1. FILE 1. GENERAL INFORMATION

File 1 is the first part of any set of evaluated cross-section data for a material. Each material must have a File 1 that contains at least one section. This required section provides a brief documentation of how the data were evaluated and a directory that summarizes the files and sections contained in the material. In the case of fissionable materials, File 1 may contain up to four additional sections giving fission neutron yields and energy release information. Each section has been assigned an MT number (see below), and the sections are arranged in order of increasing MT number. A section always starts with a HEAD record and ends with a SEND record. The end of File 1 (and all other files) is indicated by a FEND record. These record types are defined in detail in Section 0.7.

1.1. Descriptive Data and Directory (MT=451)

This section is always the first section of any material and has two parts:

1) a brief documentation of the cross-section data, and
2) a directory of the files and sections used for this material.

In the first part, a brief description of the evaluated data set is given. This information should include the significant experimental results used to obtain the evaluated data, descriptions of any nuclear models used, a clear specification of all the MT numbers defined to identify reactions, the history of the evaluation, and references. The descriptive information is given as a series of records, each record containing up to 66 characters.

The first three records of the descriptive information contain a standardized presentation of information on the material, projectile, evaluators, and modification status. The following quantities are defined for MF=1, MT=451:

ZA,AWR Standard material charge and mass parameters.

LRP Indicates whether resolved and/or unresolved resonance parameters are given in File 2:
LRP=−1, no File 2 is given (not allowed for incident neutrons);
LRP=0, no resonance parameter data are given, but a File 2 is present containing the effective scattering radius;
LRP=1, resolved and/or unresolved parameter data are given in File 2 and cross sections computed from them must be added[1] to background cross sections given in File 3;
LRP=2, parameters are given in File 2, but cross sections derived from them are not to be added to the cross sections in File 3. The option LRP=2 is to be used for derived files only.

LFI Indicates whether this material fissions:
LFI=0, this material does not fission;
LFI=1, this material fissions.

NLIB Library identifier: NLIB= 0 for ENDF/B.
Additional values have been assigned to identify other libraries using ENDF format.

[1] In the unresolved region, it is also possible to compute a self-shielding factor from File 2 and multiply it by a complete unshielded cross section in File 3.

NMOD Modification number for this material:
NMOD=0, evaluation converted from a previous version;
NMOD=1, new or revised evaluation for the current library version;
NMOD≥2, for successive modifications.

ELIS Excitation energy of the target nucleus relative to 0.0 for the ground state.

STA Target stability flag:
STA=0, stable nucleus;
STA=1 unstable nucleus. If the target is unstable, radioactive decay data should be given in the decay data sub-library (NSUB=4).

LIS State number of the target nucleus. The ground state is indicated by LIS=0.

LISO Isomeric state number. The ground state is indicated by LISO=0. LIS is greater than or equal to LISO.

NFOR Library format.
NFOR=6 for all libraries prepared according to the specifications given in this manual.

AWI Mass of the projectile in neutron units. For incident photons or decay data sub-libraries, use AWI=0.0.

EMAX Upper limit of energy range for evaluation.

LREL Release number.

NSUB Sub-library number. See Section 0.4 for a description of sub-libraries.

NVER Library version number; for example, NVER=6 for version VI.

TEMP Target temperature (Kelvin) for data that have been generated by Doppler broadening. For derived data only; use TEMP=0.0 for all primary evaluations.

LDRV Special derived material flag that distinguishes between different evaluations with the same material keys (*i.e.*, MAT, NMOD, NSUB):
LDRV=0, primary evaluation:
LDRV≥1, special derived evaluation (for example, a dosimetry evaluation using sections (MT) extracted from the primary evaluation).

NWD Number of records used to describe the data set for this material. Each record can contain up to 66 characters.

NXC Number of records in the directory for this material. Each section (MT) in the material has a corresponding line in the directory that contains MF, MT, NC, and MOD. NC is a count of the number of records in the section (not including SEND), and MOD is the modification flag (see below).

ZSYNAM Character representation of the material charge, chemical symbol, atomic mass number, and metastable state in the form Z-cc-AM with
Z, right justified in col. 1 to 3,
- (hyphen) in col. 4,
two-character chemical name left justified in col. 5 and 6,
- (hyphen) in col. 7,
A, right justified in col. 8 to 10 or blank,
M for the indication of a metastable state in col. 11,
for example, 94-PU-239, 1-H - 2, *etc*.

