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DEVELOPMENT AND TEST VALIDATION OF A COMPUTATIONAL SCHEME FOR HIGH-FIDELITY FLUENCE ESTIMATIONS OF THE SWISS BWRs

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ABSTRACT

One of the current objectives within reactor analysis related projects at the Paul Scherrer Institut is the establishment of a comprehensive computational methodology for fast neutron fluence (FNF) estimations of reactor pressure vessels (RPV) and internals for both PWRs and BWRs. In the recent past, such an integral calculational methodology based on the CASMO-4/SIMULATE-3/MCNPX system of codes was developed for PWRs and validated against RPV scraping tests. Based on the very satisfactory validation results, the methodology was recently applied for predictive FNF evaluations of a Swiss PWR to support the national nuclear safety inspectorate in the framework of life-time estimations. Today, focus is at PSI given to develop a corresponding advanced methodology for high-fidelity FNF estimations of BWR reactors. In this paper, the preliminary steps undertaken in that direction are presented. To start, the concepts of the PWR computational scheme and its transfer/adaptation to BWR are outlined. Then, the modelling of a Swiss BWR characterized by very heterogeneous core designs is presented along with preliminary sensitivity studies carried out to assess the sufficient level of details required for the complex core region. Finally, a first validation test case is presented on the basis of two dosimeter monitors irradiated during two recent cycles of the given BWR reactor. The achieved computational results show a satisfactory agreement with measured dosimeter data and illustrate thereby the feasibility of applying the PSI FNF computational scheme also for BWRs. Further sensitivity/optimization studies are nevertheless necessary in order to consolidate the scheme and to ensure increasing continuously, the fidelity and reliability of the BWR FNF estimations.

Key Words: Fast neutron fluence, BWR, CASMO-4/SIMULATE-3/MCNPX.

1. INTRODUCTION

Very recently, the oldest Swiss BWR received unlimited operating permit from the national safety inspectorate. With that, all the Swiss PWR and BWR operating reactors have now been granted the possibility to continue operation as long as it can be constantly proven that the plant continues to meet all relevant criteria for safe operation. To that aim, in addition to rigorous periodic safety assessments, comprehensive lifetime management programs are being implemented including both continuous modernization measures as well as maintenance of strict surveillance/monitoring programs related to material/structural ageing and degradation mechanisms of large reactor components. In that context, embrittlement of the reactor pressure vessel and internals due to fast neutron fluence (FNF) is given particular attention.

To support these plant ageing programs, one of the R&D activities carried out at the Paul Scherrer Institut (PSI) is the development of advanced high-fidelity computational methodologies for accurate fluence estimations. Recently, such a modern FNF computational scheme was developed for PWR reactors and a first validation on the basis of RPV scraping tests showed a rather satisfactory performance [1]. The two central pillars of the PSI scheme are 1) to ensure an accurate fission neutron source distribution and 2) to allow for a comprehensive realistic 3-D modelling of the neutron transport to any location of interest within the RPV e.g., at the core shroud, top guide, core plate, etc. For the first pillar, PSI CMSYS reference 3-D core models that are continuously developed and validated against plant data for all the Swiss reactors [2], are used to provide the neutron source distribution. For the second pillar, the state-of-the-art MCNPX code with continuous-energy neutron data libraries is used to ensure accurate neutron/particle transport calculations with maximum modelling flexibility.

Nowadays, emphasis is given at PSI to develop a corresponding FNF calculation scheme for BWRs. Similarly as for other countries [3], this is indeed a relevant task that might imply additional complexities and challenges due to the inherently different aspects of BWRs, e.g. with larger RPV core/internal volumes, in-vessel steam production, two-phase flow core thermal-hydraulics and harder spectrum, complex fuel assembly designs with axial/radially heterogeneous enrichments and Burnable-Absorber zonings. In this paper, the preliminary steps that have been undertaken towards adapting and further developing the PSI fluence scheme for BWR applications are presented.

First, the concepts and principles of the PSI methodology are briefly outlined in Section 2 where specific adaptations that might be required for BWR applications are also discussed. Thereafter, the general principle of the MCNPX FNF modelling adopted for one of the Swiss BWRs is presented. And along this, the specific modelling principles applied for RPV and core shroud FNF estimations are outlined. Finally, a first validation test case of the BWR scheme under development is presented, including a) a brief description of the analysed dosimeters in terms of location and irradiation conditions; b) an overview of the neutron source accuracy achieved with the 3-D core models and c) a comparison of the FNF and the dosimeter activities calculated with the MCNPX model versus experimentally evaluated data, noting that the presented MCNPX calculation results are obtained with version 2.4.0 [4] and with two distinct neutron data libraries, namely JEFF-3.1.1 [5] and ENDF/B-VII.0 [6] .

2. PSI LWR FLUENCE METHODOLOGY OUTLINE

2.1. Principles

The PSI methodology under development for BWR FNF analysis is illustrated in Fig. 1 and based on the same principles as the one used for PWR analyses [1]. Basically, from the CMSYS core models, the 3-D power density, fuel compositions and in-channel coolant densities for all fuel assemblies and axial elevations in a selected core region are directly transferred to an MCNPX model via the SOURCE4MC linking code. As shown in Fig. 1, this module includes a conversion of the 3-D power distributions P_m [W] and the assembly-averaged nuclide compositions $\bar{\rho}_m^i$ into a volumetric neutron source distribution. This comprises 1) a specification of the neutron source strength S_m at the 3-D assembly or pin-by-pin level, and 2) a specification for each individual fuel assembly of the fission neutron spectrum $\chi_m(E)$, based on the related recoverable energy per fission E_{Rm} [eV], and the number of neutrons per fission ν_m, noting that "*m*" is the fuel assembly identifier, "*i*" is the fissionable nuclide identifier; σ_f is the effective microscopic fission neutron cross section [cm^2], and C=1.6019·10^{-19} is the energy unit conversion factor [J/eV].

