NEACRP-L-338

# NEACRP COMPARISON OF CODES FOR THE RADIATION PROTECTION ASSESSMENT OF TRANSPORTATION PACKAGES. SOLUTIONS TO PROBLEMS 5 AND 6

by

Mrs. H.F. LOCKE

Safety Engineering Systems Division
RPSCD Department
AEA Technology
Winfrith

March 1992

| | NAME | SIGNATURE | POSITION | DATE |
|---|---|---|---|---|
| Lead Author | H. F. Locke | H F Locke | Calculator | 25/11/92 |
| Checked | A. F. Avery | A. F. Avery | Section Leader | 9/12/92 |
| Approved | G. J. Lloyd | G J Lloyd | Division Manager | 17.12.92 |

NEACRP-L-338

# NEACRP COMPARISON OF CODES FOR THE RADIATION PROTECTION ASSESSMENT OF TRANSPORTATION PACKAGES. SOLUTIONS TO PROBLEMS 5 AND 6

by

Mrs. H.F. LOCKE

Safety Engineering Systems Division
RPSCD Department
AEA Technology
Winfrith

March 1992

# NEACRP COMPARISON OF CODES FOR THE RADIATION PROTECTION ASSESSMENT OF TRANSPORTATION PACKAGES. SOLUTIONS TO PROBLEMS 5 AND 6

by

Mrs. H.F. LOCKE

Summary

The results for Problems 5 & 6 of the NEACRP code comparison as submitted by six participating countries are presented in summary. These problems concentrate on the prediction of the neutron and gamma-ray sources arising in fuel after a specified irradiation, the fuel being uranium oxide for problem 5 and a mixture of uranium and plutonium oxides for problem 6.

In both problems the predicted neutron sources are in good agreement for all participants. For gamma-rays however there are differences, largely due to the omission of bremsstrahlung in some calculations. This discrepancy occurs at low energies and is not significant for shielding calculations, although it would be important for predicting heating rates in some circumstances.

## 1.0 INTRODUCTION

The Reactor Physics Committee of the Nuclear Energy Agency set up a project in 1985 to exchange information and experience on shielding calculations for the transportation of spent reactor fuel. This took the form of an intercomparison of codes for carrying out such calculations, and it followed the pattern of similar exercises which had been set up for criticality and heat transfer assessments of fuel transport. The proposal (1) contained six theoretical benchmark problems for which participants were invited to submit results together with the methods of the calculation employed. The problems could be divided into two groups; the first four had sources which were defined in terms of their strength and spectra while problems 5 and 6 involved the calculation of the sources from data provided on the fuel assemblies and their radiation histories. The purpose of this present paper is to summarise the results obtained for problems 5 and 6, and to draw conclusions from a comparison of the source calculations. A list of the participants and the references from which their results were taken are given at the end of this report.

## 2.0 DESCRIPTION OF PROBLEMS 5 AND 6

The novel feature of these problems is the calculation of the neutron and gamma-ray sources arising in $UO_2$ and a mixed oxide (MOX) fuel, for problems 5 and 6 respectively. The fuel compositon (table 1), irradiation history (table 2) of the fuel assemblies and all necessary operational reactor data (table 3) are provided. Axial variation of burn-up along the active length of the fuel rods was also to be taken into account and the distribution is shown in table 4.

After the 1987 meeting (2) it was decided that the earlier problems (1-4) had provided adequate information concerning the shielding calculations and that problem 5 and 6 should emphasise the calculation of sources from the spent fuel assembly. At that time the simplified cask model of problem 1 was chosen for those who wished to perform dose calculations using their predicted sources.

## 3.0 METHODS OF CALCULATION

Two codes were used by the participants to calculate source strengths. These were FISPIN (UK) and ORIGEN (USA, Italy, Belgium, Japan and Germany) although not always the same version of the latter.

### 3.1 FISPIN Calculations

The UK calculations were performed using FISPIN (3). This code calculates the production and removal of three groups of nuclides , the fission product group, the actinide group and the structural material group. Modes of production and removal of a nuclide are those of decay and of neutron reaction (of any type including fission). Energies and intensities of the decaying nuclides are calculated for the various modes of decay ie. alpha, beta, gamma rays and spontaneous fission.

For the UK calculations the actinide cross-section data were taken from a burn-up dependent library for a PWR (PWRBUG28) and the decay data were taken from a separate library (PWRDAT4). Fission product data were taken from the UKFPTR4 library containing UKFPDD2 decay data and CROUCH3-I fission yields.

The remaining necessary input data are the atomic density of each heavy isotope and the power for each time step of the history (tables 1 and 2).

### 3.2 ORIGEN Calculations

#### 3.2.1 SAS-H/ORIGEN-S

The basic computational tool, used by Parkes et al in the US, for these problems was the Shielding Analysis Sequence 2H (SAS2H) of the SCALE code system (4). This sequence uses the point depletion code ORIGEN-S to calculate burn-up and decay and to produce the radiation source strengths and spectra.

Figure 1 shows the calculation procedure used by SAS2H to produce cross-section data, calculate spent fuel sources and subsequently use the data in a shipping cask shielding analysis. The sequence starts with data describing a particular fuel assembly including compositions, temperatures, geometry and time dependent specific power (tables 1 to 3) . A standard cross-section library (SCALE-2 27n-18g shielding library) was used. BONAMI-S and NITAWL-S are then used to perform resonance self-shielding treatments for nuclides with the appropriate data. Cell weighted cross-sections are computed for a fuel pin cell (part A) using XSDRNPM-S. A second unit cell (part B) is defined to be representative of the fuel assembly. The Part B model has an assembly guide tube or other non-fuel type rod at its centre and is surrounded by a region of the cell-weighted fuel. Cross-sections are computed for this larger unit cell and the sequence invokes COUPLE to update the ORIGEN-S working library with these cell- weighted data. ORIGEN-S uses these cross-section data to compute the number densities of the various nuclides in the fuel as a function of time, and the process is repeated for each step in the specified power history (specific power and

exposure times). In the final pass, after COUPLE has updated the ORIGEN-S working library, an ORIGEN-S depletion case using these burn-up dependent cross-sections is used to calculate the discharge composition of the fuel assembly, and a decay only subcase is used to predict the composition for the cooling time at which the fuel is to be loaded and shipped. This final composition is used in the determination of neutron and gamma-ray sources to be applied in the shipping cask analysis.

In the shielding analysis , cell weighted cross-sections for the fuel zone in the shipping cask are calculated using the BONAMI-S , NITAWL-S, XSDRNPM-S sequence. These fuel zone cross-sections are used in the shipping cask model and the XSDRNPM-S calculated fluxes are then used in XSDOSE to obtain dose-rates.

### 3.2.2 ORIGEN2

Parkes et al have also applied ORIGEN2 to problem 5 for comparison with ORIGEN-S. The ORIGEN2 and ORIGEN-S (used in SAS2H) codes use the same basic matrix exponential expansion method and numerical scheme as the original ORIGEN code (5). The primary differences between ORIGEN-S and ORIGEN2 are the cross-section and fission product yield data used by the two codes. Both codes use essentially identical decay data (half lives, branching ratios etc.) and photon data (energies and intensities of gamma-ray and X-ray spectra).

ORIGEN2 uses burn-up dependent cross-section and fission product yield libraries developed by performing multigroup reactor physics and depletion calculations for typical reactor models instead of carrying out such calculations for the specific reactor. These libraries include a uranium-fuelled PWR using a once-through fuel cycle at a typical burn-up (33GWd/tonne) and at an extended burn-up (50GWd/tonne) (6). The average burnup for problems 5 and 6 was given as 40GWd/tonne. The library used for problem 5 is the PWR library with extended burn-up as there is no provision for reactor problems where the burn-up exceeds the library maximum. The 33GWd/tonne was also used to illustrate the differences that can be expected as a result of improper cross-section data.

ORIGEN-2 calculations were also performed by Renard using the standard data library provided with the code for a PWR fuelled with U235 enriched $UO_2$ and a burn-up of 33GWd/tonne. For problem 6 the ORIGEN-2 cross-section library for a PWR fuelled with Pu enriched $UO_2$ in a self-generating Pu recycle reactor, was used.

### 3.2.3 SAS-2

The burn-up calculation was also performed by Gualdrini et al using ORIGEN in the SAS-2 automatic sequence of the SCALE-2 modular system. SAS-2 does not allow for axial variation.

### 3.2.4 OREST

Calculations of the source strengths were performed by Gewehr using OREST (7) which comprises the HAMMER code for fuel lattice cell simulation to derive the cross-sections and the ORIGEN code for the calculation of the inventories of actinides and fission products. OREST thus generates problem dependent neutron fluxes during the burn-up simulation. The ORIGEN code used in OREST is an enlarged version of ORIGEN-73 (5).

### 3.2.5 ORIGEN2-82

A further version of ORIGEN, ORIGEN2-82, was used by Tanaka.

## 4.0 RESULTS

### 4.1 Problem 5

Table 5 shows the total neutron and gamma-ray source strengths for 5 sub-assemblies. This includes the separate contributions from spontaneous fission and $\alpha$-n reactions for neutrons, and from actinides and decay fission products for gamma-rays.

The agreement for the neutron source is good for spontaneous fission, which provides ~98% of the total and also for $\alpha$-n reactions except for the OREST value which is higher by a factor of ~2. Table 6 compares the actinide contributions to the neutron source for each participant together with decay rates. Table 7 compares the decay rates and neutron yields per decay. The decay rates are in agreement except for Pu240, Cm243 and Cm248 which give negligible contributions to the source. The individual source contributions agree for the dominant isotope Cm244 and in general for the other isotopes although the ORIGEN-S and OREST results have slightly higher values for Pu238 and Pu239.

The agreement for the gamma-ray source is not so good with factors as large as 2.2 between the total source strengths (in gammas/s). The results fall into two groups: ORIGEN-S, ORIGEN2-82 and SAS2H and the rest. (The ORIGEN-S result is ~20%

lower than the others in their group.) Table 8 compares the gamma-ray spectra produced by each calculation and the total source for 0.4-3.0 MeV as well as the complete range. The agreement in the totals over this reduced range is very good and suggests that the inclusion of bremsstrahlung in some libraries, which yields gamma-rays with energies predominantly below 0.4 MeV, may explain the overall differences. Gamma-rays with these low energies do not contribute to dose-rates outside the flask. The bremsstrahlung contribution arrives from the radiative interactions in the fuel of energetic electrons emitted in beta decay.

Table 9 compares the contributions to the gamma-ray source from important nuclides, together with decay rates, with source strengths in gamma/s and MeV/s. There is much better agreement when total energy emitted is compared; however individual values differ widely (eg. Pr144 and Rh106). The decay rates show reasonable agreement (see table 10) and therefore suggest the discrepancies are due to the gamma-ray yield data in the various libraries.

Parkes et al carried out a study to compare contributions from Pr144 and Rh106 for source calculations with and without bremsstrahlung and these are shown in table 11. The group schemes are different but the total source strengths can be compared. The calculations including bremsstrahlung give a source strength ~5 times higher in energy and ~60 times higher in photons for Pr144. The corresponding values for Rh106 are ~2 and ~6. These values are consistent with the differences between the ORIGEN-S and FISPIN results for these nuclides.

A further explanation of differences between ORIGEN-S and SAS2H results, using the same method, may be the effect of group structure for gamma-rays. In ORIGEN-S the 27n-18g group structure of the SCALE-3 library with a lower limit of 0.01MeV was used whilst in SAS2H the 18 group ORIGEN-S structure was used with a lower limit of 0.00MeV.

All the above results correspond to a uniform distribution of burn-up along the length of the fuel assemblies. Table 12 gives the axial distribution of neutron and gamma-ray sources (for 5 sub assemblies) for a range of burn-ups covering the axial variation shown in table 3. As above the neutron sources are in good agreement and show the same axial variation. The gamma-ray source strengths are again different due to the low energy contributions but the relative strengths agree. The relative source strength i.e. the ratio of source strength to the source at mean burn-up (axial peaking=1.0) is similar to the peaking factor for the burn-up itself; however for neutrons there is a marked difference eg. for a peaking factor of 1.33 the relative source is 3.37 and for 0.31 it is 0.01. This means that the axial variation of burn-up cannot be treated by a linear multiplication factor to determine the variation of the neutron source.

