

BURNUP-CREDIT CRITICALITY BENCHMARK

**Part 1. Simple PWR Spent Fuel Cell
(Problem Specification)**

by

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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

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	Am243 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.

The absorption reaction rate (A_i), the production reaction rate (P_i) and the neutrons per fission of nuclide i (F_i) are defined as follows;

$$A_i = \frac{\iint \Sigma_a^i \phi \, dE \, dV}{V},$$

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

$$P_i = \frac{\iint \nu^i \Sigma_f^i \phi \, dE \, dV}{V}$$

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

$$F_i = \frac{\iint \nu^i \phi \, dE \, dV}{V}$$

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

i : nuclide, E : energy, V : volume

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)

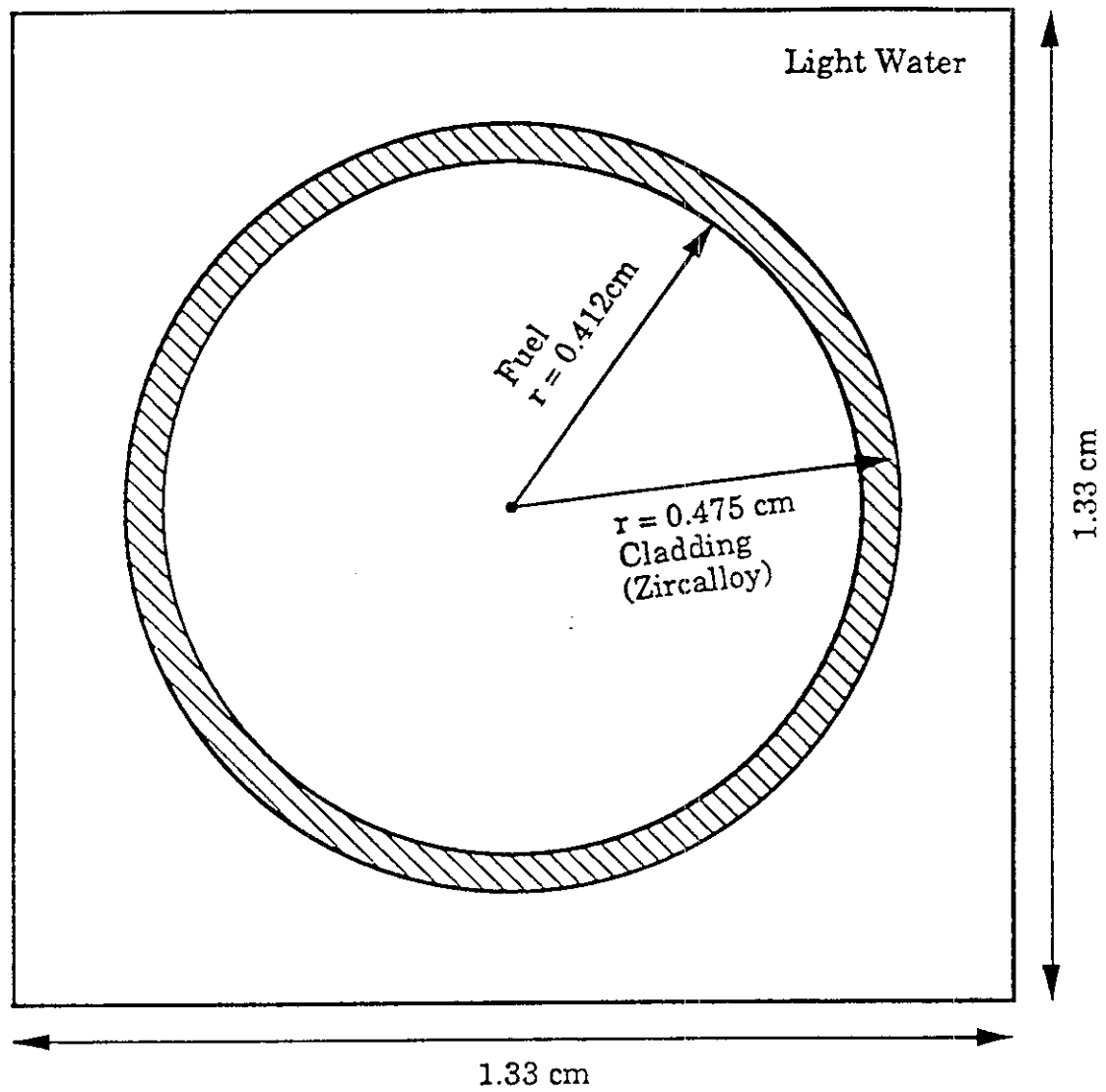


Fig. 1 Cell Configuration

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
Pellet 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, K	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
0	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
0	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

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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
0	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-95	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 COOLING 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
GD-155	4.7429E-09

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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	3.0687E-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

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BURNUP-CREDIT CRITICALITY BENCHMARK

Part I-B. Isotopic Prediction

(Problem Specification)

by

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November 1992

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Part I-B. Isotopic Prediction

(Problem Specification)

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November 1992

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BURNUP-CREDIT CRITICALITY BENCHMARK

Part I-B. Isotopic Prediction (Problem Specification)

by

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Objective:

The objective of this part of the benchmark is to check the accuracy of the depletion codes used to predict the isotopic composition of the fuel as a function of burnup.

Background:

