Part II

JEF-2.2 VALIDATION STUDIES

Introduction to Part II

Part II presents a summary of the results of analyses of integral experiments. This is followed in Part III by the conclusions drawn from a cross-section adjustment study carried out by E. Fort, *et al*. at CEA Cadarache.

Many of the thermal and fast reactor integral measurements which have been calculated have not been published in the open literature, being proprietary. The openly published compilations which have been taken into account include the CSEWG Benchmark Book [1], the compilation of criticality benchmarks and the NEA Data Bank compilation of shielding benchmarks. A problem noted in Chapter 5 is the inconsistency between benchmark specifications given in different sources. Different models in the same source (pin cells with a buckling versus reactor models) can also be inconsistent, the problem in this case perhaps being associated with the specification of the cell boundary condition in the calculational model.

A review of some of these benchmark specifications would be valuable and an extension of the openly published compilations to include the benchmarks analysed here, and also other benchmarks which have not been available to the participants in the JEF-2.2 validation studies. There is a need to compile reactor benchmark information from the laboratories which are no longer carrying out such measurements, before the information is lost.

REFERENCES

[1] CSEWG Benchmark Book, BNL 19302, ENDF 202, Revised 11-81.

Chapter 5

THERMAL REACTOR VALIDATION STUDIES

Introduction

In this chapter, we are mainly concerned with JEF-2.2 calculations made to compare with reactor lattice experiments, as distinct from the criticality related uranium and plutonium solutions which are reviewed in a separate section.

We must note that the method of calculation can influence the results. Most confidence is placed in whole reactor Monte Carlo calculations, but even these are not always in agreement. Different Monte Carlo codes (i.e. MCNP, MONK, TRIPOLI-4) can give results which differ by perhaps 100 pcm. In the deterministic methods cell cylindricalisation can introduce a significant error, and one does not have the same confidence in a calculation for a lattice cell (with the measured buckling) as for a whole reactor calculation. Errors of several hundred pcm can arise from these approximations, as is illustrated by calculations made for TRX-1 and also for the pin cell benchmark, for which intercomparison calculations have been made to explore the effects of methods approximations. The TRX-1 results (Annex 2) have also shown that there are significant differences between different models used to specify the experiments, pin cell versus whole reactor, and differing whole reactor specifications. In particular, the large difference in k_{eff} between the Winfrith and Petten Monte Carlo calculations has been ascribed to differences between the specifications given in two different references. Another effect which can give rise to differences is the treatment of solid state effects in the calculation of resonance Doppler broadening effects. The APOLLO calculations are usually made using an effective temperature (rather than the true temperature) to allow for the effects. The effects are characterised by a Debye temperature and the value used (in particular for ^{238}U in UO_2) has changed during the period of the validation studies.

Over the course of the studies made since JEF-2.2 was produced, improvements have been made to several of the code systems and differences between the earlier and later versions can be quite significant. Many of the benchmarking studies have been made with earlier versions of the code systems and also with simplified models.

We consider uranium fuelled lattices and mixed uranium-plutonium lattices. There has been no general study of ^{233}U-thorium fuelled systems, although a limited assessment of the data for ^{233}U and ^{232}Th has been made.

We note that the cross-sections for hydrogen and oxygen are essentially the same in ENDF/B-VI as in JEF-2.2 and the data for ^{235}U (earlier ENDF/B-VI versions) and ^{238}U are very similar. However, there are some differences. For ^{235}U the thermal region data and ν are different, and for ^{238}U the inelastic scattering (above about 200 keV) is different. The evaluation for ^{239}Pu is independent of ENDF/B-VI. There are also similarities with the ^{235}U and ^{238}U data in JENDL-3.2, but the oxygen, zirconium and ^{239}Pu data are different. Nevertheless, some general indications can be drawn from calculations made using these data libraries, combined with intercomparisons made for a few benchmarks.

The properties considered in this summary review are:

- k-effective.
- Variation of reactivity (or water height at critical) with temperature.
- Fission and capture reaction rates measured in the lattices.
- Relative capture reaction rates for actinide isotopes deduced from irradiation experiments.
- Fission product reactivity worth measurements.
- Coolant voiding reactivity changes.
- Reactivity worths of absorbers, and the associated fission rate perturbation effects.
- Results of cross-section adjustment studies in the thermal and resonance range.

k_{eff} and reaction rate ratio measurements for uranium fuelled H_2O and D_2O moderated systems

DIMPLE S01A

The thermal reactor lattice DIMPLE S01A has been chosen as a benchmark for intercomparison calculations relevant to uranium oxide fuelled PWR lattices. Results obtained by different contributors are as seen in Table 5.1.

Table 5.1. DIMPLE S01A

Slowing down density at 4 eV = 0.49, values of {(C/E) - 1} %

	MCNP-4A/ECN ***Whole core***	**APOLLO-2** ***Pin cell***	**MONK-8 (point)** ***Whole core***	**MONK-8 (group)** ***Whole core***	**WIMS** ***Pin cell***
k_{eff} (pcm)	0 ± 20 pcm	-32 pcm	-93 ± 60 pcm	-57 ± 60 pcm	47 pcm
F8/F5*	-9.96 ± 3.5				
F9/F5	1.0 ± 1.2				
C8/F5	0.0 ± 0.9				

* Note: The value of F8/F5 in DIMPLE S01A has always been considered suspect.

The calculations reported here are the MCNP-4A Monte Carlo calculations made at ECN Petten [1,2], the MONK Monte Carlo and WIMS multi-group calculations made at Winfrith [3] and the APOLLO-2 calculations made at CEA Cadarache and Saclay [4,5].

The only value which is significantly outside the 1 s.d. range is for the suspect measured value of F8/F5.

These comparisons illustrate the range of values to be expected from the approximations in these methods. We can also conclude that there is no evidence from this particular experiment that the data in JEF-2.2 should be modified for UOX fuelled PWR k_{eff} calculations.

CEA Cadarache studies for a set of lattices with spectra ranging from intermediate to thermal

Alet, *et al*. [6] have made calculations using APOLLO-2 version 3 for a series of LWR lattices having measured bucklings. The lattices Cristo 1, 2 and 3, and Caméléon were measured in the EOLE facility at Cadarache. These use PWR type UOX fuel pins and have ^{235}U enrichments in the range 3-3.5% and water to UO_2 ratios in the range 0.45-5.5. In addition calculations have been made for the intermediate spectrum system ZPR HiC6 which emphasises the importance of the ^{235}U resonance range.

	P	R	q	B^2	s.d.	(C-E) pcm
Cristo 3	0.96	0.45	0.37	1.95	500	1 307
ZPRHIC6	1.24	0.96	0.51	4.747	400	-87
UH1.2	1.26	1.27	0.51	6.05	300	309
Caméléon	1.26	1.80	0.57	5.085	300	818
Cristo2	1.58	3.56	0.76	3.575	300	-291
Cristo1	1.86	5.46	0.89	-0.09	300	231

P: Denotes the lattice pitch.
R: The water to UO_2 volume ratio.
B^2: The critical buckling ($10^{-3}/cm^2$).
q: The slowing down density.
s.d.: The experimental uncertainty (pcm).

