Designation: E 844 – ~~97~~03

Standard Guide for Sensor Set Design and Irradiation for Reactor Surveillance, E 706(IIC)[1]

This standard is issued under the fixed designation E 844; the number immediately following the designation indicates the year of original adoption or, in the case of revision, the year of last revision. A number in parentheses indicates the year of last reapproval. A superscript epsilon (ε) indicates an editorial change since the last revision or reapproval.

1. Scope

1.1 This guide covers the selection, design, irradiation, post-irradiation handling, and quality control of neutron dosimeters (sensors), thermal neutron shields, and capsules for reactor surveillance neutron dosimetry.

1.2 The values stated in inch-pound units are to be regarded as the standard. The values given in parentheses are for information only.

1.3 *This standard does not purport to address all of the safety problems, if any, associated with its use. It is the responsibility of the user of this standard to establish appropriate safety and health practices and determine the applicability of regulatory limitations prior to use.*

2. Referenced Documents

2.1 *ASTM Standards:*

[1] This guide is under the jurisdiction of ASTM Committee E~~-~~10 on Nuclear Technology and Applications and is the direct responsibility of Subcommittee E10.05 on Nuclear Radiation Metrology.

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E 170 Terminology Relating to Radiation Measurements and Dosimetry[2]
E 261 Practice for Determining Neutron Fluence Rate, Fluence, and Spectra by Radioactivation Techniques[2]
E 854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, E 706(IIIB)[2]
E 910 Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, E 706(IIIC)[2]
E 1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E 706(IIIA)[2]
E 706(IIID) Analysis of Damage Monitors for Reactor Vessel Surveillance[3]
E 706(IIIE) Analysis of Temperature Monitors for Reactor Vessel Surveillance[3]
E 706(IIE) Benchmark Testing of Reactor Vessel Dosimetry[3]

3. Terminology

3.1 *Definitions:*

3.1.1 *neutron dosimeter, sensor, monitor*—a substance irradiated in a neutron environment for the determination of neutron fluence rate, fluence, or spectrum, for example: radiometric monitor (RM), solid state track recorder (SSTR), helium accumulation fluence monitor (HAFM), damage monitor (DM), temperature monitor (TM).

3.1.2 *thermal neutron shield*—a substance (that is, cadmium, boron, gadolinium) that filters or absorbs thermal neutrons.

3.2 For definitions or other terms used in this guide, refer to Terminology E 170.

4. Significance and Use

4.1 In neutron dosimetry, a fission or non-fission dosimeter, or combination of dosimeters, can be used for determining a fluence-rate, fluence, or neutron spectrum, or both, in nuclear reactors. Each dosimeter is sensitive to a specific energy range, and, if desired, increased accuracy in a flux-spectrum can be achieved by the use of several dosimeters each covering specific neutron energy ranges.

4.2 A wide variety of detector materials is used for various purposes. Many of these substances overlap in the energy of the neutrons which they will detect, but many different materials are used for a variety of reasons. These reasons include available analysis equipment, different cross sections for different flux levels and spectra, preferred chemical or physical properties, and, in the case of radiometric dosimeters, varying requirements for different half-life isotopes, possible interfering activities, and chemical separation requirements.

5. Selection of Neutron Dosimeters and Thermal Neutron Shields

5.1 *Neutron Dosimeters*:

5.1.1 The choice of dosimeter material depends largely on the dosimetry technique employed, for example, radiometric monitors, helium accumulation monitors, track recorders, and damage monitors. At the present time, there is a wide variety of detector materials used to perform neutron dosimetry measurements. These are generally in the form of foils, wires, powders, and salts. The use of alloys is valuable for certain applications such as (*1*) dilution of high cross-section elements, (*2*) preparation of elements that are not readily available as foils or wires in the pure state, and (*3*) preparation to permit analysis of more than one dosimeter material.

5.1.2 For neutron dosimeters, the reaction rates are usually deduced from the absolute gamma-ray radioanalysis (there exist exceptions, such as SSTRs, HAFMs, damage monitors). Therefore, the radiometric dosimeters selected must have gamma-ray yields known with good accuracy (>98 %). The half-life of the product nuclide must be long enough to allow for time differences between the end of the irradiation and the subsequent counting. Refer to Method E 1005 for nuclear decay and half-life parameters.

5.1.3 The neutron dosimeters should be sized to permit accurate analysis. The range of high efficiency counting equipment over which accurate measurements can be performed is restricted to several decades of activity levels (5 to 7 decades for radiometric and SSTR dosimeters, 8 decades for HAFMs). Since flux levels at dosimeter locations can range over 2 or 3 decades in a given experiment and over 10 decades between low power and high power experiments, the proper sizing of dosimeter materials is essential to assure accurate and economical analysis.