ALAB Mnemonic for the originating laboratory(s) left adjusted in col. 12-22.

EDATE Date of evaluation given in the form "EVAL-DEC74" in col. 23-32.

AUTH Author(s) name(s) left adjusted in col. 34-66.

REF Primary reference for the evaluation left adjusted in col. 2-22.

DDATE Original distribution date given in the form "DIST-DEC74" in col. 23-32.

RDATE Date and number of the last revision to this evaluation in col. 34-43 in the form "REV2-DEC74", where "2" is the revision number IREV; IREV is computer retrievable.

ENDATE Master File entry date in the form *yyyymmdd* right adjusted in col. 56-63. The Master File entry date will be assigned by NNDC for ENDF/B-VI.

HSUB Identifier for the library contained on three successive records.
The first record contains four dashes starting in col. 1, directly followed by the library type (NLIB) and version (NVER). For example,
"---- ENDF/B-VI", followed by MATERIAL XXXX starting in col. 23 where XXXX is the MAT number, and REVISION 2 (starting in col. 45 only if required) where "2" is the revision number IREV.

The second record contains five dashes starting in col. 1, and followed by the sub-library identifier (see Table 0.1). For example,
"----- DECAY DATA,"
"----- PHOTO-ATOMIC INTERACTION DATA," or
"----- INCIDENT NEUTRON DATA."

The third record contains six dashes starting in col. 1 and followed by ENDF-6 where "6" is the library format type (NFOR).

Note: the three HSUB records can be generated by a utility program.

MF_n MF of the n^{th} section.

MT_n MT of the n^{th} section.

NC_n Number of records in the nth section. This count does not include the SEND record.

MOD_n Modification indicator for the nth section. The value of MOD_n is equal to NMOD if the corresponding section was changed in this revision. MOD_n must always be less than or equal to NMOD.

1.1.1. Formats

The structure of this section is

```
[MAT, 1,451/    ZA,   AWR,   LRP,    LFI, NLIB, NMOD]HEAD
[MAT, 1,451/  ELIS,   STA,   LIS,   LISO,    0, NFOR]CONT
[MAT, 1,451/   AWI,  EMAX,  LREL,      0, NSUB, NVER]CONT
[MAT, 1,451/  TEMP,   0.0,  LDRV,      0,  NWD,  NXC]CONT
[MAT, 1,451/ZSYMAM,  ALAB, EDATE,   AUTH            ]TEXT
[MAT, 1,451/   REF, DDATE, RDATE, ENDATE            ]TEXT
[MAT, 1,451/  HSUB                                  ]TEXT
   -------------------------------------
   continue for the rest of the NWD descriptive records
   -------------------------------------
[MAT, 1,451/ blank, blank,   MF1,    MT1,  NC1,  MOD1]CONT
[MAT, 1,451/ blank, blank,   MF2,    MT2,  NC2,  MOD2]CONT
   -------------------------------------
   -------------------------------------
[MAT, 1,451/ blank, blank, MFNXC,  MTNXC,NCNXC,MODNXC]CONT
[MAT, 1,  0/   0.0,   0.0,     0,      0,    0,     0]SEND
```

1.1.2. Procedures

Note that the parameters NLIB, NVER, NSUB, MAT, NMOD, LDRV, and sometimes TEMP define a unique set of "keys" that identifies a particular evaluation or "material" in the ENDF system. These keys can be used to access materials in a formal data base management system if desired.

The flag LRP indicates whether resolved and/or unresolved resonance parameter data are to be found in File 2 (Resonance Parameters) and how these data are to be used with File 3 to compute the net cross section. For incident neutrons, every material will have a File 2. If LRP=0, the file contains only the effective scattering radius; the potential cross section corresponding to this scattering radius has already been included in the File 3 cross sections. If LRP=1, File 2 contains resolved and/or unresolved resonance parameters. Cross sections or self-shielding factors computed from these parameters are to be combined with any cross sections found in File 3 to obtain the correct net cross section. For other sub-libraries (decay data, incident photons, incident charged particles, fission product yields), File 2 can be omitted (use LRP=-1). A number of processing codes exist which reconstruct resonance-region cross sections from the parameters in File 2 and output the results in ENDF format. Such a code can set LRP=2 and copy the original File 2 to its output ENDF tape. Other processing codes using such a tape will know that resonance reconstruction has already been performed, but the codes will still have easy access to the resonance parameters if needed. The LRP=2 option is not allowed in primary evaluations.

