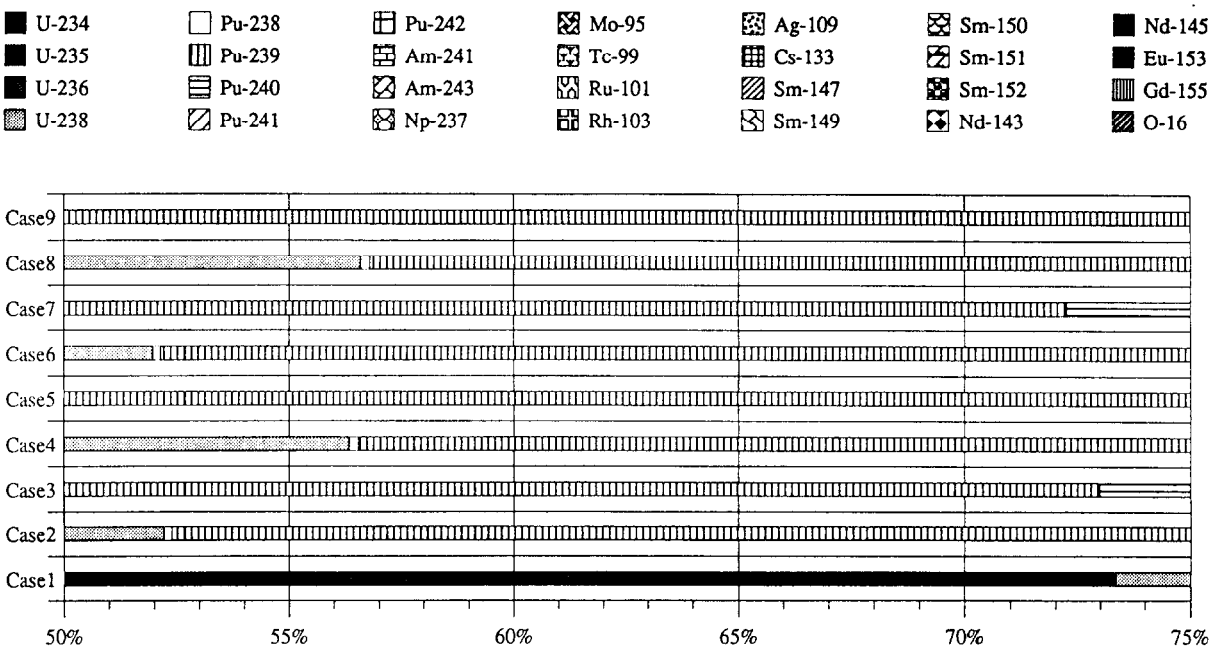


Fig. 4.19 (Continued)

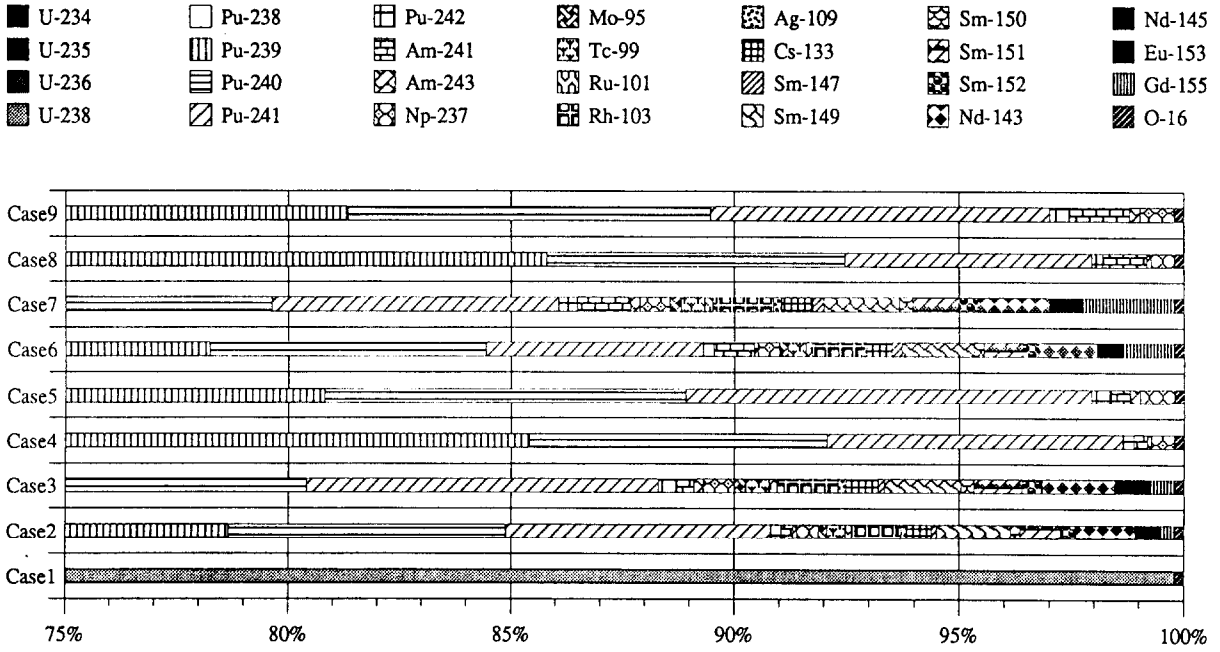


Fig. 4.19 (Continued)

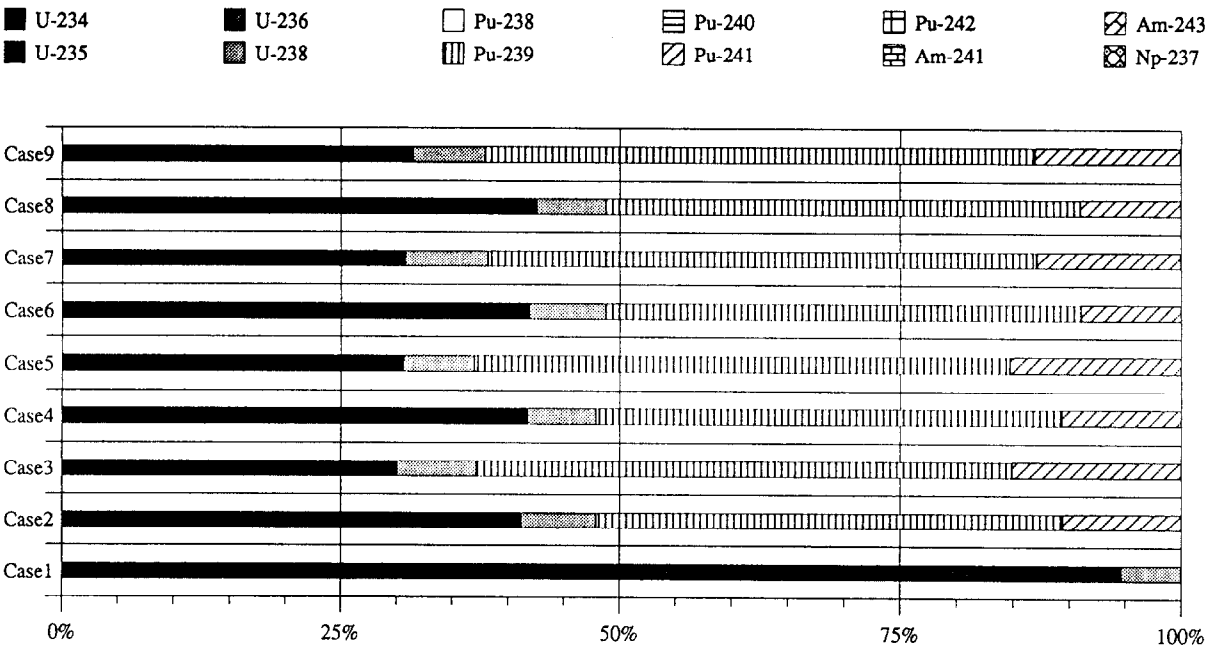


Fig. 4.20 Production-Rate of Each Nuclide (Example)

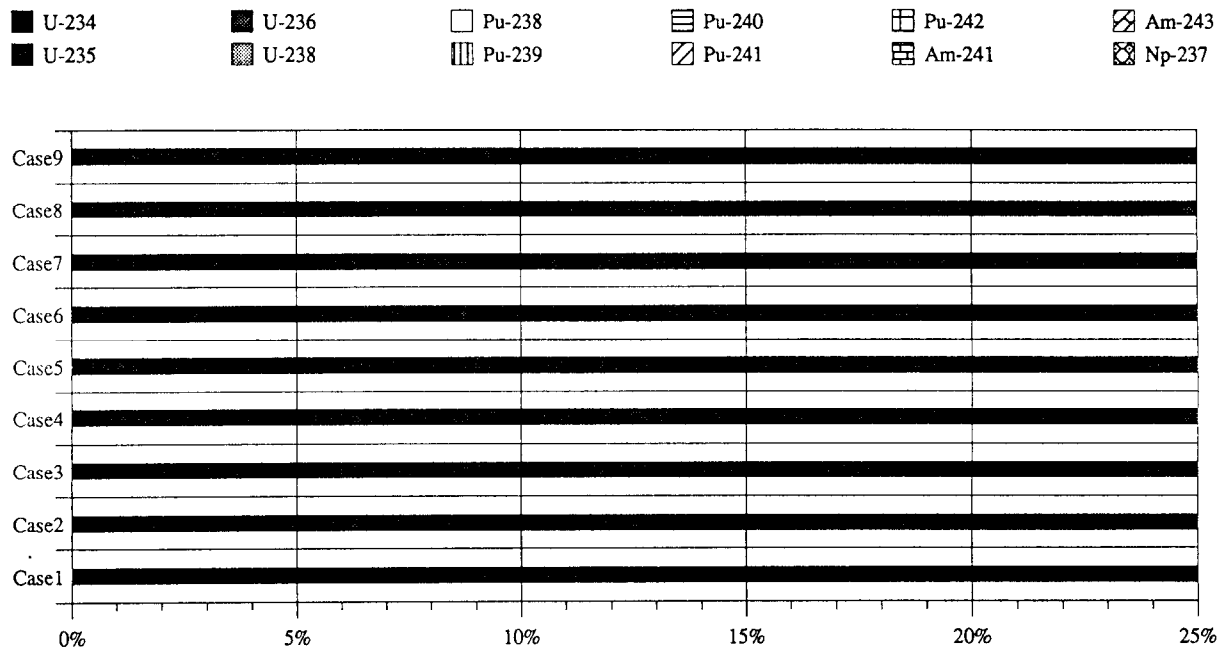


Fig. 4.21 Production-Rate of Each Nuclide (Enlarged)

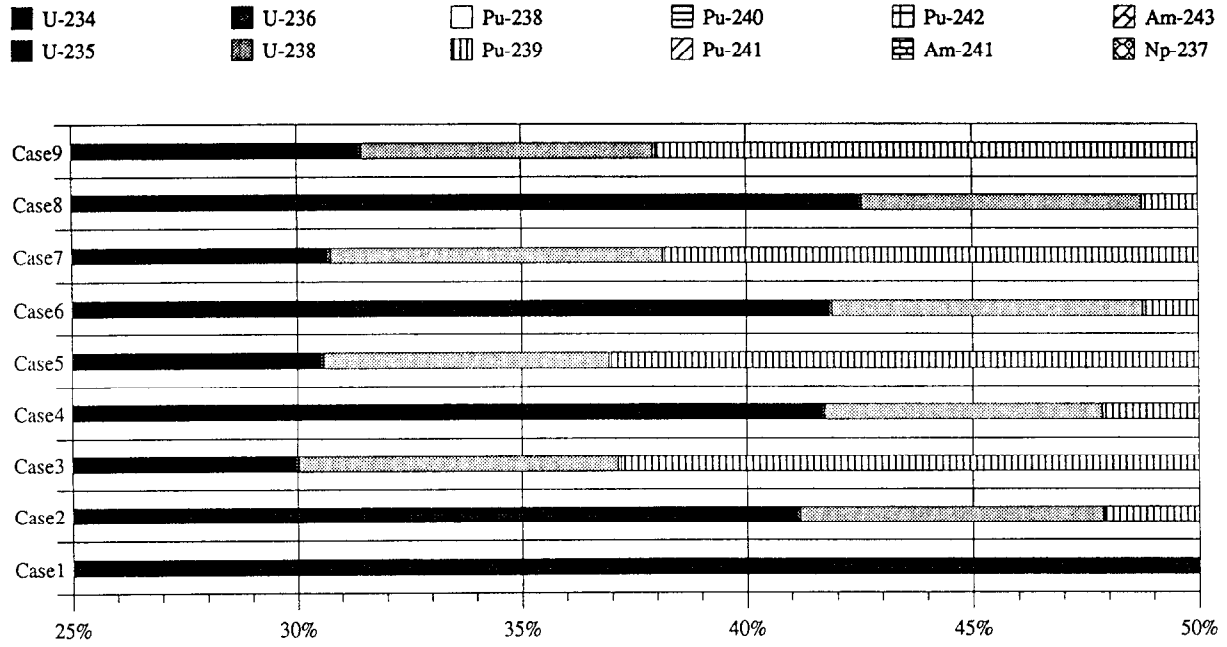


Fig. 4.21 (Continued)

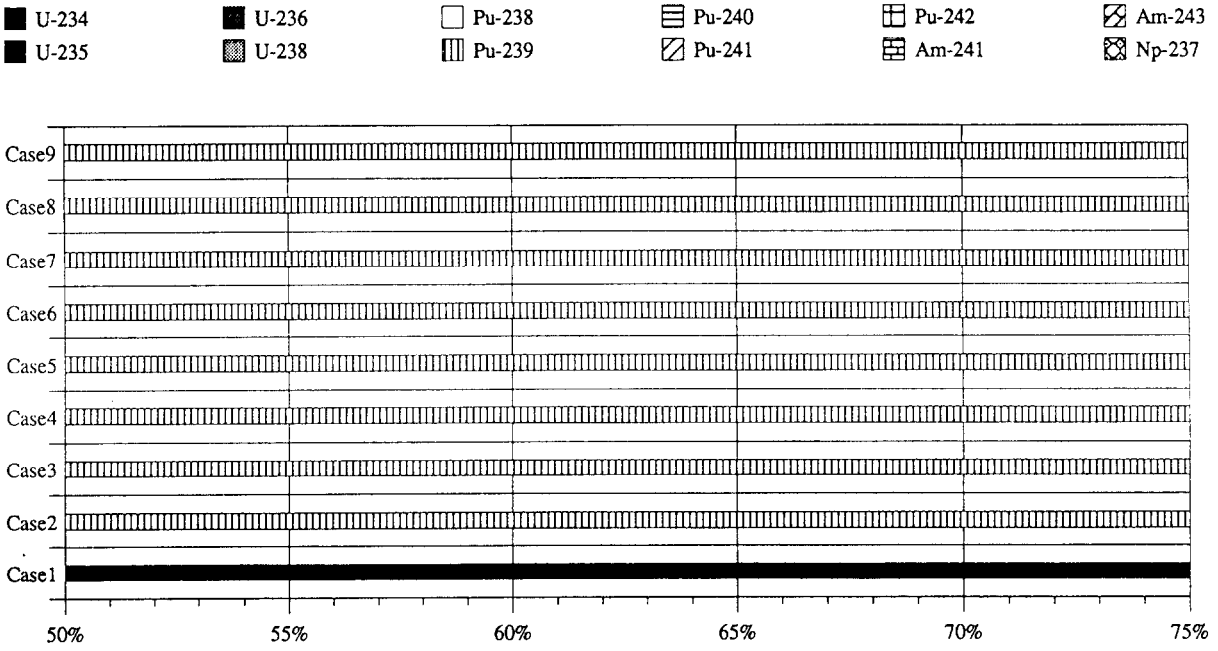


Fig. 4.21 (Continued)

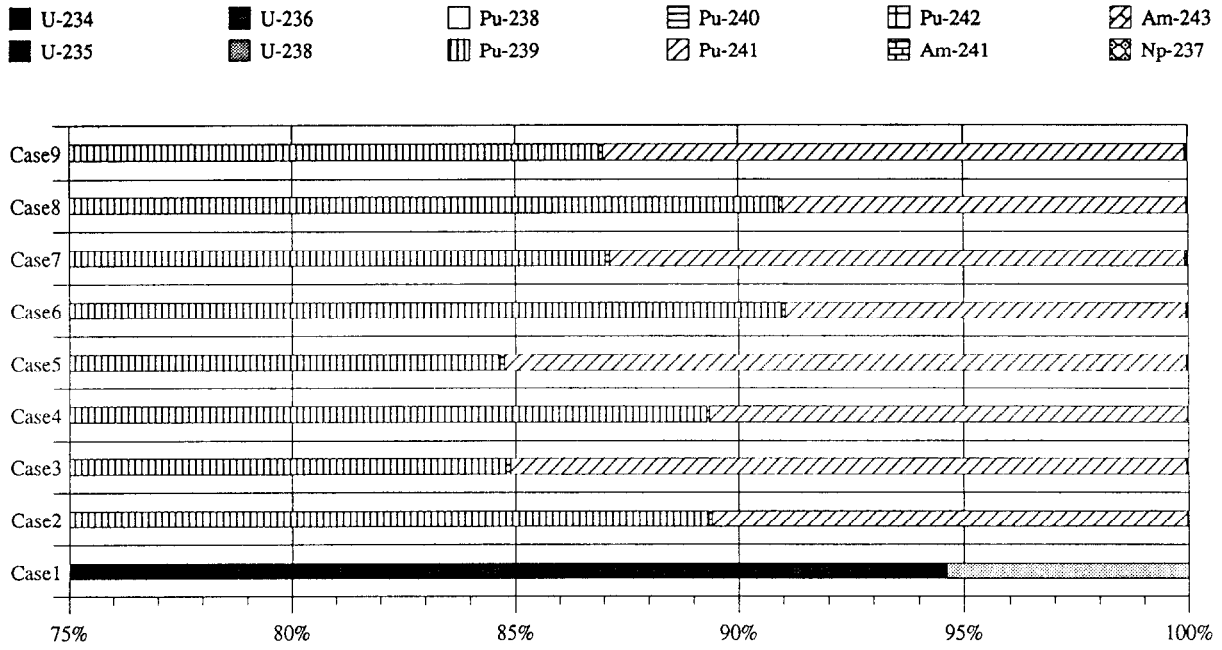


Fig. 4.21 (Continued)

Appendix 1 Specification of Benchmark Problem

Burnup-Credit Criticality Benchmark

Part 1 Simple PWR Spent Fuel Cell

(Problem Specification) NEACRP-L-337 (Revision 1)

BURNUP-CREDIT CRITICALITY BENCHMARK

Part 1 Simple PWR Spent Fuel Cell

(Problem Specification)

b y

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JAERI
Tokai-mura, Japan

and

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ORNL
Oak Ridge, USA

January 1992

OECD Nuclear Energy Agency

Burnup-Credit Criticality Benchmark
Part 1. Simple PWR Spent Fuel Cell
(Problem Specification)

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1. Overview of the Benchmark Program

In many countries, the criticality evaluation of spent fuel storage or transport casks assumes that fresh fuel is loaded instead of spent fuel. For criticality safety analysis, this assures a safety margin by simple calculations. However, this results in an excessively large criticality safety margin for typical burned light water reactor (LWR) fuel. There are indications that the nuclear power industry is moving to increased fuel enrichments in order to attain even higher burnups in LWRs. The resulting spent fuel can no longer be treated as if it were fresh. Current interest in plutonium recycle also provides an incentive to modify the fresh fuel transport assumption in that it inherently requires that spent LWR fuel be transported to and from reprocessing facilities on a timely and efficient basis. A balance needs to be sought between the two requirements: reduction of cost and optimal safety.

Conventional reactor codes and data used for in-core physics calculations can be used to evaluate the criticality state of burned LWR fuel. However, these codes involve complicated models and have large computational and data requirements. The objective of this benchmark is to verify that simple models using established away-from-reactor codes (KENO, MCNP, etc.) can be used to evaluate the criticality safety margin for spent fuel systems. The benchmark program includes PWR, BWR and MOX fuels and is designed to study the various effects on safety margins, including the effect of various fission products and axial burnup distributions taking fuel enrichment, burnup and cooling time as parameters.

Part 1 of the benchmark series can be classified as a preliminary survey of Burnup Credit where burnup, cooling time and fission products are taken as parameters. The results submitted by participants will be compared with each other in graphical form to identify sources of discrepancies. The identification of the origin of the differences in this phase (Part 1) is very important so that one can proceed with confidence into the next phase where a more complex problem will be tackled. Part 2 of the Benchmark will be the analysis of a PWR fuel assembly and a transport cask taking axial burnup distribution into account.

2. General Specification of Spent Fuel Cell (see Fig. 1)

Fuel Enrichment	3.6 w/o
Fuel Cell Pitch	1.33 cm
Fuel Radius	0.412 cm
Cladding Inner Radius	0.412 cm
Outer Radius	0.475 cm
Material	Zircalloy
Moderator	Water
Fuel Length	Infinite
Axial Burnup Distribution	Uniform
Radial Burnup Distribution	Uniform
Temperature	300 K
Atomic Number Densities	As given in Appendix A

3. Parameters

Cooling Time	1 and 5 year(s)
Fuel Burnup	0, 30 and 40 GWd/t
Fission Product	Selected and Omitted

Table 1 Reference Case Numbers

Cooling Time	Considered F.P.s	Burnup (GWd/t)		
		Fresh	30	40
1 (year)	Selected	Case 1	Case 2	Case 3
	No F.P.s		Case 4	Case 5
5 (years)	Selected	(Case 1)	Case 6	Case 7
	No F.P.s		Case 8	Case 9

4. Nuclides Specification

4.1 Actinides (12 nuclides)

U... 234, 235, 236, 238

Pu...238, 239, 240, 241, 242

Am..241, 243, Np237

4.2 Fission Products (Selected..15 nuclides)

Table 2 List of Fission Products

Nuclide	Selected	No F.P.s
Mo-95	Yes	No
Tc-99	Yes	No
Ru-101	Yes	No
Rh-103	Yes	No
Ag-109	Yes	No
Cs-133	Yes	No
Sm-147	Yes	No
Sm-149	Yes	No
Sm-150	Yes	No
Sm-151	Yes	No
Sm-152	Yes	No
Nd-143	Yes	No
Nd-145	Yes	No
Eu-153	Yes	No
Gd-155	Yes	No

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5. Requested Information and Results

(Please forward the results by electronic mail to JAERI, otherwise send a diskette or a magnetic tape by mail. We are able to read the 3.5-inch diskette recorded by Macintosh and MSDOS.)

Data	Contents
1	* General Data *
2	Date
3	Institute
4	Participants
5	Computer Code
6	Data library identification, origin, description
7	No of Energy Groups (NEG), supply 1 for continuous energy
8	Upper Energy Limit, High -> Low of each energy group(i=1,NEG)
9	* Result of Case 1 *
10	Multiplication Factor, (For Monte Carlo..No. of histories, Deviation)
11	Neutron Spectrum in water (i=1,NEG)
12	Neutron Spectrum in fuel (i=1,NEG)
13	- Reaction rates - (Total of all energy ranges)
14	U-234 Production, Absorption, Neutrons per fission
15	U-235 Production, Absorption, Neutrons per fission
16	U-236 Production, Absorption, Neutrons per fission
17	U-238 Production, Absorption, Neutrons per fission
18	Pu-238 Production, Absorption, Neutrons per fission
19	Pu-239 Production, Absorption, Neutrons per fission
20	Pu-240 Production, Absorption, Neutrons per fission
21	Pu-241 Production, Absorption, Neutrons per fission
22	Pu-242 Production, Absorption, Neutrons per fission
23	Am-241 Production, Absorption, Neutrons per fission
24	Am-243 Production, Absorption, Neutrons per fission
25	Np-237 Production, Absorption, Neutrons per fission
26	Mo-95 Absorption
27	Tc-99 Absorption
28	Ru-101 Absorption
29	Rh-103 Absorption
30	Ag-109 Absorption
31	Cs-133 Absorption
32	Sm-147 Absorption

33	Sm-149	Absorption
34	Sm-150	Absorption
35	Sm-151	Absorption
36	Sm-152	Absorption
37	Nd-143	Absorption
38	Nd-145	Absorption
39	Eu-153	Absorption
40	Gd-155	Absorption
41	O-16	Absorption

Repeat Data 9 to 41 for other Cases.

In each Case, the total of all the production rates as well as all the absorption rates should be normalized to unity. The absorption reaction rate (A_i), the production reaction rate (P_i) and the neutrons per fission (F_i) of nuclide i are defined as follows;

$$A_i = \frac{\iiint \Sigma_a^i \phi \, dEdV}{V} ,$$

where

$$\sum_{i=1}^{all} A_i \equiv 1 ,$$

$$P_i = \frac{\iiint \nu^i \Sigma_f^i \phi \, dEdV}{V} ,$$

where

$$\sum_{i=1}^{all} P_i \equiv 1 ,$$

$$F_i = \left(\frac{P_i}{\iiint \Sigma_f^i \phi \, dEdV} \right)$$

6. Atomic Number Densities to be used for Cases 1 to 9 (prepared by ORNL) will be distributed from NEA Data bank by a diskette or by an electronic mail. (See Appendix A)

7. Schedule

End November	Distribution of the Draft Specification
Mid December	Comments on Available Nuclides from each participant
End December	Distribution of Final Specification
End April	Results from all participants to JAERI
1-2 June	CWG meeting in Paris (tentative)

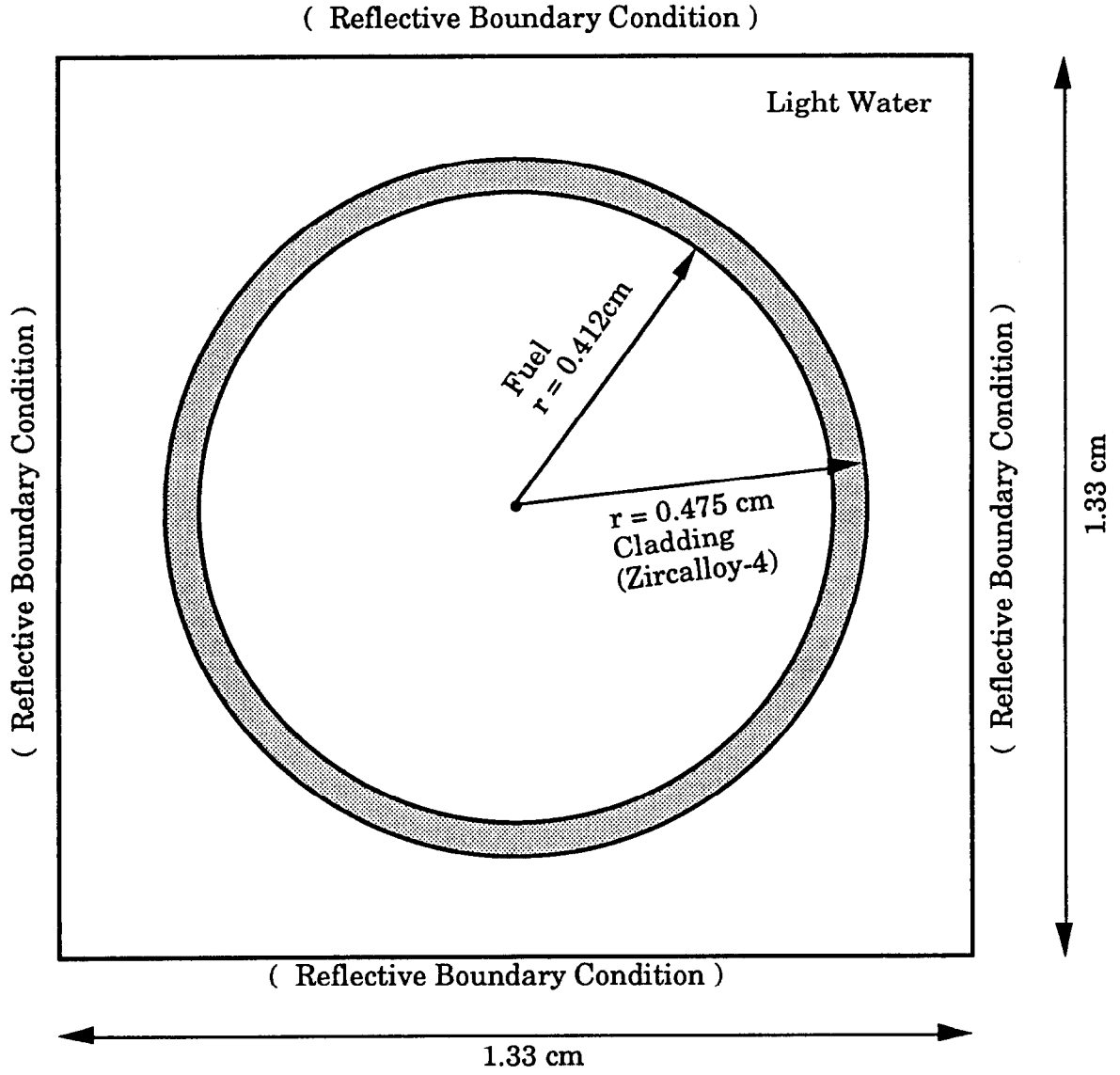

Fig. 1 Cell Configuration

Appendix A

Atomic Number Densities
(by ORNL using ORIGEN-S/SAS2H)

INFORMATION REGARDING THE CALCULATION OF ORIGEN-S/SAS2H ISOTOPICS

PWR assembly design description for tabulated cases

Parameter	Data
Assembly general data	
Designer	Westinghouse
Lattice	17 x 17
Water temperature,K	570
Water density, av, g-cm-3	0.7295
Soluble boron cycle av, ppm (wt)	550
Number of fuel rods	264
Number of guide tubes	24
Number of instrument tubes	1
Fuel rod data	
Type fuel pellet UO2	
Fellet stack density, % TD	94.5
Rod pitch, cm	1.25984
Rod OD, cm	0.94966
Rod ID, cm	0.83566
Pellet diameter, cm	0.81915
Active fuel length, cm	365.8
Effective fuel temperature, cm	811
Clad temperature, K	620
Clad material	Zircalloy
Guide tube data	
Inner radius, cm	0.5715
Outer radius, cm	0.61214
Tube material	Zircalloy

Operating History Data and Fuel Isotopic Content of PWR Cases

Specific Power	40 kW/kgU
Number of cycles	4 (for the 40 MWd/kgU case) *
Cycle duration (days)	
Uptime	250
Downtime	62.50
Uranium Content (wt%)	
U-234	0.032
U-235	3.600
U-236	0.016
U-238	96.352

Standard Burn Profile Description

The first cycle has a downtime of 20% at the middle of the cycle. The second and third cycles have a 10% downtime at both the midpoint of the cycle and at the end of the cycle. The fourth and final cycle has 100% uptime.

* Note: The 30 MWd/kgU results are taken from the third cycle results for the 40 MWd/kgU case.

Isotopics for Fresh Fuel

U-234	7.5174E-06
U-235	8.4209E-04
U-236	3.7268E-06
U-238	2.2254E-02
O	4.6215E-02

Isotopics for 30 GWd/t case

W17X17, 30000.0 MWD USING X.0WH27286.P4040F72

ORIGEN-S ISOTOPIC RESULTS 30 GWD/T FOR COOLING TIME (YR) = 0.00

U-234	4.8632E-06
U-235	2.8372E-04
U-236	9.9338E-05
U-238	2.1760E-02
NP-237	9.9766E-06
PU-238	2.6582E-06
PU-239	1.3299E-04
PU-240	4.2903E-05
PU-241	2.9424E-05
PU-242	7.4125E-06
AM-241	7.0268E-07
AM-243	1.3632E-06
O	4.6215E-02
MO-95	3.3937E-05
TC-99	4.0340E-05
RU-101	3.8222E-05
RH-103	2.1198E-05
AG-109	2.7266E-06
CS-133	4.3640E-05
ND-143	3.0248E-05
ND-145	2.4358E-05
SM-147	2.3225E-06
SM-149	1.1634E-07
SM-150	1.0403E-05
SM-151	5.8965E-07
SM-152	4.2632E-06
EU-153	3.3460E-06
GD-155	2.9042E-09

ORIGEN-S ISOTOPIC RESULTS 30 GWD/T FOR COOLING TIME (YR) = 1.00

U-234	4.8852E-06
U-235	2.8372E-04
U-236	9.9343E-05
U-238	2.1760E-02
NP-237	1.0252E-05
PU-238	2.8474E-06
PU-239	1.3523E-04
PU-240	4.2909E-05
PU-241	2.8069E-05
PU-242	7.4128E-06
AM-241	2.0558E-06
AM-243	1.3655E-06
O	4.6215E-02
MO-95	4.1050E-05
TC-99	4.0589E-05
RU-101	3.8224E-05
RH-103	2.4041E-05
AG-109	2.7336E-06
CS-133	4.4238E-05
ND-143	3.1281E-05
ND-145	2.4370E-05
SM-147	3.9702E-06
SM-149	1.7886E-07
SM-150	1.0403E-05
SM-151	5.9556E-07
SM-152	4.2633E-06
EU-153	3.3831E-06
GD-155	6.0123E-08

ORIGEN-S ISOTOPIC RESULTS 30 GWD/T FOR COOLING TIME (YR) = 5.00

U-234	4.9750E-06
U-235	2.8374E-04
U-236	9.9361E-05
U-238	2.1760E-02
NP-237	1.0281E-05
PU-238	2.8060E-06
PU-239	1.3521E-04
PU-240	4.2930E-05
PU-241	2.3243E-05
PU-242	7.4129E-06
AM-241	6.8518E-06
AM-243	1.3650E-06
O	4.6215E-02
MO	4.1250E-05
TC-99	4.0589E-05
RU-101	3.8224E-05
RH-103	2.4046E-05
AG-109	2.7336E-06
CS-133	4.4238E-05
ND-143	3.1281E-05
ND-145	2.4370E-05
SM-147	7.5335E-06
SM-149	1.7886E-07
SM-150	1.0403E-05
SM-151	5.7749E-07
SM-152	4.2635E-06
EU-153	3.3831E-06
GD-155	2.2348E-07

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Isotopics for 40 GWd/t case

W17X17, 40000.0 MWD USING X. OWH27286. P4040F72

ORIGEN-S ISOTOPIC RESULTS 40 GWD/T FOR COOLIG TIME (YR) = 0.00

U-234	4.1072E-06
U-235	1.8489E-04
U-236	1.1185E-04
U-238	2.1567E-02
NP-237	1.4196E-05
PU-238	5.2154E-06
PU-239	1.3844E-04
PU-240	5.5197E-05
PU-241	3.7809E-05
PU-242	1.3919E-05
AM-241	1.0793E-06
AM-243	3.4210E-06
O	4.6215E-02
MO-95	4.5670E-05
TC-99	5.1447E-05
RU-101	5.0605E-05
RH-103	2.6872E-05
AG-109	3.9802E-06
CS-133	5.5515E-05
ND-143	3.5868E-05
ND-145	3.0676E-05
SM-147	3.2232E-06
SM-149	1.1225E-07
SM-150	1.3915E-05
SM-151	6.6561E-07
SM-152	5.4153E-06
EU-153	4.7800E-06

ORIGEN-S ISOTOPIC RESULTS 40 GWD/T FOR COOLING TIME (YR) = 1.00

U-234	4.1503E-06
U-235	1.8489E-04
U-236	1.1186E-04
U-238	2.1567E-02
NP-237	1.4518E-05
PU-238	5.5721E-06
PU-239	1.4086E-04
PU-240	5.5231E-05
PU-241	3.6067E-05
PU-242	1.3920E-05
AM-241	2.8177E-06
AM-243	3.4254E-06
O	4.6215E-02
MO-95	5.2483E-05
TC-99	5.1694E-05
RU-101	5.0607E-05
RH-103	2.9917E-05
AG-109	3.9889E-06
CS-133	5.6108E-05
ND-143	3.6860E-05
ND-145	3.0678E-05
SM-147	4.9000E-06
SM-149	1.7846E-07
SM-150	1.3915E-05
SM-151	6.7154E-07
SM-152	5.4154E-06
EU-153	4.8278E-06
GD-155	9.9550E-08

ORIGEN-S ISOTOPIC RESULTS 40 GWD/T FOR COOLING TIME (YR) = 5.00

U-234	4.3261E-06
U-235	1.8491E-04
U-236	1.1188E-04
U-238	2.1567E-02
NP-237	1.4557E-05
PU-238	5.4910E-06
PU-239	1.4084E-04
PU-240	5.5350E-05
PU-241	2.9867E-05
PU-242	1.3920E-05
AM-241	8.9793E-06
AM-243	3.4241E-06
O	4.6215E-02
MO-95	5.2674E-05
TC-99	5.1693E-05
RU-101	5.0607E-05
RH-103	2.9921E-05
AG-109	3.9889E-06
CS-133	5.6108E-05
ND-143	3.6860E-05
ND-145	3.0687E-05
SM-147	8.5258E-06
SM-149	1.7845E-07
SM-150	1.3915E-05
SM-151	6.5116E-07
SM-152	5.4156E-06
EU-153	4.8279E-06
GD-155	3.7023E-07

NEA/NSC/DOC(92)11

ANNEX (NEACRP-L-337)

Part I-A Specification (M. Brady)

The purpose of this problem is to evaluate the contributions of major and minor actinides, and major and minor fission products in the reduction of reactivity for burned PWR fuel.

(1) Cases 10 and 1 should be calculated using the specification of NEACRP-L-337 with only the major actinides (i.e. delete Pu-242, Pu-238, Am-241, Am-243 and Np-237 from the previous case 4 and case 8 calculations).

(2) Using the fuel assembly and operating history descriptions on page 10 of Part I specification (NEACRP-L-337), calculate the isotopic inventory for fuel burned to 30 GWd/MTU and cooled to 1 yr. and 5 yr.

The number densities for minor fission products (all F.P. other than the 15 major F.P. previously selected) should be taken from this calculation - all other number densities should be taken from the previous specification This new subset of isotopics will be used in cases 12 and 13.

Cooling	MAJ ACT	ALL ACT	MAJ FP	ALL FP
1 yr	CASE 10	CASE 4	CASE 2	CASE 12
5 yr	CASE 11	CASE 8	CASE 6	CASE 13

* Only infinite multiplication factors will be calculated.

Appendix 2 Reports and Comments from Participants

(The data listings of the original reports have been omitted in order to reduce pages.)

Appendix 2.1 Reports from Technical Research Center of Finland

"The Analysis of NEACRP Burnup-Credit Criticality Benchmark, Part 1: Simple PWR Spent Fuel Cell"

VTT YDINVOIMATEKNIIKAN LABORATORIO

TYÖRAPORTTI
TECHNICAL REPORT

Projekti/työ -Project identification	Sivuja - Pages	Päiväys - Date	Raportin nro - Report No.
PERUS	5	25.9.1992	RFD-22/92

Otsikko ja tekijä - Title and author

THE ANALYSIS OF NEACRP BURNUP-CREDIT CRITICALITY BENCHMARK
PART I: SIMPLE PWR SPENT FUEL CELL

T. Roine, M. Anttila

Hyväksynyt - Approved by
Liitta Kyrki-Rajamäki

Julkisuus - Availability statement
free

Jakelu - Distribution

K.Haule	STUK
K.Valtonen	STUK
P.Siltanen	IVO
M.Antila	IVO
M.Solala	TVO
J.Peltonen	TVO
L.Mattila	VTT/YDI
O.Tiihonen	"
R.Kyrki-Rajamäki	"
R.Höglund	"
F.Wasastjerna	"

Kierto:

K.Helariutta	VTT/YDI
E.Kaloinen	"
T.Narumo	"
M.Rajamäki	"
H.Räty	"
M.Saarinen	"
T.Vanttola	"

Abstrakti, sisällysluettelo tms. - Abstract, list of contents etc.

