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Meeting Cross Section Requirements for Nuclear Energy Design

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R. T. Santoro

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MEETING CROSS SECTION REQUIREMENTS FOR NUCLEAR ENERGY DESIGN*

C. R. Weisbin, D. Gilai†, G. deSaussure, and R. T. Santoro

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ABSTRACT

The purpose of this report is to summarize and explain current requirements in cross section data that are essential to nuclear energy programs and to provide some insight into how these data might be obtained. The report is divided into six sections that describe:

1. Design parameters and target accuracies,
2. Data collection, evaluation, and analysis,
3. Determination of high accuracy differential nuclear data for technological applications,
4. Status of selected evaluated nuclear data,
5. Analysis of benchmark testing, and
6. Identification of important cross sections and inferred needs.

It is generally accepted that the uncertainties in the prediction of reactor properties can result in excessive design margins and, as a result, costs. There are, however, only a few good published studies that demonstrate the consequences of uncertainties in nuclear data. Significant complexity is introduced in such studies when factors such as feedback of operating data, design flexibility in accommodating errors, calibration of prototypic experiments, methods and modeling uncertainties, etc., are included in a systematic way. The cross-section data, and the uncertainties in these data, have changed dramatically in recent years and the necessary updating must be completed and made available. High accuracy measurements often require several years of planning, data collecting, and analyses. Also, the needs of nuclear energy programs change and as data requirements are met, new problems identify different and new data needs. An attempt is made in this report to update and identify nuclear data requirements and to summarize the state of the data as of 1981.

1.0 INTRODUCTION

The purpose of this paper is to summarize and explain current requirements in cross-section data that are essential to nuclear energy programs and to provide some insight into how these data might be obtained. The paper is divided into six sections that describe:

A. Design parameters and target accuracies,

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It is generally accepted that the uncertainties in the prediction of reactor properties will result in excessive design margins and, as a result, excessive costs. There are, however, only a few good published studies that demonstrate the consequences of uncertainties in nuclear data. Significant complexity is introduced in such studies when factors such as feedback of operating data, design flexibility in accommodating errors, calibration of prototypic experiments, methods and modeling uncertainties, etc. are included in a systematic way. The work of Greebler, et al.[1] for fast reactors was completed ten years ago and an update of this work is very much needed. The investigations of Becker and Harris[2] for thermal reactors show there are substantial impacts of nuclear data uncertainties on the light water fuel cycle. Some work has

also been carried out to assess the impacts of cross-section uncertainties on fusion reactor design,[3,4] but this work is still in its early stages.

The cross-section data, and the uncertainties in these data, have changed dramatically and the necessary updating must be completed and made available. High accuracy measurements often require several years of planning, data collecting, and analyses. Also, the needs of nuclear energy programs change and as data requirements are met, new problems identify different and new data needs. An attempt is made here to update the nuclear data requirements and to summarize the state of the data as of 1981 for the current generation of reactors.

In Section 2.0, design parameters and target accuracies are reviewed. Discussions of fuel cycles for both fission and fusion reactors are included along with the energy ranges over which the data are required. The design parameters and target accuracies for carrying out physics, shielding, and fuel handling and waste management analyses of light water and fast reactors are reviewed. A brief discussion of fusion reactor design parameters is also included.

The interrelationship between data collection, analyses, and evaluation is discussed in Section 3.0. The elements of differential measurements and calculations, data evaluation, data processing, and integral measurements that are used to validate data for inclusion in the ENDF/B files are summarized. The organization of the Cross Section Evaluation Working Group (CSEWG) and its role in these processes are also reviewed.

In Section 4.0 the several aspects of the measurement and evaluation of high accuracy differential nuclear data for technological applications are presented. The interaction between measurements and nuclear theory is stressed along with the interrelationship between different measurements. The nature and origin of uncertainties or errors in the data are identified. The relationship between nuclear theory and the results of measurements is discussed. A brief discourse is also presented on the planning of neutron measurements for nuclear applications focusing on the requirements for establishing a broad data base rather than the specific requests for particular applications. The Oak Ridge Electron Linear Accelerator (ORELA) is discussed as an example of a cross section measurement facility currently used in the acquisition and establishment of accurate nuclear data for reactor design and in nuclear physics studies.

Section 5.0 gives a review of the present status, in relation to accuracy requirements, of some specific evaluated nuclear data for the design and calculation of thermal and fast reactor performance parameters.

The analysis of benchmark testing for thermal and fast reactor data and shielding data is discussed in Section 6.0. A review of the experiments, the elements of the data-testing process, and the conclusions realized from these benchmarks are given. Finally, the identification of the important cross sections which can play a major role in the calculation/experiment discrepancies (identified in Section 6.0) is reviewed in Section 7.0 for both fission and fusion applications.

2.0 DESIGN PARAMETERS AND TARGET ACCURACIES

Cross section measurements are performed in order that calculations of nuclear design parameters can be made with acceptable accuracy. The purpose of this section is to present available target accuracies for nuclear design parameters for a number of reactor physics applications. These are particularly appropriate in the area of core physics, where safety and even feasibility considerations demand that the core be designed within tight, well defined tolerances. The situation is somewhat different in the shielding area since imprecision in design can sometimes be accommodated after the fact, albeit with significant economic penalty. "In contrast to reactor physics, approximations in the methods and geometric modeling (rather than the uncertainties in the basic nuclear data) remain the principal sources of error in shielding calculations..., there is no generally accepted statement of target accuracies laid down for shielding calculations which, by their nature, are more closely related to specific design issues and therefore more strongly dependent on the nature of, and progress with, a particular design."[5]

2.1 Fuel Cycles

First, we very briefly describe some fission and fusion reactor fuel cycles; the characterization of the fission reactor fuel cycles is taken directly from the review of Rowlands.[6]

"The reactor systems currently being used and developed are:

A. Uranium Fueled Thermal Reactors

Principal fissile nuclide ^{235}U; fertile nuclide ^{238}U. A proportion of the ^{239}Pu produced by neutron capture in

^{238}U is fissioned in the reactor; plutonium is not currently recycled in thermal reactors but this is a possibility for the future.

B. Plutonium-Uranium Fueled Fast Reactors

Principal fissile nuclide ^{239}Pu; fertile nuclide ^{238}U. Plutonium produced in thermal reactors is extracted from the irradiated fuel by chemical reprocessing and is then mixed with natural or depleted uranium for use in fast reactors. The fast reactor core is surrounded by a natural or depleted uranium breeder region and more plutonium is bred from the ^{238}U component of the fuel in the core and from the uranium breeder region. The fuel in the core is irradiated until about 10% of the uranium plus plutonium atoms have been fissioned, and is then removed for reprocessing to extract the fission products and produce fuel of the composition required to maintain criticality. This cycle produces more fissile plutonium than is consumed and so the number of plutonium fueled fast reactors can be increased using only a supply of natural or depleted uranium, and the uranium utilization is high. It is also possible to fuel fast reactors initially with ^{235}U and convert to plutonium fueling using the plutonium bred in the fast reactors.

C. Thorium Cycle

Naturally-occuring thorium consists only of the fertile isotope ^{232}Th from which ^{233}U can be produced by neutron capture. Some ^{232}Th must first be converted to ^{233}U in a reactor fueled with ^{235}U or ^{239}Pu and then the cycle ^{233}U - ^{232}Th can be self-sustaining.

As well as these primary cycles, mixed cycles are being studied; for example, fast reactors with plutonium-uranium cores surrounded by thorium breeder regions in which ^{233}U is produced for use in thermal or fast reactors."

D. Fusion Cycles

Four fuel cycles offer potential extrapolation to power producing fusion reactors

D-T-Li Cycle	$D + T \rightarrow n + {}^{4}He + 17.6$ MeV
D-D Cycle	$D + D \rightarrow H + T + 4.03$ MeV (50%)
	$D + D \rightarrow {}^{3}He + n + 3.78$ MeV (50%)
D-^{3}He Cycle	$D + {}^{3}He \rightarrow H + {}^{4}He + 18.3$ MeV
p-^{11}B Cycle	$p + {}^{11}B \rightarrow 3{}^{4}He + 8.66$ MeV.

Current emphasis is on the D-T-Li Cycle with only a modest effort devoted to pursuing reactor designs with advanced fuel cycles. The D-T cycle will be used in all fusion ignition experiments and in the first generation of reactors. It may be considered as the reference fusion fuel. The time scale for advanced fuel reactors is uncertain although preliminary studies of large tokamaks burning a catalyzed D-D fuel are encouraging.

2.2 Energy Ranges of Interest

Rowlands[6] also discusses the energy ranges which contribute to reactions of significance in fast and thermal fission reactors which is discussed below and summarized in Table 2.1.

> "The mean energy of neutrons produced in fission is 2 MeV. In a fast reactor these are moderated, on average, through a decade in energy before causing fission. The mean energy of neutrons causing fission in ^{239}Pu being about 200 KeV. The mean energy of neutrons causing radiative capture is about 20 KeV. This is because the fission cross section of ^{239}Pu is approximately flat, whereas capture cross sections mainly decrease with increasing neutron energy, being roughly proportional to $E^{-1/2}$. The mean energy of the ^{238}U Doppler effect is a decade lower, at 2 KeV. Cross sections are of significance throughout the energy range 100 eV to 15 MeV. Above about 3 MeV the reactor neutron spectrum has a similar shape to the fission spectrum and so, for reactions with an effective threshold above this energy, a fission spectrum averaged value or a reactor spectrum averaged value of the cross section is sufficient when high accuracy is not required. Because they decrease with increasing neutron energy, radiative capture cross sections are not required so accurately above 1 MeV (the region where highest accuracy is required being 1 to 100 keV).
>
> In a thermal reactor most neutrons are absorbed at thermal energies (in the range 1 mV to 1 eV). Above 1 KeV the slowing down source is only slowly varying up to the energy of the fission neutron source and the spectrum is similar in shape to the fission spectrum above about 5 KeV. Resonances in oxygen and carbon cross sections can modify the shape of the spectrum at low MeV energies. In the heavy element resonance region, from 1 eV to 1 KeV, the spectrum is strongly influenced by individual resonances, particularly those in ^{238}U (or ^{232}Th) and resonance shielding effects are very large."

Table 2.1 Incident Neutron Energy Ranges of Importance

Fast Reactor Spectra (100 eV to 15 MeV)	
Mean energy of ^{239}Pu fission	200 KeV
Mean energy of ^{238}U capture	20 KeV
Mean energy of Doppler effect	2 KeV
Shape above 3 MeV similar to fission spectrum	
Thermal Reactor Spectra (1 mV to 15 MeV)	
Thermal region 1 mV to 1 eV (Maxwellian)	
Resonance region, 1 eV to 1 KeV (Strong resonance shielding)	
Fast Region, 1 KeV to 15 MeV	
Spectrum approximatly uniform in lethargy,	
i.e. (1/E) in energy, up to 1 MeV,	
Proportional to fission spectrum above 5 MeV	

The D-T fusion reaction delivers 80% of its power via a 14.1 MeV neutron and 20% via the charged alpha particle. The neutron will further generate 2 to 4.8 MeV from exothermic reactions when the reactor incorporates lithium as the fertile material in the blanket. From the point of this paper, the neutron production is of principal concern, although a large fraction of the alpha particle energy will be converted to electromagnetic energy, deposited at the first wall, and transferred to the thermal energy recovery system.

2.3 Design Parameters and Target Accuracies

"Nuclear designers supply such information as composition (hence enrichment) for fuel specification, power distributions for thermal-hydraulics design and for verifying that limits on cladding and fuel are not exceeded, control enrichments necessary to meet safety and burnup requirements, and

power coefficients for operational use and safety verification."[7] Other nuclear design parameters include inherent source strength, activation, breeding ratio and radiation damage. Design target accuracies reflect judgment in the compromise between design conservatism and excessive cost. No attempt is made in this paper to explain or justify individual designer judgment.

A. LWR Physics

Table 2.2 indicates target and achieved accuracies for light water reactor physics analysis as of 1971,[8] as well as those assumed. In 1971, uncertainties needed to be reduced by almost a factor of two relative to target goals. Note that this information reflects consensus opinion of generic designs; accuracy targets and capability for specific light water reactor design is often proprietary. Also predictive accuracies tabulated reflect the results of calculations using design methods as opposed to benchmark methods. By 1981, calculational capability substantially improved relative to these original objectives; the objectives themselves remain largely unchanged.

B. Fast Reactor Physics

McFarlane[9] tabulates uncertainties and design targets for demonstration sized LMFBR's in the Federal Republic of Germany as well as in the United States. The status of uncertainties in 1972 as well as the present status for SNR-300 and CRBR is described. Table 2.3 is reproduced below indicating that current LMFBR designs appear close to meeting their targets.

Table 2.2. Estimated Power-Reactor[a] Physics Accuracy[8]

	Predictive Accuracy, %		Predictive Accuracy, %[b]	
	Objective	(1971) Capabilities	Objective	(1981) Capabilities
Steady-State Power				
Distribution				
Within a a Fuel Pin	± 5	± 10-20	± 5	± 8
Fuel Pin Relative to Assembly	± 2	± 3-5	± 2	± 2-4
Axial, Within an Assembly	± 3-5	± 6-10	± 3-5	± 4-8
Radial, Between Assemblies	± 1-3	± 3-8	± 1-3	± 2-5
Overall, Pellet to Average	± 3-5	± 8-13	± 3-5	± 5-9
Steady-State Reactivity				
Initial Neutron Multiplication	± 0.25 0.5	≤± 1	± .2	± .3
Reactivity Life-Time	± 2-5	± 2-10	± 2-5	± 2-6
Fuel Burn-up				
Peak Pellet	± 3	± 5	Little change since 1971 because of lack of reprocessing	
Fuel Assembly	± 2	± 4		
Discharge Batch	± 2	± 3-5		
Isotopic Composition[c]				
Local (Pellet)				
^{235}U-Depletion	± 2	± 5		
$^{239}Pu/U$ Ratio	± 2	± 4		
Net Fissile Atoms Produced/U	± 2	± 4		
Fuel Assembly				
^{235}U Depletion	± 2	± 5		
$^{239}Pu/U$ Ratio	± 2	± 5		
Net Fissile Atoms Produced/U	± 2	± 5		
Discharge Batch				
^{235}U Depletion	± 2	± 5		
$^{239}Pu/U$ Ratio	± 2	± 5		
Net Fissile Atoms Produced/U	± 2	± 5		

[a]Estimates are for generic designs, not appropriate for any specific reactor.
[b]Private communication, from R. Crowther to C. R. Weisbin, July 1981.
[c]Assuming that the burn-up is known in predictive comparisons.

Table 2.3. Uncertainties in Demonstration Reactor Design Parameters

Parameter	Uncertainty,[a] %					
	Status				Target[e]	
	FRG (1972)[b]	USA (1972)[c]	SNR-300[d] (1978)	CRBR (1980)	FRG[b]	USA[c]
k_{eff}	1.1	1.6	0.4	0.7	0.9	0.6
peak/average power	2.7	12	2.5	~4.7[f]	2-3	3
Control rod worth	6	15	4-10	5	6	3
Doppler coefficient	6-12	15	--	10[g]	6-12	10
Sodium void reactivity	12-18	~50	15	20	9	17

[a] 1 σ level, where σ is the standard deviation of a normal distribution.
[b] Federal Republic of Germany, see Ref. 10.
[c] See Ref. 11.
[d] See Ref. 12.
[e] As of 1972 as defined in Refs. 10 and 11.
[f] Includes many engineering uncertainties, only about 2% comes from ability to calculate power distribution in the criticals. Also just for peak power--may be correlated to average.
[g] Mainly from SEFOR experiments.

In comparing goals and achieved accuracies across countries one must be particularly careful, since arbitrary multiples of the uncertainties are chosen in order to add what is thought to be a proper amount of conservatism. McFarlane[9] notes that CRBR typically uses 1 σ for design margins and 3 σ for safety margins, while the German program typically uses 1.7 σ, corresponding to 90% probability.

Another picture of an LMFBR Program is reproduced below (see Table 2.4) from the paper of Hammer,[13] where uncertainties on design parameters are compared to target accuracies for commercial fast power plants. In

this case, target accuracies are considerably more stringent than those stated in Table 2.3 for demonstration plants.

Table 2.4. Uncertainties* and Target Accuracies for the Next French Commercial Fast Power Plant[13]

Design Parameters	Present Accuracy	Target Accuracy
Reactivity for a fresh core	± 0.5 % $\frac{\Delta K}{K}$	± 0.3 % $\frac{\Delta K}{K}$
Critical mass	± 1.2 %	± 1 %
Reactivity loss per cycle	± 0.7 % $\frac{\Delta K}{K}$	± 0.5 % $\frac{\Delta K}{K}$
Global breeding gain	± 0.04	± 0.03
Doppler effect	≃ 20 %	± 15 %
Sodium void effect	± 40 %	± 20 %
Control rod anti-reactivity	± 20 %	± 10 %
Peak/average power	± 4 %	± 3 %
β_{eff}	± 10 %	± 6 %
TCF	± 5 %	± 3 %
Displacements per atom	± 10 %	± 5 %
Decay heat	± 10 %	± 5 %

*All uncertainties quoted correspond to two standard deviations.

Table 2.3 indicates that the French analysts are more confident in their ability to predict reactivity swing and peak/average power compared

to the projections for the U.S. CRBR plant; on the other hand, they are less confident of their ability to predict control rod worth. Different estimates of present uncertainty can arise from different approaches to uncertainty estimation as well as different sources of information.

Further discussion of the definitions and target accuracy requirements for fast reactor design parameters has been given by Rowlands.[6] His summary is given below in Table 2.5.

Table 2.5. Typical Fast Reactor Requirements[6]

Property	Percentage Accuracy
k_{eff}	0.5 to 1.0
Breeding Ratio	2
Doppler Effect	10 to 15
Maximum sodium void effect	10 to 15
Control requirements	5
Control rod reactivities	5
Power distributions	1
Reactivity scales (Delayed Neutron Data)	3
Decay heat	2 to 5
Activity of components	10
Fluence	5
Radiation damage dose	5
Dose gradients	10

C. LWR Shielding

Butler[5] has described in some detail various fission reactor shielding applications and their relationship to required accuracies of nuclear data. Some of these applications are illustrated in Table 2.6 presented below.

Table 2.6. Shielding Applications: Energy Deposition and Radiation Penetration[5]

Shielding Applications	Design/Analysis Problem Areas	Response Predictions
Fuel Cycle Plant Design	Irradiated Fuel Transport (Flasks) Storage Ponds Reprocessing Plant Vitrification Plant	External neutron dose-rates External gamma dose-rates Radiolysis
Reactor Plant Design (including fusion reactors)	Pressure Vessel Internals	Core gamma heating Heating/Damage in steel-work near the core Flux levels for Power Monitoring Internal Boiler or IHX activation Heating/Damage in steel RPV or PCRV liners/concrete
	External Plant (Balance-of-Plant)	Dose-rates for operational access to reactor and auxiliary plant Site dose-rates
Reactor Operations Support	On-line analysis of radiation fields for operational planning including refueling and maintenance	Radiation dose-rates throughout the plant site. Activation of components removed from reactor Heating for gag-setting, SA bowling, DMSA test rigs. Displacement-rates for damage monitoring in structureal components.
Decommissioning	Access to dismantle vessel and external plant Techniques for dismantling Storage of irradiated items Disposal of active materials	Reaction-rates leading to production of long-lived isotopes

Table 2.7 reproduces[14] some of the shielding accuracy goals for light water reactors as estimated by several different organizations (A through F). In general, there is a desire to reproduce prototypic shielding design experiments to within 10-20% while deep penetration dose rates are desired to within factors of 2-3. Each column corresponds to a different user need for the given response of interest.

Table 2.7. Some Required Target Accuracies in Power Reactor Shielding[14]

No.	Target		Required Accuracies (%)					
1	Gamma-ray heating in materials testing rigs	Without fissile heaters	±5	±5	±20	+ 20		20-50
		With fissile heaters	20	20	-	-	-	10-20
2	Gamma-ray heating in	sub-assemblies	±10	±10	20	±10	-	10-20
		thermal shield	-	-	20	--	20	10-20
		biol. shield	-	-	-	--	30	20-30
3	Control and shut-off rod heating		±10	±10	-	±10 (±15)	-	10-20
4	Displacement rates in the	diagrid	±15	±50	±10	-	-	10-20
		reflector region						20-30
5	LP instrument response		±15	±15	-	±15	-	10-20
6	Displacement rates in the	supporting struct.	-	-			40	10-20
		int. shield system	±25	±(50-100)			30	
7	Activation in the	heat exchangers	±50	Fact. 2-3			30	20-50
		cooling circuits	-	-	-	±25	30	10-20
		turbines	-	-			50	20-50
8	External biological dose rate		Fact. 2-3	Fact. 2-3	Fact. 2-3	±50	40	20-30
	Response from		A	B	C	D	E	F

D. Fuel Handling and Waste Management

Estimation of target uncertainties for fuel handling and waste management may be even more complicated; the values listed in Tables 2.8 and 2.9 are taken from Ref. (15), the proceedings of an advisory group. The

Table 2.8. Accuracy Requirements for Fuel Handling and Waste Management[15]

Parameter	Required Accuracy (%)
Fuel Parameters	
Neutron source in subcritical state	50
Heat production in discharged fuel	10
Neutron output from discharged fuel (for transportation)	20
Buildup of long-lived nuclides	factor 10
Health Physics Parameters	
Measurements of body burden	$\leq$ 5
Estimation of internal dose	15

third column in Table 2.9 shows the error component resulting only from nuclear data (i.e. in the determination of reaction rate as a product of cross section and flux spectrum, the spectrum is assumed known) while the figures in brackets also take into account the error of the neutron flux, assumed to be equal to 10%.

Table 2.9. Accuracies Attained and Required in Calculating Transactinide Build-up

Isotope	Accuracy Required (%)	Estimated Accuracy Attained Fast reactor shield (%)	Estimated Accuracy attained Fast reactor core (%)
Pu-236	15 - 30	100	100
Pu-238	10 - 20	50	35
Pu-240	2 - 5	12(22)	4.5(5)
Pu-241	2 - 4	25(37)	6.5(8)
Pu-242	5 - 10	30(50)	9(9)
Am-241	3 - 5	25(37)	6.5(9)
Am-242	10 - 20	30(47)	18(18)
Am-243	10 - 20	60(77)	47(48)
Cm-242	10 - 20	30(48)	13(14)
Cm-244	15 - 30	70(98)	67(70)

Table 2.10[16] tabulates nuclear data status and requests for actinide production and depletion calculations.

Table 2.10. Thermal Neutron Cross Sections

Element	Isotope		$\sigma_{n\gamma}^{2200}$ Recommended Experimental Value, barns	Precision, % Experimental	Precision, % Requested	σ_{nf}^{2200} Recommended Experimental Value, barns	Precision, % Experimental	Precision, % Requested
Pa(91)	231		210	10	10	0.01	50	-
	233		41	12	5	≤1	-	-
U(92)	232		73.1	2	-	75.2	7	-
	234		100.2	1.5	3	≤0.65	-	-
	236		5.2	6	10	-	-	-
	237		378	33	-	≤0.35	-	-
Np(93)	237		169	2	3	0.019	16	-
	238		-	-	-	2070	2	-
Pu(94)	236		-	-	20	162	20	20
	237		-	-	-	2200	20	-
	238		559	4	20	17.3	3	20
	240		289.5	0.5	3	0.03	150	-
	241		362	3	3	1015	<1	3
	242		18.5	4-5	3	0	-	-
	243		87.4	15	-	180.0	15	-
	244		1.7	6	-	-	-	-
Am(95)	241	to 242	748	3	10	3.14	4	-
		to 242m	83.8	8	10			
	242		-	-	10-20	2100	10	10-20
	242m		-	-	10	7600	4	10-20
	243		77	5	5-10	0	-	-
Cm(96)	242		20	50	20	≤5	-	-
	243		-	-	-	690	7	-
	244		10.6	20	20	1.1	50	-
	245		383	10	10	2161	5	10
	246		1.44	20	10	0.17	60	-
	247		58	10-15	5-10	72.3	10-15	5-10
	248		2.89	10	-	0.34	30	-
Bk(97)	249		1600	50	10	0(?)	-	-
Cf(98)	249		481.4	6	10	1665	3	10
	250		1701	15	10	-	-	10
	251		2849	10	10	4801	10	10
	252		20.4	8	10	32.0	10	10
	253		12	20	-	1100	20	-
Es(99)	253	to 254	<3	-	-	-	-	-
		to 254m	155	13	-			
	254		-	-	-	2900	5	-
	254m		-	-	-	1840	5	-

E. Fusion Design Parameters

The nuclear data bases were initially constructed for fusion reactor neutronics and shielding studies by extending the data bases compiled for fission reactor technology over the energy range required for fusion systems analysis and by including the materials of interest. The nuclear data assembled for fission reactor studies were compiled over a period of many years and were evaluated and checked by extensive comparisons between measurements and calculation of integral experiments. The energy range of interest for fusion extends from thermal energies to the average energy of the D-T reaction of 14.1 MeV. For reactors where the plasma is heated or driven by the injection of neutral deuterium, the kinematics of the D-T reaction produces neutrons with energies up to ~16 MeV. The effective energy range of interest for fission reactors extends to ~5-6 MeV, so most of the data for fusion applications must be collected and evaluated above these energies.

The nuclear data requirements for the magnetic fusion energy program were delineated by Head[17] at the IAEA Advisory Group Meeting on Nuclear Data for Fusion Technology in December 1978. Detailed discussions on the specific data needs and target accuracies for fusion reactor design,[18] nuclear heating,[19] activation,[20] radiation damage,[21] shielding,[22] and hybrid reactor studies[23] were also presented at this meeting. More recent reviews summarizing the changes and modifications to these data requirements have been published by Haight,[24] Browne and Lisowski,[25] and Bhat and Abdou.[26]

The nuclear data requirements, design parameters, and accuracies can be separated, to some extent, by comparing neutronic calculations as they

pertain to the blanket, shield, and other reactor systems. Unlike fission reactor calculations which are basically eigenvalue problems, fusion reactor calculations are source problems wherein the spatial, energy, and time distributions of neutron and gamma radiation arising from a known source distribution are calculated through a physical assembly. The transport of radiation in the blanket can be classified as a shallow penetration problem compared to the analysis of radiation behavior in the shield which is a deep penetration problem. The function of the blanket is to convert the energy of the neutrons and gamma rays to recoverable energy and to breed tritium via neutron reactions with lithium in some form. The shield serves to significantly attenuate the radiation leaking from the blanket in order to minimize the heating, radiation damage, and activation in vital components outside the shield. Although there is an overlap between the analyses of radiation transport in the blanket and shield, the methods and nuclear data requirements differ for each application.

The accuracies necessary to estimate the important nuclear parameters in fusion reactors are given in Table 2.11. More precise data are required for analyzing the nuclear performance of the blanket and superconducting magnets than the other components of the reactor and reactor building. The tritium recovery and nuclear heating rates in the blanket must be accurately determined since these strongly impact the economics of the reactor. While the data accuracies for the shield can be less than those for the blanket, accurate prediction of the radiation leaking through the shield and incident on the cryogenic magnets (particularly in tokamak reactors) is necessary for estimating nuclear responses.

Table 2.11. Some Required Accuracies* in Fusion Reactor Shielding

Location/Response	Desired Accuracy
First Wall/Blanket	
Nuclear Heating	total 2%, spatial distribution 10%
Tritium Producing	breeding ratio 5%, local 10%
Atomic Displacements	10%
Helium Production	10%
Transmutations	20%
Induced Activation	50%
Bulk Shield	
Nuclear Heating	gross 20%, local 30%
dpa and He and H production	factor of 2
Activation	factor of 2
Tritium Production	factor of 3
Main (Superconducting) Magnets	
Nuclear Heating	gross 10%, local 20%
dpa	gross 10%, local 20%
H, He production	gross 40%, local 80%
Activation	factor of 2
Penetration Duct Walls	
Nuclear Heating	local 20%
dpa and He and H production	local 50%
Penetration Functional Equipment (e.g. vacuum pumps, neutral beam injectors)	
Nuclear Heating	gross 30%, local 50%
dpa and He and H production	50%
Activation	factor of 2
Tritium production	factor of 2
Reactor Floor (outside the shield and inside the containment building)	
Biological Dose during operation	factor of 3
Biological Dose after shutdown	factor of 2
Coolant Manifolds and Heat Exchangers	
Biological Dose after Shutdown	factor of 2
Containment Building	
Nuclear Heating	factor of 2
dpa and He and H production	local factor of 4
External Biological Dose (outside containment building)	factor of 3

*These accuracies are approximate and they may change as our knowledge and experience deepen and expand. Some of these accuracies (e.g. that for the nuclear heating in the blanket) may be relaxed for near-term experimental machines.

2.4 Implications of Design Uncertainties

"Uncertainties in the prediction of reactor properties result in:[6]

(a) The provision of design margins to ensure a guaranteed power output.

(b) Design margins to provide adequate control of the reactor.

(c) Provision of funds to cover the cost of possible corrective actions.

(d) Higher guaranteed generation costs which cover the uncertainties.

To ensure a guaranteed power output it might be necessary to build a larger reactor core and associated shielding and containment, and to provide extra control to cover uncertainties in the required fuel enrichment. Provision of control margins might require extra control elements and, in consequence, a larger core and associated structure.

Uncertainties in the required fuel enrichment might be catered for by accepting a possible lower burnup of the fuel in the first few fuel cycles and then adjusting the enrichment to the value found to be necessary. This results in higher fuel costs for the first few cycles."

We dwell on the relationship of cross sections to design savings (economics) for the following reasons: (1) Although there are many nuclear plant related costs which may far outweigh the savings from physics analysis, the question addressed here is whether such savings could exceed the cost of the cross section research program which is relatively small. Possible savings are for example: possibility for extended burnup coupled with improved fuel management, optimization of the blanket/shield configuration for a fusion plant, avoidance of premature shutdown due to concern relating to pressure vessel damage, etc. (2) The current state-of-the-art is perceived to be such that economic related parameters can be calculated with more confidence while safety related parameters

can introduce methods and modeling problems which obscure the relationship to cross section needs.

There are few good published studies which demonstrate the consequences of uncertainties of differential nuclear data. Complexity is introduced through calibration to benchmark and/or prototypic integral experiments, feedback of operating data, and the means by which the designer accommodates inaccuracy in calculation (e.g. by providing a different number of fuel assemblies, different enrichments, different means of control, etc.). The consequences are usually manifested in excessive design margins and ultimately cost.

An early study was made by Greebler, et al.[1] in which LMFBR fuel cycle cost data were directly related to nuclear data uncertainties. Table 2.12 presents some of the highlights of this study. At that time, it was shown that projected costs could be reduced by a factor of about 4, if prominent nuclear data uncertainties could be reduced by a similar factor. A more recent study of Becker and Harris[2] investigated the sensitivity of nuclear fuel cycle costs for PWRs and BWRs to nuclear data. The results of Becker and Harris show some substantial cost implications of nuclear data which depend directly on the reactor type and means of recycle.

