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Standard Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results, E 706(IC)[1]

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INTRODUCTION

LWR power plant safety analysis report and subsequent neutron exposure parameter calculations for the PV wall and critical welds need to be verified using modern codes and information from surveillance dosimetry. That is, the location of critical welds relative to the axial and azimuthal flux map should be taken into account, as well as changes in fuel loading during periods when surveillance capsules are exposed.

This practice is intended to be used together with other E 706 LWR Matrix Standards to provide estimates of the neutron exposure and exposure rate (together with uncertainties) at positions at the inner diameter and within the pressure vessel wall of a light water reactor. Also provided will be estimates of gamma-ray exposure and exposure rates to interpret dosimetry sensor photo-reaction and other gamma-ray induced effects. Information used to make these estimates is obtained from neutron-gamma transport calculations and from neutron and other sensor monitors located in surveillance positions on the core side of the vessel and in the cavity outside the vessel wall **(1)**[2]. Benchmark field irradiations of similar monitors also provide valuable information used in the verification of the accuracy of the calculations (a type of cross section covariance and dosimetry monitor counting calibration) **(1).**

Knowledge of the time-dependent relationship between exposure parameters at surveillance locations and selected (r, θ, z) locations within the pressure vessel wall is required to allow determination of the time dependent radiation damage to the pressure vessel. The time dependency must be known to allow proper accounting for complications due to burn-up, as well as, changes in core loading configurations **(2-5).** An estimate of the uncertainty in the neutron exposure parameter values at selected (r, θ, z) points in the vessel wall **(1)** is also needed to place an upper bound on the allowable operating lifetime of the reactor vessel without remedial action **(6-9).** (See Guide E 509).

1. Scope

1.1 This practice covers analytical and analytical-experimental approaches that can be used to determine the variation in neutron exposure (fluence E > 1.0 MeV, dpa, etc.) and exposure rate and energy spectrum between surveillance locations and points in the pressure vessel wall. Procedures for reporting[2] the results of these analyses with assigned uncertainties are also suggested. This practice also provides information and reference to other Matrix E 706 standards and procedures currently being developed and tested for the correlation, extrapolation, and interpolation of all available physics-dosimetry-metallurgy test reactor and power reactor surveillance data. That is, the relationship of the neutron damage observed at surveillance locations to that occurring within the pressure vessel wall, which at present is the purview of other Matrix E 706 standards discussed in 1.2. This practice, therefore, deals primarily with the physics-dosimetry aspects of surveillance programs.

1.2 The physics-dosimetry relationships determined from this practice may be used to estimate pressure vessel damage through application of Matrix E 706 (ID), (IE), and Guide E 900 (IIF) standards, using fluence (E > 1.0 MeV), dpa, or damage function derived exposure parameters as independent exposure variables. Supporting the applications of these standards is a set of Matrix E 706 (IIA–IIE) and Matrix E 706 (IIIA–IIIE) standards, identified in 2.1.

1.3 *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.*

[1] This practice 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 appended to this practice.

2. Referenced Documents

2.1 *ASTM Standards:*

C 859 Terminology Relating to Nuclear Materials[3]

E 170 Terminology Relating to Radiation Measurements and Dosimetry[4]

E 184 Practice for Effects of High-Energy Neutron Radiation on the Mechanical Properties of Metallic Materials, (IB)[4,5]

E 185 Practice for Conducting Surveillance Tests for Light Water-Cooled Nuclear Power Reactor Vessels, (IF)[4,5]

E 482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, (IID)[4,5]

E 509 Guide for In-Service Annealing of Light-Water Cooled Nuclear Reactor Vessels[4]

E 636 Practice for Conducting Supplemental Surveillance Tests for Nuclear Power Reactor Vessels, (IH)[4,5]

E 693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), (ID)[4,5]

E 706 Master Matrix for Light-Water Reactor Pressure Vessel Surveillance Standards[4]

IE Damage Correlation for Reactor Vessel Surveillance[6]

IG Determining Radiation Exposure for Nuclear Reactor Vessel Support Structures[6]

II Analysis and Interpretation of Physics Dosimetry Results for Test Reactors[6]

IIA Application of Spectrum Adjustment Methods[6]