## 4.2 Problem 6

Tables 13 to 16 give the source strengths, important contributions to the neutron source and gamma-ray spectrum calculated for problem 6. The neutron source strength for MOX fuel is greater than that for $UO_2$ by almost a factor of 4. As for problem 5, the calculated neutron sources are in agreement but with a ~10% variation and the contributions from spontaneous fission and α-n reaction also agree except for the OREST result for the latter, which is again a factor of 2 higher. The decay rates and source strengths for the important nuclides show some variation apart from the dominant isotope Cm244 (see table 15).

The gamma-ray sources are similar to those in problem 5 and fall into the same two groups according to whether bremsstrahlung is included. (The ORIGEN-S result is again ~20% lower than those from SAS2H probably due to the group structure.) A comparison of the gamma-ray spectrum shows good agreement, for all participants, for the total source above 0.4MeV.

## 5.0 DOSE-RATE CALCULATIONS

Shielding calculations for flasks containing fuel elements with the derived source strengths have been performed by the participants from US, Belgium and Italy. The simplified cask model of problem 1 was used and the source smeared over the flask cavity. Tables 17 to 19 show the results for problem 5 and tables 20 to 22, for problem 6. These are not considered in detail here, as the comparisons of the method of calculation are given in the results for problems 1-4 (8). It is important to note however, the increase in the dose-rates when the axial distribution of sources is used instead of the mean value, particularly for neutrons.

## 6.0 COBALT ACTIVATION IN THE END FITTINGS

Typical thermal neutron flux distributions were specified for end fittings of fuel elements so that sources arising from activation of cobalt-59 could be calculated. These are shown in table 23. The cobalt-59 content of material 1.4541, which makes up the end fittings, was assumed to be 0.1% by weight. This calculation was performed by only Gualdrini et al and Renard and this section summarises their results.

Renard performed a calculation using ORIGEN-2 with the standard library. He found that the photon source due to activation was concentrated around 1.25MeV (Co60 line energies) and a neutron flux of 3.4 x $10^{12}$ neutron/$cm^2$.s at the top end induced a gamma-ray source of about 6.6 x $10^{12}$ photon/s after 2040 days. Similarly

for a neutron flux of 2.4 x $10^{12}$ neutron/$cm^2$.s at the bottom end the gamma-ray source was about 4.0 x $10^{12}$ photon/s.

Gualdrini et al calculated the source from the following equations and looked at the distribution over the top fitting.

$$\frac{dN_{60}}{dt} = \sigma_c N_{59} \Phi - \lambda N_{60}$$

$$N_{60}(T,t) = \frac{\sigma_c}{\lambda} N_{59} \Phi (1 - e^{-\lambda T}).e^{-\lambda T}$$

$$S(z) = f N_{59}(z) \Phi(z)$$

where $f = \sigma_c (1 - e^{-\lambda T}).e^{-\lambda T}$ and a value of 37 barns was used for $\sigma_c$. The results are shown in table 24.

There should be no difficulty in predicting the cobalt activity if the neutron fluxes are provided because the cross-section, half life and decay scheme are well known. Moreover the capture cross-section is small (2 barns) so that burn-up of Co60 is not important. In the specification of problems 5 and 6, however, the definition of thermal flux was not precise so that there is some confusion in its interpretation. If it is assumed to be given in the Westcott convention (9) then the appropriate cross-section for activation is that at 2200m/s, whilst if it is the integrated product of neutron density and velocity then the cross-section must be derived for a Maxwellian spectrum of the appropriate temperature. The latter was not quoted for the problem.

The study of cobalt activation was not pursued further. In the practical situation the major difficulty for the shielding designer is in obtaining calculations of the neutron fluxes in the end fittings of the fuel assemblies during their lives in the reactor, and that was outside the scope of this intercomparison. Codes such as ORIGEN and FISPIN cannot be used to give flux levels in the absence of power ratings.

## 7.0 CONCLUSIONS

Contributions to problems 5 and 6 have been provided by six participants and these involve the use of two codes to calculate source strengths; FISPIN and ORIGEN. Several versions of ORIGEN were used which involved different data libraries and group structures.

Calculations of total neutron source were in good agreement; however there is some variation in the contributions from individual actinides except for the dominant

isotope Cm244. This implies that there are some differences in the data used in the calculation of the build up of actinides. The neutron sources per decay as used in FISPIN and ORIGEN-S are in general agreement for the actinides, although those for $\alpha$,n events in OREST show marked differences (see table 24). The differences in actinide production may be more significant when calculating neutron sources in fuel with irradiation and decay periods where isotopes other than Cm244 are important.

Calculations of the gamma-ray source fall into two categories, those with and without bremsstrahlung. This makes a factor of 2 difference to the total source in photons/s; but if energy is considered there is better agreement. Further comparison showed agreement between photon source strengths for the energy range 0.4 to 3.0 MeV where bremsstrahlung is not significant. Photons with energies below this threshhold would not contribute to dose-rates outside the flask and therefore are not significant for shielding calculations; however they could be important for energy deposition within the basket and radiolysis in water for wet flasks. There was also a 20% difference in source strength (gamma-rays/s) when the lower energy limit was 0.01MeV compared to 0.0MeV for results including bremsstrahlung.

There are differences between results for the contributions to the gamma-ray source which cannot be explained entirely by the effect of bremsstrahlung. In general the decay rates are similar so that the discrepancies are attributed to differences in gamma-ray yield data from the various libraries.

There is no significant difference, other than those already mentioned, in the calculated source strengths from the two methods of calculation i.e. those which carry out detailed cell calculations to provide cross-sections (ORIGEN-S and OREST) or those which use a standard library (FISPIN and ORIGEN2). Use of two different libraries by Parkes et al also has no significant effect. The neutron source strength for problem 5 is ~9% higher when the 50GWd/tonne library is used compared to the 33GWd/tonne library. The gamma-ray source strength is ~3% higher. In fact the 33GWd/tonne library, which was also used by Renard, gives results for the neutron source more in line with the other participants.

The difficulties in predicting the activation of the end fittings arise in practice from the lack of information on the neutron fluxes in these regions, and they have not been pursued in this study.

## REFERENCES


1. Dent W. "An FRG Proposal for an International Intercomparison of codes for Radiation Protection Assessment of Transportation Packages". NEACRP-L-290

2. Avery A.F. "Progress with the NEACRP Intercomparison of codes for the Shielding Assessment of Transportation Packages" NEACRP-A-895

3. Burstall R.F. and Endicott D.A.J. FISPIN User Input fot the FISPIN code AERE-M3508

4. SCALE A modular code system for performing standardized computer analyses for licensing evaluation, vols. 1-3, NUREG/CR-0200, U.S. Nuclear Regulatory Commission (originally issued July 1980, reissued January 1982, Revision 1 issued July 1982, Revision2 issued July 1983, Revision 3 issued December 1984)

5. Bell M.J. ORIGEN-The ORNL isotope generation and depletion code, ORNL-4628, Union Carbide Corp.,Nuclear division, Oak Ridge National Lab., May 1973

6. Croff A.G., Bjerke M.A., Morrison G.W. and Petrie L.M. Revised Uranium-Plutonium Cycle PWR and BFR Models for the ORIGEN Computer code ORNL/TM-6051, Union Carbide Corp., Nuclear division, Oak Ridge National Lab., 1978

7. Hesse U. OREST The Hammer-ORIGEN Burn-up Program System: Method and results, Nuclear Technology Vol. 82 Aug 1988

8. Avery A.F. and Locke H.F. NEACRP Intercomparison of codes for the radiation protection assessment of transportation packages. Solutions to problems 1-4. NEACRP-L-331

## REFERENCES


9. Westcott C.H., Walker W.H. and Alexander T.K.
2nd International Conference on the Peaceful uses of Atomic Energy.
Geneva 16(1958)70


## Contributions in alphabetical order of author


Broadhead B.L., Brady M.C. and Parks C.V
Benchmark Shielding Calculations for the NEACRP working group on shielding assessment of transportation packages.
ORNL/CSD/TM-272 July 1990

La Fuente A. and Renard A.
OECD/NEA Cask Shielding benchmarks. January 1988

La Fuente A. and Renard A.
OECD/NEA Cask Shielding benchmarks. Revised results of problems 5 and 6.
June 1990

Gewehr K.
Intercomparison of Codes for Shielding Assessment of Transportation Packages.
Jan. 1990

Gewehr K.
Intercomparison of Codes for Shielding Assessment of Transportation Packages.
(Additional Information to the results.) Sept. 1990

Gualdrini G.F. and Peerani P.
Some discussions and results on the burn-up calculation discrepancies concerning problem 5 ($UO_2$ fuel).
Letter to A.F.Avery, C.V.Parks and E.Sartori (Dec. 1988)

Gualdrini G.F., Burn K.W., Casalani L. and Peerani P.
Compendium of ENEA Results for Benchmark Problems 1 to 6.
(NEACRP Working group on computer codes for Radiation Protection Assessment of spent fuel transport flasks.)
Paris Feb 28 to March 2 1990

Gualdrini G.F., Casalani L., Monti S. and Peerani P.
International Intercomparison of Codes for Radiation Protection Assessment of Transportation Packages (NEA/CRP).
Updates to ENEA Results (June 1990)

Gualdrini G.F., Burn K.W., Casalani L. and Peerani P.
The ENEA Contribution to the NEACRP International Intercomparison of Codes for Radiation Protection Assessment of Spent Fuel Transport Flasks.
DRAFT ENEA/RT(AMB)()1991

Locke H.F.
NEACRP Intercomparison of Codes for Shielding Assessment of Transport Packages. Solutions to Problems 5 and 6.
RP&SG/HL/P(88)21

Tanaka S. and Sakamoto Y.
Intercomparison of Codes for Shielding Assessment of Transportation Packages.
The Source Strength of $UO_2$ and MOX spent Fuels for problems 5 and 6.
April 1988.

