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FUSION DIVISION
NEUTRONICS SECTION

Task T426 - Neutronics Experiments

EXPERIMENTAL VALIDATION OF SHUT DOWN DOSE RATES

1st Intermediate Report on Experiment Pre-analysis

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1. Background and Objectives

Neutronics experiments are important to validate neutronics analyses for ITER which are based on code calculations using more or less sophisticated models, and nuclear data with inherent uncertainties. Experimental validation is, in particular, required for responses which affect the feasibility and economics of machine components and systems, and such which are safety relevant. Within the latter category of problems, present dose rate calculations for complex geometries suffer from high uncertainties which are unacceptable for guaranteeing occupational safety during hands-on mainenance inside the cryostat, also in presence of diagnostics penetrations through the blanket. Therefore, a neutronics experiment shall be performed, using a 14 MeV neutron generator and a proper experimental set-up, in which a neutron spectrum is generated similar to that occurring in the ITER vacuum vessel. The mock-up will be irradiated for sufficiently long time to create a level of activation which can, after shut down, be followed by dosemeters for periods of time typical for the cooling time assumed to be required for allowing personal access. Proper and redundant measurement techniques shall be applied to obtain the data necessary for a precise analysis and calculation. C/E comparison will be provided using the available numerical tools, i.e. MCNP, FISPACT codes and FENDL-2 nuclear data libraries.

2. Task Description

The experimental assembly will consist of a block of stainless steel and water equivalent material with total thickness of 70 cm, and a lateral size of 1 m x 1 m (see Fig.1). The assembly will be exposed to neutrons generated by a 14 MeV neutron generator. A cavity will be realised within the shield assembly (about 12 cm in the beam direction x 15 cm x 12 cm) behind about 23-cm-thick shield. A void channel (28 mm inner diameter) will be included in front of the cavity to include the effect of streaming paths in the bulk shield. Measurements will be taken in the cavity, during the irradiation and after shut down, to obtain the local neutron flux, the decay γ-ray spectra and the dose rate after shutdown for a cooling time of interest for maintenance and repair purposes (i.e., about 2 weeks).
The experiment configuration is designed to provide validation of present dose rate calculations in a typical and complex shield geometry, compatibly with the intensity of the available neutron source and its capability to induce sufficiently high dose levels in the shield mock-up.

The following measurements will be carried out in the cavity:

- Neutron flux measurement during irradiation by activation foils and by scintillator detector.
- Dose rate measurements by two independent dosemeters.
- Decay gamma-ray spectra measurements by scintillator detector.
- Gamma-ray dose distribution by TLD.

Associated analyses for comparison will be performed using MCNP, FISPACT codes and FENDL-2 nuclear data libraries. C/E values will be provided.

The time plan is as follows:

Completion of experimental set up	June 2000
Completion of measurements	October 2000
Completion of evaluation of results	June 2001

3. Mock-up configuration

The experiment consists in the irradiation with 14-MeV neutrons of the same assembly used for the Bulk Shield Experiment (Task T.218) and for the Streaming Experiment (Task T.362), but modified in order to obtain sufficient doserate after irradiation: in particular, it is now provided with a channel with high aspect ratio (inner diameter a = 28 mm, length l = 22.47 mm, wall thickness 1 mm of stainless steel AISI316). At the end of the channel, a cavity is also realised : a parallelepipedal box with inner size 118 mm (h) x 148 mm (l) x 120 mm (in the beam direction). The thickness of the box walls is 1 mm (stainless steel AISI316). The cavity is symmetrically located with respect to the channel axis (see Fig.1).

The stainless steel layers have been re-arranged in the the new mock-up in order to have close the dose measurement position (i.e. the cavity), those layers which were exposed to low neutron flux in previous experiments, i.e. those with no residual activation. The layers positions, as well as their size, is given in Table I.

The configuration described above is considered in the present experiment pre-analysis. However, since some new parts of the mock-up are being constructed in this moment, the size of the cavity may be subject to sligth modifications and will be exactly specified when the mock-up will have been assemblied.

The chemical composition of materials employed in the mock-up is specified in the following:

<u>Stainless steel (SS316)</u> - It is a AISI 316 type steel; the density is 7.89 g/cm^3. The chemical composition, guaranteed by the furnisher and also confirmed by analysis made in ENEA laboratory, is : 68.1% Fe, 16.8% Cr, 10.7% Ni, 2.12% Mo, 1.14% Mn, 0.45% Si, 0.09% Cu, 0.14% Co, 0.16% Va, 0.04% C, <0.006% S, 0.022 P, 0.004% Sn, 0.001% Pb, 0.0035% B.

<u>Perspex (Polymethylmetacrilate)</u> - The composition is $(C_5O_2H_8)_n$ and the density is 1.18 g/cm^3. A sample of this material has been tested in ENEA laboratories and turned out to be of a high degree of purity (Si, S $\approx$ 4 appm; Ca $\approx$ 2.6 ppm; Mg, Ti, Fe, K, Na, Al < 1 appm).

Table 1 - Dimensions of mock-up layers (layer numbers refer to those shown in Fig.1)

Material /layer n.	**Thickness (mm)**	**Lateral size**
Stainless steel layer 11	43.5	100 x 100 cm^2
12	46.5	”
6	46.5	”
2	46.0	”
15-new	73.0 (nominal)	”
3	49.0	”
1	41.7	”
14	49.5	”
13	48.0	”
10	48.0	”
8	49.0	”
7	50.0	”
Perspex layers	20.6	”

