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Standard Practice for Determining ~~Radiation Exposures~~ NeutronExposures for Nuclear Reactor Vessel Support Structures[1]

[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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1. Scope

1.1 This practice covers procedures for monitoring the neutron radiation exposures experienced by ferritic materials in nuclear reactor vessel support structures located in the vicinity of the active core. This practice includes guidelines for:

1.1.1 Selecting appropriate dosimetric sensor sets and their proper installation in reactor cavities.

1.1.2 Making appropriate neutronics calculations to predict neutron radiation exposures.

1.2 This practice is applicable to all pressurized water reactors whose vessel supports will experience a lifetime neutron fluence ($E > 1$ MeV) that exceeds 1×10^{17} neutrons/cm^2 or 3.0×10^{-4} dpa.[2] (See Terminology E 170.)

1.3 Exposure of vessel support structures by gamma radiation is not included in the scope of this practice, but see the brief discussion of this issue in 3.2.

1.4 *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:*

E 170 Terminology Relating to Radiation Measurements and Dosimetry[3]

E 482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, E 706 (IID)[3,4]

E 693 Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacements Per Atom (DPA), E 706 (ID)[3,4]

E 844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance, E 706 (IIC)[3,4]

E 854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, E 706 (IIIB)[3,4]

E 910 Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, E 706 (IIIC)[3,4]

E 944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, E 706 (IIA)[3,4]

E 1005 Test Method for Application and Analysis of Radiometric Monitors for Reactor Vessel Surveillance, E 706 (IIIA)[3,4]

E 1018 Guide for Application of ASTM Evaluated Cross Section Data ~~File (ENDF/A),~~ File, E 706 (IIB)[3,4]

2.2 *ASME Standard:*

Boiler and Pressure Vessel Code, Section III[5]

2.3 *Nuclear Regulatory Documents:*

Code of Federal Regulations, "Fracture Toughness Requirements," Chapter 10, Part 50, Appendix G[6]

Code of Federal Regulations, "*Reactor Vessel Materials Surveillance Program Requirements*," Chapter 10, Part 50, Appendix H[6]

[2] Based on data from Table 5 of Master Matrix E 706 and Reference 5.

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

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

[5] Available from American Society of Mechanical Engineers, 345 E. 47th St., New York, NY 10017.

[6] Available from Superintendent of Documents, U.S. Government Printing Office, Washington, DC 20402.

Regulatory Guide 1.99, Rev. 1, "*Effects of Residual Elements on Predicted Radiation Damage on Reactor Vessel Materials*," U. S. Nuclear Regulatory Commission, April 1977[6]

3. Significance and Use

3.1 Prediction of neutron radiation effects to pressure vessel steels has long been a part of the design and operation of light water reactor power plants. Both the federal regulatory agencies (see 2.2) and national standards groups (see 2.1) have promulgated regulations and standards to ensure safe operation of these vessels. Recently, it has become apparent that the support structures for pressurized water reactor vessels may also be subject to similar neutron radiation effects **(1, 2, 3, 4, 5)**.[7] The objective of this practice is to provide guidelines for determining the neutron radiation exposures experienced by individual vessel supports.

3.2 It is known that high energy photons can also produce displacement damage effects that may be similar to those produced by neutrons. These effects are known to be much less at the belt line of a light water reactor pressure vessel than those induced by neutrons. The same has not been proven for all locations within vessel support structures. Therefore, it may be prudent to apply coupled neutron-photon transport methods and photon induced displacement cross sections to determine whether gamma-induced dpa exceeds the screening level of 3.0×10^{-4}, used in this practice for neutron exposures. See 1.2.

4. Irradiation Requirements

4.1 *Location of Neutron Dosimeters*—Neutron dosimeters shall be located along the support structure in the region where the maximum dpa or fluence (E > 1 MeV) is expected to occur, based on neutronics calculations outlined in Section 5. Care must be taken to ensure that reactor cavity structures not modeled in the neutronics calculation offer no additional shielding to the dosimeters. The neutron dosimeters will be analyzed to obtain a map of the neutron fields within the actual location of the support structures.

