A. Artmann, G. Bruhn, S. Schneider and *E. Strub*

Simulation of the occupational radiation dose caused by contamination of primary circuit media in pressurized water reactors

The occupational radiation exposure of workers in NPPs during overall maintenance and refueling inspections and decommissioning is determined by numerous parameters. Radiation exposure caused by contamination of components may be minimised by the chemical operation mode and by applying systematic decontamination techniques. Data on occupational exposure in German NPPs as well as information about the radionuclide concentration in the coolant are available. The generic 3D model of the primary circuit presented is based on the analysis of technical documentation of German PWRs. Tasks are modeled as a combination of retention times at related local positions in the surroundings of work areas. The generic model allows the calculation of the resulting occupational doses generated by definable jobs and tasks. The KWU/Siemens-PWR generations are characterised by nuclide vectors, the thickness of shielding, and the material composition of components. It was possible to show that for a pre-Konvoi plant, the calculated occupational dose caused by a specific working task is close to measurements.

Simulation der beruflichen Strahlenexposition hervorgerufen durch Kontamination im Primärkreislaufmedium von Druckwasserreaktoren. *Die berufliche Strahlenexposition von Arbeitern in KKWn während Revisionen und bei der Stilllegung wird durch eine Vielzahl von Parametern beeinflusst. Die Strahlenexposition hervorgerufen durch Kontamination von Komponenten kann durch die chemische Fahrweise und durch die systematische Anwendung von Dekontaminationstechniken minimiert werden. Daten zur beruflichen Strahlenexposition in deutschen KKWn und Informationen über die Radionuklidkonzentrationen im Primärkühlmittel sind verfügbar. Das vorgestellte generische 3D-Modell eines Primärkreislaufs basiert auf der Auswertung technischer Dokumentationen von deutschen DWRn. „Tasks" werden als eine Kombination von Aufenthaltszeiten an den zugehörigen Aufenthaltsorten in der Umgebung des Einsatzortes modelliert. Das generische Modell erlaubt die Berechnung der erwarteten beruflichen Strahlendosis, die durch eine definierte Task oder einen Job hervorgerufen wird. Die KWU/Siemens-DWR-Generationen sind durch ihre Nuklidvektoren, die Dicke der Abschirmungen und die Materialzusammensetzung der Komponenten charakterisiert. Es konnte gezeigt werden, dass für ein Vor-Konvoi-Kraftwerk die berechnete berufliche Strahlendosis, die durch eine spezifizierte Task hervorgerufen wird, nahe bei gemessenen Werten liegt.*

1 Introduction

The occupational radiation exposure of workers in nuclear power plants (NPPs) during overall maintenance and refuelling inspections and decommissioning is not only determined by the activation and contamination of structural elements of the primary circuit, but also by a number of additional parameters, such as the geometry of shielding, self-shielding of components, deposits of radionuclides, the planning of working tasks within the controlled area, or even the behaviour of workers.

However, the actual radiation exposure can be attributed to two different sources: i) systems, structures or components (SSCs) that are activated by neutron irradiation or ii) SSCs that are contaminated by radioactive substances that have been produced elsewhere. The reduction of radiation exposure by directly neutron-activated SSCs can only be achieved by using appropriate shielding, but this is relevant mainly for work on components of the core structure or the pressure vessel of an NPP. Activated parts of the primary circuit can seldomly be removed or exchanged during the lifetime of an NPP. Therefore, a minimisation of radiation exposure attributed to neutron-activated SSCs can mainly be achieved in the design phase of an NPP. In contrast, there are many areas in an NPP where the main share of the occurring dose rates is caused by contamination. Hence, a major share of occupational doses is also directly linked to radiation exposure caused by the contamination of components. Thus, it is promising to minimise occupational doses by optimising the chemical operation mode of reactors as well as by the application of systematic decontamination techniques. Influencing the contamination within the primary circuit insofar offers the greater potential for improving radiation protection. The goal of this work was to quantitatively understand the relationship between contamination formation, its deposition, and occupational doses as a basis for further improvements of radiation protection. The number of parameters influencing occupational radiation exposure leads to a complex problem which in this work is addressed by a comprehensive generic model. The aim of the presented model is to deliver a quantitative, realistic estimation of the influence of contamination of the primary circuit of PWRs on the occupational doses of workers performing specific tasks.

