CONSISTENCY AND COMPLEMENTARITY
OF MICROSCOPIC AND INTEGRAL DATA

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As a consequence of improvements both in the microscopic technics and in the formalisms of nuclear reactions, the recent evaluated data appear to be more and more consistent with integral data.

That is a phenomenon rather recent, of importance since it will give rise in the future to data libraries closer to the "physical reality" and with wider utilization areas consequently.

It was necessary to give some highlight to this evolution, and that explains the topic of the conference.
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Cross Section Adjustments using Integral Data

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The written version of this paper is almost the same as the one, presented at the Workshop on Evaluation Methods and Procedures, held at Brookhaven National Laboratory, New York, September 22-25, 1980.
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CROSS SECTION ADJUSTMENTS USING INTEGRAL DATA

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ABSTRACT

Adjustment methods currently used in France, the Netherlands and the U.S.A. to adjust neutron capture cross sections in the fission-product mass range are reviewed. The methods include least-squares fitting of multi-group constants, multi-group cross section adjustment, model-parameter adjustment and direct point cross-section adjustment. Additional comments are given on logarithmic adjustment and on other recent approaches, which stress the inclusion of "method" uncertainties and the treatment of systematic ("negligence") errors. The evaluation of experimental data, a-priori cross sections and their covariance matrices is shortly discussed. Finally some conclusions and recommendations are summarized.

1. PREFACE

In this paper a review\footnote{This review is only slightly different from the one given at the Workshop on Evaluation Methods and Procedures, held at Brookhaven, September 22-25, 1980.} is given on the use of adjustment methods in neutron cross section evaluation. This review is mainly based upon the experience of the authors in the field of fission-product cross section adjustment and refer mostly to work performed at ECN [1-7], CEA-Cadarache [8,9] and HEDL [10-12]. At these laboratories adjusted fission-product data files have been obtained. Results of a first intercomparison between adjusted multi-group capture cross sections (RCN-2A and CARNIVAL-IV) were recently published [9]. The HEDL adjustment code [11] was used to obtain ENDF/B-V fission-product cross sections [12,31] which are partly based upon integral data obtained at ECN and Idaho. The integral data measured at EBR-II on fission products have also been analysed with the above-mentioned code [13]; the results will be reviewed by Anderl at this meeting [14]. Finally, extensive testing of JENDL-1 evaluated cross sections against integral data was reported by Iijima et al. [15]. These results will be incorporated in a future Japanese evaluation (JENDL-2).
2. GENERAL LINEAR LEAST-SQUARES ADJUSTMENT TECHNIQUE

A quite general linear least-squares adjustment formalism can be derived from Bayes' theorem assuming Gaussian distributions for the measured and adjustable quantities [1]. Combining the measured quantities in a vector $\mathbf{R}^\text{exp}$ with covariance matrix $\mathbf{Y}$ and the a-priori known parameters in a vector $\mathbf{P}^0$ with covariance matrix $\mathbf{Q}$, the "best" estimate of the parameters is found as the vector $\mathbf{P}'$ that minimizes

$$q^2(\mathbf{P}) = (\mathbf{R} - \mathbf{R}^\text{exp})^T \mathbf{Y}^{-1} (\mathbf{R} - \mathbf{R}^\text{exp}) + (\mathbf{P} - \mathbf{P}^0)^T \mathbf{Q}^{-1} (\mathbf{P} - \mathbf{P}^0),$$

where

$$\mathbf{R} - \mathbf{R}^0 = \mathbf{G}(\mathbf{P} - \mathbf{P}^0).$$

In these equations $\mathbf{R}^0$ stands for quantities calculated from $\mathbf{P}^0$ and the "sensitivity" matrix $\mathbf{G}$ relates the variations $\mathbf{R} - \mathbf{R}^0$ and $\mathbf{P} - \mathbf{P}^0$. It is further assumed that $\mathbf{R}^\text{exp}, \mathbf{P}^0$ are statistically independent and that there are no uncertainties associated with $\mathbf{G}$ (see Sect. 3).

Eqs. (1) and (2) can be considered as a standard least-squares fitting problem (rather than an "adjustment" problem) when the a-priori vector $\mathbf{P}^0$ is supposed to result from direct measurements. Then the "number of degrees of freedom" $n$ equals the number of experimental data given in $\mathbf{R}^\text{exp}$.

The solution of the above-mentioned minimization problem can be denoted as follows [1]:

$$\mathbf{P}' - \mathbf{P}^0 = \mathbf{A} \mathbf{X},$$

$$\mathbf{Q}' - \mathbf{Q} = -\mathbf{A} \mathbf{Y}^{-1} \mathbf{A}^T,$$

with

$$\mathbf{A} = \mathbf{Q} \mathbf{G},$$

$$\mathbf{X} = \mathbf{Y}^{-1} (\mathbf{R}^\text{exp} - \mathbf{R}^0),$$

where $\mathbf{Y}$ is the covariance matrix of the difference of experimental data ($\mathbf{Y}$) and a-priori data ($\mathbf{Y}$):

$$\mathbf{Y} = \mathbf{V} + \mathbf{Y} = \mathbf{V} + \mathbf{G} \mathbf{C} \mathbf{G}^T.$$ 

A "goodness of fit" parameter is obtained from a $\chi^2$-test, which leads to the inner product:

---

\* Superscript $T$ indicates the transpose of a matrix.

\* Also called "model matrix", "design matrix" or "method matrix".
\[ x^2 = X^T (\mathbf{R}^{\text{exp}} - \mathbf{R}^0) \]  

(8)

When \( x^2 \) exceeds \( n \) it is advisable to increase the uncertainties in the adjusted data by multiplying the adjusted covariance matrix with \( x^2/n \); see also Sect. 7.

If the second term in Eq. (1) is dropped no a-priori information is used and the system is only overdetermined when \( n \) exceeds the number of parameters \( p \) to be fitted to the experimental data. This fitting method was followed by the French [8], with the additional constraint that \( p - p^0 \) should not exceed twice the standard deviation of \( p^0 \); otherwise the excessive parameters are fixed at the error limits. This means that the statistical distribution of \( p \) is assumed to be rectangular. Moreover, correlations between the parameters \( p^0 \) are used in some way [8]. In this application the parameters \( p^0 \) are group constants (4-5 groups) which have been collapsed to obtain the condition \( n > p \). After adjustment the a-priori multi-group constants are used to translate the adjusted collapsed data into "adjusted" 25-group constants.

In solving Eqs. (1) and (2) one should take advantage of the statistical independence of various components of \( \mathbf{R}^{\text{exp}} \) and \( p^0 \) ("partitioning" [10]). When there are several independent subsets of \( \mathbf{R}^{\text{exp}} \) it is easily demonstrated that one could reduce the adjustment problem to one subset first before adding new subsets sequentially. On the other hand, when \( p^0 \) can be partitioned it is possible to reduce the problem to a number of separate adjustment problems. This could be useful in particular when one is only interested in a "partial adjustment" [1] of \( p \) (see Sect. 3). An interesting example of partitioning both \( \mathbf{R}^{\text{exp}} \) and \( p^0 \) is given by Schmitzroth [10] in an application of neutron spectrum unfolding using integral dosimetry results ("elimination of subsidiary parameters").

Another way to reduce the size of the adjustment problem is to redefine the parameters to a vector of smaller size. Suppose that the old vector \( \mathbf{K} \) can be calculated from the new vector \( \mathbf{P} \). In that case the adjusted vector \( \mathbf{P}' \) can be used to obtain \( \mathbf{K}' \). When only the reverse relation \( \mathbf{P} = \mathcal{S} \mathbf{K} \) is known, the calculation of \( \mathbf{K}' \) is more involved. This situation occurs when \( \mathcal{S} \) defines a multi-group collapsing scheme. The formal solution of this problem is given in Sect. 4, Eqs. (16-18). Approximative methods for this unfolding process could also be adopted. However, in many instances the collapsed group constants \( p^0' \) are useful for the required application. Schmitzroth [10] notes that when \( p^0 \) and \( p^0' \) are a-priori known to be statistically independent sequential evaluation can be applied, \( \mathbf{K}' \) playing the role of integral data.
3. METHOD UNCERTAINTY

Multi-Group Cross Section Adjustment

Eqs. (1) and (2) are quite general. In a simple application, \( \bar{\mu}_{\exp} \) and \( \bar{\mu}_0 \) represent measured reaction rates and multi-group cross sections \( \bar{\mu}_j \) respectively, \( \mu \) being a sensitivity matrix containing well-known group fluxes corresponding to the neutron fields in which the reaction rates were measured. Assuming that \( \mu \) has no "method" uncertainties, the solution of the adjustment problem is given by Eqs. (3-7).

Multi-Group Neutron Flux adjustment

Another possibility is to assume that \( \mu \) stands for multi-group fluxes \( \bar{\mu} \) while the sensitivity matrix \( \mu \) is filled with well-known group cross sections. In this case the aim is to obtain adjusted flux spectra. Again no uncertainties are assumed in \( \mu \).

Adjustment of Cross Sections and Fluxes

Since both \( \bar{\mu} \) and \( \bar{\mu} \) usually contain uncertainties, a straightforward approach is to store them in one parameter vector

\[
\mu = \begin{pmatrix} \mu_1 \\ \mu_2 \end{pmatrix} = \begin{pmatrix} \bar{\mu} \\ \bar{\mu} \end{pmatrix}.
\]

(9)

This approach was followed by Perey [16] in his dosimetry unfolding code STAR'SL. This code forces the user to introduce covariance matrices for cross sections and flux spectra (as well as possible correlations). It is assumed that the matrix \( \mu \) can be written as

\[
\mu = \begin{pmatrix} \mu_1 & \mu_2 \\ \mu_2 & \mu_2 \end{pmatrix}
\]

(10)

with \( \mu_1 \) and \( \mu_2 \) expressed in terms of \( \bar{\mu}_j \) and \( \bar{\mu}_j \), respectively, i.e. without uncertainties. From straightforward application of Eqs. (3-7) adjusted parameters \( \bar{\mu}' \) and \( \bar{\mu}' \) and their covariances are obtained. However, it is not needed to follow this approach when \( \mu_1 \) and \( \mu_2 \) are statistically independent, i.e.

\[
\mu = \begin{pmatrix} \mu_1 & 0 \\ 0 & \mu_2 \end{pmatrix}
\]

(11)

and when one is only interested in adjustment of \( \mu_1 \). In this case the scheme of Eqs. (3-6) can be followed for the quantities labeled with index 1, replacing Eq. (7) by

\[
\bar{\mu} = \bar{\mu} + \bar{\mu} + \bar{\mu},
\]

(12)

where \( \bar{\mu} \) is the contribution of \( \mu_2 \) to the uncertainty in the cal-
culated integral data: \( Y = G \alpha_2 G^T \). This was called "partial adjustment" in Ref. [1], since the adjustments in \( \alpha_2 \) remain implicit. In recent literature \( Y \) is called "method uncertainty" [21,23].

**Definition of Method Error**

In the above-mentioned example the matrix \( Y \) contains the "method error". A quite general definition of method error is obtained when we assume that \( Y \) accounts for any error in the calculated integral data which is not already contained in \( Y \). In this definition it is not required that the uncertainties in \( G \) are explicitly known [16,23], in which case one could always add the elements of \( G \) to the parameter vector \( \alpha \) and reformulate the adjustment problem [16]. For instance, when there is an uncertainty in the calculational method used to generate integral data (e.g. because of a multi-group approximation) this uncertainty cannot easily be connected to elements of \( G \). Another example provides the adjustment of model parameters, where "inherent" statistical model errors cannot be attached to parameters, although they constitute an essential uncertainty in the calculated integral data (see Sect. 4). In the last two examples one could assume that Eq. (2) is replaced by

\[
(R - R^0) = G(P - P^0) + \Delta - \Delta^0 \tag{13}
\]

where \( \Delta \) is a "noise" vector \( (\Delta^0 = 0) \) with covariance matrix \( \Upsilon \).

When Eq. (13) is denoted as

\[
(R - R^0) = (G \Delta^0) \begin{pmatrix} \alpha - \alpha^0 \end{pmatrix} \tag{14}
\]

it is easily seen that we have reduced our problem to Eqs. (1,2) by adding a noise vector \( \Delta \) to the adjustable parameters. Thus, it is always possible to reduce the adjustment problem to Eqs. (1,2).

The parameter vector \( \Delta \) was introduced in this section as a "noise" vector. In Refs. [23,24] relation (13) is also adopted, where the vector \( -\Delta \) is interpreted as a "bias" of which the a-priori value \( -\Delta^0 \) is known from previous comparisons between integral data and cross sections.

**Correlations Between A-Priori Data and Method**

In the previous subsection we did not assume correlations between a-priori data and method, but these could easily be included when the elements of \( G \) are added to \( \alpha \). A quite general formulation of the problem was given by Marable and Weisbin [23], which in our notation can be written as
\[ q^2 = \begin{pmatrix} 1 - \frac{P_0}{P} \\ \frac{C_0}{\gamma - \Delta^0} \\
\frac{\gamma - \Delta^0}{1 - \Delta^0} \end{pmatrix} T \begin{pmatrix} 1 - \frac{P_0}{P} \\ \frac{C_0}{\gamma - \Delta^0} \\
\frac{\gamma - \Delta^0}{1 - \Delta^0} \end{pmatrix} \]

(15)

with the constraint of Eq. (13). In Eq. (15) a tilde means that matrix elements are written as a vector. The covariance matrix \( \xi \) is supposed to contain no correlations between \( P^{\exp} \) and the a-priori data. It is noted that Marable and Weisbin [21,32] follow an alternative geometric approach to solve the adjustment problem.

In this subsection we give an example of a correlated a-priori vector (neutron spectra) and method matrix (cross sections). This was the case in the adjustment of STEK flux and adjoint fluxes [17] of which the a-priori values were obtained from a core calculation using multi-group cross sections of the reactor materials (e.g. 235U). The integral data used for the adjustment included reactivity worths of B, 235U and various fission rate ratios. Obviously there are strong correlations between the neutron spectra and 235U cross sections \( \Sigma_u \), indicating that Perez's scheme [16] should be followed, i.e. a combination of spectra and cross sections in one parameter vector. However, since the spectra are a function of heavy metal cross sections (mainly 235U) it is possible to express \( \psi \) in terms of \( \Sigma_u \), such that a new parameter vector can be defined with only group cross sections which are statistically independent for each sample. Thus, partial adjustment can be applied on this new vector, leading to an adjusted vector \( \Sigma'_u \), the adjustments in the other dosimetry cross sections remaining implicit. From the vector \( \Sigma'_u \), "adjusted" STEK neutron spectra were calculated [17]. This example shows that a careful selection of a set of independent parameters and application of partial adjustment can be very advantageous in adjustment calculations.

**Unwanted Implicit Adjustments**

Once the neutron spectrum has been determined with corresponding covariance matrices it can be used as a "reference" spectrum for the analysis of a large class of integral cross section measurements. For instance, the STEK spectra [17] were used in an extensive series of fission-product cross section adjustments [3,4]. From the previous discussion it is evident that in each cross section adjustment calculation the neutron spectra will be re-adjusted. Although normally this adjustment is small it is an unwanted implicit adjustment, since one usually prefers to use very accurate cross section data for this purpose. Therefore, in the case of analysis of STEK data the vector \( \psi \) was constrained to the reference value, although the corresponding method uncertainty was included in the calculation of the covariance matrix of \( \Sigma \).

Implicit adjustments of cross sections are also unwanted when the "method" is a neutronics code to be used for a large number of calculations without changes. In that case possible shortcomings of the method are included in the cross sections. Again, uncertainties due to the method should be included in the adjusted cross sections.
4. MODEL PARAMETER ADJUSTMENT

Eqs. (1) and (2) are not restricted to the adjustment of multi-
group constants. Gandini and Salvatores [18] and Dragt et al. [1]
have suggested to adjust the model parameters of the cross sections,
from which adjusted point cross sections could be calculated ("con-
sistent method" [18]). In this application the vector \( \mathbf{P}^0 \) contains
these model parameters and \( \mathbf{Q} \) is their co-variance matrix. It is
advisable to adopt a set of statistically independent parameters
(or independent subsets of parameters) as far as possible.

The above-mentioned method is particularly useful when the
evaluation is entirely based upon nuclear-model calculations and
when the uncertainties in the parameters can be easily derived.
This could be the case in capture cross section calculations [1,2,
6] utilizing a statistical model, of which the main model para-
eters are deduced from "external" sources (see also Sect. 8). In
practice, these parameters are often "tuned" to fit differential
measurements and it becomes more difficult to estimate the uncer-
tainties and correlations of the "tuned" parameters. Apart from
this difficulty model parameter adjustment is attractive to the
evaluator, because before recalculating the cross sections with
adjusted parameters he may interfere, avoiding unphysical ad-
justments. Another possibility is to improve the systematics of
important parameters such as the mean level spacing, average
capture width [6] or the \( \gamma \)-ray strength function.

In model parameter adjustment a notable "method error"
(Sect. 3) is encountered. This error arises from inherent statisti-
cal-model uncertainties caused by fluctuations in the neutron
widths or in the number of levels per energy interval [2]. These
uncertainties allow for so-called "non-statistical effects",
which often reflect the uncertainty in the statistical-model
estimate. A disadvantage of parameter adjustment with respect to
multigroup cross section adjustment is that these method errors
may lead to implicit adjustments, which are not noted by the
evaluator. A mixed approach is possible, however (see below).

Another drawback of parameter adjustment is that it is not
easy to apply it in the resolved resonance range, where the num-ber of parameters can be quite large. In our application of par-

ter adjustment [6] we have assumed a number of important para-

eters only. By performing a normal multi-group cross section
adjustment calculation the adjusted model parameters \( \mathbf{K}' \) and their
covariance matrix \( \mathbf{Q}' \) are obtained \emph{a-poster]or} from the relations

\[
\mathbf{K}' = \mathbf{K}^0 + \mathbf{B} \mathbf{X} \\
\mathbf{Q}' = \mathbf{Q}^0 + \mathbf{B} \mathbf{W}^{-1} \mathbf{B}^T
\]

with

\[
\mathbf{B} = \mathbf{L} \mathbf{G}^T.
\]

(16)

(17)

(18)
In Eq. (18) $\xi$ is the sensitivity matrix for parameter variations. The adjustment of the parameters corresponds with an amount $\Delta' = \xi' - \xi''$, expressed in multi-group constants. The residual adjustment $\Delta'' = \xi'' - \xi''_0$ accounts mainly for adjustments in the resolved resonance range. Also adjustments in the range where there are large inherent statistical model errors are included in $\Delta''$. The parameters $\xi''$ are used to re-evaluate point cross sections, the remaining problem being the inclusion of residual adjustments in the evaluation.