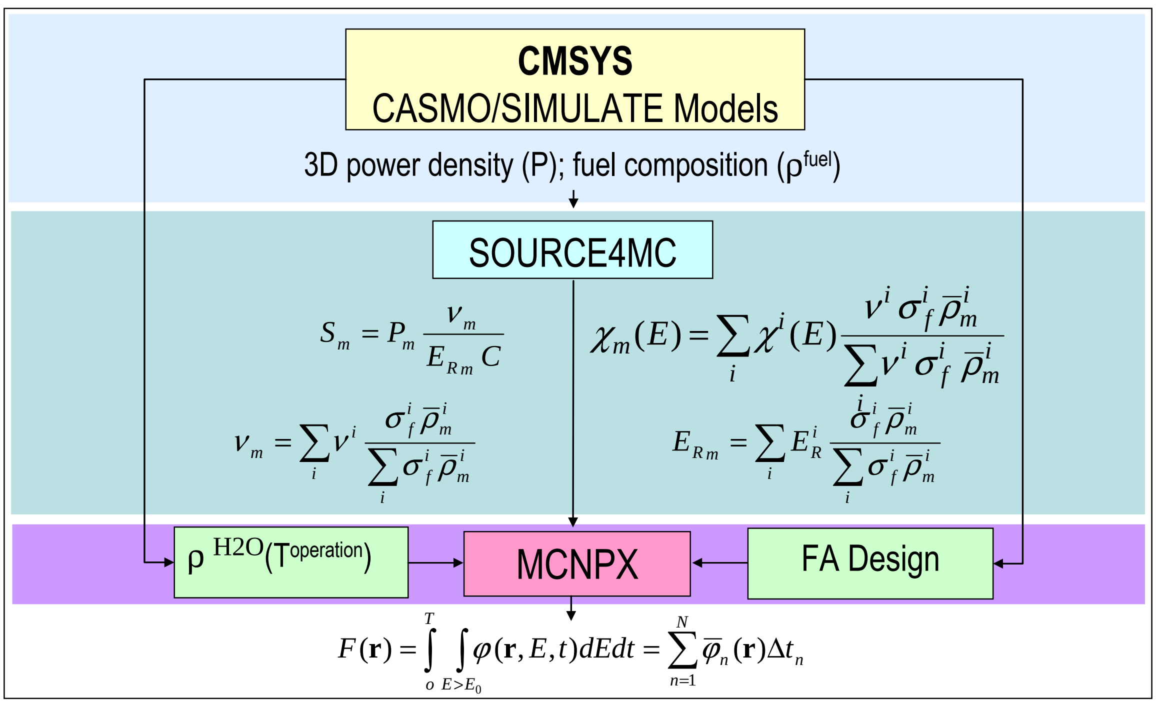

Figure 1. PSI computational scheme for LWR fluence estimations.

The above automated procedure allows to account for the spatial/temporal evolution of the source and can thus provide a "cycle-average" source (i.e. for cycle-averaged conditions) or a more (or a less) detailed source specification at any arbitrary number of operating steps/cycles. The "cycle-by-cycle" source approach has in fact been shown to provide satisfactory results for an integral fluence estimation of the PWR RVP scraping tests made after 10 cycles of irradiation. However, for the analysis of reactor surveillance dosimeters, more frequent updates of the source distributions might be required to take into account accurately the spatial-spectral redistributions of the neutron source for activity evaluations of short-lived dosimeter reaction products.

2.2. Adaptations for BWRs

Regarding the transfer of the neutron source, the previously described procedure can in principle be applied also for BWRs. However, additional adaptations of the scheme might be needed due to three main BWR specific aspects that might deserve further attention. These adaptation needs are described below along with the simplifications that are in that context currently made and hence applied for the first validation test case presented in this paper.

- A first limitation is that whereas the detailed neutron source distribution can directly be transferred to MCNPX, the assembly geometrical/structural modelling is made stand-alone in MCNPX. That is, while a representative uniform assembly geometrical model is usually sufficient for PWRs, for BWRs the SOURCE4MC code will need to be updated to also ensure, in an automatic manner, modeling in MCNPX input of a core loaded with fuel assemblies (FA) of different types, consistently with the source specifications for each FA type. Indeed, modern BWR core designs are characterized by very complex and heterogeneous fuel assembly designs both radially and axially. A proper account of these heterogeneities is currently not fully implemented in the BWR fluence scheme. The needed levels of modelling details for the production calculations are currently under analysis.

- Secondly, the BWR core is characterized by a complex set of bypass flow patterns including principally inter-assembly bypasses (i.e. outside the channel boxes) and intra-assembly bypasses structures (e.g. water rods, crosses, wings). Now while the SOURCE4MC will allow transferring the 3-D in-channel node-average void distribution into assembly water mixture-density axial profiles for MCNPX, the detailed bypass thermal-hydraulic conditions will remain not available from the core models. For this reason, it is currently assumed that for all intern-assembly bypasses, coolant is at saturation.

