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Corrections to ACER Module (NJOY-91.38)

by

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Processing EFF-2.3with ACER of NJOY to produce an MCNP library Status report

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At the last meeting in Paris (Dec. 1992), we reported about the status of processing EFF-2 (version 2) with ACER for an MCNP library (see EFF-Doc-198). That work was performed mainly with the last , at the time, released version of NJOY (91.13). Now the new released version NJOY 91.38 is available through the NEA DATA Bank, with many interesting modifications just for the file 6 handling and double differential cross section treatment. Moreover version 3 of EFF-2 has been distributed in late Feb. 1993 (see EFF-Doc-203). This document is an update as partial completion of the work performed till now in producing an MCNP library from EFF-2.

Resuming, ACER is the module specifically written to prepare data library for the Los Alamos continous energy Monte Carlo code MCNP. To process the new release of EFF-2 (version 3) the NJOY 91.38 version was used. In ACER the integral thinning option was asked, for a maximum 5000 number of energy points, keeping the total and capture integral cross sections at 0.5% of their original PENDF value.

In the 91.38 version most of the updates relate to the file 6 processing. In the previous version file 6 data could be treated by ACER, only if they were given in the Kalbach Mann formalism . This because MCNP (version 4) was expected to have a sampling algorithm compatible with such a formalism. Many evaluations in ENDF-6 have, however, file 6 in Legendre polinomial expansion or as tabulated data; so a conversion to an equivalent Kalbach formalism was inserted in 91.38 version of ACER. This proposed approach for the authors has the advantage that no additional changes have to be made to MCNP. (for more details on this subject see R.E. MacFarlane "How to NJOY ENDF-6" presented at the Int. Workshop on NJOY, Saclay 6-10 April 1992).

Improvements to this approach have been studied inside the EFF project by A. Hogenbirk et al. (see EFF-Doc-180 "Improvements in the processing of MCNP-libraries using NJOY 91.38"). Their modifications, however were not released outside their laboratory, so they could not be used.

Our experience anyway with file 6 processing is that most of the EFF-2 evaluations (with file 6) can be in some way handled by NJOY 91.38 which is a step up compared to 91.13 version, but many problems are still to be solved.

1) The sampling according to the Kalbach systematics is not available in the MCNP 4 package distributed through the official centers (RSIC or NEA DATA Bank). A subroutine able to do that has been given to us outside the official channels, by Jacek Arkuszewski of PSI (subroutine ACECS2, with a special "MCNP patch") who for a period worked at Los Alamos very close to the MCNP authors. Neverthless in our experience, all revisions and updates should undergo a quality assurance procedure before.

2) All the processed evaluations, with file 6, which had apparently a normal end (no warning messages or error detection) can be used to run neutron transport problems only. If a coupled neutron and photon MCNP run is asked, the code goes always in an endless loop consuming all the time available.

Nevertheless, just for a simple try, we tested B^{11} (MAT 528 of EFF-2 with file 6) with other previous evaluations available in ACE format in a very simple MCNP problem. A series of 4 concentric spherical surfaces was considered with radius 360, 390, 420, 450 cm respectively. The inner sphere is void. The outer 3 spherical shells have density 1 g/cm^3. A point isotropic 14 Mev neutron source is placed in the center of the spheres. Every neutron reching the outer surface is lost. The other B^{11} cross sections available (with the usual file 4 and 5) are from EFF-1 (processed by ENEA in 1989, see EFF-Doc-168), from ENDL85 and ENDFB-3 (released with the MCNP package). The results are summarized in table 1. The quantities recorded and compared, are the total neutron flux on each spherical surface (F2) in n/cm^{2}s per neutron source, the total neutron flux avaraged on a spherical shell (F4) and the neutron heating in MeV/g s per neutron source. Results have a statistical error at maximum of 3% (runs of 10^5 neutrons on IBM 3090).