## Table 1. Composition of fresh fuel for problems 5 and 6

**Problem 5 $UO_2$**

| Isotope | % by weight | Atomic Density (atoms/tonne heavy isotopes) |
|---|---|---|
| U-234 | 0.01 | 2.9200E+23 |
| U-235 | 2.86 | 8.3158E+23 |
| U-236 | 0.12 | 3.4743E+24 |
| U-238 | 85.15 | 2.4446E+27 |
| O | 11.86 | - |

**Problem 6 Mixed Oxide Fuel**

| Isotope | % by weight | Atomic Density (atoms/tonne heavy isotopes) |
|---|---|---|
| U-234 | 0.01 | 2.9200E+23 |
| U-235 | 0.59 | 1.7155E+25 |
| U-236 | 0.1 | 2.8953E+24 |
| U-238 | 83.95 | 2.4102E+27 |
| Pu-238 | 0.01 | 2.8710E+23 |
| Pu-239 | 2.7 | 7.7192E+25 |
| Pu-240 | 0.64 | 1.8221E+25 |
| Pu-241 | 0.12 | 3.4023E+24 |
| Pu-242 | 0.02 | 5.6470E+23 |
| O | 11.86 | - |

## Table 2. Irradiation History

| Time (days) | Specific Power (MW/tonne heavymetal) | Burn-up (MWd/tonne heavymetal) |
|---|---|---|
| 300 | 32.0 | 9.60E+03 |
| 40 | 0.0 | 9.60E+03 |
| 300 | 26.0 | 1.74E+04 |
| 40 | 0.0 | 1.74E+04 |
| 300 | 38.0 | 2.88E+04 |
| 40 | 0.0 | 2.88E+04 |
| 300 | 27.3 | 4.00E+04 |

2 years Cooling time

**Operational Data of Model Reactor:**

| | |
|---|---|
| Coolant pressure | 160 bar |
| Average temperature | 300°C |
| Average boron content | 450ppm |

# Table 3. Characteristics of Model Fuel Assembly

## Dimensions

| | | |
|---|---|---|
| total assembly length | 445.0 | cm |
| width over flats | 23.0 | cm |
| total rod length | 393.0 | cm |
| length of active zone | 342.0 | cm |
| length of lower rod plug | 1.8 | cm |
| length of bottom expansion space | 31.4 | cm |
| length of top expansion space | 16.0 | cm |
| length of upper rod plug | 1.8 | cm |
| height of bottom fitting | 23.0 | cm |
| height of top fiting | 27.0 | cm |
| void between botom fitting and rod | 1.0 | cm |
| void between top fitting and rod | 1.0 | cm |
| top number of rod positions | 225 | |
| fuel rods | 210 | |
| control rods | 15 | |
| pitch (square lattice) | 1.53 | cm |
| pellet outer diameter | 0.969 | cm |
| cladding inner diameter | 1.00 | cm |
| cladding outer diameter | 1.15 | cm |
| guide tube inner diameter | 1.25 | cm |
| guide tube outer diameter | 1.40 | cm |

## Materials and Weights per Assembly

| | Material | Weight/Assembly (kg) |
|---|---|---|
| fuel | $UO_2$ | 513.7 |
| fuel cladding | Zr-4 | 135.9 |
| guide tube | Zr-4 | 11.9 |
| lower rod plug | 1.4541 | 0.74 |
| upper rod plug | 1.4541 | 0.74 |
| expansion spaces | void | - |
| bottom fitting | 1.4541 | 12.3 |
| top fitting | 1.4541 | 14.4 |

## Material Composition

| Material | Density | Element | % by weight |
|---|---|---|---|
| 1.4541 | 7.8 | Mn | 2.0 |
| | | Cr | 18.0 |
| | | Ni | 10.0 |
| | | Fe | rest |
| Zr-4 | 6.5 | Zr | 97.9 |
| | | Sn | 1.6 |
| | | Fe | 0.5 |

## Table 4. Axial burn-up distribution

| rod position (cm) | | peaking* | rod position (cm) | | peaking* |
|---|---|---|---|---|---|
| from | to | | from | to | |
| 0 | 10 | 0.42 | 170 | 180 | 1.23 |
| 10 | 20 | 0.60 | 180 | 190 | 1.21 |
| 20 | 30 | 0.78 | 190 | 200 | 1.19 |
| 30 | 40 | 0.92 | 200 | 210 | 1.16 |
| 40 | 50 | 1.04 | 210 | 220 | 1.14 |
| 50 | 60 | 1.15 | 220 | 230 | 1.11 |
| 60 | 70 | 1.21 | 230 | 240 | 1.08 |
| 70 | 80 | 1.27 | 240 | 250 | 1.04 |
| 80 | 90 | 1.29 | 250 | 260 | 1.00 |
| 90 | 100 | 1.33 | 260 | 270 | 0.94 |
| 100 | 110 | 1.33 | 270 | 280 | 0.88 |
| 110 | 120 | 1.32 | 280 | 290 | 0.78 |
| 120 | 130 | 1.31 | 290 | 300 | 0.68 |
| 130 | 140 | 1.29 | 300 | 310 | 0.56 |
| 140 | 150 | 1.28 | 310 | 320 | 0.44 |
| 150 | 160 | 1.26 | 320 | 330 | 0.31 |
| 160 | 170 | 1.25 | 330 | 340 | 0.18 |

* defined as local burn-up/average burn-up

# Table 5. Problem 5. Total neutron and gamma-ray sources (for 5 sub assemblies)

## NEUTRON

| COUNTRY | CODE | SPONTANEOUS FISSION n/s | α - n n/s | TOTAL n/s |
|---|---|---|---|---|
| Belgium | ORIGEN 2 | 1.38E+09 | 3.42E+07 | 1.41E+09 |
| Italy | ORIGEN S | 1.40E+09 | 3.23E+07 | 1.43E+09 |
| UK | FISPIN | 1.38E+09 | 2.86E+07 | 1.41E+09 |
| Japan | ORIGEN 2-82 | 1.37E+09 | 3.27E+07 | 1.41E+09 |
| USA | SAS2 H | 1.32E+09 | 3.07E+07 | 1.35E+09 |
| USA | ORIGEN 2* | 1.50E+09 | 3.65E+07 | 1.54E+09 |
| USA | ORIGEN 2** | 1.37E+09 | 3.36E+07 | 1.41E+09 |
| FRG | OREST | 1.35E+09 | 6.63E+07 | 1.42E+09 |

## GAMMA - RAYS

| COUNTRY | CODE | ACTINIDES gamma/s | DECAY FISSION PRODUCTS gamma/s | TOTAL gamma/s |
|---|---|---|---|---|
| Belgium | ORIGEN 2 | 2.15E+14 | 5.03E+16 | 5.05E+16 |
| Italy | ORIGEN S | 1.80E+14 | 8.15E+16 | 8.17E+16 |
| UK | FISPIN | 1.61E+15 | 4.76E+16 | 4.92E+16 |
| Japan | ORIGEN 2-82 | 2.18E+14 | 1.07E+17 | 1.08E+17 |
| USA | SAS2H | 2.16E+14 | 1.03E+17 | 1.05E+17 |
| USA | ORIGEN 2* | 2.39E+14 | 1.09E+17 | 1.10E+17 |
| USA | ORIGEN 2** | 2.17E+14 | 1.07E+17 | 1.07E+17 |
| FRG | OREST | 8.15E+13 | 5.07E+16 | 5.08E+16 |

* 50 GWd/tonne library
** 33 GWd/tonne library

## Table 6. Problem 5. Important contributions to the neutron source in $UO_2$. (5 sub-assemblies)

| NUCLIDE | UK | | | US | | | US | | | US | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | FISPIN | | | SAS2H | | | ORIGEN2 (50)* | | | ORIGEN2 (33)* | | |
| | Decay rate (Bq) | Total source n/s | % | Decay rate (Bq) | Total source n/s | % | Decay rate (Bq) | Total source n/s | % | Decay rate (Bq) | Total source n/s | % |
| PU238 | 3.86E+14 | 1.00E+07 | 0.71% | 4.14E+14 | 1.16E+07 | 0.86% | | 1.30E+07 | 0.84% | | 1.26E+07 | 0.90% |
| PU239 | 2.81E+13 | 4.88E+05 | 0.03% | 2.90E+13 | 5.30E+05 | 0.04% | | 5.93E+05 | 0.04% | | 5.23E+05 | 0.04% |
| PU240 | 5.13E+13 | 7.19E+06 | 0.51% | 4.68E+14 | 5.81E+06 | 0.43% | | 6.09E+06 | 0.40% | | 6.38E+06 | 0.45% |
| PU242 | 2.63E+11 | 3.11E+06 | 0.22% | 2.26E+11 | 2.69E+06 | 0.20% | | 2.79E+06 | 0.18% | | 2.55E+06 | 0.18% |
| AM241 | 5.04E+13 | 1.10E+06 | 0.08% | 5.33E+13 | 1.25E+06 | 0.09% | | 1.55E+06 | 0.10% | | 1.28E+06 | 0.09% |
| AM243 | 2.64E+12 | 5.08E+04 | | 2.81E+12 | 5.83E+04 | | | 6.61E+04 | | | 6.04E+04 | |
| CM242 | 2.41E+14 | 4.94E+07 | 3.50% | 2.42E+14 | 5.29E+07 | 3.92% | | 5.92E+07 | 3.84% | | 5.25E+07 | 3.74% |
| CM243 | 1.71E+12 | 4.61E+04 | | 2.89E+12 | 2.24E+04 | | | 1.21E+05 | 0.01% | | 1.02E+05 | 0.01% |
| CM244 | 3.65E+14 | 1.33E+09 | 94.34% | 3.35E+14 | 1.27E+09 | 93.72% | | 1.45E+09 | 94.06% | | 1.32E+09 | 94.07% |
| CM246 | 1.03E+10 | 7.93E+06 | 0.56% | 9.79E+09 | 7.45E+06 | 0.55% | | 7.59E+06 | 0.49% | | 7.31E+06 | 0.52% |
| CM248 | 1.84E+05 | 4.82E+04 | | | 6.40E+04 | | | 2.64E+04 | | | | |
| CF252 | 5.37E+06 | 6.25E+05 | 0.04% | | 2.49E+06 | 0.18% | | 5.34E+05 | 0.03% | | | |
| TOTAL | | 1.41E+09 | | | 1.35E+09 | | | 1.54E+09 | | | 1.40E+09 | |

| NUCLIDE | ITALY | | | BELGIUM | | | FRG | | | JAPAN | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | ORIGEN-S | | | ORIGEN2 | | | OREST | | | ORIGEN2 -82 | | |
| | Decay rate (Bq) | Total source n/s | % | Decay rate (Bq) | Total source n/s | % | Decay rate (Bq) | Total source n/s | % | Decay rate (Bq) | Total source n/s | % |
| PU238 | 4.36E+14 | 2.37E+07 | 1.65% | | 1.26E+07 | 0.90% | 3.51E+14 | 2.13E+07 | 1.50% | | | |
| PU239 | 3.09E+13 | 1.19E+06 | 0.08% | | 5.23E+05 | 0.04% | 2.54E+13 | 1.07E+06 | 0.08% | | | |
| PU240 | 4.68E+13 | 6.94E+06 | 0.49% | | 6.32E+06 | 0.45% | 2.56E+13 | 8.90E+06 | 0.63% | | | |
| PU242 | 2.28E+11 | 3.11E+06 | 0.22% | | 2.57E+06 | 0.18% | 2.87E+11 | 4.11E+06 | 0.29% | | | |
| AM241 | 5.68E+13 | 2.91E+06 | 0.20% | | 1.27E+06 | 0.09% | 4.35E+13 | 2.35E+06 | 0.17% | | | |
| AM243 | 2.92E+12 | 1.29E+05 | 0.01% | | 6.06E+04 | | 2.29E+12 | 1.21E+05 | 0.01% | | | |
| CM242 | 2.65E+14 | 6.43E+07 | 4.49% | | 5.52E+07 | 3.93% | 2.33E+14 | 5.92E+07 | 4.18% | 2.81E+14 | | |
| CM243 | 7.51E+11 | 5.76E+04 | | | 1.03E+05 | 0.01% | 2.21E+12 | 1.68E+05 | 0.01% | | | |
| CM244 | 3.59E+14 | 1.32E+09 | 92.11% | | 1.32E+09 | 93.89% | 3.37E+14 | 1.31E+09 | 92.54% | 4.01E+14 | | |
| CM246 | 1.03E+10 | 7.84E+06 | 0.55% | | 7.30E+06 | 0.52% | 9.61E+09 | 7.74E+06 | 0.55% | | | |
| CM248 | 2.80E+05 | 7.16E+04 | 0.01% | | | | 8.30E+04 | 2.29E+04 | | | | |
| CF252 | 2.50E+07 | 2.75E+06 | 0.19% | | | | 5.36E+06 | 6.36E+05 | | | | |
| TOTAL | | 1.43E+09 | | | 1.41E+09 | | | 1.42E+09 | | | | |