Table 2.12. 1975 Data Uncertainties and Recommended Accuracy Goals[1]

Data Type and Incident Neutron Energy Range	Uncertainty (±%)[a]		Cost Uncertainty (±mill/KW(e)h)[b]	
	Present[c]	1975 Goal	Present	1975 Goal
^{238}U $\sigma(n,\gamma)$ 100 eV to 1 MeV	10	2	.065	.013
^{238}U $\sigma(n,f)$ 1 to 10 MeV	6	3	.013	.006
^{238}U $\sigma(n,n')$ 100 keV to 10 MeV	20	5	.025	.008
^{239}Pu $\sigma(n,\gamma)$ 0.1 to 500 keV	20	3	.045	.007
^{239}Pu $\sigma(n,f)$ 0.1 keV to 10 MeV	10	2	.034	.007
^{239}Pu $\overline{\nu}$ >0.1 keV	2	0.5	.050	.013
^{239}Pu $\overline{E_\chi(E)}$ >0.1 keV	10	2	.033	.006
^{240}Pu $\sigma(n,f)$ >1 keV	20	10	.008	.004
^{240}Pu $\sigma(n,\gamma)$ 0.1 keV to 1 MeV	30	10	.004	.001
^{241}Pu $\sigma(n,f)$ >0.2 keV	40	10	.007	.003
Fiss. Prod. $\sigma(n,\gamma)$ >0.1 keV	40	10	.038	.010
Fe,Ni,Cr $\sigma(n,\gamma)$ >0.1 keV	30	10	.041	.010
Combined Data Uncertainties			0.13[d]	0.03[d]

(a) Each independently listed data uncertainty is assumed to be correlated over the entire energy range indicated.
(b) For 1000 MW(e) reactor based on cost assumptions stated in Ref. (1) and estimated 90% confidence limits in data uncertainties.
(c) Corresponds to values listed in Tables III and IV for estimated 90% confidence limits.
(d) Statistical combination assuming separately listed data uncertainties are not correlated. (The assumptions regarding correlation of data uncertainties stated in footnotes (a) and (d) are respectively too pessimistic and too optimistic, and the combination hopefully gives a reasonably correct correlation effect.)

2.5 Conclusions

Concluding this section, it can be seen from Tables 2.2-2.4 that much progress was made during the last decade, approaching the required target accuracies for a variety of design parameters. Although inconsistencies still may exist between different sources as to the current accuracies of some design parameters, the estimated current accuracies seem to be approaching stated target accuracies. There are, however, a few points which should be noted. First, most of the target accuracies which were recently achieved have been set at the early seventies and are based on the capabilities and expectations at that time. The improvements made during the years also make it possible to achieve even higher accuracies so that the fact that a certain target accuracy was achieved does not necessarily imply that one could not or should not do better. The request for a revision of the target accuracies may become even more important in view of the high costs and saving possibilities associated with each uncertainty, as manifested by the work of Becker and Harris.[2] Such a revision could have a significant impact on the nuclear cross section community because it would require a reevaluation of many key cross sections by very precise and costly measurements. Based on the data presented above, it seems that a large scale nuclear cross section measurement program should depend on a cost/benefit analysis which has to take into account on a national basis the savings in energy production against the investments in uncertainty reductions.

As a second concluding remark, it should be considered again that many target accuracies, either required or achieved, are regarded as

proprietary so that the figures presented above should be regarded more as guidelines than as actual targets. Therefore, some care must be taken when quoting or interpreting the above figures.

3.0 DATA COLLECTION, EVALUATION, AND ANALYSIS

The interrelationship between data collection, evaluation, and analysis is shown schematically in Fig. 3.1. Nuclear data are obtained from both measurements and model calculations and the data for given nuclides are evaluated to provide best sets which are subsequently processed into formats directly useable in nuclear analysis. Simultaneously, transport methods are developed and adapted to the particular systems under study. However, before the data and methods are applied in nuclear analysis, they are frequently tested through procedures that involve the analysis of integral experiments and sensitivity studies in which critical areas of the analysis system (data, methods, experiments) are examined in detail. Depending upon the results of the testing procedure, iteration of some of the activities may be required before the data and methods are confidently applied to the particular system.

3.1 Differential Measurements and Evaluation

The data that are acquired from experiments (see Section 4) are collected, stored, and disseminated on an international basis by four data centers, including the National Nuclear Data Center (NNDC) where they are stored as unevaluated data in the Cross Section Information Standard Retrieval System (CSISRS), a worldwide collection of experimental data available from the Nuclear Data Centers. The number and size of the data sets are formidable. For example, CSISRS has more than 200 data sets for $^{235}U(n,f)$. Some data sets have $\sim 10^4$ "points." Hence, the need for an international organization to collect, store, retrieve and

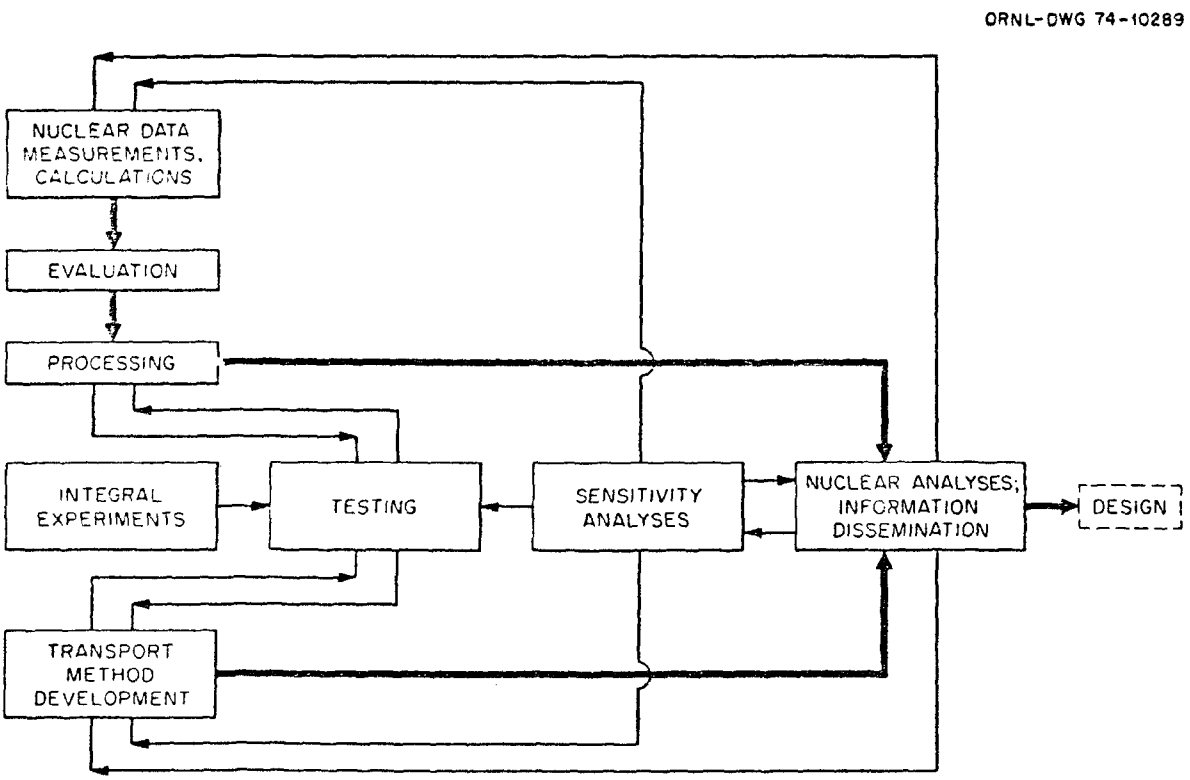

Fig. 3.1. The interrelationship between data collection, evaluation and analysis.

these data is essential. The nuclear data required for analysis but not available directly from measurements may also be obtained from model calculations. Evaluating a set of nuclear data entails developing and using nuclear models both to provide preliminary tests of all available measured cross sections and to calculate cross sections for intermediate energies (see Section 4). The evaluations are coordinated through the Cross Section Evaluation Working Group (CSEWG). The main evaluation in the U.S. is ENDF/B which is generated by CSEWG and is a file that uniquely represents each cross section at each energy. According to Pearlstein,[27] "The process of evaluation involves decision making. The objective of a nuclear data evaluator is to recommend values for nuclear data and also indicate the degree of confidence that can be placed on those recommendations." Each evaluation considers all available neutron and gamma ray interaction data on a given nuclide and these parameters are collected in BNL-325 (by the Brookhaven National Laboratory). The uncertainties associated with the evaluated data are included as covariance files.

The data evaluators are expected to provide a complete information base from incomplete and uncertain information. Differential and integral data are often obtained without a sharp definition where the two data forms are coupled. In 1966, the Cross Section Evaluation Working Group (CSEWG) was formed with representatives from over 20 federal, industrial, and university laboratories. The CSEWG structure consists of committees responsible for evaluations, data testing and applications, and methods and formats. Each of these committees reports to the Executive Committee.

The Executive Committee consists of the CSEWG chairman, funding agency representatives, committee chairmen, three additional members appointed for limited terms by the CSEWG chairman, and one elected member-at-large. The Executive Committee sets policy and gives final approval to recommendations by the other committees.

The Evaluation Committee recommends evaluation responsibilities, schedules and oversees completion of individual evaluations, selects reviewers, reviews physics content, recommends suitability of evaluations, maintains a discrepancy list, reviews requests for nuclear data, recommends new nuclear data measurements, and organizes seminars, workshops, etc. to solve specific evaluation problems.

The Data Testing and Applications Committee recommends data testing responsibilities, schedules and oversees completion of individual testing, reviews integral data experiments, analyzes integral data calculations, selects integral data benchmarks, maintains an integral data discrepancy list, recommends suitability of evaluations, collects needs of applied users, recommends priorities for measurements based on discrepancies between calculation and integral experiments, and organizes seminars, workshops, etc. to solve specific data testing problems.

The Methods and Formats Committee develops ENDF formats for data and covariances, develops ENDF utility codes, develops ENDF processing codes, recommends standard interfaces, investigates analysis methods, and organizes seminars, workshops, etc. to solve specific methods and format problems.

3.2 Nuclear Data Processing

The evaluated data sets available from NNDC are comprised of microscopic cross sections for specific neutron or gamma ray interactions with individual nuclides at discrete ("point") energies. When these data are used within a transport calculation, they are usually processed into an appropriate energy structure and converted to macroscopic cross sections corresponding to the various materials that comprise the system under analysis. In some transport codes, the point cross sections are processed and mixed within subroutines in the main code, with the energy structure being determined by the detail to which the cross sections are known. In other codes, the point cross sections must be averaged and weighted before they can be used directly by the code. These are referred to as "multigroup" cross sections that are represented over rather broad energy intervals. Multigroup cross sections are, in most cases, problem dependent since they are averaged over particle fluxes determined by the geometry and composition of the problem under study. In order to avoid numerous recalculations of average cross sections, the Bondarenko method is sometimes employed to calculate energy averaged cross sections for an isotope over a range of cross sections representing that of the remaining isotopes in the mixture.

3.3 Integral Experiments

Ideally, the use of processed cross sections and the method selected for a particular nuclear analysis would be delayed until both are tested for a given application. In reality, however, the degree of testing is determined by several factors including the uncertainty associated with

the data and/or method and the importance and urgency of the proposed problem, not to mention the availability of funding for the tests.

A. Data-Checking Experiments. A test of a cross section set can be carried out by using the data in a transport calculation performed as an analysis of an integral experiment including the material of interest. Two types of integral experiments may be characterized. Those involving relatively small samples of materials are of the type performed at facilities like ORELA and those involving bulk quantities of materials that are carried out at facilities like the Tower Shielding Facility or the Zero Power Plutonium Reactor. The small integral experiments are designed to emphasize the differential data whereas the large experiments are designed as severe test of the adequacy of data and methods for more complex systems. The uncertainties associated with the differential data should be included in the calculation of integral quantities to determine the confidence level that can be placed on calculated results. Only integral benchmark experiments that are very precise, free from systematic errors and thoroughly documented should be used to test the adequacy of evaluated nuclear data. CSEWG gives official recognition to selected "benchmark" specifications in Ref. 104.

B. Experiments for Methods Testing. Other integral experiments are designed for method testing - or specifically, for the adaptibility of a particular method for transport calculations involving complicated geometric regions and/or various combinations of materials. In such experiments, a region similar to that in question is mocked up; however, in general, prototypic mockups limited to a single design are avoided.

3.4 Development of Transport Methods

The Monte Carlo and discrete ordinates methods are the most appropriate for complex shielding calculations and diffusion theory is used most often in reactor physics analyses. The discrete ordinates method tends to be more practical since it yields the most information about a system, but it is currently limited in the United States to systems whose geometries can be described in one or two spatial dimensions. Problems that require three-dimensional descriptions are solved by using Monte Carlo methods.

3.5 Sensitivity Analysis

Although the term "sensitivity analysis" has several meanings, its basic definition implies the study of the relative contributions to the calculated response by various particle interactions. In its most sophisticated context, however, it refers to a computerized sensitivity analysis program that determines the rate of change of a calculated response with respect to a change in an individual cross section as a function of energy. The analysis indicates how important particular cross sections are to a calculation.

When an experiment is designed primarily for data testing, it is assumed that both the experimental techniques and the transport method as adapted for the analysis have already been verified or at least, the uncertainties associated with these components are understood. Significant disagreements that go beyond these uncertainties are then attributed to inadequacies of the cross section set. A sensitivity analysis can help identify potential sources of discrepancy.

If the calculated response is shown to have a large sensitivity to cross sections in particular energy groups, the question arises as to whether those cross sections, which in the calculation are averaged and weighted values, are truly representative of their groups. Thus, it should not be assumed that the cross sections themselves are in error until the effect of using other multigroup structures has been investigated. This will provoke an iteration with the cross section processing activity and eventually a reanalysis of the experiment. Continued disagreement between the experiment and the analysis will ultimately call for a reevaluation of the cross section data and possibly a new series of measurements, both differential and integral. The iteration continues, in many cases over a period of years, until resolution of the problem is obtained.

4.0. DETERMINATION OF HIGH ACCURACY DIFFERENTIAL NUCLEAR DATA FOR TECHNOLOGICAL APPLICATIONS

4.1 Introduction

In this section we discuss several aspects of the measurement and evaluation of high accuracy differential nuclear data for technological applications. Our purpose is to stress the interaction between measurements and nuclear theory in the evaluation of nuclear data and to emphasize the interrelationship of different measurements. We also comment on the nature and origin of uncertainties or errors in the data and suggest that often the best strategy to improve accuracy and eliminate errors is based on a comparison of data obtained by different methods or a study of different cross sections obtained by the same technique. Such considerations suggest that an efficient long-range program of nuclear data measurements should not be limited to a specific request for a particular application, but should address a broad range of problems by a variety of methods. This is indeed the approach followed by the nuclear data community in constructing a general purpose evaluated nuclear data file used for a variety of applications.

Within the context of this paper the discussion of measurement and evaluation problems is necessarily brief and somewhat superficial and mostly confined to illustrative examples and references to more extensive treatments.

We note here that the results of measurements are practically never used directly for reactor design; designers work with evaluated data usually processed into group cross sections. The evaluated data may be

based on one or several direct measurements, or where no measurements exist on nuclear theories and models. Most often evaluated data are based on direct measurements and theoretical models, hence it may be appropriate to illustrate with a few examples the role of nuclear theory in nuclear data evaluation.

4.2 The Role of Nuclear Theory

It is well recognized that nuclear theory and nuclear models play a central role in the evaluation of high accuracy data.[27] It is instructive to illustrate this role by a few examples.

A. Some data required for nuclear energy applications cannot be measured directly because the required samples are not available or have such short half-lives that their specific radioactivity prevents accurate cross-section measurements. Well known examples of such nuclides are the high Z actinides and the fission products. In a recent book J. E. Lynn[28] describes in detail the theoretical methods for calculating the cross sections of fissionable nuclei. These methods can be validated by comparing the results of calculations with measured data, where available; this comparison suggests that within the group of fissile isotopes of U, Pu, and Am calculations are accurate to within 25%. The method used in the evaluation of the cross section of the fission products has been described in numerous papers by R. E. Schenter,[29] H. Gruppelaer,[30] and many others. The method is based on a study of systematic trends in such nuclear parameters as strength functions, level densities, radiation widths, etc. The accuracy of the theoretical prediction can be considerably improved by adjustments to integral data.[31]

B. Several important cross sections can be measured directly or computed from nuclear models with comparable accuracy. By combining the results of measurements and calculations the evaluator reduces the uncertainties and improves the confidence in the data. An example of an evaluated cross section based on direct measurements and model calculation is the $^{238}U(n,\gamma)$ cross section above 10 keV. The calculational model and the evaluation methodology are described by Poenitz.[32,33] As noted by Guenther et al.[34] the uncertainty of the calculated $^{238}U(n,\gamma)$ cross section can be estimated to be 6% at 10 and 30 keV, and 7% at 100 keV. These uncertainties are not much larger than those obtained from an evaluation of differential experimental data in this range.

Another important example of combined use of nuclear theory and direct measurements is in the evaluation of the neutron induced cross sections of the light elements. G. M. Hale[35] has described the details of the multichannel R-matrix methodology to combine the data from neutron- and charged-particle-reaction measurements.

Many other examples of the use of nuclear models to evaluate neutron data or to supplement direct measurements can be found in the proceedings of recent conferences, especially the 1976 Consultant Meeting on The Use of Nuclear Theory in Neutron Nuclear Data Evaluation[36] and the 1980 Conference on Nuclear Data Evaluation Methods and Procedures.[37]

C. Under (1) we have given examples of cross sections which cannot be measured directly hence must be obtained from nuclear models, under (2) we have given examples of cross sections that can best be evaluated by combining the results of direct measurements with nuclear model

calculations. Before concluding this subsection we want to stress that measurements and nuclear theory are interwoven also at a much more fundamental level; indeed the results of most direct measurements are usually interpreted, fitted and represented with some parametrization based on nuclear models, such as the Wigner Eisenbud reaction theory,[38] the optical model,[39] the Watt-spectrum,[40] etc. Even at the level of data reduction, continuous use is made of nuclear theory to compute corrections to the measurements (due to changes in efficiency of detectors, multiple scattering, angular distributions) and to interpolate and extrapolate the data. Similarly throughout the evaluation process the results of nuclear theory are used as tools to check the internal consistency of the data base. The use of these "tools" in evaluating data for the interaction of neutrons with light elements has been described in a recent paper of L. Stewart.[41]

4.3 Nuclear Theory is Based on the Results of Measurements

In the previous subsection we have indicated how nuclear theory may be used to evaluate data where no direct measurements are available or how the results of nuclear models and direct measurements can be combined to obtain an improved evaluation. It should be obvious, however, that all nuclear theory is ultimately based on the results of experimental measurements and in particular, it should be clear that the determination of the parameters of nuclear models rests on the results of accurate experiments. Thus the use of nuclear models does not obviate the need for accurate measurements; on the contrary the study of systematic trends

necessitates an extensive data base. The use of nuclear models makes possible the replacement of one kind of measurement by another.

Returning to the examples of the previous subsection, Lynn[28] describes how the compound nucleus model utilized in the calculation of the cross sections of the heavy nuclei is based on considerable experimental information on level densities as a function of energy and angular momentum and on accurate measurements of fission cross sections, especially in the threshold and subthreshold regions. Similarly, the calculation of the cross sections of the fission products are based on study of the systematic trends in the distributions of resonance parameters.[29,30] These trends can be obtained from compilations of the results of measurements such as BNL-325[42] and Nuclear Data Tables.[43]

The optical model used by Poenitz[32,33] to obtain the $^{238}U(n,\gamma)$ cross section above 10 keV is based on the systematic measurements of neutron total cross sections by Barschall[44] and others. The parameter Γ_γ/D which is essential in deriving the high energy capture cross sections must be obtained from precise measurements below 10 keV. Similarly the R-matrix parameters used by G. Hale[35] to obtain the neutron induced cross sections of the light nuclei are based, among other things, on the precise measurements of charged-particle-induced reactions.

The point of the discussion so far is to show that improvements in the general data base lead to improvements in nuclear models, which in turn, allow a more accurate determination of specific data. It is not easy to determine a priori whether direct measurement or general improvements in the data base are more likely to yield the desired data to the

required accuracy. This point will be reinforced in the next subsections where the nature of the data uncertainties and the interrelationship of measurements are explored.

4.4 The Nature of Experimental Uncertainties

In this subsection we collect a few comments on the nature and origin of experimental uncertainties, which are relevant to the discussion given in this section.

Experimenters often distinguish between statistical and systematic errors. This separation may not be as clear cut as often assumed,[33,45] but for our purposes here a rough distinction between the two kinds of errors is useful. (We use here the term "error" as meaning "uncertainty", not "mistake"). When nuclear data measurement programs were first developed, errors in the data were mainly statistical and the methodology adopted to deal with errors was adapted to statistical errors. However most of the experimental errors in the more recent measurements are systematic, because the neutron sources used in measurements now, have such high intensities that a large number of counts can be accumulated in a relatively short time, making statistical errors essentially negligible.

Two important characteristics of systematic errors must be noted: (1) these errors induce strong correlations in data obtained by similar techniques and (2) these errors are not necessarily reduced by extending the duration of the measurement or by repeating a measurement without changing the technique.

It is customary to speak of "errors in the measurement," a more appropriate terminology would be "errors in the analysis (or interpretation) of the measurement." Indeed, if a measurement is defined as a set of observed detector "counts", then the measurement is essentially free of errors. The errors, or uncertainties, arise from assumptions or inferences made in deriving a cross section from the observed counts. These assumptions have to do with a multitude of corrections (for background, changes in efficiencies, contaminants, multiple scattering, etc.) and with approximations in fitting the data to a nuclear model. This observation on the nature of errors is important because it follows that if a measurement is accurately described and if the detector "counts" are preserved, the measurement may be reanalyzed with different sets of assumptions. Evaluators usually renormalize old measurements to new values of standards[27] and sometimes reanalyze differential data with new techniques;[46] this is of course less costly than repeating the measurement.

It is not our purpose here to describe in detail the sources of experimental errors and how these errors can be reduced. This subject is discussed in many books and articles dealing with measurement techniques.[47] Two comments on the subject may be relevant to our discussion: (1) It is generally not possible to make the uncertainties associated with a given experimental technique arbitrarily small because some basic assumptions or inferences cannot be completely eliminated, and (2) a good method to improve error estimates and to identify unknown sources of errors is based on comparing the values of a given cross section obtained by different techniques, or conversely, on comparing

measured values for several cross sections, all obtained by similar techniques, with their evaluations.[48] This interrelation between measurements is discussed in the next subsection.

4.5 The Interrelation of Cross Section Measurements

In previous subsections we have indicated how measurements and nuclear models are closely interwoven. Here we want to stress the interdependence of cross section measurements and show how improvements in the accuracy of a particular cross section may be dependent on improvements in a large data base.

Most measurements are relative measurements normalized at some energy. Clearly the accuracy of the data derived from such measurements cannot be better than the accuracy with which the standard and the normalization are known. In addition most measurements require corrections for sample impurities, parasitic reactions in the sample backing or sample holder, changes of detector efficiency with energy and angle. These corrections require an accurate knowledge of a large number of secondary data. A single example will be given here. The value of $\bar{\nu}$ for ^{239}Pu is a fundamental parameter of fast reactor design. This parameter is usually measured relative to $\bar{\nu}$ for spontaneous fission in ^{252}Cf, the standard, so that any uncertainty in $\bar{\nu}$ for ^{252}Cf is reflected in an uncertainty in $\bar{\nu}$ for ^{239}Pu. Recently the sulfur thermal absorption cross section has emerged as a possible major source of uncertainty in manganese bath measurements of $\bar{\nu}$ for ^{252}Cf.[49] This illustration shows how the uncertainty in $\bar{\nu}$ for ^{239}Pu, a parameter of major importance for reactor design, may be related to a lack of knowledge of a "secondary cross section," namely the sulfur capture cross section.

4.6 Planning of Neutron Data Measurements for Nuclear Applications

It follows from the discussion in the previous subsections that to be efficient a program of data measurements should be directed to improve a broad data base, rather than focus narrowly on a specific request for a particular application. This is not intended to imply that all measurements are equally important and relevant but a measurement program should not be too narrowly focused. Other considerations tend to support this point of view. Again this will be illustrated by examples.

If the equipment and techniques are developed to measure the resonance parameters of ^{238}U, because these are of interest for a certain reactor application, it is then efficient to apply the same equipment and techniques to ^{232}Th, a nucleus with characteristics similar to ^{238}U, and it would be unproductive, in the long run, to neglect ^{232}Th because that isotope is not a priority item at the time when the measurement on ^{238}U is completed. It could even be argued that it would be desirable to extend the same technique to the gold cross sections, because gold is a relatively well known standard, and a measurement with gold might be viewed as a validation of the measurement technique.

Since the cost per cross section measured decreases as the number of cross sections measured with the same equipment increases it is efficient in a data measurement program to address as many actual or anticipated "requests" as feasible and even to look for possible theoretical applications of the results of measurements since, as previously stated, improvements in nuclear models are likely to lead to improved data accuracies.

It follows from the above considerations that the cost/benefit analysis of a data measurement program should best be assessed on the basis of

a broad class of long range applied problems rather than on the basis of a narrow specific goal.

4.7 Cross Section Measurement Facilities: ORELA as an Example

Differential cross section measurements require either a monoenergetic neutron source or an intense neutron source with a wide range of energies (white source) which are sorted into energy bins by the time-of-flight technique. Both types of neutron sources have their applications.

Monoenergetic neutron sources can be used with absolute measurement of neutron flux by the associated-particle or associated-activity techniques. They also allow passive detection methods such as the activation technique or the detection of nuclear tracks. Monoenergetic neutron sources have been extensively described in the excellent book of Marion and Fowler.[47]

White neutron sources provide good energy resolution data over a wide energy range, particularly in the resonance region below 20 keV where monoenergetic neutron sources do not exist. Several recent powerful time-of-flight facilities have been dedicated to the acquisition of neutron data. The major such facility in the U.S. is the Oak Ridge Electron Linear Accelerator (ORELA). Very similar facilities are in operation at Geel (Belgium-Euratom), Harwell (UK), Kyoto (Japan), and other places.

Because these facilities play an essential role in neutron data acquisition, the facility, program and contributions of ORELA will briefly be reviewed, as an example.

The ORELA was designed to measure neutron cross sections, especially in the 10 eV to 1 MeV energy range, by the time of flight technique. For the last decade it has been the most powerful and useful pulsed-neutron time-of-flight facility in the world. The reasons for selecting an electron Linac, vs. other pulsed-neutron sources have been reviewed by Maienschein[50] and Harvey and Maienschein.[51] Dabbs[52] has recently described the present facility, the ORELA specifications and performance and the neutron measurement program. Peelle et al.[53] have reviewed the measurements performed during the past decade and future improvements under study. The material in this section is abstracted from these and other sources.[54,55] However, since this report is a survey, the review of the ORELA facility and of the measurement programs are very curtailed and the references should be consulted for a more complete description.

4.8 Brief Description of the ORELA Facility

Table 4.I is a list of the characteristics of the electron accelerator and its neutron production rates. In general all of the specifications shown on the table have been met in actual practice of the largest part of the life of the facility.

Figure 4.1 shows a layout of the facility. The facility comprises the accelerator and its target, 10 evacuated neutron flight paths have 18 measurement stations at flight path distances 8.9 to 200 meters, a complex 4-computer data acquisition system capable of handling some 17000 32-bit "events"/sec. from a total of 12 data input ports. A dedicated PDP-10 timesharing system with a 250 megabyte disk system and 4 PDP-15 graphic display satellites permits on-site data reduction analysis.

ORNL DWG. 66-5210

ORELA SPECIFICATIONS

L-band, 1300 megacycles
10-140 MeV electrons
15 amp peak current (τ <24 nsec)
2-1000 nsec burst width
5-1000 pulses per second
50 kw electron beam (τ >24 nsec)
10^{11} neutrons/pulse (τ >24 nsec, Ta target)
10^{14} neutrons/sec (average; 50 kw)
4×10^{18} neutrons/sec (peak, 15 amps)

Table 4.1. A List of the Characteristics of the Electron Accelerator and its Neutron Production Rates

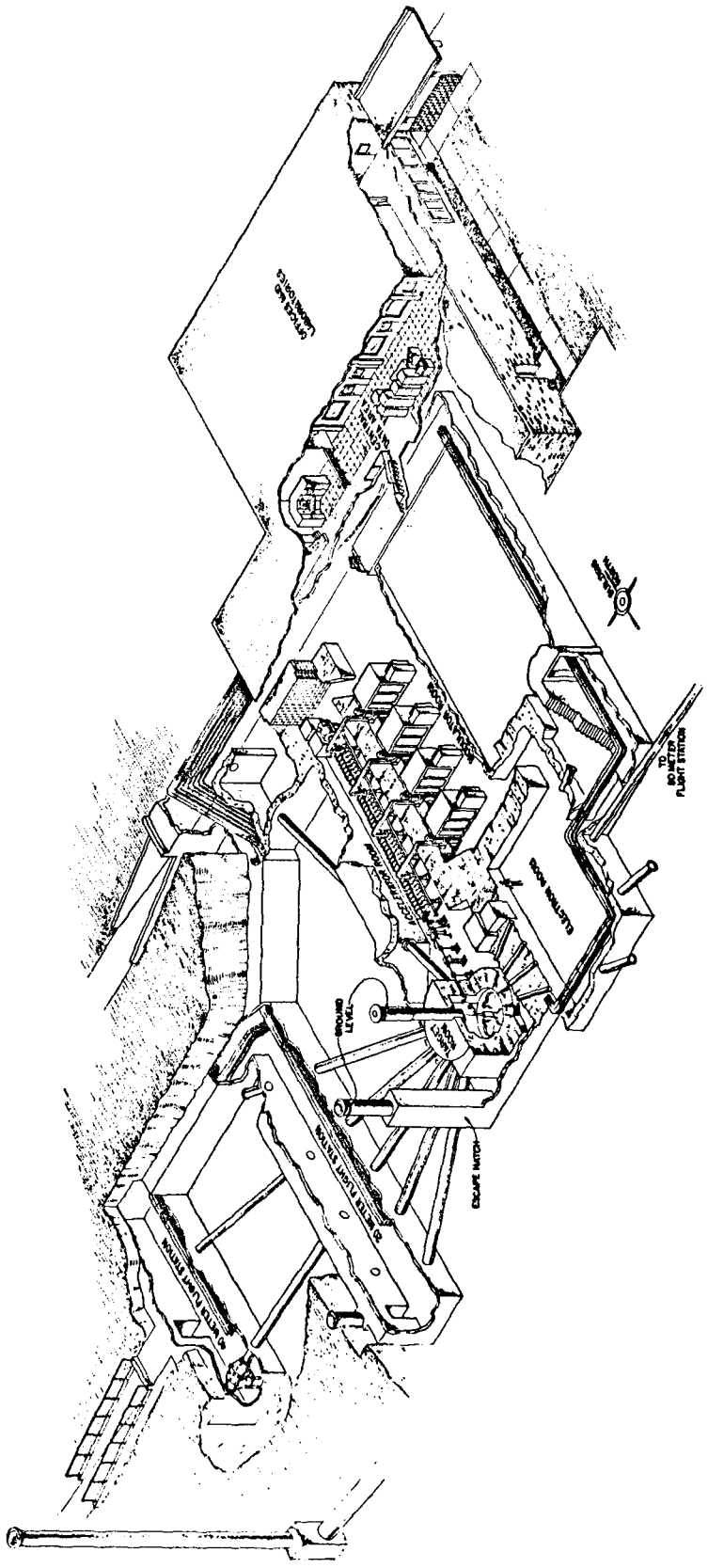

Fig. 4.1. A layout of the ORELA facility.

