Technical note on using JEFF-3.1 and JEFF-3.1.1 data to calculate neutron emission from spontaneous fission and (α,n) reactions with FISPIN.

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EXECUTIVE SUMMARY

This paper describes the production of new heavy element decay data files for FISPIN using the JEFF-3.1 and JEFF-3.1.1 data (released by the JEFF project for testing) with revised spontaneous fission and (α, n) reaction terms. In addition to the heavy element decay data files for metal and uranium dioxide fuel a new file of data for the FISPIN Graphical User Interface (FISGUI) was produced to allow (α, n) neutron emission to be calculated for arbitrary compounds. The new data was then compared to neutron emission measurements showing good agreement. Finally, it is shown that the new data increases the neutron emission for typical PWR fuel by ~3%. In addition, a review was carried out for the nuclides which most changed between JEF-2.2 and JEFF-3.1, U232, U235, U236, Pu239 and Am241. This showed that the U232 and Pu236 differ significantly from the values currently in ENDSF and probably require a review by an evaluator.

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1 Introduction

The handling, transport, storage and reprocessing of spent nuclear fuel requires the estimation of the radiation source terms of the fuel. The radiation consists of two long range components that need to be determined so that radiation doses to workers or the general pubic can be determined. The first radiation type is high energy photons; x- and gamma- rays, typically in the region of keV to 10 MeV. As the distance a photon will travel is determined by its energy and the material through which it is travelling, it is important to know both the number of photons and their energy spectra, as source terms for the radiation transport calculation to estimate the radiation dose outside of the fuel's containment. The second component is neutrons. In the case of neutrons, their initial energy spectra is of much less importance, due to the different physics governing their interactions with matter, and the neutron production rate is the important source term to radiation transport calculations of dose.

In spent nuclear fuel the radiation results from radioactive decay processes. In the case of photons, delayed neutrons and spontaneous fission neutrons these arise directly from the decay processes. However, other indirect radiation production methods are possible. In spent nuclear fuel the dominant indirect methods result from reactions that the direct particles from decay can produce (e.g. alpha and beta particles, neutrons, protons or photons).

In practice, a short time after irradiation the neutron emission is dominated by spontaneous fission and alpha particle reactions on light elements producing neutrons, (α, n) reactions.

Almost all delayed neutron emitters with significant fission product yields have half-lives of less than two minutes, and thus virtually no delayed neutrons are emitted after a few minutes. In practice, this would be too short a time to remove fuel from a commercial reactor.

Similarly, photo nuclear reactions have very small cross-sections and are usually trivial in commercial fuel. Work by A. J. Mill (Ref 1) showed that this is only a significant contribution at very low burnups (<20% of total emission at 1 GWd/t) falling rapidly to 0.1% of total at 7 GWd/t. Also unlike spontaneous fission and (α, n) reaction components which increase with cooling, the nuclides producing photons with sufficient energy to cause photo nuclear reactions all have half-lives less than 1 year.

As no significant quantity of low burnup fuel with short cooling is produced in current commercial reactors it was decided that the current FISPIN assumptions of only including spontaneous fission and (α, n) reaction neutron emission is adequate at the current time.

This paper describes the production of a new heavy element decay data files for FISPIN (Ref 2) using JEFF-3.1 and JEFF-3.1.1 data (released for testing by the JEFF project, Ref 3) including revised spontaneous fission and (α, n) reaction terms derived from these new data. In addition to the heavy element decay data files for metal and uranium dioxide fuel a new file of data for the FISPIN Graphical User Interface (FISGUI) (Ref 4) was produced to allow (α, n) neutron emission to be calculated for arbitrary compounds. The new data was then compared to neutron emission measurements showing good agreement. Finally, it is shown that the new data for spontaneous fission increases the neutron emission for typical PWR fuel by ~3%. In addition, a review was carried out for the nuclides which most changed between JEF-2.2 and JEFF-3.1; U232, U235, U236, Pu239 and Am241. This showed that the U232 and Pu236 differ significantly from the values currently in ENDSF and probably require a review by an evaluator.

2 FISPIN neutron emission calculation methodology

2.1 Spontaneous fission.

FISPIN calculates spontaneous fission neutron emission by multiplying the nuclide activities by a set of spontaneous fission neutron emission per decay values for each nuclide stored in the heavy element decay data file.