5.1.4 The estimate of radiometric dosimeter activity levels at the time of counting include adjustments for the decay of the product nuclide after irradiation as well as the rate of product nuclide buildup during irradiation. The applicable equation for such calculations is (in the absence of flux perturbations) as follows:

$$A = N_o \bar{\sigma} \phi \alpha (1 - e^{\lambda t1})(e^{-\lambda t2}) \quad (1)$$

where:

A = expected disintegration rate (dps) for the product nuclide at the time of counting,
N_o = number of target element atoms,

[2] *Annual Book of ASTM Standards*, Vol 12.02.
[3] For standards that are in the draft stage and have not received an ASTM designation, see Section 5 as well as Figures 1 and 2 of Matrix E 706.

ϕ = estimated flux density level,
$\bar{\sigma}$ = spectral averaged cross section,
α = product of the nuclide fraction and (if applicable) of the fission yield,
$1 - e^{-\lambda t1}$ = buildup of the nuclide during the irradiation period, t_1,
$e^{-\lambda t2}$ = decay after irradiation to the time of counting, t_2, and
λ = decay constant for the product nuclide.

5.1.5 For SSTRs and HAFMs, the same type of information as for radiometric monitors (that is, total number of reactions) is provided. The difference being that the end products (fission tracks or helium) requires no time-dependent corrections and are therefore particularly valuable for long-term irradiations.

5.1.6 Fission detectors shall be chosen that have accurately known fission yields. Refer to Method E 1005.

5.1.7 In thermal reactors the correction for neutron self shielding can be appreciable for dosimeters that have highly absorbing resonances (see 6.1.1).

5.1.8 Dosimeters that produce activation or fission products (that are utilized for reaction rate determinations) with half-lives that are short compared to the irradiation duration should not be used. Generally, radionuclides with half-lives less than three times the irradiation duration should be avoided unless there is little or no change in neutron spectral shape or fluence rate with time.

5.1.9 Tables 1-3 present various dosimeter elements. Listed are the element of interest, the nuclear reaction, and the available forms. For the intermediate energy region, the energies of the principal resonances are listed in order of increasing energy. In the case of the fast neutron energy region, the 95 % response ranges (an energy range that includes most of the response for each dosimeter is specified by giving the energies E_{05} below which 5 % of the activity is produced and E_{95} above which 5 % of the activity is produced) for the ^{235}U neutron thermal fission spectrum are included.

5.2 *Thermal Neutron Shields*:

5.2.1 Shield materials are frequently used to eliminate interference from thermal neutron reactions when resonance and fast neutron reactions are being studied. Cadmium is commonly used as a thermal neutron shield, generally 0.020 to 0.050 in. (0.51 to 1.27 mm) thick. However, because elemental cadmium (m.p. = 320°C) will melt if placed within the vessel of an operating water reactor, effective thermal neutron filters must be chosen that will withstand high temperatures of light-water reactors. High-temperature filters include cadmium oxide (or other cadmium compounds or mixtures), boron (enriched in the ^{10}B isotope), and gadolinium. The thicknesses of the shield material must be selected to account for burnout from high fluences.

5.2.2 In reactors, feasible dosimeters to date whose response range to neutron energies of 1 to 3 MeV includes the fission monitors ^{238}U, ^{237}Np, and ^{232}Th. These particular dosimeters must be shielded from thermal neutrons to reduce fission product production from trace quantities of ^{235}U, ^{238}Pu, and ^{239}Pu and to suppress buildup of interfering fissionable nuclides, for example, ^{238}Np and ^{238}Pu in the ^{237}Np dosimeter, ^{239}Pu in the ^{238}U dosimeter, and ^{233}U in the ^{232}Th dosimeter. Thermal neutron shields are also necessary for epithermal spectrum measurements in the 5×10^{-7} to 0.3-MeV energy range. Also, nickel dosimeters used for the fast activation reaction $^{58}Ni(n,p)^{58}Co$ must be shielded from thermal neutrons in nuclear environments having thermal fluence rates above 3×10^{12} n·cm^{-2}·s^{-1} to prevent significant loss of ^{58}Co and ^{58m}Co by thermal neutron burnout **(1)**.[4]

6. Design of Neutron Dosimeters, Thermal Neutron Shields, and Capsules

6.1 *Neutron Dosimeters*—Procedures for handling dosimeter materials during preparation must be developed to ensure personnel safety and accurate nuclear environment characterization. During dosimeter fabrication, care must be taken in order to achieve desired neutron flux results, especially in the case of thermal and resonance-region dosimeters. A number of factors must be considered in the design of a dosimetry set for each particular application. Some of the principal ones are discussed individually as follows:

[4] The boldface number in parentheses refers to the list of references at the end of the guide.