The staff associated with the cross section measurements and analysis prorgram consists of approximately 30 ORNL scientists. In addition numerous visitors and collaborators from other institutions and universities throughout the world, contribute to this program.

4.9 Measurements for Technological Applications

The dual goals of the ORELA program are to determine neutron cross sections for use in reactor design and other technological applications and in nuclear physics studies. It was shown in another section of this report that these two areas are closely interrelated and cannot always be clearly separated. In this discussion we shall, however, concentrate on those measurements which address directly data requests of the U.S. applied nuclear energy program. It is not our intent to review here all the contributions of the ORELA applied physics program during the past 12 years, but rather to illustrate with selected examples the scope and impact of the ORELA measurements. The results relevant to the applied program are reported annually to the DOE Nuclear Data Committee;[56] these reports should be consulted for a more complete overview of ORELA's contributions.

The most relevant nuclei for the nuclear energy programs are surely the main fissile and fertile isotopes: ^{232}Th, $^{233,235,238}U$, $^{239,240,241}Pu$. Table 4.2 summarizes the measurements on these isotopes which have been performed at ORELA. The table includes also a few measurements, the so called ORNL-RPI measurements, which were done with neutrons from the RPI Linac during the construction of ORELA. These measurements were done with ORNL equipment and were analyzed by ORELA

Table 4.2. Fissile and Fertile Nuclide Measurements at ORELA

Nuclide	Type of Measurement	Neutron Energy Range
^{232}Th	Transmission	.008 eV - 40 MeV
	Fission Cross Section	.7 - 10 MeV
	Capture	20 eV - .8 MeV
	Gamma Production	.2 - 20 MeV
^{233}U	Transmission	.01 -2.0 eV
	Fission Cross Section	.02 eV - .2 MeV
	Capture	.02 - 2000 eV
	$\bar{\nu}$	500 eV - 6 MeV
^{235}U	Fission Cross Section	.01 eV - .2 MeV
	Capture	.01 eV - .2 MeV
	$\bar{\nu}$	.005 eV - 10 MeV
	Spin determination	1 - 25000 eV
^{238}U	Transmission	.5 eV - .1 MeV
	Fission Cross Section	5 eV - 25 MeV
	Capture	1 eV - .1 MeV
	Inelastic scattering	82 keV
^{239}Pu	Transmission	100 eV - .5 MeV
	Fission Cross Section	.01 eV - .2 MeV
	Capture	.01 eV - .2 MeV
	$\bar{\nu}$	.005 eV - 10 MeV
^{240}Pu	Transmission	2.5 eV - 6 MeV
	Fission Cross Section	20 eV - 20 MeV
	Capture	.01 eV - .35 MeV
	$\bar{\nu}$	Spontaneous Fission
^{241}Pu	Fission	.01 - 30 eV
	α	.01 eV - .25 MeV

Table 4.3. Measurements on Minor Transactinides at ORELA

Nuclide	Transmission	Fission	Other
^{231}Pa	.01 - 1000 eV	.5 eV - 12 MeV	
^{234}U		3 eV - 9 MeV	
^{237}Np	.5 eV - 1 MeV		capture, spin determination
^{241}Am		.03 eV - 20 MeV	capture: .01 - 37000 eV
^{242m}Am		.03 eV - 20 MeV	
^{243}Am	.5 - 1000 eV		
^{242}Cm			$\bar{\nu}$, spontaneous fission
^{243}Cm		.03 eV - 10 MeV	
^{244}Cm	.5 - 400 eV		
^{245}Cm		preliminary	
^{248}Cm	.01 - 100 eV		
^{249}Bk	.01 - 100 eV		
^{249}Cf	.01 - 100 eV	preliminary	
^{252}Cf			$\bar{\nu}$, spontaneous fission

staff; some were repeated with improved techniques at ORELA. The measurements listed in Table 4.2 had a decisive impact on the U.S. ENDF/B evaluations; for instance the Version IV and V evaluations of the capture cross section for the fissile isotopes below 25 keV are based exclusively on the measurements listed in the table;[57-60] we have discussed elsewhere in this report the impact of the measurements of the low energy resonance parameters of ^{238}U and of $\bar{\nu}$ for the fissile isotopes.

In addition to the measurements on the main actinides listed in Table 4.2, measurements were done at ORELA on a number of "secondary transactinides." These measurements are summarized in Table 4.3. The cross sections of secondary transactinides are required to compute the radioactivities of irradiated nuclear fuel,[6] but are particularly difficult to measure because of the small quantities available and because of their short half-lives and high specific activities.[61] These measurements are also very useful in providing model parameters for the theoretical calculation of the cross sections which cannot be directly measured,[28] as was outlined in another section.

Peelle et al.[53] have recently listed the measurements on structural, shielding, and fission product nuclides done at ORELA. We reproduce this list in Table 4.4. As may be seen on the table, many of these measurements had very strong or substantial impact on the ENDF/B evaluations.

ORELA measurements to meet fusion energy neutron cross section needs have recently been reviewed by Larson.[62] ORELA has produced the bulk of the available cross section data in the production of gamma rays resulting from neutron induced interactions. Table 4.5 lists the elements on which data have been taken, the angles at which the data were

Table 4.4. Measurements on Structural, Shielding and Fission Product Nuclides at ORELA

Nuclide	Total Cross Section	Capture Cross Section	Gamma Production Cross Section	Other
^{6}Li	10 eV - 20 MeV (1)			(n,α) 70 - 3000 keV (2)
^{7}Li	0.1 - 20 MeV (R)	2.6 keV - 1 MeV		(n,xn) 1 - 20 MeV
^{9}Be(N)		2.6 keV - 1 MeV		
C	0.2 - 80 MeV (2)		Thresh - 20 MeV (1)	
N			Thresh - 20 MeV (1)	(n,p),(n,α) 0.5 - 15 MeV (R)
^{16}O	0.2 - 80 MeV (2)		Thresh - 20 MeV (3)	
F	5 eV - 20 MeV (1)	2.6 keV - 1 MeV (2)	0.6 - 20 MeV (1)	
^{23}Na	.032 - 37 MeV (1)	3 - 600 keV (1)	0.35 - 10.5 MeV (1)	(n,n'γ),.44 - 2 MeV (1) (n,n) 0.5 - 2 MeV (1)
Mg	0.009 - 39 MeV (1)	2.6 keV - 2 MeV (3)	0.6 - 20 MeV (3)	
^{27}Al	0.025 - 80 MeV (2)	2.6 keV - 1 MeV (2)	0.6 - 20 MeV (1)	(n,n'γ), 0.6 - 20 MeV (2) (n,xn) 1 - 20 MeV (2)
Si	0.025 - 80 MeV (2)	2.6 keV - 1 MeV (3)	0.6 - 20 MeV (1)	(n,n'γ), 1.78 - 4 MeV (1)
^{31}P		2.6 keV - 1 MeV (R)		
S		2.6 keV - MeV (R)		(n,α) 10 - 700 keV (R)
Cl		2.6 keV - 1 MeV (R)		
Ca	0.2 - 80 MeV (2)	2.6 keV - 1 MeV (2)	0.6 - 20 MeV (1)	
Ti	0.02 - 30 MeV (R)	2.6 keV - 1 MeV (3)	0.3 - 20 MeV (R)	(n,xn) 1 - 20 MeV (R)
V		2.6 keV - 1 MeV (3)		

Table 4.4. (cont'd.)

Nuclide	Total Cross Section	Capture Cross Section	Gamma Production Cross Section	Other
Cr	0.2 - 20 MeV (1)	2.6 keV - 1 MeV (2)	0.6 - 20 MeV (2)	
^{55}Mn		2.6 keV - 1 MeV (2)		
Fe	0.2 - 80 MeV (1)	0.4 - 1 MeV (1)	0.6 - 20 MeV (2)	$(n,n'\gamma)$, up to 2 MeV (2) (n,n) 0.04 - 20 MeV (1)
^{59}Co		2.6 keV - 1 MeV		
Ni	0.2 - 80 MeV (2)	2.6 keV - 1 MeV (R)	0.6 - 20 MeV (1)	
Cu	0.2 - 80 MeV (1)	2.6 keV - 1 MeV (R)	0.6 - 20 MeV (1)	(n,xn) 1 - 20 MeV (2)
^{86}Kr		2.6 keV - 1 MeV (R)		
Zr		2.6 keV - 1 MeV (2)		
^{93}Nb	0.025 - 30 MeV (R)	2.6 keV - 1 MeV (2)		(n,xn) 1 - 20 MeV (R)
Mo	0.025 - 80 MeV (R)	2.6 keV - 1 MeV (2)		
Ru		2.6 keV - 1 MeV (N)		
^{103}Rh		2.6 keV - 1 MeV (R)		
Pd		2.6 keV - 1 MeV (R)		
Cd		2.6 keV - 1 MeV (2)		
Sn	1 - 80 MeV (R)			
Te		2.6 keV - 1 MeV (N)		
I		2.6 keV - 1 MeV (N)		
^{133}Cs		2.6 keV - 1 MeV (R)		
Ba		2.6 keV - 1 MeV (2)		
La,Ce,Pr,Nd		2.6 keV - 1 MeV (N)		

Table 4.4. (cont'd.)

Nuclide	Total Cross Section	Capture Cross Section	Gamma Production Cross Section	Other
^{181}Ta		2.6 keV - 1 MeV (R)	0.6 - 20 MeV (1)	
$^{182,3,4,6}W$		2.6 keV - 1 MeV (R)	0.6 - 20 MeV (1)	
^{197}Au	0.025 - 80 MeV	2.6 keV - 1 MeV (1)		
Pd	1 - 80 keV	2.6 keV - 1 MeV (1)	0.5 - 20 MeV (1)	
^{206}Pb	15 - 900 keV (R)			
^{207}Pb	10 - 500 keV (R)			
^{231}Pa	0.5 - 100 eV (R)			
^{243}Am	0.5 - 1000 eV (R)			
^{244}Cm	0.5 - 400 eV (R)			
^{248}Cm	0.5 - 100 eV (2)			
^{249}Bk	0.5 - 100 eV (R)			
^{249}cf	0.5 - 100 eV (R)			

*Numbers in parentheses indicate approximate impact on ENDF/B according to the following:
(1) Very stong impact.
(2) Substantial impact.
(3) Weak impact.
(R) Data too recent to be included in evaluations.
(N) No ENDF evaluation at present.

Table 4.5. ORELA (n,xγ) Measurements

Element	90°	125°	Used in ENDF/B-V	ORNL Report
Li		X	N	TM-4538
C	X	X	Y	TM-3702
N	X	X	Y	ORNL-4864
O		X	N	ORNL-5575
F		X	Y	TM-4538
Na		X	Y	TM-6281
Mg	X	X	Y	TM-4544
Al	X	X	Y	TM-4232
Si	X	X	Y	TM-4389
Ca		X	Y	TM-4252
Ti		X	N	TM-6323
V		X	Y	TM-5299
Cr		X	Y	TM-5098
Mn		X	Y	TM-5531
Fe		X	Y	TM-5416
Ni		X	Y	TM-4379
Cu		X	Y	ORNL-4846
Zn		X	N	TM-4464
Nb	X		N	TM-4972
Mo		X	N	TM-5097
Ag		X	N	TM-5081
Sn		X	N	TM-4406
Ta	X	X	Y	TM-3702
W		X	Y	ORNL-4847
Au		X	N	TM-4973
Pb		X	Y	TM-4822
Th		X	N	TM-6758

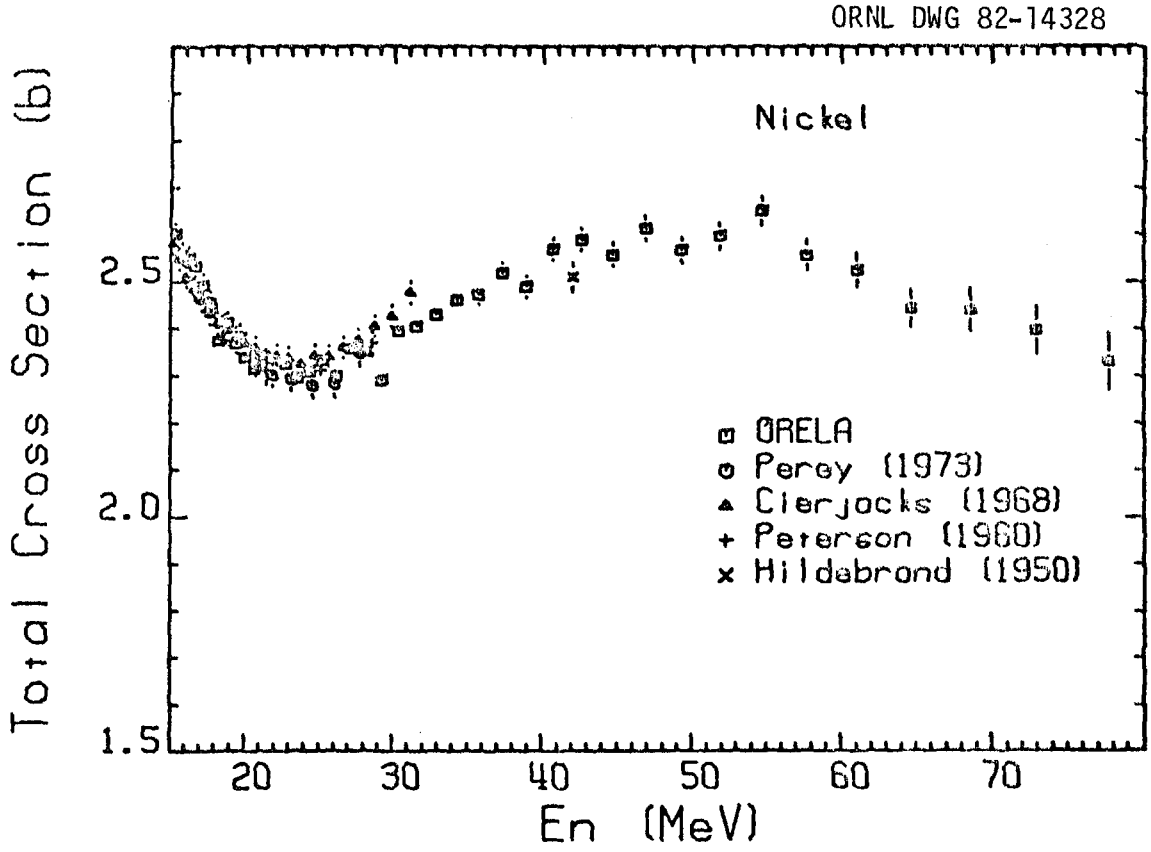

Fig. 4.2. Comparison of ORELA total cross section data for nickel with results of other measurements.

acquired, whether or not the data have been utilized in the ENDF/B-V evaluation for the material, and the ORNL report number in which the measurement is described and numerial values of the data are given.

ORELA was initially designed to cover the neutron energy range from 10 eV to 1 MeV, however recent developments in fast timing techniques and fast neutron detectors have permitted great extension of this range. A number of total cross section measurements were recently completed, in support of the shielding design needs for the FMIT project. These measurements extend to 80 MeV and thus greatly extend the range where data are available. As an illustration, Fig. 4.2 shows a comparison of the ORELA data for nickel with the other data from the CSISRS file.[62]

4.10 Conclusions

In this section we have discussed some of the factors that should be considered in selecting a program of measurements. We have indicated that such a program should be directed towards a general improvement of a data base relevant to many different technological applications and even to improvements of nuclear models. We have shown that measurement of different data are closely interrelated and that the interpretation, evaluation and representation of the results of measurements is intimately interwoven with the development of nuclear models.

We suggest that it is cost effective to address improvements of the data base in general, and to concentrate on those measurements for which the equipment at hand and the available technology are most likely to lead to improved accuracies of relevant data.

The validity of the approach suggested here has been generally recognized by data measurers and funding agency. The programs of all the important experimental facilities and the ORELA program in particular, are addressing the dual goals to determine neutron cross sections for use in reactor design and in basic nuclear physics studies.[50-53] The general improvement of an extended data base is embodied in the development of an application independent evaluated nuclear data file.[27]

Finally we suggest that the best prospect for improving the data base for reactor design is in a close interaction between the data measurer, the evaluator and the user. Such a cooperation will increase the probability that most of the information obtained from a measurement is being used, and correctly used in the design.

5.0 STATUS OF SELECTED EVALUATED NUCLEAR DATA

5.1 Introduction

In the previous section we have discussed several aspects of the measurement and evaluation of high accuracy differential nuclear data for technological applications. In particular we have commented on the general nature of data uncertainties and on the interrelation of different cross-section measurements. In this section we propose to review the present status, in relation to accuracy requirements, of some specific evaluated nuclear data important for technological applications, particularly for the design and calculation of thermal and fast reactor performance parameters.

A precise numerical comparison of achieved versus required accuracies of nuclear data is beyond the scope of this paper, since, as indicated in the next subsection, the characterization of evaluated nuclear data uncertainties is a complicated matrix which cannot be reduced to a few numbers without severe oversimplification. Furthermore, such a comparison would not be very meaningful because the presently available information on uncertainties is very incomplete and often unreliable. In addition, as was stated in Section 2, the accuracy requirements for the prediction of reactor properties are a matter of judgment and there are different ways of meeting these requirements, including the use of integral measurements;[6] the magnitude of the accuracy requirements also depends on the correlations in the evaluated data, and these correlations are often poorly known.

In the next two subsections we briefly expand on the characterization of evaluated nuclear data and nuclear data requirements. In the subsequent

three subsections we summarize qualitatively some perceived outstanding data problems of ENDF/B-V.

5.2 Characterization of Evaluated Data Uncertainties

The methodology for the estimation and representation of uncertainties in evaluated nuclear data is still very much under development.[63] An excellent review and discussion of the subject can be found in the recent article "Uncertainty in Nuclear Data Used For Reactor Calculations" by R. W. Peelle.[64]

The ENDF/B-V uncertainty covariance files have been processed into multigroup covariance matrices for many of the important reactions.[65,66] For a detailed characterization of the estimated uncertainties in the evaluated data and their correlations, these covariance matrices should be examined, remembering however that the field is still in its infancy: ENDF/B-V is the first version of ENDF/B for which extensive covariance files are available, but these files have not been systematically validated and are often incomplete and of dubious quality. As an example of the complexity of details of the covariance data, we show in Fig. 5.1 a 52-group standard deviation and correlation matrix description of the $^{240}Pu(n,\gamma)$ reaction.[66]

5.3 Nuclear Data Accuracy Requirements for Technological Applications

Nuclear data accuracy requirements for technological applications are determined by the accuracy requirements for the prediction of reactor performance parameters and the sensitivity of these parameters to the nuclear data. The Nuclear Data Section of the International Atomic Energy

ORNL DWG 82-14329

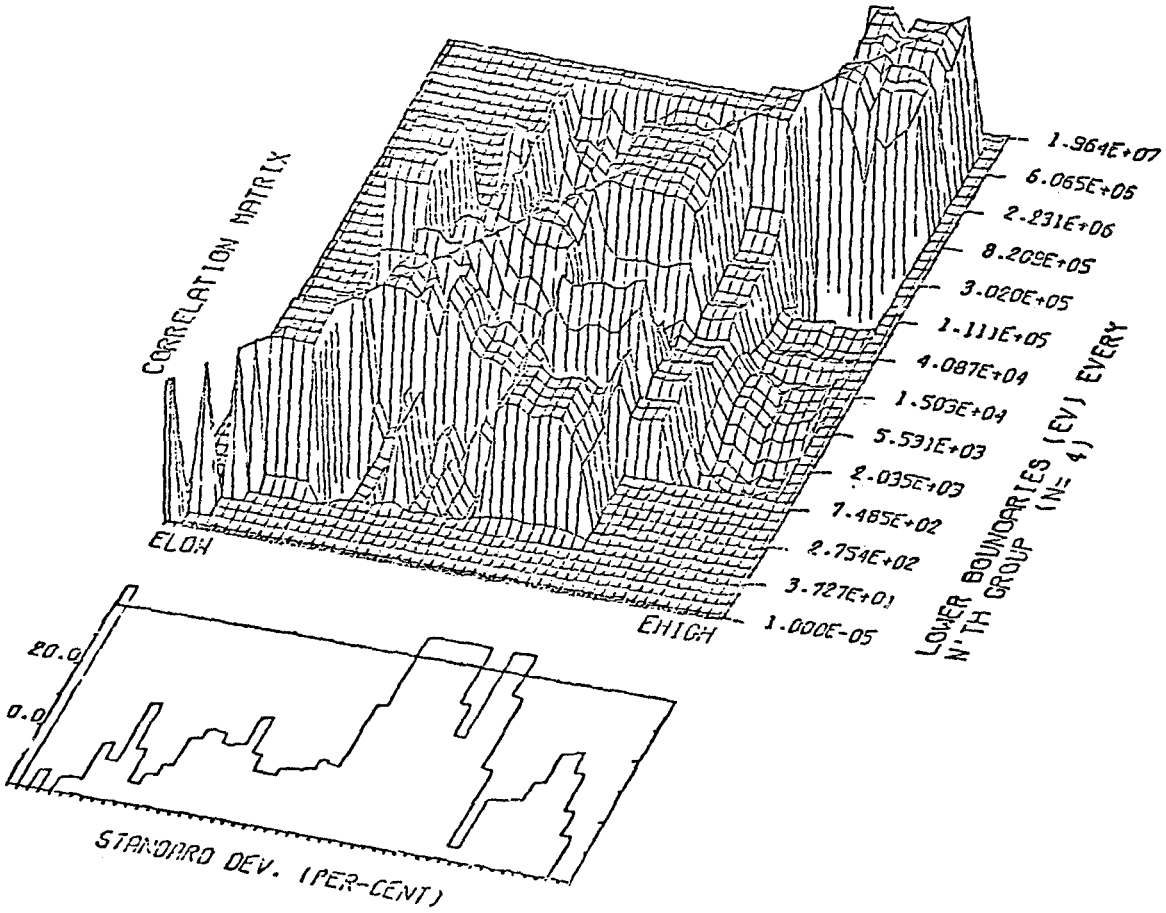

Fig. 5.1. A 52-group standard deviation and correlation matrix description of the $^{240}Pu(n,\gamma)$ reaction.[66]

Agency maintains a World Request List for Nuclear Data (WRENDA)[67] which is updated every other year and which is intended to serve as a guide to experimentalists, evaluators, and administrators when planning nuclear data measurement and evaluation programs. The U.S. Department of Energy Nuclear Data Center also publishes a "Compilation of Requests of Nuclear Data;"[68] the purpose of this compilation is to summarize the current needs of U.S. Nuclear Energy Programs and other applied technologies for nuclear data. The compilation is the result of a biennial review (the responsibility for this activity has recently been directed to CSEWG) in which the DOE and contractors, and other interested groups have been asked to review and revise their requests for nuclear data. These request lists define priorities, energy range, accuracy desired and also indicate the sponsor of the request and the purpose. These request lists are, of course, authoritative with data measurers and evaluators. As such effort is now underway in CSEWG to make them even more precise and meaningful.

In a recent and exellent review paper on Nuclear Data for Reactor Design, Operation, and Safety, Rowlands[6] summarizes the data requirements for different categories of applications. In enumerating the sources of nuclear data requirements, Rowlands stresses the fact that although the use of integral measurements to meet certain nuclear data needs has been widely adopted, one would like the prediction of safety related parameters to be based on differential nuclear data with confirmatory integral measurements. There is also a natural desire for a sound understanding of the fundamental nuclear physics underlying reactor operation and a need for the resolution of discrepancies found when evaluating nuclear data and when interpreting integral measurements.

As an illustration, typical accuracy requirements for the principal actinides, as summarized by Rowlands, are shown in Table 5.1. The accuracies shown in that table are for the data averaged over the relevant neutron spectrum.

As discussed in this and the previous subsections, the estimated uncertainties of evaluated nuclear data and the accuracy requirements cannot be characterized with a few numbers without drastic oversimplification. In order to illustrate the present status of nuclear data it seems more appropriate to list a few selected problems which are still perceived to exist, with the latest U.S. evaluation, ENDF/B-V. This is done in the next subsections.

5.4 The Status of Nuclear Data for Thermal Reactor Calculations

In previous ENDF/B evaluations, including ENDF/B-IV, the major problem concerning thermal reactor applications was the inability of the data to predict observed ^{238}U resonance capture rates in critical lattice experiments.[69,70] With ENDF/B-V the quality of the ^{238}U resonance parameters has been substantially improved and the longstanding ^{238}U resonance capture overprediction is much reduced. As discussed in Section 6 the difference between computed and measured values of the ratio of epithermal to thermal capture in ^{238}U, ρ^{28}, for the TRX-1 and TRX-2 assemblies is now covered by the uncertainties assigned to the comparison. Calculations based on ENDF/B-V data are in good agreement with measurements, both for K_{eff} and for integral parameters of ^{235}U - ^{238}U - H_2O assemblies.[71] The designer requirements of a ± 1 meV accuracy

Table 5.1. Accuracy* Requirements for the Cross Sections of the Primary Transactinides

PRINCIPAL ACTINIDES

Fissile Isotopes
^{233}U, ^{235}U, ^{239}Pu, ^{241}Pu
Fertile Isotopes
^{232}Th, ^{238}U, ^{240}Pu

Typical Requirements for the Thermal Region
$\nu \pm 0.3\%$; $\nu \pm 0.5\%$;
σ_f, σ_c or α, σ_t, σ_s $\pm 1\%$

Typical Requirements for Fast Reactor Spectrum Region
(and Fast Resonance Regions for Thermal Reactors)

Parameter	Fissile Isotopes	Fertile Isotopes
ν_f	0.3%	1%
σ_f	2%	2%
σ_c or α	4%	3% (1.5% in ratio to ^{239}Pu σ_f)
σ_t, σ_s	2%	5%
σ_{in}	10%	5%
σ_n, 2n	10%	10%
Resonance Parameters	10%	3%
ν delayed	3%	5%

*The accuracy refers to the cross section averaged over an appropriate neutron energy spectrum. The requirements arise primarily for the prediction of k_{eff} and breeding.

in the capture widths of the first few ^{238}U large levels, and of 2 to 5% accuracies in their neutron widths are considered to be met.[72]

An area where discrepancies are still perceived to exist (see Section 6) in the benchmark testing of ENDF/B-V data for thermal reactors is in the epithermal value of α (capture to fission ratio) for ^{235}U, where calculations yield a value lower than direct integral measurements.[71,73] Perceived inadequate predictions of the moderator temperature coefficient of reactivity in LWR suggest improved measurements and evaluations of the shape of the capture cross section of ^{238}U and of eta for ^{235}U and ^{239}Pu in the thermal region (below 1 ev).[74,75] The need for improved uranium data in the thermal region is also suggested by a persisting 2% inconsistency between the evaluated 2200 m/s ^{233}U and ^{235}U cross sections and the cross sections averaged over a Maxwellian neutron spectrum.[75,76]

5.5 The Status of Nuclear Data for Fast-Breeder-Reactor Applications

The contemporary status of nuclear data for fast breeder reactors has been concisely and clearly summarized in a recent paper by L. LeSage.[77] Because this characterization of the status is brief and complete we quote it here in its entirety:

"Uncertainties in basic nuclear data continue to be the largest cause of uncertainty in predictions of the physical performance of FBR systems." Differential data issues are characterized by both specific matters of precision and more general questions of physical concept. The status and recent trends of some of the most FBR-relevant basic-nuclear-data areas are outlined below.

Standard constants: The key quantities are $^{235}U(n,f)$ and $^{252}Cf(nubar)$ which, together, essentially govern the neutron-source term. $^{235}U(n,f)$ remains uncertain by 2-3%, implying larger uncertainties in all other fission cross sections. $^{252}Cf(nubar)$ has recently varied by 0.5-1.0% and with it essentially all actinide nubar values. The trend in $^{235}U(n,f)$ is downward but partly compensated for by a trend toward increased $^{252}Cf(nubar)$ values. In order to significantly improve the situation several independent measurements to accuracies of 0.5-1.0% in the case of $^{235}U(n,f)$ and 0.2-0.3% for $^{252}Cf(nubar)$ are needed.

Actinide nuclides: Recent experimental results indicate increased neutron total cross sections in the unresolved resonance region, in some cases (e.g., ^{238}U) by 5-10%. Recent measurements of neutron inelastic scattering from fertile nuclides have led to generally larger cross-section values while at the same time changing the energy transfer matrix in a manner that makes the larger cross sections more acceptable in integral calculations. It remains very difficult to measure fissile-nuclide inelastic cross sections and, consequently, the results are uncertain. However, fissile-nuclide elastic-scattering cross sections are now determined with good accuracies that suggest lower inelastic scattering cross sections relative to those generally employed in FBR calculations. Fission cross sections of prominent actinides are known relative to those of ^{235}U to 1-2% accuracies, thus the ^{235}U reference standard remains the governing factor. Fast-neutron capture cross sections of fertile nuclides are known to 5-8% and, in the case of ^{238}U, remain larger than suggested by integral studies. Nubar uncertainties

are dominated by those of the reference standard, as noted above. The most recent evaluations have generally increased the average fission-neutron energies although there are large uncertainties in these data. Knowledge of delayed-neutron emission from specific precursors is much improved but the average delayed-neutron yields have not markedly changed. Recent measurements have provided new transplutonium data, notably neutron total and fission cross sections, and the corresponding evaluated data sets are improving. The latter make extensive use of theoretical extrapolations that probably are sufficiently accurate for most FBR use.

Structure and coolant nuclides: In this area the problems involve the measured data, the evaluation and the calculational usage. Uncertainties are reflected in the wide variation between group-transport cross sections as derived in various national programs. Experimental knowledge of resonance cross sections has very much improved particularly with respect to radiative capture. This improvement is not necessarily reflected in the evaluated files due to physically inappropriate representations. The quality of the basic data in the unresolved resonance region is variable. Measurements have often emphasized resolution to the detriment of absolute magnitude and self-shielding effects have frequently been ignored with consequent distortion of 10% or more in some cases. Inelastic neutron-scattering cross sections in this region are generally large and measurements are now providing accuracies approaching 5%.

Fission products: A data base of wide scope is available. However, for applications sensitive to fission-product data, there is concern for data quality. The fission products are neutron-rich nuclides characterized by inversion of the shell structure and/or large static deformations. They are usually very radioactive and thus the data are largely obtained by theoretical extrapolation from measured values in neighboring regions. This shortcoming is probably of minor note but fast-neutron capture is a more serious concern and discrepancies of factors of 2-5 in fission-product capture cross sections are known to exist and are supported by fragmentary integral results. It is not reasonable to soon expect large improvements in the measurement of active fission-product cross sections. It is reasonable to look forward to better knowledge of the stable cases with improved theoretical extrapolation to the active products."