The flag LFI indicates that this material fissions in the context of the present sub-library. In this case, a section specifying the total number of neutrons emitted per fission, $\bar{\nu}(E)$, *must* be given as MF=1, MT=452. Sections *may* also be given that specify the number of delayed neutrons per fission (MT=455) and the number of prompt neutrons per fission (MT=456), and that specify the components of energy release in fission (MT=458).

The flag LDRV indicates that this material was derived in some way from another evaluation; for example, it could represent an activation reaction extracted from a more complete evaluation, it could be part of a gas production library containing production cross sections computed from more fundamental reactions, it could represent a reconstructed library with resonance parameters expanded into detailed point-wise cross sections, and so on.

The data in the descriptive section must be given for every material. The first three records are used to construct titles for listings, plots, *etc*., and the format should be followed closely. The remaining records give a verbal description of the evaluated data set for the material. The description should mention the important experimental results upon which the recommended cross sections are based, the evaluation procedures and nuclear models used, a brief history and origin of the evaluation, important limitations of the data set, estimated uncertainties and covariances, references, and any other remarks that will assist the user in understanding the data. For incident neutron evaluations, the 2200 m/s cross sections contained in the data should be tabulated, along with the infinite dilution resonance integrals for capture and fission (if applicable). For charged-particle and high-energy reactions, the meaning of each MT should be carefully explained using the notation of Section 0.5.

1.2. Number of Neutrons per Fission, $\bar{\nu}$, (MT=452)

If the material fissions (LFI=1), then a section specifying the average *total* number of neutrons per fission, $\bar{\nu}$ (MT=452), must be given. This format applies to both particle induced

$$\bar{\nu}(E) = \sum_{n=1}^{NC} C_n E^{n-1} \tag{1.1}$$

and spontaneous fission, each in its designated sub-library. Values of $\bar{\nu}$ may be tabulated as a function of energy or coefficients provided for the following polynomial expansion of $\bar{\nu}$(E),

where $\bar{\nu}$(E) = the average total (prompt plus delayed) number of neutrons per fission produced by neutrons of incident energy E(eV),

C_n = the n^{th} coefficient, and

NC = the number of terms in the polynomial.

MT=452 for an energy-dependent neutron multiplicity cannot be represented by a polynomial expansion when MT=455 and MT=456 are utilized in the file.

1.2.1. Formats

The structure of this section depends on whether values of $\bar{\nu}$ (E) are tabulated as a function of energy or represented by a polynomial. The following quantities are defined:

LNU	Test that indicates what representation of $\bar{\nu}$ (E) has been used: LNU=1, polynomial representation has been used; LNU=2, tabulated representation.
NC	Count of the number of terms used in the polynomial expansion. (NC≤4).
C_n	Coefficients of the polynomial. There are NC coefficients given.
NR	Number of interpolation ranges used to tabulate values of $\bar{\nu}$(E). (See 0.6.2.)
NP	Total number of energy points used to tabulate $\bar{\nu}$(E).
E_{int}	Interpolation scheme (see 0.6.2 for details).
$\bar{\nu}$(E)	Average number of neutrons per fission.

If **LNU=1**, the structure of the section is

```
[MAT, 1,  452/  ZA, AWR,  0, LNU,  0,  0]HEAD (LNU=1)
[MAT, 1,  452/ 0.0, 0.0,  0,   0, NC,  0/ C1, C2, ...CNC]LIST
[MAT, 1,99999/ 0.0, 0.0,  0,   0,  0,  0]SEND
```

If **LNU=2**, the structure of the section is

```
[MAT, 1,  452/  ZA, AWR,  0, LNU,  0,  0]HEAD (LNU=2)
[MAT, 1,  452/ 0.0, 0.0,  0,   0, NR, NP/E_int / ν̄(E)]TAB1
[MAT, 1,99999/ 0.0, 0.0,  0,   0,  0,  0]SEND
```

1.2.2. Procedures

If a polynomial representation (LNU=1) has been used to specify $\bar{\nu}$ (E), this representation is valid over any range in which the fission cross section is specified (as given in Files 2 and 3). When using a polynomial to fit $\bar{\nu}$ (E), the fit shall be limited to a third-degree polynomial (NC≤4). If such a fit does not reproduce the recommended values of $\bar{\nu}$ (E), a tabulated form (LNU=2) should be used.

