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2067-12

Joint ICTP/IAEA Workshop on Irradiation-induced Embrittlement of Pressure Vessel Steels

23 - 27 November 2009

NEUTRON DOSIMETRY

for RPV and surveillance programmes

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NEUTRON DOSIMETRY
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Milan Brumovský

Joint ICTP/IAEA Workshop on Effects of Mechanical Properties and Mechanisms Governing the Irradiation-induced Embrittlement of Pressure Vessel Steels
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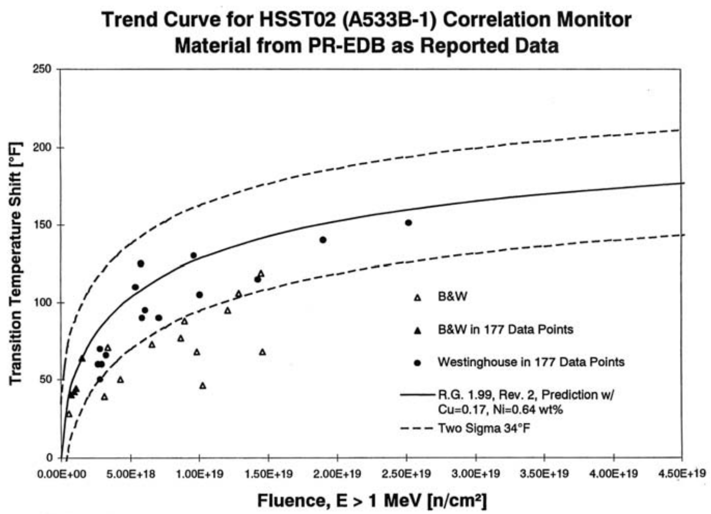
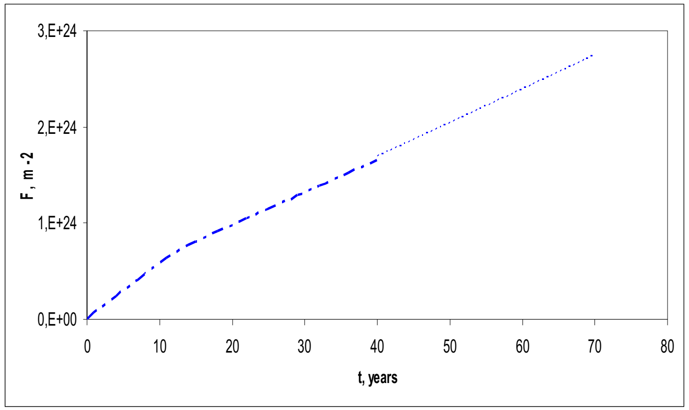
INTRODUCTION

NEUTRON DOSIMETRY IS A VERY IMPORTANT AND LEADING PARAMETER FOR RPV INTEGRITY AND LIFETIME EVALUATION

- **DETERMINATION OF FLUENCE DEPENDENCE OF MATERIAL TRANSITION TEMPERATURE CHANGES**



- **DETERMINATION OF FLUENCE TREND IN RPV**



DEFINITIONS

Designation: E 170 – 05

Standard Terminology Relating to Radiation Measurements and Dosimetry[1]

neutron activation detector—a neutron fluence sensing device, well-characterized with respect to geometry, mass, composition and cross section which produces activity with a sufficiently long half life to permit its measurement after withdrawal from the neutron field.

threshold neutron activation detector—a device containing a particular nuclide that is not significantly activated by neutrons below a certain threshold energy.

Designation: E 170 – 05 **Standard Terminology Relating to Radiation Measurements and Dosimetry**[1]

activity, A—of an amount of radioactive nuclide in a particular energy state at a given time, the quotient of dN by dt, where dN is the expectation value of the number of spontaneous nuclear transitions from that energy state in the time interval dt (ICRU).

$$A = \mathrm{d}N/\mathrm{d}t \tag{4}$$

Unit: s^{-1}

decay constant, $\boldsymbol{\lambda}$—of a radioactive nuclide in a particular energy state, the quotient of dP by dt, where dP is the probability of a given nucleus undergoing a spontaneous nuclear transition from that energy state in the time interval dt (ICRU).

$$\lambda = \mathrm{d}P/\mathrm{d}\,t \tag{11}$$

Unit: s^{-1}

DISCUSSION—The quantity (ln 2)/λ is commonly called the $\boxed{\text{half-life,}}$ $T½$, of the radioactive nuclide, that is, the time taken for the activity of an amount of radioactive nuclide to become half its initial value.

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Designation: E 170 – 05

Standard Terminology Relating to Radiation Measurements and Dosimetry[1]

cross section, σ—the quotient of P by Φ, where P is the probability of the interaction for one target entity when subjected to the particle fluence Φ (ICRU).

$$\sigma = P/\Phi \tag{9}$$

Unit: m^2

The special unit of cross section is the barn, b.

$$1\ \mathrm{b} = 10^{-28}\ \mathrm{m}^2 \tag{10}$$

spectrum averaged cross section, $\bar{\sigma}$—the cross section averaged over the energy distribution of the neutron fluence, where the energy limits of integration are chosen according

to the neutron spectrum and reaction cross section considered. It is defined as:

$$\bar{\sigma} = \int \sigma(E)\,\Phi(E)\mathrm{d}E \Big/ \int \Phi(E)\mathrm{d}E \tag{31}$$

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Designation: E 170 – 05

Standard Terminology Relating to Radiation Measurements and Dosimetry[1]

particle fluence rate, ϕ—the quotient of $d\phi$ by dt, where $d\Phi$ is the increment of particle fluence in the time interval dt (ICRU).

$$\phi = \frac{d\Phi}{dt} = \frac{d^2N}{da\,dt} \quad (28)$$

Unit: $m^{-2} \cdot s^{-1}$

DISCUSSION—The term **particle flux density** may be used but the term fluence rate conforms to the adoption of a uniform set of terms and units as prescribed by ICRU and SI units. Historically, the term *neutron flux* has been understood to mean neutron flux density (fluence rate). This term still is widely used in the nuclear community.

integral neutron fluence—the fluence of neutrons integrated over all energies.

$$\Phi = \int_0^\infty \Phi(E)\,dE \quad (17)$$

particle fluence spectrum, ϕ(E)—the quotient of $d\phi$ by dE, where $d\Phi$ is the particle fluence of particles in the interval E to $E + dE$.

$$\phi(E) = \frac{d\Phi}{dE} \quad (29)$$

Unit: $m^{-2} \cdot eV^{-1}$

DISCUSSION—The term **particle spectrum** is commonly used in E10 standards to indicate particle fluence spectrum. It should not be confused with the particle density spectrum, defined as *the quotient of the volumetric density of particles in dE by dE ($m^{-3} \cdot eV^{-1}$).*

Designation: E 170 – 05

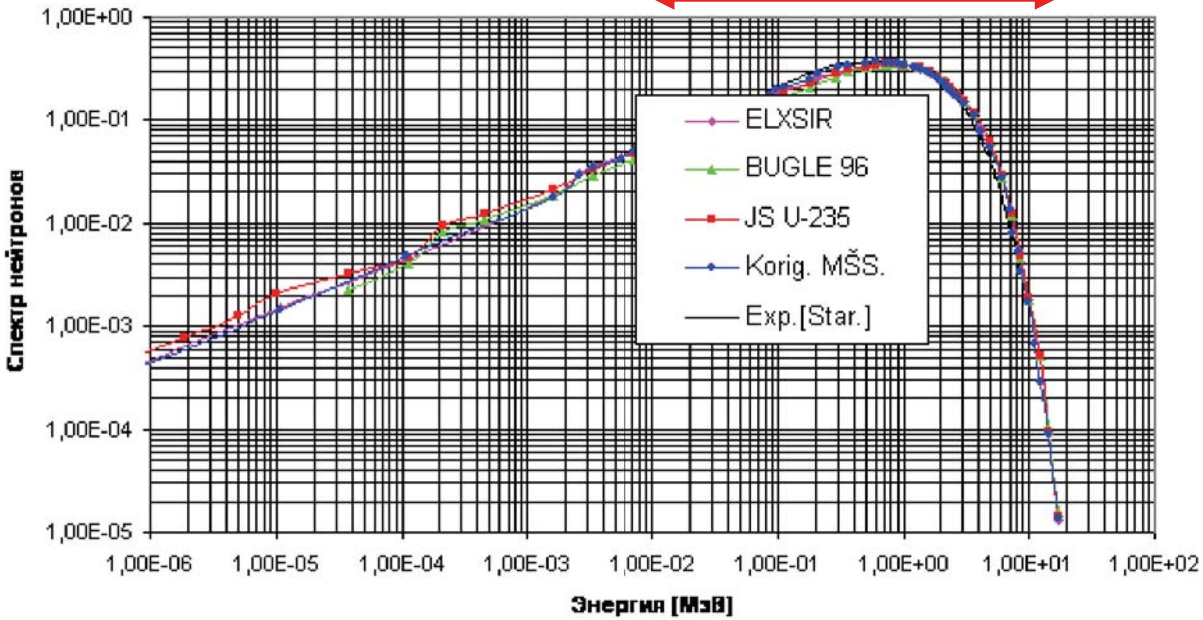
Standard Terminology Relating to Radiation Measurements and Dosimetry[1]

fast neutrons—a term for designating neutrons of energy exceeding some threshold that must be specified (typically 0.1 or 1 MeV); often associated with those neutrons predominantly responsible for displacement damage of materials in neutron radiation fields.

thermal neutrons—neutrons in thermal equilibrium with the medium through which they are traveling or diffusing.

2200 m/s cross section, $\sigma(v_o)$—the neutron cross section at v_o = 2200 m/s (*E* is about 0.0253 eV).

Epithermal neutrons



U235 Fission spectrum

Designation: E 170 – 05

Standard Terminology Relating to Radiation Measurements and Dosimetry[1]

displacements per atom (dpa)—the mean number of times each atom of a solid is displaced from its lattice site during an exposure to displacing radiation, as calculated following standard procedures (see **displacement dose**).

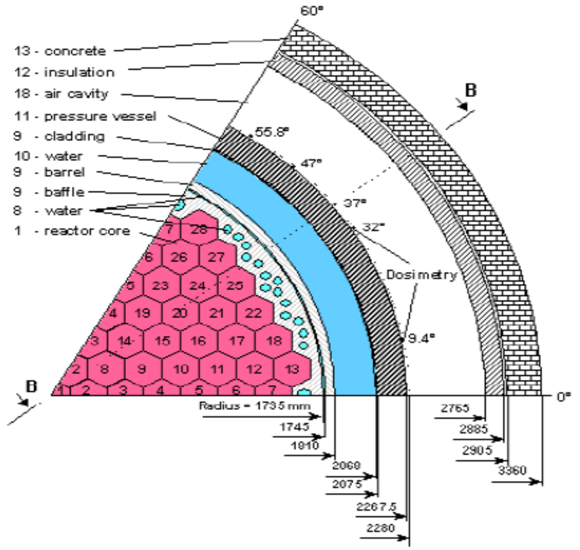
CALCULATIONS

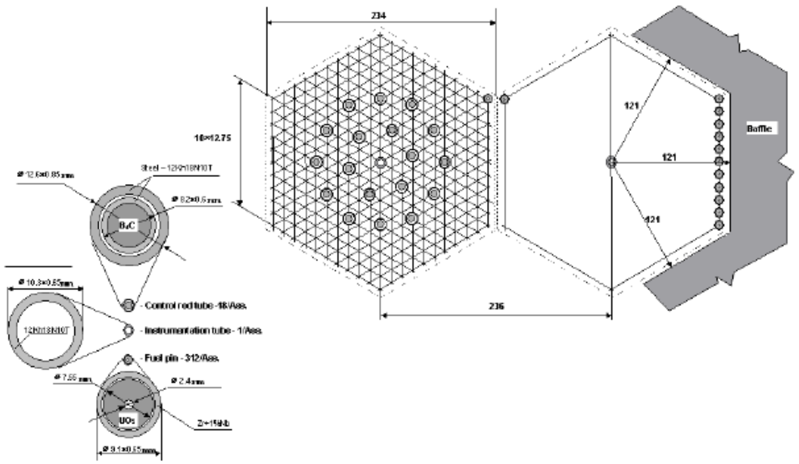
- CALCULATIONS ARE PERFORMED MOSTLY FOR DETERMINATION OF NEUTRON FLUENCE ON RPV WALL
- CALCULATION PROCEDURES MUST BE VALIDATED BY EXPERIMENTAL DATA – FROM SURVEILLANCE SPECIMEN PROGRAMMES AND, IF POSSIBLE, FROM EX-VESSEL MEASUREMENTS
- CALCULATION IS BASED ON:
  - TRANSPORT THEORY
  - MONTE CARLO METHOD
- THE FOLLOWING DATA MUST BE AVAILABLE FOR ANY CALCULATIONS:
  - DETAILED GEOMETRY OF ACTIVE CORE INCLUDING FUEL ELEMENTS AND ALL MATERIALS
  - DETAILED GEOMETRY BETWEEN ACTIVE CORE, SURVEILLANCE SPECIMENS AND RPV WALL INCLUDING MATERIALS
  - EVERYDAY OUTPUT OF INDIVIDUAL FUEL ELEMENTS

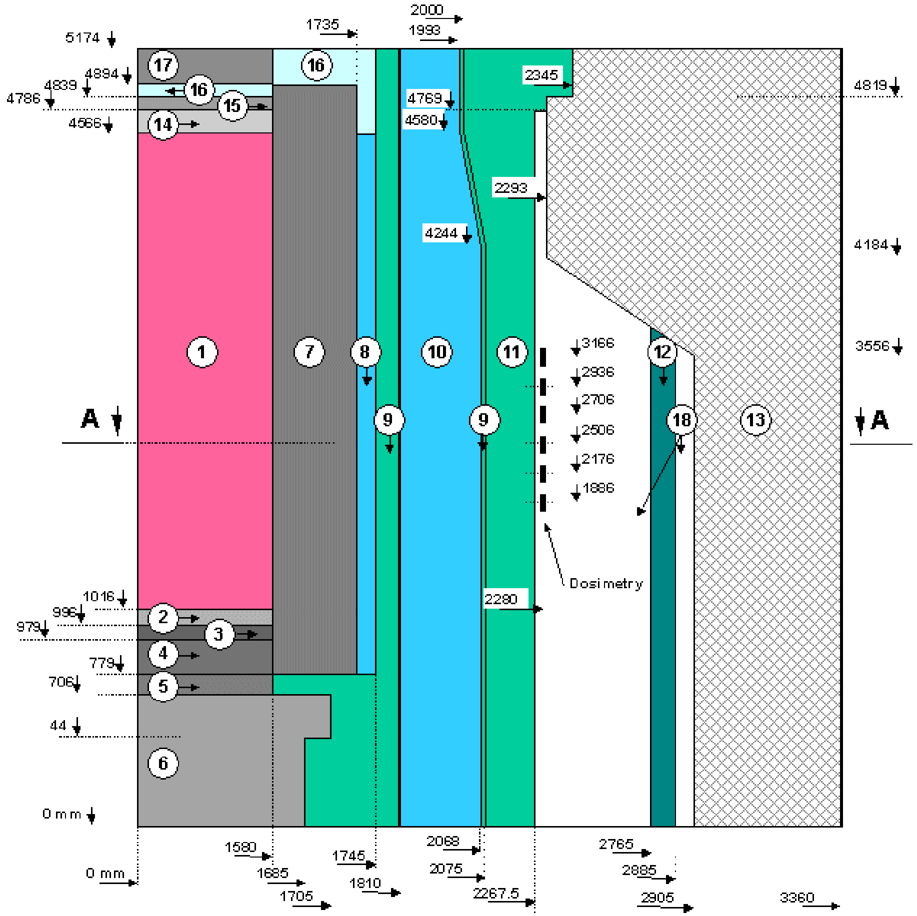
CALCULATIONS

- CALCULATIONS ARE PERFORMED EITHER IN 2-D OR 3-D GEOMETRY BY SEVERAL PROGRAMMES, LIKE DORT, TORT, ANISN etc.
- OUTPUT FROM CALCULATIONS:
  - NEUTRON ENERGY SPECTRA IN DIFFERENT LOCATIONS INSIDE RPV AND THROUGH RPV
  - NEUTRON FLUXES IN THESE LOCATIONS
  - NEUTRON FLUENCES IN THESE LOCATIONS
  - ACTIVITY OF NEUTRON MONITORS – FOR COMPARISON AND VALIDATION OF THE CALCULATION PROGRAMME
- COMPARISON OF CALCULATED AND MEASURED ACTIVITIES SERVES FOR A VALIDATION OF THE PROGRAMME AS WELL AS FOR ITS FITTING