Evaluating the reactivity of a spent fuel system has two fundamental requirements: (1) predicting the isotopic composition of the spent fuel and (2) analyzing the system reactivity based on these isotopics. Part I and Part I-A of this benchmark addressed the principal concern of this working group, the criticality calculation. However, the importance of the depletion calculation cannot be ignored. Depletion codes are often utilized to predict radiation source terms for shielding, and decay heat calculations. Validation sources for depletion codes include measurements of decay heat and chemical assay data. Chemical assay data are ideal for validating the ability of the depletion codes to accurately predict the quantities of specific nuclides in spent fuel as they are a direct measurement of these quantities.

Reference 1 describes one of a series of experiments designed to characterize spent fuel for light water reactors. The experiments were performed at the Materials Characterization Center (MCC) at Pacific Northwest Laboratories (PNL) as part of the United States Department of Energy (US/DOE) Office of Civilian Radioactive Waste Management (OCRWM) program. The spent fuel used in these measurements was designated as Approved Testing Material (ATM) - 104. The chemical assay data measured in these experiments are of particular value in validating the isotopic predictions used in burnup credit.

Problem Specification:

This calculational benchmark problem will compare the computed nuclide inventories for a simple pin cell calculation. The fuel and operating specifications given in Tables 1-5 of this problem are based on data given in Ref. 1 and 2 for the Combustion Engineering (CE) 14 x 14 assembly designated as ATM-104. The fuel pin pitch has been modified for the pin-cell calculation in order to represent a fuel-to-moderator ratio equivalent to that in the two dimensional fuel assembly. Table 6 lists the measured data for ATM-104 at three burnups.

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Participants should perform three independent calculations of the spent fuel inventory resulting from the power history data given in Table 3. Results for all nuclides listed in Table 7 should be compiled and forwarded to M. C. Brady either by electronic mail, computer diskette (Macintosh, DOS and tar formats acceptable on 3.5" diskettes, DOS only on 5.25" diskettes) or 8mm tapes written in tar format.

References:

- 1 . R. J. Guenther, et al., Characterization of Spent Fuel Approved Testing Material-ATM-104, Pacific Northwest Laboratory report, PNL-5109-104, Richland, Washington (December 1991).
2. S. R. Bierman, Spent Reactor Fuel Benchmark Composition Data for Code Validation, Proceedings of the International Conference on Nuclear Criticality Safety - ICNC'91, September 9-13, 1991, Oxford, United Kingdom, p. II-113.