If tabulated values of $\bar{\nu}$ (E) are specified (LNU=2), then pairs of energy- $\bar{\nu}$ values are given. Values of $\bar{\nu}$ (E) should be given that cover any energy range in which the fission cross section is given in File 2 and/or File 3.

The values of $\bar{\nu}$ (E) given in this section are for the average total number of neutrons produced per fission event. When another section (MT=455) that specifies the delayed neutrons from fission is given, the average number of delayed neutrons per fission ($\bar{\nu}_d$) must be added to $\bar{\nu}_p$ given in section MT=456 and included in the value of $\bar{\nu}$(E) given in this section (MT=452). In this case, only LNU=2 is allowed for MT=452.

For spontaneous fission, the polynomial representation (LNU=1) is used with NC=1 and Cl = $\bar{\nu}_{total}$. There is no energy dependence.

1.3. Delayed Neutron Data, $\bar{\nu}_d$, (MT=455)

This section describes the delayed neutrons resulting from either particle induced or spontaneous fission. The average total number of delayed neutron precursors emitted per fission, $\bar{\nu}_d$, is given, along with the decay constants, λ_i, for each precursor family. The fraction of $\bar{\nu}_d$ generated for each family is given in File 5 (section 5 of this report). The energy distributions of the neutrons associated with each precursor family are also given in File 5.

For particle-induced fission, the total number of delayed neutrons is given as a function of energy in tabulated form (LNU=2). The energy dependence is specified by tabulating $\bar{\nu}_d$(E) at a series of neutron energies using the same format as for MT=452. For spontaneous fission LNU=1 is used with NC=1 and C_1= $\bar{\nu}_d$ as for MT=452.

The total number of delayed neutron precursors emitted per fission event, at incident energy E, is given in this file and is defined as the sum of the number of neutrons emitted for each of the precursor families,

$$\bar{\nu}_d = \sum_{i=1}^{NNF} \bar{\nu}(E),$$

where *NNF* is the number of precursor families. The fraction of the total, P_i(E), emitted for each family is given in File 5 (see section 5) and is defined as

$$P_i(E) = \frac{\bar{\nu}_i(E)}{\bar{\nu}_d(E)}$$

1.3.1. Formats

The following quantities are defined.

LNU	Test indicating which representation is used: LNU=1 means that polynomial expansion is used; LNU=2 means that a tabulated representation is used.
NC	Number of terms in the polynomial expansion. (NC≤4).
NR	Number of interpolation ranges used. (NR≤20).
NP	Total number of energy points used in the tabulation of $\nu(E)$
E_{int}	Interpolation scheme (See Section 0.6.2)
$\overline{\nu}_d$ (E)	Total average number of delayed neutrons formed per fission event.
NNF	Number of precursor families considered.
λ_i	Decay constant (sec^{-1}) for the i^{th} precursor.

The structure when values of $\overline{\nu}_d$ are tabulated (LNU=2) is:

```
[MAT, 1,455/  ZA, AWR,  0, LNU,   0,  0]HEAD   (LNU=2)
[MAT, 1,455/ 0.0, 0.0,  0,   0, NNF,  0/λ1,λ2,...λNNF]LIST
[MAT, 1,455/ 0.0, 0.0,  0,   0,  NR, NP/ Eint / ν̄d(E)]TAB1
[MAT, 1,  0/ 0.0, 0.0,  0,   0,   0,  0]SEND
```

If LNU=1 (spontaneous fission) the structure of the section is:

```
[MAT, 1,455/  ZA, AWR,  0, LNU,   0,  0]HEAD   (LNU=1)
[MAT, 1,455/ 0.0, 0.0,  0,   0, NNF,  0/λ1,λ2,...λNNF]LIST
[MAT, 1,455/ 0.0, 0.0,  0,   0,   1,  0/ ν̄d]LIST
[MAT, 1,  0/ 0.0, 0.0,  0,   0,   0,  0]SEND
```

1.3.2. Procedures

When tabulated values of $\overline{\nu}_d(E)$ are specified, as is required for particle-induced fission by Section 1.2., they should be given for the same energy range as that used to specify the fission cross section.

