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MCBEND - A Fluence Modeling Tool from AEA Technology

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Abstract: Accurate assessment of neutron fluence at reactor pressure vessels and other internal components such as core shrouds is a crucial component of radiation damage studies which are of fundamental importance in determining plant life. In recent years the Monte Carlo calculation method has become recognised as an accurate tool for such assessments. AEA Technology, through its development of the MCBEND Monte Carlo code in collaboration with BNFL, has been a key innovator in the development of the Monte Carlo method and in making these developments easily accessible to the user. Recent developments aimed at fluence modeling include a source processor, a point energy adjoint capability and geometric sensitivities. The code has been applied to fluence modeling in Magnox gas cooled reactors and PWRs for a number of years : with the work described in this paper it has now also been successfully applied to BWR fluence analysis.

Keywords: Monte Carlo, MCBEND, BWR, fluence modeling

INTRODUCTION

Damage caused by neutron irradiation to the pressure vessel and internals of a nuclear reactor over its lifetime leads to embrittlement of the components, with consequent effects on the operating life of the plant. The economic consequences associated with assessment of neutron fluence in these situations are large and hence accurate assessment of iron damage and neutron fluence using validated methods is of prime importance. Accurate estimation of the total uncertainty in the calculated result is also vital.

The different types of reactor system present the fluence analyst with a range of technical problems. Assessment of PWR vessel fluences essentially involves penetration through a set of annular water/steel shields with around 10^4 attenuation between core and reactor Pressure vessel (RPV). Accurate representation of nuclear data is thus essential. Accurate modeling of source strengths is also vital, particularly at the core periphery.

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BWRs have the added complication of variable coolant density in the core. This must be accurately modeled, especially for prediction of fluences at above-core components. Gas-cooled reactors contain important streaming paths from the core to the vessel and these must be treated in detail.

Historically, multigroup discrete ordinates methods have been used to assess reactor fluence and a large amount of experience of using these methods has been built up. However, over the past decade, the massive increase in readily available computing power has facilitated the use of the Monte Carlo method for such analyses. This method offers some advantages over the discrete ordinates method such as more accurate modeling of reactor geometry and more accurate representation of cross-section data. Use of the Monte Carlo method for fluence modeling analyses, in parallel with the discrete ordinates method or by itself, is increasing and this increase is likely to continue as requirements are tightened.

MCBEND is a general geometry, point energy, Monte Carlo code used for radiation transport calculations for neutrons, gamma-rays and electrons. Over the thirty year lifetime of the code significant contributions to the development of Monte Carlo methods, such as automatic acceleration [*1*] and sensitivity analysis [*2*] have been made. These allow the user to run calculations easily and to obtain an accurate estimate of the uncertainty on the calculated fluence or damage. Recent developments include a source processor for reactor calculations, implementation of a point energy adjoint method and calculation of geometric sensitivities. MCBEND has been extensively applied to fluence analysis for UK gas cooled reactors [*3*,*4*] and to a number of PWR reactors [*5*,*6*,*7*]. This paper describes recent developments in MCBEND of particular use in fluence modeling and the recent application of the code to fluence analysis for a BWR reactor.

FLUENCE MODELING WITH MCBEND

MCBEND has been used for fluence analysis for many years. It contains a flexible, comprehensive geometry modeling package that allows the reactor components to be modeled accurately. Nuclear data from the ENDF/B-VI, JENDL3.2 and JEF2.2 evaluations may be used, as well as the older UKNDL data. Cross-sections are represented in over 13,000 hyperfine groups, which eliminates spectrum dependent effects. Secondary angle and energy distributions are also accurately represented. The code includes an automatic acceleration module, based on importances calculated in an overlaid orthogonal mesh using adjoint diffusion theory. The diffusion coefficients have been adjusted such that the resulting solution agrees well with more accurate methods such as discrete ordinates. MCBEND and its associated data have been validated for reactor fluence analysis by comparison with dosimetry measurements [*5*,*6*] and by analysis of benchmark experiments [*8*].

Estimation of uncertainties on the calculated neutron fluxes and damage rates is of crucial importance in reactor dosimetry. Uncertainties are mainly due to uncertainties in the basic nuclear data, material compositions and densities, as built dimensions, source strengths and spectra together with the stochastic uncertainty of the Monte Carlo calculation. Calculation of the resulting uncertainty in the neutron flux or damage rate requires the sensitivities of the flux or damage rate to the parameters to be known.

MCBEND includes a module which calculates sensitivity to cross-sections and these can be used to find the sensitivity to material compositions and densities as well. This facility has been used extensively in the early benchmarking of MCBEND as part of the NESDIP programme for PWR fluence assessment [*9*] and in other benchmark assessments. The stochastic uncertainty can be reduced to whatever level is required by running the calculation for longer : typically standard deviations of less than 2% can be achieved in an overnight run on a modern workstation. Analysis of other uncertainties is also accessible using MCBEND, with the recently developed geometric sensitivity option being used to assess the effect of uncertainty in as built dimensions.