5.6 Need for Accurate Uncertainty Covariance Files

Because differential cross sections measurements do not have sufficient accuracy to meet reactor design requirements, evaluated data are often adjusted using the results of integral benchmark experiments.[78,79] This adjustment process requires reliable and complete uncertainty covariance data for both the evaluated differential cross sections and the integral measurements. The ENDF/B-V covariance files are often incomplete and generally perceived as inadequate (current effort is directed toward improving the situation).[78] This is particularly true in the resonance regions where appropriate formats have not yet been developed.[75,80]

Poenitz[33] has recently suggested that in order to improve the accuracy and the uncertainty assessment of the important actinide cross sections, a cost effective approach would be to make a new generation of measurements usable to test the significance of prior knowledge. For $^{235}U(n,f)$ and $^{238}U(n,\gamma)$ measurements with less than 1% and 2% uncertainty would be needed.

5.7 Comparison Between Selected ENDF/B-IV and ENDF/B-V Data

The accuracies in neutron cross sections discussed in the previous subsections are the result of a continuous effort to improve measurement and evaluation techniques.[6] In the U. S. this effort results in the generation of the general purpose Evaluated Nuclear Data File which is updated every few years. The two latest versions of ENDF/B, version IV and version V, were released in 1975 and 1979 respectively. The changes between version IV and version V illustrate the impact of recent measurements and recent evaluations on our knowledge of the nuclear data. A list of all the changes between the two ENDF/B versions would be tedious and beyond the scope of this paper, but a discussion of a few selected changes provides additional insight on the present status of the nuclear data.

The data changes which are important for fast reactor applications are very well summarized in the quotation of L. LeSage[77] given in subsection V.E. Some changes from ENDF/B-IV to ENDF/B-V which are expected to have the greatest impact on thermal reactor applications have been listed by O. Ozer:[81]

- Reduced capture widths for the low-energy resonances of ^{238}U.
- Revised thermal-energy-range cross sections for ^{235}U and ^{232}Th and increased capture and neutron widths for ^{240}Pu.
- Higher estimates for the number of neutrons produced in a fission event for all fissile nuclei.
- New evaluations of the amounts of energy released in a fission event for 16 fissionable nuclei.
- New data files for structural or control materials, such as zirconium, hafnium, and gadolinium.
- Considerably expanded data files for the heavy actinide nuclides of interest to studies of burnup, alternate cycles, and waste management.

We shall now discuss in more detail some data included in the first three items of this list.

A. Change in the ^{238}U Resonance Parameters

As previously stated, the reevaluation of the parameters of the first three important levels of ^{238}U has greatly improved the calculation of the ^{238}U capture rate in thermal critical lattices.[70-72] The ENDF/B-IV and ENDF/B-V values of the parameters of the first few levels are compared in Table 5.2. As can be seen the changes in the neutron widths of the levels at 20.9 and 36.7 eV and in the capture widths of the first three levels exceed considerably the present uncertainty estimates. The ENDF/B-V values are consistent with other recent evaluations,[82-84] and the changes from ENDF/B-IV result from a series of new precise measurements at ORELA[85] and other laboratories[86,87] and from a reevaluation of the results of older measurements.[88,89]

Table 5.2. Comparison of Evaluated ^{238}U Neutron and Capture Widths

E_0 (eV)	6.67	20.9	36.7	66.0	80.7	102.5	10.2
			Neutron Widths (meV)				
BNL-325(65)	1.52 ±.02	8.5 ±.5	31.0 ±.9	25.0 ±1.2	2.0 ±.2	68.0 ±3	.0014
BNL-326(73)	1.52 ±.02	8.7 ±.5	32.0 ±1	26.0 ±1.5	2.0 ±.2	70.0 ±3	.00156
MOXON(74)	1.510±.009	8.9 ±.175	31.6 ±.5	24.0 ± .4	1.96 ±.07	70.8 ±.4	.00156
ENDF/B-IV(75)	1.50	8.8	31.1	25.3	2.0	71.0	.00156
ENDF/B-V(77)	1.510±.015	10.12±.10	33.9 ±.4	24.6 ± .4	1.91 ±.04	71.6 ±.4	.00167
KEDAK(81)	1.495	9.94	33.64	24.61	1.907	71.64	.001674
JENDL-2(81)	1.50	10.1	33.3	24.9	1.87	70.9	.00159
BNL-325	1.50 ±.02	10.04±.20	34.0 ±.4	24.0 ±1.0	1.9 ±.1	70 ±3	.00165
			Capture Widths (meV)				
BNL-325(65)	26.0 ±2	26.0 ±4	26.0 ±4	24.0 ± 2	21.0 ±15	24.0 ±3	
BNL-325(73)	26.0 ±2	25.0 ±3	25.0 ±2	22.0 ± 2		26.0 ±2	
MOXON(74)	26.9 ±.37	25.7 ±1.0	26.55±1.20	23.56± .76	21.17±8.9	25.78±.94	
ENDF/B-IV(75)	25.6	26.8	26.0	23.5	23.5	26.0	23.5
ENDF/B-V(77)	22.5 ±.6	23.1 ± .5	22.9 ±.3	23.7 ± .3	24.2 ±1.2	24.4 ±.3	23.5
KEDAK(81)	23.7	23.67	23.64	23.69	24.17	24.41	23.5
JENDL-2(81)	23.7	23.0	23.5	22.9	24.4	25.1	23.6
BNL-325(81)	22.8 ±.6	23.5 ±.8	23.5 ±.3	23.6 ± .1	25 ±1	24.5 ±.7	

Note the high degree of consistency in the evaluations done after 1976 and the significant changes form the older evaluations, particularly in the capture widths of the first three levels listed and in the neutron widths of the levels at 20.9 and 36.7 eV.

Another improvement in the ^{238}U resolved resonance evaluation is in the use of a multilevel formalism, versus a single level formalism in Version IV. The need for a multilevel formalism was first established by a comparison between calculated and measured transmission data obtained with ORELA.[90] It has since been confirmed by careful analyses of several other measurements.[86,91]

A third change between the Version IV and V of the ^{238}U parameters is an increase of from 10 to 20% in the strength function between 1.5 and 4 keV. This change is suggested by the analysis of careful new measurements;[92] it results in an increase of the $^{238}U(n,\gamma)$ infinitely dilute cross section, but not necessarily of the self-shielded cross section, as is illustrated in Table 5.3. In fact the observation from the Table, that the changes in the shielded group constants are not proportional to changes in the dilute group constants, not even of the same sign, stresses the importance of an accurate determination of the ^{238}U resonance parameters in the keV range.

B. Changes in the ^{235}U Fission Cross Section

The most important changes between the ENDF/B-IV and ENDF/B-V $^{235}U(n,f)$ cross sections are reviewed by Bhat,[93,94] and occur at high energies, between 12 and 20 MeV. However, the energy region where uncertainties are greatest and where more work is sorely required is the resonance region. This is illustrated (1) by the inability of evaluators to perform a reliable normalization of the high energy data to the thermal data,[93] and (2) by the discrepancy between the fission and alpha resonance integrals computed from the evaluated cross sections and their

Table 5.3. Comparison of ENDF/B-IV and V $^{238}U(n,\gamma)$ Group Cross Sections Over the Resolved Energy Range

Group	EL, EH eV	Dilute[1] IV b	Dilute[1] V b	Shielded[2] IV b	Shielded[2] V b	$\frac{V - IV}{IV}$ Dilute	$\frac{V - IV}{IV}$ Shielded
1	.4 - 100	45.55	45.65	1.700	1.636	+0.22	-3.8
2	100 - 170	23.80	22.91	1.220	1.235	-3.7	+1.2
3	170 - 230	11.46	10.96	.8472	.8867	-4.4	+4.7
4	280 - 450	3.578	3.489	.6686	.6754	-2.5	+1.0
5	450 - 750	3.538	3.521	.7720	.8120	-0.48	+5.2
6	750 - 1230	2.692	2.777	.8167	.8066	+3.2	-1.2
7	1230 - 2040	1.753	1.774	.7037	.6889	+1.2	-2.1
8	2040 - 3360	1.352	1.401	.7541	.7874	+3.6	+4.4

(1) $$\frac{\int_{EL}^{EH} \sigma_{n\gamma} \frac{dE}{E}}{\int_{EL}^{EH} \frac{dE}{E}}$$

(2) $$\frac{\int_{EL}^{EH} \frac{\sigma_{n\gamma}}{\sigma_t + \sigma_0} \frac{dE}{E}}{\int_{EL}^{EH} \frac{1}{\sigma_t + \sigma_0} \frac{dE}{E}} \quad (\sigma_0 = 10 \text{ b})$$

Table 5.3 is a comparison of infinitely dilute and strongly self-shielded (σ_0 = 10 b) group cross sections computed with ENDF/B versions IV and V. The comparison is over a somewhat arbitrary 8 group structure covering the resolved range. The values were obtained by R. Q. Wright, at ORNL, using MAT 1262 and 1398, respectively. The two last columns of the table indicate that the differences between version IV and version V are typically a few percent for either unshielded or shielded group constants. However, it is important to note that the changes in the shielded group constants are not proportional to the changes in the dilute group constants (the signs are not even the same in most cases). This indicates that an overprediction of the ^{238}U capture in strongly self-shielded critical lattices does not necessarily imply that the evaluated infinitely dilute capture cross section is too high.

direct integral measurements.[73] The values of the resonance integrals obtained with Version IV data, Version V data and by direct measurements are compared in Table 5.4. The discrepancy, as can be seen is not significantly improved with Version V. In view of the importance of the ^{235}U resonance integrals, Hardy[73] recommends a reevaluation of the ^{235}U resonance cross sections. A similar recommendation has been made, on somewhat different grounds, by several authors.[95,96]

Table 5.4. Comparison of Computed and Measured ^{235}U Resonance Integrals*

	I_f(b)	I_γ(b)	α
ENDF/B-IV	282.0	137.7	.488
ENDF/B-V	281.7	139.2	.494
Experimental	274 ± 5	140.6	.513 ± .015

*All these values refer to a low energy cut-off of 0.5 eV. The experimental value of I_f is a weighted average of eight measurements, the experimental value of α is a weighted average of three measurements. The experimental value of I_c is derived: $I_c = \alpha I_f$. For more detail see the comments of J. Hardy, Jr. in ENDF-300(1979).

C. Reevaluation of $\bar{\nu}$ of ^{252}Cf (Spontaneous Fission)

The spontaneously fissioning ^{252}Cf value of $\bar{\nu}$, the average number of neutrons emitted per fission is the standard from which $\bar{\nu}$ values for the fissile nuclei are determined by ratio measurements. The discrepancy among measured values of $\bar{\nu}$ for ^{252}Cf has long constituted one of the most vexing problems in the field of nuclear data interpretation.[97] This

discrepancy is not yet fully resolved but the status of our knowledge of $\bar{\nu}$ has been significantly improved by (1) a reevaluation of older ANL[97] and UK[98] measurements, (2) the recent completion of the most accurate measurement of $\bar{\nu}$ done so far, by R. Spencer[99] at ORELA and (3) the investigation by R. Gwin[100] at ORELA of a foil thickness effect in $\bar{\nu}_p$ ratio measurements; this effect does not affect the ^{252}Cf $\bar{\nu}$ value per se but it directly affects the derived values of $\bar{\nu}$ for the fissile nuclei, as well as the indication of a discrepancy between $\bar{\nu}$ and η.[101] Although additional work on $\bar{\nu}$ is still required, the recent developments just described have resulted in an increase of the evaluated value of ^{252}Cf $\bar{\nu}$ from the Version IV value of 3.757 to the Version V value of 3.766. This 0.24% increase is reflected on the value of $\bar{\nu}$ for the fissile nuclei and is bound to have a significant impact on thermal as well as fast reactor calculations.

5.8 Conclusions

In this section we have commented on the status of some selected evaluated nuclear data for technological applications. The discussion was limited to a few important actinide nuclei and was purely illustrative. A more complete review should include the cross sections of standards, structural materials, fission products, heavy actinides, and so on. We have also examined only one application: the calculation of reactor performance parameters. Review of the status of nuclear data for reactor operation and safety and for the nuclear fuel cycle have been presented by J. L. Rowlands[6] and M. Becker and D. R. Harris.[2] The status of shielding research and the needs of data for reactor shielding have been reviewed

by F. C. Maienschein[102] and by J. Butler.[5] Data needs for the fusion program and for alternate fuel cycles are discussed elsewhere in this paper.

From the brief illustrative discussion in this section it seems clear that although considerable progress was recently achieved in reducing the uncertainties in nuclear data, further work is still needed. For thermal reactors longstanding problems may have been resolved with ENDF/B-V, yet substantial savings in fuel costs can still be obtained from further improvements.[2] For fast reactors, "uncertainties in basic nuclear data continue to be the largest cause of uncertainty in predictions of the physical performance."[77]

6.0 ANALYSIS OF BENCHMARK TESTING

6.1 Objective

CSEWG data testing is primarily intended to assess the adequacy of ENDF/B nuclear data for use in nuclear design and applications. The purpose of this discussion is to:

- Describe the results of the benchmark testing of ENDF/B-V
- Identify areas of basic nuclear data and computational methods and integral experiments that may require additional research

The text draws directly from the extensive work of the chairmen and subcommittees of the Data Testing and Applications Committee of the Cross Section Evaluation Working Group. The summary presented in this section was prepared by C. R. Weisbin based upon the ENDF/B-V Data Testing Reports of R. D. McKnight (Fast Reactor Data Testing), J. D. Hardy, Jr. (Thermal Reactor Data Testing), R. W. Roussin (Shielding Data Testing), R. E. Schenter (Fission Product and Actinide Data Testing), Ben Magurno (Dosimetry and Special Applications), and their respective subcommittees. The full text of their reports appear in the two-volume edition of Ref. 103; only the highlights are given here with references added for those less familiar with the CSEWG Data Testing process.

References to specific experiments are given in the text below. However, a current compendium of CSEWG Benchmark experiments and specifications can be found in Ref. 104.

6.2 Fast Reactor Data Testing

A. Integral Experiments Analyzed

Data testing of ENDF/B-V focused on the analysis of five benchmark integral experiments: JEZEBEL,[105-109] GODIVA,[105-109] ZPR-6/7,[110-112] ZPR-6/6A, and ZPR-9/31[113-119] measurements including criticality, central reaction rate ratios and central reactivity worths. JEZEBEL is a bare sphere of plutonium metal; GODIVA is a bare sphere of enriched uranium metal. The single region, simple geometry, and uniform composition of these assemblies greatly facilitates calculational modeling and data testing. ZPR-6 Assemblies 6A and 7 each have large uniform cores (of approximately the same volumes) with identical single drawer core unit cell loadings with the simple exchange of ^{239}Pu (Assembly 7) for ^{235}U (Assembly 6A) in the fuel column. ZPR-9 Assembly 31 is an intermediate sized (~1000 liter) fast critical assembly with mixed (Pu, U) fuel and graphite, built to provide information regarding carbide fueled LMFBR designs. These assemblies are relatively simple in overall geometry and composition, and were deemed most suitable to test current fast reactor methods and nuclear data. For purposes of this discussion standard deviations in these measurements are typically:[104]

Criticality (k_{eff})	± 0.1-0.3%
Reaction Rate Ratios*	
$^{25}f/^{49}f$	± 2%
$^{28}f/^{49}f$	± 2-5%
$^{28}c/^{49}f$	± 2-5%
Reactivity Coefficients	± 1-5%

*The notation $^{25}f/^{49}f$ corresponds to the averaged fission rate ratio $\langle\sigma_f(^{235}U)/\sigma_f(^{239}Pu)\rangle$ in the core central location.

B. Current Limitations in the Data Testing Process

1. Quantity and Quality of Available CSEWG Benchmarks

Existing fast reactor benchmarks test only a limited domain of important nuclear data (e.g., burnup benchmarks, high temperature experiments, etc. are unavailable). The benchmark collection should be extended to include several more recent experiments (e.g., heterogenous designs,[120] larger cores,[121] GCFR assemblies,[122] HCDA configurations,[123] diagnostic cores).[124] The quality of experimental uncertainties for the existing benchmarks is very uneven. Some of the older data are now based on obsolete techniques; the nature of quoted uncertainties is imprecisely specified (often experimental extrapolations, corrections are made which have not been clearly described).

2. Determination of Calculational Corrections and Associated Uncertainties

(a) Calculational Methods Uncertainties

Notwithstanding the significant effort during the past five years culminating in improved methodologies and comparative analysis, there has been no attempt herein to quantify uncertainties due to calculational methods (e.g., cross section processing techniques, neutronics/perturbation calculations, etc.). This has been done to a limited extent in subsequent sections related to thermal reactor data testing. More effort in this area is required if we are to try and infer specific data related needs from calculation/experiment comparison.

(b) Modeling Corrections and Uncertainties

Traditionally, data testers have applied multigroup diffusion or transport theory methods to homogeneous one- or two-dimensional benchmark

models for these inherently heterogeneous three-dimensional configurations. Once the integral experiment has been reduced to such a simple model it is then practical to obtain either diffusion theory, transport theory, or Monte Carlo solutions to which the appropriate correction factor(s) must be applied. Typically, corrections might be required to account for heterogeneity, transport, or streaming effects, etc. These corrections (and uncertainties) need to be reevaluated for older assemblies and determined for newer experiments.

(c) Concluding Remarks

The uncertainty in calculated eigenvalues based on benchmark models is estimated to be ± 0.5% - ± 1%. Therefore, for data testing purposes, a computed k_{eff} within 1% of unity may yield little data-related specific information.

Regarding reaction rate ratio tests, the major limitation is the understanding and interpretation of older experiments. Modeling corrections required for calculation of cavity reaction measurements obtained with fission chambers in the ZPR criticals can be large - perhaps 3% on ^{238}U capture and 10% on threshold fission rates.

For small sample reactivity worths, the 1-D homogenous models are not adequate. Heterogeneity effects were shown to affect the low energy absorber worths a few percent. The reactivity conversion factor is geometry dependent and conversion factors derived from 1-D models may be in error. The current ENDF-202 specifications[104] do not include corrections to be applied to reactivity worth calculations for these factors, and hence, tests of reactivity worths are limited to identification of only gross trends.

3. Estimation of Uncertainties Due to Nuclear Data

With the issue of ENDF/B-V, a first attempt has been made at a quantitative characterization of nuclear data uncertainties. Together with the extensive set of sensitivity coefficients computed for CSEWG benchmarks,[125] uncertainties in integral responses due to uncertainties in nuclear data may be estimated. This has not been a formal part of the ENDF/B-V data testing process (although a limited amount of such information is available from individual laboratory studies) and needs to be an integral part of the data testing process for Version VI.

C. Overall Conclusions

1. Areas where ENDF/B-V Data Appear Consistent With Fast Reactor Measured Benchmarks.

(a) Criticality

The calculated k_{eff} values range from 0.986 to 1.015. (The calculated k_{eff} values obtained in Version IV data testing which included a larger number of benchmarks and data testers ranged from 0.985 to 1.015.) The k_{eff} values predicted with Version V were slightly better than results obtained with Version IV. Whereas, with Version IV, the k_{eff} of the Pu fueled assemblies was underpredicted relative to the U fueled assemblies, the situation with Version V has reversed this trend (e.g., the k_{eff} for JEZEBEL is calculated 0.5 - 1.0% higher than GODIVA and ZPR-6/7 is calculated 0.8 - 1.2% higher than ZPR-6/6A). There does not appear to be any apparent bias according to spectral hardness as there was with Version IV. Since most calculation/experiment k_{eff} lie well within 1% of unity, it is difficult to draw specific data related

inferences since modeling and calculational uncertainties are certainly of this order. For example, the dispersion in calculated eigenvalues for ZPR-6/7 resulting from independent analysis are ~0.6%. This dispersion of values among analysts is typical for each of the benchmarks.

(b) Relative Fission Rates

The fission ratios $^{49}f/^{25}f$ and $^{28}f/^{25}f$, both poorly predicted with Version IV data, are now well-predicted in the ZPR critical assemblies.

2. Areas Where Discrepancies Are Still Perceived to Exist Between Measured and Computed Fast Reactor Benchmarks.

(a) ^{238}U Capture Rate Relative to Fission

The capture rate in ^{238}U relative to ^{239}Pu (or ^{235}U) fission is overpredicted in the large, dilute assemblies (by ~3-6% relative to ^{25}f and 4-8% relative to ^{49}f). This overprediction, which was also observed with Version IV data, is only slightly reduced with Version V and is most important in the benchmarks which are sensitive to the low energy portion of the spectrum (≤50 keV).

(b) Central Worths

The fissile material reactivity worths in the LMFBR benchmarks are overpredicted with Version V data (as consistent (within ~1-3%) with previous ENDF/B versions), i.e., calculation/experiment values are in the range of 1.10 - 1.20 for ^{239}Pu and ^{235}U worths. The fact that the fissile worths are well calculated in harder assemblies (GODIVA, JEZEBEL, BIG-10) also point to phenomena associated with the lower energy portion of the spectrum.

The calculated reactivity worth is a complex function of a large set of nuclear data. It is a bilinear weighting of the sample cross sections

where both the forward and adjoint spectra are themselves integrally related to the cross sections of all materials in the assemblies. Thus, the information pertaining to data testing obtained from small sample reactivity worths is not as directly interpretable as criticality and reaction rate tests.

(c) Spectral Indices

The $^{28}f/^{25}f$ spectral index in the smaller harder spectrum assemblies range from underprediction in JEZEBEL by 8-10% to overprediction by ~2-4% in GODIVA. Hence, although the $^{28}f/^{25}f$ differences between measurement and calculation for lower energy LMFBR assemblies are not considered problematic, these differences would surely not be considered acceptable for higher energy applications. The discrepancies point toward re-evaluation of our state of knowledge for the neutron emission spectrum from neutron induced reactions in fissile materials.

6.3 Thermal Reactor Data Testing

A. Integral Experiments Analyzed

Of the CSEWG thermal reactor benchmarks all are either H_2O or D_2O moderated, and except for two (MIT and BNL) all are critical assemblies. The ORNL 1,2,3,4,10 series[126,127] are five unreflected spheres of uranyl nitrate in H_2O, three of them poisoned with boron. Critical compositions and volumes were determined. The TRX-1,2,3,4 series[128-133] represent fully reflected fuel rods of uranium (enriched to 1.3%) clad in aluminum and H_2O-moderated. The following integral parameters were measured at the center of each of the TRX lattices: the epithermal/thermal ratio of ^{238}U capture ($^{28}\rho$) and ^{235}U fission ($^{25}\delta$), the ratio of ^{238}U capture to ^{235}U

fission (C*), and the ratio of ^{238}U fission to ^{235}U fission ($^{28}\delta$). Also measured were axial bucklings, and radial bucklings in TRX 1 and 2. The BAPL-UO_2-1,2,3 assemblies[134-136] were also H_2O moderated, fully reflected UO_2 rods (1.3% ^{235}U) clad in aluminum, and the measurements themselves were similar to those for TRX.

The MIT-4,5,6 assemblies[137-140] are D_2O moderated exponential lattices of uranium metal rods and the measurements made were for integral parameters similar to those measured in TRX. These assemblies supersede an earlier series (MIT-1,2,3), but none in this series were analyzed with Version V data. The PNL 1-12 series[141-148] represent twelve homogeneous aqueous plutonium nitrate assemblies with H/^{239}Pu ratios ranging from $\sim 10^2$ to $\sim 10^3$; critical volumes and compositions were determined. PNL-30 through 35 are H_2O moderated lattices[149-152] of UO_2-2 wt% PuO_2 fuel rods, clad in Zircaloy; critical configurations were determined with borated and unborated moderators.

Finally, the BNL-ThO_2-1,2,3 assemblies[153-155] were $^{233}UO_2$-ThO_2 clad in Zircaloy 2 and H_2O moderated. Measured parameters include axial relaxation length, radial buckling, and $^{02}\rho$ (the ratio of epithermal to thermal thorium capture).

B. Current Limitations in the Data Testing Process

1. Quantity and Quality of Available CSEWG Benchmarks

With regard to benchmark coverage, the following comments can be made:

System Type	Benchmark Coverage
^{235}U-H_2O	Good. Small assemblies should be added to benchmark set and analyzed.
^{235}U-^{238}U-H_2O	Good
^{235}U-^{238}U-D_2O	Poor. MIT-4, 5, 6 are old and exponentials. LTRIIA[156] is complex and has some inconsistencies with $^{28}\rho$. ZEEP[157] is being added.
^{233}U-H_2O	Good. Full range of ORNL and small assemblies should be added to benchmark set.
^{233}U-Th-H_2O	Good. BNL's are exponentials. Would be better to have simple critical lattices.
Pu-H_2O	Poor. Need some well determined experiments.
UO_2-PuO_2-H_2O	Fair. Could use additional experiments to resolve inconsistencies and provide lattice parameters.

In addition, it seems appropriate to extend the range of data testing, to include isotopics from reactor discharge. As in the case of fast reactors, complete characterization of experimental uncertainties is hindered by the lack of documentation and the vintage of the experiments. Additional testing seems also worthwhile for the moderator temperature coefficient of reactivity; this area has also received recent attention by the British and French.

Other CSEWG benchmarks and related integral experiments have not yet been analyzed with ENDF/B-V data. Among these systems are ^{233}U-H_2O, Th-^{233}U-H_2O, and ^{235}U-^{238}U-D_2O assemblies. It is important that additional analyses be done in order to obtain a more comprehensive and accurate picture of the adequacy of ENDF/B-V.

2. Determination of Calculational Corrections and Associated Uncertainties

(a) Calculational Methods and Uncertainties

Many of the thermal reactor benchmark analyses have used Monte Carlo and transport theory methods that are considered by members of the Thermal Reactor Data Testing Subcommittee to be highly accurate. The nuclear data and experimental geometry are described in detail; a number of Monte Carlo calculations described the entire benchmark assembly explicitly in three dimensions. An intercomparison[158] of calculated four-group reaction rates for the TRX lattice cell has shown excellent agreement among the different methods, including four different Monte Carlo calculations. This has yielded an estimate of "methods uncertainty" for calculated integral parameters reflecting the scatter of calculated results obtained by different methods. The characterization of "methods uncertainty" based on methods scatter is an important first step, but still subject to common mode failure. Thus, continued effort to improve the characterization of such uncertainties is essential for all future benchmark analysis.

(b) Modeling Corrections and Uncertainties

Calculational corrections have not usually been required since the best available calculational methods have been used and these have described the experimental geometry explicitly.

3. Estimation of Uncertainties Due to Nuclear Data

Sensitivities of measured results to cross section data have been extensively studied by direct rerun with varied data, and by using

perturbation theory. ENDF/B-V covariance files are available and offer the potential for a number of useful applications for thermal reactor analysis. A covariance analysis of the TRX-2 assembly was completed using preliminary data, but ENDF/B-V covariance data has not yet been applied in thermal reactor data testing. This is an area for future investigations.

C. Overall Conclusions

1. Areas where ENDF/B-V Data Appear Adequate for Thermal Reactor Needs

(a) ^{235}U-H_2O Assemblies (ORNL 1,4,10)

The average k_{eff} for these CSEWG benchmarks using ENDF/B-V data is 1.0002. The marked trend observed using ENDF/B-IV for the calculated k_{eff} to increase with spectral hardness and total leakage fraction is no longer apparent using ENDF/B-V. A hardening of the fission spectrum (ENDF/B-V mean energy $\overline{E}$ = 2.030 MeV; ENDF/B-IV mean energy $\overline{E}$ = 1.985 MeV) and change in shape removed a leakage underprediction; similarly the analysis of these assemblies at the zero leakage limit suggested the need for a ~0.6% increase relative to ENDF/B-IV of thermal $\overline{\nu}$ for ^{235}U. A larger $\overline{\nu}$ was also indicated by recent differential experiments, which led to the adoption of a 0.74% increased $\overline{\nu}$ for ^{235}U in ENDF/B-V.

(b) ^{235}U-^{238}U-H_2O Assemblies

Calculations using ENDF/B-V are significantly improved over corresponding analyses using Version IV and give good agreement with experiments, both for k_{eff} and for integral parameters such as $^{28}\rho$. The quality of ^{238}U resonance parameters has been substantially improved by

the new data incorporated in ENDF/B-V. Furthermore, the long standing ^{238}U resonance capture overprediction is much improved--for TRX-1 and TRX-2, EPRI and ORNL calculations predict $^{28}\rho$ 2% above experiment on the average, a reduction of 1-2% from ENDF/B-IV to ENDF/B-V. Although capture is still overpredicted; (e.g., for TRX-1) the difference is now covered by the uncertainties assigned to the comparison (from integral experiment, differential data and calculational methods). This is indicated in Table 6.1 which displays estimated sources of uncertainties for integral parameters in TRX-2.

Table 6.1. Summary of Estimated Uncertainties and Calculation/Experiment Differences for Integral Parameters in TRX-2

	$^{28}\rho$	$^{25}\delta$	$^{28}\delta$	C*	k_{eff}
$(\frac{C}{E} - 1) \times 100\%$					
ENDF/B-V (Average)	+1.1	0.0	+0.9	-0.6	-0.13
ENDF/B-IV Nominal	+2.9	-1.0	+2.6	0	-0.64
Uncertainties (%)					
Integral Experiment	1.9	1.3	5.1	0.9	~0
Methods Scatter	0.6	0.6	0.5	0.3	0.3
Differential Data	0.9	2.0	0.4	0.8	0.4

Calculated k_{eff} values are improved by the reduced ^{238}U capture and the increased $\bar{\nu}$ of ^{235}U with a resulting increase of~ 1% in k_{eff} going from ENDF/B-IV to ENDF/B-V (increased $\bar{\nu}$ and decreased ^{238}U capture). For

TRX-1 and TRX-2 the average k_{eff} = 0.9977; for BAPL-UO_2-1,2,3 the average k_{eff} = 1.0038. These values are in nominal agreement with k_{eff} for the ^{235}U-H_2O assemblies. $^{28}\delta$ increased 3% due to the harder ^{235}U fission spectrum, and shows good agreement with experiment. $^{25}\delta$ continues to agree well with experiment in these assemblies. In general, analysis of the BAPL-UO_2 assemblies show similar trends as discussed for the TRX lattices, except for $^{28}\delta$ which is systematically high relative to TRX. (Of the two, the TRX results are considered more reliable.)

(c) UO_2-PuO_2-H_2O Assemblies

Calculated k_{eff} for PNL30 (0.9986 ± 0.0019) is reasonable. Assemblies PNL31-35 need to be analyzed to obtain a better picture.

2. Areas Where Discrepancies Are Still Perceived to Exist for Thermal Reactor Applications

(a) ^{235}U Capture/Fission

Although $^{25}\delta$ in the TRX assemblies hardly changed in going from ENDF/B-IV to ENDF/B-V, and the comparison with integral experiment remains good, it should be noted that the ENDF/B-V ^{235}U fission resonance integral is 282 b (above 0.5 eV), while the best integral result is 275 ± 5b. A re-evaluation of ^{235}U fission and capture taking into account that stated uncertainties of the integral measurements may be smaller than those of differential capture measurements, appears worthwhile.