4.2 *Neutron Dosimeters*:

4.2.1 Information regarding the selection of appropriate sensor sets for support structure application may be found in Guide E 844, Test Method E 1005, and Test Methods E 854 and E 910.

4.2.2 In particular, Test Method E 910 also provides guidance for the additional possibility that operating plants may use existing copper bearing instruments and cables within the reactor cavity as a priori passive dosimeter candidate.

5. Determination of Neutron Exposure Parameter Values

5.1 *Neutronics Calculations*—All neutronics calculations for (*a*) the analysis of integral dosimetry data, and (*b*) the prediction of irradiation damage exposure parameter values shall follow Guide E 482, subject to these additional considerations that may be encountered in reactor cavities:

5.1.1 If the vessel supports do not lie within the core's active height, then an asymmetric quadrature set must be chosen for discrete ordinates calculations that will accurately reproduce the neutron transport in the direction of the supports. Care must be exercised in constructing the quadrature set to ensure that "ray streaming" effects in the cavity air gap do not distort the calculation of the neutron transport.

5.1.2 If the support system is so large or geometrically complex that it perturbs the general neutron field in the cavity, the analysis method of choice may be that of a coupled discrete ordinates/Monte Carlo calculation. The normal coupling for this type of problem would be to perform the two dimensional discrete ordinates analysis only within the vessel. The neutron currents generated by this analysis would be used to create the appropriate cumulative distribution functions in the final Monte Carlo analysis. For details of such analyses see Refs **(6), (7), and (8). In this instance, the above caveats still hold for the discrete ordinates calculation, but in addition, the variance of the Monte Carlo results must now be included with the overall assessment of the variance of the dosimetry data.**

5.2 *Determination of Damage Exposure Values and Uncertainties*—Adjustment procedures outlined in Guide E 944 and Guide E 1018 shall be performed to obtain damage exposure values dpa and fluence (E > 1 Mev) using the integral data from the neutron dosimeters and the calculation in 5.1. The cross sections for dpa are found in Practice E 693. Dpa shall be determined for this application rather than just fluence (E > 1 MeV) because Ref **(5)** notes an increase in the ratio of dpa to fluence (E > 1 MeV) by a factor of two in going from the surveillance capsule position inside the reactor vessel to a position out in the reactor cavity.

[7] The boldface numbers in parentheses refer to a list of references at the end of this practice.

REFERENCES

(**1**) Docket 50338-207, North Anna Power Station, Units 1 and 2, *Summary of Meeting Held on September 19, 1975 on Dynamic Effects of LOCAs*, Sept. 22, 1975.
(**2**) Sprague, J. A., and Hawthorne, J. R., "Radiation Effects to Reactor Vessel Supports," U. S. Naval Research Laboratory Report NRC-03-79-148 for the U. S. Nuclear Regulatory Commission, Oct. 22, 1979.
(**3**) *Unresolved Safety Issues Summary*, NUREG-0606, Vol 4, No. 4, Task A-11: *Reactor Vessel Materials Toughness*, November, 1982.
(**4**) *Asymmetric Blowdown Loads on PWR Primary Systems*, NUREG-0609, U.S. Nuclear Regulatory Commission, 1981.
(**5**) Hopkins, W. C., "Suggested Approach for Fracture-Safe PRV Support Design in Neutron Environments," *Transactions of the American Nuclear Society*, Vol 30, 1978, pp. 187–188.
(**6**) Cain, V. R., "The Use of Monte Carlo with Albedos to Predict Neutron Streaming in PWR Containment Buildings," *Transactions of the American Nuclear Society*, Vol 23, 1976, p. 618.
(**7**) Straker, E. A., Stevens, P. N., Irving, D. C. and Cain, V. R., "The MORSE Code—A Multigroup Neutron and Gamma-Ray Montre Carlo Transport Code," ORNL-4585, September 1970.
(**8**) Emmett, M. B., Burgart, C. E., and Hoffman, T. J., "DOMINO: A General Purpose Code for Coupling Discrete Ordinates and Monte Carlo Radiation Transport Calculations," ORNL-4853, July 1973.