other words, no standard yield (except for a few newly evaluated ones) was used in the evaluation. The data and their errors have been updated. Among them, 31 are recommended and 7 need to be improved. The recommended data can be used as standard in the evaluation and measurement or as monitor yield in the industry application.

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The Evaluation for Reference Fission Yield of ^{238}U Fission

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Introduction

Some fission products are taken as internal standard nuclides in relative yield and R-value measurement, and some are used as monitors in the nuclear industry, these kind of fission yield (generally, cumulative yields) is refered to the reference yields.

The reference yields are not only the basis of all fission yields evaluation and measurement, but also have widely applications in the nuclear industry.

In the fission yield data evaluation and measurement, the reference yield is very important, good or poor recommended or measurement values depend upon the reference data to a great extent. The fission products to be used as internal reference nuclides in fission yield evaluation and measurement are shown in Table 1.

Table 1 Fission products whose fission yields used as reference data

nuclide	half-life	nuclide	half-life
^{83g}Kr	stable	^{85g}Kr	10.72y
^{85m}Kr	4.480h	^{86}Kr	stable
^{87}Kr	76.3m	^{88}Kr	2.84h
^{89}Sr	50.55d	^{90}Sr	28.6y
^{91g}Y	58.51d	^{95}Zr	64.02d
^{96}Zr	stable	^{95g}Nb	34.97d
^{95m}Nb	3.61d	^{95}Mo	stable
^{99}Mo	66.0h	^{100}Mo	stable
^{101}Ru	stable	^{102}Ru	stable
^{103}Ru	39.26d	^{106}Ru	371.63d
^{105g}Rh	35.36h	^{106g}Rh	29.80s
^{111g}Ag	7.45d	^{115g}Cd	53.46h
^{115m}Cd	44.6d	^{125}Sb	2.73y
^{132}Te	78.2h	^{131}I	8.04d
^{135}I	6.61h	^{131g}Xe	stable
^{131m}Xe	11.9h	^{132}Xe	stable
^{133g}Xe	5.245d	^{133m}Xe	2.188d
^{134g}Xe	stable	^{135g}Xe	9.09h
^{135m}Xe	15.29m	^{136}Xe	stable
^{137}Xe	3.818m	^{138}Xe	14.08m
^{133}Cs	stable	^{134g}Cs	2.062y
^{136}Cs	13.16d	^{137}Cs	30.17y
^{140}Ba	12.746d	^{140}La	40.272h
^{141}Ce	32.501d	^{143}Ce	33.0h
^{144}Ce	284.4d	^{141}Pr	stable
^{144g}Pr	17.28m	^{143}Nd	stable
^{144}Nd	stable	^{145}Nd	stable
^{146}Nd	stable	^{147}Nd	10.98d
^{148}Nd	stable	^{147}Pm	2.6234y
^{148g}Pm	5.370d	^{148m}Pm	41.29d
^{149}Pm	53.08h	^{151}Pm	28.40h
^{149}Sm	stable	^{151}Sm	90y
^{153}Sm	46.7h	^{153}Eu	stable
^{154g}Eu	8.8y	^{155}Eu	4.96y
^{156}Eu	15.19d	^{161}Tb	6.90d

In the application of nuclear industry, the reference yield data plays an important role, for example, in the reactor physics application, the reference yield data are widely used in the decay heat estimation, burn-up credit study, lose of reactivity per cycle, evaluation of energy release due to fission, transmutation studies, and calculation on the shielding, dosimetry, fuel handling, waste disposal, safety, as well as other nuclear physics calculation. In this field, about 60 reference yields were proposed, the product nuclides have short, medium and long half-life as shown in Table 2.

