PBMR FUEL SPHERE SOURCE TERMS

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ABSTRACT: A thorough evaluation of the fuel sphere source terms, that is, the determination of the nuclide inventories, photon sources and decay heat, is fundamental to any reactor design. The accurate calculation of the fuel sphere source terms depends to a great extent on the one-energy-group neutron spectrum averaged cross-sections used for the fuel depletion analyses. The continuous reloading scheme of the Pebble Bed Modular Reactor (PBMR) necessitates special considerations, since the fuel is not stationary. As a fuel sphere moves through the reactor, the neutron flux spectra changes continuously due to, amongst other reasons, temperature, neutron leakage, spatial position, material composition and reactor power. In this paper the behaviour of the one-group microscopic cross-sections, as the fuel circulates through the core, is evaluated and the effect on the fuel sphere source terms quantified. VSOP-99 and the ORIGEN-JÜl-99 code system, which uses time-dependent multi-pass reactor data generated by VSOP-99, are used. From the results it is clear that the different rates at which the fuel spheres move through the reactor core, and the different flux values and flux spectra in the different areas in the core, cause the fuel depletion to become largely dependent on the space time history of the fuel sphere. A simplified one-dimensional reactor model in SCALE was used to generate for the average equilibrium core conditions, a single ORIGEN-S cross-section library. The ORIGEN-S fuel depletion with this cross-section library is probably acceptable for scoping evaluations, but should be used with caution for specific reactor design applications.

KEYWORDS: PBMR; Decay Heat; Depletion; ORIGEN-JÜl; VSOP-99; ORIGEN-S

0. INTRODUCTION

The accurate prediction and uncertainty estimation of reactor fuel source terms is fundamental to any reactor design. For the Pebble Bed Modular Reactor (PBMR), the determination of the fuel sphere nuclide inventories, photon sources and decay heat has a direct influence on various design aspects, for example:

- Nuclide inventory evaluation for taking credit for burn-up in the subcriticality design of the used- and spent fuel tanks.
- Decay heat sources needed for core cool-down analysis and the design of the used- and spent fuel tanks.
- Photon sources needed to design radiation shields for all human activities close to any fuel sphere transport or storage facilities.
- Nuclide inventories required for the fission product release analyses for maintenance input as well as reactor licensing application.
- Design of the fuel sphere burn-up measurement system and the distinction between fuels spheres of different initial enrichment.
- Long-term waste and decommissioning.
- Non-proliferation reasons.

The PBMR design applies a continuous reloading scheme where unloaded fuel spheres, which have not reached the target burn-up, are returned to the top of the core. This multi-pass fuel circulation is often referred to as a MEDUL cycle. For a so-called six-pass equilibrium core, this implies that the

fuel spheres will circulate six times, on average, through the core. As a fuel sphere moves through the reactor, the neutron flux spectra changes continuously for reasons including temperature, spatial position, neutron leakage, material composition and reactor power. Therefore, the fuel depletion, which is directly dependent on the neutron flux spectra and the neutron flux, becomes largely spatial dependent. Thus, the fuel depletion for the same fuel sphere will be different in different positions within the reactor, even if the total neutron flux or power in the two positions is the same.

Various computer codes and data are available to perform fuel depletion analyses, thus calculating the fuel sphere source terms. The accurate calculation of the fuel sphere source terms depends to a great extent on the one-energy-group neutron spectrum weighted cross-sections used for the fuel depletion analyses. In this paper the behaviour of the one-group microscopic cross-sections, as the fuel circulates through the core, is evaluated and its effect on the final fuel sphere source terms quantified. VSOP-99 [1] (referred in this paper as VSOP), and the ORIGEN-JÜL-99 [2] (referred in this paper as ORIGEN-JÜL) code system, that uses time-dependent multi-pass reactor data generated by VSOP, is used to show the neutron spectrum effects. A preliminary evaluation was performed to compare source terms calculated with the ORIGEN-S module from the SCALE4.4 [3] code system with the VSOP and ORIGEN-JÜL source terms.

In Section 0, a summary is provided of the ORIGEN-S and ORIGEN-JÜL fuel depletion and cross-section library update. Section 0 provides the flux spectra and actinide cross-sections calculated with VSOP, illustrating the neutron spectra changes through the reactor and the effects on the spectrum weighted cross-sections. Using the detail space-time fuel history and space dependent spectrum weighted cross-sections, Section 0 provides results on the fuel depletion for fuel spheres traveling through different areas of the reactor. In Section 0, the use of a single reactor averaged cross-section library for ORIGEN-S fuel depletion is evaluated. Section 0 contains the conclusions and recommendations for future work.

1. ORIGEN-S AND ORIGEN-JÜL

The original version of the ORIGEN program [4] was developed to calculate the source terms for reactor materials for various reactor types. With the version of ORIGEN applied in the SCALE system, namely ORIGEN-S, the user has the option to update the basic ORIGEN cross-section library for the specific fuel type and reactor design. However, the SCALE neutron flux solver, XSDRNPM, only allows a one-dimensional representation of the reactor.

The ORIGEN-JÜL code is a derivation from the ORIGEN code. The main enhancement of the ORIGEN-JÜL code is the increased flexibility and accuracy by using a USER-library. This enables the user to define the irradiation history of the fuel being investigated, thus performing the fuel depletion using time-dependent neutron fluxes and cross-sections. The USER-library can be created by VSOP and includes the history (position dependent irradiation time, few-group neutron flux, few-group cross-sections) of the fuel as it passes through the reactor. In essence, a time-dependent spectrum that includes all temperature and environment effects is therefore used in the ORIGEN-JÜL depletion calculations.

The basic methodology used in ORIGEN is that neutron energy integrated interaction rate is related to a thermal input flux, three-energy group cross-sections and three spectral indices THERM, RES and FAST. The three-group energy ranges defined in ORIGEN-S is the thermal range ($E_{th} < 0.5$ eV), the resonance range (0.5 eV $< E_{res} <$ 1.0 MeV) and the fast range ($E_{fast} >$ 1.0 MeV). Typically the user defines the cross-sections of the most important actinides and selected fission product and other nuclides to be updated in the ORIGEN library. The updated three-group cross-sections must form a consistent set with the spectral indices calculated for the specific fuel and reactor type. Including a 1/v absorber material in the SCALE spectrum calculation, or B-10 as the 1/v material in the VSOP generated USER-library, the methodology followed in SCALE and ORIGEN-JÜL to calculate the spectral indices and to update the cross-sections for specific user specified isotopes is in principle similar.