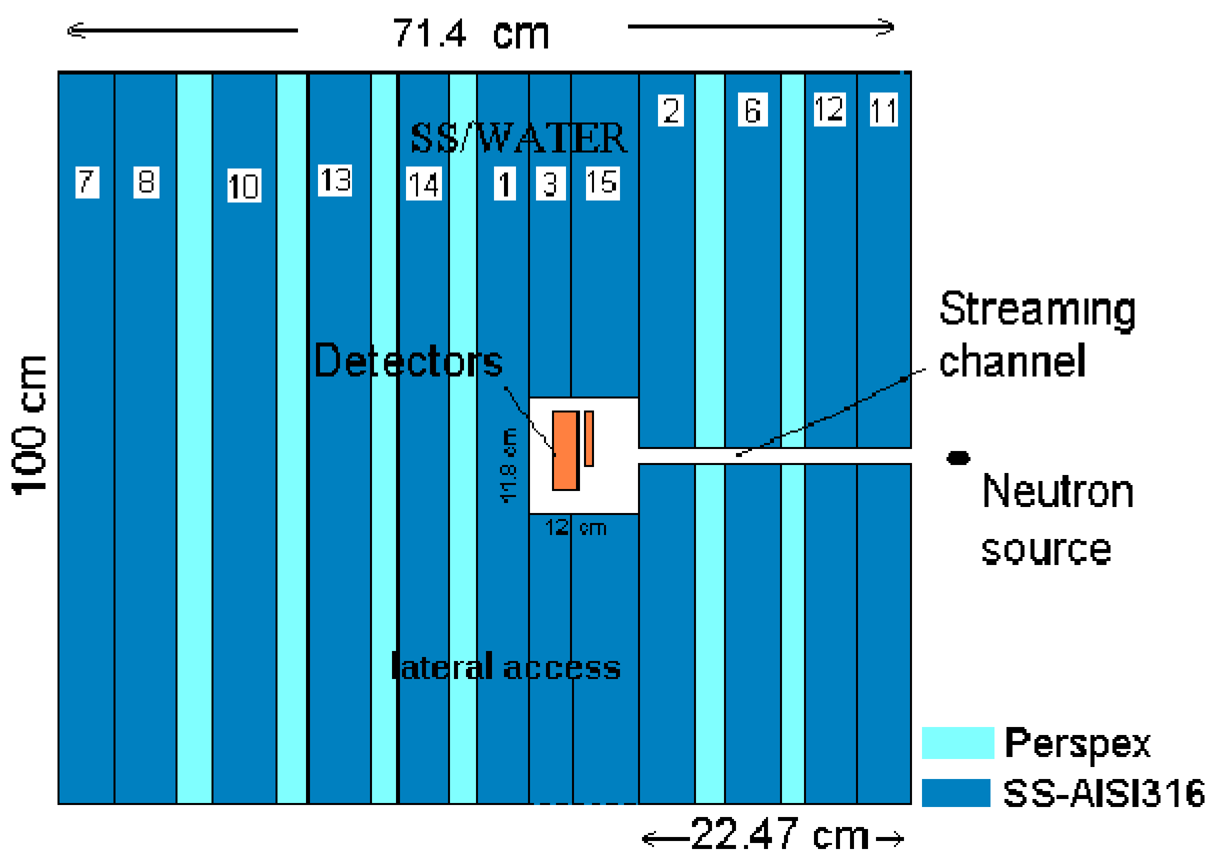


Fig 1 – Layout of experimental set-up

4. Experiment Pre-analysis

4.1 MCNP/FISPACT calculation

The neutron fluxes during irradiation inside the assembly have been calculated using MCNP-4B code and FENDL-2 nuclear data library. The assembly has been described in the MCNP geometrical model devided in small cells as shown in Fig.2. The neutron flux calculated by MCNP in each cell has then been input in FISPACT code in order to calculate the resulting activity, dose rate and decay gamma spectrum at different cooling times, using the EAF-99 activation file. A 175-group cross section library (VITAMIN-J structure) produced from EAF-99 was used as data base, collapsed with the neutron flux calculated by MCNP. Finally, the decay gamma spectra calculated by FISPACT at given cooling times have been input as gamma sources in the corresponding cells in MCNP, to calculate the dose rate vs. cooling time inside the cavity (spherical air cell in Fig.2), using tally f6 for photons.

The procedure has been applied for cooling times 1 min, 30 min, 1 hours, 12 hours, 1 day, 7 days, 14 days, 29 days, 59 days, 1 year. **An irradiation at FNG at 1 x 10^{11} n/s level for 10^4 s has been considered as a reference irradiation history**.

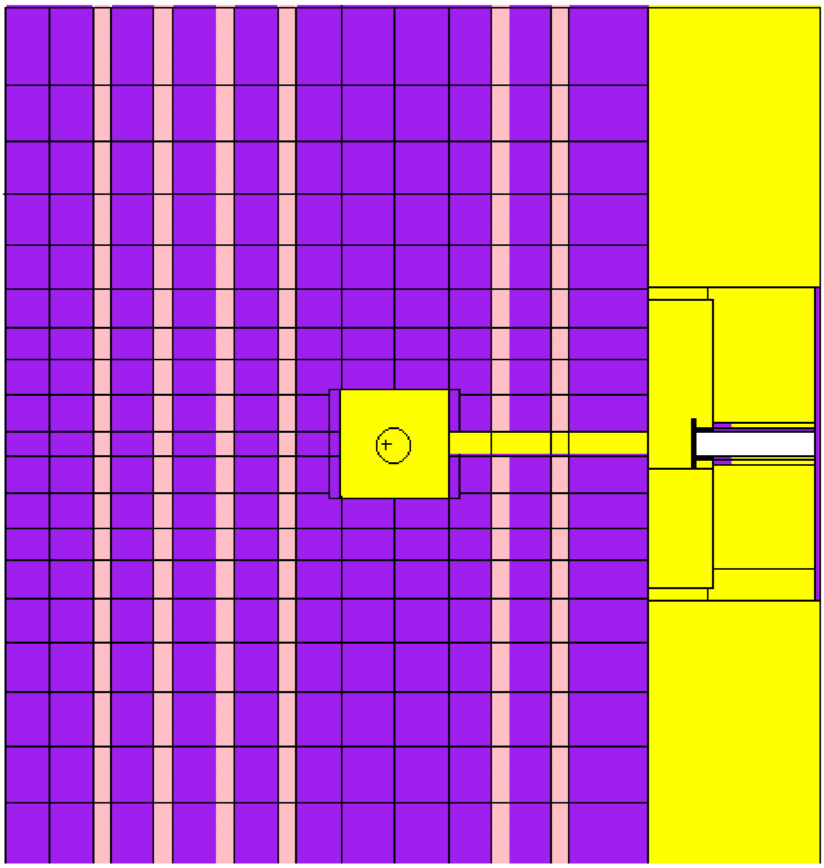


Fig. 2 - MCNP model of the Doserate Experiment assembly

Three groups of cells have been considered separately, in order to study the contributions of different parts of the assembly to the total doserate in the cavity. The three groups (shown in Fig.3, see also Fig. 12) are as follows:

1. Inner shell cells (137,160,161,226,228,230,231,242,243,251,252,253,601)
2. Next shell cells (114,115,138,139,140,162,163,232,244,254,255,274,275,276,277,278)
3. Front cells (102,103,104,105,106,107,116,117,118,119,141,142,164,165,233)
4. Channel (112,170,171,172)

The adopted cell modelling is such that the neutron fluxes and the resulting gamma activities in two contiguous cells never exceeds a factor of two difference.

