RG 1.190 - CALCULATIONAL AND DOSIMETRY METHODS FOR DETERMINING PRESSURE VESSEL FLUENCE[1]

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ABSTRACT

This paper summarizes experience and practice reflected in the recently issued Regulatory Guide RG 1.190 for the estimation of the pressure vessel fluence for US built PWR and BWR plants. Fast neutron fluence (E > 1.0 MeV) is an important component in the determination of the material properties of an irradiated vessel. Accurate knowledge of the material properties is essential in the application of the regulations for the protection of public health and safety. Reliable fluence values are needed for the current material properties and for future operation. This is becoming more important in view of present and future license extension activities. In view of experience gained from surveillance capsule measurements and the corresponding fluence calculations, a calculational methodology in which the fluence is determined using a benchmarked computer code and a measurement-to-calculation (M/C) bias, based on a qualified data base, is considered an acceptable method for the calculation of the vessel fluence. To insure a reliable fluence estimate, the qualification of the calculation methodology should include both: (1) comparisons to calculation and measurement benchmarks and (2) an analytic uncertainty analysis.

1. BACKGROUND

To assure protection of the public health and safety in its regulation of the use of nuclear power plants, the United States Nuclear Regulatory Commission (USNRC) has embraced the multibarrier concept. The reactor coolant pressure boundary (RCPB) is one of the three barriers for preventing release of radioactive materials. (The other two are the fuel cladding

1 Opinions and/or positions expressed in this paper do not represent those of the USNRC.

and the containment.) General Design Criterion (GDC) 14 requires that the RCPB "be designed, fabricated, erected , and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture" [1]. GDC 30 requires that "The reactor coolant boundary be designed, fabricated, erected, and tested to the highest quality standards practical." [1]. GDC 31 requires that the pressure boundary shall be designed with sufficient margin to assure that "...it behaves in a non-brittle manner and the design shall reflect consideration of uncertainties...on the effect of irradiation on material properties." [1].

To fulfill the requirements of the GDCs, the USNRC incorporated (in Appendix G to Title 10 of the Code of Federal Regulations (CFR) part 50 [2]) the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section III, Division 1, "Rules of Construction of Nuclear Power Plant Components," and Section Xl, Division 1 "Rules for Inservice Inspection of Nuclear Power Plant Components". In addition, Appendix H to 10 CFR part 50 requires licensees to establish a materials surveillance program to monitor those changes to material fracture toughness induced by neutron irradiation and the thermal environment [3]. This monitoring activity will make use of surveillance capsules to irradiate archival material samples. 10 CFR 50 section 61 (10 CFR 50.61) and Regulatory Guide 1.99, Revision 2, outline the material fracture toughness requirements for protecting from pressurized thermal shock [4, 5]. For a given amount of irradiation, material toughness is measured in terms of the reference temperature RT_{NDT} as described in References 2 and 5. The RT_{PTS} reference temperature (Reference-4) is used to monitor the embrittlement of the pressure vessel at future times (usually at the end of plant license). RT_{PTS} includes an allowance for uncertainties and incorporates the risk of a through-wall vessel crack during a transient which may result in pressurized thermal shock (PTS). Calculation of RT_{PTS} involves pre-irradiation material properties, the copper and nickel content, and an estimate of the future value of the neutron fluence [4]. In summary, the GDCs require that: (1) vessels have an extremely low probability of failure for normal operation and in the case of potential transients and (2) the vessel fluence used in the determination of this probability must account for uncertainties.

2. INTRODUCTION

The Code of Federal Regulations does not contain specific guidance on calculating the vessel fluence or on what would be an acceptable practice. The USNRC has recently (March 2001) published a regulatory guide delineating acceptable practices in estimating the pressure vessel fluence (Regulatory Guide 1.190) [6]. 10 CFR 50.61 requires a fluence best estimate. The USNRC, in order to satisfy the GDC requirements, needs to evaluate potential sources of uncertainty for calculated fluence values. This is interpreted to mean that uncertainty components should be evaluated and used for estimating the total uncertainty to assure it does not exceed 20% (1σ) which is implicit in the margin term of 10 CFR 50.61 [4]. These uncertainty components typically include: configuration geometry, material compositions, material densities, material cross-sections, and code numerical uncertainties. Projected fluence values in addition require knowledge (or an assumption) of future core loadings and the associated core neutron leakage to define the neutron source.