Table 2 Reference yields required for reactor physics applications

Nuclide	B. U. PWR	B. U. FBR	Monitoring	D. N.	C. E. D. H*	Radiotox
^{140}La			×××			
^{92}Sr			×××			
^{103}Ru		×××	×××		×××	
^{95}Zr			×××			
^{144}Ce			×××			
^{106}Ru			×××			
^{137}Cs			×××			
^{134}Cs			×××		×××	
^{154}Eu			×××			
^{135}Xe	×××					
^{103}Rh	×××	×××				
^{143}Nd	×××	×××				
^{131}Xe	×××	×××				
^{133}Cs	×××	×××				
^{149}Sm	×××	×××				
^{99}Tc	×××	×××				×××
^{152}Sm	×××					
^{147}Pm	×××	×××				
^{151}Sm	×××	×××				
^{153}Eu	×××	×××				
^{145}Nd	×××	×××				
^{155}Eu	×××					
^{154}Eu	×××					
^{109}Ag	×××	×××				
^{155}Gd	×××					
^{95}Mo	×××	×××				
^{147}Sm	×××					
^{150}Sm	×××					
^{101}Ru	×××	×××				
^{148m}Pm					×××	
^{148}Pm					×××	
^{156}Eu					×××	
^{136}Cs					×××	
^{79}Se						×××
^{93}Zr						×××
^{126}Sn						×××
^{129}I						×××
^{135}Cs		×××				×××
^{105}Pd		×××				
^{107}Pd		×××				
^{97}Mo		×××				
^{102}Ru		×××				
^{104}Ru		×××				
^{91}Zr		×××				
^{137}I				×××		
^{89}Br				×××		
^{94}Rb				×××		
^{90}Br				×××		
^{88}Br				×××		
^{85}As				×××		
^{138}I				×××		
^{98m}Y				×××		
^{95}Rb				×××		
^{139}I				×××		
^{97}Br				×××		
^{93}Rb				×××		
^{99}Y				×××		
^{91}Br				×××		
^{135}Sb				×××		

* Capture Effect on Decay Heat

From the mentioned above, a great number of reference yields are required in both fission yield measurement (or/and evaluation) and nuclear industry application, unfortunately, the accuracy of the reference fission yields from the existing libraries are not satisfied for the practical application so far, actually the discrepancy between these libraries for some reference yields is too large for the reactor physics application, it seems that much effort should be made in the evaluation and measurement of reference fission yields.

According to the CRP's requirement, the evaluation of reference fission yields have been and will be carried out in CNDC, as a part of the whole work (contract No.9504/R_0/Regular Budget Fund), the evaluation for 29 reference fission yields of 15 product nuclides from ^{238}U fission have been completed.

1 Data Analysis and Evaluation

1.1 Data Collection

All experimental data available up to now were collected, most of them were obtained from EXFOR master data library, some data were taken from the temporary EXFOR file, which were measured and compiled in China and haven't been merged into EXFOR master library. Some data were also collected from the publication concerned.

1.2 Data Selection

The EXFOR BIB information and papers concerned were read carefully and analysed in physics. The data were adopted or abandoned according to the measured date, method, facility, detector, monitor, data error and discrepancy situation with others. In general, the following data were abandoned:

a. The data measured is not required, for example, the data were measured at epithermal neutron source.

b. The data obtained by relatively measured, but their standards are not given. This is necessary especially for the evaluation of 'reference yield', the 'standard data' could not be based on the other standards, it will introduce more uncertainty.

c. Large discrepancy with others and measured method is not reliable or no information given in detail.

d. The data measured in fifties or earlier especially there are existing large discrepancy with others.

1.3 Error Estimate

The experimental error is a very important parameter in the analysis of data. Experimentalists report the errors in a variety of ways. Usually the fission yield value is reported together with the absolute accuracy that includes all of the uncertainties in the measurement, but some with a standard deviation corresponding to the precision of the measurement only, and some even no at all. So the reported errors sometimes should be adjusted to a reasonable level of the absolute accuracy.

For different measurement techniques, the different adjusting limits are set. No data are allowed to be more accurate than these limits.

For the absolute measurement, the adjusting limits are set as follows:

15% ~ 20% for Geiger-counter measurement pre-1960;

8% ~ 15% and 6% ~ 8% for NaI(Tl) measurement pre-1965 and after 1965, respectively;

4% ~ 8% for radiochemical measurement with GeLi detector;

3 % ~ 6% for Ge(Li) direct gamma-ray spectrometry measurement;

1% ~ 3% for mass spectrometry measurement.