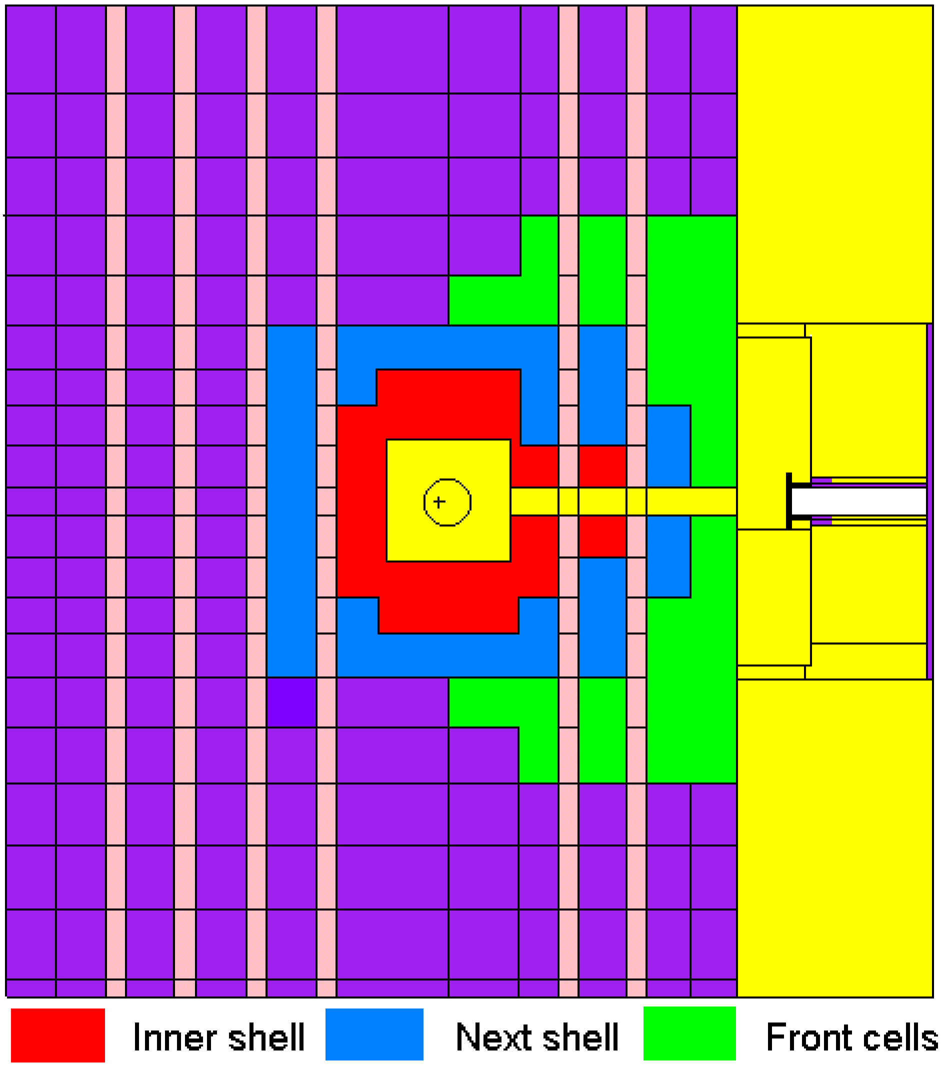


Fig.3 – MCNP model showing the cells considered in the MCNP/FISPACT calculation

These groups include only stainless steel cells, as Perspex does not present any significant activity soon after a few minutes of cooling time.
The resulting total doserate inside the cavity (full MCNP/FISPACT calculation) is shown in Fig.4 vs. cooling time: it can be seen that the time behaviuor follows very closely that of contact doserates (calculated by FISPACT) for Inner Shell cells, also presented in the figure.

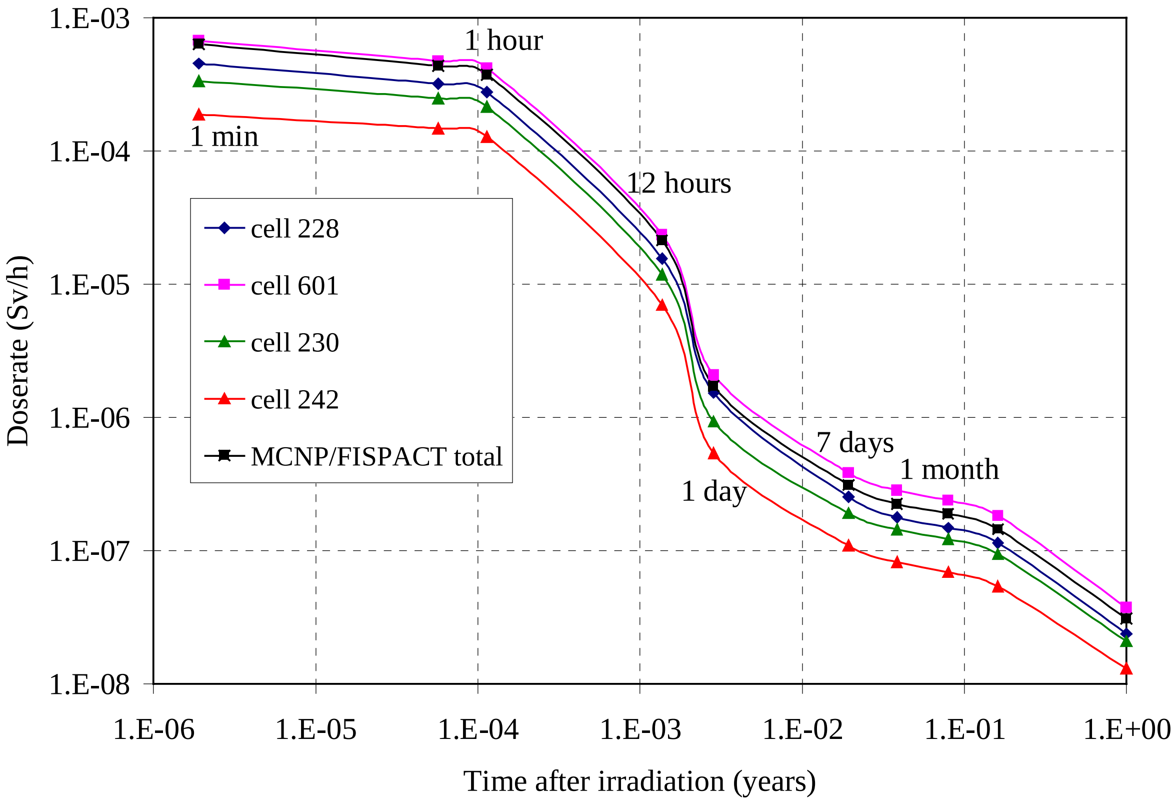


Fig.4 - Total shutdown doserate in the centre of the cavity, resulting from full MCNP/FISPACT calculation. The contact doserates for Inner Shell cells, calculated by FISPACT are also shown for time behaviour comparison. (Reference irradiation for 10^4s at 10^{11} n/s)

Fig. 5 shows the partial contributions of different cell groups to the total doserate, while the corresponding fractions for each group are given in Fig. 6. It can
be seen that Inner Shell contributes to more than 70% to the total doserate, the Next Shell contributes between 10 and 20%. Finally, both the Front and the Channel Cells contribute always to about 5%, although the mass involved is very diffferent.
Moreover, the importance of closer cells increases with time, as a consequence of the modification of the decay gamma spectra as the short term activity is dominated by high-energy gamma emitters (Fig. 7), while the medium term activity is due to gamma emitters at lower energies (Fig.8): after 30 minutes of cooling time, the activity, f.i. for cell 226 (Inner Shell, rear wall, Fig.7b) is mainly due to Mn-56 (98%, $<E_\gamma>$=1.7 MeV, where $< E_\gamma>$ is the gamma energy averaged over all lines per disintagration); after 7 days of cooling time the activity, again in cell 226 (Fig 8b), is