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1. INTRODUCTION

The first part of the Burnup-Credit Criticality Benchmark organized by NEACRP (at present NEANSC) /1/ was calulated both with CASMO-3 fuel assembly burnup program and with KENO5A-PC Monte Carlo Criticality code. Two sets of CASMO-3 calculations were performed using both the 40-group (E4LTJ40) and 70-group (E4LTJ70) data library based mainly on ENDF/B-IV /2/. In KENO calculations both the 16-group Hansen-Roach (XN16, SCALE version with some additional ENDF/B-IV -data) and the 27-group burnup library based on ENDF/B-IV (XN27BURN,SCALE) /3/ were utilized. Only the CASMO-3 results from the 40-group library calculations were send to the comparison. In this report the multiplication factors of the CASMO and the KENO calculations are given. For the comparison the average values of the k_∞s of the benchmark participants are included.

2. INPUT SPECIFICATION

The NEACRP Burnup-Credit Criticality Benchmark Part 1 (Simple PWR Spent Fuel Cell) consists of the analyses of a pin cell in an infinite lattice having burnup, cooling time and fission products as parameters. The pin cell geometry was specified as follows:

- fuel pellet radius	0.412 cm
- outer cladding radius	0.475 cm
- pitch of the square unit cell	1.330 cm

The material compositions of fresh fuel, cladding and water are given in Table 1.

Table 1. Atomic number densities (1/b·cm) of fresh fuel, cladding and water

Nuclide	Fuel pellet	Cladding	Water
^{1}H			$6.6630 \cdot 10^{-2}$
^{16}O	$4.6215 \cdot 10^{-2}$		$3.3317 \cdot 10^{-2}$
Zircalloy		$4.3244 \cdot 10^{-2}$	
^{234}U	$7.5174 \cdot 10^{-6}$		
^{235}U	$8.4209 \cdot 10^{-4}$		
^{236}U	$3.7268 \cdot 10^{-6}$		
^{238}U	$2.2254 \cdot 10^{-2}$		

The problem was subdivided to eight (nine) cases according to the burnup, cooling time and selection of fission products as shown in Table 2.

Table 2. Case numbers

Cooling time (a)	Considered F.P.s	Fresh	Burnup (GWd/tU) 30	40
1	Selected	Case1	Case 2	Case 3
	No F.P.s		Case 4	Case 5
5	Selected		Case 6	Case 7
	No F.P.s		Case 8	Case 9

The atomic number densities of the isotopes in the fuel are given in Table 3.

Table 3. Atomic number densities (1/b·cm) in the fuel

Nuclide	Case 1	Case2 and Case4	Case 3 and Case 5	Case 6 and Case 8	Case 7 and Case 9
^{234}U	$7.5174\cdot10^{-6}$	$4.8852\cdot10^{-6}$	$4.1503\cdot10^{-6}$	$4.9750\cdot10^{-6}$	$4.3261\cdot10^{-6}$
^{235}U	$8.4209\cdot10^{-4}$	$2.8372\cdot10^{-4}$	$1.8489\cdot10^{-4}$	$2.8374\cdot10^{-4}$	$1.8491\cdot10^{-4}$
^{236}U	$3.7268\cdot10^{-6}$	$9.9343\cdot10^{-5}$	$1.1186\cdot10^{-4}$	$9.9361\cdot10^{-5}$	$1.1188\cdot10^{-4}$
^{238}U	$2.2254\cdot10^{-2}$	$2.1760\cdot10^{-2}$	$2.1567\cdot10^{-2}$	$2.1760\cdot10^{-2}$	$2.1567\cdot10^{-2}$
^{237}Np		$1.0252\cdot10^{-5}$	$1.4518\cdot10^{-5}$	$1.0281\cdot10^{-5}$	$1.4557\cdot10^{-5}$
^{238}Pu		$2.8474\cdot10^{-6}$	$5.5721\cdot10^{-6}$	$2.8060\cdot10^{-6}$	$5.4910\cdot10^{-6}$
^{239}Pu		$1.3523\cdot10^{-4}$	$1.4086\cdot10^{-4}$	$1.3521\cdot10^{-4}$	$1.4084\cdot10^{-4}$
^{240}Pu		$4.2909\cdot10^{-5}$	$5.5231\cdot10^{-5}$	$4.2930\cdot10^{-5}$	$5.5350\cdot10^{-5}$
^{241}Pu		$2.8069\cdot10^{-5}$	$3.6067\cdot10^{-5}$	$2.3243\cdot10^{-5}$	$2.9867\cdot10^{-5}$
^{242}Pu		$7.4128\cdot10^{-6}$	$1.3920\cdot10^{-5}$	$7.4129\cdot10^{-6}$	$1.3920\cdot10^{-5}$
^{241}Am		$2.0558\cdot10^{-6}$	$2.8177\cdot10^{-6}$	$6.8518\cdot10^{-6}$	$8.9793\cdot10^{-6}$
^{243}Am		$1.3655\cdot10^{-6}$	$3.4254\cdot10^{-6}$	$1.3650\cdot10^{-6}$	$3.4241\cdot10^{-6}$
^{16}O[1]	$4.6215\cdot10^{-2}$	$4.6215\cdot10^{-2}$	$4.6215\cdot10^{-2}$	$4.6215\cdot10^{-2}$	$4.6215\cdot10^{-2}$
^{95}Mo[2]		$4.1050\cdot10^{-5}$	$5.2483\cdot10^{-5}$	$4.1250\cdot10^{-5}$	$5.2674\cdot10^{-5}$
^{99}Tc[2]		$4.0589\cdot10^{-5}$	$5.1694\cdot10^{-5}$	$4.0589\cdot10^{-5}$	$5.1693\cdot10^{-5}$
^{101}Ru[2]		$3.8224\cdot10^{-5}$	$5.0607\cdot10^{-5}$	$3.8224\cdot10^{-5}$	$5.0607\cdot10^{-5}$
^{103}Rh		$2.4041\cdot10^{-5}$	$2.9917\cdot10^{-5}$	$2.4046\cdot10^{-5}$	$2.9921\cdot10^{-5}$
^{109}Ag		$2.7336\cdot10^{-6}$	$3.9889\cdot10^{-6}$	$2.7336\cdot10^{-6}$	$3.9889\cdot10^{-6}$
^{133}Cs		$4.4238\cdot10^{-5}$	$5.6108\cdot10^{-5}$	$4.4238\cdot10^{-5}$	$5.6108\cdot10^{-5}$
^{143}Nd		$3.1281\cdot10^{-5}$	$3.6860\cdot10^{-5}$	$3.1281\cdot10^{-5}$	$3.6860\cdot10^{-5}$
^{145}Nd		$2.4370\cdot10^{-5}$	$3.0687\cdot10^{-5}$	$2.4370\cdot10^{-5}$	$3.0687\cdot10^{-5}$
^{147}Sm		$3.9702\cdot10^{-6}$	$4.9000\cdot10^{-6}$	$7.5335\cdot10^{-6}$	$8.5258\cdot10^{-6}$
^{149}Sm		$1.7886\cdot10^{-7}$	$1.7846\cdot10^{-7}$	$1.7886\cdot10^{-7}$	$1.7845\cdot10^{-7}$
^{150}Sm		$1.0403\cdot10^{-5}$	$1.3915\cdot10^{-5}$	$1.0403\cdot10^{-5}$	$1.3915\cdot10^{-5}$
^{151}Sm		$5.9556\cdot10^{-7}$	$6.7154\cdot10^{-7}$	$5.7749\cdot10^{-7}$	$6.5116\cdot10^{-7}$
^{152}Sm		$4.2633\cdot10^{-6}$	$5.4154\cdot10^{-6}$	$4.2635\cdot10^{-6}$	$5.4156\cdot10^{-6}$
^{153}Eu		$3.3831\cdot10^{-6}$	$4.8278\cdot10^{-6}$	$3.3831\cdot10^{-6}$	$4.8279\cdot10^{-6}$
^{155}Gd[1]		$6.0123\cdot10^{-8}$	$9.9550\cdot10^{-8}$	$2.2348\cdot10^{-7}$	$3.7023\cdot10^{-7}$

1) reaction rates not available in CASMO
2) excluded in CASMO calculations

3. GENERATION OF KENO CROSS SECTIONS

The cross sections for KENO were processed with BONAMI and NITAWL codes. BONAMI performs the resonance self-shielding calculations in the unresolved resonance energy range applying the Bondarenko method. NITAWL applies the Nordheim integral method to perform the resonance shielding calculations in the resolved resonance energy range. It also changes the cross section library format suitable for KENO. Both programs have been taken from the SCALE-4 program package and used as stand-alone programs. /3/

4. RESULTS

In the Table 4 the infinite multiplication factors and standard deviation of KENO results in different cases are listed. In the KENO calculations the XN27BURN library was used in the cases where fission products were requested. Otherwise XN16 library was chosen because it seemed to yield better results than XN27BURN, which can be seen in Case 1 in Table 4. XN16 library has no data for fission products.

The larger differences in the cases where fission products were included may result from the selected data library (XN27BURN). Those other participants using the SCALE-4 package have reported results which are in agreement with our k_{∞}s.

Table 4. The infinite multiplication factors

Case	CASMO		KENO		AVERAGE
no.	E4LTJ40	E4LTJ70	XN16	XN27BURN.	
1	1.4366	1.4370	1.4354 ±0.0028	1.4239 ±0.0025	1.4409
2	1.1488	1.1492		1.1285 ±0.0027	1.1454
3	1.0740	1.0744		1.0570 ±0.0026	1.0673
4	1.2478	1.2483	1.2446 ±0.0029		1.2466
5	1.1903	1.1909	1.1910 ±0.0023		1.1882
6	1.1209	1.1212		1.0981 ±0.0024	1.1181
7	1.0337	1.0340		1.0161 ±0.0025	1.0300
8	1.2307	1.2312	1.2350 ±0.0028		1.2297
9	1.1669	1.1675	1.1692 ±0.0027		1.1662

Also the original Hansen-Roach cross section library was utilized in benchmark calculations. This library systematically yielded greater than average values for multiplication factors. Partially the differences were due to the lack of minor actinide data in the Hansen-Roach library. This was tested calculating the Case 4 (Case 10) with CASMO and KENO (using XN16 library) omitting isotopes ^{237}Np, ^{241}Am and ^{243}Am. The Case 8 (Case 11) was also calculated with CASMO. The following values were received:

	Case 10	Case 11
CASMO	1.2639	1.2552
KENO (XN16)	1.2601	
KENO (H-R)	1.2809	

There is still a difference of 2000 pcm, which might be caused by the improper selection of the cross section data set from the Hansen-Roach library. It is somewhat unclear how the resonance correction term σ_p should be defined /4/. However, the direct use of Hansen-Roach data (i.e. no BONAMI and NITAWL calculations) seems to yield fairly good results.

It can be assumed that the more detailed study of the KENO options, the use of different data libraries and the data processing capabilities could give results which are close to the average.

REFERENCES

1. Takano, M., Brady, M.C., Burnup-Credit Criticality Benchmark, Part 1. Simple PWR Spent Fuel Cell (Problem Specification). January 1992, NEACRP-L-339 (Revision 1).

2. Edenius, M., Forssén, Bengt H., CASMO-3. A Fuel Assembly Burnup Program. User's Manual. Version 4.4 Studsvik/NFA-89/3. (Restricted distribution)

3. RSIC Computer Code Collection: SCALE-4, A Modular Code System for Performing Standardized Computer Analyses for Lincensing Evaluation. Vols. I-III (Draft January 1990). (CCC-545)

4. Landers, N.F., Petrie, L.M., Uncertainties Associated With The Use of The KENO Monte Carlo Criticality Codes.

Appendix 2.2 Report from JINS

"A Comparative Study of Neutron Group Constants through the NEA/NSC Burnup Credit Criticality Benchmark"

A Comparative Study of Neutron Group Constants through the NEA/NSC Burnup Credit Criticality Benchmark

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Abstract

Two sets of nuclear group constants, MGCL-JINS and SCALE-4, were comparatively studied using the NEA/NSC Burnup Credit Criticality Benchmark. The effective multiplication factor with MGCL-JINS resulted to be about 1.1% Δk larger than that with SCALE-4. Through the Benchmark, it may be concluded that characteristics of the reactivity decrease with burnup are mainly defined by the evaluated nuclear data files used for group constant preparations.

INTRODUCTION

In November 1991, the Burnup Credit Criticality Benchmark was proposed by the Nuclear Science Committee of the Nuclear Energy Agency. The objective of the benchmark was defined to verify that simple models employed in the away-from-reactor codes, KENO, MCNP, etc., can be used to evaluate the criticality safety margin for spent fuel systems[1]. Evaluation of the criticality safety of fuel cycle facilities has been studied in the Institute of Nuclear Safety for the last several years, and a set of commonly used criticality analysis codes, ANISN, KENO-V.a, the MGCL-JINS neutron group constants, and the MAIL program for providing the effective cross sections, have been prepared[2]. The SCALE code systems[3], including its latest version 4.1, have been also installed as an alternate method for the criticality evaluation. Then, using the Burnup Credit Criticality Benchmark, a comparative study of the neutron group constants, MGCL-JINS and SCALE-4, has been made.

ANALYSIS

Benchmark Specifications:

As shown in Table 1, eleven cases out of the thirteen Phase-IA problems were selected for the comparative study of the neutron group constants. Case Nos. 12 and 13, which require nuclear data for all fission products, were omitted. In Case Nos. 2, 3, 6 and 7, 'Major Fifteen' fission products are specified to be taken into consideration. All these nuclides are included in SCALE-4, but only five of them are prepared in MGCL-JINS. So that, the four cases of 'Available Five' were added to the calculations with SCALE-4 and was made the comparison with the MGCL-JINS calculations. However, our study laid much emphasis on the effects due to the actinoids and the comparisons were made mostly for the 'No' fission product cases.

Neutron Group Constants:

Table 2 shows key features of the neutron group constants used in this study, SCALE-4 and MGCL-JINS, including the lists of nuclides specified and the criticality analysis methods employed. Both of these neutron group constants are based on the ENDF/B-IV data file. Differences in the group structures and the methods for evaluating the effective cross sections may bring some effects on the results.

RESULTS AND DISCUSSIONS

Multiplication Factors:

The effective multiplication factors evaluated with SCALE-4 and MGCL-JINS for the all eleven cases are summarized in Table 3. All results but two cases for 'Major Actinoids' are plotted in Fig. 1, and selected seven results for 'No' fission product cases in Fig. 2 respectively. It is clearly shown that the effective multiplication factors calculated with the MGCL-JINS neutron group constants are larger than those with SCALE-4: about 1.1% Δk for the fresh fuel case. From the results for the burnup of 30 GWd/t, it is indicated that the difference of actinoid nuclides included, all 12 or major 7, brings nearly same difference among the multiplication factors evaluated by MGCL-JINS and SCALE-4. Detailed investigation of the calculated results reveals discrepancies of the fission rates with U-238 in higher energy groups: the smaller

neutron production from the U-238 fast fission in the SCALE-4 calculations. The typical of them found in the case for the fresh fuel is shown in Fig. 3. Differences of the group structures in the energy range above 1MeV may bring the discrepancies.

Reactivity Decreases:

Reactivity decreases with burnup are easily calculated as the difference of effective multiplication factors, and shown in Table 4. As indicated in Figs. 1 and 2, the reactivity decreases of MGCL-JINS and SCALE-4 are in good agreement, not only for the cases without fission product but for the cases with five fission products. But, the neutron production rates by fissionable nuclides in the burnup fuel are not same. Table 5 shows the neutron production rates of each fissionable nuclides for the SCALE and the MGCL calculations and the ratio of these values for the fresh and the irradiated fuels. In the fresh fuel, the neutron production rate of U-235 is exactly same for two methods. But, in the irradiated fuels, two methods evaluates different production rates for fissionable nuclides. As clearly shown in the SCALE/MGCL ratio for U-235 and Pu-239, MGCL-JINS brings greater U-235 and less Pu-239 contributions, and less U-235 and greater Pu-239 contributions in SCALE. Differences in the reactivity decreases between MGCL and SCALE may appear for higher burnup fuels.

Dependence on Nuclear Data File:

In Reference One, the multiplication factors of all participants to the Benchmark and origin of the nuclear data used by them were tabulated. The nuclear data files can be classified in three categories: ENDF/B-IV, JEF and JENDL-3. So that, we have selected six typical cases, including our two, and calculated reactivity changes from fresh fuel with burnup. Results are shown in Fig. 4. The reactivity changes with burnup show a similar characteristic when the same nuclear data file, ENDF/B-IV or JEF-1, are used, not affected with the group structures, the correction methods for resonance self-shieldings or the criticality calculation codes. However, the results based on JENDL-3 do not agree with this finding.

CONCLUSIONS

Using the Burnup Credit Criticality Benchmark proposed by the NEA/NSC, a comparative study of the neutron group constants, MGCL-JINS and SCALE-4, has been made.

The effective multiplication factors resulted with MGCL-JINS are about 1.1% Δk larger than those with SCALE-4. Precise comparison of of the results from MGCL-JINS and SCALE-4 shows differences in the neutron production rates by fissionable nuclides in burnup fuels; MGCL-JINS calculates greater U-235 and less Pu-239 contributions, and less U-235 and greater Pu-239 contributions by SCALE-4.

The reactivity decrease characteristics with burnup are mainly defined by the evaluated nuclear data files used for group constant preparations, ENDF/B-IV or JEF (except JENDL-3), not affected strongly with the group structures, the correction methods for resonance self-shieldings or the criticality calculation codes.

REFERENCES

[1] M. TAKANO and E. SARTORI, "NEACRP Burnup Credit Criticality Benchmark, Result of Phase-1A: DRAFT," (1993).

[2] S. MITAKE, et al., "Analysis of Benchmark Critical Experiments using MGCL-JINS Cross-section Library with P_3 Scattering Matrix," Proc. of ICNC'91, V-18 (1991).

[3] "SCALE-4: A Modular Code System for Performing Standard Computer Analysis for Licensing Evaluation," ORNL/NUREG/CR-0200.

Table 1 Parameters and Case Numbers

Cooling Time (Years)	Actinoids	Fission Products	Burnup(GWd/t)		
			Fresh	30	40
1	Major 7	No		Case 10	
	All 12	No	Case 1	Case 4	Case 5
	All 12	Major 15 / Availalbe 5		Case 2	Case 3
	All 12	All availables		(Case 12)	
5	Major 7	No		Case 11	
	All 12	No	(Case 1)	Case 8	Case 9
	All 12	Major 15 / Availalbe 5		Case 6	CAse 7
	All 12	All availables		(Case 13)	

Case 12 and 13 are not included in this study.

Table 2 Neutron Group Constants

Name	SCALE-4(27 groups)		MGCL-JINS
Nuclear Data File	ENDF/B-IV		
No. of groups	27		137
Actinoids	Major 7: U-234, -235, -236, -238, Pu-239, -240, -241 All 12 : Major 7 and Np-237, Pu-238, -242, Am-241, -243		
Fission Products	Major 15 (Standard Case)	Available 5	
Mo- 95	Yes	No	
Te- 99	Yes	Yes	
Ru-101	Yes	No	
Rh-103	Yes	No	
Aq-109	Yes	Yes	
Cs-133	Yes	Yes	
Sm-147	Yes	No	
Sm-149	Yes	Yes	
Sm-150	Yes	No	
Sm-151	Yes	No	
Sm-152	Yes	No	
Nd-143	Yes	No	
Nd-145	Yes	No	
Eu-153	Yes	No	
Gd-155	Yes	Yes	
Effective cross-section calc.	NITAWL-II (Nordheim) (SCALE-4/CSAS1X)		MAIL (Bondarenko)
Criticality calc. (1-D Sn)	XSDRNPM-S (SCALE-4/CSAS1X)		ANISN-JR

Table 3 Multiplication Factors

CASE	SCALE-4 (Standard Case)	SCALE-4 (Available 5 FPs)	MGCL-JINS (*:Available 5 FPs)
1	1.4273		1.4486
2	1.1321	1.1954	1.2139 *
3	1.0568	1.1320	1.1491 *
4	1.2395		1.2580
5	1.1833		1.2005
6	1.1049	1.1670	1.1844 *
7	1.0177	1.0898	1.1050 *
8	1.2225		1.2411
9	1.1601		1.1174
10	1.2571		1.2757
11	1.2502		1.2687

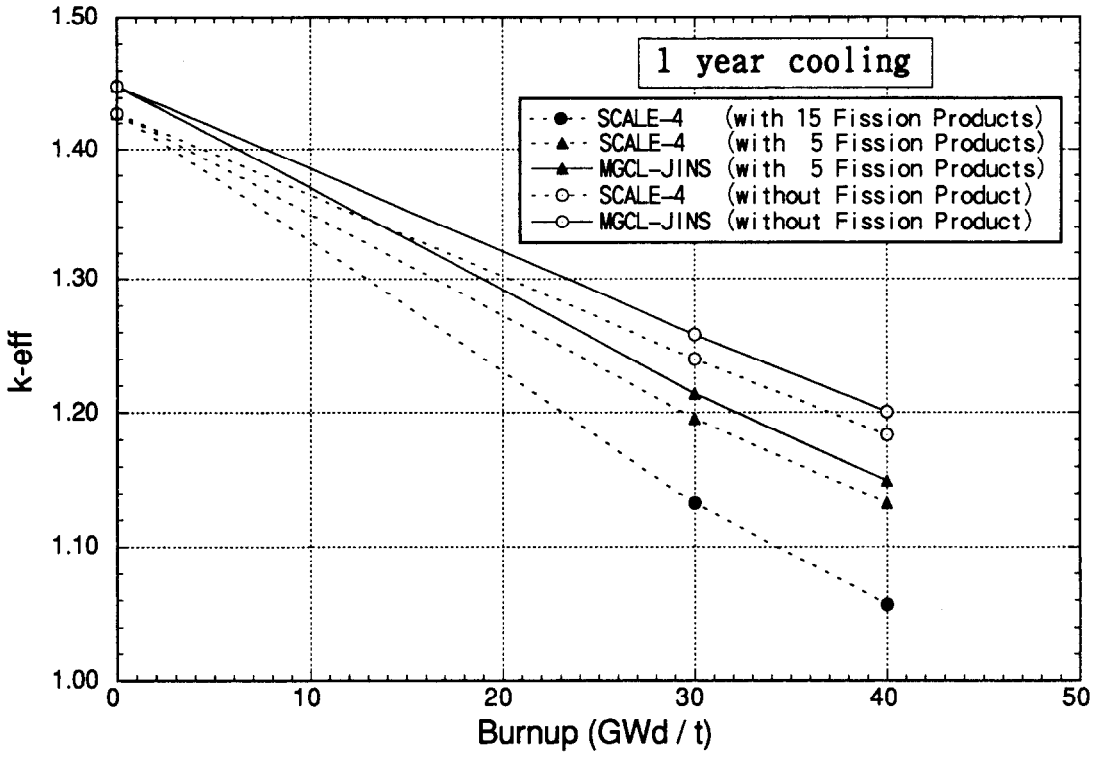
Table 4 Reactivity Decreases (Δk)

CASE	SCALE-4 (Standard Case)	SCALE-4 (Available 5 FPs)	MGCL-JINS (*:Available 5 FPs)
1	0.0		0.0
2	-0.2952	-0.2319	-0.2347 *
3	-0.3705	-0.2953	-0.2995 *
4	-0.1878		-0.1906
5	-0.2440		-0.2481
6	-0.3224	-0.2603	-0.2642 *
7	-0.4096	-0.3375	-0.3436 *
8	-0.2048		-0.2075
9	-0.2672		-0.2712
10	-0.1702		-0.1729
11	-0.1771		-0.1799

Table 5 Comparison of Neutron Production Rates Calculated by SCALE-4 and MGCL-JINS

Nuclide	Fresh fuel			Irradiated (30GWD/MTU)			Irradiated (40GWD/MTU)		
	SCALE	MGCL	SCALE/MGCL	SCALE	MGCL	SCALE/MGCL	SCALE	MGCL	SCALE/MGCL
U-234	9.15E-05	9.99E-05	0.916	6.08E-05	6.52E-05	0.933	5.19E-05	5.55E-05	0.935
U-235	1.36E+00	1.36E+00	1.000	5.16E-01	5.26E-01	0.982	3.60E-01	3.68E-01	0.979
U-236	2.09E-05	2.44E-05	0.857	5.75E-04	6.54E-04	0.879	6.52E-04	7.38E-04	0.883
U-238	6.50E-02	7.90E-02	0.823	6.59E-02	7.76E-02	0.849	6.58E-02	7.71E-02	0.853
Np-237				1.70E-04	1.83E-04	0.930	2.42E-04	2.60E-04	0.933
Pu-238				2.66E-04	2.71E-04	0.979	5.43E-04	5.55E-04	0.978
Pu-239				5.23E-01	5.17E-01	1.013	5.74E-01	5.69E-01	1.009
Pu-240				8.29E-04	8.81E-04	0.942	1.07E-03	1.13E-03	0.944
Pu-241				1.32E-01	1.32E-01	0.995	1.80E-01	1.81E-01	0.992
Pu-242				1.12E-04	1.21E-04	0.923	2.11E-04	2.28E-04	0.926
Am-241				9.92E-05	1.02E-04	0.972	1.39E-04	1.43E-04	0.972
Am-243				1.73E-05	1.89E-05	0.915	4.36E-05	4.76E-05	0.916
	1.4273	1.4410	0.990	1.2394	1.2549	0.988	1.1833	1.1990	0.987

Cooling Time : 1 year



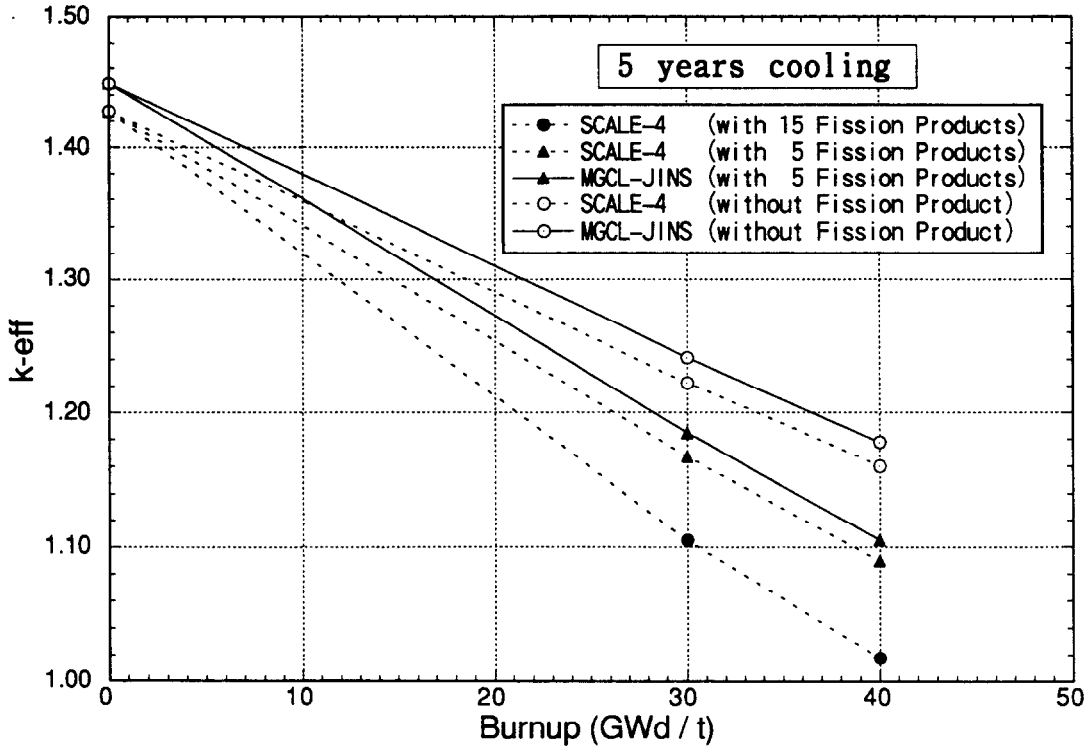


Fig. 1 Multiplication Factors (k-eff) with Burnup; Dependence on group constants and fission products (including all 12 actinoids).

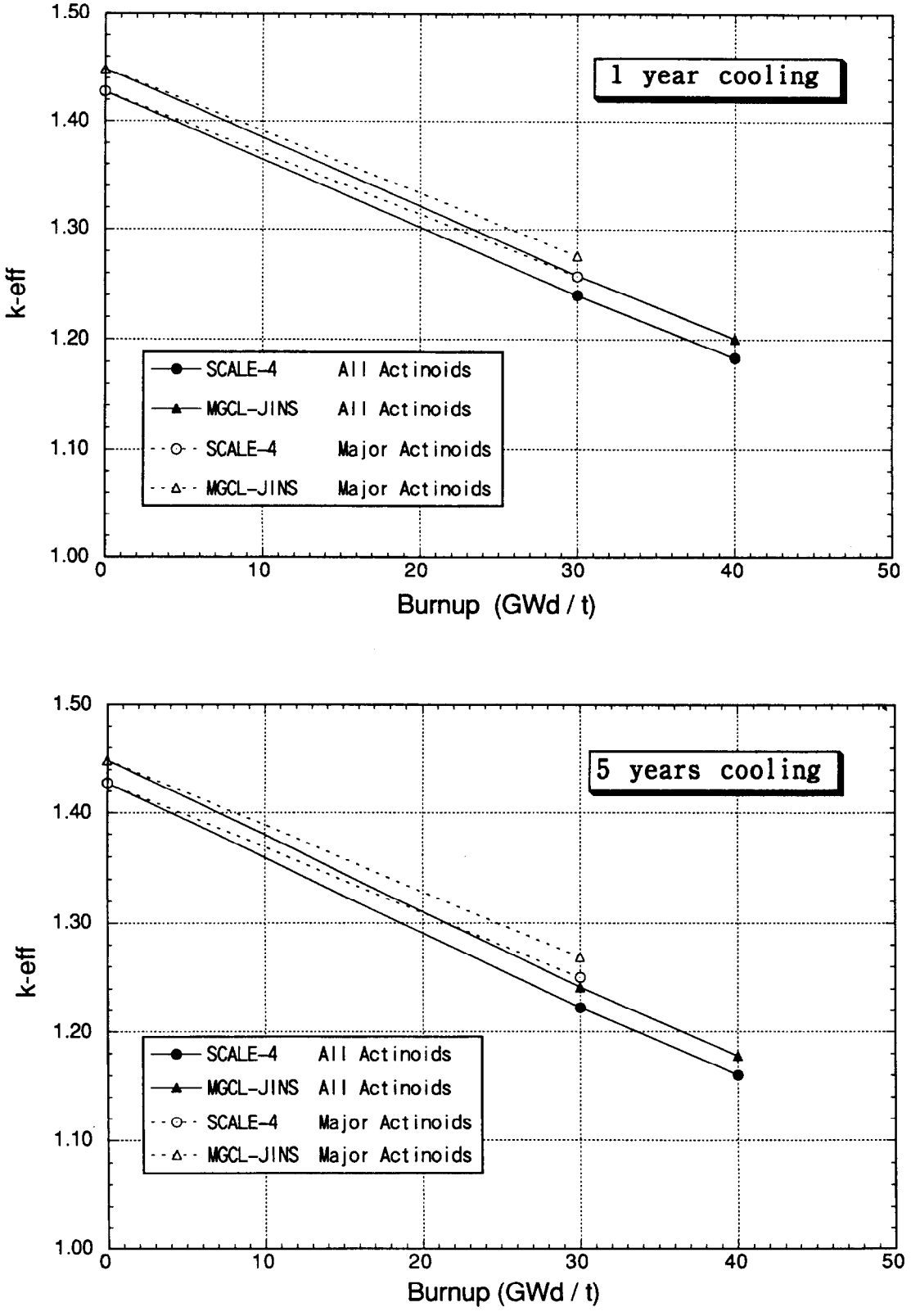


Fig. 2 Multiplication Factors (k-eff) with Burnup;
Dependence on group constants (without fission product).
Major Actinoids: U-234, 235, 236, 238, Pu-239, 240, 241
All Actinoids: Major Actinoids + Np-237, Pu-238, 242, Am-241

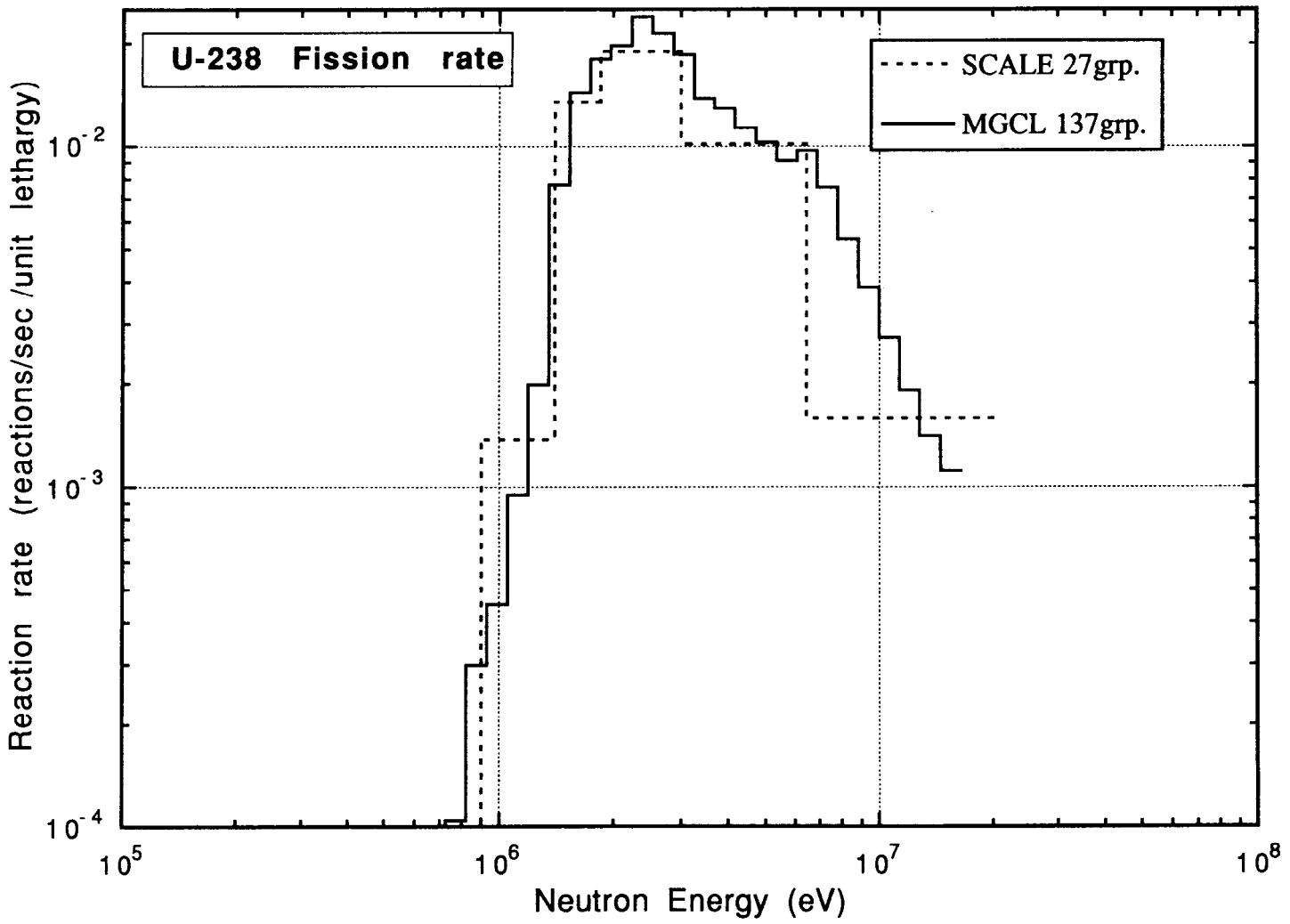


Fig. 3 U-238 Fission Rates at the Energy above 100keV

EMS **DRAFT REPORT**

Burnup-Credit and Nuclear Criticality Safety

DENNIS MENNERDAHL
NOVEMBER 6, 1992

This report includes the results
of participation in part 1 of an
OECD/NEA working group on
Burn-up credit criticality benchmark

The project is sponsored
by the Swedish Nuclear
Power Inspectorate

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Summary

The SCALE-4 calculation code system has been used to calculate some problems that were specified by an OECD/NEA Working Group. The Working Group study is titled "Burnup-credit criticality benchmark".

The calculation of neutron multiplication factors, k_∞ gave results that agree very well with those members using similar methods. Also the other parameters (neutron production rates, absorption rates, neutrons per fission, neutron spectrum in fuel and water) are, after a recent revision, in close agreement with results of other members using similar methods. The revision was not caused by code implementation problems but by errors in the treatment of the generated output.

During discussions at the last meeting of the Working Group, I informed the other members that I was somewhat disappointed with the current coverage of the study. The determination of the contents of spent fuel is normally carried out by specialists at the reactor sites. The quality of that work is the basis for later criticality safety applications. Reporting and verifying the contents and the history of the fuel is also essential. The Working Group has not taken advantage enough of the wide interest in the current study, the availability of specialists and good calculation methods to simplify a future necessary communication between fuel management specialists and criticality safety specialists.

A short discussion and proposals for future studies are included in the report. The lack of calculation cases using different burnup codes is one area that should be expanded. Another, related area is that the most realistic cases, those with all fission products in the fuel are not studied at all. This is a serious omission since any real verification (reactivity, residual heat, radiation) will be based on the real contents.

The Working Group has initiated a second part of the study. It includes two-dimensional effects of similar cases as in part one. It is not yet clear whether I will participate in all calculations of this second part. It will require use of the KENO-Va code in SCALE-4. KENO-Va has not yet been implemented and verified as a part of the current SCALE-4 installation here. This is expected to be completed before the end of the year.

Introduction

OECD/NEA has initiated a working group study of methods that could be used for taking advantage of the burnup of spent nuclear fuel for nuclear criticality safety control. A large number of organizations from many countries are directly or indirectly involved in the working group. Some of the best specialists and methods from industrial organizations, research laboratories and national authorities are available to the working group.

The basic advantage in taking advantage of burnup for criticality safety control is that current designs could be used with new fuel types and that storage and transport of spent fuel could be made more economical. This solution will require less natural resources while not reducing the overall safety. Some countries have additional interests such as reprocessing plants for spent fuel. Sweden is not supporting reprocessing.

The main subject of the study is nuclear criticality safety. To find safe procedures it is not always necessary to use the most accurate tools available. What is necessary is that the responsible organization and authority have a complete understanding of all significant parameters influencing the safety.

Discussion

A nuclear criticality safety specialist has to cover several areas. The numerical calculation of the neutron multiplication factor is just one such area. Asking for the right input parameters and verifying their accuracy is another area. This is where the major obstacle to taking advantage of burnup for criticality safety control can be found. It can not be separated from the problems of verifying the contents of each spent fuel assembly or even of different sections of the same assembly.

The interest in taking advantage of burnup of spent fuel for criticality safety control is not new. The OECD/NEA working group has started its work with some very detailed work. In my opinion there is a lack of perspective. Before a benchmark study can be carried out successfully, it is necessary to evaluate the quality of the input data. The quality refers to the accuracy and the precision of the input data and its applicability to realistic situations. During the last meeting in June, 1992 there were references made to national studies that cover various aspects of the subject. They may be valuable as background material but may also be insufficient. With all the different organizations that can give input to the working group, it is not too early to get started on preparing background information.