- Third, the surveillance dosimeters that are available for validation are located at the RPV wall separated from the core shroud by the downcomer which includes internal components such as e.g. jet pumps. The coolant flow in the downcomer is also complex since consisting of feedwater flow as well as water mixture returning from the steam-separators with a certain steam carry-over. The complex coolant velocity/temperature patterns in the downcomer are certainly not within the reach of the deterministic core models and therefore can not at this stage be accurately modelled. For that reason, it is assumed that the downcomer flow is at core inlet temperature conditions.

3. MCNPX MODELLING OF SELECTED SWISS BWR REACTOR

3.1. Situation and Validation Target

To comprehensively qualify and validate a BWR FNF computational methodology, it is clear that a wide range of experimental data from commercial BWRs in operation would constitute a very solid and strong basis. For the methodology under development here, a principal source of experimental data stems for reactor dosimeters and surveillance capsules irradiated in the vicinity of the RPV inner wall of a Swiss BWR/6 reactor. Therefore, this reactor along with the available RPV dosimeter data sets was selected as primary "situation and validation target" i.e. for the development and first validation cases of a BWR FNF scheme. The MCNPX model set-up to that aim along with geometrical truncations related to the available validation data are described in the next subsections. The results for a first validation case are then presented in the Section 4.

3.2. General Modelling Approach

The general radial and axial views of the MCNPX model set-up for the given Swiss BWR reactor is shown in Fig. 2 where for simplicity, a fictitious core loaded with fuel assemblies of a unique design is represented.

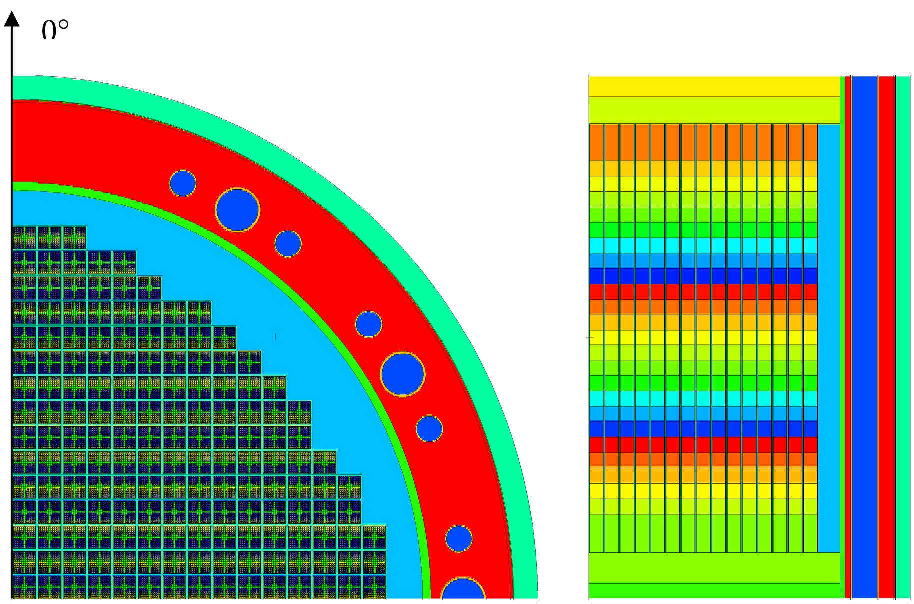

Figure 2. MCNPX calculation model of a Swiss BWR reactor.

It should be underlined that for the available validation target(s), irradiation towards the RPV wall is of primary relevance and consequently, a detailed modelling of the inner core regions is not really necessary although as can be seen, the adopted MCNPX modelling approach thus allows to account for an entire quarter core region, something that might become necessary for future FNF estimations at other locations than the RPV wall. In any case, for the validation target presented in this paper, the number of required fuel assemblies will be discussed in the next sections. Concerning the structures situated axially outside the core, these play a very insignificant role at least for the current validation targets and may therefore be modelled only approximately (or even neglected). For the present MCNPX model, these regions are therefore simply modelled as “homogeneous regions” in accordance with the guidelines proposed in a BWR benchmark [7]. Finally, because the dosimeter sets are located on the RPV wall close to the 0^{o} azimuthal direction (See Fig. 2), a detailed modelling of the jet pumps and risers do not play any significant role for the present validation targets. Therefore, these are also currently modelled only in a very primitive manner. Regarding the axial discretization, it is schematically shown in the figure that the MCNPX core model is subdivided in 25 nodes which was selected to be consistent with the axial nodalisation employed for the CMSYS core models of the given reactor.

In order to achieve a satisfactory statistical precision of the results on the fast neutron flux at the RPV, the one-energy bin, 1-D cylindrical Weigh Windows variance reduction option was utilized. The flux surface tally at the entire RPV inner wall was used as a target for the Weight Windows generator. It was found that this option is rather convenient for all consequent calculations discussed later. However, accurate FNF estimations might also be required at other locations (e.g. weld of support plates, nozzles of safety injection systems) and the computational model should therefore be equally applicable for FNF analyses at any of these locations. In the case of using a Monte Carlo code for production FNF estimations at several RPV locations, this might not be straightforward as the model must be optimised to achieve a reasonable variance and hence statistical precision for any of the desired locations. The guiding principle is therefore to start with a model that describes/includes all important reactor structures but with a different level of modelling details for the various components, depending on the validation target.

