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Modeling of Pool Critical Assembly Pressure Vessel Facility Benchmark

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ABSTRACT

A Pool Critical Assembly Pressure Vessel Facility Benchmark (PCA benchmark) has been chosen for qualification of our methodology for pressure vessel neutron fluence calculations, as required by the U.S. Nuclear Regulatory Commission regulatory guide. The SCALE (Standardized Computer Analysis for Licensing Evaluation) code package, developed at Oak Ridge National Laboratory, was used for modelling of the PCA benchmark. The CSAS6 sequence of the SCALE code package, which includes KENO-VI Monte Carlo code, was used for calculation of equivalent fission fluxes. The calculations were performed using three different models for unit cell of PCA benchmark facility core. The comparison of calculational results and benchmark data showed a good agreement of the calculated and measured equivalent fission fluxes. No systematic decrease of agreements between calculations and measurements with increasing distance of detector from the PCA benchmark facility core was observed for any of unit cell models.

1 INTRODUCTION

Calculational methods for determining the fast neutron fluence are necessary to estimate the fracture toughness of the pressure vessel materials. A Pool Critical Assembly Pressure Vessel Facility Benchmark (PCA benchmark) [1] has been chosen for qualification of our methodology for pressure vessel neutron fluence calculations, as required by the U.S. Nuclear Regulatory Commission regulatory guide [2]. The scope of PCA benchmark is to validate the capabilities of the calculational methodologies to predict the reaction rates in the region outside the core when the neutron source, material compositions, and relatively simple geometry configuration are well defined and given. The PCA benchmark provides measured reaction rates inside the simulated pressure vessel, as well as in the water gap in front of the pressure vessel. This allows an assessment of the accuracy with which the calculations predict the neutron flux attenuation inside the pressure vessel. The PCA benchmark is described in Section 2.

The SCALE (Standardized Computer Analysis for Licensing Evaluation) [3] code package, developed at Oak Ridge National Laboratory, was used for modelling of the PCA benchmark. The description of the SCALE code package is given in Section 3. The analysis of the PCA benchmark includes calculational model, results of calculation, and comparison of calculated values with benchmark data. The analysis of the PCA benchmark is presented in Section 4. Conclusions are given in Section 5.

2 PCA BENCHMARK FACILITY DESCRIPTION

The PCA benchmark facility consists of the reactor core and the components that mock up the reactor–to-cavity region in light water reactors. These components are the thermal shield (TS), the reactor pressure vessel simulator (RPVS) and the void box (VB), which simulates the reactor cavity. An overall view of the PCA benchmark facility is shown in Figure 1. An aluminium plate, referred to in Fig.1 as the reactor window simulator, was added to the facility for operational reasons. The thicknesses of the water gaps between the aluminium window and thermal shield, and between the thermal shield and pressure vessel, are approximately 12 cm and 13 cm, respectively. The materials used for the components outside the core were aluminium for the reactor window simulator, stainless steel for the thermal shield, and carbon steel for the pressure vessel. The facility is located in a large pool of water, which serves as reactor core coolant and moderator and provides shielding. The PCA benchmark facility core is a light water moderated, enriched uranium fueled critical assembly. It consists of 25 material test reactor (MTR) plate type elements.