2.2 (α,n) reactions

FISPIN calculates the neutron emission from (α,n) using the same approach with average neutron yield per decay values, calculated using information on the nuclide's alpha particle emission and the mixture of elements present. The standard library uses values associated with uranium dioxide, but using the FISGUI, or the earlier RUNFISP code (Ref 5), any compound can be calculated using the West approach (Ref 6). This uses the (α, n) yields per element and the alpha stopping powers per element to calculate the neutron yield per alpha particle in an arbitrary mixture or compound. The approach and data used are described in the RUNFISP report.

3 Production of files using JEFF-3.1 and JEFF-3.1.1 data to calculate neutron emission

The JEFF-3.1 and JEFF-3.1.1 decay data format is described in the ENDF-6 format guide report (Ref 7). This uses the same format as the JEF-2.2 file (Ref 8) that is the basis of the current heavy element decay data. The neutron emission is given by two values; the spontaneous fission branching fraction and the average number of neutrons produced per fission ($\bar{\nu}$). It should be noted that the neutron emission in these files do not follow the current format, as $\bar{\nu}_p$ and $\bar{\nu}_d$ are not given as the FC and FD parameters as required. However, the field that should contain the $\bar{\nu}_p$ value contains the $\bar{\nu}_p$ multiplied by the branching ratio, i.e. the neutron emission per decay. This appears to be so that the neutron spectra can be given normalised to unity, which is implied, but not explicitly stated as a requirement for neutrons.

It should also be noted that the JEFF-3.1 website includes an error report that the U238 spontaneous fission branching ratio is 100 times too big, as is the neutron emission. In this work this was corrected. In JEFF-3.1.1 this is corrected and no special treatment was required.

A comparison of the existing JEF-2.2 and new JEFF-3.1 and JEFF-3.1.1 spontaneous fission neutron emission rates are given in the following table 1.

It should be noted that U232, U235, U236, Pu239 and Am241 change by greater than 10%. To investigate these changes N.J. Thompson and A.B. Garnsworth, students of the University of Surrey were asked to compare the spontaneous fission branching ratios in JEFF-3.1 with those in ENDSF. Their results are given in Table 2. There are only two significant differences between the data sets. First, they noted that the large discrepancy for U232 possibly originates from the branching ratio of spontaneous fission and for (208Pb + 24Ne) cluster decay being combined. Secondly, the difference in Pu236 is about a factor of two. It is recommended that these differences are investigated further by an evaluator.

Table 1: Neutron emission rate per decay from JEF-2.2, JEFF-3.1 and JEFF-3.1.1.

Nuclide	Neutron emission per decay JEF-2.2	JEFF-3.1	JEFF-3.1.1	Ratio JEFF-3.1/ JEF-2.2	Ratio JEFF-3.1.1/ JEF-2.2
Th230	3.475E-13	3.475E-13	3.475E-13	1.00	1.00
Th232	2.1E-11	2.1E-11	2.1E-11	1.00	1.00
Pa231	5.13E-12	5.13E-12	5.13E-12	1.00	1.00
U232	1.539E-16	1.539E-12	1.539E-12	**10000.00**	**10000.00**
U234	3.06E-11	3.06E-11	3.06E-11	1.00	1.00
U235	3.74E-10	1.3464E-10	1.3464E-10	**0.36**	**0.36**
U236	2.28E-09	1.71E-09	1.71E-09	**0.75**	**0.75**
U238	1.08E-06	1.092E-06	1.092E-06	**1.01**	**1.01**
Pu236	1.802E-09	1.7384E-09	1.7384E-09	**0.96**	**0.96**
Pu238	4.111E-09	4.1106E-09	4.1106E-09	1.00	1.00
Pu239	1.021E-11	7.192E-12	7.192E-12	**0.70**	**0.70**
Pu240	1.226E-07	1.22607E-07	1.2261E-07	1.00	1.00
Pu242	1.178E-05	1.17755E-05	1.1775E-05	1.00	1.00
Pu244	2.863E-03	2.8625E-03	2.8625E-03	1.00	1.00
Am241	9.425E-12	1.075E-11	1.075E-11	**1.14**	**1.14**
Am242m	4.096E-10	4.096E-10	4.096E-10	1.00	1.00
Am243	9.657E-11	9.657E-11	9.657E-11	1.00	1.00
Cm242	1.6E-07	1.54208E-07	1.5421E-07	**0.96**	**0.96**
Cm244	3.62E-06	3.70875E-06	3.7088E-06	**1.02**	**1.02**
Cm246	7.706E-04	7.70607E-04	7.7061E-04	1.00	1.00
Cm248	2.611E-01	2.61099E-01	2.611E-01	1.00	1.00
Cm250	2.31E+00	2.31E+00	2.31E+00	1.00	1.00
Bk 249	1.595E-09	1.5946E-09	1.5946E-09	1.00	1.00
Cf249	1.768E-08	1.768E-08	1.768E-08	1.00	1.00
Cf250	2.710E-03	2.7104E-03	2.7104E-03	1.00	1.00
Cf252	1.161E-01	1.16136E-01	1.1614E-01	1.00	1.00
Es253	4.089E-07	4.089E-07	4.089E-07	1.00	1.00