Source Processor for Reactor Calculations

Fluence analysis calculations require an accurate representation of neutron source strengths in the core for the cycles or sub-cycles being considered. The source strength (neutrons/second/unit volume) at a particular state point of the reactor in a particular core subdivision depends on the reactor power, the relative power fraction (rpf) and the power fractions arising from different actinides. The source strength at each state point is evaluated assuming full reactor power and these strengths and rpf data are averaged over the cycles or sub-cycles being considered using the power generated corresponding to each state point as weights.

The power fractions arising from different actinides are a function of assembly type (materials and geometry), coolant density and burnup. They can be derived using reactor physics lattice calculations. Usually, simply the power fraction due to fission in plutonium is quoted and the source strength is then derived by assuming average values of power per fission and neutrons per fission in uranium and plutonium. Greater accuracy can be achieved if the lattice calculation is used to provide the source strength and fraction of neutrons arising from plutonium directly since then the power per fission and neutrons per fission for all of the different actinides are taken into account. The plutonium power fraction is important because it affects both the source strength and the source spectrum. For example, in a typical BWR fuel assembly, a burnup of 40GWD/Te produces around 12% more neutrons than fresh fuel for the same nominal power with over 80% of the neutrons arising from fission in plutonium. In addition the Pu239 spectrum is slightly harder than the U235 spectrum, though this is a secondary effect.

For fluence analysis the core is typically divided into 25 axial subdivisions per assembly and source strengths are given as averages over each internal assembly and explicitly over each pin region for external assemblies. The greater accuracy in the external assemblies is necessary because of their importance to fluences at radial components such as shroud and RPV and because the power and hence source strength profile varies markedly across the external assemblies. Source strengths are typically held in an orthogonal mesh and the inclusion of source strengths at the pin level in outer assemblies can produce impractical memory requirements. For example a typical BWR with an 8x8 assembly would need around 45 million locations or 180Mb to model the source strengths in a 90 degree sector with source weighting included. It is also worth noting that some parts of the resulting distribution would never be sampled because the number of different random numbers available is 2^{24} , i.e. 16.8 million.

This problem has been solved in MCBEND by developing a source processor with a relatively small memory requirement. The code reads in and processes source strength and Pu fraction data on an assembly basis and also reads in the pin power data for external assemblies from a SIMULATE output. The memory requirement is thus reduced to around 0.6Mb. A source particle is sampled from the distribution on an assembly basis and MCBEND then determines whether or not the particle is in an external assembly. If it is then the pin to mean data for the relevant axial subdivision of the assembly are used to sample the neutron's position within the assembly, with positions outside the fuel pins being rejected. This explicit treatment can be used for other assemblies provided that pin power data are available for them. If the neutron is not in an external assembly then it is started from the initial sampled position. An option to use coarser axial subdivisions is also available. The user may also choose which external assemblies are to be treated explicitly. The small number of locations required to store the assembly source strengths and the pin data means that all parts of the source distribution will be sampled, thus providing improved source sampling. MCBEND includes both U235 and Pu239 fission spectra and thus is able to sample the neutron energy correctly according to the Pu neutron fraction present.

Point Energy Adjoint Calculations

In reactor dosimetry assessments there is often a requirement to calculate a detector reaction-rate or neutron flux for a number of different source distributions such as distributions at different state points in a cycle or cycles. Examples are surveillance detector reaction-rates or on-line instrument response. If standard forward calculations are used in such assessments then one calculation is required for each source distribution. The individual calculations may also have very long running times if the detector region is small. To overcome these problems an alternative approach is to use the adjoint method. In this method the cross-section of the desired response at the required location becomes the adjoint source and the calculation produces an adjoint function over the model. The adjoint function over the core is then integrated with the real source distribution to give the required result. Thus if a detector reaction-rate or flux is required for many source distributions then a single adjoint calculation may be performed to obtain the adjoint function over the source which can be combined in turn with the different source distributions. In addition, for problems with a small detector and a large source it can be more efficient to perform an adjoint calculation rather than a forward calculation. This is because in an adjoint calculation the roles of source and detector are reversed, so whereas a forward calculation may have difficulty in scoring sufficient particles at a small detector to give good statistics an adjoint calculation has no difficulty in scoring sufficient particles across the reactor core.

A limitation of adjoint calculations until now is that they have been restricted to multigroup data and thus suffer from the approximations inherent in such methods. To overcome this problem a point energy adjoint capability has recently been developed in MCBEND [10]. This major development has involved rewriting parts of the neutron collision processor and also producing specific adjoint nuclear data libraries. At present modeling of thermal neutron collisions is performed using a one group treatment, which is adequate for fluence calculations.

