



Designation: E 706 – 87 (Reapproved 1994)

Standard Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards, E 706(0)¹

This standard is issued under the fixed designation E 706; 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 master matrix standard describes a series of standard practices, guides, and methods for the prediction of neutron-induced changes in light-water reactor (LWR) pressure vessel (PV) and support structure steels throughout a pressure vessel's service life (Figs. 1 and 2). Some of these are existing ASTM standards, some are ASTM standards that have been modified, and some are newly proposed ASTM standards. General requirements of content and consistency are discussed in Section 7. More detailed writers' and users' information, justification, and specific requirements for the nine practices, six guides, and five methods are provided in Sections 3-5. Referenced documents, references, and a matrix standard steering committee task group are discussed in Sections 2 and 6. The summary-type information that is provided in Sections 3 and 4 is essential for establishing proper understanding and communications between the writers and users of this set of matrix standards. It was extracted from the referenced documents, Section 2 and Table 1 and references **(1-104)**² for use by individual writers and users.

1.2 This master matrix is intended as a reference and guide to the preparation, revision, and use of standards in the series and for planning and scheduling purposes, particularly as may concern future activities of E10.05 and E10.02. This index is to help ensure the accomplishment of an objective irrespective of the time required, the number of ASTM committees concerned, or the complexity of the issues involved.

1.3 This master matrix standard provides a guide to ASTM standards related to the energy-critical areas that are to be developed under the category of Fission Reactor Development, Section 10, of Guide E 584 and as discussed in Practice E 583.³ With reference to Fig. 1 of Guide E 584, this (E 706) master matrix standard is identified with the horizontal and vertical

subdivisions, respectively:

1.3.1 *Energy Source-Research, Development, and Application Areas:*

1.3.1.1 *Fission Reactor Development Converter:*

(a) LWR Pressurized Water (PWR).

(b) LWR Boiling Water (BWR).

1.3.2 *Standards-Needed Area:*

1.3.2.1 Definition of Function Characteristics.

1.3.2.2 Provision to Achieve Function.

1.3.2.3 Quality Assurance and Reliability.

1.4 To account for neutron radiation damage in setting pressure-temperature limits and making fracture analyses (see Refs **2-7, 9-14, 21-57, 63, 69-71, 77, 78, 83-104**, and Recommended Guide E 509), neutron-induced changes in reactor pressure vessel steel fracture toughness must be predicted, then checked by extrapolation of surveillance program data during a vessel's service life. Uncertainties in the predicting methodology can be significant. Techniques, variables, and uncertainties associated with the physical measurements of PV and support structure steel property changes are not considered in this master matrix, but elsewhere (**1, 3, 4, 10-13, 17, 21, 22-27, 32-39, 42, 43, 45, 49-57, 71, 77, 78, 83, 91-104**, and Recommended Guide E 509). The techniques, variables and uncertainties related to **(1)** neutron and gamma dosimetry, **(2)** physics (neutronics and gamma effects), and **(3)** metallurgical damage correlation procedures and data are addressed in this master matrix **(2, 3 4)**. The main variables of concern to **(1)**, **(2)**, and **(3)** are as follows:

1.4.1 Steel chemical composition and microstructure,

1.4.2 Steel irradiation temperature,

1.4.3 Power plant configurations and dimensions, from the core edge to surveillance positions and into the vessel and cavity walls,

1.4.4 Core power distribution,

1.4.5 Reactor operating history,

1.4.6 Reactor physics computations,

1.4.7 Selection of neutron exposure units,

1.4.8 Dosimetry measurements,

1.4.9 Neutron spectral effects, and

1.4.10 Neutron dose rate effects.

1.5 A number of potential methods and standards exist for assuring the adequacy of fracture control of reactor pressure vessel belt lines under normal and accident loads **(1-4, 7, 13, 14, 21-28, 29-34, 52-57, 71, 77, 78, 91, 93, Recommended**

¹ This master matrix 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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² The boldface numbers in parentheses refer to the list of references at the end of this standard.

³ With reference to the E583 matrix approach, Subcommittee E10.05 on Nuclear Radiation Metrology assumes the lead responsibility for developing and maintaining this LWR Pressure Vessel Surveillance Master Matrix with the liaison assistance of Subcommittee E10.02 on Behavior and Use of Nuclear Metallic Materials in Nuclear Systems.