Within ORIGEN-S the effective cross-section, σ_{eff} , is defined and approximated as

$$\sigma_{eff} = \int_0^\infty \sigma(E)\phi(E)\,dE \Big/ \int_{th} \phi(E)\,dE$$

$$\cong THERM \cdot \sigma_0 + RES \cdot I + FAST \cdot \sigma_1 \qquad (1)$$

where σ_0 the 2 200 m.s^{-1} neutron reaction cross-section, **I** the resonance integral (> 0.5 eV) and σ_1 the fission spectrum averaged cross-section (> 1.0 MeV). With the flux defined as

$$\phi_x = \int_x \phi(E)\,dE \qquad (2)$$

THERM, RES and FAST are defined in ORIGEN-S as:

$$THERM = \sigma_{1/v}^{th} = \int_{th} \sigma_{1/v}(E)\,\phi(E)\,dE \cdot \frac{1}{\phi_{th}}$$

$$RES = \frac{1}{\lambda n(10^6/0.5)} \cdot \frac{\phi_{res}}{\phi_{th}} \qquad (3)$$

$$FAST = 1.45\,\frac{\phi_{fast}}{\phi_{th}}$$

Calculating ϕ_{th}, ϕ_{res} and ϕ_{fast}, using for example the SCALE module XSDRNPM, the THERM, RES and FAST factors in the ORIGEN-S library are updated with the SCALE module COUPLE, thus having an updated σ_{eff} cross-section when using Eq. 1.

Furthermore, having calculated the flux spectrum averaged cross-section σ_{th}, σ_{res} and σ_{fast} as

$$\sigma_x = \int_x \sigma(E)\phi(E)\,dE \Big/ \int_x \phi(E)\,dE \qquad (4)$$

the ORIGEN-S library cross-sections σ_0, I and σ_1 can be replaced for particular user-specified isotopes. Using the THERM, RES and FAST factors defined in Eq. 3, σ_0, I and σ_1 are then calculated as

$$\sigma_0 = \sigma_{th}/THERM = \sigma_{th}/\sigma_{1/v}^{th}$$

$$I = \sigma_{res}\frac{\phi_{res}}{\phi_{th}} \Big/ RES = \sigma_{res}\,\lambda n(10^6/0.5) \qquad (5)$$

$$\sigma_1 = \sigma_{fast}\frac{\phi_{fast}}{\phi_{th}} \Big/ FAST = \frac{1}{1.45}\sigma_{fast}$$

In this study, no further attention has been given to evaluate the implementation differences between SCALE and ORIGEN-JÜL/VSOP for the calculation of the spectral indices and the isotope cross-section update of the ORIGEN library. Also, differences may exist between the specific isotopes that were updated, and this selection must still be evaluated for the PBMR design. The basic data that originated from different cross-section evaluations also needs to be compared. Since it is understood that a complete data update has been performed for the ORIGEN-S basic libraries in the SCALE-5 release, this should also be evaluated for use in future evaluations and comparisons.

2. VSOP PBMR REACTOR ANALYSIS

The PBMR core design consists of an annular fuel region with a fixed central graphite column and outer graphite reflector. The neutron behaviour and fuel management within the reactor is calculated with VSOP. More detail on the reactor design and VSOP calculational models can be found in [5].

The VSOP two-dimensional axisymmetric computational model in *FIGURE 1* represents the PBMR core and internals layout. For the neutronics calculations, the layout of the reactor is modelled up to the reactor core barrel. In this model, five channels have been specified to simulate the fuel flow paths and flow rates as the fuel circulates through the reactor. The reactor core is divided into 79 regions, each of which contains a mixture of six different fuel element batches. These batches represent the various passes through the core, i.e. from the fresh state to a total of six times through the core. The burn-up of each of these fuel element batches is followed individually. The 79 regions also define the flow rates, or in other words the residence time, in each of the five channels. To simulate the slower flow of the fuel spheres adjacent to the reflector walls, the two outer channels are each made up of 20 regions compared to the 13 regions for the three inner channels. The fuel residence time in channels 1 and 5 will thus be 54% longer (20/13) than for Channel 2 to 4. For this model, each of the 79 core regions was specified as a separate spectrum zone for the fuel sphere spectrum calculations. The VSOP reactor calculations are performed in four energy groups, namely from 10 MeV – 0.1 MeV – 29 eV – 1.86 eV – 0 eV.

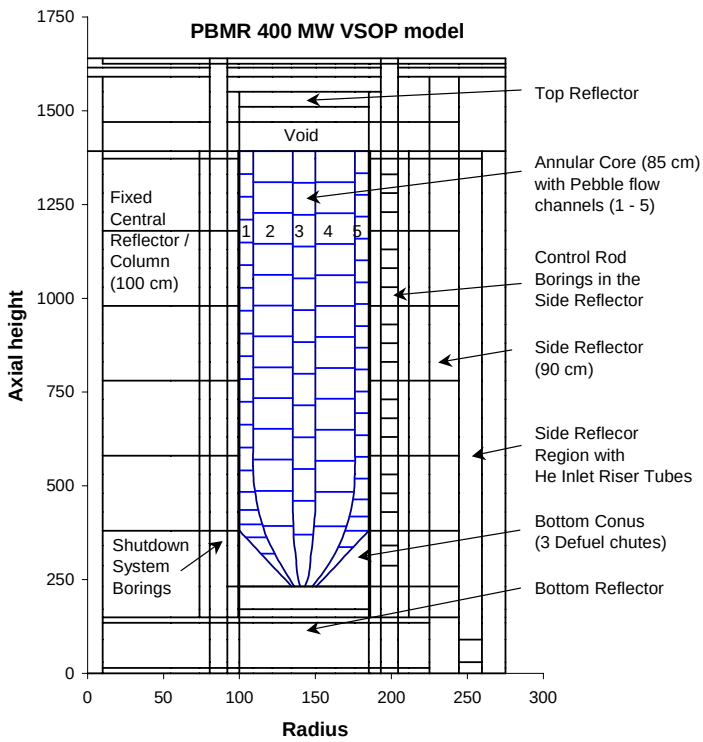

FIGURE 1. VSOP core geometrical model (R-Z cut) showing five flow channels in the annular core, the reflector regions and positions of reactivity control systems.

