NEA/NSC/DOC(96)5

COMPUTING RADIATION DOSE TO REACTOR PRESSURE VESSEL AND INTERNALS

STATE-OF-THE-ART REPORT

NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

FOREWORD

The OECD/NEA Nuclear Science Committee Task Force on Computing Radiation Dose and Modelling of Radiation-induced Degradation of Reactor Components was set up in November 1995 in response to a request by the NEA Committee on Safety of Nuclear Installations. The purpose of this Task Force was to:

- evaluate the accuracy of the calculation methods used in NEA Member countries for predicting long-term radiation doses to reactor pressure vessels and internal structures;
- identify points for improvement and validate the performance of improved methods for fluence calculations;
- initiate a study on the modelling of radiation-induced damage in metals.

This report fulfils the first and second objectives of the Task Force by providing a critical discussion of the most recently published literature on computational methods of reactor dosimetry, and a detailed overview of the computational techniques currently used in the dosimetry programs of NEA Member countries. The analysis presented suggests additional factors whose importance should be considered in modelling these phenomena. Proposals are made for further work on these problems.

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EXECUTIVE SUMMARY

Adequate knowledge of cumulative radiation doses, or fluence, is an essential step in developing and validating more effective models for prediction of radiation damage to reactor components. The present report reviews the computation techniques for calculating neutron/gamma radiation doses to reactor components, and describes in considerable detail the methods presently used in NEA Member countries for computing long-term cumulative dose rates.

Although the median of results reported in national calculations appears to lie within 20 per cent difference between calculations and measurements, significantly higher and lower values are also reported. Moreover, the numbers reported are difficult to compare, since each country has its own methodology including different reactors, computer codes, nuclear data sets and measurement procedures. On the basis of these country reports, the working group concludes that no firm judgement can be formed on the current international level of accuracy in pressure vessel fluence calculations.

To identify the range of differences between calculations and measurements, the NEA/NSC Task Force on Computing Radiation Dose and Modelling of Radiation-induced Degradation of Reactor Components (TFRDD) is conducting an international "blind" intercomparison exercise. The varying methodologies are applied to predicting dose rates in the Belgian Venus test reactor, calculating in both two and three dimensions for comparison with measured data. This comparison of methods for dosimetry calculations should lead to consensus on:

- The level of accuracy of methods currently used in the NEA Member countries in calculating radiation dose to reactor components;
- The relative merits of different calculation methods;
- Possible improvements to these methods;
- The advantages of fully 3-D methods over 2-D methods.

Additional problem areas were identified during the review of recent literature:

- Thermal dosimetry, which is not yet well established, although there is mounting evidence of its importance in some reactors (desirable action: study of the importance and magnitude of errors in thermal fluence estimates on metal degradation in some reactors);
- Gamma dosimetry, which is clearly less well established than neutron dosimetry (desirable action: validation of gamma transport codes, nuclear data, and gamma-metal interaction models for estimating metal damage);
- Poor accuracy of the metal damage models used to compute damage parameters from known fluence (desirable action: further work on development of models relating particle fluence to metal damage – a critical activity in improving the accuracy of metal damage prediction).

Introduction

Currently, the issues of reactor components ageing due to irradiation present some important challenges for nuclear engineering. As nuclear reactors are ageing across the world, the ability to accurately predict the condition of various reactor components is becoming a key factor in nuclear safety assessment prior to extension of reactor operating life.

If reliable and accurate techniques can be developed to accurately predict doses incurred by various reactor components and the resulting reduction in their metal toughness (in other words, to predict metal ageing), then utilities and reactor vendors could justify present design life of their reactors and analyse feasibility of its extension. This potentially implies considerable financial savings.

Reactor components sustain damage due to irradiation. Damage is believed to result mainly [1,2] from neutron interaction with atoms of the metal lattice, causing displacement and displacement cascades which can block the movement of dislocations and thus lead to reduction in metal toughness.

In addition, precipitates can be formed (e.g. Cu, P, Ni) leading to precipitation hardening [1,2] of the metal. The net effect of these processes is irradiation embrittlement [1,2] of the reactor component metals (including welds). There are several clearly defined parameters [1,2] contributing to radiation embrittlement. These are:

- neutron fluence;
- gamma fluence (if significant);
- irradiation time;
- metal temperature (low temperature neutron exposure strongly increases the neutron damage);
- steel composition (e.g. P, Ni, Cu segregate at the grain boundaries);
- steel microstructure.

Additional factors in embrittlement include:

- thermal ageing;
- strain ageing;
- hydrogen, helium, and gamma embrittlement.

The previously listed parameters are an object of research of four intimately linked subfields of the reactor component ageing field:

- materials research on mechanisms of metal degradation due to irradiation and development of its semi-empirical models (the models of metal damage due to irradiation are not the main focus of this document),

- the experimental techniques of measuring and the computational techniques (i.e. neutron and gamma transport methods) of calculating neutron and when appropriate gamma doses incurred by various reactor components;
- recovery of material properties by annealing;
- re-embrittlement after annealing.

In this report, the emphasis is placed on the second of the research areas listed above, that of computational techniques of calculating neutron and gamma doses to reactor components with some discussion of material science problems related to translating the knowledge of radiation fluence to metal damage.

In particular, the discussion includes some critical issues of computational methods and their uncertainties, gamma and thermal neutron induced embrittlement, models relating particle fluence to metal damage used in predicting embrittlement, benchmarking of present day transport methods, and a detailed description of computational methodologies of the NEA Member countries. In conclusion, areas of possible improvements that could benefit from international collaboration are identified.

Neutron flux and irradiation embrittlement

The determination of metal damage due to irradiation involves determination of energy dependent neutron fluence (and where significant, of gamma fluence) for each location of interest by combining theoretical calculations and measurements (usually integral detectors). The fluence(s) are then used in some damage model [1,2,3] to predict the ageing. The general practice is to derive the metal toughness reduction versus accumulated fluence predictive formulae (for use in predicting metal ageing in power reactors) from controlled environment experiments in test reactors. This procedure is considered to be acceptable for pool type light water moderated reactors.

Energy dependent neutron fluence above a certain threshold energy level is widely used in estimating metal radiation damage. The question of what this threshold energy should be so that the “right” part of the spectrum (i.e. neutrons at energies causing metal damage) is chosen is still causing a controversy [1]. In general, only neutrons with energies above 1 MeV are considered to contribute to metal damage; the threshold value of E_0=1 MeV in fluence calculations is accepted in France being the easiest part of the flux spectrum to measure experimentally. In Russia, the threshold is set at E_0=0.5 MeV (then the spectrum contains about 30 per cent more neutrons). In the USA, the light water reactor surveillance analysis is based on fluence above E_0=1 MeV threshold [5]. Further, the US NRC DG-1025 Draft Regulatory Guide [5] recommends that for pressure vessel inner walls the calculated fluence above E_0=1 MeV and E_0=0.1 MeV be reported. The threshold E_0=0.1 MeV is used in the USA for reporting fluence in fast reactors. No standard practice exists for derivation of threshold fluence in test or power reactors. Nevertheless, uncertainties in derivation of fluence should be lowered when test reactor experiments are involved and recommended practices are employed (ASTM standards, EWGRD recommendations [6,7]).

The assumption that neutrons below certain threshold energy do not cause any metal damage is a somewhat crude approximation of the reality as the threshold energy [8] (set somewhat arbitrarily) does not reflect the physics of the metal lattice. Moreover, the threshold energy concept implies that all neutrons of energies greater than this particular value produce the same damage – yet another crude approximation [8]. Hence, a different fluence parameter called DPA (Displacements Per Atom) was introduced [8] to represent the metal damaging effects of neutrons at all energy levels.

Unfortunately, DPA as much as it accounts for the full neutron energy spectrum is not a precise parameter either [8,9,10]. DPA is based only to a limited extent on actual damage mechanism and cannot be expected to produce accurate damage estimates. It should be regarded as a theoretical construct that can be used as a correlation parameter and tested against experimental data to determine its precision.

In some situations DPA is considered a more conservative parameter than neutron fluence above E_0=1 MeV, e.g. in predictions of metal damage within the reactor vessel relative to its surface. Likewise, DPA may not be conservative in other situations (such as relating surveillance capsules to the vessel inner surface) as compared with flux above E_0=1 MeV. Regarding the precision of the DPA model it should be further noted that at reactor operating temperatures a large number of DPA are restored immediately and only an unknown number of DPA actually remain to cause the residual damage.

In current practice, the DPA cross-section approach is typically employed only [9,10] for monoatomic crystalline materials (and not for composite materials such as alloys, for example). However, the polyatomic versions of the DPA model have been developed [11,12]. The polyatomic DPA model results can be significantly different than the sum of monoatomic displacement terms. In fact, a computer code exists to treat displacements in compound materials [11]. The reason for which the new polyatomic DPA models are not routinely used in reactor dosimetry calculations is that they have not been demonstrated to yield a better correlation with measured metal damage and that details on material compositions for previously irradiated samples in the damage database are not readily available. Hence, a correlation could not be easily established with the historical database to verify the polyatomic DPA models.

It should be noted that in light water reactors the neutron spectra at the surveillance capsule positions and at the corresponding inner surfaces of the reactor vessel are very similar. Experience shows that in this case as long as predictions of vessel embrittlement are based on surveillance specimens from power reactors, the result is not sensitive to the correlation parameter used in metal damage calculations. In fact, no significant improvement in metal damage estimates is observed when using other parameters (such as DPA) rather than fluence above E_0=1 MeV. This does not hold for extrapolations through the vessel wall or for correlation of data from other sources (such as research reactors). In this instance, DPA should be used in estimating metal damage. Recent investigations in France show that fluence above 1 MeV is an adequate correlation parameter in pressure vessel surveillance at positions from surveillance capsule to 1/4T (where T is the pressure vessel thickness) [14].

In recent studies [10], it was determined that a knowledge of the PKA (Primary Knock-on Atom) spectra in a crystalline medium is required for in-depth investigation of the neutron radiation damage effects. In particular, the dimensions of the cascades and stable defects, after the temperature induced recombination of vacancies and interstitials, are strongly influenced by the PKA spectra. These spectra can be further used [10] to define damage parameters (e.g. inter-displacement model) for polyatomic metals which cannot be treated correctly with the DPA approach. In other words, the PKA spectra are a more general double differential quantity (with respect to the incident neutron energy and the outgoing particle energy) unlike the DPA damage function which is differential in energy. Consequently, the PKA spectra constitute a better input for the correlation parameters in the damage models than the neutron spectra used in DPA calculations. A code SPECTER was developed at the Argonne National Laboratory in the USA which calculates the PKA spectra. Another code is being developed [10] at ENEA Bologna (PRISMA – Primary Spectrum in Materials). PRISMA is a database interface to the NJOY code whose HEATER module is used to produce PKA spectra. The PKA spectra are not part of the normal NJOY output, but they have been routinely extracted and used [13] to calculate improved displacement damage models in materials such as Si and GaAs electronic components. It would be useful [10] to extend this method to nuclides of steels.

At this point, it should be stressed that the information about the effects of neutron damage to metals based on any theoretical model should be provided (by material scientists) in terms of the cross-section concept so that the fluence information can be readily converted into the metal damage information as the end result of reactor physics computations using neutron transport codes. This is because even the most accurate fluence calculations are not very useful without reliable neutron damage model expressed in terms of metal damage cross-sections. It is also important that the fluence spectrum, magnitude, and variation with time be thoroughly documented so that future developments in the materials area will have these data for validation of new damage models (several such models already exist but they are not routinely used [3] due to lack of well documented historical database allowing their validation). In addition, it must be recognised that a simple cross-section approach to metal damage calculation may not be adequate. The current level of precision in fluence calculations would potentially be sufficient from the practical point of view if accompanied by an improvement in the radiation-induced metal damage model. This statement would have to be supported by a careful benchmarking study establishing the degree of accuracy of the current fluence computation methods at the international level.

Additional confusion [9] arises from the fact that different response functions of DPA according to different models have been used by different laboratories making the comparison of irradiation effects on metals in different reactors and locations very difficult. Hence, there is a need [9] for an international standard for a displacement dose unit. The establishment of a standard for the displacement dose unit is not a simple task. Historically, a standard model for converting displacement kerma (energy) into DPA was established by IAEA over twenty years ago [15]. Since then displacement models have improved in their fidelity, but it is thought that the IAEA model was still in use as a standard. The displacement kerma functions for iron have changed as iron cross-sections were improved. The DPA functions are a result of combining the displacement kerma (or energy) with a displacement production model. Thus, the DPA damage functions in Europe and the USA are different because they use different iron cross-sections and different models for computing the recoil spectra (and partitioning of damage energy between ionisation and displacements). For example [16], the standard model (TRN) used in the UK for steel is different from the model by Doran [17] used in the USA (the UK formalism uses an anisotropic model for the elastic scattering in iron whereas the Doran model uses an isotropic elastic scattering model). The question then arises: should one standardise a formalism – or a specific tabulated response function? What happens when iron cross-sections are changed again? Does one retain the tabulated response functions (based on what one would then consider to be a poorer cross-section model) or does one make frequent changes in the “standard model”? These questions are currently faced in the USA with regard to the ASTM recommended iron DPA model. The USA community is divided on whether or not to change the US standard to (hopefully) improve the correlation of DPA with observed damage or to stay with the current model because changing it puts at risk much of the historical database that could not be easily re-analysed and evaluated with the new model based on available documentation.

Furthermore, the evidence of the higher-than-expected irradiation damage of the RPV (Reactor Pressure Vessel) and reactor components of the HFIR reactor [8] (High Flux Isotope Research reactor – a light-water-cooled, low temperature, high performance research reactor at ORNL in the USA) seemed to indicate that the importance of the thermal neutron flux has long been underestimated.

The HFIR neutron spectrum consists in 95 per cent of thermal neutrons and its vessel operating conditions are that of low temperature of 70-130ºC and low neutron flux. Considering that thermal neutron dosimetry has not been a focus of the RPV surveillance programs, it was longed believed that accelerated damage of the HFIR reactor vessel was induced by unaccounted for thermal neutron damage [8]. Recent HFIR analyses [18] indicate that the thermal flux (at the pressure vessel) is much smaller than was previously estimated and is not the cause of higher-than-expected radiation embrittlement. Instead, a high gamma field present at the vessel region of the HFIR is now believed to cause the accelerated damage of HFIR metals. The HFIR gamma flux (E_0>1 MeV) was found to be approximately four orders of magnitude higher than neutron flux (E_0>1 MeV). When gamma-induced DPA were added to neutron DPA the HFIR surveillance results were brought into agreement with results from other test-reactor irradiations. To relate the HFIR findings to LWR reactors it should be added that the photo-fission correction in LWRs is typically 5 per cent for ^{238}U (n,f) and 2.3 per cent for ^{237}Np [19]. In spite of these recent findings about the HFIR reactor indicating that the gamma flux rather than thermal neutrons contributed to its enhanced material damage, thermal neutron damage can be important in some situations and more work needs to be done to establish thermal neutron effects on metal degradation. This must be regarded as a special situation, however, and thermal effects are not considered to have significant effects on the accuracy of damage estimates in light water reactors. The effect of thermal neutrons may be important in power reactors other than HFIR. For example, the surveillance data for the Atucha 1 reactor are greatly affected by thermal neutron damage.

Studies [8] have shown that the mechanism of the thermal neutron induced metal damage is that of (n,γ) reactions characterised by recoil energies of about 500 eV and two known (n,α) reactions (i.e. with ^{10}B and ^{6}Li) characterised by average recoils of approximately 1 keV. In contrast, the mechanism of the fission neutrons (E>1 MeV) induced metal damage [8] is that of elastic and inelastic (n,n') reactions or transmutation reactions (n,p) or (n,α) characterised by recoil energies in the range of 10 keV to 100 keV. In particular, the fast neutron-induced recoils produce a large number of metal defects of which only a small fraction survive to cause permanent metal damage whereas the thermal neutron induced reactions produce a small number of point defects of which most survive to contribute to permanent metal damage. In fact, the point defects survival rate [8] for low energy recoils is about 30 per cent in comparison to 5 per cent survival rate for the recoils originating from fast neutrons of energies above 1 MeV. Models have been developed to treat the displaced atom motion and recombination. The binary collision approximation (BCA) model is used in, e.g. MARLOWE [20] code and the molecular dynamic approach [21] was developed. Heinisch has developed some empirical annealing codes that use the MARLOWE output displacement maps as a starting point [22]. Several authors have introduced advanced damage models that use annealing damage measures for the damage correlation. Many models have been developed that treat subsequent interactions and predict the resulting defect clusters, cascade loops, and other measures of permanent damage. Some successful correlations with non-reactor material damage have been reported [22] for the measure of "freely migrating interstitials".

The computational problems of precision in calculating thermal neutron flux/fluence (i.e. few group thermal energy range group structure in multigroup calculations, neglect of up-scattering in thermal energy range) can affect dosimetry and more detailed calculations should be attempted to define the magnitude of the errors. They are, however, limited to research reactors specifically designed for high thermal-to-fast flux ratio. Transfer of information about metal damage in such reactors to commercial PWR power reactors is not direct due to differences in spectrum, flux density, and operating temperature of such reactors versus commercial PWR reactors.

At this point a distinction should be made between neutron (or gamma) transport calculations and calculations to derive exposure estimates based on dosimetry measurements. In the first case, calculations of neutron fluence and spectrum are made for points of interest in the problem. In the second case, measurements are interpreted using the calculated neutron spectrum. These may be combined (e.g. by using a least squares procedure) to produce a "best estimate" value for the exposure (fluence, DPA, etc.) at the dosimetry point and also at all other points using the calculation for extrapolation or interpolation. The relative weighting of the dosimetry data to produce the fluence magnitude may range from 0 to 1. If zero weighting is used, the dosimetry data is only used to check the adequacy of the calculation and, if the result is satisfactory, the estimates are based solely on the calculation (e.g. as in US NRC Draft Guide-1025). If a weighting of one is used, the calculated results are normalised to the dosimetry measurements and the magnitude of the fluence from the transport calculation is ignored. In practice, both of these approaches have some merit. In the first case, it is assumed that error between the results is due to errors in measurement or in the local dosimetry geometry. In the second case, errors are assumed to be due to such errors as in the calculated neutron leakage from the core or geometrical errors around the core or vessel that are sensed by the measurements but not included in the transport model.

Another problem of irradiation embrittlement is that of gamma ray induced metal degradation. The gamma flux metal damage may be an important factor in the overall metal degradation when gamma flux levels are high as is the case in the BWR reactors or research reactors (see page 13 and page 18 for more details). On the other hand, in the PWR reactors gamma flux levels are not high and hence the issue of gamma ray induced metal degradation in PWR environment is expected to be a minor one, but further studies are needed to support this statement.

The evidence presented indicates that:

- more effort should be directed towards improved experimental and computational dosimetry of thermal neutrons for reactor components (even though the issue of thermal neutrons metal degradation is largely specific to research reactors with high thermal-to-fast flux ratio, a better understanding of the phenomenon and improved ability of calculating and measuring thermal flux/fluence may lead to a more complete knowledge of the irradiation induced metal degradation in commercial reactors).

- there is a need for development of gamma particles damage models to determine the importance of this damage mechanism to reactor components metal degradation.

Computational problems of reactor components dosimetry

The computational schemes for evaluating fast neutron fluence at the RPV (Reactor Pressure Vessel) and reactor cavity (i.e. a volume outside of the RPV locations filled with air) are quite well established yielding results typically within 20 per cent for in-vessel surveillance capsules and to within 30 per cent for cavity dosimetry. These estimates are based on calculations with pre-ENDF/B-VI cross-sections. The agreement obtained using ENDF/B-VI cross-sections is expected to be much better. The computations for the positions in the reactor cavity are generally more difficult than for the positions within the reactor vessel and it is not obvious how to extrapolate the ex-vessel results for the in-vessel positions. In general, the uncertainties of the ex-vessel transport calculations are higher than uncertainties of the in-vessel transport calculations. For example, in the ex-vessel computations (and measurements), the problem of backscattering of low energy neutrons (especially thermal neutrons) from concrete present in the reactor cavity should be accounted for.