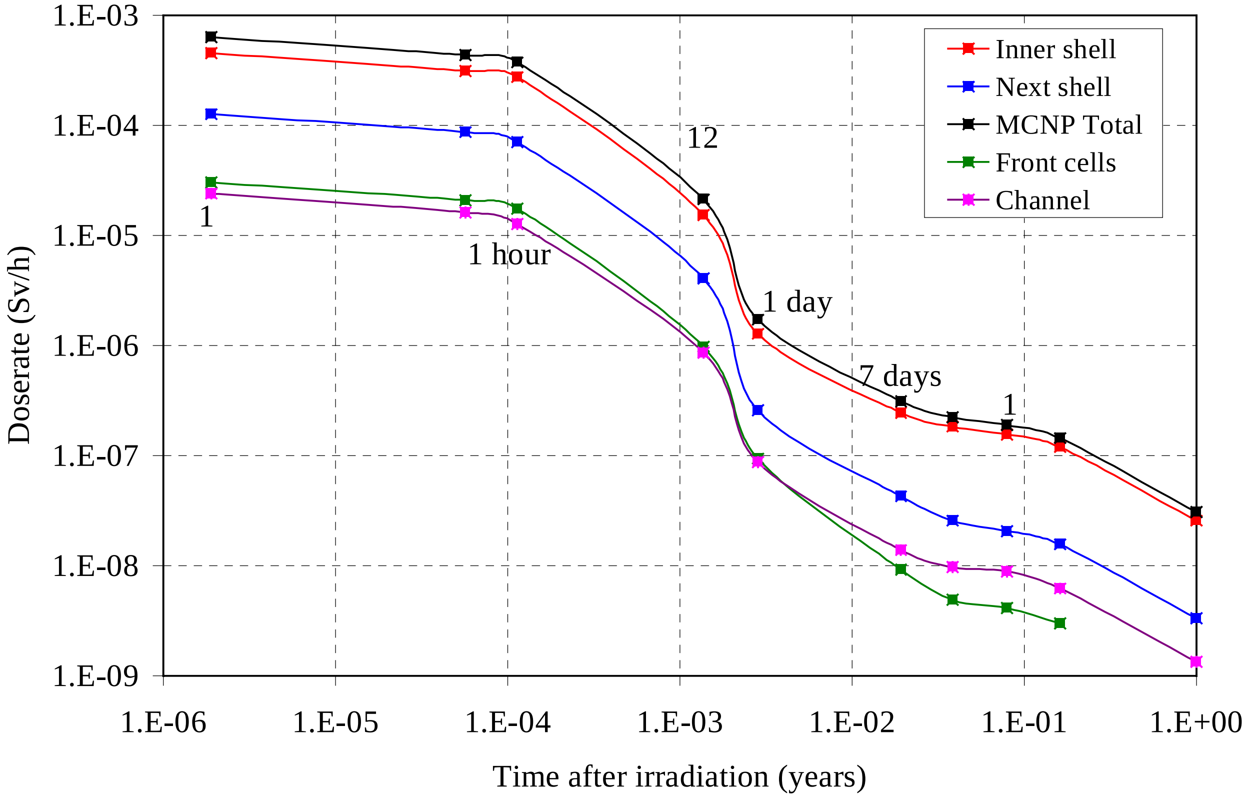


Fig.5 - Total shutdown doserate in the centre of the cavity from full MCNP/FISPACT calculation and contributions of the cells groups. (Reference irrad. for 10^4s at 10^{11} n/s)

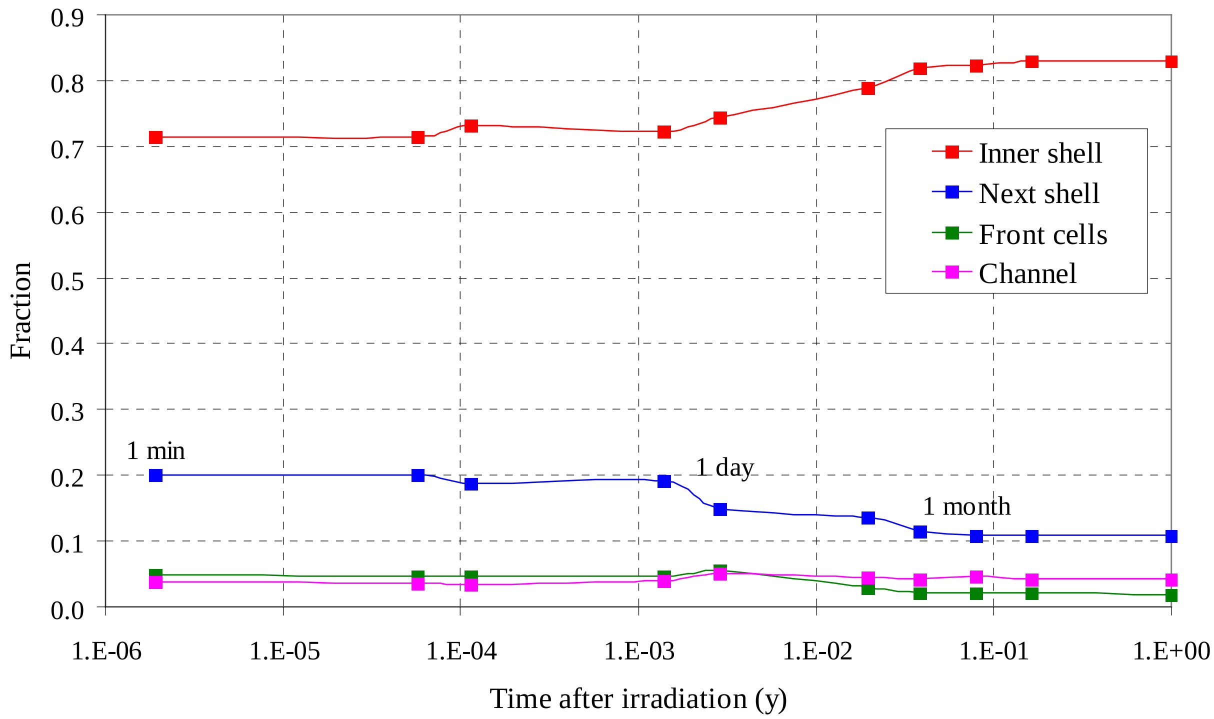
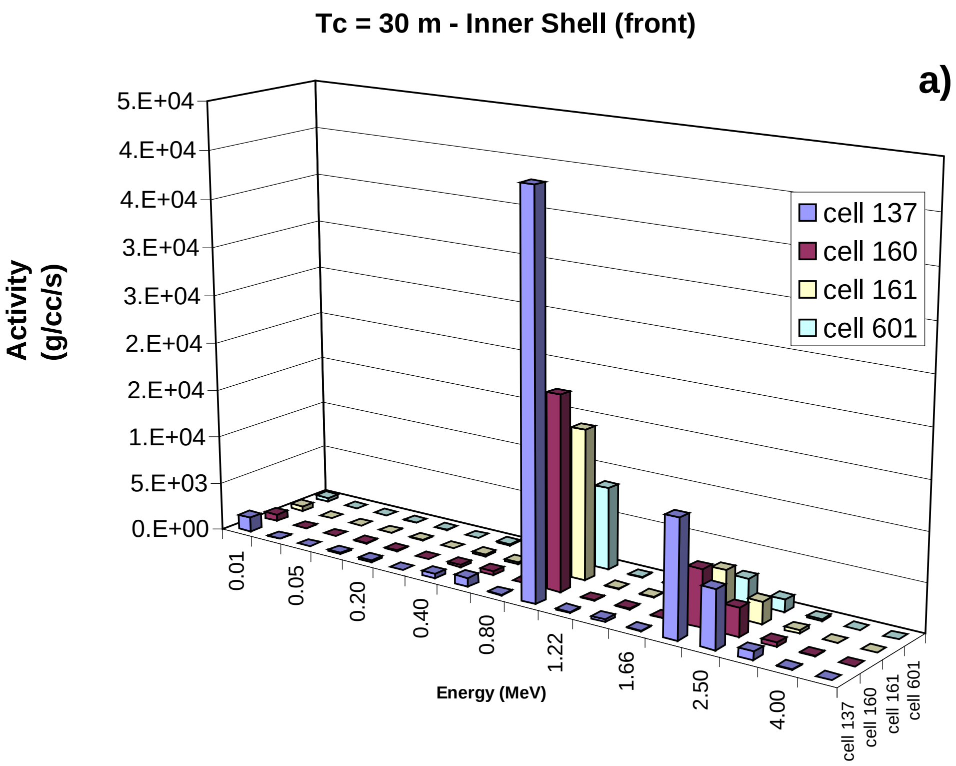