3.2. Geometry Truncation for RPV Validation Targets and Core Shroud Evaluations

At present, the responses of main interest for validation and future applications are 1) fast neutron flux at the locations of the dosimeters and 2) maximum fast neutron flux at the core shroud. For the former, two different positions of dosimeters will therefore be considered, both located around the active core centreline and relatively close to the 0° azimuthal degree core symmetry axis. Regarding the fast neutron flux at the core shroud, it takes a maximum value also relatively not far from 0° azimuthal degrees as shown in Fig.3 based on scoping calculations (for a certain cycle) with the simplified quarter core model also shown in that figure.

It is thus evident that only some fuel assemblies will play an important role for FNF calculations at the core shroud or at the dosimeters locations. Thus, at first the inner parts of the core may therefore be neglected in order to achieve better calculation efficiency (as shown in Fig. 3) while still providing unbiased results. To distinguish precisely the important vs. less relevant core regions, the contributions of the individual fuel assemblies with respect to the considered responses and referred to as the “importance factors“ (IF) were evaluated (see examples of such calculations in, e.g., [8, 9]).

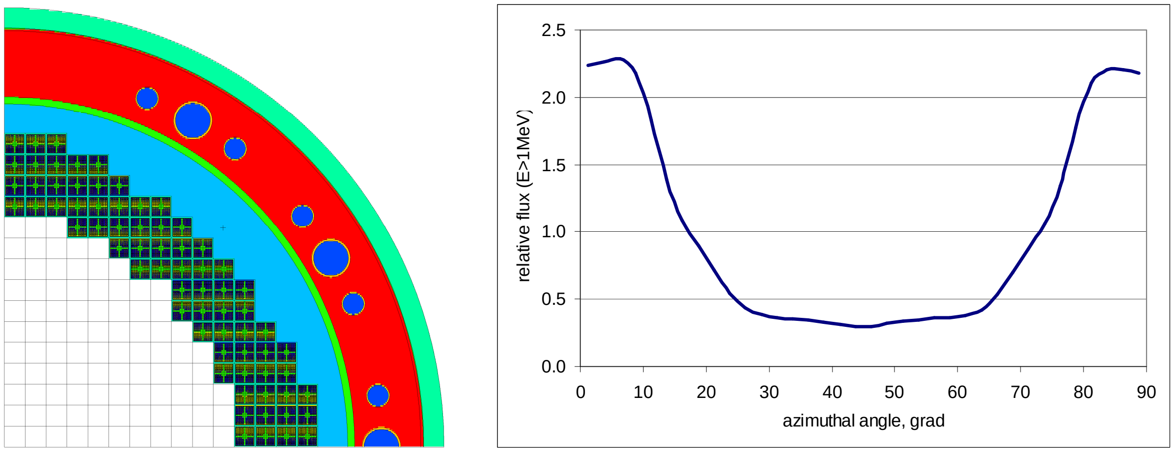

Figure 3. Scoping FNF model (left) for assessment of fast neutron flux shape at core shroud (right).

Because the FNF at the core shroud may be more sensitive to central fuel assemblies compared to the response at the dosimeter locations in the RPV vicinity, the core shroud FNF was chosen as the response of interest for an optimized model truncation. On the left part of Fig. 4, the IF values for the locations corresponding to FAs located in the first 4 top rows of the north core quarter (see left-hand side of Fig.3) are shown. The right side of Fig. 4 shows respectively the relative impact of each individual FA located on the reactor symmetry axis (i.e. at 0° azimuthal degrees), on the neutron flux at the 0° azimuthal location on the inner surface of the core shroud. There, the values on the axis of abscises correspond to the row positions of the FAs on the symmetry axis starting from -1 at the core periphery and going inside the core. Four values marked on the plot were actually calculated and the value for the fifth fuel assembly, which was not actually presented in the model, was simply extrapolated, as a near perfect exponential behaviour of the FA IF vs. radius position can indeed be noted.

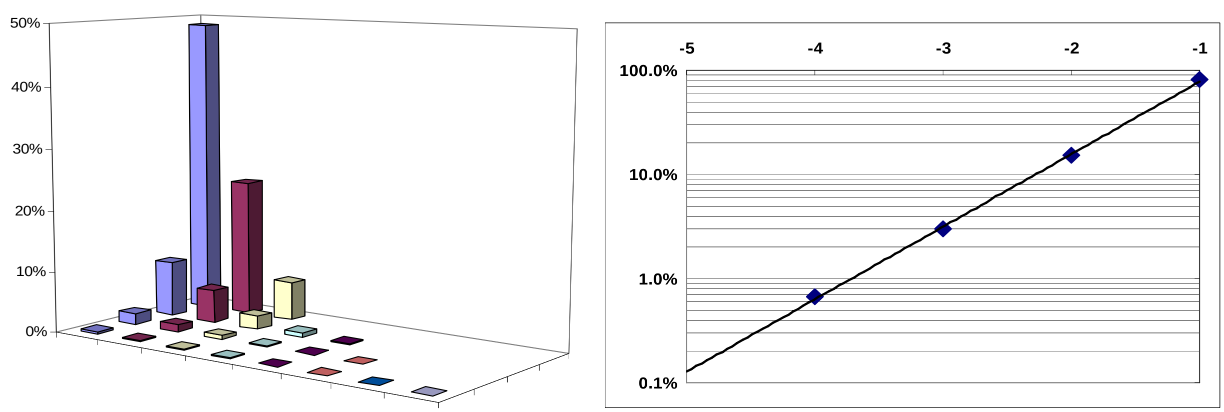

Figure 4. IF values (left) and relative contributions of the FAs on the 0° azimuthal degrees axis (right) on the FNF at the 0° azimuthal location of the inner core shroud surface.