(b) Pu-H_2O Critical Experiment Analysis

For Pu-H_2O assemblies, calculated k_{eff} using ENDF/B-V averages 1.011. The overall picture is unchanged from Version IV. The large scatter of k_{eff} (2.7%) shows no convincing trends and is attributed to uncertainties of plutonium content. This scatter makes it difficult to draw any

detailed conclusions about data inadequacies. In any case, sensitivities are sufficiently small that it is hard to explain the high k_{eff} on the basis of obvious data deficiencies. Thus, despite a good deal of work and the analysis of more benchmarks, the picture is about the same as it was at the time of the ENDF/B-IV data report. There remains a need for several highly accurate benchmark critical experiments to be performed. In addition, careful studies should be made to identify the source of the high k_{eff} values.

6.4 Shielding Data Testing

It is widely accepted that gamma-ray interaction cross sections (Compton scattering, pair production, photoelectric effect) are very well known compared to neutron and secondary gamma-ray production cross sections. Thus, CSEWG shielding data testing has focused on testing data associated with neutron and secondary gamma-ray production processes.

A. Integral Experiments Analyzed

The current CSEWG integral experiments used for shielding data testing are listed in Table 6.2; additional experiments and computations which have been useful in testing ENDF/B-V are listed in Table 6.3. The estimated accuracy of the integral experiment data is summarized in Table 6.4. The SDT1-SDT5 "broomstick" experiments provide tests of total cross sections and, as such, are of general importance for shielding applications. SDT11 provides tests of neutron transport through iron and stainless steel and is of general value because of the widespread use of steel and iron as shielding materials. SDT9, SDT12, and AB1 are all

Table 6.2. Integral Experiments Recommended by CSEWG for Shielding Data Testing

SDT Series

SDT1. Iron Broomstick Experiment - an experimental check of neutron total cross sections, R. E. Maerker (ORNL-3867, ENDF-166, revised)

SDT2. Oxygen Broomstick Experiment - an experimental check of neutron total cross sections, R. E. Maerker (ORNL-3868, ENDF-167, revised)

SDT3. Nitrogen Broomstick Experiment - an experimental check of neutron total cross sections, R. E. Maerker (ORNL-3869, ENDF-168, revised)

SDT4. Sodium Broomstick Experiment - an experimental check of neutron total cross sections, R. E. Maerker (ORNL-3870, ENDF-169, revised)

SDT5. Stainless Steel Broomstick Experiment - an experimental check of neutron total cross sections, R. E. Maerker (ORNL-3871, ENDF-170, revised)

SDT9. Neutron Attenuation Measurements in a Mockup of the FFTF Radial Shield, P. Rose, H. Alter, R. Paschall, A. W. Thiels (AI-AEC-13048, ENDF-181)

SDT10. Calculational Models for LLL Pulsed Spheres, E. F. Plechaty, R. J. Howerton (UCID-16372) - ^{6}Li, ^{7}Li, Be, C, N, O, Mg, Al, Ti, Fe, Pb, ^{235}U, ^{238}U, ^{239}Pu

SDT11. The ORNL Benchmark Experiment for Neutron Transport Through Iron and Stainless Steel, Part 1, R. E. Maerker (ORNL-TM-4222, ENDF-188)

SDT12. The ORNL Benchmark Experiment for Neutron Transport Through Sodium, R. E. Maerker (ORNL-TM-4223, ENDF-189)

SB Series

SB2. Experiment on Secondary Gamma-Ray Production Cross Sections Arising from Thermal-Neutron Capture in Each of 14 Different Elements Plus Stainless Steel, R. E. Maerker (ORNL-TM-5203, ENDF-227)

SB3. Experiment on Secondary Gamma-Ray Production Cross Sections Averaged Over A Fast-Neutron Spectrum for Each of 13 Different Elements Plus Stainless Steel, R. E. Maerker (ORNL-TM-5204, ENDF-228)

Table 6.3. Additional Experiments and Computations Which Have Provided Useful Information for Testing ENDF/B-V

AB1/SB4. Measurement and Calculations of the ORNL CRBR Upper Axial Shield Experiment, R. E. Maerker, F. J. Muckenthaler, C. E. Clifford (ORNL-5259, ENDF-258)

(Note this is approved as a benchmark candidate pending its removal from "Applied Technology" classification. Experimental results are currently restricted to U. S. distribution.)

AB2. Status of ENDF/B-V Neutron Emission Spectra Induced by 14 MeV Neutrons, D. M. Hetrick, D. C. Larson, C. Y. Fu (ORNL/TM-6637, ENDF-280)

AB3. Negative Heating Numbers, Los Alamos Scientific Laboratory Office Memorandum from E. C. Little and R. E. Seamon to P. G. Young and E. D. Arthur, Chairmen, CSEWG Evaluations Committee, June 17, 1981. See also "Energy Balance of ENDF/B-V", R. E. MacFarlane (Trans. Am. Nuc. Soc. 33, 681 (1979).

Table 6.4. Estimated Accuracy of Integral Experiment Data of Interest for Shielding Data Testing and Indication of Participation in ENDF/B-V Data Testing

Experiment	Estimated Accuracy	Participating Lab
SDT1-SDT5	± 10% in total cross section	BNL, ORNL
SDT9 (FFTF Radial Shield)	± 1% to 50% depending on amount of attenuation	None
SDT10 (Pulsed Spheres)	± 10%	LANL, LLNL
SDT11 (ORNL Fe, SS)	± 15%	LANL
SDT12 (ORNL Sodium)	From ± 5% to ± 15% depending on amount of attenuation	None
SB2 (Thermal Gamma Rays)	± 15%	LANL, ORNL
SB3 ("Fast" Gamma Rays)	± 30%	ORNL
AB1 (SB4) (CRBR Upper Axial)	± 5-15% Bonner Ball count rates, greater uncertainties in unfolded spectra	ORNL
AB2 (14 MeV Neutron Emission)	± 15% integrated cross section	ORNL
AB3 (Negative Heating)	(Computation)	LANL

mockup experiments relevant to sodium-cooled fast reactor design and provides tests of transport through combinations of sodium, iron, and steel. SDT10 and AB2 provide tests of data for neutrons at 14 MeV and are of value for fusion applications.

B. Current Limitations in the Data Testing Process

1. Quantity and Quality of Available CSEWG Benchmarks.

Limited resources have hindered acquisition and documentation of additional shielding benchmarks. Potential candidates exist including measurements in concrete and air. Only recently has a major series of fusion shielding integral experiments been documented as CSEWG benchmarks,[104] and efforts are currently under way to test the performance of ENDF/B-V relative to these measurements.

2. Determination of Calculational Corrections and Associated Uncertainties

The SDT1-SDT5 broomstick experiments require no processing or transport approximation and require only an uncollided flux calculation with point energy values of cross sections. The SB2, SB3, and AB2 experiments are, in effect, measures of production cross sections which require a minimum of processing of data; uncertainties of ~10-15% in the fast neutron flux spectrum still remain.

The SDT10 "pulsed sphere" experiments and the SDT11 "iron" benchmark are normally calculated with Monte Carlo and the uncertainty associated with calculated results is generally assumed to be the statistical uncertainty on the computed value.

The AB1 CRBR Upper Axial Shield mockup is calculated with one- and two- dimensional discrete ordinates calculations using fine- and broad-multigroup data sets. No attempt has been made to characterize the uncertainty in the calculated results due to the discrete ordinates method or to the cross section generation. In general, the CSEWG shielding studies have not attempted to explicitly characterize calculational uncertainties.

The AB1 CRBR Axial Shield neutron penetration studies proved to be disappointing in that the recalculated C/E values disagreed with the expected predictions based on previously calculated sensitivity coefficients. The effects of the changes in cross sections are large enough to be nonlinear, thus casting doubt on the use of these sensitivity coefficients for estimating the effect of future proposed changes.

3. Estimation of Uncertainties Due to Nuclear Data

For the purposes of the initial contributions (May 1981) to this data testing report, ENDF/B-V shielding data testing has not made use of the uncertainty files available with ENDF/B-V. Multigroup covariance files do exist for some isotopes and offer a potentially useful data base for future application. These include a 26 neutron group and a 24 neutron group set of data based on ENDF/B-V and private evaluations which were derived for fast reactor and light water reactor pressure vessel damage applications. Both multigroup sets are available from the Radiation Shielding Information Center (RSIC), Oak Ridge National Laboratory as DLC-77/COVERV-81.

There also has been some potentially useful work which attempts to assign uncertainties to secondary energy distributions as part of fusion reactor studies. None of the above unfortunately was actually used as part of the data testing analysis process.

C. Overall Conclusions

1. Areas Where ENDF/B-V Appear Adequate for Shielding Needs (i.e. air, steel, iron, concrete).

In this section, "adequate" agreement is intended to imply consistency to within the stated uncertainties given in Table 6.4 unless otherwise stated.

The SDT1-SDT5 results for N, O, Na, Fe, and stainless steel indicate adequate total cross section values. The SB2 experiments indicate adequate thermal γ-ray production data for N, Na, Al, Si, S, K, Ca, Ti, Fe, Ni, and stainless steel. The SB3 experiments indicate adequate data for fast neutron gamma-ray production for O, Na, Al, Si, Ti, Fe, Ni, and stainless steel. The SDT11 iron benchmark results show adequate results with calculations generally within the stated experimental uncertainties and with an apparent improvement over ENDF/B-IV for the thicker slabs. The SDT10 pulsed sphere results indicate adequate agreement for H_2O, ^{6}Li, ^{7}Li, C, CH_2, N, O, Fe, Au, Th, ^{235}U, ^{238}U, and ^{239}Pu. This assessment is made in a somewhat arbitrary fashion by accepting integral results which deviate from the experimental value by no more than 15% in either the transmission and elastic peak or the non-elastic component of the results. This indicates acceptable 14 MeV cross sections and secondary emission spectra.

The AB2 14 MeV neutron emission experiments indicate fair agreement for Na, Al, and Si and good agreement for Fe and Pb. This experiment provides a test of non-elastic data at 14 MeV and supplements the SDT10 pulsed sphere results mentioned above.

The AB3 energy balance calculations indicate good agreement for light elements ^{1}H, ^{2}H, ^{6}Li, ^{7}Li, ^{9}Be, ^{10}B, C, ^{14}N, ^{15}N, ^{16}O for most applications.

It must be emphasized that data sets labeled "adequate" are still with respect to computation of relatively simple benchmarks, certainly no guarantee of performance in complex shields.

2. Areas Where Discrepancies Are Still Perceived to Exist for Shielding Applications

(a) Total Cross Sections - C, Ni, Cr.

The SDT1-SDT5 broomstick results indicate the C total cross section may be too high in the 8 MeV region by ~8%. Recent GCFR experiments at the Tower Shielding Facility also point to possible carbon total cross sections discrepancies in the 5-8 MeV region. The Ni and Cr total cross sections are thought to be too high by about 20% in the 1-3 MeV region. This is consistent with underproduction of transmission through stainless steel experiments observed in the GCFR program.

(b) Gamma-ray Production.

Table 6.5 shows poor agreement for thermal capture gamma-ray production in Cl and Cu below 6 and 4.5 MeV respectively.

Table 6.5. Comparisons of Calculated and Measured Photon Production Cross Sections Arising from Thermal-Neutron Capture, in Millibarns

Photon Energy	Copper		Chlorine	
Range (MeV)	SB2	Calc.	SB2	Calc.*
~1.0 - 1.5	501	80	13500	4420
1.5 - 2.0	518	115	13600	9800
2.0 - 2.5	246	71	4280	1130
2.5 - 3.0	324	99	5770	2410
3.0 - 3.5	317	87	3680	427
3.5 - 4.0	290	98	2720	283
4.0 - 4.5	378	195	1940	171
4.5 - 5.0	159	182	2690	1590
5.0 - 5.5	324	308	1020	453
5.5 - 6.0	74	72	3500	2660
6.0 - 6.5	183	182	7840	8480
6.5 - 7.0	349	423	5470	5100
7.0 - 7.5	534	567	3710	6260
7.5 - 8.0	1510	1855	2990	0
8.0 - 8.5			0	0
8.5 - 9.0			960	851
9.5 -10.0				
10.0 -10.5				
10.5 -11.0				

*Unchanged from Version-IV.

(c) Spectra from 14 MeV Interaction

Pulsed sphere studies indicate that problems (15-30% discrepancies) remain in the analysis of spectra from 14 MeV neutron induced reactions in Mg, Ca, Ti, V, Cr, Ni, Cu, Nb, and W. Calculated fast neutron gamma-ray production for S, K, and Cu should be substantially increased if experimental results of benchmark SB3 are valid (see Table 6.6).

(d) Missing Data and General Evaluation Difficulties

Many materials available from ENDF/B-V have no photon production data. Among them are ^{3}He, ^{11}B, Zr, Cd, ^{152}Eu, ^{154}Eu, the Gd isotopes, ^{197}Au, ^{233}Pa, ^{233}U, ^{234}U, ^{236}U, ^{237}Np, and ^{238}Pu. These represent areas in which data are needed.

Energy balance study suggests that evaluators should (a) do isotopic evaluations when possible, (b) use model codes wherever practical, (c) avoid placing experimental data directly into the files without analysis, (d) use yields rather than production cross sections where possible, (e) use discrete photons rather than continuous distributions where possible, and (f) check the energy balance at each stage in the evaluation.

6.5 Fission Product and Actinide Data Testing

A. Integral Experiments Analyzed

1. Capture and Fission Cross Sections in Fast Reactor Spectra

Integral measurements of capture cross sections of fission products include central reactivity worth and reaction rates performed in fast critical assemblies, fast-thermal coupled reactors, and fast reactors. The primary CSEWG benchmark facility for fission product and actinide testing is the Coupled Fast Reactivity Measurement Facility (CFRMF);[159]

Table 6.6 Comparisons of Calculated and Measured Photon Production Cross Sections Averaged Over a Fast-Neutron Spectrum above 1 MeV, in Millibarns

Photon Energy	Copper		Potassium		Sulfur	
Range (MeV)	SB3	Calc.*	SB3	Calc.*	SB3	Calc.*
~1.0 - 1.5	1090	886	133	16	73	5.7
1.5 - 2.0	307	196	96	9.0	44	10
2.0 - 2.5	127	92	39	5.7	199	199
2.5 - 3.0	87	61	96	77	16.4	9.0
3.0 - 3.5	45	36	26	28	11.2	2.4
3.5 - 4.0	35	22	23	39	3.3	3.5
4.0 - 4.5	24	14	6.1	2.3	10.8	9.0
4.5 - 5.0	13.5	7.2	8.7	1.5	3.8	0.7
5.0 - 5.5	9.1	4.5	3.6	0.9	1.2	0.6
5.5 - 6.0	<5.9	3.0	4.6	0.5	<3.0	0.4
6.0 - 6.5	<6.2	2.0	<3.1	0.2	<0.7	0.3
6.5 - 7.0			<1.4	0.1	<1.2	0.2
7.0 - 7.5			<1.5	0	<0.6	0.2

*Unchanged from Version-IV.

additional fission product measurements are obtained from the Experimental Breeder Reactor (EBR-II) experiments[160] and the STEK 500-4000 cores.[161,162]

Integral capture and/or fission cross-section measurements for nine actinides ranging from ^{232}Th to ^{243}Am have been made for samples irradiated in the CFRMF.[163-165] Fission-rate measurements for several isotopes of plutonium, americium and curium have been made in the Zero Power Fast Reactor, ZEBRA.[166] In addition, integral capture cross sections for ^{241}Am and ^{243}Am have been determined based on measurements of the ^{242}Cm and ^{244}Cm production cross sections irradiated in ZEBRA.[167-168]

In sum, testing has been performed on 34 fission product neutron capture reactions and for 7 higher actinide fission and capture reactions.

2. Decay Heat and Beta and Gamma Spectra Test of Decay Data and Fission Yields

A number of experiments at Los Alamos and Oak Ridge, in which nuclear fuel samples were irradiated with thermal neutrons and the decay energy and spectra of the resulting fission products subsequently observed were selected for calculational comparisons. The experiments used in this study include the following.

- Oak Ridge spectral experiments[169,170] in which ^{235}U and ^{239}Pu fuels were irradiated with thermal neutrons for times of 1, 10, (5 for ^{239}Pu), and 100 s, and both aggregate fission-product gamma-ray and beta-ray decay-energy spectra were measured for a range of averaging cooling times from 2.2 s (for the 1 s irradiation time) to 12 000 s (for the 100 s irradiation time).

- Los Alamos calorimetric experiments[171,172] in which ^{233}U, ^{235}U, and ^{239}Pu were irradiated with thermal neutrons for 20,000 s and total decay heat (gamma plus beta) measured for a range of cooling times from 29 to 190,000 s.
- Los Alamos spectral experiments[173] in which fuels, irradiation time, and cooling time ranges were the same as for the calorimetric experiments, but aggregate fission-product gamma-ray decay-energy spectra were measured.

All of the experiments were included in formulating the ANSI/ANS-5.1 Decay Power Standard.[174]

B. Current Limitations in the Data Testing Process

1. Quantity and Quality of Available CSEWG Benchmarks

The existing integral experiment facilities have been utilized productively for CSEWG data testing of fission products and actinides. A continuing program of measurements at CFRMF and/or STEK is needed which focuses upon measurements of important absorbing fission products, not heretofore tested (e.g., ^{135}Cs, ^{151}Sm, etc.). It is also considered possible that reactivity measurements at these facilities might lend insight in the testing of inelastic cross sections of fission products.

2. Determination of Calculational Corrections and Associated Uncertainties

Both measurements and calculations were used to characterize the CFRMF spectrum and associated uncertainty.

Although several cross section libraries were generated and intercompared, uncertainties due to approximations in cross section

processing methodologies were not taken into account, neither were any uncertainties due to transport methods or geometric modeling.

Calculational uncertainties for summation calculations for the point irradiations of various time duration were assumed to be negligible.

3. Estimation of Uncertainties Due to Nuclear Data

Nuclear data induced uncertainties in the flux spectrum in CFRMF were computed by combining sensitivity coefficients for the flux spectrum with a 26 x 26 energy group covariance matrix for the important material constituents. Uncertainties for the fission products and actinides tested were not available, so that the only nuclear data uncertainty information considered was that for the major materials comprising the CFRMF assembly.

Characterization of the neutron spectra for the samples irradiated in EBR-II is based upon passive neutron dosimetry and spectrum unfolding. The midplane spectrum of EBR-II is similar to that for CFRMF but has a larger component of low-energy neutrons. Adjusted flux covariance matrices which describe the best estimates of the uncertainties and correlations for the spectrum were obtained from the unfolding analysis.

C. Overall Conclusions

A brief summary of the testing results is presented in Table 6.7. In Part A, fission product capture cross sections are assessed. The second and third columns list the measured and calculated integral cross sections. The numbers in parenthesis are estimated uncertainties in percent; the calculational uncertainty reflects only the uncertainty due

to imprecision in the flux spectrum and does not consider calculational errors or data uncertainties in the isotopic reaction of interest. The fourth column indicates the calculated/measured ratio and an uncertainty determined from the second and third columns by quadrature (certainly a lower limit). The last column indicates the degree of current difference of calculation to experiment relative to estimated uncertainties. Daggers are indicated for those reactions for which the calculational to experiment difference appears to be significantly in excess of estimated uncertainties (the latter uncertainties are known to be low).

Changes made in the cross sections in going from ENDF/B-IV to ENDF/B-V resulted in more consistency for 28 of the integral data and less consistency for 14 of the integral data.

The gamma and beta-ray decay energies were assessed with calculations using (1) the ENDF/B-IV fission product file, (2) the ENDF/B-V fission product file, (3) a file derived by substituting decay energies from the Japanese (JNDC) fission product file into the ENDF/B-V energies, and (4) a file derived by substituting decay energies and spectra from the British (UK) data file into the ENDF/B-V file.

Table 6.7. ENDF/B-V Testing of Fission Products and Actinides

Isotope	Meas. (mb)	Calc. (mb)	Calc./meas.	(Calc./Meas.) Relative to Uncertainties
	A. Fission Product Capture Cross Sections (CFRMF)			
^{87}Rb	12 (10)[a]	12.7 (2.0)	1.05 (11)	< 1
^{98}Mo	56.4 (6.4)	65.7 (1.8)	1.09 (7)	1-2
^{100}Mo	55 (17)	48.8 (1.8)	.86 (17)	< 1
^{99}Tc	267 (15)	319.9 (2.0)	1.15 (15)	1-2
^{102}Ru	88.9 (6.6)	100.5 (1.1)	1.12 (7)	1-2
^{104}Ru	82.6 (6.3)	86.4 (1.6)	1.04 (7)	< 1
^{108}Pd	144 (6.7)	198.5 (1.8)	1.20 (7)	2-3[b]
^{107}Ag	380 (19)	384.9 (1.9)	.97 (20)	< 1
^{109}Ag	507 (9.7)	450.2 (2.0)	.86 (10)	2-3[b]
^{121}Sb	251 (8.3)	273.4 (1.5)	1.09 (8)	< 1
^{123}Sb	154 (7.0)	140.0 (1.6)	.91 (7)	1-2
^{127}I	298 (10)	345.5 ___	1.16 (10)	1-2
^{129}I	184 (6.6)	196.3 (1.8)	1.07 (7)	1-2
^{132}Xe	43.9 (7.7)	48.3 ___	1.10 (8)	1-2
^{134}Xe	14.6 (6.9)	24.9 ___	1.70 (7)	> 4[b]
^{133}Cs	276 (6.6)	285.1 (1.9)	1.02 (7)	< 1
^{137}Cs	90 (25)	7.3 (---)[c]	.08 (25)	> 4[b]
^{139}La	17.6 (5.2)	21.8 (1.9)	1.10 (6)	1-2
^{142}Ce	18.4 (7.4)	24.6 (1.0)	1.34 (8)	> 4[b]
^{141}Pr	73 (15)	79.6 (2.0)	1.01 (15)	< 1
^{146}Nd	58 (6.5)	72.5 (1.0)	1.24 (7)	3-4[b]
^{148}Nd	89 (14)	118.9 (---)	1.30 (14)	2-3[b]
^{150}Nd	66.6 (12)	98.0 (1.7)	1.45 (12)	3-4[b]
^{147}Pm	641 (13)	730.1 (2.0)	1.12 (13)	< 1

Table 6.7 (Cont.)

Isotope	Meas. (mb)		Calc. (mb)		Calc./meas.	(Calc./Meas.) Relative to Uncertainties
^{152}Sm	277	(6.4)	342.3	(1.8)	1.18 (7)	2-3[b]
^{151}Eu	2390	(5.8)	2306	(---)	.96 (6)	< 1
^{153}Eu	1450	(7.0)	1444	(---)	1.00 (7)	< 1
	B. Fission Product Capture Cross Sections (EBR-II)					
^{143}Nd	400	(5.8)	341.3	(5.4)	.85 (8)	1-2
^{144}Nd	73.6	(5.6)	58.1	(4.2)	.79 (7)	2-3[b]
^{145}Nd	607	(5.7)	468.0	(6.0)	.77 (8)	2-3[b]
^{147}Sm	1840	(5.7)	1606	(7.3)	.87 (9)	1-2
^{149}Sm	3250	(6.1)	2517	(6.2)	.77 (9)	2-3[b]
^{151}Eu	3750	(7.9)	2842	(5.3)	.76 (10)	2-3[b]
^{152}Eu	3610	(11.2)	3346	(4.9)	.93 (12)	< 1
^{153}Eu	2060	(7.2)	1832	(5.3)	.89 (9)	1-2
^{154}Eu	2660	(9.6)	2006	(5.2)	.76 (11)	2-3[b]
	C. Actinide Cross Sections					
^{240}Pu(n,f)	573	(3.8)	620.3	(-)	1.08 (4)	1-2
^{242}Pu(n,γ)	146	(15)	269.1	(-)	1.84 (15)	> 4[b]
^{242}Pu(n,f)	557	(10)	474.4	(-)	.85 (10)	1-2
^{241}Am(n,γ)	1530	(10)	1109	(-)	.72 (10)	2-3[b]
^{241}Am(n,f)	504	(10)	522.6	(-)	1.04 (10)	< 1
^{243}Am(n,γ)	1000	(15)	594.0	(2.2)	.59 (15)	2-3[b]
^{243}Am(n,f)	352	(10)	418.1	(5.3)	1.19 (10)	1-2

[a] ( ) Values in percent reflecting estimated uncertainties; Calculated uncertainties are due only to imprecision in spectra.

[b] Indicates areas for further study

[c] (---) Indicates information not available

Specifically, the following conclusions can be drawn from the spectral comparisons.

(1) The experimental data are consistent except for the ^{239}Pu beta energy inferred from two Los Alamos experiments and compared to the Oak Ridge experiment.

(2) No method of calculating the decay energy fits the experimental spectral data very well (aggregate results using the combined JNDC/ENDF data agree with the Oak Ridge experiments).

(3) In general, experimental gamma decay energies at short cooling times (<100 s) are low at low energies (<0.8 MeV) and high at high energies (>1.6 MeV) in comparison to calculated energies.

(4) In general, experimental beta decay energies are high at all cooling times for low energies (<1.4 MeV) and low for high energies (>1.8 MeV).

6.6 Dosimetry Data Testing

Dosimetry data testing involves comparisons between calculated spectrum averaged cross sections (i.e., the evaluated energy dependent cross section integrated over and normalized to a specified calculated spectrum) and an experimentally derived spectrum averaged cross section (i.e., the measured reaction rate in the specified spectrum weighted by that spectrum).

A. Integral Experiments Analyzed

Reaction rates and reaction rate ratios in five spectra have been defined for use in the data testing of the Dosimetry File. The CFRMF[159] facility is a benchmark which is also used for the testing of fission products and actinides (see Section 5). The Big-10 facility[198] is a CSEWG benchmark appropriate for fast reactor as well as dosimetry data testing. It is a cylindrical system consisting of a uranium-metal core, averaging 10% ^{235}U, reflected by depleted uranium metal. The ^{235}U thermal fission spectrum[199] is represented in ENDF/B-V in the form of a Watt shape, i.e.,

$$\phi(E) = e^{-(E/a)} \sinh (bE)^{1/2}$$

where

$$a = 0.988 \text{ MeV}$$

$$b = 2.249 \text{ (MeV)}^{-1}$$

Two other fields were considered as part of dosimetry testing. The first is the ^{252}Cf spontaneous fission spectrum and the second is the

Intermediate Energy Standard Neutron Field[200] (ISNF). The ISNF configuration uses only materials whose cross sections are well characterized and the flux spectrum, therefore, is relatively well known. The geometry is a spherical cavity in the graphite thermal column of the NBS reactor with a B-10 shell suspended at the void center and ^{235}U fission source disks placed symmetrically around the inner surface of the graphite.

B. Current Limitations in the Data Testing Process

1. Quantity and Quality of Available CSEWG Benchmarks

There is no CSEWG experimental dosimetry data for $^{23}Na(n,\gamma)$, $^{65}Cu(n,2n)$ or $^{60}Ni(n,p)$. High energy threshold reactions (i.e., $^{58}Ni(n,2n)$) should be dropped or a suitable spectrum adopted. Other reactions (e.g., $^{93}Nb(n,n')$, $^{103}Rh(n,n')$, $^{199}Hg(n,n')$) are in demand and should be considered for future libraries. Uncertainties should be developed which quantitatively characterize the spectrum of all fields used in CSEWG dosimetry testing.

2. Determination of Calculational Corrections and Associated Uncertainties

Uncertainties in the benchmark spectra due to modeling, cross section processing methods or transport methods approximations were not available or developed for use in the data testing process.

3. Estimation of Uncertainties Due to Nuclear Data

A sensitivity and uncertainty analysis was completed[201] as part of the spectral characterization of CFRMF. A 26x26 energy group flux

covariance matrix related to uncertainties and correlations in the nuclear data was generated for the materials which comprise the facility. These were folded with sensitivity coefficients to estimate the flux spectrum uncertainties in CFRMF due to nuclear data uncertainties. The uncertainties in the dosimetry reaction cross sections themselves were not processed although dosimetry reaction covariance files are available on the ENDF/B-V tape (i.e., the uncertainty in the estimated calculational reaction rate includes uncertainties due to nuclear data of the CFRMF matrix, but exclude uncertainties in methods, modeling, or nuclear data of the dosimetry reaction under consideration).

Flux spectrum uncertainties due to nuclear data were not developed for any of the other benchmark fields. Comparisons are given in Tables 6.8 and 6.9 which employ different model representations for the ^{252}Cf fission spectrum. The functional representation makes a considerable difference.

C. Current Estimates of Calculation/Experiment

A comparison of the C/E ratios of the data results in ^{235}U, BIG-10, and CFRMF are given in Table 6.10 and 6.11. The nuclear data induced uncertainty in the flux spectrum is given in the last column of these tables. The results of ISNF testing are given by Wagschal[202] and are reproduced below.