Fig. 6- Fractions of partial contributions of cells groups to the total doserate in the centre of the cavity. (Reference irradiation for 10^4s at 10^{11} n/s at FNG)



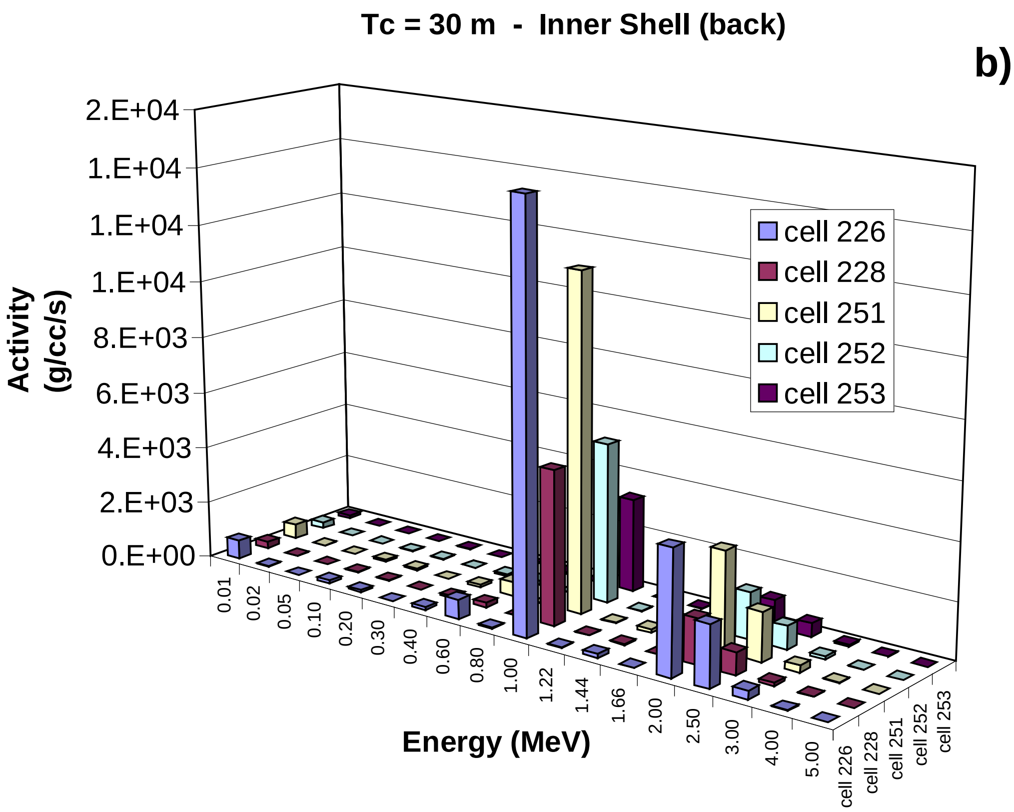
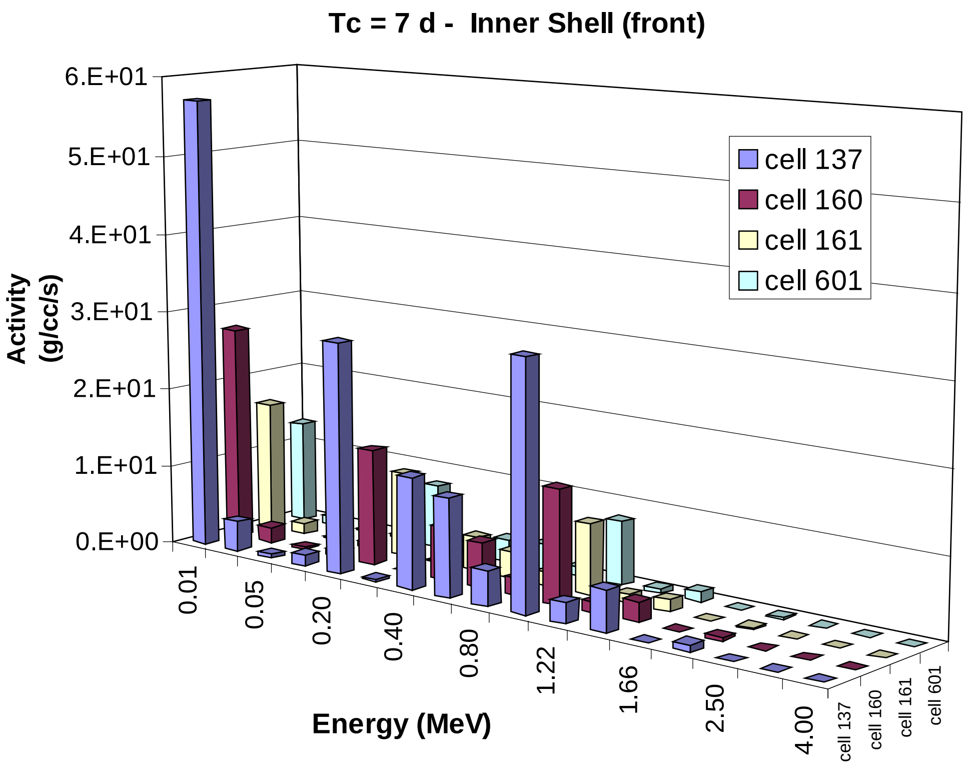


Fig.7 Decay gamma spectra after 30 min of cooling time from cells in the Inner Shell: a) front wall and b) rear wall.