The fuel depletion in a region is dependent on the neutron flux, the fuel exposure time and the effective one-group neutron interaction cross-section (calculated from the few-group microscopic cross-section and neutron flux spectrum) within the region. The rest of the figures in this section illustrate the spatial dependence of these parameters. The region numbering in these figures represents the region numbers in the different flow channels. Regions 1 to 20 show the axial profile (from the top of the reactor to the bottom) of Channel 1 that is the annulus adjacent to the central reflector column. Regions 21 to 33 represent Channel 2, and so on. Channel 3 represents the annulus in the centre of the fuel annulus, and Channel 5 is the fuel annulus adjacent to the outside reflector (represented by regions 60 to 79).

FIGURE 2 presents the four-group fluxes (from high to low energy) for the 79 regions. Each of the five peaks shows the axial flux profile within the respective channel. From *FIGURE 2* it is clear that the axial neutron flux profile peaks towards the upper part of the reactor (where the fuel is cooler and less burned). It is also clear that the radial fast energy (groups 1 and 2) neutron flux profile peaks in the centre of the fuel annulus (channel 3), while the thermal energy (group 4) neutron flux profile peaks towards the sides of the fuel annulus (channels 1 and 5) adjacent to the graphite reflectors.

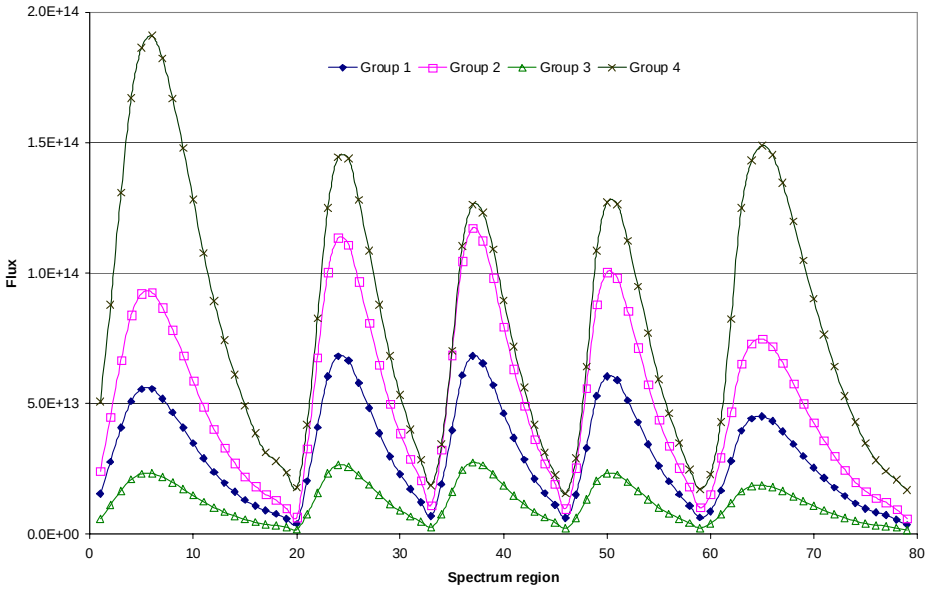

FIGURE 2. VSOP 4-energy group neutron flux for the five fuel sphere flow channels.

To illustrate the spectrum dependence within the different areas of the reactor more clearly, *FIGURE 3* presents the fraction of the thermal (E < 1.86 eV) neutrons for the different regions in the five channels. The increase in the thermal spectrum towards the bottom of each channels is due to the larger amount of graphite present as fuel enters the defuel cone, for which a more thermal spectrum can be expected. Similarly, the neutron spectrum is more thermal for channels 1 and 5 next to the graphite reflector, with a harder spectrum (less thermal) for channels 2 to 4, which are towards the centre of the fuel annulus.

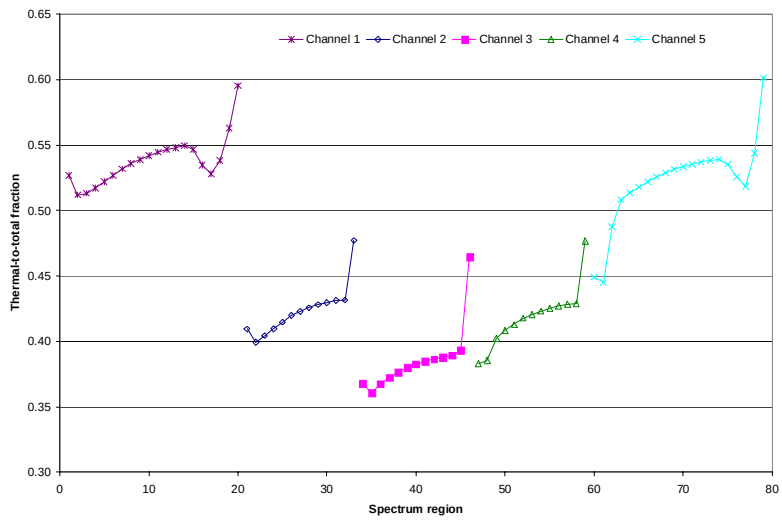

FIGURE 3. Thermal neutron flux fraction for the five fuel sphere channels.

The spatial dependence of the one-group spectrum weighted cross-sections is illustrated for a few important actinides in *FIGURE 4* to *FIGURE 6*.

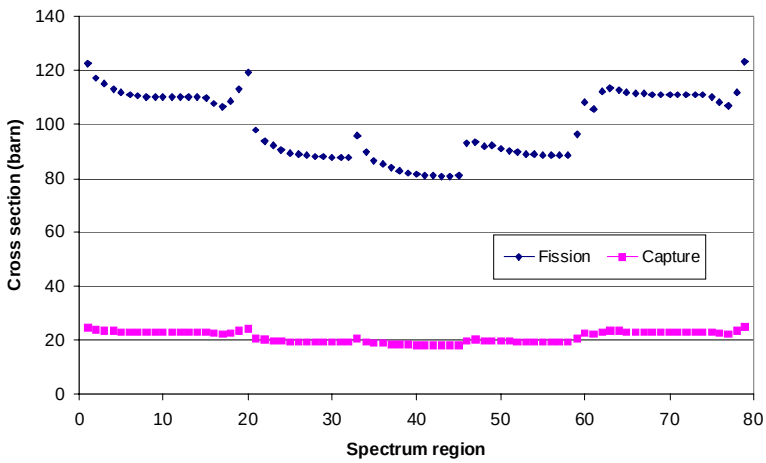

FIGURE 4. U-235 one-energy group spectrum weighted cross-sections in the reactor core.