CALCULATIONS – EXAMPLE OF GEOMETRY







CHOICE OF MONITORS

- EXPERIMENTAL NEUTRON DOSIMETRY IN EXPERIMENTAL/ POWER REACTORS DEPENDS ON A PROPER AND EFFECTIVE CHOICE OF MONITORS
- MONITORS MUST BE CHOSEN PREFERABLY REGARDING THEIR HALF-TIME AND TIME OF IRRADIATION
  - AT LEAST ONE MONITOR SHOULD COVER THERMAL NEUTRON REGION
  - SEVERAL MONITORS (IF POSSIBLE) SHOULD COVER FAST NEUTRON REGION
  - MOST OF REACTORS FOR FAST NEUTRON REGION ARE OF THRESHOLD TYPE WITH THRESHOLD MUCH HIGHER THAN 0.5/1 MeV, THUS FISSION MONITORS AND Nb ARE VERY USEFUL
  - FOR DETERMINATION OF NEUTRON FLUENCE, MONITORS CAN BE PRACTICALLY USED UP TO 5-TIMES THEIR HALF LIFE

CHOICE OF MONITORS

REACTION	ISOTOPIC ABUNDANCE, %	^{235}U FISSION SPECTRUM AVERAGED CROSS SECTION (mb)	ENERGY RESPONSE RANGE, MeV			HALF-TIME	Eγ, MeV
			LOW, $E_{0.5}$	MEDIAN, E_{50}	HIGH, E_{95}		
^{59}Co(n,γ)^{60}Co	100.0	37 233	thermal			5.271 y	1.332
^{237}Np(n,f) ^{137}Cs	100.0	1344.0	0.684	1.96	5.61	30.17 y	0.662
^{238}U(n,f) ^{137}Cs	depleted	309.0	1.44	2.61	6.69	30.17 y	0.662
^{93}Nb(n,n')^{93m}Nb	100.0	146.2	0.97	4.5	11	16.13 y	16.6 and 17.7 keV
^{93}Nb(n,γ)^{94}Nb	100.0					20 300 y	704 and 897 keV
^{58}Ni(n,p)^{58}Co	68.27	108.5	1.98	3.94	7.51	70.82 d	0.8108
^{54}Fe(n,p)^{54}Mn	5.9	80.5	2.27	4.09	7.54	312.5 d	0.8348
^{46}Ti(n,p)^{46}Sc	8.1	19.0	3.70	5.72	9.43	83.81 d	0.8893 1.1205
^{63}Cu(n,α)^{60}Co	69.17	0.50	4.53	6.99	11.0	5.271 y	1.1732 1.3325

CHOICE OF MONITORS

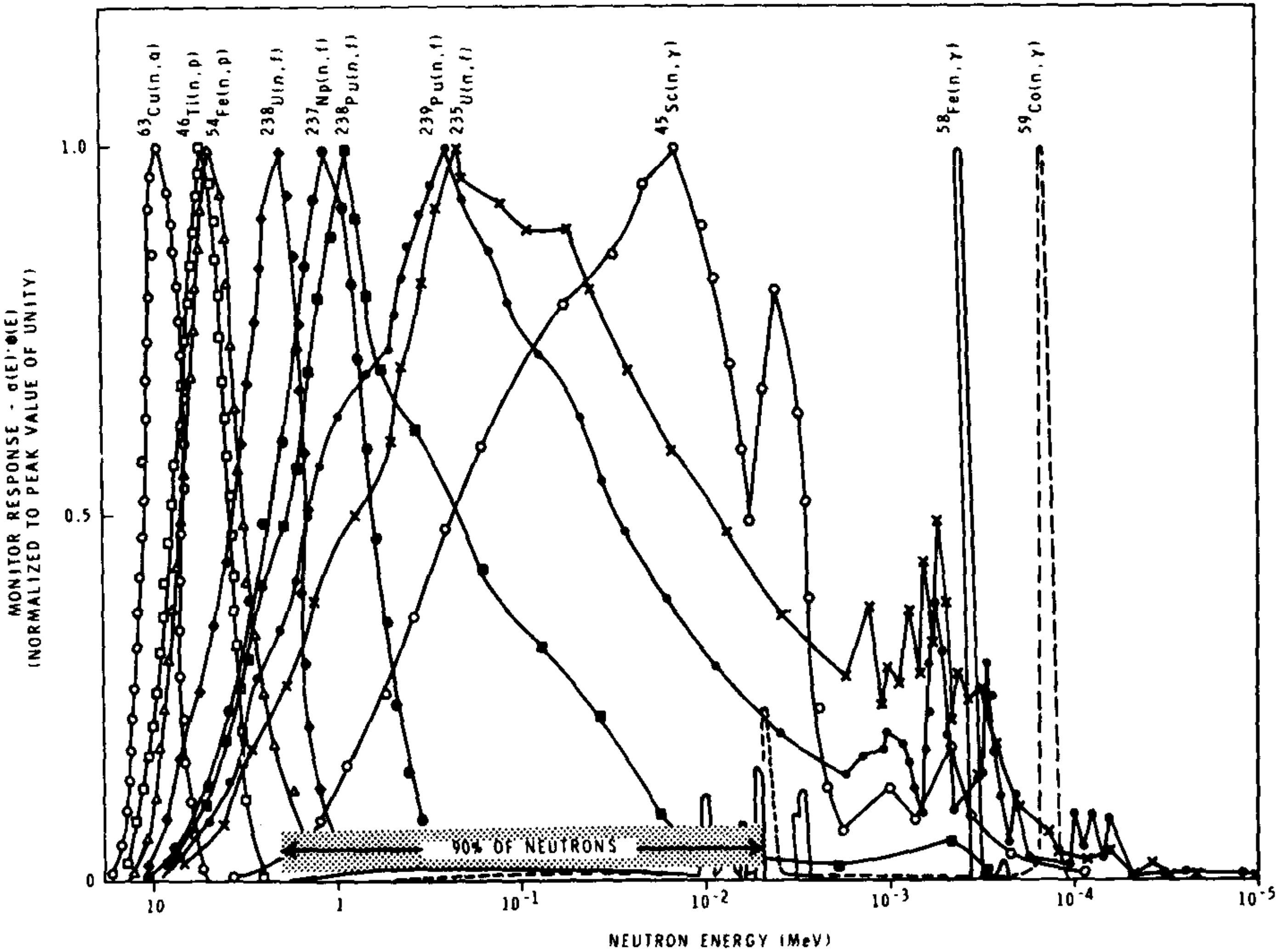


FIG. 34. Monitor response in the Fast Test Reactor driver element spectrum [9].

CHOICE OF MONITORS

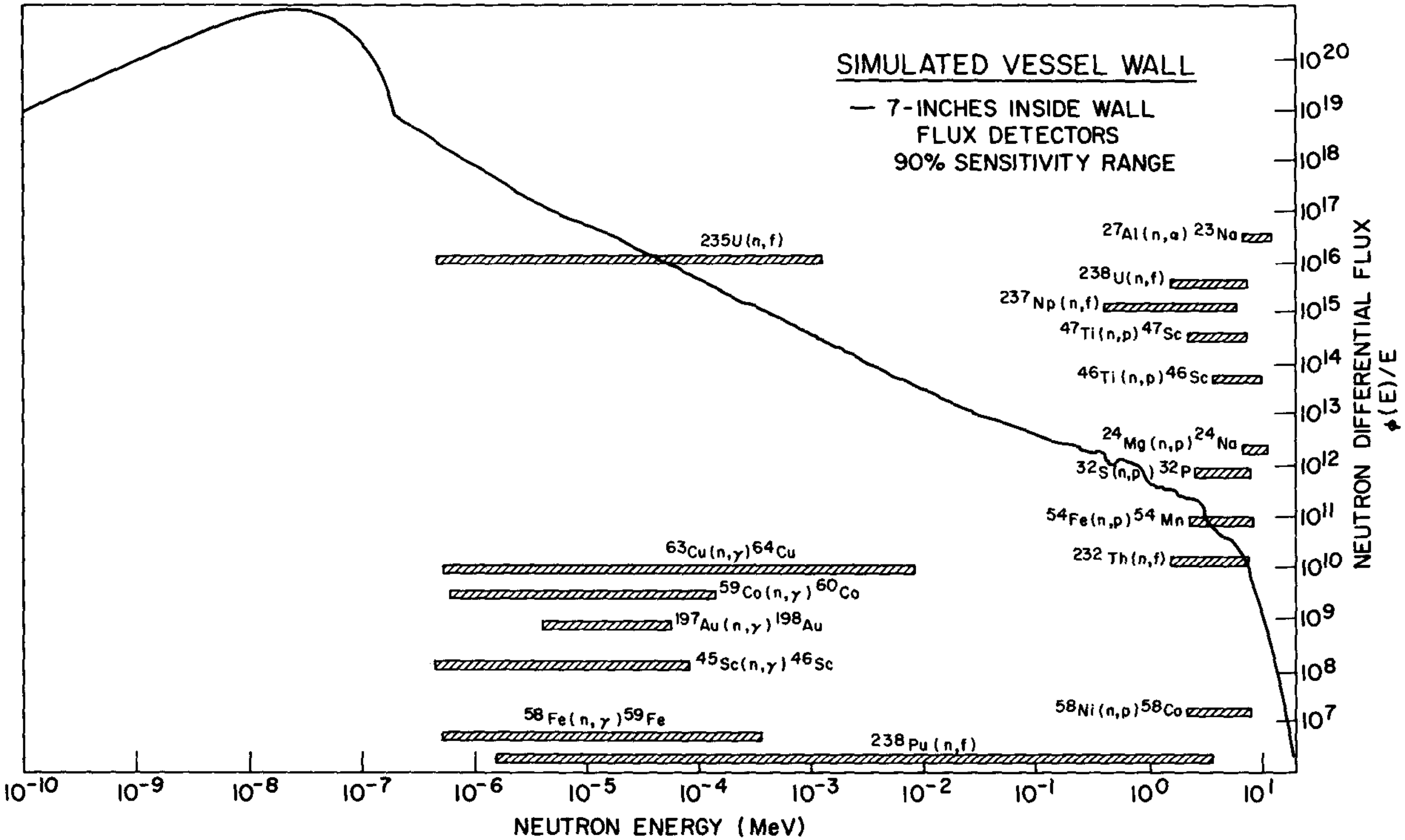


FIG. 33. Schematic representation of the energy range of neutrons effective in causing 90% of the activation of a series of materials used for neutron flux detection. The reference spectrum represents deep penetration into a pressure vessel wall [8].

Designation: E 1297 – 02

Standard Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Niobium[1]

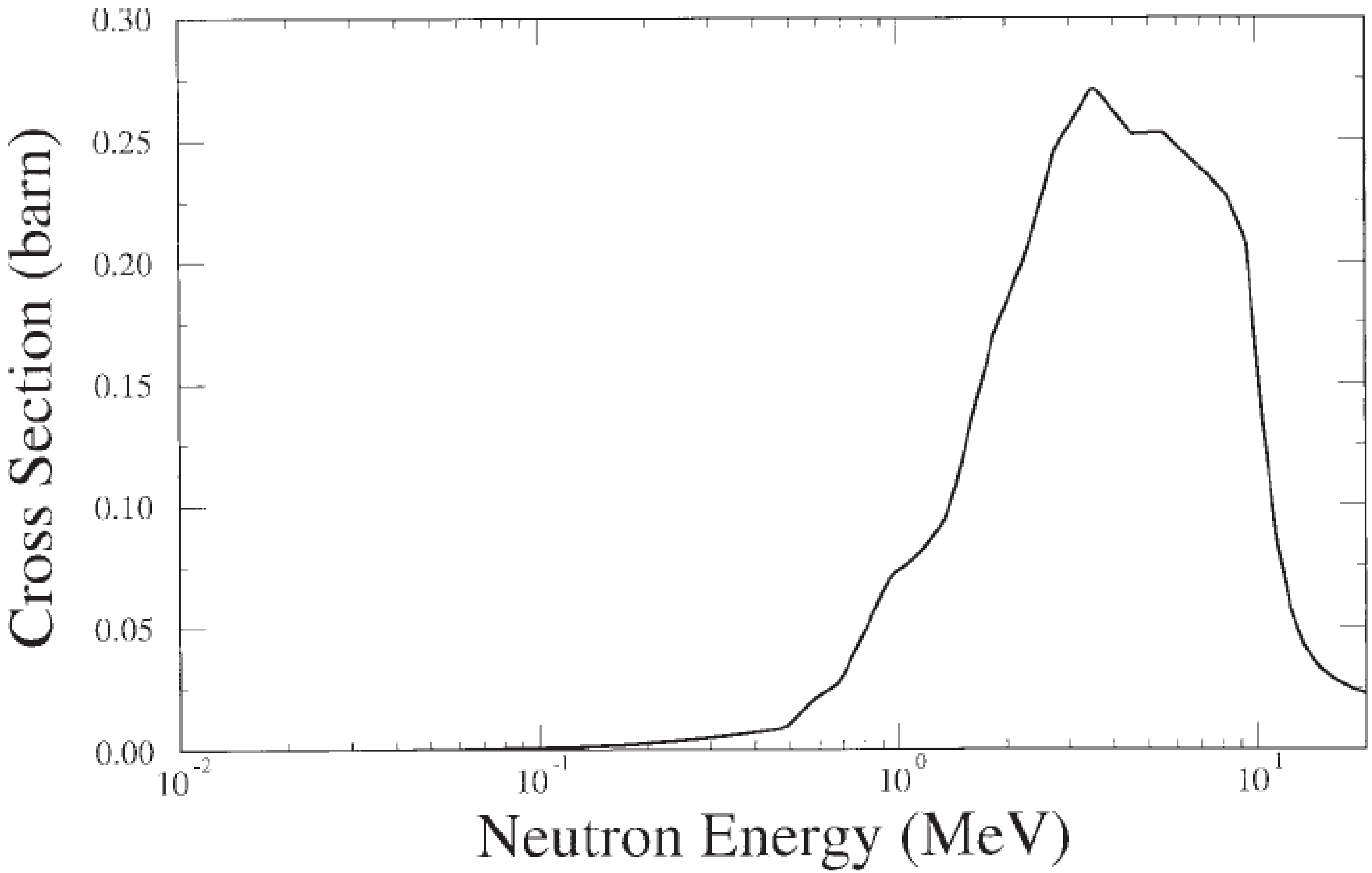


FIG. 1 IRDF-90 Cross Section Versus Energy for the $^{93}Nb(n,n')^{93m}Nb$ Reaction