Similarly, Fig 5 below shows the IF values with respect to the fast neutron flux response at the considered two dosimeter locations.

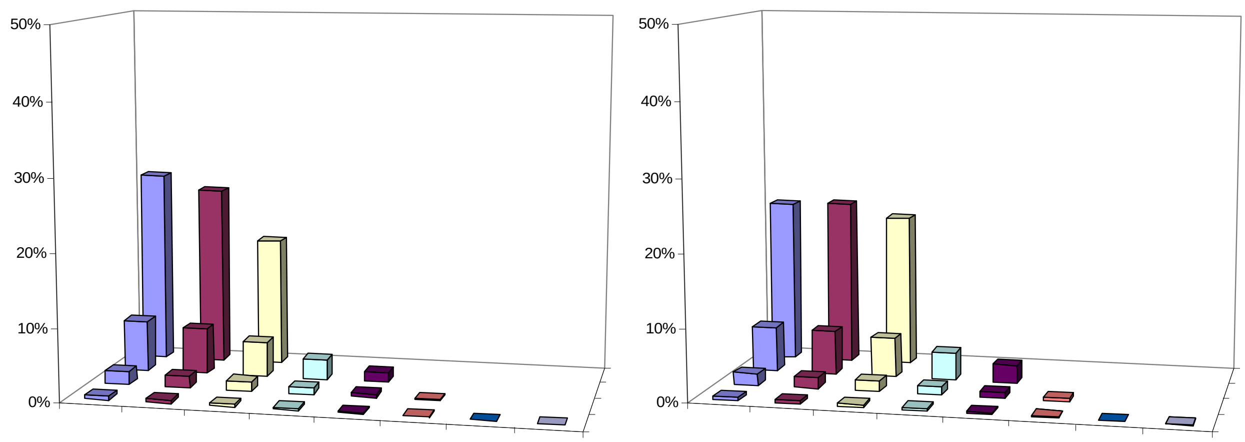

Figure 5. IF values for dosimeters at two different azimuthal locations.

The presented results in Figs. 4 and 5 clearly illustrate that if the model is intended for calculations of the FNF distribution at the core shroud surface, then there is no need to include more than 4 peripheral rows of FAs into the model. And if only some specific azimuthal locations are of interest, the number of modelled FAs might be even further reduced to significantly improve the calculation efficiency while retaining the same level of accuracy. This approach will precisely be adopted for the test validation case presented and described in more details in the next section.

Now it must be underlined that although the IF values presented here were obtained with an approximate FA modelling (i.e. based on a specific uniform assembly design), it was verified by means of additional scoping studies that the IFs are indeed relatively weekly dependent on the FA design. Therefore, the optimized geometrical model truncation presented here should remain applicable for azimutahl core shroud/RPV FNF calculations of any cycle for the given reactor, i.e. more or less independently of the cycle-specific core and FA design.

4. VALIDATION TEST CASE

4.1. Description of Selected Dosimeter Data and Associated MCNPX Modelling

As first validation target, the most recent dosimeter set evaluated at the PSI Hotlab was selected. From this experimental analysis, measured activities for two dosimeters, viz. for ^{54}Fe and ^{93}Nb and both irradiated during two recent cycles, are available. In addition, experimental-based FNF estimates are also provided. Therefore, it was considered appropriate to select this recent dosimeter data set to perform a first assessment of the BWR FNF methodology currently under development, the principal objective being at this stage mainly to verify its applicability and to provide a first quantitative estimation of achieved accuracy.

It must indeed be recalled that the current MCNPX model as well as the associated SOURCE4MC methodology is still at an early phase of development and will most likely be gradually updated/enhanced as function of extending the validation basis. Therefore, the validation target presented here is referred to as a test validation case. The location of the analysed dosimeters is schematically shown in Fig. 6.