2 A generic model for the influence of contamination on occupational doses

A generic model for the influence of contamination on occupational doses for specific jobs must comprise at least 4 elements that will be addressed in the following paragraphs:

1. Determination of the relevant representative nuclide vectors,
2. 3D modeling of the PWR primary circuit,
3. Definition of jobs (locations, retention times within 3D model)
4. Dose rate calculations.

The general procedure of modelling is schematically demonstrated in Fig. 1.

2.1 Determination of the relevant nuclide vectors

Technical literature of the past three decades was analysed to gather existing knowledge about the formation and deposition of contamination within the primary circuit. A broad overview of the relevant microscopic processes can be derived e. g. from Neeb [3], see Fig. 2. Although the important chemical processes in different locations in a PWR are generally well understood, the currently available approaches for the actual modelling of the water chemistry and the transport of radionuclides in the primary circuit usually turn out to be restricted. Several examples can be found in the literature [e. g. 1, 2], but in most cases the high complexity of chemical and thermodynamic processes within the primary circuit requires a large number of modelling parameters. On the other hand, there are only relatively few data on actual local dose rates available. This leads to the consequence that such simulations with many degrees of freedom and only a few measured data to fit generally exhibit increased systematic uncertainties. It is hence difficult to apply these models to simulate the processes such that precise results can be expected only from first principles. Therefore the resulting models tend to be rather specific to individual facilities – thus not generic – and do not lead to general conclusions. In [2] it is deduced that a step back to simpler models might be more successful.

Conclusively, the approach of the work presented here was not to model the relevant nuclide vectors from microscopic models. Rather, general scientific considerations were used as a basis to develop plausible nuclide vectors for different scenarios. Radiochemical and physical experiences and the evaluation of geometrical circumstances first lead to the choice of the *components* of three different nuclide vectors: (for a more detailed information on the construction of nuclide vectors, see [8]).

1. The nuclide vector for operation can be reduced to the nuclide ^{16}N alone. This nuclide is short-lived (half-life 7 s) and emits high energy gamma radiation. Its formation during operation cannot be avoided.
2. The nuclide vector for maintenance and refuelling outages contains a relatively large number of nuclides. Each of these nuclides might contribute a significant part to the contamination of the primary circuit: ^{51}Cr, ^{54}Mn, ^{59}Fe, ^{58,60}Co, ^{124}Sb (activated metals/metal oxides) and ^{131,133}I, ^{133}Xe, ^{134,137}Cs (fission products)
3. Finally, the nuclide vector for decommissioning is commonly determined by a few longer-lived nuclides: ^{60}Co, $^{110\,m}$Ag, ^{124}Sb

To determine the *relative contributions* of the components to the nuclide vectors, data on the concentration of radionuclides dissolved in the primary coolant of German NPPs of the past 15 years were analysed (Fig. 3 and 4). For the quantitative determination of nuclide vectors to describe contamination caused by deposits, only metal oxides are considered as the main dose contributors [3]. The ranking order was based on the radiological impact of each radionuclide. Finally, for a specific NPP, these nuclide vectors were scaled using local dose rate measurements, i. e. a reverse simulation from known local dose rates. Such values can be obtained from the ISOE database [5] for a limited number of relevant locations in an NPP.

There are extensive data sets of actually measured local dose rates, but they are limited to a few working positions (steam generator water chambers and reactor coolant pump outside hot/cold) and points in time (overall maintenance and refuelling inspection). Additional measured data have been collected by concrete enquiries, e. g. during visits to facilities.

Although the resulting nuclide vectors are highly dependent on the individual NPP, it turned out that mainly the ^{60}Co content has to be adjusted to adapt the nuclide vectors to a specific NPP or a specific NPP generation. The weighting of the other components of the nuclide vector has only a minor influence on the final simulation results.