In the forthcoming RCN-3 evaluation [25], which has been completed now for 30 materials, the above-mentioned procedure was followed starting from the unadjusted RCN-2 evaluation [26]. The adjustments were made to fit integral STEK and CFRMF data, reviewed in Ref. [5]. In addition some revisions were applied (e.g. for Mo [7]), based upon recent differential data. For most materials the multi-group cross section adjustments $\xi''$ in the resolved resonance range were small, so that no revisions were needed in the corresponding point cross sections which are stored in KEDAK format. In the case of $^{133}$Cs a correction was applied in the highest part of the resolved resonance range by multiplying the capture cross section with an exponentially increasing smooth correction factor, which was obtained from a "rough fit" through the relative group cross section adjustments, see Fig. 1. In general the evaluator should be conservative in applying corrections in this energy range, but in the case of $^{133}$Cs there are other indications for missed strength in the highest part of the resolved resonance range, e.g. the argument that the statistical model predicts higher average capture cross sections than those calculated from resolved resonance parameters [27], see also Fig. 2. In Figs. 3 and 4 portions of the unadjusted and adjusted group cross sections $\xi''$ and corresponding point sections are shown, together with available experimental data measured at laboratories indicated in the legend (see CINDA literature index [28]). The adopted RCN-3 curve [25] is slightly different from the adjusted one. This was due to the fact that a revision was made in the calculation of the total cross section. This also affected the capture cross section since the same optical model was used to obtain the neutron transmission coefficients in the capture cross section calculation.

Instead of adopting a physical model parametrization one could also use a mathematical parametrization. This approach was followed at HEDL, using the following "finite-element" representation [11,30]:

$$
\sigma(E) = \sum_i h_i(E) \sigma_i, \tag{19}
$$

where $h_i(E)$ is a triangle or "roof" function with triangle coordinates $(\log E_{i-1}, 0), (\log E_i, 1)$ and $(\log E_{i+1}, 0)$. In this representation additional and points have to be defined. Since the coefficients are just the point cross sections $\sigma_i$ at neutron energies $E_i$ their treatment is discussed in the next section.
5. DIRECT POINT CROSS SECTION ADJUSTMENT

Instead of multi-group cross sections the vector $\mathbf{P}$ in Eqs. (1, 2) could also contain point cross sections $\sigma_i$ of which the elements $\sigma_i$, energies $E_i$ and an appropriate interpolation scheme [e.g. Eq. (19)] define the cross section at each energy. The main advantage of this approach is that differential experimental data could easily be included in $R$.

The sensitivity matrix relating differential cross sections and $\sigma$ is given by Eq. (19); for the relation with integral data a similar expression is used [11,30]:

$$ R = \sum_i H_i \sigma_i $$  \hspace{2cm} (20)

with

$$ H_i = \int h_i(E) \phi(E) dE. $$ \hspace{2cm} (21)

Adjustment calculations of cross sections in an extended resolved resonance region become very tedious in this scheme. Therefore, average cross sections are introduced [12] in this region, e.g. by introducing multi-group cross sections in $\Sigma$. In the thermal range the cross section is usually smooth and Eq. (19) can be used. The histogram part of the cross section could be parametrized by means of block functions or, approximatively, with triangle functions.

In the last-mentioned case adjustments do not conserve the block shape of the histograms, as was the case in the application of adjustment of ENDF/B fission-product cross sections [12].

The above-mentioned scheme is very convenient to obtain adjusted point cross sections fitted to both integral data and differential data. As in the previous section, adjustments in the resolved resonance range are difficult to include in practice.

A drawback of the method is that still the number of parameters can be quite large. Moreover, the evaluator does not gain insight in the adjustments of underlying physical model parameters.

Another approach to direct point cross section adjustment has been followed by Pazy et al. [19] who have formulated the adjustment problem in terms of continuous functions. In their discrption the a-priori cross section $\sigma^0(x)$ is a continuous function of energy and the integral data $r$ are functionals of $\sigma(x)$ and may also be a continuous function of another parameter $E'$. Their minimization problem can be denoted as:

$$ q^2(\sigma) = \int \left[ \frac{\sigma(E) - \sigma^0(E)}{\Delta \sigma^0(E)} \right]^2 w_{\sigma}(E) dE + \int \left[ \frac{r(E', \sigma(E)) - r^0(E')}{\Delta r^0(E')} \right]^2 w_r(E') dE', $$(22)

where $\Delta \sigma^0$ and $\Delta r^0$ are standard deviations and $w_{\sigma}$ and $w_r$ are density functions which reflect the number of measurements per energy...
interval on which the a-priori quantities are based. In this picture correlations are not included, although in applications w could be interpreted as a constant over a wide energy interval. This incomplete uncertainty treatment limits the applicability of the above-mentioned method.

6. LOGARITHMIC ADJUSTMENT

A possible drawback of the methods discussed before is due to the assumed Gaussian distribution of p, which may lead to unrealistic adjustments, such as negative values for cross sections. This could be avoided by considering lognormal distributions \[1,10,11\] or constraining the vector \(\mathbb{P}\) within prescribed error limits \([5]\), which could be asymmetric.

It is most appropriate to assume the logarithms of the cross-sections \(\mathbb{P}\) to be normally distributed. In the minimization of Eq. (1) the second term is then to be replaced by

\[
(Z - Z^0)^T Q^{-1} (Z - Z^0)
\]

(23)

with

\[
Z = \ln \mathbb{P},
\]

\[
Z^0 = \ln \mathbb{P}^0
\]

(24)

The first term is not changed: integral data are still assumed to be normally distributed. Several approximations with regard to the dependence of \(\mathbb{R}\) on \(Z\) (or \(\mathbb{P}\)) are possible, e.g.: (a) Assume \(\mathbb{R}\) to be linear in \(Z\) (i.e. variations in \(\mathbb{R}\) proportional to relative variations in \(\mathbb{P}\)). Then the whole adjustment procedure remains the same as before, with \(\mathbb{P}\) replaced by \(Z\) in the equations.

The a-posteriori distribution for \(Z\) is normal again, i.e. the adjusted cross sections follow a logarithmic normal distribution. This idea was followed in Ref. [11].

(b) Assume - as before - that \(\mathbb{R}\) is linear in \(\mathbb{P}\) (or can be linearized). Then the a-posteriori distribution for \(\mathbb{P}'\) is no longer logarithmic normal. The most probable value for \(\mathbb{P}'\) cannot be found from a closed expression, but must be computed by iteration. This approach has been worked out in much detail by Schmiettbroch [10].

It seems to be reasonable to assume a lognormal distribution for most a-priori cross sections, particularly in case of large uncertainties. The distribution is especially appropriate if the a-priori data originate from a measurement that contains certain relative errors and short-range correlations, together with a relative normalization error from an independent normalization measurement (see Sect. 8). The product of the two lognormal quantities is again lognormal.

The best choice for the dependency of \(\mathbb{R}\) on \(Z\) or \(\mathbb{P}\) depends on the
character of the integral data. In case of reaction rates the linear dependency on $P$ (case b) seems to be most realistic: variations in reaction rates are proportional to absolute cross section variations rather than relative ones. So case b seems to be preferable in this application, in spite of the mathematical and statistical complexity.

It is very important to note that in these approaches the most probable values are used as estimates for the cross sections (both a-priori and a-posteriori), and not the mean (or expectation) values. For lognormal distributions (i.e. the prior distribution in both cases and the posterior distribution in case a) the following relation exists:

$$<P_k> = P_k \exp \left( Q_z \right)_{kk} \text{,} \tag{25}$$

where $P_k$ is the $k^{th}$ element of the most probable vector $P$ used in Eq. (24). Such a simple relation does not exist for the posterior distribution in case b.

Likewise some care is needed to translate an error matrix of $P$ into the corresponding error matrix of $Z$ or vice versa. In case of the lognormal distribution this relation is $[10]$

$$\left( Q_z \right)_{kl} = <P_k><P_l> \left( \exp Q_{kl} - 1 \right). \tag{26}$$

The relation can be used to obtain the error matrix of the prior data in the minimization expression (23), and the reverse relation produces adjusted cross section errors in case a. The formula does not hold for adjusted data in case b. It has been shown in Ref. $[10]$ that the linear approximation

$$\left( Q_z \right)_{kl} = <P_k><P_l> Q_{kl} \tag{27}$$

is always reasonable in practice. This simple linear relation was used throughout in Ref. $[1]$. 

7. SYSTEMATIC ERRORS

It is basically true that Eqs. (3-7) should only be applied when there is statistical consistency between integral data and a-priori information. In the formulation of Sect. 2 this means that $\chi^2/n$ should be close to 1. The confidence interval of this quantity follows from the $\chi^2$-distribution for $n$ degrees of freedom. When $\chi^2/n$ exceeds unity it is "common practice" to multiply the initial or final covariance matrices with $\chi^2/n$ to "force" consistency.

However, it should be stressed that before doing so the evaluator should try to find the origin of the discrepancy. In some cases one could a-priori assume that (most of) the discrepancy is due to systematic errors in either the integral data or the a-priori cross sections. The French $[8]$ implicitly assume that systematic errors in the integral data are small by adopting a rectangular uncertainty distribution of the a-priori data with wide error limits. In this way the a-priori information is almost not used. When, for instance,
the shape of the a-priori cross sections is well-known from differential data and the main uncertainty is in its normalization, it might be better to estimate this normalization from a comparison between integral and a-priori data. (see Sect. 8, Eq. (37) for the usual approach).

Recently, Chao [20, 21] has investigated the situation of significant discrepancies between integral data and a-priori information in much detail. He introduced the concept of "negligence" to deal with neglected systematic errors. A simple estimate of this negligence $E$ and its covariance matrix $F$ can be expressed in terms of integral data by [21, 22]

$$E^O = \Delta^\exp - R^O = \Delta^\exp - \Delta^O,$$

$$F^O = N + N.$$  

(28)

(29)

The transformation of the negligence in terms of a-priori parameters $F^O$ is much more involved [21, 22]. The "best" estimates of $E$ and $F$ are obtained from

$$E = \frac{1}{1+\lambda} E^O,$$

$$F = \frac{1}{1+\lambda} F^O,$$

(30)

(31)

where $\lambda$ is related to $\chi^2/n$ by

$$\frac{\chi^2}{n} = \frac{\lambda+1}{\lambda},$$

(32)

assuming that $\chi^2/n$ exceeds 1. These quantities could be used to correct either the integral data or the a-priori data (Chao calls this "adjustment", departing from the usual definition) before combining them in the adjustment process like discussed in Sect. 2.

Also when it is not a-priori known which quantity should be corrected, Chao [22] gives estimates for the corrections. These estimates are as follows:

negligence in integral data:

$$E^\Delta = \frac{1}{1+\lambda} (N + N)^{-1} E^O,$$

(33)

negligence in a-priori data (in terms of integral data):

$$E^\beta = -\frac{1}{1+\lambda} (N + N)^{-1} F^O.$$

(34)

Thus, the total negligence is still given by the previous Eq. (30).

Let us to note that Chao needed an assumption in order to arrive at a division between the two negligences Eqs. (33, 34). His "model" was to weight the negligence components in the probability distributions by the error matrices of initial experimental and a-priori integral data, $\Sigma$ and $\Sigma$, respectively. This leads immediately to the proportionality with $\Sigma$ and $\Sigma$ in Eqs. (33, 34). This assumption leads to nice symmetrical results. It

$^+$Or, more generally, to correct only the suspected parts of $R^\exp$ or $R^O$. 
should be noted, however, that there is no physical reason whatsoever for systematic errors to be related to the corresponding experimental uncertainties. So the mathematically logical assumption of Chao is physically completely arbitrary, and may even be misleading. One should use Chao's formulas with some care; they present a convenient statistical tool to be used if inconsistencies arise to gain quantitative understanding of the discrepancies, but they can never be a substitute for the real task of the physicist, namely to find the physical source of the discrepancy.

Using the corrections given in Eqs. (33,34) the results of the adjustment of corrected integral data and a-priori values leads exactly to the results of Sect. 2, provided that the posterior covariance matrices are multiplied with $\chi^2/n$. This gives a foundation to the practice of scaling-up covariance matrices with the factor $\chi^2/n$ when this quantity exceeds 1.

In Ref. [22] Chao has also included model uncertainties, which leads to differences compared to the above treatment, provided that $\chi^2 >> 1$. His final expression for adjusted integral data in the presence of method uncertainties can be denoted as:

$$ R' - R^\text{exp} = -V(N + \bar{V} + \frac{\lambda}{2+\lambda} \bar{V})^{-1} E^O, \quad (35) $$

which differs from the usual solution by a scaling factor $(\lambda+2)/\lambda$ of matrices $\bar{V}$ and $N$. The corresponding covariance matrix is much more complicated [24]. It is remarkable to note the asymmetry in this equation. This is due to the fact that $\lambda$ is determined by imposing a $\chi^2$-test on [22]:

$$ n = E^O T [((1+\lambda)(N + \bar{V}) + \bar{V})^{-1} E^O. \quad (36) $$

Evidently, no "method negligence" is included, i.e. it is assumed that $\bar{V}$ has the character of statistical "noise" (Sect. 3). If a "negligence" is included in the method as well, reformulation of the adjustment problem as discussed in Sect. 3 reduces the problem to the previous situation, i.e. a solution given by Eqs. (3-7), with covariance matrices multiplied with $\chi^2/n$.

8. DATA EVALUATION AND UNCERTAINTY ASSESSMENT

Prior to any adjustment procedure all data, sensitivity matrices and covariance matrices need to be evaluated. A few comments on the determination of neutron spectra have already been given (Sect. 3). Some other remarks on the data evaluation problems follow below.

Experimental data: self shielding

As an example of the evaluation of integral data we mention here the analysis of the STEK integral data which are small-sample reactivity worths in five reactor cores [5]. Most STEK samples were chemical and isotopic mixtures, subjected to self-shielding.

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† This assumption is not made in a new report of Chao and Mir, see ref. [33].
Another complication was that part of the reactivity worth is caused by scattering rather than capture. When these effects are small, corrections could be applied to obtain "clean" data, referring to the capture reactivity effect of a pure isotope measured in infinite dilution. In our application [1-5] we have corrected only for chemical admixtures and scattering effects. In the adjustment a number of isotopic mixtures having various degrees of self-shielding were considered. This approach avoids iterative procedures which would be needed otherwise. On the other hand, the size of the matrices involved in the adjustment process became quite large and strong correlations were introduced in the a-priori cross sections belonging to the various samples. By extending the a-priori cross section vector with isotopic cross sections in infinite dilution the required adjusted data were obtained ("indirect adjustment" [1]). The covariance matrix $\mathbf{V}$ of the (corrected) experimental data was obtained by including uncorrelated statistical experimental errors, errors due to uncertainties in sample composition, normalization errors and errors due to the scattering correction [1,5]. Another - quite different - example of integral data evaluation is reported in Ref. [29]. See also Refs. [13-15].

A-priori cross sections

The assessment of uncertainties to the a-priori cross sections should be made during the evaluation process. In the resolved resonance range a straightforward way to obtain uncertainties in multi-group constants is to use the uncertainties of the resolved resonance parameters and to apply the error propagation law [1]. Complications arise because of self-shielding and correlations between various resonance parameters [1]. At higher energies three evaluation procedures could be distinguished, which are based on:

(a) nuclear model calculations with parameters from external sources (i.e. no differential cross section measurements available),
(b) selection, averaging, smoothing, interpolation and extrapolation of differential data [34],
(c) model calculations with parameters "tuned" to available differential data.

In the first approach [1,2] the parameters could come from averaged resolved resonances (e.g. mean level spacing, average capture width, neutron strength function), level scheme data, Q-values, differential data from related cross sections (e.g. optical-model parameters from total cross section data), systematic or theory. These "external" sources can be used to assign parameters and their uncertainties (with possible correlations). From these parameters the covariance matrix of (group) cross sections can be calculated [1,2]. Additional uncertainties in multi-group cross sections dealing with the validity of the model cannot be attached to parameters, but need to be included also. Here we refer to Ref. [2] for a discussion of these statistical-model errors. An advantage of this approach is that both "short-range" and "long-range" correlations are introduced in quite a natural way.

In the second approach [34] mentioned above the main task of the evaluator is to review the various differential data and to trace
back the uncertainties and correlations. Usually the evaluator can only indicate for each measurement the statistical uncertainties and a common normalization error, expressing the measured points by

$$\sigma_i = \eta f_i,$$

(37)

where $\eta$ is the normalization constant. For lognormal distributed quantities the relative covariance can be expressed as \[11\]

$$\left< \frac{q_{ij}}{q_i q_j} \right> = (1 + c^2)(1 + r_i r_j \rho_{ij}) - 1,$$

(38)

where $c$ and $r_i$ are the fractional uncertainties in $\eta$ and $f_i$ and $\rho_{ij}$ is the (unknown) correlation coefficient of $f_i$ and $f_j$. Assuming that $\rho_{ij}$ is a short-range correlation it could be parametrized as \[11\]

$$\rho_{ij} = (1 - \Theta) \delta_{ij} + \Theta \exp \left[ - \frac{(i-j)^2}{2\gamma^2} \right].$$

(39)

It is the task of the evaluator to assume reasonable values for $\Theta$ and $\gamma$ which refer to the fraction of short-range correlation and its range, respectively. This information should follow from the experimental method (resolution). From Eqs. (37-39) or similar representations the co-variance matrix of the (group) cross sections can be calculated.

In the last-mentioned approach, which is probably closest to common evaluation practice, the parameters are modified (mostly within their uncertainties) to improve the agreement with differential data. This may lead to smaller uncertainties in the parameters. However, these uncertainties are difficult to estimate when the parameters are "tuned" by means of "trial and error" methods. Therefore, it seems better to introduce the differential data in an adjustment process, like described in Sect. 5. Integral data could be included, if needed, at the same time. Thus we advocate to base the a-priori data and their uncertainties upon nuclear model calculations and to adjust the calculated data to differential and/or integral data.

9. CONCLUSIONS

Most cross section adjustment problems can be formulated by Eqs. (1,2) with the solution \[1\] given by Eqs. (3-7), assuming no uncertainties in $G$ \[Eq. (2)\] and no correlations between experimental data and a-priori data. Important simplifications may be obtained by taking advantage of the statistical independence of the various components (partitioning, partial adjustment) or a transformation of the parameters \[1,10\].

If the relation between a-priori integral data and parameters is not rigid, "method" errors $\bar{v}$ need to be included in the covariance matrix of the difference of experimental and calculated integral data \[Eq. (12)\]. This is easily shown by extending the adjustable vector
with elements of $G^{[1,10,16,23]}$ and/or a "noise" term $\Delta$ to account for those statistical uncertainties which are not included in the elements of $G$ (Sect. 3). In this reformulated adjustment problem the adjustments in $G$ and $\Delta$ are obtained explicitly from the usual solution Eqs. (3-7).

Instead of adjusting multi-group cross sections one could also adjust the underlying physical model parameters $[1,13]$ when their number is not too large. In a practical application $[6,25]$ a selection of important statistical-model parameters has been adjusted by applying Eqs. (16-18) $[1]$ after completing the multi-group cross section adjustment. The parameters were used to obtain "adjusted" point cross sections $[25]$. Adjustments in the resolved resonance range are much more difficult to transfer to point cross sections (Sect. 4).

"Direct" point cross-section adjustment has been applied at HEDL $[10-12]$. This method is also problematic in the resolved resonance range, but it has the important advantage that differential experimental data could be easily used together with the integral data $[12]$.

In principle it is also possible to formulate the adjustment problem for continuous functions $[19]$, but it seems rather difficult to include correlations in a proper way (Sect. 5).