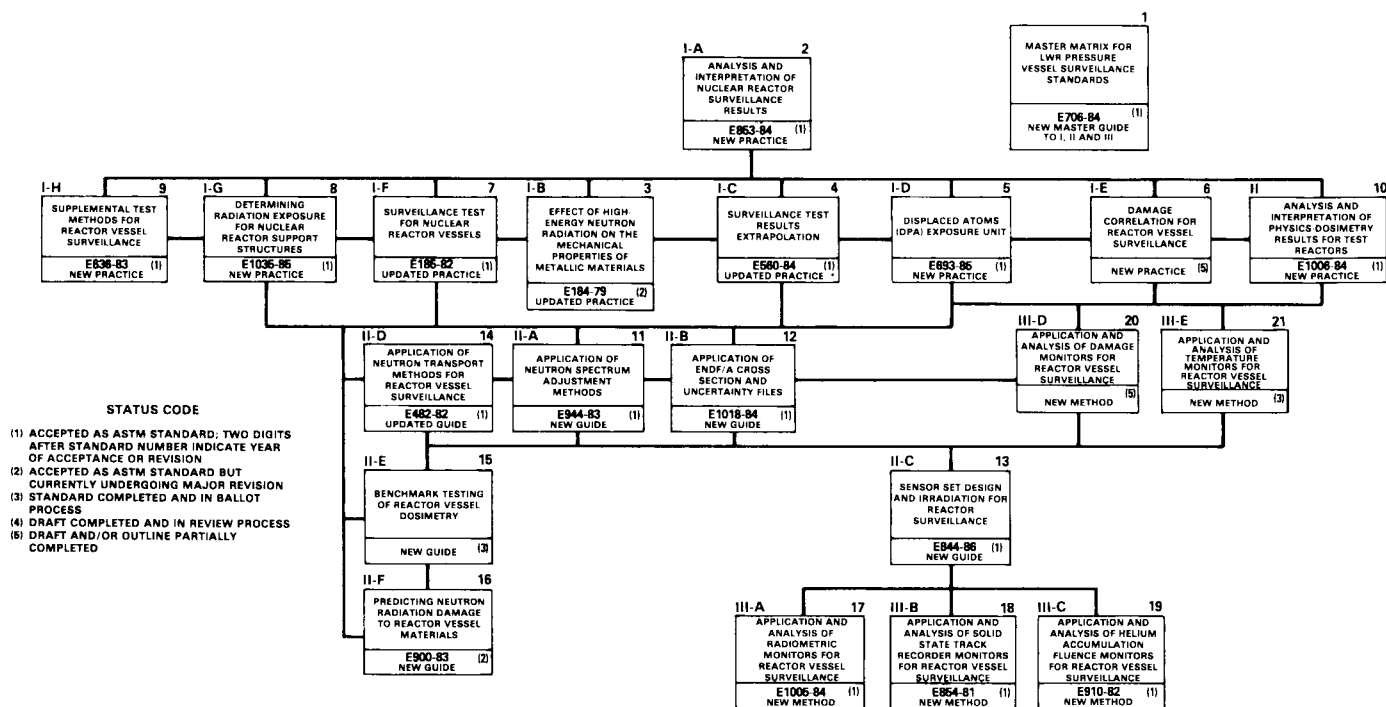


FIG. 1 ASTM Standards for Surveillance of LWR Nuclear Reactor Pressure Vessels and Support Structures

Guide E 509, and 2.3 ASME Standards). As older LWR pressure vessels become more highly irradiated, the predictive capability for changes in toughness must improve. Since during a vessel's service life an increasing amount of information will be available from test reactor and power reactor surveillance programs, better procedures to evaluate and use this information can and must be developed (1-4, 6, 7, 9-15, 17, 21-34, 52-57, 69, 71-73, 77 and 78, 91-104, and Recommended Guide E 509). This master matrix, therefore, defines the current (1) scope, (2) areas of application, (3) interrelationships, and (4) status and time table of development, improve-

ment, validation, and calibration for a series of 20 ASTM standards, as shown in Figs. 1 and 2.

1.6 Chronology of Past Events and Reference Documentation:

1.6.1 Table 1 provides a chronological listing of important events, results, benefits, and references related to the establishment of this Master Matrix for LWR Pressure Vessel Surveillance Standards. It is intended for use by authors and users of this set of 20 ASTM standards, see Section 7.