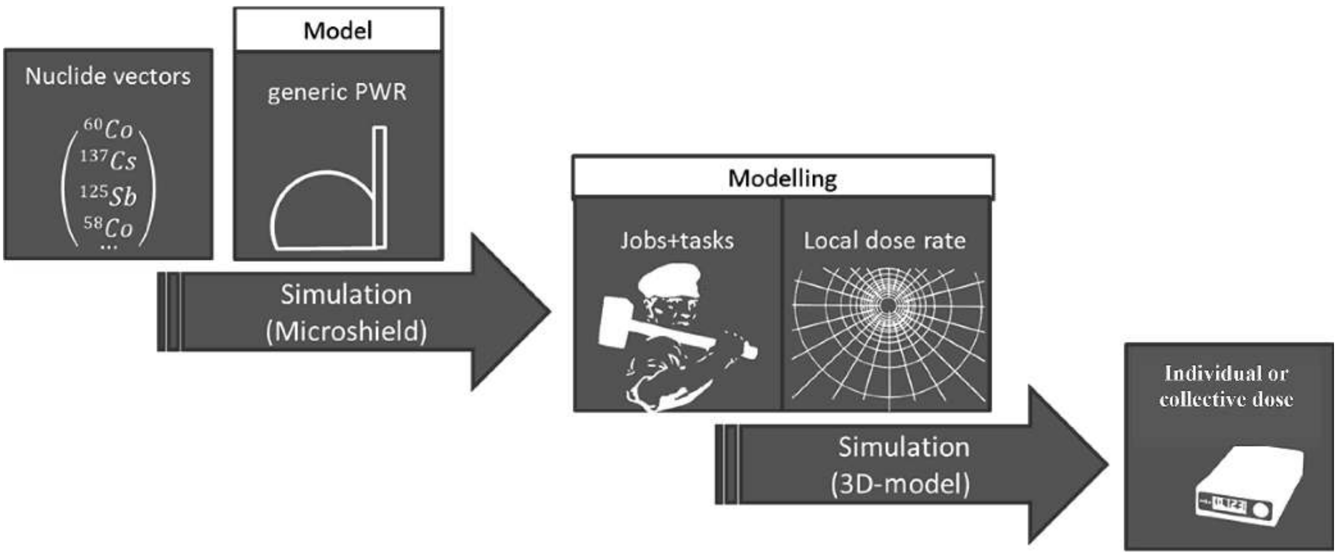

Fig. 1. Scheme of the method for simulating job-related doses in nuclear power plants with pressurised water reactors

2.2 The 3D model

The generic 3D model of the primary circuit presented here is based on the analysis of technical documentation of German nuclear power plants. With the aid of engineering drawings of KWU/Siemens NPPs with pressurised water reactors of the 1 200 MW+ power class, a 3D CAD model was constructed (Fig. 6) with SketchUp [6]. This model contains all relevant components from the point of view of radiation protection. With it, the spatial relations during the performance of working tasks can be well illustrated and investigated. Within this model it can be decided which SSCs act as relevant radiation sources for any working point and in what cases shielding from which components of the primary circuit has to be considered. Within the model, the NPP is represented as a number of elementary radiation sources (mainly cylinders) and elementary shieldings (cylinders and cuboids). Each of these construction elements is defined such that it can be easily fit to a single MicroShield [4] calculation (see below).

Due to geometrical limitations in MicroShield, it was necessary to simplify components within the model to simple geometrical forms, like cylinders (hollow or full). These simplifications followed some rules. The main priority was to keep the radiological impact of a source, then the outer geometrical dimensions. A very low priority was the original inner structure of a component and details of the component form. An example of the modelling of a steam generator as a set of sources and shielding is shown in Fig. 5. In this simplification, all shapes used are cylinders. The top, secondary circuit part is simplified to a shielding-only cylinder. The heat exchange part is a fictive, yet radiologically exact material mix and is a source with an outer shielding. The inner structure of the heat exchanger is not modelled in detail but simplified