Table 1. Physical data for isotopics calculation for PWR pin-cell

Parameter	Data
Type fuel pellet	UO2
Fuel density, g-cm-3	10.045
Rod pitch, cm	1.5586
Rod OD, cm	1.118
Rod ID, cm	0.986
Pellet diameter, cm	0.9563
Active fuel length, cm	347.2
Effective fuel temperature, K	841
Clad temperature, K	620
Clad material	Zircalloy-2 (97.91 wt% Zr, 1.59 wt% Sn, 0.5 wt% Fe)
Water temperature, K	558
Water Density, av, g-cm-3	0.7569
Cycle 1 avg boron concentration	331 ppm

Table 2. Operating history data for isotopic calculation

OPERATING CYCLE	BURN days	DOWN days	BORON %cycle 1
1	306.0	71	100
2	381.7	83.1	141.9
3	466.0	85	152.3
4	461.1	1870	148.8

BURN is the fuel irradiation time in days.

DOWN is the downtime in days between cycles except for cycle 4 where it includes the decay time from reactor to measurement (cooling time) of 1870 days.

BORON is the cycle-average boron concentration as a percent of the cycle 1 concentration.

Table 3. Specific power (kW/kgU) for the three cases.

OPERATING CYCLE	POWER(a) kW/kgU	POWER(b) kW/kgU	POWER(c) kW/kgU
1	17.24	24.72	31.12
2	19.43	26.76	32.51
3	17.04	22.84	26.20
4	14.57	18.87	22.12

(a) Spent fuel sample 1 with a cumulative burnup of 27.35 Gwd/MTU.

(b) Spent fuel sample 2 with a cumulative burnup of 37.12 Gwd/MTU.

(c) Spent fuel sample 3 with a cumulative burnup of 44.34 Gwd/MTU.

Table 4. Initial fuel number densities

Nuclide	Number Density
U-234	6.15165E-06
U-235	6.89220E-04
U-236	3.16265E-06
U-238	2.17104E-02
C-12	9.13357E-06
N-14	1.04072E-05
O	4.48178E-02

Table 5. Cycle 1 coolant number densities

Nuclide	Number Density
H-1	5.06153E-02
O-16	2.53076E-02
B-10	2.75612E-06
B-11	1.11890E-05

Table 6. Fuel radiochemical analyses results, mg/g fuel (UO₂), from Ref. 1.

Nuclide	Cumulative burnup		
	27.35 GWd/MTU	37.12 GWd/MTU	44.34 GWd/MTU
U-234	1.6 E-01	1.4 E-01	1.2 E-01
U-235	8.47 E+00	5.17 E+00	3.54 E+00
U-236	3.14 E+00	3.53 E+00	3.69 E+00
U-238	8.425E+02	8.327E+02	8.249E+02
Pu-238	1.012E-01	1.893E-01	2.688E-01
Pu-239	4.264E+00	4.357E+00	4.357E+00
Pu-240	1.719E+00	2.239E+00	2.543E+00
Pu-241	6.812E-01	9.028E-01	1.020E+00
Pu-242	2.886E-01	5.761E-01	8.401E-01
Np-237	1.89 E-04*	2.51 E-04*	3.31 E-04*
Am-241	8.56 E-01*	1.18 E+00*	1.31 E+00*
Se-79	4.55 E-05*	6.036E-05*	6.49 E-05*
Sr-90	4.59 E+01*	5.9 E+01*	6.58 E+01*
Tc-99	9.59 E-03*	1.23 E-02*	1.35 E-02*
Sn-126	1.25 E-04*	1.82 E-04*	2.2 E-04*
Cs-135	4.16 E-04*	4.59 E-04*	4.95 E-04*
Cs-137	6.71 E+01*	9.01 E+01*	1.09 E+02*

* These values are in mCi/g fuel (UO₂)

Table 7. Nuclides whose isotopic concentrations will be compared

Actinides	Fission Products
U-234	Mo-95
U-235	Tc-99
U-236	Ru-101
U-238	Rh-103
Pu-238	Ag-109
Pu-239	Cs-133
Pu-240	Sm-147
Pu-241	Sm-149
Pu-242	Sm-150
Am-241	Sm-151
Am-243	Sm-152
Np-237	Nd-143
	Nd-145
	Eu-153
	Gd-155

Table 8. Data to be included in results

- 1 Date
- 2 Institute
- 3 Participants
- 4 Computer code(s)
- 5 Data library identification, origin, description
(includes fission yields and decay data as well as cross-section data).
- 6 General results for all nuclides included in Table 7. Please list actinides first, followed by fission products in the order given in Table 7. **Nuclide inventories should be given in units of mg/g-fuel, i.e., relative to the initial mass of UO₂ for the uranium and plutonium isotopes and as mCi/g-fuel for all other nuclides (e.g., as in Table 6).