According to my point of view, the current Part 1 of the working group study involves too many details and calculations while limiting the study to some artificial numbers that may never be qualified as benchmarks. It also seems strange to me that so much emphasis is put

on specified fission products when the real and experimentally verified cases involve fresh fuel or spent fuel with all fission products present. Work involving verification of fissile materials for nonproliferation purposes has also included spent fuel for a long time and is mandatory. This work could be of value if the actual presence of uranium and plutonium was accounted for. The presence of only certain fission products is not normally verified.

Some proposals for the next phases are included in this report.

Criticality Safety Perspectives

It is the responsibility of the criticality safety specialist to request the right information and to verify that it is correct and complete. With this information available it still remains to show that criticality safety is acceptable. With spent fuel it seems logical that the reactor physicists, whose main responsibility has been to maintain criticality using the same fuel, will be the best people to find out the properties of the spent fuel. They use good calculational and measurement methods on a routine basis and significant errors would be discovered during operation. When it comes to determining the criticality safety of the fuel outside of the reactor there are other factors to be considered. Clean water (no boron) and better reflector materials than water are some of those factors. However, the most important difference is probably the human factor and the lack of measurements to verify calculations.

The first place where burnup can be taken advantage of outside the reactor itself is in the spent fuel pool at the reactor. The operators may be very confident that each fuel assembly has the documented history and that the associated burnup is accurately determined. Still, there are possibilities for mistakes. If burnup is used for criticality safety control, it is necessary to restrict it to certain sections of the spent storage pool. Fuel that is unloaded from the reactor should not be allowed directly into these sections.

During shipment of the spent fuel there are other questions. The reactor operators may be as confident about the contents of the fuel as when it is handled in the storage pools. The shipper and the receiver of the shipment normally can not be as confident without additional controls. Examples of controls are the documentation and the physical verification of fuel assemblies that are established for nonproliferation purposes. Additional controls could be the measurement of heat and radiation levels on the transport package (if the package surface is room temperature and the radiation is close to the background level somebody might question whether the package is loaded with spent fuel assemblies). Such measurements could be a part of the criticality safety procedures. Simplified computer models of the fuel package for determination of heat and radiation levels could be developed, adjusted and verified through experiments.

At the receiving end of the shipment (could be in another part of the world) similar concerns apply. If burnup is not taken advantage of for criticality safety control at the reactor site or during shipment, the receiver will have more reasons to verify the reported contents. This can be done before the shipment but some additional possibilities may also be of interest. A large central storage facility for spent fuel may invest in techniques such as reactivity, isotopic and/or heat measurements of each fuel assembly.

The final step for spent fuel is either final disposal of the fuel (somewhere, sometime) or reprocessing of the fuel. In both cases it may be of interest to take advantage of the burnup. During final storage it may be required that a future criticality should not be possible (even after many thousand, maybe million years). In such cases it may be of interest to take advantage of various combinations of fissionable nuclides and fission products.

If fission products are taken advantage of, there is also a question whether all fission products still remain in the fuel. Could some volatile fission products have escaped already without being noticed? What are the possibilities of losing such fission products during the current operation, through normal handling or incidents? What is the maximum effect on criticality safety?

Current Working Group Study

The first part of the working group is now being completed. It involved a simplified, 1-D (1-dimensional) geometric model with an infinite number of identical, homogeneous spent fuel pins. The atomic number densities were given for a number of nuclides and for a few different burnup rates and cooling times. The number densities were calculated with a simplified burnup calculation method for typical PWR fuel. The fission products were limited to 15.

The method used by the Working Group to calculate the atomic number densities is of the type suitable for a criticality specialist in that it is expected to give adequate results without requiring too much detailed information on reactor operating data. It is not representative for the quality of normal methods used for

reactor design and management. Such methods are expected to give more accurate results.

Results for the first part was reported to the coordinator before the previous meeting. Some of the data were not reported in a consistent form and some of the Working Group members were asked to supply revised numbers.

The first part of the Working Group was to be finished and reported to OECD/NEA during the autumn 1992.

Unfortunately, I have not had the opportunity to revise my own results earlier. The neutron multiplication factors that I obtained previously seem to agree well with those of other members using similar methods. However, the reaction rates are not consistent. It is obvious that something is wrong when the different cases are compared. The calculation system that I have used (based on SCALE-4) is currently being implemented and has not been validated or applied to calculate reaction rates, only neutron multiplication factors. Further work has now been carried out but unfortunately not in time to be included in the first Working Group report. The result of the revision is that there was no error in the calculation system. All can be referred to the "human factor". The production rates previously reported were fission rates. Data for the fission products had been misplaced. The formula specified by the coordinator for normalized neutrons per fission turned out to be wrong.

The revised EMS results are given in tables that follow the text.

During the meeting of the Working Group in June 1992, some additional cases were added to part 1. They involved calculation of fission products using other methods. There was very little interest in the calculation of fissionable nuclides using other methods. Since the burnup calculation method in SCALE-4 is not yet implemented here, these additional cases have not been carried out.

The second part of the study was determined during the last meeting of the Working Group. It involves 2-D models of spent fuel assemblies. The axial distribution of atoms is not constant. The real axial burnup varies, with a reduced rate at the ends of the fuel assembly. The study is limited to atomic number densities, calculated with the same method as in part one. The second part is also limited to the same fission products as in part one. The real situation with all fission products in the fuel is not included. Different cooling times and burnup rates are included, similar to part one. This means that the number of calculation problems is quite large.

The calculations for the second part of the working group should be reported in the spring 1993, before the next meeting.. Specifications will be sent to the members of the Working Group.

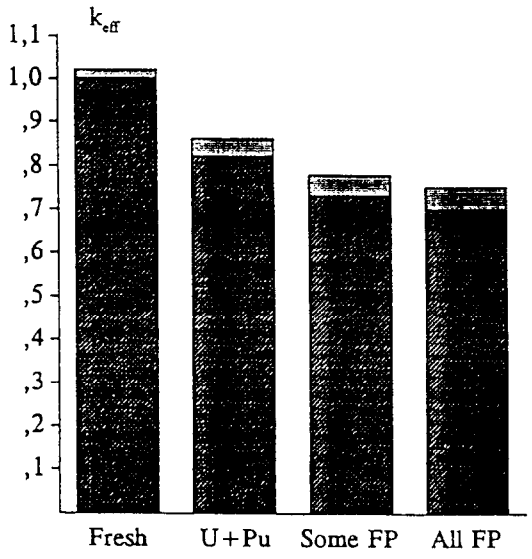
My own participation in the next part is not clear yet. Before the end of the year I will decide whether all of the second part or just a subset should be calculated.

Future Work - Proposals

A typical example of taking advantage of burnup for criticality safety control is shown in the following figure. Four cases are included representing

1. Fresh fuel
2. U+Pu in spent fuel
3. Some actinides and some fission products (FP)
4. Spent fuel with all nuclides present.

Each bar includes the calculated effective neutron multiplication factor k_{eff} and an addition to cover uncertainties and known errors.



It would be logical to include at least three of these cases in all applications taking advantage of burnup. If fission products are taken advantage of, all four cases are essential. Some of the reasons are given below.

1. Fresh fuel. This would be the first reference case. The uncertainties should be small and similar cases may be found in previous applications or even in experiments. The fuel has been used in a reactor and

its original properties have been verified at various stages.

2. Spent fuel with contents limited to uranium and plutonium isotopes. This is information of vital interest to nonproliferation (safeguards) authorities. It has been documented and often to some extent verified that the fuel and documentation are in agreement.

3. Spent fuel with some actinides and some fission products. If some of the fission products are to be taken advantage of this case is necessary. The reason is that some of the fission products in the fuel may be missing for some reason. This is particularly true for volatile fission products.

4. Spent fuel with real contents (all fission products). This is in almost all cases the real composition of the spent fuel. As a reference case it will give information on the spent fuel and about safety margins. The neutronic properties of the spent fuel has, to some extent, been verified during the operation of the reactor. If there is going to be any real verification of the contents of the spent fuel (isotopic, residual heat generation, reactivity, radiation etc.) it would have to be compared with the real contents. It is true that in reprocessing, final waste disposal or accident conditions other compositions may be found. The case with all fission products present will still be the starting point for determining the current contents.

The case number 4 above is not included in the current parts 1 or 2 of the Working Group study. I find it hard to understand why the members did not want to include the case that, except the fresh fuel, is most realistic and easiest to measure. There are probably practical reasons. One reason could be that methods to calculate neutron multiplication factors for spent fuel are not normally used by criticality specialists. Without a good understanding of how well the burnup for the studied fuel is calculated and verified, the criticality specialist can't really base any credible safety analysis on it.

When fresh fuel is shipped or being stored, the authorities and receivers normally can trust the numbers given by the shipper or manufacturer of the fuel. The U-235 enrichments and weights are controlled by several organizations.

How will the contents of spent fuel be determined, reported and verified if all the changes during radiation in a reactor are taken advantage of for criticality safety control?

The first step of a burnup study should really be to evaluate how well the burnup can be determined, reported and verified. All possible variations of fuel types and reactor operation should be considered. For example: will a fuel assembly taken out of the reactor after 10 months have the same axial burnup if the reactor is operated with an 18 month cycle as if it was operated with a 12 month cycle? What types of errors can be possible in the burnup determination?

It is possible that some of the members have participated in previous national studies that dealt with questions like these. This could be helpful but it is not a replacement for an international study. Especially, for the safe international transport of spent fuel, credibility of the methods used in the safety reports are essential for the approvals of the national competent authorities.

The Working Group should consider these questions when selecting future calculation cases.

Step 1 - Burnup determination

First, some simple cases as in part 1 of the current Working Group study should be calculated with the same methods as are normally used for burnup determination in various countries. The input would be the specifications for fresh fuel and the design and history of the reactor. In the first part the design was really simple; an infinitely long fuel pin cell with reflecting boundaries.

A criticality safety specialist does not have to carry out these burnup calculations. It is more realistic to leave the calculations to specialists who are familiar with burnup calculations and who will be responsible for the specifications of the spent fuel from each reactor.

The results of the burnup calculations can then be compared by the Working Group. Multiplication factors, fuel contents and other parameters can be compared. As mentioned before, residual heat generation and radiation levels may also be of value.

A realistic full scale reactor operation could then be simulated as the next step.

Finally, with the conclusions from the previous steps and with the help from reactor management specialists, some extreme cases should be selected. Examples of parameters are reactor size (MW), fuel assembly position in the reactor (middle, peripheral), internal variations of fuel assembly (axial, radial), power levels (fixed, reduced, frequent ups/downs), fuel (burnable poison, advanced fuel), temperatures etc.

During the first step, the Working Group would collect information on typical methods and their capabilities. Also, an inventory of different parameters related to the reactor design and operation would be compiled. Spent fuel contents, multiplication factors and other parameters can be compared.

Step 2 - Criticality safety analysis

This is basically what the Working Group is doing now. One of the methods from Step 1 is used as a reference method for further studies of fuel operations outside the reactor (transport, storage). To make comparisons of cross sections possible, each member will use the atomic densities calculated with the reference method. The exception would be the most realistic cases; those including all fission products. They should be calculated, if possible, using the same methods as in Step 1.

A very important part of this proposal is to identify parameters that are essential for the evaluation of criticality safety when taking advantage of burnup. Is it possible to use just the average fuel burnup per assembly? What are the uncertainties? What could and should be verified?

EMS Proposal - Step 1 - Standard burnup calculations				
Step 1	Case a	Case b	Case c	Case ...
k_{eff}				
U-235				
...				
Axial profile				
Residual heat generation				

EMS Proposal - Step 2 - Criticality safety calculations				
Step 2	Case A	Case B	Case C	Case ...
k_{eff}				
Neutron production rates				
Neutron absorption rates				

Appendix 2.4 Comments of Kim Ekberg, Studsvik Core Analysis

Studsvik Core Analysis

1992-05-15
Page 1/15

From: Kim Ekberg

To: M Takano , Jaeri
E Sartori, NEA

BURN-UP CREDIT CRITICALITY BENCHMARK

Initially I would like to make some comments to the benchmark, as proposed. Doubtless the evaluation of criticality of spent fuel for storage and transportation is an important problem. In order to avoid having to make excessively conservative assumptions methods should be developed and verified, which allow the actual burn-up status of the fuel to be taken into account.

It is an advantage if simple methods can be employed. However, "simple" means different things to different people, and also other considerations are important. Many reactor operators already have a functioning away-from-reactor code system for core follow and reload and operations planning. They will probably find it preferable to use their existing code system for several reasons. Their staff are used to the system and know it. Further, the system already contains all relevant data of the burn-up status of their fuel. From a QA point of view it is important to minimize data transfer between code systems.

Several such away-from-reactor code systems exist. The one which is probably most spread among utilities and safety authorities around the world is the Studsvik Core Management System (CMS). The Studsvik-CMS already now is capable of handling most regular storage racks, and work is underway to develop a version which can handle more general rack and cask arrangements.

The fact that the physical model of a core management program can be characterized as more "complicated" than that of e.g. KENO is of no consequence to the user. What matters to the user is the ease of handling . Generally the handling of core management programs is much simpler than of Monte Carlo and similar programs. Especially is this true for the Studsvik CMS, and in addition the computer resource requirements are very modest. An added advantage is that Studsvik-CMS is already extensively benchmarked with respect to reactivity and isotopic content of the burnt fuel.

I feel that in calculations where you want to take credit for the burnup of spent fuel it will be necessary to have access to a data file with the burnup status of each fuel bundle. Such a file is one of the files being created and used within CMS (and also in other code systems for core follow and predictive calculations). However, as I do not know the plans for the continuation of the benchmark, I will not comment further at this time.

I will now go on to describe briefly how the benchmark probrem has been treated. The problem described in NEACRP-L-337 has been set up in CASMO-3, which is

/extra32/kim/fram/memo92/neacrp.13may

the fuel bundle (and, if you wish, pincell) code of the Studsvik-CMS. For the benchmark I have used the pincell option of the code.

The number densities given in the Appendix to NEACRP-L-337 **have not been used**. It is not meaningful to introduce into CASMO number densities from other sources, and therefore number densities at all exposures have been calculated with CASMO. For number densities after 1 and 5 years of decay an option of CASMO, Shut Down Cooling, has been used.

To evaluate the reactivity value of fission products the number densities of all FPs in CASMO have been put equal to zero. This method has been chosen because not all FPs specified in NEACRP-L-337 are explicitely represented in CASMO. Those missing are Mo-95, Tc-99 and Ru-101. On the other hand CASMO represents explicitly several FPs which are not specified in NEACRP-L-337. In CASMO all fission products, which are not separately represented, are lumped together in two lumped fission products, one non-saturating (LFP 1) and one slowly saturating (LFP 2). Putting the number densities of all FPs equal to zero thus means to eliminate all FP absorption in the calculation, and this method has been preferred when the specifications could not be fully adhered to.

The reaction rates given in the following are normalized to 1.0000 fission neutrons.

The CASMO run reported here did take 4 min 46 sec on a SUN Sparc IPC Work Station.

A summary of the results is given in the Appendix. Unfortunately I did not have the time to prepare all the detailed results requested in the benchmark specification. The tables given are extracts from the standard CASMO output list, and transferring them from the Fortran output list has required some editing.

To check the validity of the simplification of running the calculation as a pincell case I also ran a full bundle calculation. The results agree well between the pincell and the bundle case, which is to be expected for a normal PWR bundle.

The agreement with the ORIGEN results is also fairly good. One can see a tendency that CASMO gives 8-10 % lower fissile Pu content at 30 and 40 MWd/kgU. Taking into account that about half the fissions take place in Pu at 40 MWd/kgU this is probably the most significant difference. Isotopics at higher burn-ups are strongly spectrum dependent, and the cross section sets provided in ORIGEN may not always represent the neutron spectrum during burnup adequately.

Generally a good deal of confidence should be placed in the CASMO results. This code is used worldwide by more than 50 utilities, safety authorities and others in the nuclear industry. It is extremely well benchmarked against critical experiments and operating reactor data. The reactivity level for normal reactor lattices and normal storage rack lattices can be expected to differ from one at criticality by at most a few tenths of a per cent.

Appendix 2.5 Report from BNFL

"BNFL Criticality Calculations on OECD/NEACRP
Burn-up Credit Benchmark, Phase 1"
(The data listings of the original report have been omitted.
The omitted pages are 3, 4 and 12 to 48.)

BNFL	TITLE PAGE	Project Number CR921 Page 1 of 48

SECURITY CLASSIFICATION: None

Project Title: Methods Development

DOCUMENT TITLE :
BNFL CRITICALITY CALCULATIONS ON OECD/NEACRP BURN-UP CREDIT BENCHMARK, PHASE 1.

Author: P E Broome

Location: R102, Risley Ext: 3087

Reference: CR921 Rev: 0

Date: June 1992

Approved By: J M Hobson

2					
1					
Orig	PEB	Jun 92	LMF/PRT	JMH	Original document
	By	Date	Chkd.	Appr.	Description of Issue or Revision

BNFL 5896A (Rev 11/84)

BNFL	Title: BNFL CRITICALITY CALCULATIONS ON OECD/NEACRP BURN-UP CREDIT BENCHMARK, PHASE 1.	Project Number CR921 Page 2 of 48

Classification None

Author P E Broome

Department Safety and Nuclear Technology

Approved by J M Hobson

Report No.	Project	Building	File No.
CR921	Methods Development	-	CRF3008/3
Date	**Revision**	**Issue**	**Pages**
June 1992	0	1	48

Keywords
MONK6B, burn-up, irradiated fuel, fission products

Summary
The criticality working group of the Nuclear Energy Agency Committee on Reactor Physics (NEACRP) has proposed an exercise to investigate the calculational capability of evaluating burn-up credit in criticality calculations.
Phase 1 of the study is intended to establish common ground between the participating groups by defining a relatively simple system for evaluation. The model consists of a simple PWR fuel cell lattice of infinite extent with fuel material which varies in irradiation history and isotopic inventory between the cases studied.
BNFL Risley's contribution to the study is reported here and was performed using MONK6B, with a specially modified version of the point nuclear database which incorporates fission product data taken from the latest JEF2 evaluations.

2					
1					
Orig	PEB	Jun 92	LMF/PRT	JMH	Original document
	By	Date	Chkd.	Appr.	Description of Issue or Revision

BNFL 5896 A/1

BNFL	Title: BNFL CRITICALITY CALCULATIONS ON OECD/NEACRP BURN-UP CREDIT BENCHMARK, PHASE 1.	Project Number CR921 Page 5 of 48

Contents

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1					
Orig	PEB	Jun 92	LMF/PRT	JMH	Original document
	By	Date	Chkd.	Appr.	Description of Issue or Revision

BNFL 5896D (Rev 11/84)

1 INTRODUCTION

In many countries, the criticality evaluation of spent fuel transportation, storage and reprocessing has historically, for the sake of pessimism and ease of calculation, assumed the presence of fresh rather than spent fuel. The increasing pressure on costs and efficiency and the tendency towards higher and higher initial enrichments, however, is compelling the industry to consider more seriously the adoption of burned-up fuel in the calculational assumptions. This clearly could dramatically increase existing safety margins with, generally, an accompanying increase in plant efficiency and decrease in operational costs. As with all matters such as this, a balance needs to be sought between the two requirements of reducing costs and optimal safety.

As part of its continuing study into the various computational techniques of its member groups, the criticality working group of the Nuclear Energy Agency Committee on Reactor Physics (NEACRP, but now superceded by the NEA Nuclear Science Committee, NEANSC) has proposed an international burn-up credit benchmark (Reference 1).

This document formally reports the results of BNFL's Nuclear Technology Group (Risley) assessment of the criticality benchmark Phase 1 exercise.

2 DESCRIPTION OF PHASE 1 EXERCISE

As indicated in the preliminary instructions to participants (Reference 2), the first phase of an international benchmark should not be too ambitious if it is to be successful. The first phase is intended to 'tune [the participant's] working tools so that inconsistencies do not creep in later or are propagated further into the more complex phases' (Reference 2).

With this aim in mind, a simple PWR fuel cell lattice, infinite in extent, was chosen as the system to be studied. A notional initial enrichment of 3.6w/o U-235 is used. The actual geometrical model is shown in Figure 1 and consists of a UO_2 fuel pellet region (radius 0.412cm) surrounded by zircalloy clad (radius 0.475cm) with no intervening air gap. An infinite square array of these fuel pins at a pitch of 1.33cm is then modelled by surrounding these annulii with a square of light water (side 1.33cm) and using a reflective

boundary condition. The model is also made infinite in the axial direction by use of the same boundary condition.

The above geometrical model is used in a total of 9 cases involving differing degrees of burn-up, cooling time and fission product inventory. A complete specification of the cases to be studied is given in Table 1, which has been reproduced from Reference 2.

Fuel isotopic number densities for each case were provided (Reference 2) for 12 actinides, 15 fission products and oxygen. These are reproduced here in Table 2. The number densities for the remaining materials (zircalloy and water) were taken from Reference 8.

In recording the results of the exercise, a number of data items, in addition to the calculated k-infinity value, were requested by the benchmark organisers. These were to be presented in a specified fixed format as shown in Table 3. Among other things the required data consisted of neutron spectra, absorption and production rates and estimates of the number of neutrons per fission. These were defined formally in Reference 2 as:-

$$\text{Absorption reaction rate,} \qquad A_i = \frac{\iint \Sigma_a^i \phi \, dE \, dV}{V}$$

where

$$\sum_{i=1}^{\text{all}} A_i \equiv 1$$

$$\text{Production reaction rate,} \qquad P_i = \frac{\iint \nu^i \Sigma_f^i \phi \, dE \, dV}{V}$$

where

$$\sum_{i=1}^{\text{all}} P_i \equiv 1$$

$$\text{Neutrons per fission,} \qquad F_i = \frac{\iint \nu^i \phi \, dE \, dV}{V}$$

where

$$\sum_{i=1}^{\text{all}} F_i \equiv 1$$

i = nuclide, E = energy, V = volume

3 CALCULATIONAL DETAILS

The UK Monte Carlo criticality code MONK6B (Reference 3) was used for the calculations. Since the current point energy nuclear database does not contain data for a number of the required fission products, a special version of the database was created (Reference 4) with new data being incorporated from the latest JEF-2 evaluations (Reference 5).

Using the geometrical model and isotopic number densities identified in Section 2, MONK models of the 9 required cases were constructed. To ease the analysis of the results, the default 3-group energy partition used in the output was over-ridden to produce action counts in a single group covering the full energy range (0 - 15MeV).

A target standard deviation on the eigenvalue of 0.002 was set and each case was run 3 times in order to further reduce the statistical uncertainty. The calculations were performed on a Research Machines 486/20 SystemBase 25 operating under SCO UNIX System 5 Release 3.2.2.

A sample MONK6B input listing is given in Figure 2 and the remaining listings can be found in Reference 7.

4 RESULTS AND ANALYSIS

The k-infinity values for each run with the corresponding mean values for each case are given in Table 4.

To obtain the absorption and production reaction rate data the following procedure was adopted. The material action counts for material 1 (the fuel) were extracted from the 3 outputs corresponding to each case and loaded into a spreadsheet (Reference 6). For the present exercise the action counts of particular interest were CAPTURE, FISSION and

FISSION CHILDREN. It is noted that in MONK these parameters are given normalised to 10,000 source neutrons.

An average value for the action counts was first obtained by taking the mean of the 3 runs performed for each case. Using the definition Absorption = Capture + Fission, the ABSORPTION 'action counts' were calculated. Summing these over all nuclides gives a total which can be used to normalise the individual reaction rates to unity as requested in the specification. The production reaction rates were obtained by doing a similar process for the FISSION action counts.

The formal definition for the number of neutrons per fission as requested is given in Section 3. This definition, however, was considered to be suspect in that it required a normalisation involving the neutron flux which seemed inappropriate. The requirement to normalise these values to unity also seemed to be a strange request in that it would actually mask the information on the average number of neutrons per fission for a particular nuclide, which is what is presumably desired. Following discussions with the benchmark organisers, therefore, it was decided to leave this parameter unnormalised and so as such this strictly represents a departure from the originally requested specification. Practically, this data item is obtained by forming the ratio FISSION CHILDREN / FISSION for each (fissile or fissionable) nuclide.

The data thus obtained for each of the 9 cases is shown in Tables 5 - 13. These individual spreadsheet files were then loaded into a 'master' file in the format required by the benchmark specification, and this is reproduced here as Table 14.

5 DISCUSSION OF RESULTS

The first obvious trend to be noticed is that the calculated k-infinity decreases with increasing burn-up in each instance (see Table 4). By comparison of Cases 4 with 5 and 8 with 9, where the fission products have been omitted, it is apparent that the reduction of about 0.06 in k-inf is due to the burning-up of U-235 which is not compensated for by the build-up of other fissile nuclides (e.g. Pu-239).

The inclusion of the fission products in these cases (compare Cases 2 with 3 and 6 with 7) brings about a further reduction in k-inf of about 0.02 when the burn-up is increased from 30 to 40GWd/te. This appears to be due to the build-up of fission products in general rather than the accumulation of any nuclide in particular.

When considering the effects of increased cooling time, it is noticed that even when the fission products are omitted (compare Cases 4 with 8 and 5 with 9) there is a drop in reactivity of about 0.02 for the longer cooling time. Examination of the reaction rate data for these cases indicates that this can be attributed principally to the increased build-up of Am-241, which acts as a neutron absorber.

When the fission products are included in these cases (compare Cases 2 with 6 and 3 with 7) then k-inf is seen to drop by about 0.03 to 0.04 overall. Again, detailed examination of the reaction rate data indicates that the probable principal contributors to the difference are a reduction in the Pu-241 absorption and production rates (about 1%), increased Am-241 absorption (about 0.7%) and the build-up over time of Gd-155 (about 1% increase in absorptions).

6 CONCLUSIONS

The criticality working group of the Nuclear Energy Agency Committee on Reactor Physics (NEACRP) has proposed a benchmark exercise to investigate the calculational capability of evaluating burn-up credit in criticality calculations.

Phase 1 of the study consists of the evaluation of a simple PWR fuel cell lattice of infinite extent, with fuel material which varies in irradiation history and isotopic inventory between the cases studied.

BNFL Risley's contribution to this initial study has been reported here. It was performed using MONK6B, with a specially modified version of the point nuclear database which incorporates fission product data taken from the latest JEF-2 evaluations.

Oct 13, 1993

Revised Results for Burnup-Credit Criticality Benchmark Part 1 Using SCALE 4.2 and 44 Group ENDF/B-V Cross-Sections

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Introduction

Results submitted by the Oak Ridge National Laboratory (ORNL) for Phase 1A of the NEACRP Burnup-Credit Criticality Benchmark report were obtained using SCALE 4.1 and a 27-group library of ENDF/B-IV cross-sections. (Note: most fission product cross-sections in the 27-group library were obtained from pre-release ENDF/B-V data.) Revised calculations have been completed using version 4.2 of SCALE combined with a new 44-group ENDF/B-V based cross-section library. The new 44-group library has been specifically developed for the use in criticality analysis of LWR-type spent fuel assemblies.

A developmental DECstation version of SCALE 4.2 was used in these calculations; however, subsequent recalculation of Case 7 specifications using a configuration-controlled version of the production version of SCALE 4.2 on an IBM RISCstation showed no change in the computed value for k_{eff}.

Results using SCALE 4.2 and the new cross-section library show marked improvement over those obtained in the earlier set of calculations. The following section describes the new ENDF/B-V based results relative to the earlier ENDF/B-IV results and the Group C (16 participant) average results reported in the 4/30/93 draft of the NEACRP Burnup-Credit Criticality Benchmark report.

Results

Results of the revised calculations have been assembled in the format requested for benchmark results, and are included as an attachment. Values of k_{eff} computed for each of the 13 variations of the benchmark are given in Table 1. The average k_{eff} column in the table refers to the 16-Participant average value, the "Previous..." column provides the previously reported SCALE 4.1/27-group ENDF/B-IV value of k_{eff}, and the "Revised..." column gives k_{eff} based on calculations using SCALE 4.2 and the new 44-group ENDF/B-V cross-sections. Note that in all cases the revised k_{eff} is closer to

*Managed by Martin Marietta Energy Systems, Inc., under contract DE-AC05-84OR21400 with the U. S. Deparment of Energy.

the average than the previous k_{eff}. Previous k_{eff}'s were approximately 0.3 to 0.7% low, while the revised values were approximately 0.0 to 0.3% low. Also, note that revised Case 13 results contain the correction to an error discovered in the original calculation.

Table 1 Comparison of Results Previous and Revised Calculations

Case Number	Average k_{eff}	Previous ENDF/B-IV k_{eff} / (% difference)	Revised ENDF/B-V k_{eff} / (% difference)
1	1.4378	1.4276 (-0.71%)	1.4369 (-0.06%)
2	1.1404	1.1323 (-0.71%)	1.1366 (-0.33%)
3	1.0640	1.0570 (-0.66%)	1.0604 (-0.34%)
4	1.2457	1.2397 (-0.48%)	1.2436 (-0.17%)
5	1.1887	1.1836 (-0.43%)	1.1859 (-0.24%)
6	1.1125	1.1051 (-0.67%)	1.1093 (-0.29%)
7	1.0242	1.0180 (-0.61%)	1.0209 (-0.32%)
8	1.2284	1.2228 (-0.46%)	1.2273 (-0.09%)
9	1.1658	1.1604 (-0.46%)	1.1634 (-0.21%)
10	1.2619	1.2576 (-0.34%)	1.2619 (0.00%)
11	1.2550	1.2505 (-0.36%)	1.2552 (0.02%)
12	1.1104	1.1026 (-0.70%)	1.1088 (-0.14%)
13	1.0767	1.0809* (0.39%)*	1.0753 (-0.13%)

* This case inadvertently omitted ^{103}Rh. The correct 27-group ENDF-4 k_{eff} value is 1.0679 (-0.82%).

Figures 1-6 show the flux spectrums computed using both the 27-group and 44-group libraries for Cases 1, 7, and 9, in both the fuel and reflector regions. These figures show good agreement between the previous and revised results. Note that the 44-group spectrum varies from the 27-group spectrum at both the far upper and far lower energy regions. The differences are most likely due to the increased number of energy groups in the 44-group library, but are likely to have little overall effect. However, significant differences are noted in two key regions. First, the 44-group spectrum is depressed in the 6-8 eV range, while no such depression is seen in the 27-group spectrum. This depression is most notable in the fuel, and occurs in both fresh and spent fuel cases; it is therefore most likely to be the result of absorption in the lowest ^{238}U resonance, which occurs in this range. This difference may result from either the refined 44-group structure or improvements to the resonance processing capabilities of SCALE-4.2. The second difference between the two sets of calculations is a dip in the 27-group spectrum around 1 eV. Again, this dip is seen primarily in the fuel, but in this case occurs only in spent fuel.

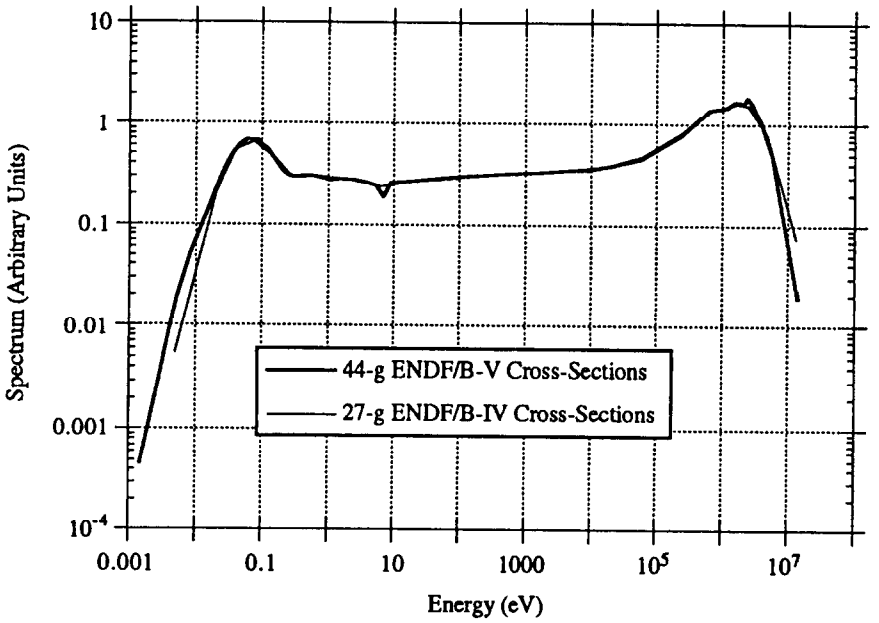

Fig. 1 Case 1 Flux in Fuel (Fresh Fuel)

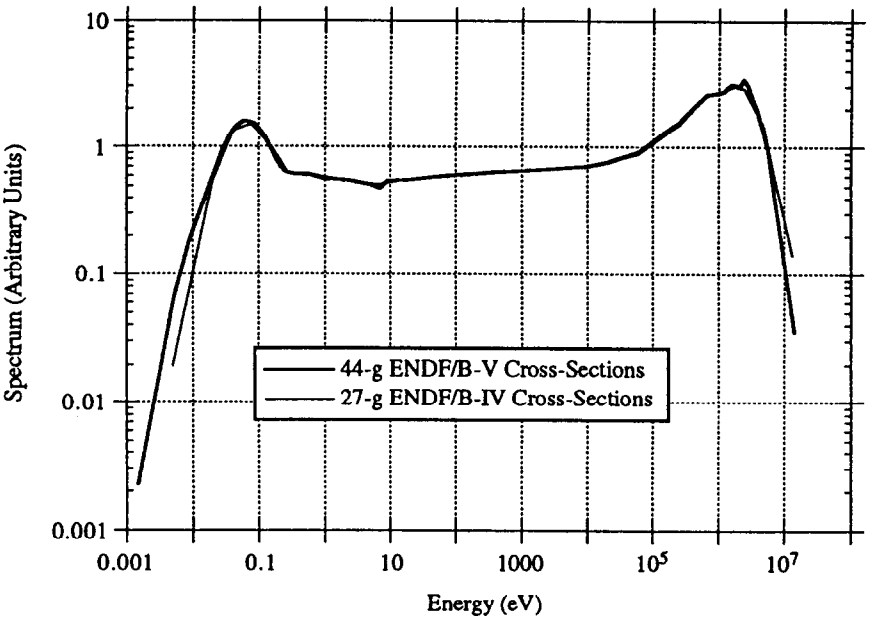

Fig. 2 Case 1 Flux in Reflector (Fresh Fuel)

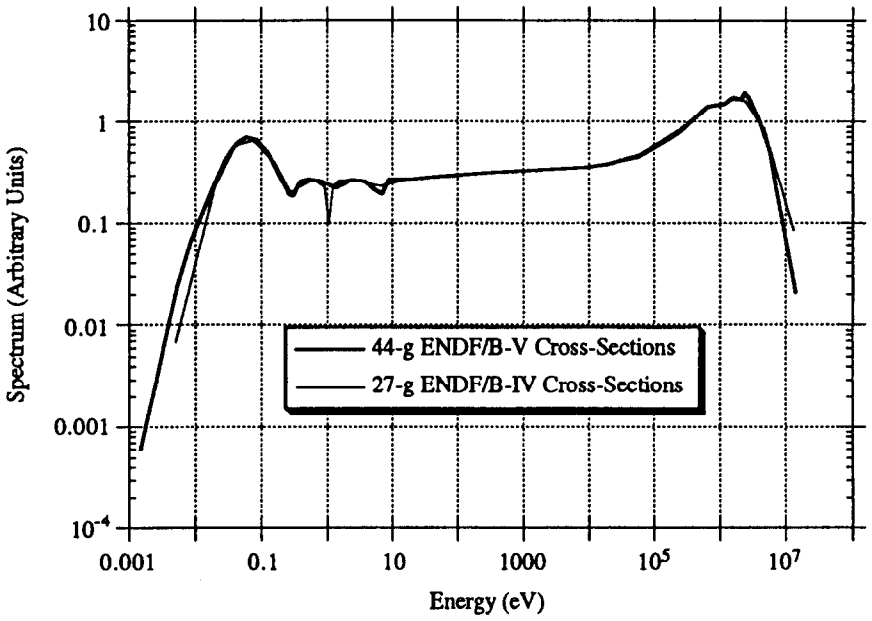

Fig. 3 Case 7 Flux in Fuel
(40 GWD/t, 5y, Major FPs)

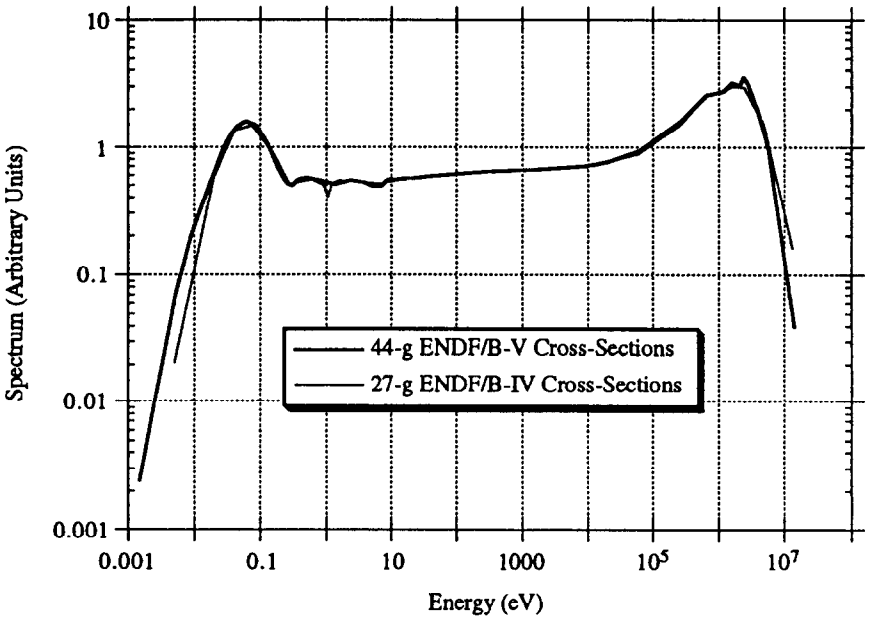

Fig. 4 Case 7 Flux in Reflector
(40 GWD/t, 5y, Major FPs)

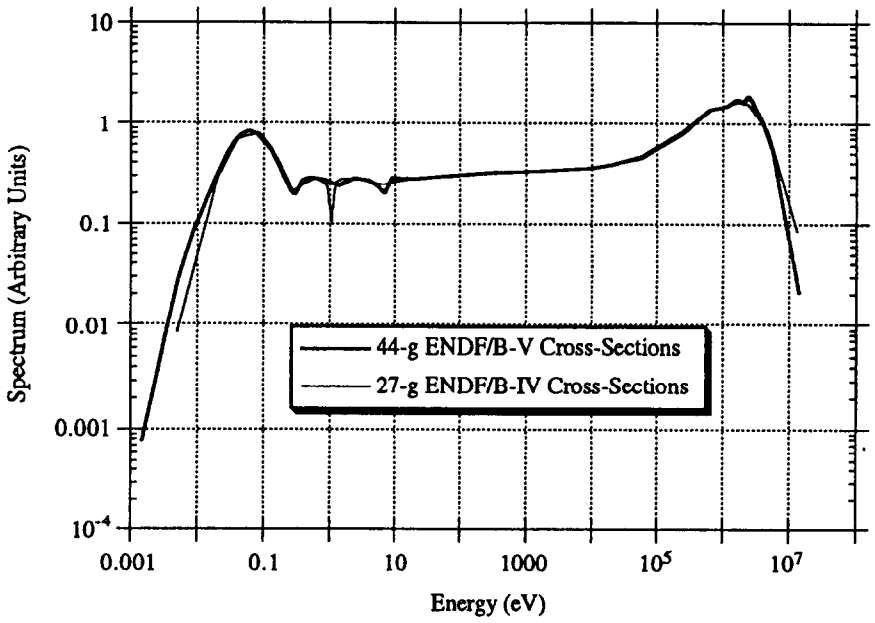

Fig. 5 Case 9 Flux in Fuel
(40 GWD/t, 5y, No FPs)

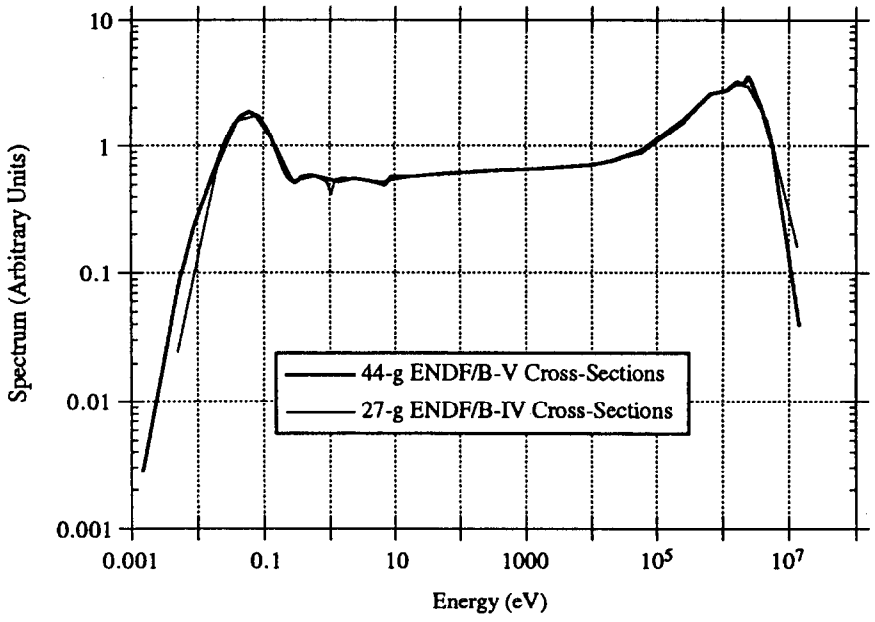

Fig. 6 Case 9 Flux in Reflector
(40 GWD/t, 5y, No FPs)

The depression is present in both 44-group and 27-group results, although it is significantly more pronounced in the 27-group spectrum. This anomaly is most likely due to the 1 eV resonance in ^{241}Pu absorption, which appears to be overestimated in the 27-group calculation.