Ratio	Experimental Value	% Uncertainty	Calculated Value	% Uncertainty
σ_f(U-238)/σ_f(U-235)	0.0926	1.39	0.091	4.29
σ_f(Np-237)/σ_f(U-235)	0.510	2.03	0.513	10.30
σ_f(Pu-239)/σ_f(U-235)	1.155	1.28	1.143	2.66

Table 6.8. ^{252}Cf Spectrum Averaged Cross Sections of Threshold Reactions

Isotope	Mat#	Reaction	$\bar{\sigma}$(Watt) CALC.	$\bar{\sigma}$(Max)[a] CALC.	$\bar{\sigma}$(NBS)[b] CALC.	$\bar{\sigma}$(LANL)[c] CALC.	$\bar{\sigma}$ [d] EXP.
^{27}Al	6313	(n,p)	5.58	5.27	5.14	5.62	5.1 ± 0.5
		(n,α)	1.07	1.16	1.06	1.03	1.006 ± 0.022
^{55}Mn	6325	(n,2n)	0.352	0.545	0.440	0.530	0.58 ± 0.06
^{59}Co	6327	(n,2n)	0.320	0.538		0.490	0.57 ± 0.06
		(n,α)	0.220	0.235		0.250	0.20 ± 0.01[e]
^{232}Th	6390	(n,f)	83.78	78.47	78.07		89 ± 9
^{238}U	6398	(n,f)	355.72	314.56	313.60		320 ± 9
^{46}Ti	6427	(n,p)	14.64	13.82	13.47	14.70	13.8 ± 0.3
^{47}Ti	6428	(n,np)	0.015	0.027			
		(n,p)	22.63	24.27	24.07	25.70	18.9 ± 0.9
^{48}Ti	6429	(n,np)	0.004	0.004			
		(n,p)	0.415	0.446	0.409	0.465	0.42
^{54}Fe	6430	(n,p)	97.09	88.89	88.26	94.6	84.6 ± 2.0
^{56}Fe	6431	(n,p)	1.48	1.50	1.41	1.59	1.45 ± 0.003
^{58}Ni	6433	(n,2n)	0.005	0.0009			
		(n,p)	124.92	114.58	113.80	122.00	118 ± 3
^{60}Ni	6434	(n,p)	3.36	3.62	3.44		
^{63}Cu	6435	(n,α)	0.791	0.807	0.758	0.850	
^{65}Cu	6436	(n,2n)	0.513	0.796			
^{115}In	6437	(n,n')	194.16	182.06	181.9	190.00	198 ± 5
^{127}I	6438	(n,2n)	2.02	2.75			
^{32}S	6439	(n,p)	83.64	76.47	76		

Note: Watt and Maxwellian calculations from BNL and HEDL.
[a] T = 1.42 MeV
[b] P. C. Eisenhauer, et al.[205]
[c] Madland, et al.[203] (Using the theory of Ref. 204)
[d] Except where noted Grundl, et al.[206]
[e] McElroy, et al,[207]
[f] Csikai, et al.[208] as noted in Mannhart[209]

Table 6.9. ^{252}Cf Spectrum Averaged Cross Sections (mb) of Exoergic Reactions

Isotope	Matt #	Reaction	$\bar{\sigma}$(Watt)[a] CALC.	$\bar{\sigma}$(Max)[b] CALC.	$\bar{\sigma}$(NBS)[c] CALC.	$\bar{\sigma}$[d] EXP.
^{23}Na	6311	(n,γ)	0.260	0.274	0.271	0.335 ± 0.015[e]
^{59}Co	6327	(n,γ)	5.806	6.096	6.028	6.97 ± 0.34[e]
^{197}Au	6379	(n,γ)	71.269	77.059	76.33	79.9 ± 2.9
^{232}Th	6390	(n,γ)	84.684	90.181	89.68	
^{235}U	6395	(n,f)	1234.87	1238.55	1236	1203 ± 30
^{238}U	6398	(n,γ)	64.328	68.916	68.34	
^{237}Np	6337	(n,f)	1393.46	1354.18	1352	1332 ± 37
^{239}Pu	6399	(n,f)	1800.32	1782.97	1792	1804 ± 45
^{6}Li	6424	(n,tot He)	454.33	462.69	464.60	
^{10}B	6425	(n,tot He)	477.30	493.89	488.90	
^{45}Sc	6426	(n,γ)	4.885	5.381	5.260	
^{58}Fe	6432	(n,γ)	1.613	1.680	1.660	
^{63}Cu	6435	(n,γ)	9.230	9.760	9.649	10.95 ± 0.51[f]
^{115}In	6437	(n,γ)	116.36	122.00	121.20	125.3 ± 4.3

NOTE: Watt and Maxwellian calculations from BNL and HEDL.

a a = 0.983 MeV; b = 3.181 MeV^{-1} (203)

b T = 1.42 MeV

c Eisenhauer, et al.[205]

d Experimental values (except where noted) from Grundl, et al.[206]

e Csikai, et al.[208] from the compilation of Mannhart[209]

f Green[210] calculated from the ratio $\bar{\sigma}Cu/\bar{\sigma}Au = 0.137$

Table 6.10. Comparison of Ratios of Calculated-to-Experimental (C/E) Integral Cross Sections Threshold Reactions

Isotope	Mat #	Reaction	X25	BIG 10	CFRMF	CFRMF Integral Test Uncertainty %
^{27}Al	6313	(n,p) (n,α)	1.10 1.02	1.01	1.09 0.99	8.9 9.2
^{55}Mn	6325	(n,2n)	0.82			
^{59}Co	6327	(n,α)	1.05			
^{232}Th	6390	(n,f)	0.93		0.98	9.1
^{238}U	6398	(n,f)	0.99	1.07	1.06	8.1
^{46}Ti	6427	(n,p)	0.97	0.96	0.95	8.9
^{47}Ti	6428	(n,p)	1.31	1.24	1.29	9.2
^{48}Ti	6429	(n,p)	1.05	0.87	0.91	9.2
^{54}Fe	6430	(n,p)	0.97	1.03	1.07	8.6
^{56}Fe	6431	(n,p)	1.00			
^{58}Ni	6433	(n,2n) (n,p)	0.52 0.96	1.00	1.02	8.6
^{60}Ni	6434	(n,p)				
^{63}Cu	6435	(n,α)	0.96			
^{65}Cu	6436	(n,2n)				
^{115}In	6437	(n,n')	0.95	0.98	1.02	7.7
^{127}I	6438	(n,2n)	1.15			
^{32}S	6439	(n,p)	1.06			

Table 6.11. Comparison of Ratios of Calculated-to-Experimental (C/E) Integral Cross Sections Exoergic Reactions

Isotope	Mat #	Reaction	x25	BIG 10	CFRMF	CFRMF Integral Test Uncertainty %
^{23}Na	6311	(n,γ)				
^{59}Co	6327	(n,γ)		0.96	0.94	4.5
^{197}Au	6379	(n,γ)	0.94	0.93	0.91	3.5
^{232}Th	6390	(n,γ)			0.87	4.1
^{235}U	6395	(n,f)	1.02	1.00	1.01	3.1
^{238}U	6398	(n,γ)		1.00	1.00	4.0
^{237}Np	6337	(n,f)	1.01	1.12	1.11	4.9
^{239}Pu	6399	(n,f)	0.98	0.99	0.99	2.2
^{6}Li	6424	(n,tot He)		0.90	0.95	3.1
^{10}B	6425	(n,tot He)		0.85	0.86	3.3
^{45}Sc	6426	(n,γ)		1.04	1.01	3.8
^{58}Fe	6432	(n,γ)		0.83	1.09	3.6
^{63}Cu	6435	(n,γ)	1.06	0.98	1.02	6.5
^{115}In	6437	(n,γ)	0.93		1.02	3.9

D. Overall Conclusions

1. Areas Where ENDF/B-V Data Appear Adequate for Fast Reactor Needs

The ENDF/B-V Dosimetry File has 18 evaluated cross sections which are consistent with integral measurements for the facilities tested. These are $^{27}Al(n,\alpha)$, $^{59}Co(n,\alpha)$, $^{232}Th(n,f)$, $^{238}U(n,f)$, $^{46}Ti(n,p)$, $^{54}Fe(n,p)$, $^{56}Fe(n,p)$, $^{58}Ni(n,p)$, $^{115}In(n,n')$, $^{32}S(n,p)$, $^{59}Co(n,\gamma)$, $^{197}Au(n,\gamma)$, $^{235}U(n,f)$, $^{238}U(n,\gamma)$, $^{239}Pu(n,f)$, $^{45}Sc(n,\gamma)$, $^{63}Cu(n,\gamma)$, and $^{63}Cu(n,\alpha)$.

The fact that ^{238}U capture is consistent indicates that the CFRMF and BIG-10 spectrum is significantly different from the ANL criticals (see results of fast reactor testing), the calculational representation of one or more of these facilities is in error, or that the integral measurements in one or more of the facilities is incorrect. Furthermore, a consistent representation for $^{239}Pu(n,f)$ would indicate that a breeding ratio calculation would not likely be discrepant in a spectrum like that for CFRMF or BIG-10.

2. Areas Where Discrepancies Are Still Perceived to Exist

Discrepancies are defined to exist for reactions where the C/E (see Tables 6.10 and 6.11) deviates from unity by more than 10% (the required amount specified by the ILRR program.) The reactions include Al(n,p), Mn(n,2n), $^{47}Ti(n,p)$, $^{48}Ti(n,p)$, $^{58}Ni(n,2n)$, $^{127}I(n,2n)$, $^{232}Th(n,\gamma)$, $^{6}Li(n,$ tot He), $^{10}B(n,$ tot He), $^{58}Fe(n,\gamma)$ and $^{237}Np(n,f)$.

The discrepancies for Mn(n,2n) and ^{58}Ni(n,2n) are probably caused by thresholds of high energies, i.e., in the region of the "high energy tail" of the spectrum which is not well defined.

The discrepancies for Al(n,p), ^{63}Cu(n,α), ^{127}I(n,2n), ^{232}Th(n,γ), ^{6}Li(n,tot He), ^{10}B(n, tot He), ^{58}Fe(n,γ) and ^{237}Np(n,f) indicate the need for reanalysis or reevaluation of the energy dependent cross section evaluations.

^{47}Ti(n,p) (and probably ^{48}Ti(n,p)) present a mystery not easily solved. Version IV data testing of the Ti isotopes resulted in much the same discrepancy as that shown in Table 6.10. D. L. Smith[211] of Argonne measured and reevaluated the Ti isotopes. ^{46}Ti(n,p) and ^{47}Ti(n,p) for versions IV and V remain essentially equivalent evaluations and ^{48}Ti changes drastically but the discrepancy remains.

Fabry[207b] has suggested using bias factors with cross sections that do not meet the requirements of the ILRR program. This stratagem has been adopted by the ASTM and Euratom for the issuance of their Reactor Dosimetry File.

7.0 IDENTIFICATION OF IMPORTANT CROSS SECTIONS AND INFERRED NEEDS

7.1 Introduction

In the previous section it was shown that for some integral parameters, significant differences remain between calculated and experimentally measured values. These discrepancies may arise from either inaccurate measurements of the integral parameters themselves, unjustified approximations in calculational methods, and/or uncertainties in the basic nuclear data underlying the calculations. Often, the stated accuracy of the measured value of an integral parameter is much greater than the corresponding calculated value (e.g. 0.2% estimated uncertainty in measured k_{eff} vs 2% uncertainty in the calculated value). The measurement of an integral parameter is usually a more direct determination of the quantity of interest whereas calculation can involve a broad range of cross section, methods, and modeling uncertainties which must be propagated through a calculational model.

When trying to analyze the sources of the discrepancies between measured and calculated parameters, the cross sections which primarily affect the computed result (either because the result is very sensitive to such cross section variations, or because the current state of knowledge of such cross sections is very imprecise) are initially considered as possible contributors to the source of disagreement. The interpretation of the sources of discrepancy is clearly not unique.

Regardless of the specific criteria used, serious consideration must be given to those cross sections for which results of interest are most sensitive. An output parameter is considered to be sensitive to a

particular input parameter if a relatively small variation of the latter causes a relatively large change in the former. The role of sensitivity theory is to provide the vehicle through which this type of information is quantified. A sensitivity coefficient of a performance parameter R with respect to an input variable q is defined as the relative change in R due to a relative change in q. That is

$$S_{Rq}(\underline{\rho}) \equiv \frac{dR}{R} \Big/ \frac{dq\ (\underline{\rho})}{q\ \ (\underline{\rho})} \tag{1}$$

where $\underline{\rho}$ stands for a subset of phase space variables (space, energy angle). The theoretical basis for the efficient determination of sensitivity coefficients is provided by perturbation theory, and particularly, by the developments in recent years of perturbation theory for ratios of reaction rates and of other linear and bilinear functions. This theory will not be repeated herein but can be found in many references including Refs. 175-184 of this paper. Finally, recall that the sensitivities S_{Rq} are appropriate for a given system and result (R) so that cross sections (q) which have low sensitivity for some result R may be quite important for other systems or results.

The Cross Section Evaluation Working Group has compiled a report[185] listing and graphically displaying some of the more important sensitivities for a selection of its benchmarks. Only a few of these are presented below for illustrative purposes.

7.2 Selected Fast Reactor Sensitivity Information

Recall from Section 6 that there remains with Version V, discrepancies relating to the overprediction of ^{238}U capture and fissile worths in

fast critical assemblies as well as spectral indices in small fast assemblies such as GODIVA and JEZEBEL. Table 7.1 lists[185] some of the important integral sensitivities (i.e. the percent change in a response with respect to a percent change in a cross section over the entire energy range; changes in any partial cross section result in changes to the total cross section since all other reaction probabilities are unchanged) for the ZPR-6/7 critical.

Table 7.1 Integral Sensitivities of ZPR-6/7 Performance Parameters to Various Cross Section Reaction Types

$\langle {}^{28}C/{}^{49}f \rangle_C$		^{239}Pu central reactivity worth		^{235}U central reactivity worth	
Reaction	Integral Sensitivity	Reaction	Integral Sensitivity	Reaction	Integral Sensitivity
$^{239}Pu(n,f)$:	-1.073	$^{239}Pu(n,f)$:	0.320	$^{235}U(n,f)$:	1.168
$^{238}U(n,\gamma)$:	+0.896	$^{238}U(n,\gamma)$:	0.198	$^{239}Pu(n,f)$:	-0.893
O(n,n):	+0.109	$^{238}U(n,f)$:	-0.127	$^{235}U(n,\gamma)$:	-0.183
238U(n,n'):	+0.068	$^{238}U(n,n')$:	0.110	O(n,n):	0.180
Na(n,n):	+0.028	Fe(n,n):	0.083	$^{238}U(n,n')$:	0.142
Fe(n,n'):	+0.025	$^{239}Pu(n,\gamma)$:	-0.081	$^{238}U(n,\gamma)$:	0.118
Fe(n,n):	+0.024	O(n,n):	0.077	Fe(n,n):	0.110
$^{239}Pu(n,\gamma)$:	-0.020	Na(n,n):	0.056	$^{238}U(n,f)$:	-0.106
		$^{235}U(n,f)$:	-0.025	Na(n,n):	0.102
				$^{239}Pu(n,\gamma)$:	0.040
				Cr(n,n):	0.028
				$^{241}Pu(n,f)$:	-0.028
				Ni(n,n):	0.020

The overriding importance of the heavy material cross sections, including inelastic scattering in ^{238}U seems apparent. More detailed information can be obtained about the sensitivities by examining the energy dependent sensitivity profile; one such example is the sensitivity of

$\langle {}^{28}c/{}^{49}f \rangle_c$ to $^{238}U(n,\gamma)$. This value, listed as 0.896 in Table 7.1, is disaggregated in Fig. 7.1 as a function of energy. Note that there is high sensitivity at low energies, including the resonance region ($\leq$ 150 keV). It is quite important that these energy dependent profiles be examined since in many cases the sign of the sensitivity coefficient changes as a function of energy leading to cancellation and a deceptively low total sensitivity.

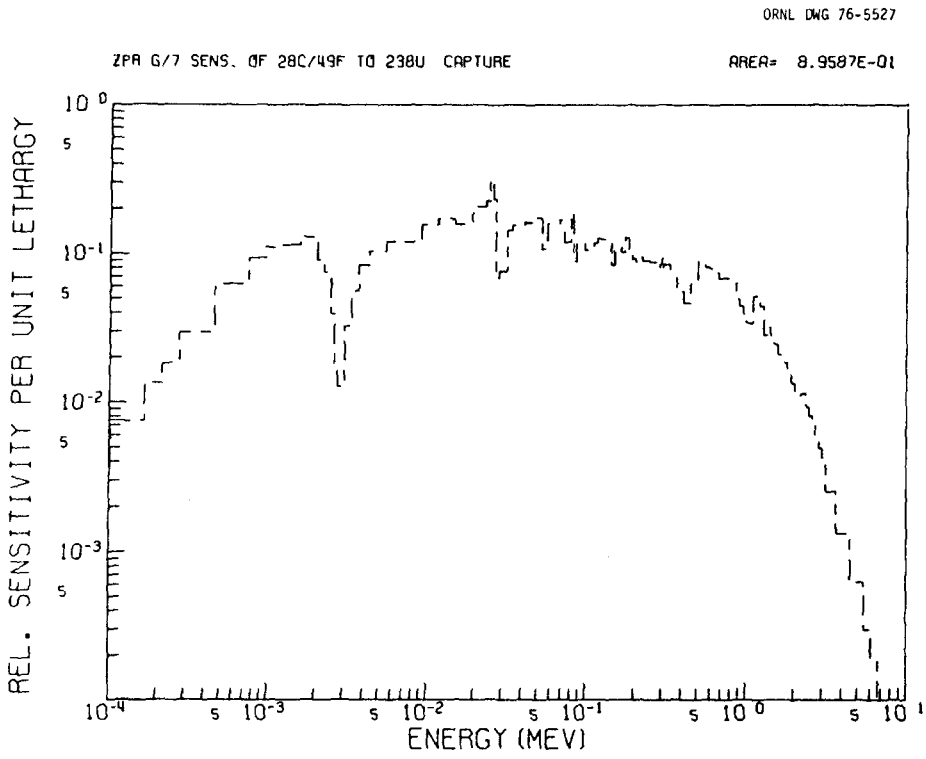

Fig. 7.1. Sensitivity of ^{238}U Capture/^{239}Pu Fission Central Reaction Ratio in Assembly ZPR-6/7 to the ^{238}U Capture Cross Section.

Table 7.2 Total Sensitivities of the $^{28}\sigma_f/^{25}\sigma_f$ Central Spectral Index in GODIVA and JEZEBEL to Various Cross Sections

JEZEBEL		GODIVA	
$\langle^{28}\sigma_f/^{25}\sigma_f\rangle_c$		$\langle^{28}\sigma_f/^{25}\sigma_f\rangle_c$	
Reaction	Relative Sensitivity	Reaction	Relative Sensitivity
$^{28}\sigma_f$	1.0	$^{238}U(n,f)$	0.998
$^{25}\sigma_f$	-1.0	$^{235}U(n,f)$	-0.819
$^{49}\sigma_{n,n',d}$	-0.192	$^{235}U(n,n',d)$	-0.269
$^{49}\sigma_{n,n',c}$	0.101	$^{235}U(n,n',c)$	-0.164
$^{49}\sigma_{n,n}$	-0.078	$^{235}U(\nu)$	0.068
$^{49}\nu$ —	0.054	$^{235}U(n,\gamma)$	0.034
$^{49}\sigma_c$	0.012	$^{238}U(n,n',d)$	-0.026
$^{40}\sigma_{n,n',d}$	-0.010		
$^{40}\sigma_{n,n',c}$	-0.007		

Note the large sensitivities to the inelastic interactions in the fissile nuclides. (n,n',d represents discrete inelastic interactions while n,n'c represents inelastic scattering to a continuum of states).

7.3 Selected Thermal Reactor Sensitivity Information

Table 7.3 displays the effect of cross section variations upon integral parameters measured for the thermal benchmark series TRX 1-4. The results were obtained by recalculation[185] rather than through the use of perturbation theory. $^{28}\rho$ is sensitive to the ^{238}U thermal capture and the shielded capture resonance integral. Since this integral is only about 15 b, unshielded components of capture, which are small compared to

Table 7.3 Effect of Cross Section Variations on Integral Parameters

Integral Parameter	Cross Section Variation*	TRX-3 (V_w/V_u=1.00)	TRX-1 (V_w/V_u=2.35)	TRX-2 (V_w/V_u=4.02)	TRX-4 (V_w/V_u=8.11)
		Percent Change of Parameter			
ρ^{28}	1	+8.78	+7.27	+6.56	+6.02
	2	+7.41	+6.36	+5.87	+5.61
	3	+ .45	+ .18	+ .11	+ .05
δ^{25}	1	- .18	+ .14	+ .08	+ .08
	2	+2.37	+1.13	+ .66	+ .36
	3	+3.52	+3.69	+3.68	+3.88
δ^{28}	1	+2.29	+1.50	+ .89	+ .52
	2	+1.99	+1.33	+ .81	+ .48
	3	- .20	- .14	- .10	- .08
C*	1	+6.77	+4.26	+3.09	+2.06
	2	+5.24	+3.63	+2.74	+1.91
	3	- .35	- .23	- .17	- .11
k_{eff}	1	-1.03	-1.40	- .83	- .63
	2	- .89	-1.21	- .76	- .57
	3	+ .10	+ .13	+ .10	+ .08

*Variation 1: 1.0b increase of ^{238}U smooth capture integral (5.5 keV - 25 keV).

2: 1.0b increase of ^{238}U smooth capture integral (.625 eV-6 eV).

3: 10b increase of ^{235}U smooth fission integral (.625 eV-6 eV).

the dilute capture integral become very important in the TRX lattices. $^{25}\delta$ is sensitive to the ^{235}U thermal fission cross section and the dilute fission integral, thus any proposed modification to the fission resonance integral must be done carefully. Some of the more important group energy-dependent sensitivities for $^{25}\delta$ are given in Table 7.4. $^{28}\delta$ is

primarily sensitive to the ^{238}U and ^{235}U fission cross sections, the ^{238}U inelastic scattering, and the ^{235}U fission spectrum.

Table 7.4 Sensitivities for $^{25}\delta$ in the TRX-2 Thermal Lattice

		dR/R/dσ/σ			
Nuclide	Item	Group 1[a]	Group 2[b]	Group 3[c]	Group 4[d]
H	σ_s	-0.074	-0.042	-0.929	0.002
^{235}U	σ_f	0.098	0.040	0.859	-0.460
^{238}U	σ_c	0.0004	0.001	0.002	0.198
H	σ_c	0.0	0.0	0.0	0.177
^{235}U	σ_c	0.00001	0.00003	-0.001	0.094
Moderator	DB^2	0.001	0.001	0.010	0.017
O	σ_s	-0.003	-0.0004	-0.011	0.0009

[a]10 MeV-67.37 keV

[b]67.37keV-3.35 keV

[c]3.35 keV-0.625 eV

[d]0.625 eV-10^{-5} eV

7.4 Selected Shielding Sensitivity Information

A. Fusion Reactors

The Cross Section Evaluation Working Group has, only this year, incorporated a first fusion reactor benchmark experiment in its Data Testing and Application Activity. Thus, there has been little CSEWG sensitivity analysis of experiments; however, there have been several sensitivity studies for fusion design concepts and some experimental analysis

performed by individual laboratories. These are discussed below in some detail since there is potentially large amounts of additional data which may be required for future applications.

The nuclear data needs and their required accuracies were systematically reviewed by Abdou.[187] At the present, fusion reactor concepts vary widely and are rapidly changing which complicates the task of determining the relative importance of materials for which measurements need be made to extend the data base for fusion applications. Table 7.5 lists the elemental materials that are currently accepted as having the greatest potential for use in the blanket, shield, and other components of fusion reactors. (The list excludes materials for hybrid systems.) A large number of materials comprise the list but this is necessary to assure diversity in studying reactor concepts and to have a sufficiently broad data base for comparative analyses to determine design options and appropriate compositions. The types of data that are required in fusion nuclear design are summarized in Table 7.6.[186,188]

Shielding plays a major role in fusion design so the data requirements for shielding must be emphasized. These requirements are generally different from those for the blanket. In shielding, the main interest is neutron and gamma attenuation, deep penetration and streaming. This requires emphasizing the 1) accuracy of basic cross sections such as total and elastic and inelastic scattering, 2) gamma-ray production and interactions, 3) energy and angular distribution of secondary neutrons and gammas; scattering anisotropy is generally crucial to streaming and deep penetration problems, and 4) good resolution of cross section minima.[187,188]

Table 7.5 Elementary Materials of Potential Use in Fusion Reactors[187]
A qualitative judgement on the probability of using the material is shown
(H = high, M = average, L = Low)

	Demo and Commercial			Experimental Reactors			
Material	Blanket	Shield	Others	Blanket	Shield	Others	Comments
Hydrogen	M	H	H		H	H	plasma, H_2, in organic insulators
Helium	M	M	H		M	H	coolant, also in S.C. magnets
Lithium	H	M		L			tritium breeding, neutron absorption
Beryllium	M						neutron and energy multiplication
Boron		H		M	H		probably in B_4C form
Carbon	L	H			H	H	B_4C, reflector, wall coating, in insulators
Nitrogren			M		M		reactor building atmosphere, insulators
Oxygen		H			H		H_2O, reactor building atmosphere
Fluorine	L						in molten salts
Sodium			M				secondary coolant
Aluminum	L	L	H	L	L	H	S.C. magnet stabilizer and structure
Silicon	M	M	M				SiC, Stainless steel
Argon			M			M	reactor building atmosphere
Titanium	M	L		L			structural material
Vanadium	M	L		L			structural material
Chromium	H	H	H	H	H	H	in steel and super alloys
Manganese	H	H	H	H	H	H	in steel and super alloys
Iron	H	H	H	H	H	H	in steel and super alloys
Nickel	H	H	H	H	H	H	in steel and super alloys
Copper			H			H	normal and superconducting magnets
Niobium	M		H				structural material, also in superconductor
Molybdenum	L	L					
Tin			M			L	in Nb_3Sn superconductor
Tantalum		L			L		
Tungsten		H			H		in space-restricted shield region
Lead	M	H			H		in lithium-lead compounds and lead shield
Bismuth		L					

The production of energetic neutrons above 5 MeV results in reactions leading to the production of charged particles that affect the magnitude of the nuclear heating, radiation damage, and induced activation. The thresholds for many important reactions are well above MeV, so there has been little need for measuring these data accurately for fission application and for some elements, the data may not exist at all. The gamma-ray production data are necessary for determining the nuclear heating in components outside the shield where photons dominate the energy deposition.

Table 7.6 Types of Nuclear Data Required in Fusion Nuclear Design[186,188]

Application	Tritium Breeding	Nuclear Heating	Radiation Damage Indicators	Induced Activation	Radiation Shielding
Data Type					
Total	x	x	x		x
Elastic	x	x	x		x
Inelastic	x	x	x		x
Neutron Emission (σ_{em}, $d\sigma/d\Omega$, P(E')	x	x	x		x
(n,2n), (n,3n)	x	x		x	x
(n,α), (n; n'α)		x	x	x	
(n,p), (n;n'p)		x	x	x	
(n,d), (n; n'd)		x	x	x	
(n,t), (n; n't)	x	x	x	x	
(n,γ), (n; n'γ), (n;xγ)		x		x	x
Gamma Production {σ^P, $d\sigma/d\Omega$, $P(E_n \rightarrow E_\gamma)$}		x			x

In Section 2.0 it was pointed out that the first generation fusion reactors will likely be based on the DT reaction and, in the case of power reactors, with tritium breeding taking place in Lithium in a blanket surrounding the plasma. The cross section sensitivity of the D-T fusion probability and the cross section sensitivities of the D-T and T-T reaction rates have been studied by Santoro and Barish.[189] For the range of incident deuteron source energies and plasma ion temperatures considered, it appears that better D-T and T-T cross section data are needed in the energy range for these reactions between 50 and 200 keV.

Alsmiller et al.[190] used perturbation techniques to compare various fusion reactor designs and showed that the sensitivities are very system dependent. The differences between the sensitivities of the tritium breeding to changes in Li partial cross sections for two reactor concepts are shown in Table 7.7.

Table 7.7 Percent Change in Breeding Ratio Due to a 1% Increase at All Energies in the Indicated Partial Cross Sections of ^{7}Li

^{7}Li Cross-Section Type	Breeding Material J	$\frac{\delta R_j}{R}$ in Percent ($R = R_{^{6}Li} + R_{^{7}Li}$)		
		ORNL Design With Nb	ORNL Design with V	LASL Design
$\Sigma_{(n,n')\alpha,t}$	^{6}Li	1.7×10^{-2}	2.2×10^{-2}	3.1×10^{-3}
	^{7}Li	2.8×10^{-1}	2.7×10^{-1}	1.2×10^{-1}
	$^{6}Li + ^{7}Li$	3.0×10^{-1}	2.9×10^{-1}	1.2×10^{-1}
$\Sigma_{ABSORPTION}$	^{6}Li	-5.9×10^{-3}	-6.0×10^{-3}	-2.3×10^{-3}
	^{7}Li	-3.4×10^{-3}	-3.4×10^{-3}	-2.8×10^{-4}
	$^{6}Li + ^{7}Li$	-9.3×10^{-3}	-9.4×10^{-3}	-2.6×10^{-3}
$\Sigma_{ELASTIC}$	^{6}Li	1.8×10^{-2}	2.9×10^{-2}	6.4×10^{-3}
	^{7}Li	-1.8×10^{-2}	-1.9×10^{-2}	2.4×10^{-4}
	$^{6}Li + ^{7}Li$	0.0	1.0×10^{-2}	6.6×10^{-3}
$\Sigma_{INELASTIC}$	^{6}Li	4.3×10^{-3}	5.5×10^{-3}	1.3×10^{-3}
	^{7}Li	-1.1×10^{-2}	-1.1×10^{-2}	-3.9×10^{-4}
	$^{6}Li + ^{7}Li$	-6.7×10^{-3}	-5.5×10^{-3}	-6.5×10^{-3}
$\Sigma_{(n,2n)}$	^{6}Li	3.7×10^{-2}	3.9×10^{-2}	1.1×10^{-2}
	^{7}Li	-1.9×10^{-2}	-1.9×10^{-2}	-1.6×10^{-3}
	$^{6}Li + ^{7}Li$	1.8×10^{-2}	2.0×10^{-2}	9.4×10^{-3}
$\Sigma_{TOTAL\ COLLISION}$	^{6}Li	7.0×10^{-2}	9.0×10^{-2}	2.0×10^{-2}
	^{7}Li	2.3×10^{-1}	2.2×10^{-1}	1.1×10^{-1}
	$^{6}Li + ^{7}Li$	3.0×10^{-1}	3.1×10^{-1}	1.3×10^{-1}

The first is the ORNL concept proposed by Fraas[191] employing natural lithium metal as the breeder with either niobium or vanadium structure; the second, a Reference Theta Pinch Reactor.[192] This study, and similar studies conducted elsewhere, indicate that a most important consideration in the accuracy of calculated tritium breeding ratios for fusion reactor blankets is the uncertainty associated with the ^{7}Li (n,n',T) ^{4}He reaction cross section. Gerstl et al.[193] computed sensitivities in order to estimate uncertainties in activation of shield components for the Tokamak Fusion Test Reactor. Two of the largest contributors to the potential exposure are ^{58}Co and ^{54}Mn; their production reactions are of considerable interest. Alsmiller et al.[194] have studied the impact of cross section uncertainties in the nuclear heating and radiation damage in the toroidal field coils of an Experimental Power Reactor. Table 7.8 illustrates the total sensitivity (i.e. energy integrated change in response per percent change in a given cross section made uniformly over all energy) and the sensitivity for high energies (13.5 to 14.9 MeV) of two response functions (i.e. energy deposition per unit volume, and displacements per atom) with respect to the various partial cross sections of iron. It is clear that appreciable contributions to the total sensitivity for each of the responses come from the elastic; inelastic, (n,2n) and absorption cross sections of iron. The contributions to the relative variance in the energy deposition per unit volume, and the displacements per atom in the TF coil due to the cross sections identified in Table 7.8 are estimated in Table 7.9 (see Ref. 194).

Table 7.8 Sensitivity of the Energy Deposition Per Unit Volume, and the Displacements Per Atom in the TF coil to Various Partial Neutron Cross Sections and the Total Neutron Cross Section of Iron

Iron Cross-Section Type	$\frac{\delta R_E}{R_E} \Big/ \frac{\delta \Sigma}{\Sigma}$		$\frac{\delta R_D}{R_D} \Big/ \frac{\delta \Sigma}{\Sigma}$	
	All Energies	Energy Interval 13.5 to 14.9 MeV[a]	All Energies	Energy Interval 13.5 to 14.9 MeV[a]
$\Sigma_{elastic}$	-1.75	-0.58	-1.88	-0.64
$\Sigma_{inelastic}$	-3.59	-1.70	-4.03	-1.88
$\Sigma_{(n,2n')}$	-1.28	-1.18	-1.39	-1.29
$\Sigma_{(n,n')p}$	-0.12	-0.11	-0.13	-0.12
$\Sigma_{(n,n')a}$	-0.01	-0.01	-0.01	-0.01
$\Sigma_{absorption}$	-0.80	-0.51	-0.78	-0.55
Σ_{total}	-7.55	-4.10	-8.23	-4.49

[a] eV·1.60·10^{-19}=J.