As is described in Chapter 15 in Part IV, using the new ORNL ^{235}U resonance region data, with increased capture, incorporated in ENDF/B-VI revision 5, results in an improved agreement for most cases and in particular for the harder spectrum systems.

Studies made at Winfrith relating to the ^{235}U resonance region

C. Dean, D. Hanlon and R. Perry studied a set of 10 benchmarks covering a range of intermediate and thermal spectrum systems [7]. They included two TOPSY reflected uranium hydride systems, UH3Ni and UH3UR, which have hard spectra, the Winfrith Hector Intermediate Spectrum uranium fuelled System, HISS, two uranium flouride systems (hardest spectrum, UF1 and softest spectrum, UF6), a UO_2 lattice studied at CEA Valduc, DIMPLE SO1, TRX-1 and the hardest and softest spectrum ORNL spheres, ORNL1 and ORNL-10. The calculations were made using the MONK-8 hyperfine Monte Carlo code and the statistical accuracy is typically ±80 pcm.

System	q	MONK-8/JEF-2.2 (C-E) in pcm
UH3-UR	0.031	1 643
UH3-Ni	0.058	2 577
HISS(HUG)	0.141	3 103
UF1	0.391	700
Valduc UOX lattice	0.488	27
DIMPLE SO1	0.503	-93
TRX-1	0.628	-580
UF6	0.792	20
ORNL-1	0.843	-370
ORNL-10	0.932	-217

One notes the large discrepancies for the harder spectrum systems and the more acceptable results for the lattices with q values larger than 0.5, typical of LWRs and thermal systems. It is shown in Chapter 15 (Part IV) that there is a much improved agreement for the four hardest spectrum systems resulting from the use of the ENDF/B-VI revision 5 resonance region data for ^{235}U, with its increased resonance capture. The agreement is worsened, however, for DIMPLE SO1.

Studies made by Cathalau and Blaise

S. Cathalau and P. Blaise (CEA Cadarache) made k_{eff} calculations using APOLLO-2 version 2 for 55 thermal and intermediate spectrum systems using pin cell models [4]. They include the Budapest VVER lattice experiments. Measurements were made for several of these VVER lattices at three different temperatures, 20°C, 80°C, and 130°C. The largest discrepancies between calculation and experiment are for the measurements at the highest temperature (see Table 5.2).

Table 5.2. Values of C-E in pcm for the Budapest temperature coefficient experiments

Temperature	V1273658	V1273650	V1273600	V1103600
q value	0.57	0.55	0.52	0.44
20°C	504	185	980	-491
80°C	348	93	-207	-1 637
130°C	-330	-160	-334	-1 931
Range	**-834**	**-345**	**-1 314**	**-1 440**

There is clearly a problem associated with the interpretation of these temperature dependent experiments.

The above results for V1103600 at 80°C and 130°C are the most discrepant C-E values. If we exclude the experiments at temperatures above ambient from the considerations the range of C-E values reduces to (1 452 to -1 174 pcm) for the remaining 45 lattice systems. Removing all of the Budapest VVER lattice experiments reduces the overall variance, although the range remains from (939 to -1 174 pcm) for the remaining 38 lattice experiments. Part of the variance is due to experimental uncertainty, and a part is also due to modelling approximations in the method used.

Analysis of BWR-simulated critical experiments

An analysis was made by Jung-Do Kim and Choong-Sup Gil of two series of BWR simulation experiments, with and without absorber arrays [8]. In the first series the absorbing material is in the form of a curtain of borated stainless steel and in the second selected fuel elements contain Gd_2O_3 as a burnable poison. The cores have been modelled in detail in MCNP calculations. Fission density measurements are available for selected fuel rods. The measurements for the burnable absorber criticals were made at three temperatures, about 20°C, 90°C and 243°C. The standard deviation of the Monte Carlo calculations is ±70 pcm. The range of C-E values for the reactivity is from -230 to 446 pcm. For the five measurements at 20°C the range is -186 to 371 pcm.

For the assemblies measured at three temperatures the average value of C-E for the reactivity is -6 at 20°C, -79 at 90°C and +211 at 243°C. (Note: there are three cases at 20°C and 90°C and four cases at 243°C.) These differences in C-E values are probably not significant.

For the fission rate distributions the agreement is within ±1.0% for the cases without absorber and ±1.7% for the cases with absorber, well within the uncertainty of measurement (about ±2%) and the Monte Carlo statistical uncertainties (<±1.3%).

Analysis of the TCA water moderated lattices using JENDL-3.2, ENDF/B-VI and JEF-2.2

H. Takano has reported calculations of k_{eff} for the Japanese TCA uranium fuelled and plutonium fuelled water moderated lattices [9]. The JEF-2.2 results are intermediate between those obtained using JENDL-3.2 and ENDF/B-VI (version 2 and version 5 ^{235}U data). For the four uranium fuelled lattices the JEF-2.2 values of (C-E) range from -10 to +190 pcm and for the four plutonium fuelled lattices they are in the range -480 to -230 pcm.

Stuttgart calculations for heterogeneous uranium and plutonium fuelled lattices

W. Bernnat, *et al*. [10] have presented the results of calculations of k_{eff} using JENDL-3.2, ENDF/B-V and VI, and JEF-2.2 for a number of benchmarks documented in the International Handbook of Evaluated Criticality Safety Benchmark Experiments [11]. The calculations were made using MCNP. For the heterogeneous, highly enriched uranium metal fuelled lattices the JEF-2.2 values of (C-E) are in the range -600 pcm to +500 pcm. For the low enriched uranium fuelled LWR lattices the values of (C-E) are in the range -1 000 pcm to -300 pcm The JENDL-3.2 values are 400 to 700 pcm higher. For mixed uranium-plutonium oxide fuel the range is from -1 000 to +500 pcm. For these uranium/plutonium fuelled cores JENDL-3.2 is on average 400 pcm higher and ENDF/B-VI about 100 pcm lower. Results are also given for three ANS uranium fuelled LWR benchmarks. For these the JEF-2.2 k_{eff} (C-E) values are in the range -400 to -200 pcm.

Analyses of the KRITZ temperature coefficient measurements

The results of the analyses made by K. Ekberg using the CASMO system (Studsvik), C. Mounier using APOLLO-2 (CEA Saclay) and using WIMS at Winfrith, are summarised in Annex 3. In contrast to the large discrepancies seen in the above analyses of the Budapest VVER lattice experiments, which show discrepancies of the order of -10 pcm/°C, the agreement is broadly consistent with the experimental uncertainties, the largest discrepancy being 1.2 ± 0.4 pcm/°C for the Winfrith analysis of Core 2.19. The analyses for the Cores 2.1 and 2.13 show no significant discrepancies, although there are some differences between the different methods. (This is in contrast to the discrepancies which were being found about 15 years ago using earlier data sets and calculation methods.) A possible explanation for the differences between the different methods could be the treatment of solid state effects. This effect is treated in the Saclay calculations but not in the Winfrith calculations.