table 1

B^{11} MCNP data comparison

	surface or cell 1	surface or cell 2	surface or cell 3	surface or cell 4	
F2	$1.59\ 10^{-5}$	$1.26\ 10^{-5}$	$6.30\ 10^{-6}$	$4.23\ 10^{-7}$	EFF-2
	$1.67\ 10^{-5}$	$1.30\ 10^{-5}$	$6.60\ 10^{-6}$	$4.30\ 10^{-7}$	EFF-1
	$1.62\ 10^{-5}$	$1.23\ 10^{-5}$	$6.48 10^{-6}$	$4.36\ 10^{-7}$	ENDL 85
	$1.60\ 10^{-5}$	$1.18\ 10^{-5}$	$6.34 10^{-6}$	$4.36\ 10^{-7}$	ENDF B4
F4	$1.71\ 10^{-5}$	$1.46\ 10^{-5}$	$9.34\ 10^{-6}$	$3.33\ 10^{-6}$	EFF-2
	$1.81\ 10^{-5}$	$1.52\ 10^{-5}$	$9.63\ 10^{-6}$	$3.40\ 10^{-6}$	EFF-1
	$1.73\ 10^{-5}$	$1.45 10^{-5}$	$9.36\ 10^{-6}$	$3.31\ 10^{-6}$	ENDL 85
	$1.72 10^{-5}$	$1.42 10^{-5}$	$9.14\ 10^{-6}$	$3.29\ 10^{-6}$	ENDF B4
F6		$7.69\ 10^{-8}$	$2.21\ 10^{-8}$	$4.92\ 10^{-9}$	EFF-2
		$1.86\ 10^{-7}$	$5.29\ 10^{-8}$	$1.35\ 10^{-8}$	EFF-1
		$9.09\ 10^{-8}$	$2.78\ 10^{-8}$	$7.24\ 10^{-9}$	ENDL 85
		$1.79\ 10^{-7}$	$5.66 10^{-8}$	$1.51\ 10^{-8}$	ENDF B4

Major differences concern heating values.

A similar excercise was done with Pb (taking a 1. g/cm^3 material density), comparing the EFF-2 data with the previous available EFF-1 (table 2).

table 2

Pb^{nat} MCNP data comparison

	surface or cell 1	surface or cell 2	surface or cell 3	surface or cell 4	
F2	$2.49\ 10^{-6}$	$2.18\ 10^{-6}$	$1.62\ 10^{-6}$	$8.68\ 10^{-7}$	EFF-2
	$2.43\ 10^{-6}$	$2.15\ 10^{-6}$	$1.59\ 10^{-6}$	$8.60\ 10^{-7}$	EFF-1
F4	$3.49\ 10^{-6}$	$2.37\ 10^{-6}$	$1.91\ 10^{-6}$	$1.27\ 10^{-6}$	EFF-2
	$3.45\ 10^{-6}$	$2.33\ 10^{-6}$	$1.88\ 10^{-6}$	$1.25\ 10^{-6}$	EFF-1
F6		$2.37\ 10^{-8}$	$1.67\ 10^{-8}$	$1.15\ 10^{-8}$	EFF-2
		$5.91\ 10^{-10}$	$5.48\ 10^{-10}$	$3.99\ 10^{-10}$	EFF-1

3) For other elements the conversion procedure is not effective (Li^7, Be^9, Al^{27} ,Si^{28}). The Kalbach parameters derived (***a*** and ***r***) are often not reasonable, and, running MCNP, severe problems are met in subroutine ACECS2 with outgoing neutron cosine >> 1 (!?!?).Other sampling laws in MCNP are maybe necessary.

In case of D , file 6 MT 16 (n,2n), the energy available in the center of mass of the bodies involved (LAW 6)is given, but it is not allowed by ACER; in this case also a special MCNP treatment should maybe given. These troubles (except Al27 and Si28) are not new to the authors of ACER and they were already mentioned at the NJOY meeting in 1992. At the time, supposed solutions involve additional work on ACER and on MCNP.