Table 7.9. Contributions to the Relative Variance and the Relative Deviation of the Neutron Scalar Flux, the Energy Deposition Per Unit Volume, and the Displacements Per Atom in the TF Coil from Iron Neutron Cross-Section Uncertainties

Variance of $(\delta R_j/R_j)$ in Percent Squared Due to Various Cross-Section Errors	Response	
	Energy Deposition Per Unit Volume (i=E)	Displacements Per Atom (i=D)
Uncertainties in $\Sigma_{elastic}$	5.17×10^2	6.32×10^2
Correlations between errors in $\Sigma_{elastic}$ and all other cross sections	-5.62×10^3	-6.80×10^3
Uncertainties in all cross sections other than $\Sigma_{elastic}$	1.53×10^4	1.83×10^4
Uncertainties in $\Sigma_{inelastic}$	4.61×10^3	5.59×10^3
Correlations between errors in $\Sigma_{inelastic}$ and all other cross sections	4.47×10^3	5.26×10^3
Uncertainties in all cross sections other than $\Sigma_{inelastic}$	1.08×10^3	1.24×10^3
Uncertainties in $\Sigma(n,2n')$	1.02×10^3	1.21×10^3
Correlations between errors in $\Sigma(n,2n')$ and all other cross sections	4.40×10^3	5.23×10^3
Uncertainties in all cross sections other than $\Sigma(n,2n')$	4.75×10^3	5.65×10^3
Uncertainties in $\Sigma(n,n')p$	3.53×10^1	4.15×10^1
Correlations between errors in $\Sigma(n,n')p$ and all other cross sections	1.13×10^3	1.33×10^3
Uncertainties in all cross sections other than $\Sigma(n,n')p$	9.00×10^3	1.07×10^4
Uncertainties in $\Sigma(n,n')a$	2.58×10^1	3.04×10^1
Correlations between errors in $\Sigma(n,n')a$ and all other cross sections	1.02×10^2	1.21×10^2
Uncertainties in all cross sections other than $\Sigma(n,n')a$	1.01×10^4	1.20×10^4

Table 7.9. (Contd.)

Variance of $(\delta R_i/R_i)$ in Percent Squared Due to Various Cross-Section Errors	Energy Deposition Per Unit Volume (i=E)	Displacements Per Atom (i=D)
Uncertainties in $\Sigma_{absorption}$	2.99×10^2	3.44×10^2
Correlations between errors in $\Sigma_{absorption}$ and all other cross sections	2.89×10^3	3.39×10^3
Uncertainties in all cross other than $\Sigma_{absorption}$	6.98×10^3	8.34×10^3
Standard deviation of $(\delta R_i/R_i)$ in percent due to all cross-section errors	101	110

These results must be accepted with much caution because of the very approximate manner in which the cross section uncertainties for iron were developed. However, they are still useful in identifying areas where cross section uncertainties may have significant impact (elastic, inelastic, (n,2n) and absorption cross sections of iron).

B. Fast Breeder Reactor

The sensitivity analysis for the CRBR upper axial shield experiment[195] indicates high sensitivities to the total cross section data for iron and sodium. This is discussed in some detail in the sensitivity analysis[196] of this series of deep penetration steel/sodium/iron configurations. Especially important is the energy region in the vicinity of the 300-keV and 500-keV sodium total cross section minima (see Fig. 7.2) and the 24-keV iron minima. Calculated sensitivity profiles for the CRBR mockup are available from the Radiation Shielding

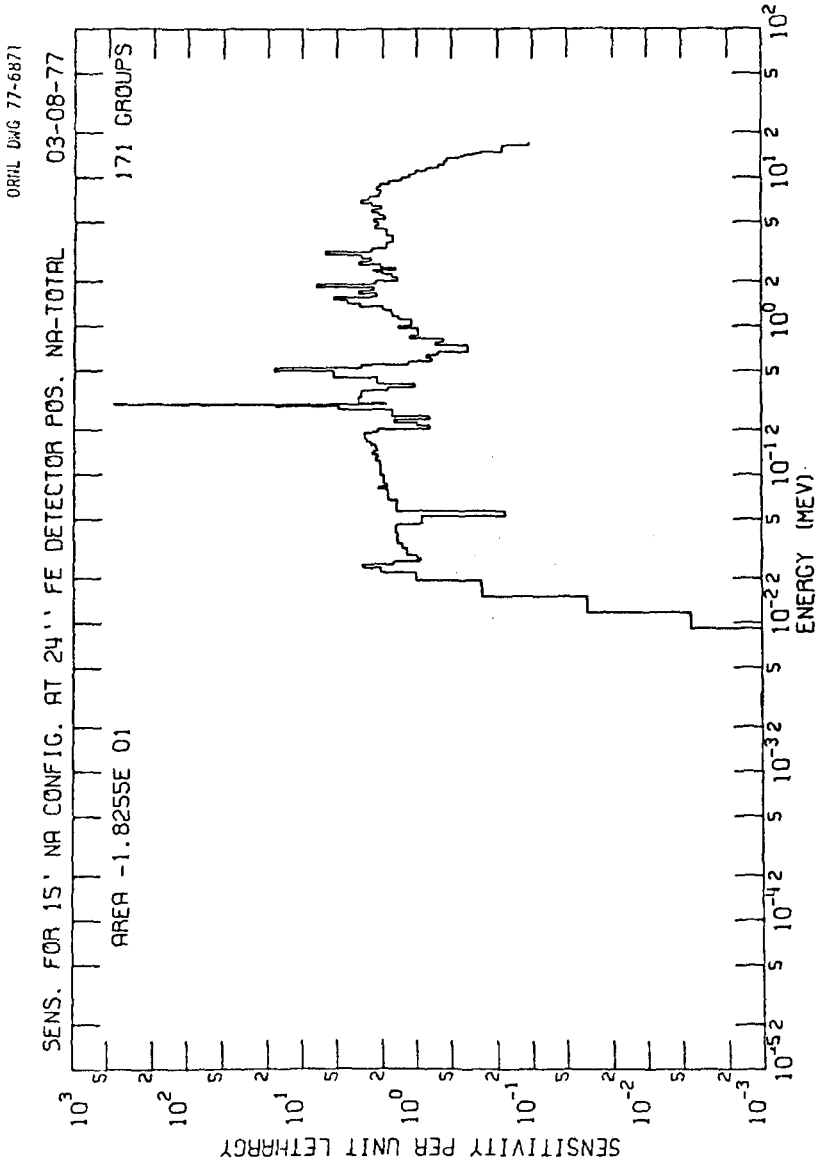

Fig. 7.2. Total sensitivity profile for sodium with detector at 24" iron position in 18"SS+15'Na+24"Fe configuration.

Information Center (RSIC) at ORNL as part of the DLC-45/SENPRO package. However, recalculation of the CRBR mockup with ENDF/B-V cross sections produced results which are inconsistent with the projections based upon sensitivity coefficients and changes from Version IV to Version V. It has been assumed, but not substantiated, that for this deep penetration problem, the system is highly nonlinear and first order analysis is simply not adequate.

7.5 Selected Fission Product and Actinide Sensitivity Information

Schenter[197] tabulated the important energy regions for each of the fission product and actinide reactions tested. Table 7.10 below lists the mean energies, median energies, and energy region which contributes 90% of the reaction response for those reactions starred in Chapter 6 for further consideration.

Table 7.10. Energy Regions of Importance for Reactions With Significant Calculation/Measurement Difference

Isotopic Reaction	Mean Energy (keV)	Median Energy (keV)	← (90% response) → Lower Energy	Upper Energy
(CFRMF)				
^{108}Pd	246	16.6	0.03	1473
^{109}Ag	177	34.6	0.11	974
^{134}Xe	373	135.	0.95	1708
^{137}Cs	177	2.5	1.35	1116
^{142}Ce	458	165.	2.45	2399
^{146}Nd	457	127.	.61	2780
^{148}Nd	344	64.5	.12	1917
^{150}Nd	233	41.2	.23	1343
^{152}Sm	225	29.5	.07	1297

Table 7.10. (Contd.)

Isotopic Reaction	Mean Energy (keV)	Median Energy (keV)	← (90% response) → Lower Energy	Upper Energy
(EBR-2)				
^{144}Nd	302	64.6	.31	1765
^{145}Nd	112	.89	.004	807
^{149}Sm	58.8	.70	.007	406
^{151}Eu	111.	9.16	.006	702
^{154}Eu	87.9	9.80	.009	500
(CFRMF)				
^{242}Pu(n,γ)	202	32.4	0.11	1147
^{241}Am(n,γ)	190	56.6	0.17	913
^{243}Am(n,γ)	99.2	10.5	0.10	605

7.6 Conclusions

The sensitivity coefficients and profiles described in this chapter are useful in identifying potential sources of calculational/experiment differences stemming from nuclear data, and in clarifying the energy regions of importance. In general, a great amount of structure is possible in the energy dependence of the sensitivity coefficients. Frequently, such structure can be traced back to particular cross section resonances, thresholds, etc.

Sensitivity coefficients can be used in the perturbation theory to estimate changes in calculated integral parameters which would accompany proposed cross section changes. It is in this context, that the measure of importance is established.

8.0 CONCLUSIONS

As is implied by the title, the main goal of this paper is to identify some of the outstanding cross section data required by the nuclear energy programs and to discuss how these data might be obtained.

In Section 2.0 we review published target accuracies for nuclear design and the economic implications of uncertainties in performance parameters. Target accuracies and achievable accuracies for power reactor parameters are proprietary so that published figures must be regarded as mere guidelines. Furthermore, as pointed out by Crowther et al., "the accuracy target is a moving one. The standards for predictive accuracy have progressively become tighter as reactor performance has been improved and constraints have been sharpened." For fast reactors, figures from different sources are not always consistent, illustrating that target accuracies reflect subjective judgement in the compromise between design conservatism and cost, and that there are different standards by which to estimate achieved accuracies. For instance, adjusted data may meet some requirements but some would prefer the prediction of safety parameters to be based only on differential data with integral measurements playing a confirmatory role. In a recent study McFarlane suggests that currently achieved accuracies for fast reactors are close to target; however the targets were set in the early seventies and based on the capabilities at that time. A revision of the target accuracies is probably desirable.

Target accuracies for shielding calculations are often related to specific design issues and hence are strongly dependent on a particular

design. Approximations in methods and modeling rather than uncertainties in basic nuclear data may dominate shielding analysis.

Within the context of the observations summarized above, the target accuracies and achievable accuracies for thermal and fast reactors, shielding, fuel handling and fusion reactors are summarized in Tables 2.2 to 2.10 of Section 2.0.

There are few published studies which analyze fully the economic implications of design uncertainties that include feedback from operating experience, design alternatives, etc. Uncertainties in the prediction of reactor properties result in excessive design margins and ultimately in cost. The fundamental study of Greebler, Hutchins, and Cowan on the economic implications of design uncertainties is now more than ten years old and in need of an update. The recent investigation of Becker and Harris shows substantial cost implications of nuclear data uncertainties on the LWR fuel cycle.

The determination of high accuracy data for technological applications is a complex process of measurements and evaluations discussed in Sections 3.0 and 4.0. Measurements of different data are closely interrelated. The interpretation, evaluation, and representation of experimental results are intimately interwoven with the development of nuclear models. These models are based on studies of systematic trends which necessitate an extensive data base. Experimental uncertainties are dominated by systematic errors which are not necessarily reduced by repeated measurements. The reduction of uncertainties requires new experimental techniques or improved methods of data reduction, interpretation or evaluation.

On the basis of these observations, an efficient long-range program of nuclear data measurements should not be limited to a specific request for a particular application but rather should address a broad range of measurements most likely to lead to improved accuracies of relevant data. This general improvement of an extensive data base is embodied in the cooperative development of an application-independent evaluated nuclear data file.

The status of selected data important for the design of thermal and fast reactors is examined in Section 5.0. The methodology for the estimation and representation of data uncertainties is still very much under development: the proper characterization of the uncertainties requires a very large variance-covariance matrix which cannot be reduced to a few numbers without severe oversimplification; hence a precise numerical comparison of achieved vs. required accuracies is somewhat elusive. One approach to determine the adequacy of evaluated nuclear data is through data testing and sensitivity analysis, as discussed in Sections 6.0 and 7.0.

Experimenters and evaluators consult request lists intended to guide measurement and evaluation programs. The IAEA maintains a World Request List of Nuclear Data (WRENDA) updated every other year. The U.S. Department of Energy publishes every year a Compilation of Request for Nuclear Data which defines priority, energy range, and accuracy desired, and also indicates the originator and purpose of the request. An effort is underway in CSWEG to make the U.S. request list even more precise and meaningful. These lists include requests for a large number of data over all energy ranges. Some of those requests are probably beyond present technology.

Requests for data originate from a broad area of technical concerns, including a desire for a sound understanding of the fundamental nuclear physics underlying reactor operation and for a resolution of discrepancies found in evaluating data or in interpreting integral measurements.

Several recent papers have reviewed the status of the data for thermal and fast reactor design, operation and shielding, for nuclear fuel cycle studies and for the fusion program. Many of these papers are listed in Section 5.0.

The analysis of the CSWEG benchmark data testing is reported in Section 6. The primary objective of the CSWEG data testing is to assess the adequacy of the ENDF/B-V nuclear data and to identify nuclear data or computational methods in need of further work. However, it is important to recognize that existing benchmarks test only a limited domain of important data. Furthermore the benchmark testing of ENDF/B-V is very incomplete: for instance ^{233}U and ^{232}Th were not tested at all and several series of thermal benchmark experiments were not analyzed with Version V data. Also, the quality of the analyzed uncertainty estimates for existing benchmarks is very uneven. Finally, there has been only a minimal effort for thermal benchmarks and no effort elsewhere to quantify uncertainties due to calculational methods.

It is important to stress that benchmark testing can only assess the adequacy of the data. Where discrepancies are perceived, sensitivity analysis as described in Section 7.0 can indicate to what data the discrepancy is sensitive, but as pointed out in Section 4.0 the data are interrelated through standards and their evaluation is based on nuclear models,

so that a recommendation for reevaluation or remeasurement can be made only on the basis of a detailed investigation of the ingredients of the evaluations: measurements, assumptions, models, etc. This observation explains partly why request lists often involve data, particularly standards, which are not tested by benchmark testing.

As was illustrated in Section 5.0, recent measurements and new evaluations had a significant impact in our ability to confidently predict nuclear fuel cycle parameters. Some of the remaining perceived discrepancies pointing to specific areas of nuclear data are summarized below.

A. Fast Benchmarks:

(a) The ^{238}U capture to ^{239}Pu (or ^{235}U) fission ratio is overpredicted. This discrepancy is most sensitive to the $^{238}U(n,\gamma)$ cross section in the range 10-100 keV. This is a long-standing discrepancy. The ENDF/B-V ^{238}U evaluators have stated emphatically that the measurements do not support the adjustment required to remove the observed overprediction.

(b) The central fissile worth is overpredicted. The interpretation of this discrepancy is not clear.

(c) The ^{238}U to ^{235}U fission ratios in small fast assemblies is not predicted correctly. The discrepancy is usually interpreted as pointing to an inadequate evaluation of the neutron-induced neutron emission spectra in fissile materials.

B. Thermal Benchmarks:

A discrepancy between the measured and computed ^{235}U capture to fission ratio suggests a reevaluation of the ^{235}U resonance data.

C. Shielding Applications

Data testing suggests inadequate total cross sections of C, Ni, and Cr in the 1-10 MeV region and inadequate capture gamma-ray-production data for Cl and Cr. Other important materials do not have photon-production data in the Version V evaluations.

D. Fission Products and Actinide Testing

The data for which the calculated over measured ratio is significantly large have been flagged with a dagger (†) in Table 6.7 of Section 6.0.

E. Dosimetry

Discrepancies are defined to exist for reactions where the C/E (see Tables 6.10 and 6.11) is $\geq$ 10% (the required amount specified by the ILRR program.) The reactions include Al(n,p), Mn(n,2n), ^{47}Ti(n,p), ^{48}Ti(n,p), ^{58}Ni(n,2n), ^{63}Cu(n,α), ^{126}I(n,2n), ^{232}Th(n,γ), ^{6}Li(n, tot He), ^{10}B(n, tot He), ^{58}Fe(n,γ) and ^{237}Np(n,f).

The discrepancies for Mn(n,2n) and ^{58}Ni(n,2n) are probably caused by thresholds of high energies, i.e., in the region of the "high energy tail" of the spectrum which is not well defined.

F. Fusion Applications

Total cross section and photon-production data with accuracies between ±10 - 15% in the energy range below a few MeV to ~50 MeV are required to meet shielding needs for both test facilities and reactors. Data for radiation damage and dosimetry studies are required with accuracies of ±10 - 40% over this same energy region (see Table 7.6). The

Li(n,n')T cross section must also be reevaluated to resolve problems of tritium breeding (see Table 7.7).

As noted previously, benchmarks test only a limited domain of important data and perceived discrepancies cannot always point clearly to the specific data in need of improvement. From a study of the request lists, the review papers listed in Section 5.0, and the results of benchmark testing, we identify some important nuclear data measurements and evaluations required in the near term for reactor design. These include:

(1) The ^{235}U fission cross section (a standard) requested to 1% accuracy up to at least 14 MeV (an energy where absolute measurements have been performed with neutrons from the D-T reaction). (2) The value of $\bar{\nu}$ (S.F.) for ^{252}Cf requested to .25% accuracy. (The recent measurement of R. R. Spencer, when finalized, might satisfy this request.) (3) The cross sections of the fertile and fissile nuclei in the resonance region (below 10 keV). (4) The cross sections of the fissile and fertile nuclei below 1 eV. (5) The spectra of secondary neutrons emitted from scattering and fission in ^{235}U, ^{238}U, and ^{239}Pu.

In addition, there is a consensus of measurers, evaluators, and designers that data covariance files need to be improved. For shielding and fusion applications, measurements and evaluations need to be extended in the MeV region, as discussed in Section 6.0. The above list is illustrative but not exhaustive.

We also perceive a need for a more extensive documentation of the process by which design requirements and results of data testing lead to specific requests for measurements and evaluation.

REFERENCES

1. P. Greebler et al., "Implications of Nuclear Data Uncertainties in Reactor Design," 2nd International Conference on Nuclear Data for Reactors, Helsinki, CN-26/102 (June 1970).

2. M. Becker and D. R. Harris, "Sensitivity of Nuclear Fuel Cycle Cost to Uncertainties in Nuclear Data," EPRI NP-1632 (1980).

3. S. A. W. Gerstl, D. J. Dudziak and D. W. Muir, "The Application of Sensitivity Analysis to Nuclear Data Assessment," Proceedings Specialists Meeting on Sensitivity Studies and Shielding Benchmarks, Paris, France (October 7-9, 1975).

4. R. G. Alsmiller, J. Barish, C. R. Weisbin, "Uncertainties in Calculated Heating and Radiation Damage in the Toroidal Field Coil of a Tokamak Experimental Power Reactor Due to Neutron Cross Section Errors," Nucl. Tech., Volume 34 (August 1977).

5. J. Butler, "Nuclear Data for Reactor Shielding," Neutron Physics and Nuclear Data, Proceedings of International Conference, p. 28, Harwell, England (September 1978).

6. J. L. Rowlands, "Nuclear Data for Reactor Design, Operation, and Safety," Neutron Physics and Nuclear Data, Proceedings of International Conference, p.7, Harwell, England (September 1978).

7. "Nuclear Fission and Neutron-Induced Fission Cross Sections," Chapter 3, A. Michaudon, editor, Pergamon (1981).

8. R. L. Crowther et al., "Feedback of Reactor Operating Data to Nuclear Methods Development," Advances in Reactor Physics, CONF-780401 (April 1978).

9. H. McFarlane, "The Use of Critical Experiments in CRBR Design: Status and Historical Perspective," ANL-80-119 (November 1980).

10. H. Kusters, "Progress in Fast Reactor Physics in the Federal Republic of Germany," KFK-1632 (August 1973).

11. P. Greebler and B. A. Hutchins, "Significance of Integral Parameters in Design and Performance of Fast Breeder Reactors," New Developments in Reactor Physics and Shielding, CONF-720901, pp. 928-939.

12. F. Helm, "Summary of Results for the SNEAK-9 Series of Critical Experiments and Conclusions for the Accuracy of Predicted Physics Parameters of the SNR-300," KFK-2586 (August 1978).

13. P. Hammer, "Nuclear Data Needs for Plutonium Breeders," Nuclear Cross Sections for Technology, NBS Special Publication 594, p. 6 (October 1979).

14. V. Herrnberger et al., "Required Target Accuracies," Paper presented at the Specialists Meeting on Sensitivity Studies and Shielding Benchmarks, Paris, (October 1975).

15. "Transactinium Isotope Nuclear Data (TND)" Vol. 1, Proceedings of an Advisory Group Meeting on Transactinium Isotope Nuclear Data, Karlsruhe 3-7, (IAEA-186)(November 1975).

16. Richard W. Benjamin, "Status of Measured Neutron Cross Sections of Transactinium Isotopes for Thermal Reactors," Proceedings of an Advisory Group Meeting on Transactinium Isotope Nuclear Data, Karlsruhe, Vol. II, IAEA-186 (1976).

17. C. R. Head, "Nuclear Data Requirements of the Magnetic Fusion Power Program of the United States of America," Proceedings of the Advisory Group Meeting on Nuclear Data for Fusion Reactor Technology organized by IAEA, Vienna, IAEA-TECDOC-223, p. 1 (1979).

18. G. Constantine, "Nuclear Data Requirements for Fusion Reactor Design, Neutronics Design, Blanket Neutronics and Tritium Breeding," Proceedings of the Advisory Group Meeting on Nuclear Data for Fusion Reactor Technology organized by IAEA, Vienna IAEA-TECDOC-223, p. 11 (1979).

19. Y. Seki, "Review of Nuclear Heating in Fusion Reactors," Proceedings of the Advisory Group Meeting on Nuclear Data for Fusion Reactor Technology organized by IAEA, Vienna, IAEA-TECDOC-223, p. 31 (1979).

20. O. N. Jarvis, "Nuclear Data Requirements for Transmutation and Activation of Reactor Wall and Structural Materials," Proceedings of the Advisory Group Meeting on Nuclear Data for Fusion Reactor Technology organized by IAEA, Vienna, IAEA-TECDOC-223, p. 47 (1979).

21. S. M. Qaim, "Nuclear Data Needs for Radiation Damage Studies Relevant to Fusion Reactor Technology," Proceedings of the Advisory Group Meeting on Nuclear Data for Fusion Reactor Technology organized by IAEA, Vienna, IAEA-TECDOC-223, p. 75 (1979).

22. M. A. Abdou, "Nuclear Data Requirements for Fusion Reactor Shielding," Proceedings of the Advisory Group Meeting on Nuclear Data for Fusion Reactor of the Advisory Group Meeting on Nuclear Data for Fusion Reactor Technology organized by IAEA, Vienna, IAEA-TECDOC-223, p. 91 (1979).

23. D. V. Markovskii and G. E. Shatalov, "Neutron Data for Hybrid Reactor Calculations," Proceedings of the Advisory Group Meeting on Nuclear Data for Fusion Reactor Technology organized by IAEA, Vienna, IAEA-TECDOC-223, p. 111 (1979).

24. R. C. Haight, "Neutron Cross Sections for Fusion," Proceedings of the International Conference on Nuclear Data for Technology, NBS Spec. Pub. 594, Knoxville, TN, pp. 228-238 (October 1979).

25. J. C. Browne and P. W. Lisowski, "Status of High Energy Neutron Cross Sections," BNL-NCS-51245, p. 215 (1980).

26. M. R. Bhat and M. A. Abdou, "Nuclear Data Requirements for Fusion Reactor Nucleonics," Proc. of the Fourth Topical Meeting on the Technology of Controlled Nuclear Fusion, King of Prussia, PA. (October 14-17, 1980).

27. S. Pearlstein, "Evaluation and Processing of Nuclear Data," BNL-NCS-27892 (February 1980).

28. J. E. Lynn, "Theoretical Methods for Calculating the Cross Sections of Fissionable Nuclei," in Nuclear Fission and Neutron-Induced Fission Cross Sections, A. Michaudon, Editor, Pergamon Press (1981).

29. R. E. Schenter et al., "Evaluations of Fission Product Cross Sections for ENDF/B-V," Proceedings International Conference on Nuclear Cross Sections and Technology, Knoxville (October 1979).

30. H. Gruppelaar, "Uncertainty Estimates of Statistical Theory Calculations of Neutron Capture Cross Sections of Fission Products," Nuclear Theory in Neutron Nuclear Data, IAEA-190, II 61 (1976).

31. H. Gruppelaar and J. B. Dragt, "Cross Section Adjustments Using Integral Data," Proceedings of the Conference on Nuclear Data Evaluation Methods and Procedures held at BNL September 22-25, 1980 INDC(USA)-85, p. 133 (1981).

32. W. P. Poenitz, "Fast Neutron Cross Sections of Importance in Technological Applications," Neutron Cross Sections for Technology, NBS Special Publication 594, p. 368 (1980).

33. W. P. Poenitz, "Data Interpretation, Objective Evaluation Procedures and Mathematical Techniques for the Evaluation of Energy-Dependent Ratio, Shape and Cross Section Data," Proceedings of the Conference on Nuclear Data Evaluation Methods and Procedures held at BNL September 22-25, 1980 INDC(USA)-85, p. 249 (1981).

34. P. T. Guenther, W. P. Poenitz and A. B. Smith, "Interpretation and Normalization of Experimental Data for Total, Scattering, and Reaction Cross Section," Proceedings of the Conference on Nuclear Data Evaluation Methods and Procedures held at BNL September 22-25, 1980 INDC(USA)-85, p. 567 (1981).

35. G. M. Hale, "Use of R-Matrix Methods for Light Element Evaluations," Proceedings of the Conference on Nuclear Data Evaluation Methods and Procedures held at BNL September 22-25, 1980 INDC(USA)-85, p. 509 (1981).

36. "Nuclear Theory in Neutron Nuclear Data Evaluation," Proceedings of a Consultant Meeting held at Trieste, December 8-11, 1975, IAEA-190 (1976).

37. Proceedings of the Conference on Nuclear Data Evaluation Methods and Procedures held at Brookhaven National Laboratory, Sept. 22-25, 1980, INDC(USA)-85 (1981).

38. E. P. Wigner and L. Eisenbud, "Higher Angular Momenta and Long Range Interaction in Resonance Reaction, Phys. Rev. 72, p. 29 (1947).

39. H. Fesbach, C. E. Porter and V. F. Weisskopf, "Model for Nuclear Reactions with Neutrons," Phys Rev. 96, p. 448 (1954).

40. D. G. Madland, "Prompt Fission Neutron Spectra and $\bar{\nu}_p$," Proceedings of the Conference on Nuclear Data Evaluation Methods and Procedures held at BNL September 22-25, 1980, INDC(USA)-85, p. 861 (1981).

41. L. Stewart, "Methods Used in Evaluating Data for the Interaction of Neutrons with Light Elements," Proceedings of the Conference on Nuclear Evaluation Methods and Procedures held at BNL September 22-25, 1980, INDC(USA)-85 (1981).

42. S. F. Mughabghab and D. I. Garber, "Neutron Cross Sections," BNL-325 (1981).

43. K. Way, editor, "Nuclear Data Tables," Academic Press.

44. H. H. Barschall, "Regularities in the Total Cross Sections for Fast Neutrons," Phys. Rev. 80, p. 431 (1952).

45. J. W. Muller, "Some Second Thoughts on Error Statements," J. of Nucl. Instr. and Meth. 163, 241 (1979).

46. H. Derrien and P. Ribon, "On the Shape Analysis of Various ^{232}Th and ^{238}U Transmission Data," Specialists Meeting on Resonance Parameters of Fertile Nuclei and ^{239}Pu, NEANDC(E)-163/U, p. 63 (1975).

47. "Fast Neutron Physics," Editors J. B. Marion and J. L. Fowler (Interscience Publishers, 1960), "Experimental Neutron Spectroscopy," Editor J. A. Harvey (Academic Press, 1970), "Neutron Physics and Nuclear Data in Science and Technology," (Series published by the OECD).

48. W. J. Youden, Technometrics 14, 1 (1972), see also H. H. Ku, NBS Special Publication 300 (1969)

49. J. R. Smith, "Status of the Quest for ^{252}Cf $\bar{\nu}$," EPRI, NP-1258 (1979).

50. F. C. Maienschein, "Neutron Cross-Section Measurements of Reactor Interest with the Oak Ridge Electron Linear Accelerator (ORELA), Energia Nuclear, 79, 533 (1972).

51. J. A. Harvey and F. C. Maienschein, "Comparison of Pulsed-Neutron Sources for Time-of-Flight Neutron Spectroscopy," ORNL/TM-582 (1963).

52. J. W. T. Dabbs, "Neutron Cross Section Measurements at ORELA," in Proceedings of the Intern. Conf. on Nuclear Cross Sections for Technology, NBS-SP-594, p. 929 (1980).

53. R. W. Peelle et al., "Future Neutron Reactor Research and Facility Developments at ORELA," ORNL/TM-8225 (1981).

54. S. Raman, "Basic and Applied Research with Neutrons at ORELA," unpublished (1979).

55. E. R. Rae and W. M. Good, "Pulsed Accelerator Time-of-Flight Spectrometers," in Experimental Neutron Resonance Spectroscopy, Academic Press (1970).

56. Reports to the DOE Nuclear Data Committee, compiled by NNDC, BNL.

57. R. W. Peelle, "An Evaluation for ENDF/B-IV of the Neutron Cross Sections for ^{235}U from 82 eV to 25 eV," ORNL-4955 (1976).

58. ENDF/B-V file for ^{233}U (MAT-1393); principal evaluators: L. Stewart, et al., (LANL); see also BNL-NCS-17541 (ENDF-201), Third Edition (ENDF/B-V, edited by R. Kinsey, available from Brookhaven National Laboratory, Upton, NY.) (July 1979).

59. ENDF/B-V file for ^{235}U (MAT-1395); principal evaluator: M. R. Bhat (BNL); see also M. R. Bhat, "Evaluation of ^{235}U Neutron Cross Section and Gamma Ray Production Data for ENDF/B-V," BNL-NCS-51184 (1980).

60. ENDF/B-V file for ^{239}Pu (MAT-1399); principal evaluators: E. Kujawski (NUTECH, San Jose, CA), L. Stewart (LANL), and R. Labauve (LANL); see also BNL-NCS-17541 (ENDF-201), Third Edition (ENDF/B-V) edited by R. Kisney.

61. G. D. James and G. de Saussure, "Measurements of Fission Cross Sections," in Nuclear Fission and Neutron Induced Fission Cross Sections, A. Michaudon, Editor, Pergamon Press (1981).