In general, the flow of reactor dosimetry calculations can be divided into five fundamental steps:

- calculation of the multigroup fast neutron fluence using a transport code;
- calculation of dosimeter reaction rates from measured decay rates using the flux history (the flux history may also be used together with the transport result to calculate the expected decay rates);
- comparison of the dosimetry measurements with the calculated results either by:
  1) derivation of fluence from the measurements and the calculated neutron spectrum, or
  2) derivation of dosimeter reaction rates from the calculations;
- combination of the calculated and measured results to produce the multigroup fluence to be used for damage estimates;
- calculation of exposure parameters such as DPA from the fluence and spectrum as defined in previous point.

However when the relationship between power and flux changes significantly during the period of irradiation due to variations in the power distribution within the core, in the relative source strengths from U and Pu, or modifications to the shield, it is necessary to compare calculated and measured activation of the dosimeters rather than reaction rates.

Calculation of the multigroup neutron flux is carried out using a transport code. The transport methods of choice are: state-of-the-art S_N or Monte Carlo codes with advanced geometrical modelling capabilities. This calculated multigroup neutron flux may then be adjusted by least squares fitting [23] within its uncertainty interval by using the responses of dosimeters (sensitive to the desired energy ranges) irradiated at locations of interest in the reactor and reactor cavity. It is rather commonly believed that when the source, geometry, and material data are known within a high degree of accuracy and when extreme care is taken in modelling geometry and generating macroscopic cross-section library then no adjusting by least square fitting is necessary. In this case, the use of calculated neutron spectra at the fluence detector positions within the irradiation capsules is generally sufficient in calculating the effective metal damage parameters.

It should be pointed out here that using the adjustment procedure makes sense since it further reduces the uncertainties of the flux estimates. This is because the calculated uncertainties are in the order of 20 to 30 per cent (1σ) (for example, for the calculated reaction rates or flux), while uncertainties of the measurements (e.g. specific activities, reaction probabilities) are typically around 5 per cent for threshold dosimeters and slightly higher (~ 8 per cent to ~ 10 per cent) for fission dosimeters. It is an intrinsic property of the least-squares adjustment procedures that the adjusted fluxes have smaller uncertainties than the (initial) calculated fluences, thanks to the information obtained from the measurements. It should be noted, however, that in spite of progress in the accuracy of the cross-section data the inelastic scattering cross-section of iron is still a source of significant computational discrepancies in the energy range 10 to 100 keV (for reactors other than PWR), especially at positions in reactor cavity.

More information about the latest evaluations of iron cross-sections can be found in the following reports: “Comparison of Evaluated Data for Chromium-52, Iron-56 and Nickel-58”, “Generation of Covariance Files for Iron-56 and Natural Iron”, and “Cross-section Fluctuations and Self-shielding Effects in the Unresolved Resonance Region” [23,24,25].

Typical detectors (and reactions) used are [26]:

- $^{59}Co(n,\gamma)^{60}Co$, $^{58}Fe(n,\gamma)^{59}Fe$, $^{63}Cu(n,\gamma)^{64}Cu$ – in the thermal energy range;
- $^{237}Np(n,f)FP$, $^{238}U(n,f)FP$, $^{232}Th(n,f)FP$ – in the energy range 0.67 to 1.52 MeV to estimate the fission reaction rates;
- $^{93}Nb(n,n')^{93m}Nb$, $^{115}In(n,n')^{115m}In$, $^{47}Ti(n,p)^{47}Sc$, $^{58}Ni(n,p)^{58}Co$, $^{54}Fe(n,p)^{54}Mn$, $^{46}Ti(n,p)^{46}Sc$, $^{48}Ti(n,p)^{48}Sc$, $^{63}Cu(n,\alpha)^{60}Co$, $^{27}Al(n,\alpha)^{24}Na$ – for estimating threshold reactions in the energy range 0.5 MeV to 6.5 MeV.

In a recent paper [27], the thermal neutron activation in Fe is reported to be underestimated by 25 per cent in computations. In general, the accuracy of computations in the thermal energy range is lower than in the fast energy range. This is due to the energy group structure and approximations used in the thermal energy range (e.g. no upscattering is considered) which was normally neglected in the dosimetry calculations.

The measured activation of irradiated detectors is used to calculate the fluence of neutrons which, combined with precalculated multigroup neutron flux and displacement cross-sections, allows to calculate the number of DPA [26].

Uncertainties implicit in transport calculations

There are a number of uncertainties [27-30] associated with the computations using transport codes. These include:

- Numerical approximations (quadrature used in S_N calculations, scattering cross-section expansion, mesh spacing, energy groups, statistical convergence criteria, etc.);
- Modelling approximations (capsule placement, PV thickness variations, cavity streaming, 3-D flux synthesis, peripheral subassembly source distribution, dimension and material uncertainties);
- Nuclear data uncertainties (cross-sections, dosimeter cross-sections used in calculating dosimeter responses, ^{239}Pu fission spectrum and ^{235}U fission spectrum for E>6 MeV).

The above listed uncertainties and recent improvements in their reduction are discussed in more detail in the following section of the report.

Most recent advances in dealing with computational uncertainties

In recently reported publications [27-40], all of the above defined computational uncertainties have been addressed and many have been reduced to acceptable levels. Thus, based on most recent publications [27-40], we have:

- S_8 quadrature sets in S_N computations are reported to be sufficient [35] in the in-vessel dosimetry computations; S_8 quadrature sets are generally considered not sufficient in the ex-vessel dosimetry computations;

- P_3 (for neutrons) and P_5 (for gamma rays) scattering cross-section expansion is commonly used [32] although a higher anisotropy order quadrature may be needed [35-37];

- recently, a claim was made that the S_N differencing scheme may introduce uncertainties in the results of about 3 per cent [39]; consequently, a new differencing scheme was proposed to remedy this situation [38];

- as fine as possible group structure should be used and attention should be paid to group boundaries; currently, cross-section libraries from 47 energy groups (BUGLE) up to "continuous" Monte Carlo data sets are used [27-40]; it was considered by the TFRDD group that the 47 energy group structure is sufficiently fine for practical applications [43] (providing that self-shielding is taken into account). In our experience it may not be possible to derive group-averaged cross-sections which are appropriate at all penetrations through a thick region of steel. This was demonstrated in the NEACRP study which considered the attenuation of neutrons through the steel body of a fuel transport flask (NEACRP-L-331), and similar problems may be expected to arise in the wall of a pressure vessel;

- fine spatial mesh structure is needed in computations;

- an uncertainty of about 20 per cent in the flux in the capsule is reported [31] if the radial position of the capsule is known within 0.5 inch;

- cavity streaming is reported [35] not to be a significant issue (S_N codes) when positions at core midplane are considered; conversely, at positions off core midplane ray effects may cause errors in S_N calculations;

- the 3-D flux synthesis based on 2-D and/or 1-D calculations is a reliable procedure [27,32,34,35,39] for in-vessel dosimetry at core midplane positions (S_N vs. Monte Carlo comparisons); conversely, the flux synthesis method is not an acceptable procedure [27] for ex-vessel dosimetry and/or above core midplane;

- peripheral subassembly source distribution may differ about 3-5 per cent from the average [31];

- current required tolerances on the manufacture of reactor components result in a 2.5 per cent uncertainty in the calculated reactor damage exposure parameters [31];

- material uncertainties in steel compositions have no significant effect on flux predictions but they can significantly affect the steel displacement damage function (trace impurities affect the damage annealing and the residual defect level), and thus the DPA exposure parameter; coolant density variations may cause about 6 per cent errors in the flux predictions at measurements positions [31] but smaller values are also reported [40];

- cross-sections are another source of uncertainties [27-40] with the inelastic scattering cross-section for iron contributing as much as 9 to 13 per cent errors (ENDF/B-IV and V) or about 6 per cent (ENDF/B-VI) to the high energy neutrons reactions predictions in the RPV and/or cavity (the problem does not exist for the low energy reactions); in many in-vessel dosimetry calculations the ENDF/B-IV and ENDF/B-V cross-section data are accurate enough; in the ex-vessel dosimetry calculations, the improved iron inelastic scattering cross-section data [41] (important in any deep penetration problem through the RPV wall) given in the ENDF/B-VI or EFF-3 data files should be used (the ENDF/B-IV and V iron cross-section data are not accurate enough in these ex-vessel dosimetry calculations);

- the uncertainty in the total hydrogen and oxygen cross-sections results in uncertainty in the flux prediction (1 to 3 per cent for hydrogen, and about 1 per cent for oxygen);

- certain dosimeter cross-sections may be a source of significant uncertainties in the computations, as the cross-sections for measured reactions have covariance matrices (for cross-sections) in reduced energy schemes resulting in the overall uncertainty in calculated responses of about 1-4 per cent for the dosimeters of interest in reactor surveillance dosimetry (except for the ^{237}Np and ^{232}Th fission monitors which are of the order of 10 per cent); the ^{237}Np(n,f) reaction is often underpredicted due to neglect of the photo-fission effects resulting in errors about 10 per cent or 20-30 per cent in some cases [31]; there are several current projects to improve the ^{237}Np(n,f) cross-section evaluation. The photo-fission effect for ^{238}U is also reported important [40]; overall, it is considered that in practical computations dosimeter cross-sections as given in the IRDF-90/Rev.2 data sets are accurate enough but that there is a need for a better representation of the photo-fission response; the data for these dosimeters should be provided with a photon energy-dependent covariance matrix, and incorporated into standard dosimetry cross-section libraries, such as IRDF-90 (energy-dependent photo-fission cross-sections can be found in Reference 42). An additional problem of computing photo-fission reactions arises from the fact that for most reactor facilities their photon spectra are not known, nor have their photon transport codes been validated for predicting the photon spectra; many cross-section libraries do not even include secondary photon production data. We are not aware of any good source of (γ,γ') cross-sections for dosimeters such as ^{93}Nb and ^{115}In;

- in fission detectors (^{238}U, ^{237}Np) large uncertainties can arise from impurities (e.g. ^{235}U in ^{238}U) and ^{239}Pu build-up;

- various ways of dealing with the ^{235}U fission spectrum are reported ranging from using a slightly harder spectrum [27] through adjustment procedures [40] to analytic fits to experimental data [34]; in the current state-of-the-art computations, a mixed high burnup fission spectrum is used (rather than the pure ^{235}U fission spectrum) which considers all important uranium and plutonium isotopes such as: ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, and includes the variation of the contributions during the irradiation when comparisons are being made with measurements.

Computational uncertainties – discussion

In the previous section, the state-of-the-art minimal values of various uncertainties associated with the computations of flux/fluence in the dosimetry calculations were reported. It should be emphasised that the reported values of uncertainties were picked from publications in the list of references. Often, the computations presented utilise procedures, data sets, and modelling assumptions which are not optimal (arising from availability of particular data sets and tools at particular organisations). Consequently, the computations that follow contain extra errors which adversely affect the results. The problem is dealt with by devising "corrective factors" to force a better agreement of the results with measurements. This approach often leads to ambiguities and difficulties in interpreting the results. In fact, the use of corrective factors or adjusted calculated and experimental values makes an objective comparison of calculated and experimental data very difficult or impossible. A more rigorous approach would be to start with the best data available and modelling assumptions as accurate as possible and then proceed with calculations. Of course, this is a very time consuming process but correcting and adjusting procedures are time consuming too and the results are not satisfying. The way of conducting dosimetry computations varies from organisation to organisation. In some approaches, no adjusting of the computational results takes place [43]. Instead, the unadjusted calculations are compared with measurements to determine the order of computational uncertainties. For example, the results published by Siemens/KWU in fluence reports are always obtained without using corrective factors and without adjusting calculated fluences.

Based on the state-of-the-art literature it is difficult to infer the relative (with respect to each other) merits and the magnitude of associated uncertainties of various computational techniques (and nuclear data sets) used as these are very strongly problem dependent.

Clearly, one of the biggest problems of the state-of-the-art reactor dosimetry is that of how to translate the computed fluences into metal damage. The limited accuracy of basic physics models of material degradation folded with estimated fluences in an effort to assess metal damage often yields results to which little confidence is given. An example can be provided of the BR3 reactor in Belgium which was shut down because the RPV embrittlement prediction methods used had not demonstrated the RPV integrity [43]. Hence, there is a strong need for the improved irradiation induced metal damage models.

Benchmarking of computational tools

Table 1. Important RPV dosimetry system validation benchmarks

Benchmark	Description
Winfrith Iron Benchmark Experiment	Determination of neutron spectra and detector reaction rates as a function of depth within an iron shield 1 m thick.
Winfrith Water/Iron Experiment (ASPIS-PCA Replica)	Determination of neutron spectra and detector reaction rates as a function of depth in the ASPIS water/iron shield. A mock-up of a PWR ex-core radial geometry and a replica of the ORNL PCA experiment with a fission plate in place of the core source.
Wuerelingen Iron Benchmark Experiment (PROTEUS)	Determination of neutron spectra and reaction rates as a function of depth in iron and stainless steel shield 80 cm thick.
Winfrith Iron 88 Benchmark Experiment (ASPIS)	Determination of the neutron transport up to 67 cm in steel.

Table 1. Important RPV dosimetry system validation benchmarks (cont.)

Benchmark	Description
Winfrith Water Benchmark Experiment (ASPIS)	Determination of fast neutron spectra above 1 MeV and detector reaction rates up to 50 cm in water.
Ispra Iron Benchmark Experiment (EURACOS)	Study of neutron deep penetration in homogeneous construction materials of advanced reactors: Fe and Na (flux and spectrum measured up to 130 cm in Fe).
Karlsruhe Iron Sphere Benchmark Experiment	Determination of neutron leakage spectra from a set of iron spheres of 15, 20, 25, 30, 35, and 40 cm diameters with a ^{252}Cf neutron source at the centre. The goal was to verify Fe inelastic scattering cross-sections. Pure Fe and spherical geometry were considered.
SDT11 Iron and Stainless Steel Experimental Benchmark (TSF-ORNL)	Determination of neutron spectra and transport through combined thicknesses of 1.5, 4, 6, 12, 24, 36 in of Fe and 12 and 18 in of stainless steel.
Winfrith NESDIP2 and NESDIP3 Radial Shield and Cavity Experiments	Determination of reaction rates in an array consisting of a water cell, thermal baffle, water cell, RPV, a 29 cm cavity, water cell, and 61 cm thick concrete slab.
VENUS (SCK/CEN, Mol)	Determination of typical PWR neutron spectrum and dosimeter reaction rates in a realistic core configuration near the fuel region and core barrel/thermal shield region.
SAINT-LAURENT B (cavity midplane) (France)	Determination of dosimeter reaction rates at cavity midplane.
H.B. Robinson Power Reactor (USA)	Special cavity dosimetry benchmark.

In order to estimate how well computational tools using processed data libraries can predict fluence and doses in reactor components and to judge sources and magnitude of computational uncertainties, results from measurements on clean experimental configurations need to be compared against calculations.

Over the past 20 years a number of such experiments have been conducted at various laboratories in Europe and the USA. The high-priority experimental benchmarks widely used in dosimetry codes validation are listed in table above together with a brief description of their scope (the detailed description of these benchmarks can be found in Reference 45; see also the French, Belgian, and US sections).

Status of computing radiation dose to reactor components in the TFRDD Member countries

Belgium

The Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) [44]

The US Nuclear Regulatory Commission (NRC) established the Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) in 1977 to improve, maintain, and standardise neutron dosimetry, damage correlation, and the associated reactor analysis procedures used for predicting the integrated effect of neutron exposure to LWR-PV. Four principal laboratories collaborated in the conduct of this program: the Belgian Nuclear Research Centre

(SCK•CEN, Belgium), the Hanford Engineering and Development Laboratory (HEDL, USA), the National Institute for Science and Technology (NIST, the former NBS, Gaitherburg, Maryland, USA), and Oak Ridge National Laboratory (ORNL, Tennessee, USA). Subsequently, collaboration with additional European laboratories was established: Rolls-Royce & Associates Ltd. (RR&A, UK), the Atomic Energy Authority (AEA-Winfrith, UK), and the German Nuclear Research Centre (KFA-Jülich, Germany).

The LWR-PV-SDIP adopted specific experimental and calculational strategies to meet the challenge of the complex radiation-induced PV embrittlement phenomenon. The major benefit of this program has been and continues to be a significant improvement in the accuracy of the assessment of the current metallurgical condition and the remaining safe operating lifetime of LWR-PV.

The LWR-PV-SDIP has produced a broad range of technical accomplishments such as:

- establishment of ASTM standards for reactor dosimetry [46],
- establishment of benchmark fields for reactor dosimetry validation and improvement [47-51],
- definition of the state-of-the art LWR-PV surveillance dosimetry,
- identification of deficiencies in the neutron transport cross section of iron,
- quantification of active detector perturbations,
- assessment of the photoreactions contributions,
- observation of radiation induced embrittlement in LWR-PV steels, and
- recommendation for ex-vessel dosimetry using SSTR and HAFM.

This is a non exhaustive list in which all the metallurgical findings of the LWR-PV-SDIP program are not reported; additional items can be found in the reference list of documents [44].

Validation of the Belgian PV-surveillance dosimetry calculational schemes: state-of-the-art

The Belgian Utility ELECTRABEL operates seven PWR units. The PV surveillance dosimetry measurements of these reactors are currently carried out by SCK•CEN and the calculations are shared between SCK•CEN using the LEPRICON code system [52] and TRACTEBEL EE using the Monte Carlo code MCBEND.

Both methods have been extensively validated on a range of experimental benchmarks and power plant dosimetry results. Nevertheless, SCK•CEN and TRACTEBEL EE decided in 1993 to validate respectively their calculational scheme on the same valuable benchmark experiment. Among the available experiments, the VENUS configurations offered outstanding advantages of realistic core radial shape and typical PWR emerging neutron spectrum. The results of these experimental validations [53,54] showed the ability of both methodologies to predict the experimental results within less than 10 per cent. The global uncertainties of the calculated values are within 20 per cent.

Results of the computational analysis of a 900 MWe PWR (TIHANGE-2) surveillance capsule

Both calculational schemes discussed above were used to analyse the second surveillance capsule unloaded from the 900 MWe PWR TIHANGE-2 [55-57]. The comparison of the calculated and measured reaction rates (^{63}Cu(n,a), ^{54}Fe(n,p), ^{58}Ni(n,p), ^{46}Ti(n,p), ^{238}U(n,f), and ^{237}Np(n,f)) at the surveillance capsule position showed the ability of the two schemes to predict the experimental values within a maximum discrepancy of 10 per cent. The uncertainty of the calculated fluence is estimated at 20 per cent when using the LEPRICON code system. This uncertainty can be reduced to 4 per cent when using the adjustment module of the LEPRICON system. The uncertainty of the calculated values using the MCBEND methodology ranges from 13-20 per cent depending on the response one considers.

Denmark

In Denmark, there are two reactors operating:

- the 10 MW heavy water moderated research reactor DR3 with an aluminium vessel, and
- the 2 kW homogeneous spherical research reactor DR1 with a stainless-steel vessel.

The problems of irradiation induced degradation of the vessels of these two reactors are negligible. In the first reactor, the aluminium tank is weakened insignificantly by the formation of silicone. In the second reactor, the stainless-steel vessel receives negligible neutron doses. The degradation of the vessel in the second reactor is hence due primarily to the chemical corrosion; annual chemical analysis of the core solution specimens are conducted to monitor the corrosion effects in this reactor vessel. Therefore, in Denmark no particular problems in the context of this report are to be dealt with.

The experience in reactor vessel fluence computations in Denmark stems from performing several estimations of induced activities in reactor components of Swedish reactors leading to a rather surprising conclusion that quite simple calculational schemes yielded acceptable results. In these studies, the flux distributions were obtained using standard diffusion theory code and twenty energy groups cross-sections. The activation calculations were then performed under the assumption of constant flux spectrum (but varying amplitude). In spite of strong streaming effects in the air cavity around the reactor vessel and flux attenuation factors of the order of 10^{10} both total neutron fluxes and flux spectra were confirmed by Monte Carlo calculations. The recorded discrepancies between the diffusion theory and Monte Carlo calculations were of the order of factor of 5 and in most cases within a factor of 2.

Finland

In Finland, VTT Chemical Technology mainly performs surveillance neutron dosimetry for the Loviisa VVER-440 reactors and to a lesser extent for the BWR reactors at Olkiluoto. In the near future beam characterisation measurements for the Finnish BNCT facility will be performed. Neutron dosimetry transport calculations are performed at VTT Energy.