The capability has been applied to analysis of the H B Robinson reactor and also to a PWR cavity streaming problem [*11*]. In the latter problem an efficiency gain of around 6 was achieved halfway up the 4 metre cavity and at the top the gain was of the order of 1000. For reactor dosimetry the relative efficiency of forward and adjoint calculations depends on the number of source distributions and detectors which are being used. The authors of reference 11 estimate an efficiency gain of a factor of 5 if 10 detectors and 10 source distributions are required.

Geometric Sensitivities

Sensitivity analysis is a key component of any neutron fluence assessment. For many years MCBEND has had the capability to calculate sensitivity to basic nuclear data, as described previously. A recent innovation is the development of a geometric sensitivity option. This allows the user to assess the effect of small changes in positions or dimensions of components in the geometry model on specific results without recourse to an extra calculation. This is useful as it obviates the need for the user to repeat his model verification process after making a small change explicitly to a complex geometry model and reduces the computational effort. The facility in MCBEND is currently being developed and calculates both first and second order sensitivities. Achieving low stochastic error on the first order sensitivities is straightforward but for the second order sensitivities long cpu times are currently required.

The geometric sensitivity capability has been applied to a simplified PWR model to assess the effect of a change in the thickness of the thermal shield on the Ni58(n,p) reaction-rate at the RPV and in the cavity. A 1cm increase in the thickness of the shield was estimated by the geometric sensitivity option, using second order sensitivities, to give a 6.6% decrease in reaction-rate at the RPV and a 7.1% decrease in the cavity. If only the first order sensitivities were used then these figures were 9.6% and 9.8%, respectively. These results compare with decreases of 7.5% $\pm$ 1.1% and 9.1% $\pm$ 1.6% obtained by explicit calculation. Thus the geometric sensitivity option gives results which agree with the explicit results to within the Monte Carlo statistics and can thus be used to give reasonable estimates of the effect of geometric uncertainties within a reasonable additional computing time (~20% for first order & ~ 50% for second order)

APPLICATION OF MCBEND TO BWR FLUENCE ANALYSIS

Although much of the historic development of fluence analysis has been for PWRs, the use of Monte Carlo codes for BWR fluence analysis has increased in recent years. AEA Technology has recently used MCBEND in a fluence assessment for a typical Swedish BWR. The project is a joint venture between the Swedish utilities. The reactor analysed started commercial operation in 1976. The core contains 648 fuel assemblies and these assemblies have contained 8x8 or 10x10 arrays of pins with water coolant channels. Fluence analysis will be used to establish the neutron irradiation embrittlement of the RPV and reactor internals. The results will be used to make recommendations for in service inspections of these regions in Swedish BWRs. For this purpose fluences and neutron damage were calculated throughout the reactor. Dosimetry in a capsule just in board of the shroud, which was extracted in 1997, was used to validate the MCBEND calculations.

The geometry model used in MCBEND was derived from engineering drawings and included detail of the core, shroud, shroud lid, RPV, feedwater sparger and diffuser, steam separators, control rods and guide tubes and the above core spray system. The shroud lid, steam exchangers and the pipework above the core were modeled in considerable detail so that fluences at the consoles carrying the core spray pipes could be determined accurately. A single model was set up and used for all of the analysis with calculations optimised for different parts of the reactor. Views of the MCBEND model are shown in Figures 1 and 2. The model included accurate representation of the variation of coolant density within the core.

The source data which were supplied by Vattenfall included SIMULATE outputs for a number of state points (relative power fractions, burnup and coolant density at 25 axial subdivisions in each assembly), pin powers for external assemblies, power history of the reactor and details of the different assembly types. The AEA Technology lattice code WIMS [*12*] was used to calculate neutron source strength (normalised to full power) and fraction of neutrons from plutonium as a function of assembly type, burnup and coolant density. Thus power per fission and neutrons per fission were treated exactly for each actinide. These results were combined with the SIMULATE data to produce neutron source strengths (for full power), relative power fractions and plutonium neutron fractions at each state point. These data were read into MCBEND at run-time and averaged over the relevant cycles for the calculation being performed using the power generated corresponding to each state point as weights. Thus the neutron source strengths on an assembly basis were produced. The pin power data were also read by MCBEND at run-time, processed to give pin-to-mean ratios and used to refine the source sampling in external assemblies as described in the previous section. Thus a fine representation of neutron source strength was achieved.