Logarithmic adjustment $[1,10,11]$, i.e. assuming lognormal distributed adjustable quantities, has to be recommended for most cross sections, to ensure positivity and to cope with asymmetric error limits. The problem can be reduced to Eqs. (1,2) when the relation between integral data and the logarithm of cross sections is assumed to be linear $[1]$. In that case the distribution of the adjusted cross sections is again lognormal $[1]$. Otherwise, iterations may be needed and the statistical distribution of adjusted parameters is more complicated $[10]$. Furthermore, interpretation problems may arise due to the difference between the expectation and most probable values (Sect. 6).

The statistical $\chi^2$-test (Eq. 8) is a useful tool to detect inconsistencies, of which the source should be investigated by means of a re-analysis of all data involved. When this source cannot be detected, but it is a-priori known that systematic errors are absent (or small) in either the experimental or a-priori data still a correction should be applied to the suspected data. A statistical estimate of this "negligence" $[21]$ follows from Eqs. (30,31). If the origin of the inconsistency is completely unknown a possible - but arbitrary - approach follows from a distribution of the negligence over both experimental and a-priori data in a ratio according to their respective uncertainties $[22]$. The result of this approach leads to the usual solution Eqs. (3-7), except that the covariance matrix of the adjusted data should be multiplied with the value of $\chi^2$ per degree of freedom. Inclusion of model parameters complicates this picture. However, reformulation of the adjustment problem as discussed in Sect. 3 reduces the problem to the previous situation (Sect. 7). Chao $[22]$ treats the special case that it is a-priori known that the negligence is not due to the method, assuming method errors with a statistical ("noise") character.

In practice a large effort is needed to evaluate experimental

$^+\text{See Ref. [33] for a more recent report on this problem.}$
data, a-priori data, sensitivity matrices and covariance matrices. Usually the a-priori data are evaluated cross sections based upon model calculations and differential data. A recommended approach is to consider model parameters (based upon "external" sources) as a-priori data. The calculated point cross sections could then be adjusted to experimental data as described in Sect. 5 [11]. In such a procedure the three independent data sources, i.e. model parameters, differential cross section data and integral data are used in a consistent way, provided that the covariance matrix of the calculated cross sections is derived from the parameter covariance matrix [1,2,18]. Other information used in this process is contained in the theoretical relations and their method uncertainties.
10. REFERENCES


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Recent references:


FIGURE CAPTIONS

Fig. 1. Adopted [25] smooth correction factor for the capture cross section of $^{133}$Cs in the resolved resonance range (up to 3.5 keV) compared with relative multi-group cross section adjustments [4], mainly based upon STEK integral data.

Fig. 2. Comparison of $^{133}$Cs capture cross section calculated from resolved resonance parameters [26] (full-line histogram) with statistical-model prediction and broad-resolution data [28]. Also shown are adjustments based upon STEK [4], the p- and d-wave contribution and the additive correction applied by Ribon et al. [27].

Fig. 3. Unadjusted and adjusted multi-group capture cross sections of $^{133}$Cs based upon STEK and CFRMF integral data [4] compared with experimental data [28] at 3.5 to 200 keV.

Fig. 4. Unadjusted [26], adjusted and adopted [25] evaluated curves for the $^{133}$Cs capture cross section compared with experimental data [28] at 3.5 to 200 keV.
Fig. 3.
A NEW METHOD OF CORRELATION OF FUEL IRRADIATION EXPERIMENTS AND BASIC NUCLEAR DATA

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I - INTRODUCTION -

High quality experiments are becoming available on transactinide isotopes, with the analysis of irradiation experiments in power reactors.

In particular, several experiments have been performed in the PHENIX reactor, related to the irradiation of isolated isotope samples (PROFIL experiments /1/) or the irradiation of actual fuel pins and their analysis.

The measure of the build-up of transactinide isotopes can provide very detailed information on several cross-section and decay data of all the isotopes connected to the build-up scheme.

II - IRRADIATION EXPERIMENT AND PERTURBATION THEORY ANALYSIS -

Let us consider a fuel pin with a composition represented by an isotope density vector \( \mathbf{n} \):

\[
\mathbf{n} = (n_1, n_2, \ldots, n_J)
\]  

(\(J\) : total number of isotopes considered).

The initial value of \( \mathbf{n} \) will be represented by:

\[
\mathbf{n}(t=t_0) = (n_{j_0}, \ldots, n_{j_0})
\]  

The evolution with time of \( \mathbf{n} \) will be described in the usual formalism:

\[
\frac{d\mathbf{n}(t)}{dt} = -A\mathbf{n}(t)
\]
After an irradiation from \( t = t_0 \) to \( t = t_f \), we obtain a "final" value of vector \( \mathbf{n} \):

\[
\mathbf{n}(t = t_f) = (n_1^F, \ldots, n_j^F) \tag{4}
\]

Each component of the vector \( \mathbf{n}(t = t_f) \) can be subject to measurement. We can then obtain:

\[
\mathbf{n}'(t = t_f) = (n_1'^F, \ldots, n_j'^F) \tag{5}
\]

If we suppose that the discrepancy between the calculated and the experimental value of \( \mathbf{n} \) is mainly due to the uncertainties in basic data (i.e. mainly due to \( \delta A \)), we try to correlate \( \delta \mathbf{n}(t = t_f) \) to \( \delta A \). The same problem is a well known approach in the case of the integral experiments in the neutron field /2,3/.

To obtain a linear relation between \( \delta \mathbf{n} \) and \( \delta A \), we will use a perturbation technique. If we define \( A' \) as:

\[
A' = A + \delta A \tag{6}
\]

the data matrix that would give rise to the measured \( \mathbf{n}'(t = t_f) \) vector, we will have:

\[
\frac{d \mathbf{n}'(t)}{dt} = -A' \mathbf{n}'(t) \tag{7}
\]

Now, let us define the following adjoint systems:

\[
\frac{d \mathbf{n}^*_i}{dt} = A^T \mathbf{n}^*_i(t) \tag{8}
\]
where at each $n^*_i$ is associated a "final" condition:

$$n^*_i(t=t_F) = (\delta_{i1}, ..., \delta_{iJ})$$

(9)

where $\delta_{ji}$ are Kronecker symbols; and $i=1...J$ (total number of isotopes under consideration).

If we now multiply (7) by $n^*_i$, and (8) by $n'_i$, integrate between $t_0$ and $t_F$, and we take the difference of the two expressions, we obtain:

$$n'_i n^*_i \bigg|_{t_F}^{t_0} - n'_i n^*_i \bigg|_{t_0}^{t_F} = \int_{t_0}^{t_F} n'_i \delta A n^*_i \, dt$$

(10)

or:

$$n'_i F - n'_i F + n'_i F - n'_i n^*_i \bigg|_{t_0}^{t_F} = \int_{t_0}^{t_F} n'_i \delta A n^*_i \, dt$$

(11)

Recalling that:

$$n'_i F = n n^*_i \bigg|_{t_F}^{t_0} = n n^*_i \bigg|_{t_0}^{t_F}$$

(12)

we obtain:

$$n'_i F - n'_i F = \delta n'_i F = \int_{t_0}^{t_F} n'_i \delta A n^*_i \, dt + (n'_i - n) n^*_i \bigg|_{t_0}^{t_F}$$

(13)

At first order, we can finally write:

$$\delta n'_i F = \int_{t_0}^{t_F} n'_i \delta A n^*_i \, dt$$

(13')
Each of the $\delta n^F_i$ is a component of the vector $\delta n(t=t_F)$, that we are now able to express as a linear function of possible $\delta A$.

Similar results have been already obtained $^{14,5}$, in relation to burn-up problems.

For an actual irradiation experiment, the interval $(t_0,t_F)$ will be subdivided in $K$ intervals, where the flux $\varphi$ can be considered well known. In this case, we will have:

$$\delta n^F_i = \sum_K \delta A^K \int_{t_0}^{t_F} \frac{n \cdot n^*}{t_{OK}} dt$$  \hspace{1cm} (14)

The $\delta A^K$ (constant in the time interval $K$) is in general, of the type:

$$\delta A^K = \delta \left( \sum_j \sigma_j \varphi^K_j (r) \right)$$  \hspace{1cm} (15)

where the sum is over $J$ energy groups.

$\varphi^K_j (r)$ is the flux in group $J$, during the irradiation time interval $K$, when the flux is considered to be constant, at the reactor position $r$ where the irradiation takes place. If the flux is considered to be well known, and the perturbation due to irradiation to be small (as it is the case in actual pin irradiation), we can rewrite (15) as:

$$\delta A^K = \sum_j \varphi^K_j (r) \delta \sigma_j$$  \hspace{1cm} (16)
and then:

\[
\left( \frac{\delta n_i^F}{n_i^F} / \delta \sigma_j / \sigma_j \right) = \frac{1}{n_i^F} \sum_k^F \frac{\varphi_j^k}{\sigma_j} \int_{t_{0K}}^{t_{FK}} n_i^* n_i^* \, dt \tag{16'}
\]

represents the so-called sensitivity coefficient for the build-up of isotope \( i \) to the generic cross-section \( \sigma_j \).

III - THE CONSISTENT METHOD -

The main principle of this method /6/, is to correlate an integral experiment to a set of basic nuclear parameters of the isotopes which contribute to the discrepancy in the integral experiment comparison between calculation and experimental values.

The application of this method to the irradiation experiments can be envisaged according to the two following steps:

1°/ express \( \sigma \) as a function of \( m \) parameters. In such a way:

\[
\delta \sigma = \sum_m \frac{\partial \sigma}{\partial p_m} \delta p_m \tag{17}
\]

2°/ find the correlation among corresponding parameters in different isotopes (family correlations).
For what concerns point 1°, we can distinguish between resonance parameters (in particular the average unresolved resonance region parameters, $<D>$, $<\Gamma_n>$, $<\Gamma_f>$, $<\Gamma_Y>$) and the parameters in the continuum region.

In this energy region, possible candidates (related to the optical model of the nucleus) are, in the case of capture cross-section, which are of a major interest in irradiation experiments:

a) $K_E$ and $K_M$ parameters involved in the capture transmission coefficients, computed according to Holmes and Woosley /7/; the high sensitivity to the $K_E$ parameter has been stressed /8/;

b) as an alternative, the average capture width and observed $D$ could be considered;

c) the density parameters in the level density $\rho(E)$ function, if the Gilbert and Caméron /9/ representation is used.

For what concerns point 2°, it is necessary to establish a correlation matrix for basic parameters of different isotopes, in particular inside the same family (i.e. Cm isotope family or Am isotope family, etc...) according to nuclear parameter systematics.
This procedure will be able to correlate the high precision measurements of the $n_i^F$ final density of isotope $i$ after irradiation, with the basic parameters of the isotopes which take part to the build-up scheme.

Moreover, direct informations on some isotopes of a family, will be able to obtain informations on other isotopes of the same family.

A practical application of this method will be presented in the following paragraphs.

IV - IRRADIATION EXPERIMENTS AND THEIR SENSITIVITY TO BASIC DATA

As a practical case of interest, we will consider the analysis of the irradiation of fuel pins in a three-region core configuration in the PHENIX reactors.

The three fuel irradiation experiments are related to two central Plutonium regions, with different enrichments, and to an external U 235 fuel region. The irradiations covered approximately 300 days, with a loading factor of about 0.7.

The irradiation conditions were very satisfactory and few calculated factors (affected by low uncertainties) need to be applied to the experimental results, to reduce them to fundamental mode results.

These corrections are essentially related to:

- space effects, to take into account that at the irradiation location the fundamental mode of the flux, typical of the core region composition, is perturbed by higher order modes. This space effect correction is very small for the central Plutonium region and increases in the two external regions;
. variation of self-shielding with burn-up. These corrections, even if small, are not negligible and introduce small uncertainties on the corrected result.

In practice, the terms of the matrix $A^K$ are calculated as follows:

$$f_{s, l} f_{sh, l} \sum_j \sigma_j^{(t_0)} \varphi_{j, fund, l}^K$$

(18)

where $\sigma_j^{(t_0)}$ are cross-sections calculated for the $t=t_0$ self-shielding situation (i.e., related to the $n(t=t_0)$ density vector), and $\varphi_{j, fund, l}^K$ is the fundamental mode of the flux of region $l$, normalized to the constant power during time step $K$. $f_s$ is the correction factor due to spatial effects in region $l$, and $f_{sh, l}$ is the correction factor due to the variation of the self-shielding with build-up in region $l$. These factors are different for each term of the matrix $A^K$.

A sensitivity analysis has been performed, to point out the most relevant data for the analysis of each experiment.

In the present case, the vector $n(t=t_F)$ is represented by the density of the different isotopes in each fuel type, after the irradiation time $t_F-t_0$.

The calculation of the sensitivity coefficients (16') gives the results shown in Table I. In this table only sensitivity coefficients higher than 0.05 are indicated.
V - EFFECTS OF BASIC DATA UNCERTAINTIES

The present status of data uncertainty assessment can be taken into account to obtain standard deviations on each of the integral experiment results, i.e. on each quantity of isotope build-up.

As a starting point, we considered the following rough uncertainties on one group cross-sections and on the decay constants:

<table>
<thead>
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<th>NUCLIDE</th>
<th>TYPE OF DATA</th>
<th>UNCERTAINTY (%)</th>
<th>NUCLIDE</th>
<th>TYPE OF DATA</th>
<th>UNCERTAINTY (%)</th>
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</thead>
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<td>Pu 239</td>
<td>$\sigma_c$</td>
<td>10</td>
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<tr>
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<td>$\sigma_f$</td>
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<td></td>
<td>$\sigma_f$</td>
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<tr>
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<tr>
<td></td>
<td>$\sigma_f$</td>
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<td></td>
<td>$\beta^-$</td>
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</tr>
<tr>
<td>Np 237</td>
<td>$\sigma_c$</td>
<td>25</td>
<td>Cm 242</td>
<td>$\alpha$</td>
<td>1</td>
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<tr>
<td></td>
<td>$\sigma_{n,2n}$</td>
<td>100</td>
<td>Cm 244</td>
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<td>Np 239</td>
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<td>1</td>
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<tr>
<td></td>
<td>$\sigma_f$</td>
<td>20</td>
<td></td>
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<td></td>
</tr>
</tbody>
</table>
These data are derived from current evaluation of uncertainties /10, 11, 12/.

Table II shows the results of a standard deviation calculation on the isotope build-up experiments of Table I.

Table III shows similar results, but obtained in the somewhat artificial case of complete correlation among all the data inside each isotope family. This results show the relevance of taking into account properly the possible correlations among different isotopes. More physical hypothesis can ben taken into account to establish correlations. But, in this case, a detailed energy dependence of the sensitivity coefficients is needed.

VI - THE EFFECTS OF ENERGY SPLITTING OF THE SENSITIVITY COEFFICIENTS -

The sensitivity coefficients of Table I can be expanded according to (16').

For practical purpose we will show the expansion only for the case of the Pu 241 and Am 241. The energy group structure adopted, is shown in the following table:

<table>
<thead>
<tr>
<th>GROUP</th>
<th>LOWER ENERGY BOUNDARY</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1.35 MeV</td>
</tr>
<tr>
<td>2</td>
<td>500 KeV</td>
</tr>
<tr>
<td>3</td>
<td>40 KeV</td>
</tr>
<tr>
<td>4</td>
<td>10 KeV</td>
</tr>
<tr>
<td>5</td>
<td>0.5 KeV</td>
</tr>
<tr>
<td>6</td>
<td>Thermal</td>
</tr>
</tbody>
</table>
Tables IV and V show the isotope build-up sensitivities to Pu 241 and Am 241 multigroup cross-section data. These results show the relative importance of different energy ranges, for the series of experimental results under investigation.

VII - SENSITIVITY TO BASIC PARAMETERS - CASE OF UNRESOLVED RESONANCE PARAMETERS OF Am 241 -

As an example of the use of the consistent method in connection with irradiation experiments, we will consider the case of the Am 241 data.

Let us consider only the experimental isotope build-up data which are related to Am 241 data. These experiments are those indicated in Table V. For each of these experiments, the sensitivity can be expressed as:

\[
\frac{dn_i^F}{n_i^F} = \sum_{J \neq \text{Am241}} a_{iJ}J \frac{\delta \sigma_j}{\sigma_j} + \left[ a_{iC}^C \frac{\delta \sigma_c}{\sigma_c} + a_{iF} \frac{\delta \sigma_f}{\sigma_f} \right]_{\text{Am241}}
\]  \hspace{1cm} (19)

where the first sum is related to all the data other than Am 241 data.

If we consider now, that the capture and fission cross-section variations between 10 KeV and 500 eV can be expressed in term of unresolved resonance parameters, we can write (19) as:
\[
\frac{dn_j^F}{n_i^F} = \sum_{J \neq 241} a_{ij} \frac{\delta \sigma_j}{\sigma_j} + \sum_{K=1,4} \left[ b_{i,k}^c \frac{\delta \sigma_j^K}{\sigma_c^K} + b_{i,k}^f \frac{\delta \sigma_j^K}{\sigma_f^K} \right]_{241} + \nonumber \\
+ \left[ b_{i,5}^c \sum_{m=1,3} \frac{\partial \sigma^5}{\partial \rho_m} \frac{\delta \rho_m}{\rho_m} + b_{i,5}^f \sum_{m=1,3} \frac{\partial \sigma^5}{\partial \rho_m} \frac{\delta \rho_m}{\rho_m} \right]_{241} + \nonumber \\
+ \left[ b_{i,6}^c \frac{\delta \sigma_c^6}{\sigma_c} + b_{i,6}^f \frac{\delta \sigma_f^6}{\sigma_f} \right]_{241}
\]

(20)

where:
\[ b_{i,k}^c, b_{i,k}^f \] are the group components of the sensitivity coefficients \[ a_{i}^c \] and \[ a_{i}^f \], respectively.

The \( p_m \) coefficients are: \( \langle D \rangle, \langle \Gamma_\gamma \rangle \) and \( \langle \Gamma_n \rangle \).

The \( \frac{\partial \sigma_c}{\partial \rho_m} \) and \( \frac{\partial \sigma_f}{\partial \rho_m} \) where calculated taking into account the conservation relations:

\[ \langle \Gamma \rangle = \langle \Gamma_n \rangle + \langle \Gamma_c \rangle + \langle \Gamma_f \rangle \]

(21)

\( \langle \Gamma_f \rangle \) variations were neglected in this particular case, in view of their negligible effects.

Only l=0 data were considered at this stage.

In Table VI, we show the result of the sensitivity analysis which involves the unresolved resonance parameter and should be compared to the previous Table V, where only multigroup data were present (only data relevant to capture cross-section variations are shown).
- 43 -

VIII - UNCERTAINTY ANALYSIS INVOLVING BASIC PARAMETERS -

The Table VI is actually a "blow-up" of only a few data of Table I.

The method adopted is being applied to other isotopes and to the continuum energy region, as it was mentioned in paragraph III.

As far as uncertainties are concerned, in Table VII we show the results of an uncertainty analysis restricted to the Am 241 data and to the integral experiments which are affected by these data (see Tables I and VI).