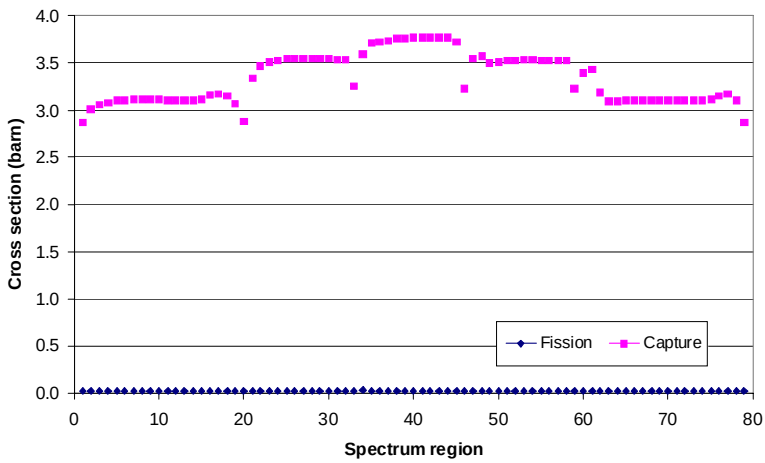

FIGURE 5. U-238 one-energy group spectrum weighted cross-sections in the reactor core.

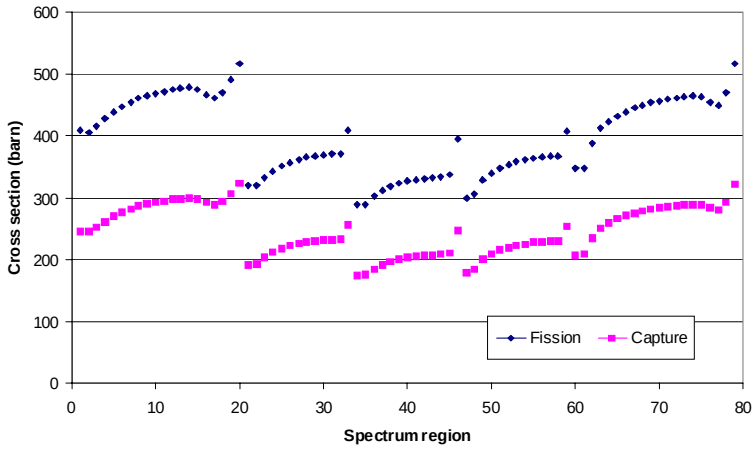

FIGURE 6. Pu-239 one-energy group spectrum weighted cross-sections in the reactor core.

FIGURE 4 to *FIGURE 6* clearly illustrate the spatial dependence of the one-group cross-sections as calculated with VSOP. Two effects can be noticed, namely the difference in the cross-sections from channel to channel, as well as within a channel. Furthermore, the U-235 and Pu-239 cross-sections increase towards the graphite reflector, while the U-238 cross-section peaks towards the centre of the annulus.

From the VSOP results illustrated in *FIGURE 4* to *FIGURE 6*, it should be clear that the fuel depletion will be different in the different regions. In the next section, the effect of the spatially dependent fuel depletion will be quantified in terms of the fuel movement through the different channels.

3. PBMR FUEL DEPLETION

From Section 0 it is clear that the spectrum changes within the reactor have a large influence on the spectrum weighted cross-sections. Although the effects were only shown for U-235, U-238 and Pu-239, it can be expected that the one-group spectrum weighted cross-sections for most other isotopes will have similar spectrum dependencies.

For the evaluation of the fuel sphere source terms, it is important to determine to what extent the actinide masses in a fuel sphere can differ due to irradiation in the different fuel channels. In other words, to what extent the actinide masses and source terms differ for fuel spheres that have achieved the same burn-up, but have been irradiated in different channels in the reactor. With VSOP, the fuel sphere masses as a function of burn-up have been calculated for the following scenarios:

- Core: The ‘normal’ VSOP equilibrium model where fuel spheres being recycled are mixed before being reintroduced in all the flow channels. Fuel depletion histories are thus restricted to a single pass with earlier burn-up histories being averaged.
- Channel x: Fuel spheres are repeatedly recycled in Channel x until the discharge burn-up is achieved, where x is respectively 1 to 5. No mixing of fuel being reintroduced takes place, and a fuel sphere’s complete burn-up history was spent in the single channel spectrum and residence time.

The isotopic masses calculated for these fuel-recycling scenarios are presented in *FIGURE 7* to *FIGURE 9* for U-238, Pu-239 and Pu-241.