4) It is not clear how to handle **MT 10** in file 6. In theory it is a brilliant way to describe all reactions emitting neutrons, all together. The point is that, unfortunally, ACER does not allow a neutron yield dependent on the energy except for the fission reactions.

5) For some materials for which resonance parameters are given in the Unresolved Range (URR) some questions arose if MCNP was able to handle self-shielded data. At the moment, the unresolved range is still a problem in MCNP, the code has no probability table capability (version 4.2) (probability and energy ladders given by the PURR module of NJOY). At the moment the URR data which MCNP is using , treated by UNRESR, are unshielded. An improvement has been proposed by A. Hogenbirk at the previous EFF Meeting in December (see EFF-Doc-180), but his modifications to NJOY for the proposed solution were not distributed.

A resume of the status of the processing of EFF-2.3 is given in tables 3 through 6. In table 7 the dictionary of the elements from the whole EFF-2.3, with their identifier, for MCNP usage, is shown. In the list, the isotopes of W (182, 183, 184, 186) taken from JEF-2.3 were added. Mo isotopes evaluations from ENEA Bologna are not yet available, so still Mo^{nat} is in the list.

Over the **82** which EFF-2.3 consists of (see "Status of European Fusion File" EFF-Doc-203), **74** evaluations are really available taking away Ba^{132} and replacing the 8 Mo isotopes with Mo^{nat}. Of these 74: **51** are successfully processed and fully available, uccessfully processed; **9** are available with ddx data in K.M. formalism, but only for neutron alone transport problem and not submitted to a quality assurance procedure of the sampling routine; **4** meet severe problems inside MCNP.

Besides the EFF-2 derived MCNP **transport** library, to satisfy the needs of MCNP calculations for the Benchmark experiment with FNG (task NDB 2.1, see also the document of P.Battistoni at this meeting) a consistent MCNP **dosimetry** library was derived (for few evaluations) from EFF-2, when available or from JEF-2, in place.

At the scope ACER module was used with a different user input selection run option. Many original modifications were inserted trying to reduce the number of output data which often are redundant. As distributed, ACER, in dosimetry run option, takes all the MTs from the PENDF file (no thinning); so for each MT you have a couple of set of data (energies and the relative cross section values). With this procedure, dosimetry files can be very big, but ,with our modifications, the user can select by input the desired MTs. In table 8 is the content of the dosimetry file derived and in table 9 is the relative dictionary.

table 3

content of sublibrary 200 of EFF-2.3, element identifier and processing comments
in *italics* are the new entries in respect to the latest report (Dec. 92)

elem.	MAT	ZAID	origin	comments
Ti^{nat}	2200	22000.92c	JENDL-3	O.K.; neg. neutr. kerma .2-11. MeV
V^{nat}	2300	23000.92c	ENDF/B-6	O.K.
Mn^{55}	*2525*	*25055.93c*	*ENDF/B-6*	***MF6*** *,not able for n+γ transport*
Co^{59}	2725	27059.92c	ENDF/B-6	O.K.
Cu^{63}	*2925*	*29063.93c*	*ENDF/B-6*	***MF6*** *,not able for n+γ transport*
Cu^{65}	*2931*	*29065.93c*	*ENDF/B-6*	***MF6*** *,not able for n+γ transport*
Zr^{nat}	*4000*	*40000.93c*	*JENDL-3*	*O.K.*
Nb^{93}	4125	41093.92c	ENDF/B-6	O.K.,neg. kerma 2.8-12.6 MeV
Mo^{nat}	4200	42000.92c	ENDF/B-6	O.K.,neg. kerma .01-20. MeV; URP
Mo isotopes		*not available*		
In^{113}	*4925*	*49113.93c*	*JENDL-3*	*no γ*
In^{115}	*4931*	*49115.93c*	*JENDL-3*	*no γ*
Ba^{130}	*5625*	*56130.93c*	*JENDL-3*	*no γ; URP*
Ba^{134}	5637	56134.92c	JEF-2	O.K.; no γ
Ba^{135}	5640	56135.92c	JEF-2	O.K.; no γ
Ba^{136}	5643	56136.92c	JEF-2	O.K.; no γ
Ba^{137}	5646	56137.92c	JEF-2	O.K.; no γ
Ba^{138}	5649	56138.92c	ENDF/B-6	O.K.;neg. neutr. kerma 2.25-9.5 MeV
Ta^{181}	7328	73181.92c	ENDF/B-6	O.K.;neg. neutr. kerma .035-20. MeV; URP