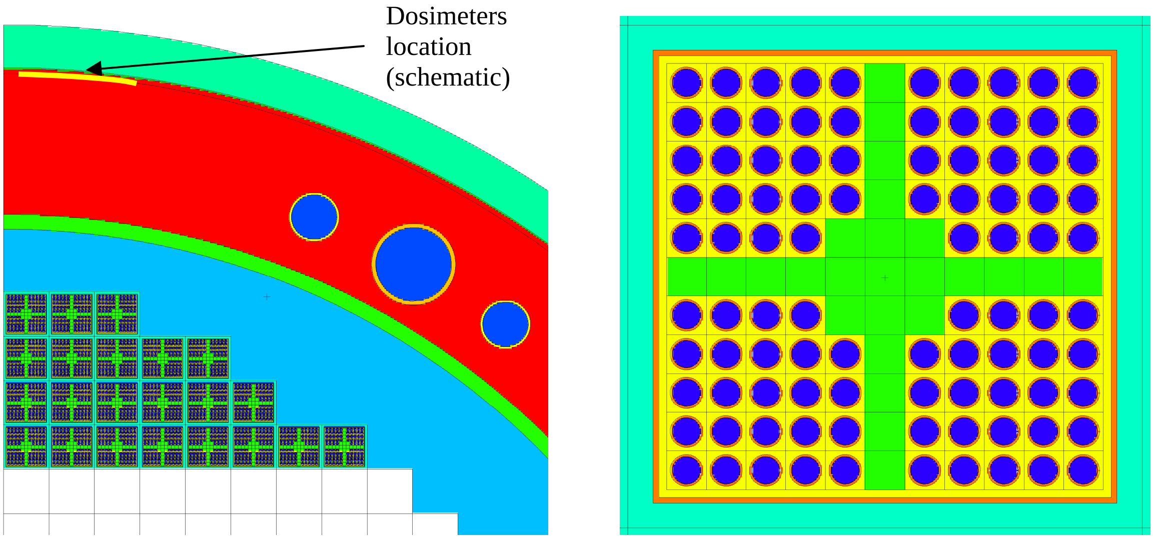

Figure 6. Core (left) and fuel assembly (right) modelling for dosimeters irradiation simulation.

Based on the truncation and IF studies presented in the previous section, the modelled core region is, as can be seen on the left-hand side of Fig. 6, restricted to all FAs of the four first rows in the northern quadrant zone facing the RPV dosimeter locations.

For each of the FA, the simplified assembly model shown on the right-hand side of Fig. 6 is applied. This corresponds to a crude approximation of a modern 10x10 BWR fuel type with ‘representative’ fuel rod dimensions and a simplified description of the intra-assembly bypass water paths modelled as a homogenous mixture of Zircaloy and coolant. To verify this geometrical design approximation, a sensitivity calculation was performed replacing the intra-assembly homogenized Zircaloy-coolant volumes (green zones in Fig.6) simply by “dummy” fuel rods, the impact on the FNF estimations at the location of dosimeters would be in the order of +3% or below (note that the statistical precision of the related calculations was around ~1%). Further, in the next test calculation the intra-assembly homogenized Zircaloy-coolant volumes were replaced by pure coolant volumes, neglecting thus the Zircaloy structures inside the fuel assembly. The observed difference in the FNF values was only ~+1.5%. This tends to justify that at least, at this stage of development, a more exact modelling of the various and complex FA designs is not really needed.

Concerning the FA fuel compositions, a ‘representative’ composition corresponding to the average fuel burnup over the peripheral assemblies and over the irradiation time was used for all fuel rods. It was found in the previous studies [1] that the fast neutron flux calculations with a fixed neutron source are almost insensitive to the actual fuel composition and therefore it is assumed that the currently used approximations are acceptable. Note however that for the source specification, the individual FA fuel compositions, as well as the complex FA design details, are taken into account (see Section 2.1).

Regarding the T-H conditions, the bypass and downcomer flow conditions are approximated as described in Section 2.2. For the FA in-channel density, core-average axial profiles are applied and obtained together with the neutron source data transfer (see next section). The more detailed SOURCE4MC option to provide individual FA axial density/fuel temperature profiles is hence not applied at this stage. And with regards to thermal expansion, it is mostly the changes in distance between the core periphery and the RPV dosimeters that are important. Therefore, the thermal expansion was approximated by an effective increase of the RPV radiuses while keeping all other dimensions. The RPV inner radius change was approximated by assuming that both the RPV and the reactor core are expanding as function of the coolant temperature with the same linear thermal expansion coefficient, chosen as $1.7*10^{-5}$[1/°C].

Finally, the above model is used for MCNPX version 2.4.0 calculations with both the JEFF-3.1.1 and ENDF/B-VII.0 libraries. However, in both cases, the same fast neutron spectrum approximations as described in e.g. [1] are employed. The neutron source for the modelled core region is obtained from the core models (see next section) noting that the option to use a "cycle-average" source is applied here.

4.2. Neutron Source and Validation

To provide the neutron source in the modelled FAs, the PSI methodology relies on the SOURCE4MC code (see Section 2) which uses as basis reference CMSYS 3-D core steady-state models developed and validated after each completed cycle. The validation against plant data of the 3-D power distribution obtained with these models provide thus an estimation of the achieved accuracy in terms of the neutron source used for the MCNPX FNF estimations.

For the BWR reactor analysed here, the validation is made using as basis, Transverse In-Core Probe (TIP) measurements that for each cycle are periodically carried out at hot operating conditions. The gamma response is measured by a radiation detector within the TIP probe mechanism. In total, 35 detector strings are employed and gamma signal measurements are made at 25 axial locations. Concerning the core model computations, pin power reconstruction is performed to calculate the homogeneous 2-group fluxes in the narrow-narrow gap of each assembly and axial level. These fluxes are then combined with microscopic detector cross-sections (steel material for gamma detectors) and detector-to-cell flux ratios, obtained from 2-D lattice transport calculations as function of local conditions (e.g. exposure, void history etc..), to calculate the TIP response (i.e. reaction rate) of each assembly/axial plane. Finally, at a given TIP radial/axial location, the "final" calculated TIP signal is estimated as the average of the 4 neighbouring assembly TIP responses. After normalization of all calculated TIP signals, a comparison to the (normalized) measured TIP signals is made. To quantify the accuracy, Root-Mean-Square errors of the local deviations between calculations and measurements are then computed and used a principal performance metrics.