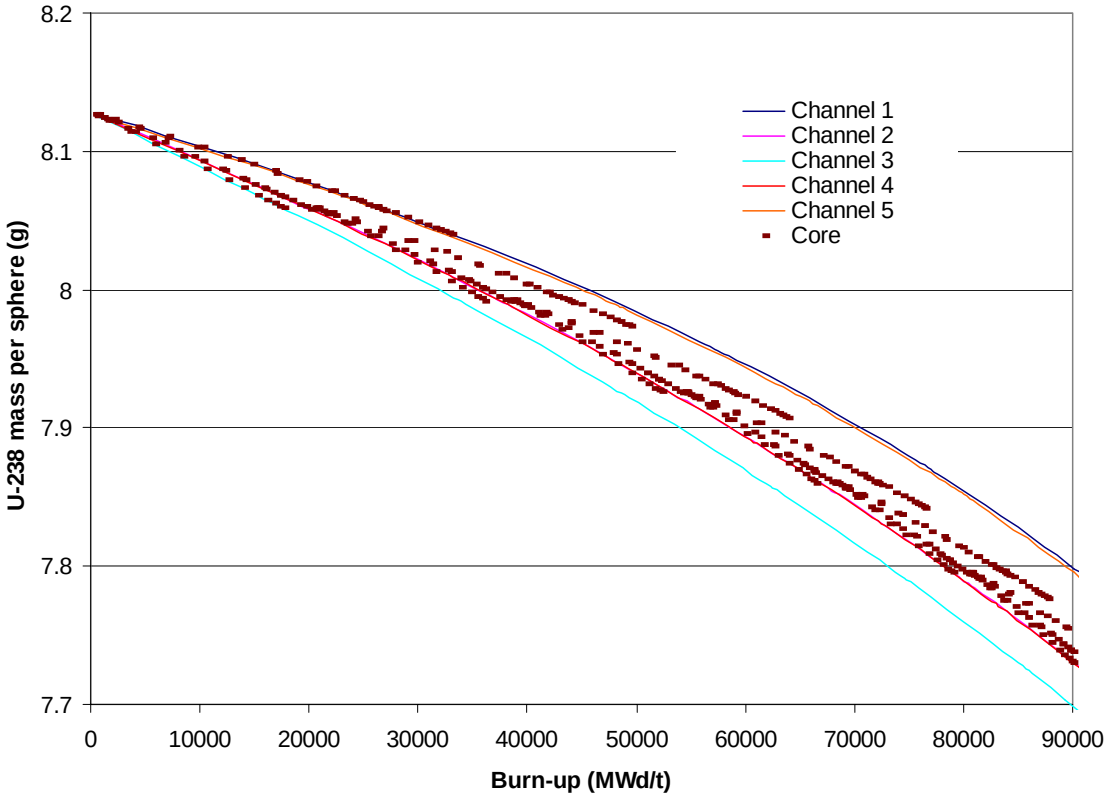

FIGURE 7. U-238 mass per fuel sphere as a function of burn-up for the equilibrium core fuel cycling and for fuel passing multiple times through a single channel.

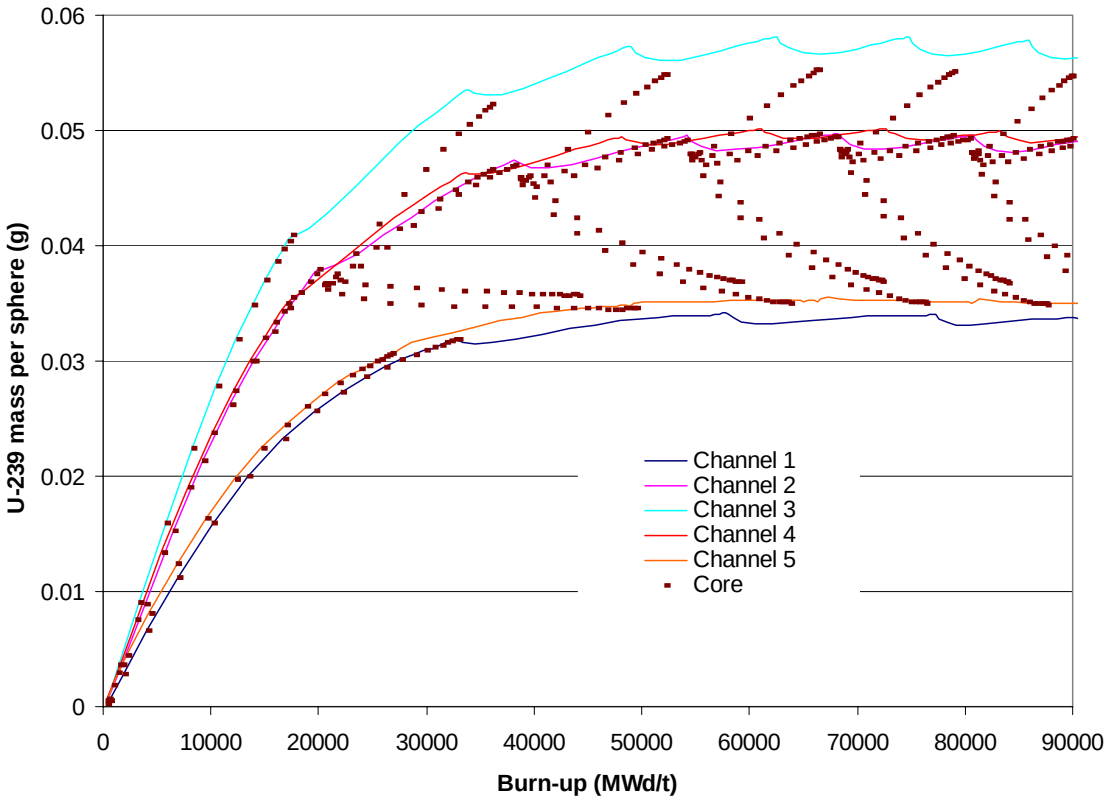

FIGURE 8. Pu-239 mass per fuel sphere as a function of burn-up for the equilibrium core fuel cycling and for fuel passing multiple times through a single channel.

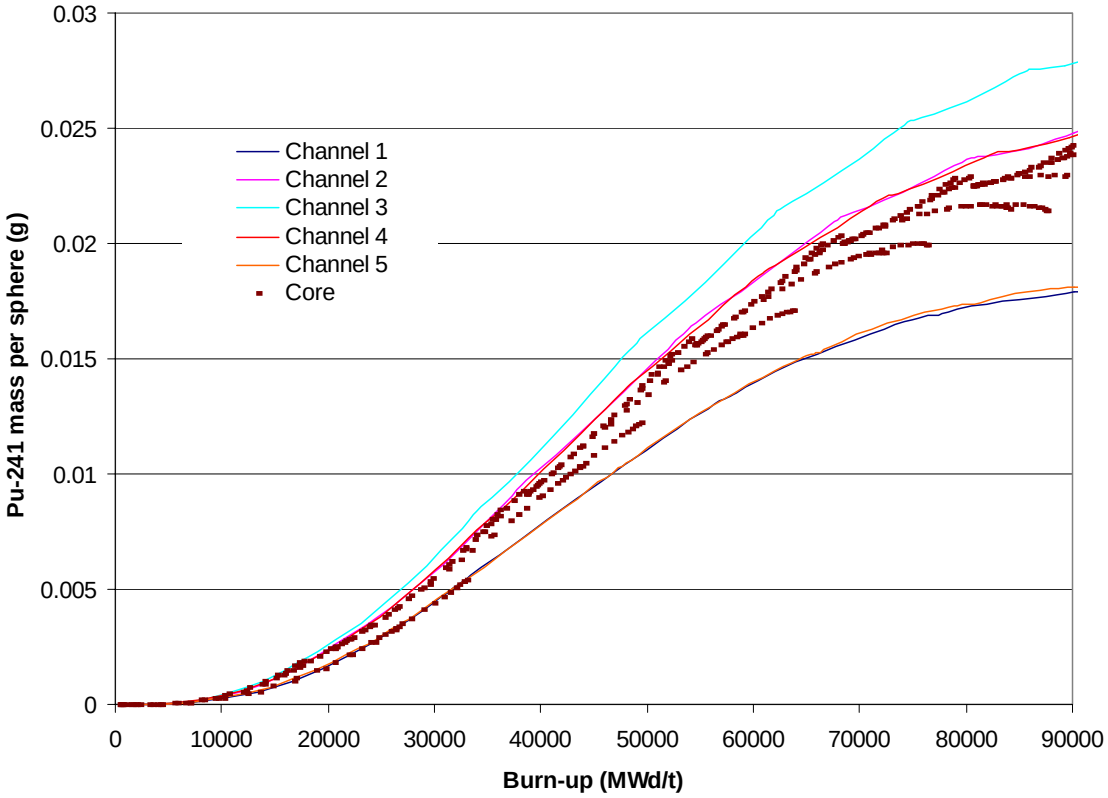

FIGURE 9. Pu-241 mass per fuel sphere as a function of burn-up for the equilibrium core fuel cycling and for fuel passing multiple times through a single channel.