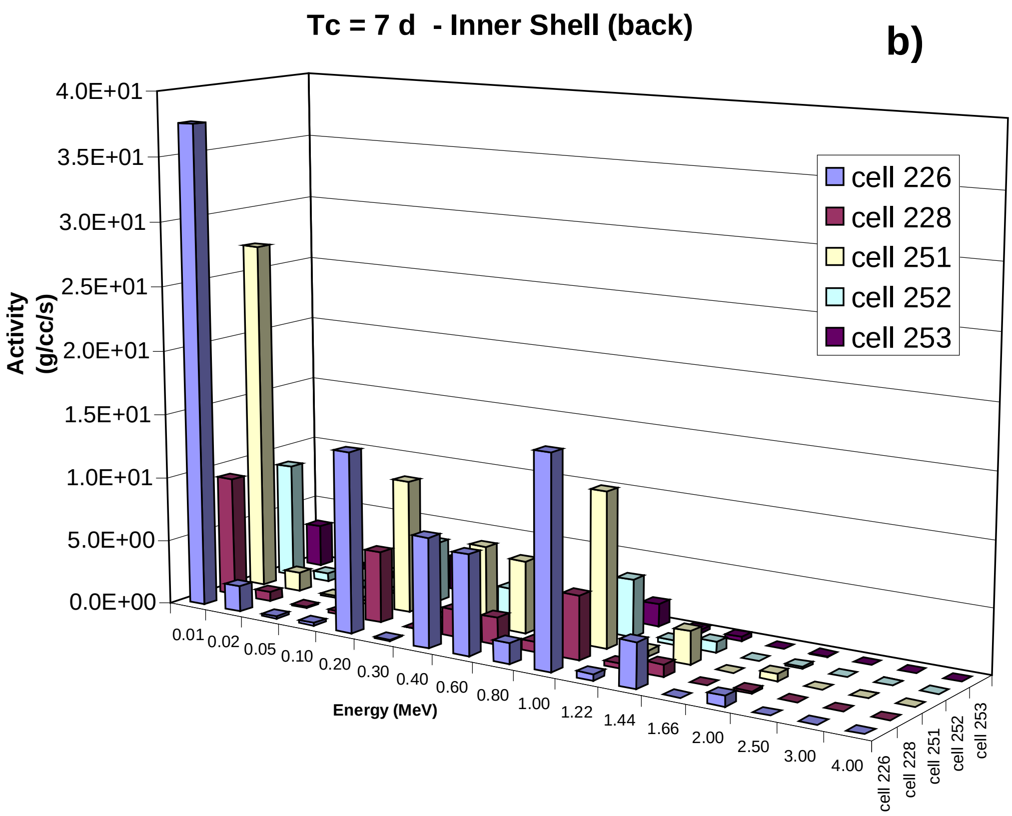


Fig.8 Decay gamma spectra after 7 days of cooling time from cells in the Inner Shell: a) front wall and b) rear wall.

mainly due to Cr-51 (61.45%, <E_γ>=0.033 MeV), Fe-55 (9.28%, <E_γ>=0.0017 MeV), Co-58 (8.85%, <E_γ>=0.976 MeV), Mo-99 (5.04%, <E_γ>=0.147 MeV), Tc-99m (4.89%, <E_γ>=0.126 MeV), Co-57 (4.86%, <E_γ>=0.124 MeV), Ni-57 (2.19%, <E_γ>=1.96 MeV), Mn-54 (1.94%, <E_γ>=0.836 MeV). Major contributions to total contact dose rates (calculated by FISPACT) for Inner Shell cells are shown in Fig.9.

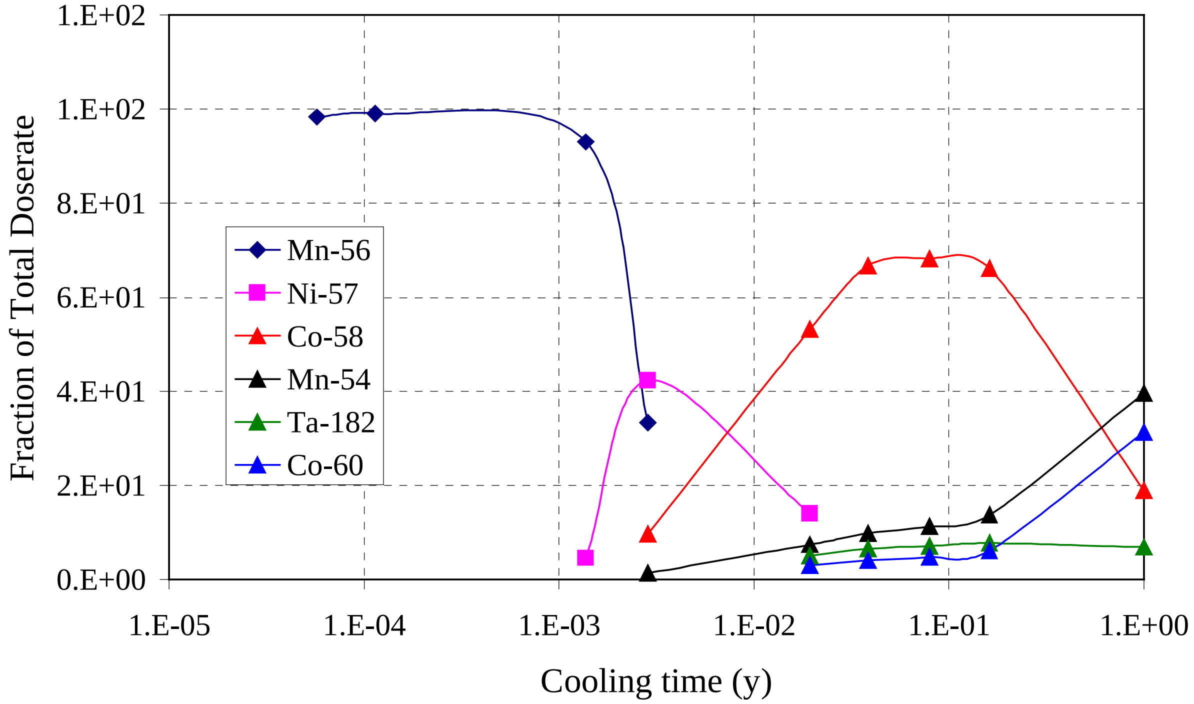


Fig. 9 - Major contributions to total contact doserate in Inner Shell cells, as calculated by FISPACT

4.2 Modified MCNP calculation

A new implemented MCNP version has also been used to calculate the dose rate after shutdown using the same geometry model shown in Fig.2. The modification implemented enables MCNP to couple the event of creating a radiactive isotope, through a neutron reaction, to the subsequent gamma decay. The emitted delayed gamma is then followed and, at the occurrence, scored. This MCNP version uses *ad hoc* libraries in which the cross sections are taken from FENDL.2, except for V-51 and for Ta-181, for which they are taken from EFF-2. Decay gamma spectra and yields are taken from EAF-99.

The dose values so obtained by tally f6 of MCNP need to be multiplied by adjusting factors, for desired cooling times, that take into account the time dependence, essentially the decay rate of each particular nuclide, according to its decay constant. The activity A of a given nuclide at cooling time t_c is given by

$$A(t_c) = <\sigma\phi> \; N_t \; \beta \, (1-e^{-\lambda T}) \, e^{-\lambda t_c}$$

$$N\,(0) = <\sigma\phi> N_t \, \lambda^{-1}(1-e^{-\lambda T})$$

where $<\sigma\phi>$ is the reaction rate (product of neutron flux times the cross section of reaction producing the nuclide), N_t is the concentration of target nuclei, β is the branching ratio of gamma channel, T is the irradiation duration, λ is the nuclide decay constant ($\lambda = \ln 2/\tau_{1/2}$, where $\tau_{1/2}$ is the half life) and $N(0)$ is the nuclide concentration at the end of the irradiation.
In practice, the modified-MCNP output is multiplied by the total number of neutrons produced during the irradiation, by $(1-e^{-\lambda T})$ to take into account saturation during irradiation, and by the adjusting factor f_{adj}

$$f_{adj} = A(t_c) \,/\, N\,(0) \; = \beta \, \lambda \, e^{-\lambda t_c}$$

This adjusting factor, i. e. the ratio beteewn the decay rate at a given cooling time with respect to the isotope concentration at shutdown, can be calculated analytically, or can be extracted from the output of the FISPACT calculation.

