Reactor Vessel Dosimetry , Lessons Learned[1]

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Abstract

This paper summarizes the experience gained from extensive reviews of reactor dosimetry and describes preferred practices in the estimation of pressure vessel fluence as it relates to licensing actions. The applicable regulations require high confidence in the knowledge of vessel fluence which in turn defines the material properties and the range of temperature and pressure for plant operation. This paper supports the view that a suitable calculated fast neutron (E > 1.0 MeV) vessel fluence value is determined using a benchmarked computer code. The benchmarking should be supported by a qualified data base of measured to calculated fluence ratios. The same data allow the determination of any bias that may exist on the calculated values and the associated uncertainty. Potential fluence adjustments are discussed and it is concluded that fluence values determined in this manner are preferable to plant specific measurements and/or measurement adjustments that are not based on a benchmarked code.

Keywords: Pressure vessel, neutron fluence, fluence methodology

Background

The United States Nuclear Regulatory Commission (USNRC), to carry out its mandate of protecting public health and safety, adopted the principle of multi-barrier protection in preventing radioactive fission products from being released into the environment during the course of power production using nuclear reactors. 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 and the containment.

To implement the desired protection, Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50) Appendix A, includes General Design Criteria (GDCs) 14, 30 and 31. GDC 14, requires that the RCPB "be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, from a rapidly propagating failure, and/or from gross rupture" [*1*]. GDC 30 requires that "The reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical." GDC 31 requires that the pressure boundary 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." In addition to the GDCs, the USNRC incorporated into 10 CFR 50 the American Society of Mechanical

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1 Opinions and/or positions expressed or implied in this paper do not represent those of the US Nuclear Regulatory Commission.

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Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1, "Rules of Construction of Nuclear Power Plant Components," and Section XI, Division 1, "Rules for In-service Inspection of Nuclear Power Plant Components" [*2*]. Furthermore, 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 makes use of surveillance capsules to irradiate archival material samples. 10 CFR 50 Section 61 (10 CFR 50.61) and Regulatory Guide (RG) 1.99, Revision 2, outline the material fracture toughness requirements for protecting the reactor vessel from pressurized thermal shock [*4, 5*]. RG 1.190 outlines the attributes of neutron transport methodologies for the calculation of the vessel fluence required to determine material properties.

The metric for material toughness for a given irradiation is the reference temperature for the nil ductility transition temperature (RT_{NDT}) as described in References 2 and 5. However, the projected pressure vessel reference temperature at some future time, (usually at the end of the plant's current operating license) is called RT_{PTS} and is described in Reference 4. RT_{PTS} is a reference temperature used to monitor the vessel, includes an allowance for uncertainties and incorporates the risk of a through-wall vessel crack during a pressurized thermal shock (PTS) transient. Calculation of RT_{PTS} involves material properties, material chemistry particularly the copper and nickel content of each RCPB component, and an estimate of future values of the neutron fluence (E > 1.0 MeV) [*4*]. The objective of this paper is to outline and justify a methodology for the calculation of vessel fluence as projected into future nuclear power plant operation. This paper is based on and is limited to the technical experience of the USNRC staff and reflects the evolution of the last three decades of the reactor vessel dosimetry for domestic power plants.

Pressure vessel fluence is the starting point in determining future vessel material properties which in turn determines the allowable values of temperature and pressure for safe operation. Therefore, fluence is the basic parameter needed to satisfy the requirements of the GDCs and assure that the vessel will have the extremely low probability of failure mandated by the GDCs. To this end, the mean fluence value and its uncertainty must be quantified. It should be pointed out that fast neutron fluence (E > 1.0 MeV) is calculated for other components and different applications in addition to the pressure vessel, however, in this paper we limit our attention to the pressure vessel.

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 staff issued RG 1.190 "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence" delineating acceptable methodology attributes and acceptable practices in calculating the pressure vessel fluence [*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 uncertainties that are of known origin and are quantifiable should be evaluated and used for estimating the total uncertainty to assure that it does not exceed 20% (1σ) which is the uncertainty implicit in the margin term (M in equation 1) of 10 CFR 50.61 [*4*]. In this category, we find such uncertainties as configuration geometry, material compositions, material densities, material cross-sections, and neutron transport code numerical uncertainties. However, there are instances where the observed measured to calculated fluence ratio (M/C) deviations are large and cannot be attributed to known physical causes. In such cases, the NRC staff recommends a conservative approach.