NEUTRON FLUENCE INDEXATION

- ENGINEERING APPROACH USES INDEXATION FOR PRACTICAL PURPOSES – i.e. NEUTRON FLUX AND NEUTRON FLUENCE IS DETERMINED/CALCULATED FOR ENERGIES HIGHER THAN SOME INDEX/LIMIT
- THIS LIMIT IS SUPPOSED TO BE A MINIMUM NEUTRON ENERGY FOR SUBSTANTIAL RADIATION DAMAGE IN MATERIALS, i.e. FOR CREATION OF LARGE CASCADES OF KNOCK-ON ATOMS (AND HAS NOTHING TO DO WITH THRESHOLD ENERGIES OF ACTIVATED MONITORS)
- THIS ENERGY FOR RPV MATERIALS WAS CHOSEN BY DIFFERENT WAYS:
  - FOR LWR (PWR AND BWR) : En = 1 MeV
  - FOR VVER : En = 0.5 MeV

RELATION BETWEEN FLUXES/FLUENCES WITH THESE TWO ENERGIES DEPENDS ON REACTOR TYPE AND LOCATION, BUT FOR SURVEILLANCE/RPV POSITION IS USUALLY APPROXIMATELY:

$$\Phi(E_n \geq 0.5\ \text{MeV}) / \Phi(E_n \geq 1\ \text{MeV}) \approx 1.6$$

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ACTIVITY OF MONITORS

- ACTIVITY OF MONITORS ARE A STARTING POINT FOR DETERMINATION OF NEUTRON ENERGY SPECTRUM, NEUTRON FLUX AND NEUTRON FLUENCES
- ACTIVITY OF MONITORS IS CALCULATED FROM MONITOR CONSTANT (HALF-LIFE) AND NUMBER OF COUNTS MEASURED DURING A CERTAIN PERIOD
- THIS MEASUREMENT MUST BE PERFORMED ON A STANDARD EXPERIMENTAL FACILITY WITH FIXED GEOMETRY, USING STANDARDIZED/VALIDATED COUNTING EQUIPMENT
- IMPORTANT PARAMETERS ARE:
  - MASS OF THE MONITOR
  - TIME OF IRRADIATION
  - TIME DEPENDENCE OF REACTOR OUTPUT
  - TIME SINCE THE END OF IRRADIATION

Designation: E 261 – 03

Standard Practice for Determining Neutron Fluence, Fluence Rate, and Spectra by Radioactivation Techniques[1]

9.1 The activity of the sample, A, at the end of the exposure period is calculated as follows:

$$A = \lambda D/[(1 - \exp(-\lambda\, t_c))\exp(-\lambda\, t_w)] \qquad (1)$$

where:

λ = decay constant for the radioactive nuclide,

t_c = time interval for counting,

t_w = time elapsed between the end of the irradiation period and the start of the counting period, and

D = number of disintegrations (net number of counts corrected for background, random and true coincidence losses, efficiency of the counting system, and fraction of the sample counted) in the interval t_c.

9.1.1 If, as is often the case, the counting period is short compared to the half-life (= 0.693/λ) of the radioactive nuclide, the activity is well approximated as follows:

$$\boxed{A = D/[t_c \exp(-\lambda\, t_w)]} \qquad (2)$$

9.2 For irradiations at constant fluence rate, the saturation activity, A_s, is calculated as follows:

$$\boxed{A_s = A/(1 - \exp - \lambda' t_i)} \qquad (3)$$

where:

t_i = exposure duration, and
λ' = effective decay constant during the irradiation.

9.3 The reaction rate is calculated as follows:

$$R_s = A_s \lambda' / N\lambda \qquad (5)$$

where:

N = number of target nuclei in the detector at time of irradiation.

9.3.1 The number of target nuclei can often be assumed to be equal to N_o, the number prior to irradiation.

$$N_o = N_A F m / M \qquad (6)$$

where:

N_A = Avogadro's number
= 6.022×10^{23} mole^{-1},

F = atom fraction of the target nuclide in the target element,

m = mass of target element, g, and

M = atomic mass of the target element.

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Designation: E 261 – 03

Standard Practice for Determining Neutron Fluence, Fluence Rate, and Spectra by Radioactivation Techniques[1]

9.4 The neutron fluence rate, ϕ, is calculated as follows:

$$\phi = R_s/\bar{\sigma} \tag{7}$$

where:

$\bar{\sigma}$ = the spectral weighted neutron activation cross section.

9.5 The neutron fluence, Φ, is related to the time varying differential neutron fluence rate $\phi(E,t)$ by the following expression:

$$\Phi = \int_0^\infty \int_{t_1}^{t_2} \phi\,(E,t)\,\mathrm{d}t\,\mathrm{d}E \tag{8}$$

where:

$t_2 - t_1$ = duration of the irradiation period.

Designation: E 261 – 03 Standard Practice for Determining Neutron Fluence, Fluence Rate, and Spectra by Radioactivation Techniques[1]

9.5.2 It is usual to assume, however, that the neutron fluence rate is directly proportional to reactor power; under these conditions, the fluence can be well approximated by:

$$\Phi = \left(\frac{\phi}{P}\right) \sum_{i=1}^{n} P_i t_i \tag{9}$$

where:

ϕ/P = average value of the neutron fluence rate, ϕ, at a reference power level, P,

t_i = duration of the i^{th} operating period during which the reactor operated at approximately constant power, and

P_i = reactor power level during that operating period.

9.7.2 The total irradiation period can be divided into a continuous series of periods during each of which $\phi(E)$ is essentially constant. Then the activity generated during the i^{th} irradiation period is:

$$A_i = [\lambda N_i (R_s/\lambda')_i](1 - \exp(-\lambda'_i t_i)) \qquad (15)$$

where:

N_i = number of target atoms, and

t_i = duration of the i^{th} period.

9.7.2.1 The activity remaining from the i^{th} period at the end of the n^{th} period can be calculated as the following equation:

$$(A_n)_i = A_i \exp\left(-\sum_{j=i+1}^{n} \lambda'_j t_j\right) \qquad (16)$$

9.7.2.2 The total activity of the foil at the end of the irradiation duration is thus the sum of all the $(A_n)_i$ terms.

9.7.3.2 It is normally assumed that the fluence rate is directly proportional to the power generation rate in the adjacent fuel.

9.7.4 Under the conditions assumed in 9.7.3, Eq 15 can be written as:

$$A_i = A_s (P_i/P) (1 (- \exp (- \lambda' t_i)), \tag{17}$$

and Eq 16 can be written as:

$$(A_n)_i = A_s \left(\frac{N_i}{N_o}\right) K_i (1 - \exp (-\lambda' t_i)) \tag{18}$$

where:

A_s = the saturation activity corresponding to a reference power level, P,

P_i = actual power generation rate during the irradiation period,

K_i = $(P_i/P) \exp\left(-\lambda' \sum_{j=i+1}^{n} \left[1 + \frac{P_j}{P}\left(\frac{\lambda'}{\lambda} - 1\right)\right] t_j\right)$, and

N_i = $N_o \exp\left(- R_s \sum_{j=1}^{i-1} \frac{P_j}{P} t_j\right)$.

DETERMINATION OF NEUTRON ENERGY SPECTRUM

- NEUTRON FLUENCE CAN BE DETERMINED ONLY WITH THE PROPER KNOWLEDGE OF NEUTRON ENERGY SPECTRUM

- NEUTRON ENERGY SPECTRUM CAN BE DETERMINED BY:
  - CALCULATIONS THAT MUST BE VALIDATED BY EXPERIMENTS
  - EXPERIMENTALLY ON MOCK-UP/ZERO POWER REACTOR (DIRECT MEASUREMENTS USING e.g. Ge DETECTORS)
  - EXPERIMENTALLY USING A SET OF ACTIVATION/FISSION MONITORS AND UNFOLDING CODE

DETERMINATION OF NEUTRON ENERGY SPECTRUM

- DETERMINATION OF NEUTRON ENERGY SPECTRUM FROM MEASUREMENT OF ACTIVITIES OF A SET O MONITORS IS USUALLY PERFORMED BY APPLICATION OF SO-CALLED „UNFOLDING METHOD"
- THIS METHOD REPRESENTS SOME SORT OF ITERATIONS OR STATISTICAL EVALUATIONS. IT STARTS WITH AN „INPUT SPECTRUM" THAT COULD BE EITHER FISSION ONE OR A SPECTRUM THAT WAS OBTAINED FROM CALCULATIONS OR OTHER SIMILAR MEASUREMENTS
- COMPUTER PROGRAMME BY ITERATION OR STATISTICAL METHODS TRIES TO FIND THE SPECTRUM THAT FITS BEST TO THE MEASURED ACTIVITIES OF THIS SET OF MONITORS
- RESULT OF THIS COMPUTATION VERY MUCH DEPENDS ON ITS INPUT SPECTRUM ESPECIALLY WHEN NUMBER OF MONITORS IS SMALL OR THEY COVER ONLY A LIMITED AREAS OF ENERGY SPECTRUM (e.g. MONITORS WITH HIGH THRESHOLD ENERGIES)

DETERMINATION OF NEUTRON ENERGY SPECTRUM

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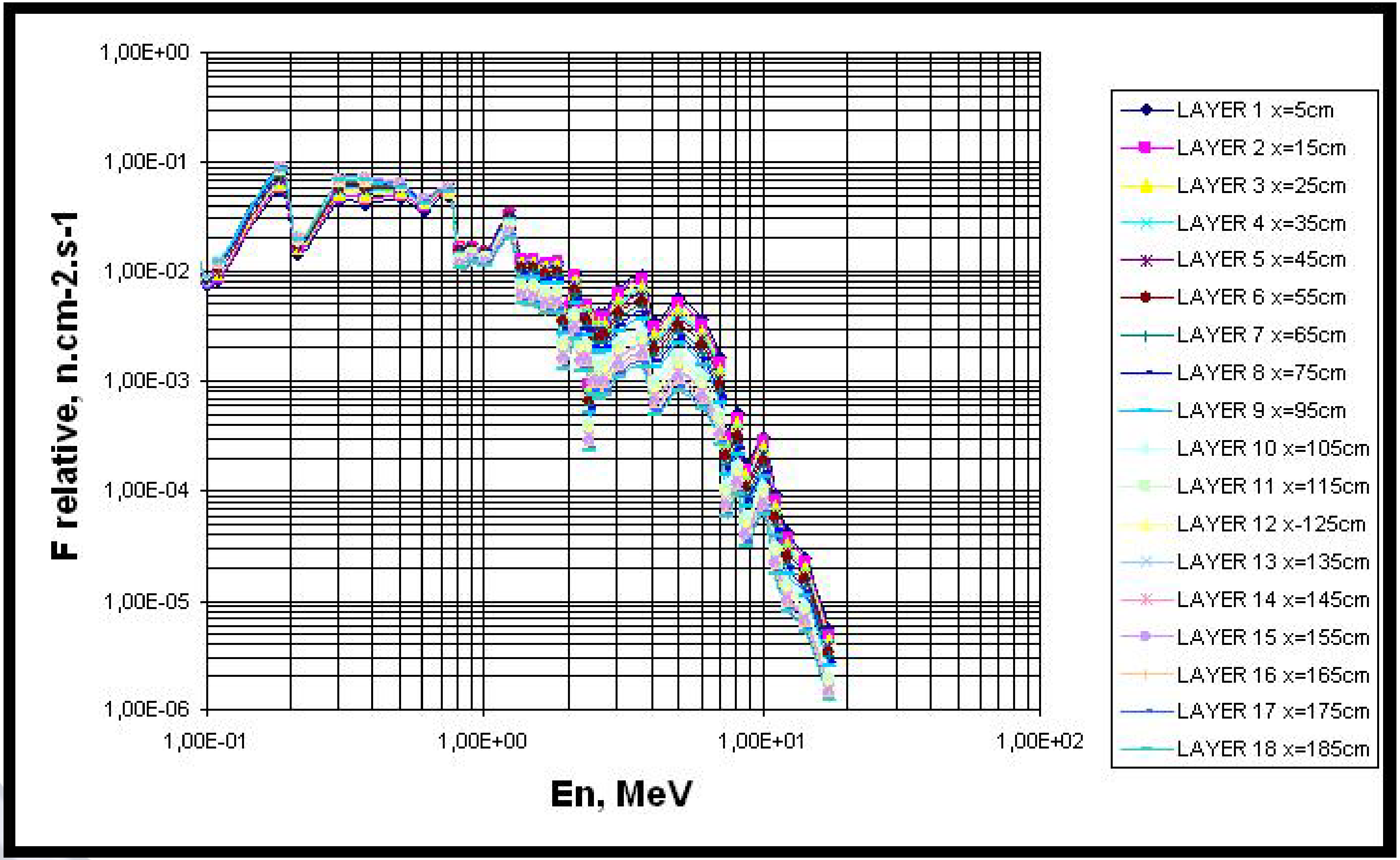
E 944 – 02