The shutdown dorerates calculated by modified MCNP in the spherical volume inside the cavity are shown in Fig.10, where also the MCNP/FISPACT results are shown for comparison.

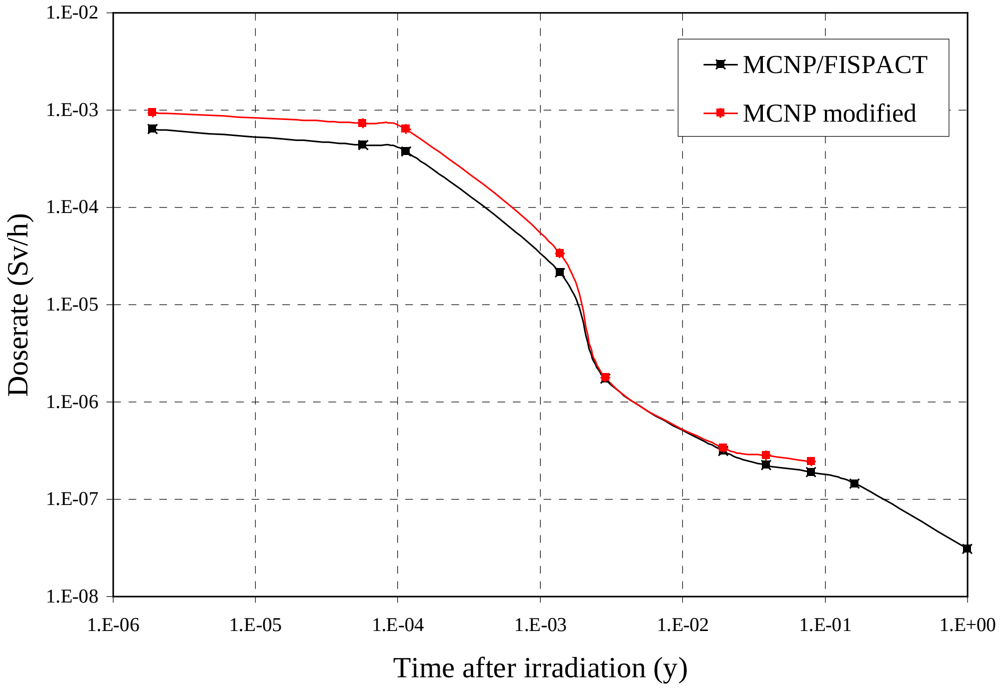


Fig.10 - Shutdown doserate in the centre of the cavity: comparison between full MCNP/FISPACT calculation and modified MCNP calculation

5. Discussion

As shown in Fig. 10, the MCNP/FISPACT calculation and the modified MCNP calculation give similar values within a factor 1.7 at short times, when Mn-56 dominates, and <1.3 at longer times, when Co-58 dominates. The total statistical errors are $\pm$6% in both cases. The transport cross sections are taken from FENDL-2 in both cases. As far as the activation cross sections for most contributing nuclides are concerned, they are

Reaction	Cross section used in FISPACT	Cross section used in MCNP modified
Fe-56 (n,p) Mn-56	EAF-99 from IRK	FENDL-2 from EFF-3
Ni-58 (n,2n) Ni-57	EAF-99 from IRDF-90.2 rev	FENDL-2 from ENDF/B-VI.1
Ni-58 (n,p) Co-58 Ni-58 (n,p) Co-58m	EAF-99 from ADL-3	FENDL-2 from ENDF/B-VI.1

The decay data are taken from EAF-99 in both cases. Comparison of cross section data used in the two approaches, indicates that only a minor part of the observed discrepancy could be due to different cross sections.
On the other hand, the geometrical modeling of cells close to the cavity, whose tipical size is <5 cm in the present pre-analysis, is probabily responsible of part of the observed discrepancy since in the MCNP/FISPACT method it introduces values of the decay gamma sources averaged over the cell volume. Based on the present analysis, the cell description in the Inner Shell will be improved.

6. Conclusions of pre-analysis

The following conclusions may be drawn from the present pre-analysis:

- The Inner Shell cells dominate the dose rate inside the cavity to more than 70%.

- There is presently a discrepancy by factor of 1.3 - 1.7 between the two calculation methods, MCNP/FISPACT and modified MCNP, that must be understood. In particular, comparison of activation cross sections must be performed and the geometrical description of cells close to the channel and to the cavity, that mostly contribute to the doserate in the measuring position, must be further refined.

- Finally, an irradiation at FNG at 1 x 10^{11} n/s source intensity level for 10^4 s, studied here as a reference irradiation history, produces a doserate level reaching the natural background level (0.25 µSv/h) in about 15 days. It will be therefore necessary to irradiate the mock-up for a longer time. For instance, Fig.11 shows, for an Inner Shell cell (cell 228), that a two-interval irradiation, 2x10^4 s each interval in two subsequent days, at 7 x 10^{10} n/s source intensity average level (to take into account tritium consumption in the target), produces a dorerate level higher by a factor 2-3 with respect to that produced in the reference irradiation, and sufficiently higher than the background limit up to more than one month of cooling time.