We can conclude that for the experiments for which whole core calculations have been made (albeit with axial bucklings) there are no temperature coefficient discrepancies. The Budapest VVER experiments require more detailed study.

Mixed uranium plutonium oxide lattices

Eleven of the series of thirteen lattice experiments in the ESADA programme [12] have been analysed by C. Mounier (Cores 2 and 5 being omitted) [13]. Cores 1 to 7 have an increasing water to fuel volume ratio, V_m/V_fincreasing from 1.125 to 8.257. Cores 8 and 9 have the same V_m/V_f ratio as

1 and 3 but with borated water and Cores 10 and 11 have double the boron content. The ^{240}Pu content of the fuel is 8% in Cores 1 to 11 and 24% in Cores 12 and 13, which have the same V_m/V_f ratio as Cores 4 and 6.

The analysis made using the most refined method in APOLLO-2, a 172 group calculation, treating the double heterogeneity and allowing for streaming effects, results in a mean (C-E) value for the reactivity of -412 pcm with a standard deviation of 466 pcm. The range of results is from -1 287 to 205 pcm. The variations in C-E with V_m/V_f, boron content and ^{240}Pu content are not systematic and it is concluded that there is no evidence from these experiments for a need to revise the ^{239}Pu or ^{240}Pu data.

Summary of k_{eff} calculations using JEF-2.2

We can note that for UOX and MOX fuelled PWR type lattices and for well thermalised systems, the k_{eff} values calculated using JEF-2.2 are within about ±1 500 pcm of the measured values (at room temperature).

Results of the analyses of irradiation experiments

Analyses of the compositions of irradiated UOX and MOX fuel pins

Analyses of the isotopic compositions of irradiated UOX fuel pins from the GRAVELINES and BUGEY PWRs have been compared with APOLLO-2 calculations at CEA Cadarache by R. Dorel, C. Chabert and A. Santamarina [14]. The burn-up of the BUGEY fuel extends to 38.3 GWd/t and that of the GRAVELINES fuel to 59.7 GWd/t. The analyses are subject to a number of uncertainties which result in a variation of several percent for some of the values of (C-E)/E. In particular there is a larger range of discrepancies for the BUGEY values than for the GRAVELINES values. However, some trends are consistent between different samples and reactors. It is only these consistent patterns which are summarised here.

Isotopic ratio	Mean (C-E)/E	Ranges in GRAVELINES
^{236}U/^{238}U	-4%	-3.5 to -4.8
^{238}Pu/^{238}U	-7.5%	-6.9 to -8.7
^{240}Pu/^{238}U	-1.5%	-1.1 to -2.3
^{241}Pu/^{238}U	-3.0%	-0.8 to -4.8
^{242}Pu/^{238}U	-7.5%	-6.4 to -8.7

For ^{238}Pu there is a trend to underestimate the production, but the variation is large.

An analysis of PWR fuel irradiations (to 34.2 GWd/t) made using the SRAC system, has been reported by Takano, Akie and Kaneko [15] and by H. Takano [9]. The measurements are relative to ^{238}U (ranges and uncertainties are not quoted). Results are given for JENDL-3.2, ENDF/B-VI and JEF-2.2.

The large underestimation of the ^{232}U build-up is associated with the underestimation of the ^{236}Pu build-up, the JENDL-3.2 and ENDF/B-VI values being in better agreement with the measurements. There are similar discrepancies for ^{237}Np, ^{238}Pu, ^{240}Pu, ^{242}Pu and ^{242m}Am in JENDL-3.2 and ENDF/B-VI but somewhat better agreement for ^{243}Cm and ^{244}Cm in ENDF/B-VI.

Isotope	JENDL-3.2 (C-E)/E %	ENDF/B-VI (C-E)/E %	JEF-2.2 (C-E)/E %
^{232}U	-17%	-10%	-36%
^{235}U	-2%	-2%	-1%
^{236}U	-2%	+1%	-2%
^{237}Np	-8%	-7%	-7%
^{236}Pu	-5%	+5%	-28%
^{238}Pu	-17%	-13%	-14%
^{239}Pu	-1%	0	0
^{240}Pu	-7%	-7%	-7%
^{241}Pu	-2%	-2%	-4%
^{242}Pu	-9%	-11%	-10%
^{241}Am	-4%	-4%	-6%
^{242m}Am	-30%	-32%	-35%
^{243}Am	-2%	+8%	-4%
^{242}Cm	-6%	-4%	-6%
^{243}Cm	+4%	+5%	+16%
^{244}Cm	-23%	-15%	-25%

Some deviations will vary depending on the burn-up, the rating and on the initial fuel enrichment and isotopic composition. We can note that some of the trends are in the same sense as those found in the CEA Cadarache studies. That is, the (C-E)/E values are negative for ^{236}U and the plutonium isotopes.

Analyses of irradiated MOX fuel assemblies have been made in France and Belgium. Calculations have been made at CEA Cadarache by C. Chabert, A. Santamarina and R. Jacqmin, using APOLLO-2, to compare with measurements of the compositions of MOX fuel irradiated in SLB-1, a 900 MWe PWR. The MOX fuel is at three plutonium enrichments, 5.636 mass % (central region), 4.420% (intermediate region) and 2.913% (edge fuel). The Pu vectors are slightly different in the three cases, being:

	(^{238,239,240,241,242}Pu, ^{241}Am)
Centre	(0.87, 66.71, 20.61, 7.66, 2.89, 1.27 atoms %)
Intermediate	(0.83, 66.78, 20.53, 7.66, 2.83, 1.37 atoms %)
Edge	(0.85, 66.69, 20.69, 7.57, 2.91, 1.29 atoms %)

The uranium is depleted (^{234,235,236,238}U) (0.002, 0.229, 0.004, 99.765). Three irradiations were studied, one cycle, two cycles and three cycles, and the burn-up was determined from the ^{145}Nd build-up. A maximum burn-up of 45 GWd/t is achieved in the central fuel rods after three cycles. The compositions are given relative to the ^{238}U content.

A databank has been established at SCK•CEN Mol containing the results of isotopic analyses of irradiated samples with burn-up levels up to 90 GWd/t. Calculations have been made at Mol by Winckel, Aoust and de Raedt, and at BN by Maldague, Brusselaers and Pilate and also by Kuijper, *et al.* at Petten, using WIMS-7, of measurements on MOX fuelled samples irradiated in the BR3 reactor and analyses of fuel from the BEZNAU-1 reactor. The results have been summarised by Pilate, *et al*., Belgonucléaire [16].

The C/E values obtained from the MOX fuel analyses are summarised in Table 5.3.