IIB Application of ENDF/A Cross Section and Uncertainty Files[6]

IIE Benchmark Testing of Reactor Vessel Dosimetry[6]

IIIA Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance[6]

IIID Application and Analysis of Damage Monitors for Reactor Vessel Surveillance[6]

E 844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, (IIC)[4,5]

E 853 Practice for Analysis and Interpretation of Light-Water Reactor Surveillance Results, (IA)[4,5]

E 854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, (IIIB)[4,5]

E 900 Guide for Predicting Neutron Radiation Damage to Reactor Vessel Materials, (IIF)[4,5]

E 910 Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, (IIIC)[4,5]

2.2 *American Society of Mechanical Engineers Standard:*

Boiler and Pressure Vessel Code, Sections III and XI[7]

2.3 *Nuclear Regulatory Document:*

Code of Federal Regulations, Chapter 10, Part 50, Appendixes G and H[8]

[3] *Annual Book of ASTM Standards*, Vol 12.01.

[4] *Annual Book of ASTM Standards*, Vol 12.02.

[5] The reference in parentheses refers to Section 5 as well as Figs. 1 and 2 of Matrix E 706.

[6] For standards that are in the draft stage and have not received an ASTM designation, see Section 5 as well as Figs. 1 and 2 of Matrix E 706.

3. Significance and Use

3.1 *Regulatory Requirements*—The Code of Federal Regulations (10CRF Part 50, Appendix H) requires the implementation of a reactor vessel materials surveillance program for all operating LWR's **(10).** The purpose of the program is to (*1*) monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and the thermal environment, and (*2*) make use of the data obtained from the surveillance program to determine the conditions under which the vessel can be operated with adequate margins of safety throughout its service life. Matrix E 706 (IF), Practice E 185, derived mechanical property data, and (r, θ, z) physics-dosimetry data (derived from the calculations and cavity and surveillance capsule measurements **(1)** using Matrix E 706 physics-dosimetry standards) can be used together with information in Matrix E 706 (IIF) and Refs. 411-18to provide a relation between property degradation and neutron exposure, commonly called a "trend curve." To obtain this trend curve at all points in the pressure vessel wall requires that the selected trend curve be used together with the appropriate (r θ, z) neutron field information derived by use of this practice to accomplish the necessary interpolations and extrapolations in space and time.

3.2 *Neutron Field Characterization*—The tasks required to satisfy the second part of the objective of 3.1 are complex and are summarized in the Annex of Matrix E 706 (IA) and Practice E 853. In doing this, it is necessary to describe the neutron field at selected (r, θ, z) points within the pressure vessel wall. The description can be either time dependent or time averaged over the reactor service period of interest. This description can only be obtained by combining neutron transport calculations with cavity and surveillance capsule measurements, benchmark irradiations of dosimeter sensor materials, and a knowledge of the core power distribution, including either the time dependence, or time averaged. Because core power distribution may change with time, the cavity or surveillance capsule measurement obtained early in plant life may not be representative of long-term reactor operation. Therefore, a simple normalization of neutron transport calculations to dosimetry data from a given capsule is unlikely to give a satisfactory solution to the problem over the full reactor lifetime. Matrix E 706 (IID), Guide E 482, and Matrix E 706 (IIA) standards provide detailed information related to the characterization of the neutron field for BWR and PWR power plants.

3.3 *Fracture Mechanics Analysis*—Currently, operating limitations for normal heat up and cool down transient imposed on the reactor pressure vessel are based on the fracture mechanics techniques outlined in the ASME Boiler and Pressure Vessel Code. This code requires the assumption of the presence of a surface flaw of depth equal to one fourth of the pressure vessel thickness. In addition, the fracture mechanics analysis of accident-induced transients (Pressurized Thermal Shock, (PTS)) may involve evaluating the effect of flaws of varying depth within the vessel wall **(4).** Thus, information is required regarding the distribution of neutron exposure and the corresponding radiation damage within the pressure vessel, both in space and time **(4).** In this regard, Matrix E 706 (IF) and Practice E 185 standards provide guidelines for designing a minimum surveillance program, selecting materials, and evaluating metallurgical specimen test results for BWR and PWR power plants.