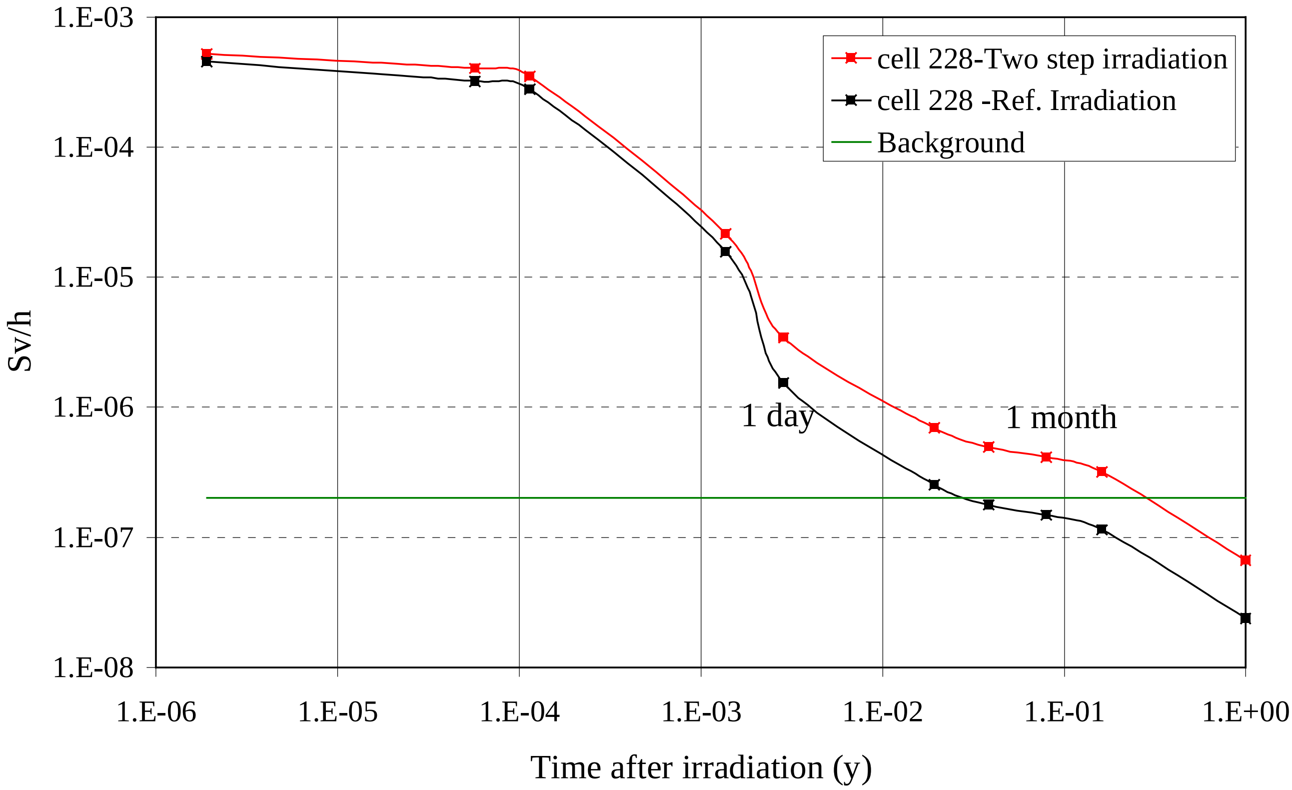
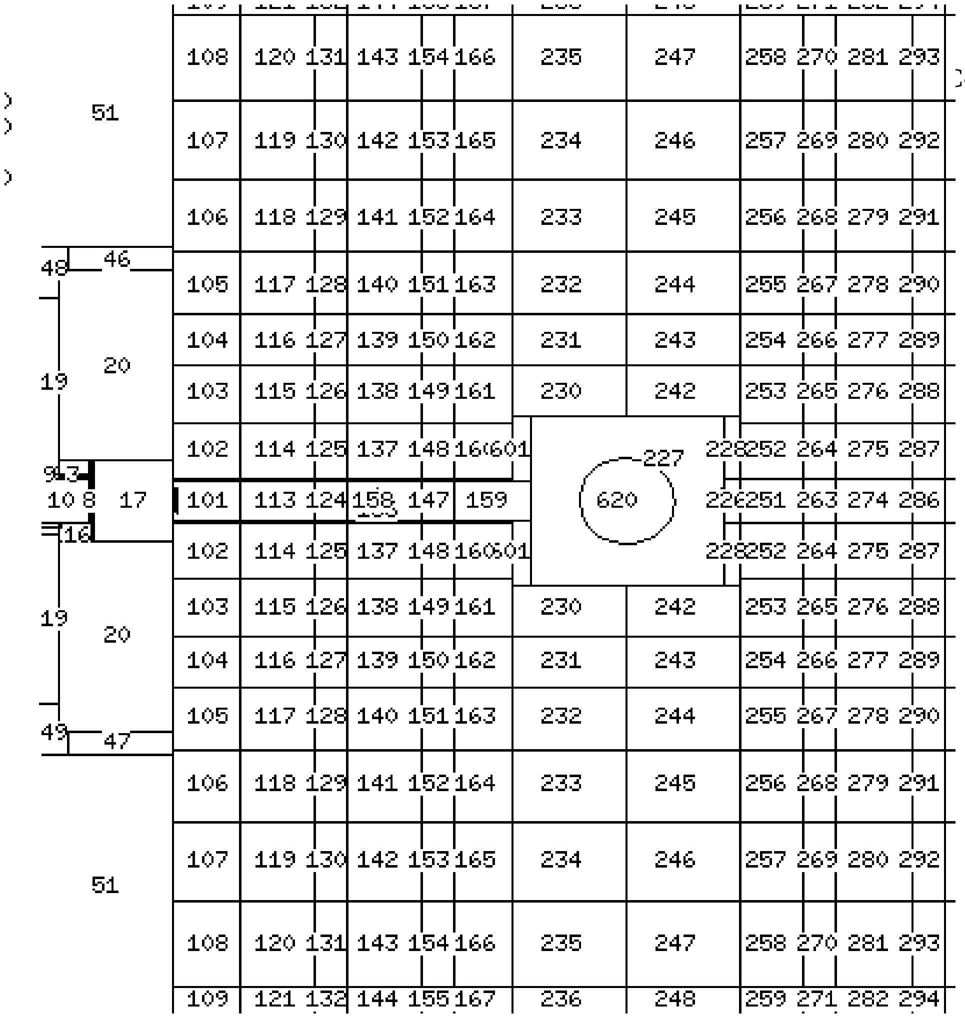


Fig. 11 – Contact dose rate, calculated by FISPACT for Inner Shell cell 228, for reference irradiation (one irradiation of 10^4 s at 10^{11} n/s) and for a two-interval irradiation, 2x10^4 s each interval in two subsequent days, at 7 x 10^{10} n/s average source intensity level.



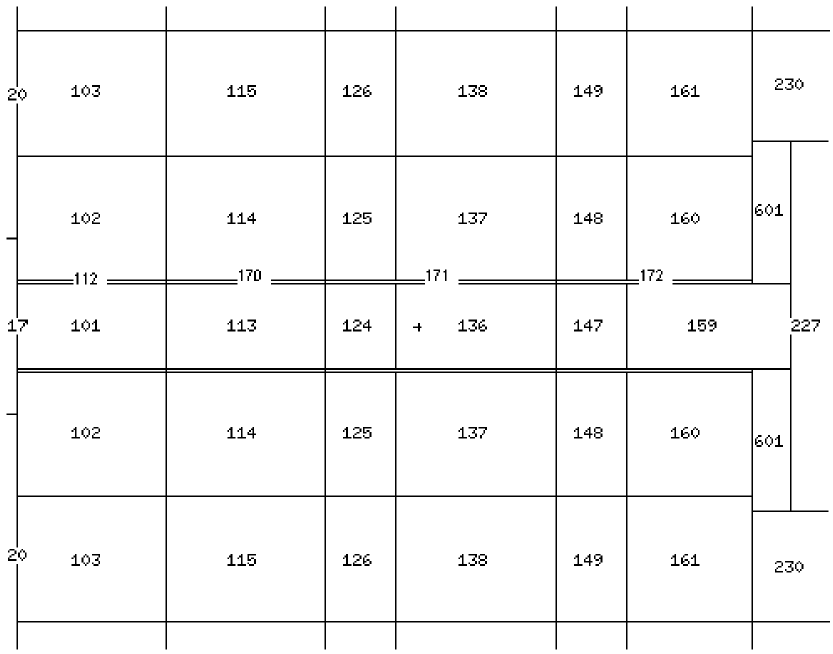


Fig.12 – Details of the MCNP model for the assembly and for the channel, showing the cell numbers