table 4

content of sublibrary 300 of EFF-2.3, element identifier and processing comments
in *italics* are the new entries in respect to the latest report (Dec. 92)

elem.	MAT	ZAID	origin	comments
C^{nat}	600	6000.92c	ENDF/B-6	O.K.
N^{14}	725	7014.92c	ENDF/B-6	O.K.
N^{15}	728	7015.92c	ENDF/B-6	O.K.
O^{16}	825	8016.92c	ENDF/B-6	O.K.
Mg^{nat}	1200	12000.92c	JENDL-3	O.K.; neg. neutr. kerma 1.37-1.57 MeV
P^{31}	1525	15031.92c	ENDF/B-6	O.K.
S^{nat}	1600	16000.92c	ENDF/B-6	O.K.
Ca^{nat}	2000	20000.92c	JENDL-3	O.K.; neg. neutr. kerma .308-1.186 MeV
Sn^{112}	*5025*	*50112.93c*	*JENDL-3*	*O.K., no γ; URP*
Sn^{114}	*5031*	*50114.93c*	*JENDL-3*	*O.K.no γ; URP*
Sn^{115}	*5034*	*50115.93c*	*JENDL-3*	*O.K.,no γ; URP*
Sn^{116}	*5037*	*50116.93c*	*JENDL-3*	*O.K.,no γ; URP*
Sn^{117}	*5040*	*50117.93c*	*JENDL-3*	*O.K.,no γ; URP*
Sn^{118}	*5043*	*50118.93c*	*JENDL-3*	*O.K.,no γ; URP*
Sn^{119}	*5046*	*50119.93c*	*JENDL-3*	*O.K.,no γ; URP*
Sn^{120}	*5049*	*50120.93c*	*JENDL-3*	*O.K.,no γ; URP*
Sn^{122}	*5055*	*50122.93c*	*JENDL-3*	*O.K.,no γ; URP*
Sn^{124}	*5061*	*50124.93c*	*JENDL-3*	*O.K.,no γ; URP*
W^{nat}	7400	74000.92c	JENDL-3	O.K.,;neg. neutr. kerma .0022-2.87 MeV
Re^{185}	7525	75185.92c	ENDF/B-6	O.K.;no γ production; URP
Re^{187}	7531	75187.92c	ENDF/B-6	O.K.;no γ production; URP

table 5

content of sublibrary 103 of EFF-2.3, element identifier and processing comments

in *italics* are the new entries in respect to the latest report (Dec. 92)