Figures 7 and 8 show the relative absorption rates for all of the selected fission products for both the previous and revised results. Figure 7 compares results from the 27-group ENDF/B-IV and the 44-group ENDF/B-V based absorption rates after a 1 year cooling period following 40 GWD/t burnup. Figure 8 shows the same information after a 5 year cooling period. Absorption and production rates in the actinides were found to be relatively close for the two sets of calculations, and are therefore not shown here. As illustrated in the two charts, the largest differences among the fission products are in the isotopes of samarium and gadolinium. These differences are small relative to the magnitude of the absorption rate. Such small changes are expected since the fission product cross-sections the fission product cross-section data was changed little, if any, between the pre-release ENDF/B-V data in the 27-group library and the ENDF/B-V data in the 44-group library. The differences are most likely due to changes in actinide data (which can alter both fission product production rates and the energy spectrum), and changes in the flux spectrum due to the additional energy groups.

Note that in Figure 5.18 of the draft Burnup-Credit report, ORNL's previously computed ^{155}Gd absorption rate is significantly less than that of other participants. While this isotope shows little relative change in Figures 7 and 8, the magnitude of the change is such that ORNL absorption rates are in significantly better agreement with the results of other participants than was found using previous results.

Discussion and Conclusions

Improvements in cross-section processing portions of SCALE-4 together with a spectrum-tailored 44-level broad-group cross-section library based on ENDF/B-V data have improved ORNL's accuracy for burnup-credit type calculations. This is evidenced by the closer agreement of ORNL results with the average of results obtained by other benchmark participants. For all but a single case, the ORNL-computed k_{eff} values in the previously submitted results were lower than the 16-participant average. Case 13, the single exception, was found to be an erroneous result, due to the inadvertant omission of the fission product ^{103}Rh; recalculation with the addition of this isotope brought this computed k_{eff} down below the average. The revised calculations gave increased values of k_{eff} by roughly 0.3% for all cases, moving closer to the multiparticipant averages.

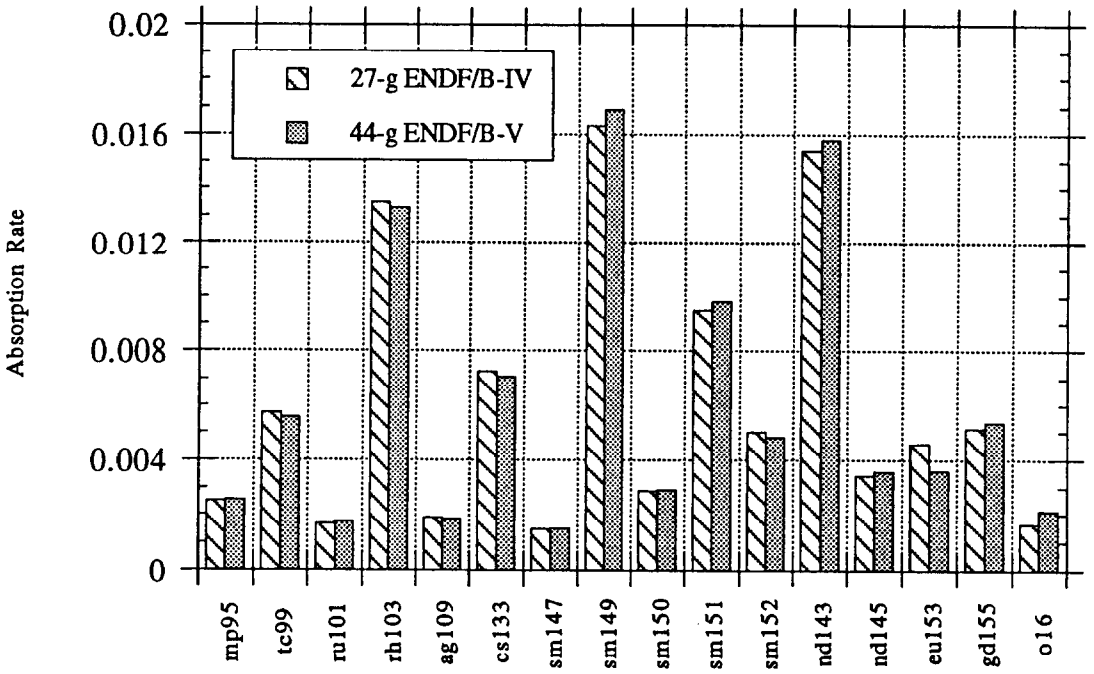

Fig. 7 Absorption in FPs, Case 3
(40 GWD/t, 1y Cooling Time, Major FPs)

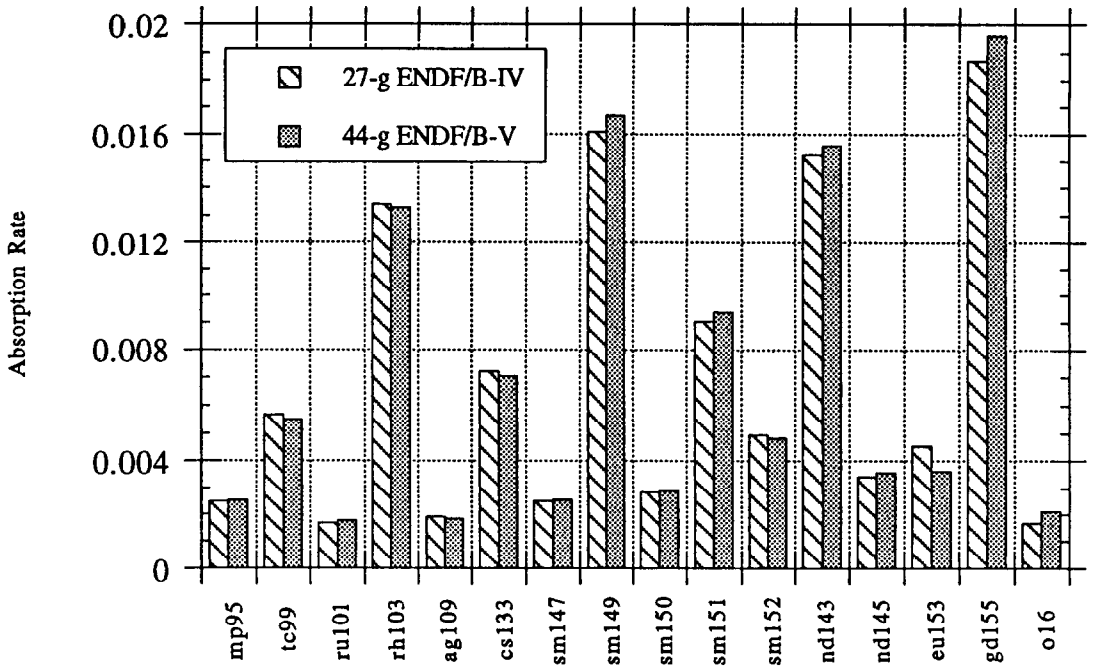

Fig. 8 Absorption in FPs, Case 7
(40 GWD/t, 5y Cooling Time, Major FPs)

k_{eff}, reaction rates, and ν-values for cases 1 to 9, along with the energy spectrum for the 44-group library, are provided in benchmark format in the attachment (Table 1 of this report provides revised values of k_{eff} for cases 1 to 13). Note that the production and absorption rates are normalized to unity as requested for the benchmark results; this normalization was not applied in the previous submission.

Finally, one comment should be made on the presentation of data in this draft of the benchmark report. In evaluating the computed reaction rates relative to those of the other participants and to the 16-particant average, it was found difficult to make a direct comparison due to the lack of numerical values for these rates. The bar charts of Figures 13 - 64 illustrate individual variations relative to the collected average, but nowhere is the average explicitly specified. Furthermore, it is not possible to assess these variations relative to the magnitude of the average, to obtain the percentage variation in reaction rates for the various isotopes. It might be useful to include the average value of the reaction rates for each of the actinides and fission products, either as a table, or by noting the value of the average on each of the isotopic comparison bar charts.