3.4 *Neutron Spectral Effects and DPA*—Analysis of the neutron fields of operating power reactors has shown that the neutron spectral shape changes with radial depth into the pressure vessel wall **(2, 3).** The ratio of dpa/ϕt (E > 1.0 MeV) changes by factors of the order of 2.0/1.0 in traversing from the inner to the outer radius. Since dpa has been found to provide a more satisfactory correlation with property degradation than fluence (E > 1.0 MeV) (see Refs **1** and **19**) it is necessary to calculate and report both quantities; see Matrix E 706 (IA), Practice E 853, Matrix E 706 (ID), and Practice E 693.

3.5 *In-Vessel Surveillance Program*:

3.5.1 The neutron dosimetry monitors used in reactor vessel surveillance capsules provide measurements of the neutron fluence and fluence rate at single points within the reactor and near the vessel wall; that is, at the surveillance capsule locations **(1).** In actual practice, the surveillance capsules may be located within the reactor at an azimuthal position that differs from that associated with the maximum neutron exposure (or that differs from the azimuthal and axial location of the assumed flaw); and at a radial position a few centimetres or more from the flaw and pressure vessel wall **(4, 5).** Although the surveillance capsule dosimetry does provide points for normalization of the neutron physics transport calculations, it is still necesary to use analytical methods that provide an accurate representation of the spatial variation of the neutron fluence, see Matrix E 706 (IID) and Guide E 482.

3.5.2 The neutron fluence calculation on the PV inner surface can be further verified by means of the "scratch sampling" neutron fluence measurement method. During the reactor shut down periods, small samples (50–100 mg) can be taken from the PV inner steel plating. From the measured $_{54}$Mn, ^{58}Co, and eventually ^{93m}Nb activities, the fast neutron fluence distribution and its maximum on the PV inner surface can be determined. By comparison of these data to the dosimetry data of the surveillance capsules, the lead factor can also be obtained.

3.6 *Ex-Vessel Surveillance Program*—Calculations of neutron fields in commercial reactors show that the neutron exposure (dpa) at the inner diameter of the pressure vessel varies by factors of the order of 3.0/1.0 for various azimuthal positions **(2, 3).** Dosimetry monitors in the cavity outside the pressure vessel are a useful tool, therefore, in determining the accuracy of the neutron field calculations at points inside the pressure vessel wall. Matrix E 706 (IA) and Practice E 853 standards recommend the use of ex-vessel cavity dosimetry measurements for verification of the physics transport calculations. The status of benchmark field and power reactor applications as well as studies of this approach are discussed in Refs **1, 18-36.**

4. Analytical Procedures

4.1 *Basic Approach*—Several auxiliary ASTM practices cover various aspects of the extrapolation problem (see 2.1). The basic approach is that a benchmarked Matrix E 706 (IID) and Guide E 482, transport calculation is to be used to supply the neutron field information at the (r, θ, z) points in the pressure vessel wall where property deterioration information will be calculated using Matrix E 706 (IIF), Guide E 900, or other trend curves **(411-18).** 4The dosimetry information obtained from cavity and surveillance capsule measurements is to be used to adjust the transport results and ensure that the transport calculation is valid. The adjustments are to be accomplished using the guidelines presented in Matrix E 706 (IIA). Dosimetry from monitors in the cavity and surveillance capsules will be used in establishing uncertainties for the calculated neutron field at selected (r, θ, z) positions in the pressure vessel wall. Time dependence of the core power distribution (due to burnup within a given cycle, or due to variations in cycle to cycle loading), surveillance capsule perturbation effects, and dosimetry monitor experimental effects must be recognized as complications, and these effects must be accounted for in the calculation and adjustment methods chosen **(1-6, 11).**

4.2 *Spatial Extrapolations*:

4.2.1 *Transport Codes*—In general, a two dimensional [(r, θ), (x, y)] transport code is needed for the calculation of the neutron and gamma fields in the region from the core to the interior of the biological shield beyond the pressure vessel. The methods of Matrix E 706 (IID) and Guide E 482 should be followed for the calculations and Matrix E 706 (IIA) for measured dosimetry adjustments. The mesh should be fine enough in all regions of importance so that diamond difference breakdown difficulties are avoided in a discrete ordinate method. Methods of ensuring that the mesh is sufficiently fine are the province of Guide E 482. If cavity dosimeter measurement results are used, the modeling in the cavity and external shield should be adequate to provide usable calculations for the neutron field in the cavity region. This requires an attention to mesh size in the ex-vessel region and an accurate representation of the chemical makeup of the external shield. Adequacy of methods of calculation and adjustments for the cavity region are also the province of Guide E 482 and Matrix E 706 (IIA).