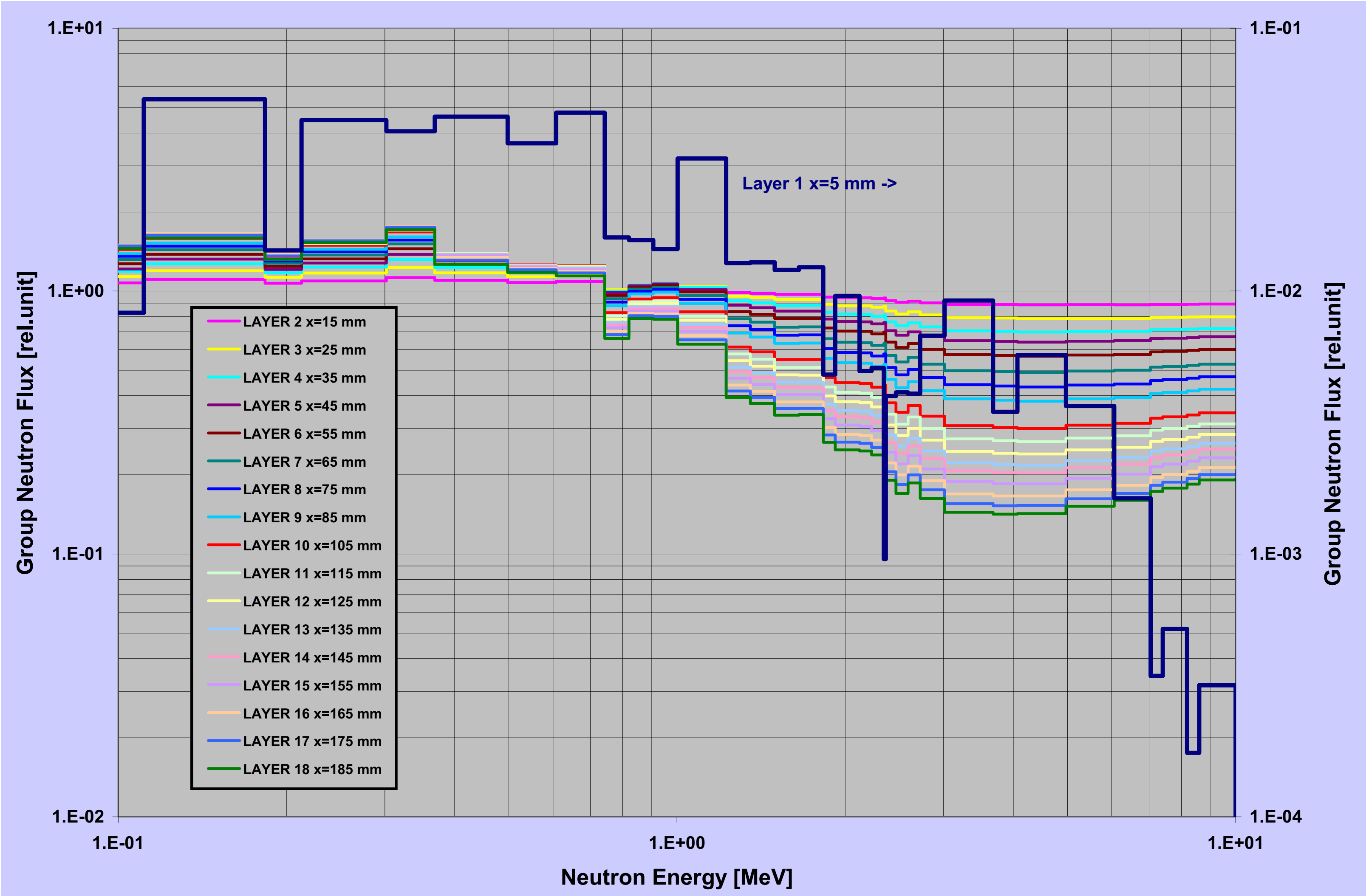
TABLE 1 Available Unfolding Codes

Program	Solution Method	Code Available From	References	Comments
SAND-II	semi-iterative	RSIC Prog. No. CCC-112, CCC-619, PSR-345	1	contains trial spectra library. No output uncertainties in the original code, but modified Monte Carlo code provides output uncertainties (**12, 22**)
SPECTRA	statistical, linear estimation	RSIC Prog. No. CCC-108	**2, 3**	minimizes deviation in magnitude, no output uncertainties.
IUNFLD/ UNFOLD	statistical, linear estimation		**5**	constrained weighted linear least squares code using B-spline basic functions. No output uncertainties.
WINDOWS	statistical, linear estimation, linear programming	RSIC Prog. No. PSR-136, 161	**6**	minimizes shape deviation, determines upper and lower bounds for integral parameter and contribution of foils to bounds and estimates. No statistical output uncertainty.
RADAK, SENSAK	statistical, linear estimation		**7, 8**	RADAK is a general adjustment code not restricted to spectrum adjustment.
STAY'SL	statistical linear estimation	RSIC Prog. No. PSR-113	**9**	permits use of full or partial correlation uncertainty data for activation and cross section data.
NEUPAC(J1)	statistical, linear estimation	RSIC Prog. No. PSR-177	**10, 11**	permits use of full covariance data and includes routine of sensitivity analysis.
FERRET	statistical, least squares with log normal a priori distributions	RSIC Prog. No. PSR-145	**12, 22**	flexible input options allow the inclusion of both differential and integral measurements. Cross sections and multiple spectra may be simultaneously adjusted. FERRET is a general adjustment code not restricted to spectrum adjustments.
LEPRICON	statistical, generalized linear least squares with normal a priori and a posteriori distributions	RSIC Prog. No. PSR-277	**14, 15, 18**	simultaneous adjustment of absolute spectra at up to two dosimetry locations and one pressure vessel location. Combines integral and differential data with built-in uncertainties. Provides reduced adjusted pressure vessel group fluence covariances using built-in sensitivity database.
LSL-M2	statistical, least squares, with log normal a priori and a posteriori distributions	RSIC Prog. No. PSR-233	**19**	simultaneous adjustment of several spectra. Provides covariances for adjusted integral parameters. Dosimetry cross-section file included.

TYPICAL SPECTRUM FOR VVER-1000 WALL THICKNESS = 190 mm



TYPICAL SPECTRUM FOR VVER-1000 WALL THICKNESS = 190 mm



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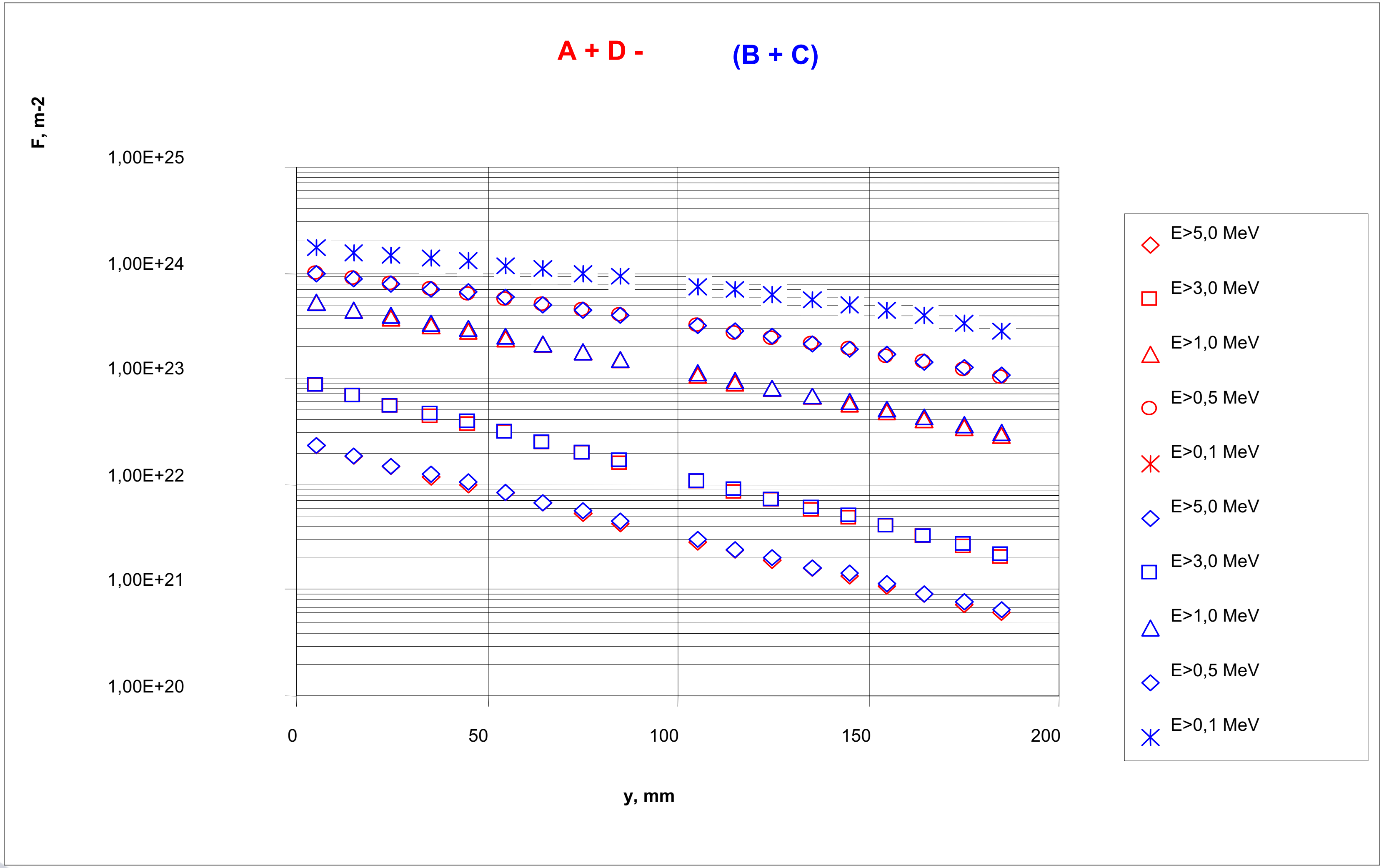
CHANGES IN FLUXES THROUGH A RPV WALL THICKNESS = 190 MM



SPECTRAL INDEXES

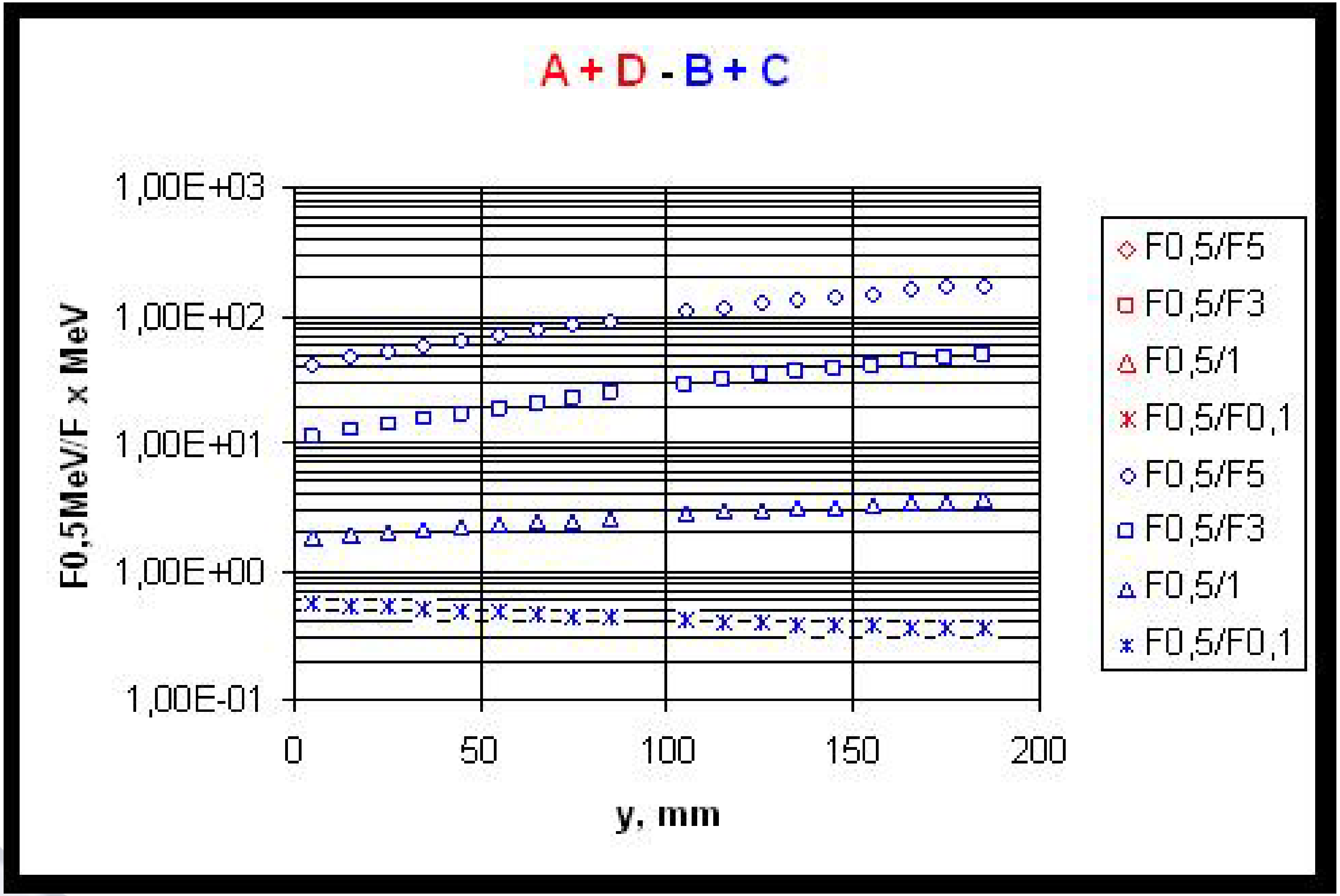
- IN DIFFERENT LOCATIONS IN THE RPVs, NEUTRON ENERGY SPECTRA ARE DIFFERENT, AS WELL AS NEUTRON FLUXES ARE CHANGED
- CHANGES IN NEUTRON ENERGY SPECTRA ALSO CHANGES VALUES OF NEUTRON FLUX DEFINED FOR DIFFERENT THRESHOLD ENERGIES OF NEUTRONS, e.g. 0.5 OR 1 MeV
- THIS EFFECT CAN BE DESCRIBED MBY SO-CALLED „SPECTRAL INDEXES“ DEFINED AS A RATIO OF TWO FLUXES WITH DIFFERENT NEUTRON ENERGY THRESHOLDS, e.g.

$$\text{SPECTRAL INDEX} = \Phi(E_n \geq 0.5\ \text{MeV})/\Phi\ (E_n \geq 1\ \text{MeV})$$

OR SOMETIMES IT IS DEFINED AS

$$SI(3\ \text{MeV})_{X\ \text{MeV}} = \Phi(E_n \geq X\ \text{MeV})/\Phi\ (E_n \geq 3\ \text{MeV})$$

SPECTRAL INDEXES



Designation: E 261 – 03

Standard Practice for
Determining Neutron Fluence, Fluence Rate, and Spectra by Radioactivation Techniques[1]

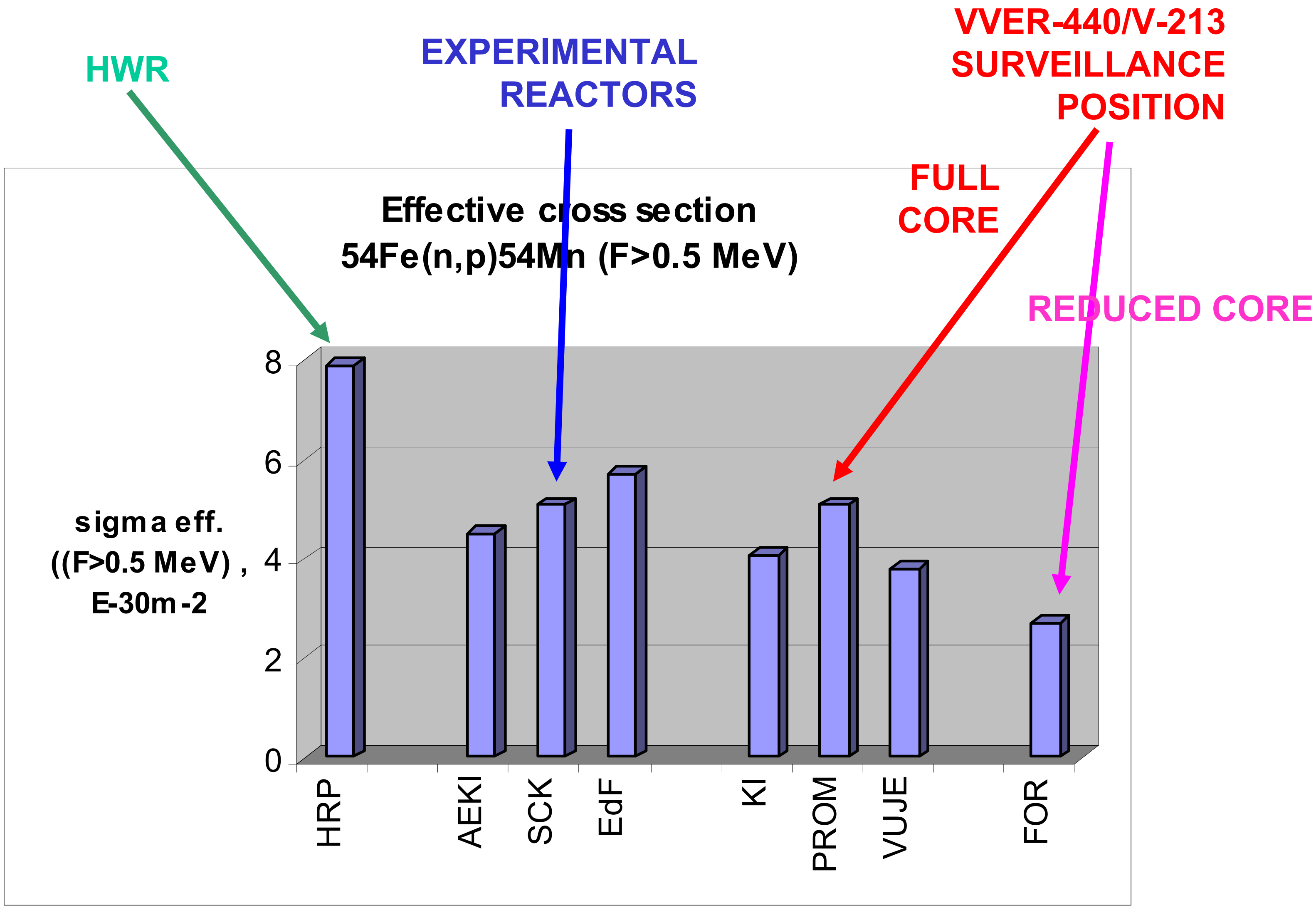
9.8 *Spectral-Averaged Cross Sections:*

9.8.1 As a general practice, the spectral-averaged cross sections will be used in these calculations. Since a spectral-averaged cross section is defined as follows:

$$\bar{\sigma} = \frac{\int_0^\infty \sigma(E)\,\phi(E)\,dE}{\int_0^\infty \phi(E)\,dE} \qquad (19)$$

the differential cross section of the nuclide and the neutron spectrum over the neutron energy range for which the nuclide has an effective cross section must be known. When cross-section and spectrum information are not available alternative procedures may be used; suggested alternatives are discussed in 9.11-9.13, and in the methods for individual detectors.