For the 5 last modelled cycles of the given plant model, the achieved RMS are shown on the left-hand side of Fig. 7, reflecting a satisfactory accuracy with nodal RMS around or below 4%. This also applies to the two cycles during which the dosimeters analysed in this paper were irradiated and which are highlighted on the left hand side of Fig. 7 as Cycles 2-3. For these two cycles, the bias and RMS of the differences against measurements (averaged over all measurements of both cycles) is in addition shown as function of axial elevation on the right hand side of Fig. 7. Principally, the bias curve can be considered as the “average” axial error shape of the neutron source transferred to the MCNPX FNF models and as can seen, a typical overprediction is obtained at the core axial peripheries while around core mid-plane, the source will tend to be underpredicted by around -2 to -3%.

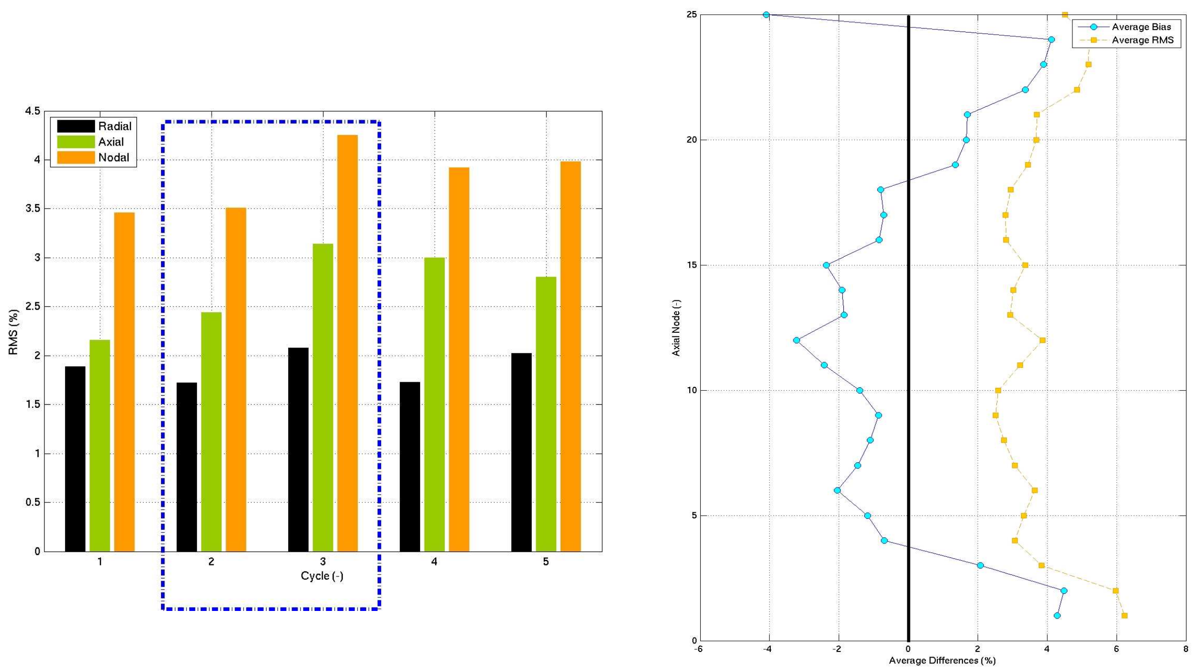

Figure 7. RMS errors for 5 last cycles (left) and axial distribution of error (averaged overall measurements of 2 selected cycles, right).

4.2. Calculated FNF and Activities versus Measurements

With the previously described MCNPX model and associated neutron source, the fast neutron flux at the location of the detectors and the dosimeters reaction rates (R) are as last step calculated. Based on Eq. (1) below, the calculated reaction rates can then be translated into the specific activities of the dosimeter irradiation products (as was similarly applied for a PWR dosimetry benchmark analysis [10]). Here, the inclusion of a weighting factor based on the IF ratios is applied in order to take into account the neutron source redistribution during the irradiation period. In the given study intermediate neutron source distributions at 5 time steps per one cycle were utilised[1].

$$A = R\,\frac{N_A x}{M}\sum_{j=1}^{J}\sum_{i=1}^{I_j} P_i\,(1-e^{-\lambda\Delta t_i})\,e^{-\lambda(T-t_i)}\cdot\frac{\sum_{n=1}^{N} S_n^{j} IF_n}{\sum_{n=1}^{N} S_n^{av} IF_n}\;/1000 \qquad (1)$$

where:

- A [Bq/mg] – Specific activity;
- R [1/s] – Reaction rate;
- N_a [1/mol] – Avogadro number, $6.022\cdot10^{23}$;
- M [g/mol] – Mass number;
- J – Number of irradiation time intervals for which the data on power distributions are given (here - 5);
- I – Number of time steps for which the relative power load factors are given (daily);
- N – Number of fuel assemblies present in the model;
- S – Relative neutron source (in the present analysis substituted by relative power);
- S^{av} - Relative neutron source averaged over the cycle (which was used for the *IF* calculations);
- P_i – Relative power;
- *IF* – FA importance factor;
- x – Here: isotope abundance;
- T – Time from beginning of irradiation to the time of activity counting;
- t_i –Time from beginning of irradiation to the end of time interval i;
- Δt_i – Length of time interval i (here – one day);
- $\lambda[s^{-1}] = \ln(2)/T_{1/2}[s]$ - decay constant of the product isotope.