For the ratio measurement, the adjusting limits are set as follows:

5% ~ 6% for Geiger-counter measurements;

3% ~ 5% for radiochemical measurements;

2% ~ 3% for Ge(Li) direct gamma-ray spectrometry measurements;

1% ~ 2% for mass spectrometry measurements.

1.4 Data Correction

The data were corrected for standard yield data, (intensity, fission cross section, and standard cross section used for neutron flux monitor such as Al(n,α) cross section if necessary.

1.4.1 Standard Fission Yield

As mentioned above, in general case, the fission yield obtained by relatively measured were not directly adopted, only the data taken ^{99}Mo(F,H) from ^{238}U and ^{135}Xe(F), ^{140}La(T) from ^{235}U as standards were adopted, for which the fission yield data have just been evaluated by us[1], respectively. And these relatively measured yield were corrected by using these newly standard data.

1.4.2 Gamma intensity

If the γ intensity used is given by an author, the data were corrected by using following new intensity data in order: evaluated decay data at CNDC, Table of Radioactive Isotopes, ENDSF computer library.

1.4.3 Fission Cross Section and other Standard Cross Section

If the fission cross section and standard cross section used for neutron flux monitor were given by the author of the measurement, the fission yields were corrected by using the new cross sections taken from ENDF/B-6.

2 Data Processing

After making the selection, analysis, and correction of the experimental data as mentioned above, all the adjusted experimental data were obtained, then the data processing was made. The method for data processing are the same as the fission yields evaluation of ^{235}U, the more detail please refer to the "Fission Yield Data Evaluation System-FYDES" by Liu Tingjin, here only the important content are repeated.

2.1 Data Average

If only the absolute yield are existing and there are several measurement data sets with the same incident neutron energy, target and product nuclide, the data are averaged with code AVERAG.

For a set of n measurements of value x_i and assessed standard deviation Δx_i, the following averages and the errors are calculated:

Weighted means yield

$$y_w = \frac{\sum_{i=1}^{n} \frac{1}{(\Delta x_i)^2} x_i}{\sum_{i=1}^{n} \frac{1}{(\Delta x_i)^2}}$$

The internal and external standard deviations are, respectively:

$$\sigma_1 = \sqrt{\frac{1}{\sum_{i=1}^{n} \frac{1}{(\Delta x_i)^2}}}$$

and

$$\sigma_2 = \sigma_1 \varepsilon = \sigma_1 \sqrt{\frac{\sum_{i=1}^{n} \frac{(x_i - y_w)^2}{(\Delta x_i)^2}}{n-1}}$$

A useful test of the consistency of the data is the χ^2 test:

$$\chi^2 = \sum_{i=1}^{n} \frac{(x_i - y_w)^2}{(\Delta x_i)^2} \Big/ n-1$$

The weighted mean yield was taken as the recommended value, and its error was taken to be the σ^2.

2.2 Simultaneous Evaluation

If there are several absolute fission yields and their ratio measurements for some product nuclides at the same energy and target, in order to avoid the introduction of other 'standards', the simultaneous evaluation is necessary, and is completed with the code ZOTT, taking account of measured absolute fission yields, their ratios and the measured errors as well as their correlations, making them consistent. In order to avoid too complicated, in the case of existing several absolute yields or ratio values for the same product nuclide, the data should be averaged with code AVERAG before the simultaneous evaluation, then the weighted means were used for the input data of code ZOTT. Using this code, not only the adjusted fission yields and ratios, but also their covariance matrix can be calculated.

3 Result, Comparison and Discussion

The evaluated results are as shown in Table 3.

3.1 The Errors of Evaluated Data

For the total 35 evaluated fission yield values in the Table 3, we can see that the errors are smaller than 3% for 29 values, 3.1% ~ 5% for 5, and the largest one is 5.2%.

Comparison with the main libraries, our errors are as the same level as those of ENDF/B-6. The CENDL-FY(87) and JEF-2/FY are in another same error level, and much larger than both present work and ENDF/B-6, the distribution of error for the main libraries are as shown in Table 4. No data errors were given in JENDL-3/FY.