TABLE 1 Dosimeter Elements—Thermal Neutron Region

Element of Interest	Nuclear Reaction	Available Forms
B	$^{10}B(n,\alpha)^{7}Li$	B, B_4C, B-Al, B-Nb
Co	$^{59}Co(n,\gamma)^{60}Co$	Co, Co-Al, Co-Zr
Cu	$^{63}Cu(n,\gamma)^{64}Cu$	Cu, Cu-Al, $Cu(NO_3)_2$
Au	$^{197}Au(n,\gamma)^{198}Au$	Au, Au-Al
In	$^{115}In(n,\gamma)^{116m}In$	In, In-Al
Fe	$^{58}Fe(n,\gamma)^{59}Fe$	Fe
Fe	$^{54}Fe(n,\gamma)^{55}Fe$	Fe
Li	$^{6}Li(n,\alpha)^{3}H$	LiF, Li-Al
Mn	$^{55}Mn(n,\gamma)^{56}Mn$	alloys
Ni	$^{58}Ni(n,\gamma)^{59}Ni(n,\alpha)^{56}Fe$	Ni
Pu	$^{239}Pu(n,f)FP$	PuO_2, alloys
Sc	$^{45}Sc(n,\gamma)^{46}Sc$	Sc, Sc_2O_3
Ag	$^{109}Ag(n,\gamma)^{110m}Ag$	Ag, Ag-Al, $AgNO_3$
Na	$^{23}Na(n,\gamma)^{24}Na$	NaCl, NaF, NaI
Ta	$^{181}Ta(n,\gamma)^{182}Ta$	Ta, Ta_2O_5
U (enriched)	$^{235}U(n,f)FP$	U, U-Al, UO_2, U_3O_8, alloys

TABLE 2 Dosimeter Elements—Intermediate Neutron Region

Energy of Principal Resonance, eV (17)	Dosimetry Reactions	Element of Interest	Available Forms
[A]	$^{6}Li(n,\alpha)^{3}H$	Li	LiF, Li-Al
[A]	$^{10}B(n,\alpha)^{7}Li$	B	B, B_4C, B-Al, B-Nb
[A]	$^{58}Ni(n,\gamma)^{59}Ni(n,\alpha)^{56}Fe$	Ni	Ni
1.457	$^{115}In(n,\gamma)^{116m}In$	In	In, In-Al
4.28	$^{181}Ta(n,\gamma)^{182}Ta$	Ta	Ta, Ta_2O_5
4.906	$^{197}Au(n,\gamma)^{198}Au$	Au	Au, Au-Al
5.19	$^{109}Ag(n,\gamma)^{110m}Ag$	Ag	Ag, Ag-Al, $AgNO_3$
21.806	$^{232}Th(n,\gamma)^{233}Th$	Th	Th, ThO_2, $Th(NO_3)_4$
[B]	$^{235}U(n,f)FP$	U	U, U-Al, UO_2, U_3O_8, alloys
132	$^{59}Co(n,\gamma)^{60}Co$	Co	Co, Co-Al, Co-Zr
1038	$^{58}Fe(n,\gamma)^{59}Fe$	Fe	Fe
337.3	$^{55}Mn(n,\gamma)^{56}Mn$	Mn	alloys
579	$^{63}Cu(n,\gamma)^{64}Cu$	Cu	Cu, Cu-Al, $Cu(NO_3)_2$
0.2956243	$^{239}Pu(n,f)FP$	Pu	PuO_2, alloys
2810	$^{23}Na(n,\gamma)^{24}Na$	Na	NaCl, NaF, NaI
3295	$^{45}Sc(n,\gamma)^{46}Sc$	Sc	Sc, Sc_2O_3
7788	$^{54}Fe(n,\gamma)^{55}Fe$	Fe	Fe

[A] This reaction has no resonance that contributes in the intermediate energy region and the principle resonance has negative energy (i.e. the cross section is 1/v).
[B] Many resonances contribute in the 1 – 100 eV region for this reaction.

6.1.1 *Self-Shielding of Neutrons* —The neutron self-shielding phenomenon occurs when high cross-section atoms in the outer layers of a dosimeter reduce the neutron flux to the point where it significantly affects the activation of the inner atoms of the material. This is especially true of materials with high thermal cross sections and essentially all resonance detectors. This can be minimized by using low weight percentage alloys of high-cross-section material, for example, Co-Al, Ag-Al, B-Al, Li-Al. It is not as significant for the fast region where the cross sections are relatively low; therefore, thermal and resonance detectors shall be as thin as possible. Mathematical corrections can also be made to bring the material to "zero thickness" but, in general, the smaller the correction, the more accurate will be the results. Both theoretical treatments of the complex corrections and experimental determinations are published **(2-13,20)**.