Appendix 3 Comparison of Reaction Rate

Appendix 3.1 Absorption Rate

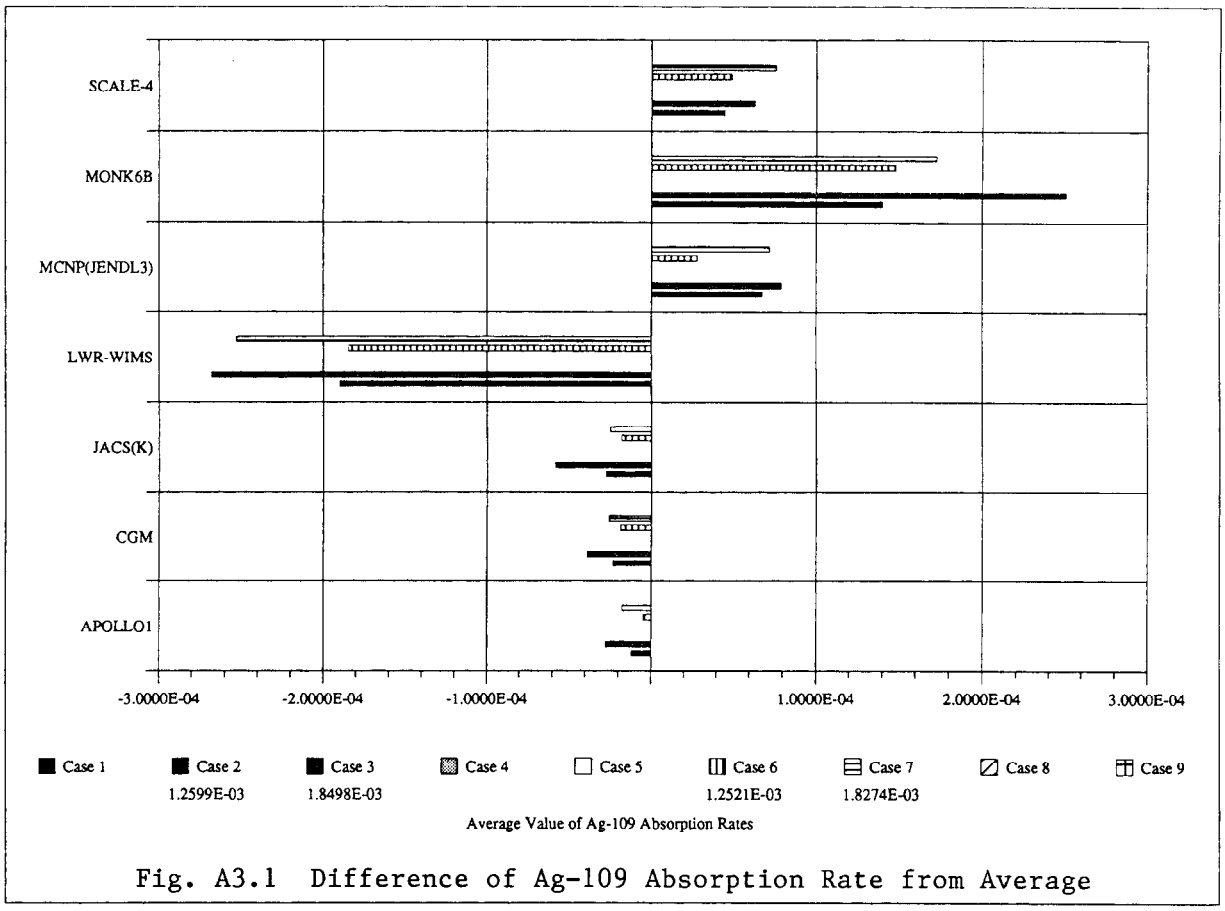

Fig. A3.1 Difference of Ag-109 Absorption Rate from Average

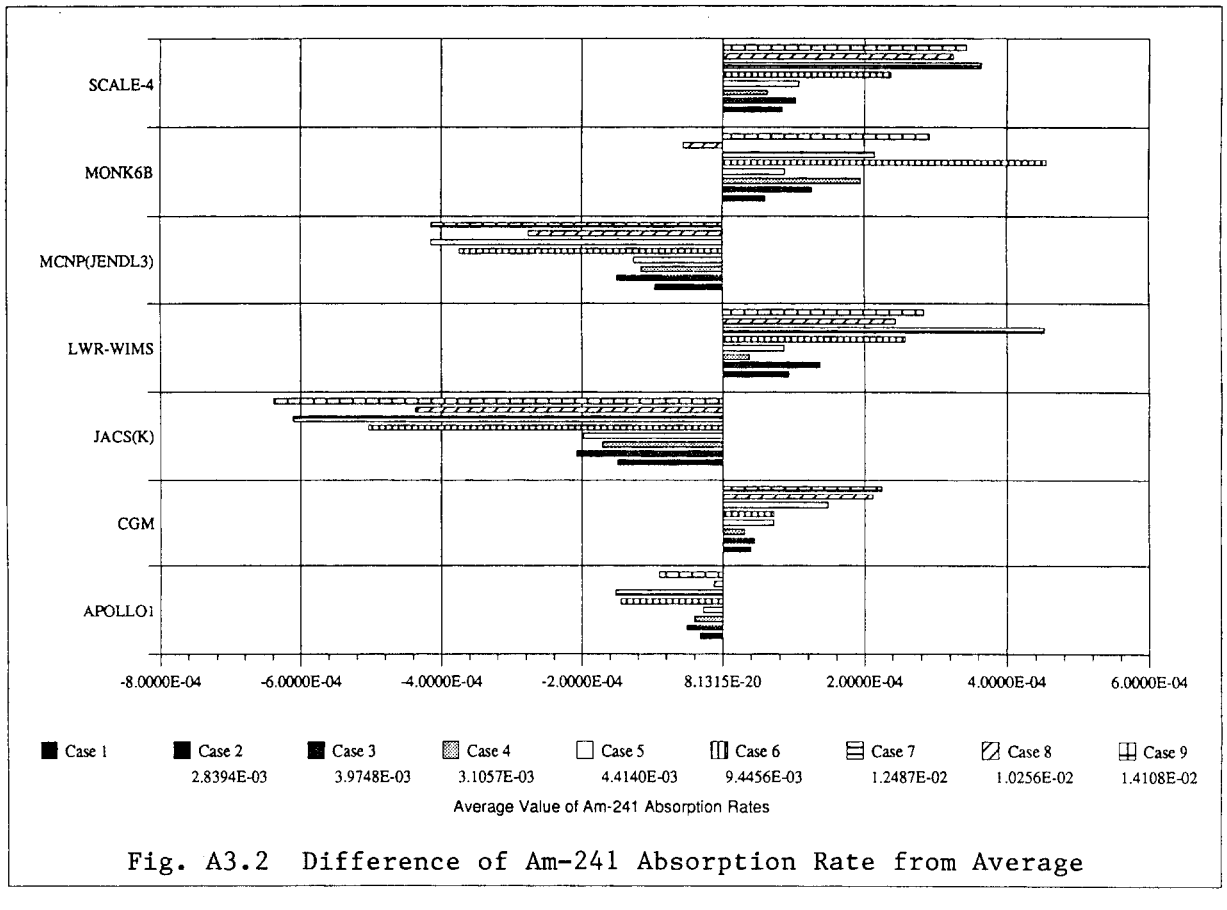

Fig. A3.2 Difference of Am-241 Absorption Rate from Average

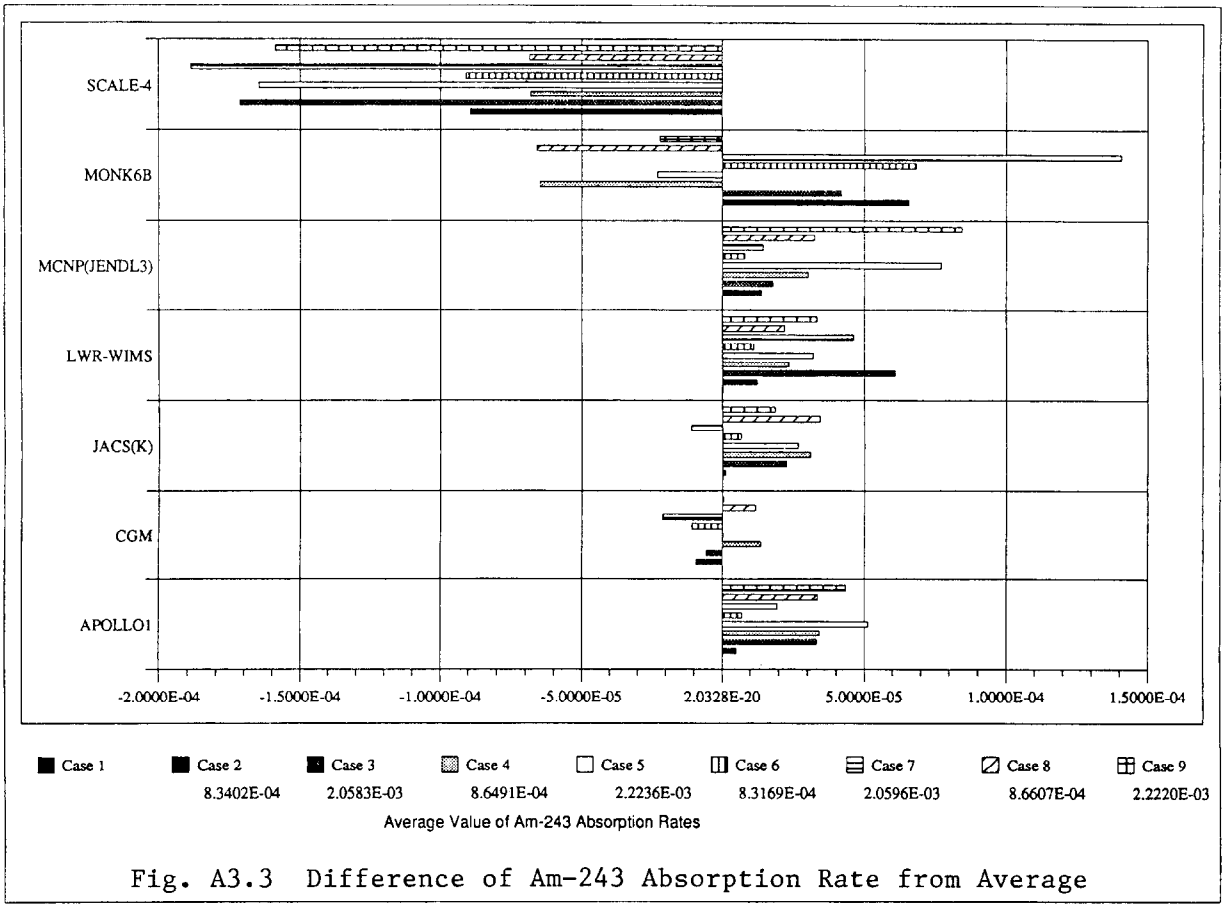

Fig. A3.3 Difference of Am-243 Absorption Rate from Average

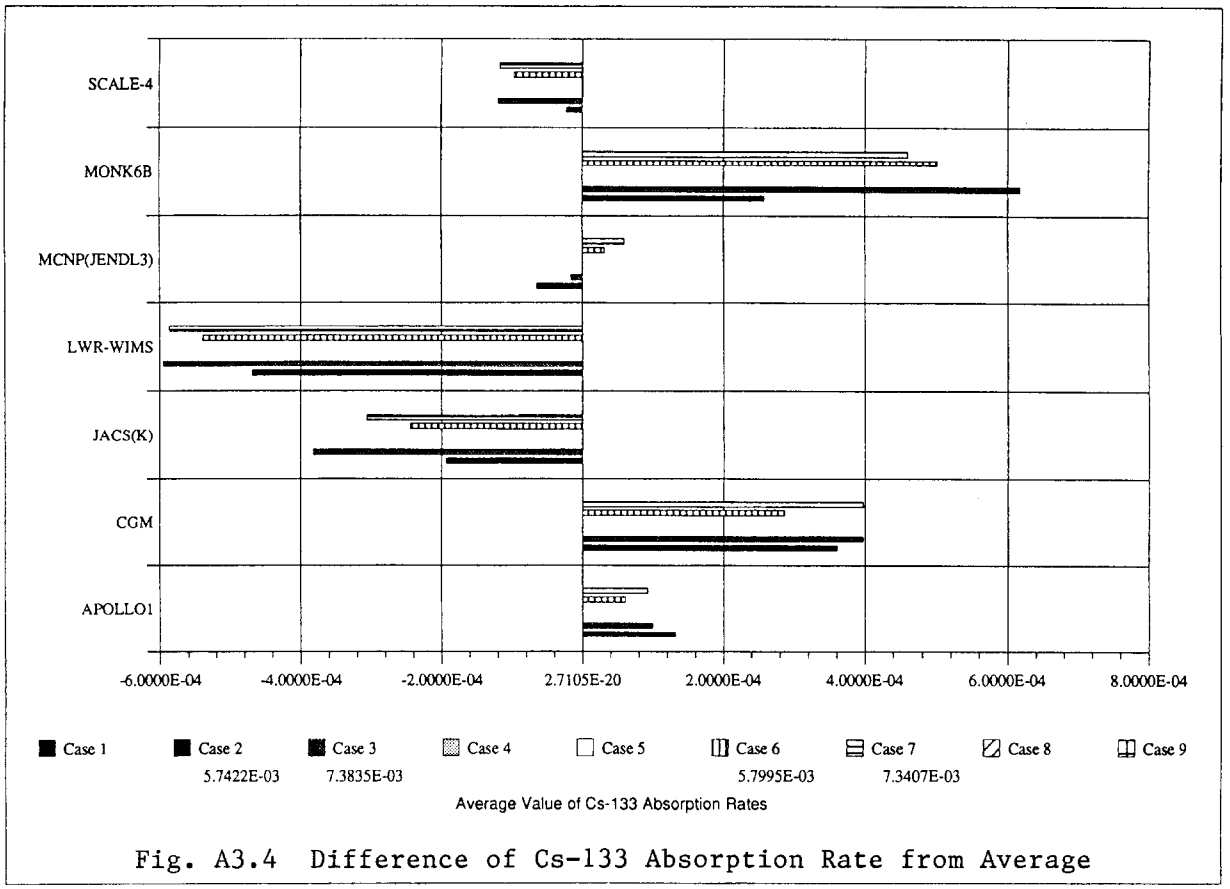

Fig. A3.4 Difference of Cs-133 Absorption Rate from Average

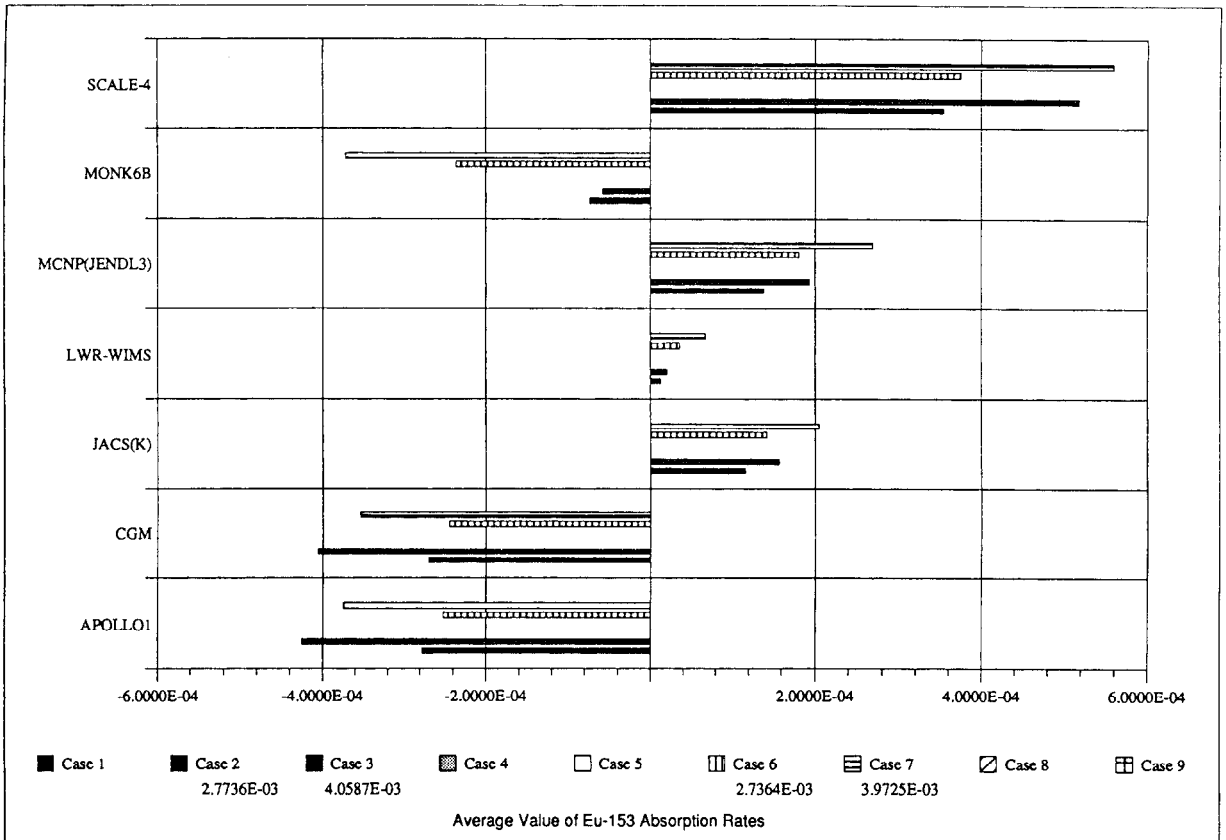

Fig. A3.5 Difference of Eu-153 Absorption Rate from Average

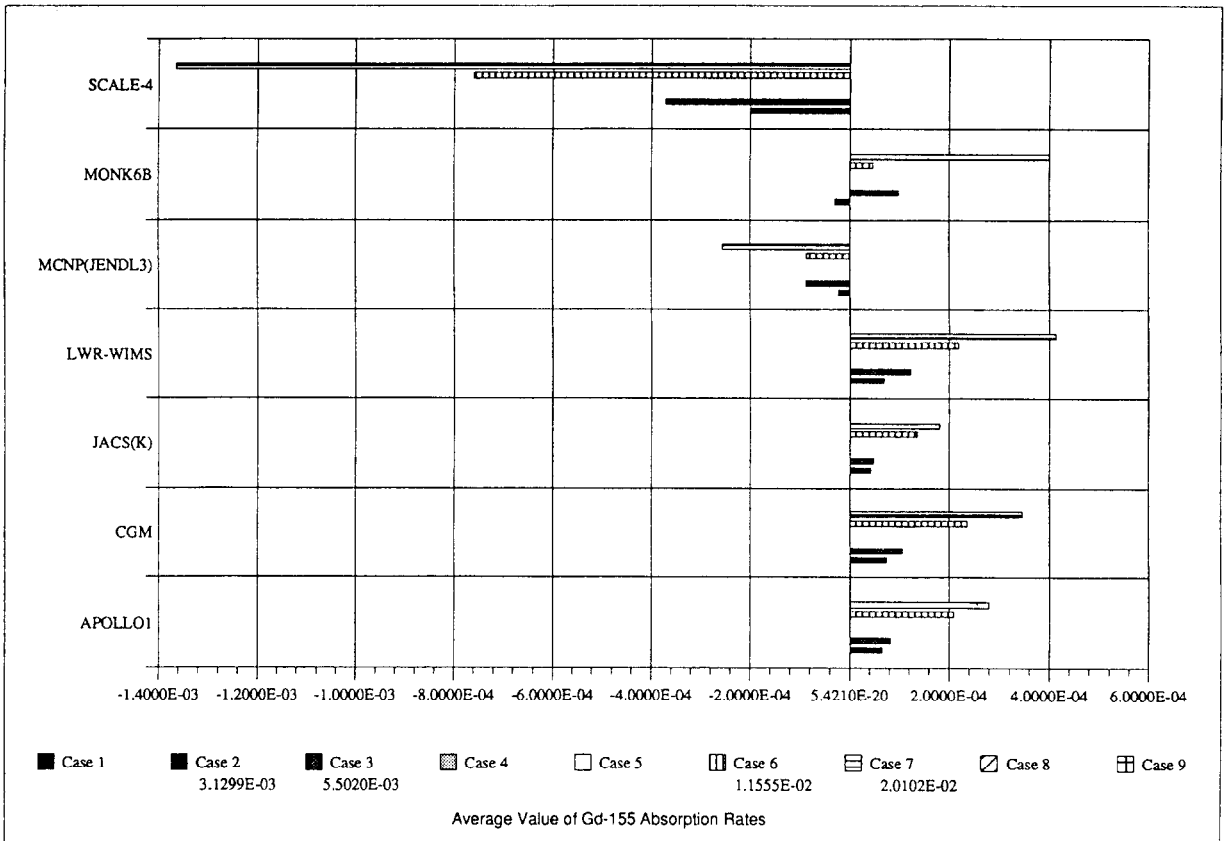

Fig. A3.6 Difference of Gd-155 Absorption Rate from Average

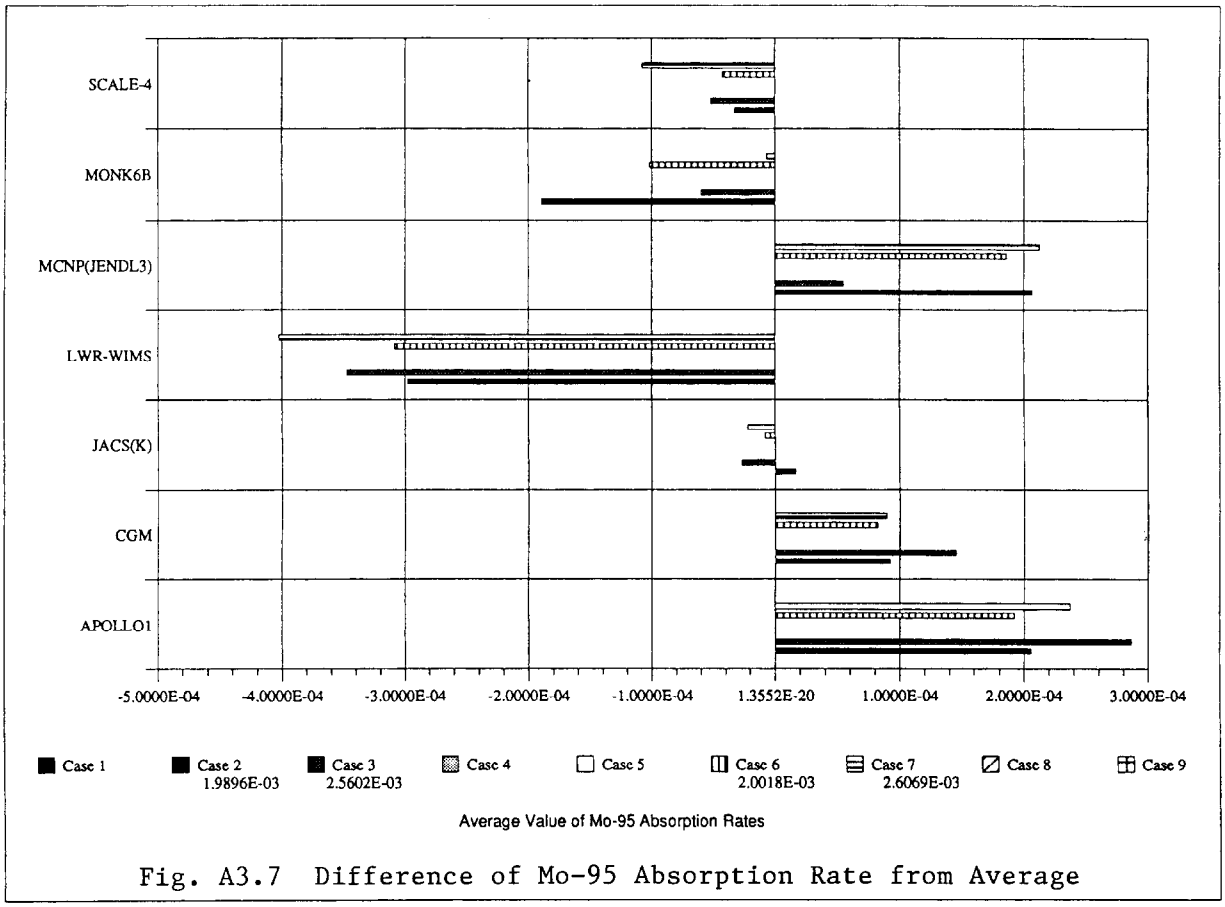

Fig. A3.7 Difference of Mo-95 Absorption Rate from Average

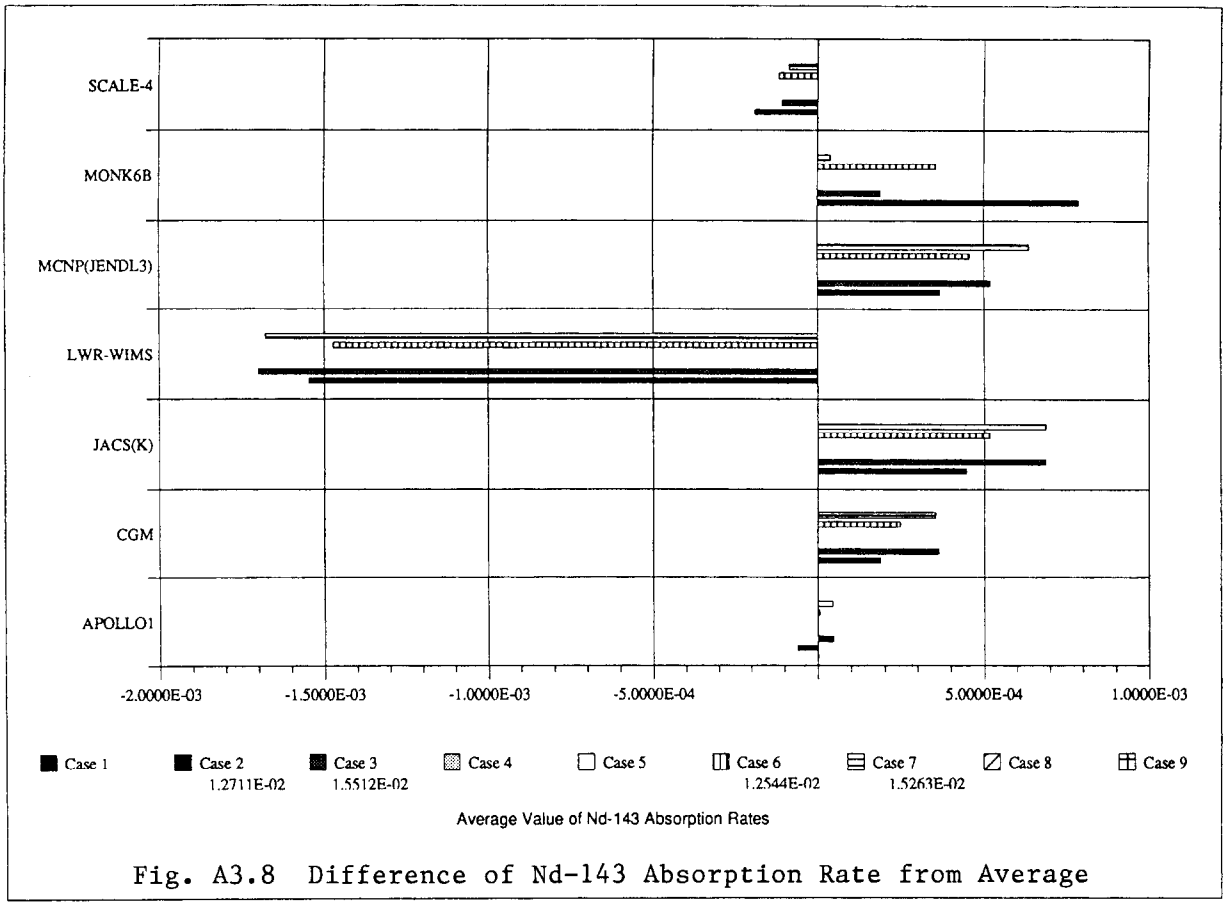

Fig. A3.8 Difference of Nd-143 Absorption Rate from Average

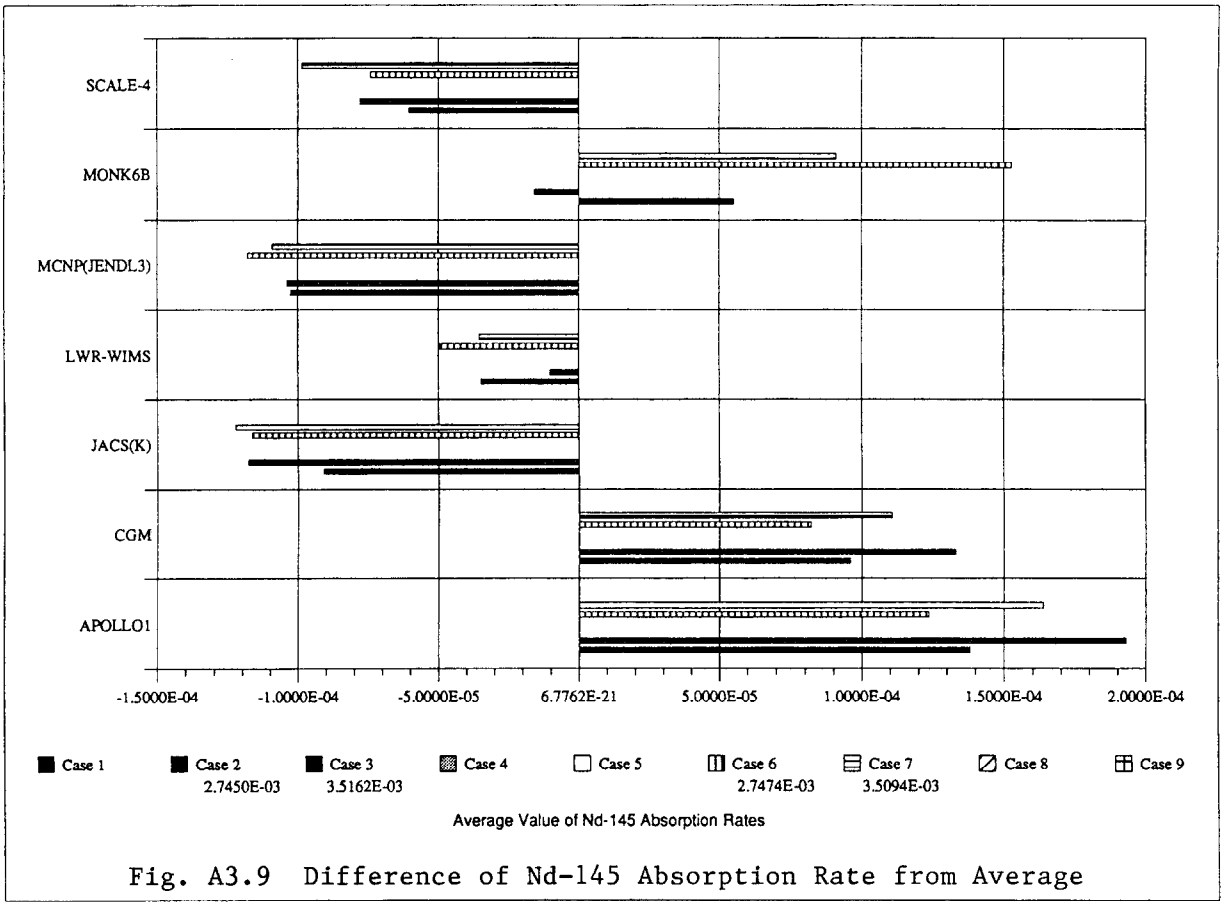

Fig. A3.9 Difference of Nd-145 Absorption Rate from Average

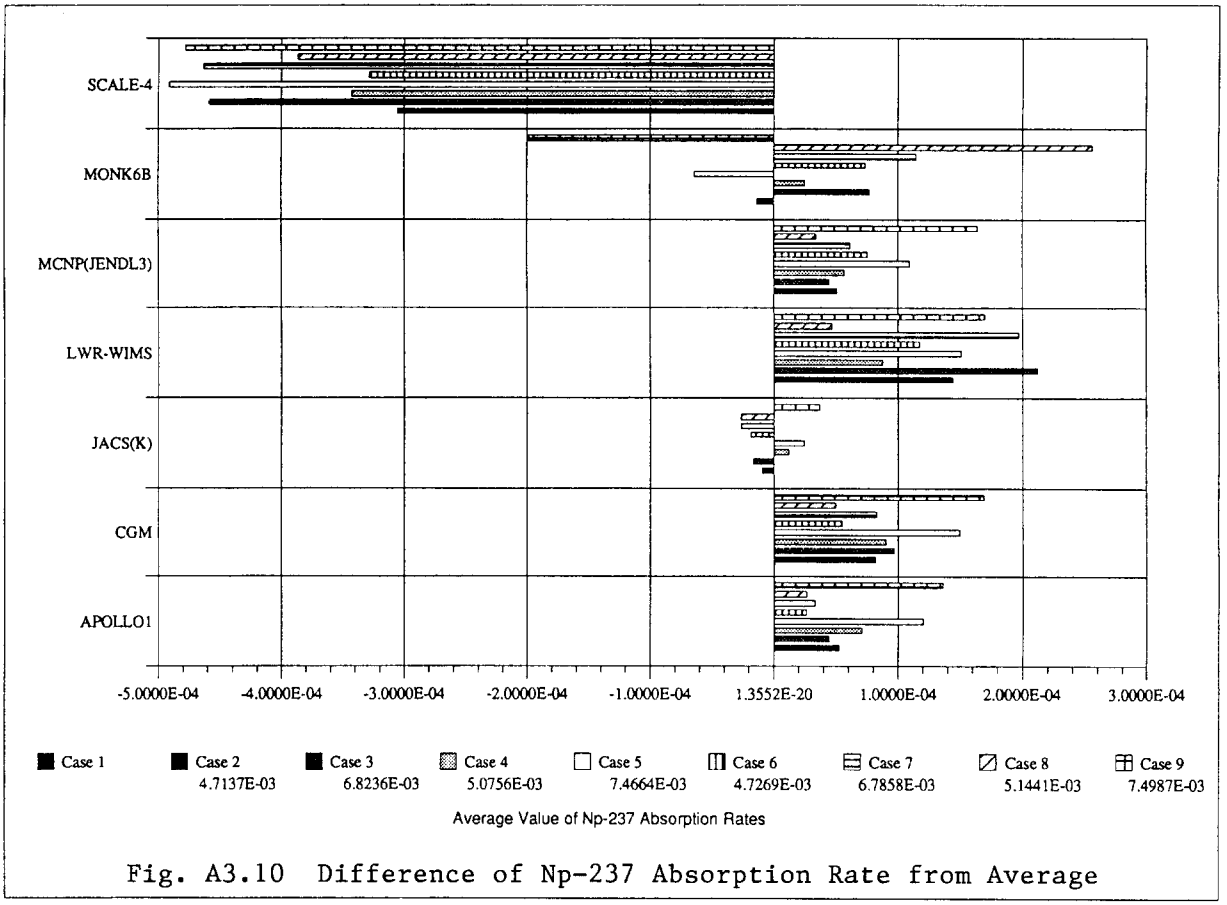

Fig. A3.10 Difference of Np-237 Absorption Rate from Average

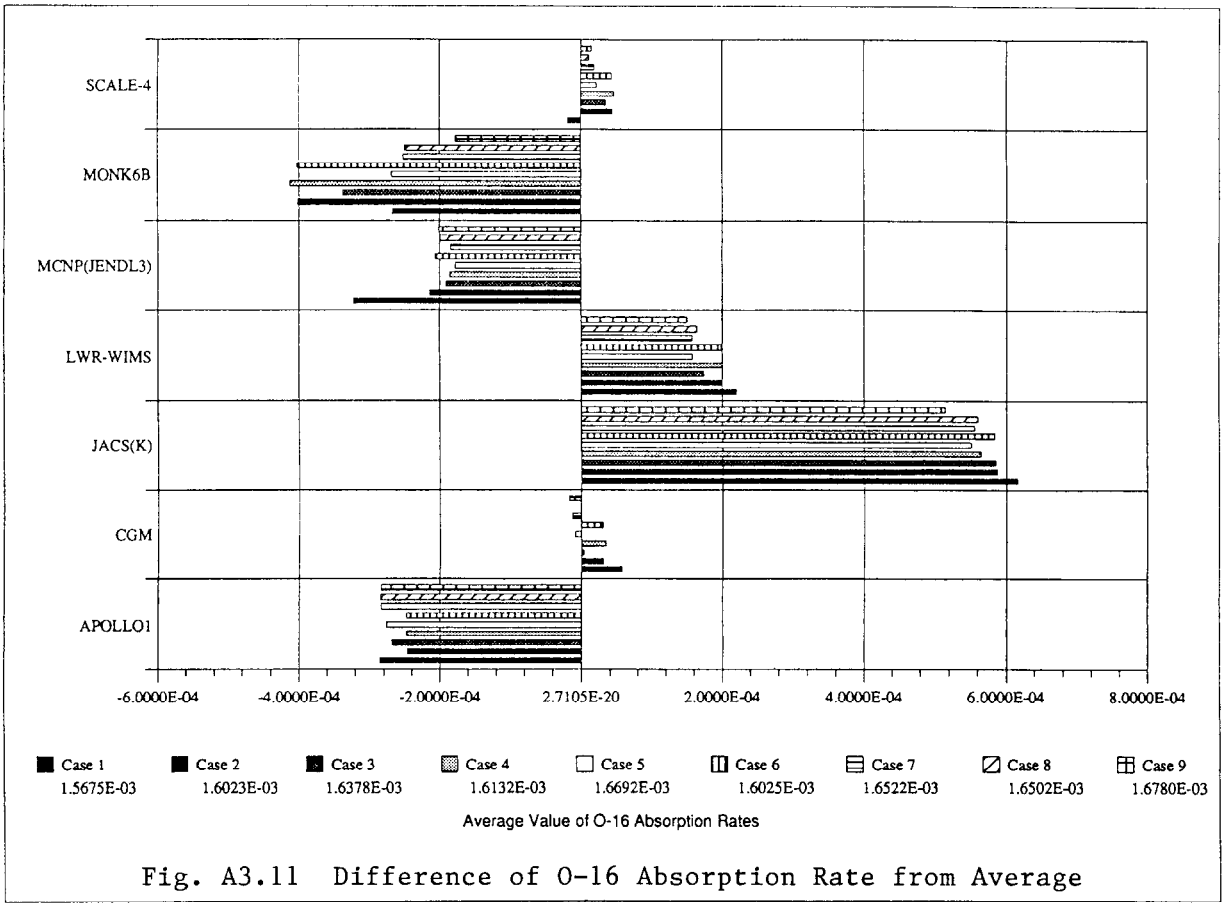

Fig. A3.11 Difference of O-16 Absorption Rate from Average

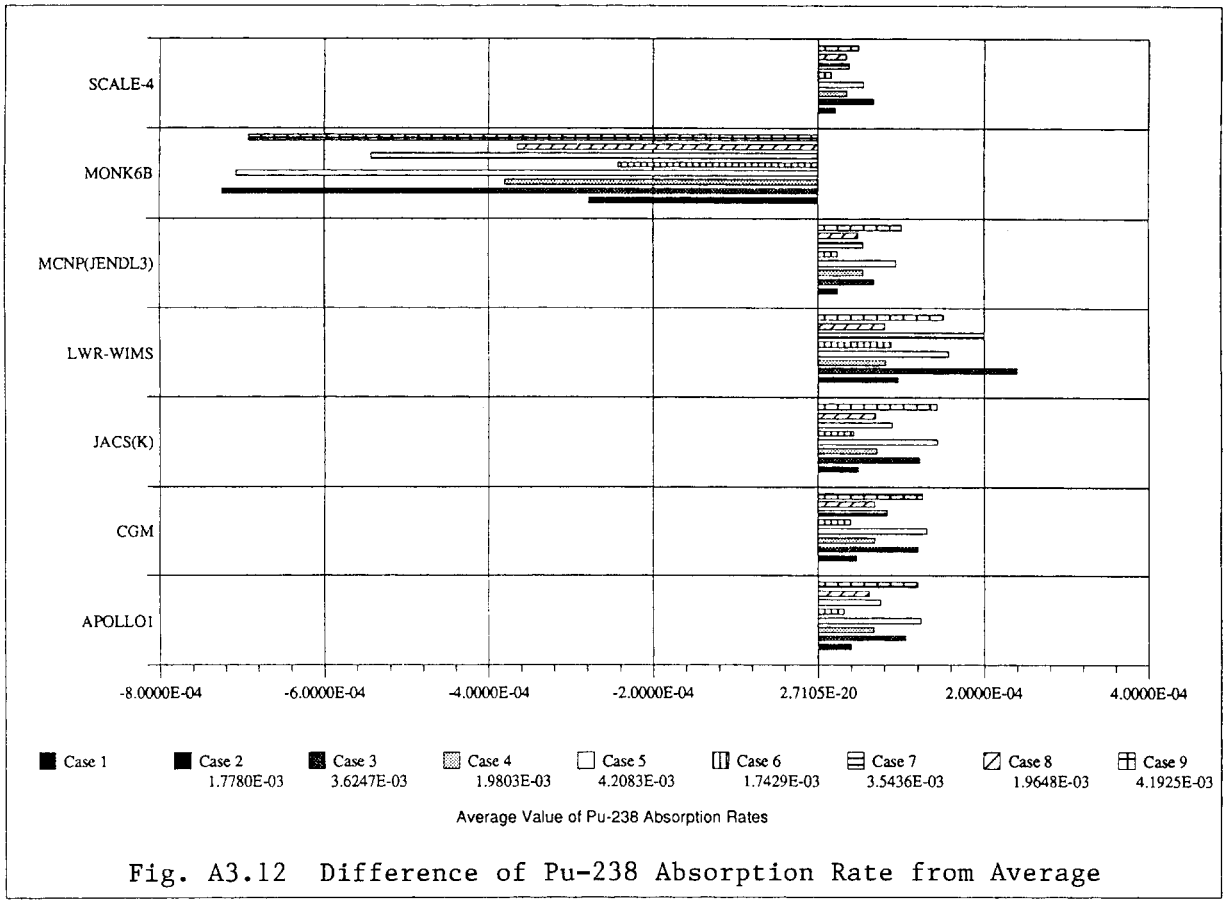

Fig. A3.12 Difference of Pu-238 Absorption Rate from Average

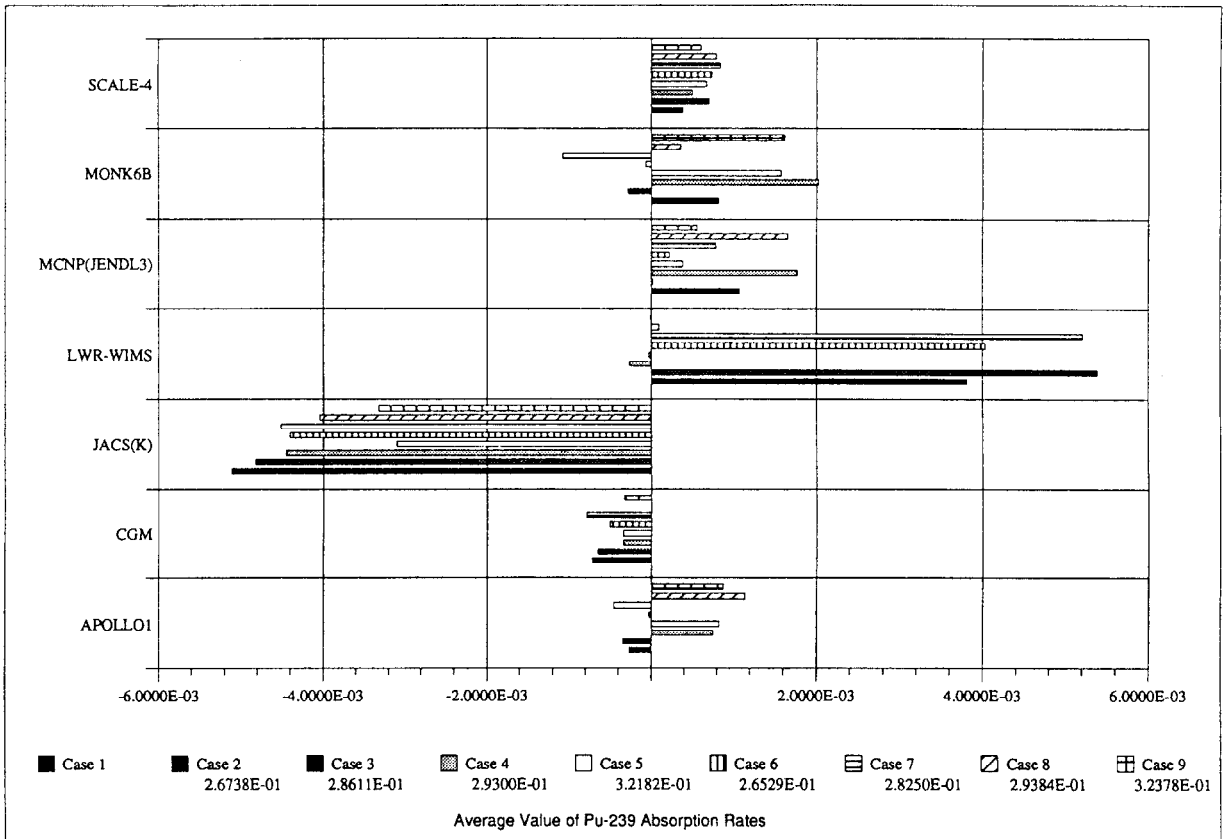

Fig. A3.13 Difference of Pu-239 Absorption Rate from Average

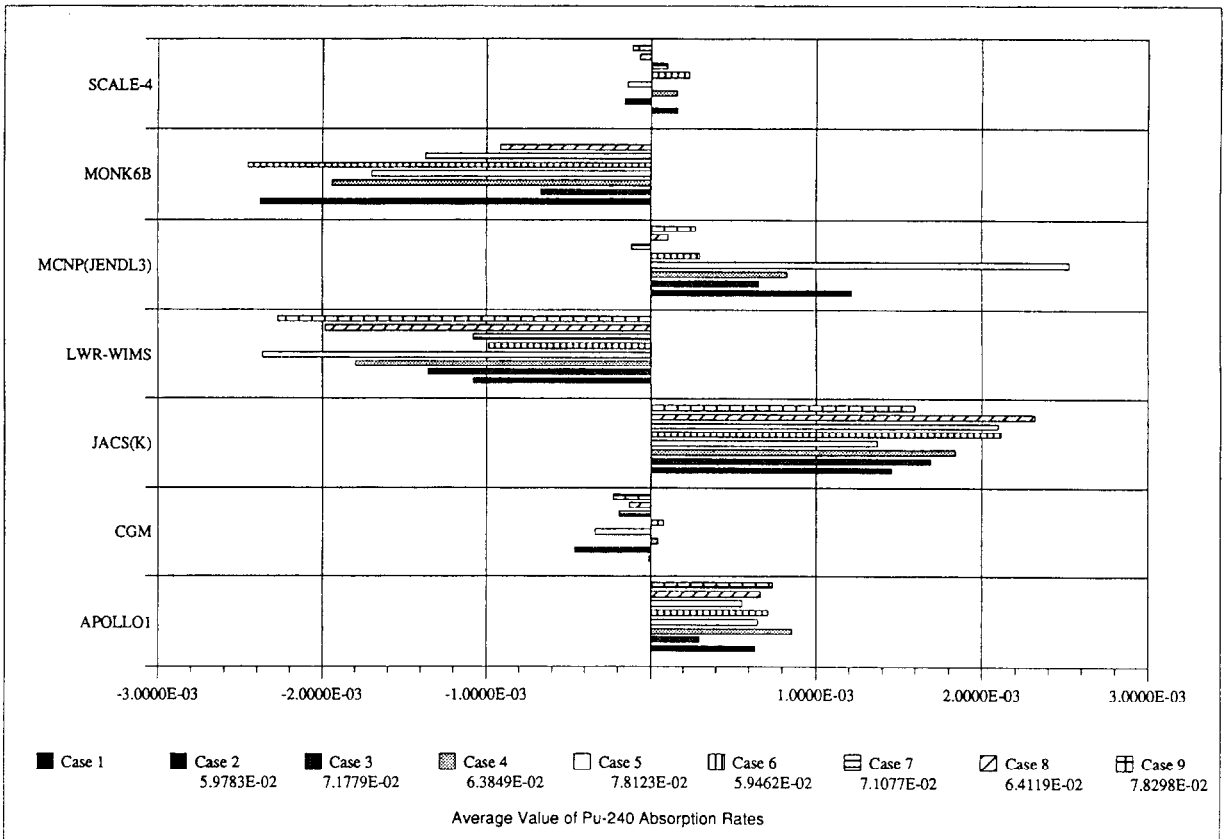

Fig. A3.14 Difference of Pu-240 Absorption Rate from Average

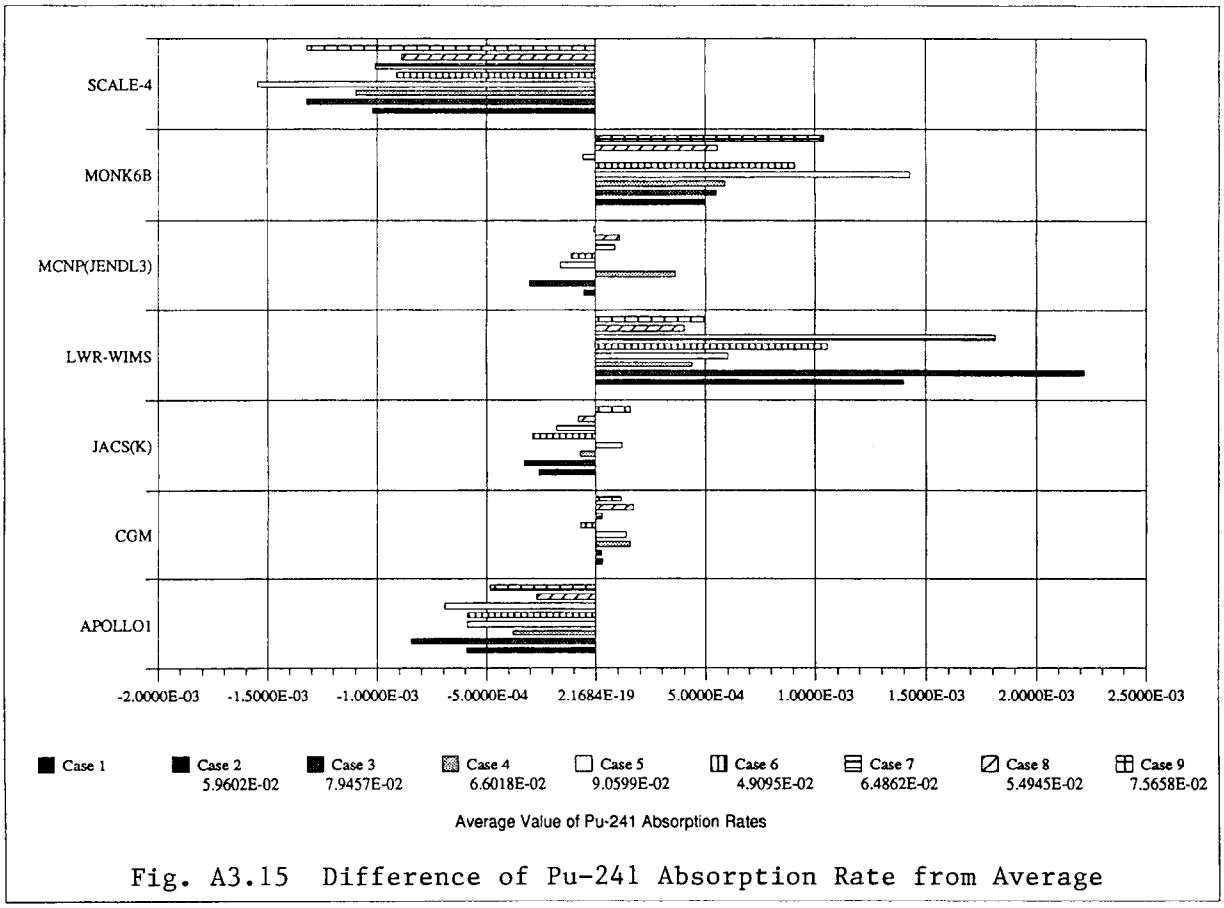

Fig. A3.15 Difference of Pu-241 Absorption Rate from Average

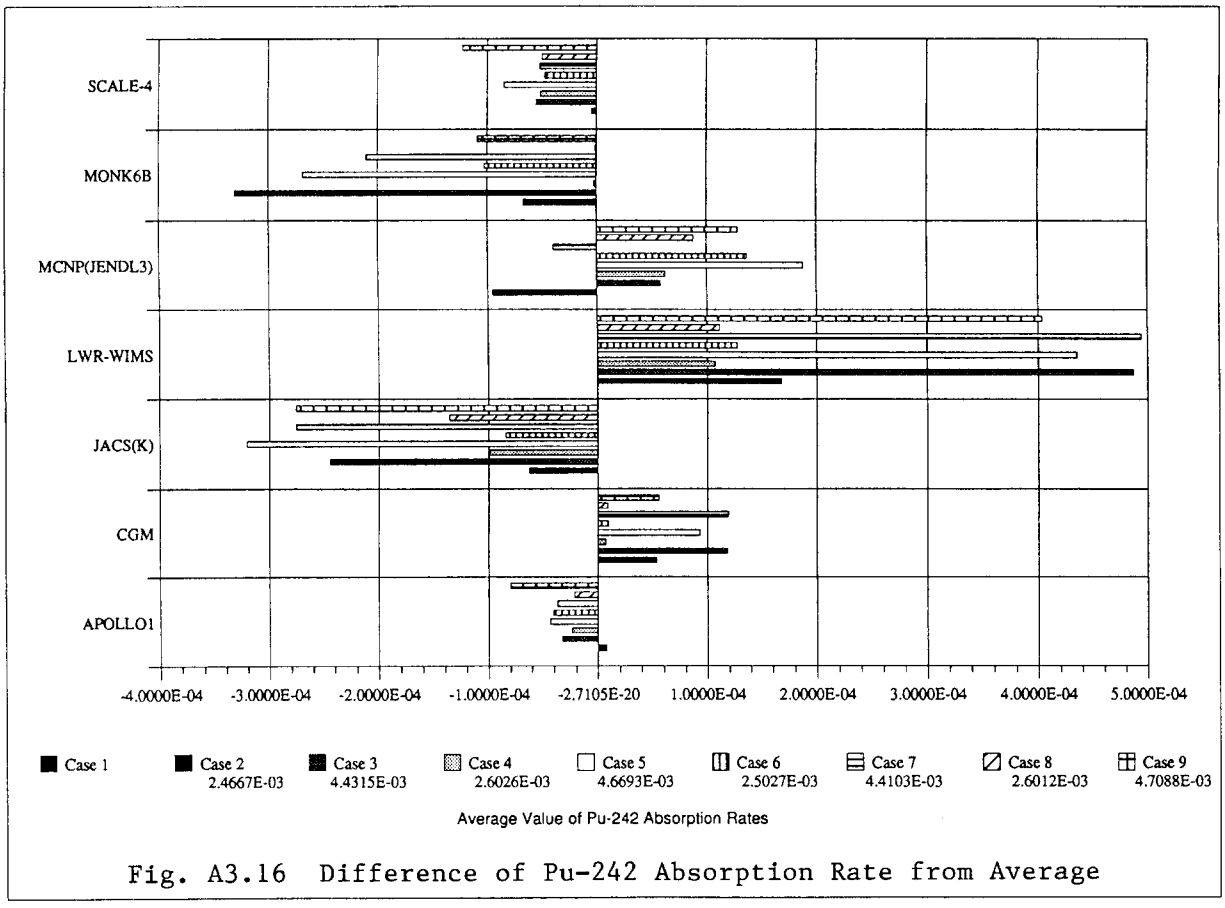

Fig. A3.16 Difference of Pu-242 Absorption Rate from Average

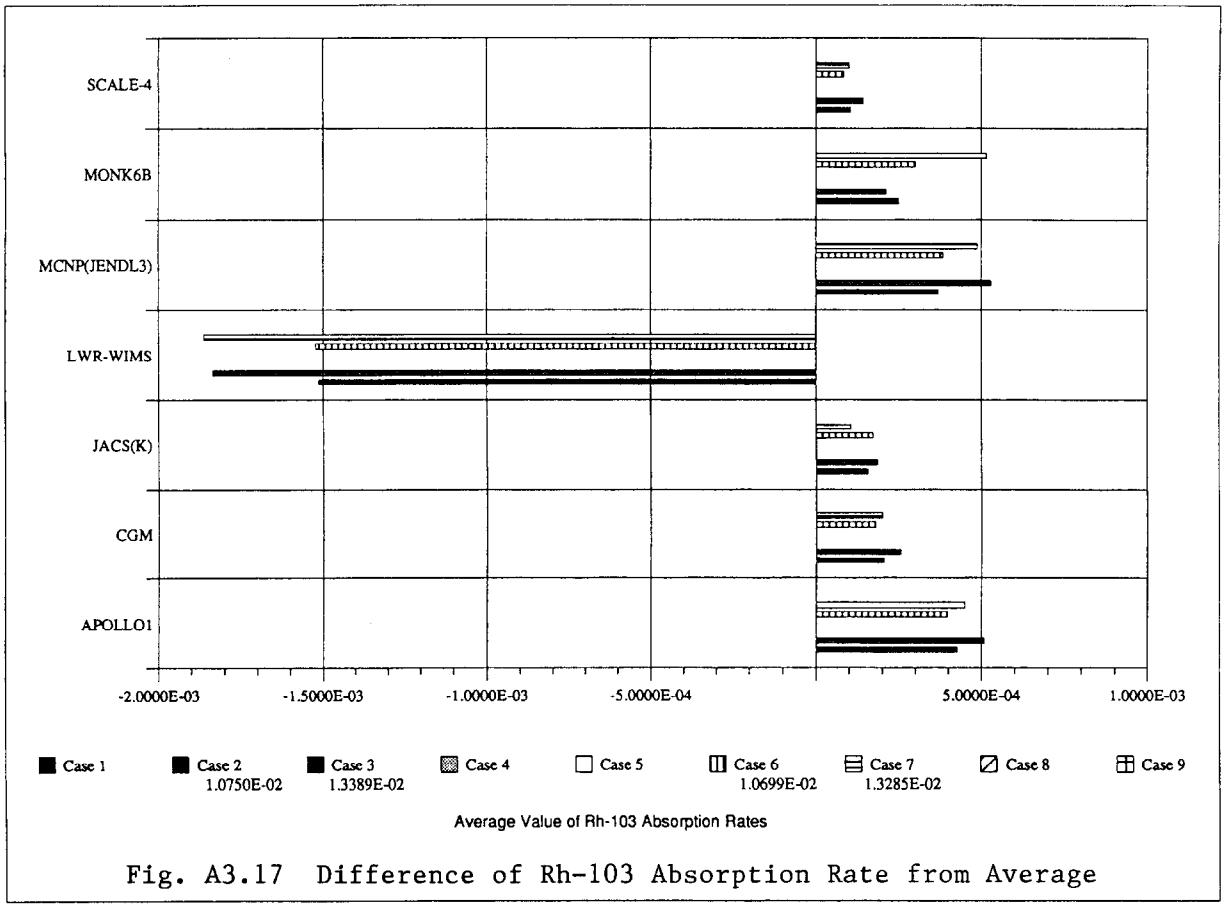

Fig. A3.17 Difference of Rh-103 Absorption Rate from Average

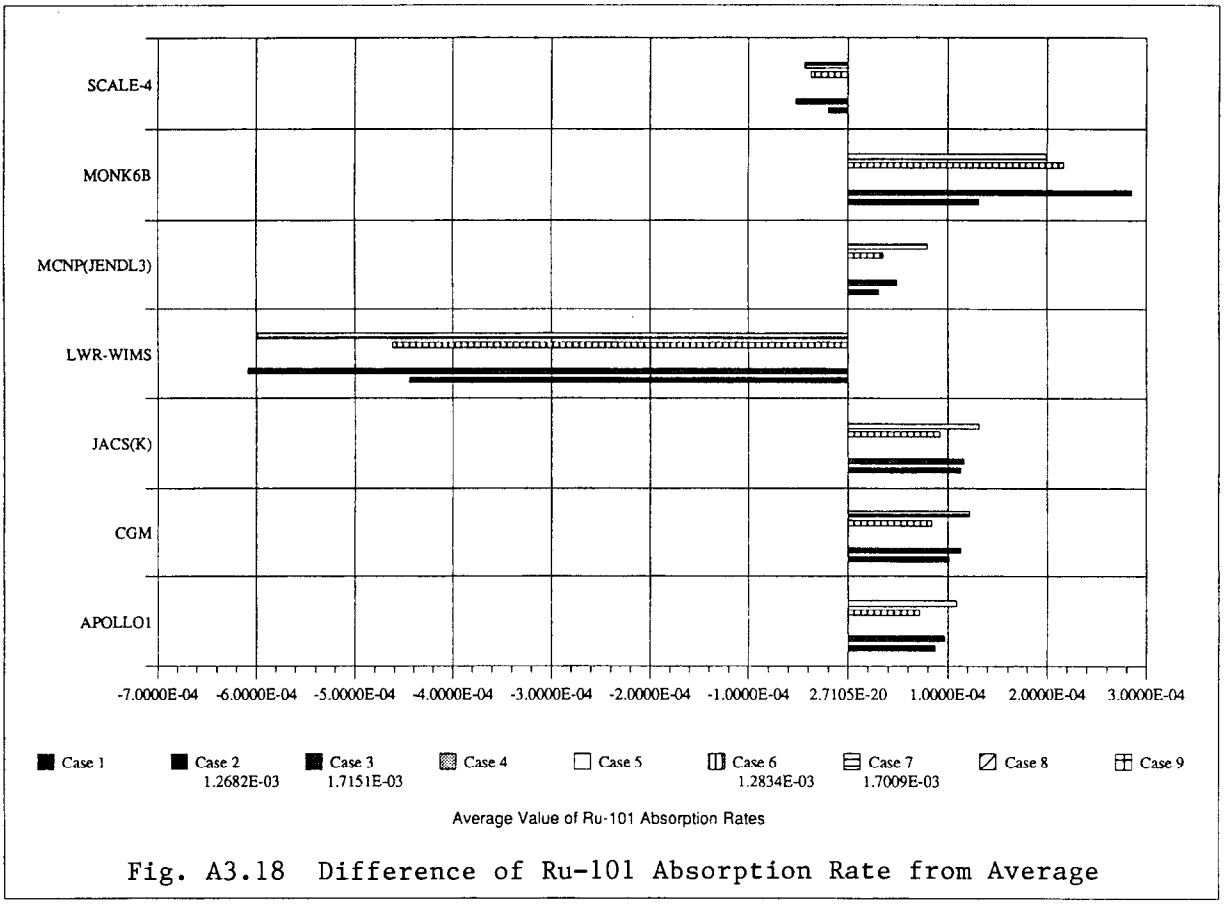

Fig. A3.18 Difference of Ru-101 Absorption Rate from Average

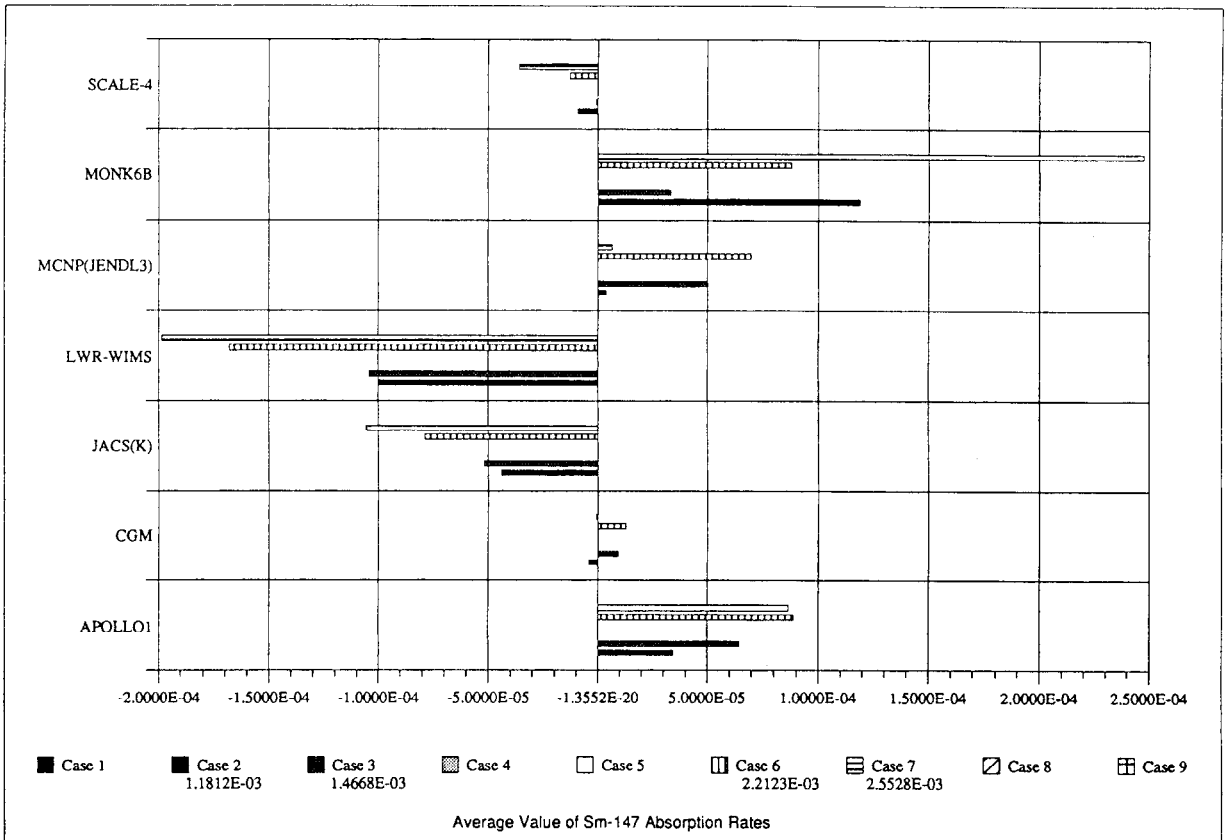

Fig. A3.19 Difference of Sm-147 Absorption Rate from Average

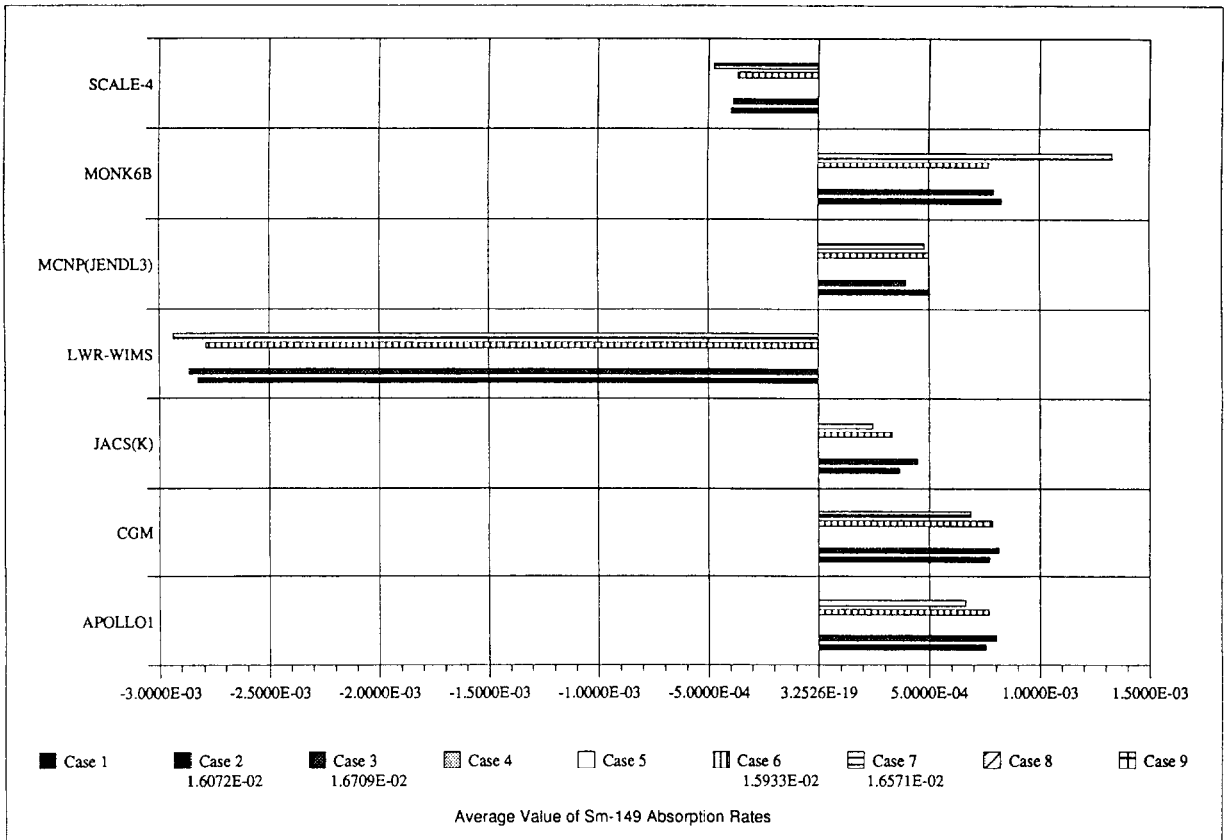

Fig. A3.20 Difference of Sm-149 Absorption Rate from Average

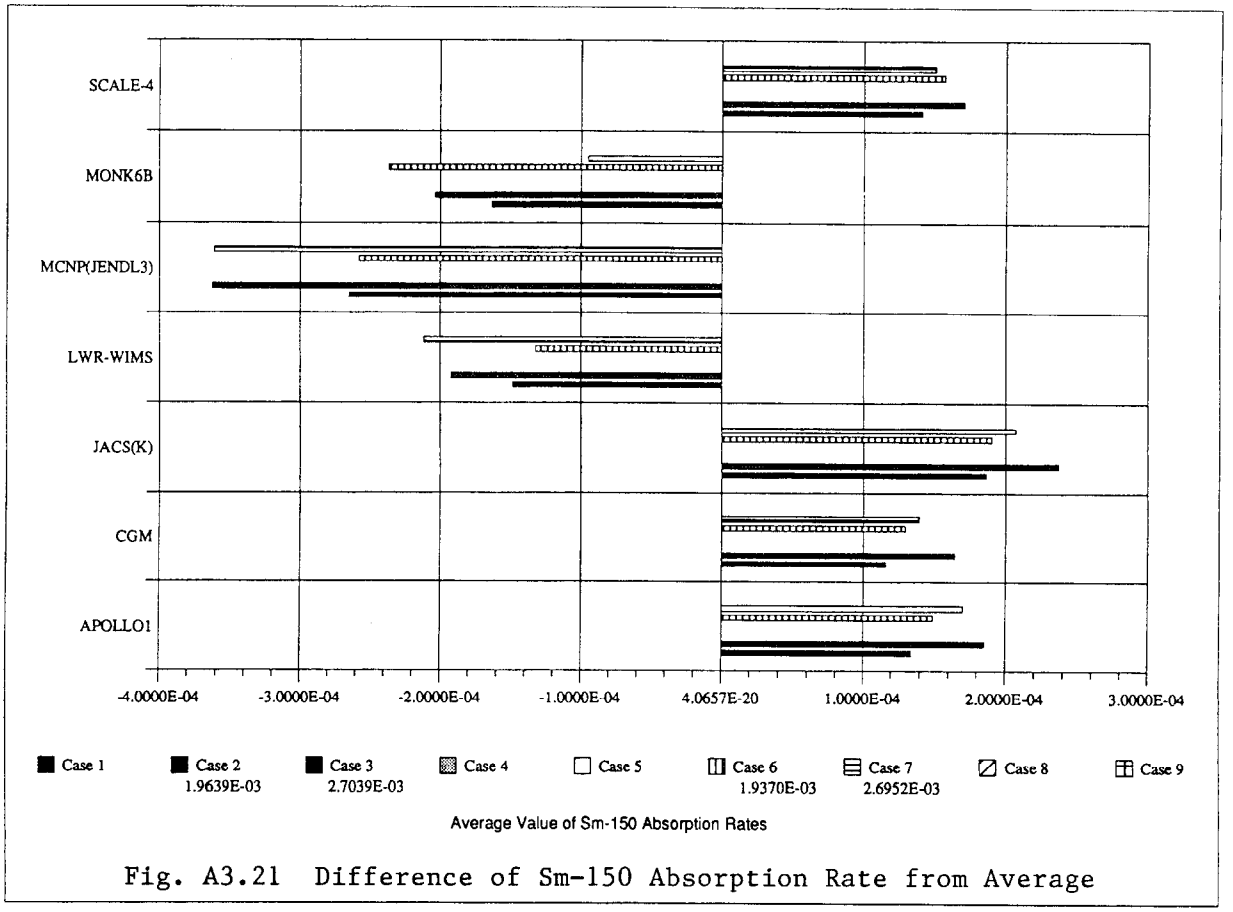

Fig. A3.21 Difference of Sm-150 Absorption Rate from Average

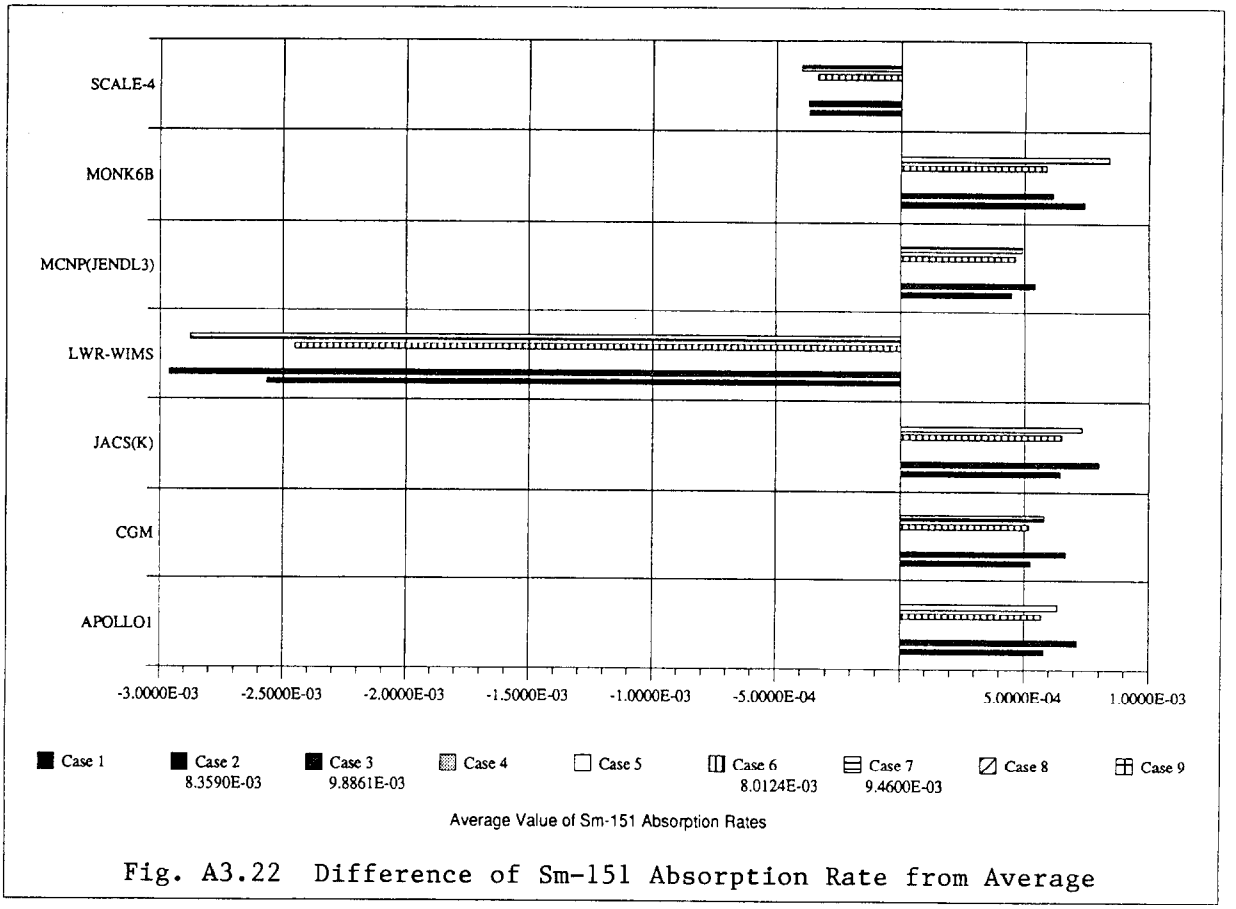

Fig. A3.22 Difference of Sm-151 Absorption Rate from Average

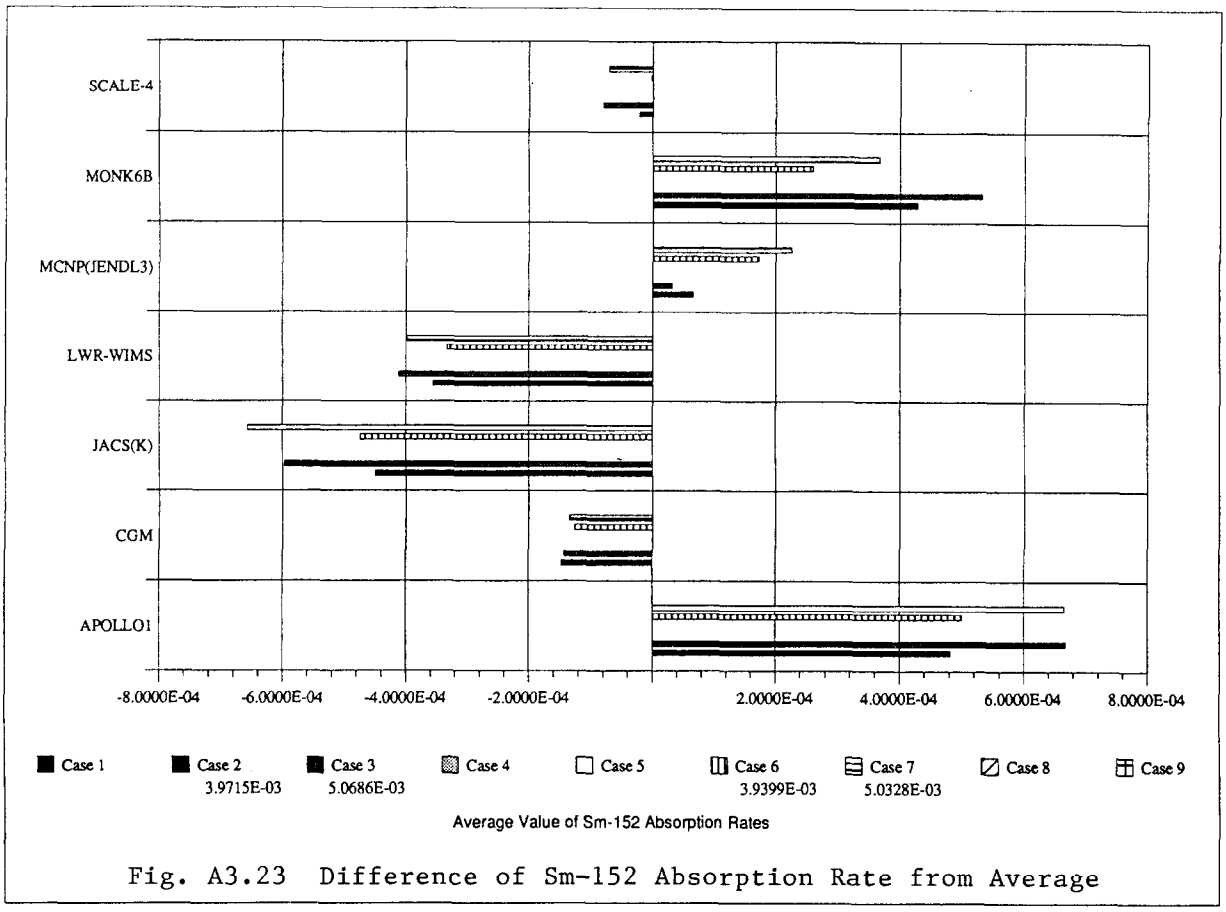

Fig. A3.23 Difference of Sm-152 Absorption Rate from Average

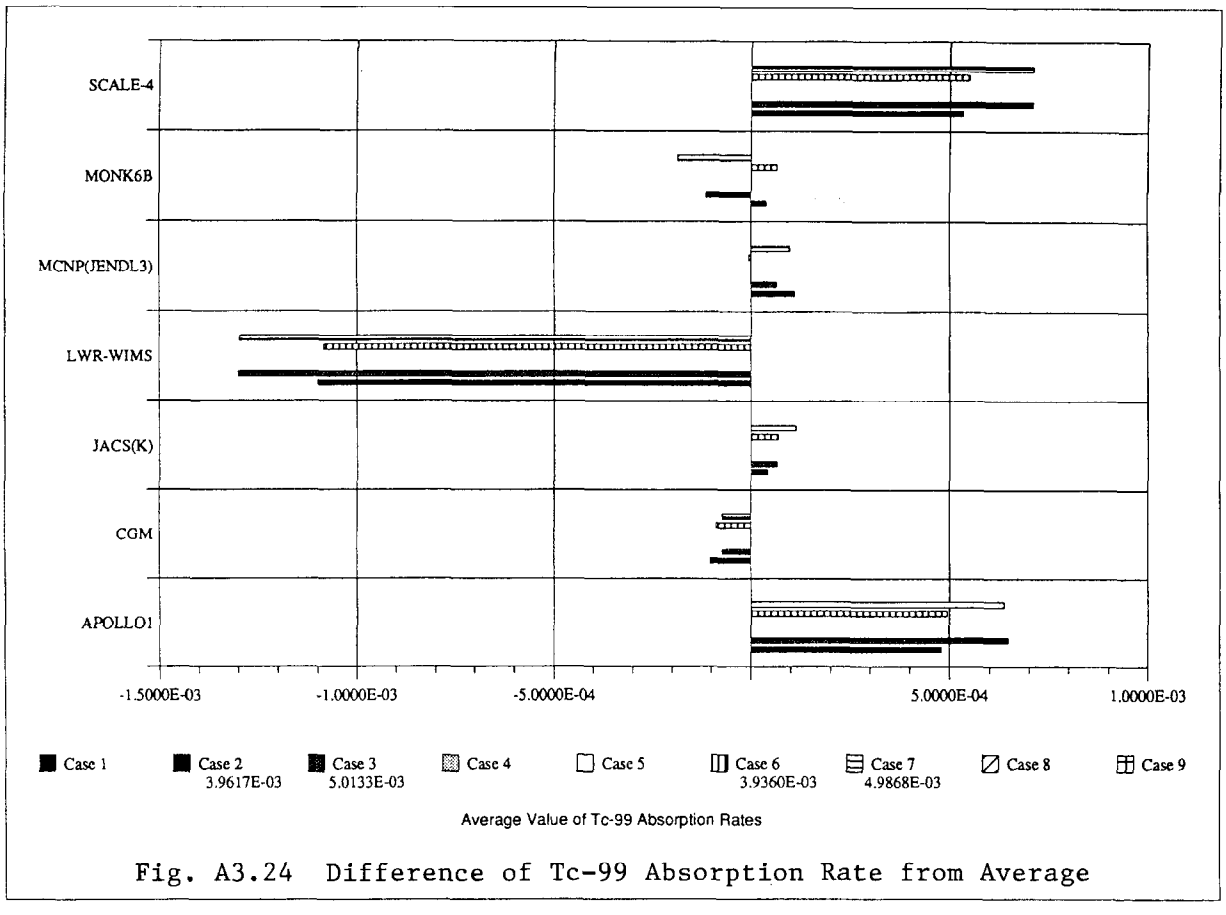

Fig. A3.24 Difference of Tc-99 Absorption Rate from Average

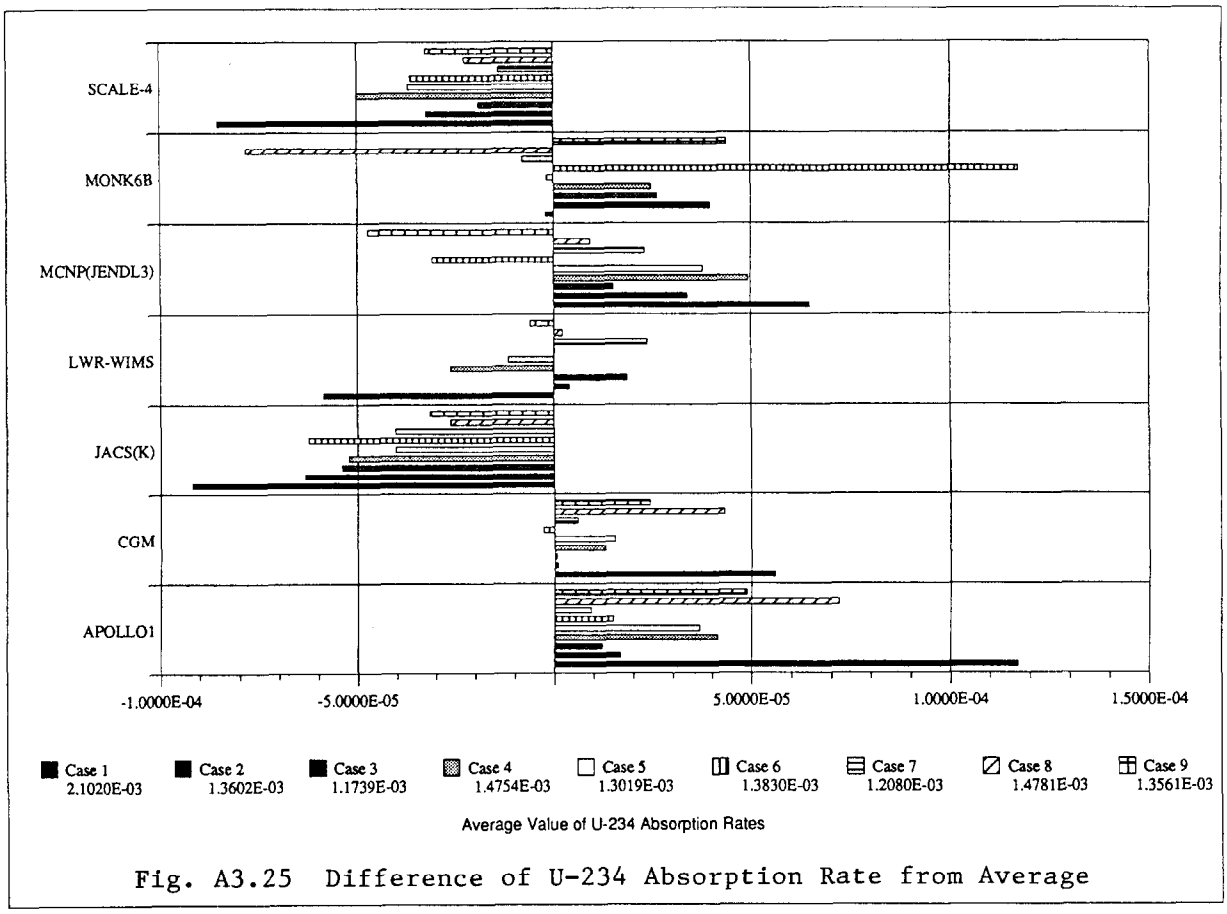

Fig. A3.25 Difference of U-234 Absorption Rate from Average

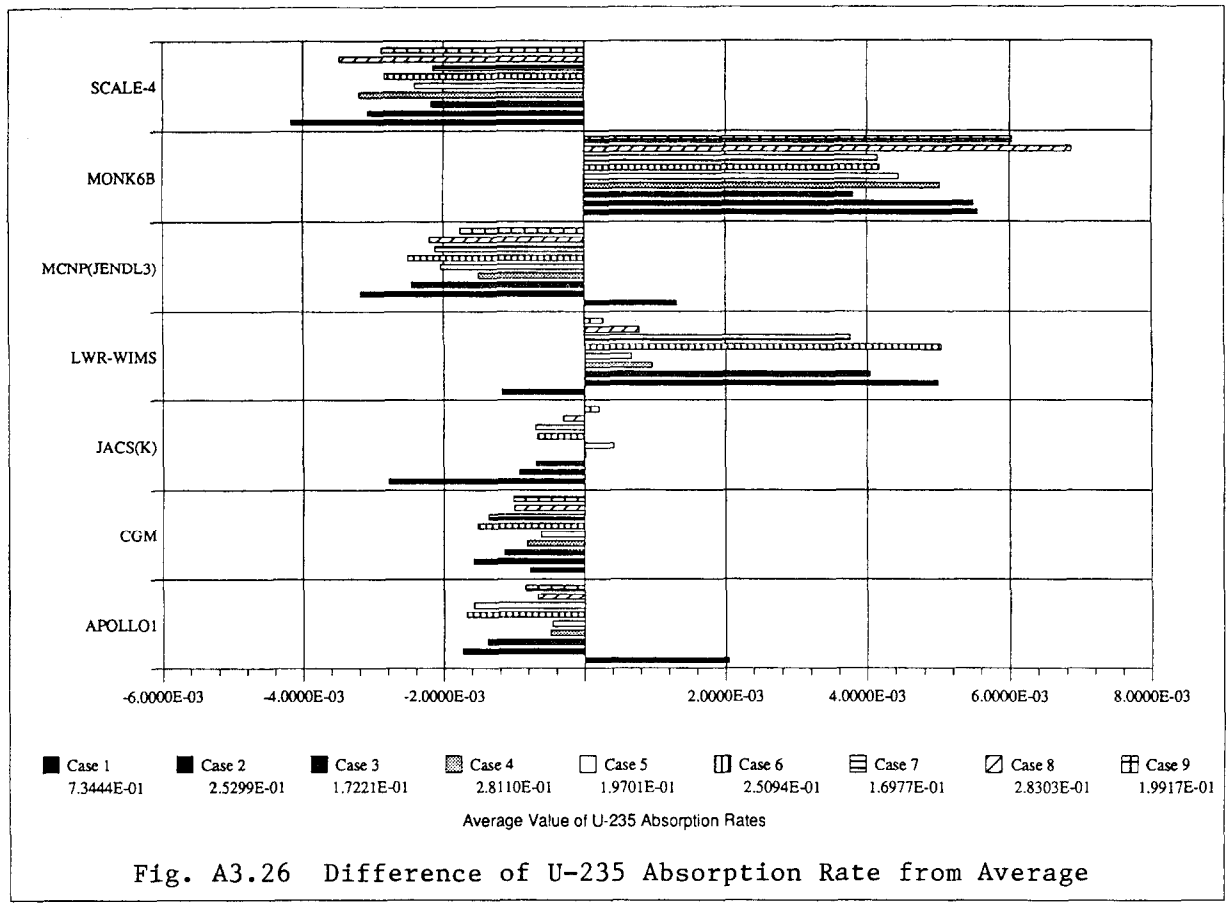

Fig. A3.26 Difference of U-235 Absorption Rate from Average