*PWR library used

## Table 7 Relative decay rates and yields for nuclides contributing to the neutron source in $UO_2$.

| NUCLIDE | Mean decay rate (Bq) | UK FISPIN | US SAS2H | ITALY ORIGEN-S | FRG OREST | JAPAN ORIGEN2-82 | UK FISPIN | US SAS2H | ITALY ORIGEN-S | FRG OREST |
|---|---|---|---|---|---|---|---|---|---|---|
| | | Ratio of decay rate to the mean | | | | | Neutron source per decay | | | |
| **PU238** | 3.97E+14 | 0.97 | 1.04 | 1.10 | 0.88 | | 2.60E-08 | 2.79E-08 | 5.44E-08 | 6.07E-08 |
| **PU239** | 2.84E+13 | 0.99 | 1.02 | 1.09 | 0.90 | | 1.73E-08 | 1.83E-08 | 3.85E-08 | 4.21E-08 |
| **PU240** | 1.48E+14 | 0.35 | 3.16 | 0.32 | 0.17 | | 1.40E-07 | 1.24E-08 | 1.48E-07 | 3.48E-07 |
| **PU242** | 2.51E+11 | 1.05 | 0.90 | 0.91 | 1.14 | | 1.18E-05 | 1.19E-05 | 1.36E-05 | 1.43E-05 |
| **AM241** | 5.10E+13 | 0.99 | 1.05 | 1.11 | 0.85 | | 2.17E-08 | 2.34E-08 | 5.12E-08 | 5.40E-08 |
| **AM243** | 2.66E+12 | 0.99 | 1.05 | 1.10 | 0.86 | | 1.93E-08 | 2.07E-08 | 4.42E-08 | 5.28E-08 |
| **CM242** | 2.52E+14 | 0.96 | 0.96 | 1.05 | 0.92 | 1.11 | 2.05E-07 | 2.18E-07 | 2.43E-07 | 2.54E-07 |
| **CM243** | 1.89E+12 | 0.91 | 1.53 | 0.40 | 1.17 | | 2.69E-08 | 7.74E-09 | 7.67E-08 | 7.60E-08 |
| **CM244** | 3.59E+14 | 1.02 | 0.93 | 1.00 | 0.94 | 1.12 | 3.65E-06 | 3.78E-06 | 3.68E-06 | 3.89E-06 |
| **CM246** | 9.99E+09 | 1.03 | 0.98 | 1.03 | 0.96 | | 7.72E-04 | 7.61E-04 | 7.61E-04 | 8.05E-04 |
| **CM248** | 1.82E+05 | 1.01 | | 1.53 | 0.45 | | 2.62E-01 | | 2.56E-01 | 2.76E-01 |
| **CF252** | 1.19E+07 | 0.45 | | 2.10 | 0.45 | | 1.16E-01 | | 1.10E-01 | 1.19E-01 |

The ratios are given relative to the mean of all calculated values for each isotope.

## Table 8. Problem 5. Gamma-ray source spectrum (gamma/s) for 5 sub assemblies

| E max | E min | UK FISPIN | ITALY ORIGEN-S | US SAS2H | US ORIGEN2-50 | US ORIGEN2-33 | BELGIUM ORIGEN-2 | JAPAN ORIGEN2-82 | FRG OREST |
|---|---|---|---|---|---|---|---|---|---|
| 4.250 | 4.000 | | | | | | | | 1.88E+07 |
| 4.000 | 3.500 | | | 2.26E+08 | | | 3.37E+11 | | 2.16E+08 |
| 3.500 | 3.130 | 3.37E+11 | 3.70E+11 | 3.98E+11 | 4.00E+11 | 3.95E+11 | | 3.96E+11 | 4.23E+11 |
| 3.130 | 3.000 | | | | | | | | |
| 3.000 | 2.600 | | | 2.43E+12 | | | 2.34E+12 | 3.14E+12 | 2.58E+12 |
| 2.600 | 2.500 | 2.06E+12 | 3.03E+12 | | 3.22E+12 | 3.13E+12 | | | |
| 2.500 | 2.200 | | | 1.51E+13 | | | | | 2.78E+12 |
| 2.200 | 2.000 | 1.33E+14 | 1.50E+14 | | 1.46E+14 | 1.46E+14 | 1.42E+14 | 1.46E+14 | 1.69E+14 |
| 2.000 | 1.800 | | | 1.71E+14 | | | | | |
| 1.800 | 1.660 | 1.47E+13 | 4.10E+13 | | 1.12E+14 | 1.09E+14 | 6.94E+13 | 1.09E+14 | |
| 1.660 | 1.500 | | | 4.35E+14 | | | | | 4.55E+14 |
| 1.500 | 1.440 | 1.11E+14 | 4.56E+14 | | | | | | |
| 1.440 | 1.350 | | | | | | | | |
| 1.350 | 1.220 | 8.00E+14 | | | 2.63E+15 | 2.65E+15 | 1.44E+15 | 1.60E+15 | |
| 1.220 | 1.000 | 8.30E+14 | 1.40E+15 | 2.89E+15 | | | | | 1.40E+15 |
| 1.000 | 0.900 | | | | | | | | |
| 0.900 | 0.800 | 1.52E+15 | 4.60E+15 | | 9.67E+15 | 9.25E+15 | 8.87E+15 | 9.18E+15 | |
| 0.800 | 0.700 | | | | | | | | |
| 0.700 | 0.600 | 3.43E+16 | 1.98E+16 | 3.54E+16 | | | | | 3.96E+16 |
| 0.600 | 0.475 | | | | 2.81E+16 | 2.74E+16 | 2.65E+16 | 2.74E+16 | |
| 0.475 | 0.450 | 6.45E+15 | 1.23E+16 | | | | | | |
| 0.450 | 0.400 | | | | | | 5.69E+14 | | |
| 0.400 | 0.375 | 1.74E+13 | 1.47E+15 | | 2.33E+15 | 2.31E+15 | | 2.26E+15 | |
| 0.375 | 0.300 | | | | | | | | 1.19E+16 |
| 0.300 | 0.225 | 7.83E+13 | 1.95E+15 | 3.29E+15 | | | | | |
| 0.225 | 0.200 | | | | | | 5.37E+14 | 4.02E+15 | 2.29E+11 |
| 0.200 | 0.175 | | | | 4.05E+15 | 4.02E+15 | | | |
| 0.175 | 0.150 | | | 2.83E+15 | | | | | |
| 0.150 | 0.140 | 2.51E+15 | 7.47E+15 | | | | 2.71E+15 | | 4.91E+10 |
| 0.140 | 0.125 | | | 5.20E+15 | 5.16E+15 | 5.13E+15 | | 5.12E+15 | |
| 0.125 | 0.100 | | | | | | | | 3.28E+12 |
| 0.100 | 0.070 | | | 4.35E+15 | 4.62E+15 | 4.58E+15 | 9.60E+14 | 4.58E+15 | |
| 0.070 | 0.050 | 4.61E+14 | 7.80E+15 | | 6.63E+15 | 6.58E+15 | 7.43E+14 | 6.57E+15 | 1.77E+13 |
| 0.050 | 0.045 | | | 7.13E+15 | | | | | |
| 0.045 | 0.040 | | | | | | 3.41E+15 | | 5.86E+11 |
| 0.040 | 0.035 | 3.13E+14 | 2.43E+16 | | 7.42E+15 | 7.37E+15 | | 7.35E+15 | |
| 0.035 | 0.030 | | | | | | | | 5.84E+13 |
| 0.030 | 0.025 | | | 1.29E+16 | 7.45E+15 | 7.39E+15 | 1.21E+15 | 7.26E+15 | |
| 0.025 | 0.020 | | | | | | | | |
| 0.020 | 0.010 | 3.81E+11 | | | | | | | |
| 0.010 | 0.000 | 5.97E+13 | | 3.00E+16 | 3.19E+16 | 3.17E+16 | 3.34E+15 | 3.18E+16 | |
| 3.000 | 0.400 | 4.42E+16 | 3.87E+16 | 3.89E+16 | 4.07E+16 | 3.96E+16 | 3.76E+16 | 3.84E+16 | 4.16E+16 |
| TOTAL | | 4.76E+16 | 8.17E+16 | 1.05E+17 | 1.10E+17 | 1.09E+17 | 5.05E+16 | 1.07E+17 | 5.36E+16 |

## Table 9. Problem 5. Contributions to the gamma-ray source for 5 sub assemblies (all energies).

| Isotope | U UK | | | | | ITALY | | | | | USA(SAS2H) | | | | | FRG | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | decay rate (Bq) | fission product gammas | | | | decay rate (Bq) | fission product gammas | | | | decay rate (Bq) | fission product gammas | | | | decay rate (Bq) | fission product gammas | |
| | | MeV/s | | gamma/s | | | MeV/s | | gamma/s | | | MeV/s | | gamma/s | | | MeV/s | gamma/s |
| | | | % of total | | % of total | | | % of total | | % of total | | | % of total | | % of total | | | |
| KR85 | 7.79E+14 | 4.49E+13 | 0.1% | 3.46E+12 | - | 7.84E+14 | 4.86E+12 | - | 9.37E+13 | 0.1% | 7.85E+14 | 1.73E+12 | - | | | 7.84E+14 | | |
| SR90 | 2.99E+15 | - | | - | | 6.85E+15 | 3.77E+13 | 0.1% | 6.48E+14 | 0.8% | 6.65E+15 | | | 1.12E+15 | 1.1% | 6.66E+15 | | |
| Y90 | 6.75E+15 | 1.72E+12 | - | 1.04E+12 | - | 6.85E+15 | 5.12E+14 | 1.7% | 4.67E+15 | 5.7% | 6.65E+15 | 1.12E+10 | - | 7.11E+15 | 7.1% | 6.66E+15 | | |
| ZR95 | 5.85E+13 | 4.29E+13 | 0.1% | 6.14E+13 | 0.1% | 4.92E+13 | 3.64E+13 | 0.1% | 5.59E+13 | 0.1% | 4.89E+13 | 3.62E+13 | 0.1% | | | 4.86E+13 | | |
| NB95 | 1.26E+14 | 9.63E+13 | 0.3% | 1.38E+14 | 0.3% | 1.09E+14 | 8.37E+13 | 0.3% | 1.28E+14 | 0.2% | 1.09E+14 | 8.30E+13 | 0.3% | 1.28E+14 | 0.1% | 1.11E+14 | | |
| RH106* | 1.40E+16 | 2.88E+15 | 9.3% | 4.70E+15 | 10.0% | 1.34E+16 | 4.71E+15 | 15.6% | 1.87E+16 | 23.0% | 1.32E+16 | 2.70E+15 | 10.9% | 2.66E+16 | 26.5% | 1.38E+16 | | |
| AG110M | 6.72E+13 | 1.85E+14 | 0.6% | 2.13E+14 | 0.5% | 6.75E+13 | 1.85E+14 | 0.6% | 2.32E+14 | 0.3% | 6.53E+13 | 1.80E+14 | 0.7% | 4.45E+13 | | 6.97E+13 | | |
| SB125 | 5.38E+14 | 2.30E+14 | 0.7% | 4.50E+14 | 1.0% | 6.01E+14 | 2.58E+14 | 0.9% | 7.31E+14 | 0.9% | 5.92E+13 | 2.55E+14 | 1.0% | 3.38E+14 | 0.3% | 4.85E+14 | | |
| TE125M | 5.81E+13 | 3.48E+11 | - | 9.04E+12 | - | 1.47E+14 | 5.08E+12 | - | 1.69E+14 | 0.2% | 1.45E+14 | 5.15E+12 | - | 1.67E+14 | 0.2% | 6.89E+13 | | |
| CS134 | 1.27E+16 | 1.98E+16 | 63.7% | 2.85E+16 | 60.7% | 9.14E+15 | 1.42E+16 | 46.9% | 2.20E+16 | 27.0% | 8.71E+15 | 1.36E+16 | 55.1% | 2.16E+16 | 21.5% | 8.89E+15 | | |
| CS137 | 4.44E+15 | - | | - | | 1.03E+16 | 4.75E+13 | 0.2% | 8.24E+14 | 1.0% | 9.95E+15 | | | 1.44E+15 | 1.4% | 9.88E+15 | | |
| BA137M | 9.50E+15 | 5.65E+15 | 18.2% | 8.08E+15 | 17.2% | 9.71E+15 | 5.78E+15 | 19.1% | 9.67E+15 | 11.9% | 9.42E+15 | 5.62E+15 | 22.8% | 9.37E+15 | 9.3% | 9.36E+15 | | |
| CE144 | 1.68E+16 | 2.71E+14 | 0.9% | 2.01E+15 | 4.3% | 1.66E+16 | 3.60E+14 | 1.2% | 4.91E+15 | 6.0% | 1.66E+16 | 3.40E+14 | 1.4% | 5.58E+15 | 5.6% | 1.67E+16 | | |
| PR144 | 1.68E+16 | 4.86E+14 | 1.6% | 3.87E+14 | 0.8% | 1.66E+16 | 2.42E+15 | 8.0% | 1.56E+16 | 19.2% | 1.66E+16 | 5.27E+14 | 2.1% | 2.44E+16 | 24.3% | 1.67E+16 | | |
| PR144M | 1.12E+14 | 1.64E+11 | - | 3.50E+11 | - | 1.98E+14 | 1.98E+12 | - | 6.60E+13 | 0.1% | 2.00E+14 | 2.53E+12 | 0.0% | | | 2.17E+14 | | |
| EU154 | 1.08E+15 | 1.31E+15 | 4.2% | 1.67E+15 | 3.6% | 1.37E+15 | 1.68E+15 | 5.6% | 2.81E+15 | 3.4% | 1.05E+15 | 1.29E+15 | 5.2% | 2.05E+15 | 2.0% | 8.69E+14 | | |
| EU155 | 7.18E+14 | 4.49E+13 | 0.1% | 6.71E+14 | 1.4% | 2.44E+14 | 1.46E+13 | - | 1.71E+14 | 0.2% | 5.94E+14 | 3.59E+13 | 0.1% | 4.20E+14 | 0.4% | 4.67E+14 | | |
| Total | | 3.10E+16 | | 4.69E+16 | | | 3.03E+16 | | 8.15E+16 | | | 2.47E+16 | | 1.00E+17 | | 5.05E+16 | | |