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SUMMARY OF MEETING HELD AT OECD, 17-18 June 1992
and
CONCLUSIONS OF THE NUCLEAR SCIENCE COMMITTEE, 5 October 1992

Introduction

Chairman G.E. Whitesides opened the meeting. For the many newcomers to the group he recalled briefly the history, purpose and achievements of the nuclear criticality safety benchmark group.

The main purpose of the group is to validate computational methods through international exercises.

The composition of the group has changed over the years in order to meet the evolving requirements from the criticality safety community. It has started as a group overseen by CSNI; during this phase criticality codes for spent LWR fuel transport containers and for large arrays of packages of fissile materials were investigated.

The next phase addressed criticality of fuel undergoing dissolution. Because more physics understanding is demanded in modelling these problems the group was transferred to NEACRP.

The work carried out in these two phases resulted in several reference reports and publications at conferences.

The new phase concerning burnup credit criticality was discussed during a short meeting held at the international conference on nuclear criticality safety (Oxford, September 1991) and the resulting work plan was submitted to the NEACRP. Following the recommendations of NEACRP the newly established Nuclear Science Committee (NSC) endorsed this activity.

Burnup credit criticality moves ahead of the historically pessimistic criticality calculation approach in which fresh fuel is assumed to be present rather than spent fuel. This new approach shows how large the safety margins are and how plant or storage efficiency can be achieved together with a decrease of operational costs. The end goal is to find an approach which balances the two requirements of reducing costs and optimal safety.

The work plan is divided into 3 phases.

- The first looks at the effects of first order:
the effect of taking into account burnup and build-up of major and minor actinides, the effect of presence of major and minor fission products.
- The second addresses the effects of spatial distribution of burnup. This is particularly important to verify whether or which simplifications are conservative; the effects observed here are of second order.
- The third phase addresses other special questions: it maybe a revisiting of Phase I and II for fuel other than for PWRs (Examples: BWR, MOX, etc.)

The objectives of this work are the intercomparison and assessment of quality of analysis computer codes, methods and data used in burnup credit criticality calculations.

Participants

The list of participants is provided as the last section. Not all were able to attend the meeting, but will exchange their views via correspondence.

Agenda

The agenda was adopted without modifications.

Discussion of Phase I - Simple PWR Fuel Cell

The specification of the benchmark concerned an infinite array of PWR spent fuel cells in which 12 actinides and 15 among the most important fission products were considered. Makoto Takano reviewed the submitted results. For Phase I, 23 sets of results were submitted from 19 institutions in 11 countries. Additional institutions expressed the intention to participate in Phase II. France has participated with only one set of results although several were calculated, which were all highly consistent with each other. The results were presented both in tabular and in graphical form. The first part displayed multiplication factors, the second the different group structures and the energy spectra, parts 3-6 displayed absorption rates production rates and $\bar{\nu}$. Participants have each in addition commented their results. Results obtained with modelling not strictly conforming with the specification were identified. Some discrepancies may still be due to the fact that not all participants have submitted the results in the strict format requested, so that some data provided may be interchanged. It is important that these are identified so that a meaningful comparison can be made.

The following was observed for the results submitted by the time of the meeting:

the effect of burnup at 30 GWd/t results in a reduction of reactivity of 18 percent and of 24 percent at 40 GWd/t. This includes the effect of fission products which amounts approximately to 7 percent (15 nuclides). The effect of cooling from 1 to 5 years reduces reactivity by another 2 percent.