Table 5.3. The C/E values obtained from the MOX fuel analyses

	SLB1	BEZNAU-1 (ARIANE)	BR3
^{235}U	1.01-1.05	0.98-1.06	0.99
^{236}U	0.90-0.94	0.92-0.95	0.83
^{238}Pu	0.91-0.94	0.94-0.97	(0.37)
^{239}Pu	1.01-1.05	1.02-1.06	0.99
^{240}Pu	0.99-1.02	0.98-1.00	1.04
^{241}Pu	0.97-0.99	1.00-1.02	1.05
^{242}Pu	0.94-0.98	1.02-1.03	1.02
^{241}Am	0.98-1.01	1.12-1.15	(1.00-1.10)
^{242m}Am	0.6-0.8	0.6-0.7	
^{243}Am	0.93	1.00-1.01	
^{243}Cm	0.7-0.86	0.8-0.89	
^{244}Cm	0.92-0.94	0.93-0.95	0.84-0.89
^{245}Cm	0.87-0.94	0.88-0.94	

There are some wide dispersions of the C/E ratios as a consequence of differences in initial isotopic compositions, irradiation spectra and burn-up levels. The only satisfactory way to draw conclusions would be to carry out a cross-section adjustment analysis. Nevertheless, some trends are clear. In particular we note significant discrepancies for the following isotopes in MOX fuel:

	Typical values of (C-E)/E
^{236}U	-8%
^{238}Pu	-6%
^{242m}Am	-30%
^{243}Cm	-20%
^{244}Cm	-7%
^{245}Cm	-10%

We also note a larger underestimation of ^{236}U production in MOX fuel than that found for UOX fuel. This is attributed to the larger contribution of the ^{235}U resonance region capture in the case of the MOX fuel (about 70% of the capture in ^{235}U is in the resonance range in the case of MOX fuel) and can be related to a 10% underestimation of the ^{235}U resonance capture.

Cross-section modifications required to improve the prediction of the compositions of irradiated UOX fuel

C. Mounier has examined the cross-section changes required to improve the calculation of compositions of irradiated fuel, the changes considered being to the capture and fission cross-sections in the thermal and epithermal ranges. The analysis was made using the irradiated fuel measurements in the GRAVELINES reactor. The calculations were made using APOLLO-2 and selected results are summarised in Table 5.4. It should be pointed out that the changes to the fission cross-sections include ^{242m}Am at thermal energies (5% reduction) and ^{235}U at epithermal energies (1.6% reduction). For capture

cross-sections one can note the following tendencies, although there are possible approximations in the methods which could be affecting the results.

Table 5.4. Percentage increases indicated to be applied to selected capture cross-sections

Isotope	Thermal range	Epithermal range
^{235}U	+0.4	+8.5
^{236}U		-4
^{238}U		-0.9
^{239}Pu	+0.4	+1.3
^{240}Pu		+1.9
^{241}Pu	+0.6	+6.8
^{241}Am	+0.4	+6.9
^{242m}Am	-62	-4.6
^{242}Cm	+17	+11.4

Analyses of the ICARE/S and SHERWOOD sample irradiation experiments

S. Cathalau and A. Benslimane-Bouland have published an analysis of the irradiation experiments performed in the SHERWOOD (PWR spectrum) and ICARE/S (high conversion LWR spectrum, emphasising the resonance region) experiments performed in the MELUSINE reactor at Grenoble [17]. Only the ICARE/S measurements have been analysed using JEF-2.2. These give capture cross-sections for ^{238}U, ^{239,240,241}Pu and ^{241}Am. The results indicate some significant discrepancies, some of which appear to be unacceptably large and probably imply difficulties of analysis. The interpretation of the ^{241}Am capture measurement is complicated by many factors, such as the uncertainty in the branching ratio, but nevertheless it is considered that the capture cross-section is underestimated by about 20%.

Analyses were also reported for calculations made using the CEA-86 library. This contains JEF-1 data, but the cross-sections are the same in JEF-2.2 for ^{242}Pu, ^{241,243}Am and ^{244}Cm, although the calculated neutron spectrum will be different. The same discrepancy is found for ^{241}Am for the measurements in both SHERWOOD and ICARE/S. There is a similar discrepancy for ^{243}Am.

Cross-section adjustment studies based on analyses of lattice experiments

Studies have been published by Cathalau, *et al*. [18] and by Blaise and Fort [19]. A feature of the latter analysis is that the adjustments are applied to the resonance parameters, rather than to cross-sections, thus treating resonance shielding effects. However, the integral measurements have been analysed using simple cell models and this could be influencing the results obtained.

A summary of cross-section changes indicated in the different studies

The studies carried out at different times and using different types of integral measurement (lattice criticality, spectral index measurements, irradiations of isotopic samples and the isotopic compositions of standard fuel pins after irradiation) have given different indications concerning the changes required to cross-sections. The differences can arise because of differences in the spectra in which the measurements were made, between the assumed uncertainties in the differential cross-sections (relative to those of the integral measurements) and differences in the sensitivities of the different integral measurements to particular items of nuclear data. Nevertheless, some trends are consistent

between the different analyses, such as the need to increase ^{235}U resonance capture by about 10%. There is also strong evidence that the capture cross-section of ^{241}Am is substantially underestimated in JEF-2.2.

Tendency research methods

These methods, which were developed by Reuss and Tellier, have been used at CEA Saclay to obtain best estimates for thermal and resonance region data (i.e. [20,21]. In Ref. [21] H. Tellier studies the thermal parameters of ^{233}U and concludes that the data in JEF-2, ENDF/B-VI and JENDL-3 are all in need of improvement. However, there is also a need for more integral measurements relating to ^{233}U.

Fission product nuclear data studies

The studies are of two types:

- Analyses of irradiated fuel samples.
- Reactivity worth measurements for samples of irradiated fuel, samples of elements in the fission product range and samples of important fission product isotopes.

There has been a joint France/UK/USA programme of measurements of the reactivity worths of samples of fission product isotopes and irradiated fuel samples, together with analyses of the compositions of the fuel samples, the CERES programme.

Work in this area has been summarised by A. Santamarina, N. Thiollay and C. Chabert [22], and also by N. Gulliford and D. Hanlon [23].

Conclusions drawn from the analyses of irradiated fuel

Conclusions relating to yield data

APOLLO-2 calculations have been made to compare with measurements of the isotopic compositions of UO_2 fuel irradiated in BUGEY3 and GRAVELINES. Conclusions relating to the yield data are drawn from the isotopic ratio measurements relative to ^{238}U of Nd and Cs isotopes and are given in JEF/DOC-784 [22]. For the Nd isotopes the accuracy of measurement is better than ±1% and there are (C-E)/E discrepancies of -3.1% for $^{144}Nd/^{238}U$ and +2.0% for $^{148}Nd/^{238}U$. For the Cs isotopes the discrepancies are larger, up to -10% for $^{135}Cs/^{238}U$, calculation underestimating the ratio by 3% or more for all the Cs isotopes.

It is concluded from the result for ^{133}Cs that the Meek and Rider yield values for ^{133}Xe in ^{235}U and ^{239}Pu fission are more satisfactory than the JEF-2.2 values. The same conclusion is drawn for ^{148}Nd.