The results obtained in that manner for the two measured dosimeters, ^{54}Fe ($T_{1/2}$=313days) and ^{93}Nb ($T_{1/2}$=16.1years), are presented in Fig. 8. On the left the activities calculated versus measured values are shown. On the right – the values of calculated FNF versus FNF previously evaluated at PSI based on the measured activities. These older evaluations were done using dosimetry cross-sections based on the ENDF/B-V library. Calculation results are actually compared with average measured-based values, which experimentalists reported respectively based on 3 and 6 individual measurements for ^{54}Fe and ^{93}Nb dosimeters. It should be mentioned that the same ^{93m}Nb production cross section (originated from ENDF/B-VI MOD 3 library), was used in calculations with both JEFF-3.1.1 and ENDF/B-VII.0 libraries.

Note that the experimental uncertainties were estimated as ~±7% while the MCNPX statistical precision is less then 1%, yielding thus that the total uncertainty is ~±7%.

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[1] This is actually needed only for short-living radioactive nuclides. In fact, for the case of $^{54}Fe(n,p)^{54}Mn$ dosimeter discussed here, the impact of the source redistribution correction was found to be only ~1.5%

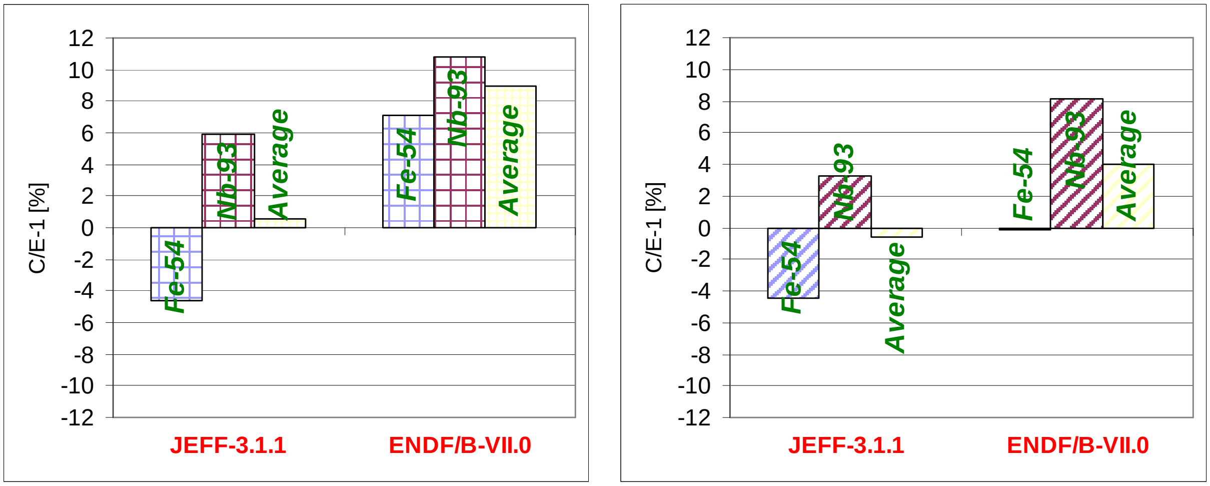

Figure 8. Obtained C/E values for specific activities (left) and FNF (right).

The above results show that despite the current modelling simplifications/approximations, a satisfactory accuracy may be achieved. It is further noted that in line with the previous findings regarding the performance of the JEFF-3.1.1 and ENDF/B-VII.0 libraries for dosimetry calculations, the main discrepancies are seen for the Fe dosimeter activity and are due to the library differences in terms of the ^{54}Fe(n,p) reaction cross-section [11]. The present results are also in line with observations from [12], where ENDF/B-VII.0 library was found to give higher values for the FNF at PWR RPV comparing to JEFF-3.1 library.

5. CONCLUSIONS

The paper has presented one of the R&D activities currently conducted at the Paul Scherrer Institute and aimed at the establishment of a high-fidelity computational methodology for accurate FNF analyses of reactor pressure vessel and internals of the Swiss operating BWR reactors. The overall principles of the PSI methodology for LWR fluence analyses and based on the CASMO-4/SIMULATE-3/MCNPX system of codes, have been outlined along with a discussion on specific adaptation needs that might be required for BWR applications. The general approach for the MCNPX FNF modelling adopted for a selected Swiss BWR reactor was then presented and complemented by a description of an optimised model truncation proposed for FNF estimations specific to the core shroud surface and dosimeters' locations. On that basis, a first validation test has been presented by analysing two dosimeter monitors irradiated at the inner RPV wall during two recent cycles of the selected BWR reactor. These first validation results indicate that despite the current modelling simplifications/approximations, a satisfactory accuracy is already achieved when comparing the calculation results to measured data, both in terms of dosimeter activities as well as FNF estimations. Although this hence provides a certain confidence in the developed BWR scheme, further sensitivity/optimisation studies are planned to be conducted along with an extension of the validation basis in order to achieve a comprehensive BWR FNF fluence scheme applicable to any location of interest for ageing or safety related aspects.

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