EXAMPLE OF EFFECTIVE CROSS SECTIONS



Designation: E 261 – 03 Standard Practice for Determining Neutron Fluence, Fluence Rate, and Spectra by Radioactivation Techniques[1]

9.10 *Neutron Spectra*:

9.10.1 A reactor neutron spectrum can be considered as being divided into three idealized energy ranges describing the neutrons as thermal, resonance or epithermal, and fast. Since these ranges have distinctive distributions, they are a natural division of neutrons by energy for thermal reactor spectra.

9.11 *Thermal-Neutron Fluence Rate:*

9.11.1 A solution of the activation equation, Eq 11, leads to the following result:

$$R_s = (nv)\sigma_{eff} \quad (22)$$

where:

(nv) = true thermal-neutron fluence rate;
n = neutron density, and
σ_{eff} = effective cross-section value that will give the correct activation.

It has become conventional to tabulate cross sections for thermal neutrons as the value for a neutron velocity of V_o = 2200-m/s (see Table 1). This is the most probable velocity of the Maxwellian distribution for a standard temperature whose value is 20.44°C (293.6 K).

9.12 *Resonance-Neutron Fluence Rate*:

9.12.1 In this section, we consider the detection of neutrons with energies extending from those of thermal neutrons to about 0.1 MeV. These neutrons are called resonance neutrons or epithermal neutrons. In this range of energies, the neutron absorption may be divided into two parts. For the first, the cross section varies as the reciprocal of the neutron velocity. The second is "resonance absorption," that is characterized by a large increase in cross section over a narrow energy range.

9.13 *Fast-Neutron Fluence Rate*:

9.13.1 The energy at which to separate "fast neutrons" from "resonance energy neutrons" is rather arbitrarily chosen here to be 0.1 MeV. The spectral shape as given by the differential fluence rate, $\phi(E)$, can in principle be determined from the measured reaction rates of several detectors that are activated by different parts of the neutron energy spectrum. The effective cross section $\bar{\sigma}_j$ is calculated from Eq 19 for a known spectrum similar to the spectrum for the unknown field being measured;

9.13.2 An alternate procedure is to consider the detectors as having threshold properties. For an ideal threshold detector, the cross section for activation is a step-function; that is, it is zero for neutrons with energies below a certain energy E_i (the "threshold energy") and constant for neutron energies above E_i. The constant value at energies above E_i is the "threshold cross section," $\bar{\sigma}_j(E > E_i)$. Then the effective threshold cross section for the assumed spectrum is given as follows:

$$\bar{\sigma}_j(E > E_i) = [\int_0^\infty \sigma(E)\phi(E)\,dE] / [\int_{E_i}^\infty \phi(E)\,dE] \qquad (36)$$

The integral in the denominator of this equation is the integral neutron fluence rate with energies above E_i, $\phi(E > E_i)$. Hence, the integral fluence rate above energy E_i is given, in this ideal case, as follows:

$$\phi(E > E_i) = R_s/\bar{\sigma}(E > E_i) \qquad (37)$$

where, as before, R_s, the reaction rate, is determined experimentally from Eq 5. If $\phi(E > E_i)$ is determined for a number of detectors, that is for a number of values of E_i, the differential fluence rate, $\phi(E)$, can be deduced by differentiating the curve of $\phi(E > E_i)$ versus E_i.

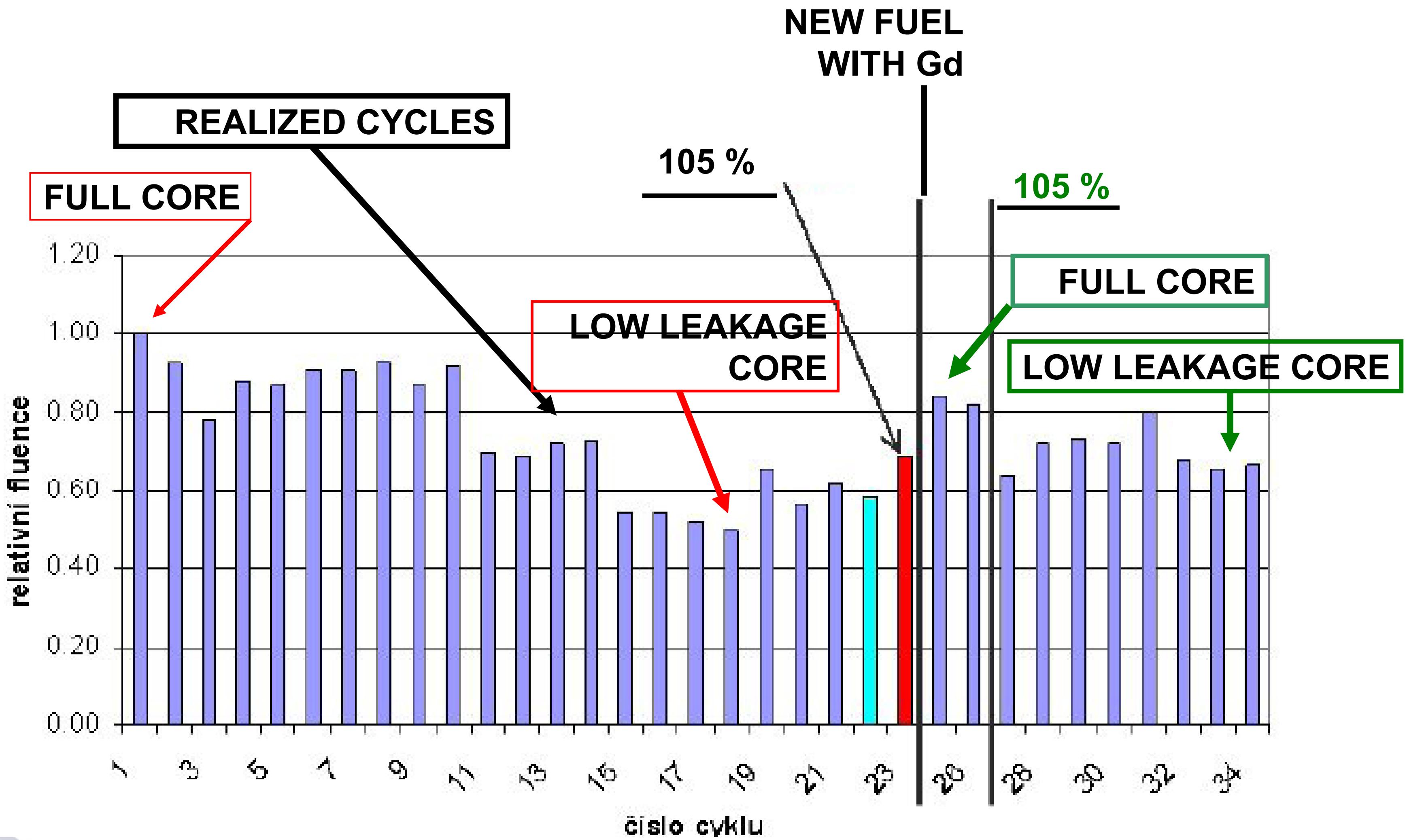
NEUTRON FLUENCE

- NEUTRON FLUENCE IS THEN DETERMINED FROM THE NEUTRON FLUX AND TIME OF IRRADIATION (SUMMATION OF INDIVIDUAL IRRADIATION PERIODS)
- NEUTRON FLUENCE USUALLY DEPENDS LINEARLY ON REACTOR OUTPUT UNDER NORMAL OPERATION CONDITIONS
- INCREASE IN REACTOR OUTPUT BY CHANGES OF THE REACTOR CORE (DIFFERENT FUEL ELEMENTS, INCREASE OF REACTOR OUTPUT BY OF HEAT PRODUCTION etc.) CAN EITHER BE LINEARLY DEPENDENT WITH REACTOR OUTPUT INCREASE OR, IS CASES WITH CHANGES OF FUEL RE-LOADING SCHEME, BE SUBSTANTIALLY HIGHER

NEUTRON FLUENCE

- NEUTRON FLUX ON RPV AND/OR SURVEILLANCE SPECIMENS DEPENDS STRONGLY ON:
  - DESIGN OF THE RPV WITH RESPECT THICKNESS OF THE WATER MODERATOR
  - DESIGN OF SURVEILLANCE SPECIMEN CAPSULES AND THEIR LOCATION
  - FUEL LOADING SCHEME
    - FULL CORE
      - ALL FUEL ELEMENTS ARE FRESH OR NEW OR PARTIALLY BURNT FUELD ELEMENTS ARE LOADED EVEN INTO PERIPHERY REGION
    - LOW-LEAKAGE CORE
      - ONLY PARTIALLY BURNT (IN THEIR LAST CYCLE) FUEL ELEMENTS ARE LOADED INTO PERIPHERY
    - REDUCED CORE
      - DUMMY ELEMENTS (I.E. ELEMENTS WITH SHIELDING MATERIALS LIKE AUSTENITIC STEEL ARE LOADED INTO PERIPHERY REGION OF ACTIVE CORE)

TYPICAL NEUTRON FLUENCE OSCILATIONS



EFFECT OF DUMMY ELEMENTS

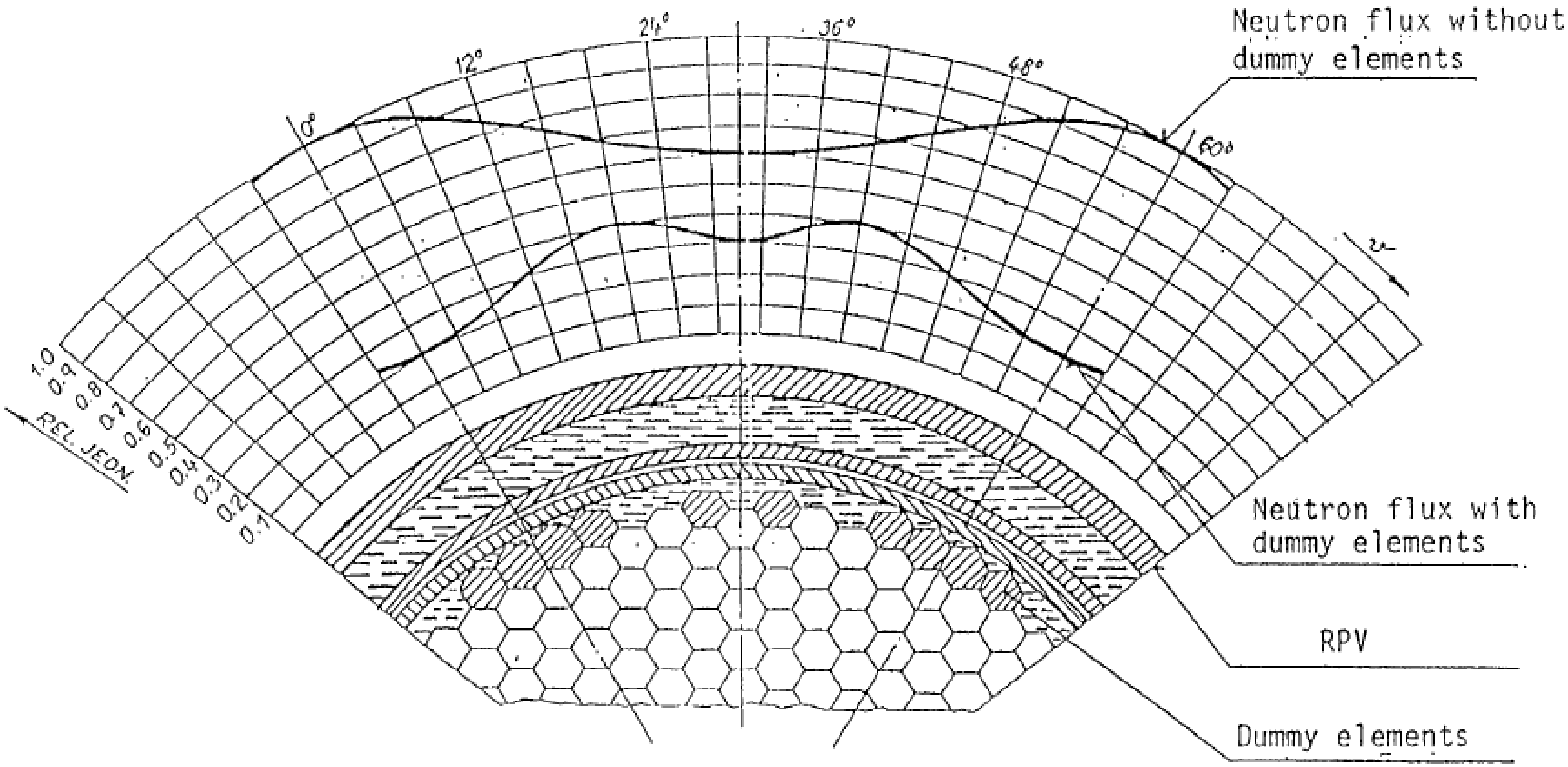


FIG. 43. WWER Flux distributions in low leakage cores.