Capture cross-sections

By examining the variation of isotopic ratios with burn-up conclusions can be drawn concerning the capture cross-sections. It is concluded that whereas the capture cross-sections of ^{145}Nd and ^{134}Cs are satisfactory that of ^{143}Nd is underestimated by 4% ± 1%.

Reactivity worth measurements

Small sample reactivity perturbation measurements have being made in several different spectra and using different measurement techniques. The samples consist of natural UO_2 doped with the fission product isotope. The APOLLO-2 analysis of the measurements made in MINERVE has been described by A. Santamarina, N. Thiollay and C. Chabert [22]. A summary of the WIMS-7 analysis of the measurements made in the DIMPLE reactor has been presented in JEF/DOC-648 by N. Gulliford and D Hanlon [24]. The (C-E)/E values are presented in Table 5.5.

Table 5.5. Summary of CERES fission product reactivity worth analysis

Values of (C-E)/E in %

Isotope	Winifrith results (DIMPLE)		Cadarache results (MINERVE)	
	Dimple II	Dimple III	R1-UO2	R2-UO2
^{95}Mo	+6	+11	-3.1 ± 3.4	-3.7 ± 3.8
^{99}Tc	+7	-3	+4.1 ± 3.8	+3.4 ± 3.5
^{103}Rh	+11	+11	+11.0 ± 4.0 –	+8.0 ± 4.2 +14 ± 9.0
^{109}Ag	+2	+3	-3.6 ± 4.3 -4.6 ± 9.0	-4.5 ± 4.3 +2.8 ± 6.9
^{133}Cs	+11 +12	+9 +7	+8.5 ± 3.2 +7.6 ± 3.5 -11.0 ± 7.3 -0.6 ± 3.8 +4.1 ± 8.5	+7.6 ± 3.8 +9.3 ± 3.8 -0.4 ± 5.7 -2.4 ± 4.3 +9.1 ± 7.3
^{143}Nd	-6	-4	-7.1 ± 3.1	-8.5 ± 3.8
^{145}Nd	-1	0	+0.4 ± 3.8	+1.1 ± 4.4
^{147}Sm	+1	+3	+1.3 ± 4.3	+2.7 ± 4.7
^{149}Sm	-5	-2	-6.0 ± 2.9	-4.9 ± 3.6
^{152}Sm	+1	-2	-1.6 ± 2.9	-1.8 ± 3.7
^{153}Eu	-9	-10	-4.2 ± 4.0	-1.3 ± 4.6
^{155}Gd	+4	+2	-2.5 ± 2.9	-6.1 ± 4.0

The uncertainty in the DIMPLE measured values is quoted as ±2%, with which must be combined uncertainties in compositions and corrections for contaminants bringing the uncertainties close to the MINERVE values. All the (C-E)/E values are within ±12%. The ^{133}Cs might be affected by resonance overlap effects. There are significant discrepancies for ^{103}Rh (11% high), ^{133}Cs (9% high) and ^{143}Nd (7% low).

Local coolant voiding in MOX fuelled LWRs

Experiments have been performed in the EPICURE programme, carried out in the EOLE facility at CEA Cadarache, of the reactivity effects of 2-D and 3-D local coolant voids. They have been analysed using the APOLLO-2 code and the CEA93 library, which is based on JEF-2.2. The analysis is described in a paper to the PHYSOR-96 conference, by Cathalau, *et al*. [25]. The main conclusions reached relate to the methods approximations, rather than the nuclear data. It is necessary to treat P_1

scattering and to use S_8 to get satisfactory agreement in general, although diffusion theory is adequate for 2-D voids. Overall the agreement is satisfactory.

Fission rate distributions across the boundary between a MOX fuelled assembly and a UOX fuelled assembly

Measurements have been made in the EPICURE programme in EOLE. A problem in the interpretation relates to the derivation of the fission rate from the foil activation. The decay of selected fission products is monitored to determine the fission rates and there are significant differences between the yields found in the JEF-2 fission yield library and the B-VI library for the mass chains 92 and 140, in particular for the ratios of yields in ^{239}Pu to ^{235}U. These differences should be investigated. However, methods approximations can also have a significant effect in the calculation of these fission rate distributions.

On the basis of the assumptions made about the yields satisfactory agreement is found for the fission rate distributions.

Analyses of measurements for absorbers in PWRs

A study was carried out in the EPICURE programme in the EOLE facility of the reactivity effects of absorber rods and clusters in UOX and MOX fuelled cores. The analysis was made using APOLLO-2. The single rods consisted of B_4C, Ag-In-Cd and Pyrex. The absorber clusters studied are formed from:

- 24 natural B_4C rods.
- 24 Ag-In-Cd rods.
- 12 Ag-In-Cd rods and 12 stainless steel rods (the grey cluster).

Fission rate distributions were also studied.

The results are described in Ref. [26]. The reactivity effects are all calculated within about 1 s.d. except for the grey cluster for which the C/E discrepancy is 5.9% ± 2.5% (2 s.d.). For the fission rate distributions most points are calculated within the 2 s.d. limits of ±5%, the average error being typically less than ±1.5%.

Reactivity worth measurements for hafnium

An analysis of measurements carried out at Cadarache was published by J.-M. Palau [27]. Based on an analysis using TRIPOLI-4 it is concluded that the JEF-2.2 data for hafnium overestimate the absorption. An adjustment to the data is proposed based on a sensitivity analysis and an assessment of uncertainties. The proposed adjustment is mainly applied to the resonance region cross-sections of ^{177}Hf. The adjustment is interpreted in terms of the modifications to be applied to resonance parameters. The main changes proposed are reductions in the values of Γ_n for the resonances of ^{177}Hf at 6.6 eV (-5.0%) and at 22.26 eV (-2.6%), the reductions in the other resonances being about 1%.

Intercomparisons of calculations made using different nuclear data libraries

JEF/DOC-699 by S. Pelloni, intercompares calculations made for a uranium free fuel consisting of Er absorber and Pu in a Zr matrix [28]. The range of values of k_∞ is about 1% and the differences between the values calculated for void coefficients in some cases exceeds 50%.

REFERENCES

[1] A. Hogenbirk, "The TRX-1 and TRX-2 Benchmarks – Validation of the JEFF-3.0 Evaluation for ^{235}U", JEF/DOC-541, January 1996.

[2] T.T.J.M. Peeters, A. Hogenbirk, "The DIMPLE S01A Critical Assembly – Results of MCNP-4A Calculations", JEF/DOC-548, January 1996.

[3] R.J. Perry, C.J. Dean, "DIMPLE S01A Models", JEF/DOC-518, March 1995.

[4] S. Cathalau, P. Blaise, "Synthesis of the Comparison of JEF-2.2 (ENDF/B-VI) and LEAL-DERRIEN/95 Evalautions for ^{235}U", JEF/DOC-589.

[5] Saclay document

[6] M.C. Alet, A. Benslimane, C. Chabert, C. Mounier, A. Santamarina, G. Willermoz, "Qualification of the ^{235}U Leal-Derrien-Larson Evaluation Using French Integral Experiments", JEF/DOC-707, December 1997.