Table 2: Comparison of JEFF-3.1 SF branching ratios with ENDSF

Nuclide	**JEFF-3.1 S.F. Branching ratio (%)**	**ENSDF S.F. Branching Ratio (%)**
230TH	2.50E-13	≤5E-13
231PA	3.00E-12	≤3E-12
232TH	1.40E-11	1.1E-11 (4)
232U	9.00E-13	8.5E-22 (18)
234U	1.70E-11	1.73E-11 (10)
235U	7.20E-11	7E-11 (2)
236PU	8.20E-10	1.9E-09 (4)
236U	9.00E-10	9.4E-10 (4)
238PU	1.86E-09	1.9E-09 (1)

Nuclide	**JEFF-3.1 S.F. Branching ratio (%)**	**ENSDF S.F. Branching Ratio (%)**
238U	5.46E-07	5.45E-07 (7)
239PU	3.10E-12	3.1E-12 (6)
240PU	5.70E-08	5.7E-08 (2)
241AM	4.30E-12	4.3E-12 (9)
242AM	1.60E-10	Not given
242CM	6.10E-08	6.2E-08 (3)
242PU	5.50E-06	5.5E-06 (6)
243AM	3.70E-11	3.7E-11 (2)
244CM	1.38E-06	1.37E-06 (3)

Nuclide	**JEFF-3.1 S.F. Branching ratio (%)**	**ENSDF S.F. Branching Ratio (%)**
244PU	1.25E-03	0.00121 (4)
246CM	2.61E-04	0.000262 (7)
248CM	8.26E-02	0.0839 (16)
249BK	4.69E-10	4.7E-10 (8)
249CF	5.20E-09	5E-09 (4)
250CF	7.70E-04	0.00077 (3)
250CM	0.7	≈0.74
252CF	3.09E-02	0.03092 (8)
253ES	8.70E-08	8.7E-08 (3)

The neutron emission data from (α, n) reactions is calculated using the West approximation (Ref 6) that requires neutron yields and stopping powers for the alpha particles from each nuclide. The only new parameters available from JEFF-3.1 (and JEFF-3.1.1) in this approximation are the alpha particle energies. Thus for these libraries the mean alpha energy from each nuclide was calculated.

In previous work, these files included 76 alpha emitting heavy element nuclides. However, for this work this list was extended to all alpha emitters. In should be noted that this included Pb204 with a very low alpha particle energy (1.933 MeV). The evaluator of this nuclide's data reported that this energy was calculated with inconsistent data, and as it is below the threshold energy for (α, n) reactions, it was decided that it was not necessary to include this nuclide, leaving 100 nuclides to be included. These nuclides are compared in table 3. It should be noted that only a few values change by greater than 1% from JEF-2.2 to JEFF-3.1, these are high-lighted. It should be noted that the only difference between JEFF-3.1 and JEFF-3.1.1 is a difference for U238.

Table 3: Comparison of mean alpha energies from JEF-2.2, JEFF-3.1 and JEFF-3.1.1.