6.1.2 *Self-Absorption of Emitted Radiation*—This effect may be observed during counting of the radiometric dosimeter. If the radiation of interest is a low-energy gamma ray, an X ray, or a beta particle, the thickness of the dosimeter may be of appreciable significance as a radiation absorber (especially for higher atomic number materials). This will lower the counting rate, which would then have to be adjusted in a manner similar to that for the "zero thickness" correction in the case of self-shielding. Therefore, it would again be desirable to use thin dosimeters in cases where the count rate is affected by dosimeter thickness. In the case of thick pellets, it is usually possible to perform chemical separation of the radionuclide.

6.1.3 *Fission Fragment Loss*—It has been observed that fission foils of 0.001-in. (0.03-mm) thickness lose a significant fraction (approximately 7 %) of the fission fragments. Increasing the thickness to 0.005 in. (0.13 mm) will reduce this loss to about 1 %.

6.1.4 *Dosimeter Size*:

6.1.4.1 The size of dosimeters and dosimetry sets is often limited by space available, especially in reactor applications where volume in high flux regions is very limited and in great demand for experimental samples. This fact, coupled with the desirability of minimizing perturbations to the reactor environment due to the presence of the dosimetry set, of minimizing self-shielding corrections, and of minimizing corrections to obtain reaction rates at a common point in space, creates the need for miniaturized dosimeters.

6.1.4.2 The larger the dosimeter, the higher the counting rate of the activated nuclide or the higher the amount of stable product. This would be desirable in low-flux regions, but probably undersirable in high fluxes for radiometric dosimeters, since the excessive count rate may result in dead-time losses. Excess activity may result in a radiation hazard. Certain types of dosimeters (for example, HAFMs, foils, wires, and dissolvable samples) can be segmented or diluted prior to analysis. The lower limit on dosimeter size would be governed by a size that could be readily handled and would not be easily lost or overlooked.

6.1.5 *Temperature*— In high-power reactor irradiations, dosimeters must be constructed to withstand the adverse environment. The temperature, as determined by gamma and neutron reaction heating and heat transfer, will often be too high for simple bare dosimeters. At high temperatures, migration of reaction products, melting, or diffusion bonding may occur, necessitating encapsulation in a high-temperature material with non-interfering or short half-life products.

6.1.6 *Burnup*:

6.1.6.1 Long irradiations can introduce additional problems. Burnup of the dosimeters and burn-in or burn-out of the product nuclide, or both, may occur. Calculation of burn-up corrections may be complicated by reactions other than the one measured, such as neutron capture by fission or threshold dosimeters. Long irradiations also admit the possibility of two-stage competing reactions, the best examples being ^{238}U, ^{237}Np, and ^{232}Th. Fig. 1 schematically shows the production of ^{137}Cs by ^{238}U, ^{237}Np, and ^{232}Th reactions.

6.1.6.2 At moderately high fluences, fission products from two-step reactions can dominate those produced directly by ^{238}U, ^{237}Np, or ^{232}Th fission, which limits their usefulness as fluence threshold dosimeters. Fig. 2 graphically shows semi-empirical