[7] C. Dean, D. Hanlon and R. Perry, Reactor Physics Conference, Long Island, October 1998, p. 1413.

[8] Jung-Do Kim, Choong-Sup Gil, "Testing of the JEF-2.2 and EDF/B-VI.2 Through MCNP Analysis of BWR-Simulated Critical Experiments", JEF/DOC-543, January 1996.

[9] H. Takano, "", proceedings of the International Conference on Nuclear Criticality Safety (ICNC'99), Versailles (September 1999).

[10] W. Bernnat, S. Langenbuch, M. Mattes and W. Zwermann, "Influence of Cross-Section Data Bases on Criticality Safety Calculations with the MCNP Monte Carlo Code", proceedings of the International Conference on Nuclear Criticality Safety (ICNC'99), Versailles (September 1999).

[11] International Handbook of Evaluated Criticality Safety Benchmark Experiments, NEA/NSC/DOC(95)03, September 1998.

[12] ESADA, WCAP-3726-1, 1967.

[13] C. Mounier, “Contribution to the Benchmark of JEF-2: Results for ESADA Lattices”, JEF/DOC-542.

[14] R. Dorel, C. Chabert and A. Santamarina, “Qualification du Schéme de Calcul de Référence APOLLO-2 des Assemblages REP UOX en Evolution à Partir des Analyses de Combustibles Irradiés BUGEY 3 et GRAVELINES”, CEA Internal Note.

[15] Takano, Akie, Kaneko, Reactor Physics Conf., Long Island, October 1998, p. 58.

[16] Pilate, *et al*., “Supporting Nuclear Data for Advanced MOX Fuels”, report of a study in the framework of the European Union R&D Programme 1995-1998, on Nuclear Fission Safety, Radioactive Waste Management.

[17] S. Cathalau, A. Benslimane-Bouland, GLOBAL’95 Conference, Versailles, Sept. 1995, and thesis by A. Benslimane-Bouland, Université de Provence, Aix-Marseille I, 1997.

[18] S. Cathalau, A. Benslimane, A. Maghnouj, P. Fougeras, “Qualification of the JEF-2.2 Cross-Sections in the Epithermal and Thermal Energy Ranges Using a Statistical Approach”, *Nucl. Sci. & Eng*. 121, p. 326, 1995.

[19] P. Blaise, E. Fort, “Resonance Parameter Adjustment Methodology Based on Integral Experiment Analysis”, *Nucl. Sci. & Eng*. 133, p. 235, 1999.

[20] H. Tellier, C. van der Gucht, J. Vanuxeem, “Integral Validation of the JEF-2 Major Actinides for Thermal Neutron Reactors”, JEF/DOC-349, also contribution to the ANS Conference, Pittsburgh, USA, 28 April-3 May 1991.

[21] H. Tellier, “Remarks about the Uranium-233 Evaluations”, JEF/DOC-357.

[22] A. Santamarina, N. Thiollay, C. Chabert, “JEF-2 Fission Product Qualification Based on French Integral Experiments”, JEF/DOC-784, April 1999.

[23] N. Gulliford, D. Hanlon, “Measurement of Reactivity Loss with Burn-up in PWR, BWR and MOX Fuel Made in the CERES Collaborative Programme”, PHYSOR’96, p. E-151.

[24] N. Gulliford, D. Hanlon, “Summary of CERES Fission Product Analysis and Some Observations on ^{239}Pu Fission Spectra”, JEF/DOC-648, December 1996.

[25] S. Cathalau, J.-P. Chauvin, P. Fougeras, A. Maghnouj, “Calculation of 2-D and 3-D Local Void Effects in MOX Reactors: Analysis of the EPICURE Results”, PHYSOR’96, p. C-11.

[26] P. Fougeras, S. Cathalau, P. Klenov, P.J. Finck, “Improved Caluclational Scheme for Absorber Clusters in 30% MOX-Fuelled PWRs”, PHYSOR’96, A-258.

[27] J.M. Palau, “Correlations entre données nucléaires et expériences integrales à plaques”, CEA-R-5843; see also paper presented to the Mathematics and Computation Conf., Madrid, Sept. 1999).

[28] S. Pelloni, “Non-Fertile Fuel Benchmark: Sensitivity of the Calculational Results Against the Basic Cross-Section Libraries”, JEF/DOC-699, December 1997.

Annex 1

JEF/DOC Documents Relating to JEF-2.2 Thermal Reactor Benchmark Studies

JEF/DOC-309 Thermal Neutron Region Slowing Down Sources; Memo by D. Mathews.

JEF/DOC-311 JEF Integral Benchmark Testing in the UK; A.D. Knipe.

JEF/DOC-327 JEF Working Group on Benchmark Testing: UK Progress; A.D. Knipe.

JEF/DOC-329 Data and Results for KRITZ Experiments on Regular H_2O/Fuel Pin Lattices at Temperatures up to 245°C; E. Johansson (STUDSVIK/NS-90/133).

JEF/DOC-349 Integral Validation of the JEF-2 Major Actinides for Thermal Neutron Reactors; H. Tellier, C. van der Gucht, J. Vanuxeem (Contribution to the ANS Conference, Pittsburgh, USA, 28 April to 3 May 1991).

JEF/DOC-355 Resonance Parameter Representation in JEF-2.2; NEA Data Bank, Dec. 1991.

JEF/DOC-357 Remarks about the Uranium-233 Evaluations; H. Tellier.

JEF/DOC-362 Nuclear Data Improvements in the Decade with Special Emphasis on Very Recent Data Evaluations, and their Application to Thermal and Fast Reactor Analysis; A. Mateeva-Küsters and E. Kiefhaber.

JEF/DOC-363 Calculations to Compare with Measurements of Thermal Neutron Spectra in Borated Water; J Rowlands.

JEF/DOC-396 UK Programme/Progress; N.T. Gulliford.

JEF/DOC-400 Integral Data Testing for Thermal Reactors and Feedback into JEF-2; H. Tellier and J. Vanuxeem.

JEF/DOC-402 JEF-2 Validation Work in Cadarache December 1992; E. Fort and S. Cathalau.

JEF/DOC-425 Benchmarking of the JEF-2.2 Bases EJ2_XMAS Neutron Cross-Section Library; R.C.L. van der Stad, H. Gruppelaar, J.L. Kloosterman, Y. Wang.

JEF/DOC-428 Complete Analysis of the ERASME Experiments Using CEA93 Libraries; S. Cathalau.

JEF/DOC-429 SHERWOOD Benchmark Analysis Using JEF-2.2 Cross-Section Data; A. Benslimane, S. Cathalau.

JEF/DOC-430 Benchmarking JEF-2.2 with MONK; N.R. Smith, D. Hanlon and A.K. Ziver.

JEF/DOC-431 Summary of an Assessment of JEF-2.2 Nuclear Data Library for LWR Lattice Calculations; D. Hanlon, N.T. Gulliford.

JEF/DOC-433 Influence of Delayed Neutron Importance on Calculated K-eff for Thermal Systems; N.T. Gulliford.