Methods

For the Loviisa reactors a semi-empirical method based on measured ^{54}Mn activities from the $^{54}Fe(n,p)$ reaction and utilisation of the kernel-based PREVIEW program (developed at VTT Energy) is used [58,59]. The detailed local irradiation history is accounted for (variations in axial flux distribution and neutron spectrum during an operating cycle).

The chain arrangement of the surveillance specimen capsules in the Loviisa reactors results in an effectively random azimuthal orientation of the specimens. This, in combination with a large radial flux gradient makes it necessary to determine the fluences individually for each specimen. This was achieved by irradiating Fe plates close to the steel specimens in the irradiation capsules.

In addition, samples were milled from the RPV cladding (Loviisa 1: 1980, Loviisa 2: 1986) and from the RPV outer surface (Loviisa 1: 1986). A cavity rack with several dosimeter capsules and vertical and horizontal Fe and Ni wires was irradiated in Loviisa 1 in 1984 - 1985. For the cladding samples the reactions $^{54}Fe(n,p)$, $^{58}Ni(n,p)$, $^{93}Nb(n,n')$, and $^{58}Fe(n,\gamma)$ (thermal neutron reaction) were utilised; for the RPV outer surface only the Fe reactions were considered.

For the Olkiluoto BWR reactors a conventional method with fixed calculated neutron spectra (ANISN) combined with global power history and measured activities are used.

For the activity measurements HPGe spectrometers with carefully calibrated measurement geometries are used. ^{93m}Nb activities are measured with Liquid Scintillation Counters.

PREVIEW code methodology

PREVIEW (PREssure Vessel Irradiation Evaluation Working code) calculates the flux, fluence or various dosimetry reactions at one or more “detector points”. These may be located at the surveillance chains or at the inner surface, one quarter thickness or the outer surface of the pressure vessel. A full core or reduced core with dummy elements can be calculated. The permitted axial range for the detector points is from half a meter below the core bottom to half a meter above core top. Their azimuthal locations can be chosen arbitrarily except at the surveillance chains.

Comparisons with measured activities have shown the accuracy of PREVIEW to be very good, with errors in the fast flux mostly amounting to less than 15 per cent – and still much less at the inner surface of the pressure vessel.

PREVIEW is currently in routine use for estimating pressure vessel irradiation of the Loviisa Nuclear Power Station (VVER-440) and at VTT for calculations needed in the evaluation of pressure vessel dosimetry measurements.

The flow of calculations with PREVIEW code is presented in Figures 1 and 2. The code starts from a three-dimensional power and burn-up history file calculated with core simulator code HEX-3D. Based on this, the source in each node is calculated and multiplied by pre-calculated kernels, with the contributions from all nodes summed to obtain the total flux. The change in the fission spectrum with changing fuel composition is taken into account, as is the effect of the movement of the fuel followers of control rods on the source distribution. PREVIEW gives the flux or fluence and reaction rates at the specified detector points. The fluxes and fluences can be obtained in the 7-group scheme used by PREVIEW itself, in a fine-group scheme with 47 groups or in a broad group scheme with 5 groups.

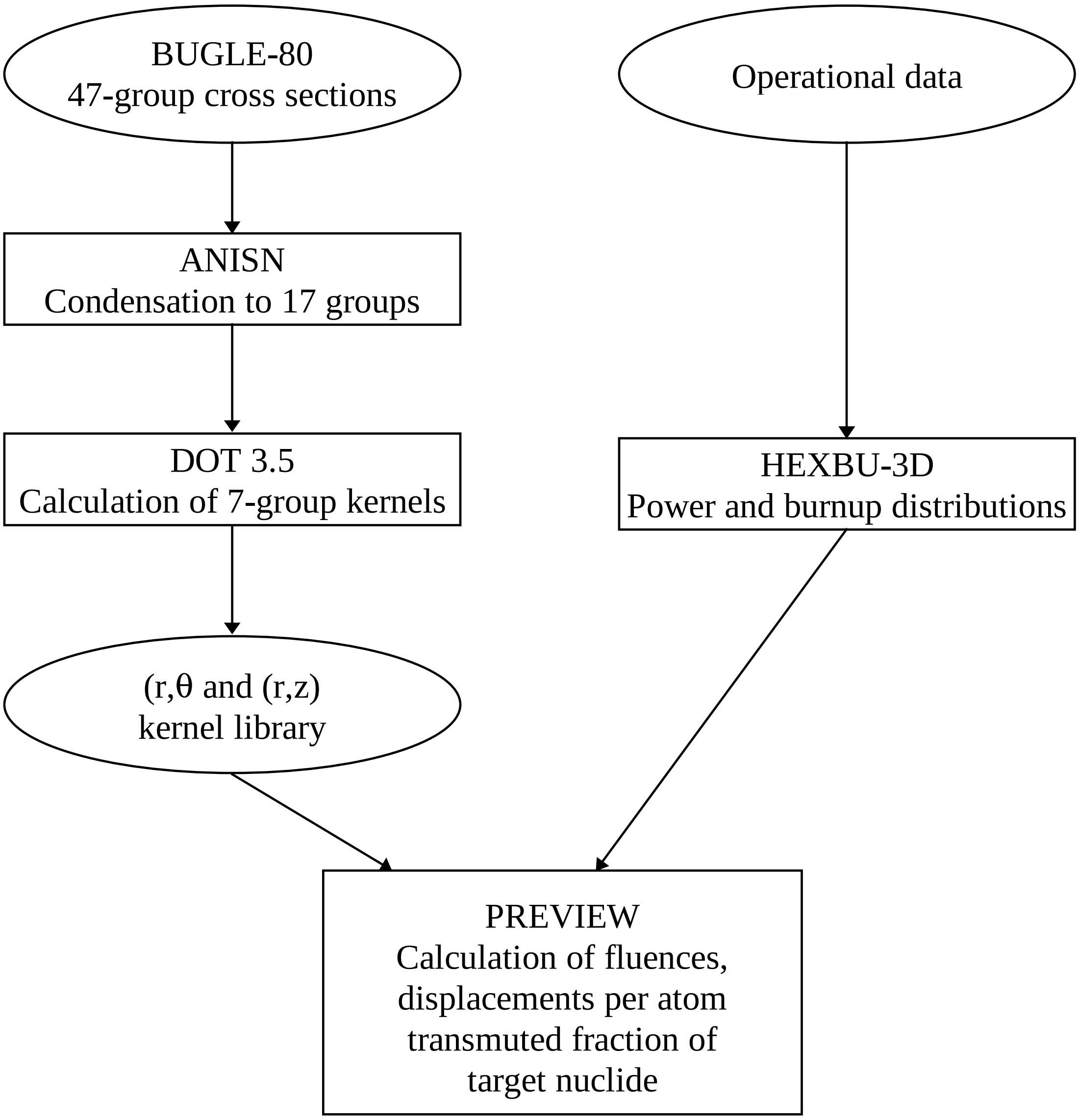


Figure 1. Simplified antecedents of PREVIEW results

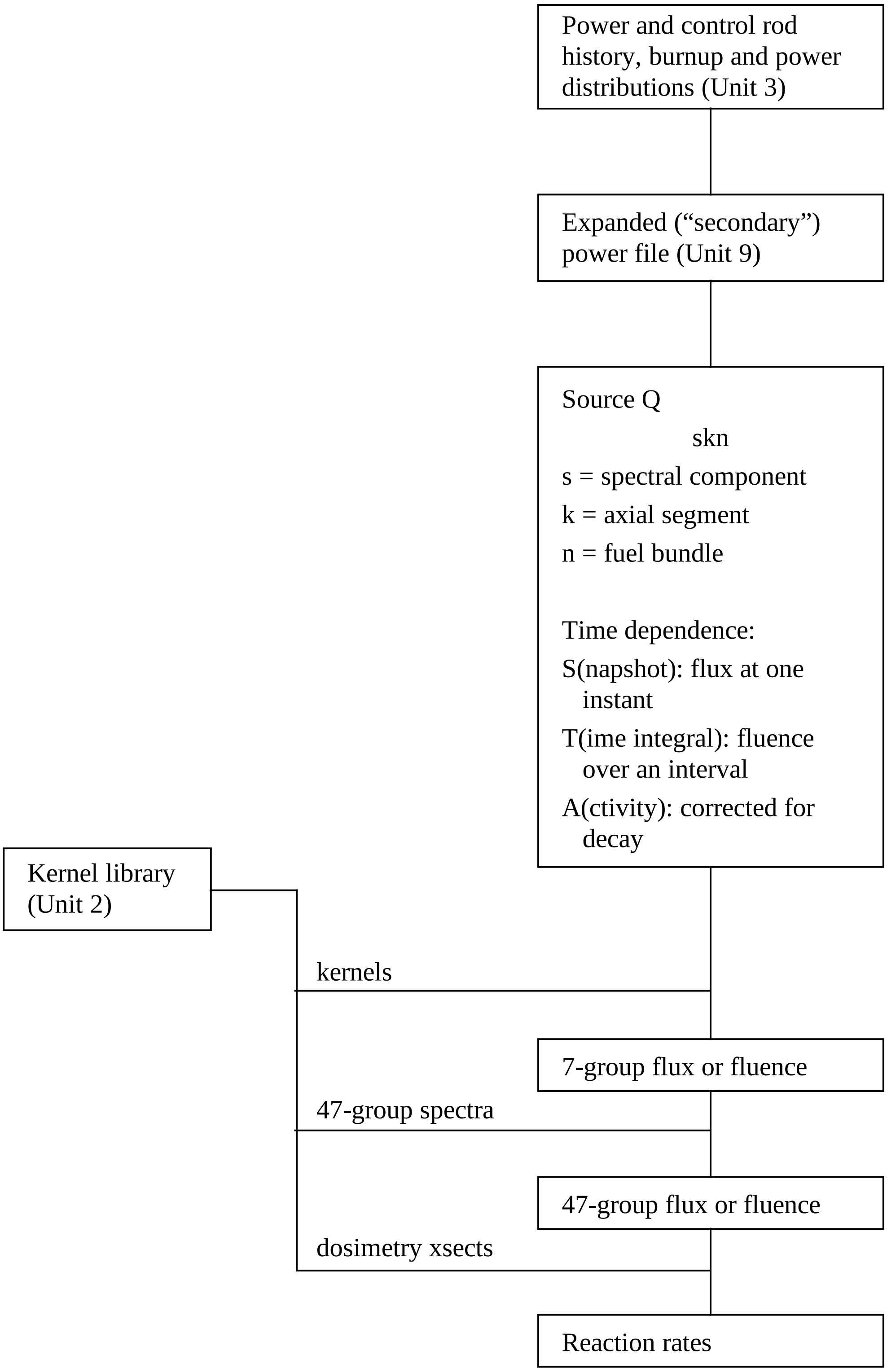


Figure 2. Calculation flow in PREVIEW

Since most of the irradiation of the pressure vessel comes from the outer core (defined here as consisting of the two outermost layers of fuel assemblies), with the inner core making a very small contribution, it is not necessary to treat the contribution of the inner core in detail. Consequently, PREVIEW uses somewhat different methods for the outer and inner core.

In both cases, however, calculation starts with determining the source in each node from nodal power, taking into account the fact that the source-to-power ratio is a function of burn-up, due to change in contributions from different fissionable nuclides as the composition of fuel changes. This also affects the fission spectrum which influences both the probability that a neutron will reach the surveillance chain or the pressure vessel and the damage it will do. This is taken into account by treating the two main contributors, thermal fission in ^{235}U and ^{239}Pu separately and using different transport kernels for them. Contributors from less important nuclides and from other incident energy ranges are added to these two components in accordance with the hardness of their fission spectra.

For the outer core each spectral component of the source in each node is multiplied by an (r,θ) kernel giving its contribution to the flux in each of the 7 groups and to the displacement rate at all desired radial and azimuthal locations. The result is multiplied by an axial kernel depending on the horizontal and vertical distances from the node to the detector points. The contributions from all outer core nodes in a 60 degree sector are then summed.

The inner core is treated in a simpler manner. There is a single (r,θ) multigroup kernel for the inner core but the contribution from each node is weighted by a factor dependent on its distance from the core axis. The inner core is also subdivided into axial layers with their contributions multiplied by an axial kernel.

The summations over fuel assemblies are carried out separately for the outer core and the inner core. Once the 7-group fluxes are obtained, they are unfolded into a fine group structure (i.e. 47-groups of BUGLE-80). The fine group fluxes may then be multiplied by suitable cross-sections to get reaction rates which may include fluxes in a broad group structure.

The kernels were obtained using DOT 3.5 and BUGLE-80 cross-sections condensed to 17 groups by ANISN (see Figure 1). The (r,θ) kernels were calculated separately for each fission spectrum component in each outer core fuel assembly position for a 60 degree sector of both the full core and the reduced core using periodic boundary conditions, so that actually each kernel gives the flux from 6 assemblies at 60 degree intervals. The axial kernels were obtained from (r,z) and one-dimensional radial calculations and are of the form $K_z(z,r)=\phi(r,z)/\phi(r)$.

In both ANISN and DOT 3.5 calculations the quadrature order was S_8P_3. The space mesh was sufficiently fine in the ANISN calculations not to be a likely source of significant errors (about 0.4 cm to 0.6 cm near material boundaries and in other places where a fine mesh was desirable, increasing to 2 cm in the concrete shield and 8.487 cm in the innermost part of the core). In the DOT 3.5 calculations it was necessary, due to budget limitations, to use a slightly coarser mesh than optimal – the (r,θ) calculations used a mesh interval of 1.5 degrees for θ and a radial interval of 2 cm or less in the important locations, increasing in the interior of the core. In the (r,z) calculations, the radial mesh interval was about 2.5 cm and the axial interval was 3.125 cm.

Further details on the PREVIEW code can be found in References 60, 61, and 62.

Comparison of calculations and measurements

A large data base of measured activities from several locations in the Loviisa reactors exists. An extensive comparison of calculated (PREVIEW) and measured activities [59] and revision of all dosimetry results for the Loviisa reactors based on the above mentioned method were carried out in 1993.

The results of the comparisons can be summarised as follows:

- At the surveillance chain positions (r=162.5 cm) the ^{54}Mn activities are underestimated by about 10 per cent over most of the core height and slightly overestimated at the ends. The calculated neutron spectrum shape is slightly on the hard side.

- At the inner pressure vessel surface (r=177.55 cm, 0.45 cm inside cladding) the agreement between calculated and measured ^{54}Mn, ^{58}Co, and ^{93m}Nb activities is very good.

- In the cavity outside the pressure vessel the activities are underestimated by about 10-15 per cent.

Estimated uncertainties

At present, the uncertainties (1σ) in the semi-empirical fluence estimates for the surveillance specimens are about ±11 per cent for fluence >1 MeV and slightly larger (order of 13 per cent) for DPA and fluence >0.5 MeV (Eastern European Standard). For the pressure vessel the corresponding uncertainties are within the range of 20-25 per cent.

An improvement is expected and more rigorous basis for the uncertainty estimates will be established in the near future with adjusted neutron spectra.

Dosimetry data

The dosimetry cross-sections (built into the PREVIEW kernel library) are taken from IRDF-90 v. 2 (IAEA 1993) and from recent Obninsk evaluations (Nb and Ti, Zolotarev *et al*., 1994). For the DPA cross-sections both ASTM and Euratom data with NUREG/CR-5530 at low energies are available.

Half-lives and gamma emission probabilities are taken from “XG Standards” (IAEA-NDS-112 Rev. 2, 1991). Isotopic compositions, atomic weights etc. have been taken from EUR 12354 EN (J.H. Baard *et al*., 1989), except for Fe (Wille & Jede, EUR 14356 EN, 1992).

Future improvements

The activation results from several surveillance chains and a cavity irradiation have been used to adjust the calculated (PREVIEW) neutron fluence spectra using the LSL-M2 code (F.W. Stallman, NUREG/CR-4349, ORNL/TM-9933, 1986). This provides a basis for reduced and more rigorous uncertainty estimates.

The results will be incorporated into an optimised adjustment library which will be interfaced directly to the PREVIEW program without changing the kernel library itself. This leaves open the option of calculating unadjusted fluences for direct comparison with previous results.

France

In France, fluence calculations in reactor pressure vessels and irradiation samples of different standardised PWR reactor series are carried out by three organisations: FRAMATOME and CEA (Commisariat à l'Énergie Atomique) on a contractual basis following their withdrawal from the program in 1974 and the electricity producer EDF in the framework of the maintenance support services.

The designer FRAMATOME determines pressure vessel fluence and the lead factor (defined as the ratio of the capsule flux to flux at most exposed azimuthal location); in both cases fluence above 1 MeV is considered. Also, FRAMATOME established the pressure vessel surveillance program. For its part, the CEA performs on behalf of the EDF utility reference calculations of neutron spectra (CEA/DRN/DMT/SERMA) necessary for the analysis of surveillance capsules (CEA/DRN/DRE/SRS). These reference calculations provide both the lead factor and the pressure vessel fluence under nominal operating conditions. Reference support irradiations are performed at OSIRIS reactor (CEA/DRN/DRE/SRO). Finally, the EDF services responsible for fuel management carry out the monitoring of each power plant reactor pressure vessel fluence, taking into account operational history and fuel reload (EDF/EPN/GCN). In this monitoring, calculations using a simplified reactor model are adjusted via reference calculations.

The fundamental characteristics of the French commercial PWR plants is a great degree of standardisation of core management schemes of only a few standardised plant series: at nominal power, neutron flux at pressure vessel does not vary significantly from one plant to another. This variation tends to become even less significant in recent years, most of it resulting from low leakage core management configurations, use of MOX or URT fuels, or extended fuel cycle schemes.

Fluence calculations at reactor designer FRAMATOME

FRAMATOME performs fluence calculations to estimate future radiation doses of reactor vessel and internals already at the reactor design stages as well as calculates doses to pressure vessels of operating plants.

In fluence calculations, FRAMATOME uses a system of codes whose core constitutes the 2-D S_N code DORT [63]. DORT allows modelling of propagation of neutrons and/or photons between the point of their generation and all points of modelled domain by solving transport equation using discrete ordinates method. The advantage of discrete ordinates method is that it provides flux distribution in all points of modelled system in one calculation. The interactions of particles with matter are modelled by means of microscopic cross-sections from recent nuclear cross-section data files (ENDF/B-IV and partially ENDF/B-VI). The cross-sections of isotopes most important in flux calculations (Fe, Cr, Ni) are supplied by CEA (a 100 energy groups neutron library and 30 energy groups gamma library) and take into account the self-shielding phenomena in considered materials. In standard calculations, the particle sources are described pin-by-pin in peripheral assemblies and assembly-by-assembly in the interior of a reactor core. Since DORT solutions are two-dimensional, FRAMATOME uses DOTSYN [64] code to perform a three-dimensional reconstruction of the flux. This reconstructed three-dimensional flux is then combined with dosimetry cross-sections to deduce reaction rates which can be compared directly with dosimeter responses (see page 32, *Experimental flux above 1 MeV*).

The combined DORT and DOTSYN calculation yields in all point of the model considered:

- neutron flux above threshold energy E_0 (e.g. E_0=1 MeV or 100 keV),
- neutron and/or gamma spectra,
- saturation activities of different dosimeters (Fe, Cu, Ni, etc.).

Reference fluence calculations at CEA

Reference fluence calculations are carried out using the multigroup 3-D Monte Carlo code TRIPOLI 3 [65-67]. The choice of the Monte Carlo method allows to take into account exactly fine details of the 3-D geometry of the reactor and in particular to model the stiffners (see Figure 3) which are the 4 cm thick plaques of steel perturbing axially neutron spectrum at surveillance capsules positions(see Figure 4) [68,69].

Neutron source densities are described pin-by-pin in peripheral assemblies and assembly-by-assembly in the interior of a reactor core. The characteristics of fission neutrons are that of uranium (^{235}U, ^{238}U) and plutonium (^{239}Pu, ^{241}Pu) fissions. The rate of fissions of each type depends on assembly burn-up and is determined in cell burn-up calculations using APOLLO code [71].

In commercial EDF reactors the reference calculations are done using ENDF/B-IV cross-section data files. In Reference 80 it is shown that the Fe inelastic scattering cross-section in ENDF/B-IV data file is not sufficiently accurate, its value in the ENDF/B-VI data file is significantly improved. Hence, CEA uses the ENDF/B-VI library since 1994 in its reference calculations for different French standardised PWR plant series and in international benchmarks.