Nuclide	Mean alpha energy (eV)			Comparisons		
	J22	J31	J311	J31/J22	J311/J22	J311/J31
Pb-210	3720000	3720000	3720000	1.00000	1.00000	1.00000
Bi-210	4591970	4656488	4656488	**1.01405**	**1.01405**	1.00000
Bi-210m	4913734	4913734	4913734	1.00000	1.00000	1.00000
Bi-211	6566635	6566635	6566635	1.00000	1.00000	1.00000
Bi-212	6053892	6051310	6051310	0.99957	0.99957	1.00000
Bi-212m	6322800	7523620	7523620	**1.18992**	**1.18992**	1.00000
Bi-213	5846222	5846222	5846222	1.00000	1.00000	1.00000
Bi-214	5457243	5457243	5457243	1.00000	1.00000	1.00000
Po-208	5115313	5115010	5115010	0.99994	0.99994	1.00000
Po-209	4881359	4881359	4881359	1.00000	1.00000	1.00000
Po-210	5304550	5304442	5304442	0.99998	0.99998	1.00000
Po-211	7442228	7442228	7442228	1.00000	1.00000	1.00000
Po-211m	7406716	7406716	7406716	1.00000	1.00000	1.00000
Po-212	8784600	8785060	8785060	1.00005	1.00005	1.00000
Po-212m		11562042	11562042	**Not in JEF-2.2**		1.00000
Po-213	8375978	8375978	8375978	1.00000	1.00000	1.00000
Po-214	7687005	7686961	7686961	0.99999	0.99999	1.00000
Po-215	7385893	7385863	7385863	1.00000	1.00000	1.00000
Po-216	6778486	6778585	6778585	1.00001	1.00001	1.00000
Po-218	6002541	6002491	6002491	0.99999	0.99999	1.00000
At-214	8814808	8814808	8814808	1.00000	1.00000	1.00000
At-215	8025800	8025800	8025800	1.00000	1.00000	1.00000
At-216	7793004	7809363	7809363	1.00210	1.00210	1.00000
At-217	7067143	7067143	7067143	1.00000	1.00000	1.00000
At-218	6693348	6685736	6685736	0.99886	0.99886	1.00000
At-219	6270000	6270000	6270000	1.00000	1.00000	1.00000
At-220		5943000	5943000	**Not in JEF-2.2**		1.00000
Rn-217	7738761	7738761	7738761	1.00000	1.00000	1.00000
Rn-218	7132043	7128447	7128447	0.99950	0.99950	1.00000
Rn-219	6759055	6759055	6759055	1.00000	1.00000	1.00000
Rn-220	6287590	6287589	6287589	1.00000	1.00000	1.00000
Rn-221	5991360	5988890	5988890	0.99959	0.99959	1.00000
Rn-222	5489299	5489234	5489234	0.99999	0.99999	1.00000
Fr-218	7808120	7808227	7808227	1.00001	1.00001	1.00000
Fr-219	7292305	7307053	7307053	1.00202	1.00202	1.00000
Fr-220	6626544	6574027	6574027	0.99207	0.99207	1.00000
Fr-221	6354216	6354216	6354216	1.00000	1.00000	1.00000
Fr-223	5340000	5340000	5340000	1.00000	1.00000	1.00000
Ra-220	7450430	7450481	7450481	1.00001	1.00001	1.00000
Ra-222	6543652	6546643	6546643	1.00046	1.00046	1.00000
Ra-223	5677889	5677889	5677889	1.00000	1.00000	1.00000
Ra-224	5673287	5672830	5672830	0.99992	0.99992	1.00000