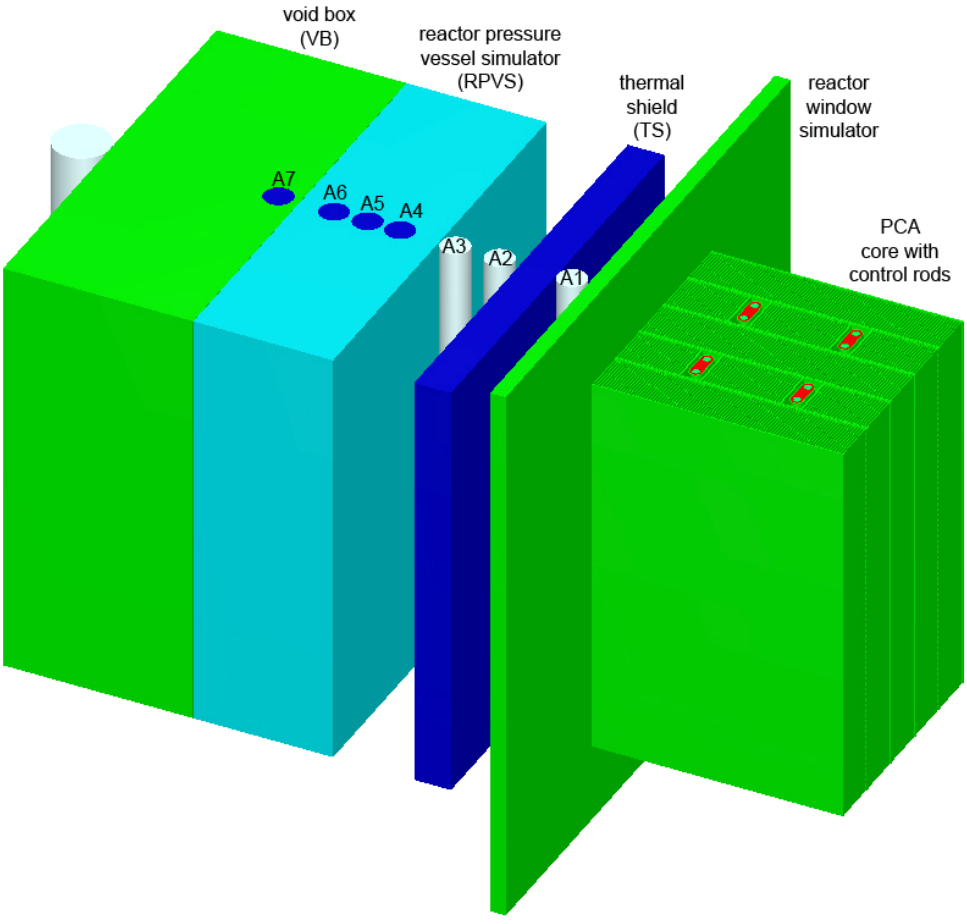

Figure 1: PCA benchmark facility

All measured quantities provided for comparison with calculated values are given per unit PCA benchmark facility core neutron source. Therefore, the calculated responses need to be normalized to the source strength of one fission neutron per second being born in the whole PCA core. Measurements were performed at core midplane at several locations, labelled in Fig. 1. as A1 to A7. Measured quantities, used in PCA benchmark, are given in terms of the equivalent ^{235}U fission fluxes. To complete the PCA benchmark analysis the analyst must determine the calculated to measured ratios of the equivalent ^{235}U fission fluxes for all the locations and all the dosimeters for which the measured values are provided.

3 SCALE CODE PACKAGE

The SCALE code system was developed for the U.S. Nuclear Regulatory Commission to satisfy a need for a standardized method of analysis for the evaluation of nuclear facilities and package designs. In its present form, the system has the capability to perform criticality, shielding, radiation source term, spent fuel depletion/decay, reactor physics, and sensitivity analyses using well established functional modules tailored to the SCALE system. The SCALE system consists of easy to use analytical sequences which are automated to perform the necessary data processing and manipulation of well established computer codes required by the sequence. The user is able to select an analytical sequence characterized by the type of analysis to be performed and the geometric complexity of the system being analyzed. The user then prepares a single set of input for the control module corresponding to this analytical sequence. The control module input is in terms of easily visualized engineering parameters specified in a simplified, free form format. The control modules use this information to derive additional parameters and prepare the input for each of the functional modules in the analytical sequence. A variety of control modules (such as CSAS) have been developed to automate various analytic sequences.