FIGURE 7 to *FIGURE 9* clearly show the build-up (Pu-239, Pu-241) and depletion (U-238) as a function of burn-up and the dependence on the actual channel in which the fuel irradiation took place. They also show how the mixing in the 'Core' case leads to a reduced spread in values. In all cases the channel 1 and channel 3 actinide masses provide the boundary cases for the build-up and depletion. The Pu-239 case shows the VSOP equilibrium core fuel cycling behaviour quite clearly, indicating the averaging of the fuel sphere compositions when being recycled back to the top of the core. This example also shows the effect of a balance between the Pu-239 formed and removed.

From these few graphs it is clear that the actinide masses within a fuel sphere can differ considerably for the same burn-up. The spread of the scattered graph for the equilibrium VSOP equilibrium core case provides an indication that a large number of the fuel spheres at the same burn-up will have a large spread in mass for a specific isotope. In the case of Pu-239, it can probably be concluded that the Pu-239 content of all fuel spheres with a burn-up above 4 000 MWd/t will fall within a flat but wide range with almost a factor 2 between the minimum and maximum concentration.

Using ORIGEN-JÜL, different fuel sphere source terms have been calculated for the same VSOP fuel recycling cases as described above. For illustration purposes, *TABLE 1* provides the activities for a few isotopes, the total activity, the photons per second and the decay heat. The table also provides the number of fuel cycles that was needed to reach a minimum burn-up of 91.37 GWd/t. The percentage values were determined with respect to the core average values.

TABLE 1. Fuel sphere discharge activities (Bq) and other parameters for zero days' decay.

Isotope	Core (Bq)	Channel 1		Channel 2		Channel 3		Channel 4		Channel 5	
		(Bq)	%	(Bq)	%	(Bq)	%	(Bq)	%	(Bq)	%
Sr-90	8.04E+10	8.81E+10	10	8.46E+10	5	7.97E+10	-1	8.38E+10	4	8.59E+10	7
Zr-95	9.48E+11	6.54E+11	-31	9.44E+11	0	9.64E+11	2	8.35E+11	-12	5.39E+11	-43
Ru-103	6.28E+11	3.92E+11	-38	6.43E+11	2	6.63E+11	6	5.89E+11	-6	3.37E+11	-46
Ag-110m	1.41E+09	1.19E+09	-16	1.41E+09	0	1.65E+09	17	1.59E+09	13	9.92E+08	-30
I-131	1.77E+11	9.77E+10	-45	1.79E+11	1	1.75E+11	-1	1.66E+11	-6	9.49E+10	-46
I-132	2.02E+11	1.21E+10	-94	1.96E+11	-3	1.92E+11	-5	1.85E+11	-8	1.21E+11	-40
Cs-134	1.20E+11	1.36E+11	13	1.39E+11	16	1.28E+11	7	1.34E+11	12	1.24E+11	3
Cs-137	9.71E+10	1.01E+11	4	9.91E+10	2	9.84E+10	1	1.03E+11	6	9.86E+10	2
La-140	4.52E+11	2.34E+11	-48	4.70E+11	4	4.60E+11	2	4.28E+11	-5	2.23E+11	-51
Pr-144	1.00E+12	9.49E+11	-5	1.05E+12	5	9.78E+11	-2	8.84E+11	-12	7.57E+11	-24
Total	3.08E+13	2.00E+13	-35	3.01E+13	-2	2.98E+13	-3	2.81E+13	-9	1.91E+13	-38
# Passes	6	4		6		7		8		5	
Burn-up (MWd/tu)	91.37	95.46		93.10		92.74		97.48		93.67	
Photon/s	1.86E+13	1.17E+13 (-37%)		1.82E+13 (-2%)		1.77E+13 (-5%)		1.75E+13 (-6%)		1.15E+13 (-38%)	
Decay heat (W)	8.69	5.39 (-38%)		8.38 (-4%)		8.24 (-5%)		7.94 (-9%)		5.36 (-38%)	
Volume fraction	-	0.08572		0.25453		0.17674		0.34109		0.14192	

From *TABLE 1* it is clear that fuel sphere source terms can differ considerably for fuel spheres travelling through different flow channels. Since the fuel spheres have different burn-up values, they cannot be compared directly. However, in practice, not all fuel spheres will be classified as spent fuel at the same burn-up, but between a certain minimum and maximum burn-up. Therefore, the different values provide an indication of typical differences that can be expected. As expected, the total activity, the total photon/s source and the decay heat show the same trend.

A direct comparison can be made between the Channel 2 and Channel 5 values, for which the fuel spheres have reached almost the same burn-up. The reason for the much lower decay heat in Channel 5 than in Channel 2 is as follows. The Channel 5 fuel spheres' core residence time is 1.3 times longer than that of Channel 2 fuel spheres and 1.5 times longer per pass. Therefore, the fission products had more time to decay after passing through the high flux area towards the upper part of the core. Also, the flux, just before discharge, is slightly lower in Channel 5 than in Channel 2, thus the short-lived fission products make a smaller contribution to the decay heat.

From the results in Section 0, it can be concluded that the spread in the source terms for spent fuel, or fuel spheres at the same burn-up, can be considerable. This needs to be taken into account for every design aspect of the PBMR. For example, when shielding needs to be considered for a fuel sphere transfer pipe, it will not be sufficient to use the core average fuel sphere source term. In this case the maximum first pass source term will have to be determined from all the flow channels (probably channel 1). The specific source term to be used will also depend on the decay time. For longer decay times, higher burn-up will provide a conservative source term. For the storage of the fuel spheres of a complete core, the importance of hot spots (thermal or radiation) needs to be evaluated in terms of the fuel sphere space-time history in the reactor. In case credit is taken for fuel sphere burn-up in criticality calculations for fuel sphere storage design, the maximum Pu-239 and Pu-241 values as a function of burn-up (Channel 3 graphs in *FIGURE 8* and *FIGURE 9*) will be used to reduce

calculational uncertainty in comparison to the real world.