RECOMMENDED E10 ASTM STANDARDS

0 MASTER MATRIX GUIDE TO I, II, III

I. METHODS OF SURVEILLANCE AND CORRELATION PRACTICES

A. ANALYSIS AND INTERPRETATION OF NUCLEAR REACTOR SURVEILLANCE RESULTS

B. EFFECTS OF HIGH-ENERGY NEUTRON RADIATION ON MECHANICAL PROPERTIES

C. SURVEILLANCE TEST RESULTS EXTRAPOLATION

D. DISPLACED ATOM (DPA) EXPOSURE UNIT

E. DAMAGE CORRELATION FOR REACTOR VESSEL SURVEILLANCE

F. SURVEILLANCE TESTS FOR NUCLEAR REACTOR VESSELS(*)

G. DETERMINING RADIATION EXPOSURE FOR NUCLEAR REACTOR SUPPORT STRUCTURES

H. SUPPLEMENTAL TEST METHODS FOR REACTOR VESSEL SURVEILLANCE(*)

I. ANALYSIS AND INTERPRETATION OF PHYSICS-DOSIMETRY RESULTS FOR TEST REACTORS

II. SUPPORTING METHODOLOGY GUIDES

A. APPLICATION OF NEUTRON SPECTRUM ADJUSTMENT METHODS

B. APPLICATION OF ENDF/A CROSS SECTION AND UNCERTAINTY FILES

C. SENSOR SET DESIGN AND IRRADIATION FOR REACTOR SURVEILLANCE

D. APPLICATION OF NEUTRON TRANSPORT METHODS FOR REACTOR VESSEL SURVEILLANCE

E. BENCHMARK TESTING OF REACTOR VESSEL DOSIMETRY

F. PREDICTING NEUTRON RADIATION DAMAGE TO REACTOR VESSEL MATERIALS(*)

III. SENSOR MEASUREMENTS METHODS

APPLICATION AND ANALYSIS OF:

A. RADIO-METRIC MONITORS FOR REACTOR VESSEL SURVEILLANCE

B. SOLID STATE TRACK RECORDER MONITORS FOR REACTOR VESSEL SURVEILLANCE

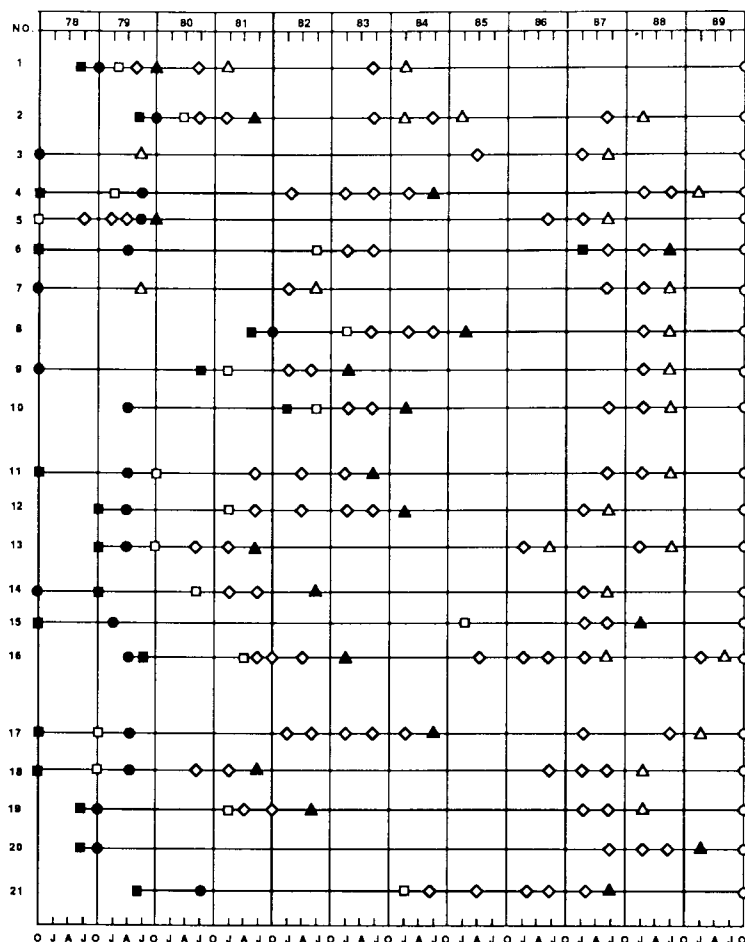
C. HELIUM ACCUMULATION FLUENCE MONITORS FOR REACTOR VESSEL SURVEILLANCE

D. DAMAGE MONITORING FOR REACTOR VESSEL SURVEILLANCE

E. TEMPERATURE MONITORS FOR REACTOR VESSEL SURVEILLANCE(*)

-  DRAFT OUTLINE DUE TO ASTM E10 SUBCOMMITTEE TASK GROUPS
 1ST DRAFT TO APPROPRIATE ASTM E10 SUBCOMMITTEE TASK GROUPS
 REVISED DRAFT FOR ASTM E10 SUBCOMMITTEES, ASTM E10 COMMITTEE, AND/OR ASTM SOCIETY BALLOTING(*)
 ACCEPTANCE AS ASTM STANDARD
 REVISION AND ACCEPTANCE AS ASTM STANDARD
 PRIMARY TIME INTERVAL FOR ROUND ROBIN VALIDATION AND CALIBRATION TESTS

(*) INDICATES THAT THE LEAD RESPONSIBILITY IS WITH SUBCOMMITTEE E10.02 INSTEAD OF WITH SUBCOMMITTEE E10.05.