The uncertainties used for this analysis are shown in the following table:

<table>
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<th>PARAMETER</th>
<th>UNCERTAINTY (%)</th>
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<tr>
<td>$\sigma^1_c$</td>
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<td>$\sigma^2_c$</td>
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<tr>
<td>$\sigma^3_c$</td>
<td>30</td>
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<tr>
<td>$\sigma^4_c$</td>
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<td>$\langle\sigma\rangle_{l=0}$</td>
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<tr>
<td>$\langle\Gamma_Y\rangle_{l=0}$</td>
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<td>$\langle\Gamma_n\rangle_{l=0}$</td>
<td>25</td>
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<tr>
<td>$\sigma^6_c$</td>
<td>30</td>
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</tbody>
</table>
In Table VIII, we show the standard deviations obtained on the same integral parameters if we use a full correlation hypothesis both on the $\sigma_{c}^{i}$ values and on unresolved resonance parameters. Again, it is clear that an appropriate hypothesis on the parameter correlation play a major role in the uncertainty analysis and on possible parameter adjustments.

For what concerns the data at energies higher than 10 KeV, the optical model parameter will supply the method correlations in energy and will be used as far as possible to correlate different isotopes of the same family.

IX - CONCLUSIONS -

A comprehensive method is proposed for the correlation of irradiation experiments and basic nuclear parameters. This method can be used to take into account both irradiation and standard reaction rate (fission rate in particular) experiments.

A few practical examples of the use of this method have given, in connection with the analysis of PHENIX pin irradiation experiments. A typical example of the relation between isotope build-up experimental values and nuclear parameter uncertainties has been given.

The method will be used in the framework of a program of actinide cross-section adjustments for the future version of the CARNIVAL formulaire for fast reactor core analysis.
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INDC (NDS)-121/NE.
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</table>

a - First core configuration. Enrichment in Pu : 17 %
b - Second core configuration. Enrichment in Pu : 24 %
c - Third core configuration. Composition : U-235 : 27% ; U-238 : 73 %

**TABLE 1 :** Sensitivity coefficients for nuclide build-up experiments.
<table>
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<td></td>
<td>U-238</td>
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<tr>
<td>Pu-239</td>
<td></td>
</tr>
<tr>
<td>a</td>
<td>27</td>
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<tr>
<td>b</td>
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<td></td>
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<tr>
<td>a</td>
<td>27</td>
</tr>
<tr>
<td>b</td>
<td>-8</td>
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<td>b</td>
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</tr>
<tr>
<td>c</td>
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a - First core configuration. Enrichment in Pu : 17 %
b - Second core configuration. Enrichment in Pu : 24 %
c - Third core configuration. Composition : U-235 : 27 %; U-238 : 73 %

Table 1 (continued) - Sensitivity coefficients for nuclide build-up experiments.
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<td>4.7</td>
</tr>
<tr>
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<td>2.5</td>
</tr>
<tr>
<td>U-234 c</td>
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<tr>
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<td>2.2</td>
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<tr>
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<td>1.5</td>
</tr>
<tr>
<td>U-235 c</td>
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<tr>
<td>Pu-240 c</td>
<td>13.7</td>
</tr>
<tr>
<td>Pu-241 a</td>
<td>10.3</td>
</tr>
<tr>
<td>Pu-241 b</td>
<td>8.6</td>
</tr>
<tr>
<td>Pu-241 c</td>
<td>24.2</td>
</tr>
<tr>
<td>Pu-242 a</td>
<td>7.8</td>
</tr>
<tr>
<td>Pu-242 b</td>
<td>7.6</td>
</tr>
<tr>
<td>Pu-242 c</td>
<td>31.6</td>
</tr>
<tr>
<td>Am-241 a</td>
<td>7.0</td>
</tr>
<tr>
<td>Am-241 b</td>
<td>4.9</td>
</tr>
<tr>
<td>Am-241 c</td>
<td>24.8</td>
</tr>
<tr>
<td>Am-242m a</td>
<td>19.0</td>
</tr>
<tr>
<td>Am-242m b</td>
<td>18.8</td>
</tr>
<tr>
<td>Am-242m c</td>
<td>31.8</td>
</tr>
<tr>
<td>Am-243 a</td>
<td>10.6</td>
</tr>
<tr>
<td>Am-243 b</td>
<td>10.6</td>
</tr>
<tr>
<td>Am-243 c</td>
<td>33.6</td>
</tr>
<tr>
<td>Cm-242 a</td>
<td>15.4</td>
</tr>
<tr>
<td>Cm-242 b</td>
<td>16.8</td>
</tr>
<tr>
<td>Cm-242 c</td>
<td>31.6</td>
</tr>
<tr>
<td>Cm-244 a</td>
<td>25.4</td>
</tr>
<tr>
<td>Cm-244 b</td>
<td>25.9</td>
</tr>
<tr>
<td>Cm-244 c</td>
<td>41.9</td>
</tr>
</tbody>
</table>

a - First core configuration. Enrichment in Pu : 17 

b - Second core configuration. Enrichment in Pu 24% 

c - Third core configuration. Composition : U-235 : 27 % ; U-238 : 73% 

**TABLE 2:** Standard deviation for nuclide build-up experiments (case of no correlation among nuclear data).
<table>
<thead>
<tr>
<th>NUCLIDE</th>
<th>STANDARD DEVIATION (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>U-234 a</td>
<td>6.5</td>
</tr>
<tr>
<td>U-234 b</td>
<td>2.5</td>
</tr>
<tr>
<td>U-234 c</td>
<td>0.0</td>
</tr>
<tr>
<td>U-235 a</td>
<td>3.0</td>
</tr>
<tr>
<td>U-235 b</td>
<td>2.1</td>
</tr>
<tr>
<td>U-235 c</td>
<td>1.4</td>
</tr>
<tr>
<td>U-236 a</td>
<td>7.0</td>
</tr>
<tr>
<td>U-236 b</td>
<td>8.5</td>
</tr>
<tr>
<td>U-236 c</td>
<td>9.2</td>
</tr>
<tr>
<td>U-238 a</td>
<td>0.6</td>
</tr>
<tr>
<td>U-238 b</td>
<td>0.0</td>
</tr>
<tr>
<td>U-238 c</td>
<td>0.0</td>
</tr>
<tr>
<td>Np-237 a</td>
<td>35.9</td>
</tr>
<tr>
<td>Np-237 b</td>
<td>36.8</td>
</tr>
<tr>
<td>Np-237 c</td>
<td>34.4</td>
</tr>
<tr>
<td>Np-239 a</td>
<td>4.5</td>
</tr>
<tr>
<td>Np-239 b</td>
<td>4.7</td>
</tr>
<tr>
<td>Np-239 c</td>
<td>4.8</td>
</tr>
<tr>
<td>Pu-236 a</td>
<td>136.1</td>
</tr>
<tr>
<td>Pu-236 b</td>
<td>102.5</td>
</tr>
<tr>
<td>Pu-236 c</td>
<td>131.4</td>
</tr>
<tr>
<td>Pu-238 a</td>
<td>22.4</td>
</tr>
<tr>
<td>Pu-238 b</td>
<td>25.6</td>
</tr>
<tr>
<td>Pu-238 c</td>
<td>64.9</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>NUCLIDE</th>
<th>STANDARD DEVIATION (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pu-239 a</td>
<td>0.3</td>
</tr>
<tr>
<td>Pu-239 b</td>
<td>0.5</td>
</tr>
<tr>
<td>Pu-239 c</td>
<td>9.4</td>
</tr>
<tr>
<td>Pu-240 a</td>
<td>1.2</td>
</tr>
<tr>
<td>Pu-240 b</td>
<td>1.3</td>
</tr>
<tr>
<td>Pu-240 c</td>
<td>19.0</td>
</tr>
<tr>
<td>Pu-241 a</td>
<td>8.1</td>
</tr>
<tr>
<td>Pu-241 b</td>
<td>6.8</td>
</tr>
<tr>
<td>Pu-241 c</td>
<td>41.8</td>
</tr>
<tr>
<td>Pu-242 a</td>
<td>8.3</td>
</tr>
<tr>
<td>Pu-242 b</td>
<td>8.3</td>
</tr>
<tr>
<td>Pu-242 c</td>
<td>62.9</td>
</tr>
<tr>
<td>Am-241 a</td>
<td>4.7</td>
</tr>
<tr>
<td>Am-241 b</td>
<td>2.4</td>
</tr>
<tr>
<td>Am-241 c</td>
<td>45.8</td>
</tr>
<tr>
<td>Am-242m a</td>
<td>5.8</td>
</tr>
<tr>
<td>Am-242m b</td>
<td>10.2</td>
</tr>
<tr>
<td>Am-242m c</td>
<td>64.4</td>
</tr>
<tr>
<td>Am-243 a</td>
<td>11.7</td>
</tr>
<tr>
<td>Am-243 b</td>
<td>12.5</td>
</tr>
<tr>
<td>Am-243 c</td>
<td>74.2</td>
</tr>
<tr>
<td>Cm-242 a</td>
<td>16.6</td>
</tr>
<tr>
<td>Cm-242 b</td>
<td>17.0</td>
</tr>
<tr>
<td>Cm-242 c</td>
<td>65.8</td>
</tr>
<tr>
<td>Cm-244 a</td>
<td>34.3</td>
</tr>
<tr>
<td>Cm-244 b</td>
<td>35.5</td>
</tr>
<tr>
<td>Cm-244 c</td>
<td>100.0</td>
</tr>
</tbody>
</table>

a - First core configuration. Enrichment in Pu : 17 %
b - Second core configuration. Enrichment in Pu : 24 %
c - Third core configuration. Composition : U-235 : 27 %; U-238 : 73 %

**TABLE 3** : Standard deviation for the nuclide build-up experiments (case of complete correlation among nuclear data).
<table>
<thead>
<tr>
<th>BUILD-UP OF</th>
<th>SENSITIVITY (%) TO Pu-241 DATA</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>σ_C IN GROUP</td>
</tr>
<tr>
<td></td>
<td>1 2 3 4 5 6</td>
</tr>
<tr>
<td>Pu-241 a</td>
<td>-0.2 -1.9 -1.8 -1.9 -0.2</td>
</tr>
<tr>
<td>Pu-241 b</td>
<td>-0.2 -1.6 -1.4 -1.3 -0.1</td>
</tr>
<tr>
<td>Pu-241 c</td>
<td>0.4 -0.7 -2.6 -1.2 -0.9</td>
</tr>
<tr>
<td>Pu-242 a</td>
<td>0.2 1.3 12.0 11.3 12.0 1.0</td>
</tr>
<tr>
<td>Pu-242 b</td>
<td>0.2 1.4 12.1 10.3 9.8 0.5</td>
</tr>
<tr>
<td>Pu-242 c</td>
<td>0.5 3.4 32.5 29.2 27.5 1.8</td>
</tr>
<tr>
<td>Am-241 a</td>
<td>0.1 0.8 7.5 7.0 7.5 0.6</td>
</tr>
<tr>
<td>Am-241 b</td>
<td>0.1 0.9 7.5 6.4 6.1 0.3</td>
</tr>
<tr>
<td>Am-241 c</td>
<td>0.5 3.5 33.2 29.8 28.1 1.8</td>
</tr>
<tr>
<td>Am-243 a</td>
<td>0.1 0.6 5.2 4.9 5.2 0.4</td>
</tr>
<tr>
<td>Am-243 b</td>
<td>0.1 0.6 5.6 4.7 4.5 0.2</td>
</tr>
<tr>
<td>Am-243 c</td>
<td>0.5 3.5 33.2 29.8 28.1 1.8</td>
</tr>
<tr>
<td>Cm-244 a</td>
<td>0.1 0.6 5.2 4.9 5.2 0.4</td>
</tr>
<tr>
<td>Cm-244 b</td>
<td>0.1 0.6 5.6 4.7 4.5 0.2</td>
</tr>
<tr>
<td>Cm-244 c</td>
<td>0.5 3.5 33.2 29.8 28.1 1.8</td>
</tr>
</tbody>
</table>

- **a** - First core configuration. Enrichment in Pu: 17%
- **b** - Second core configuration. Enrichment in Pu: 24%
- **c** - Third core configuration. Composition: U-235: 27%; U-238: 73%

**TABLE 4** - Nuclide build-up sensitivities to Pu-241 multigroup cross section data.
<table>
<thead>
<tr>
<th>BUILD-UP OF</th>
<th>$\sigma_c$ IN GROUP</th>
<th>$\sigma_f$ IN GROUP</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1</td>
<td>2</td>
</tr>
<tr>
<td>Pu-238 a</td>
<td>0.2</td>
<td>2.0</td>
</tr>
<tr>
<td>Pu-238 b</td>
<td>0.2</td>
<td>2.0</td>
</tr>
<tr>
<td>Am-241 a</td>
<td>-0.2</td>
<td>-2.2</td>
</tr>
<tr>
<td>Am-241 b</td>
<td>-0.2</td>
<td>-1.6</td>
</tr>
<tr>
<td>Am-242m a</td>
<td>0.5</td>
<td>5.2</td>
</tr>
<tr>
<td>Am-242m b</td>
<td>0.6</td>
<td>6.0</td>
</tr>
<tr>
<td>Am-242m c</td>
<td>0.6</td>
<td>6.5</td>
</tr>
<tr>
<td>Cm-242 a</td>
<td>0.5</td>
<td>5.1</td>
</tr>
<tr>
<td>Cm-242 b</td>
<td>0.6</td>
<td>5.9</td>
</tr>
<tr>
<td>Cm-242 c</td>
<td>0.6</td>
<td>6.5</td>
</tr>
</tbody>
</table>

a - First core configuration: Enrichment in Pu: 17%
b - Second core configuration: Enrichment in Pu: 24%
c - Third core configuration: Composition: U-235: 27%; U-238: 73%

Table 5: Nuclide build-up sensitivities to Am-241 multigroup cross section data.
<table>
<thead>
<tr>
<th>BUILD-UP OF</th>
<th>SENSITIVITY (%) OF AM-241 CAPTURE CROSS SECTION</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>$\sigma_1$</td>
</tr>
<tr>
<td>Pu-238 a</td>
<td>0.2</td>
</tr>
<tr>
<td>Pu-238 b</td>
<td>0.2</td>
</tr>
<tr>
<td>Am-241 a</td>
<td>-0.2</td>
</tr>
<tr>
<td>Am-241 b</td>
<td>-0.2</td>
</tr>
<tr>
<td>Am-242m a</td>
<td>0.5</td>
</tr>
<tr>
<td>Am-242m b</td>
<td>0.6</td>
</tr>
<tr>
<td>Am-242m c</td>
<td>0.6</td>
</tr>
<tr>
<td>Cm-242 a</td>
<td>0.5</td>
</tr>
<tr>
<td>Cm-242 b</td>
<td>0.6</td>
</tr>
<tr>
<td>Cm-242 c</td>
<td>0.6</td>
</tr>
</tbody>
</table>

*a* - First core configuration. Enrichment in Pu : 17 %

*b* - Second core configuration. Enrichment in Pu : 24 %

*c* - Third core configuration. Composition : U-235 : 27 % ; U-238 : 73 %

\[Table 6 - Nuclide build-up sensitivities to Am-241 multigroup cross section data and basic parameters (case of unresolved resonances)\].
<table>
<thead>
<tr>
<th>NUCLIDE</th>
<th>STANDARD DEVIATION (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pu-238 a</td>
<td>4.6</td>
</tr>
<tr>
<td>Pu-238 b</td>
<td>4.2</td>
</tr>
<tr>
<td>Am-241 a</td>
<td>5.3</td>
</tr>
<tr>
<td>Am-241 b</td>
<td>3.4</td>
</tr>
<tr>
<td>Am-242m a</td>
<td>12.1</td>
</tr>
<tr>
<td>Am-242m b</td>
<td>12.8</td>
</tr>
<tr>
<td>Am-242m c</td>
<td>14.7</td>
</tr>
<tr>
<td>Cm-242 a</td>
<td>11.8</td>
</tr>
<tr>
<td>Cm-242 b</td>
<td>12.4</td>
</tr>
<tr>
<td>Cm-242 c</td>
<td>14.7</td>
</tr>
</tbody>
</table>

- **First core configuration.** Enrichment in Pu: 17%
- **Second core configuration.** Enrichment in Pu: 24%
- **Third core configuration.** Composition: U-235: 27%; U-238: 73%

**Table 7** - Standard deviation for nuclide build-up experiments due to multigroup cross section data and unresolved resonance parameters of Am-241 (case of no correlation among nuclear data).
<table>
<thead>
<tr>
<th>NUCLIDE</th>
<th>STANDARD DEVIATION (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pu-238 a</td>
<td>7.1</td>
</tr>
<tr>
<td>Pu-238 b</td>
<td>6.4</td>
</tr>
<tr>
<td>Am-241 a</td>
<td>8.0</td>
</tr>
<tr>
<td>Am-241 b</td>
<td>5.3</td>
</tr>
<tr>
<td>Am-242m a</td>
<td>18.6</td>
</tr>
<tr>
<td>Am-242m b</td>
<td>19.6</td>
</tr>
<tr>
<td>Am-242m c</td>
<td>22.5</td>
</tr>
<tr>
<td>Cm-242 a</td>
<td>18.2</td>
</tr>
<tr>
<td>Cm-242 b</td>
<td>19.1</td>
</tr>
<tr>
<td>Cm-242 c</td>
<td>22.5</td>
</tr>
</tbody>
</table>

a - First core configuration. Enrichment in Pu : 17 %
b - Second core configuration. Enrichment in Pu : 24 %
c - Third core configuration. Composition : U-235 : 27 %; U-238 : 73%

Table 8 - Standard deviation for nuclide build-up experiments due to multigroup cross section data and unresolved parameters of Am-241 (case of complete correlation both for $\sigma_i^C$ values and for unresolved resonance parameters taken separately).
INTRODUCTION

The structural materials for which nuclear data are required are those present in the reactor core, in the shield and as trace quantities in the coolant. Nuclear data for core structural materials are required because absorption in them affects the neutron balance. A knowledge of the scattering cross-sections is required because these affect the reactor neutron spectrum and the neutron leakage probability. The gas production cross-sections, \( (n,\alpha) \) and \( (n,p) \), can be factors influencing the irradiation damage of structural materials. The \( (n,p) \) reaction is also a significant component of the absorption. Atomic displacement cross-sections are functions of the scattering cross-sections at energies above about 10 keV and can also depend on the low energy capture cross-sections. Transmission of neutrons through shields depends primarily on scattering cross-sections; the strength of the moderation and the effective transport cross-sections. The activation of structural materials must be predicted so that handling facilities and long term disposal can be planned. The activity of the circuit also determines the accessibility of components for maintenance.

The structural materials used in the cores of current designs of commercial fast reactors are stainless steels and nimonic alloys. The principal elements in these steels are Fe, Ni, and Cr. The small proportions of Mn, Co, and Mo can also have significant effects. Small pads of stellite, which contains a high proportion of Co, can also be present in the core or breeder regions. The activation of these pads can present handling problems. Stainless steel is also used in the cores of the UK Advanced Gas Cooled Reactors. In light water reactors the fuel canning material is zircaloy, which is predominantly zirconium. Inconel and stainless steel are used for grids and niobium is being used for pressure tubes in CANDU.

Steels are major constituents of reactor shields and of components in the coolant circuit. Inconel, which has a high nickel content, and stellite are used in circuit components.

The build-up of strong gamma emitters within the coolant circuit and on reactor components causes maintenance problems. The predominant gamma activity in the coolant circuits of light water reactors arises from the isotopes Co58 and Co60. These are reaction products which result from trace quantities of cobalt and nickel leached out of circuit components, irradiated in the core and then placed out in various parts of the circuit, such as pumps and heat exchangers. Circuit activation is also of importance in fast reactors.