Table 3 The results of evaluation for reference yield from ^{238}U fission

Nuclide	Energy	FY	Error	Data Sets*	Processed
^{92}Sr	F	4.4403E+00	1.0979E-01	2(2)	A S
	H	3.9440E+00	8.2076E-02	4(3)	A S
^{97}Mo	F	5.6021E+00	3.8838E-02	3(2)	A S
	H	5.4470E+00	9.7386E-02	3(1)	A S
^{103}Ru	F	6.0778E+00	5.8911E-02	8(4)	A S
	H	4.6604E+00	9.0389E-02	6(4)	A S
^{106}Ru	F	2.3922E+00	1.0967E-01	2(1)	A S
	H	2.5478E+00	1.1710E-01	2	A
^{131}I					
	H	3.8538E+00	6.7381E-02	5(2)	A S
^{133}I					
	H	5.9714E+00	9.6900E-02	6(3)	A S
^{131}Xe	F	3.2330E+00	6.9613E-02	4(2)	A S
	H	3.8143E+00	1.4630E-01	3	A
^{134}Xe					
	H	6.4616E+00	1.0870E-01	4	A S
^{135}Xe	F	6.7282E+00	1.5930E-01	6	A
	H	5.8049E+00	1.2020E-01	6	A
^{133}Cs	F	6.7631E+00	7.1388E-02	2(2)	A S
	H	6.0174E+00	1.6800E-01	2	A
^{137}Cs	F	5.9376E+00	5.0172E-02	4	A S
	H	5.1737E+00	1.8120E-01	4	A
^{140}Ba	F	5.8798E+00	3.5758E-02	12(1)	A S
	H	4.5693E+00	5.3390E-02	7	A S
^{140}La	F	5.8948E+00	8.8467E-02	3(2)	A S
	H	4.6529E+00	1.3010E-01	3	A
^{143}Nd	F	4.5338E+00	3.2080E-02	4(3)	A S
	H	4.1917E+00	1.1580E-01	4	A
^{145}Nd	F	3.7548E+00	3.2806E-02	2(1)	A S
	H	3.0894E+00	1.6000E-01	1	
^{148}Nd	F	2.1015E+00	1.8580E-02	3(1)	A S
^{147}Sm	F	2.5266E+00	3.0405E-02	2(2)	A S
	H	2.0365E+00	5.3261E-02	3	A S
^{149}Sm	F	1.5850E+00	1.9000E-02	2	A
	H	1.2038E+00	3.4174E-02	1(1)	S
^{151}Sm	F	7.8691E-01	2.4570E-02	2(2)	A S
	H	6.5409E-01	1.7017E-02	1(1)	S
^{152}Sm	F	5.2168E-01	5.7700E-03	2	A

Meaning of the symble in the table:

* Number of absolutely measured data sets, and the number of ratio measured data sets in parentheses ( )

A The average with weight

S Simultaneous evaluation

F Fission spectrum average

H High energy (around 14 MeV)

Table 4 The comparison of error distribution for main libraries

Libraries	Data Sets for Different Errors Arange							
	<1.0%	1.1%~2%	2.1%~3%	3.1%~5%	5.1%~7%	7.1%~9%	9.1%~20%	>20%
This work	7	11	11	5	1	0	0	0
ENDF/B-6	9	19	4	2	1	0	0	0
CENDL-FY(87)	1	8	4	8	3	3	1	7
JEF-2/FY	0	1	6	9	5	7	7	0