NOTE 1—All ASTM standards must be reviewed, updated as required, and reballoted every 5 years.

FIG. 2 Preparation, Validation, and Calibration Schedule for LWR-PV and Support Structure Surveillance Standards

TABLE 1 Chronological Listing of Important Events, Results, Benefits, and References

Date - Event - Reference	Result or Benefit, or Both
June 1961—Publication of E 185 – 61, ASTM Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels.	First U.S. standard for LWR pressure vessel surveillance.
May 1971—Interlaboratory Reaction Rate (ILRR) program established	Start of coordinated interlaboratory integral measurements in selected standard and reference benchmark neutron fields.
October 1972—IAEA Specialist's Meetings on Radiation Damage Units for Ferritic and Stainless Steels [Proceedings of Seattle Meeting, <i>Nuclear Engineering Design</i> , Vol 33, 1975]	Validation/calibration of integral and differential dosimetry data for fission reactors.
September 1973—IAEA Consultant's Meeting on Nuclear Data for Reactor Dosimetry, INDC(NDS)56/U, 1973	First serious international attempt to standardize exposure units for materials irradiations. Calculational model adopted and engineering damage cross sections defined for ferritic steels.
December 1973—Publication of Recommended Procedures for Displacement Damage Calculations for U.S. FBR Program, HEDL-TME 73-76, 1973	International agreement on need to develop reference dosimetry files for LWR, FBR, and MFR programs and direction established for future interlaboratory program work.
September 1974—Cross-Section Evaluation Working Group (CSEWG) Dosimetry File Task Force Meeting	Procedures widely adopted in U.S. FBR programs, with carryover to LWR and MFR programs.
December 1974—ASTM E-10 Mini-Symposium on Neutron Dosimetry and Spectrum Analysis for Materials Applications, <i>Journal of Testing and Evaluation</i> . Vol 3, 1975	Format identification of dosimetry, gas production, etc. Cross-section requirements. Specific needs established for ENDF/B for LWR, FBR, and MFR programs.
	Early attempt to encourage improved understanding and interaction between dosimetrists and metallurgists.