Fluence Measurements

It is not feasible to measure fluence directly in a commercial reactor environment and in the positions required for vessel calculations, i.e., at the 1/4T and 3/4T locations through the vessel thickness T. Therefore, in accordance with Appendix H, dosimeters must be placed in the vessel and periodically removed and counted. Dosimeter activity can be converted to neutron flux (and fluence) at the irradiation location using an effective cross-section. However, this conversion requires detailed knowledge of the neutron spectrum at the dosimeter irradiation location. The spectra can be calculated based on a description of the geometry, materials, power distribution, neutron source fission spectra and a qualified neutron transport code. Consequently, the conversion of the measured activity to fluence involves a substantial amount of additional data and a relatively complex calculation. The amount of information from the measurement used in the vessel fluence evaluation is small with respect to the information provided by the calculated spectrum. A true comparison of measured and calculated results may be obtained from the direct ratio of measured to calculated dosimeter activities. However, such comparisons can only be carried out at locations other than the 1/4T and 3/4T of the vessel thickness and outside the vessel. Therefore, the fluence values to be used at 1/4T and 3/4T must be calculated and this underscores the importance and need for computational tools of demonstrated accuracy.

In the early calculations of vessel fluence, often large variations in the M/C ratios of the dosimeter activities and in the associated estimates of the neutron flux and fluence were encountered. There is a large number of potential contributors to such deviations. Many of these contributors are not amenable to analysis, and licensees (or their contractors) did not provide a physical explanation for these M/C differences. In many instances, we have no accurate knowledge of the capsule location because of eccentricity or ovality of the vessel, etc., which have not been measured for the as-built vessel. In some of the early plants, the surveillance capsules were installed after plant startup which introduced another source of geometrical uncertainty.

As an example, in some instances the M/C ratios measured for copper and titanium dosimeters [^{63}Cu (n, α) ^{60}Co, E > 4.7 MeV and ^{46}Ti (n, p) ^{46}Sc , E > 3.8 MeV] are in reasonable agreement with the calculated values, but the iron and nickel dosimeters [^{54}Fe (n, p) ^{54}Mn, E > 1.0 MeV and ^{58}Ni (n, p) ^{58}Co, E > 1.0 MeV] show measured count rates that are lower than the corresponding calculated rates and correspondingly lower M/Cs. If we assume that the surveillance capsule is closer to the core, agreement of measured and calculated values would improve. This is because the E > 4.7 MeV fluxes decay (vs. radial distance) slower than the E > 1.0 MeV fluxes. However, the hypothesis of the capsule mislocation cannot be proven because it is not practically feasible to get a more accurate measurement of the capsule's distance from the core. Industry submittals to the USNRC typically estimate an error for capsule mislocation but do not attempt to interpret or account for dosimeter discrepancies beyond the expected error estimate. In such instances, it would seem prudent to take a conservative approach in order to fulfill the requirements of GDCs 14, 30, and 31.

Often submittals to the USNRC use adjustment codes such as FERRET [8] to obtain a least squares average using the neutron spectra as a weighing function. The early version of FERRET used (among others) a precalculated covariance matrix to implement the adjustments. A recent version of FERRET used only the spectral based least squares adjustments and has been reviewed and accepted by the USNRC staff. However, the major contribution from the approval process of the FERRET code stems from the benchmarking performed to demonstrate its capability. In this process, a large portion of the existing PWR surveillance capsule dosimetry data were carefully re-analyzed. In the process, it was shown that the data base can be improved by implementing careful recalculation, correcting cross sections and where feasible performing geometrical adjustment. This process reduced the uncertainties and showed that the data were not biased. The FERRET safety review made a significant contribution to the regulatory aspects of reactor dosimetry by demonstrating that the historical inconsistencies in the dosimetry data bases can be significantly improved by careful reanalyses.

Fluence Calculations

Fluence calculations should be performed using a benchmarked neutron transport code. There are several levels of code benchmarking, all of which contribute to increase the assurance and confidence in the code's performance. The NRC staff recommends code benchmarking to measurements from two pressure vessel simulation experiments, i.e., the PCA (pool critical assembly) [*8*] or the PSF (poolside facility) [*9]* or both. The PCA and the PSF experiments were carefully conducted and measured to small quantified uncertainties. Favorable results of these comparisons will assure the user that the programming and the cross sections are correct, with high probability. However, they do not provide any information regarding the code's performance or the existence of biases in the case of actual plant measurements. That kind of information can only be acquired from plant measurements. Normally, a data base of M/C values is compiled containing a statistically significant number of measurements and the corresponding calculations. Conventional statistical analyses of the data allows the determination of the existence of any potential bias and the uncertainty associated with the given data base. Normally, such data bases are formed with measurements from similar plants regarding geometry, size, fuel loadings, operation and other plant parameters.