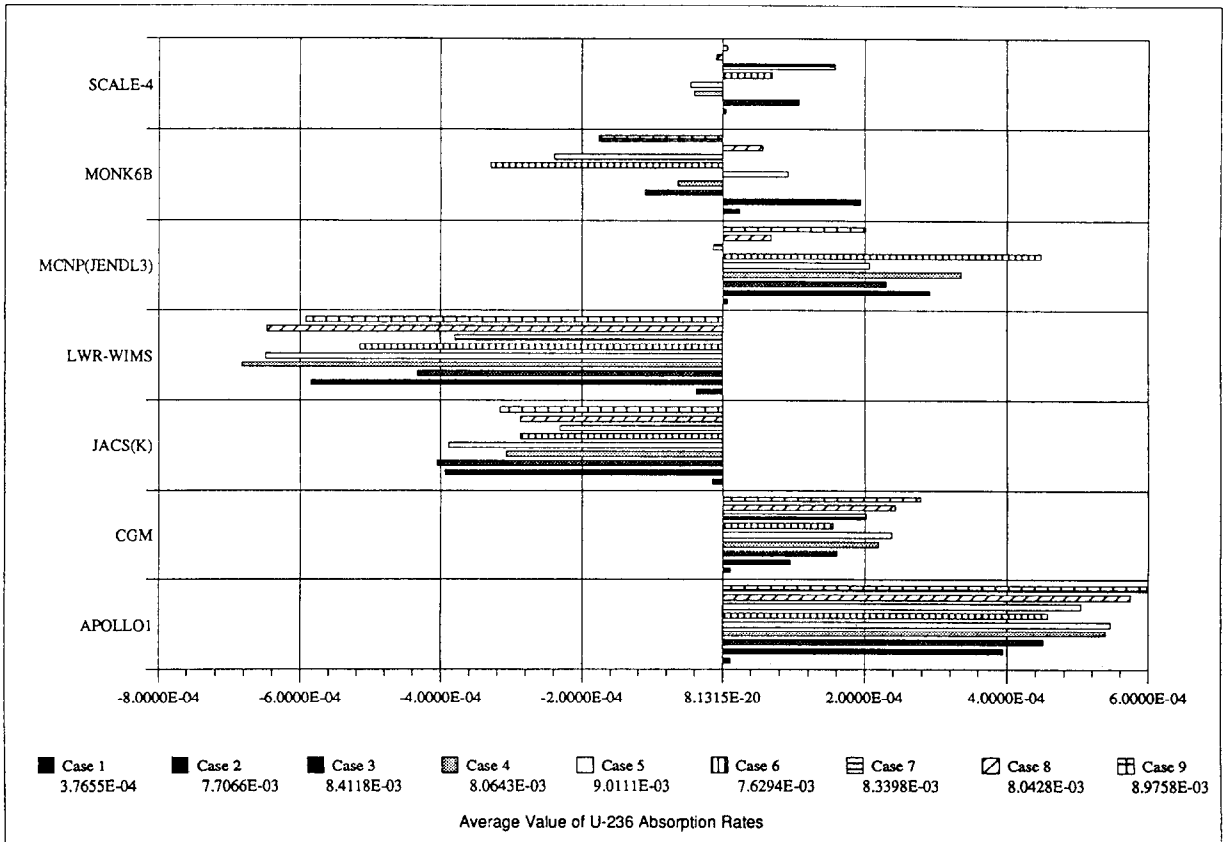

Fig. A3.27 Difference of U-236 Absorption Rate from Average

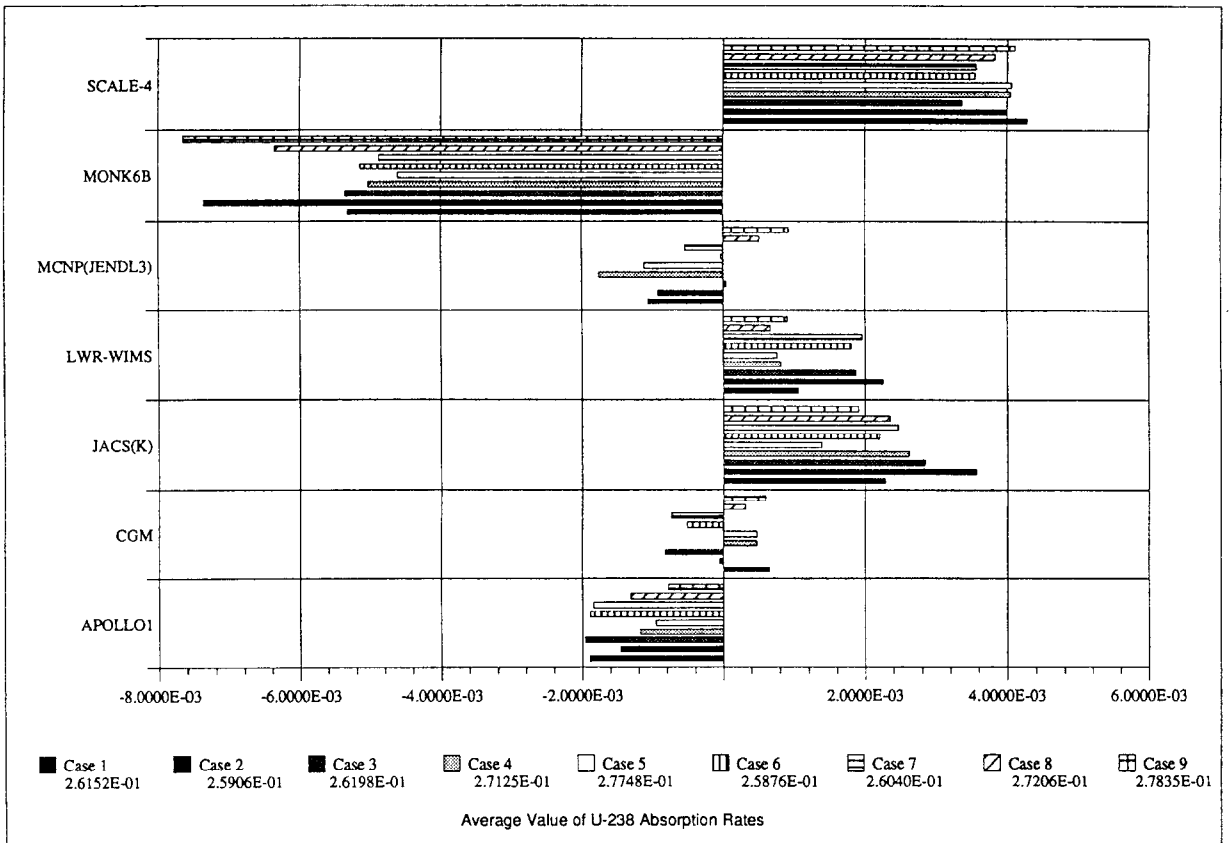

Fig. A3.28 Difference of U-238 Absorption Rate from Average

Appendix 3.2 Production Rate

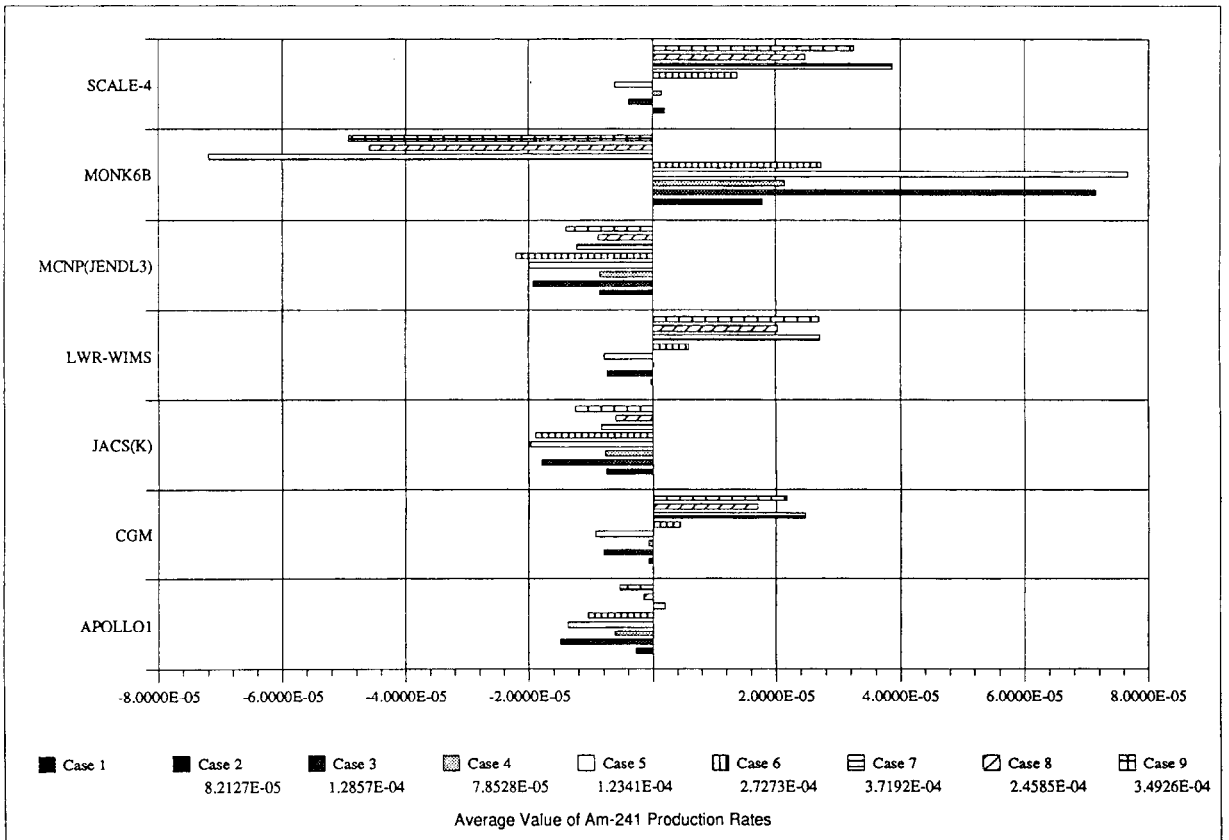

Fig. A3.29 Difference of Am-241 Production Rate from Average

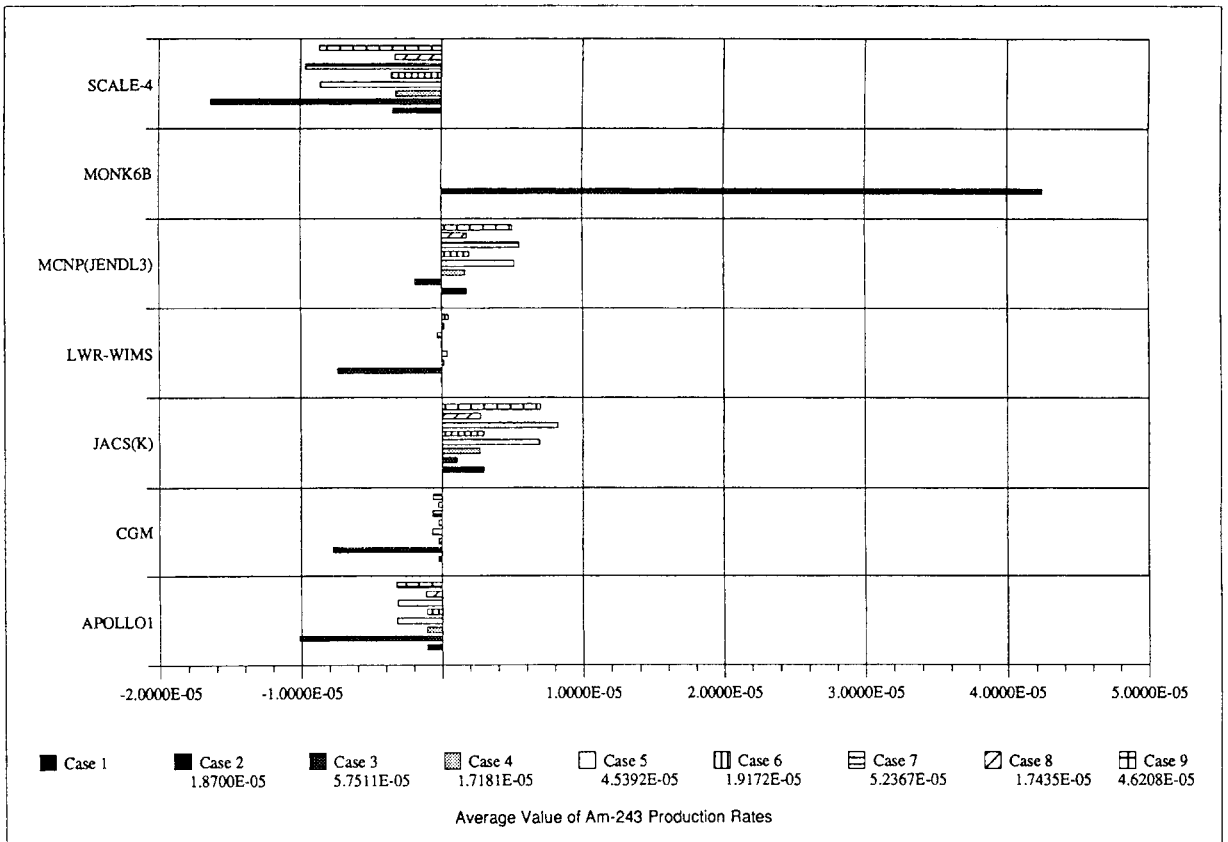

Fig. A3.30 Difference of Am-243 Production Rate from Average

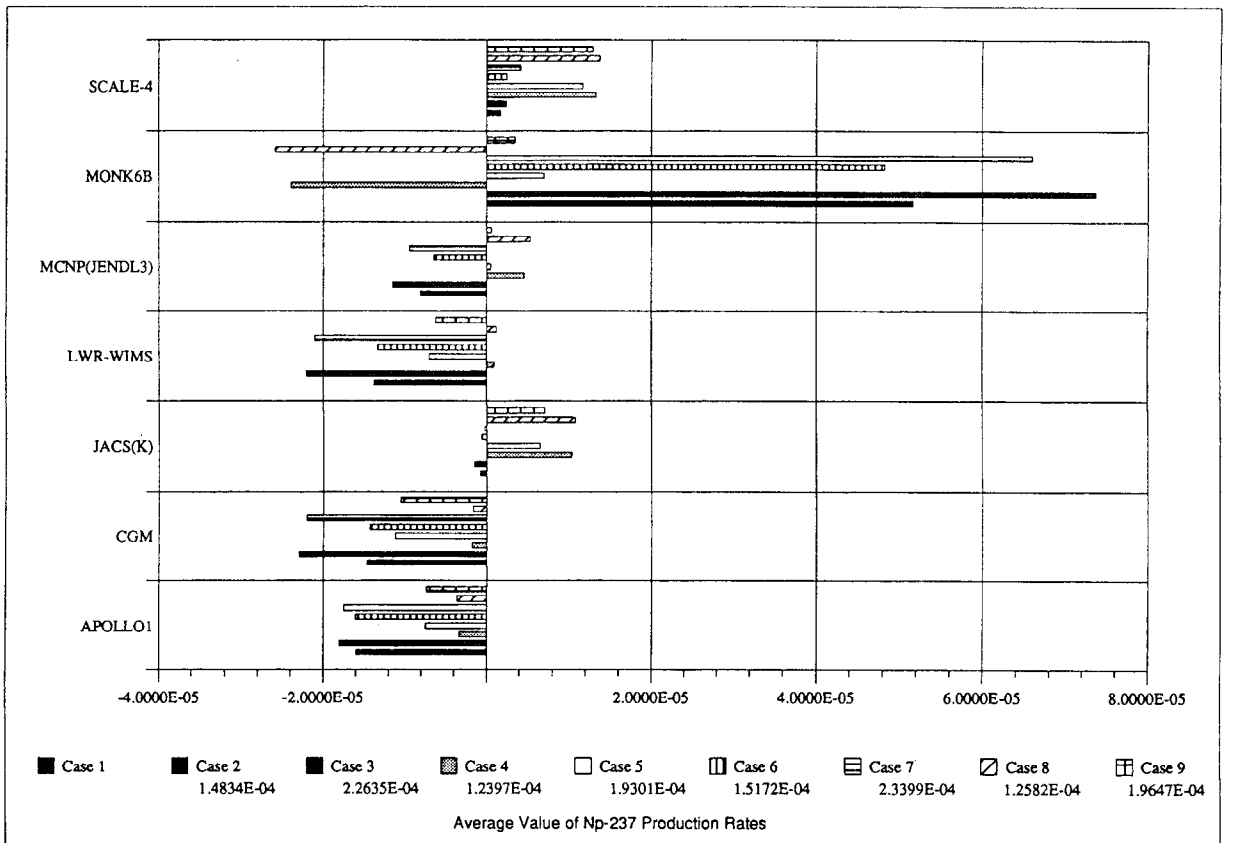

Fig. A3.31 Difference of Np-237 Production Rate from Average

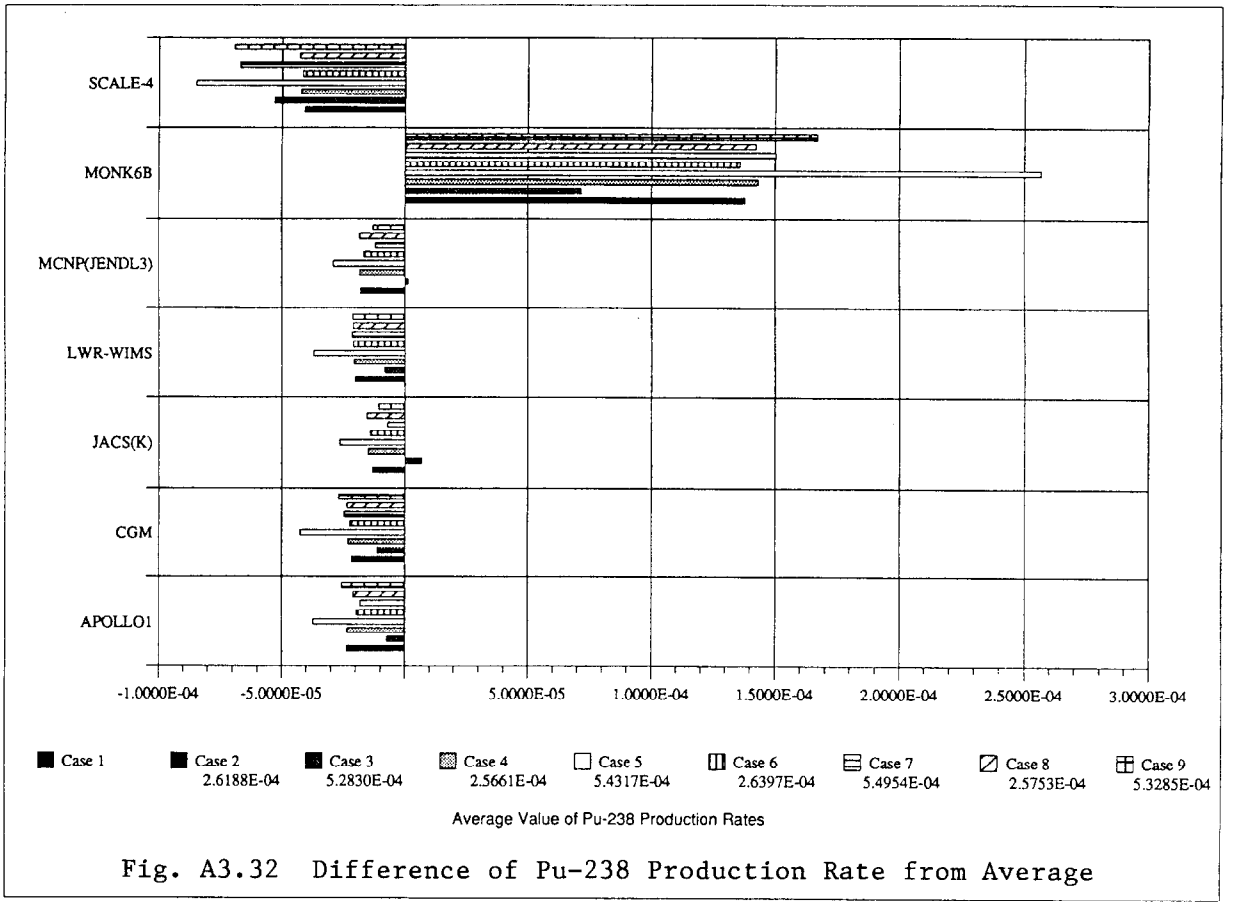

Fig. A3.32 Difference of Pu-238 Production Rate from Average

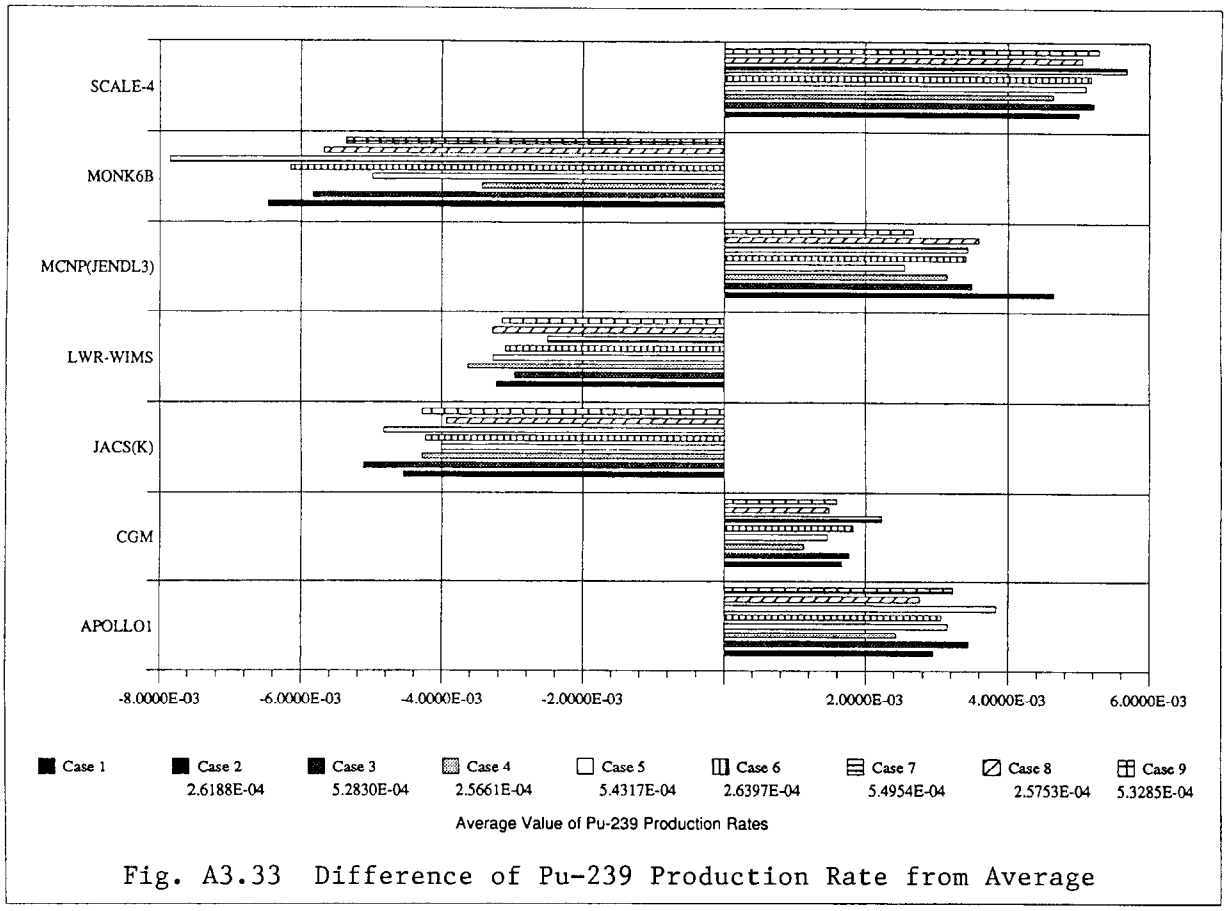

Fig. A3.33 Difference of Pu-239 Production Rate from Average

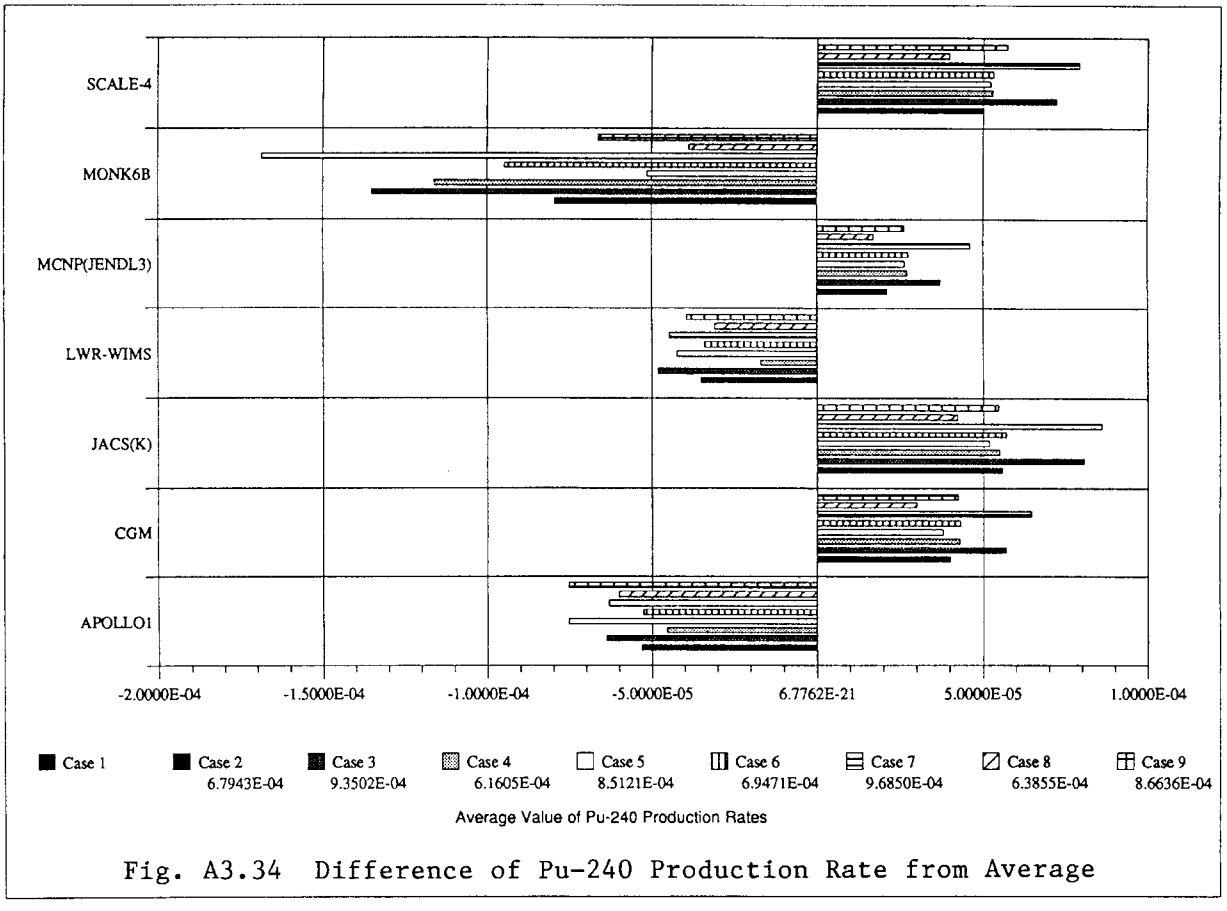

Fig. A3.34 Difference of Pu-240 Production Rate from Average

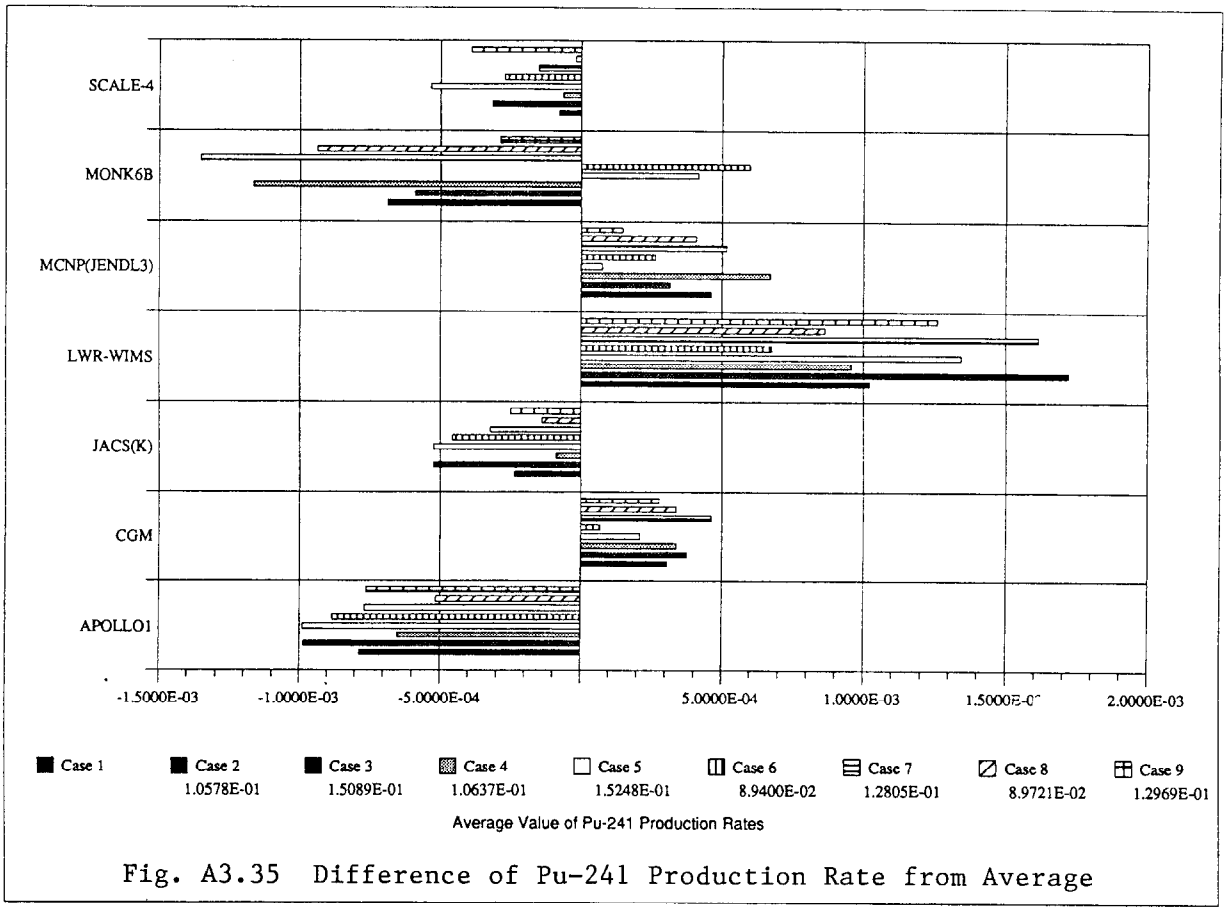

Fig. A3.35 Difference of Pu-241 Production Rate from Average

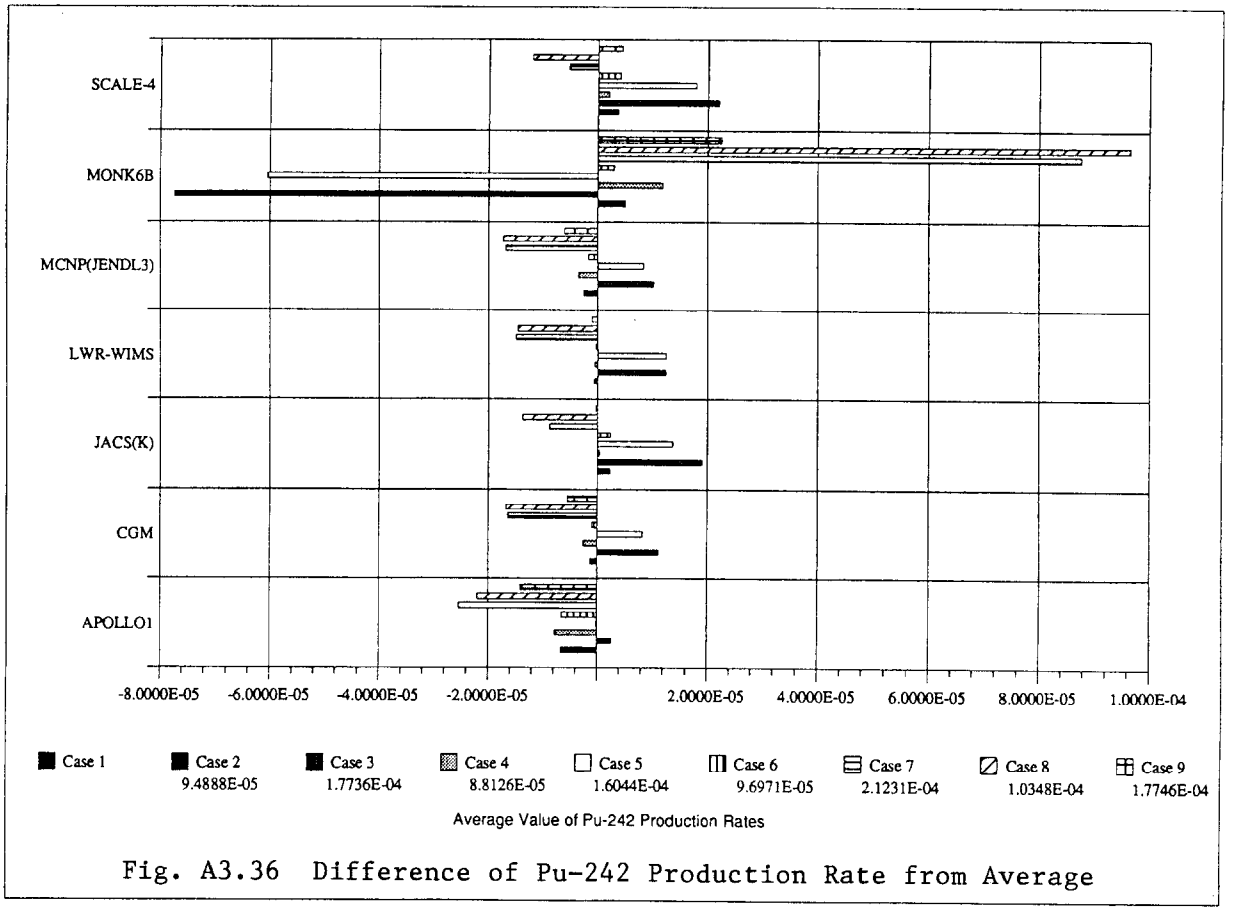

Fig. A3.36 Difference of Pu-242 Production Rate from Average

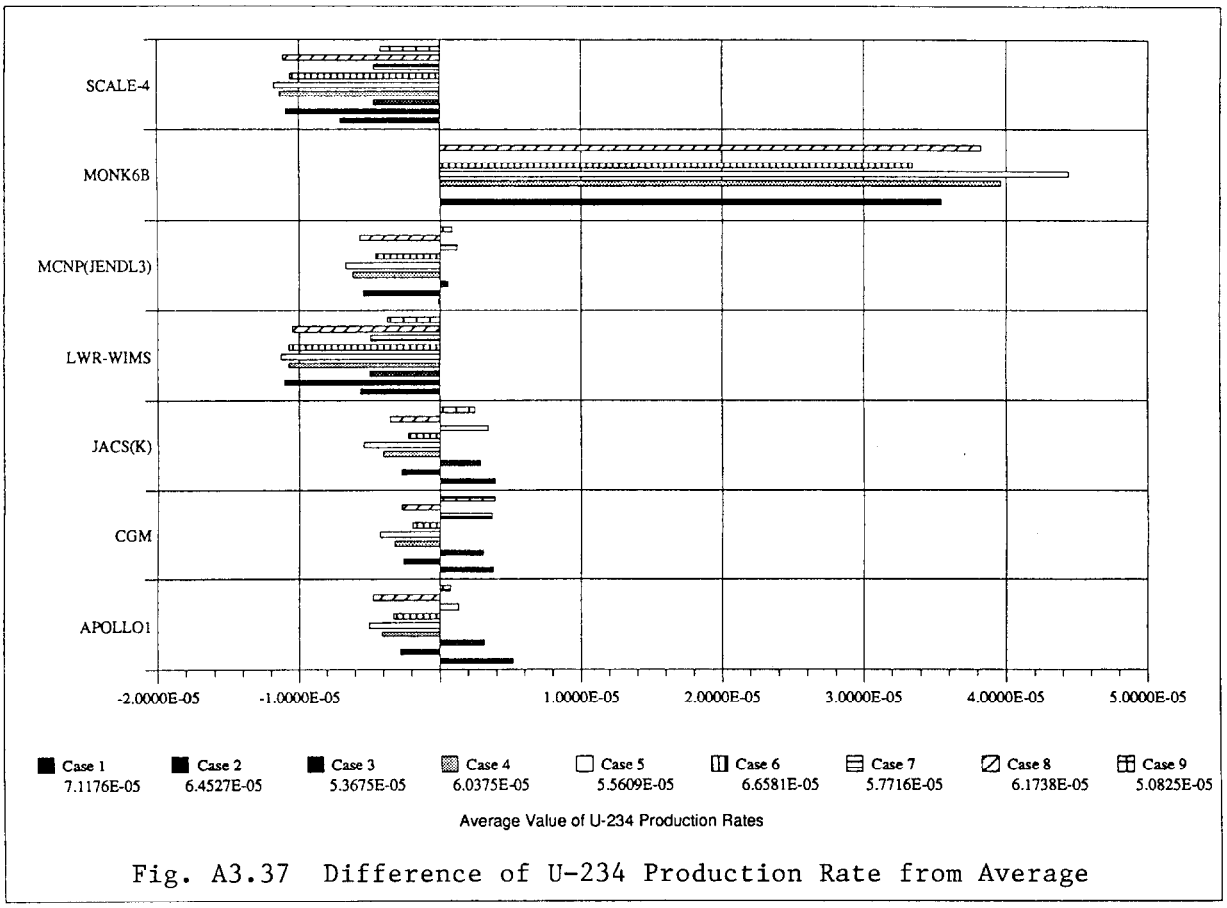

Fig. A3.37 Difference of U-234 Production Rate from Average

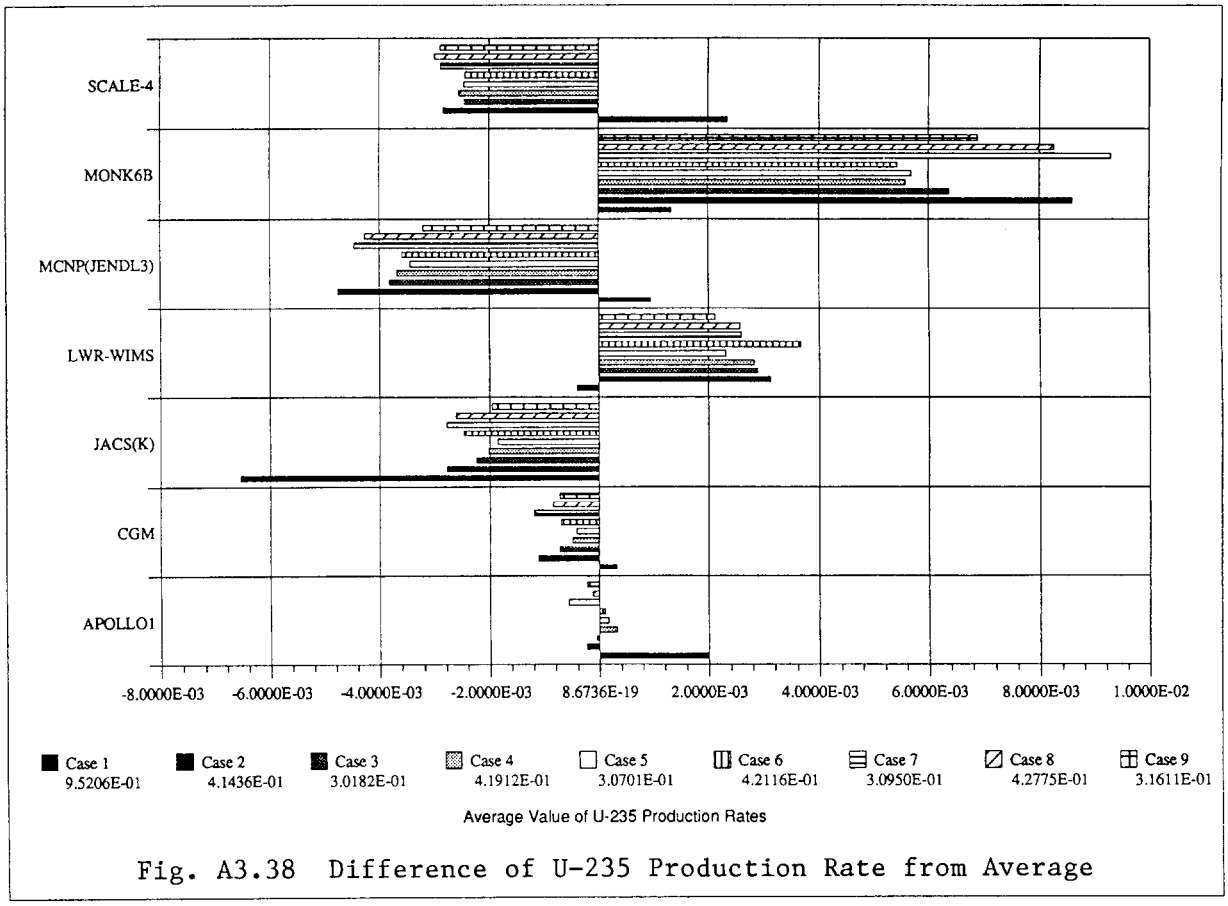

Fig. A3.38 Difference of U-235 Production Rate from Average

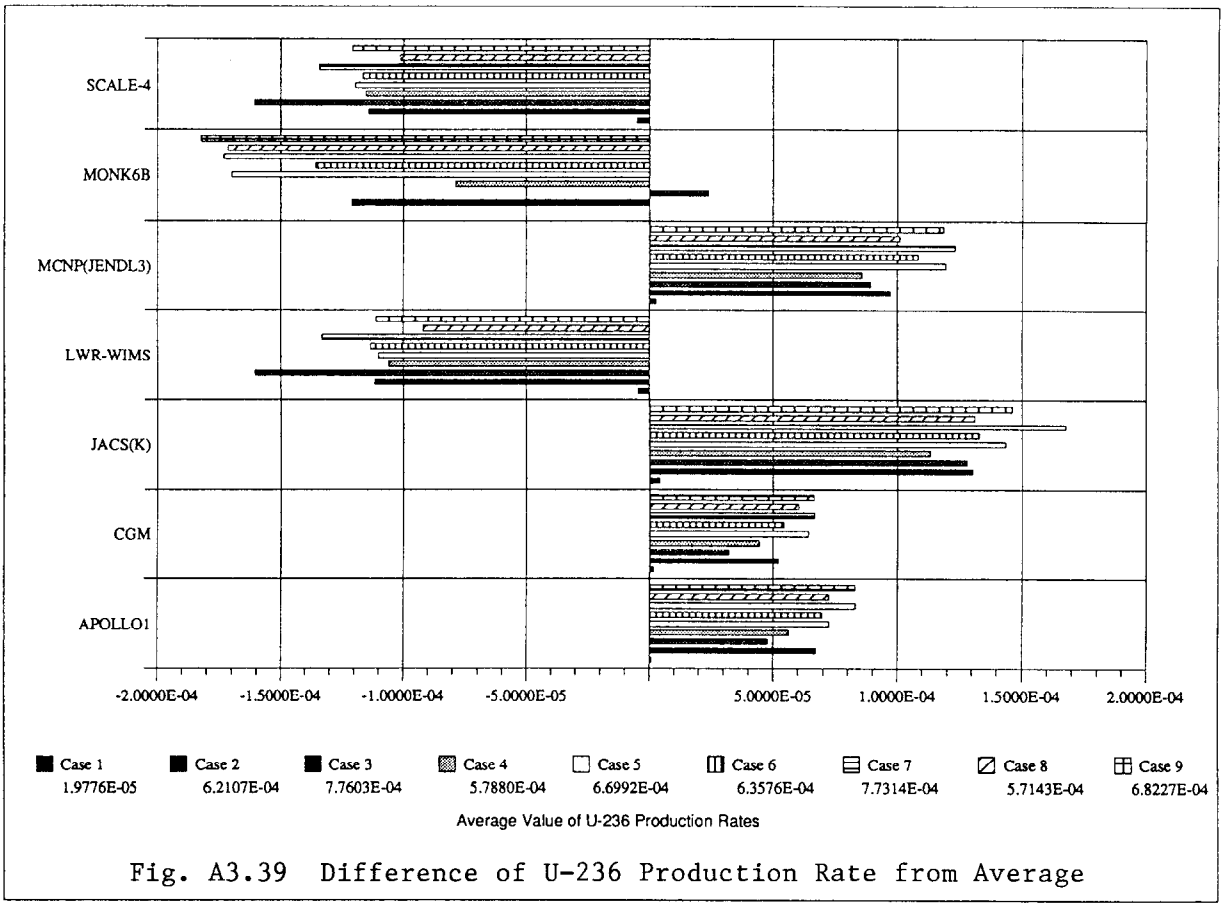

Fig. A3.39 Difference of U-236 Production Rate from Average

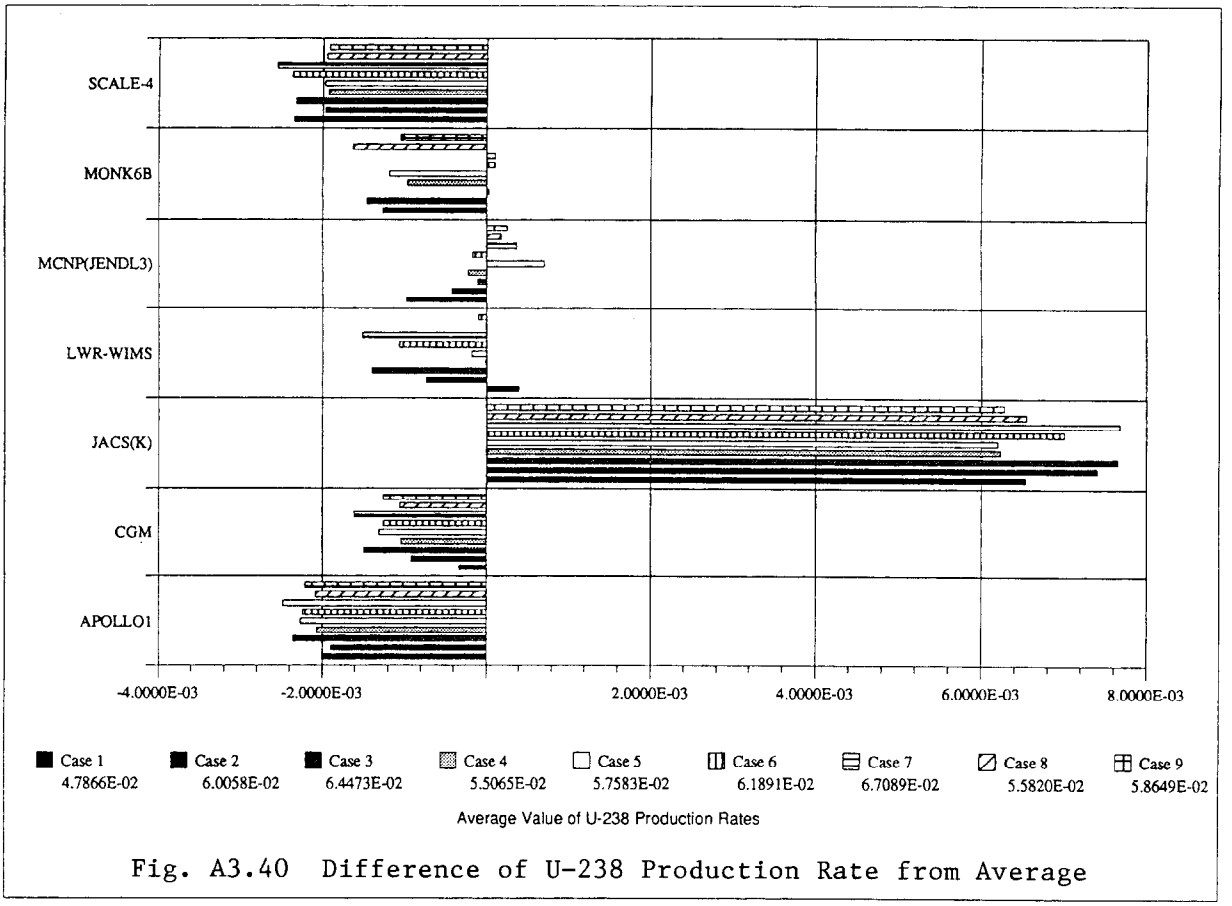

Fig. A3.40 Difference of U-238 Production Rate from Average

Appendix 3.3 Nue-Value

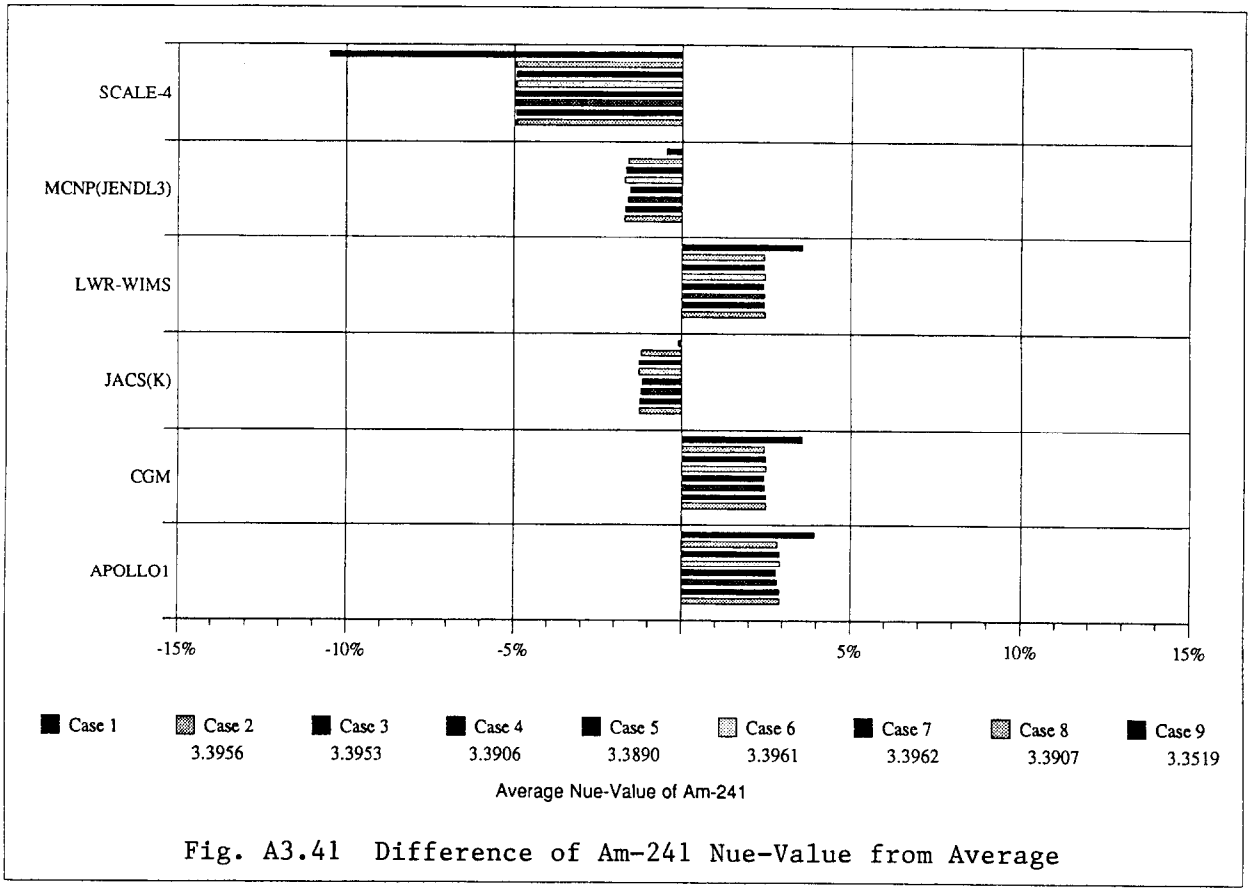

Fig. A3.41 Difference of Am-241 Nue-Value from Average

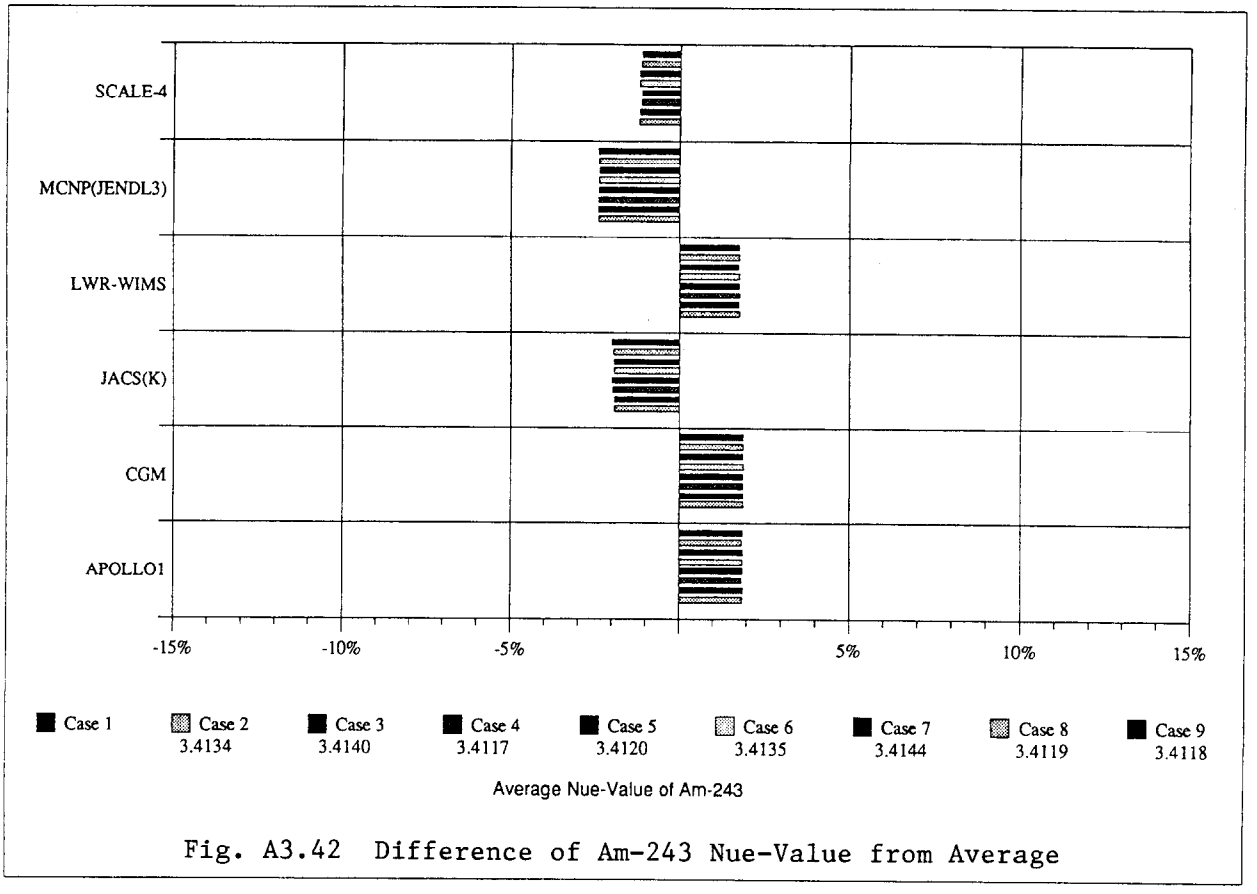

Fig. A3.42 Difference of Am-243 Nue-Value from Average

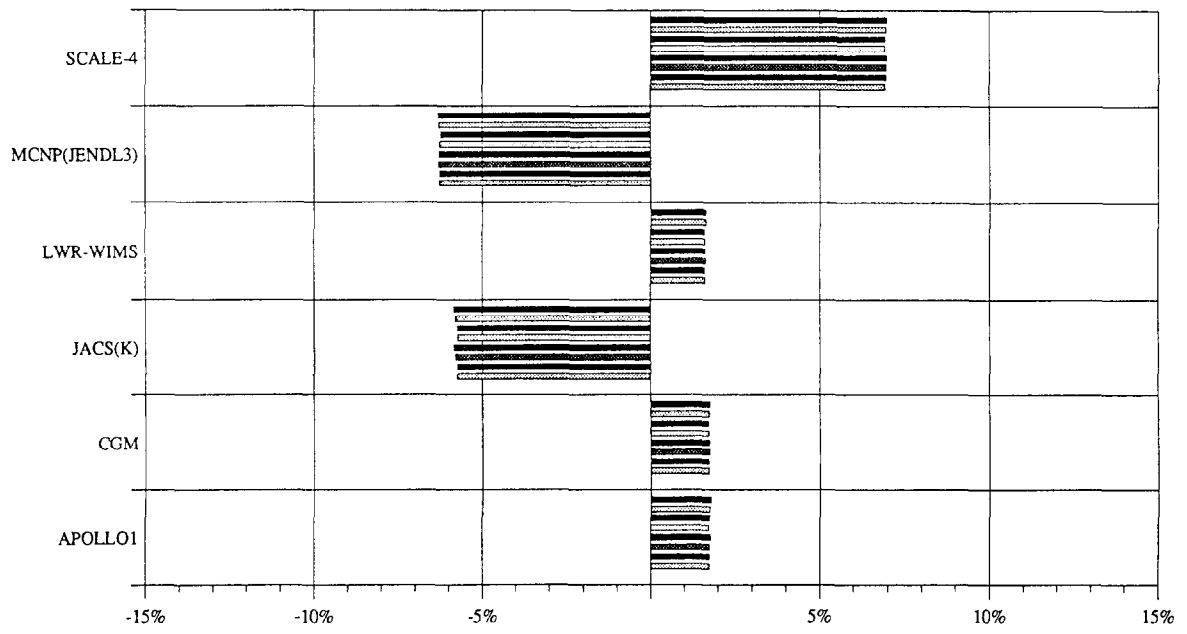

Fig. A3.43 Difference of Np-237 Nue-Value from Average

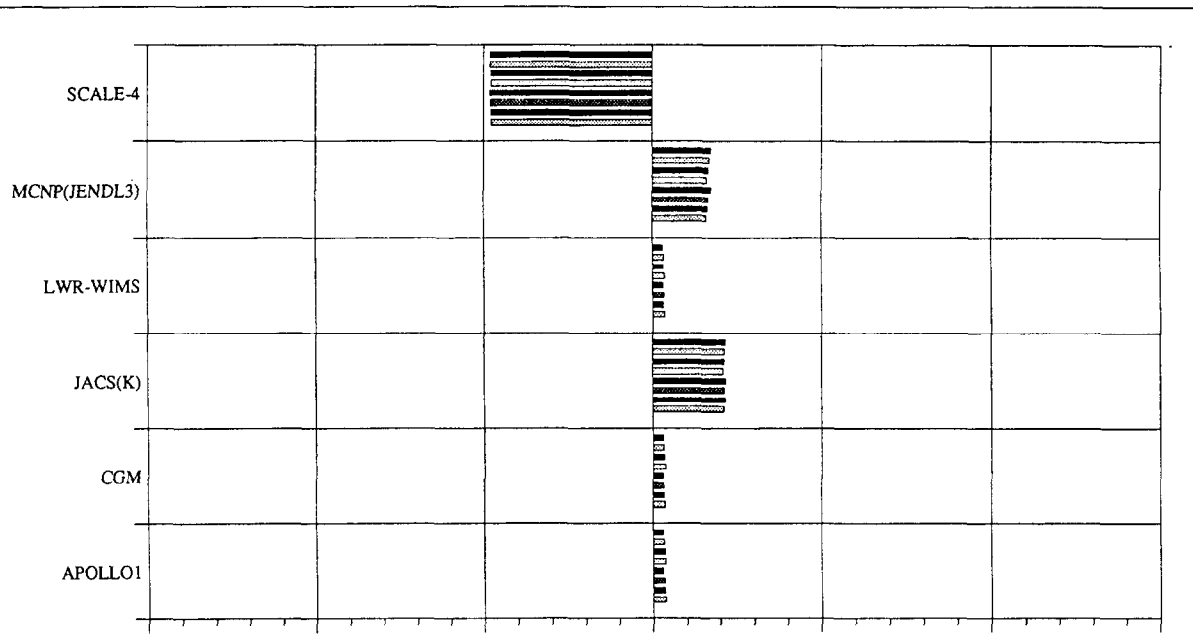

Fig. A3.44 Difference of Pu-238 Nue-Value from Average

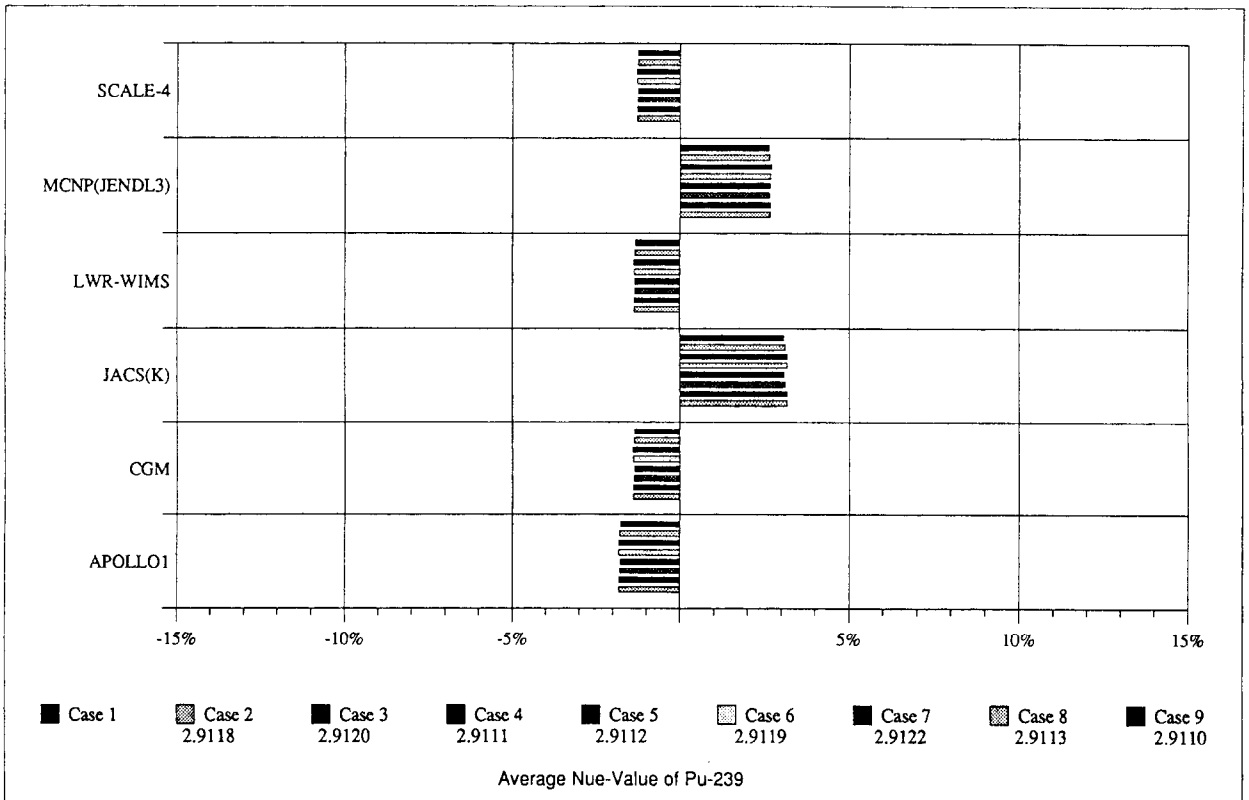

Fig. A3.45 Difference of Pu-239 Nue-Value from Average

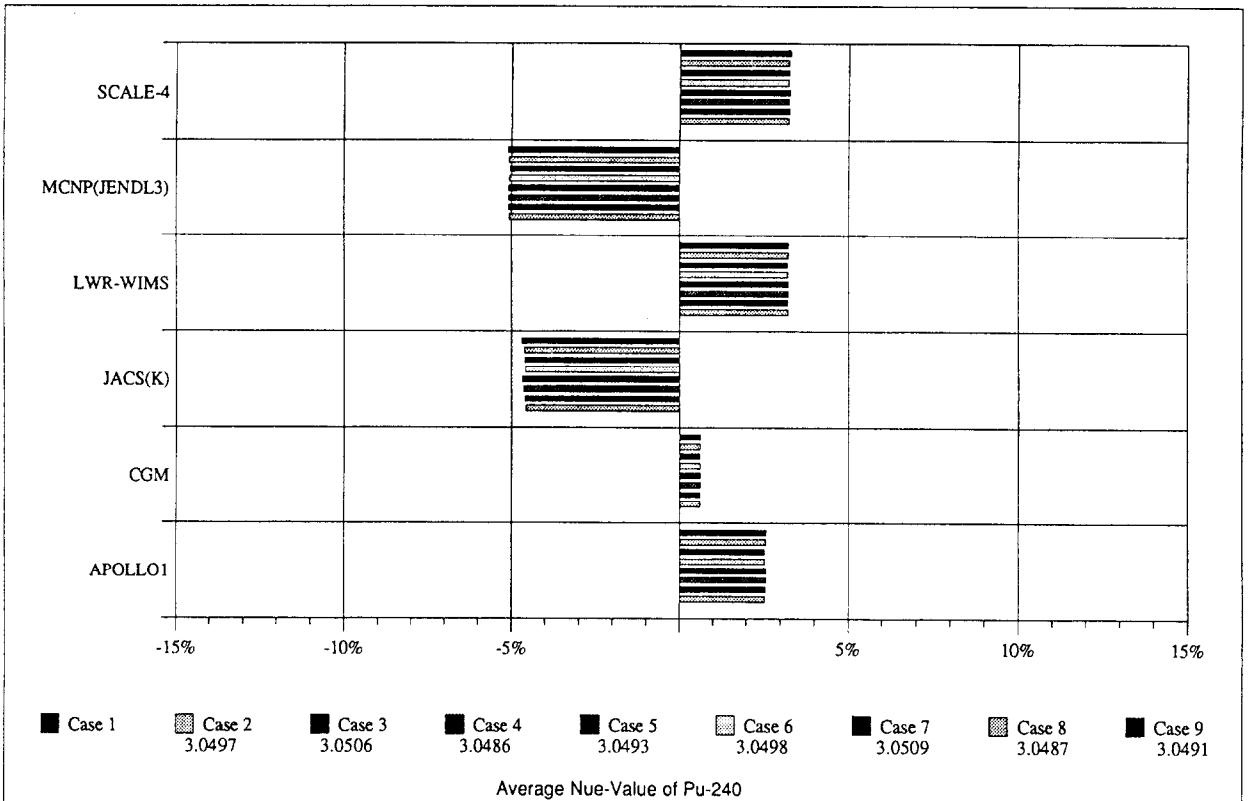

Fig. A3.46 Difference of Pu-240 Nue-Value from Average

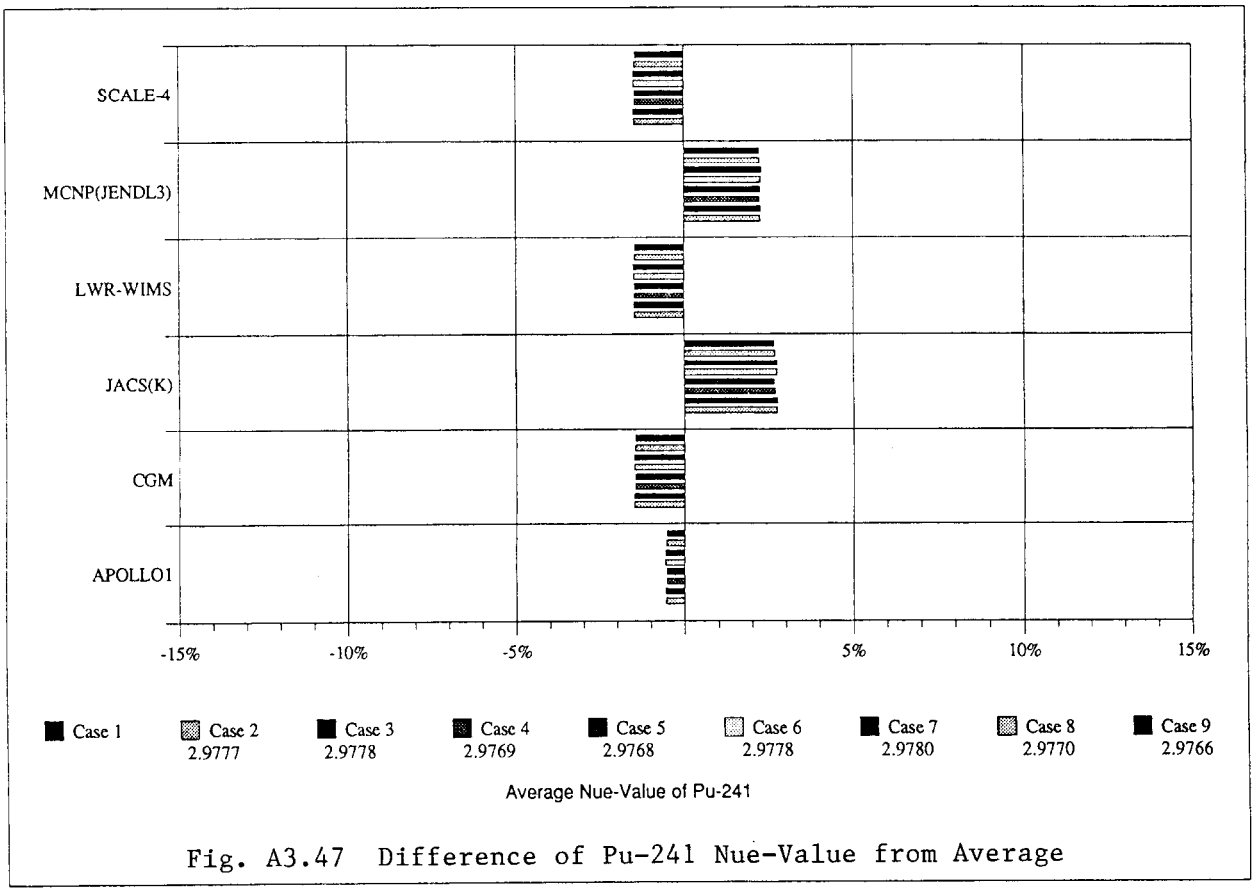

Fig. A3.47 Difference of Pu-241 Nue-Value from Average

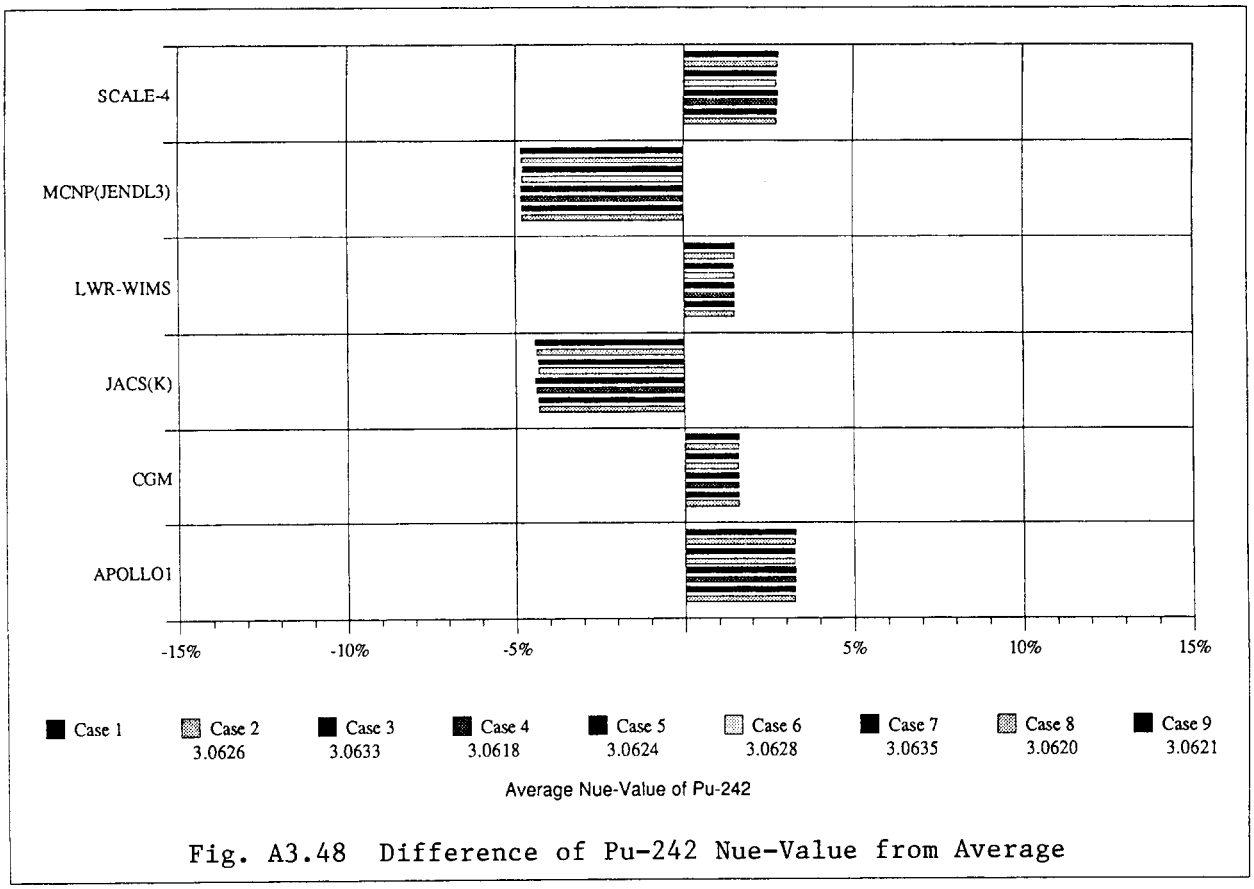

Fig. A3.48 Difference of Pu-242 Nue-Value from Average

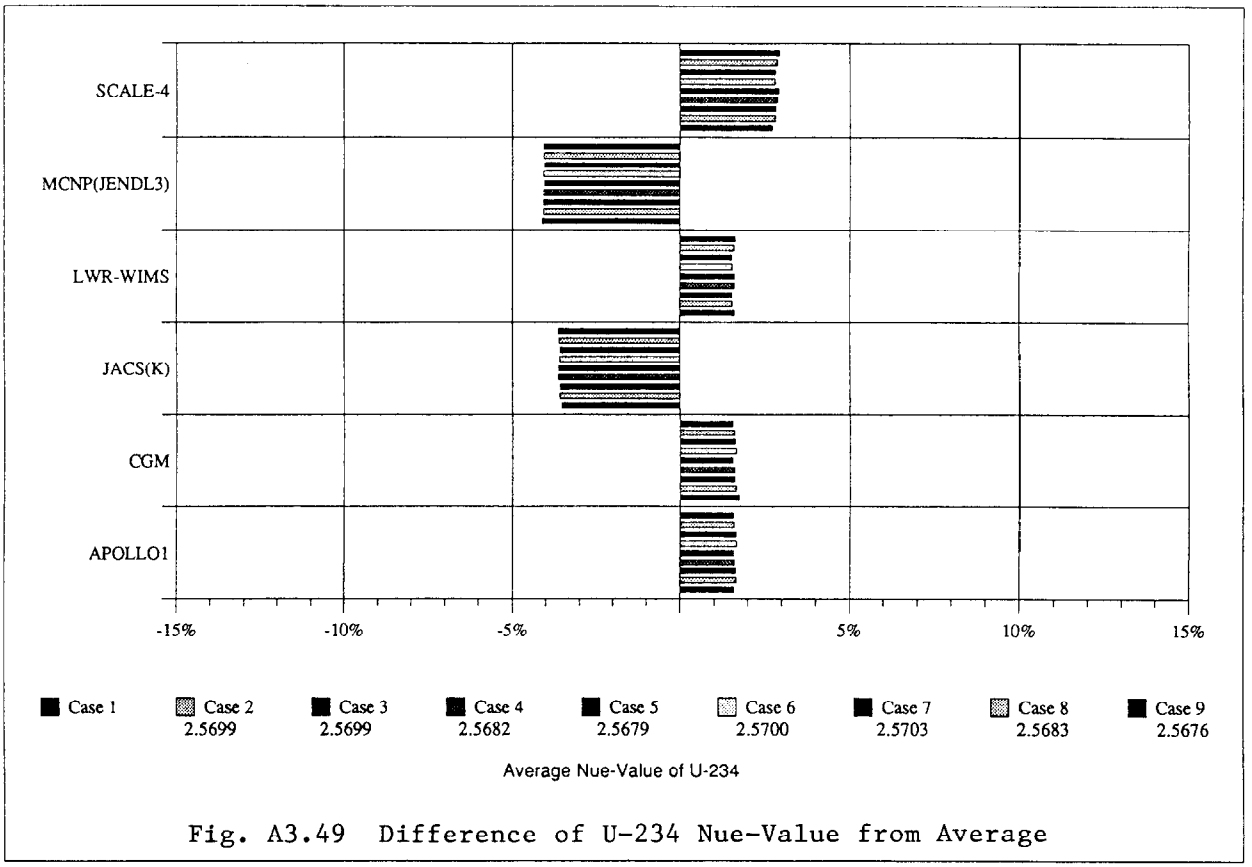

Fig. A3.49 Difference of U-234 Nue-Value from Average

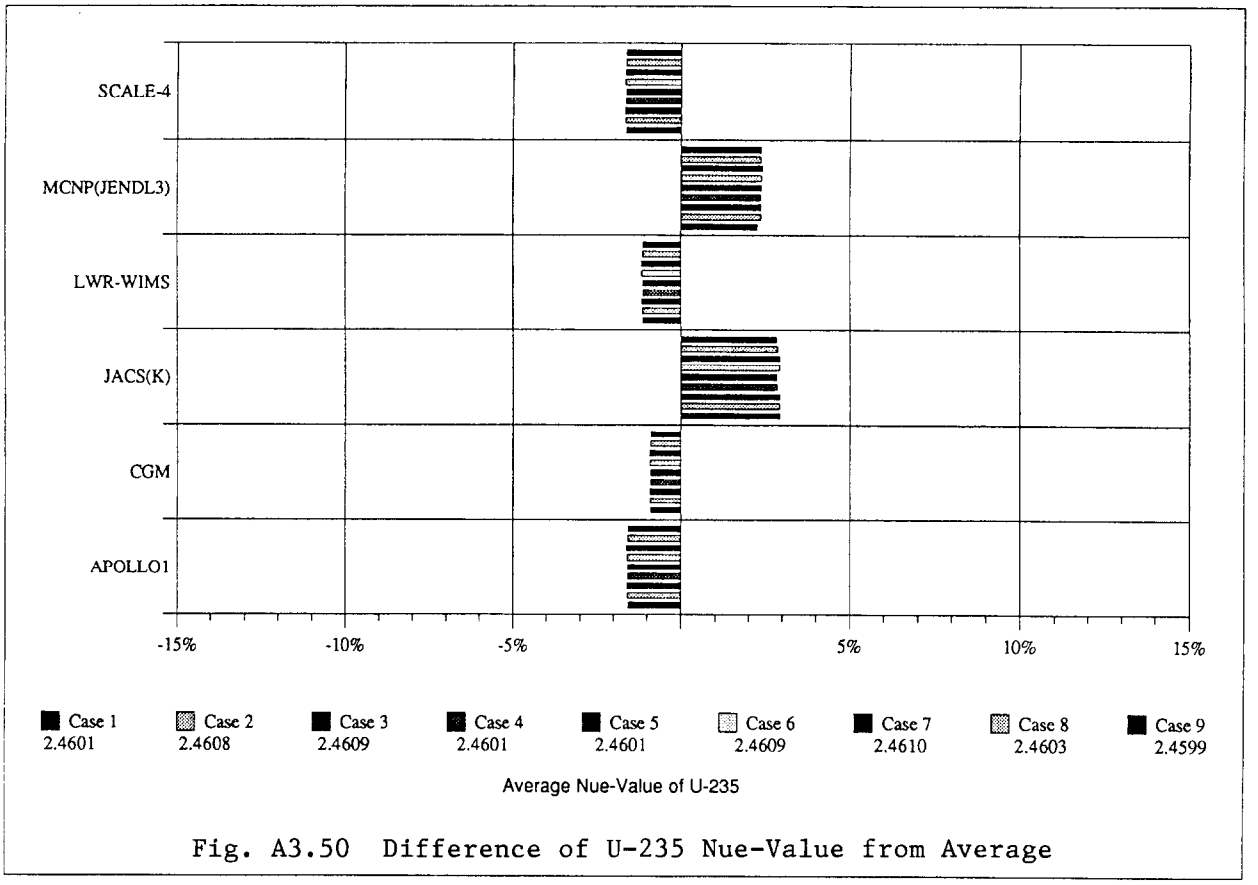

Fig. A3.50 Difference of U-235 Nue-Value from Average

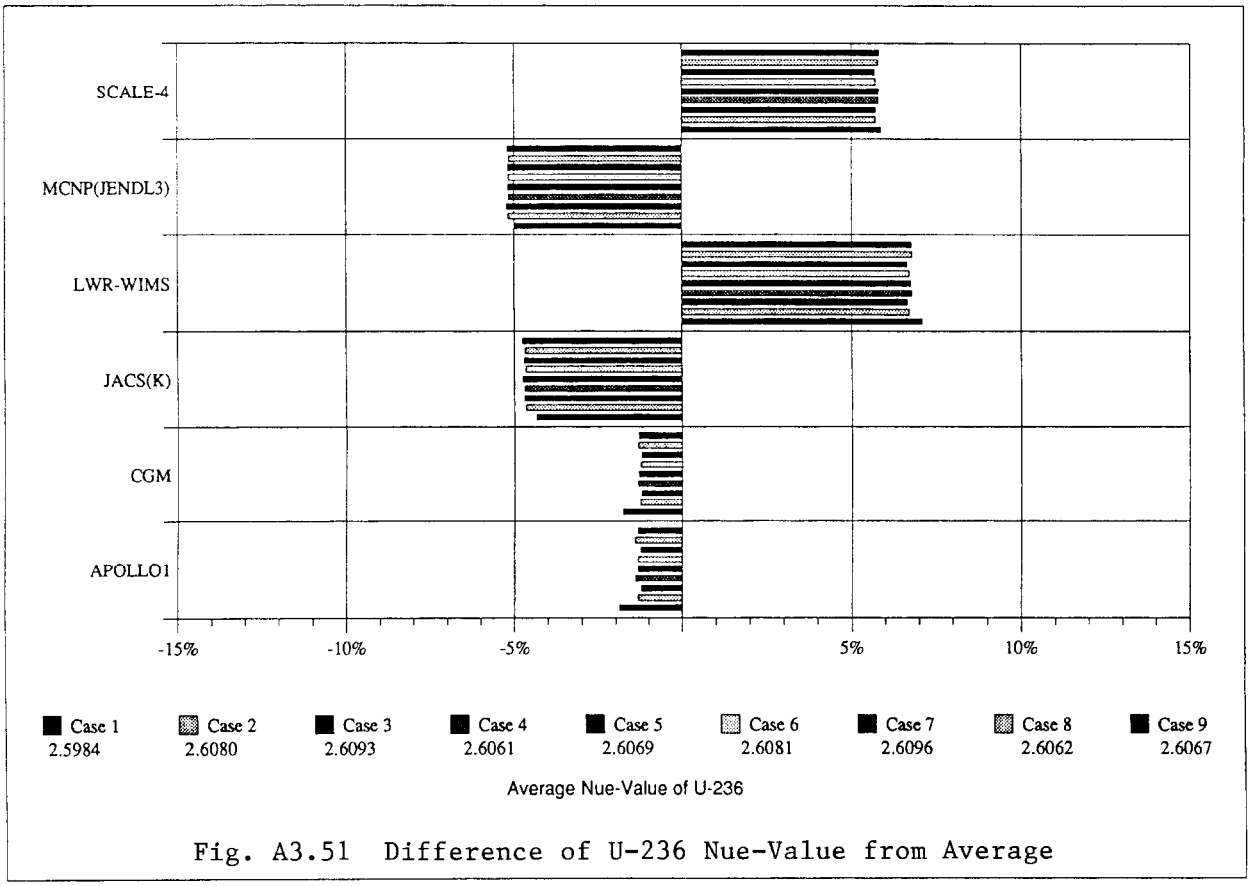

Fig. A3.51 Difference of U-236 Nue-Value from Average

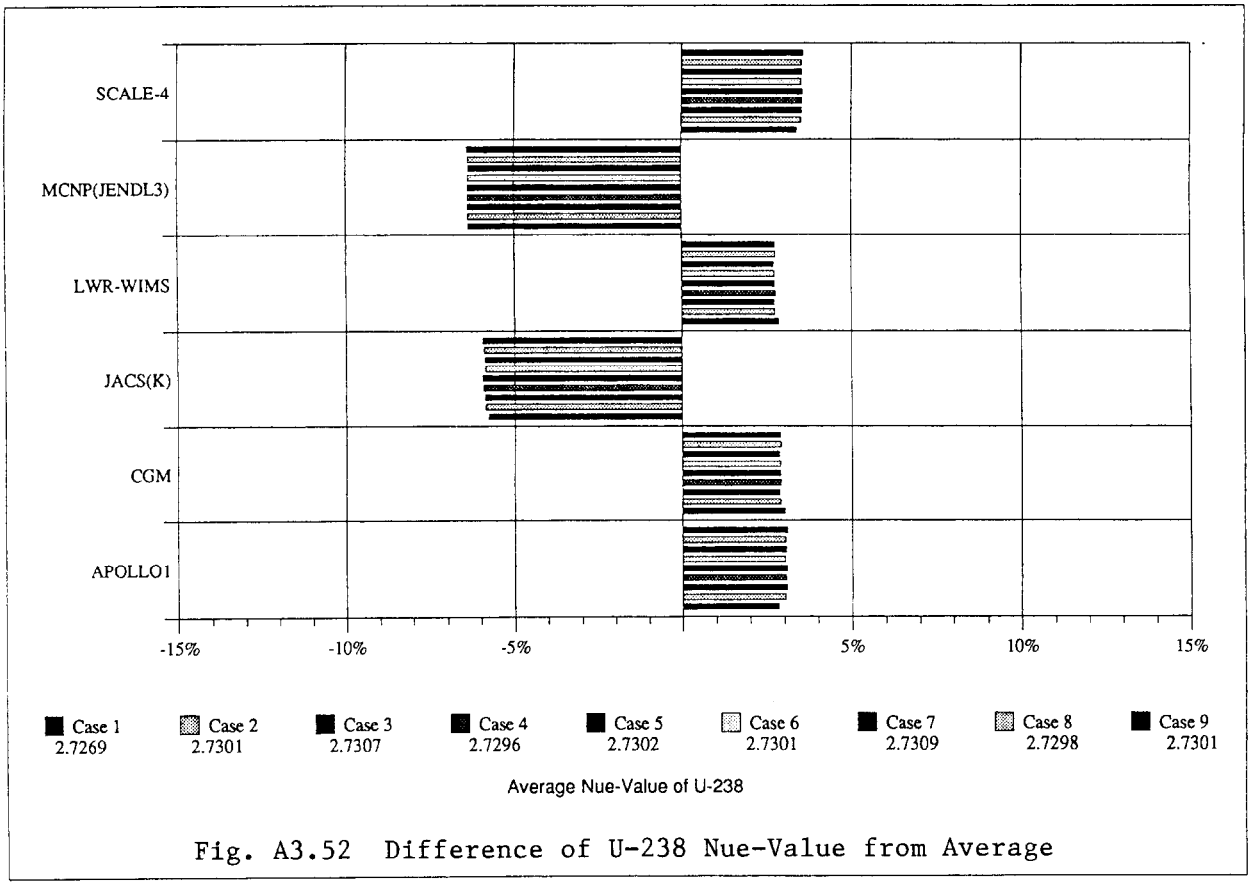

Fig. A3.52 Difference of U-238 Nue-Value from Average

Appendix 4 On Quality Assurance by Jim Stewart