EFFECT OF DUMMY ELEMENTS AND ANNEALING

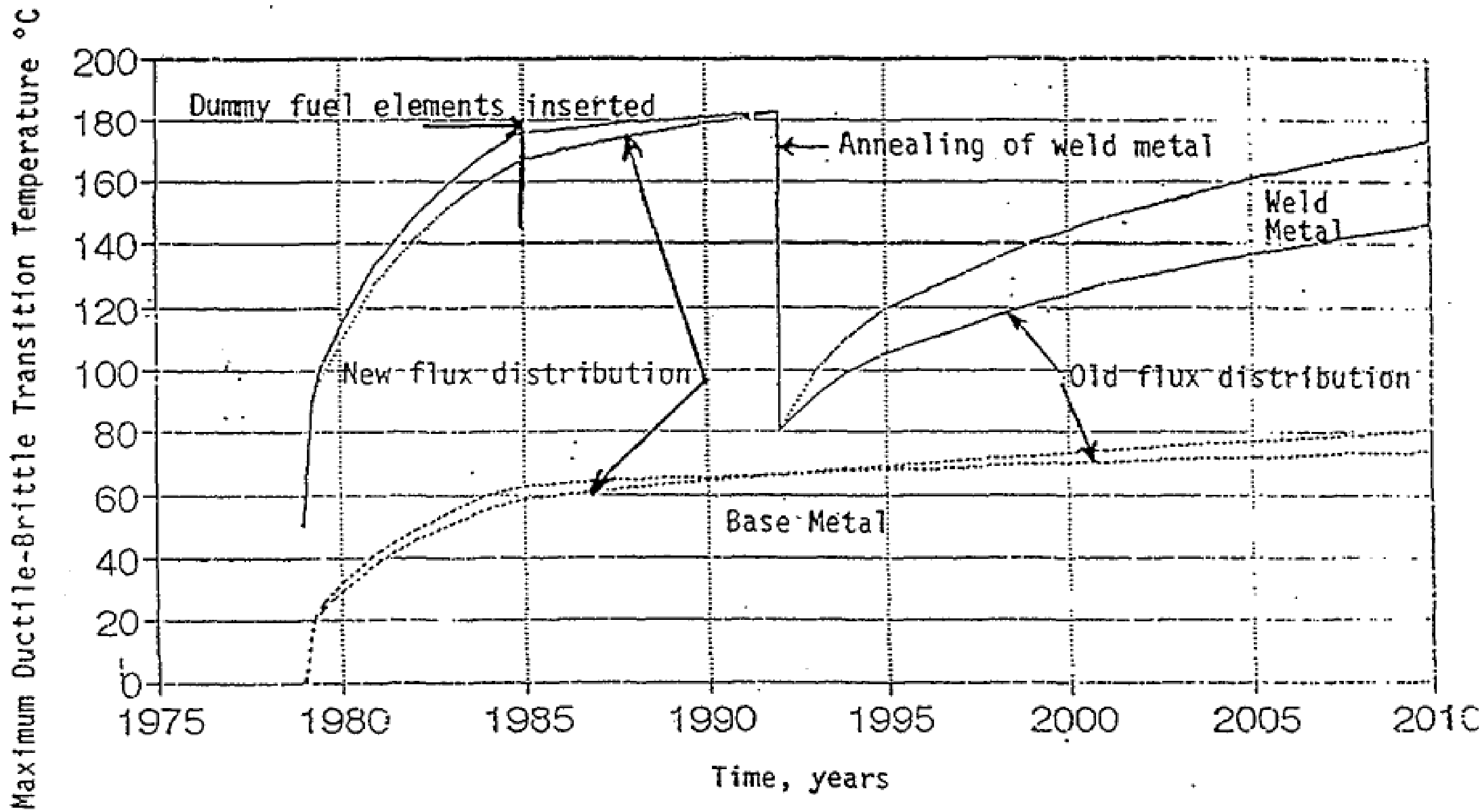


FIG. 25. Transition temperature values as a function of operation time for Bohunice Unit 2. The upper lines are the weld material and the lower lines are the base metal material.

Designation: E 170 – 05

Standard Terminology Relating to Radiation Measurements and Dosimetry[1]

displacements per atom (dpa)—the mean number of times each atom of a solid is displaced from its lattice site during an exposure to displacing radiation, as calculated following standard procedures (see **displacement dose**).

FOR LWR – SURVEILLANCE POSITION AND RPV:
10^{23} m^{-2} ≈ 0.015 dpa

dpa

- IN SOME CASES, dpa IS USED AS A MEASURE OF RADIATION DAMAGE OF MATERIALS
- THIS PARAMETER OF NEUTRON FIELD IS MOSTLY APPLIED IN THE FOLLOWING CASES:
  - NEUTRON EXPOSURE OF RPV MATERIALS IN GCR (DIFFERENT NEUTRON SPECTRA)
  - NEUTRON EXPOSURE OF MATERIALS OF INTERNALS (VERY HIGH NEUTRON FLUENCES, SPECTRUM VERY CLOSE TO FISSION SPECTRUM)
  - NEUTRON EXPOSURE OF MATERIALS IRRADIATED BY PARTICLES (IN ACCELERATORS)
  - COMPARISON OF RADIATION EXPOSURE OF MATERIALS BY NEUTRONS AND BY PARTICLES
  - PRACTICALLY FOR ALL MATERIALS IN GEN IV REACTORS (FAST AND THERMAL)
-

ASTM Designation: E 693 – 01 Standard Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E 706(ID)[1]

5.1 The displacement rate at time t is calculated as follows:

$$\text{dpa/s} = \int_0^\infty \sigma_d(E)\phi(E,t)\,\mathrm{d}E \quad (1)$$

where:

$\sigma_d(E)$ = the displacement cross section for a particular material, and

$\phi(E,t)\,\mathrm{d}E$ = the fluence rate of neutrons in the energy interval E to $E + dE$.

5.2 The exposure index, dpa, is then the time integrated value of the displacement rate, calculated as follows:

$$\text{dpa} = \int_0^{t_r} \phi_{tot}(t) \int_0^\infty \sigma_d(E)\psi(E,t)\,\mathrm{d}E\,\mathrm{d}t \quad (2)$$

where:

$\phi_{tot}(t)$ = the time dependent fluence rate intensity, and

$\psi(E,t)$ = the fluence rate spectrum normalized to give unit integral fluence rate at any time when integrated over energy.

ASTM Designation: E 693 – 01

Standard Practice for
Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E 706(ID)[1]

- THE LINHARD MODEL OF ENERGY PARTITION BETWEEN ATOMS AND ELECTRONS AND THE NORGETT-ROBINSON-TORRENS (NRT) RECOMMENDED CONVERSION OF DAMAGE ENERGY TO DISPLACEMENTS WITH AN EFFECTIVE DISPLACEMENT THRESHOLD ENERGY OF E_d = 40 Ev (FOR IRON) AND AN ATOMIC SCATTERING CORRECTION FACTOR OF β = 0.8 IS USED FOR CALCULATION OF NUMBER OF DISPLACEMENT N_d, CORRESPONDING TO A GIVEN DAMAGE ENERGY, T_d, THROUGH THE EQUATION

$$N_d(T_d) = \left[\begin{array}{ll} 0 & T_d < E_d \\ 1 & E_d \le T_d < 2\,E_d/\beta \\ \dfrac{\beta T_d}{2E_d} & 2E_d/\beta \le T_d < \infty \end{array}\right] \qquad (10)$$

Designation: E 693 – 01

Standard Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E 706(ID)[1]

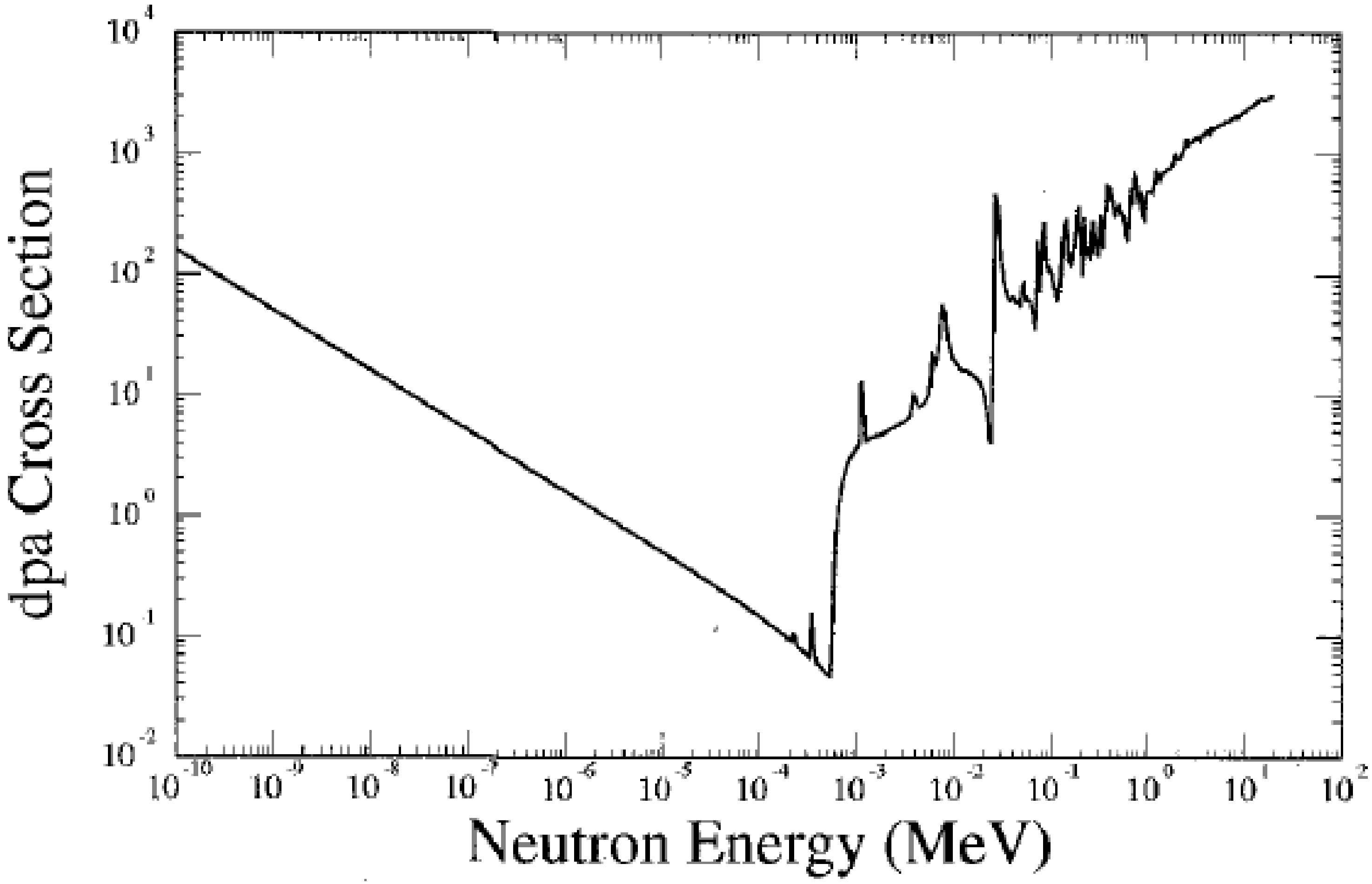


FIG. 1 ENDF/B-VI-based Iron Displacement Cross Section

dpa

- AS RPV STEELS USUALLY CONTAIN MORE THAN 95 % OF IRON, dpa FOR STEELS IS CALCULATED USING CROSS SECTIONS GIVEN IN ASTM E 693-01
- dpa CROSS SECTIONS HAVE BEEN ALSO DETERMINED FOR SOME OTHER ELEMENTS – CHROMIUM AND NICKEL – TO BE APPLIED MAINLY FOR AUSTENITIC MATERIALS WITH HIGH CONTENT OF BOTH ELEMENTS (18/8, 25/10 etc.)