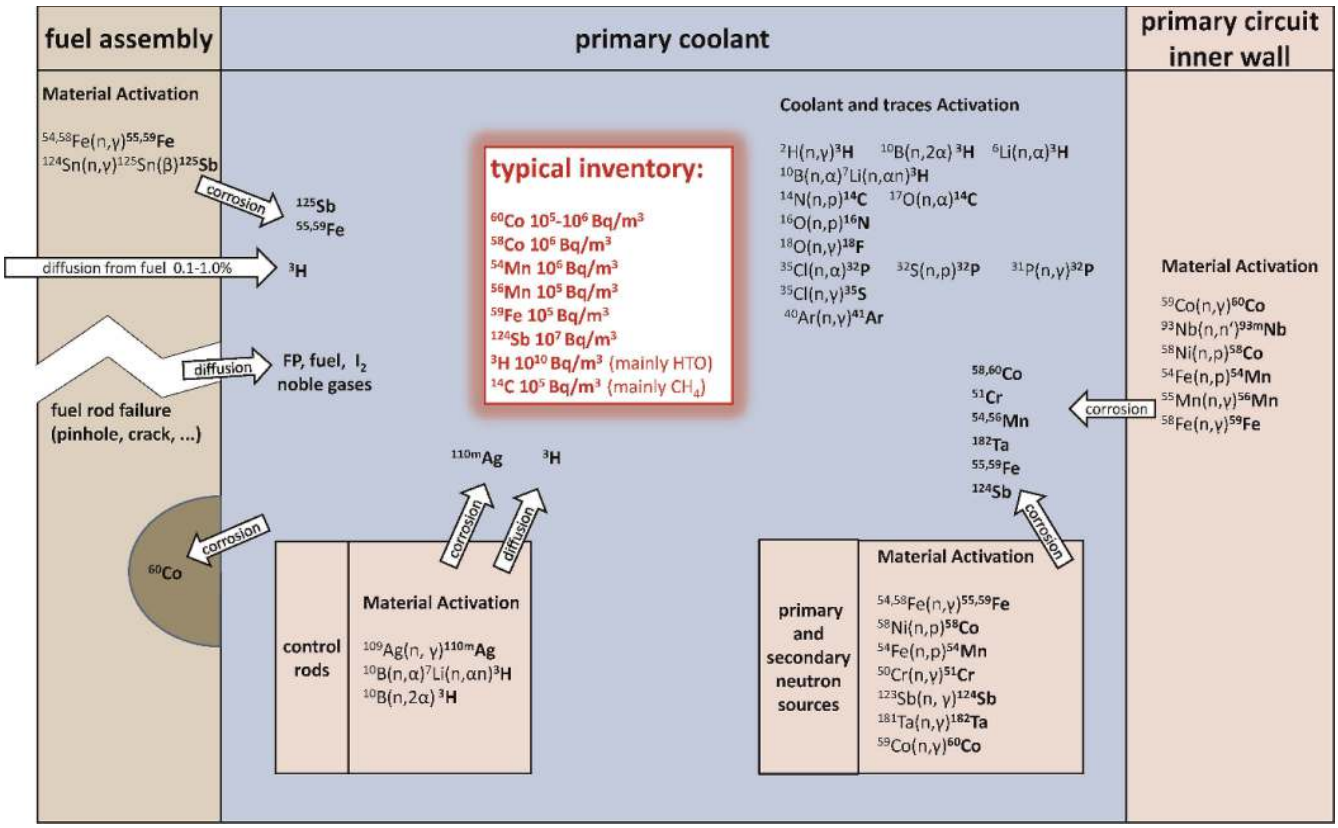

Fig. 2. Physico-chemical processes, resulting concentrations of radionuclides in PWR primary loops (adapted from [3]). The typical inventory comprises only the radiologically most important nuclides during revision assuming of no fuel defects