TABLE 3 Dosimeter Elements—Fast Neutron Region

Dosimetry Reactions	Element of Interest	Energy Response Range (MeV)[A,B] Low E_{05}	Median E_{50}	High E_{95}	Cross Section Uncertainty (%)[A,C]	Available Forms
~~^{237}Np(n,f)FP~~	~~Np~~	~~0.684~~	~~1.96~~	~~5.61~~	~~9.33~~	~~Np_2O_3, alloys~~
^{237}Np(n,f)FP	Np	0.684	1.96	5.61	9.34	Np_2O_3, alloys
~~^{103}Rh(n,n′)^{103m}Rh~~	~~Rh~~	~~0.731~~	~~2.25~~	~~5.73~~	~~3.1~~	~~Rh~~
^{103}Rh(n,n′)^{103m}Rh	Rh	0.731	2.25	5.73	3.10	Rh
~~^{93}Nb(n,n′)^{93m}Nb~~	~~Nb~~	~~0.951~~	~~2.57~~	~~5.79~~	~~3.06~~	~~Nb, Nb_2O_5~~
^{93}Nb(n,n′)^{93m}Nb	Nb	0.951	2.57	5.79	3.01	Nb, Nb_2O_5
~~^{115}In(n,n′)^{115m}In~~	~~In~~	~~1.12~~	~~2.55~~	~~5.86~~	~~2.17~~	~~In, In-Al~~
^{115}In(n,n′)^{115m}In	In	1.12	2.55	5.86	2.16	In, In-Al
^{14}N(n,α)^{11}B	N	1.75	3.39	5.86	—	TiN, ZrN, NbN
~~^{238}U(n,f)FP~~	~~U (depleted)~~	~~1.44~~	~~2.61~~	~~6.69~~	~~0.53~~	~~U, U-Al, UO_3, U_3O_8, alloys~~
^{238}U(n,f)FP	U (depleted)	1.44	2.61	6.69	0.319	U, U-Al, UO_3, U_3O_8, alloys
~~^{232}Th(n,f)FP~~	~~Th~~	~~1.45~~	~~2.79~~	~~7.21~~	~~5.09~~	~~Th, ThO_2~~
^{232}Th(n,f)FP	Th	1.45	2.79	7.21	5.11	Th, ThO_2
^{9}Be(n,α)^{6}Li	Be	1.59	2.83	5.26	—	Be
~~^{47}Ti(n,p)^{47}Sc~~	~~Ti~~	~~1.70~~	~~3.63~~	~~7.67~~	~~2.17~~	~~Ti~~
^{47}Ti(n,p)^{47}Sc	Ti	1.70	3.63	7.67	3.77	Ti
~~^{58}Ni(n,p)^{58}Co~~	~~Ni~~	~~1.98~~	~~3.94~~	~~7.51~~	~~2.43~~	~~Ni, Ni-Al~~
^{58}Ni(n,p)^{58}Co	Ni	1.98	3.94	7.51	2.44	Ni, Ni-Al
~~^{54}Fe(n,p)^{54}Mn~~	~~Fe~~	~~2.27~~	~~4.09~~	~~7.54~~	~~2.17~~	~~Fe~~
^{54}Fe(n,p)^{54}Mn	Fe	2.27	4.09	7.54	2.12	Fe
~~^{32}S(n,p)^{32}P~~	~~S~~	~~2.28~~	~~3.94~~	~~7.33~~	~~4.0~~	~~$CaSO_4$, Li_2SO_4~~
^{32}S(n,p)^{32}P	S	2.28	3.94	7.33	3.63	$CaSO_4$, Li_2SO_4
^{32}S(n,α)^{29}Si	S	1.65	3.12	6.06	—	Cu_2S, PbS
^{58}Ni(n,α)^{55}Fe	Ni	2.74	5.16	8.72	—	Ni, Ni-Al
~~^{46}Ti(n,p)^{46}Sc~~	~~Ti~~	~~3.70~~	~~5.72~~	~~9.43~~	~~2.46~~	~~Ti~~
^{46}Ti(n,p)^{46}Sc	Ti	3.70	5.72	9.43	2.48	Ti
~~^{56}Fe(n,p)^{56}Mn~~	~~Fe[D]~~	~~5.45~~	~~7.27~~	~~11.3~~	~~2.33~~	~~Fe~~
^{56}Fe(n,p)^{56}Mn	Fe[D]	5.45	7.27	11.3	2.26	Fe
^{56}Fe(n,α)^{53}Cr	Fe	5.19	7.53	11.3	—	Fe
~~^{63}Cu(n,α)^{60}Co~~	~~Cu[E]~~	~~4.53~~	~~6.99~~	~~11.0~~	~~2.85~~	~~Cu, Cu-Al~~
^{63}Cu(n,α)^{60}Co	Cu[E]	4.53	6.99	11.0	2.36	Cu, Cu-Al
~~^{27}Al(n,α)^{24}Na~~	~~Al~~	~~6.45~~	~~8.40~~	~~11.9~~	~~1.40~~	~~Al, Al_2O_3~~
^{27}Al(n,α)^{24}Na	Al	6.45	8.40	11.9	1.19	Al, Al_2O_3
~~^{48}Ti(n,p)^{48}Sc~~	~~Ti~~	~~5.92~~	~~8.06~~	~~12.3~~	~~6.66~~	~~Ti~~
^{48}Ti(n,p)^{48}Sc	Ti	5.92	8.06	12.3	2.56	Ti
^{47}Ti(n,α)^{44}Ca	Ti	2.80	5.10	9.12	—	Ti
^{60}Ni(n,p)^{60}Co[F]	Ni[E]	4.72	6.82	10.8	10.3	Ni, Ni-Al
~~^{55}Mn(n,2n)^{54}Mn~~	~~Mn[G]~~	~~11.0~~	~~12.6~~	~~15.8~~	~~13.4~~	~~alloys~~
^{55}Mn(n,2n)^{54}Mn	Mn[G]	11.0	12.6	15.8	13.54	alloys

[A] Energy response range was derived using the ENDF/B-VI ^{235}U fission spectrum, reference **(17)** , MT = 9228, MF = 5, MT = 18. The cross section and associated covariance sources are identified in standard E 1018 and in Reference **(18)**.