The cross-sections used are in the 315 energy groups multigroup form [71] and include probability tables. The probability tables [72-74] allow to account for variations in cross-section values within the groups by assigning a probability to each possible within the group cross-section value. The resulting multigroup Monte Carlo calculations yield the same results as when 3857 continuous Monte Carlo energy group structure is considered. In neutron/matter interactions, the degree of anisotropy and final energy of scattered particle relative to its initial energy are determined for each particle interaction.

The results of TRIPOLI calculations give:

- neutron spectra in all surveillance capsules and pressure vessel,
- neutron flux above 1 MeV and 0.1 MeV as well as DPAs in pressure vessel and irradiation samples (determination of the lead factor). The lead factor is used to translate the irradiation sample damage to pressure vessel damage,
- saturation reaction rates in capsules τ_k for different dosimeters k ($^{63}Cu(n,a)^{60}Co$, $^{54}Fe(n,p)^{58}Co$, $^{238}U(n,f)^{137}Cs$, $^{237}Np(n,f)^{137}Cs$); the average cross-sections $\sigma_{1,k}=\tau_k/\phi_k$ are also given.

The statistical error of this obtained flux above 1 MeV (ϕ_1) using 5 million particle histories is of 1 per cent in the vessel and 1.5 per cent in the surveillance capsule. The older results presented in Figure 4 were obtained using several hundred thousand particle histories.

Figure 3. Tripoli geometry

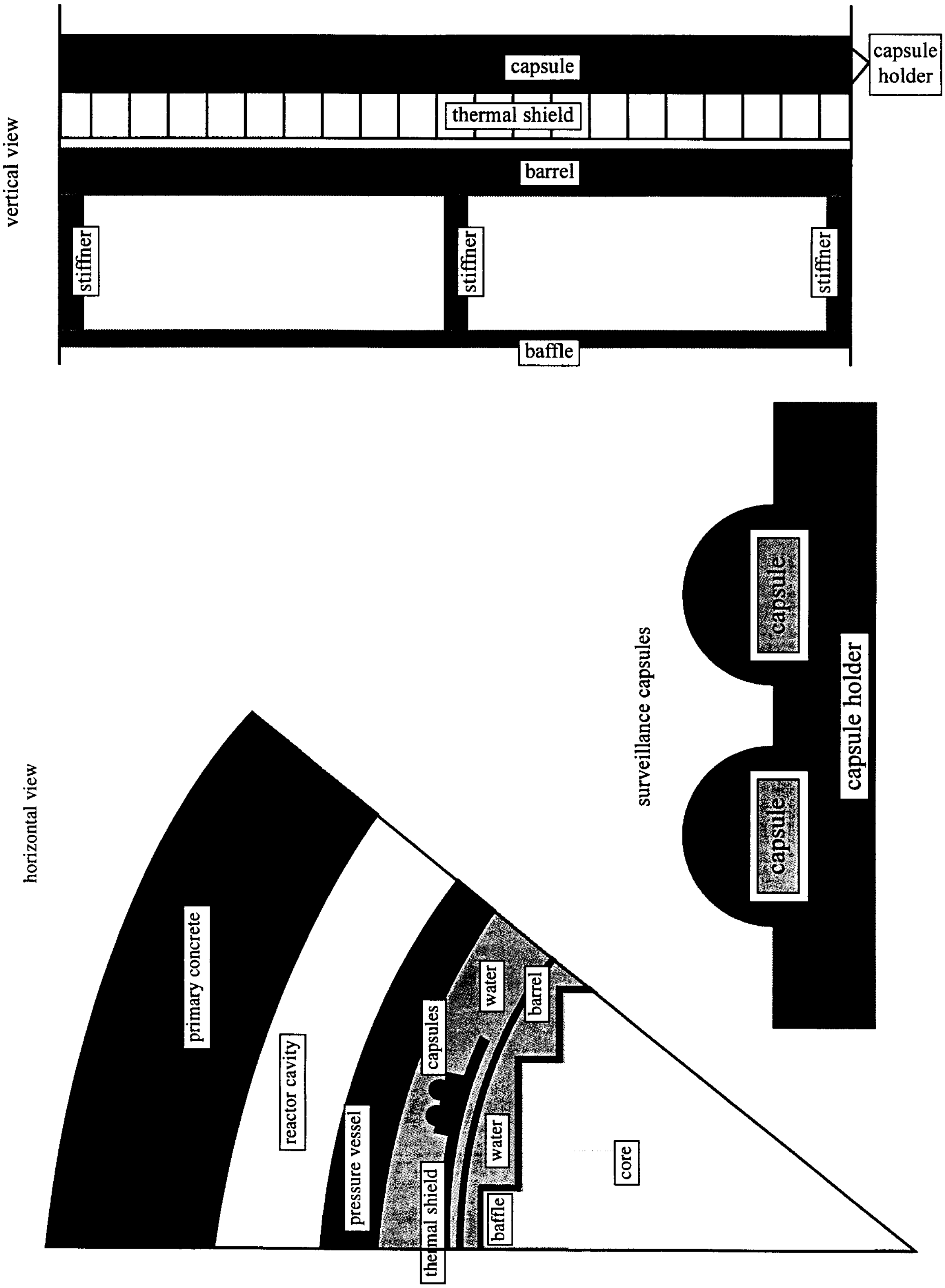
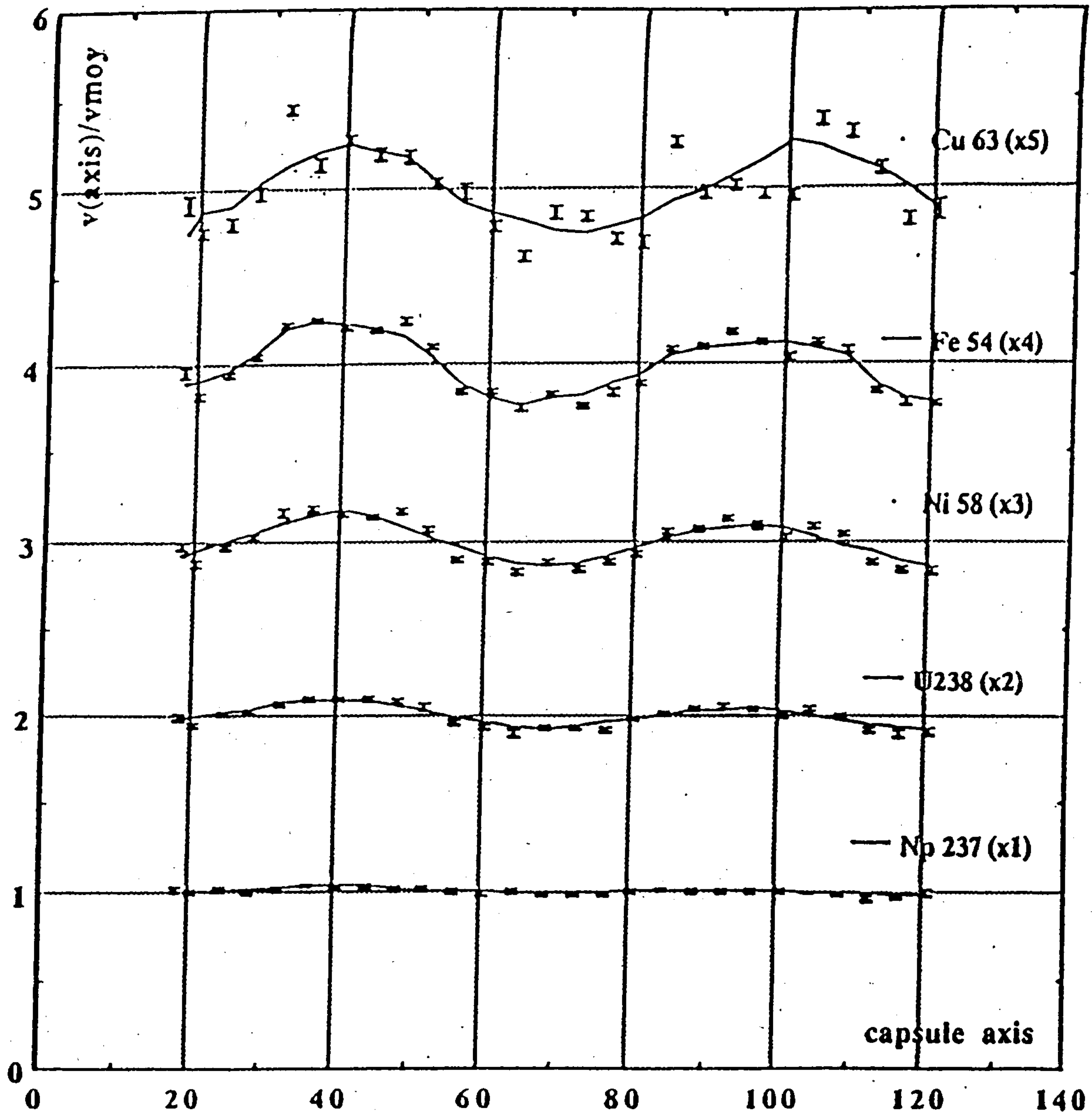


Figure 4. Axial fluctuations of reaction rates

Cb6-b calculation G1-capsules



Experimental flux above 1 MeV

Neutron flux above 1 MeV is not a measurable quantity. It can be however, deduced from the measurements.

By considering the power history of a reactor and by accounting for the parasitic reactions (i.e. impurities, burn-up, photofission, traces of fissile isotopes ^{238}U and ^{237}Np in dosimeters), the code measured activity. The experimental saturation activity is then compared with calculated saturation activity τ_k. From experimental activities $\tau_{k,exp}$ and calculated average dosimeter cross-sections $\sigma_{1,k}$ neutron flux above 1 MeV seen by the detector k can be deduced as $\phi_{1,k,exp}=\tau_{k,exp}/\sigma_{1,k}$. It should be noted that the average dosimeter cross-sections $\sigma_{1,k}$ resulting from calculations are quite sensitive to neutron spectra variations dependent on nuclear data chosen.

The uncertainty analysis is supplied with the experimental results as well as the standard deviation of three measurements carried out on surveillance capsule specimens. These are given in Table 2.

Table 2. Uncertainties and standard deviations on measurements

	^{63}Cu	^{54}Fe	^{58}Ni	^{238}U	^{237}Np
Uncert.	2.7%	2.80%	2.70%	6.80%	4.30%
G1 stan. dev.	2.67%	3.00%	4.69%	8.44%	3.59%
G2 stan. dev.	2.35%	3.45%	4.58%	8.18%	2.82%
G3 stan. dev.	2.54%	2.72%	3.58%	10.5%	2.75%

G1: 24 capsules – working time: 4 years, azimuthal location: 20°
G2: 20 capsules – working time: 7 years, azimuthal location: 20°
G3: 12 capsules – working time: 9 years, azimuthal location: 17°

From the uncertainties in activities of different dosimeters and the uncertainties in the calculated average dosimeter cross-sections $\sigma_{1,k}$ the uncertainty in $\phi_{1,k,exp}$ can be deduced. Then the average uncertainty can be derived from the above uncertainties using weighting depending on their value to determine the final experimental value of the neutron flux above 1 MeV denoted as $\phi_{1,exp}$. In France, $\phi_{1,exp}$ obtained from the analysis of surveillance capsule samples is used in characterising metal damage.

Recent experimental data produced in test reactors for CEA/EDF specific program (ESTEREL) indicate fluence above 1 MeV as the most relevant exposure parameter for surveillance program [14].

Monitoring of pressure vessel damage

Fluences received by pressure vessels of different power plants vary from fluences obtained as a product of: flux calculated assuming ideal core management, load factor, and time of reactor operation. The reason lies in variations in core management schemes of power reactors (i.e. fuel reload pattern, load follow, fuel cycle extension, etc.). EDF uses the code EFLUVE [75] for fast coupled fluence and core management calculations.

During the studies of core fuel management schemes this code allows to consider minimisation of fluence to pressure vessel as a criterion of choosing the fuel management pattern. During plant operation EFLUVE allows to follow the fluence received by the pressure vessel by taking into account the actual power history of the reactor.

The code utilises the following source attenuation law: $\phi = Se^{-\mu\tau}/4\pi\tau^2$. In EFLUVE, the source is given pin-by-pin in core calculations taking properly into account fission rates in uranium/plutonium of considered assemblies. The attenuation coefficient μ is adjusted as that EFLUVE follows the results of reference TRIPOLI calculations (pressure vessel and irradiation capsules flux) for standardised plant series.

Finally, EFLUVE is benchmarked with respect to TRIPOLI code for other (than standard) types of core management schemes (low fluence, extended fuel cycles, MOX, etc.); the uncertainty of this obtained lead factor without taking into account the effects of uncertainties in cross-sections is estimated at 4 per cent. The effects of temperature variations during irradiation and of changes in fuel plutonium content are accounted for and adjusted using the deterministic finite difference transport code SN1D [86].

Comparison of calculations and measurements

Experiments

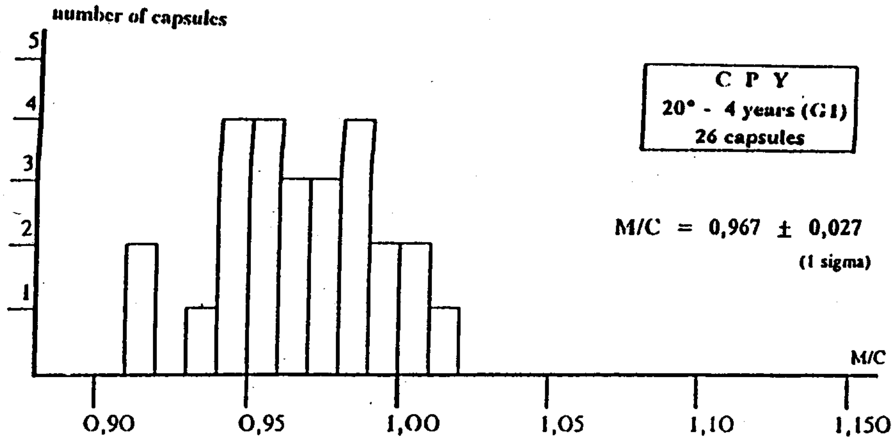
The surveillance capsules containing typical dosimeters were removed from the following reactors: DAMPIERRE, TRICASTIN, GRAVELINES, BLAYAIS, CHINON B, CRUAS, and SAINT-LAURENT B. For these capsules, experimentally determined flux above 1 MeV - $\phi_{1,exp}$ was compared against calculations using TRIPOLI code. The results of these comparisons are presented in Figure 5 in the form of histograms of the number of capsules versus M/C ratio of measured-to-calculated flux above 1 MeV. The TRIPOLI calculations were performed within the reference calculations scheme described previously.

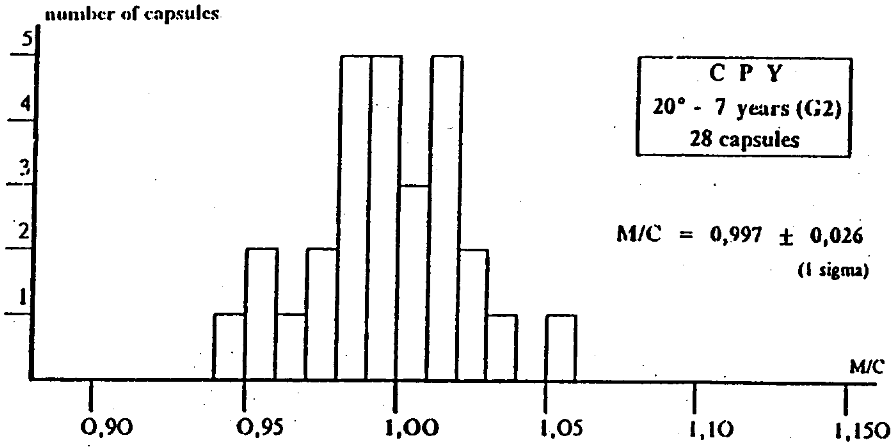
For the CHOOZ A power plant scraping samples of the stainless steel pressure vessel lining were taken in the past. The ^{54}Mn(^{54}Fe(n,p)^{54}Mn reaction) dosimetry measurements were carried out on pulverised samples. The initial ^{54}Fe content in the stainless steel was obtained (after the measurements) by re-irradiating the samples in the presence of pure iron indicator. The experiments were analysed by FRAMATOME and CEA, and a fairly good correlation between calculations and experimental results was obtained (~10 per cent). In addition, an attempt was made to perform ^{93}Nb(n,n') dosimetry measurements but too low content of ^{93}Nb in samples made it impossible.

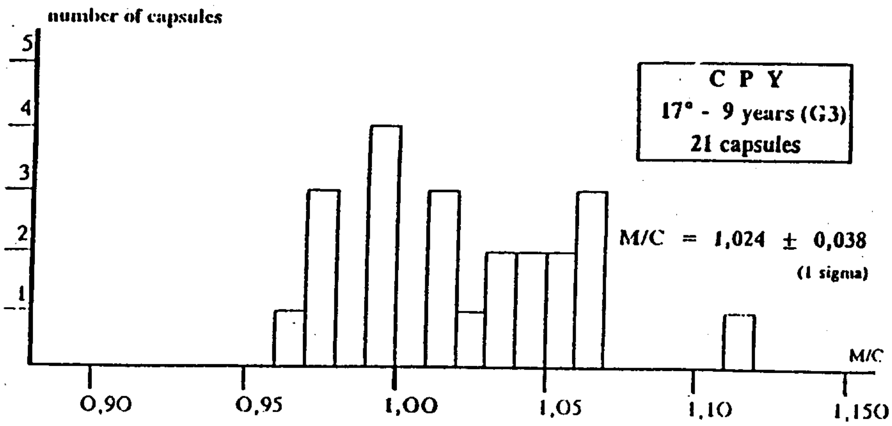
Recently, during an outage of the CHOOZ A reactor new scraping samples were taken from the reactor pressure vessel stainless steel lining. CEA will repeat the old analysis of samples (described above) with increased statistical precision.

Figure 5. CPY reactor series: Histogram of the RPV surveillance results

Measurements/calculations ratio for the flux above 1 MeV







CEA/DRN
CEA/Grenoble
24/05/1996

Validation by benchmarks

Benchmarks allow to validate cross-sections as well as their processing before using them in transport codes (energy group structure, anisotropy, etc.). The following benchmark experiments related to reactor pressure vessel dosimetry were analysed at CEA using TRIPOLI code with ENDF/B-IV, JEFF, and ENDF/B-VI nuclear data:

- REPLICA,
- ASPIS,
- NEEDS,
- SAINT-LAURENT B (cavity),
- WINFRITH light water,
- MOL light water (in progress).

To illustrate these benchmarks some results of calculations/experiments carried out using TRIPOLI code with ENDF/B-VI data are given in Tables 3, 4, 5, and 6 (see also References 76-79).

Table 3. REPLICA (PWR mock-up configuration)

Medium	Distance to Source (cm)	Uncertainty* C/E $^{103}Rh(n,n')^{103m}Rh$	C/E	Uncertainty* C/E $^{115}In(n,n')^{115m}In$	C/E	Uncertainty* C/E $^{32}S(n,p)^{32}P$	C/E
12 cm of water	1.91	7.2%	0.99				
	7.41	7.2%	0.89				
	12.41	7.2%	0.89				
	14.01	7.2%	0.85				
13 cm of water	19.91	7.3%	0.94				
	25.41	8.2%	0.88				
	30.41	8.2%	0.87				
RPV T/4	39.01	5.8%	0.91	5.2%	0.91	7.0%	0.96
RPV 3T/4	49.61	6.4%	0.95	5.5%	0.88	7.5%	0.99
Cavity	58.61	6.3%	0.91	5.6%	0.91	7.0%	1.06

*Cross-section uncertainties not included.
T/4 = ¼ of pressure vessel thickness T.
3T/4 = ¾ of pressure vessel thickness T.