| Isotope | USA(ORIGEN2) | | | | | JAPAN | | | | | BELGIUM | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | decay rate (Bq) | fission product gammas | | | | decay rate (Bq) | fission product gammas | | | | decay rate (Bq) | fission product gammas | | | |
| | | MeV/s | | gamma/s | | | MeV/s | | gamma/s | | | MeV/s | | gamma/s | |
| | | | % of total | | % of total | | | % of total | | % of total | | | % of total | | % of total |
| KR85 | | | | | | | | | | | | | | | |
| SR90 | | 3.58E+13 | 0.1% | 1.20E+15 | 1.1% | | | | | | | | | 8.43E+13 | 0.2% |
| Y90 | | 4.89E+14 | 1.6% | 7.38E+15 | 6.8% | | | | | | | | | 6.82E+14 | 1.4% |
| ZR95 | | | | | | | | | | | | | | 4.52E+13 | 0.1% |
| NB95 | | 8.35E+13 | 0.3% | 1.01E+14 | 0.1% | | | | | | | | | 1.03E+14 | 0.2% |
| RH106* | | 5.12E+15 | 16.6% | 3.03E+16 | 28.0% | 1.61E+16 | | | | | | | | 7.05E+15 | 14.3% |
| AG110M | | 1.98E+14 | 0.6% | 2.49E+14 | 0.2% | | | | | | | | | 1.44E+14 | 0.3% |
| SB125 | | 3.82E+14 | 1.2% | 1.33E+15 | 1.2% | | | | | | | | | 1.23E+15 | 2.5% |
| TE125M | | 7.68E+12 | - | 2.87E+14 | 0.3% | | | | | | | | | 2.70E+14 | 0.5% |
| CS134 | | 1.47E+16 | 47.7% | 2.23E+16 | 20.6% | 1.03E+16 | | | | | | | | 2.02E+16 | 41.0% |
| CS137 | | 4.69E+13 | 0.2% | 1.58E+15 | 1.5% | | | | | | | | | 1.08E+14 | 0.2% |
| BA137M | | 5.61E+15 | 18.2% | 1.04E+16 | 9.6% | | | | | | | | | 9.76E+15 | 19.8% |
| CE144 | | 3.57E+14 | 1.2% | 5.18E+15 | 4.8% | | | | | | | | | 4.31E+15 | 8.8% |
| PR144 | | 2.39E+15 | 7.8% | 2.49E+16 | 23.0% | 1.87E+16 | | | | | | | | 2.82E+15 | 5.7% |
| PR144M | | | | | | | | | | | | | | 5.99E+13 | 0.1% |
| EU154 | | 1.33E+15 | 4.3% | 2.31E+15 | 2.1% | 1.16E+15 | | | | | | | | 1.92E+15 | 3.9% |
| EU155 | | 3.76E+13 | 0.1% | 5.18E+14 | 0.5% | | | | | | | | | 4.33E+14 | 0.9% |
| Total | | 3.08E+16 | | 1.08E+17 | | 4.63E+16 | | | | | | 2.71E+16 | | 4.92E+16 | |

## Table 10. Relative decay rates and yields for nuclides contributing to the gamma-ray source.

| | Mean decay rate (Bq) | UK FISPIN | ITALY ORIGEN-S | US SAS2H | FRG OREST | JAPAN ORIGEN2-82 | UK FISPIN | ITALY ORIGEN-S | US SAS2H |
|---|---|---|---|---|---|---|---|---|---|
| | | ratio of decay rate to mean | | | | | gamma-ray source (MeV/s) per decay | | |
| **KR85** | 7.83E+14 | 0.99 | 1.00 | 1.00 | 1.00 | | 0.06 | 0.01 | 0.00 |
| **SR90** | 5.79E+15 | 0.52 | 1.18 | 1.15 | 1.15 | | | 0.01 | 0.00 |
| **Y90** | 6.73E+15 | 1.00 | 1.02 | 0.99 | 0.99 | | 0.00 | 0.07 | 0.00 |
| **ZR95** | 5.13E+13 | 1.14 | 0.96 | 0.95 | 0.95 | | 0.73 | 0.74 | 0.74 |
| **NB95** | 1.14E+14 | 1.11 | 0.96 | 0.96 | 0.98 | | 0.76 | 0.77 | 0.76 |
| **RH106*** | 1.41E+16 | 0.99 | 0.95 | 0.94 | 0.98 | 1.14 | 0.21 | 0.35 | 0.20 |
| **AG110M** | 6.74E+13 | 1.00 | 1.00 | 0.97 | 1.03 | | 2.75 | 2.74 | 2.76 |
| **SB125** | 4.21E+14 | 1.28 | 1.43 | 0.14 | 1.15 | | 0.43 | 0.43 | 4.31 |
| **TE125M** | 1.05E+14 | 0.55 | 1.40 | 1.38 | 0.66 | | 0.01 | 0.03 | 0.04 |
| **CS134** | 1.16E+16 | 1.09 | 0.79 | 0.75 | 0.76 | 0.89 | 1.56 | 1.55 | 1.56 |
| **CS137** | 8.64E+15 | 0.51 | 1.19 | 1.15 | 1.14 | | | 0.00 | 0.00 |
| **BA137M** | 9.50E+15 | 1.00 | 1.02 | 0.99 | 0.99 | | 0.59 | 0.60 | 0.60 |
| **CE144** | 1.67E+16 | 1.01 | 1.00 | 1.00 | 1.00 | | 0.02 | 0.02 | 0.02 |
| **PR144** | 1.71E+16 | 0.98 | 0.97 | 0.97 | 0.98 | 1.09 | 0.03 | 0.15 | 0.03 |
| **PR144M** | 1.82E+14 | 0.62 | 1.09 | 1.10 | 1.19 | | 0.00 | 0.01 | 0.01 |
| **EU154** | 1.11E+15 | 0.98 | 1.24 | 0.95 | 0.79 | 1.05 | 1.21 | 1.23 | 1.23 |
| **EU155** | 5.06E+14 | 1.42 | 0.48 | 1.17 | 0.92 | | 0.06 | 0.06 | 0.06 |

Ratios are given relative to the mean of all the calculated decay rates for each isotope.

# Table 11. US comparison of Gamma source contributions from Pr144 and Rh106 for calculations with and without bremsstrahlung.

| Without Bremsstrahlung | | | | | | With Bremsstrahlung | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Pr 144 | | Rh 106 | | | | Pr 144 | | Rh 106 | |
| Lower Energy | Upper Energy | gamma/s | MeV/s | gamma/s | MeV/s | Lower Energy | Upper Energy | gamma/s | MeV/s | gamma/s | MeV/s |
| 1.00E-02 | 5.00E-02 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 2.00E-02 | 2.78E+20 | 2.78E+18 | 1.12E+22 | 1.12E+20 |
| 5.00E-02 | 1.00E-01 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 2.00E-02 | 4.00E-02 | 9.19E+19 | 2.77E+18 | 3.75E+21 | 1.13E+20 |
| 1.00E-01 | 2.00E-01 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 4.00E-02 | 7.00E-02 | 6.45E+19 | 3.55E+18 | 2.66E+21 | 1.46E+20 |
| 2.00E-01 | 3.00E-01 | 0.00E+00 | 0.00E+00 | 2.25E+17 | 5.63E+16 | 7.00E-02 | 1.00E-01 | 3.80E+19 | 3.23E+18 | 1.58E+21 | 1.34E+20 |
| 3.00E-01 | 4.00E-01 | 0.00E+00 | 0.00E+00 | 2.34E+18 | 8.20E+17 | 1.00E-01 | 1.40E-01 | 2.72E+19 | 3.26E+18 | 1.14E+21 | 1.37E+20 |
| 4.00E-01 | 6.00E-01 | 0.00E+00 | 0.00E+00 | 2.97E+21 | 1.48E+21 | 1.40E-01 | 2.00E-01 | 2.87E+19 | 4.88E+18 | 1.22E+21 | 2.08E+20 |
| 6.00E-01 | 8.00E-01 | 5.95E+18 | 4.16E+18 | 1.31E+21 | 9.17E+20 | 2.00E-01 | 4.00E-01 | 3.36E+19 | 1.01E+19 | 1.48E+21 | 4.43E+20 |
| 8.00E-01 | 1.00E+00 | 2.43E+16 | 2.18E+16 | 5.70E+19 | 5.13E+19 | 4.00E-01 | 9.00E-01 | 2.30E+19 | 1.49E+19 | 4.55E+21 | 2.96E+21 |
| 1.00E+00 | 1.33E+00 | 0.00E+00 | 0.00E+00 | 2.54E+20 | 2.96E+20 | 9.00E-01 | 1.35E+00 | 2.94E+18 | 3.30E+18 | 4.24E+20 | 4.77E+20 |
| 1.33E+00 | 1.66E+00 | 1.23E+18 | 1.84E+18 | 2.98E+19 | 4.46E+19 | 1.35E+00 | 1.80E+00 | 1.83E+18 | 2.88E+18 | 8.00E+19 | 1.26E+20 |
| 1.66E+00 | 2.00E+00 | 0.00E+00 | 0.00E+00 | 1.35E+19 | 2.47E+19 | 1.80E+00 | 2.20E+00 | 3.54E+18 | 7.09E+18 | 2.60E+19 | 5.20E+19 |
| 2.00E+00 | 2.50E+00 | 3.03E+18 | 6.81E+18 | 1.52E+19 | 3.41E+19 | 2.20E+00 | 2.60E+00 | 3.35E+16 | 8.03E+16 | 1.45E+19 | 3.49E+19 |
| 2.50E+00 | 3.00E+00 | 8.05E+14 | 2.21E+15 | 2.01E+18 | 5.52E+18 | 2.60E+00 | 3.00E+00 | 2.70E+15 | 7.55E+15 | 2.45E+18 | 6.87E+18 |
| 3.00E+00 | 4.00E+00 | 0.00E+00 | 0.00E+00 | 2.93E+17 | 1.03E+18 | 3.00E+00 | 3.50E+00 | 0.00E+00 | 0.00E+00 | 4.22E+17 | 1.37E+18 |
| 4.00E+00 | 5.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 3.50E+00 | 4.00E+00 | 0.00E+00 | 0.00E+00 | 1.86E+14 | 6.97E+14 |
| 5.00E+00 | 6.50E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 4.00E+00 | 4.50E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 |
| 6.50E+00 | 8.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 4.50E+00 | 5.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 |
| 8.00E+00 | 1.00E+01 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 5.00E+00 | 6.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 | 0.00E+00 |
| TOTAL = | | 1.02E+19 | 1.28E+19 | 4.65E+21 | 2.86E+21 | | | 5.93E+20 | 5.88E+19 | 2.82E+22 | 4.95E+21 |