A benchmarked code will provide accurate predictions if the geometry, materials, and source information are within their uncertainty band. Regarding vessel geometry, the analyst should bear in mind that the mean vessel "as built" diameter is frequently very close to the "blueprint" value. Core eccentricity and vessel ovality, on the other hand, are not usually reported as measured values and may be responsible for some of the observed differences between measured and calculated fluence values. Other methods for benchmarking, the associated geometry, and cross section requirements are described in RG 1.190 [*6*].

Recently, the USNRC staff has reviewed and approved for use in licensing actions a new code (designated RAMA) based on a different principle than the conventional discrete ordinates codes. RAMA seems to be performing much better than the conventional discrete ordinates codes, i.e., results in M/C values closer to 1.0 (smaller uncertainties) without discernable biases [10]. However, similar results were obtained using conventional discrete ordinates codes when older data were reanalyzed implementing updates and corrections in the cross sections, code updates, geometry refinements, etc. It should be noted that such updates are labor and computer-time intensive. As such, licensees are not motivated to undertake such revisions unless there exists a specific regulatory requirement. It should also be noted that the mandated acceptable uncertainty is 20% (1σ). However, it is generally clear to the reactor dosimetry practitioners that the attainable accuracy now (compared to the time in the mid-80s when the 20% was instituted) is much better and the dosimetry community could support a lower value, for example 10% (1σ).

Measurements vs. Calculations

Having reviewed a large number of pressure vessel measured dosimetry activations and the corresponding calculated values (over a period of three decades) of domestic PWR and BWR dosimetry results, we concluded that:

> significant improvements have been achieved based on improved analytical methods, cross section improvements, better analytical practices and vastly improved computer capabilities,
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> there are inconsistencies between measured and calculated dosimeter activities,
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> inconsistencies exist between different dosimeters from a given surveillance capsule, and
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> inconsistencies exist between the same dosimeter of different capsules.

We also conclude that measured plant specific dosimetry should not be used to adjust calculated fluence values unless it is the only available value and the accuracy requirements are judged suitable. Measured values should be used to form a statistically qualified data base to benchmark a neutron transport code, estimate potential bias and the associated uncertainty.

Calculational Bias

As stated above, the important use of measurements is that they provide the means to estimate the magnitude (when present) of any bias in the calculated values, with respect to the measurements. Consider the case in which a large number of reliable fluence measurements are available from a class of reactors of similar design. The following steps could be used to determine bias and uncertainty. The data should first be scrutinized for physical relevance and statistical consistency. The M/C values then could be used to estimate a bias in the calculated values. The bias accounts for (and is a metric of) the calculational model/method shortcomings to correctly represent the physical features or the material characteristics of the class of reactors represented in the data base.

A calculated fluence value determined using a benchmarked code and supported by a qualified data base is the most justifiable estimate of the pressure vessel fluence considering the requirements of GDCs 14, 30 and 31. Alternate estimates of the vessel fluence, in order of descending confidence level are:

> a calculated value by a benchmarked code and confirmed by plant-specific measured values,
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> a calculated value determined with a benchmarked code, and
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> plant-specific measured values.

Any of the four methods described above could be used, depending on the quality of the analysis, the physical basis and input data used in the calculations, and the requirements of the proposed licensing action. For example, if the RT_{PTS} value for a plant at the end of its current license is estimated to be about 85°C (about 150°F) below the 10 CFR 50.61 screening criteria, the proposed method would be of little concern because of the large margin in RT_{PTS}. If, however, for another plant the end-of-license RT_{PTS} value is within a few degrees centigrade from the 10 CFR 50.61 screening criteria, the most reliable method should be used.

Summary

On the basis of the requirements of General Design Criteria 14, 30, and 31, the regulations in 10 CFR 50.61, the guidance in RG 1.99, Revision 2, and RG 1.190 we conclude that a calculated fluence determined using a benchmarked code and a M/C bias, based on a qualified data base, is the most appropriate method for determining pressure vessel fluence.

References

[1] Code of Federal Regulations, Title 10, Part 50, Appendix A, "General Design Criteria for Nuclear Reactors," January 1, 2006.

[2] Code of Federal Regulations, Title10, Part 50, Appendix G, "Fracture Toughness Requirements," January 1, 2006.

[3] Code of Federal Regulations, Title 10, Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," January 1, 2006.

[4] Code of Federal Regulations, Title 10, Part 50, Section 61 "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," January 1, 2006.

[5] Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," US Nuclear Regulatory Commission, May 1988.

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