TABLE 1 Continued

Date - Event - Reference	Result or Benefit, or Both
February 1975—ILRR Program Results Published in Special Issue of Nuclear Technology, <i>Nuclear Technology</i> , Vol 25, 1975	Important initial step completed in validation/calibration of integral and differential dosimetry data for fission reactors. Preliminary data testing completed of evaluated dosimetry cross-sections on ENDF/B-IV and Sand II files.
September 1975—First ASTM-EURATOM International Symposium on Reactor Dosimetry EUR 5667 elf, Parts I and II, 1977	First major world wide effort to identify and assess dosimetry accuracies for LWR, FBR, and MFR programs and establish improved interlaboratory cooperative work.
October 1976—CSEWG-ENDF/B Special Applications File (Safe) Subcommittee Established	Established following CSEWG task groups: · dosimetry file · gas production file· activation file · radioactive decay file· users needs and test recommendations Major milestone-important step towards international standardization and usage of consistent set of ENDF/B data files. Identified computational problem areas [factor of two or more differences] requiring future resolution.
October 1976—Joint IAEA-OECO (NEA) Technical Committee Meeting, Includes IAEA Calculational Intercomparison of PWR PV Damage Rate Calculations: Reactor Shielding Benchmark No. 2 [IAEA-] November 1976—IAEA Consultant's Meeting on Integral Cross-Section Measurements in Standard Neutron Fields, IAEA-208, 1978	First major step to establish international agreement on interlaboratory effort of benchmark field studies.
November 1976—IAEA Specialist's Meeting on Radiation Damage Units, Proceedings of Harwell Meeting, IAEA Conference-761146, 1977	LWR, FBR, and MFR program needs identified, reviewed, and updated. Identified two specific displacement cross-section sets for fission reactor structural metals based on damage model recommended in October 1972. Recommended that one set be adopted, based on single set of evaluated reaction cross sections, applicable to both fission and fusion reactors.
March 1977—Interlaboratory LWR Pressure Vessel Surveillance Dosimetry Improvement Program, LWR-PV-SDIP Established.	Start of coordinated interlaboratory effort to prepare new and updated ASTM standards and validate and calibrate recommended procedures and data in selected "benchmark neutron fields" and reactor "test and surveillance regions." New and updated validated/calibrated ASTM standards expected to be established in the period 1979 to 1987 for LWR fission reactors. Data Testing of ENDF/B-IV Dosimetry File. Direction for establishment of ENDF/B-V dosimetry file.
March 1977—CSEWG Dosimetry Task Group Meeting	Assessment of progress in neutron standards and applications. Assessment established for future work for LWR, FBR, and MFR Programs. Assessment of status of neutron cross sections from 10-40 MeV for MFR program high flux neutron sources. Nuclear data needs and research direction established for MFRs, with carry-over to LWR and FBR programs.
March 1977—International Specialist's Symposium on Neutron Standards and Applications NBS Special Publication 493, 1977 May 1977—Symposium on Nuclear Reaction Data in the Energy Range 10-40 MeV BNL-NCS-50681, 1977	Major step to further establish working international laboratory cooperative efforts for LWR, FBR, and MFR programs.
October 1977—Second ASTM-EURATOM International Symposium on Reactor Dosimetry, NUREG/CP-0004, Vols 1 to 3, 1978	Major step towards implementation of international standardization and better coordination of joint efforts of dosimetrists, metallurgists, and reactor analysts. Major step to establish international agreement on use of more standardized procedures of determining flux/fluence spectra.
October 1977—IAEA Specialist's Meeting at ORNL on current status of Neutron Spectrum Adjustment, IAEA-TECDOC-221, 1979	Guidance for preparation of ASTM multiple sensor adjustment code standard. Important step to inform metallurgists and dosimetrists on state-of-the-art practices in use for LWR-PV surveillance programs.
November 1977—ASTM E10 Mini-Symposium on Reactor Pressure Vessel Surveillance Programs: Recent Experiences and Future Directions August 1978—RSIC Workshop on Theory and Application of Sensitivity and Uncertainty Analysis, ORNL/RSIC-42, 1979	State-of-the-art review of sensitivity and uncertainty analysis for fast, thermal, and fusion reactor applications.
November 1978—IAEA Advisory Group Meeting on Nuclear Data for Reactor Dosimetry, INDC(MDS)-103/M, 1978	IAEA agreed to establish and maintain an international reactor dosimetry file (IRDF).
November 1978—Publication of E521 – 77 ASTM Standard Recommended Practice for Neutron Radiation Damage Simulation by Charged Particle Irradiation. Section on Correlation of Damage Produced by Neutrons is being developed. December 1978—ASTM E-10 Mini-Symposium on Interfaces Between Neutron Dosimetry and Metallurgical Properties in Nuclear Pressure Vessel Surveillance.	First U.S. standard for using and correlating charged particle irradiation effects data, including DPA exposure units.
February 1979—IAEA Specialist's Meeting on Irradiation Embrittlement, Thermal Annealing and Surveillance of Reactor Pressure Vessels Proceedings. Published as Reference 103.	First major step to encourage improved understanding and interactions between dosimetrists, metallurgists, and reactor analysts for LWR pressure vessel surveillance programs.
September 1979—IAEA Specialist's Meeting on Accuracies in Correlation Between Property Change and Exposure Data from Reactor Pressure Vessel Steel Irradiations Proceedings. Published as Reference 4.	Second major step to encourage improved understanding and interactions between dosimetrists, metallurgists, and reactor analysts for LWR pressure vessel surveillance programs.
October 1979—Third ASTM-EURATOM International Symposium on Reactor Dosimetry, EUR6813, Vols 1 and 2, European Atomic Energy Community, Brussels, Belgium, 1980.	Third major step to encourage improved understanding and interactions between dosimetrists, metallurgists, and reactor analysts for LWR pressure vessel surveillance programs.
March 1982—Fourth ASTM-EURATOM International Symposium on Reactor Dosimetry, NUREG/CP-0029, Vols 1 and 2, Nuclear Regulatory Commission, 1982.	Major step to further establish working international interlaboratory cooperative efforts for LWR, FBR, and MFR programs.
	Major step towards implementation of international standardization and better coordination of joint efforts of dosimetrists, metallurgists, and reactor analysts. Reports on results and significant progress of international interlaboratory cooperative efforts for LWR, FBR, and MFR programs. LWR Master Matrix E 706 updated to describe an expanded series of 20 ASTM standard practices, guides and methods. The need for two additional standards identified. LWR pressurized thermal shock becomes a major area of study for PWR power plants and declared an unresolved safety issue by USNRC. PCA Experiments and Blind Test, Melusine Test, and PWR and BWR Physics-Dosimetry results reported and requirements for the VENUS and NESDIP benchmark tests established.