The probability of producing the precursors for each family and the energy distributions of neutrons produced by each precursor family are given in File 5 (section 5 of this report). It is extremely important that the same precursor families be given in File 5 as are given in File 1 (MT=455), and the ordering of the families should be the same in both files. It is recommended that the families be ordered by decreasing half-lives ($\lambda_1<\lambda_2<\ldots<\lambda_{NNF}$).

For spontaneous fission, the polynomial form (LNU=1) is used with only one term (NC=1, $C_1= \overline{\nu}_d$).

If MT=455 is used, then MT=456 must also be used as well as MT=452.

1.4. Number of Prompt Neutrons per Fission, $\overline{\nu}_p$, (MT=456)

If the material fissions (LFI=1), a section specifying the average number of prompt neutrons per fission, $\overline{\nu}_p$, (MT=456) can be given using formats identical to MT=452. For particle-induced fission, $\overline{\nu}_p$ is given as a function of energy. The prompt $\overline{\nu}$ for spontaneous fission can also be given using MT=456, but there is no energy dependence.

1.4.1. Formats

The following quantities are defined:

LNU	Indicates what representation of $\bar{\nu}_p(E)$ has been used: LNU=1, polynomial representation has been used; LNU=2, tabulated representation.
NC	Count of the number of terms used in the polynomial expansion. (NC≤4)
NR	Number of interpolation ranges used to tabulate values of $\bar{\nu}_p(E)$. (See 0.7.7.)
NP	Total number of energy points used to tabulate $\bar{\nu}_p(E)$.
E_{int}	Interpolation scheme (see section 0.6.2.)
$\bar{\nu}_p(E)$	Average number of prompt neutrons per fission.

If LNU=2, (tabulated values of $\bar{\nu}$), the structure of the section is:

```
[MAT, 1,456/  ZA, AWR,  0, LNU,  0,  0] HEAD       (LNU=2)
[MAT, 1,456/ 0.0, 0.0,  0,   0, NR, NP/Eint/ νp(E)]TAB1
[MAT, 1,  0/ 0.0, 0.0,  0,   0,  0,  0]SEND
```

If LNU=1 (spontaneous fission) the structure of the section is:

```
[MAT, 1,456/  ZA, AWR,  0, LNU,  0,  0]HEAD   (LNU=1)
[MAT, 1,456/ 0.0, 0.0,  0,   0,  1,  0/ νp]LIST
[MAT, 1,  0/ 0.0, 0.0,  0,   0,  0,  0]SEND
```

1.4.2. Procedures

If tabulated values of $\bar{\nu}_p(E)$ are specified (LNU=2), then pairs of energy-$\bar{\nu}$ values are given. Values of $\bar{\nu}_p(E)$ should be given that cover any energy range in which the fission cross section is given in File 2 and/or File 3. The values of $\bar{\nu}_p(E)$ given in this section are for the average number of prompt neutrons produced per fission event. The energy independent $\bar{\nu}_p$ for spontaneous fission is given using LNU=1 with NC=1 and $C_1=\bar{\nu}_p$ as described for MT = 452.

If MT=456 is specified, then MT=455 must also be specified as well as MT=452.

1.5. Components of Energy Release Due to Fission (MT=458)

The energy released in fission is carried by fission fragments, neutrons, gammas, betas (+ and –), and neutrinos and anti-neutrinos. The term fragments includes all charged particles that are emitted promptly, since for energy-deposition calculations, all such particles have short ranges and are usually considered to lose their energy locally. Neutrons and gammas transport their energy elsewhere and need to be considered separately. In addition, some gammas and neutrons are delayed, and in a shut-down assembly, one needs to know the amount of energy tied up in these particles and the rate at which it is released from the metastable nuclides or precursors. The neutrino energy is lost completely in most applications, but is part of the Q-value. As far as the betas are concerned, prompt betas, being charged, deposit their energy locally with the fragments, and their prompt energies are correctly included with the fragment energies.

ET	Sum of all the partial energies that follow. This sum is the total energy release per fission and equals the Q value.
EFR	Kinetic energy of the fragments.

ENP Kinetic energy of the "prompt" fission neutrons.
END Kinetic energy of the delayed fission neutrons.
EGP Total energy released by the emission of "prompt" γ rays.
EGD Total energy released by the emission of delayed γ rays.
EB Total energy released by delayed β's.
ENU Energy carried away by the neutrinos.
ER Total energy less the energy of the neutrinos (ET - ENU); equal to the pseudo-Q in File 3 for MT=18.