Integral measurements can be of one of the following types:

(i) Measurements which give information about a single reaction in a known spectrum.
The spectrum can be one of the following:

- Thermal Maxwellian
- Resonance Integral
- Fission spectrum
- Standard Benchmark field
- Reactor spectra

The spectrum should either be measured, reliably calculated or close to the reactor spectrum for which predictions are to be made. Such measurements can be made when the reaction results in a radioactive product, or some other product which can be measured (e.g. He). A reactivity worth measurement can also provide reaction data when the worth depends predominantly on one reaction. Generally, reactivity worth measurements depend on both the absorption and scattering cross sections and, in a fast reactor spectrum, the neutron importance spectrum is a source of uncertainty in calculating the moderation component. In a thermal reactor the absorption effect can predominate. Absorption cross-sections can also be derived from null-reactivity test zones when all the other reactions can either be measured or calculated accurately.

Thick sample broad resolution transmission measurements can give information about the average resonance structure in the total cross section.

(ii) Measurements which give information about a single substance.

Reaction rate scans and spectrum measurements through a large block of the substance give information about a combination of reactions for the substance.

(iii) Integral properties of mixtures and critical assemblies.

More general types of integral data give information about a combination of reactions and substances and it is only by simultaneously analysing a set of measurements that information relating to single reactions can be obtained.

The present paper discusses some measurements which are in the first two categories, in particular:

- thermal reactor resonance integrals;
- helium production reactions;
- fast reactor activation reactions;
- structural material absorption reactions deduced from null-reactivity test zone measurements;
- iron block transmission measurements;
Activation reaction measurements in a reactor spectrum can provide directly the required data for use in predicting activity in commercial power reactors. Adjustment factors can be derived which can be applied to power reactor calculations. However any differences in the resonance shielding of the reaction and in the overall spectrum shape must be allowed for. Resonance shielding can be very important. For example, shielding in the cobalt resonance at 132 KeV is very large in a satellite pad but is negligibly small for the small quantities of cobalt in steel or in the coolant. The neutron spectrum in a zero power critical assembly can differ from that in a power reactor because of the effects of fuel burn-up and the higher temperature of the power reactor. The temperature affects resonance shielding and, consequently, the overall neutron spectrum shape. The differential cross section data must be sufficiently accurate to permit measurements made in a zero power facility to be extrapolated to the power reactor spectrum and for any necessary adjustments to be made in the resonance shielding.

CHARACTERISTICS OF CROSS-SECTIONS FOR Cr, Mn, Fe, Co AND Ni

Table 1 gives isotopic abundances of the elements Cr, Mn, Fe, Co and Ni together with the energies of the lowest resonances and the Q values for the reactions (n,p) and (n,α).

For Cr, Fe and Ni the only resonances below 1 KeV are in the isotope Fe58 (0.3% abundance) which has resonances at 230 eV and 359 eV. The capture cross sections have a 1/√ form up to above 100 eV.
Resonance integrals have been compiled by Gryntakis and Kim (1975) (1). Calculated values of the resonance integrals, \( I_0 \), and the resonance components, \( I'_0 \), are:

<table>
<thead>
<tr>
<th>Element</th>
<th>( I_0 )</th>
<th>( I'_0 )</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cr</td>
<td>1.51</td>
<td>0.17</td>
</tr>
<tr>
<td>Fe</td>
<td>1.86</td>
<td>0.71</td>
</tr>
<tr>
<td>Ni</td>
<td>3.0</td>
<td>1.0</td>
</tr>
</tbody>
</table>

It is seen that the 1/√ component predominates over the resonance component, \( I_0 \). Resonance integrals for these elements must be obtained by reactivity perturbation measurements. Because the resonances are so high in energy corrections must be made for departure of the flux spectrum from 1/E form and resonance shielding effects can be significant. Measured values of \( I'_0 \) have large uncertainties, typically ± 30%.

For the monoisotopic elements Mn and Co the (n,γ ) reaction results in radioactive products and the resonances are at lower energies. Because of this the values of \( I'_0 \) can be measured more accurately. Again it is necessary to allow for departures of the flux spectrum from 1/E form and to use thin foils or deposits to reduce resonance shielding effects. Typical values are:
\[
\begin{array}{|c|c|c|}
\hline
\text{Isotope} & I_o \text{(barns)} & I'_o \text{(barns)} \\
\hline
\text{Mn55} & 14 & 8 \\
\text{Co59} & 40 & 31.5 \\
\text{Co60m} & 31 & 23.0 \\
\text{Co60g} & 71 & 54 \\
\hline
\end{array}
\]

For these isotopes the accuracy of \( I'_o \) is typically ±10% to 20%.

Resonance integrals have also been measured for some of those isotopes of Cr, Fe and Ni which activate, but the uncertainties in \( I'_o \) are large. The measurements might, nevertheless, provide some confirmation that no small low energy resonances are undetected.

Although the thresholds of \((n,p)\) and \((n,\alpha)\) reactions have positive \( Q \) values in some isotopes the effective thresholds are at MeV energies. Fission spectrum averaged values of \((n,\alpha)\) reactions can be related to the reactor spectrum averaged values to within the required accuracies for helium production rates of about ± 20%.

Table 2 shows the reactions which result in active products and those which have been measured in either reactor spectra, fission spectra or benchmark fields. The table suggests that integrals \((n,p)\) cross-sections for the elements could be obtained by combining activation measurements for the constituent isotopes.

**STRUCTURAL MATERIAL TRANSMISSION MEASUREMENTS - THE IRON BLOCK EXPERIMENT**

Benchmark experiments for reactor shielding were reviewed at a recent NEA Specialists Meeting (2). The experiments include measurements of transmission of neutrons through large blocks of iron. A fission plate provides a source of neutrons on one face of the iron block and flux spectra are measured at points through the block. In the ASPIS experiment (3) threshold detectors were used to measure the high energy flux, proton recoil counters measure the spectra between about 7 KeV and, typically, about 4 MeV (with some data up to 10 MeV) and resonance sandwich foils measure the low energy flux. The threshold activation reactions used were \(\text{Al}_{27}(n,\alpha)\text{Na}_{24}, \text{Si}_{28}(n,p)\text{P}_{32}, \text{In}_{115}(n,n')\text{In}_{115m} \) and \(\text{Rh}_{103}(n,n')\text{Rh}_{103m}\). The resonance sandwich foil reactions were \(\text{Cd}(\text{Au}(n,\gamma)), \text{Cu}_{63}(n,\gamma), \text{W}_{186}(n,\gamma) \) and \(\text{Au}(n,\gamma)\). A range of proton recoil detectors were used. For high energies (> 0.8 MeV) an NE213 organic scintillation counter was employed and for the range 7 KeV to 1.75 MeV seven different gas proportional counters were used. Spectrum measurements were made at 4 distances from the neutron source plate, 20.3cm, 50.8cm, 76.2cm and 101.6cm.
In an analysis of the ASPIS measurements by M C G Hall (4) a comparison was made between the adjusted iron cross-sections obtained by fitting to the ASPIS measurements and the adjustments to the iron data incorporated in the FGL5 cross-section set. These latter adjustments were based on measurements of the neutron balance, reactivity worths and spectra in critical compositions containing iron and so they are more sensitive to iron absorption and moderation cross-sections and not so sensitive to the neutron transmission properties. The transmission through the iron block is more sensitive to minima in the total cross-section and provides data relating more specifically to iron.

Differences in the two sets of adjusted cross-sections might arise for one of the following reasons:

(a) The adjustments might partly compensate for approximations in the calculation methods, such as group averaging and the use of the transport approximation in core physics calculations.

(b) The adjustments to individual cross-sections and energy ranges are not separately significant and it is only the combined set of adjustments which is significant. The different sensitivities of the two types of experiment might mean that adjustments made to fit one type do not significantly improve the accuracy of prediction of the other type.

(c) Some of the assumed uncertainties in the cross-sections might be unrealistic and this might have resulted in adjustments to fit the two types of experiment being different.

The analyses of the ASPIS measurements were performed using discrete-ordinates multigroup transport codes. The total, inelastic and absorption cross-sections of iron (UKNDL DFN908) were adjusted in 100 groups according to the Principle of Maximum Likelihood.

The adjusted cross section set, FGL5, took into account integral measurements of $k_{eff}$, $k_{s}$, central buckling, central reaction-rate-ratios, central spectra and central reactivity worths. The calculations were performed using collision-probability cell-averaging with anisotropic scattering treated using the transport approximation. The scattering and absorption cross-sections of iron (UKNDL DFN980) were adjusted, along with other cross-sections, to give a least squares fit to differential cross-section data and integral measurements. The ten-group adjustments were interpolated to produce a smooth set of adjustments.

Calculations of the neutron spectra in the iron block were made by M Hall using the three iron cross-section sets; DFN908, the ASPIS adjusted set and the FGL5 set. The McBEND Monte Carlo code was used, and the spectra were calculated in eleven energy groups (from 10 keV to 15 MeV) at the two penetrations 75cm and 100cm. Figure 1 and Figure 2 present the fractional deviations of the fluxes calculated using cross-sections relative to the unadjusted values (with the Monte Carlo statistical uncertainty bands). Features of the results are:
In the first three groups the flux calculated using FCL3 iron cross-sections is the same as for DFM908. This is because these groups are outside the range of the ten-group adjustment scheme; this is an example of the different energy range of sensitivities of the integral data used to produce FCL5 and the ASPIS measurements.

There is quite good agreement between the calculations using FCL5 and the ASPIS adjusted cross-sections in groups four to eleven. In both cases the flux is reduced at energies above the inelastic threshold and increased at energies below. This is consistent with the increase in the inelastic cross-section found in both adjusted sets.

This agreement suggests that it should be possible to produce an adjusted cross-section set which takes into account both core physics and shielding integral data. It also gives confidence that adjusted cross-sections will have wide ranges of validity and need not be restricted to applications within the range of the types of integral data used to produce them.

4 MATERIAL ACTIVATION REACTIONS

Structural material reactions which produce strong gamma emitting products are important because the activity which they induce in core components presents handling problems. Trace quantities of structural materials can be carried around the coolant circuits, activated and plated out on components. This can present handling or maintenance problems.

Such activation of structural materials can be measured directly, both in zero power critical facilities and operating power reactors. Measurements have been made in fast reactor spectra in Zebra, both for reactions in separate elements and in the alloys used in fast reactors. The alloys for which measurements have been made are the stainless steels M316 and FV548, the nimonic alloy PE16 and the cobalt alloy stallita.

Following long irradiations in the Zebra assemblies the metallic samples were dissolved and liquid sources made for high-resolution Ge-Li gamma spectrometry. The measurements were made relative to Pu239 and U238 fission and normalised to the calculated values of these reactions. Measurements have been made in different assemblies and in both core and breeder region spectra. Measured values and the calculated spectra have been reported by Murphy et al (5) and later measurements have been reported by Sanders et al (6). Further measurements have been made and a report on this work is to be published at the end of 1981. The accuracy of these measurements is about ± 7% (1σ) and the agreement with calculation is typically ± 20%. It is also proposed to analyse samples of materials removed from the UK Prototype Fast Reactor. These measurements will indicate whether burnup of activation products and two stage reaction products are significant.
These activation measurements, together with the calculated spectra, provide simple integral cross section measurements which have an accuracy of about ± 10%.

The reactions for which measurements have been made are given in Table 3.

Measurements have also been made in fission spectra and in Benchmark Fields, such as CFFRF and the MolSE facility. These measurements relate primarily to reactions which can be applied in dosimetry (7).

Integral measurements would be possible for many of the remaining reactions which result in radioactive products. The (n,p) reactions in Cr52, 53 and 54, Fe56 and 58 and Ni61, 62 and 64 result in products with very short half-lives but measurements should be possible. Measurements for the (n,p) reactions in Mn55 and Fe57 would be more difficult. Activation measurements should also be possible for Fe54(n,γ), the short lived activities in Cr54 (n,α), Ni62 (n,γ) and Ni64 (n,α). Measurements of Cr54 (n,γ), Fe58 (n,α) and Ni62 (n,γ) would be more difficult. The most worthwhile measurements are probably the (n,p) reactions because these would cover the main contributions for the elements.

HELUM PRODUCING REACTIONS

Integral measurements of (n,α) reactions in structural materials have been made by measuring the helium in the samples following a long irradiation in a power reactor. Measurements were made by Freeman et al. (8) in the Dounreay Experimental Power Reactor and by Weitman et al (9) under a boral shield in a thermal reactor. The accuracy of the measurements is typically ± 20%. Because the effective thresholds for the reactions are above about 3 MeV it was considered acceptable to derive fission spectrum averaged values from the reactor spectrum averaged values using the threshold dosimetry monitors and calculated relationships. These are the values usually quoted. Measurements have also been made by Lippincott et al. (10). Values are summarised in Table 4 which is taken from the compilation by Grytaakis (11).

In such measurements care must be taken to ensure that the samples are sufficiently pure and free from small quantities of elements with very high (n,α) cross sections such as boron and nitrogen.

As well as the measurements for pure elements, the structural materials used in power reactors have also been irradiated and analysed. This is necessary because of the possible contributions of trace elements and two stage reactions such as Ni58(n,γ) Ni59(n,α). Such reactions were observed by Weitman et al. (9) in measurements made in a thermal reactor spectrum.

It will be seen from Table 4 that the measurements, have a spread of about ± 15% for Cr, ± 30% for Fe and ± 5% for Ni.

Measurements made in an EBR11 spectrum have been reported by McElroy et al. (12). For Fe and Ni these values are about a factor of 5 lower than the fission spectrum averaged values but the value for Cr then appears to be too large.

Recent advances in mass spectrometric methods for measuring small quantities of helium could now make more accurate measurements possible.
MEASUREMENTS IN NULL-REACTIVITY TEST ZONES

Test zones of material combinations for which the measured value of $K_\infty$ is close to unity have been built in several critical facilities. By measuring relative reaction rates, values which cannot be measured directly can be derived from the neutron balance. In this way integral values of the absorption cross-sections of iron and steel have been derived.

A series of unit $K_\infty$ test zones was studied in the Assembly 8 programme of zones built in the ZEBRA Facility (13). Zone 8 C contained plates of plutonium, natural uranium, graphite and steel. U238 capture and fission rates were measured relative to Pu239 fission. Test zones 8 A and 8 B contained only plutonium, uranium and graphite and so provided a check of the Pu239 capture cross-section and $\gamma$ values. The neutron spectra were also measured. These measurements were included in the set of integral measurements used to produce the FGL5 cross-section set.

In the PROTEUS facility zones consisting of mixed pin cells comprising uranium-plutonium oxide fuel pins and diluent pins, were studied (14). These differ from the Zebra test zones in that they contain oxygen as moderator, in place of carbon, and pins instead of plates.

The heterogeneity effects are also smaller because of the use of mixed uranium-plutonium oxide fuel instead of separate plates of plutonium metal, uranium metal and graphite. One ZEBRA test zone, 8 F, contained plates of mixed uranium-plutonium oxide and uranium oxide, together with graphite. The U238 fission and capture rates relative to Pu239 fission and neutron spectra were also measured in the PROTEUS test zones. It is of particular interest to compare the calculations made using FGL5 data (15) for the ZEBRA test zones and the PROTEUS test zones to see how the comparison might be influenced by heterogeneity effects and the replacement of carbon by oxygen in the PROTEUS experiments. The results of calculations are compared in Table 5. Two sets of results are given for PROTEUS, using different oxygen data.

In FGL5 the oxygen cross-sections were not adjusted because the integral data used to produce FGL5 were not sensitive to oxygen data and the uncertainties assumed for the oxygen cross-sections were small. However, there is a marked difference between the ENDF/B-IV oxygen evaluation and the ENDF/B-VI evaluation which was used to derive FGL5. There is a strong backward scattering anisotropy below the resonance at 450 keV in the ENDF/B-IV data and the (n,p) cross-section is about 20% higher. The effect of changing to this evaluation is to increase the moderation and transport cross-sections in the energy range 200 - 400 keV. The net effect of the change depends on the leakage fraction in the core. For these zero leakage test zones it is the moderation effect which is significant. A more detailed treatment of resonance shielding in iron also affects the comparison of the present data with earlier published results. This effect is negligibly small for normal LMFBR compositions but in the test zones with a high iron fraction resonance shielding in iron is much larger, as is seen from Table 6. This resonance shielding affects the relative contributions of different energy ranges to the spectrum averaged cross-section compared with that of an LMFBR. Accurate resonance parameter data are required to relate the integral measurements in a test zone to the LMFBR spectrum averaged value.
We can see from Table 5 that the use of ENDF/B-IV oxygen data in place of the FGL5 oxygen has a large effect in the calculations for the PROTEUS cores, reducing $K_p$ by 1.0% for Core 6 and by 0.6% for Core 10. The values of $K_p$ for the PROTEUS cores using the more recent oxygen evaluation are about 1.0% higher than the values for the ZEBRA cores which, is consistent with the "pin/plate discrepancies" observed in other ZEBRA studies. The less heterogeneous ZEBRA Core 8F has a C/E value closer to PROTEUS Core 6.

When combined with the U238 fission and capture rate ratio measurements relative to Pu239 fission, an iron capture and a stainless steel capture cross section can be derived. This has been done by Seth et al (14), the accuracy of the derived spectrum averaged cross-sections being about $\pm 5\%$. The adjusted FGL5 cross-sections agree with the measured values to within the uncertainties but the ENDF/B-IV values are about 45% high.

The comparison illustrates some of the current problems and limitations of this type of integral data for structural materials:

(i) Geometrical modelling of the cells

(ii) The strong resonance shielding

(iii) Possible uncertainties in diluent data (e.g., oxygen)

An extensive programme of integral measurements on structural materials is being undertaken in the RB-2 Reactor in Italy. These employ both the Null Reactivity and Reactivity Worth Techniques (16). In this way the ratios of the capture cross sections to the U235 fission cross-sections are derived. The null-reactivity test zone compositions comprise enriched uranium oxide, borated graphite, graphite, and the structural material being investigated, in the form of microspheres having diameters of about 1 mm. Heterogeneity effects are, therefore, much reduced. The accuracy of the measurements of the ratio of structural material capture cross-section to U235 fission cross section is typically $\pm 5\%$ and measurements are made in a range of different spectra. For the intermediate spectrum measurements for Cr, Fe, Ni and stainless steel agreement with ENDF/B-IV is within or close to the measurement uncertainty.

### SMALL SAMPLE REACTIVITY WORTH MEASUREMENTS IN FAST REACTORS

For Fe, Cr and Ni the moderation component is roughly equal to the absorption component. The former component is sensitive to uncertainties in the neutron importance spectrum, which cannot be measured directly. Resonance shielding effects are also important. Nevertheless, good agreement between measured and calculated central reactivity worths have been reported for FGL5 (15) and JENDL 1 (17).

### CONCLUSIONS

Cross sections of iron, chromium and nickel.

Capture cross-sections.

Integral measurements of the total capture or $(n,\gamma)$ reactions must be derived from reactivity worth or null reactivity test-zone measurements. The $1/\gamma$ component dominates the resonance integral and consequently there are large uncertainties ($\pm 30\%$) in the resonance contribution.
This contribution is further complicated by deviations of the spectrum from 1/E form and resonance shielding effects. Accuracies of about ±5% are claimed for epithermal and fast reactor spectrum averaged values derived from null-reactivity test zone measurements but resonance shielding effects can be large in these. The cell heterogeneity of some of these experiments also complicates the interpretation. In fast reactor reactivity worth measurements the moderation contribution is of about the same magnitude as the capture contribution and consequently an accurate knowledge of the neutron importance spectrum is required.