4. ORIGEN-S FUEL DEPLETION

From Section 0 it is clear that the different rates at which the fuel spheres move through the reactor core, and the different flux values and flux spectra in the different areas in the core, cause the fuel depletion to become largely dependent on the space-time history of the fuel sphere. Using VSOP and ORIGEN-JÜL, all aspects of the PBMR fuel sphere source terms can be calculated and evaluated. In addition, the use of a flexible stand-alone code such as ORIGEN-S provides the user with the option to do independent comparison calculations and also calculate core average fuel sphere source terms more easily. Furthermore, since ORIGEN-S is a more widely used code, it is exposed to much more evaluation, and therefore the basic cross-section data is more frequently updated. However, when using ORIGEN-S, the accuracy of an approximate fuel recycling with ORIGEN-S in comparison with a more realistic fuel recycling in VSOP, needs to be evaluated.

In this section a simple model to simulate the PBMR fuel depletion with ORIGEN-S was used. More realistic modelling is probably possible, but has not been further explored. The ORIGEN-S model used and the results compared with VSOP and ORIGEN-JÜL should merely be seen as a first attempt from which further analysis and evaluations can follow.

Using SCALE, an ORIGEN-S cross-section library was generated for the annular fuel region of the PBMR, making use of a one-dimensional cylindrical PBMR reactor model. *TABLE 2* provides a summary of the model for which the fuel region was divided in the five annular channels. The channel widths were selected to correspond with the VSOP model upper most channel widths, i.e. the meshes where the fuel is introduced. All fuel spheres were specified at the VSOP equilibrium core burn-up (~53 GWd/t) using the core averaged VSOP isotopic atom densities. The creation of the ORIGEN-S library for the PBMR is similar to the description in [6], with the main difference being the change from a dynamic central column to a fixed central column.

The PBMR ORIGEN-S library was evaluated by comparing different ORIGEN-S source terms with VSOP. Knowing the spent fuel target burn-up, the ORIGEN-S fuel sphere depletion calculations were performed by normalizing the axial VSOP thermal flux profile (radial averaged over channels 2 to 4) so that the target burn-up is reached after six passes. The VSOP axial thermal flux profiles for the five channels as well as the average axial thermal flux profile used for the ORIGEN-S calculation (not normalized) is presented in *TABLE 3*. Note that the flux profile itself mainly influences the source terms for short decay periods, due to the time-dependence of the fission rate on the fission products with short half-lives.

TABLE 2. PBMR cylindrical model used in SCALE.

Region	Radius (cm)	Material
Central column hole	10	Void
Central column	100	Graphite (10 % void)
Fuel region – channel 1	109	Fuel spheres (39% void)
Fuel region – channel 2	135	Fuel spheres (39% void)
Fuel region – channel 3	150	Fuel spheres (39% void)
Fuel region – channel 4	176	Fuel spheres (39% void)
Fuel region – channel 5	185	Fuel spheres (39% void)
Reflector	275	Graphite (10% void)
Barrel and RPV	328	Steel (0.55% void)

TABLE 3. VSOP channels axial thermal (< 1.86 eV) flux values.

Channel 1	Channel 2	Channel 3	Channel 4	Channel 5	Average over Channels 2 to 4
5.07E+13	4.19E+13	3.44E+13	2.90E+13	2.28E+13	3.45E+13
8.79E+13	8.28E+13	7.02E+13	6.41E+13	4.30E+13	7.17E+13
1.31E+14	1.25E+14	1.10E+14	1.09E+14	8.24E+13	1.14E+14
1.67E+14	1.44E+14	1.26E+14	1.27E+14	1.25E+14	1.33E+14
1.86E+14	1.44E+14	1.23E+14	1.27E+14	1.43E+14	1.32E+14
1.91E+14	1.28E+14	1.09E+14	1.12E+14	1.49E+14	1.17E+14
1.82E+14	1.09E+14	8.95E+13	9.49E+13	1.45E+14	9.82E+13
1.67E+14	8.80E+13	7.18E+13	7.71E+13	1.35E+14	7.95E+13
1.48E+14	6.82E+13	5.61E+13	5.94E+13	1.20E+14	6.16E+13
1.28E+14	5.32E+13	4.18E+13	4.62E+13	1.05E+14	4.75E+13
1.08E+14	4.00E+13	3.13E+13	3.49E+13	9.02E+13	3.58E+13
8.94E+13	2.84E+13	2.26E+13	2.48E+13	7.65E+13	2.55E+13
7.43E+13	1.86E+13	1.56E+13	1.71E+13	6.42E+13	1.73E+13
6.11E+13				5.29E+13	
4.92E+13				4.29E+13	
3.87E+13				3.49E+13	
3.13E+13				2.84E+13	
2.81E+13				2.41E+13	
2.36E+13				2.07E+13	
1.78E+13				1.68E+13	

The following results were obtained. The fuel masses calculated with ORIGEN-S as a function of burn-up are compared with the equilibrium core masses calculated with VSOP. *FIGURE 10* provides a presentation of the Pu-239 and Pu-241 masses as a function of burn-up. The VSOP curve has been obtained by sorting the VSOP batches as a function of burn-up. The reason for the zigzag VSOP curve is that the mass for a specific isotope can be different for the same burn-up, due to its irradiation history and spectrum.

From *FIGURE 10* it is clear that the ORIGEN-S values do not fit the VSOP curve exactly, but largely fall within the VSOP spread caused by the spatial dependent cross sections. In the spent fuel masses at 91.37 GWd/tU is tabulated for U-235, U-238, Pu-239 and Pu-241.

