REACTOR PRESSURE VESSEL FLUENCE
A PROPOSED ANS STANDARD

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ABSTRACT

The purpose of the proposed standard is to delineate a technically sound method to determine the pressure vessel fast neutron fluence based on input from all stakeholders. Fast neutron irradiation (E > 1.0 MeV) of the reactor vessel contributes to material embrittlement. The amount of the embrittlement also depends on the material chemistry and in particular the presence of the trace elements Copper and Nickel. To assure that the material meets regulatory requirements (for example 10 CFR 50.61 [1] and Appendix G to 10 CFR 50 [2]) an accurate fluence value is required. Surveillance capsule dosimetry (required by Appendix H to 10 CFR 50 [3]) and/or reactor cavity dosimetry provide measured activation values at the point of the irradiation. Conversion of dosimeter activation into flux (or fluence) requires knowledge of the neutron energy spectrum at the point of the irradiation. The energy spectrum is usually derived as a byproduct of the multigroup neutron transport calculations performed to estimate fluence at the irradiation points and the reactor vessel beltline. In many cases, the measured dosimetry activation values and the corresponding calculated values show significant discrepancies: (1) between measured values at the same location and (2) between measured and the corresponding calculated values. Various adjustments methods have been devised to extract a representative measured value from measured data. The ANS 19.10 discussions were focused on PWRs and associated measurement and dosimetry practices, at this stage the standard is applicable to PWRs only. This paper summarizes the major features of the proposed standard.

1. INTRODUCTION

The purpose of this paper is to summarize the technical aspects of the proposed standard "Fast Neutron Fluence in Light Water Reactor Pressure Vessels" of the ANS 19.10 standards subcommittee [4]. Knowledge of the pressure vessel fluence is required for safe nuclear power plant operation. Neutron fluence and material chemistry are needed to quantify the embrittlement caused by the continuous fast neutron ($E > 1.0$ MeV) irradiation. The material toughness is measured in terms of the reference temperature of the nil-ductility transition RT_{NDT} which depends on the initial material properties, material chemistry and fast neutron fluence. The proposed standard provides guidance for estimating fast neutron fluence to reactor pressure vessels, however, it is not a detailed prescriptive guide, but rather it is a methodology which covers the areas of general agreement based on physical principles. The proposed standard reflects the persistence, expertise and technical capabilities of the members of the ANS 19.10 subcommittee.

The standard is divided into three sections which address: (1) Transport Calculations (2) Dosimetry Measurements and (3) determination of the Fluence Best Estimate. This organization corresponds to the technical areas involved in determining fast neutron fluence to the pressure vessel when measured data are available. The standard also addresses the situation where there are no measured data available. The following sections discuss the major issues in each of the technical areas.

2. TRANSPORT CALCULATIONS

Either a discrete ordinates or a Monte Carlo computational method is acceptable for the neutron transport calculations necessary to estimate the fluence at the pressure vessel, the neutron spectrum and the dosimeter activities. Three-dimensional transport codes are desirable if the computational facilities are available (i.e., fast CPU and large memory or parallel computing machines). However, two dimensional (r,è) and (r,z) flux solutions which are synthesized into three dimensional (r,è,z) flux (fluence) distributions are also acceptable. Expressions are given for the calculation of the dosimeter material reaction rates. When performing cycle-by-cycle vessel fluence analyses, adjoint calculations can be used to precalculate the fluence contribution from each fuel assembly in order to avoid repeating forward calculations. The required input data for material composition, geometry and cross sections are described. Recommendations are made for the use of the pin power distribution in determining the core neutron source data. The transport code should be qualified by comparison of the calculated results to benchmark measurements and calculations. Code benchmarking is described along with the estimation of the calculational uncertainties. The transport calculation section concludes with a description of the reporting requirements, a description of the output data, a sample of a neutron spectrum and bibliography.

3. DOSIMETRY MEASUREMENTS

Neutron dosimeters are typically in the form of a thin foil or encapsulated powder and are strategically located inside or outside the pressure vessel to record the local neutron activation. The full power neutron flux is determined from the measured activation using the calculated neutron activation cross section, core power history and calculated local neutron spectrum. The proposed standard describes the general physical, chemical and nuclear characteristics to be considered in the selection of reactor dosimeters. Physical and chemical properties such as the melting point, chemical stability and resistance to corrosion are discussed. Similarly, the required nuclear properties, such as cross sections, the decay half life and the fission yield for fissionable type dosimeters are discussed. Comments and guidance are provided on dosimeter mass, isotopic composition, photon yield, photon energy and potential sources of contamination. With respect to dosimeter suitability for counting, the size (thickness in particular) and the geometrical configuration are discussed. The important aspect of spectral coverage is discussed and dosimeter threshold energies are presented for typical dosimeters. The standard discusses the irradiation location and possible effects of the surrounding structures. The standard describes dosimeter encapsulation and dosimeter filtering.

The irradiation parameters play an important role in the dosimeter activation and the subsequent counting and evaluation. The reactor power level and the temporal history should be known and included. Likewise, the presence of dosimeter reaction products and dosimeter burn-up should be considered. Finally, the standard discusses equipment requirements for dosimeter counting. The ever present problem of interference from other isotopes with decay gammas in the same energy range as the desired isotope is also discussed. Methods of resolving such problems are mentioned; for example, by radio-chemical separation or by solvent extraction. Reporting requirements are also enumerated.

The special case of the stable product neutron monitors are described. This category includes the solid state track recorders and the Helium accumulation fluence monitors. These are direct fluence recorders and provide a permanent record and reference for the accumulated fluence.

Determination of the fluence at the location of the dosimeter is obtained using elaborate transport calculations. The dosimeter activity is one of the quantities which can be calculated and, thus, it offers the means for a direct comparison of experiment and calculation. However, dosimetry measurements must be corrected for the effect of interfering (nuclear) reactions and when applicable detector burn-up and photo-fission. A meaningful comparison of the measured and calculated quantities requires an uncertainty evaluation. Examples of dosimetry uncertainties are: dosimeter mislocation, dosimeter physical characteristics, irradiation spectrum, nuclear data and nuclear counting. When using the measurements to validate the calculation, the measurement process should be validated with standard neutron fields.

4. BEST ESTIMATE FLUENCE

As indicated above, the transport methodology yields calculated values while the dosimetry (in combination with the transport results) yields “measured” values. Either dosimeter activation or fluence can be used to form the measured-to-calculated (M/C) data bases. Experience has shown that the M/Cs are sensitive to a multiplicity of parameters such as capsule location, dosimeter material, counting procedures, etc. In addition, in many cases M/C values show inconsistencies between sister plants for the same dosimeters irradiated under similar conditions. Because of the many sources of uncertainty in the M/Cs, data bases used to determine a fluence best estimate should be drawn from a broad set of measurements. The measured data must be of sufficient quantity and quality, and must be subjected to appropriate qualification tests to assure that the data is uniform and belong to a single statistical group. The calculational bias may be determined using the qualified data base together with a qualified least squares method. In the absence of measured data the calculated fluence value derived using a benchmarked code is acceptable.

5. REFERENCES

1 Code of Federal Regulations, Title 10, Part 50, Section 61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events" Office of the Federal Register, January 1, 1999.

2 Code of Federal Regulations, Title 10, Part 50, Appendix G, "Fracture Toughness Requirements" Office of the Federal Register, January 1, 1999.

3 Code of Federal Regulations, Title 10, Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements" Office of the Federal Register, January 1, 1999.

4 ANS 19.10 Standards Subcommittee, Proposed Draft Standard on "Reactor Pressure Vessel Fluence" January, 2000.