elem.	MAT	ZAID	origin	comments
Li^7	*328*	*3007.93c*	*Un. Birm.*	***MF6*** *: translation to K.M. not " well done"+ problems in running MCNP with emission cosine >> 1*
Be^9	*425*	*4009.93c*	*Un. Birm.*	,, ,, ,,
Al^{27}	*1325*	*13027.93c*	*ENEA Bo*	,, ,, ,,
Si^{28}	*1425*	*14028.93c*	*ENEA Bo*	,, ,, ,,
Cr^{52}	*2431*		*IRK Vienna*	***MF6****, MT 10: it is not possible in ACER to have a neutron yield energy dependent for reactions other then fission*
Fe^{56}	*2631*		*IRK Vienna*	,, ,, ,,
Ni^{58}	*2825*		*IRK Vienna*	,, ,, ,,
Ni^{59}	2828		ENDF/B-6	no MF4, MF5, MF6
Pb^{nat}	*8200*	*82000.93c*	*ECN Petten*	***MF6*** *,not able for n+γtransport*
Cr^{50}	2425	24050.92c	JEF-2	O.K.
Cr^{53}	2434	24053.92c	JEF-2	O.K.
Cr^{54}	2437	24054.92c	JEF-2	O.K.
Fe^{54}	2625	26054.92c	JEF-2	O.K.
Fe^{57}	2634	26057.92c	JEF-2	O.K.
Fe^{58}	2637	26058.92c	JEF-2	O.K.; neg. neutr. kerma .015-.2 MeV;URP
Ni^{61}	*2834*	*28061.93c*	*ENDF/B-6*	***MF6*** *,not able for n+γtransport*
Ni^{62}	*2837*	*28062.93c*	*ENDF/B-6*	,, ,, ,,
Ni^{64}	*2843*	*28064.93c*	*ENDF/B-6*	,, ,, ,,
Ni^{60}	*2831*		*IRK Vienna*	***MF6, MT 10***

table 6

content of sublibrary 400 of EFF-2.3. element identifier and processing comments
in *italics* are the new entries in respect to the latest report (Dec. 92)

elem.	MAT	ZAID	origin	comments
H	*125*	*1001.93c*	*ENDF/B-5*	*O.K.*
D	*128*		*ENDF/B-6*	*in **MF6, MT16** only LAW=1 is allowed, not LAW=6 (energy available in the c.m. system)*
T	*131*	*1003.93c*	*ENDF/B-4*	*it was necessary to add MT102=0. to make it pass subroutine UNIONX of ACER*
He^3	*225*		*ENDF/B-5*	*as above but with no success*
He^4	*228*		*ENDF/B-6*	,, ,, ,,
Li^6	*325*	*3006.93c*	*ENDF/B-5*	*O.K.*
B^{10}	*525*	*5010.93c*	*ENDF/B-6*	***MF6** ,not able for n+γ transport*
B^{11}	*528*	*5011.93c*	*ENDF/B-6*	,, ,, ,,
F^{19}	*925*	*9019.93c*	*ENDF/B-4*	*running MCNP some problems in generating photons*
Na^{23}	*1125*	*11023.93c*	*JENDL-3*	*O.K.*
Cl	*1700*	*17000.93c*	*ENDF/B-4*	*O.K.*
Ar^{36}	*1825*		*RCN-2*	*no MF5 for MT22*
Ar^{38}	*1831*		*RCN-2*	,, ,, ,,
Ar^{40}	*1837*	*18040.93c*	*RCN-2*	*O.K.; no γ*
K	*1900*	*19000.93c*	*ENDF/B-4*	*O.K.*
Bi^{209}	*8325*	*83209.93c*	*BRC*	*O.K.; no γ*