4.2.1.1 *Benchmarking*—It is not the purpose of this practice to dictate the type of transport calculation to be used in the region between the core and the outer radius of the pressure vessel or the adjustment procedures, but any such calculations or adjustment procedures should be adequately benchmarked by a test calculation of well defined problems (for example, PCA Blind Test **(21),** VENUS **(32),** NESDIP **(33),** BWR **(25, 26),** and PWR **(1, 20, 27-30)**). For further details see Matrix E 706 (IID) and IIA).

4.2.1.2 *Calculation Steps*—With reference to Matrix E 706 (IA) and Practice E 853, the steps to be taken in the overall calculations are as follow:

4.2.1.3 *Power Distribution*—As discussed in Matrix E 706 (IA) and Practice E 853, obtain a valid time averaged core power distribution using a diffusion calculation, or a transport calculation, but in either case obtain experimental verification of the accuracy of the results **(20, 21, 32).** A time dependent approach is also acceptable, with appropriate documented procedures for the remaining parts of the extrapolation.

4.2.1.4 *Ex-Core Regions*—Perform a transport calculation for the neutron field in all ex-core regions, using adequate modeling of the surveillance capsules, and adequate modeling of the ex-vessel region (adequacy depending on whether or not ex-vessel dosimetry has been used in the verification of the extrapolation). The neutron balance should be checked in all regions to make sure the calculation has converged. Further, the transport calculation should be benchmarked following requirements of Matrix E 706 (IID).

4.2.2 *Dosimetry Sensor Analysis*—For analysis of any given set of cavity or surveillance capsule dosimetry sensors, the integral reactions or reaction rates of the individual sensors, or both, should be calculated, using the results of the transport calculation. The measurement and analysis procedures for individual Radiometric Monitors (RM), Solid State Track Recorders (SSTR), Helium Accumulation Fluence Monitors (HAFM), and Damage Monitors (DM) should be benchmarked for each sensor type, using reference neutron fields (for example, NBS or MOL 235 fission spectrum cavities), see Matrix E 706 (IIC), (IIE), (IIIA), (IIIB), (IIIC), and (IIID) (See 2.1). If the calculated and experimental integral results (C/E ratios) agree to within the required accuracy ($\pm$ 5 to 15 %, 1σ being the best attainable, see Ref **21**) expected from the benchmark calibration of the transport code, the transport calculation may be used directly to calculate the neutron field at all (r, θ, z) points in the pressure vessel wall. If the C/E ratios do not agree within acceptable accuracy limits, a physics-dosimetry adjustment code analysis should be performed as outlined in 4.2.3.

4.2.3 *Physics-Dosimetry Adjustment Code Analysis*—Matrix E 706 (IIA) should be used to combine the transport calculation with the dosimeter results. Matrix E 706 (IIA) adjustment procedure should be used to indicate whether the dosimeter measurements and associated uncertainties are consistent with the transport calculation and with uncertainties implied from benchmark tests of the transport code (PCA, VENUS, NESDIP, and an appropriate Commercial BWR or PWR; see Refs **1, 20, 21, 25-30**). Having established the required consistency, the adjusted transport code results may be used to calculate the neutron field at all points in the pressure vessel wall with the uncertainty estimates derived from the application of the adjustment codes. Direct use of the transport code results with appropriate bias factors and uncertainties is another acceptable approach.

4.2.3.1 *Surveillance Capsule Results*—If the calculated neutron field at the surveillance capsule is inconsistent with the experimental dosimetry results, an attempt should be made to uncover and correct errors in order to obtain consistency. Particular attention will be required to sensor monitor correction factors for perturbation, photo-reaction, impurity, burn-in, and other effects.

4.2.3.2 If the transport result indicates a higher flux than that indicated by the dosimetry, the transport result can be used for extrapolation purposes, but with an appropriate increase in the stated uncertainty for the results.