Quality Assurance was discussed at the meetings, with particular regard to the effect on use of the inter comparison for validation purposes (comparison against an independently developed calculation method is one means of validation). Although details of the application of quality assurance by individual participating organizations is not recorded in this report, this does not indicate that quality assurance was not being applied at the time of the inter comparison. Further information may be provided on quality assurance in a report of a later phase of this work. For those who wish to use this phase for validation purposes the following guidance is given. In order for the inter comparison of codes to be of use as a means of validating calculation methods against each other it is necessary to know details of quality assurance applied by each code controller and/or participant. Where this is not known it will prove difficult to make use of the work to satisfy those regulators who require a high degree of quality assurance in the area of criticality safety. Validation can be seen as the foundation on which calculation studies are built. If quality assurance is lacking at the validation stage it will result in acceptance problems at all further stages of a project with regard to demonstrable quality assurance. The minimum information required is :

A. For the inter comparison,

1 . It must be shown that the input information is complete and controlled. This includes the problem specification.
2. It must be shown that the computer and all software used is controlled. This could involve the running of standard test cases.
3. It must be shown that the methods used in assessment are controlled. This will involve working to approved procedures and the creation of assessment records.
4. It must be shown that the output of information is controlled and verifiable. This will include collating of results.

B, For the validation.

1 . Those procedures identified below (*).

The information which should be requested from the participant should include:

1 . Reference to the organization's quality assurance system including its top tier document, and confirmation that it was in place and fully functioning.
2. Reference to the applicable national or international QA standard.
3. Reference to work procedures.*
4. Reference to document control procedures.*
5. Reference to computer hardware and software control procedures.*
6, Reference to any instructions from organizations delegating the work to sub-contractors, along with a reference to the delegating organizations QA arrangements.

国際単位系 (SI) と換算表

表 1　SI基本単位および補助単位

量	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
平面角	ラジアン	rad
立体角	ステラジアン	sr