Calculations were run using ENDF/B-VI data. Results were produced at all locations of the reactor and uncertainty analysis was performed using sensitivities from MCBEND. Azimuthal variation of damage and fluence over welds and other points of interest was calculated. Figure 3 shows a scan of the iron damage at the inner surface of the RPV. An upper bound for neutron dose of 5×10^{20} n/cm^2, $E_n > 1$MeV before yearly in service inspections are required has been established by the regulatory authority. For the RPV this limit will never be reached during the operational lifetime of the plant but for some reactor internals, i.e. the core shroud, it will be reached.

A full uncertainty analysis was completed for each result taking into account cross-section, material density and source uncertainties. These gave uncertainties (1 sd) of around 10% at the shroud and 13% at the RPV. The major component was the uncertainty on the source strength with the stochastic Monte Carlo uncertainty reduced to between 1 and 3%. Comparison of detector activation for a Cu63(n,α)Co60 detector just in-board of the shroud gave a C/M value of 1.03, thus demonstrating the validity of the the calculations .

CONCLUSIONS

Monte Carlo fluence analyses are a practical proposition and are becoming common place. The MCBEND Monte Carlo code continues to be developed to improve the accuracy of such assessments . The code has been successfully applied to gas cooled and PWR reactors for a number of years and the work reported here demonstrates its successful application to BWR fluence analysis as part of an in service inspection programme for Swedish BWRs.

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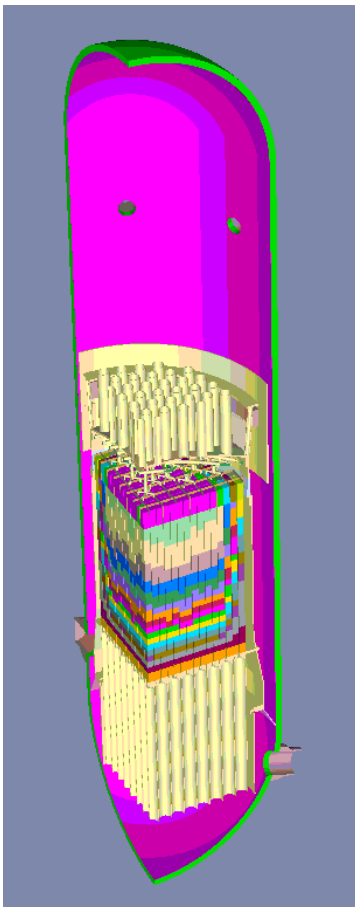

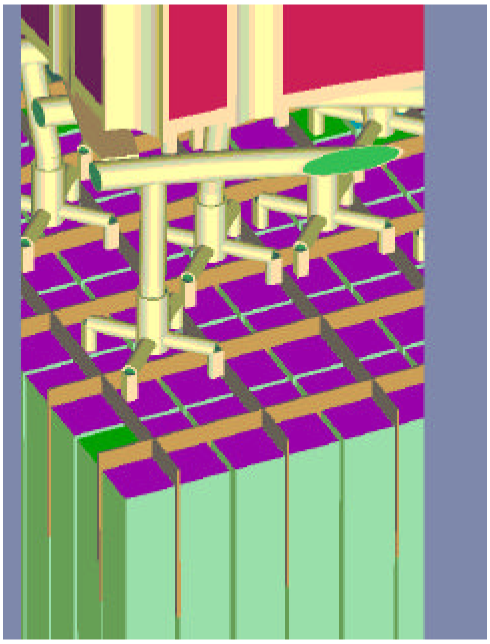

Figure 2 View of the Above Core Spray System in the MCBEND Model of the BWR

Figure 1 View of the MCBEND Model of the BWR

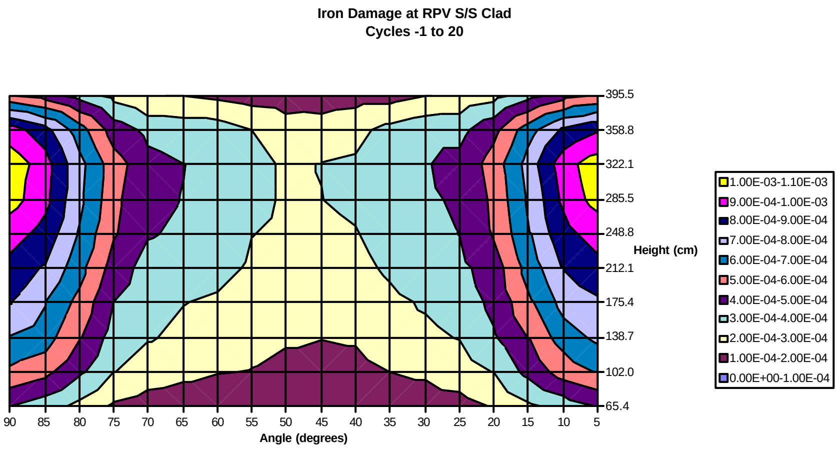

Figure 3 Iron Damage at RPV Cladding in the BWR.