3.1 The CSAS6 sequence

The Criticality Safety Analysis Sequence with KENO-VI (CSAS6) was developed within the SCALE code system to provide automated, problem-dependent, cross-section processing followed by calculation of the neutron multiplication factor for the system being modelled using KENO-VI.

SCALE includes two versions of KENO: KENO-V.a and KENO-VI. Both versions are 3D multi-group Monte Carlo codes for determination of the k_{eff} of multidimensional systems. The same systems are presented in SCALE by geometry models which consist of units that are constructed by nesting cylinders, rectangular cuboids and spheres. These units can be combined in rectangular arrays in other units.

KENO-VI was introduced in 1995 (SCALE4.3) which has more general geometry description known as SGGP (SCALE Generalized Geometry Package) for constructing complex models. Also, it has much larger assortment of elementary bodies than KENO-V.a.

The flowchart of the CSAS6 sequence is given in Figure 2.

At the centre of the CSAS6 is the library of subroutines referred to as Material Information Processor Library (MIPLIB). The purpose of MIPLIB is to allow user to specify problem-dependent materials, mixtures, elements and nuclides provided in the Standard Composition Library. MIPLIB also provides specification of geometry and prepares input for modules that perform the problem-dependent multi-group cross section processing: BONAMI, NITAWL, CENTRM/PMC and XSDRNPM.

The CSAS6 sequence activates the cross section processing code BONAMI, to provide resonance corrected cross sections in the unresolved resonance range, and either: NITAWL or

WORKER, CENTRM and PMC, to provide resonance corrected cross sections in the resolved resonance range. For the unit cell modelling XSDRNPM is utilized to provide a cell weighted mixture cross sections for use in KENO-VI. KENO-VI uses the processed cross sections and calculates the effective multiplication factor of the three dimensional system models.

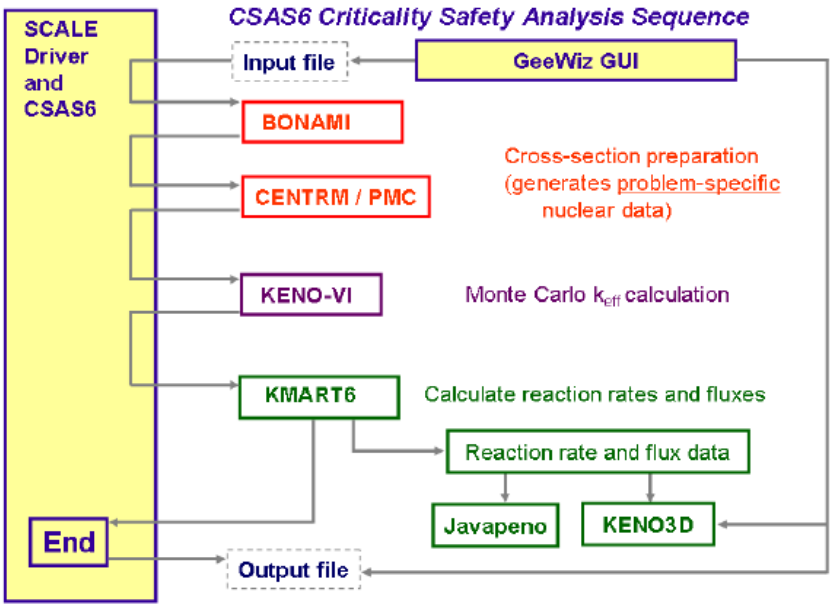

Figure 2**:** Flowchart of the CSAS6 sequence

3.2 Processors for problem-dependent multi-group cross sections generation

BONAMI performs resonance shielding through the application of the Bondarenko shielding factor method. It is typically used to process data in the unresolved resonance energy range.

NITAWL uses the Nordheim Integral Treatment to process neutron cross sections in the resolved energy range. It implements a fine-energy-group calculation for the slowing down flux across each resonance with flux weighting. This processor is compatible only with ENDF/B-V library.