All of these energies are given for an incident energy of zero.[2]

$$E_i(0) = E_i(E_{inc}) + \delta E_i \tag{1.2}$$

where E_i is any of the energy release components;
$E_i(0)$ is the value at $E_{inc} = 0$; $E_{inc} = 0$ is fictitious and represents an artifice by which it is possible to recover the values at any E_{inc}.
$E_i(E_{inc})$ is the value at incident energy E_{inc}.

The δE_i's are given by the following:

$$\begin{aligned}
\delta\mathrm{ET} &= -\left(1.057 E_{inc} - 8.07\left(\bar{\nu}(E_{inc} - \bar{\nu}(0)\right)\right) \\
\delta\mathrm{EB} &= 0.075 E_{inc} \\
\delta\mathrm{EGD} &= 0.075 E_{inc} \\
\delta\mathrm{ENU} &= 1.000 E_{inc} \\
\delta\mathrm{EFR} &= 0 \\
\delta\mathrm{ENP} &= -\left(1.307 E_{inc} - 8.07\left(\bar{\nu}(E_{inc}) - \bar{\nu}(0)\right)\right) \\
\delta\mathrm{EGP} &= 0
\end{aligned}$$

1.5.1. Formats

The structure of this section always starts with a HEAD record and end with a SEND record. The section contains no subsections and only one LIST record.

The structure of a section is:

```
[MAT, 1,458/  ZA,  AWR,   0,    0,   0,    0]HEAD
[MAT, 1,458/ 0.0,  0.0,   0,    0,  18,    9/
             EFR, ΔEFR, ENP, ΔENP, END, ΔEND,
             EGP, ΔEGP, EGD, ΔEGD,  EB,  ΔEB,
             ENU, ΔENU,  ER,  ΔER,  ET,  ΔET]LIST
[MAT, 1,  0/ 0.0,  0.0,   0,    0,   0,    0]SEND
```

where the Δ's allow the error estimates on the quantities listed above.

[2] Taken from R. Sher and C. Beck, Fission Energy Release for 16 Fissioning Nuclides, EPRI-NP-1771 (1981).

1.5.2. Procedures

This section should be used for fertile and fissile isotopes only.

Consistency should be maintained between the Q values in File 3, the energies calculated from File 5 and 15 and the energies listed in File 1. Note that ER = the pseudo-Q for fission (MT=18) in File 3.

Other components are not so readily determined or checked. The procedure should be that File 5 and File 15 data take precedent, whenever available. That is, "prompt" fission neutron energy calculated from File 5 spectra from MT=18 should be used in File 1; the same holds true for the delayed neutron spectra given in File 5, MT=455. The "prompt" gamma energy calculated from File 15 (MT=18 for fission) should be input into File 1, that is the prompt gammas due to the fission process.

These quantities should be calculated at the lowest energy given in the Files for MT=18 except for fissile isotopes for which the thermal spectra should be used. For fertile materials, the spectrum given at threshold would be appropriate. Note that the File 5 spectra for MT=18 should be used with $\bar{\nu}$ prompt (not $\bar{\nu}$ total) for the fission neutrons. MT=455 in File 5 contains the delayed fission neutron spectra.

In many reactor applications, time dependent energy deposition rates are required rather than the components of the total energy per fission which are the values given in this MT. Time-dependent energy deposition parameters can be obtained from the six-group spectra in File 5 (MT=455) for delayed neutrons. Codes such as CINDER, RIBD, and ORIGEN must be used, however, to obtain more detailed information on the delayed neutrons and all time-dependent parameters for the betas and the gammas due to the fission process.

The time-integrated energies for delayed neutrons, delayed gammas, and delayed betas as calculated from the codes listed above may not always agree with the energy components given in File 1. The File 1 components must sum to ET (the total energy released per fission).

In heating calculations, the energy released in all nuclear reactions besides fission, principally the gamma-energy released in neutron radiative capture, enters analogously to the various fission energy components. Thus the (n,γ) energy-release would be equal to the Q-value in File 3, MT=102, of the capturing nuclide. The capture gammas can be prompt or delayed, if branching to isomeric states is involved, and this is relevant to various fission- and burnup-product calculations. The "sensible energy" in a heating calculation is the sum of ER, defined previously, and the energy released in all other reactions.