As far as the consistency of the results are concerned one observes a standard deviation of 500 pcm for the cases with no fission products. This is of the same order as observed in other benchmark studies concerning criticality calculations. However for the cases with fission products this discrepancy is much higher. This is an indication that there is a need to give a closer look at the fission product cross sections used. As previously observed in other criticality studies, also in this case we find that calculations using ENDF/B-IV data predict uniformly a lower reactivity when compared with those using other evaluations. Not all participants have used the same approach in resonance self shielding nor have they applied it to the same set of isotopes. This is an additional source of discrepancy. The final report will show the effect of using resonance self shielding deterministic methods as compared to continuous energy Monte Carlo.

It was suggested that for describing the variation of reactivity it would be better to use the logarithmic formula $\log(k_f/k_i)$ in the tables of the final report (suffixes f and i stand for final and initial).

The chairman expressed thanks to M. Brady and M. Takano for their effort in providing the specifications of this benchmark and to M. Takano in particular for the excellent work in summarizing the results.

**Further Actions decided for Phase I
of the Burnup Credit Criticality Benchmark (BUC)**

During the meeting, held on 17th and 18th June, it was decided to structure Phase I into two parts: one, called I-A, which extends Phase I of the BUC benchmark to include the following additional calculations:

Phase I-A: 4 additional cases as outlined in the enclosed annex (M. Brady).

Purpose: assess the contribution of major and minor actinides as well as the major and minor fission products in the reduction of reactivity for burned PWR fuel.

Actions:

- * The results should be sent to M. Takano following the instructions provided in NEACRP-L-337.
- * Participants are asked to check whether their results as displayed in the tables and graphs summarising the results of all participants (Parts 1-6) are correctly represented. It was noted that results not provided in the exact sequence as described in NEACRP-L-337 may in fact have been processed automatically, assuming the specified sequence. Results not conforming to that sequence should be resubmitted in the right format.
- * Participants should provide additional written information on how they have carried out calculations, with the aim of shedding light on the summary tables and to facilitate interpretation of discrepancies. All deviations from the specification in NEACRP-L-337 must be reported. For example, participants should provide information as to which nuclides were resonance self shielded in their calculation and the method they have used. In addition as the composition of Zircalloy was not provided in the specification, participants should also specify the composition they have used. Note the incorrect specification of $\bar{\nu}$ - if necessary submit corrected results.
- * The results for the additional cases and revised results including comments must reach M. Takano by the end of August 1992.
- * M. Takano agreed to revise tables and graphs with additional or new results and collect the comments, to form a first draft of the text. Participants willing to write part of the text should let the Secretariat know so that it can be coordinated. E. Whitesides and M. Brady have expressed their availability to write parts of it. The report should include some statement describing the validation and QA of the codes used. It was agreed that results not conforming to the specification will not be used to calculate averages or standard deviations because they would bias the conclusions.

Drafts of the final report will be sent to participants for comments and approval. Those who feel uncomfortable with the publishing of their results should say so and state whether their results should be withdrawn. Participants should refrain from publishing the results of other participants without their approval. The first draft

will be circulated by the end of 1992. The status of the benchmark plus the proposed subsequent phases will be reported to the NSC on 3rd and 4th November 1992.

The final report containing the analysis and conclusions of Phase I and I-A is scheduled for the beginning of 1993.

The overall coordination of the publication for Phase I-A is carried out by M. Takano with support from the Secretariat.

Phase I-B: Isotopic Prediction (M. Brady)

Objective: to check the accuracy of depletion codes.

This part of the benchmark was renamed from Phase II-A to I-B. The purpose of Phase I is in fact to identify the origin of possible discrepancies in the different computational paths. I-A looks at discrepancies due to different cross section sets and their use. I-B looks at the other components: the isotopic compositions at different burnups. Predictions of isotopic compositions will be primarily dependent on the cross-section and fission yield data utilized in the calculations.