[B] One half of the detector response occurs below an energy given by E^{50}; 95 % of the detector response occurs below E_{95} and 5 % below E_{05}.

[C] Uncertainty metric only reflects that component due to the knowledge of the cross section and is reported at the 1σ level.

[D] Low manganese content necessary.

[E] Low cobalt content necessary.

[F] This reaction does not appear in the ENDF/B-VI evaluation. The JENDL evaluation **(19)** was used.

[G] Low iron content necessary.

calculations of ^{137}Cs produced from the irradiation of infinitely dilute bare and cadmium-covered ^{238}U, ^{237}Np, and ^{232}Th **(14)**. These curves can be used as a guide to estimate corrections and are based on a neutron spectrum distribution of Thermal/Intermediate/>1 MeV = 1/1/0.25. The abscissa scale corresponds to the number of neutrons·cm^{-2} either in the total spectrum (bare) or in the epithermal spectrum (Cd). Accurate corrections for these types of reactions are often very difficult to calculate. Epithermal fluences of greater than 1×10^{20} n·cm^{-2} for thermally-shielded fast fission dosimeters should be avoided.

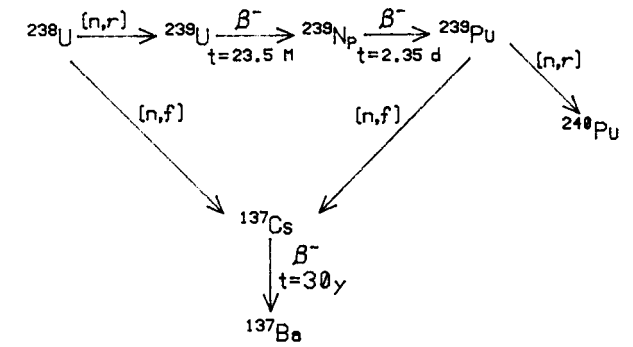
6.1.7 *Cross-Contamination*—During fabrication, and subsequent post-irradiation handling, materials must not cross-contaminate one another.

6.2 *Thermal Neutron Shields*—Powder metallurgy techniques can be used to produce thermal neutron shield hollow cylinders of compound mixtures that lend themselves to pressing. Examples of these are boron, gadolinium, cadmium, cadmium oxide, and cadmium oxide-copper. If encapsulated powders are used, care must be taken to prevent redistribution of material. Shield radiographs are recommended.

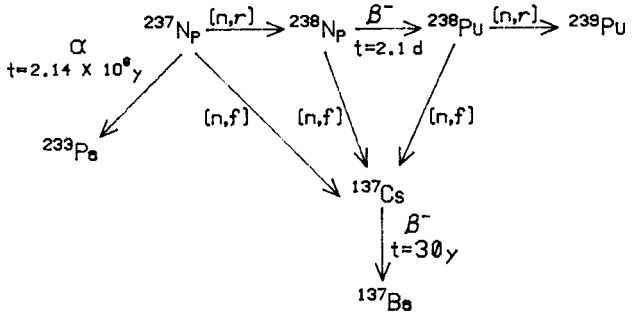
6.3 *Capsules*:

6.3.1 Five important criteria shall be met by the dosimeter capsule design: (*1*) it must not interfere with the function of the irradiation experiment (for example, stainless steel should not be used to contain thermal flux dosimeters); (*2*) it must position all dosimeters of the dosimetry set in close proximity to the experiment (see 7.1); (*3*) it must be easy to load; (*4*) it must be easy to unload since this is often a hot-cell operation; and (*5*) it must not perturb the neutron environment excessively.

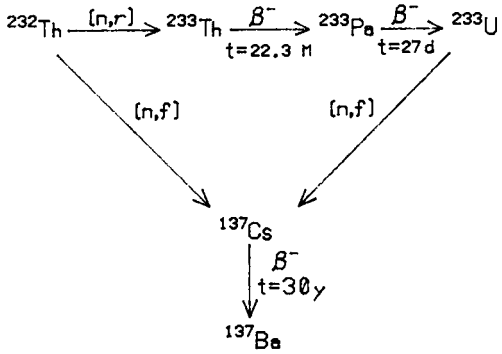
6.3.2 Fission dosimeters may be encapsulated in hermetically sealed containers to avoid oxidation and loss of materials, and for health-hazard requirements.