62. D. C. Larson, "ORELA Measurements to Meet Fusion Energy Cross Section Data Needs," BNL-NCS-51245, I, 277 (1980).

63. F. G. Perey, "Logical Inference and Evaluation," BNL-NCD-51363, I, 58 (1981).

64. R. W. Peelle, "Uncertainty in Nuclear Data Used for Reactor Calculations," see C. R. Weisbin et al., Chap. 2 of Sensitivity and Uncertainty Analysis of Reactor Performance Parameter, Plenum Press, to be published (1982).

65. D. W. Muir and R. J. LaBauve, COVFILS, A 30-Group Covariance Library Based on ENDF/B-V, LA-8733-MS (1981).

66. J. D. Smith, III and B. L. Broadhead, "Multigroup Covariance Matrices for Fast Reactor Studies," ORNL/TM-7389 (1981).

67. WRENDA 81/82, World Request List for Nuclear Data, INDC(SEC)-78/URSF, N. Day, IARA, Editor.

68. Compilation of Requests for Nuclear Data, BNL-NCS-51354 (1981).

69. O. Ozer, Seminar on 238U Resonance Capture, BNL-NCS-50451 (1975).

70. E. T. Tomlinson et al., "Sensitivity Analysis of TRX-2 Lattice Parameters with Emphasis on Epithermal ^{238}U Capture," EPRI-NP-346 ENDF-252 (March 1977).

71. J. Hardy, Jr., Benchmark Testing of ENDF/B-V Data for Thermal Reactors.

72. H. Tellier, "Effective Integral and Reactivity Effects of ^{238}U and ^{240}Pu Resonance Parameter Uncertainties," IAEA Consultant Meeting, September 28-October 2, 1981.

73. J. Hardy, Jr., ^{235}U Resonance Fission Integral and Alpha Based on Integral Measurements, ENDF-300 (1979).

74. B. Zolotar, "Interaction of Methods and Data in Industrial Experience," EPRI-NP-1098, 7975-1, June Summary Report (1979).

75. J. Rowlands, Working Group on Accuracy Requirements on U and Pu Resonance Parameters for Reactor Applications (1981).

76. H. Lemmel, NBS Special Publication-493, p. 170 (1977).

77. L. S. LeSage, Status of Fast Reactor Physics and Shielding, p. 51 (1980).

78. J. H. Marable et al., "Uncertainty in the Breeding Ratio of a Large LMFBR Theory and Results," Nucl. Sci. Eng., 75, pp. 30-55 (1980).

79. J. L. Rowlands, BNL-NCS-51363, I, 23 (1981).

80. M. Salvatores et al., Resonance Parameter Data Uncertainty Effects on Integral Characteristics of Fast Reactors (1981).

81. O. Ozer, EPRI-Journal, December 1979; see also O. Ozer in NBS Special Publication 594, p. 1 (1980).

82. S. F. Mughabghab and M. Divadenum, "Some Highlights in the Evaluations of the Thermal Cross Sections and Resonance Parameters of the Actinides," BNL-NCS-30130 (1981).

83. T. Nakagowa, "Evaluation of the Resonance Parameters of ^{238}U, ^{240}Pu, and ^{242}Pu for JENL-2," (1981).

84. G. de Saussure and R. Q. Wright, "Comparison of the ENDF/B-V and SOKRATOR Evaluations of ^{235}U, ^{249}Pu, ^{240}Pu, and ^{242}Pu at Low Neutron Energies," (1981).

85. D. K. Olsen et al., "Precise Measurement and Analysis of Neutron Transmission Through Uranium-238," Nucl. Sci. Eng., 62, p. 479, (1977).

86. H. I. Liou and R. E. Chrien, "Epithermal Neutron Capture in Uranium-238," Nucl. Sci. Eng., 62, 463-478 (1977).

87. P. Stavelos and E. Cornelis, "Precise Measurement of the Capture Width of the 6.67 - eV Neutron Resonance of Uranium-238," Nucl. Sci. Eng., 66, 349 (1978).

88. H. E. Jackson and J. E. Lynn, "Resonance Absorption of Neutrons by Crystals," Phys. Rev., 127, 461 (1962).

89. G. de Saussure and M. Bhat, "Experimental Differential Data and Resonance Analysis," BNL-NCS-50451, p. XXIV (1975).

90. G. de Saussure et al., "The ENDF/B-IV Representation of the Uranium-238 Total Neutron Cross Section in the Resolved Resonance Energy Region," Nucl. Sci. Eng., 61, 496 (1976).

91. R. C. Block et al., Uranium-238 Resonance Self Indication Capture Measurements," Nucl. Sci. Eng., 80 (2) 263-281 (February 1982).

92. G. de Saussure et al., "Evaluation of the ^{238}U Neutron Cross Sections for Incident Neutron Energies up to 4 keV," Prog. in Nucl. Energy, 3, 87 (1979).

93. M. R. Bhat, "^{235}U Fission Cross Section," ENDF-300 (1979).

94. M. R. Bhat, "Evaluation of ^{235}U Neutron Cross Section and Gamma Ray Production Data for ENDF/B-V," BNL-NCS-51184 (1980).

95. G. de Saussure and R. B. Perez, "Representation of the Neutron Cross Sections of Several Fertile and Fissile Nuclei in the Resonance Regions," ORNL/TM-7945 ENDF-312 (1981).'

96. M. S. Moore et al., "Problems and Progress Regarding Resonance Parameterization of ^{235}U and ^{239}Pu for ENDF/B Submitted to Uranium and Plutonium Isotope Resonance Parameters," INDC(NDS) - 129/G.J. LA-UR-81-2549, P. 74 (1981).

97. A. DeVolpi and K. G. Porges "Neutron Yields of ^{252}Cf Based on Absolute Measurements of the Neutron Rate and Fission Rate," Phys. Rev. C-1, p. 683 (1970).

98. E. J. Axton et al., J. Nucl. Energy A/B23, 457 (1968).

99. R. R. Spencer, "Absolute Measurement of $\bar{\nu}_p$ for ^{252}Cf by the Large Liquid Scintillator Tank Technique," NBS Special Publication, p. 728, 594 (1980).

100. R. Gwin et al., "Measurements of the Average Number of Prompt Neutrons Emitted per Fission of ^{239}Pu and ^{235}U," ORNL/TM-6246 ENDF-264 (1978).

101. J. R. Smith, "$\bar{\nu}$ of ^{252}Cf (Spontaneous Fission)," ENDF-300 (1979).

102. F. C. Maienschein, "The Status of Reactor Shielding Research in the United States," 198 Advances, p. 70 (1980).

103. C. R. Weisbin, R. D. McKnight, J. D. Hardy, Jr., R. W. Roussin, R. E. Schenter and B. Magurno, "Benchmark Testing of ENDF/B-V," (to be published in 1982).

104. ENDF-202 Cross Section Evaluation Working Group Benchmark Specifications, BNL-19302 ENDF-202 (November 1974).

105. G. E. Hansen and H. C. Paxton, "Reevaluated Critical Specifications of Some Los Alamos Fast-Neutron Systems," Los Alamos National Laboratory, LA-4208 (1969).

106. (a) G. E. Hansen, "Status of Computational and Experimental Correlations for Los Alamos Fast Neutron Critical Assemblies," Proc. Seminar Physics of Fast and Intermediate Reactors, Vienna, August 3-11, 1961 (IAEA, Vienna, 1962) Vol. I, pp. 445-455. (b) J. A. Grundl and G. E. Hansen, "Measurement of Average Cross Section Ratios in Fundamental Fast-Neutron Spectra," Proc. Paris Conf. Nuclear Data for Reactors (IAEA, Vienna, 1967) Vol. I, pp. 321-336. (c) P. I. Amundson, A. M. Broomfield, W. G. Davey, and J. J. Stevenson, "An International Comparison of Fission Detector Standards," Proc. Inter. Conf. Fast Critical Experiments and Their Analysis, Argonne, ANL-7320, pp. 679-687, ANL-7320 (October 1966). (d) D. M. Gilliam, "Integral Measurement Results in Standard Fields," proc. Intern. Specialist's Symposium Neutron Standards and Applications, Gaithersburg, NBS Special Publication 493, pp. 293-303 (1977). (e) G. A. Lineberger and L. L. Lowry, "Neutron Detector Traverses in the Topsy and Godiva Critical Assemblies," Los Alamos National Laboratory, LA-1653 (1953).

107. C. E. Byers, "Cross Sections of Various Materials in the Godiva and Jezebel Critical Assemblies," Nucl. Sci. Eng., 8, pp. 608-614, (1960).

108. L. B. Engle, G. E. Hansen and H. C. Paxton, "Reactivity Contributions of Various Materials in Topsy, Godiva, and Jezebel," Nucl. Sci. Eng., 8, pp. 543-569, (1960).

109. L. Stewart, "Leakage Neutron Spectrum from a Bare ^{239}Pu Critical Assembly," Nucl. Sci. Eng., 8, pp. 595-597 (1960).

110. C. E. Till, L. G. LeSage, R. A. Karam et al., "ZPR-6 Assemblies 6A and 7: Benchmark Specifications for the Two Large Single-Core-Zone Critical Assemblies - ^{235}U-Fueled Assembly 6A and Plutonium-Fueled Assembly 7 - LMFBR Demonstration Reactor Benchmark Program," Applied Physics Division Annual Report, ANL-7910, pp. 86-101, (1971).

111. C. E. Till, J. M. Gasidlo, E. F. Groh et al., "Null Reactivity Measurements of Capture/Fission Ratio in ^{235}U and ^{239}Pu," Nucl. Sci. Eng., 40, p. 132 (1970).

112. L. G. LeSage, E. M. Bohn and J. E. Marshall, "Sodium-Void and Small-Sample Reactivity-Worth Measurements in ZPR-6 Assembly 7." Applied Physics Division Annual Report, ANL-7910, pp. 414-454, (1971).

113. R. D. McKnight, "Benchmark Specifications in CSEWG Format for ZPR-9 Assembly 31, The Advanced Fuels Program Carbide Benchmark Critical Assembly," ZPR-TM-281 (1977).

114. L. G. LeSage, E. M. Bohn, D. C. Wade and R. B. Pond, "Program Description-Advanced Fuels Critical Experiments on ZPR-9," ZPR-TM-256 (1976). [See also ZPR-TM-275].

115. J. A. Morman, I. K. Olson and W. R. Robinson, "Reaction Rate Measurements in the Carbide Benchmark Assembly of the Advanced Fuels Program," ZPR-TM-299 (1977).

116. M. M. Bretscher and I. K. Olson, "Measurement of Absorption-to-Fission Ratios and the Central Core Conversion Ratio in the Carbide Benchmark Assembly of the ZPR-9 Advanced Fuels Program," ZPR-TM-296 (1977). [See also ZPR-TM-309].

117. R. G. Bucher, D. M. Smith and R. Nepomechie, "Small-Sample Reactivity-Worth Measurements in Advanced Fuels Program," ANL-RDP-63 (1977). [Also to be issued as ZPR-TM report].

118. R. W. Schaefer and D. C. Wade, "The Advanced Fuels Critical Experiments Program: Calculational Models and Results," ZPR-TM-307 (1978).

119. R. D. McKnight, "Benchmark Testing Using ENDF/B-III and -IV," Nucl. Sci. Eng., 62, pp. 309-330 (1977).

120. P. J. Collins et al., "Experimental Studies of 350 MWe Heterogeneous LMFBR at ZPPR," Proceedings of Symposium - Fast Reactor Physics, (Aix-em-Provence, France) IAEA-SM-244/70 (1979).

121. M. L. Lineberry, "Experimental Studies of Large-Conventional LMFBR Cores at ZPPR," Proceedings of Symposium-Fast Reactor Physics, (Aix-em-Provence, France) IAEA-SM-244/86 (1979).

122. S. K. Bhatacharyya, "An Experimental Study of the Neutronics for the First Gas-Cooled Fast Reactor Benchmark Assembly (GCFR-Phase 1 Assembly)" ANL-76-36 (1976). [See also ANL-76-108 (1976) and ANL-79-35 (1979)].

123. S. K. Bahbatt et al., "A Critical Experimental Study of Integral Physics Parameters in Simulated Liquid Metal Fast Breeder Reactor Melt-Down Cores," Nuclear Technology, V. 46, pp. 517-524 (1979).

124. R. E. Schaefer and R. G. Bucher, "Material Worths in the U-9 Diagnostic Critical Assemblies," to be published in the Proceedings of ANS Topical Meeting on Advances in Reactor Physics and Core Thermal Hydraulics to be held September 22-24, 1982, Lake Kiamesha, NY.

125. C. R. Weisbin, J. H. Marable, J. Hardy, Jr. and R. D. McKnight, "Sensitivity Coefficient Compilation for CSEWG Data Testing Benchmarks," ENDF-265 BNL-NCS-24853 (1978).

126. R. Gwin and D. W. Magnuson, "Eta of ^{233}U and ^{235}U for Critical Experiments," Nucl. Sci. Eng., 12, p. 364 (1962).

127. A. Staub et al., "Analysis of a Set of Critical Homogeneous U-H_2O Spheres," Nucl. Sci. Eng., 34, p. 263 (1968).

128. J. Hardy, Jr., D. Klein and J. J. Volpe, "A Study of Physics Parameters in Several H_2O-Moderated Lattices of Slightly Enriched and Natural Uranium," WAPD-TM-931 (March 1970).

129. J. Hardy, Jr., D. Klein and J. J. Volpe, "A Study of Physics Parameters in Several H_2O-Moderated Lattices of Slightly Enriched and Natural Uranium," Nucl. Sci. Eng., 40, pp. 101-115 (1970).

130. J. J. Volpe, J. Hardy, Jr. and D. Klein, "A Comparison of Thermal-Neutron-Activation Measurements and Monte Carlo Calculations in Light-Water Moderated Uranium Cells," Nucl. Sci. Eng., 40, pp. 116-127 (1970).

131. J. Hardy, Jr., D. Klein and R. Dannels, "Spatial Variation of the Fast Neutron Flux in Cells of Slightly ^{235}U-Enriched-Uranium Water-Moderated Lattices," Nucl. Sci. Eng., 26, pp. 462-471 (1966).

132. J. R. Brown et al., "Kinetics and Buckling Measurements in Lattices of Slightly Enriched U or UO_2 Rods in H_2O." WAPD-176 (January 1958).

133. R. Sher and S. Fiarman, "Studies of Thermal Reactor Benchmark Data Interpretation: Experimental Corrections," EPRI-NP-209 (1976).

134. R. L. Hellens, G. A. Price, "Reactor Physics Data for Water-Moderated Lattices of Slightly Enriched Uranium," Reactor Technology Selected Reviews, p. 529 (1964).

135. J. R. Brown and D. R. Harris et al., "Kinetic and Buckling Measurements on Lattices of Slightly Enriched Uranium and UO_2 Rods in Light Water," WAPD-176 (1958).

136. W. Rothenstein, "Thermal Reactor Lattice Analysis Using ENDF/B-IV Data with Monte Carlo Resonance Reaction Rates," BNL-204346 (1975); Nucl. Sci. Eng., 59, pp. 337-349 (1976).

137. T. J. Thompson et al., "Heavy Water Lattice Project Annual Report, September 30, 1966," MIT-2344-09 (1966).

138. P. F. Palmedo et al., "Measurements of the Material Bucklings of Lattices of Natural Uranium Rods in D_2O," NYO-9660 (MITNE-13) (1962).

139. T. J. Thompson et al., "Heavy Water Lattice Project Final Report," MIT-2344-12 (1967).

140. D. R. Finch and W. E. Graves, "Quantitative Consistency Testing of Thermal Benchmark Lattice Experiments," Trans. of ANS, 27, p. 888 (1977). Also DP-MS-77-33, E. I. du Pont, Savannah River Laboratory (1977).

141. R. C. Lloyd et al., "Criticality Studies with Plutonium Solutions," Nucl. Sci. Eng., 25, p. 165 (1966).

142. L. E. Hansen and E. D. Clayton, "Theory-Experiment Tests Using ENDF/B Version II Cross-Section Data," Trans. Am. Nucl. Soc., 15, p. 309 (June 1972).

143. F. E. Kruesi, et al., "Critical Mass Studies of Plutonium Nitrate Solution," HW-24514 (1952).

144. R. C. Lloyd, C. R. Richey, E. D. Clayton and D. R. Skeen, "Criticality Studies with Plutonium Solutions," Nucl. Sci. Eng., 25, 165-173 (1966).

145. F. F. Kruesi, J. O. Erkman and D. D. Lanning, "Critical Mass Studies of Plutonium Solutions," HW-24514 DEL, p. 67 (1952).

146. F. Abbey, P. J. Hemmings and E. R. Woodlock, "Data Checks on Plutonium Systems," HSRC/CSC p. 24 (UK).

147. R. C. Lloyd, E. D. Clayton and L. E. Hansen, "Criticality of Plutonium Nitrate Solution Containing Soluble Gadolinium," Nucl. Sci. Eng., 48, pp. 300-304, (1972).

148. R. C. Lloyd and E. D. Clayton, "The Criticality of High Burnup Plutonium," Nucl. Sci. Eng., 52, pp. 73-75 (1973).

149. U. P. Jenquin and S. R. Bierman, "Benchmark Experiments to Test Plutonium and Stainless Steel Cross Section," NUREG/CR-0210, PNL-2273 (June 1978).

150. R. I. Smith and G. J. Konzek, "Clean Critical Experiment Benchmarks for Plutonium Recycle in LWR's," EPRI-NP-196, Vol. 1 (April 1976).

151. R. C. Liikala, V. O. Uotinen and U. P. Jenquin, "Lattices of Plutonium Enriched Rods in Light Water - Part II Theoretical Analysis of Plutonium Fuel Systems," Nucl. Tech., 15, p. 272 (August 1972).

152. R. Sher and S. Fiarman, "Analysis of Some Uranium Oxide and Mixed-Oxide Lattice Measurements," EPRI-NP-691 (February 1978).

153. H. H. Windsor, W. J. Tunney and G. A. Price, "Exponential Experiments with Lattices of Uranium-233 Oxide and Thorium Oxide in Light and Heavy Water," Nucl. Sci. Eng., 42, p. 150 (1970).

154. G. A. Price, Uranium - Water Lattice Compilation Part I, BNL Exponential Assemblies, BNL-50035 (T-449) Reactor Technology - TID 45007, B. R. Sehgal, "Analysis of Brookhaven ^{233}U - ^{232}Th Oxide, Light and Heavy Water Lattice Experiments," Trans. Am. Nucl. Soc., 10, p. 596 (1967).

155. J. J. Ullo, J. Hardy, Jr. and N. M. Steen, "Review of Thorium-^{233}U Cycle Thermal Reactor Benchmark Studies," BNL/EPRI Symposium on Nuclear Data Problems for Thermal Reactor Applications (May 22-24, 1978).

156. D. J. Pellarin et al., "Reactivity and Parameter Measurements in a Coaxial Uranium Fuel-D_2O Moderated Critical Lattice," in Symposium Proceedings in Nuclear Data Problems for Thermal Reactor Applications, EPRI-NP-1098 (June 1979).

157. B. G. Chidley, T. J. Manual and P. R. Tunnicliffe, "Initial Conversion Ratios and Fast Fission Factors in Heavy-Water Natural-Uranium Lattices Using 19-Element UO_2 Fuel Rods," Nucl. Sci. Eng., 17, p. 47 (1963).

158. "Benchmark Testing of ENDF/B-V Data for Thermal Reactors," CSEWG Thermal Reactor Data Testing Subcommittee, ENDF-313 (July 1981).

159. Y. D. Harker, J. W. Rogers and D. A. Millsap, "Fission-Product and Reactor Dosimetry Studies at Coupled Fast Reactivity Measurements Facility," USDOE Report TREE-1259, Idaho National Engineering Laboratory (1978).

160. R. A. Anderl, F. Schmittroth and W. D. Harker, "Integral-Capture Measurement Cross-Section Adjustments for Nd, Sm, and Eu," EGG-THYS-5182 (July 1981).

161. J. W. M. Dekker, "Tables and Figures of Adjusted and Unadjusted Capture Cross Sections Based on the RCN-2 Evaluation and Integral Measurements in STEK," Vol. 1, ECN-14 (1977), Vol. 2, ECN-30 (1977) and Vol. 3, ECN-54 (1979).

162. J. W. M. Dekker and H. Ch. Rieffe, "Adjusted Capture Cross Sections of Fission-Product Nuclides from STEK Reactivity Worths and CFRMF Activation Data," Vol. 1, ECN-28 (1977) and Vol. 2, ECN-55 (1979).

163. R. A. Anderl and Y. D. Harker, "Measurement of the Integral Capture and Fission Cross Sections for ^{232}Th in the CFRMF," in Proc. International Conference on Nuclear Cross Sections for Technology, Knoxville, Tennessee, October 22-26, 1979, NBS Special Publication 594 (September 1980) 475.

164. D. M. Gilliam and J. W. Rogers, "Spectrum-Averaged Fission Cross Sections for ^{235}U, ^{232}Th, and ^{240}Pu in the Coupled Fast Reactivity Measurements Facility (CFRMF)," in Reports to the DOE Nuclear Data Committee, USDOE Report DOE-NDC-19, 124 (May 1979).

165. Y. D. Harker, R. A. Anderl, E. H. Turk and N. C. Schroeder, "Integral Measurements for Higher Actinides in CFRMF," in Proc. International Conference on Nuclear Cross Sections for Technology, Knoxville, Tennessee, October 22-26, 1979, NBS Special Publication 594 (September 1980) 548.

166. K. M. Glover et al., "Cm Production Cross-Section Measurements in ZEBRA," UKNDC (UK) P71/INDC (UK)-25/U, 51, (July 1975).

167. R. A. P. Wiltshire et al., "The Cross Section for the Production of ^{242}Cm from ^{241}Am in a Fast Reactor," AERE-R-7363, (1973).

168. K. M. Glover et al., "Measurement of the Integral Capture Cross Section of ^{243}Am," UKNDC (78) P88/NEANDC (E) 192U. Vol. 8/INDC (UK) 30/U, 96 (June 1978).

169. J. K. Dickens, T. R. England, T. A. Love, J. W. McConnell, J. F. Emery, K. J. Northcutt and R. W. Peelle, "Delayed Beta- and Gamma-Ray Production Due to Thermal-Neutron Fission of ^{235}U, Spectral Distributions for Time After Fission Between 2 and 14,000 sec: Tabular and Graphical Data," Oak Ridge National Laboratory report NUREG/CR-0162, ORNL/NUREG-39 (August 1978).

170. J. K. Dickens, T. R. England, T. A. Love, J. W. McConnell, J. F. Emery, K. J. Northcutt and R. W. Peelle, "Delayed Beta- and Gamma-Ray Production Due to Thermal-Neutron Fission of ^{239}Pu: Tabular and Graphical Spectral Distributions for Times After Fission Between 2 and 14,000 sec.," Oak Ridge National Laboratory report NUREG/CR-1172, ORNL/NUREG-66 (January 1980).

171. J. L. Yarnell and P. J. Bendt, "Decay Heat from Products of ^{235}U Thermal Fission by Fast-Response Boil-Off Calorimetry," Los Alamos Scientific Laboratory report LA-NUREG-6713 (September 1977).

172. J. L Yarnell and P. J. Bendt, "Calorimetric Fission Product Decay Heat Measurements for ^{239}Pu, ^{233}U, and ^{235}U," Los Alamos Scientific Laboratory report NUREG/CR-0349, LA-7542-MS (September 1978).

173. E. T. Jurney, P. J. Bendt and T. R. England, "Fission Product Gamma Spectra," Los Alamos Scientific Laboratory report LA-7620-MS (January 1979).

174. "American National Standard for Decay Heat Power in Light Water Reactors," prepared by the American Nuclear Society Standards Committee working Group ANS-5.1, ANSI/ANS-5.1 (1979).

175. L. N. Usachev, "Perturbation Theory for the Breeding Ratio," J. Nucl. Energy A.B., 18, p. 571 (1964).

176. A. Gandini, "A Generalized Perturbation Method for Bi-Linear Functionals of the Real and Adjoint Neutron Fluxes," J. Nucl. Energy, 21, p. 755 (1967).

177. W. M. Stacey, Jr., "Variational Methods in Nuclear Reactor Physics," Academic Press, New York (1974).

178. E. Greenspan, "Developments in Perturbation Theory," in Advances in Nuclear Science and Technology, Vol. 9, Academic Press, New York, (1976).

179. C. R. Weisbin, R. W. Peelle, J. H. Marable, P. Collins, E. Kujawski, E. Greenspan and G. de Saussure, "Sensitivity and Uncertainty Analysis of Reactor Performance Parameters," Advances in Nuclear Science and Technology, Plenum Press, Vol. 14, (1982).

180. D. G. Cacuci, C. F. Weber, E. M. Oblow and J. H. Marable, "Sensitivity Theory for General Systems of Nonlinear Equations," Nucl. Sci. Eng., 75, 88 (1980).

181. D. G. Cacuci, "Sensitivity Theory for Nonlinear Systems, Part I: Nonlinear Functional Analysis Approach, p. 2794; Part II: Extensions to Additional Classes of Responses," p. 2803, J. Math. Phys., 22 (1981).

182. J. R. White, "The Depth-Charge Static and Time-Dependent Perturbation/Sensitivity System for Nuclear Reactor Core Analysis," ORNL/CSD-78 (September 1981).

183. E. M. Oblow, "Sensitivity Theory for Reactor Thermal-Hydraulic Problems," Nucl. Sci. Eng., 68, 332 (1978).

184. E. Greenspan, M. L. Williams, and J. H. Marable, "Time-Dependent Generalized Perturbation Theory for Coupled Neutron Nuclide Problems," Nucl. Sci. Eng., 73, 210 (1980).

185. C. R. Weisbin, J. H. Marable, J. Hardy, Jr. and R. D. McKnight, "Sensitivity Coefficient Compilation for CSEWG Data Testing Benchmarks," ENDF-265 (August 1978).

186. C. R. Head, "Nuclear Data Requirements of the Magnetic Fusion Power Program of the United States of America," Proc. of the Advisory Group Meeting on Nuclear Data for Fusion Reactor Technology organized by IAEA, Vienna, p. 1, IAEA-TECDOC-223 (1979).

187. M. A. Abdou, "Nuclear Data Requirements for Fusion Reactor Shielding," Ibid p. 91.

188. D. Steiner, "Nuclear Data Needs for Fusion Reactor Design," NBS Special Publication 425 (1975).

189. R. T. Santoro and J. Barish, "Cross-Section Sensitivity of the D-T Fusion Probability and the D-T and T-T Reaction Rates," Nucl. Sci. Eng., 59, 189-194 (1976).

190. R. G. Alsmiller et al., "Comparison of the Cross Section Sensitivity of the Tritium Breeding Ratio in Various Fusion Reactor Blankets," Nucl. Sci. Eng., 57, 2, 122 (1975).

191. A. P. Fraas. "Conceptual Design of the Blanket and Shield Region and Related Systems for a Full Scale Toroidal Fusion Reactor," ORNL/TM-3096 (1973).

192. D. J. Dudziak, "Discrete Ordinates Neutronic Analysis of a Reference Theta Pinch Reactor," (RTPR) LA-DC-72-1427 (1972).

193. S. A. W. Gerstl, D. J. Dudziak and D. W. Muir, "The Application of Sensitivity Analysis to Nuclear Data Assessment," Proc. Specialist's Mtg. on Sensitivity Studies and Shielding Benchmarks, Paris, France, 7-10 (Oct. 1975).

194. R. G. Alsmiller, J. Barish and C. R. Weisbin, "Uncertainties in Calculated Heating and Radiation Damage in the Toroidal Field Coil of a Tokamak Experimental Power Reactor Due to Neutron Cross Section Errors," Nuclear Technology, Vol. 34 (August 1977).

195. R. E. Maerker, F. J. Muckenthaler and C. E. Clifford, "SB4 Measurements and Calculations of the ORNL CRBR Upper Axial Shield Experiment," ORNL-5259 ENDF-258 (June 1977).

196. E. M. Oblow and C. R. Weisbin, "Fast Reactor Shield Sensitivity Studies for Steel-Sodium-Iron Systems," Paper E.7, Proceedings of the Fifth International Conference on Reactor Shielding, Knoxville, Tennessee (April 18-22, 1977).

197. R. E. Schenter, "Data Testing of Fission Products and Actinides," see C. R. Weisbin, R. D. McKnight, J. D. Hardy, Jr., R. W. Roussin, R. E. Schenter and B. Magurno, "Benchmark Testing of ENDF/B-V," (to be published in 1982).

198. G. E. Hansen and H. C. Paxton, "A Critical Assembly of Uranium Enriched to 10% in Uranium-235," Nucl. Sci. Eng., 72, pp. 230-236 (1979).

199. M. R. Bhat, "Evaluation of ^{235}U Neutron Cross Section and Gamma Ray Production Data for ENDF/B-V," BNL-NCS-51184 (March 1980).

200. C. M. Eisenhauer, J. A. Grundl and A. Fabry, "Proceedings of a Symposium on Neutron Standards and Applications," NBS-SP-493.

201. R. A. Anderl et al., EGG-Phys. 5608, October 1981 and EGG-Phys. 5668 (January 1982).

202. J. J. Wagschal and B. L. Broadhead, Trans. Am. Nuc. Soc. 39, 887 (1981).

203. D. G. Madland, "Average Cross Sections in the Neutron Benchmark Field of ^{252}Cf: A Status Report," private communication to B. Magurno (April 1981).

204. D. G. Madland and J. R. Nix, LA-UR-81 2968, accepted for publication in Nucl. Sci. Eng. (October 1981).

205. Private communication, C. M. Eisenhauer to B. Magurno (October 1981).

206. J. Grundl, C. Eisenhauer and E. D. McGarry, "Compendium of Benchmark Neutron Fields for Reactor Dosimetry," NUREG/CR-0552, HEDL-TME 78-8 R5 (Sept. 1978). Revised evaluation by private communication (March 1982).

207. W. N. McElroy, Letter Report to B. A. Magurno, July 1977, with attachments:

    a) "Further Phase II Data Testing of ENDF/B-IV Dosimetry Cross Section File and Comments Relative to the Elaboration of Version V," A. Fabry, HEDL memo for Dosimetry Task Force Meeting at NBS, March 1977.

    b) "Review of Microscopic Integral Cross Section Data in Fundamental Reactor Dosimetry Benchmark Fields," Fabry et al., HEDL-SA-1148, November 1976, and Neutron Cross Sections for Reactor Dosimetry, IAEA-208, 233, Vienna, November 1976.

    c) "Spectral Characterization by Combining Neutron Spectroscopy, Analytical Calculations and Integral Measurements," McElroy et al., HEDL-SA-1149, November 1976, and Neutron Cross Sections for Reactor Dosimetry, IAEA-208, 147, Vienna, November 1976.

208. Csikai et al., "Proceedings of International Symposium of Cf-252 Utilization," Paris (April 1976), as quoted in Ref. 15.

209. W. Mannhart, PTB-FMRB-72 (April 1978).

210. L. Green, "Absorption Cross-Section Measurements for Californium-252 Spontaneous Fission Neutrons," Nucl. Sci. Eng., 58, pp. 361-370 (1975).

211. D. L. Smith and J. W. Meadows, "Response of Several Threshold Reactions in Reference Fission Neutron Fields," ANL/NDM-13 (1975).

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