Nuclide	Mean alpha energy (eV)			Comparisons		
	J22	J31	J311	J31/J22	J311/J22	J311/J31
Ra-226	4774701	4773259	4773259	0.99970	0.99970	1.00000
Ac-224	6106618	6073489	6073489	0.99457	0.99457	1.00000
Ac-225	5764040	5764040	5764040	1.00000	1.00000	1.00000
Ac-227	4935569	4935569	4935569	1.00000	1.00000	1.00000
Th-224	7130749	7095636	7095636	0.99508	0.99508	1.00000
Th-226	6307312	6306237	6306237	0.99983	0.99983	1.00000
Th-227	5911048	5901294	5901294	0.99835	0.99835	1.00000
Th-228	5398103	5401029	5401029	1.00054	1.00054	1.00000
Th-229	4860560	4860560	4860560	1.00000	1.00000	1.00000
Th-230	4664828	4664828	4664828	1.00000	1.00000	1.00000
Th-232	4007090	4007090	4007090	1.00000	1.00000	1.00000
Pa-229	5511137	5509144	5509144	0.99964	0.99964	1.00000
Pa-231	4973624	4973624	4973624	1.00000	1.00000	1.00000
U-230	5867485	5867353	5867353	0.99998	0.99998	1.00000
U-232	5303941	5303941	5303941	1.00000	1.00000	1.00000
U-233	4819896	4819896	4819896	1.00000	1.00000	1.00000
U-234	4759174	4759174	4759174	1.00000	1.00000	1.00000
U-235	4386985	4388571	4388571	1.00036	1.00036	1.00000
U-236	4486388	4494131	4494131	1.00173	1.00173	1.00000
U-238	4188247	4187304	4187078	0.99977	0.99972	**0.99995**
Np-235	5070374	4989345	4989345	**0.98402**	**0.98402**	1.00000
Np-236	4980008	4982000	4982000	1.00040	1.00040	1.00000
Np-237	4780620	4777946	4777946	0.99944	0.99944	1.00000
Pu-236	5752059	5753141	5753141	1.00019	1.00019	1.00000
Pu-237	5479363	5479363	5479363	1.00000	1.00000	1.00000
Pu-238	5485985	5485985	5485985	1.00000	1.00000	1.00000
Pu-239	5149097	5150029	5150029	1.00018	1.00018	1.00000
Pu-240	5155595	5155595	5155595	1.00000	1.00000	1.00000
Pu-241	4816332	4900333	4900333	**1.01744**	**1.01744**	1.00000
Pu-242	4890152	4890128	4890128	0.99999	0.99999	1.00000
Pu-244	4580464	4580464	4580464	1.00000	1.00000	1.00000
Am-240	5373562	5373562	5373562	1.00000	1.00000	1.00000
Am-241	5479187	5465168	5465168	0.99744	0.99744	1.00000
Am-242m	5189897	5263391	5263391	**1.01416**	**1.01416**	1.00000
Am-243	5270018	5270018	5270018	1.00000	1.00000	1.00000
Cm-240	6250847	6282165	6282165	1.00501	1.00501	1.00000
Cm-241	5929329	5929329	5929329	1.00000	1.00000	1.00000
Cm-242	6097780	6103838	6103838	1.00099	1.00099	1.00000
Cm-243	5856680	5856680	5856680	1.00000	1.00000	1.00000
Cm-244	5795250	5796763	5796763	1.00026	1.00026	1.00000
Cm-245	5359357	5359357	5359357	1.00000	1.00000	1.00000
Cm-246	5378841	5378841	5378841	1.00000	1.00000	1.00000
Cm-247	4946722	4946722	4946722	1.00000	1.00000	1.00000
Cm-248	5069527	5069527	5069527	1.00000	1.00000	1.00000
Cm-250	5087000	5087000	5087000	1.00000	1.00000	1.00000
Bk-247	5565502	5565502	5565502	1.00000	1.00000	1.00000
Bk-249	5355978	5355978	5355978	1.00000	1.00000	1.00000
Cf-246	6740545	6745598	6745598	1.00075	1.00075	1.00000
Cf-248	6253200	6249404	6249404	0.99939	0.99939	1.00000
Cf-249	5832305	5832305	5832305	1.00000	1.00000	1.00000
Cf-250	6024278	6024278	6024278	1.00000	1.00000	1.00000
Cf-251	5784172	5784172	5784172	1.00000	1.00000	1.00000
Cf-252	6119958	6119958	6119958	1.00000	1.00000	1.00000
Cf-253	5975926	5975926	5975926	1.00000	1.00000	1.00000
Es-253	6627116	6627116	6627116	1.00000	1.00000	1.00000
Es-254	6407017	6400299	6400299	0.99895	0.99895	1.00000
Fm-254	7179206	7186767	7186767	1.00105	1.00105	1.00000
Fm-255	7020307	7054687	7054687	1.00490	1.00490	1.00000

The existing (α, n) neutron yields for each element and the stopping powers were interpolated using the new JEFF-3.1 (and JEFF-3.1.1) mean alpha particle energies to produce new data files that can be used with the FISGUI to generate a heavy element decay file for arbitrary compounds and mixtures. These data files were used to generate new heavy element decay files using (α, n) yields for uranium dioxide. It should be noted that the existing routines in the FISGUI to calculate (α, n) neutron yields required to be modified to use the new files.