Table 4. ASPIS (iron slab 1.2 m long)

d (cm)	^{103}RH(n,n')^{103m}Rh		^{115}In(n,n') ^{115m}In		^{32}S(n,p)^{32}P	
↓	Uncertainty*	C/E	Uncertainty*	C/E	Uncertainty*	C/E
5.72	6.0%	0.86	5.6%	0.99	6.6%	1.00
11.43	6.0%	0.95	5.8%	1.05	6.6%	1.03
17.15	6.1%	0.95	5.9%	0.98	6.7%	1.09
22.86	6.1%	0.94	6.1%	1.03	6.6%	1.05
28.58	6.1%	0.97	5.9%	1.03	7.0%	1.10
34.29	6.3%	1.09	6.1%	1.12	6.6%	1.08
45.72	6.2%	1.11	6.9%	1.08	6.6%	1.10
51.44	6.1%	1.05	6.1%	1.02	6.7%	1.14
57.15	6.1%	1.10	7.0%	1.15	6.6%	1.02
62.87	6.1%	1.08	6.1%	1.07	6.9%	1.15
68.58	6.1%	1.15			6.7%	1.00
85.73	7.0%	1.13				
114.30	6.2%	1.05				

*Uncertainties in nuclear data not included.

Table 5. SAINT LAURENT B1 (cavity midplane)

Dosimeter	Cu	Fe	Ni	Nb	^{238}U	^{237}Np	$\varnothing_1$
C/E	1.081	1.085	1.157	1.069	1.032	1.054	1.08

Table 6. Comparison DORT/measurements of dosimeter activities of surveillance capsule SLB1

Dosimeter	^{54}Fe	^{58}Ni	^{63}Cu	^{93}Nb	^{238}U
C/E	0.97	0.94	0.93	0.92	0.90

FRAMATOME validated the code system DORT and data processing tools it had developed (e.g. automatic mesh generator, graphical tools, etc.). The validation was carried out by means of experimental benchmarks conducted by EDF and CEA at the St Laurent B1 reactor [79] as well as NESDIP experiment. The comparisons of DORT versus measurements are given in Table 6 and in Figure 6 showing the results of 3-D simulation of the effects of stiffners on surveillance capsules fast flux.

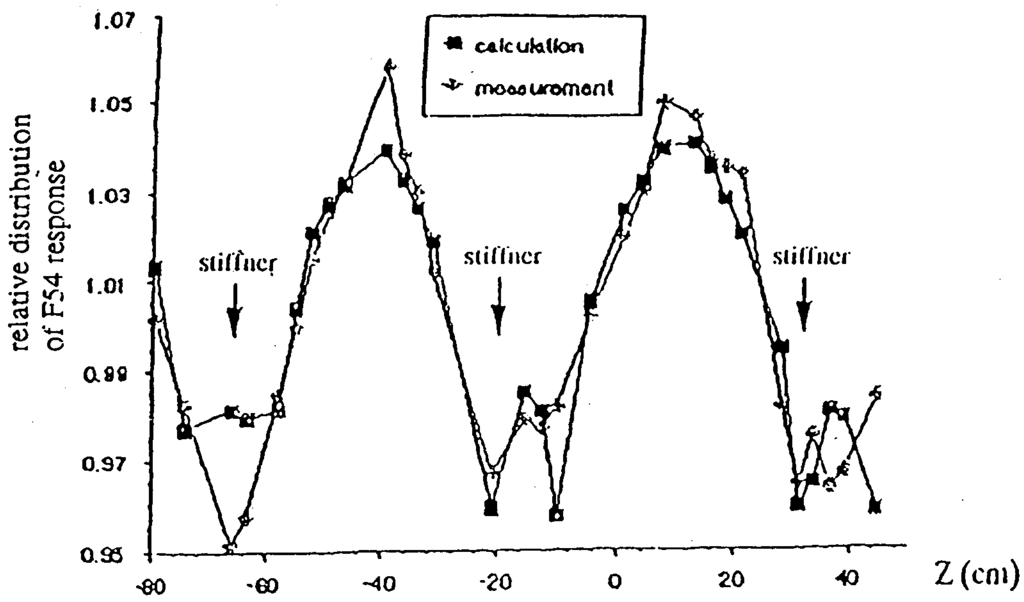
Uncertainties and adjustments

Being a matter of nuclear safety, the pressure vessel fluence calculations should be completed with as realistic as possible uncertainty analysis. At CEA the uncertainty calculations are carried out using two concepts:

- that of matrices of sensitivity to parameters uncertainty,
- and that of parameters uncertainty and their correlations in the variance-covariance matrix form.

The uncertainty calculations are carried out by CEA using 2-D deterministic neutron transport codes TWODANT [82] and SUSD [83,84].

Figure 6. Recreaction using DORT code of the effects of stiffners on the response of ^{54}Fe dosimeter



The object of the uncertainty analysis is:

- identification of most important uncertainty parameters (e.g., energy range in which cross-section data should be improved),
- adjustment of certain nuclear constants using benchmark data. It is at this point[80] that adjustment of the inelastic scattering cross-section of Fe in ENDF/B-IV data is proposed which leads to values close to those in ENDF/B-VI or use of fission spectra from ENDF/B-V rather than Cranberg is advised.

In the table below[85], uncertainty values in pressure vessel fast neutron flux (>1 MeV) obtained using ENDF/B-VI data are given.

Origin of uncertainty	Uncertainty (%)
fission spectrum	7.0
cross-section O	1.0
cross-section H	2.6
cross-section Fe, Ni, Cr	6.0
normalisation of neutron source	2.2
peripheral power gradient	4.1
water density	1.0
Monte Carlo statistics	1.8
pressure vessel radius (2 mm)	2.4
Total	**12.0**

Conclusions

The following conclusions can be drawn:

- at nominal power the effect of standardisation is remarkably apparent for different French power reactors (in terms of consistency of fluence calculations results);

- the metal damage parameter is the neutron flux at energies above 1 MeV – a non-measurable quantity. In dosimetry of surveillance samples the measured activity is translated to value of fast flux (>1 MeV) via knowledge of calculated neutron spectrum using TRIPOLI code. The same holds for DPAs;

- the lead factor is determined in DORT and TRIPOLI calculations.

Benchmark analysis and estimation of uncertainties show importance of Fe, Cr, and Ni cross-sections. The analysis of WINFRITH light water benchmark shows that there may exist a problem with these cross-sections or their anisotropy in 1 to 2 MeV energy range. The good knowledge of neutron fission spectra above 5 MeV is also important.

Tools available today allow to make reliable predictions of the level of flux and fluence in surveillance capsules and reactor vessel. However, these tools may still be improved and the estimation of uncertainties in calculations requires more precise variance-covariance matrix data for uncertain parameters.

Germany

In Germany, there are presently three ongoing activities in the field of neutron fluence calculations. For the power reactors the routine fluence calculations and measurements are normally performed by Siemens/KWU, Erlangen. If higher computational effort is necessary in generating the cross-section sets additional calculations are performed at IKE-Stuttgart. For the Russian type reactors the neutron fluence calculations are carried out by Siemens/KWU and FZR-Rossendorf. FZR-Rossendorf primarily uses the Monte Carlo method, otherwise the S_N method is favoured. The details of routine neutron fluence calculations as well as the evaluations of several fluence detectors are specified by the German standard DIN 25456.

Several years ago intercomparisons were carried out between dosimetry calculations and measurements at the material irradiation facilities of KFA-Jülich, GKSS-Geesthacht and HFR-Petten. Large specimens were irradiated and the neutron spectrum was adjusted by material filters to simulate the pressure vessel conditions of a power reactor. Special attention was devoted to deficiencies of the iron neutron cross-section which was adjusted using the results of deep penetration experiments in single material (iron). Burn-up of each fuel rod was considered in detail to provide a precise representation of the fission source in the outer core region which is of high importance in pressure vessel dosimetry.

In addition, Germany participated in the LWR-PV-SDIP program established by the US NRC in 1977 (and described in the Belgian and the US sections of the report) and profited from its results. However, the emergence of improved neutron transport codes and improved cross-section data justify a new computational benchmark exercise.

State-of-the-art in fluence evaluations at SIEMENS/KWU

Neutron spectra and neutron fluences are calculated using the S_N codes ANISN and DOT. Calculations using DORT and TORT codes are on the way as well. In routine calculations up to date the EURLIB4 cross-section library is used. A cross-section library based on the ENDF/B-VI data is also available.

The fluence evaluations are performed by using calculated neutron spectra at the fluence detector positions, the energy dependent detector cross-sections given in the IRDF90/2 file, and the irradiation history of the detectors. The irradiation history included the thermal power diagram of the reactor and the flux density variations caused by changes in the core loading schemes. Routine irradiation programs use Fe and Nb fluence detectors. The calculational procedure is described in the German Standard DIN 25456.

In order to reduce the uncertainties in the neutron spectra and fluence calculations careful attention is paid to modelling geometry and generating macroscopic cross-sections and source terms. As a result, the uncorrected and unadjusted ratios of calculated and measured fluences only slightly exceed 20 per cent.

The uncertainties in fluences evaluated via fluence detectors are determined using the Gauss law of error propagation which takes into account the correlations between the input data. The covariance matrices of the energy dependent fluence detector cross-sections are taken from the IRDF90/2 dosimetry file. The resulting standard deviations of the fluences evaluated via Fe and Nb fluence detectors are smaller than 10 per cent. Further details on methods used and results obtained can be found in References 87-90.

Three-dimensional neutron spectra and fluence calculations for the reactor pressure vessel and material surveillance capsules at IKE Stuttgart/Germany

Towards the end-of-life of a nuclear plant, when interesting possibilities of substantial life extension are studied, radiation induced metal degradation has to be determined and examined. The usually applied synthesis method for determining the neutron fluence in the pressure vessel from two- and one-dimensional transport calculations is not sufficient, especially if status of important welds has to be examined away from the midplane of the reactor core. Consequently, complex three-dimensional material configurations have to be considered for the surveillance capsules and for correlating the calculations with fluence measurements using activation and fission detectors placed mostly away from the core midplane.

For the oldest German nuclear power station at Obrigheim (PWR, 357 MWe) a detailed study was recently performed using the three-dimensional S_N program TORT with the newest nuclear data of ENDF/B-VI [91]. Since the important welds of the pressure vessel as well as most of the surveillance capsules were located in the lower half of the core, the octant between the azimuthal angles of 45 degrees and 90 degrees was simulated in full three-dimensional detail comprising the lower part of the active core region. The complex three-dimensional source distribution of fission neutron was derived from the power determined for the individual fuel rods as shown in Figure 7 for the reactor cycle 22 in (x,y) cell geometry. Additionally, axial form factors were used as given in Figure 8 for the radial corner element MO_4 of the cycles 19 to 22. The source strength was transformed into (r,θ,z) geometry which was finally applied in the neutron transport calculations with the S_N code TORT in S_8P_3 approximation.

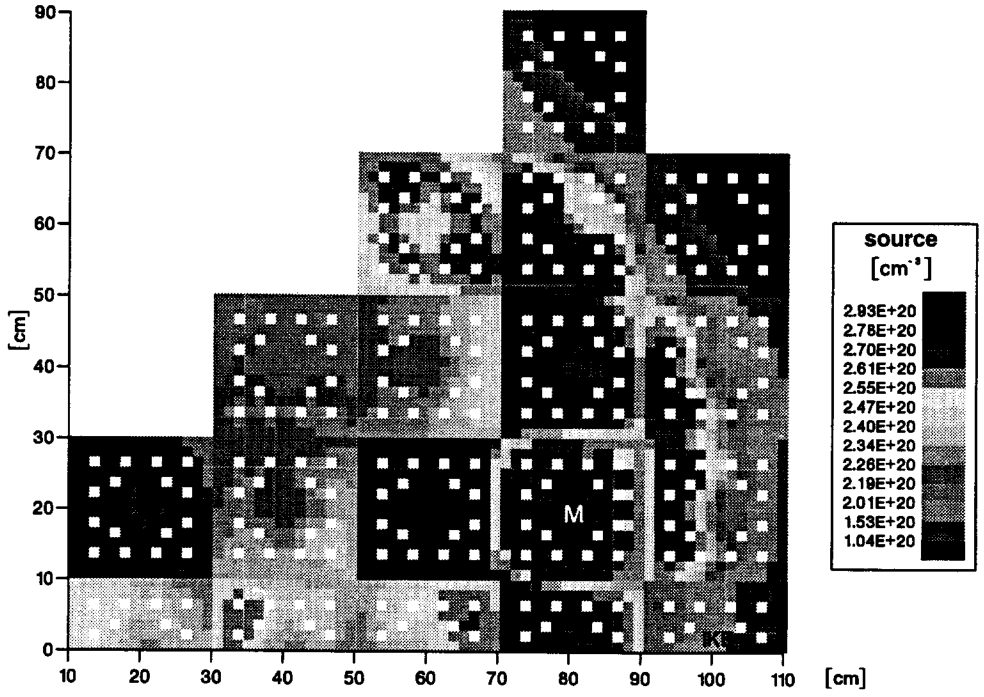


Figure 7. Axially averaged neutron source density in the pin cells of the core octant for cycle 22, MOX element M, control rods white

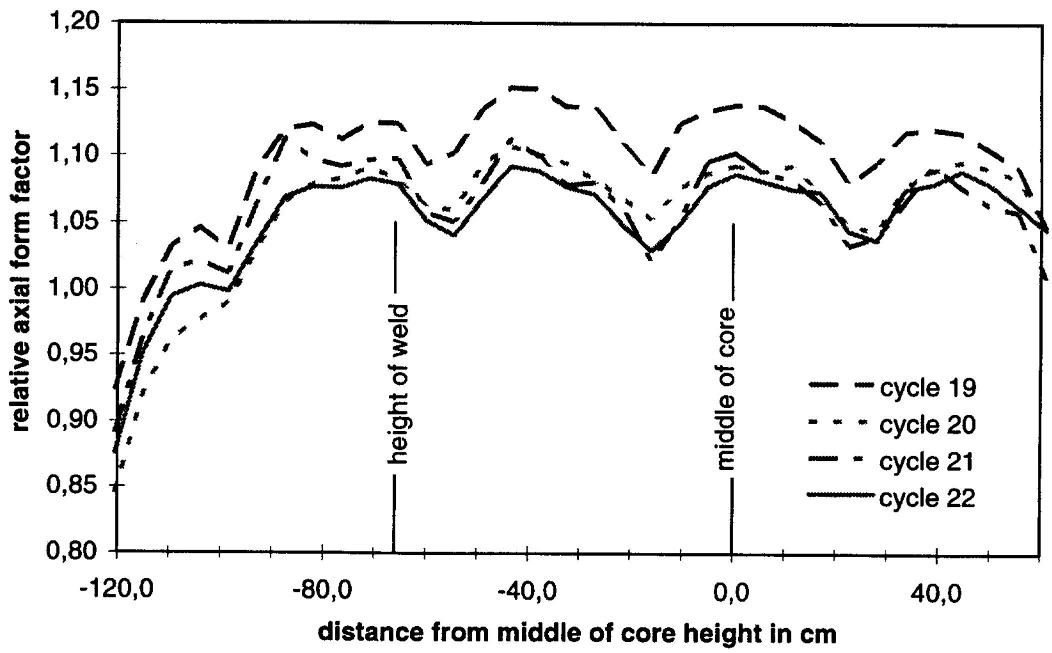


Figure 8. Relative form factors of axial power distribution in radial corner element MO_4 for cycle 19 to 22

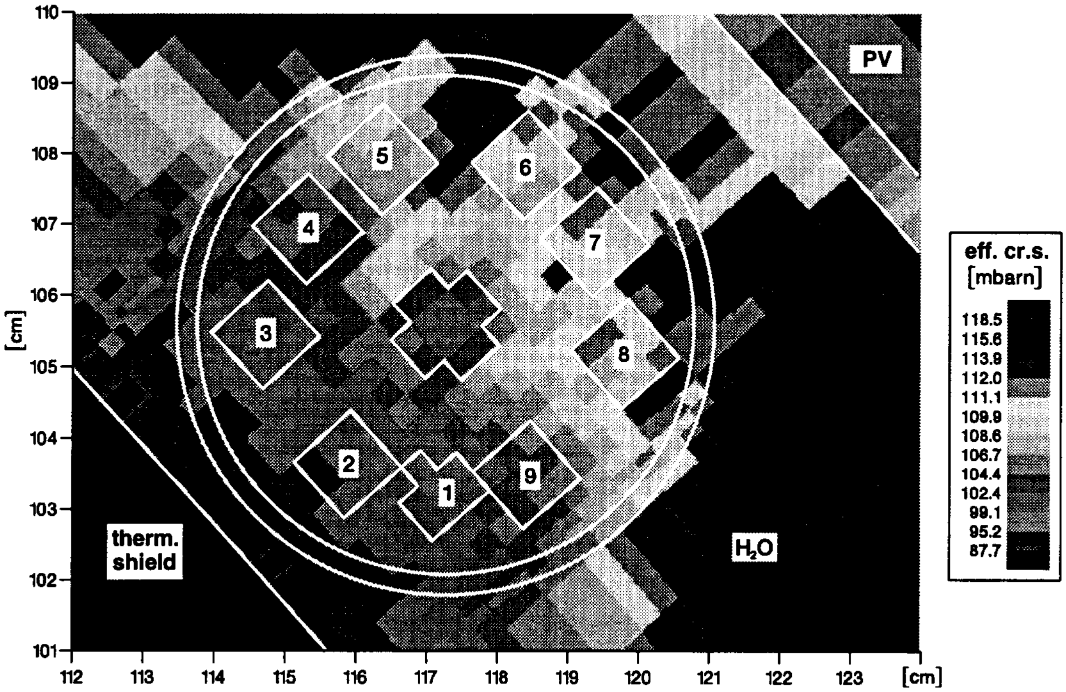


Figure 9. Horizontal distribution of the effective cross-section for $^{54}Fe(n,p)^{54}Mn$ in the surveillance capsule 12 with samples 1 to 9 for cycle 22

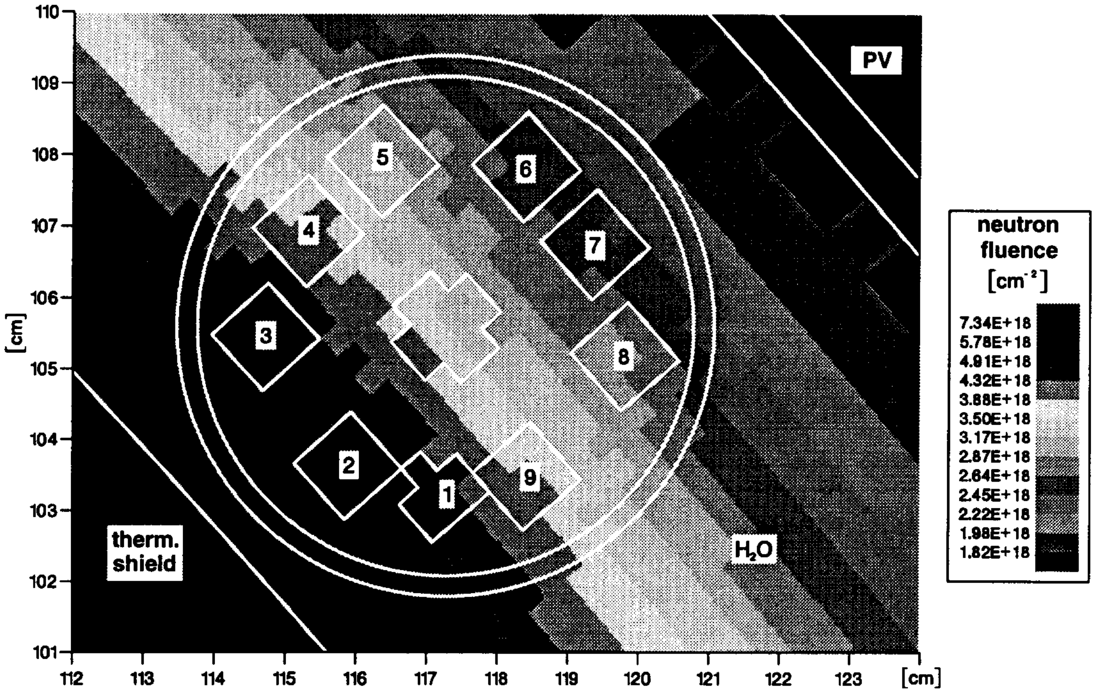


Figure 10. Horizontal distribution of the fast neutron fluence (E>1 MeV) in the surveillance capsule 12 with samples 1 to 9 for cycle 21 and 22