table 7

MCNP dictionary for EFF-2.3 library: identifier, atomic mass (in neutron mass) and data handling information

```
3007.93C 6.955732 EFF231 0 1 1 46393 0 0 2.5300E-08
4009.93C 8.934780 EFF231 0 1 11612 17874 0 0 2.5300E-08
13027.93C 26.749800 EFF231 0 1 16093 36432 0 0 2.5300E-08
14028.93C 27.736600 EFF231 0 1 25213 37720 0 0 2.5300E-08
24050.92C 49.516900 EFF231 0 1 34655 96431 0 0 2.5300E-08
24053.92C 52.485400 EFF231 0 1 58775 93679 0 0 2.5300E-08
24054.92C 53.475000 EFF231 0 1 82207 54082 0 0 2.5300E-08
26054.92C 53.476000 EFF231 0 1 95740 95030 0 0 2.5300E-08
26057.92C 56.446000 EFF231 0 1 119510 112029 0 0 2.5300E-08
26058.92C 57.436000 EFF231 0 1 147530 70976 0 0 2.5300E-08
28061.93C 60.408000 EFF231 0 1 165286 84698 0 0 2.5300E-08
28062.93C 61.396000 EFF231 0 1 186473 65035 0 0 2.5300E-08
28064.93C 63.379000 EFF231 0 1 202744 56612 0 0 2.5300E-08
82000.93C 205.430000 EFF231 0 1 216909 53545 0 0 2.5300E-08
22000.92C 47.481800 EFF232  0 1 1 67223 0 0 2.5300E-08
23000.92C 50.504000 EFF232  0 1 16819 128227 0 0 2.5300E-08
25055.93C 54.466100 EFF232  0 1 48888 142827 0 0 2.5300E-08
27059.92C 58.426900 EFF232  0 1 84607 126097 0 0 2.5300E-08
29063.93C 62.389000 EFF232  0 1 116144 71945 0 0 2.5300E-08
29065.93C 64.370000 EFF232  0 1 134143 69982 0 0 2.5300E-08
40000.93C 90.440300 EFF232  0 1 151651 37633 0 0 2.5300E-08
41093.92C 92.105100 EFF232  0 1 161072 70155 0 0 2.5300E-08
42000.92C 95.116000 EFF232  0 1 178623 21453 0 0 2.5300E-08
56130.93C 128.790000 EFF232  0 1 183999 44990 0 0 2.5300E-08
49113.93C 111.934000 EFF232  0 1 195259 44479 0 0 2.5300E-08
49115.93C 113.917000 EFF232  0 1 206391 46880 0 0 2.5300E-08
56134.92C 132.754000 EFF232  0 1 218123 12069 0 0 2.5300E-08
56135.92C 133.747000 EFF232  0 1 221153 12671 0 0 2.5300E-08
56136.92C 134.737000 EFF232  0 1 224333 7558 0 0 2.5300E-08
56137.92C 135.729000 EFF232  0 1 226235 11043 0 0 2.5300E-08
56138.92C 136.715000 EFF232  0 1 229008 7711 0 0 2.5300E-08
73181.92C 179.400000 EFF232  0 1 230948 37419 0 0 2.5300E-08
6000.92C 11.898000 EFF233 0 1 1 23699 0 0 2.5300E-08
7014.92C 13.882780 EFF233 0 1 5938 63364 0 0 2.5300E-08
7015.92C 14.871000 EFF233 0 1 21791 25817 0 0 2.5300E-08
8016.92C 15.853160 EFF233 0 1 28258 66044 0 0 2.5300E-08
12000.92C 24.096300 EFF233 0 1 44781 50124 0 0 2.5300E-08
15031.92C 30.708000 EFF233 0 1 57324 7051 0 0 2.5300E-08
16000.92C 31.788200 EFF233 0 1 59099 94608 0 0 2.5300E-08
20000.92C 39.731900 EFF233 0 1 82763 96105 0 0 2.5300E-08
50112.93C 110.944000 EFF233 0 1 106802 32337 0 0 2.5300E-08
50114.93C 112.925000 EFF233 0 1 114899 32809 0 0 2.5300E-08
50115.93C 113.916000 EFF233 0 1 123114 24525 0 0 2.5300E-08
50116.93C 114.906000 EFF233 0 1 129258 22767 0 0 2.5300E-08
50117.93C 115.899000 EFF233 0 1 134962 39943 0 0 2.5300E-08