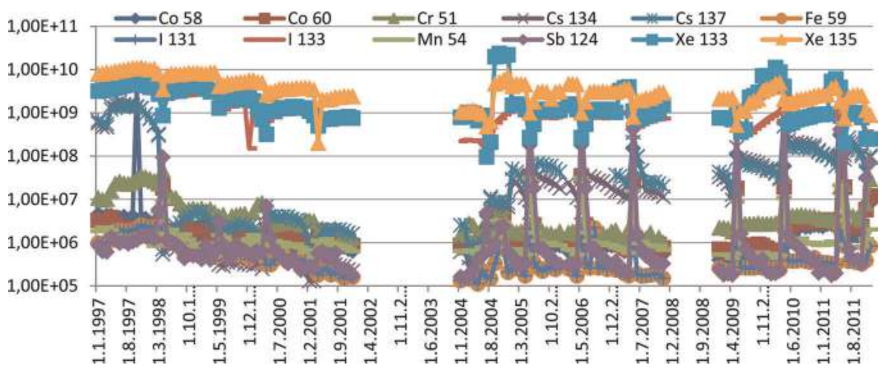

Fig. 3. Nuclide concentration of a 2nd Generation KWU/Siemens PWR in Bq/m³

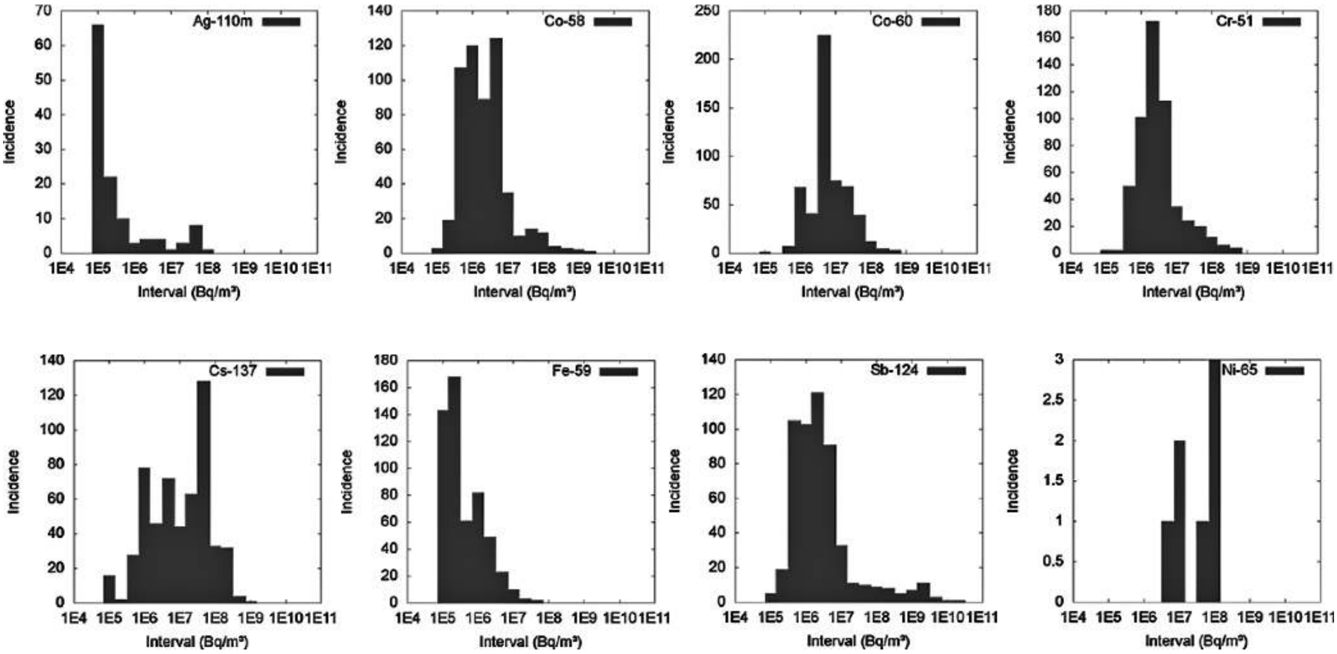

Fig. 4. Nuclide distributions for all measurements of the respective nuclides of a 2nd Generation KWU/Siemens PWRs