1(a) Production of ^{137}Cs by ^{238}U Reactions



1(b) Production of ^{137}Cs by ^{237}Np Reactions



1(c) Production of ^{137}Cs by ^{232}Th Reactions

FIG. 1 Production of ^{137}Cs

6.3.3 The technology for making vanadium capsules to contain dosimeter materials has been developed **(15)** . Vanadium was chosen as an encapsulation material because of its nuclear and high temperature properties. In addition to vanadium, copper, aluminum, and quartz encapsulation have been found satisfactory for uranium, plutonium, neptunium, thorium, and other elemental oxides or salts.

6.3.4 Procedures have been developed to make HAFMs **(16)**. Boron, lithium, and other specimens that may require encapsulation, can be sealed in miniature vanadium or Au/Pt capsules.

6.3.5 Post-irradiation recovery requires that individual dosimeters be readily identified; thus the dosimeter capsule identification and location within the experiment must be recorded along with the location of the individual dosimeters within the capsule.

7. Irradiation

7.1 Exact locations of individual dosimeters must be recorded for irradiation analysis. If the dosimeters in a set cannot be located in the same region or in a region of uniform neutron field, they can occupy a larger volume of varying fluence rate provided the neutron spectral shape is constant. Flux gradients can introduce large uncertainties in reaction rate and flux-spectral results. Gradient flux dosimeters (for example, nickel, iron, or aluminum-cobalt, or both,) must then be placed at each location. Considerations discussed in Practice E 261 apply.

7.2 Dosimeters shielded from thermal neutrons must be located apart from non-shielded (“bare”) dosimeters to avoid thermal flux depression of the bare dosimeters.

7.3 A strong resonance absorber such as thick ^{235}U,^{239}Pu, silver, cobalt, and gold cannot be placed in front of a 1/v absorber, and thick dosimeters should not be stacked so as to result in large neutron scattering corrections.

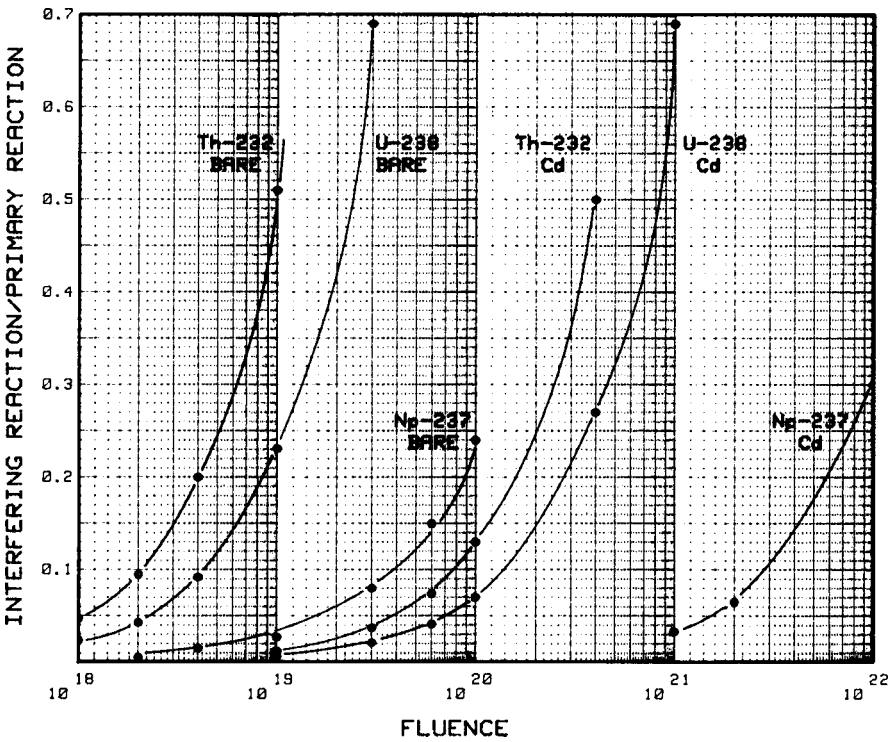


FIG. 2 Irradiation of ^{238}U, ^{237}Np, and ^{232}Th

8. Post-Irradiation Handling

8.1 Hot-cell or remote handling facilities are often required for recovery of dosimeter materials after an irradiation. Remote handling operations for dosimetry should be planned and supervised by personnel familiar with the assembly of the dosimetry capsules. Since the dosimetry recovery operation is seldom routine, complete familiarity with the construction of the capsule and identification of the dosimeters is essential. Careful planning and practice will (*1*) eliminate loss of dosimeters due to improper opening of dosimeter containers and identification mixups; (*2*) preclude decay of short half-life reactions because of excessive recovery time; and (*3*) minimize the cost of hot cell operations.