3.2 Comparison and Discussion

The present evaluated data were also compared with the ENDF/B-6, JENDL-3/FY, CENDL-FY(87), and JEF-2/FY. It was found that the present evaluated data are in good agreement with other main libraries for most nuclides within the quoted error limits. But for ^{97}Mo(H), ^{143}Nd(H), ^{149}Sm(H), and ^{151}Sm(H), the present evaluated data are obviously larger or smaller than those of other libraries. It should be emphasized that present evaluated data reflect the existing experimental data for ^{97}Mo(H)[2,3], ^{143}Nd(H)[2~5], ^{149}Sm(H)[3,6], and ^{151}Sm(H)[2,6], and these experimental data are in good agreement with each other for each product nuclide, a typical example is ^{151}Sm(H), present evaluated value is 0.6541, which is obviously smaller than the value about 0.8000 of other libraries as shown in Table 5, but there are existing two experimental data sets for ^{151}Sm(H), one absolute measurement is 0.6310[2], another ratio measurement(^{151}Sm/^{147}Sm) is 0.322[6], if taking the value for ^{147}Sm as 2.0365 of our new evaluated data (which is in good agreement with those of other libraries), then the absolute value is 0.6558 deduced from this ratio measurement, so comparing these two experimental data sets with present evaluated data, our evaluated data for ^{151}Sm(H) seems more reasonable than those of other libraries. Anyway, the fission yields for these reference product nuclides should be studied further based on more new experimental data.

4 Conclusion

The 35 cumulative fission yield for ^{238}U were evaluated based on the available experimental data up to now by using weighted average code AVERAG and simultaneous evaluation code ZOTT, only absolute yield and ratio measurements were used, i.e., no other standard yields were introduced in present evaluation. Among the 35 fission yields, 29 values are the reference fission yields required by the RCP's contract, no evaluated data for ^{152}Sm(H) was obtained due to no experimental data available, the others are needed for present simultaneous evaluation and also listed here.

Table 5 Comparison of evaluated reference fission yields for ^{238}U with other main libraries

Nuclide	Libraries	F	H
^{92}Sr	This work	4.4403±0.1098	3.9440±0.0821
	ENDF/B-6	4.3123±0.1207	3.8760±0.1085
	JENDL-3/FY	4.5092	3.9095
	CENDL-FY(87)	4.2896±0.1861	3.9739±0.2940
	JEF-2/FY	4.2263±0.0795	3.7811±0.2387
^{97}Mo	This work	5.6021±0.0388	5.4470±0.0974
	ENDF/B-6	5.5625±0.0389	5.2800±0.0739
	JENDL-3/FY	5.5746	5.3738
	CENDL-FY(87)	5.4706±0.0711	5.2373±5.2334
	JEF-2/FY	5.5235±0.5866	5.2497±0.5974
^{103}Ru	This work	6.0778±0.0589	4.6604±0.0904
	ENDF/B-6	6.2753±0.0879	4.6158±0.0923
	JENDL-3/FY	6.2096	4.6250
	CENDL-FY(87)	6.3122±0.1112	4.5116±0.1625
	JEF-2/FY	6.0728±0.2251	4.4617±0.4805
^{106}Ru	This work	2.3922±0.1097	2.5478±0.1171
	ENDF/B-6	2.4897±0.0349	2.4546±0.0687
	JENDL-3/FY	2.5311	2.4570
	CENDL-FY(87)	2.6260±0.1122	2.4274±0.2189
	JEF-2/FY	2.5491±0.2727	2.5004±0.1949
^{131}I	This work		3.8538±0.0674
	ENDF/B-6		3.9925±0.0798
	JENDL-3/FY		4.0449
	CENDL-FY(87)		3.7999±0.2027
	JEF-2/FY		3.8261±0.0997
^{133}I	This work		5.9714±0.0969
	ENDF/B-6		5.9999±0.0384
	JENDL-3/FY		6.1263
	CENDL-FY(87)		6.0199±0.1501
	JEF-2/FY5.7		440±0.3223
^{131}Xe	This work	3.2330±0.0696	3.8143±0.1463
	ENDF/B-6	3.2908±0.0329	3.9925±0.0559
	JENDL-3/FY	3.2386	4.0449
	CENDL-FY(87)	3.2035±0.0384	3.9860±0.1096
	JEF-2/FY	3.3041±0.0746	3.8257±0.0997
^{134}Xe	This work		6.4616±0.1087
	ENDF/B-6		6.4531±0.1291
	JENDL-3/FY		6.5541
	CENDL-FY(87)		6.5986±5.4260
	JEF-2/FY		6.1386±0.2162
^{135}Xe	This work	6.7282±0.1593	5.8049±0.1202
	ENDF/B-6	6.9676±0.0697	5.8393±0.1168
	JENDL-3/FY	6.8109	5.8186
	CENDL-FY(87)	6.6363±0.4964	5.9361±0.1455
	JEF-2/FY	6.5680±0.4936	5.4620±0.3320
^{133}Cs	This work	6.7631±0.0714	6.0174±0.1680
	ENDF/B-6	6.7610±0.0338	6.0172±0.0842
	JENDL-3/FY	6.6062	6.1449
	CENDL-FY(87)	6.6924±0.3416	6.1495±6.1503
	JEF-2/FY	6.7252±0.1698	5.7441±0.1200
^{137}Cs	This work	5.9376±0.0502	5.1737±0.1812
	ENDF/B-6	6.0525±0.0605	5.1460±0.1441
	JENDL-3/FY	6.0907	4.9857
	CENDL-FY(87)	5.9669±0.2089	5.0688±0.2273