TABLE 4. ORIGEN-S and VSOP actinide fuel sphere mass (g) at 91.37 GWd/t

Isotope	VSOP	ORIGEN-S	% Difference
U-235	0.175	0.166	-5.1
U-238	7.73	7.72	-0.1
Pu-239	0.0480	0.0509	6.0
Pu-241	0.0238	0.0228	-4.2

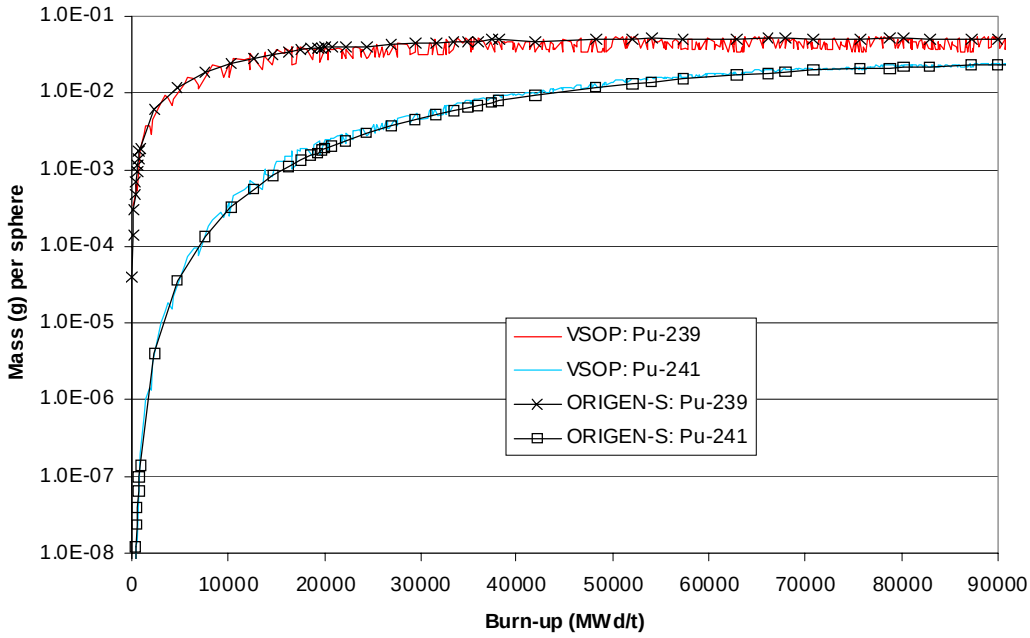

FIGURE 10. ORIGEN-S and VSOP fuel sphere mass as a function of burn-up.

The spent fuel mass differences between the ORIGEN-S and VSOP are larger than expected. Comparing the ORIGEN-S fuel region, one-group actinide isotope cross-sections with the VSOP values, the ORIGEN-S U-238 capture cross-section is about 7% larger than the VSOP value, and the ORIGEN-S U-235 fission cross-section is about 5% larger than the VSOP value. This is the main reason why the ORIGEN-S U-235 mass is lower than the VSOP value and the ORIGEN-S Pu-239 mass is higher than the VSOP value. The large differences in the one-group U-238 capture and U-235 fission cross-sections between SCALE and VSOP need further investigation, but this falls outside the scope of this paper.

Despite the differences in some of the actinide masses for the spent fuel, it is important to determine the effect on the fission product source terms. Since many of the fission product yields for different fissionable isotopes generally do not differ too much, it can be expected that the fission product inventory and source term differences between ORIGEN-S and VSOP will be smaller. The larger ORIGEN-S decay heat can partly be explained by the VSOP flux profile for Channels 2 to 4 that were used for the ORIGEN-S calculation, thus not taking the effect of the different flux profiles in Channels 1 and 5 correctly into account. This was confirmed by a reduction in the decay heat difference for 20 days' decay. The decay heat difference decreased to 9%. The differences seen for some isotopes are probably also due to differences in the one-group cross-sections of the fission product, their activation and daughter isotopes.

TABLE 5 provides, as an example, the activities (Bq) of selected isotopes and decay heat (watt) calculated for a fuel sphere at 91.37 GWd/t burn-up with ORIGEN-S and ORIGEN-JÜL.

The larger ORIGEN-S decay heat can partly be explained by the VSOP flux profile for Channels 2 to 4 that were used for the ORIGEN-S calculation, thus not taking the effect of the different flux profiles in Channels 1 and 5 correctly into account. This was confirmed by a reduction in the decay heat difference for 20 days' decay. The decay heat difference decreased to 9%. The differences seen for some isotopes are probably also due to differences in the one-group cross-sections of the fission product, their activation and daughter isotopes.

TABLE 5. ORIGEN-JÜL and ORIGEN-S fuel sphere discharge activities (Bq) and decay heat (W).

Isotope	ORIGEN-JÜL	ORIGEN-S	%
Sr-90	8.04E+10	7.63E+10	-5
Zr-95	9.48E+11	1.01E+12	7
Ru-103	6.28E+11	8.16E+11	30
Ag-110m	1.41E+09	2.33E+09	65
I-131	1.77E+11	2.11E+11	19
I-132	2.02E+11	2.39E+11	18
Cs-134	1.20E+11	1.14E+11	-5
Cs-137	9.71E+10	1.00E+11	3
La-140	4.52E+11	5.32E+11	18
Pr-144	1.00E+12	1.06E+12	6
Total	3.08E+13	3.52E+13	14
Decay heat (w)	8.69	10.28	18

In this study no further attention has been given to evaluate the ORIGEN library update implementation differences between SCALE and VSOP, that is for the calculation of the spectral indices and the isotope cross section update of the ORIGEN library. Also, the basic ORIGEN cross-section and nuclear data (example fission yields) libraries must also be compared and updated to ensure a common basis. Until such detailed evaluation has been performed, an explanation of detailed isotopic activity differences between the ORIGEN-S and ORIGEN- JÜL results will be very difficult.

5. CONCLUSIONS

From the results it is clear that the space-time neutron flux history of the fuel spheres travelling through the PBMR reactor, and the spectrum weighted cross-sections at each position, are very important for the fuel sphere depletion calculations. It was shown, for example, that the Pu-239 mass in a fuel sphere at discharge burn-up could differ with a factor 1.7.

Depending on the design application, different fuel depletion analyses need to be performed to ensure conservative, but realistic, source terms. VSOP together with ORIGEN-JÜL meet many of these calculational requirements. With special VSOP fuel reload models, variations in the fuel irradiation histories can be simulated and transferred to ORIGEN-JÜL via a USER-library. In this way representative burn-up histories, with conservative spectrum and residence times, can be constructed for each application.

To fulfil all PBMR's requirements, additional editing features such as the calculation of the core average source terms (for all fuel spheres in the core) need to be added. Also, as part of the PBMR software validation process, the basic ORIGEN-JÜL cross-section data needs to be evaluated and updated if required. The developers of SCALE have recently issued SCALE-5 with a complete data update for the ORIGEN-S basic libraries. This will probably be used as the basis for the future ORIGEN-JÜL evaluations and comparisons.

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