There was an extended discussion as to the best method for accomplishing the objective of Phase I-B. Published results for the chemical assay of spent fuel assemblies and fuel pins do exist. However there were concerns that calculations to model the experiments would be extremely complex even if all the necessary detailed parameters were included in the published reports, which is itself unlikely. In order to model these experiments with a large degree of precision could require complex codes and data to represent the fuel exposure (including in-core fuel movement) and the chemical separation and analysis. A suggestion was made to develop a simple calculational comparison problem that might be loosely tied to a published experiment. It was also noted that although prediction of isotopic composition is important, it is not the main objective of this study. M. Brady shall prepare a specification, based on experiments, which will satisfy the need of precision in the best possible way, but with necessary simplification due to lack of data. It will be either a calculation for a PWR assembly or fuel pin. The detailed specification and time schedule will be prepared in the autumn. The coordination of Phase I-B will be carried out by M. Brady.

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 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.

Phase II

Phase II aims at evaluating the effect of the axial burnup distribution. Both uniform and non-uniform burnup distributions will be considered as a function of initial enrichment, burnup, and cooling time in a two-dimensional infinite array of fuel pins and in a realistic geometry (generic 4 assembly truck cask). The final specification of this benchmark will be prepared by Makoto Takano of JAERI. The isotopic composition, including that associated with the axially distributed burnup, will be included in the problem specification and will be calculated by ORNL. A. Santamarina from Cadarache is providing the experimental axial profiles for a range of burnups from approximately 10 MWd/t to 60 MWd/t to be used in the calculation of the isotopic composition for the Phase II specification. The profiles are based on measurements of PWR fuel obtained from French utilities.

This phase will look primarily at axial end effects. These effects are particularly marked when fission products are taken into account. For low burnup this effect is difficult to see because of possible reactivity inversion burnup effects.

The investigation of radial burnup effects was also discussed and it was proposed that (1) the radial burnup profile is calculated, (2) its reactivity is estimated and (3) the radial migration of fission products is considered.

The preparation for Phase II has now been started.

Phase III

The possibility of a third phase was briefly discussed; in the course of the Phase II the need may be identified to investigate burned fuels other than for PWRs such as from BWRs or MOX fuels. A proposal for this further phase will be made to the NSC only if such a need is identified.

Discussion at the Third NSC Meeting

At the third meeting of the Nuclear Science Committee in October, Dr. T. Asaoka, in charge of keeping the liaison between the group and the NSC, presented a status report concerning Phase I and I-A which included further results received after the June burnup credit meeting. An improved agreement among participants' results was observed.

The plan for Phase I-B and II were presented. The NSC approved both the extension of Phase I and the work outlined for Phase II.

Doubts were expressed however as to the need for investigating radial burnup effects and it was suggested that a real need for it should be assessed by one laboratory first before it is proposed to the full group.

Phase III should be started only if clear objectives have been identified and if there is a real interest in participating countries. In any case, such a proposal would be reviewed by the NSC first.

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Further Developments

Phase I-B

The specifications for Phase I-B have now been finalized by M. Brady. Participants are asked to predict the isotopic composition of the fuel as a function of burnup for a simple pin cell; as in Phase I-A 12 actinides and 15 fission products are considered. These predictions will be compared against those obtained from fuel radiochemical analyses carried out at the Material Characterization Center of PNL(USA) for many of the isotopes considered in this benchmark.

The detailed specification was distributed as NEA/NSC/DOC(92)10. According to established deadlines a draft final report should be ready for June 1993. A status report will be provided to NSC at the next Meeting in May.

Phase II

The draft specification is being finalised and includes contributions from A. Santamarina, M. Brady and M. Takano. It is planned to distribute it to participants for comments at the beginning of 1993. A first summary of the results provided by participants will be presented at the next burnup credit meeting with the aim of resolving existing discrepancies and assessing the performance of the codes used on a realistic case. The overall coordinator of this work will be M. Takano from JAERI.

Next Meeting

Because of unforeseen delays in preparing the isotopic axial compositions for Phase II, it is anticipated that the meeting scheduled for beginning of June 1993, will be postponed to September 1993. At this meeting participants will compare and discuss the results of Phase II and resolve discrepancies observed among the different solutions provided.

The results of this phase will be reported to NSC in the autumn meeting of 1993.