dpa

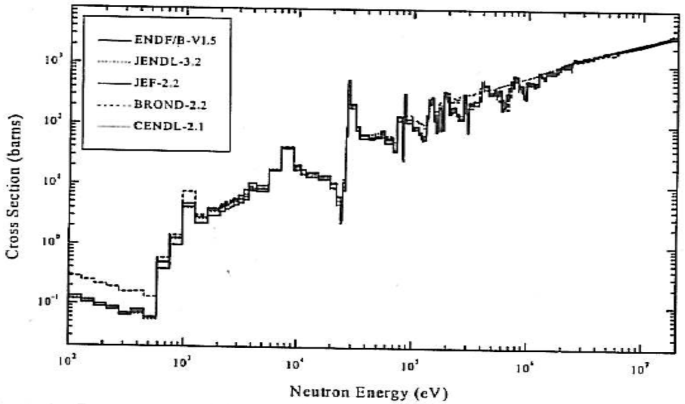


Figure 1. *Comparison of Energy Dependent DPA Cross Sections on Iron*

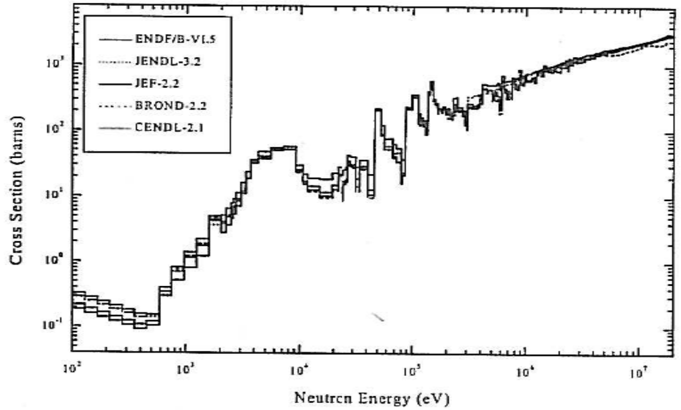


Figure 2. *Comparison of Energy Dependent DPA Cross Sections on Chrome*

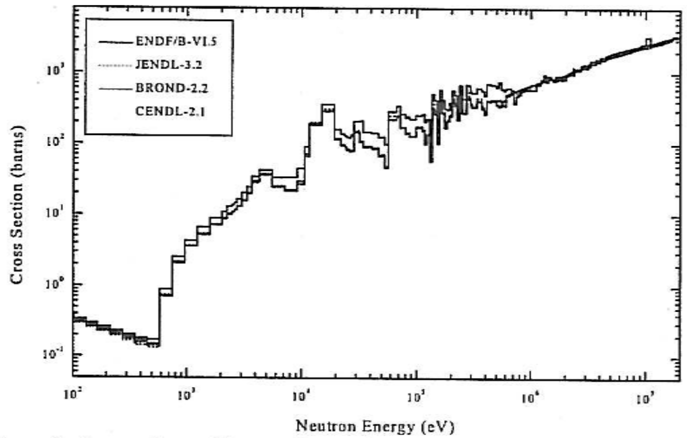


Figure 3. *Comparison of Energy Dependent DPA Cross Sections on Nickel*

COMPARISON OF dpa AND FLUENCE

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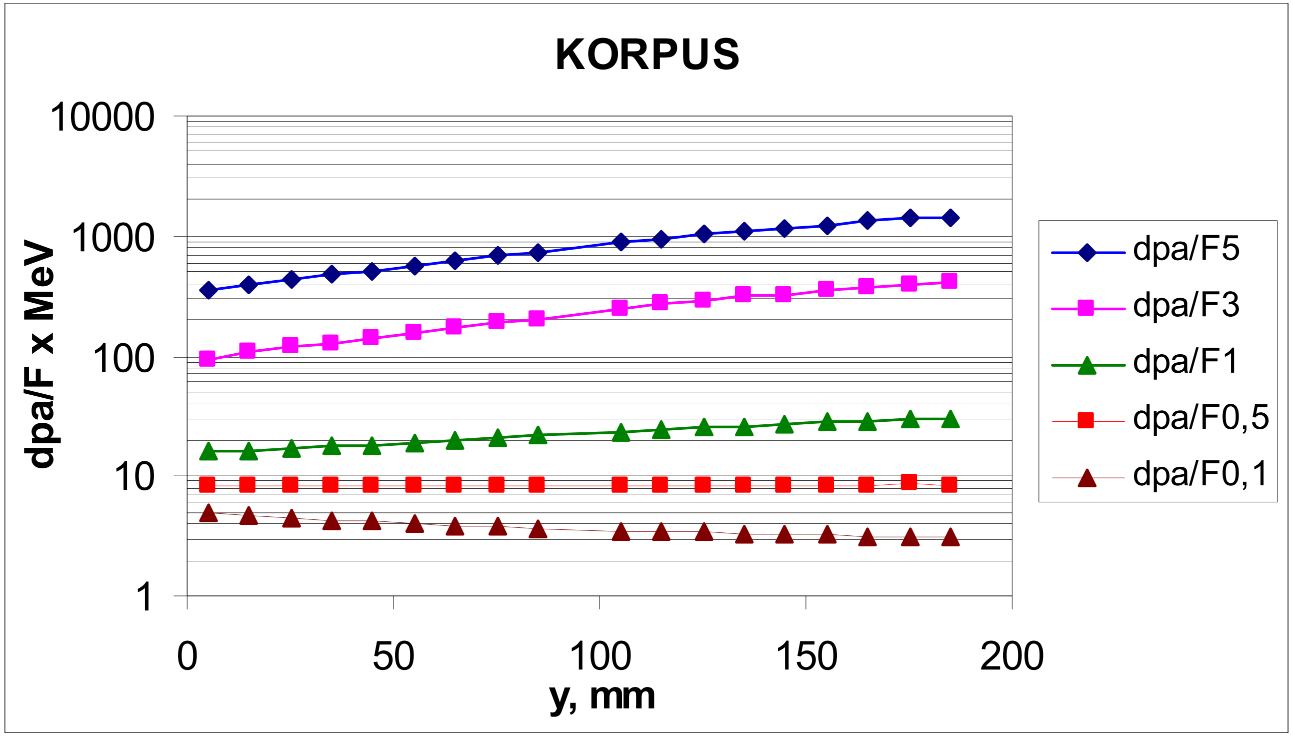
- COMPARISON OF dpa AND NEUTRON FLUENCE IN MODEL IRRADIATION EXPERIMENT WITH RPV WALL OF 190 mm THICKNESS SHOWED THAT THE CLOSEST INDEXATION OF NEUTRON FLUENCE TO THE dpa VALUES ARE THRESHOLD ENERGY E_n = 0.5 MeV AS THEIR RATIO REMAINS PRACTICALLY CONSTANT THROUGH THE WHOLE THICKNESS ON THE CONTRARY WITH OTHER INDEX ENERGIES



REQUIREMENTS FOR UNCERTAINTY

- ALL EXPERIMENTS AS WELL AS CALCULATIONS ARE CONNECTED WITH SOME UNCERTAINTY
- THIS UNCERTAINTY MAINLY DEPENDS ON:
  - UNCERTAINTY IN NEUTRON CROSS SECTIONS
  - LIMITED NUMBER OF NEUTRON MONITORS
  - LIMITATIONS IN MODELLING OF REACTOR ACTIVE CORE etc.
- UNCERTAINTY OF NEUTRON FLUENCE DETERMINATION IS A CRUCIAL PARAMETER FOR A RELIABLE AND EFFECTIVE:
  - CREATION OF A DATABASE FOR EVALUATION OF PREDICTIVE FORMULAE OF RADIATION DAMAGE
  - COMPARISON OF ANY EXPERIMENTAL DATA
  - COMPARISON OF CALCULATED AND EXPERIMENTAL DETERMINED FLUENCES
  - PREDICTION OF SAFE OPERATION LIFETIME

Designation: E 944 – 02

Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIA)[1]

5.4 *Output Uncertainties*—of damage exposure values depend on the accuracy of the fluence calculation and dosimetry measurements and on the selection of dosimetry sensors (see Ref (**20**)). The achievable accuracy depends on the neutron field under investigation. Assuming due care in calculations and measurements, the following output variances can be expected for the damage exposure parameters $\Phi > 1.0$ MeV, $\Phi > 0.1$ MeV, and dpa (1σ):

Benchmark Fields - 5 % (see NUREG/CR-1861)

Research Reactors 5 – 15 % (see Ref (**20**))

Power Reactors 5 – 30 % (see Ref (**21**))

The quoted uncertainties are for dpa as damage exposure parameter. The uncertainties for $\Phi > 1.0$ MeV are slightly lower and for $\Phi > 0.1$ MeV slightly higher given the same input data.

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U.S. NUCLEAR REGULATORY COMMISSION March 2001

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.190

(Previous drafts were DG-1053 and DG-1025)

CALCULATIONAL AND DOSIMETRY METHODS FOR DETERMINING PRESSURE VESSEL NEUTRON FLUENCE

Fluence Calculational Uncertainty. The vessel fluence (1 sigma) calculational uncertainty must be demonstrated to be $\leq$20% for RT_{PTS} and RT_{NDT} determination. In these applications, if the benchmark comparisons indicate differences greater than 20%, the calculational model must be adjusted or a correction must be applied to reduce the difference between the fluence prediction and the upper 1-sigma limit to within 20%. For other applications, the accuracy should be determined using the approach described in Regulatory Position 1.4, and an uncertainty allowance should be included in the fluence estimate as appropriate in the specific application.

MATERIAL MONITORS

- ADDITIONALLY TO NEUTRON FLUENCE MONITORS, SO-CALLED MATERIAL MONITORS CAN BE ALSO INSERTED INTO SURVEILLANCE PROGRAMMES
- SUCH MATERIAL CAN MONITOR SIMUTANEOUSLY NEUTRON FLUENCE AND IRRADIATION TEMPERATURE – THESE PARAMETERS CAN BE CHECKED IF TREND CURVE FOR THIS MATERIAL ALREADY EXISTS
- ASTM E 185 DEFINES

3.1.16 *reference material*—any steel that has been characterized as to the sensitivity of its mechanical and fracture toughness properties to neutron radiation embrittlement.

5.6.1 *Use of Reference Materials*—The use of a reference material is optional. If used, a reference material can provide an independent check for deviations from the expected surveillance capsule irradiation conditions (for example, temperature, neutron fluence rate, and neutron energy spectrum).

MATERIAL MONITORS

- SOME OF LWR REACTORS INSERTED SPECIMENS FROM HSSTP PLATE 02 AS REFERENCE MATERIALS
- WHILE IN MANY VVER SURVEILLANCE PROGRAMMES AS WELL AS IN SEVERAL IAEA CO-ORDINATED RESEARCH PROGRAMES „IAEA REFERENCE STEEL – JRQ“ HAS BEEN INSERTED

EXAMPLES OF NEUTRON DOSIMETRY IN SURVEILLANCE PROGRAMMES

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- CHOICE OF MONITORS, THEIR TYPE, SIZE, NUMBER AND LOCATION DEPENDS ON TYPE OF REACTOR, RPV DESIGN, OPERATION CONDITIONS, WITHDRAWAL SCHEDULE etc.
- IN PRINCIPLE, STANDARDS REQUIRED THAT NEUTRON MONITORS SHOULD HAVE TO BE INSERTED IN EACH IRRADIATION CAPSULE
  - THIS IS NOT A CASE OF ALL REACTORS, MAINLY VVER OF OLDER DESIGN
- NEUTRON MONITORS SHOULD BE ABLE TO DETERMINE NEUTRON FIELD IN THE WHOLE OF EACH CAPSULE, PREFERABLY IN CRITICAL POINTS OF SPECIMENS, i.e. IN THEIR NOTCHES/CRACKS

TYPICAL C-E SURVEILLANCE CAPSULE

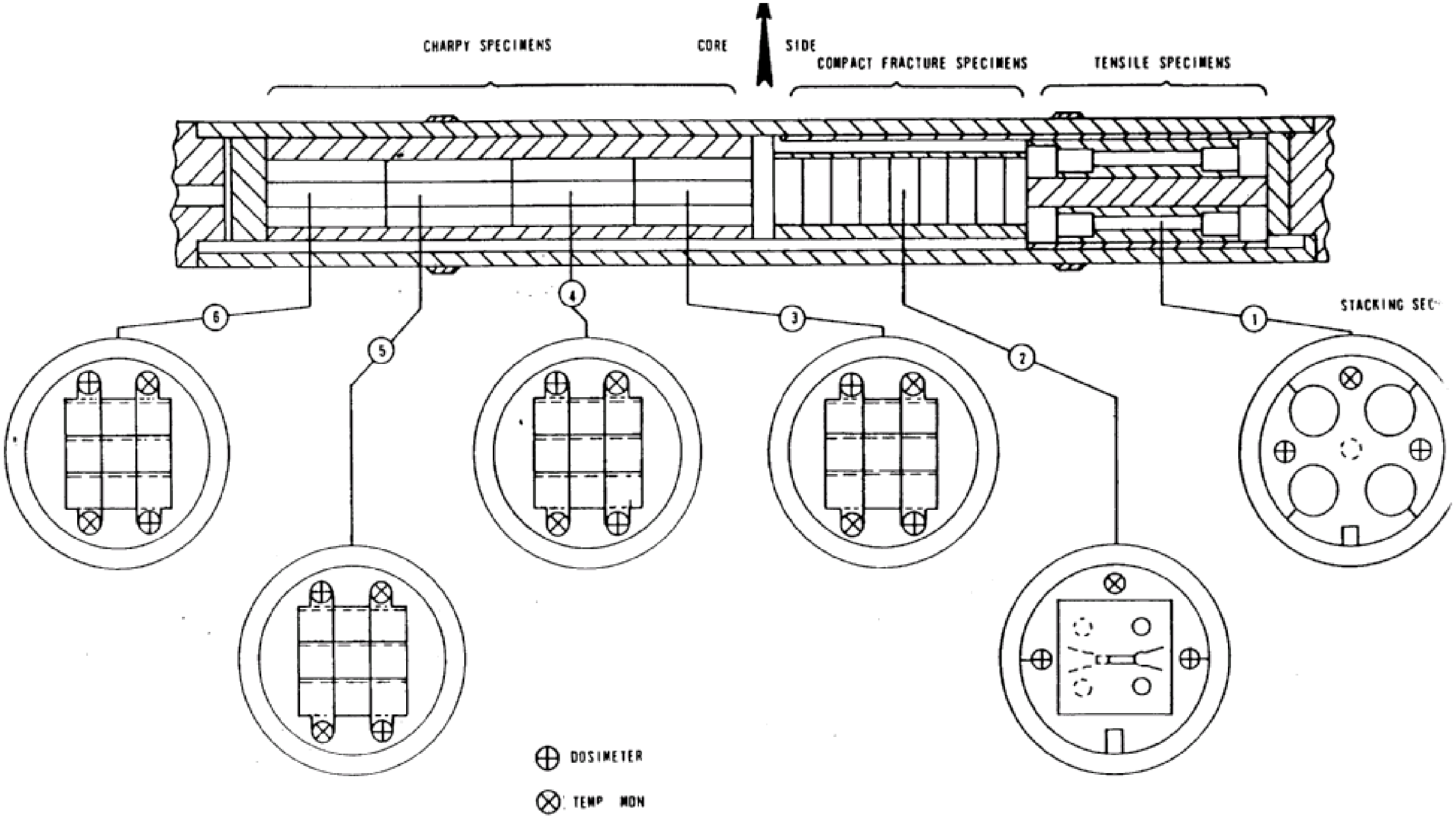

FIG. 2.12—*Typical loading diagram for test specimens in surveillance capsule containing compact and fracture specimens.*

TYPICAL C-E SURVEILLANCE CAPSULE

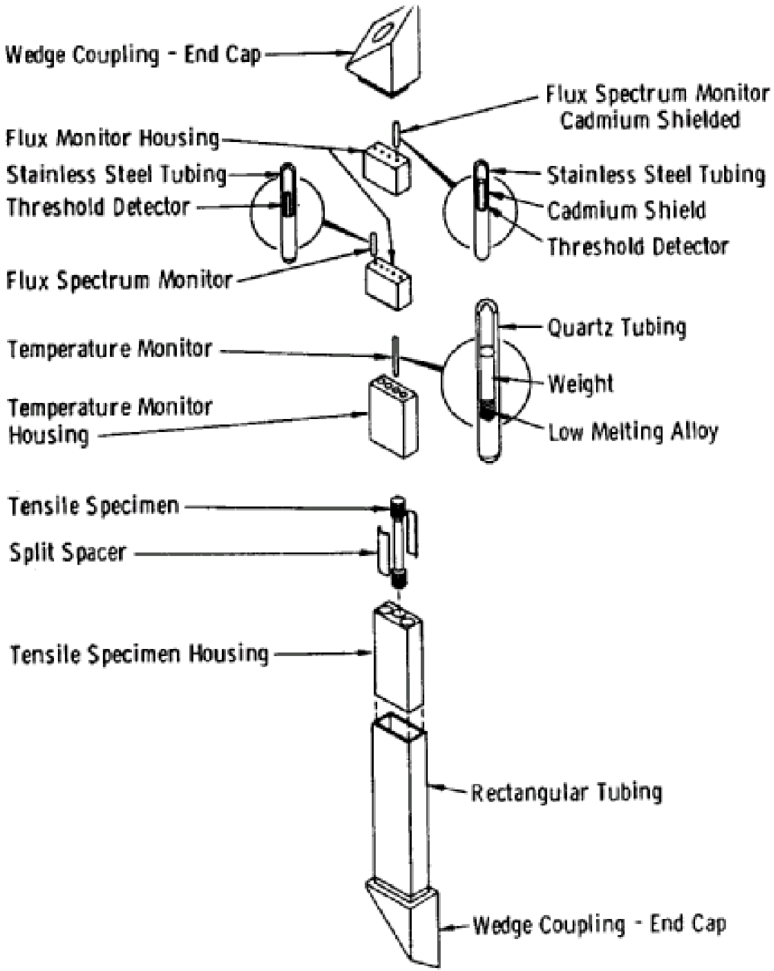

FIG. 3.6—*Typical tensile monitor compartment containing three tension specimens, eight threshold detectors, and four temperature monitors. These compartments occupy the top, center, and bottom position of each capsule assembly.*

LOCATION OF NEUTRON MONITORS IN VVER CAPSULES

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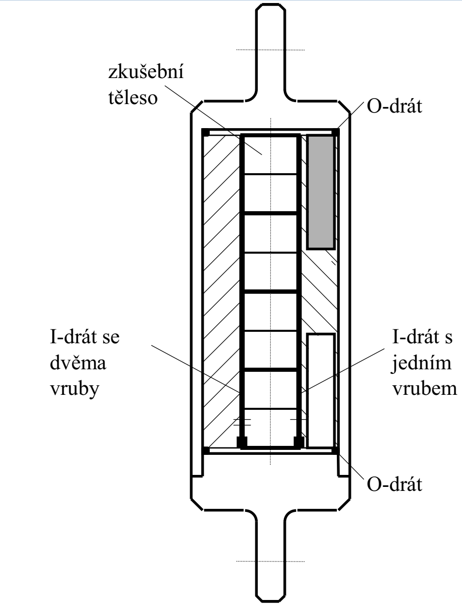
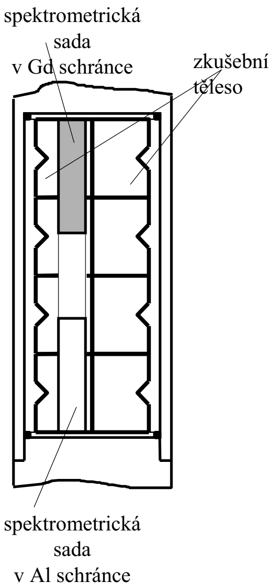