(n,p) cross-sections

Many of the constituent isotope contributions could be measured by activation analysis although most of them produce short lived activities.

(n,α) cross-sections

Reactor measurements have been made of the helium production, which is the property of interest. The quoted uncertainties and the spread in values is typically ±20% to 30%. Techniques for measuring small fractions of helium have improved and more accurate measurements could now be made.

Scattering cross-sections

Measurements of spectra in large blocks of structural materials driven by a neutron source provide data suitable for testing and adjusting scattering cross-sections. The measurements complement spectrum measurements made in critical systems with different structural material diluents.

Activation cross-sections and the (n,γ) cross-sections of Mn and Co.

The important activation cross-sections are those which are strong gamma emitters. Spectrum averaged values of these can be measured to accuracies of 5% to 10% and several sets of measurements have been published. Again resonance shielding or sample size effects can be important for (n,γ) activation reactions and these must be given careful attention. Mass spectrometry on irradiated samples might provide information about reactions which produce stable products.

Studies relating to the FGLS adjusted cross-section library

Encouraging consistency is found between iron block spectrum calculations made using the Fe data in FGLS and the data adjusted to fit the ASPIS iron block measurements. This suggests that an adjusted library could be developed to fit both types of measurements, critical core measurements and shield transmission spectrum measurements. Problems in the calculation of some cell heterogeneities have become more apparent in recent years. In particular, there is a discrepancy between the standard calculation methods for some Zebra plate cells and pin cells of about 10% K. The evaluation for oxygen used in FGLS has also been found to contain a larger error than was assumed, compared with the ENDF/B-IV evaluation.

Atomic Energy Establishment
United Kingdom Atomic Energy Authority
Winfirth

March, 1981
REFERENCES


4. M.C.G. Hall. A Simple Comparison of Adj usted Data Sets Based on Measurements in ASPIS and in the Cores of Fast Reactor Criticals. Ibid, p.79.


<table>
<thead>
<tr>
<th>Element</th>
<th>Isotopic Abundance (Percent)</th>
<th>Threshold Energies</th>
<th>Lowest Energy Resonance (KeV)</th>
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<tr>
<td></td>
<td></td>
<td>(n,p)</td>
<td>(n,α)</td>
</tr>
<tr>
<td>Cr</td>
<td>50 4.35</td>
<td>0.26 (−0.3)</td>
<td>5.48</td>
</tr>
<tr>
<td></td>
<td>52 83.79</td>
<td>3.2 1.2</td>
<td>1.63</td>
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<tr>
<td></td>
<td>53 9.50</td>
<td>2.6 (−1.8)</td>
<td>4.19</td>
</tr>
<tr>
<td></td>
<td>54 2.36</td>
<td>6.2 1.5</td>
<td>10.30</td>
</tr>
<tr>
<td>Mn</td>
<td>55 100</td>
<td>1.8 0.63</td>
<td>0.337</td>
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<td>Fe</td>
<td>54 5.8</td>
<td>(−0.08) (−0.8)</td>
<td>3.10</td>
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<tr>
<td></td>
<td>56 91.7</td>
<td>2.9 (−0.3)</td>
<td>1.15</td>
</tr>
<tr>
<td></td>
<td>57 2.14</td>
<td>1.8 (−2.4)</td>
<td>1.63</td>
</tr>
<tr>
<td></td>
<td>58 0.31</td>
<td>5.3 1.4</td>
<td>0.23</td>
</tr>
<tr>
<td>Co</td>
<td>59 100</td>
<td>0.78 (−0.3)</td>
<td>0.132</td>
</tr>
<tr>
<td>Ni</td>
<td>58 67.76</td>
<td>(−0.4) (−2.9)</td>
<td>6.89</td>
</tr>
<tr>
<td></td>
<td>60 26.42</td>
<td>2.0 (−1.4)</td>
<td>1.29</td>
</tr>
<tr>
<td></td>
<td>61 1.16</td>
<td>0.53 (−3.6)</td>
<td>1.35</td>
</tr>
<tr>
<td></td>
<td>62 3.71</td>
<td>4.4 0.44</td>
<td>4.54</td>
</tr>
<tr>
<td></td>
<td>64 0.95</td>
<td>6.2 2.4</td>
<td>9.52</td>
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Isotope abundances, threshold energies and lowest energy resonances.

**TABLE 1**
<table>
<thead>
<tr>
<th>Isotopic Abundance (percent)</th>
<th>Active Products Produced by Reactions</th>
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<tbody>
<tr>
<td></td>
<td>(n,γ)</td>
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<tr>
<td>Cr  50</td>
<td>4.35</td>
</tr>
<tr>
<td>52</td>
<td>83.79</td>
</tr>
<tr>
<td>53</td>
<td>9.50</td>
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<tr>
<td>54</td>
<td>2.36</td>
</tr>
<tr>
<td>Mn 55</td>
<td>100</td>
</tr>
<tr>
<td>Fe 54</td>
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<td>57</td>
<td>2.14</td>
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<tr>
<td>58</td>
<td>0.31</td>
</tr>
<tr>
<td>Co 59</td>
<td>100</td>
</tr>
<tr>
<td>Ni 58</td>
<td>67.76</td>
</tr>
<tr>
<td>60</td>
<td>26.42</td>
</tr>
<tr>
<td>61</td>
<td>1.16</td>
</tr>
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<td>62</td>
<td>3.71</td>
</tr>
<tr>
<td>64</td>
<td>0.95</td>
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</tbody>
</table>

○ denotes measured in Zebra

• denotes measurements in a fission spectrum or a standard

☐ benchmark field

Reactions resulting in radioactive products

**TABLE 2**
<table>
<thead>
<tr>
<th>REACTION</th>
<th>Half-life</th>
<th>PFR core spectrum cross-section (mbarns)</th>
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<tbody>
<tr>
<td>Cr50(n,γ )Cr51</td>
<td>27.7 days</td>
<td>31</td>
</tr>
<tr>
<td>Mn55(n,γ )Mn56</td>
<td>2.58 hrs</td>
<td>70</td>
</tr>
<tr>
<td>Mn55(n,2n)Mn54</td>
<td>312 days</td>
<td>0.030</td>
</tr>
<tr>
<td>Fe54(n,p)Mn54</td>
<td>312 days</td>
<td>10.5</td>
</tr>
<tr>
<td>Fe54(n,γ )Cr51</td>
<td>27.7 days</td>
<td>0.081</td>
</tr>
<tr>
<td>Fe58(n,γ )Fe59</td>
<td>45.1 days</td>
<td>9.8</td>
</tr>
<tr>
<td>Co59(n,γ )Co60</td>
<td>5.27 years</td>
<td>54</td>
</tr>
<tr>
<td>Ni58(n,p)Co58</td>
<td>70.8 days</td>
<td>14.5</td>
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<tr>
<td>Ni60(n,p)Co60</td>
<td>5.27 years</td>
<td>0.25</td>
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**Activation Reactions Measured in Core and Broad Region Spectra in Zebra**

**TABLE 3**
From Gryntakis (1979)

<table>
<thead>
<tr>
<th>Element</th>
<th>Value ± Error</th>
<th>Reference</th>
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<tbody>
<tr>
<td>Cr</td>
<td>0.187 ± 0.066</td>
<td>Freeman (1969)</td>
</tr>
<tr>
<td></td>
<td>0.192</td>
<td>Weitman (1970)</td>
</tr>
<tr>
<td></td>
<td>0.14</td>
<td>Lippincott (1975)</td>
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<tr>
<td>Fe</td>
<td>0.333 ± 0.10</td>
<td>Byrne (1965)</td>
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<tr>
<td></td>
<td>0.198 ± 0.044</td>
<td>Freeman</td>
</tr>
<tr>
<td></td>
<td>0.363</td>
<td>Weitman</td>
</tr>
<tr>
<td></td>
<td>0.30</td>
<td>Lippincott</td>
</tr>
<tr>
<td>Ni</td>
<td>5.17 ± 0.66</td>
<td>Freeman</td>
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<tr>
<td></td>
<td>4.89</td>
<td>Weitman</td>
</tr>
<tr>
<td></td>
<td>4.78</td>
<td>Lippincott</td>
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Measurements in EBR II by McElroy (1970)

<table>
<thead>
<tr>
<th>Element</th>
<th>Value</th>
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<tbody>
<tr>
<td>Cr</td>
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<td>Fe</td>
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<td>Ni</td>
<td>0.95</td>
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**TABLE 4**

Fission Spectrum Averaged Values of (n, α) Reactions
<table>
<thead>
<tr>
<th>ZEBRA Core</th>
<th>Composition (plates)</th>
<th>s.d</th>
<th>(C-Z)/E percent</th>
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<tr>
<td></td>
<td>Pu U C SS Pu/UO₂ UO₂</td>
<td></td>
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<tr>
<td>8A</td>
<td>1 8 15</td>
<td>&lt;0.6</td>
<td>-0.2</td>
</tr>
<tr>
<td>8B</td>
<td>1 10 3</td>
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<td>-0.1</td>
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<tr>
<td>8C</td>
<td>1 6 2 15</td>
<td>&lt;0.5</td>
<td>-0.9</td>
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<tr>
<td>8F</td>
<td>4 1 2</td>
<td>&lt;0.4</td>
<td>0.5</td>
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<table>
<thead>
<tr>
<th>PROTEUS Core</th>
<th>Composition</th>
<th>s.d</th>
<th>(C-Z)/E Percent</th>
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<tr>
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<td>U/PuO₂ UO₂ SS Fe</td>
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<tr>
<td>6</td>
<td>1 1</td>
<td>&lt;0.3</td>
<td>1.7 0.7 -3.5</td>
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<td>7</td>
<td>3 3</td>
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<td>0.6 -0.2 -8.1</td>
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<tr>
<td>10</td>
<td>5 11</td>
<td>&lt;0.6</td>
<td>1.8 1.2 -8.5</td>
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FGL5 calculations of K_o for ZEBRA and PROTEUS cores

TABLE 5
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Iron Shielding Factors in Different Compositions

TABLE 6
Fractional Deviation of Adjusted Fluxes at 75 cm Penetration in Iron

FIGURE 1
DFN 908 with ASPIS adjustments
DFN 906 with core adjustments
DFN 908 unadjusted

Fractional Deviation of Adjusted Fluxes at 100 cm Penetration in Iron

FIGURE 2
Complementarity of Integral and Differential Experiments for Reactor Physics Purposes

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I. INTRODUCTION

To perform the neutron calculation of a nuclear reactor core, people need the knowledge of the physical laws which manage the history of the neutron life in a multiplying medium and also the numerical values of several parameters which intervene in these laws. These parameters are mainly neutron cross sections and neutron spectra. The numerical values can be obtained either by measurements of cross sections versus neutron energy for each component of the multiplying medium, or by a more synthetical way as the measurement of a critical size or a criticality factor of a well defined medium. The former measurements are called differential measurements, the latter integral measurements. It is obvious that the result of an integral experiment depends on the formalism which is used in the calculation and on the accuracy of this computation. Formerly, neither the cross section measurements nor the reactor calculations were accurate enough. Then the reactor physicists preferred to use the synthetic parameters deduced from integral experiments. An adjustment process allowed them to obtain with the microscopic data the same result as with the integral experiments. But now, with the improvement of the computer performances, the calculation formalisms are almost exact, as far as simple geometries of the multiplying media are concerned. The cross section measurements are better too. Differential and integral results must tend both towards the same value, which is the true value. We can say that the two kinds of experiments become
complementary. If we observe a discrepancy between them, that means that it exists an inaccuracy in one of them or in both. The resorption of such a discrepancy generally leads to an improvement of the cross section values or of the computation methods. We can quote as examples the well known discrepancy of the uranium 238 effective resonance integral which seems resolved now and the cases of the $\frac{1}{V}$ dependance of thermal capture cross section of uranium 238 and of the americium 242 capture cross section which are not yet resolved.

II. URANIUM 238 EFFECTIVE INTEGRAL

In a reactor core, the uranium nuclei are not in an infinite dilution state. Then the absorption rate of uranium 238 is not managed by the absorption cross section but by the self-shielded one. This absorption rate can be synthetized by the effective integral $I_{\text{eff}} = \int \sigma_{\text{eff}} \, du$ ($\sigma_{\text{eff}}$ is the self-shielded cross section). To compute this integral we need the knowledge of the infinite dilution cross section and the use of an appropriate formalism to represent the slowing down of the neutrons in the resonance energy range. The research of the effective integral value by this way is named the differential experiment way because we use the individual resonance parameters. But it is also possible to obtain the value of the effective integral by another way. We can directly measure the critical size or the criticality factor of a lattice in a critical facility and extract the uranium absorption rate from these experimental results. It is the integral experiment way.

Several years ago, it was well known that there was a systematic discrepancy between the two values of the effective integral. The value which was computed with the resonance parameters were always larger than the value which was deduced from the critical experiments. Obviously the reactor physicists preferred the second value. So, in their calculations they used an adjustment process to obtain the same value as in the critical experiments. They reduced the computed value of the effective integral. The decrease was equal to about one barn for thermal reactors. Formerly this was allowable. Now the cross section measurements are more accurate and the calculation formalisms are almost exact. An adjustment process was no longer justified for the physical point of view. It became absolutely necessary to search for an explanation of this discrepancy. Then, new measurements of the uranium 238 cross
sections were performed with higher experimental resolution and accuracy. It was found, first that the mean radiation width of uranium 238 was lower than the previous value and second that it must be taken account of the level-level interferences [1,2]. These two results were very important. The use of both new resonance parameters and a multilevel formalism in the computation of the cross sections improved the previous state. It is the scattering cross section which is sensitive to the interference effect. But, as this scattering cross section intervenes in the calculation of the self-shielded capture cross section, the effective integral is also sensitive to the multilevel formalism. It was shown that, for a wide range of neutron spectrum, the values of the effective integral (or of the absorption rate) of uranium 238 which is computed with these two new assumptions became on an average equal to the integral experiment values [3]. So now we do not need any adjustment of the computed effective integral values (Table I).

In this particular case the complementarity of the differential and integral experiments is well displayed, since the attempts to explain the discrepancy induced an improvement of the resonance parameter knowledge and a better computation of the self-shielded cross section. It is very satisfactory.

III. THERMAL RANGE URANIUM 238 CAPTURE CROSS SECTION

Even if the result is not yet as satisfactory as for the effective integral, the study of the dependance of the uranium 238 capture cross section versus the neutron energy in the thermal range is also another good example. This problem appears when we want to compute the temperature coefficient of a thermal multiplying lattice. Several parameters are involved in the temperature coefficient calculation: the changes of the moderator thermalization properties with temperature, the Doppler broadening of the fuel nuclei resonances, mainly the uranium 238 resonances, the variations of the thermal cross sections versus neutron energy. Actually the temperature coefficient measurements, which are integral experiments, give a result which is in disagreement with the value which is computed with the differential data. Solve this discrepancy is not so easier as for the effective integral problem because the causes must be miscellaneous. And it can exist balancing and interferences between them. The study of the uranium 238 effective integral showed that neither the new resonance parameters nor the multilevel formalism
can explain the discrepancy by Doppler effect shift. It was shown in another study that modifications of uranium 235 cross sections would be able to explain the temperature coefficient disagreement [4]. Unfortunately the proposed modifications are too large in comparison with the experimental uncertainties of the more recent and accurate fission cross section measurement [5]. Then it remained a modification of the uranium 238 cross section shape in the thermal range. M. Edenius showed that with a small deformation of the uranium 238 capture cross section it is possible to explain the temperature coefficient discrepancy [6]. But the modifications were not yet experimentally observed at that time. Recently reactor physics people thought that it would be very useful to measure again the thermal capture cross section of uranium 238. Several requests on this topic occurred in WRENDA. Since twenty years no measurement was performed, and the cross section is always supposed to have and 1/v dependance up to 0.1 eV. A few months ago we were acquainted with a new measurement of the thermal capture cross section of uranium 238. This experiment was carried out by the Harwell team [7]. The results of this measurement are not yet available. But it will be very interesting to check if this new differential measurement improves or not the temperature coefficient calculation and corroborates the Edenius assumption (Figure 1). In uranium 238 thermal capture case too, differential and integral experiments seem both useful to progress towards the right result.

IV. AMERICIUM 242m CAPTURE CROSS SECTION

With the increase of burn up, the neutron shielding of the irradiated fuels becomes important for the transport and the reprocessing. The neutron emission of the irradiated fuels must be known with a very good order of magnitude. The neutron emission is mainly produced by the \( (\alpha, n) \) reactions. The \( \alpha \) particle emitters are the even curium isotopes. Then it is necessary to have an accurate forecast of the curium amount. During the irradiation of the fuel in a reactor core, the curium isotopes are obtained by neutron capture in americium isotopes which are produced by \( \beta^- \) decay of plutonium 241 and neutron capture. Among these isotopes, americium 242m takes a prominent part and its capture cross section is not well known. Because up to now, the available amount of \( {}^{242m}\text{Am} \) was very small, only few accurate cross section
measurements were done. And it was essentially the fission cross section which was investigated. At the present time it is not possible to rely upon a direct measurement of the americium 242m capture cross section. Then, this cross section must be deduced from resonance analysis at low energy and computed with theoretical nuclear models at high energy. Several computations were performed. It exists very important differences between their results that it is necessary to check. And for this purpose integral experiments can help us another time. Nowadays chemists are able to measure the amount and the isotopic percentage of americium in an irradiated fuel. This chemical analysis can be done with high burn-up fuels or with separated isotopes samples. The second type of experiment is more accurate and can be undertaken because sufficient quantities of americium 241 are now available. By depletion calculation we know to compute the isotopic composition of a sample at the end of the neutron exposure. From the observed differences between the computed and the experimental values of the isotopic composition, we can research tendencies on the numerical values of the cross section which were used in the computation. The irradiated fuel analysis can give us informations about the average capture cross section of americium 242m. If we have the result of several irradiations performed with various neutron spectra (in a thermal reactor and in a fast reactor for instance) we can also have informations about the shape of the capture cross section versus energy. So we can perhaps make a choice between the different sets of computed capture cross sections (Figure 2).

Chemical analysis of irradiated fuels are now in progress. The results of these experiments will be useful for both reactor physicists and nuclear physicists.

V. CONCLUSION

The above mentioned examples showed us that differential and integral experiments are both useful to the reactor physicists and sometimes even to the nuclear physicists. According to the case, either of these experiments can have a more important weight. But they always remain complementary and in some circumstances both are absolutely necessary.