4.2.3.3 If the transport calculation indicates a lower flux than that which would be consistent with the dosimetry (taking account of the uncertainties in both the dosimetry and transport results) and if the discrepancy cannot be resolved, then the transport results should be scaled up proportionally to obtain agreement, following which the transport results are to be used for extrapolation purposes. In this case, appropriate increases should be made in the stated uncertainties of the final result, and documented logic should be provided to defend the assigned uncertainties.

4.2.4 *Ex-Vessel Surveillance Results*—Ex-vessel cavity dosimetry is to be treated in the same manner as surveillance capsule dosimetry, but care must be exercised to ensure that the physics calculation modeling is adequate and includes the proper modeling of the cavity surveillance capsule and any covers, as well as any nearby vessel support members.

4.2.4.1 The biological shield is accurately modeled.

4.2.4.2 In the final calculation of the neutron and gamma field at any point in the vessel wall, proper statistical weight should be given to ex-vessel dosimetry, taking account of modeling problems as well as the possibility that a larger logarithmic extrapolation or interpolation in absolute flux value exists from ex-vessel positions to a ¼ T location when compared to the extrapolation or interpolation from an internal surveillance capsule position to a ¼ T location.

4.2.5 *Power Plant Dimensions*—In all calculations, as-built dimensions should be used. If they are unavailable, documented logic should be presented to defend the dimensions used, and the uncertainty in the final results should reflect the added uncertainty. It should be noted that dpa declines ~10 %/cm of radial travel, in water, and deviations of ~3 cm between design dimensions and as-built dimensions have been observed in commercial reactors.

4.3 *Time Extrapolations*—In the case where a time averaged core loading has been used to define the neutron source term, the fluence or dpa in future years is estimated by multiplying by the expected integrated time at full power. Existing problems associated with time extrapolations (for example, saturation effects and differences in the slope of trend curves for different ferritic steels) are addressed elsewhere. The reader is referred to Refs **1, 6, 11-18, 23,** Matrix E 706 (IIF), and Guide E 900 for more information on these subjects.

5. Report and Bias of Results

5.1 As a minimum, the documentation of results should include the following information:

5.1.1 A description of the analytical technique used, including a listing of pertinent input parameters that may affect the bias of the calculation. For example, if the discrete ordinates approach is used, specify or reference the cross-section preparation procedures, energy group structure, spatial mesh, S_n order, and P_l order.

5.1.2 Information indicating the bias of the analytical approach in steel-water systems, including the details of benchmark calculations used to validate the procedures, and data and the bias attained in the benchmark tests.

5.1.3 The calculated total, thermal, epi-thermal (also known as epi-cadmium flux) E > 0.1 MeV, E > 1.0 MeV neutron flux-fluence values, and energy spectrum at the surveillance capsule, and any ex-vessel dosimetry locations. Also calculated values of dpa/s and dpa at the same locations.

5.1.3.1 The location of peak flux-fluence points on the surface and in the interior of the vessel wall are calculated values that are required for all the above exposure and exposure rate parameters, except for the thermal and epithermal fluxes, which generally can be best determined by dosimetry measurements. For some damage analysis studies, all of the above information is needed **(36-41).**

5.1.3.2 At dosimetry measurement locations, gamma ray flux-fluence should be estimated to the bias required to make necessary photo reaction corrections. Similarly, gamma field parameters should be estimated to whatever bias is needed to allow temperature corrections for radiation damage in PV steels and in surveillance capsule mechanical property specimens.

5.1.4 Methods and pertinent parameters used in the physics-dosimetry analysis must be documented or referenced, including appropriate tabulations of all measured individual sensor results and uncertainties. Methods of extrapolation and interpolation must specifically be delineated.

5.1.5 Details must be given relative to the methods used to assign uncertainties for calculated values of neutron flux, fluence, dpa/s, and dpa. Uncertainties for calculated values for total, thermal, E > 0.1 MeV, and E > 1.0 MeV neutron fluxes and fluences should be provided.

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(**41**) Serpan Jr., C. Z., and McElroy, W. N., “Damage Function Analysis of Neutron Energy and Spectrum Effects Upon the Radiation Embrittlement of Steels,” *NRL Report 6925*, July 1969.