JEF/DOC-434 Summary of UK Integral Benchmark Results for JEF 1992/93; N.T. Gulliford.

JEF/DOC-459 Benchmark Testing of the EJ2-XMAS Library by PASC-4 Code System; Wang Yaoqing.

JEF/DOC-461 Some Provisional Results from the 1993/94 UK JEF2.2 Benchmarking Programme; N.T.Gulliford, D. Hanlon, G.A. Wright.

JEF/DOC-462 The UK JEF International Benchmarking Programme: Current Status and Future Programme (Draft); N.T. Gulliford.

JEF/DOC-464 Comments on Discrepancies in ^{154,155}Eu and ^{166}Er Resonance Integrals and 2 200 m/s Cross-Sections Between JEF-2.2 and Other Libraries; C. Gragg.

JEF/DOC-480 Qualification of the JEF-2.2 Cross-Sections in the Epithermal and Thermal Ranges Using a Statistical Approach; S. Cathalau, A. Benslimane, A. Maghnouj, P. Fougeras, V. Ukraintsev.

JEF/DOC-495 Results of Benchmark Calculations; J.L. Hutton, A.W. Butement, C.J. Dean, R.J. Perry, D.J. Powney.

JEF/DOC-499 Some Remarks about the ^{241}Am Capture Cross-Section and Branching Ratio; S. Cathalau, R. Soule, A. Benslimane.

JEF/DOC-500 A MCNP Analysis of PWR/BWR Critical Benchmarks by Using JEF-2.2 and ENDF/B-VI J.D. Kim, C.S. Gil.

JEF/DOC-501 Qualification of Fission Product Cross-Sections by Reactivity Worth Measurements in MINERVE Reactor; A. Santamarina, P. Albarede, B.Gastaldi.

JEF/DOC-504 Specification of the DIMPLE S01 Benchmark Assemblies; A.D. Knipe.

JEF/DOC-518 DIMPLE S01A Models; R.J. Perry, C.J. Dean.

JEF/DOC-520 Oxygen Potential Scattering; C.J. Dean.

JEF/DOC-541 The TRX-1 and TRX-2 Benchmarks – Validation of the JEFF-3.0 Evaluation for ^{235}U; A. Hogenbirk.

JEF/DOC-542 Contribution to the Benchmark of JEF-2: Results for ESADA Lattices; C. Mounier.

JEF/DOC-543 Testing of the JEF-2.2 and EDF/B-VI.2 Through MCNP Analysis of BWR-Simulated Critical Experiments; Jung-Do Kim, Choong-Sup Gil.

JEF/DOC-548 The DIMPLE S01A Critical Assembly – Results of MCNP-4A Calculations; T.T.J.M Peeters and A. Hogenbirk.

JEF/DOC-549 Description of the DIMPLE S01A Model Developed with Monte Carlo Computer Code MCNP-4A; Marko Maucec.

JEF/DOC-568 The Influence of the Basic Evaluated Data Files on Thermal Lattice Integral Parameters; A. Trkov.

JEF/DOC-569 The Nuclear Core Design Manual – Krsko NPP – Operation CORlibfile from INCORE Analysis; A. Trkov, M. Kromar.

JEF/DOC-589 Synthesis of the Comparison of JEF-2.2 (ENDF/B-VI) and LEAL-DERRIEN/95 Evaluations for ^{235}U; S. Cathalau, P. Blaise.

JEF/DOC-598 Verification of Evaluated Nuclear Data File for Vanadium; A.Trkov, G. Reffo.

JEF/DOC-599 MCNP Simulation of the TRIGA Mark II Benchmark Experiment; R. Jeraj, B.Glumac, M. Maucec.

JEF/DOC-600 The EAF-97 Library – The European Activation File, 1997 Version; J.-Ch. Sublet, J. Kopecky, A. Simpson.

JEF/DOC-627 Problems Related to the Resonance Analysis of ^{235}U; H. Derrien, L. Leal, N.M. Larson, M.C. Moxon.

JEF/DOC-639 Updated JEF-2.2 Pu data – The Effect on Several Benchmarks; A. Hogenbirk.

JEF/DOC-644 Future Evaluated Nuclear Data for ^{235}U – Progress Report; C.J. Dean.

JEF/DOC-646 Benchmark of JEF-2 Actinides and Fission Products with Isotopic Depletion in the French Power Reactor Gravelines; C. Mounier.

JEF/DOC-648 Summary of CERES Fission Product Analysis and Some Observations on ^{239}Pu Fission Spectra (copies of overheads presented); N.T. Gulliford.

JEF/DOC-649 Some Results on the Doppler Broadening of Neutron Resonances in U and UO_2 from Measurements at Low Temperature; A. Meister.

JEF/DOC-696 Overview of JEF-2.2 Qualification for Criticality Calculations; A. Nouri, N. Smith.

JEFDOC-699 Non-Fertile Fuel Benchmark: Sensitivity of the Calculational Results Against the Basic Cross-Section Libraries; S. Pelloni.

JEF/DOC-702 Analysis, Processing and Integral Testing of the ^{235} U Leal-Derrien-Larson's Evaluation; C. Mounier.

JEF/DOC-706 The Effective Temperature for Doppler Broadening in UO_2 Lattices; A. Meister, A. Santamarina.

JEF/DOC-707 Qualification of the ^{235}U Leal-Derrien-Larson Evaluation Using French Integral Experiments; M.C. Alet, A. Benslimane, C. Chabert, C. Mounier, A. Santamarina, G. Willermoz.

JEF/DOC-708 An Evaluation of the ^{235}U Resonance Parameters Using the Program REFIT; M.C. Moxon.

JEF/DOC-726 Summary Notes: Specialists Meeting on Fission Product Cross-Sections, 18 June 1997; C. Dean.

JEF/DOC-727 Remarks on Measurements on Long-Lived Radioactive Fission Products; H. Weigmann.

JEF/DOC-730 Worth of Thermal Reactor Fission Products; N.T. Gulliford, D. Hanlon.

JEF/DOC-731 Preliminary Evaluation of the LENDEL, *et al*. Model to Calculate the Delayed Neutron Yield as a Function of Energy. First Results of the JEF-2.2 Data Validation; E. Fort, V. Zammit, A. Filip, E. Dupont.

JEF/DOC-732 Notes in Support of Discussions at JEF Specialists Meeting on Fission Products Held at the NEA Data Bank, Paris, June 1997; N.T. Gulliford.

JEF/DOC-733 Notes on Energy Dependence of Capture Cross-Section for Individual Nuclides (Including 20 Highest Contribution Nuclides and 7 Nuclides with RMS (>10%)).

JEF/DOC-737 JEFF Working Group on Fission Product Cross-Sections; Summary Record of the Meeting Held on Monday, 15 December 1997; C.J. Dean.

JEF/DOC-744 Benchmark Calculations for ^{235}U; C.J. Dean.

JEF/DOC-745 Solid State Effects on the Reaction Cross-Section Below a Few eV; M.C. Moxon.