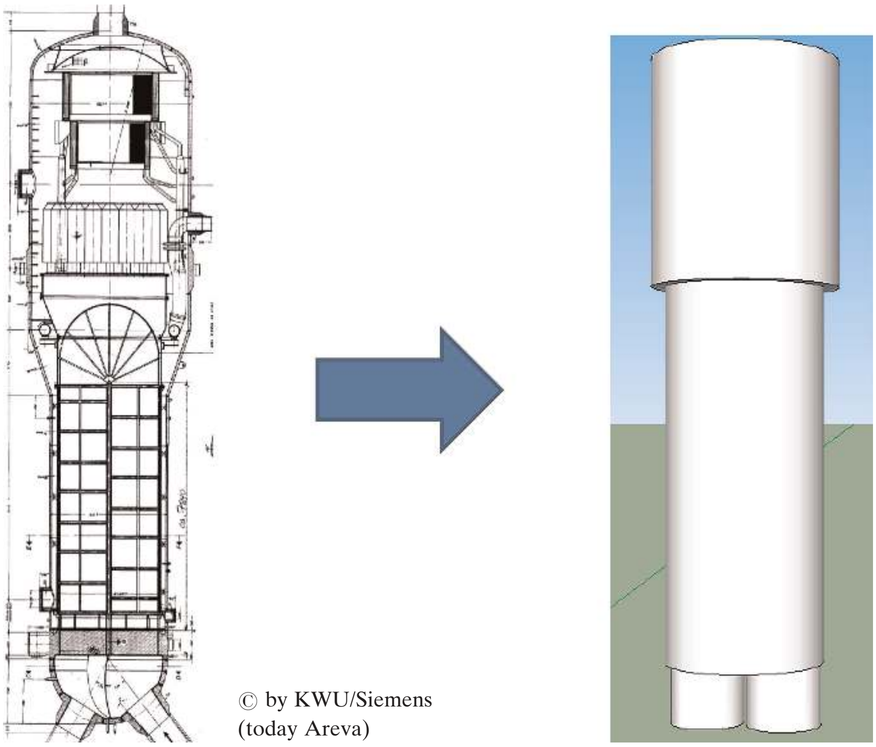

Fig. 5. Schematic (left) and simplified steam generator (right). The secondary part is modelled as shielding only, the heat exchange part is modelled as shielding and source, and the inlet and outlet are modelled as both sources and shielding

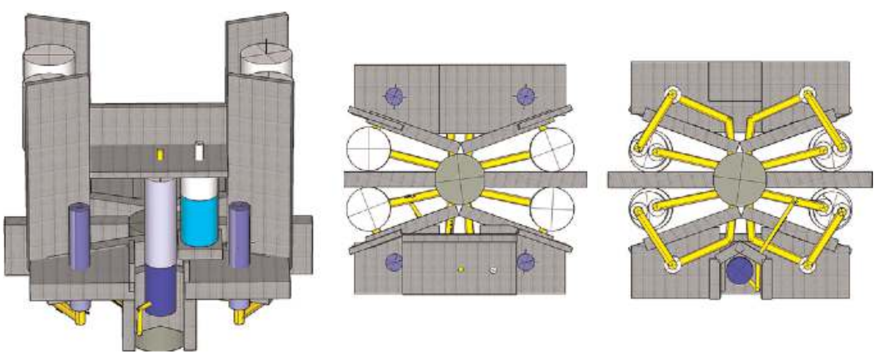

Fig. 6. Views of the 3D CAD model of the primary circuit with shielding walls (left to right: front, top and bottom view) [6]. The variant shown is based on the KWU/Siemens Konvoi design

as a virtual composite material (consisting of a mixture of metal (Incoloy 800), water and steam). The inlet and outlet from the primary circuit are split into two cylinders and act as sources with shielding.

2.3 Definition of jobs

Working tasks are modelled as a combination of retention times and related local positions in the surroundings of work areas, so that mean occupational doses can be determined. The geometrical coordinates of relevant local positions are mathematically transformed in such a way that they are suitable as input parameters for the MicroShield software used in the final simulation step [4]. Using this, the contribution of any subcomponent and subsystem to the local dose rate is calculated and summarised afterwards.

Planning of jobs in NPPs is done by the local radiation protection department. It is a combination of measurements of local dose rates, detailed planning of the tasks, and experience with similar jobs. With that, a specific duration is attributed to each task.