**Pr 144 with/without = 57.96 for gamma/s**

**4.58 for MeV/s**

**Rh106 with/without = 6.05 for gamma/s**

**1.73 for MeV/s**

# Table 12. Problem 5. Axial distribution of n and gamma-ray sources (5 sub-assemblies)

| Axial Peaking* | UK | | | | USA | | | | JAPAN | | | | BELGIUM | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Gammas/s | | | | Gammas/s | | | | Gammas/s | | | | Gammas/s | | | |
| | fission products | actinides | TOTAL | relative source** | fission products | actinides | TOTAL | relative source** | fission products | actinides | TOTAL | relative source** | fission products | actinides | TOTAL | relative source** |
| 0.15 | | | | | | | | | 1.13E+16 | 4.48E+12 | 1.13E+16 | 0.11 | | | | |
| 0.18 | 3.54E+15 | 9.35E+13 | 3.63E+15 | 0.07 | 1.36E+16 | 6.85E+12 | 1.36E+16 | 0.13 | | | | | | | | |
| 0.20 | | | | | | | | | 1.57E+16 | 8.08E+12 | 1.57E+16 | 0.15 | | | | |
| 0.30 | | | | | | | | | 2.47E+16 | 1.89E+13 | 2.47E+16 | 0.23 | | | | |
| 0.31 | 7.54E+15 | 2.62E+14 | 7.80E+15 | 0.16 | | | | | | | | | | | | |
| 0.40 | | | | | 3.35E+16 | 3.43E+13 | 3.35E+16 | 0.33 | 3.44E+16 | 3.28E+13 | 3.44E+16 | 0.32 | | | | |
| 0.44 | 1.28E+16 | 4.77E+14 | 1.33E+16 | 0.27 | | | | | | | | | 1.57E+16 | 3.78E+13 | 1.57E+16 | 0.31 |
| 0.50 | | | | | | | | | 4.48E+16 | 5.09E+13 | 4.49E+16 | 0.42 | | | | |
| 0.60 | | | | | 5.40E+16 | 7.60E+13 | 5.41E+16 | 0.53 | 5.61E+16 | 7.36E+13 | 5.62E+16 | 0.52 | | | | |
| 0.68 | 2.54E+16 | 9.22E+14 | 2.63E+16 | 0.54 | | | | | | | | | | | | |
| 0.70 | | | | | | | | | 6.79E+16 | 1.00E+14 | 6.80E+16 | 0.63 | | | | |
| 0.78 | | | | | | | | | | | | | 3.49E+16 | 1.23E+14 | 3.50E+16 | 0.69 |
| 0.80 | | | | | 7.75E+16 | 1.36E+14 | 7.76E+16 | 0.76 | 8.04E+16 | 1.32E+14 | 8.05E+16 | 0.75 | | | | |
| 0.90 | | | | | | | | | 9.37E+16 | 1.71E+14 | 9.39E+16 | 0.87 | | | | |
| 1.00 | 4.76E+16 | 1.60E+15 | 4.92E+16 | 1.00 | 1.03E+17 | 2.16E+14 | 1.03E+17 | 1.00 | 1.07E+17 | 2.18E+14 | 1.08E+17 | 1.00 | 5.03E+16 | 2.15E+14 | 5.05E+16 | 1.00 |
| 1.10 | | | | | | | | | 1.21E+17 | 2.74E+14 | 1.22E+17 | 1.13 | | | | |
| 1.20 | | | | | 1.29E+17 | 3.22E+14 | 1.29E+17 | 1.26 | 1.36E+17 | 3.37E+14 | 1.36E+17 | 1.27 | 6.56E+16 | 3.36E+14 | 6.59E+16 | 1.31 |
| 1.30 | | | | | | | | | 1.51E+17 | 4.14E+14 | 1.51E+17 | 1.41 | | | | |
| 1.32 | | | | | 1.46E+17 | 3.99E+14 | 1.46E+17 | 1.43 | | | | | | | | |
| 1.33 | 7.50E+16 | 2.58E+15 | 7.76E+16 | 1.58 | | | | | | | | | 7.60E+16 | 4.36E+14 | 7.64E+16 | 1.51 |
| 1.40 | | | | | | | | | 1.65E+17 | 4.99E+14 | 1.66E+17 | 1.54 | | | | |
| 1.50 | | | | | | | | | 1.81E+17 | 5.95E+14 | 1.81E+17 | 1.69 | | | | |
| Axial Peaking* | n/s | | | | n/s | | | | n/s | | | | n/s | | | |
| | spontaneous fission | (α,n) | TOTAL | relative source** | spontaneous fission | (α,n) | TOTAL | relative source** | spontaneous fission | (α,n) | TOTAL | relative source** | spontaneous fission | (α,n) | TOTAL | |
| 0.15 | | | | | | | | | 9.28E+05 | 6.34E+05 | 1.56E+06 | 0.00 | | | | |
| 0.18 | 1.49E+06 | 7.29E+05 | 2.22E+06 | 0.00 | 1.46E+06 | 8.11E+05 | 2.27E+06 | 0.00 | | | | | | | | |
| 0.20 | | | | | | | | | 1.93E+06 | 1.04E+06 | 2.96E+06 | 0.00 | | | | |
| 0.30 | | | | | | | | | 7.06E+06 | 2.29E+06 | 9.35E+06 | 0.01 | | | | |
| 0.31 | 8.00E+06 | 2.01E+06 | 1.00E+07 | 0.01 | | | | | | | | | | | | |
| 0.40 | | | | | 2.28E+07 | 3.82E+06 | 2.66E+07 | 0.02 | 2.21E+07 | 4.10E+06 | 2.62E+07 | 0.02 | | | | |
| 0.44 | 3.30E+07 | 4.19E+06 | 3.72E+07 | 0.03 | | | | | | | | | 3.32E+07 | 5.07E+06 | 3.83E+07 | 0.03 |
| 0.50 | | | | | | | | | 5.77E+07 | 6.61E+06 | 6.43E+07 | 0.05 | | | | |
| 0.60 | | | | | 1.36E+08 | 9.26E+06 | 1.45E+08 | 0.11 | 1.31E+08 | 9.92E+06 | 1.41E+08 | 0.10 | | | | |
| 0.68 | 2.38E+08 | 1.11E+07 | 2.49E+08 | 0.18 | | | | | | | | | | | | |
| 0.70 | | | | | | | | | 2.67E+08 | 1.41E+07 | 2.81E+08 | 0.20 | | | | |
| 0.78 | | | | | | | | | | | | | 4.41E+08 | 1.84E+07 | 4.59E+08 | 0.32 |
| 0.80 | | | | | 4.96E+08 | 1.79E+07 | 5.14E+08 | 0.38 | 4.96E+08 | 1.93E+07 | 5.15E+08 | 0.37 | | | | |
| 0.90 | | | | | | | | | 8.51E+08 | 2.58E+07 | 8.77E+08 | 0.62 | | | | |
| 1.00 | 1.38E+09 | 2.86E+07 | 1.41E+09 | 1.00 | 1.32E+09 | 3.07E+07 | 1.35E+09 | 1.00 | 1.37E+09 | 3.37E+07 | 1.41E+09 | 1.00 | 1.38E+09 | 3.42E+07 | 1.41E+09 | 1.00 |
| 1.10 | | | | | | | | | 2.11E+09 | 4.35E+07 | 2.15E+09 | 1.53 | | | | |
| 1.20 | | | | | 2.83E+09 | 4.84E+07 | 2.88E+09 | 2.13 | 3.08E+09 | 5.52E+07 | 3.13E+09 | 2.23 | 3.11E+09 | 5.57E+07 | 3.17E+09 | 2.24 |
| 1.30 | | | | | | | | | 4.35E+09 | 6.91E+07 | 4.42E+09 | 3.14 | | | | |
| 1.32 | | | | | 4.16E+09 | 6.16E+07 | 4.22E+09 | 3.13 | | | | | | | | |
| 1.33 | 4.68E+09 | 6.12E+07 | 4.74E+09 | 3.37 | | | | | | | | | 4.84E+09 | 7.42E+07 | 4.91E+09 | 3.47 |
| 1.40 | | | | | | | | | 5.93E+09 | 8.49E+07 | 6.02E+09 | 4.27 | | | | |
| 1.50 | | | | | | | | | 7.86E+09 | 1.03E+08 | 7.96E+09 | 5.65 | | | | |

* Axial peaking is the ratio of the point burn-up to the mean.

**Relative source = source/source at mean burn-up

# Table 13. Problem 6. Total neutron and gamma-ray sources (for 5 sub assemblies)

## NEUTRON

| Contributor | Code | Spontaneous Fission n/s | $\alpha$ - n n/s | TOTAL n/s |
|---|---|---|---|---|
| Belgium | ORIGEN 2 | 4.04E+09 | 9.57E+07 | 4.14E+09 |
| Italy | ORIGEN S | 5.17E+09 | 1.02E+08 | 5.27E+09 |
| UK | FISPIN | 5.17E+09 | 1.01E+08 | 5.27E+09 |
| Japan | ORIGEN 2-82 | 4.65E+09 | 1.08E+08 | 4.75E+09 |
| USA | SAS2H | 4.86E+09 | 9.63E+07 | 4.95E+09 |
| FRG | OREST | 5.05E+09 | 2.24E+08 | 5.28E+09 |

## GAMMA - RAYS

| Contributor | Code | Actinides gamma/s | Decay Fission Products gamma/s | TOTAL gamma/s |
|---|---|---|---|---|
| Belgium | ORIGEN 2 | 6.28E+14 | 5.31E+16 | 5.37E+16 |
| Italy | ORIGEN S | 5.60E+14 | 8.55E+16 | 8.60E+16 |
| UK | FISPIN | 4.45E+15 | 4.37E+16 | 4.82E+16 |
| Japan | ORIGEN 2-82 | 6.42E+14 | 1.16E+17 | 1.17E+17 |
| USA | SAS2H | 6.80E+14 | 1.08E+17 | 1.09E+17 |
| FRG | OREST | 2.87E+14 | 5.34E+16 | 5.37E+16 |