TABLE 1 Continued

Date - Event - Reference	Result or Benefit, or Both
1970 to 1983—ASTM International Symposia on Radiation Effects in Materials and ASTM Special Technical Publications (STPs).	Reports on state-of-the-art metallurgy and dosimetry. Continuing steps to encourage improved understanding and interactions between dosimetrists, metallurgists, and reactor analysts.
June 1983—Test report for ^{235}U fission spectrum neutron fluence standards completed.	Neutron fluence standards for Ni, Fe, and Ti available from NBS on a near-routine basis.
November 1983—First revision of Compendium of Benchmark Neutron Fields for Reactor Dosimetry completed.	^{252}Cf and ^{235}U fission spectra and for ISNF include reaction rate formulations appropriate for benchmark calibration of dosimetry, and measurement reaction cross sections with a cut-off date of July 1983 for evaluation.
December 1983—Cavity fission source at NBS intercalibrated with similar facility at SCK/CEN.	Fission neutron fluence scale at both facilities is made traceable to the primary ^{252}Cf neutron source by means of spectrum insensitive reaction rates.

TABLE 2 List of Neutron Fields for LWR-PV Dosimetry Validation and Calibration Studies

Neutron Field	Type of Dosimetry ^A	Status
Benchmark for Calibrations: ^{252}Cf and ^{235}U fission neutron irradiation facilities at NBS (standard)	Sensor calibrations: RM, NE, SSTR	Neutron fluence standards for RM dosimetry available on a periodic basis. SSTR and NE dosimeter types carried out on a special schedule basis. ^{252}Cf is an absolute fast neutron fluence standard. ¹⁶
^{235}U fission field at CEN/SCK (standard)	Sensor calibrations	Applied routinely to RM, DM, and other sensor calibrations for PCA, VENUS, NESDIP and other neutron field measurements.
ISNF irradiation facilities at NBS (standard)	Sensor calibrations (particularly detectors with response range below 0.5 MeV)	Fission rate ratios established with fission chambers.
CRMF irradiation facility at EG&G (reference)	Sensor calibrations	Application to RM and other sensors planned.
PCA/PSF/SDMF irradiation facilities at ORNL (reference)	Sensor validations and surveillance perturbation investigations	Availability being established
Benchmarks for Transport Calculation Validation: Iron shells field at CEN/SCK (reference)	Active and passive RM, SSTR, NE, HAFM	See References 6, 45, and 82.
PCA/PSF/SDMF at ORNL (reference)	Active and passive RM, SSTR, NE, HAFM, DM	See References 2 and 45.
VENUS at CEN/SCK (reference)	Active and passive Neutron and Gamma Technique	See References 71 and 74.
NESDIP at UKAEA, Winfrith (reference)	Active and passive Neutron and Gamma Technique	See References 75 and 83.
Test Regions for Dosimetry Method,^B Validation: BSR-HSST dosimetry-test (test reactor at ORNL)	Radiometric monitors (RM)	See References 2 and 47.
Melusine PCBT (test reactor at _____, France)	RM and DM (tungsten & graphite)	See Reference 63.
FRJ-1 and FRJ-2 (test reactor at KFA, Germany)	RM, HAFM, TM (melt wires), DM (quartz)	"Test Region" irradiations in metallurgical rigs in progress. Interlaboratory dosimetry.
DIDO, PLUTO, and HERALD (test reactors in UK)	RM, HAFM, TM, DM	See References 2, 45, 75, 83, and 92.
Buffalo NRL-NRC (test reactor at University of Buffalo)	Radiometric monitors	See References 2 and 45.
University of Virginia (test reactor)	Radiometric monitors	See References 22 and 32.
PWRs and BWRs	Active and passive monitors for Neutrons and Gamma fields	See References 2, 5, 6, 9, 10, 21, 22, 29–31, 40–41, 69, 71–73, 77, 84–90, 91, 103

^A RM—radiometric monitors; NE—nuclear emulsions; SSTR—solid state track recorders; DM—damage monitors; HAFM—helium accumulation fluence monitors; TM—temperature monitors.

^B These neutron fields are also being used for physics and metallurgical damage correlation procedures and data validation and calibration studies.

TABLE 3 Metallurgical Steel Irradiations—Test Facilities Estimated Environment Parameter Values for Assessing Temperature, Flux Level, and Spectral Effects

Test Facility	Irradiation Temperature °C	$\bar{\sigma}_{\text{Fe DPA}}^{\text{Fe}}(\phi > 1.0 \text{ MeV})$ (barns)	$\bar{\sigma}_{\text{Fe}^{54}}^{\text{Fe}}(n, p)(\phi > 1.0 \text{ MeV})$ (Milli-barns)	$\left[\frac{\phi(>0.1 \text{ MeV})}{\phi(>1.0 \text{ MeV})} \right]$	Flux ($>1.0 \text{ MeV}$) Level (n/cm ² ·s) (peak)
United States: ORR-PSF (PV Mockup) ^A (30 MW)					
Core center-in fuel		1390 ^B	107 ^C	1.85	3.8×10^{14}
Surveillance capsule-center	288 ± 5	1606	66	3.08	6.8×10^{12}
PV surface position-incident	"	1588	83	2.79	9.6×10^{11}
PV T/4 position	"	1844	61	4.10	4.6×10^{11}