3. FLUENCE CALCULATIONAL METHODS

Regulatory Guide 1.190 recommends that the modeling (geometry, materials, etc.) used in the vessel fluence calculation be based on documented and verified plant-specific data. The core neutron source input to the transport calculation should account for the local fuel isotopics and, where appropriate, the moderator density. The neutron source normalization and energy dependence must account for the fuel exposure dependence of the fission spectra, the number of neutrons produced per fission, and the energy released per fission.

The spatial representation in discrete ordinates transport calculations should incorporate a detailed radial- and azimuthal-spatial mesh of about 2 intervals per inch radially. The discrete ordinates calculations must employ (at a minimum) an S_8 quadrature and (at least) 40 intervals per octant. The adequacy of the S_8 angular quadrature used in cavity transport calculations must be demonstrated by comparison with higher order calculations.

When the transport calculation is performed using a Monte Carlo method, the estimated mean and relative error should be tested to insure that all relevant statistical criteria are satisfied. When variance reduction is used to improve the Monte Carlo sampling efficiency, the variance reduction method should be qualified by comparison with calculations performed without variance reduction.

The latest version of the Evaluated Nuclear Data File (ENDF/B) (the current version is ENDF/B-VI) should be used for determining nuclear cross-sections. Cross-section sets based on earlier or equivalent nuclear-data sets that have been thoroughly benchmarked may also be used. When the recommended cross-section data change, the effect of these changes must be evaluated and the fluence estimates updated when the effects are significant. In discrete ordinates calculations, a P_3 angular decomposition of the scattering cross-sections (at a minimum) must be employed. The adequacy of the collapsed (job library) must be demonstrated by comparing calculations for a representative configuration performed with both the master library and the job library.

To insure a reliable fluence estimate, the calculational methodology must be qualified by both: (1) comparisons to measurement and calculational benchmarks and (2) an analytic uncertainty analysis. RG 1.190 recommends specific measurement and calculational benchmarks [7-12]. The methods used to calculate the benchmarks must be consistent (to the extent possible) with the methods used to calculate the vessel fluence. The vessel fluence and the (1σ) calculational uncertainty determined by the methods qualification must be demonstrated to be . 20% for RT_{PTS} and RT_{NDT} determination. When the measurement data are of sufficient quality and quantity that they allow a reliable estimate of the calculational bias (i.e., they represent a statistically significant measurement data base), the comparisons to measurement may be used to determine a calculational bias. This bias may then be used to adjust the calculated fluence.

Predictions of the vessel end-of-life fluence should be made with a best-estimate or conservative generic core power distribution. If a best estimate is used, the power distribution must be updated if changes in core loadings, surveillance measurements, or other information indicate a significant change in projected fluence values.

Three-dimensional calculations are generally desirable. However, in most cases, the three-dimensional fluence distribution constructed by synthesis of one- and two-dimensional calculations provides an accurate fluence prediction as shown in the Reference-12 comparisons of three-dimensional Monte Carlo and synthesis calculations. In the synthesis approach, the three dimensional fluence is determined by the expression

$$\Phi_g(r, \theta, z) = \Phi_g(r, \theta)^* L_g(r, z), \tag{1}$$

where $\Phi_g(r, \theta)$ is the group-g flux in (r, θ) geometry for the plane of interest and $L_g(r, z)$ is an axial shape factor for the group-g flux. $L_g(r, z)$ can be determined from the equation

$$L_g(r, z) = \Phi_g(r, z)/\Phi_g(r), \tag{2}$$

where $\Phi_g(r)$ and $\Phi_g(r, z)$ are one- and two-dimensional group-g flux solutions, respectively. The (r, z) should represent the axial plane of interest.

An analytic uncertainty analysis is required to establish confidence in the methodology. As a first step in this analysis, the calculation methodology should be reviewed and the important sources of uncertainty identified. Typically, these include:

! Nuclear data uncertainty: transport and activation cross sections and fission spectra,

! Geometry: location of core and internal components and potential deviation from nominal values,

! Material uncertainties: density and composition of the coolant, core, core barrel, former plates, thermal shield (if present), pressure vessel, biological shield, etc.,

! Neutron source uncertainties: spatial and energy distribution and burnup dependence,

! Methods uncertainties from: angular expansion, mesh density, perturbation from a surveillance capsule, spatial synthesis, convergence criteria, cavity streaming, etc.