Nuclide	Libraries	F	H
^{137}Cs	JEF-2/FY	6.0045±0.2499	5.6732±0.5332
^{140}Ba	This work	5.8798±0.0358	4.5693±0.0534
	ENDF/B-6	5.8152±0.0407	4.6070±0.0645
	JENDL-3/FY	5.9882	4.6523
	CENDL-FY(87)	5.9284±0.0629	4.5684±0.1022
	JEF-2/FY	5.7428±0.4118	4.7048±0.2238
^{140}La	This work	5.8948±0.0885	4.6529±0.1301
	ENDF/B-6	5.8153±0.0407	4.6112±0.0646
	JENDL-3/FY	5.9882	4.6525
	CENDL-FY(87)	6.0336±0.2173	4.8061±0.2256
	JEF-2/FY	5.7428±0.4118	4.7049±0.2237
^{143}Nd	This work	4.5338±0.0321	4.1917±0.1158
	ENDF/B-6	4.6221±0.0324	3.9087±0.0782
	JENDL-3/FY	4.5666	3.9229
	CENDL-FY(87)	4.5375±0.1803	3.9333±3.9298
	JEF-2/FY	4.8241±0.4089	3.9586±0.1140
^{145}Nd	This work	3.7548±0.0328	3.0894±0.1600
	ENDF/B-6	3.8090±0.0267	3.0038±0.1202
	JENDL-3/FY	3.7559	3.0060
	CENDL-FY(87)	3.7550±0.0352	2.9807±2.9787
	JEF-2/FY	3.8920±0.1578	2.9195±0.1954
^{148}Nd	This work	2.1015±0.0186	
	ENDF/B-6	2.1125±0.0148	
	JENDL-3/FY	2.0816	
	CENDL-FY(87)	2.0944±0.0356	
	JEF-2/FY	2.2791±0.1520	
^{147}Sm	This work	2.5266±0.0304	2.0365±0.0533
	ENDF/B-6	2.5927±0.0181	2.0912±0.0418
	JENDL-3/FY	2.5298	2.0970
	CENDL-FY(87)	2.5149±0.0387	2.0617±0.2396
	JEF-2/FY	2.6632±0.0837	2.1715±0.2190
^{149}Sm	This work	1.5850±0.0190	1.2038±0.0342
	ENDF/B-6	1.6253±0.0163	1.4582±0.0875
	JENDL-3/FY	1.6076	1.4227
	CENDL-FY(87)	1.5866±0.1083	1.4303±1.4291
	JEF-2/FY	1.6647±0.0722	1.3409±0.1068
^{151}Sm	This work	0.7869±0.0246	0.6541±0.0170
	ENDF/B-6	0.7994±0.0112	0.8015±0.0321
	JENDL-3/FY	0.8006	0.8014
	CENDL-FY(87)	0.7991±0.0136	0.8169±0.8163
	JEF-2/FY	0.8091±0.0293	0.7872±0.0757
^{152}Sm	This work	0.5217±0.0058	
	ENDF/B-6	0.5302±0.0053	
	JENDL-3/FY	0.5208	
	CENDL-FY(87)	0.5192±0.0093	
	JEF-2/FY	0.5499±0.0394	

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