To do something in analogy for the calculation of the mean personal dose for a common worker, each job/working task/craft is broken down into several (usually three) specific local points. Each point is attributed to a different dose rate: One point is chosen near to the working object, and there is usually a high dose rate. A second point is chosen at some distance from the working object and is attributed to the time of moving from a shielded location (e. g. where tools and spare parts may be stored). The third point is usually chosen as a shielded point and stands for everything else a common worker has to do in a controlled area, e. g. change clothes, take breaks, stay on pathways, and so on (see Fig. 7).

After definition of all points for each job/working task, all not-negligible sources around each point need to be identified, as does all the relevant shielding. With that, it is possible to calculate the local dose rates at each point. This has to be done taking several simulation steps, one for each source, and afterwards by superposition of all calculated doses at a specific point.

Another important part for the final definition of a job/working task is to define a realistic retention time of the average worker at each of the defined local points. Starting point is the data from the ISOE [5], where for each task the working time and the number of personnel is listed. The jobs are broken down by time and personnel. From that, a mean value for the working time of one average worker for a specific task/job is derived. The distribution of the (individual) overall working times taken from the ISOE database for the designated retention points was evaluated and estimated by interviewing staff of the radiation protection sections of German NPPs.

With knowledge of dose rates at the specific local points and the respective retention times. a calculation of the occupational dose is possible.

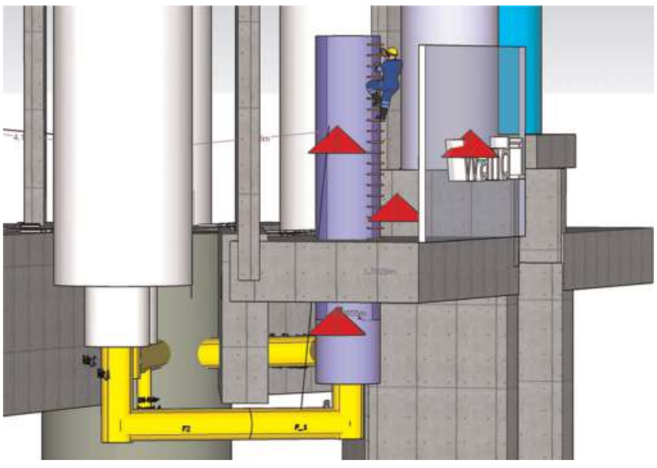

Fig. 7. Specification of representative points (tips of red pyramids) in the neighbourhood of the reactor coolant pump [7]

2.4 Dose rate calculations

The NPP generations 2, 3 and 4 of KWU/Siemens PWRs are characterised in MicroShield [4] by diverse representative nuclide vectors (especially by differences in the fraction of ^{60}Co, see above), the thickness of available shielding, and the material composition of the components.

The distribution of contamination of about $1 - 2 \cdot 10^{14}$ Bq to the components of the primary circuit is done on one hand based on reverse simulation, starting from known local dose rates, and on the other hand based on geometrical and physical considerations. An example of the latter is to consider the ratio of the inner area to the volume, which would be much greater for steam generator tubes than for the pressuriser, for example, so that a higher activity concentration (Bq/m^3) can be expected.

3 Implementation and results

The generic model allows the calculation of the resulting occupational doses generated by definable jobs and working tasks. One example of a typical and recurring job is related to the reactor coolant pumps. Figure 6 shows the associated locations for local dose rate measurement (tips of the red pyramids). For each of the four locations, the components within the direct surroundings have been considered as sources (surface contamination), including those behind shielding walls. The slightly outlying point in Fig. 7 (near wall “Wand 1”) is placed behind a wall partly blanked out here, representing paths, breaks, and time spent in the changing rooms. Simulations were performed considering deposited contamination, specified for each component of a system without primary coolant.

The highest local dose rate is generated in the lower region where the relative retention time is 2 – 10 %.

It could be shown that assuming a nuclide vector and shield arrangement of a KWU/Siemens pre-Konvoi plant, the resulting occupational dose caused by working tasks related to the reactor coolant pump is close to measured data. The calculated dose rates for different jobs are within the range of reported data from the ISOE database (see Fig. 8 and Tab. 1). This shows the validity of this generic model approach in this case.