TABLE 3 Continued

Test Facility	Irradiation Temperature	$\frac{\bar{\sigma}}{\left[\frac{\text{Fe DPA}}{\phi(>1.0 \text{ MeV})} \right]}$	$\frac{\bar{\sigma}}{\left[\frac{\text{Fe}^{54} (n, p)}{\phi(>1.0 \text{ MeV})} \right]}$	$\left[\frac{\phi(>0.1 \text{ MeV})}{\phi(>1.0 \text{ MeV})} \right]$	Flux (>1.0 MeV) Level
PV T/2 position	"	2238	49	5.77	2.1×10^{11}
PV 3T/4 position	"	2727	41	7.82	9.8×10^{10}
PV void box back face position	ambient ($\sim 44^\circ\text{C}$)	3060	41	8.82	3.1×10^{10}
ORR-30, BSR-2MW					
Core center-in fuel	288 ± 5	1386^B	107^C	1.84	3.2×10^{13}
Surface position center 4T HSST specimen	288 ± 10	1623	64	3.01	2.1×10^{12}
Buffalo (2 MW)					
Core positions in NRL/NRC steel test assembly	288 ± 5	1765^B	73^C	3.54	$\sim 3 \times 10^{13}$
Virginia (2 MW)					
Core position in Westinghouse/EPR steel test assembly	288 ± 5	~ 1765	~ 73	~ 3.50	$\sim 3 \times 10^{13}$
PWR					
Surveillance position-in steel capsule	288–321	$1800^{B,D}$	$63^{C,D}$	...	$0.1\text{--}2.0 \times 10^{11}$
PV surface	288–321	1630	110	...	$0.07\text{--}0.70 \times 10^{11}$
PV T/4 position	288–321	1960	78	...	$0.04\text{--}0.40 \times 10^{11}$
Ex-vessel cavity	ambient	...	...	...	$0.0020\text{--}0.20 \times 10^{11}$
BWR					
Surveillance position					
PV surface	280	1553^B	182^E	~ 1.6	$10^8\text{--}10^9$
PV T/4 position	270	1603	141	~ 2.5	$\sim 10^8$
Ex-vessel cavity	55	2014	98	~ 4.4	$10^7\text{--}10^8$
<i>United Kingdom:</i>					
DIDO (25 MW)					
Core center (in fuel)	$\text{XXX} \pm 5$	$1600^B (1567)^F$	112^C	2.3	5×10^{13}
2V4 position (steel)	$\text{XXX} \pm 5$	1962 (1665)	133	2.3	3×10^{12}
Pluto (25 MW)					
Core center (in fuel)	$\text{XXX} \pm 5$	1646 (1594)	117	2.3	5×10^{13}
CA6B position (steel)	$\text{XXX} \pm 5$	3094 (1700)	168	2.0	5×10^{11}
Herald (5 MW)					
Core center (in fuel)	$\text{XXX} \pm 5$	1367 (1356)	102	1.9	3.5×10^{13}
DIO position (steel)	$\text{XXX} \pm 5$	1457 (1451)	86	2.8	3.5×10^{12}
<i>France:</i>					
Melusine					
PV-simulator (PCBT position)	235 ± 5	1970	...	4.31	1.8×10^{12}
Core edge in steel (61 AVD)	235 ± 5	1450	...	2.13	1.9×10^{13}
Core edge in steel (73 AVG)	235 ± 5	1450	...	2.14	1.9×10^{13}
<i>Germany:</i>					
FRJ 1					
Core edge (G7)-without steel	290 ± 5		122^G	1.85^G	1.51×10^{13G}
FRJ2					
Core (B5)	290 ± 5	1480	88	2.29	4.8×10^{13}
Reflector 2V4-with steel	290 ± 5				
<i>Belgium:</i>					
BR2					
DGR position (core)	~ 150	1390^H	87^C	2.1	$\sim 3 \times 10^{14}$
BR3					
Surveillance position	260^I	1422^H	120^C	1.9	8.8×10^{12J}
Dosimetry position in reflector	260^I	1410	126	1.8	2.3×10^{12}
PVF (in water)	260^I	1621	94	2.9	1.7×10^{11J}
PVF (in steel)	260	1645	88	3.1	1.5×10^{11J}
PV-T/4	260	1723	75	3.6	1.2×10^{11J}
Between PV and Neutron Shield Tank	60	2314	50	6.1	2.2×10^{10J}
<i>Japan:</i>					
JMTR (50MW)					
Core center	270–320	...	81^C	1.87	$1.0\text{--}2.0 \times 10^{14}$
Core position in steel	"	1320^C	71	1.9	1.0×10^{13}
Reflector-IAEA Steel position	"	...	65	2.5	1.0×10^{13}