## Table 14. Problem 6. Important contributions to MOX source (for 5 sub assemblies)

| NUCLIDE | UK | | | US | | | JAPAN | | |
|---|---|---|---|---|---|---|---|---|---|
| | FISPIN | | | SAS2H | | | ORIGEN2 -82 | | |
| | Decay rate (Bq) | Total source n/s | % | Decay rate (Bq) | Total source n/s | % | Decay rate (Bq) | Total source n/s | % |
| PU238 | 5.67E+14 | 1.47E+07 | 0.28% | | 1.67E+07 | 0.34% | | | |
| PU239 | 3.43E+13 | 5.94E+05 | 0.01% | | 1.15E+06 | 0.02% | | | |
| PU240 | 1.03E+14 | 1.44E+07 | 0.27% | | 2.08E+07 | 0.42% | | | |
| PU242 | 1.03E+12 | 1.21E+07 | 0.23% | | 6.86E+06 | 0.14% | | | |
| AM241 | 1.79E+14 | 3.87E+07 | 0.73% | | 5.00E+06 | 0.10% | | | |
| AM243 | 1.07E+13 | 2.07E+05 | - | | 1.75E+05 | - | | | |
| CM242 | 1.41E+15 | 2.88E+08 | 5.42% | | 2.31E+08 | 4.66% | 1.18E+15 | | |
| CM243 | 1.03E+13 | 2.77E+05 | 0.01% | | 5.61E+04 | - | | | |
| CM244 | 1.35E+15 | 4.92E+09 | 92.63% | | 4.63E+09 | 93.32% | 1.18E+15 | | |
| CM246 | 2.82E+10 | 2.17E+07 | 0.41% | | 3.23E+07 | 0.65% | | | |
| CM248 | 3.13E+05 | 8.17E+04 | - | | 4.40E+05 | 0.01% | | | |
| CF250 | 1.67E+07 | 4.51E+04 | - | | 2.85E+05 | 0.01% | | | |
| CF252 | 3.64E+06 | 4.23E+05 | 0.01% | | 1.67E+07 | 0.34% | | | |
| TOTAL | | 5.31E+09 | | | 4.96E+09 | | | | |

| NUCLIDE | ITALY | | | BELGIUM | | | FRG | | |
|---|---|---|---|---|---|---|---|---|---|
| | ORIGEN-S | | | ORIGEN2 | | | OREST | | |
| | Decay rate (Bq) | Total source n/s | % | Decay rate (Bq) | Total source n/s | % | Decay rate (Bq) | Total source n/s | % |
| PU238 | 6.35E+14 | 3.51E+07 | 0.67% | | 1.67E+07 | 0.40% | 5.58E+14 | 3.40E+07 | 0.64% |
| PU239 | 6.88E+13 | | | | 1.21E+06 | 0.03% | 5.29E+13 | 2.23E+06 | 0.04% |
| PU240 | 1.62E+14 | 2.35E+07 | 0.45% | | 2.41E+07 | 0.58% | 1.74E+14 | 2.78E+07 | 0.53% |
| PU242 | 5.57E+11 | 7.79E+06 | 0.15% | | 6.60E+06 | 0.16% | 8.18E+11 | 1.17E+07 | 0.22% |
| AM241 | 2.26E+14 | 1.18E+07 | 0.22% | | 5.40E+06 | 0.13% | 1.85E+14 | 1.00E+07 | 0.19% |
| AM243 | 8.60E+12 | | | | 1.60E+05 | - | 7.21E+12 | 3.48E+05 | 0.01% |
| CM242 | 1.13E+15 | 2.82E+08 | 5.36% | | 2.27E+08 | 5.49% | 1.14E+15 | 2.91E+08 | 5.52% |
| CM243 | 1.77E+12 | | | | 4.70E+05 | 0.01% | 1.30E+13 | 9.89E+05 | 0.02% |
| CM244 | 1.31E+15 | 4.85E+09 | 92.14% | | 3.83E+09 | 92.65% | 1.24E+15 | 4.86E+09 | 92.11% |
| CM246 | 4.55E+10 | 3.42E+07 | 0.65% | | 2.24E+07 | 0.54% | 4.31E+10 | 3.47E+07 | 0.66% |
| CM248 | 2.10E+06 | | | | | - | 5.06E+05 | 1.40E+05 | - |
| CF250 | 1.30E+08 | | | | | - | 4.77E+07 | | - |
| CF252 | 1.80E+08 | 1.91E+07 | 0.36% | | | - | 2.93E+07 | 3.48E+06 | 0.07% |
| TOTAL | | 5.26E+09 | | | 4.13E+09 | | | 5.28E+09 | |

# Table 15. Relative decay rates and yields for nuclides contributing to the neutron source in the MOX fuel.

| NUCLIDE | Mean Decay rate (Bq) | UK FISPIN | ITALY ORIGEN-S | FRG OREST | JAPAN ORIGEN2-82 | UK FISPIN | ITALY ORIGEN-S | FRG OREST |
|---|---|---|---|---|---|---|---|---|
| | | Ratio of decay rate to the mean | | | | Neutron source per decay | | |
| **PU238** | 5.87E+14 | 0.97 | 1.08 | 0.95 | | 2.59E-08 | 2.77E-08 | 6.09E-08 |
| **PU239** | 5.20E+13 | 0.66 | 1.32 | 1.02 | | 1.73E-08 | 1.85E-08 | 4.22E-08 |
| **PU240** | 9.41E+13 | 1.09 | 1.72 | 0.18 | | 1.40E-07 | 1.27E-07 | 1.60E-06 |
| **PU242** | 8.00E+11 | 1.28 | 0.70 | 1.02 | | 1.18E-05 | 1.18E-05 | 1.43E-05 |
| **AM241** | 1.97E+14 | 0.91 | 1.15 | 0.94 | | 2.16E-07 | 2.32E-08 | 5.41E-08 |
| **AM243** | 8.85E+12 | 1.21 | 0.97 | 0.81 | | 1.92E-08 | 2.05E-08 | 4.83E-08 |
| **CM242** | 1.22E+15 | 1.16 | 0.93 | 0.94 | 0.97 | 2.04E-07 | 2.10E-07 | 2.55E-07 |
| **CM243** | 8.36E+12 | 1.23 | 0.21 | 1.56 | | 2.69E-08 | 2.96E-08 | 7.61E-08 |
| **CM244** | 1.27E+15 | 1.06 | 1.03 | 0.98 | 0.93 | 3.66E-06 | 3.75E-06 | 3.92E-06 |
| **CM246** | 3.89E+10 | 0.72 | 1.17 | 1.11 | | 7.71E-04 | 7.82E-04 | 8.05E-04 |
| **CM248** | 9.73E+05 | 0.32 | 2.16 | 0.52 | | 2.61E-01 | 2.55E-01 | 2.77E-01 |
| **CF250** | 6.48E+07 | 0.26 | 2.01 | 0.74 | | 2.71E-03 | 2.73E-03 | 0.00E+00 |
| **CF252** | 7.10E+07 | 0.05 | 2.54 | 0.41 | | 1.16E-01 | 1.19E-01 | 1.19E-01 |

Ratios are given relative to the mean of the calculated decay rates for each isotope.

## Table 16. Problem 6. Gamma-ray source spectrum for 5 sub-assemblies (gamma/s)

| E max | E min | UK FISPIN | ITALY ORIGEN-S | US SAS2H | BELGIUM ORIGEN-2 | JAPAN ORIGEN 2-82 | FRG OREST |
|---|---|---|---|---|---|---|---|
| 4.250 | 4.000 | | | | | | 7.02E+07 |
| 4.000 | 3.500 | | | 4.47E+08 | 5.19E+11 | | 4.02E+08 |
| 3.500 | 3.130 | 3.37E+11 | 5.44E+11 | 5.90E+11 | | 6.10E+11 | 6.61E+11 |
| 3.130 | 3.000 | | | | | | |
| 3.000 | 2.600 | | | 3.53E+12 | 3.56E+12 | 4.77E+12 | 3.95E+12 |
| 2.600 | 2.500 | 3.27E+12 | 4.37E+12 | | | | |
| 2.500 | 2.200 | | | 2.16E+13 | | | 3.37E+12 |
| 2.200 | 2.000 | 1.29E+14 | 1.44E+14 | | 1.36E+14 | 1.42E+14 | 1.63E+14 |
| 2.000 | 1.800 | | | 1.64E+14 | | | |
| 1.800 | 1.660 | 2.23E+13 | 5.42E+13 | | 1.01E+14 | 1.53E+14 | |
| 1.660 | 1.500 | | | 4.67E+14 | | | 5.03E+14 |
| 1.500 | 1.440 | 1.40E+14 | 4.91E+14 | | | | |
| 1.440 | 1.350 | | | | | | |
| 1.350 | 1.220 | 8.02E+14 | | | 1.68E+15 | 1.89E+15 | |
| 1.220 | 1.000 | 9.44E+14 | 1.85E+15 | 2.90E+15 | | | 1.77E+15 |
| 1.000 | 0.900 | | | | | | |
| 0.900 | 0.800 | 1.40E+15 | 4.55E+15 | | 8.68E+15 | 9.12E+15 | |
| 0.800 | 0.700 | | | | | | |
| 0.700 | 0.600 | 2.94E+16 | 1.99E+16 | 3.58E+16 | | | 3.78E+16 |
| 0.600 | 0.475 | | | | 2.84E+16 | 2.95E+16 | |
| 0.475 | 0.450 | 7.45E+15 | 1.32E+16 | | | | |
| 0.450 | 0.400 | | | | 7.10E+14 | | |
| 0.400 | 0.375 | 2.23E+13 | 1.63E+15 | | | 2.66E+15 | |
| 0.375 | 0.300 | | | 3.65E+15 | | | 1.32E+16 |
| 0.300 | 0.225 | 9.42E+13 | 2.17E+15 | | | | |
| 0.225 | 0.200 | | | | 6.28E+14 | 4.56E+15 | 9.26E+11 |
| 0.200 | 0.175 | | | | | | |
| 0.175 | 0.150 | | | 3.11E+15 | | | |
| 0.150 | 0.140 | 2.44E+15 | 7.96E+15 | | 2.64E+15 | | 1.79E+11 |
| 0.140 | 0.125 | | | 5.36E+15 | | 5.37E+15 | |
| 0.125 | 0.100 | | | | | | 1.33E+13 |
| 0.100 | 0.070 | 4.84E+14 | 8.34E+15 | 4.68E+15 | 1.06E+15 | 5.10E+15 | |
| 0.070 | 0.050 | | | | 8.97E+14 | 7.28E+15 | 7.24E+13 |
| 0.050 | 0.045 | | | 7.74E+15 | | | |
| 0.045 | 0.040 | | | | 3.38E+15 | | 1.84E+12 |
| 0.040 | 0.035 | 3.41E+14 | 2.58E+16 | | | 7.75E+15 | |
| 0.035 | 0.030 | | | 1.35E+16 | | | 1.94E+14 |
| 0.030 | 0.025 | | | | 1.44E+15 | 8.09E+15 | |
| 0.025 | 0.020 | | | | | | |
| 0.020 | 0.010 | 4.52E+11 | | 3.21E+16 | 3.95E+15 | 3.45E+16 | |
| 0.010 | 0.000 | 7.09E+13 | | | | | |
| 3.000 | 0.400 | 4.03E+16 | 4.02E+16 | 3.94E+16 | 3.97E+16 | 4.08E+16 | 4.02E+16 |
| TOTAL | | 4.37E+16 | 8.60E+16 | 1.09E+17 | 5.37E+16 | 1.16E+17 | 5.37E+16 |

## Table 17. Problem 5 radial neutron

| METHOD | CODE | CONTRIBUTOR | DOSE RATE AT SURFACE (microSv/h) | DOSE 1m | ATTENUATION 2m | 10m |
|---|---|---|---|---|---|---|
| **Values on the mid-plane** | | | | | | |
| 1D Sn | ANISN | Belgium (a) | 1135 | | | |
| | SAS 2 | Italy (a) | 1710 | 0.287 | 0.143 | 0.0990 |
| 2D | DORT | US (a) | 1515 | | | |
| | DORT | US (b) | 4120 | | | |
| Monte Carlo (Pt) | MCNP | Italy (a) | 1470 (1%) | 0.292 | 0.140 | 0.0100 |
| | MCNP | Italy (b) | 3960 (2%) | 0.214 | 0.093 | 0.0060 |

## Table 18. Problem 5 radial FP gamma

| METHOD | CODE | CONTRIBUTOR | DOSE RATE AT SURFACE (microSv/h) | DOSE 1m | ATTENUATION 2m | 10m |
|---|---|---|---|---|---|---|
| **Values on the mid-plane** | | | | | | |
| Kernel | QAD-CG | Belgium(b) | 479 | 0.386 | 0.213 | 0.0209 |
| 1D Sn | SAS 2 | Italy (a) | 506* | 0.368 | 0.209 | 0.0178 |
| 2D | DORT | US (a) | 403 | 0.370 | 0.210 | |
| | DORT | US (b) | 553 | | | |
| Monte Carlo (Pt) | MCNP | Italy (a) | 497 (1%) | 0.376 | 0.213 | 0.0199 |
| | MCNP | Italy (b) | 732 (2%) | 0.335 | 0.173 | 0.0148 |