4 Discussion

It could be shown that the described generic NPP model allows the prediction of expected individual and collective doses. Plausible nuclide vectors for different scenarios have been developed and have been scaled by inverse simulation to a restricted number of available dose rate measurements for specific NPPs. Now, with these nuclide vectors, individual and collective doses can be estimated reliably for given jobs and tasks on the basis of relatively simple assumptions: For

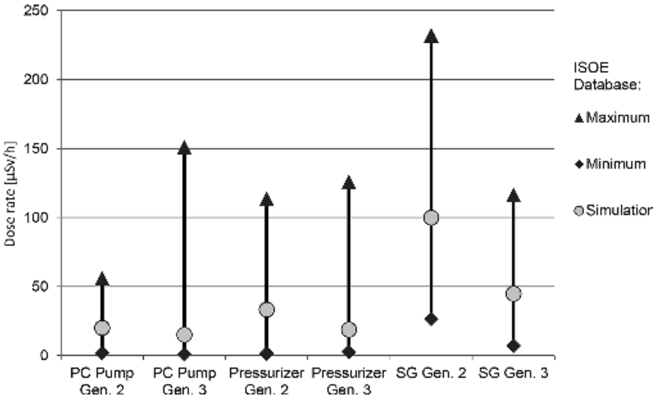

Fig. 8. Comparison between data from ISOE database (bandwidth) and calculated dose rates for specific jobs in KWU/Siemens pre-Konvoi plants

the typical duration of a job/task set, the presence of a worker is attributed to a small set of locations (normally 3 to 4) within a generic 3D model. Then, the dose is calculated by integrating relevant time intervals. The resulting values have been compared with Job/Task dose data from the ISOE database, showing good agreement between the real data and the simulated data.

- The described model is based on empirical data from German NPPs, but can be easily adapted to other 4-loop PWR reactor types. Such an adaptation can be carried out easily and comprises several steps: If necessary, nuclide vectors can be changed. This can be done either by adopting a measured nuclide vector (if available), but experience shows that in many cases already a scaling of the ^{60}Co component will yield realistic simulation results in the end.
- Within the 3D-model, the material composition and thickness of shielding have to be adapted.
- Changing of job situation (time-shares and retention times) or new jobs can be created.

In the future, the model will be applied to KWU/Siemens generation 4 NPPs (Konvoi NPPs). Since Konvoi NPPs are designed to build up less ^{60}Co activity during operation, it might turn out that the adaptation of nuclide vectors is not as straightforward as in many other cases. Another possible application of the described model is the simulation of the influence of hot spots. For example, it would be possible to estimate the additional resulting doses from a certain hot spot on the work at surrounding SSCs. This could drive an evaluation of which effort would be reasonable to mitigate the consequences of such a hot spot.

(Received on 29 February 2016)

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The authors of this contribution

Dr. *Andreas Artmann* (corresponding author),
Dr. *Gerd Bruhn*, Dr. *Sebastian Schneider*
E-Mail: Andreas.Artmann@grs.de
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH
Schwertnergasse 1
50667 Cologne, Germany

PD Dr. *Erik Strub*
Abteilung Nuklearchemie der Universität zu Köln
Zülpicher Str. 45
50674 Cologne, Germany

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DOI 10.3139/124.110728
KERNTECHNIK
81 (2016) 5; page 553–558

ISSN 0932-3902

Table 1. Simulated effective doses of jobs related to coolant pumps in comparison with measured values of KWU/Siemens pre-Konvoi plants

Item	Simulation result	Range of plant mean values	Range of measured single values
Individual mean dose Gen 2	174 µSv	194–365 µSv	2–924 µSv
Collective dose per Gen 2 per pump	8.7 man mSv	7–18 man mSv	7–56 man mSv
Individual mean dose Gen 3	73 µSv	85–301 µSv	2.5–637
Collective dose per Gen 3 per pump	4.6 man mSv	1.8–16.8 man mSv	0.36–65 man mSv