JEF/DOC-750 On the Influence of Crystalline Binding on Resonant Absorption and Reaction Rates; D. Naberejnev, C. Mounier and R. Sanchez.

JEF/DOC-751 Nuclear Data Adjustment Using Spent Fuel Analysis: Feedback to the JEFF-3 Project; C. Mounier.

JEF/DOC-752 Contribution to the Validation of the New ^{235}U Evaluation; I. Guimier and A. Nouri.

JEF/DOC-759 JEF-2.2 Validation: Part 1 – General Purpose File; E. Fort, *et al*.

JEF/DOC-761 Plots of Important Fission Product Radiative Capture Cross-Sections, EAF vs. Experimental Data; J. Kopecky.

JEF/DOC-784 JEF-2 Fission Product Qualification Based on French Integral Experiments; A. Santamarina, N. Thiollay, C. Chabert/

Annex 2

TRX-1 and TRX-2

TRX-1

Slowing down density at 4 eV = 0.62. Values of {(C/E) - 1} %.

	k_{eff} ((C/E) - 1)%	**ρ^{28} ((C/E) - 1)%**	**δ^{25} ((C/E) - 1)%**	**F8/F5 ((C/E) - 1)%**
MCNP-4A/ECN, whole core	+0.12 ± 0.20	1.5 ± 1.52	-0.27 ± 1.02	1.1 ± 4.34
MONK-8 (point), whole core	-0.580 ± 0.06			
MVP (Japan)	-0.42 ± 0.053			
APOLLO-2V-4 (core calculation)	-0.368			
APOLLO-2V-4, pin cell	-0.151			
APOLLO-2 V-2, pin cell	+0.145			
MONK-8 (group), whole core	-0.807± 0.06			
WIMS, pin cell	-0.896			

ρ^{28} and δ^{25} denote the epithermal to thermal ratios for ^{238}U capture and ^{235}U fission.

TRX-2

Slowing down density at 4 eV = 0.71. Values of {(C/E) - 1} %.

	MCNP-4A/ECN ***Whole core***	**APOLLO-2** ***Pin-cell***	**MVP (Japan)**
k_{eff}	0.13 ± 0.20	0.077	-0.63 ± 0.051
ρ^{28}	0.06 ± 1.93		
δ^{25}	-1.21 ± 1.32		
F8/F5	0.5 ± 5.05		

The MCNP-4A uncertainties are the combined measurement and calculation uncertainties but are predominantly the measurement uncertainties. The MONK uncertainties are the calculation uncertainties.

Discussion of the differences between the Winfrith and the ECN results for TRX-1 *(comments received from C.J. Dean, Winfrith)*

The JEFF-2.2 k_{eff} values for TRX-1 are calculated to be 120 ± 200 pcm high for the MCNP-4A/ECN whole core model whereas the Winfrith MONK whole reactor model gives a C/E discrepancy of -580 ± 60 pcm. The Winfrith results are thus about 0.7% lower than those being obtained in ECN using the same JEF-2.2 data library.

The Winfrith calculations were made using the 3-D model specified in the benchmark by the Cross-Section Evaluation Working Group (CSEWG) [1]. This model cross-checks with pin cell calculations made using the specified experimental buckling (within the expected differences between GROUP and POINT code methods found for other assemblies). Note that there is a small problem in that the CSEWG book specifies 764 fuel rods on page T(6-9)-4 and 763 on page T(6-9)-5. The Winfrith calculations used 763 but the effect of this difference is estimated to be very small. This has been confirmed by pin cell calculations (using the total measured buckling of 0.0057 cm^{-2} split to give a radial buckling of 0.005174 and an axial value of 0.000526 cm^{-2}).

The MCNP-4A/ECN calculations were made by A. Hogenbirk and are detailed in JEF/DOC-541. They were made using the specification in WAPD-TM-931 (and *Nucl. Sci. Eng*. 40(1970)101). This calculation is not for a full core but is a 2-D calculation which uses "leakage disadvantage factors" based on calculations made in the 1970s and given in the *Nucl. Sci. Eng*. paper. The MCNP-4A/ECN model used an equilateral triangle with segments of pins in each corner and reflected boundary conditions. The printed number density for oxygen appears to be slightly wrong (3.383E-2 instead of 3.338E-2) but this may be a typographical error in the report. A calculation using the ECN model has been repeated in WIMS and gave a k_∞ value within 200 pcm of the ECN result. This shows that the 1970 leakage correction is inconsistent with the 3-D model.

Note that the result for TRX-1 obtained by R.Q. Wright using ENDF/B-VI revision 3 (-780 pcm) is more consistent with the Winfrith JEF-2.2 results (there are some differences, however, between the ^{235}U and ^{238}U data in JEF-2.2 and B-VI revision 3). The value obtained using ENDF/B-VI by Takano, *et al*. using the MVP code is -810 ± 49 pcm, in agreement with R.Q. Wright's calculation, the value obtained using MVP and JEF-2.2 being -420 ± 53 pcm, closer to the MONK-8 value of -580 ± 60 pcm.

The above WIMS pin cell calculation is a cylindrical buckled pin cell with a special Carlvik calculation to account for the triangular pitch. (The WIMS result can be improved with more appropriate options than given in the "reference" input. Results then are close to the MONK-8 group results.)

REFERENCES

[1] "Cross-Section Evaluation Working Group Benchmark Specifications", BNL 19302-ENDF-202.

Annex 3

Results of Calculations for the KRITZ Cores

Table A.3.1. Calculations made at Studsvik using CASMO/j2lib (Kim Ekberg)

XY geometry with axial buckling

Core	Case	k_{eff}	Difference Hot-Cold (pcm)	Difference (pcm/°C)
2.1 ΔT = 228.8	Cold Hot	1.00050 0.99830	-220	-1.0
2.13 ΔT = 220.9	Cold Hot	1.00074 1.00154	80	0.4

Table A.3.2. Calculations made at CEA Saclay using APOLLO/JEF-2 (Claude Mounier)

XY geometry ($P_1.S_8$) with axial buckling

Core	Case	k_{eff}	Hot-Cold (pcm)	(pcm/°C)
2.1 ΔT=228.8	Cold Hot	0.99928 0.99889	-39	-0.2
2.13 ΔT=220.9	Cold Hot	1.00127 1.00142	15	0.1

Table A.3.3. Calculations made at Winfrith using WIMS6/JEF-2 in 172 groups

Debye Temp. = 0, XY geometry (P_1.W-MONK Monte Carlo) with axial buckling

Core	Case	k_{eff}	Hot-Cold (pcm)	(pcm/°C)
2.1	Cold	0.9997 (±0.0006)		
ΔT=228.8	Hot	0.9986 (±0.0006)	-110	-0.5 (±0.4)
2.13	Cold	1.0002 (±0.0006)		
ΔT=220.9	Hot	0.9988 (±0.0006)	-140	-0.6 (±0.4)
2.19	Cold	1.0005 (±0.0006)		
ΔT=214.8	Hot	0.9979 (±0.0006)	-260	-1.2 (±0.4)