## Table 19. Problem 5 radial secondary gamma

| METHOD | CODE | CONTRIBUTOR | DOSE RATE AT SURFACE (microSv/h) | DOSE 1m | ATTENUATION 2m | 10m |
|---|---|---|---|---|---|---|
| **Values on the mid-plane** | | | | | | |
| 1D Sn | | | | | | |
| 2D | | | | | | |
| Monte Carlo (Gp) | | | | | | |
| Monte Carlo (Pt) | MCNP | Italy (a) | 10.1 (1%) | 0.293 | 0.145 | 0.0106 |
| | MCNP | Italy (b) | 22.8 (2%) | 0.245 | 0.108 | 0.0071 |

(a) Uniform axial burn up
(b) Axially dependent burn up

## Table 20. Problem 6 radial neutron

| METHOD | CODE | CONTRIBUTOR | DOSE RATE AT SURFACE (microSv/h) | DOSE ATTENUATION 1m | 2m | 10m |
|---|---|---|---|---|---|---|
| **Values on the mid-plane** | | | | | | |
| 1D Sn | SAS 2 | Italy (a) | 6610 | 0.287 | 0.142 | 0.0100 |
| 2D | | | | | | |
| Monte Carlo (Gp) | | | | | | |
| Monte Carlo (Pt) | MCNP | Italy (a) | 5590 (1%) | 0.297 | 0.147 | 0.0109 |
| | MCNP | Italy (b) | 10400 (2%) | 0.232 | 0.104 | 0.0070 |

## Table 21. Problem 6 radial FP gamma

| METHOD | CODE | CONTRIBUTOR | DOSE RATE AT SURFACE (microSv/h) | DOSE ATTENUATION 1m | 2m | 10m |
|---|---|---|---|---|---|---|
| **Values on the mid-plane** | | | | | | |
| Kernel | QAD-CG | Belgium(b) | 506 | | | |
| 1D Sn | SAS 2 | Italy (a) | 560* | 0.364 | 0.207 | 0.0170 |
| 2D | | | | | | |
| Monte Carlo (Gp) | | | | | | |
| Monte Carlo (Pt) | MCNP | Italy (a) | 527 (1%) | 0.370 | 0.209 | 0.0200 |
| | MCNP | Italy (b) | 774 (3%) | 0.377 | 0.177 | 0.0150 |

* includes secondary gammas

## Table 22. Problem 6 radial secondary gamma

| METHOD | CODE | CONTRIBUTOR | DOSE RATE AT SURFACE (microSv/h) | DOSE ATTENUATION 1m | 2m | 10m |
|---|---|---|---|---|---|---|
| **Values on the mid-plane** | | | | | | |
| 1D Sn | | | | | | |
| 2D | | | | | | |
| Monte Carlo (Gp) | | | | | | |
| Monte Carlo (Pt) | MCNP | Italy (a) | 38.8(1%) | 0.294 | 0.144 | 0.0106 |
| | MCNP | Italy (b) | 67.1 (3%) | 0.244 | 0.108 | 0.0073 |

(a) Uniform axial burn up
(b) Axially dependent burn up

**Table 23. Thermal neutron flux distributions for the end fittings of the fuel elements.**

Thermal neutron flux at the top of the fuel element.

| Distance above the active fuel (cm) | Thermal Neutron Flux ($n/cm^2s$) |
|---|---|
| 0.0 | 1.89E+13 |
| 2.5 | 2.09E+13 |
| 5.0 | 1.62E+13 |
| 7.5 | 1.60E+13 |
| 10.0 | 1.35E+13 |
| 12.5 | 1.08E+13 |
| 15.0 | 9.58E+12 |
| 17.5 | 8.73E+12 |
| 20.0 | 6.47E+12 |
| 22.5 | 4.64E+12 |
| 25.0 | 3.67E+12 |
| 27.5 | 3.16E+12 |
| 30.0 | 2.53E+12 |
| 32.5 | 1.91E+12 |
| 35.0 | 1.50E+12 |
| 37.5 | 1.14E+12 |
| 40.0 | 6.61E+11 |
| 42.5 | 3.33E+11 |
| 45.0 | 2.44E+11 |
| 47.5 | 1.72E+11 |
| 50.0 | 1.35E+11 |
| 51.8 | 1.32E+11 |

Thermal neutron flux at the bottom of the fuel element.

| Distance below the active fuel (cm) | Thermal Neutron Flux ($n/cm^2s$) |
|---|---|
| 0.0 | 2.22E+13 |
| 2.5 | 4.10E+13 |
| 5.0 | 5.45E+13 |
| 7.5 | 5.30E+13 |
| 10.0 | 4.49E+13 |
| 12.5 | 3.60E+13 |
| 15.0 | 3.18E+13 |
| 17.5 | 2.90E+13 |
| 20.0 | 2.15E+13 |
| 22.5 | 1.54E+13 |
| 25.0 | 1.22E+13 |
| 27.5 | 1.05E+13 |
| 30.0 | 8.40E+12 |
| 32.5 | 6.35E+12 |
| 35.0 | 4.98E+12 |
| 37.5 | 3.80E+12 |
| 40.0 | 2.39E+12 |
| 42.5 | 1.40E+12 |
| 45.0 | 1.14E+12 |
| 47.5 | 8.80E+11 |
| 50.0 | 6.20E+11 |
| 52.5 | 5.07E+11 |
| 55.0 | 4.60E+11 |
| 57.5 | 3.95E+11 |
| 60.0 | 3.68E+11 |
| 60.8 | 3.46E+11 |

# Table 24. Co-60 production in top end fittings (Gualdrini et al)

| Assembly length | | Volume | Co-59 | | Flux | Activity | Source strength | |
|---|---|---|---|---|---|---|---|---|
| (cm) | (cm) | ($cm^3$) | (g) | ($10^{22}$ atoms) | ($10^{11}$ n/$cm^2$s) | ($10^{12}$ dis/s) | $10^{12}$ g/s | % |
| 171.00 | 176.00 | 25153 | 0.660 | 0.67 | 192.00 | 1.29 | 2.58 | 5.90 |
| 176.00 | 181.00 | 25153 | 0.660 | 0.67 | 154.00 | 1.03 | 2.06 | 4.71 |
| 181.00 | 187.00 | 30159 | 0.800 | 0.82 | 109.00 | 0.89 | 1.78 | 4.07 |
| 187.00 | 188.80 | 9048 | 3.700 | 3.78 | 90.00 | 3.40 | 6.80 | 15.55 |
| 188.80 | 189.80 | - | - | - | 76.00 | - | - | - |
| 189.80 | 192.25 | 12566 | 5.625 | 5.75 | 64.70 | 3.72 | 7.44 | 17.00 |
| 192.25 | 194.75 | 12566 | 5.625 | 5.75 | 46.40 | 2.67 | 5.34 | 12.20 |
| 194.75 | 197.25 | 12566 | 5.625 | 5.75 | 36.70 | 2.11 | 4.22 | 9.65 |
| 197.25 | 199.75 | 12566 | 5.625 | 5.75 | 31.60 | 1.82 | 3.64 | 8.32 |
| 199.75 | 202.25 | 12566 | 5.625 | 5.75 | 25.30 | 1.45 | 2.90 | 6.63 |
| 202.25 | 204.75 | 12566 | 5.625 | 5.75 | 19.10 | 1.10 | 2.20 | 5.03 |
| 204.75 | 207.25 | 12566 | 5.625 | 5.75 | 15.00 | 0.86 | 1.72 | 3.93 |
| 207.25 | 209.75 | 12566 | 5.625 | 5.75 | 11.40 | 0.66 | 1.32 | 3.02 |
| 209.75 | 212.25 | 12566 | 5.625 | 5.75 | 6.61 | 0.38 | 0.76 | 1.74 |
| 212.25 | 214.75 | 12566 | 5.625 | 5.75 | 3.33 | 0.19 | 0.38 | 0.87 |
| 214.75 | 217.25 | 12566 | 5.625 | 5.75 | 2.44 | 0.14 | 0.28 | 0.64 |
| 217.25 | 219.75 | 12566 | 5.625 | 5.75 | 1.72 | 0.10 | 0.20 | 0.46 |
| 219.75 | 221.80 | 10053 | 4.500 | 4.60 | 1.35 | 0.06 | 0.12 | 0.28 |

Irradiation time 1200 days, decay time 2 years

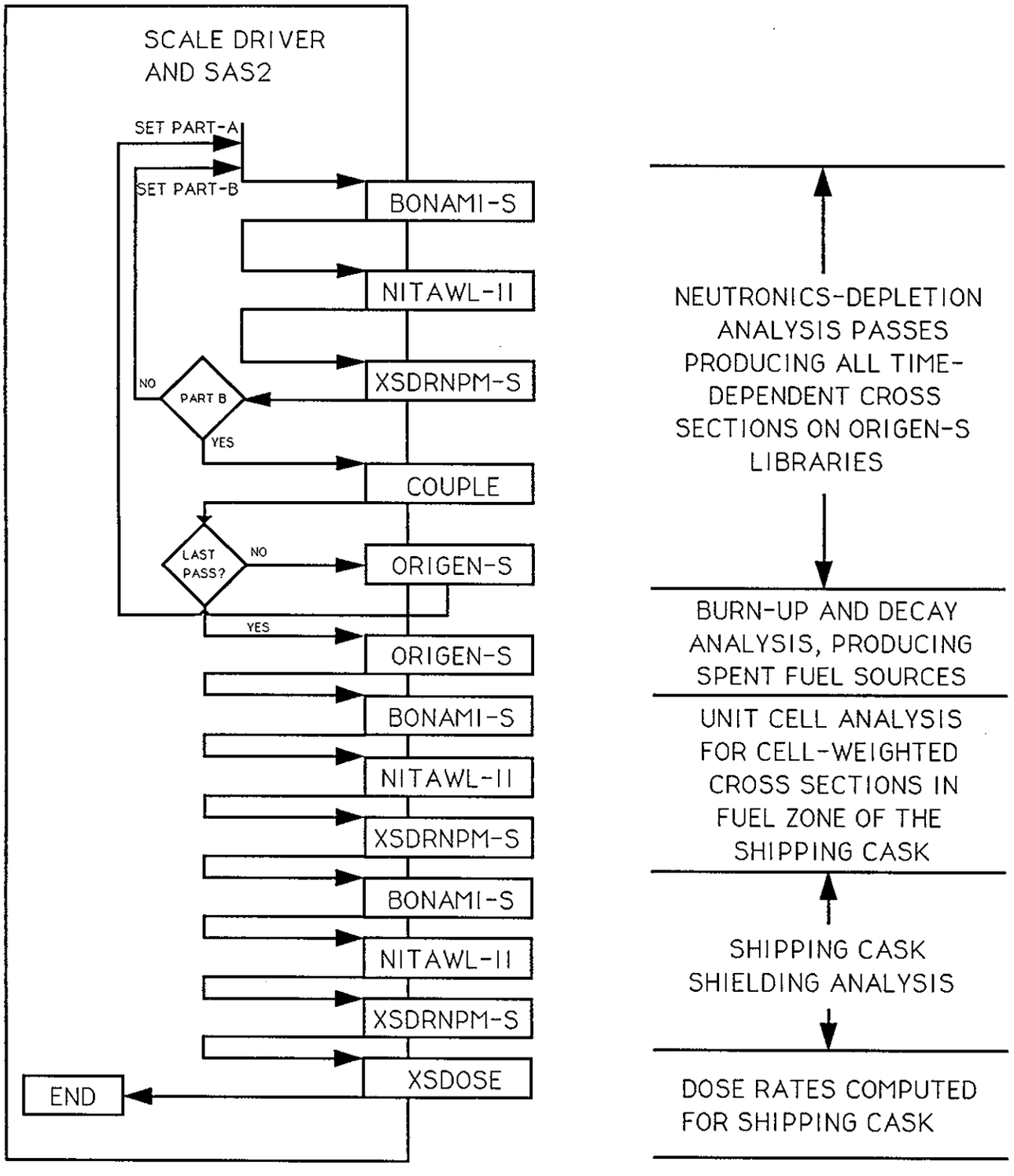


**Figure 1. Basic Flow Invoked by SAS2H**