The above uncertainties need to be supplemented by other items deemed to be important contributors on a case by case basis. However, experience suggests that there are a few contributors such as the geometry and neutron source which dominate the fluence uncertainty. Identification of the dominant uncertainty components can result in a substantial simplification of the uncertainty analysis.

Finally, let us note that there are calculational benchmarks available (e.g., Reference-12) to evaluate the performance of the computational methodology. Such benchmarks can test the numerical operations of the codes as well as the modeling assumptions, and help to assure an accurate fluence estimate.

4. FLUENCE MEASUREMENTS

Dosimetry measurements provide independent estimates of specific activities and isotopic production rates that are used for validating the neutron transport calculations. Fluence is obtained from the response of passive integral detectors placed in surveillance capsules and, more recently, in the ex-vessel cavity. RG 1.190 recommends that the set of dosimeters used to perform the dosimetry measurements provide adequate coverage of the neutron spectrum. The selection of the dosimeter materials should address melting, oxidation, material purity, total and isotopic mass assay, perturbations due to encapsulation, thermal neutron shields, and accurate dosimeter positioning. Dosimeter half-life, photon yield and interference should also be evaluated. A typical set of threshold dosimeters is given in Table 1. ASTM Standards (as those referenced in Table-1) are available for reference on the details of the dosimetry measurements and the interpretation of the results.

The dosimeter-response measurements should account for fluence rate variations, isotopic burnup effects, detector perturbations, self shielding, reaction interferences, and photofission. In order to insure measurement reliability, an uncertainty analysis must be performed for the response of each dosimeter. In addition, detector-response calibrations must be carried out periodically in a standard neutron field.

The ($E > 1.0$ MeV) fast-neutron fluence for each measurement location is determined using a calculated spectrum-averaged cross-section and the individual detector measurement.

When a statistically significant number of measurements is available and the various dosimeter measurements are self-consistent, then a calculational bias can be determined. As more data from in-vessel capsule and cavity dosimetry become available they should be incorporated into the data base and bias estimates re-evaluated as necessary.

5. SUMMARY

RG 1.190 provides guidance for the estimation of the vessel fluence which incorporates the requirements of General Design Criteria 14, 30, and 31, the requirements of 10 CFR 50.61, the guidance in RG 1.99 Revision 2 and the experience gained from reactor vessel dosimetry. RG 1.190 provides best-estimate methods for the calculation and measurement of the vessel fluence. It is recommended that the fluence be determined using a benchmarked code and an M/C bias, based on a qualified data base. To insure a reliable fluence estimate, the methodology used to determine the fluence must be qualified by both comparisons to benchmarks and an analytic uncertainty analysis.

TABLE 1. THRESHOLD DETECTORS FOR PRESSURE VESSEL DOSIMETRY

Detector	**Nominal Threshold (MeV)**	**Applicable ASTM Standards**
^{237}Np(n,f)FP*	0.69	E 705 (Ref. 13)
^{93}Nb(n,n=)^{93m}Nb	0.97	E 1297 (Ref. 14)
^{238}U(n,f)FP	1.45	E 704 (Ref. 15)
^{58}Ni(n,p)^{58}Co	2.05	E 264 (Ref. 16)
^{54}Fe(n,p)^{54}Mn	2.32	E 263 (Ref. 17)
^{46}Ti(n,p)^{46}Sc	3.76	E 526 (Ref. 18)
^{63}Cu(n,α)^{60}Co	4.65	E 523 (Ref. 19)

* FP indicates fission product.

6. REFERENCES

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2 Code of Federal Regulations, Title 10, Part 50, Appendix G, AFracture Toughness Requirements,@ January 1, 2001.

3 Code of Federal Regulations, Title 10, Part 50, Appendix H, AReactor Vessel Material Surveillance Program Requirements,@ January 1, 2001.

4 Code of Federal Regulations, Title 10, Part 50, Section 61, AFracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events,@ January 1, 2001.

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