In conclusion we can say that, with the help of the two kinds of experiments, the knowledge of the neutron data can be improved.
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Table I

Comparative results of \((k_{\text{eff}} - 1)\) for several types of thermal reactors in units of \(10^{-5}\)

<table>
<thead>
<tr>
<th>Library</th>
<th>Lattice</th>
<th>Graphite</th>
<th>Heavy water</th>
<th>Light water steel clad</th>
<th>Light water Zr clad</th>
<th>Light water critical facility</th>
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<td>-114</td>
<td>+166</td>
<td>+35</td>
<td>-62</td>
<td></td>
</tr>
</tbody>
</table>
Figure 1

Uranium 238 Capture Cross Section

Modified shape of $\sigma_c$

$2.72\sqrt{E_0/E}$
Figure 2
Evaluated Capture Cross Section of $^{242m}_{\text{Am}}$
CONSISTENCY AND COMPLEMENTARITY

OF MICROSCOPIC AND INTEGRAL DATA

E. FORT

Paper presented at the Topical Discussion

of NEANDC on 9 April 1981
ABSTRACT

In this paper the complementarity of microscopic and integral informations is stressed. The advantages resulting from a combinaison of them in the preparation of nuclear data Libraries for reactor applications are displayed through some examples.
The microscopic and integral data are, in fact, the measures by different ways of the same primary physical phenomena. Obviously these data which give complementary informations have to be consistent.

They are complementary:

- the microscopic data result from a straightforward interpretation of raw data. They give an information detailed but sometimes with short accuracy.

- Generally the situation is reverse for the integral data which result from rather long interpretation procedure involving many data and complex calculationnal methods. Nevertheless they have the advantage to have been obtained in similar conditions to the real use. But in some cases it is possible to perform integral experiments where the usefull data are directly derived from raw data. Such data are often affected by excellent statistical accuracy. Frequently they are called "simple" or "clean" integral data*. That are the only ones we will consider in this paper.

They have to be consistent:

In order to have data libraries as close as possible to the "Physical reality" and that is a necessary condition for these libraries to have areas of utilization as wide as possible, some criteria have to be respected for the selection of data:

* The term of "simple" should be prefered.
- internal consistency of microscopic data selected by evaluation,
- internal consistency of integral data,
- consistency of microscopic data with integral data.

I - INTERNAL CONSISTENCY OF MICROSCOPIC DATA

The experimental technics and the knowledge of the nucleus have been considerably improved during the years 70's.

Essentially owing to the improved performances of the electronic part of the experimental set-up it becomes more and more possible to extract the useful signal, even small, from spurious noise. Due to the existence of sophisticated codes the raw data can be, nowadays, correctly corrected for "experimental" effects.

As a result of such improvements the experimental microscopic data recently obtained have a high level of quality. That was not always the case in the past.

In addition, the improvements of the knowledge on the properties of the nucleus have been gradually introduced in the evaluation codes: deformation effects of the nucleus in neutron entrance channel, non statistical effects in the decay of the compound nucleus, preequilibrium state influence, cleaver treatment of fission process...

In other words, the consistency of microscopic data for the...
cross sections of different nature of a same nucleus (fission and n, 2n, for instance) can, in many cases, indoubtly be proved by the mean of nuclear models.

The research of such a consistency is typically an evaluation work.

II - INTERNAL CONSISTENCY OF INTEGRAL DATA

In this paper we will consider only the experiments in which the element under consideration is present in so small quantity that it as a contribution to the second order of magnitude to the local neutron spectrum. This condition restricts the experimental methods to: reaction rate measurement, activation or irradiation method.

In each case the measured quantity is an averaged cross section on a given energetic neutron spectrum.

The connexion with the microscopic information is established only if the energetic neutron spectrum is undoubtly known, which imposes restrictions on the experimental conditions so that:

- neutron spectrum is obtained by calculations in fundamental mode (small contamination by harmonics) (small heterogeneity) from accurately known data (\(^{235}\text{U}, ^{238}\text{U}, ^{239}\text{Pu}\)), the sensitivity to less accurately known data (O\(_2\), Na, structural materials...) being small.

- correction of small magnitude for flux perturbation, for selfshielding (and then for Doppler effect...).  

.../...
The information is more complete if are performed several experiments with different neutron spectra of different characteristics of hardness. (different values for the parameter \[ r = \frac{\langle v \sigma_f \rangle}{\langle v \sigma_s \rangle}, \quad \langle v \sigma_s \rangle \text{ being the integral of slowing down density.} \]

In that case the integral data have to be consistent, that means they have to lead to similar conclusions about the microscopic data of reference in the energy ranges where the sensitivities are of significant magnitude.

An integral experiment has the property to bring an highlight on the behavior of a cross section in a given energy range and this property should be more widely used for evaluation purpose, especially when the informations cannot (or not accurately) be available from microscopic experiments (fission cross section near the threshold, capture cross section above a few hundreds KeV...).

III - CONSISTENCY OF INTEGRAL DATA WITH MICROSCOPIC DATA

The situation characterized by the agreement, within their respective error bars, of data evaluated according to the conditions given in [I] with integral data obtained in the context described in [II] is an ideal situation, rarely existing nowadays. Such a situation means:

- the values of data are probably close to their "true" values.

- the calculationnal models and especially the nuclear codes are correct and can be used for evaluation of similar nuclei.
the evaluation takes benefit of the high accuracy of integral experiments.

As an illustration of these considerations some selected examples, hopefully convincing, are presented.

Example no 1 : $^{151}\text{Sm}$

The capture cross section of $^{151}\text{Sm}$ in a typical neutron spectrum of fast breeders was obtained by irradiation of pure (content > 90 %) $^{149}\text{Sm}$ sample in "PHENIX".

It is the so called "PROFIL" experiment /1/.

The conditions of the irradiation performed in the central part of the core were such that the energetic spectrum of neutrons was known with good accuracy : correction on $<\varnothing\sigma>$ :

- for harmonics contribution : 6.5 %
- for flux perturbation due to the cladding : 0.6 %
- for spatial and evolution effects : a few percents.

Concerning microscopic informations the situations is as follows : there are only data in the resolved resonance region and the capture cross section in the unresolved resonance region, where the sensitivity is maximum, results from calculations. For this nucleus we have $\Gamma_{\gamma} \ll \Gamma_n$ and $\sigma_{n,\gamma} = 2\pi^2\lambda^2 \sum_{l=0}^{1} g \frac{<\Gamma_{\gamma}^l>}{<\Gamma_{\gamma}^l>}$

Then $\sigma_c$ depends on $<D_{1=0}^1>$ if there is no error on $\Gamma_{\gamma}^1$ determination. In the case of KIROUAC and EILAND /2/ experiment .../...
we refer to the experimental conditions (sample thickness giving a potential transmission close to 1) were such that there are a loss of weak resonances and the technics of raw data analysis excludes a large error on partial $\Gamma_\gamma$ determination. Using $\Delta_3$ statistics, KIROUAC and EILAND obtained a value of 1.76 eV for $< D^{l=0} >$, claiming there were few missing resonances. An analysis of reduced neutron widths by a maximum likelihood method /3/ gives 1.05 eV.

An adjustement of an evaluation calculated with $< D^{l=0} > = 1.76$ eV on PROFIL experiment data performed with "BARRAKA" method, perfectly adapted in this case, supports strongly the value of 1.05 eV.

In this case the information coming from integral experiment was a validation of one method to derive the average spacing from a resolved resonance parameter set.

Example n° 2 : $^{241}$Am.

Both fission and capture cross sections were measured by integral experiments.

Fission cross section was obtained by the reaction rate measurement technics relatively to $^{235}$U. The experiments were carried out in critical facilities (MASURCA, ERMIN) fueled either with uranium (R3, RONA3) or plutonium [ZONAI, OP40 ($K=1$) OP 50 ($K=1$)] /5/. For these experiments, $r$ parameter was ranging between 0.2 and 0.3 approximately.

.../...
The maximum of sensitivity was between 10 KeV and 1.3 KeV. The neutron spectra were known with an accuracy similar to the one of "PROFIL" experiment: a few percents.

The evaluation /6/ was performed with deformed optical model and coupled channel calculations for neutron channel penetrabilities, BRINK-AXEL type profil fonction for gamma channel penetrabilities and a formalism similar to the one given by Lynn for fission channel penetrabilities in the approximation of the complete damping for the coupling between class I and class II states.

Concerning fission, above the resolved resonances region, the evaluation was based on the experimental values of KNITTER and BUDTZ-JORGENSEN /7/. The fit was excellent everywhere, except in the range 20 KeV-200 KeV, a region a major sensitivity for the referred integral experiments. In that region KNITTER'S and BUDTZ-JORGENSEN'S data (much lower than the other ones) are twice as large as the theoretical values. The adjustment in 6 macro-groups of the CARNAVAL scheme performed according to the statistical consistent method (LAGRANGE'S multipliers method) and using an improved version of AMARA code /15/ showed:

1°) ZONA 3 experiment was not consistent with the others and was disregarded.

2°) The consistency of the integral experiments with the evaluated curve was excellent in each macrogroup where the sensitivity was significant. In particular, in the macrogroup concerned by the discrepancy between microscopic and theoretical
values mentionned above, the agreement was within 1 %.

Some months after, experiments were repeated in the energy range of interest by HAGE, KAFPPELER and WISSHAK /8/ in different geometrical conditions, whose results confirmed, within their error bars both evaluation and adjustment [fig. (1)].

From such a nice situation we deduce that:

- the theoretical model for fission process is certainly correct,
- as a result of the adjustment procedure and method a good accuracy (± 3 %, 1σ) is affected to the evaluated fission curve.

Similar operation was made for capture cross section which was measured from integral side by irradiation in OSIRIS, RAPSODIE and PHENIX. From microscopic side there are 4 sets of data:

- 2 absorption measurements by WESTON /9/ and GAYTHER /10/.
- 2 capture measurements, in different geometrical conditions by WISSHAK /11/, /12/. WISSHAK'S and GAYTHER'S data are in good agreement, while WESTON'S are lower.

An adjustement, statistical consistent manner, shows acceptable consistency of the integral data with the microscopic data of GAYTHER and WISSAK with a systematic trend for the first ones to be higher by 7 % [fig. (2)].
As a conclusion for $^{241}$Am the integral experiments gave helpful informations to solve the discrepancy between microscopic data, a discrepancy which is very difficult to be solved on the basis of arguments of experimental technics.

Example n° 3 $^{237}$Np

We will concentrate only on the (n, 2n) cross section which is in the NEANDC-INDC "discrepancy" list.

The status of knowledge on this reaction is the following one:

- there are a few microscopic data, centred on 14.5 MeV with the exception of one datum at 9.6 MeV. All were obtained by activation method,

- there is an unique published integral datum which also results from an activation method and which needs for validation a confirmation by microscopic data.

Apparently microscopic and integral data are discrepant by a factor about 2.

At this step some comments have to be made: the (n, 2n) cross section on $^{237}$Np leads to $^{236}$Pu according to the following schema, by passing by $^{236}$Np in its shortest lived state.
The activation method consists of measuring the quantity of $^{236}_{\text{Pu}}$ formed. LANDRUM et al /13/ measured at 14.5 MeV the isomeric ratio and found a value of 0.75 for the part of (n, 2n) cross section going to the shortest lived state of $^{236}_{\text{Np}}$.

If the branching ratio for $\beta^-$ decay to $^{236}_{\text{Pu}}$ is 0.48, the ratio by which the total (n, 2n) cross section should be multiplied to obtain at 14.5 MeV the $^{236}_{\text{Pu}}$ formation cross section is $R = 0.75 \times 0.48 = 1/2.76$.

The apparent discrepancy between integral and microscopic data results from the assumption that $R$ is not energy dependent.

Actually $R$ is energy dependent as it could be showed by heavy calculation involving nuclei $^{238}_{\text{Np}}$, $^{237}_{\text{Np}}$, $^{236}_{\text{Np}}$, their level
densities and their respective excitation functions in the \((n, 2n)\) process. Exact calculations are not possible since informations are lacking for \(^{236}\text{Np}\).

Realistic hypotheses show that the consistency between integral and microscopic informations possibly exist: A recent evaluation \(^{14}\) of \((n, 2n)\) cross section was performed for which the consistency with total fission cross section was guaranted (Fig. 3, 4) through model calculations. Using the scarce experimental data it has been possible to determine 3 values for \(R\) (E) (fig. 5) from which an "academic" \(R\) (E) curve was deduced. This one allows the wanted consistency, but should be supported by additional experimental data, of microscopic nature indeed.

This example shows the obvious necessary complementarity of integral and microscopic informations.

**CONCLUSION**

With more time it would have been possible to exhibit more examples \((^{10}\text{B, }^{239}\text{Pu, }^{235}\text{U}...\)).

We are convinced that neutron nuclear data libraries for reactor application filled with data resulting from the triple consistency of microscopic data, nuclear model indications and integral data have to be preferred because:

.../...
- closer to the "truth",
- more accurate,
- they can be a common basis for different formulaires.
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$(n, 2n)$ reaction

$^{237}$Np $(n, 2n) ^{236}$Np
$(n, 3n) ^{235}$Np

With pre-equilibrium

Without pre-equilibrium

Production of $^{236}$Pu

$\Delta$ PERKIN
$\circ$ LINDEKE
$+$ LANDRUM
$\circ$ NISHI (1975)

ENDF/BV

$arbitrary$ 1975

$\rightarrow n, 3n$

$\rightarrow 0.2$

$\rightarrow 0.1$

$\rightarrow 0.0$

$\rightarrow 0.4$

$\rightarrow 0.5$

$\rightarrow 0.6$

$\rightarrow 0.7$

$\rightarrow 0.8$

barn

MeV

Energy

FIG. 3
\[ R(E) = \frac{\sigma^{236*}_{Pu}}{\sigma_{n,2n}} \]

\[ {}^{237}N_p(n,2n) {}^{236}N_p^* \rightarrow {}^{236}Pu \]

Energy (MeV) Spin

\[ {}^{236}N_p \]

- Fundamental 0 6^+ \, 1^- \, 6^-
- Excited 0.0446 1^- \, 6^- \downarrow \quad R_1 \, R_2

**Deduced from NISHI (1976)**

\[ R_4(E) = R_2(E) \]

\[ R_2(E) \quad \text{LANDRUM} \]

\[ NISHI \text{ (revised 1980)} \]

\[ R_1(E) \]

**FIG. 5**
STATUS OF THE DATA BASE AND SOME INTEGRAL-DIFFERENTIAL
COMPARISONS FOR NON-FISSION THRESHOLD DOSIMETRY REACTIONS
FROM ENDF/B-V

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Prepared as a contribution to the Topical Conference on
"Convergence of Integral and Microscopic Nuclear Data",
22nd Meeting of the NEANDC, Aix-en-Provence, France
5 - 10 April 1981
INTRODUCTION

It is a pleasure for me to be here at the NEANDC Topical Conference on Integral-Differential Convergence. The question of convergence would, of course, never arise if we had completely accurate nuclear data and suitable refined techniques for utilizing these data in applications. The fact that integral-differential convergence is a perennial topic for discussion reminds us that the achievement of this goal continues to evade us in most areas of nuclear technology.

Today I would like to address the field of dosimetry. This is a rather broad topic and one where the question of integral differential convergence is the central issue. I will only touch upon part of the field since time is limited. My talk will be a survey and overview designed to give you a qualitative feel for the current state of affairs, and it will deal mostly with the work of others. A handout which reproduces the transparencies for your permanent record is available.

Please help yourself and devote your attention to the key points and conclusions which I will indicate to you as we go along.
SCOPE OF PRESENTATION

The objective of this presentation is to briefly survey the status of the database of some integral-differential comparisons for non-fission threshold reactions from the ENDF/B-V Dosimetry File. Several reactions have been excluded from the survey because the data testing of these reactions for ENDF/B-V has been delayed by problems encountered in dealing with resonances in the calculation. Other reactions of potential interest to the dosimetry community (e.g., some to be included in the International Reactor Dosimetry File) are also excluded because of the limited time allotted for this presentation, and because information of this nature could not be obtained by the author in the short time available to prepare this survey. The emphasis of this presentation is directed toward fission reactor application.
DOSIMETRY REACTIONS

SURVEYED

$^{27}\text{Al}(n,p)^{27}\text{Mg}$

$^{27}\text{Al}(n,\alpha)^{24}\text{Na}$

$^{55}\text{Mn}(n,2n)^{54}\text{Mn}$

$^{59}\text{Co}(n,2n)^{58}\text{Co}$

$^{59}\text{Co}(n,\alpha)^{56}\text{Mn}$

$^{46}\text{Ti}(n,p)^{46}\text{Sc}$

$^{47}\text{Ti}(n,np)^{46}\text{Sc}$

$^{47}\text{Ti}(n,p)^{47}\text{Sc}$

$^{48}\text{Ti}(n,np)^{47}\text{Sc}$

$^{48}\text{Ti}(n,p)^{48}\text{Sc}$

$^{54}\text{Fe}(n,p)^{54}\text{Mn}$

$^{56}\text{Fe}(n,p)^{56}\text{Mn}$

$^{58}\text{Ni}(n,2n)^{57}\text{Ni}$

$^{58}\text{Ni}(n,p)^{58}\text{Co}$

$^{60}\text{Ni}(n,p)^{60}\text{Co}$

$^{63}\text{Cu}(n,\alpha)^{60}\text{Co}$

$^{65}\text{Cu}(n,2n)^{64}\text{Cu}$

$^{115}\text{In}(n,n')^{115m}\text{In}$

$^{127}\text{I}(n,2n)^{126}\text{I}$
<table>
<thead>
<tr>
<th>Isotope</th>
<th>t 1/2 (accuracy)</th>
<th>$E_\gamma$</th>
<th>$I_\gamma$</th>
<th>Source</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mg - 27</td>
<td>9.462 m (0.1 %)</td>
<td>0.8438</td>
<td>0.717 (0.7 %)</td>
<td>H + G</td>
<td>Accuracy probably adequate for dosimetry. Improved knowledge of $\gamma$ branch would be useful. Convenient for Ge(Li) measurements.</td>
</tr>
<tr>
<td>Na - 24</td>
<td>15.00 h (0.1 %)</td>
<td>1.3686</td>
<td>0.99993 (0.002 %)</td>
<td>H + G</td>
<td>Accuracy acceptable for dosimetry. Convenient for Ge(Li) measurements.</td>
</tr>
<tr>
<td>Mn - 54</td>
<td>312.5 d (0.2 %)</td>
<td>0.8348</td>
<td>0.9997 (0.02 %)</td>
<td>H + G</td>
<td>Accuracy acceptable for dosimetry. Convenient for Ge(Li) measurements.</td>
</tr>
<tr>
<td>Co - 58</td>
<td>70.85 d (0.2 %)</td>
<td>0.8108</td>
<td>0.9944 (0.01 %)</td>
<td>H + G</td>
<td>Accuracy acceptable for dosimetry. Convenient for Ge(Li) measurements.</td>
</tr>
<tr>
<td>Mn - 56</td>
<td>2.5785 h (0.02 %)</td>
<td>0.8468</td>
<td>0.9887 (0.04 %)</td>
<td>L</td>
<td>Accuracy acceptable for dosimetry. Convenient for Ge(Li) measurements.</td>
</tr>
<tr>
<td>Sc - 46</td>
<td>83.9 d (0.4 %)</td>
<td>0.8893</td>
<td>0.99984 (0.002 %)</td>
<td>H + G</td>
<td>Accuracy probably adequate for dosimetry. Improved knowledge of half life would be useful. Convenient for Ge(Li) measurements.</td>
</tr>
<tr>
<td>Sc - 47</td>
<td>3.40 d (0.9 %)</td>
<td>0.1594</td>
<td>0.690 (3.6 %)</td>
<td>H + G</td>
<td>Accuracy inadequate for dosimetry. Improved knowledge of $\gamma$ branch and half life required. Special care required for $\gamma$ ray measurements.</td>
</tr>
<tr>
<td>Sc - 48</td>
<td>43.8 h (0.2 %)</td>
<td>0.9834</td>
<td>0.99987 (0.002 %)</td>
<td>H + G</td>
<td>Accuracy acceptable for dosimetry. Convenient for Ge(Li) measurements.</td>
</tr>
<tr>
<td>Ni - 57</td>
<td>35.99 h (0.3 %)</td>
<td>1.3776</td>
<td>0.776 (1 %)</td>
<td>L</td>
<td>Accuracy marginal for dosimetry. Improved knowledge of $\gamma$ branch required. Convenient for Ge(Li) measurements.</td>
</tr>
<tr>
<td>Co - 60</td>
<td>5.271 y (0.04 %)</td>
<td>1.3325</td>
<td>0.99980 (0.003 %)</td>
<td>H + G</td>
<td>Accuracy acceptable for dosimetry. Convenient for Ge(Li) measurements.</td>
</tr>
<tr>
<td>Cu $\gamma$ 64</td>
<td>12.702 h (0.03 %)</td>
<td>0.511 ($\beta^+$)</td>
<td>0.38 (5.3 %)</td>
<td>H + G</td>
<td>Accuracy inadequate for dosimetry. Improved knowledge of $\gamma$ branch required. Convenient for Ge(Li) measurements.</td>
</tr>
<tr>
<td>In - 115m</td>
<td>4.486 h (0.09 %)</td>
<td>0.3362</td>
<td>0.459 (0.2 %)</td>
<td>H + G</td>
<td>Acceptable accuracy for dosimetry. Special care required in $\gamma$ ray measurement.</td>
</tr>
<tr>
<td>I - 126</td>
<td>13.02 d (0.5 %)</td>
<td>0.3886</td>
<td>0.35 (8.6 %)</td>
<td>L</td>
<td>Accuracy inadequate for dosimetry. Improved knowledge of $\gamma$ branch and half life required. Special care required in $\gamma$ ray measurements.</td>
</tr>
</tbody>
</table>

$a$: $\gamma$ energy in MeV  
$b$: $\gamma$ branch given as a fraction of total decays  
L: Lederer et al., Table of the Isotopes 7th Ed., John Wiley and Sons (1978)
There are large differences in various recent representations for the U-235 thermal-neutron-fission neutron spectrum—an important benchmark field. Significant differences in representations for the Cf-252 spontaneous-fission neutron field and other benchmarks used for integral-differential studies are also observed. These differences are most pronounced above 10 MeV. Integral-differential comparisons for high-threshold \((n,2n)\) reactions are probably meaningless for differential data testing, but may be useful to test the spectral representations.