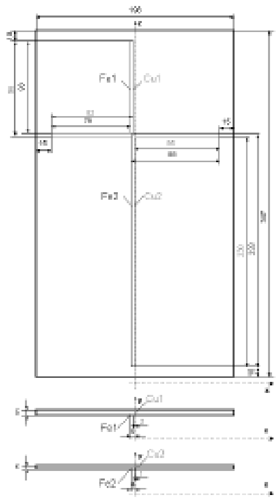


Figure 6. Fe and Cu wires in the container lid

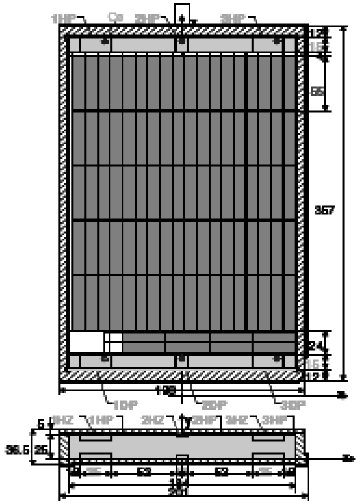
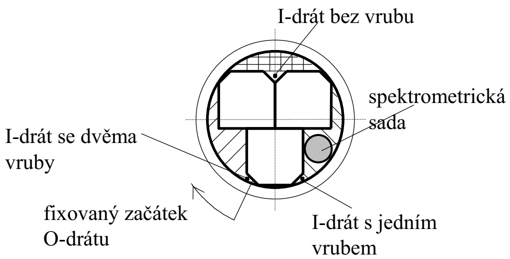
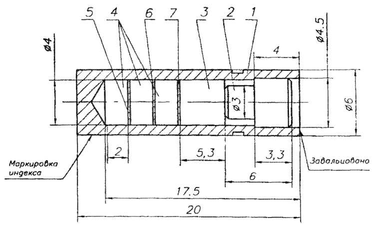


Figure 7. Chart of the container with specimens and monitor capsules





Поз.	Наименование	Материал
1	Капсула	Алюминиевый сплав
2	Пробка	Алюминиевый сплав
3	Упор	Титановый сплав
4	Прокладка	Титановый сплав
5	Элемент активируемый	Изотоп ^{54}Fe
6	Элемент активируемый	Изотоп ^{63}Cu
7	Элемент активируемый	Изотоп ^{93}Nb

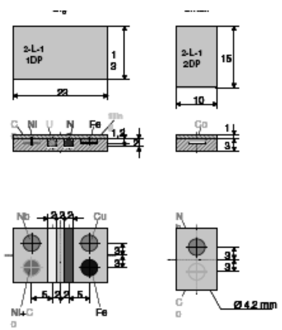


Figure 8. Small and big capsules with spectrometric neutron monitor sets

NEUTRON FIELD IN SURVEILLANCE CAPSULES

- TYPICAL DISTRIBUTION OF NEUTRON FLUX ALONG THE HEIGHT OF CHAIN OF CAPSULES IN VVER-440/V-213 TYPE

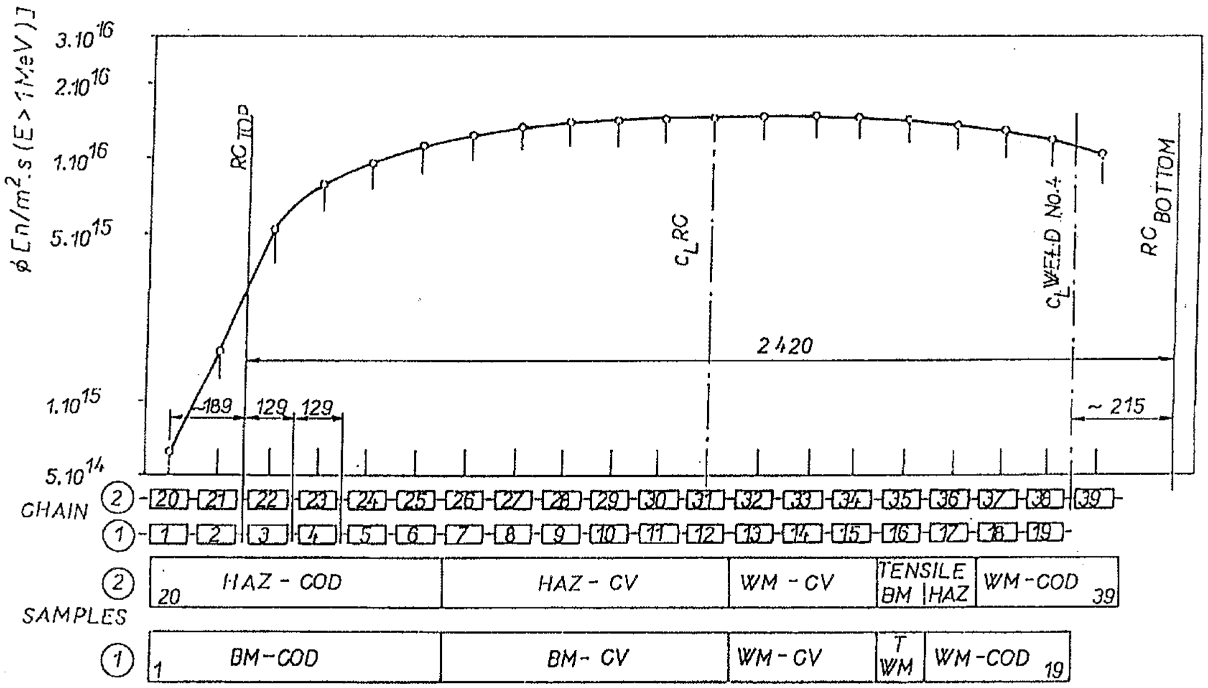
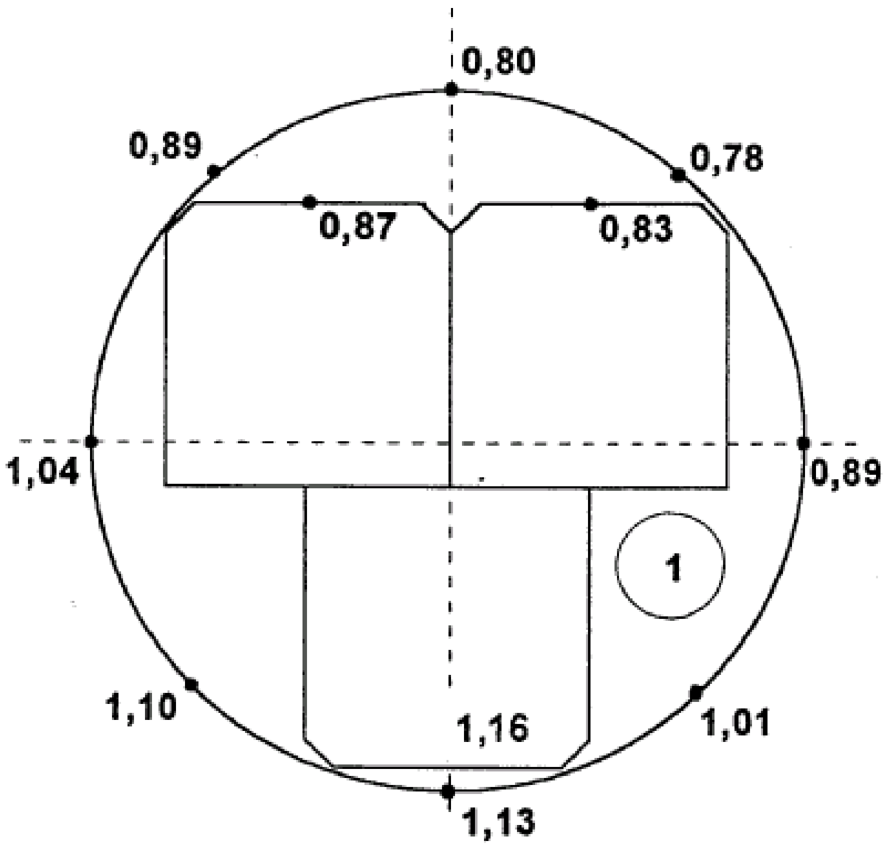
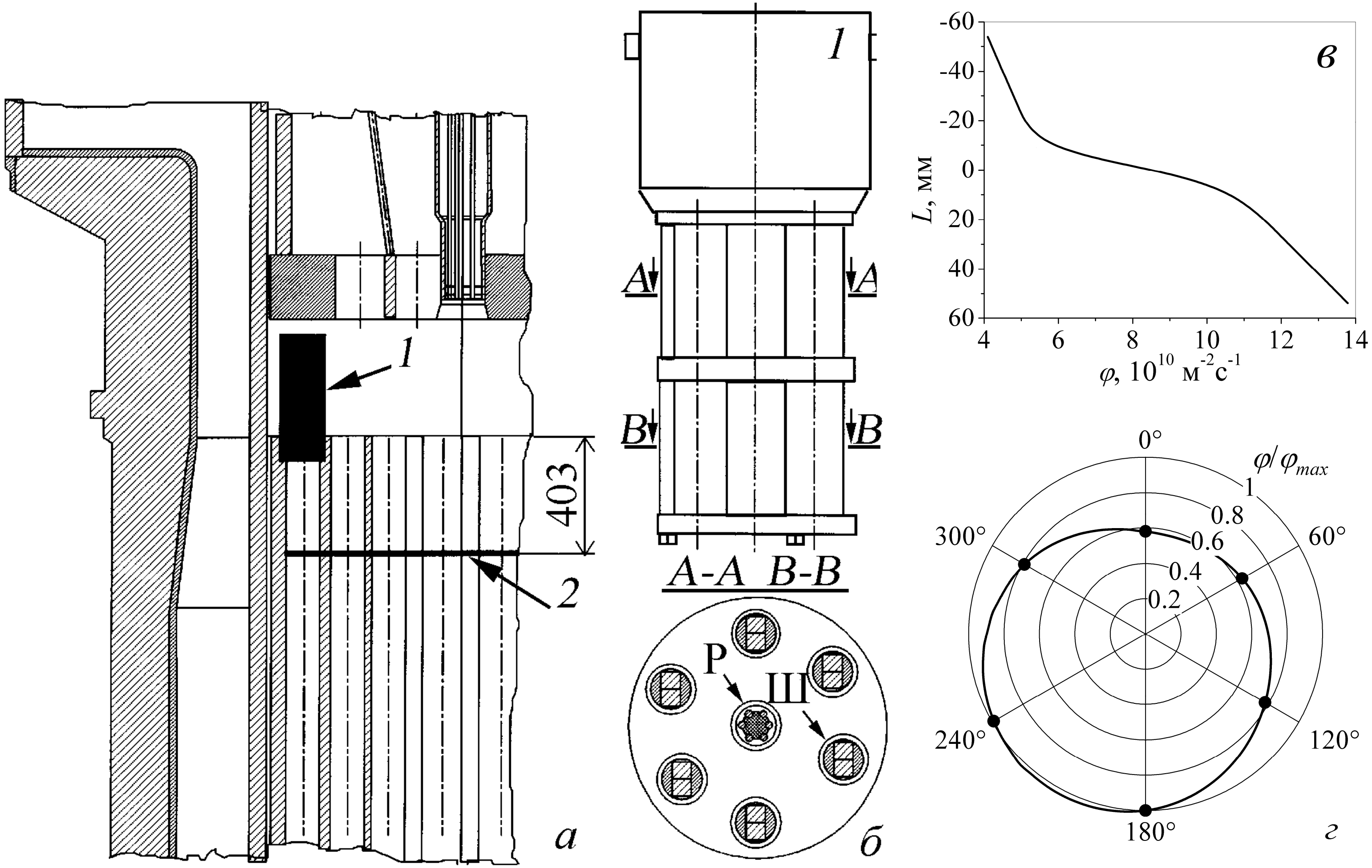


Fig. 3 Typical distribution of neutron flux along irradiation chain with containers

NEUTRON FLUX DISTRIBUTION WITHIN A CAPSULE



VVER-1000/V-320 STANDARD PROGRAMME VARIATION IN NEUTRON FIELD IN ONE LEVEL COULD BE UP TO 180 – 200 %



CONCLUSIONS

- NEUTRON DOSIMETRY IS A VERY IMPORTANT PART OF RADIATION DAMAGE STUDY AS WELL AS RPV INTEGRITY AND LIFETIME ASSESSMENT
- NEUTRON DOSIMETRY MUST BE CARRIED OUT WITH A HIGH PRECISION, NEUTRON PARAMETERS (CROSS SECTIONS etc.) MUST BE CONTINUOUSLY IMPROVED
- NEUTRON DOSIMETRY IS ALSO VERY IMPORTANT FOR RPV LIFE EXTENSION WHEN EX-VESSEL MEASUREMENTS ARE OF HIGH IMPORTANCE WHEN NO SURVEILLANCE CAPSULES WILL BE LOCATED INSIDE RPV

2. Referenced Documents

2.1 *ASTM Standards:*

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]

Designation: E 944 – 02 **Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIA)**[1]

2. Referenced Documents

2.1 *ASTM Standards:*

E 170 Terminology Relating to Radiation Measurements and Dosimetry[2]

E 262 Test Method for Determining Thermal Neutron Reaction and Fluence Rates by Radioactivation Techniques[2]

E 263 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron[2]

E 264 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Nickel[2]

E 265 Test Method for Measuring Reaction Rates and Fast-Neutron Fluences by Radioactivation of Sulfur-32[2]

E 266 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Aluminum[2]

E 343 Test Method for Measuring Reaction Rates by Analysis of Molybdenum-99 Radioactivity from Fission Dosimeters[2]

E 393 Test Method for Measuring Reaction Rates by Analysis of Barium-140 from Fission Dosimeters[2]

E 481 Test Method for Measuring Neutron Fluence Rate by Radioactivation of Cobalt and Silver[2]

Designation: E 944 – 02 **Standard Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIA)[1]**

E 523 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Copper[2]
E 526 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Titanium[2]
E 693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E 706(ID)[2,3]
E 704 Test Method for Measuring Reaction Rates by Radioactivation of Uranium-238[2]
E 705 Test Method for Measuring Reaction Rates by Radioactivation of Neptunium-237[2]
E 706 Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards[2]
E 844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, E 706(IIC)[2,3]
E 853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results, E 706(IA)[2,3]
E 854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, E 706(IIIB)[2,3]

E 910 Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, E706(IIIC)[2,3]

E 1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E 706(IIIA)[2]

E 1018 Guide for Application of ASTM Evaluated Cross Section Data File, E706(IIB)[2,3]

E 2005 Guide for the Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Field, E706 (IIE-I)[2]

E 2006 Guide for the Benchmark Testing of LWR Calculations E706 (IIE-2)[2,3]

2.2 *Nuclear Regulatory Commission Documents:*[4]

NUREG/CR-1861 PCA Experiments and Blind Test

NUREG/CR-2222 Theory and Practice of General Adjustment and Model Fitting Procedures

Thank you for your attention

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