TABLE 3 *Continued*

Test Facility	Irradiation Temperature	$\left[\frac{\bar{\sigma}}{\Phi(>1.0 \text{ MeV})} \right]$ Fe DPA	$\left[\frac{\bar{\sigma}}{\Phi(>1.0 \text{ MeV})} \right]$ Fe ⁵⁴ (n, p)	$\left[\frac{\Phi(>0.1 \text{ MeV})}{\Phi(>1.0 \text{ MeV})} \right]$	Flux (>1.0 MeV) Level
JRR-2 (10 MW)					
Core (6D)-in Hollow Fuel	Ambient (~50°C)	1590	104 ^K (100)	2.4	2.3×10^{13}
Core center (VT-1)	"	1780 (1590) ^L	113 ^L (109)	3.0	1.8×10^{13}
Position in flux trap					
Netherlands:					
HFR (45 MW)	250–650	1316 ^K	90	1.87	2.1×10^{14}
E5 core position					

^A ^{1/2}Configuration (4 cm H₂O between core and thermal shield and 12 cm of H₂O between the thermal shield and PV wall).

^B ASTM Practice E 693 – 79 iron $\sigma(E)$ with (n, γ) recoil displacements included.

^C ENDF/B-IV $\sigma(E)$.

^D Perturbed spectrum (40°) for a 4-loop reactor geometry.

^E ENDF/B-V $\sigma(E)$.

^F Without (n, γ) recoil displacement included.

^G Spectrum unfolding by L. Wiese, with more appropriate two-dimensional input spectrum.

^H Doran and Graves iron displacement cross section, HEDL SA-1058 (1976).

^I Temperature of water.

^J Based on calculation BR 3/core 4A.

^K Damage cross section library DANSIG-77, E CN-36 report.

^L JENDL-1 $\sigma(E)$.

1.7 The values stated in SI units are to be regarded as the standard.

1.8 *This standard does not purport to address all of the safety concerns, 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:

C 859 Terminology Relating to Nuclear Materials⁴

E 170 Terminology Relating to Radiation Measurements and Dosimetry⁵

E 184 Practice for Effects of High-Energy Neutron Radiation on the Mechanical Properties of Metallic Materials, E 706 (IB)^{5,6}

E 185 Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels, E 706 (IF)^{5,6}

E 380 Practice for Use of the International System of Units (SI) (the Modernized Metric System)⁷

E 482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, E 706 (IID)^{5,6}

E 509 Guide for In-Service Annealing of Light-Water Cooled Nuclear Reactor Vessels⁵

E 560 Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results, E 706 (IC)^{5,6}

E 583 Practice for Systematizing the Development of (ASTM) Voluntary Consensus Standards for the Solution of Nuclear and Other Complex Problems⁵

E 584 Recommended Guide for Developing the (ASTM) Voluntary Consensus Standards Required to Help Imple-

ment the National Energy Plan⁵

E 636 Guide for Conducting Supplemental Surveillance Tests for Nuclear Power Reactor Vessels, E 706 (IH)^{5,6}

E 646 Test Method for Tensile Strain-Hardening Exponents (N-Values) of Metallic Sheet Materials⁸

E 693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E 706 (ID)^{5,6}

E 844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance^{5,6}

E 853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results, E 706 (IA)^{5,6}

E 854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, E 706 (IIIB)^{5,6}

E 900 Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials, E 706 (IIF)^{5,6}

E 910 Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, (IIIC)^{5,6}

E 944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, (IIA)^{5,6}

E 1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E 706 (IIIA)^{5,6}

E 1006 Practice for Analysis and Interpretation of Physics Dosimetry Results for Test Reactors, E 706 (II)⁶

E 1018 Guide for Application of ASTM Evaluated Nuclear Data File (ENDF/A)—Cross Section and Uncertainty File, E 706 (IIB)⁵

E 1035 Practice for Determining Radiation Exposures for Nuclear Reactor Vessel Support Structures⁵

2.2 ASTM Draft Standards:

⁴ Annual Book of ASTM Standards, Vol 12.01.

⁵ Annual Book of ASTM Standards, Vol 12.02.

⁶ The reference Master Matrix designation in parentheses refers to Section 5, as well as Figs. 1 and 2 of Matrix E 706.

⁷ Annual Book of ASTM Standards, Vol 14.02.

⁸ Annual Book of ASTM Standards, Vol 03.01.