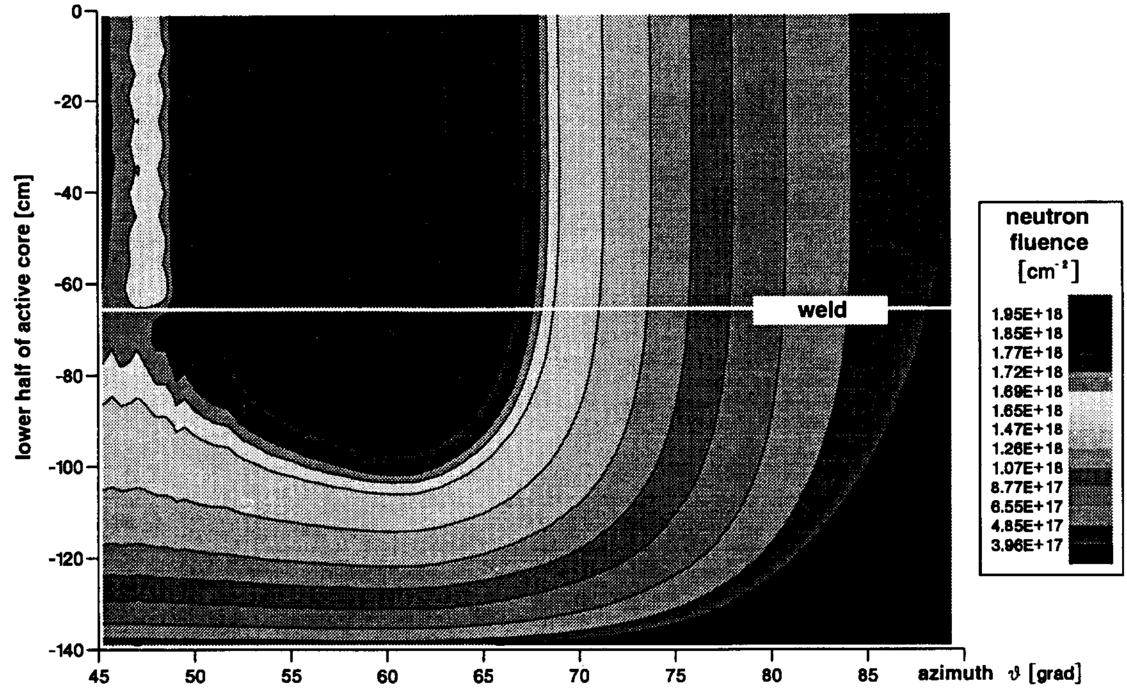


Figure 11. Azimuthal distribution of fast neutron fluence (E>1 MeV) at inner side of the pressure vessel (lower half with weld) for cycle 21 and 22

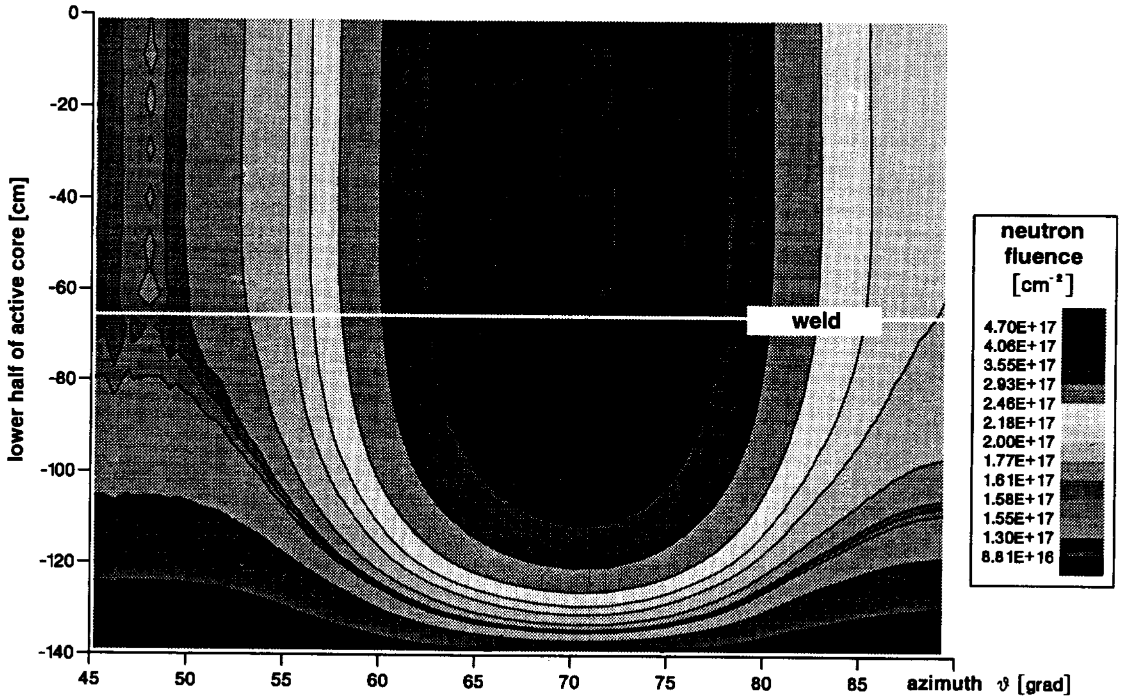


Figure 12. Distribution of fast neutron fluence (E>1 MeV) at inner side of the pressure vessel (lower half with weld) for cycle 23

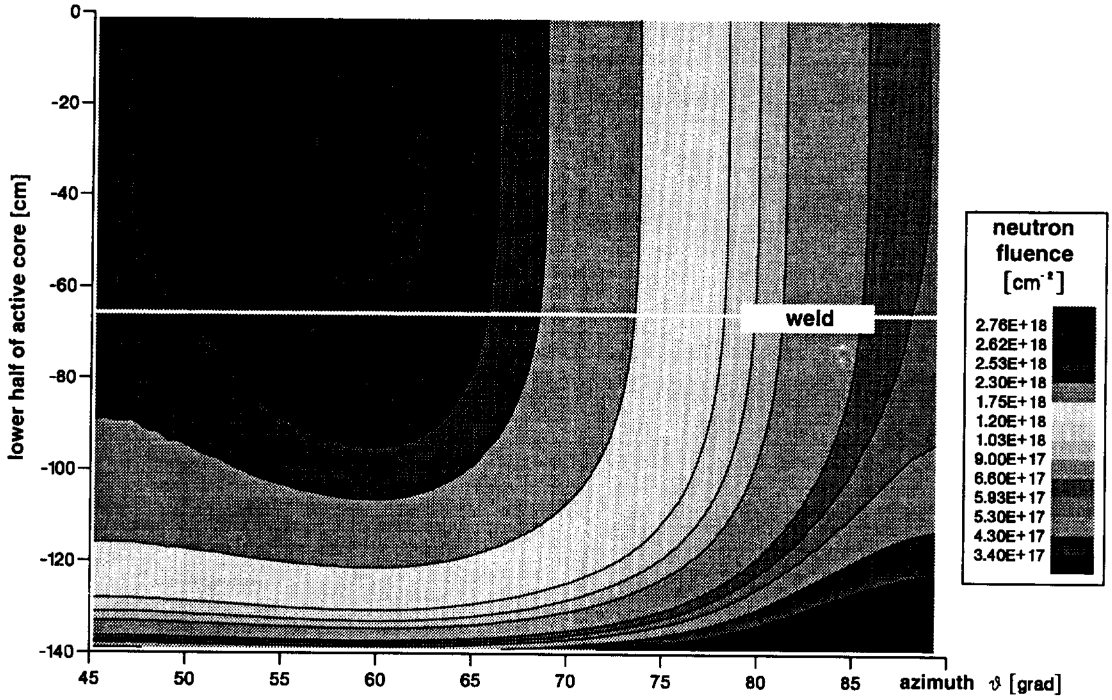


Figure 13. Azimuthal distribution of fast neutron fluence (E>0.1 MeV) at the inner side of the pressure vessel (lower half with weld) for cycle 22

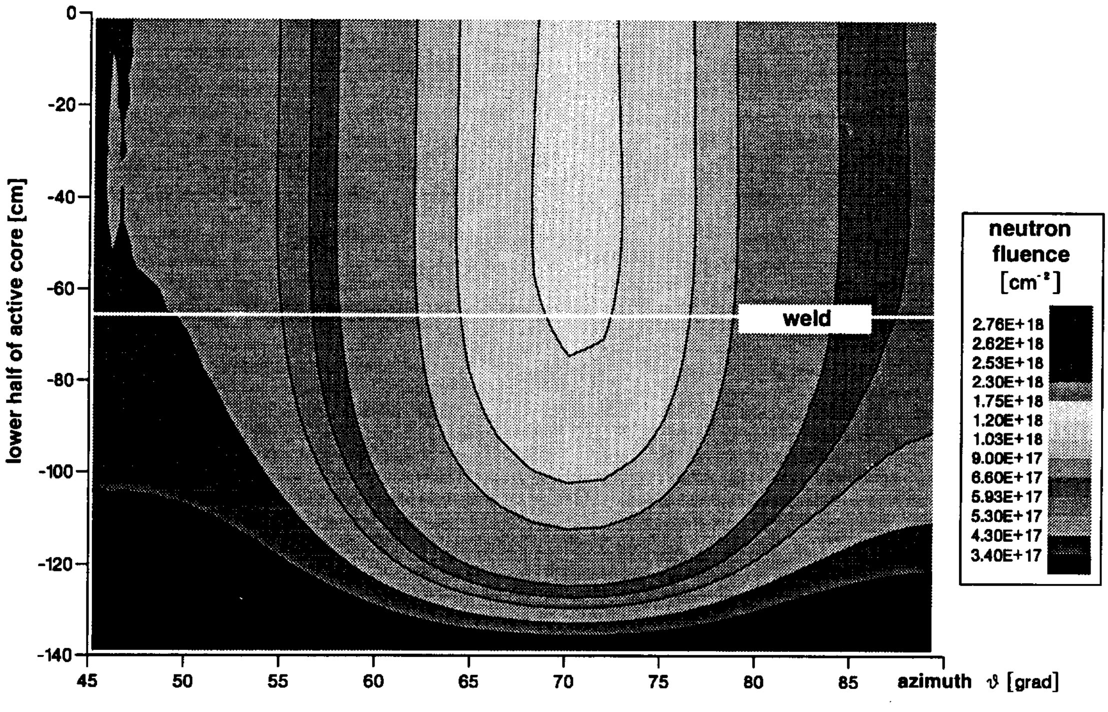


Figure 14. Distribution of fast neutron fluence (E>0.1 MeV) at the inner side of pressure vessel (lower half with weld) for cycle 23

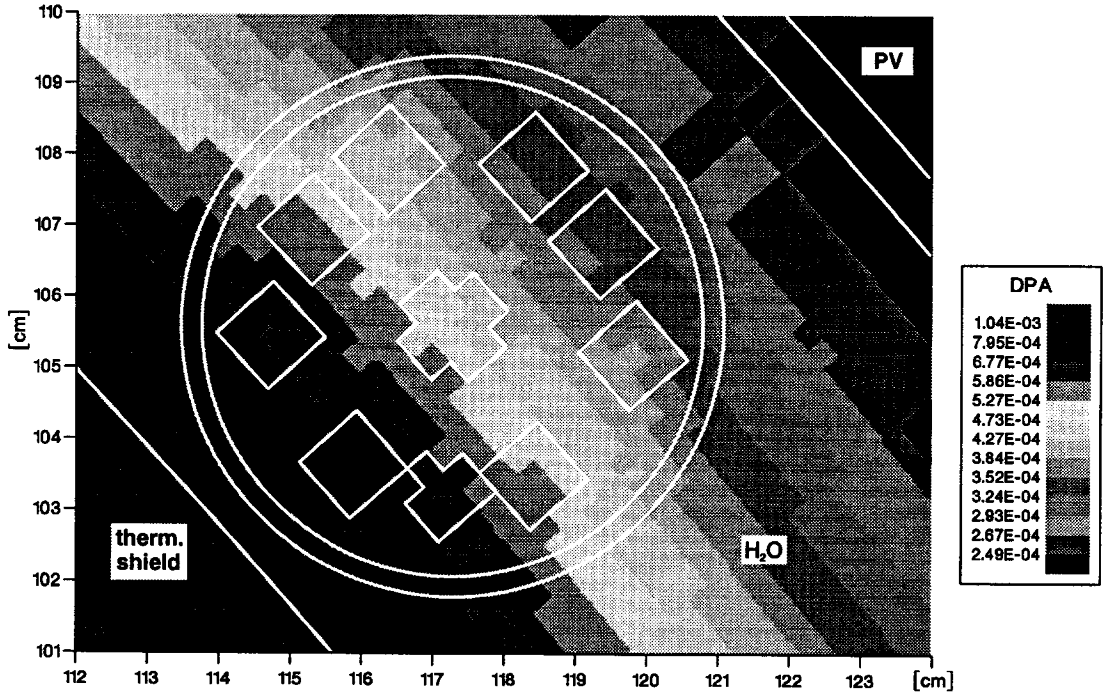


Figure 15. Horizontal distribution of DPA in the surveillance capsule 12 with samples 1 to 9 for cycle 23

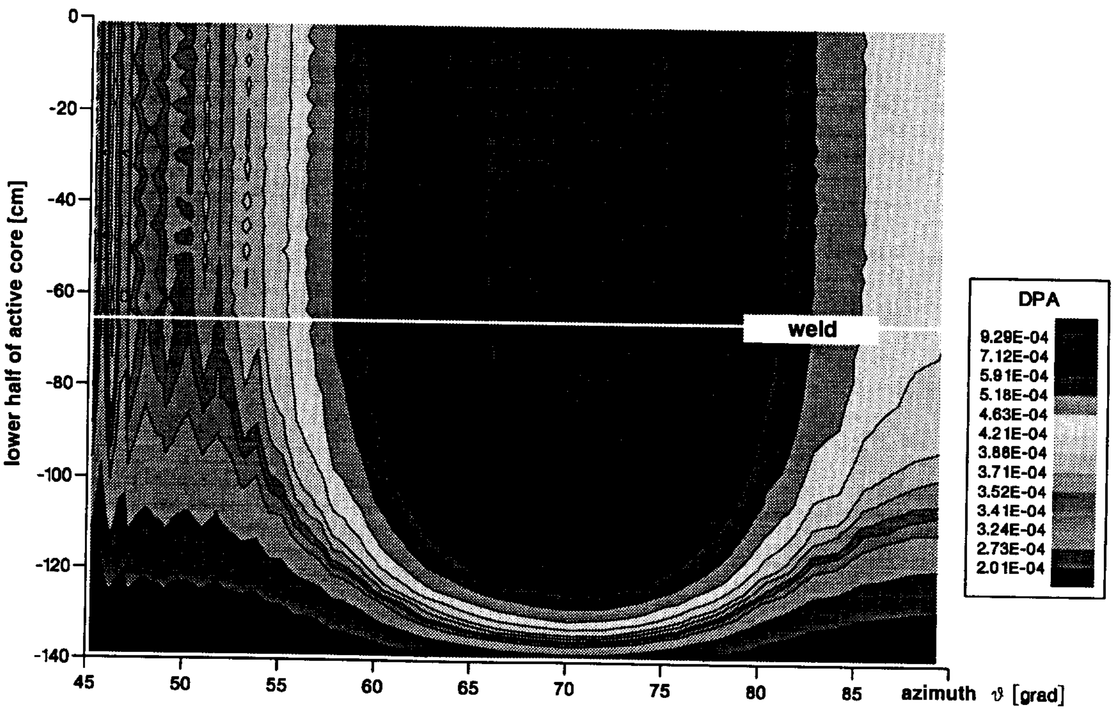


Figure 16. Azimuthal distribution of DPA at inner side of pressure vessel (lower half with weld) for cycle 23

Figures 9 and 10 show the results of the transport calculations for the surveillance capsule within the irradiation tube placed in the downcomer between the thermal shield and the pressure vessel. The horizontal cross-sectional view through the metal samples (No. 1 to No. 9) indicates in Figure 9 the energy averaged effective reaction cross-section of the detector reaction $^{54}Fe(n,p)^{54}Mn$ and in Figure 10 the fast neutron fluence (E>1 MeV) contributed by the reactor cycles 21 and 22.

The azimuthal distribution of the fast neutron fluence (E> 1 MeV) at the inner surface of the pressure vessel is compared in Figures 11 and 12 for different reactor cycles. Three-dimensional results are displayed around the vessel welds in the lower half of the active core region. The strong shift of the azimuthal fluence maximum from the angle of 60 degrees to 70 degrees is caused by addition of new dummy elements in the outer core region beginning with the reactor cycle 23. At the angle of 48 degrees the shadow of the surveillance capsules can be recognised clearly in the fast neutron fluence (E>1 MeV) of the pressure vessel. These details are lost by additional scattering below 1 MeV.

If the neutron fluence is taken down to E=0.1 MeV, a strong smoothing of the local fluence structure can be observed as shown in Figures 13 and 14. On the other hand, more local structure can be seen when the neutron fluence is weighted with cross-section for atomic displacements as given in Figures 15 and 16. The figures presented demonstrate the calculational effort needed for high precision fluence determination. A three-dimensional representation is required for the fission source in each reactor cycle. A similar effort is needed for calculating the details of neutron spectra and fluences in the complex arrangements of the surveillance capsules.

Japan

General description of reactor dosimetry in Japan

In Japan, neutron fluence (E>0.1 or 1 MeV) in PWR and BWR reactors is in general calculated using the 2-D S_N code DOT 3.5 with collapsed cross-sections generated from 100 neutron groups cross-section data of ENDF/B-IV, ENDF/B-VI, and JENDL-3 [92] libraries. The neutron fluence calculations are validated by comparison with dosimeter measurements. Several ways of refining calculations are considered and described below.

Improved results of fluence calculations in the pressure vessel surveillance of the BWR/5-1100 MWe plant are reported by treating the core void fraction precisely and by carefully modelling the core boundary [93]. The experimentally obtained fast neutron fluence evaluated using three element dosimeters showed agreement within few per cent with the calculated fluence in the considered test case.

A new approach using the 3-D S_N code TORT was applied to BWR/800 and 1000 MWe reactors [94]. The 1/8 of the reactor core was approximated with the r-θ-z model. In this model, the radius spans the core up to the pressure vessel surface. Azimuthally, the model covers an octant of the core. Vertically, effective fuel length is considered. Nineteen groups cross-sections were prepared from the JSSTDL library [95] based on the JENDL-3 data files. The ratio of calculated to measured fluence using Cu, Fe, and Ni detectors at the surveillance position turned out to be 1.7.

In order to improve the quality of fluence calculations the neutron transport, dosimeter, and DPA cross-sections as well as PKA spectra are evaluated by the Japanese Nuclear Data Committee at JAERI [96]. Continuous energy and multigroup Monte Carlo techniques are also being developed [97].

Dosimetry assessment in Material Test Reactors

The Japan Materials Testing Reactor (JMTR) at JAERI is used in materials research. Efforts have been undertaken to accurately determine the radiation fields in this reactor. The neutron fluence is evaluated using Fe and Co dosimeters contained in a surveillance capsule. The discrepancy between the measured and calculated fast neutron fluence for this reactor is at $\pm$20 per cent (1.65σ) and at $\pm$40 per cent (1.65σ) for thermal neutron fluence. The detailed neutron energy spectra were determined in the critical assembly experiments conducted on the JMTR reactor using the multifoil method [98].

Recently, to improve the accuracy of fluence prediction in thermal energy range the fluence monitor data are used. Namely, the neutron energy spectra are obtained by folding the two spectra obtained using the SRAC code [99] in thermal energy range and the ANISN code in the fast energy range. Smoothing is carried out using the experimental dosimeter data.

In addition to fluence evaluation, DPA and helium production rates are calculated using the TENJIN-2 code [100] based on the ASTM E693 standard and using the JENDL-3 and ENDF/B-VI nuclear data libraries.

Korea

In Korea, the pressure vessel dosimetry calculations and measurements are performed at KAERI. In the following, a succinct description of KAERI methodology (currently under development) employed and results obtained is presented on the example of Kori Unit 4 Westinghouse designed 2785 MWth PWR plant surveillance capsule analysis.

Kori Unit 4 surveillance capsule computational analysis

The surveillance capsule of Kori Unit 4 Westinghouse reactor is extracted at the end of cycle 8 which amounts to 7.05 effective full power years of irradiation time. The neutron fluence accumulated by the capsule over that period is determined experimentally and compared with the results of the following computational analysis.

The computational analysis of the surveillance capsule is based on the two-dimensional (r,θ) calculations, hence, it does not account for the axial configuration of the core and capsule as well as the axial power/fission source distributions.