CENTRM computes continuous energy neutron spectra using 1D discrete ordinates or infinite media geometry. It determines the problem-specific fluxes for processing resonance shielded multi-group data using 1D unit cell. PMC then reads the CENTRM continuous energy flux spectra and cross section data for calculation of problem-dependent group-averaged cross section.

XSDRNPM is a 1D S_n transport code for performing neutron or coupled neutron-gamma calculations. It has a variety of uses in SCALE. In CSAS6 sequence it is optionally used for preparation of cell-averaged cross sections for subsequent system analysis in KENO-VI.

3.3 Cross Section Libraries

There are several multi-group cross section libraries distributed with SCALE for criticality safety analyses. The 238-group ENDF/B-V library (238GROUPNDF5 or V5-238) contains data for all ENDF/B-V nuclides and has 148 fast and 90 thermal groups. The 44-group ENDF/B-V library (44GROUPNDF5 or V5-44) is a broad-group version of V5-238 designed for analysis of LWR fresh and spent fuel systems. Most late and complete are 238-group ENDF/B-VI (V6-238) and ENDF/B-VII (V7-238) libraries, with the same group structure as the ENDF/B-V version. Continuous energy cross section libraries based on both V6-238 and V7-238 are also available.

The V7-238 libraries (multi-group and continuous) are recommended, because they are based on the latest nuclear data.

When using multi-group libraries, the CSAS6 first accesses BONAMI to process nuclides with Bondarenko data. The master library output from BONAMI is input to NITAWL, where resonance data are processed via the Nordheim Integral Treatment, or to CENTRM which performs 1D continues energy calculation that is subsequently collapsed to multi-group cross sections by PMC. In either case, the result is an AMPX working-format library that is used by KENO-VI.

4 ANALYSIS OF THE PCA BENCHMARK

The analysis of the PCA benchmark has been performed using the SCALE code package. The calculational model of the PCA benchmark within CSAS6 sequence of SCALE has been determined. The results of calculations using the established model have been compared with PCA benchmark data.

4.1 Calculational model

The CSAS6 sequence of SCALE code package has been used for PCA benchmark modelling. Three different unit cell models were used for PCA benchmark facility core:

- INFHOMMEDIUM – best suited for large masses of materials in which the size of each material is large compared to the average mean free path so that it can be treated as infinite homogeneous medium.
- LATTICECELL – best suited for large arrays of fuel in slabs, pins/rods or spherical pellets. The resonances are treated differently, depending on which cross section processor is used. The isotopes comprising the mixtures specified in the cell are collapsed using point fluxes. The cell is assumed to have white boundary conditions to simulate an infinite lattice.
- MULTIREGION – is appropriate for geometric regions in which the geometry effects are important, so generalized (more flexible) description is needed. All regions of the cell require the same geometric shape. Lattice arrangements can be approximated by specifying a white boundary condition on the outer boundary.

Unit cell data applies only to the use of the multi-group cross section libraries for modelling of a heterogeneous system where resonance self-shielding is important. The MIP (Material Information Processor) is used primarily for creating the problem-dependent libraries required by the CSAS6. This is important for large regular lattices of slabs as in PCA benchmark facility core. The MIP implements 1D unit cell description to provide information about the resonance self-shielding corrections and the Dancoff corrections that are applied to the cross sections to create a working-format library.

For all three cases we used the latest V7-238 cross section library with CENTRM (SCALE default) cross section processor.

4.2 Results of calculation and comparison with benchmark data

The CSAS6/KENO-VI criticality calculations of the PCA benchmark facility were performed for 650 batches (i.e. neutron generations) with first 150 batches skipped in order for the fission source distribution to converge and with 2000 neutrons per batch. The results obtained for effective multiplication factors of PCA benchmark facility core are:

- k_{eff} = (0,99991±0,00090) for infinite homogeneous medium model,
- k_{eff} = (0,99989±0,00099) for lattice cell model,
- k_{eff} = (1,00022±0,00084) for multi-region model.