4 Validation of the FISPIN calculation of neutron emission

4.1 Neutron emission from fuel samples prior to irradiation

In the early 1980's Lees and West measured the neutron emission from some well characterised plutonium and mixed oxide fuel samples (Ref 9 and 10). Using FISPIN with the existing JEF-2.2 and new JEFF-3.1 and JEFF-3.1.1 based files the following results were obtained for these samples.

Sample	Measured neutron emission (n/s)	Library	Neutron emission			
			SF	(α,n)	Total	C/E
M120 Fast reactor MOX	1648 ± 16	JEF-2.2	1190	460	1650	1.001
		JEFF-3.1	1190	460	1650	1.001
		JEFF-3.1.1	1190	460	1650	1.001
0671 Zebra Mk XIV Pu metal plate	10024 ± 161	JEF-2.2	10198	6	10203	1.018
		JEFF-3.1	10198	6	10203	1.018
		JEFF-3.1.1	10198	6	10203	1.018
IV/R/3467 Zebra Mk IV Pu/U oxide plate	5116 ± 86	JEF-2.2	3022	2008	5029	0.983
		JEFF-3.1	3021	2004	5025	0.982
		JEFF-3.1.1	3021	2004	5025	0.982
ZMC 1616B Zebra Type 'D' Pu/U oxide pin	3949 ± 49	JEF-2.2	2669	1286	3956	1.002
		JEFF-3.1	2669	1283	3953	1.001
		JEFF-3.1.1	2669	1283	3953	1.001

These show very good agreement with experiment, the calculations being within 2% of the measurement. The neutron emission values calculated using all the libraries are very similar, with the JEFF-3.1 and JEFF-3.1.1 results being identical.

4.2 Neutron emission from fuel samples after irradiation

The neutron emission from irradiated fuel samples is dominated by the Cm242 and Cm244 activities. These nuclides usually have large uncertainties in their inventories, often greater than 30%, when comparing calculations with measurements of irradiated samples. In 1989 measurements of CAGR samples were made in the UKAEA research reactor DIMPLE to determine the neutron output and then these samples under went destructive analysis to determine their heavy element composition (Ref 11). It should be noted that it was estimated that the measured nuclides represented about 99% of neutron emission.

The results of using the existing JEF-2.2 and new JEFF-3.1 and JEFF-3.1.1 data in these calculations are shown below:

Sample	Measured neutron emission (n/s)	Library	Calculated neutron emission			
			SF	(α,n)	Total	C/E
31	10306 ± 206	JEF-2.2	9501	454	9955	0.966
		JEFF-3.1	9723	456	10179	0.988
		JEFF-3.1.1	9723	456	10180	0.988
33	19282 ± 386	JEF-2.2	18867	746	19613	1.017
		JEFF-3.1	19295	753	20048	1.040
		JEFF-3.1.1	19295	753	20048	1.040
34	4845 ± 97	JEF-2.2	4153	299	4452	0.919
		JEFF-3.1	4236	300	4536	0.936
		JEFF-3.1.1	4236	300	4536	0.936
35	6091 ± 122	JEF-2.2	5365	331	5696	0.935
		JEFF-3.1	5479	333	5812	0.954
		JEFF-3.1.1	5479	333	5812	0.954

In these cases, there is an approximately 2% increase in neutron emission using the JEFF-3.1 and JEFF-3.1.1, when compared to the original JEF-2.2 results. However, the differences between JEFF-3.1 and JEFF-3.1.1 results are too small with only the sample 31 cases showing a small difference in the above table due to rounding.

5 Estimation of the effect of using the new JEFF-3.1 and JEFF-3.1.1 data for PWR calculations

In previous work (Ref 12) JEFF-3.1 based libraries were used to calculate decay heat for a range of PWR samples which had been previously calculated with JEF-2.2. The table below shows the calculated neutron yields for the WEPCO sample at 39.8 GWd/t cooled to 4.5 years. In this work the same cross-section libraries were used for all three cases, so that the differences in neutron emission resulted only from the spontaneous fission and (a,n) reactions from the radioactive decay libraries.