50118.93C 116.889000 EFF233 0 1 144960 27456 0 0 2.5300E-08
50119.93C 117.882000 EFF233 0 1 151836 29252 0 0 2.5300E-08
50120.93C 118.872000 EFF233 0 1 159161 44999 0 0 2.5300E-08
50122.93C 120.856000 EFF233 0 1 170423 29325 0 0 2.5300E-08
50124.93C 122.841000 EFF233 0 1 177767 24766 0 0 2.5300E-08
74000.92C 182.269000 EFF233 0 1 183971 109149 0 0 2.5300E-08
74182.92C 180.390000 EFF233 0 1 211271 60739 0 0 2.5300E-08
74183.92C 181.380000 EFF233 0 1 226468 45553 0 0 2.5300E-08
74184.92C 182.370000 EFF233 0 1 237869 50951 0 0 2.5300E-08
74186.92C 184.360000 EFF233 0 1 250619 51748 0 0 2.5300E-08
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75185.92C	183.364000	EFF233	0	1	263568	37515	0	0	2.5300E-08
75187.92C	185.350000	EFF233	0	1	272959	38111	0	0	2.5300E-08
1001.93C	0.999	EFF234	0	1	1	2336	0	0	2.530E-08
1003.93C	2.990	EFF234	0	1	597	3072	0	0	2.530E-08
3006.93C	5.963	EFF234	0	1	1377	10559	0	0	2.530E-08
5010.93C	9.927	EFF234	0	1	4029	20425	0	0	2.530E-08
5011.93C	10.915	EFF234	0	1	9148	109434	0	0	2.530E-08
9019.93C	18.835	EFF234	0	1	36519	42154	0	0	2.530E-08
11023.93C	22.792	EFF234	0	1	47070	49551	0	0	2.530E-08
17000.93C	35.148	EFF234	0	1	59470	33781	0	0	2.530E-08
18040.93C	39.619	EFF234	0	1	67928	14032	0	0	2.530E-08
19000.93C	38.766	EFF234	0	1	71448	19631	0	0	2.530E-08
83209.93C	207.185	EFF234	0	1	76368	72995	0	0	2.530E-08

table 8

content of dosimetry library for MCNP for special use in the benchmark experiment FNG, element identifier

elem.	MAT	ZAID	origin	reactions
Al^{27}	1325	13027.92y	EFF-2	(n,2n) (n,p) (n,α)
Mn^{55}	2525	25055.92y	EFF-2	(n,γ)
Fe^{56}	2631	26056.92y	EFF-2	(n,p)
Ni^{58}	2825	28058.92y	EFF-2	(n,2n) (n,p)
Cu^{63}	2925	29063.92y	EFF-2	(n,2n)
Rh^{103}	4525	45103.92y	JEF-2	(n,n')
In^{115}	4931	49115.92y	EFF-2	(n,n')
In^{115}	4915	49115.93y	EAF-2	(n,n') sum of all reactions to In^{115m}
Au^{197}	7925	79197.92y	JEF-2	(n,γ)
U^{235}	9228	92235.93y	JEF-2	(n,f)
U^{238}	9237	92238.93y	JEF-2	(n,f)
Np^{237}	9346	93237.93y	JEF-2	(n,f)

table 9

MCNP dictionary for EFF-2 ,mainly based, dosimetry library: identifier, atomic mass (in neutron mass) and data handling information

13027.93Y	26.749802	EFFDOS	0	1	1	1606	0	0	3.000E+02
25055.92Y	54.466095	EFFDOS	0	1	415	11412	0	0	3.000E+02
26056.92Y	55.454498	EFFDOS	0	1	3280	856	0	0	3.000E+02
28058.92Y	57.438004	EFFDOS	0	1	3506	19496	0	0	3.000E+02
29063.92Y	62.389008	EFFDOS	0	1	8392	108	0	0	3.000E+02
45103.92Y	102.020996	EFFDOS	0	1	8431	2776	0	0	3.000E+02
49115.92Y	113.917007	EFFDOS	0	1	9137	8536	0	0	3.000E+02
79197.92Y	195.274002	EFFDOS	0	1	11283	39822	0	0	3.000E+02
49115.93Y	113.917007	EFFDOS	0	1	21251	322	0	0	3.000E+02
92235.93Y	233.025	EFFDOS	0	1	21344	45298	0	0	3.000E+02
92238.93Y	236.006	EFFDOS	0	1	32681	60530	0	0	3.000E+02
93237.93Y	235.012	EFFDOS	0	1	47826	12408	0	0	3.000E+02