The first step of the analysis consists of careful preparation of the two-dimensional reactor model including core, coolant, core baffle, barrel, neutron pad, pressure vessel, surveillance capsule, and biological shield. Uniform water density (i.e. temperature) is considered in the model and 87 x 92 meshes are used in the 1/8 reactor model for the radial and azimuthal directions.

Two-dimensional (r,θ) co-ordinate system with S_8 angular quadrature and P_3 Legendre scattering expansion is used in DOT 4.3 transport calculations.

BUGLE-93 cross-section library in 47 neutron energy groups is used. This library is based on the ENDF/B-VI data file intended for use in LWR shielding and pressure vessel dosimetry applications. BUGLE-93 dosimeter file dosimeter cross-sections are used to calculate the activities of the capsule specimens.

In modelling the fission source term core power distributions applicable to each operating cycle of the considered reactor are used. For Kori Unit 4, 20 to 25 sets of pin-power distributions are taken into account according to burn-up increase in each cycle – MEDIUM nuclear design code is used for this purpose. The core source distribution is then obtained by weighting the pin-power distribution with burn-up over the operating period. The Watt fission spectrum is used for ^{235}U fissions and about 20 per cent of ^{239}Pu fission contribution is assumed in evaluating the average number of fission neutrons per fission (n).

The calculated fast (>1 MeV) neutron fluence at the centre of the surveillance capsule is 1.8972E+11 n/cm^2. The saturation factors for the dosimeters were calculated based on the daily operating power levels. In table below, fluences and ratios of measured and calculated values are given for four detectors considered.

Detector	**Measured fluence above 1 MeV**	**Calculated fluence above 1 MeV**	**Ratio M/C*100**
$^{54}Fe(n,p)^{54}Mn$	1.5057E+11	1.89724E+11	79.37
$^{63}Cu(n,a)^{60}Co$	2.4458E+11	same	128.91
$^{58}Ni(n,p)^{58}Co$	1.4203E+11	same	74.86
$^{59}Co(n,g)^{60}Co$	8.5927E+10	same	45.29

The Kori 4 calculated fluences were compared with fluences measured at the surveillance capsule position. The fluence above 1 MeV was measured using ^{63}Cu, ^{54}Fe, ^{58}Ni, and ^{59}Co dosimeters. The comparison of measured and calculated fast neutron fluence gave 15 per cent uncertainty at the capsule position with the bias factor (i.e. the ratio of calculated to measured fluence) of 1.23. It is believed that these rather large uncertainty and bias factors have their origins in deficiency of the fission dosimeters (^{238}U and ^{237}Np).

In the future, the use of the LEPRICON methodology is foreseen in adjustment of calculated fluences to measurements and in performing the uncertainty analyses.

Netherlands

In the text that follows, the current state-of-the-art in calculating the radiation dose to reactor components at ECN Petten in the Netherlands is presented.

Calculational procedures

In order to calculate the radiation dose to reactor components two approaches are used at ECN Petten. Both approaches are based on Monte Carlo methods, as a detailed model of the reactor geometry is usually required. The codes used are MCNP4A [101] and KENO-Va.

Monte Carlo analysis of the neutronic behaviour of a LWR in combination with continuous-energy cross-section data are an attractive tool to provide a detailed description of a static LWR core. Very few limitations exist in the area of geometric modelling of a problem. Furthermore, details of the original nuclear data evaluation can be retained in the cross-section library and self-shielding in the resolved resonance range is explicitly taken into account.

The Monte Carlo code MCNP4A is a very widely used tool for performing neutron, photon, and electron transport. It allows detailed geometric modelling and use of cross-section data in complex formats. As an example, a coupled representation of the energy-angle distributions of scattering neutrons (a possibility in the ENDF/B-VI format) can be used in MCNP4A.

Cross-section data

In general, the format of the cross-section libraries for MCNP4A is defined in such a way that as much detail as possible can be retained from the original evaluated nuclear data file. However, the availability of the high quality nuclear data for radiation transport codes in general and for MCNP4A in particular is rather limited.

At ECN Petten, MCNP4A is used in conjunction with data from the EJ2-MCNPlib library [102]. This high quality cross-section library contains data from the European JEF-2.2, EFF-2.4, and EFF-3.0 evaluations. The JEF-2.2 evaluation contains data for nuclides needed in reactivity, criticality, and general purpose calculations (data for 313 nuclides are included). The EFF-2.4 and EFF-3.0 evaluations are intended for use in fusion reactor shielding calculations. Therefore, these evaluations contain data for fewer nuclides (only 78 nuclides) – particularly relevant to these calculations (e.g. structural materials and neutron multipliers).

Data from several sources are used for dosimetry and DPA cross-sections. The standard libraries are the ECNAF-96 library for dosimetry data (based on EAF-4 it contains data of more than 11 000 activation cross-sections) and the ECNDPA-96 library for DPA data (based on JEFF-2.2 contains data for 55 nuclides). It should be stressed that both libraries utilise continuous energy cross-section data.

In some cases, group cross-section data from the IRDF-90.2 [103] library are used.

Calculational method

Usually, radiation dose calculations are performed for a representative average reactor core. The core is modelled as a homogeneous region and k_{eff} calculations are performed – the expected k_{eff} value is 1.0. If k_{eff} is not 1.0 then the core model is adjusted. Care is taken to establish an agreement with available experimental information. The neutron spectrum from the k_{eff} calculations is used to determine the source in the fixed source calculations of neutron (and gamma) transport in the reactor.

Fission product gamma rays are not included in the coupled (n,γ) MCNP4A calculations. Therefore, in relevant cases a separate fission product gamma transport calculations are performed.

The procedure outlined above is used in pressure vessel dosimetry calculations of the Dodewaard plant (55 MW BWR) and of the Petten high flux reactor.

At the Technical University Delft (TUD) the procedure developed by the ECN Petten (using MCNP4A in combination with the EJ2-MCNPlib library) is used for dosimetry calculations of the HOR research reactor at the TUD site. The agreement between the resulting calculated and measured ex-vessel neutron fluence above 1 MeV is better than 15 per cent and usually even better than 10 per cent.

Sweden

In Sweden, basic data for fluence calculations are provided by the power plants and consists of blueprints, material data, and reactor operating statistics. Load maps, power distributions, burn-up maps for different cycles are also provided by the utilities.

The following methodology is employed in performing fluence calculations. The reactor load patterns are divided into a smaller number of typical power distributions according to corresponding power maps. Special attention is paid to the relative power in the two outer fuel rows. The cell-assembly code CASMO is then used together with a 70 energy groups cross-section library to generate a 23 coarse group library for each material (including the fuel) as a function of burn-up. The code DORT is then used to obtain the 2-D geometrical layout of the reactor. The mesh sizes are chosen from 0.5 to 1 cm interval. In core centre where flux gradients are low larger mesh sizes can be selected. For the fluence calculations the S_8 angular quadrature set is typically chosen. The convergence criteria are always set tight. By transferring fission cross-sections to DORT it is possible to adjust the power distribution according to the proposed power map by selecting fuel cross-sections for fuel assemblies from the burn-up dependent library. The DORT calculations yield 23 coarse group flux distributions representative for typical cores running at full power. For each cycle, fluxes from the closest radially power distribution are chosen for weighting with the relative power in the outer two fuel rows. The time of each cycle is divided into a few periods of constant power if a reactor was run at reduced power level or with a coast down period at the end of the cycle (as is common in Sweden). The next step consists of calculating the reaction rates in the surveillance capsules irradiated during corresponding time intervals using the 23 coarse group fluxes. This obtained reaction rates are then compared with measured DPS (disintigrations per second). In the end, neutron fluence (i.e. time integral of flux above 1 MeV) is a parameter used by material physicists to estimate neutron damage effects to reactor vessel. In the following table some typical DPS from calculations and measurements are presented.

Detector	**Measured**	**% Difference**	**Calc.**	**Calc./Meas.**
^{63}Cu	1.78E04	10	1.78E04	0.998
^{59}Co	1.13E10	25	1.17E10	1.04
^{54}Fe	9.63E05	2	9.00E05	0.93
^{58}Ni	3.34E06	4	3.94E06	1.18
^{237}Np	4.07E06	-	3.62E06	0.89
^{238}U	7.60E05	-	5.12E05	0.67

The agreement between the calculated and measured DPS is not always as good as shown in the table but the deviations can often be explained by uncertainties in measurements or positioning of detectors. Nevertheless, the agreement between the calculations and measurements does not necessarily mean that the fluence is indeed accurately calculated since the previous cycles are to some degree “forgotten” by the detectors and they are as important contributors to neutron fluence as the current cycles.

Switzerland

The details of the state-of-the-art in the reactor pressure vessel calculations in Switzerland can be found in an article written by F. Holzgrewe *et al.* entitled “Calculation and Benchmarking of an Azimuthal Pressure Vessel Neutron Fluence Distribution Using the Boxer Code and Scrapping Experiments” [104]. The following is a summary of the information presented in this article.

The computational method discussed by F. Holzgrewe and his co-authors is focused on as accurate as possible determination of the high energy neutron fluence at the inner surface of an RPV. The fast fluence (E>1 MeV) calculations are performed at the Paul Scherrer Institut (PSI) in Switzerland using the PSI's LWR BOXER code in the fixed source mode for the average burn-up of four different cycles. The results of the calculations are the 2-D X-Y geometry average neutron flux distributions. These flux maps are then used to obtain neutron fluxes at the RPV inner surface via linear interpolation. Fast fluences are finally calculated by integrating the neutron flux distributions over time. The computed fast fluences are compared against the fast fluences determined experimentally from the RPV scraping test samples. A maximum difference of 15 per cent is reported [104] between BOXER and experimental results. Moreover, the BOXER results are "benchmarked" against a TWODANT S_N code system. The maximum reported deviations in the 2-D fast flux distributions between the BOXER and TWODANT are less than 3 per cent.

Key elements of the PSI's RPV fluence calculations methodology

The geometrical modelling of the reactor is performed in 2-D X-Y geometry consistently with BOXER input requirements. Extreme care is taken in developing this model. The mesh size is chosen so that reactor components which can be modelled exactly are in fact modelled exactly. This minimises the number of mixed materials that have to be defined. The exact modelling cannot be accomplished for cylindrical elements, thermal shield and the reactor RPV and material mixtures have to be defined for these components. Hence the meshes containing mixed materials are modelled as homogeneous, i.e. any effect due to a lack of homogeneity is neglected. In order to further minimise the number of different materials the core model uses only one material (UO_2) regardless of burn-up. This is possible because the cross-sections of uranium and plutonium are almost the same above 1 MeV and hence plutonium is replaced with uranium when describing a core for an average burn-up of a cycle. The meshes outside the RPV are replaced by steel to eliminate the need for defining yet another material (air). This is a valid approximation since there is virtually no backscattering of neutrons from this introduced steel outside of the RPV to positions within the RPV wall.

The fixed source distributions for BOXER transport calculations are obtained from cell burn-up calculations which supply the nuclide number densities and reaction rates as a function of burn-up. The sources can be calculated from burn-ups of fuel elements at the beginning and the end of each cycle considered. All meshes (in the reactor geometrical model) corresponding to a given fuel element are assigned an identical source value which is the average source of this element. The meshes describing fuel elements at the core edge are treated differently; each mesh is assigned its own source value to account for large deviations in the source values of single meshes. The average source values are determined using simple analytical expressions [104]. The groupwise source values are then obtained in BOXER by multiplying the fission spectrum with the meshwise average sources.

The fission spectrum is obtained (using the ETOBOX code) for the average burn-up by taking into account all important uranium and plutonium isotopes: ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu (the production rates of these nuclides are obtained from a burn-up calculation using BOXER – average cycle burn-up is considered).

The energy group structure used in calculations is that of 45 energy groups with 40 groups spanning the energy range above 0.1 MeV. This assures sufficient accuracy for neutron backscattering in metal. The energy range below 0.1 MeV (of no interest in this calculational method) is divided into 4 energy groups without any loss of accuracy in calculating the flux above 1 MeV (due to using fixed source distribution and neglecting the upscattering). The chosen group structure differs from that of BUGLE and SAILOR cross-section libraries using 26 groups structure above 0.1 MeV and does not produce significant differences in the flux and fluence distributions. The cross-sections in the adopted 45 energy groups structure are generated using the ETOBOX code from the JEFF-1 data file. The important iron cross-section is taken from the ENDF/B-IV data file (no more recent data were available).

The calculations are conducted taking advantage of the symmetry planes in the reactor, no buckling option is used, the convergence criterion is set to 0.0001, and the Legendre order of scattering is set to P_1. A sensitivity study considering P_1, P_2, and P_3 Legendre orders of scattering had shown that the maximum deviations in the results were less than 1 per cent. The BOXER calculations give X-Y flux distributions in 45 energy groups. The flux above 1 MeV given in 36 energy groups is then summed up meshwise. The flux distribution at the inner surface of the RPV is obtained by linear interpolation between the appropriate meshes. The calculated fluence is then compared with fluence obtained from the scraping experiments.

In the scraping tests, small steel samples are taken from the inner plating of the RPV. The inner plating of the RPV contains niobium, nickel, and iron. These elements when irradiated produce the long-lived radioactive nuclides ^{93m}Nb, ^{58}Co, and ^{54}Mn which decay emitting gamma or X-rays which can be measured by appropriate detectors and the activities of the samples determined. The fast neutron fluence can then be calculated from the measured activities (at the PSI the Deutsche Industrienorm [105-109] is used for this calculation). In general, the RPV is irradiated by various neutron fluxes at different time periods which has to be properly accounted for when calculating fast neutron fluence from the scraping samples activities. For this purpose, the thermal reactor power and the corresponding time periods have to be provided by the utility. In the scraping test the half life of the monitored nuclides (^{93m}Nb, ^{58}Co, and ^{54}Mn) has to be considered. The small half-lives of ^{58}Co (71 days) and ^{54}Mn (312 days) imply that these nuclides "see" the neutron flux during the last 6 and 25 months of irradiation respectively. On the contrary, the ^{93m}Nb half-life is 16.1 years and hence the results for this nuclide are expected to be the most accurate for long RPV irradiation times. Finally, the scraping test is subject to errors: the statistical error E_{stat} and the systematical error E_{sys}. The statistical error consists of errors in the detector, activity, and nuclide weight measurements and is of the order of 8 per cent. The systematic error consists of error in determination of the spectral neutron flux (used to calculate the effective cross-sections of the detectors) and is of the order of 6 per cent. The total error (in the square norm) of the scraping test is therefore of the order of 10 per cent. Comparison of BOXER and scraping test results shows deviations that stay within this range. Only 20 per cent of the compared values lay outside of this range. The highest deviation is nearly 15 per cent.

The BOXER results are "benchmarked" against the TWODANT discrete-ordinates code system using the same group structure and cross-section library. Deviations are found to be less than 3 per cent. The reported results are very good, considering the factor of 1000 neutron attenuation between the reactor core and the reactor pressure vessel.

United Kingdom

In the UK "Displacements Per Atom" (DPA) as defined in ASTM E693 is used as the dose parameter. DPA cannot be measured directly and theoretical methods must be used in order to predict DPA rates from knowledge of neutron spectra.

The aim of Magnox reactor RPV dosimetry program is to be able to predict, for each station, the fast neutron dose in terms of DPA at one degree intervals around the inner surface of the pressure vessel and for each surveillance canister removed. In both cases, the best estimate and 1σ uncertainty are required. This has been achieved by establishing detailed neutron transport models for each reactor which include representations of the pressure vessel and surveillance locations. These models are highly station specific and have been validated for each station to establish uncertainties.

This aim has now been completed for the steel vessel Magnox stations, i.e. for each station MCBEND radiation transport models have been developed and validated against flux measurements. It is judged that the calculations are subject to different systematic uncertainties in the top-, side- and sub-core regions and so far for each power plant three values of C/E are quoted. These data are shown in the table below before any adjustment to the measured data. When predictions are made for power plant, the models are adjusted by the C/Es in each region of the model.

Each C/E is usually based on several measurements at different times, generally nickel and iron activation components and swarf removed from the engineering structure as well as material introduced deliberately for activation. More details are included in Reference 109. It should be noted that the design of each power plant is broadly the same and relies on the same basic graphite and steel nuclear data. However, the engineering designs are all different which makes them subject to different uncertainties for plant specific items, e.g. power data, geometric modelling approximations, etc.

All calculations have been made using the Monte Carlo code MCBEND, version 6B. The models we create are full 3-D representations using combinatorial geometry of the reactor pressure vessels with an approximate representations beyond this to the biological shield. Essentially, point nuclear data from the UKNDL library, which is supplied as part of the MCBEND code package, was used. Later versions of the code now supply JEFF 2.2 and comparisons have shown no significant difference for our application. Uncertainty allowances are typically:

a)	Power profiles	10%
b)	Source representation	5%
c)	Nuclear data (derived from benchmarks)	10-20% (mainly because we predict DPA)
d)	Modelling uncertainties	10%
e)	Stochastic uncertainties	<5%

Uncertainties a), b), and d) are reduced by adjustment to measurement. This is possible because Magnox reactors are continuously refuelled and power profiles and reactor geometry (e.g. density) do not change significantly during operation.

Table 7. MCBEND fast flux calculations for Magnox plant (before adjustment to measurement)

Station	**Calculation/Measurement**		
	Top-core	**Side-core**	**Sub-core**
Trawsfynydd	1.03	0.93	0.99
Sizewell	1.17	0.93	1.35
Dungeness	1.06	1.03	1.00
Hinkley	0.99	0.85	0.96
Bradwell	1.19	0.87	0.98

There is one PWR operating in the UK, that at Sizewell B, and data for this reactor will become known after the first planned outage in mid-1996. Calculations of the fluences at the surveillance capsules and within the vessel wall were carried out in the design stage, again using the MCBEND code. This method has been validated for calculations in PWRs by AEA Technology, who have also applied it to predict life-time fluences for reactors operating in the US.

United States of America

The state-of-the-art in dosimetry computations for reactor components is best summarised in the Draft Regulatory Guide DG-1025 titled “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” dated September 1993 and available from the US Nuclear Regulatory Commission (NRC). The fact that this text is still considered a draft document and is actively undergoing revision is a testament to the diverse views within the US reactor community on how best to combine dosimetry calculations and measurements. Work is underway within the US nuclear standards groups (ASTM and ANS) to produce standard guides on the computational methods for light water reactor calculations and on the use of benchmarks in testing reactor dosimetry methodologies. The following sections provide more specific information on the US state-of-the-art.

Comparisons of calculations and measurements

The comparison of calculations and measurements is usually done in benchmark fields. A draft ASTM standard “Guide for Benchmark Testing of Light Water Reactor Calculations,” E706-IIE2 discusses the state-of-the-art in this area. Benchmark fields take the form of standard fields, such as the ^{252}Cf spontaneous fission and the ^{235}U thermal fission fields, reference fields such as the Materials Dosimetry Reference Facility (MDRF) [110], and controlled environments or engineering benchmarks. One dedicated program effort to provide engineering benchmarks whose radiation environments closely resemble those found outside the core of an operating reactor was the NRC’s Light-Water-Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) [111]. This program has resulted in three benchmark configurations, VENUS [112], PCA/PSF [113], and NESDIP [114]. Taken together these benchmarks provide coverage for reactor locations from the fuel region to the vessel cavity. The VENUS facility was set up to measure spatial fluence distributions and neutron spectra near the fuel region and core barrel/thermal shield region. The PCA/PSF measurements look at surveillance capsule effects and the fluence fall-off within the vessel itself. The NESDIP measurements overlap the PCA/PSF measurements and extend into the cavity behind the vessel. As part of their qualification as benchmark fields, these neutron environments were well characterised both experimentally and calculationally. The difference between the measurements and calculations for exposure parameters in these engineering benchmark fields were reconciled to within 5 per cent to 10 per cent (1σ).

Work was initiated at the Arkansas Power and Light Reactor ANO-1 to use ex-vessel cavity dosimetry as a supplement or replacement for surveillance capsule dosimetry [115]. This work led to special cavity dosimetry benchmarking at the H.B. Robinson nuclear power reactor [116,117]. In general, evaluations of surveillance capsule dosimetry and ex-vessel dosimetry, when performed in a systematic fashion, has shown excellent self-consistency among plants of the same type [118]. This work indicates that the changes in neutron source with changes in fuel loading are being correctly handled, and that calculational bias is most probably due to systematic and not random effects.