File	SF neutrons/s	(α,n) neutrons/s	Total neutrons/s
JEF-2.2	4.458E+08	8.005E+06	4.538E+08
JEFF-3.1	4.576E+08	8.012E+06	4.656E+08
JEFF-3.1.1	4.576E+08	8.012E+06	4.656E+08

The results from JEFF-3.1 and JEFF-3.1.1 are identical to the 4 significant figures given in the above table. The total neutron emission increases by 2.6% from JEF-2.2, with almost all of this increase coming from changes in spontaneous fission with only a small change in the (α,n) neutrons.

It should be noted that in this case almost all of the extra neutron emission arises from the spontaneous fission neutrons from Cm244. It should be noted that Cm242 ($T_{1/2}$ = 163 days) and Cm244 ($T_{1/2}$ = 18.1 years) dominate neutron emission calculations for most commercial fuels and thus it would be expected that most neutron emission assessments, after the Cm242 has decayed, would show similar trends.

6 Conclusions

The new JEFF-3.1 and JEFF-3.1.1 data, which was released for testing, has been incorporated into FISPIN to calculate spontaneous fission and (α,n) neutron emission. The revised data still gives good agreement with the limited neutron emission measurements available to validate these calculations. However, the result of using the new data suggests that for PWR calculations the estimated neutron emission will increase by ~2-3% from currently calculated values.

It was noted that between JEF-2.2 and JEFF-3.1.1 only the spontaneous fission between U232, U235, U236, Pu239 and Am241 and the mean alpha energy of Bi212m change by greater than 10%. Comparisons with ENDSF by N.J. Thompson and A.B. Garnsworth, students of the University of Surrey, suggested that the values in JEFF-3.1.1 agree reasonably with ENDSF except for the spontaneous fission branching ratios of U232 and Pu236. It is recommended that these differences are investigated further by an evaluator. However, it is noted that these nuclides are not usually important in calculating neutron emission from spent nuclear fuels.

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8 References

1. A.J. Mill, "The Contribution from (a,n) and (g,n) Reactions to the Total Neutron Emission in Irradiated Magnox Fuel Elements" CEGB Report RD/B/N4104 (1978).

2. FISPIN is developed by Nexia Solutions Ltd., and distributed by Serco Assurance. The code is described by R.F. Burstall in "FISPIN-A computer code for nuclide inventory calculations", ND-R-328I (1979).

3. OECD/NEA, "The JEFF-3.1 Nuclear Data Library", JEFF Report 21, ISBN 92-64-02314-3. Available from the OECD Nuclear Energy Agency, Le Seine Saint-Germain, 12 Boulevard des iles, F-92130, Issy-les-Moulineaux, France (2006). The JEFF-3.1 file was released in 2005.

4. FISGUI is a graphical user interface to FISPIN that allows many types of FISPIN calculations to be generated simply. It was developed by Nexia Solutions and Serco Assurance and is distributed by Serco Assurance.

5. B. Alldred, P. Little and Miss G.A. Quayle, "RUNFISP 6- A Macro for Use with FISPIN-6 on the Risley 3982" BNFL Research Memorandum 1110. Also published at the UK DIDSG committee as paper DIDSG(90)P415.

6. D. West, "The Calculation of Neutron Yields in Mixtures and Compounds from the Thick Target (α,n) Yields in the Separate Constituents", Annals of Nuclear Energy, Vol. 6, pp. 549-552 (1979). Also published as UKAEA report AERE-R 9372.

7. V. McLane, "ENDF-102: Data Formats and Procedures for the Evaluated Nuclear Data File ENDF-6, BNL-NCS-44945-01/04-Rev" (2001).

8. OECD/NEA, "The JEF-2.2 Nuclear Data Library", JEFF Report 17, ISBN 92-64-17686-1. Available from the OECD Nuclear Energy Agency, Le Seine Saint-Germain, 12 Boulevard des iles, F-92130, Issy-les-Moulineaux, France (2000). The JEF-2.2 file was released in 1993.

9. E.W. Lees, "Comparison of the measured and calculated neutron output from Zebra fuels". UK committee paper DIDWG(81)P249 (1981).

10. E.W. Lees and D. West, "Comparison of the measured and calculated neutron output from a mixed oxide fuel pin". UK committee paper DIDWG(81)P243 (1981).

11. N.T. Gulliford, "Measurement of neutron source strengths in irradiated CAGR fuel samples in DIMPLE", UKAEA report AEEW-R2532 (1989).

12. R.W. Mills, "Validation of JEFF-3.1 for decay heat and inventory calculations" (insert full reference).

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