8.2 A list of materials commonly used in a remote handling facility for the recovery sequence include:

8.2.1 Clean paper or plastic covering a clear work area,

8.2.2 Clean manipulator fingers to prevent contamination to the dosimeters from previous hot cell work,

8.2.3 Telescopic viewer for identification of encapsulated dosimeters,

8.2.4 A cut-off wheel for opening welded containers,

8.2.5 A small vise, screwdriver, tweezers, and

8.2.6 Vials, each appropriately marked with capsule identification and dosimeter type.

8.3 After capsule disassembly, the dosimeters shall be cleaned with an appropriate solution (for example, acid or acetone, or both), smeared to ensure the absence of radioactivity contamination (if dose levels permit), and weighed (if not weighed prior to irradiation).

8.4 Information to be supplied for radioactivity and flux analysis include:

8.4.1 Dosimeter identification,

8.4.2 Isotopic assay of fission dosimeters, alloys, and mixtures,

8.4.3 Weights or concentrations,

8.4.4 Reactor spatial power history (includes time of the end of the irradiation),

8.4.5 Neutron energy region analysis desired, and

8.4.6 Description of encapsulation.

8.5 Refer to Method E 1005 for the analysis of radiometric monitors, Method E 854 for the analysis of solid state track recorder monitors, Method E 910 for the analysis of helium accumulation neutron monitors, Method E 706 (IIID) for the analysis of damage monitors, Method E 706 (IIIE) for the analysis of temperature monitors, and Guide E 706 (IIE) for a guide to benchmark neutron field referencing.

9. Quality Control

9.1 Dosimeter materials must be of adequate purity to ensure that any impurity present will not produce a significant error in the product nuclide or in the assessment of the amount of the monitor nuclide present in the monitor.

9.2 The elemental or isotopic dosimeter quantities should be checked or ~~redetermined~~ confirmed either prior to or following an irradiation. Analytical measurement methods include atomic absorption, spectrophotometry, emission spectrometry, neutron activation analysis, mass spectrometry, and radioactivity counting techniques. The dosimeter purity analysis results must be kept on permanent record for use in making dosimeter impurity corrections. These impurities must be known to an accuracy dictated by the magnitude of the correction.

9.3 Most dosimeter materials are readily available in high-purity form. Certain materials cannot be obtained in a high-purity state without excessive processing and cost. When impurities do exist, it should be realized that there are a number of impurities that will not interfere with the analysis because of (*1*) low cross section, (*2*) short half-life, or (*3*) low-detection efficiency. These include O, N, C, Si, H, Be, A1, Mg, F, S, Ca, Zr, and Pb. There are certain other elements that should be avoided as impurities, when thermal or intermediate flux analysis, or both, is desired, because of their high thermal or resonance cross sections and the resulting readily detectable activities. These include Li, B, Cd, In, Hg, Au, Mn, Ta, W, Th, U, Bi, Co, Hf, K, Sn, and rare earths. Occasionally the effect of impurity nuclides can be reduced to acceptable levels by the use of thermal neutron filters (for example, traces of Co in high purity Cu, Sc in Ti, and Ta in Nb). Specific impurities that interfere with the radioanalysis of dosimeters and should be avoided are given inTable 4 for commonly used dosimeter materials. For HAFMs, the two most important impurity elements are B and Li.

9.4 Before using a given specimen or lot of dosimeter material, the impurities should be carefully evaluated to determine that no impurity significantly contributes to the activity or parameter to be measured. For example, in a thermal neutron spectrum, a small quantity of Mn could invalidate a measurement using the reaction ^{56}Fe(n,p) ^{56}Mn.

9.5 Dosimeter nuclides should be known to better than ± 2 % (provided that the isotopic mole fraction of the nuclide is well-enough known) either by weighing or chemical assay using suitable equipment and techniques.

10. Keywords

10.1 activity; dosimeter; fission monitor; monitor; monitor foil; neutron fluence; pressure vessel; radiometric monitor; reaction rate; reactor surveillance; sensor

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TABLE 4 Impurities in Commonly Used Dosimeter Materials

Element of Interest	Interfering Impurities
Al	Na
Cu	Co (>1 ppm)[A]
^{237}Np	^{238}Pu
Ni	Co
Nb	Ta (>100 ppm)[A]
^{239}Pu	other Pu isotopes
Rh	Ir, W
Sc	Zn
S	Cl
^{238}U	^{235}U (>40 ppm)[A]

[A]Require corrections in typical thermally-shielded surveillance application. For unshielded or special cases, even lower levels of these impurities may contribute.

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