Computational path used

All US approaches, endorsed by the DG-1025 regulatory guide, require that the calculational methodology should be qualified by both 1) comparisons to measurements and calculational benchmarks and 2) an analytic uncertainty analysis. There are several different ways to use measurements in conjunction with calculations to determine the best-estimate fluence. The different approaches vary with the confidence assigned to the measurements or calculation. The calculational path varies with the methodology.

The first method is to use the measurements as a test of the calculational results. The calculations would be considered adequate if they reproduced the measurements within a specific tolerance. The calculational uncertainty is ascribed to the result. This method, while the simplest method of combining measurements and engineering benchmark data, does not produce a result with the smallest uncertainty.

The second method is to use the plant specific measurements to re-normalise the calculations. This method produces the best result at actual dosimetry measurement locations and at locations suitably close to the measurements locations. The plant specific measurements reflect unknown errors in geometry parameters used in the calculations of fluence that cannot be benchmarked in any other way. The translation of the results to locations away from the measurement points can be guided by results from the engineering benchmark comparisons.

The most sophisticated method for fluence determination is to include both the calculated results, calculational uncertainty and covariance estimates, dosimetry measurements, and uncertainty and covariance data for the measurements through a least square procedure. One common way to combine these results is to use the LEPRICON code [115,121]. This method makes the maximum use of available information, but it can be very time-consuming to perform this analysis.

In all calculations, the determination of the fixed neutron source term for the pressure vessel calculations must entail specification of the temporal, spatial, and energy dependence of the source term together with the absolute normalisation. The spatial dependence of the source should be from core-follow calculations or measured data. Core-follow calculations should be performed with a three-dimensional coarse mesh simulator code and provide the relative power over large rectangular nodes. Plant power diagnostics provide similar power distribution data from in-core instrumentation. The horizontal core geometry may be described by an (r,θ) representation in the nominal plane. However a θ mesh of 40-80 angular intervals must be applied and the (r,θ) representation should produce the true physical assembly area to within 0.5 per cent and pin-wise source gradients to within 10 per cent.

The transport of neutrons from the core to locations of interest is generally determined with a two-dimensional discrete ordinates transport code. An S_8 symmetric angular quadrature should be a minimum for determining the fluence at the vessel, however, a higher order quadrature may be needed for cavity fluence calculations. The radial mesh in the core region should be ~2 intervals per inch for peripheral assemblies. In ex-core regions, the spatial mesh should be sufficient to ensure that the flux in any energy group changes by less than a factor of two between the adjacent energy-intervals, consists of at least 3 intervals per inch of water, and consists of about 1.5 intervals per inch in steel. A point-wise convergence criteria of <0.001 should be applied. A weighted-difference model should be applied to avoid negative fluxes and to improved convergence. The adequacy of the spatial mesh and the angular quadrature should be demonstrated by tightening the numerics until the resulting changes in the transport results are negligible.

Since 3-D discrete ordinates codes are now commonly available, there have been several efforts to compare the results and to determine if there is an advantage to the 3-D modelling [123-125]. Three-dimensional calculations should be considered where strong axial and azimuthal heterogeneities exist. When 3-D calculations are not performed, a 3-D fluence representation may be constructed by synthesising calculations of lower dimensions.

Estimated uncertainties

The DG-1025 draft regulatory guide requires that the vessel fluence calculational uncertainty must be demonstrated to be <20 per cent (1σ) for RTPTS determination. If the benchmark comparisons indicate an uncertainty greater than 20 per cent, the calculational model must be adjusted or a bias applied to bring the agreement within this range.

The overall vessel fluence calculation uncertainty in US reactors must be determined by a combination of 1) analytic uncertainty and 2) the uncertainty estimate based on comparisons to benchmarks. The analytic uncertainty analysis must include uncertainty components from the nuclear data, reactor geometry/position data, isotopic composition of materials, description of neutron source terms, and modelling/methods errors. The effect of significant component uncertainties are to be determined by a sensitivity analysis. A typical sensitivity analysis might yield a 10-15 per cent decrease in the vessel >1 MeV fluence for every centimetre increase in vessel inner radius. The estimated uncertainties from these sources are to be combined to determine the expected total fluence uncertainty. The independent random components are combined in a statistical (root mean square) fashion and systematic components are combined algebraically. Often the uncertainty analysis from engineering benchmarks analyses [115] are used as initial uncertainty estimates. Typical calculational uncertainty estimates are 15-20 per cent (1σ) at the inside of the reactor vessel in the beltline area and may be as large as 30 per cent in the cavity. When the engineering benchmark results are used to eliminate systematic bias terms, the resulting uncertainty is typically 10-15 per cent.

A good summary of the US position on the assessment of uncertainty in reactor vessel fluence calculations can be found in Reference 119. Additional insights can be found in Reference 120. The best US efforts to quantify the calculational bias in PWR geometries have been made by Maerker and others at ORNL using LEPRICON code [115,121].

Cross-sections and dosimetry data

The US community generally uses dosimetry data from sources specified in ASTM standard guide E1018-95. The nuclear constants for dosimetry applications (fission yields, isotopic abundances, gamma branching ratios, half-life, atomic weights, Q-values) are generally traceable to data libraries maintained by the National Nuclear Data Center (NNDC) at Brookhaven National Laboratory (BNL) and consistent with other IAEA-supported distribution channels for recommended nuclear data. Dosimetry cross-sections are generally consistent with the IRDF-90 (revision 2) cross-sections [126].

The DG-1025 NRC draft regulatory guide recommends that the latest version of the Evaluated Nuclear Data File (ENDF/B) data be used for the transport cross-sections. Cross-section sets based on earlier or equivalent data sets that have been thoroughly benchmarked are also acceptable.

The ENDF/B cross-sections are approved and maintained by the Cross-Section Evaluation Working Group (CSEWG). The current version is ENDF/B-VI. Due to funding limitations the CSEWG data evaluation and measurement components have been eroding. The current list of cross-section evaluations under development and the "wish list" for future work can be found in the summary of the CSEWG Meeting held on 17-19 October 1995. The most significant effort is devoted to a re-evaluation of the thermal and resonance region for ^{235}U.

ENDF/B-VI data listing is continuing. Most recent testing, as reported at the ANS Winter Meeting on 2 November 1995, on the LLL Pulsed Spheres, Winfrith Water Benchmark, Illinois Iron Sphere, Fusion SS/BP Shield, NESDIP2 Radial Shield, and H.B. Robinson Cavity Dosimetry indicates that the ENDF/B-VI data provides much better calculation-to-experiment (C/E) ratios than do previous versions of ENDF data. ENDF/B-VI also results in predictions of a 15 per cent increase in the in-vessel dosimetry and 30-35 per cent increase in the cavity dosimetry. Data testing has raised some concern that, despite the improved C/E ratios, the individual results for the dosimeters used in reactor dosimetry did not all improve by the same ratio, and that the spectral shape resulting from the use of ENDF/B-VI data may not be improved. Investigations of the results from use of the ENDF/B-VI cross-sections are continuing.

Codes used

Discrete Ordinates neutron transport calculations are generally performed with the DANTSYS codes (ONEDANT, TWODANT, THREEDANT) [127] or TORT/DORT [128] codes (the successors to the DOT code). MCNP (version 4a) [129] is used for point cross-section Monte Carlo transport calculations.

Spectrum adjustment calculations most commonly use the LSL [130] or FERRET [131] codes. For sophisticated least squares spectrum analysis the LEPRICON [115,121] code is used.

Critical issues

Experience in the US characterisation of reactor exposure parameters suggests that the following issues are most critical to the improvement of the calculational methodology for the determination of reactor exposure parameters:

- Cavity dosimetry measurements are very accurate using modern dosimeters and redundant measurement techniques. Flux uncertainty due to the dosimeter location uncertainty is also limited due to the small flux gradients in the cavity. However, the extrapolation from the cavity to the vessel inner radius is much less certain due to the accuracies of the neutron transport through the vessel. This extrapolation [119] is the critical point limiting the fluence determination based on cavity dosimetry.

- In general, the dosimetry cross-sections used in the characterisation of commercial nuclear power plants are well known and have high quality energy-dependent covariance data to guide the sensitivity estimates. One of the most important limitations in today's fluence estimates is the neutron transport cross-sections [122]. The other major contributors to the calculational uncertainty are the uncertainty in the distance from the dosimetry capsule to the vessel inner radius and in the azimuthal flux shape arising from the source distribution.

- New damage models exist but are not currently used by the industry. The reason they are not routinely used is that these models have not been demonstrated to yield a better correlation with measured damage. A reactor materials damage database exists which can be used to evaluate the damage models, however this information must be updated and evaluated (poor data and/or inconsistent damage assessment methodologies need to be identified). Hence, a correlation is not easily established with the historical damage database. Work needs to be done to put the historical reactor damage database into a form that can be used to evaluate improved damage modelling.

Discussion and conclusions

Table 8 presents the different levels of precision reported in the national contributions included above, for dosimetry calculations with adjustment to measurements.

Table 8: Current status of fluence calculation in the NEA Member countries

Country	Uncertainty in fluence calculations	Comments
Belgium	VENUS experimental results < 10% global uncertainties < 20% TIHANGE-2 unadjusted < 20% with adjustment 4% unadjusted 13-20%	Using both LEPRICON and MCBEND codes LEPRICON code MCBEND code
Finland	Loviisa VVER-440 Fluence E>1 MeV 11% DPA and Fluence E>0.5 MeV 13% 20-25%	PREVIEW code - Surveillance specimens - Pressure vessel
France	EDF power reactors RPV Uncert. in Fluence E>1 MeV 12% M/C = 0.967 (4 y) M/C = 0.997 (7 y) M/C = 1.024 (9 y)	TWODANT, SUSD RPV surveillance capsules using TRIPOLI code (CEA) and ENDF/B-VI
Germany	Siemens-KWU (general) C/E unadjusted ~20% Obrigheim PWR n/a	ANISN and DOT IKE Stuttgart using TORT and ENDF/B-VI
Japan	JMTR research reactor Fluence for E>1 Mev ~20% Thermal fluence ~40%	ENDF/B-IV ENDF/B-VI ENDL-3
Korea	KORI 4 C/E differences ~23% Fluence for E>1 Mev 15%	Surveillance capsule
Netherlands	HOR research reactor Fluence for E>1 Mev <15% often <10%	 MCNP4A EJ2-MCNPlib
Sweden	General method Fluence for E>1 Mev 2-25%	CASMO and DORT codes

Table 8: Current status of fluence calculation in the NEA Member countries (cont.)

Country	Uncertainty in fluence calculations		Comments
Switzerland	Inner surface of PWR vessel		BOXER code vs. Scraping test samples
	Fluence for E>1 Mev	<15%	
	Experimental uncertainty	10%	
United Kingdom	Magnox reactors		
	Fast flux C/E	0-35%	MCBEND
United States	Standard methods for LWR		
	Typical in-vessel E>1 Mev	20%	ENDF/B-IV
	ex-vessel E>1 Mev	30%	ENDF/B-IV
	in-vessel E>1 Mev	<20%	ENDF/B-VI

An analysis of the data in table reveals that:

- only the US and French contributions provide the orders of uncertainties in calculations;
- only the Japanese contribution gives the difference between calculations and measurements of thermal fluence;
- there is a visible spread in the numbers reported for the state-of-the-art methodologies with majority of data indicating about 20 per cent differences between calculations and experimental results for the fast fluence calculations;
- the numbers reported are difficult to compare; accuracy of the analysis of one reactor system is not representative of the analysis of another system even if the same methodologies are used (in the contributions above all possible parameters are varied: each country has its own methodology involving different reactors, codes, nuclear data sets, and measurement procedures).

Hence, based on the country reports on the state-of-the-art in computational dosimetry no firm judgement can be formed on:

- the current international level of accuracy in pressure vessel fluence calculations. The “median” result appears to fall within the 20 per cent difference between calculations and measurements but significantly higher and lower values are also reported – is this because some methods are clearly worse/better than others?;
- what are the relative merits of various methodologies (the weak and strong points of each methodology with respect to others) and hence which are the areas of possible improvements in various calculational schemes.

The advantages of a blind benchmark study, based on a single reactor model for which the fluence measurements (and their uncertainties) are known, would be:

- critical analysis and verification of each of the national methodologies;
- establishment of an unequivocal international consensus regarding the current level of accuracy of the pressure vessel fluence predictions using the latest nuclear data and state-of-the-art transport codes versus the experimental data;
- identification of sources of uncertainties and hence areas in methodologies considered where improvements could be made;
- establishment of a base for the third phase of the project: improvements in the models of material damage – verification of the fluence calculation methodologies is necessary prior to addressing the problem of improvements in metal damage models.

Recommendations

Based on the evidence presented, the following conclusions can be drawn:

- a benchmarking study based on a common reference problem and utilising different computational methods is needed to establish an international consensus regarding the level of accuracy of methods currently used in the NEA Member countries in calculating radiation dose to reactor components;
- the estimation of uncertainties in flux/fluence calculations requires more precise and complete variance/covariance matrix data for uncertain parameters; moreover, the spectra adjustment using detector measurements needs improvements for variance-covariance matrix of calculated spectra;
- there is a need to validate gamma transport codes, nuclear data, and gamma induced metal damage models for estimating gamma metal damage in some reactors (see pages 13 and 18);
- there is a need for more detailed work to determine the importance and magnitude of errors in thermal fluence estimates on metal degradation in some reactors (see page 13);
- there is a need for further work on the development of models (such as the PKA [10]) relating particle flux/fluence/spectrum to metal damage.

The members of the TFRDD group agreed [43] to perform the benchmarking fluence computations in a "blind test" as a Phase II of this project (Phase I being preparation of this report), using the specifications of the VENUS critical facility at SCK/CEN Mol in Belgium. Two configurations are considered: a 2-D and a 3-D configuration of the VENUS facility.

In the first benchmark, the VENUS-1 experiment will be simulated using most recent transport codes and the results of computations will be directly compared with the VENUS-1 2-D measurements. VENUS-1 configuration is based on a fresh UOX core (i.e. no fission products will be taken into account in the computations).

In the second benchmark, the VENUS-3 experiment will be simulated using the state-of-the-art transport codes and the results will be compared with measured VENUS-3 flux distributions. This is a 3-D benchmark that will allow to verify the relative accuracy of the flux synthesis methods consisting of reconstructing the 3-D flux distributions using 1-D and/or 2-D calculations versus the full 3-D computations using 3-D transport codes. Various other issues of the multidimensional computational dosimetry such as, for example, the accuracy of calculations as a function of the core height (the calculations for positions above core midplane are considered to be less accurate), etc., will be verified as well. In these studies state-of-the-art cross-section data will be used. The results of the analysis of the VENUS benchmarks will be presented and discussed in depth in a separate report.

In Phase III of the project, the TFRDD group agreed [43] (contingent upon the approval of the NEA Nuclear Science Committee) to address the problems of the basic physics of metal damage phenomena occurring during irradiation of reactor components. In particular, the commonly used models of threshold fluence, DPA, and PKA would be considered from the point of view of their accuracy in an attempt to look for improvements in translating the knowledge of flux/fluence to knowledge of the degree of metal structure degradation. It should be stressed that at this point the program of Phase III of the project [43] still needs to be clearly defined.

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ANNEX 1

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ANNEX 2

Programs and data related to reactor dosimetry

A list of computer programs and data sets useful for reactor dosimetry studies is provided here.

Programs and data sets identified with CCC, PSR and DLC have been packaged at the RSICC (Radiation Safety Information Computational Center): http://epicws.epm.ornl.gov/.

Programs and Data identified with NESC have been packaged by the former National Energy Software Center now taken over by ESTSC (Energy Science and Technology Software Center): http://apollo.osti.gov/html/osti/estsc/

The identifications NEA and IAEA correspond to programs and data packaged at the OECD/NEA Data Bank: http://www.nea.fr/html/dbprog/.

A more detailed description can be found by accessing the URLs of the respective information centres. In addition to those listed, there are a large number of codes of a more general nature which are required for flux and fluence calculations, i.e. 2D and 3D radiation transport codes. The different WWW pages provide ample information on these as well. Instructions for requesting the various items are provided.

Index to programs and data related to reactor dosimetry

Program-name	Description	Identification
ACDOS3	n activation activities & dose rates	CCC-0442
ACFA	isotope activation of coolant & structural materials	NEA 1072
BASACF	ntegral n spectra adjustment & dosimetry	IAEA0953
CRYSTAL-BALL	n spectra calculation from activation experiments with error estimate	CCC-0233
DANTE	activation analysis n spectra unfolding by covariance matrix method	NEA 0694
DOSEFACTOR-DOE	dose rate conversion factors for photon & electron exposure	CCC-0536
ENBAL2	n & gamma kerma calculation from multigroup cross-sections	NEA 0857
HEXANN-EVALU	n irradiation of reactor pressure vessels	NEA 1125
KAOS-V	n fluence to kerma factor evaluation from ENDF/B-5 & JENDL-2	PSR-0306
LEPRICON	PWR vessel dose analysis with DORT & ANISN program	PSR-0277
LOUHI	spectra unfolding with linear & non-linear regularisation	NEA 1026
MARLOWE	atomic displacement cascades for crystals, recoil range distributions	NESC0680
NAC	n activation analysis & isotope inventory	CCC-0164
NJOY94	general ENDF processing system for reactor design problems	PSR-0171
OCA-P	PWR vessel probabilistic fracture mechanics	NESC1125
OCTAVIA	PWR vessel failure probability for routine pressure transients	NESC0898
RDMM	flux spectra from in-pile fast n activation experiments	NEA 016
REAC-2	nuclide activation & transmutation	NESC9554
RICE	energy exchange matrix, damage x-sec, recoil energy spectra from ENDF	NESC0453

Index to programs and data related to reactor dosimetry (continued)

Program-name	Description	Identification
RICKI	interactive gamma spectra unfolding with isotope identification	NESC9580
SNL-SAND-II	n flux spectra from multiple foil activation analysis	PSR-0345
SPECTER-ANL	n damage for material irradiation	PSR-0263
SPUNIT	multisphere n spectra unfolding	PSR-0266
STAY-SL	dosimetry unfolding with activation, dosimetry, flux error	PSR-0113
SUSD	sensitivity & uncertainty in n transport & detector response	NEA 1151
VISA-2	reactor vessel failure probability under thermal shock	NESC1115

Data library-name	Description	Identification
ZZ ACTIV-87	fast n activation cross-sections	IAEA1275
ZZ ACTL82	evaluated activation cross-section library	DLC-0069
ZZ-BUGLE-96	multigroup coupled n gamma x-sec for LWR shielding calculations	DLC-0185
ZZ COVFILS	30-GROUP covariances from ENDF/B-5 for sensitivity studies	DLC-0091
ZZ COVFILS-2	74-group n x-sec, scattering matrices, covariances	DLC-0137
ZZ DAMSIG84	640 group damage x-sec library for SAND-2 calculations	NEA 0791
ZZ DLC-10B AVKER	n kerma response function data library	DLC-0010
ZZ DOSCOV	24 group covariances from ENDF/B-5 for dosimetry calculations	DLC-0090
ZZ DRALIST	radioactive decay data for dosimetry & hazard assessment	DLC-0080
ZZ IRDF-90	620 group x-secs & spectra for dosimetry calculations	IAEA0867
ZZ KAOS/LIB-V	kerma factors, response functions for fission, fusion	DLC-0160
ZZ KERMAL	n & gamma kerma library from ENDL & EGDL	DLC-0142
ZZ MACKLIB	response functions for CTR & hybrid fission fusion systems	DLC-0029
ZZ MATX175/42-JEF	172 n, 42 gamma groups library in Vitamin-j structure	NEA 1205
ZZ NMF-90	database for n spectra unfolding	IAEA1279
ZZ NUCDECAY	nuclear decay data for radiation dosimetry calculation for ICRP	DLC-0172
ZZ RECOIL/B	heavy charged particle recoil spectra library for radiation damage	DLC-0055
ZZ SENPRO/45C	multigroup sensitivities for fast & thermal reactors	DLC-0045
ZZ SNLRML	dosimetry cross-section recommendations	DLC-0178
ZZ UNGER	effective dose equivalent data for selected isotopes	DLC-0164
ZZ VITAMIN J/COVA	covariance matrix data library for sensitivity analysis	NEA 1264
ZZ VITAMIN-J/KERMA	gas production x-sec, n & gamma kerma	NEA 1168