表 2　SIと併用される単位

名称	記号
分，時，日	min，h，d
度，分，秒	°，′，″
リットル	l，L
トン	t
電子ボルト	eV
原子質量単位	u

1 eV = 1.60218 × 10^{-19} J

1 u = 1.66054 × 10^{-27} kg

表 5　SI接頭語

倍数	接頭語	記号
10^{18}	エクサ	E
10^{15}	ペタ	P
10^{12}	テラ	T
10^{9}	ギガ	G
10^{6}	メガ	M
10^{3}	キロ	k
10^{2}	ヘクト	h
10^{1}	デカ	da
10^{-1}	デシ	d
10^{-2}	センチ	c
10^{-3}	ミリ	m
10^{-6}	マイクロ	μ
10^{-9}	ナノ	n
10^{-12}	ピコ	p
10^{-15}	フェムト	f
10^{-18}	アト	a

表 3　固有の名称をもつSI組立単位

量	名称	記号	他のSI単位による表現
周波数	ヘルツ	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
エネルギー，仕事，熱量	ジュール	J	N·m
工率，放射束	ワット	W	J/s
電気量，電荷	クーロン	C	A·s
電位，電圧，起電力	ボルト	V	W/A
静電容量	ファラド	F	C/V
電気抵抗	オーム	Ω	V/A
コンダクタンス	ジーメンス	S	A/V
磁束	ウェーバ	Wb	V·s
磁束密度	テスラ	T	Wb/m^2
インダクタンス	ヘンリー	H	Wb/A
セルシウス温度	セルシウス度	℃	
光束	ルーメン	lm	cd·sr
照度	ルクス	lx	lm/m^2
放射能	ベクレル	Bq	s^{-1}
吸収線量	グレイ	Gy	J/kg
線量当量	シーベルト	Sv	J/kg

表 4　SIと共に暫定的に維持される単位

名称	記号
オングストローム	Å
バーン	b
バール	bar
ガル	Gal
キュリー	Ci
レントゲン	R
ラド	rad
レム	rem

1 Å = 0.1 nm = 10^{-10} m

1 b = 100 fm^2 = 10^{-28} m^2

1 bar = 0.1 MPa = 10^5 Pa

1 Gal = 1 cm/s^2 = 10^{-2} m/s^2

1 Ci = 3.7 × 10^{10} Bq

1 R = 2.58 × 10^{-4} C/kg

1 rad = 1 cGy = 10^{-2} Gy

1 rem = 1 cSv = 10^{-2} Sv

（注）

1. 表1－5は「国際単位系」第5版，国際度量衡局 1985年刊行による。ただし，1 eVおよび1 uの値はCODATAの1986年推奨値によった。
2. 表4には海里，ノット，アール，ヘクタールも含まれているが日常の単位なのでここでは省略した。
3. barは，JISでは流体の圧力を表わす場合に限り表2のカテゴリーに分類されている。
4. EC閣僚理事会指令ではbar，barnおよび「血圧の単位」mmHgを表2のカテゴリーに入れている。

換　算　表

力	N(=10^5 dyn)	kgf	lbf
	1	0.101972	0.224809
	9.80665	1	2.20462
	4.44822	0.453592	1

粘度　1 Pa·s(N·s/m^2) = 10 P(ポアズ)(g/(cm·s))

動粘度　1 m^2/s = 10^4 St(ストークス)(cm^2/s)

圧力	MPa(=10 bar)	kgf/cm^2	atm	mmHg(Torr)	lbf/in^2(psi)
	1	10.1972	9.86923	7.50062 × 10^3	145.038
	0.0980665	1	0.967841	735.559	14.2233
	0.101325	1.03323	1	760	14.6959
	1.33322 × 10^{-4}	1.35951 × 10^{-3}	1.31579 × 10^{-3}	1	1.93368 × 10^{-2}
	6.89476 × 10^{-3}	7.03070 × 10^{-2}	6.80460 × 10^{-2}	51.7149	1

エネルギー・仕事・熱量	J(=10^7 erg)	kgf·m	kW·h	cal(計量法)	Btu	ft·lbf	eV
	1	0.101972	2.77778 × 10^{-7}	0.238889	9.47813 × 10^{-4}	0.737562	6.24150 × 10^{18}
	9.80665	1	2.72407 × 10^{-6}	2.34270	9.29487 × 10^{-3}	7.23301	6.12082 × 10^{19}
	3.6 × 10^6	3.67098 × 10^5	1	8.59999 × 10^5	3412.13	2.65522 × 10^6	2.24694 × 10^{25}
	4.18605	0.426858	1.16279 × 10^{-6}	1	3.96759 × 10^{-3}	3.08747	2.61272 × 10^{19}
	1055.06	107.586	2.93072 × 10^{-4}	252.042	1	778.172	6.58515 × 10^{21}
	1.35582	0.138255	3.76616 × 10^{-7}	0.323890	1.28506 × 10^{-3}	1	8.46233 × 10^{18}
	1.60218 × 10^{-19}	1.63377 × 10^{-20}	4.45050 × 10^{-26}	3.82743 × 10^{-20}	1.51857 × 10^{-22}	1.18171 × 10^{-19}	1

1 cal = 4.18605 J (計量法)
= 4.184 J (熱化学)
= 4.1855 J (15 °C)
= 4.1868 J (国際蒸気表)

仕事率　1 PS (仏馬力)
= 75 kgf·m/s
= 735.499 W

放射能	Bq	Ci
	1	2.70270 × 10^{-11}
	3.7 × 10^{10}	1

吸収線量	Gy	rad
	1	100
	0.01	1

照射線量	C/kg	R
	1	3876
	2.58 × 10^{-4}	1

線量当量	Sv	rem
	1	100
	0.01	1

（86年12月26日現在）