**Conclusion:** Integral-differential comparisons will continue to be of questionable usefulness until the benchmark fields are better known.
### AVAILABILITY OF INTEGRAL DATA
FOR NON-FISSION REACTIONS FROM
ENDF/B-V DOSIMETRY FILE

<table>
<thead>
<tr>
<th>Reaction</th>
<th>Standard Fields</th>
<th>Some Fast - Reactor</th>
<th>Benchmarks(^a)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Cf-252 U-235</td>
<td>(\Sigma) Big-10</td>
<td>CFRMF</td>
</tr>
<tr>
<td>27Al(n,p)(^{27})Mg</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>27Al(n,(\alpha))(^{24})Na</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>55Mn(n,2n)(^{54})Mn</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>59Co(n,2n)(^{58})Co</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>59Co(n,(\alpha))(^{56})Mn</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>46Ti(n,p)(^{46})Sc(^b)</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>47Ti(n,np)(^{46})C(^b)</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>47Ti(n,p)(^{47})Sc(^b)</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>48Ti(n,np)(^{47})Sc(^b)</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>48Ti(n,p)(^{48})Sc</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>54Fe(n,p)(^{54})Mn</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>56Fe(n,p)(^{56})Mn</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>58Ni(n,2n)(^{57})Ni</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>58Ni(n,p)(^{58})Co</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>60Ni(n,p)(^{60})Co</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>63Cu(n,(\alpha))(^{60})Co</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>65Cu(n,2n)(^{64})Cu</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>115In(n,n')(^{115})In</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
<tr>
<td>127I(n,2n)(^{126})I</td>
<td>*</td>
<td>*</td>
<td>*</td>
</tr>
</tbody>
</table>

\(^{a}\)Response is similar to U-235 standard field for endoergic reactions.

\(^{b}\)Activation measurements detect Ti(n,x)\(^{46}\)Sc and Ti(n,x)\(^{47}\)Sc

**CONCLUSION:** There are some deficiencies in the integral data base for dosimetry, but evaluation of existing results should be given priority over new measurements.
SOME RECENT EVALUATIONS OF INTEGRAL DOSIMETRY DATA

1976: A. Fabry et al., IAEA Conf. on Reactor Dosimetry, Vienna, IAEA-208(1978)
Review of integral data for U-235, Cf-252 and several reactor benchmarks.

1977: A. Fabry et al., 2nd ASTM-EURATOM Symposium on Reactor Dosimetry,
Palo Alto, NUREG/CP-0004(vol. 3)
Updates 1976 work by A. Fabry et al., but less detailed than earlier paper. More emphasis is given to benchmarks than to integral data evaluation.

1978: W. Mannhart, RSIC Seminar-Workshop on Sensitivity and Uncertainty Analysis,
Oak Ridge, ORNL/RSC-42(1979)
Evaluation of some Cf-252 integral data.

1979: W. Mannhart and F. Perey, 3rd ASTM-EURATOM Symposium on Reactor Dosimetry,
Upgrading of 1978 work of Mannhart using covariance method.

N° 594 (1980)
Careful evaluation of $^{63}$Cu(n,$\alpha$)$^{60}$Co integral data in U-235 benchmark.

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This list is surely not complete. Work in this field is often reported in documents with limited distribution beyond the Dosimetry community.

CONCLUSION: Less systematic evaluation effort has been devoted to integral data than to differential data for dosimetry.
### SOME INTEGRAL - DIFFERENTIAL COMPARISONS FOR NON-FISSION THRESHOLD REACTIONS

**FROM ENDF/B-V DOSIMETRY FILE**

<table>
<thead>
<tr>
<th>Reaction</th>
<th>(&lt;a&gt;) Exp.</th>
<th>(&lt;a&gt;) Calc.</th>
<th>Diff.</th>
<th>(&lt;a&gt;) Exp.</th>
<th>(&lt;a&gt;) Calc.</th>
<th>Diff.</th>
</tr>
</thead>
<tbody>
<tr>
<td>U - 235</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>27(^{Al}(n,p))(^{27}Ni)</td>
<td>3.86(± 6.5 (\sigma))(^a)</td>
<td>4.26(^b)</td>
<td>- 10%</td>
<td>5.3(± 9.35)(^c)</td>
<td>5.27(^d)</td>
<td>- 3.3%</td>
</tr>
<tr>
<td>27(^{Al}(n,\alpha))(^{24}Na)</td>
<td>0.705(± 5.7 (\sigma))(^g)</td>
<td>0.720(^a)</td>
<td>- 2.1%</td>
<td>1.019(± 2.4 (\sigma))(^c)</td>
<td>1.16(^b)</td>
<td>- 14%</td>
</tr>
<tr>
<td>54(^{Mn}(n,2n))(^{54}Mn)</td>
<td>0.244(± 6.1 (\sigma))(^g)</td>
<td>0.202(^a)</td>
<td>+ 17%</td>
<td>0.58(± 10 (\sigma))(^e)</td>
<td>0.554(^b)</td>
<td>+ 4.5%</td>
</tr>
<tr>
<td>58(^{Co}(n,2n))(^{58}Co)</td>
<td>0.227(± 4.8 (\sigma))(^j)</td>
<td>0.183(^a)</td>
<td>+ 19%</td>
<td>0.57(± 11 (\sigma))(^e)</td>
<td>0.512(^b)</td>
<td>+ 10%</td>
</tr>
<tr>
<td>58(^{Co}(n,\alpha))(^{58}Ni)</td>
<td>0.143(± 7 (\sigma))(^e)</td>
<td>0.150(^a)</td>
<td>- 4.3%</td>
<td>0.20(± 5 (\sigma))(^f)</td>
<td>0.229(^b)</td>
<td>- 18%</td>
</tr>
<tr>
<td>46(^{Ti}(n,p))(^{46}Sc)</td>
<td>11.90(± 6.4 (\sigma))(^c)</td>
<td>11.18(^a)</td>
<td>+ 5.3%</td>
<td>14.12(± 2.6 (\sigma))(^c)</td>
<td>13.81(^b)</td>
<td>+ 2.2%</td>
</tr>
<tr>
<td>47(^{Ti}(n,p))(^{47}Sc)</td>
<td>19.0(± 7.4 (\sigma))(^c)</td>
<td>22.48(^a)</td>
<td>- 18%</td>
<td>19.27(± 2.3 (\sigma))(^c)</td>
<td>24.22(^b)</td>
<td>- 26%</td>
</tr>
<tr>
<td>49(^{Ti}(n,p))(^{49}Sc)</td>
<td>11.90(± 6.4 (\sigma))(^c)</td>
<td>11.18(^a)</td>
<td>+ 5.3%</td>
<td>14.12(± 2.6 (\sigma))(^c)</td>
<td>13.81(^b)</td>
<td>+ 2.2%</td>
</tr>
<tr>
<td>54(^{Fe}(n,p))(^{54}Mn)</td>
<td>79.7(± 6.1 (\sigma))(^e)</td>
<td>81.08(^a)</td>
<td>- 1.7%</td>
<td>86.38(± 2.6 (\sigma))(^c)</td>
<td>88.3(^d)</td>
<td>- 2.5%</td>
</tr>
<tr>
<td>58(^{Fe}(n,p))(^{58}Mn)</td>
<td>1.035(± 7.2 (\sigma))(^d)</td>
<td>1.04(^a)</td>
<td>- 0.5%</td>
<td>1.468(± 2.3 (\sigma))(^c)</td>
<td>1.5(^d)</td>
<td>- 2.2%</td>
</tr>
<tr>
<td>58(^{Ni}(n,2n))(^{58}Ni)</td>
<td>0.0036(± 6.7 (\sigma))(^j)</td>
<td>0.003(^a)</td>
<td>+ 17%</td>
<td>No data</td>
<td>No data</td>
<td>--</td>
</tr>
<tr>
<td>60(^{Ni}(n,p))(^{60}Co)</td>
<td>108.5(± 5 (\sigma))(^g)</td>
<td>105.2(^a)</td>
<td>+ 3.1%</td>
<td>115.4(± 1.3 (\sigma))(^c)</td>
<td>114.5(^d)</td>
<td>+ 0.7%</td>
</tr>
<tr>
<td>60(^{Cu}(n,p))(^{60}Cu)</td>
<td>No data</td>
<td>2.61(^a)</td>
<td>--</td>
<td>No data</td>
<td>No data</td>
<td>--</td>
</tr>
<tr>
<td>65(^{Cu}(n,\alpha))(^{65}Cu)</td>
<td>0.524(± 2.8 (\sigma))(^h)</td>
<td>0.507(± 9.7 (\sigma))(^h)</td>
<td>+ 5.1%</td>
<td>0.709(± 2.4 (\sigma))(^f)</td>
<td>0.678(± 6.5(^f))</td>
<td>+ 4.7%</td>
</tr>
<tr>
<td>115(^{In}(n,\alpha))(^{115}In)</td>
<td>189(± 4.2 (\sigma))(^f)</td>
<td>179.5(^a)</td>
<td>+ 5%</td>
<td>198.3(± 2.6 (\sigma))(^c)</td>
<td>185.6(^g)</td>
<td>+ 6.4%</td>
</tr>
</tbody>
</table>

**CONCLUSION**: Satisfactory convergence of differential and integral results is seen for only ~ 40% of the reactions in this set. For the rest, there are data needs, uncertainties and clear discrepancies. This is a disgraceful situation considering the effort which has apparently been devoted to this field.
COMMENTS ON SPECIFIC PROBLEM REACTIONS

$^{27}\text{Al}(n,p)^{27}\text{Mg}$

There is a considerable base of both differential and integral experimental data for this reaction. Precise definition of the differential cross section is hampered by resonance structure in the MeV energy range, but this should have little effect on integral-differential comparisons. Agreement of differential and integral results is marginal, so a thorough examination of errors is warranted. ENDF/B-V is based on a 1972 evaluation. A more recent evaluation by Asami (IAEA Dosimetry Conf., Vienna, 1976) predicts lower cross sections than ENDF/B-V in the important response range (5 - 10 MeV), and, this would lead to improved integral-differential agreement.

CONCLUSION: The data base and evaluations should be examined before undertaking any new measurements.

$^{27}\text{Al}(n,\alpha)^{24}\text{Na}$

The rather large discrepancy for the $^{252}\text{Cf}$ comparison is surprising in view of the extensive differential and integral experimental data available for this reaction. ENDF/B-V is based on a 1972 evaluation. More recent evaluations by Asami (IAEA Dosimetry Conf., Vienna, 1976) and by Tageson and Vonach (Vienna, private communication, 1981) predict lower cross sections in the important response range (7 - 13 MeV), and would lead to some improvement in the integral-differential agreement. The calculated spectrum average cross section for $^{252}\text{Cf}$ reported by Magurno (3rd ASTM-EURATOM Dosimetry Conf., Ispra, 1979) is 10% larger than the calculated value provided by Zijp (ECN-80-143, 1980). Similar differences are noticed for calculated values pertaining to other reactions considered by these two authors. This illustrates the necessity for settling the matter of benchmark field representations.

CONCLUSION: The data base, evaluations and spectrum-average calculations should be examined before undertaking any new measurements.
$^{55}\text{Mn}(n,2n)^{54}\text{Mn}$

The differential data exhibit considerable scatter in the important response range (11 - 16 MeV). This is a difficult range for differential measurements. A new integral measurement by Kobayashi et al. (NEANDC(J)67U42, 1980) would appear to eliminate the discrepancy for the $^{235}\text{U}$ spectrum (the value $0.202 \pm 0.010$ mb is reported). Magurno and Zijp differ in their calculated values for $^{252}\text{Cf}$ in this reaction - see $^{27}\text{Al}(n,\alpha)^{24}\text{Mg}$ discussion. Integral-differential comparisons will remain uncertain until the matter of the fission spectra representations above 8 MeV is settled.

CONCLUSION: The new integral result is useful, but uncertainties in the fission spectra representations and in the differential data appear to be the dominant issues.

$^{59}\text{Co}(n,2n)^{58}\text{Co}$

The evaluation for ENDF/B-V appears to have been carefully done. A relatively extensive differential data base, including the liquid scintillator tank work of Frehaut and Mosinski (CEAR-4627,1974) was considered in the analysis. Model calculations by Uhl (Vienna) were used to extrapolate and interpolate the experimental data where required. The integral data base is sparse, although a recent $^{235}\text{U}$ measurement by Kobayashi et al. (NEANDC(J)67U42, 1980) serves to accentuate the existing discrepancy.

CONCLUSION: Dominant issues are probably the limited integral data base and uncertainties in the fission spectra representations above 8 MeV.

$^{59}\text{Co}(n,\alpha)^{56}\text{Mn}$

ENDF/B-V is based on a 1972 evaluation. The evaluator for ENDF/B-V stated that new differential data since 1972 did not alter the situation. Magurno and Zijp disagree in calculated value for $^{252}\text{Cf}$ - see $^{27}\text{Al}(n,\alpha)^{24}\text{Na}$ discussion. Integral data are available, but not extensive.

CONCLUSION: The matter of calculating the spectrum average cross section needs to be examined. Additional integral data would be useful.
\[ {^{47}\text{Ti}(n,p)^{47}\text{Sc}} + {^{48}\text{Ti}(n,np)^{47}\text{Sc}} \]

The (n,np) reaction is not an important factor for fission reactors. The integral data base is extensive and seems to point toward a possible normalization problem in the differential data used in the ENDF/B-V evaluation of \( {^{47}\text{Ti}(n,p)^{47}\text{Sc}} \). Uncertainties in the \( {^{47}\text{Sc}} \) decay scheme may be a factor in comparisons with other reactions.

CONCLUSION: The experimental data base needs to be examined to see if a normalization problem exists. Possibly, new differential measurements should be undertaken with special attention paid to the experimental procedure used to measure \( {^{47}\text{Sc}} \) activity.

\[ {^{58}\text{Ni}(n,2n)^{57}\text{Ni}} \]

The ENDF/B-V evaluation is based on experimental data, however the data are very uncertain near threshold and no model calculations were performed to extrapolate to threshold. Recent integral result of Kobayashi et al. (NEANDC(J)67U42, 1980) reduces disagreement between integral and differential data to \( \sim 17\% \) for \( {^{235}\text{U}} \). Magurno and Zijp differ on calculated \( {^{252}\text{Cf}} \) result - see discussion for \( {^{27}\text{Al}(n,a)^{24}\text{Na}} \). No experimental integral data are available for a comparison.

CONCLUSION: An improved differential evaluation aided by model calculations is warranted. Also, more integral data is required before meaningful integral-differential comparisons can be made.

\[ {^{60}\text{Ni}(n,p)^{60}\text{Co}} \]

Knowledge of differential cross section is limited essentially to the work of Paulsen (Nukleonik 10, 91, 1967), and the shape of the excitation function in the 10 - 12 MeV range is odd. Integral data are essentially non-existent.

CONCLUSION: This is a relatively undeveloped reaction for dosimetry. Integral-differential comparisons should be deferred until more experimental data - especially integral data - become available.
The important region near threshold is defined only by the differential data of Santry et al. (Can. J. Phys. 44, 1183, 1965). The evaluation for ENDF/B-V appears to have been thoughtfully executed. Integral data are apparently non-existent.

CONCLUSION: Integral-differential comparisons must be deferred until more integral data become available.

The uncertainties in the decay scheme for $^{126}$I are clearly a problem. No details are available on the ENDF/B-V evaluation, but a brief survey of the differential data base indicates that very few data are available to define the cross section so the evaluation must be very speculative. There are some $^{235}$U integral data available. The recent result by Kobayashi et al. (NEANDC(J)67U42, 1980) agrees with the older evaluated value from Fabry et al. given in the main table. No $^{252}$Cf data are available.

CONCLUSION: The differential and integral data bases are too sparse to permit meaningful evaluations and integral-differential comparisons.
SOME GENERAL COMMENTS

1. Integral-differential comparisons suffer from a lack of standardization of criteria for the comparisons. For example, error information on calculated quantities is often overlooked, and covariances in the errors have only recently received some attention.

2. Most of the improvement which has been observed in this field has come about as a result of recent differential measurements rather than from integral investigations. This point has not received adequate recognition by the Dosimetry community, judging from the proceedings of several recent conferences.

3. Arbitrary renormalizations of either integral or differential cross sections without identification of the origin of the discrepancies is unjustified for the reactions considered in this paper. The discrepancies should be eliminated by scientifically legitimate means.

4. With the possible exception of reactions which are able to provide spectral sensitivity in the important range 1-100 keV, e.g. $^{93}\text{Nb}(n,n')^{93}\text{Mn}$, it does not seem reasonable to add new reactions to fission reactor dosimetry files until improvement in the status of the reactions already on the files is achieved. In fact, some reactions which provide redundant information might well be candidates for elimination from the files.