For all three cases the calculated k_{eff} satisfies the Pearson's chi-square (χ^2) test for normality at the 95 % level.

The equivalent fission fluxes calculated by CSAS6/KENO-VI for all seven PCA benchmark detector positions and three different unit cell models are given in Table 1. The experimental values of equivalent fission fluxes for all seven PCA benchmark detector positions are also given in Table 1 (column 3). The Table 1 also includes in column 4 the equivalent fission fluxes for all seven PCA benchmark detector positions obtained by transport calculations using the DORT computer code [1]. The calculational results for PCA benchmark obtained by CSAS6 sequence are compared with the experimental values for all seven PCA benchmark detector positions and relative differences for equivalent fission fluxes of all three unit cell models are given in Table 2.

A good agreement of the calculated and measured equivalent fission fluxes has been obtained. No systematic decrease of agreements between calculations and measurements with increasing distance of detector from the PCA benchmark facility core was observed for any of unit cell models.

The PCA benchmark equivalent fission fluxes versus detector position obtained by experiment and three different unit cell models are depicted in Figure 3. Relative differences of calculated and measured PCA benchmark equivalent fission fluxes versus detector position for three different unit cell models are given in Figure 4.

The equivalent fission fluxes in Table 1 and Fig. 3 are given in unit of cm^{-2} as calculated neutron fluxes are normalized to the source strength of one fission neutron per second being born in the whole PCA benchmark facility core.

Table 1: Experimental and calculational results for PCA benchmark equivalent fission fluxes

Detector	Position (cm)	PCA bench. experiment	transport DORT code	CSAS6 (inf_hom)	CSAS6 (lattice_cell)	CSAS6 (multiregion)
A1	12.0	$6.30\cdot10^{-6}$	$5.77\cdot10^{-6}$	$4.85\cdot10^{-6}$	$4.71\cdot10^{-6}$	$4.84\cdot10^{-6}$
A2	23.8	$7.48\cdot10^{-7}$	$6.90\cdot10^{-7}$	$6.36\cdot10^{-7}$	$5.89\cdot10^{-7}$	$6.69\cdot10^{-7}$
A3	29.7	$2.76\cdot10^{-7}$	$2.60\cdot10^{-7}$	$1.24\cdot10^{-7}$	$1.52\cdot10^{-7}$	$1.58\cdot10^{-7}$
A4	39.5	$7.42\cdot10^{-8}$	$7.06\cdot10^{-8}$	$7.01\cdot10^{-8}$	$9.82\cdot10^{-8}$	$9.42\cdot10^{-8}$
A5	44.7	$3.54\cdot10^{-8}$	$3.27\cdot10^{-8}$	$5.53\cdot10^{-8}$	$5.52\cdot10^{-8}$	$5.29\cdot10^{-8}$
A6	50.1	$1.58\cdot10^{-8}$	$1.43\cdot10^{-8}$	$1.56\cdot10^{-8}$	$2.88\cdot10^{-8}$	$3.39\cdot10^{-8}$
A7	59.1	$7.25\cdot10^{-9}$	$6.09\cdot10^{-9}$	$1.04\cdot10^{-9}$	$8.39\cdot10^{-9}$	$5.17\cdot10^{-9}$

Table 2: Relative differences for equivalent fission fluxes of all three unit cell models

Detector	Position (cm)	Relative diff. (%) (inf_hom)	Relative diff. (%) (lattice_cell)	Relative diff. (%) (multiregion)
A1	12.0	-22.99	-25.21	-23.15
A2	23.8	-14.97	-21.26	-10.56
A3	29.7	-55.07	-44.93	-42.75
A4	39.5	-5.46	32.43	27.04
A5	44.7	56.39	56.11	49.60
A6	50.1	-1.08	82.63	114.97
A7	59.1	43.31	15.72	-28.69

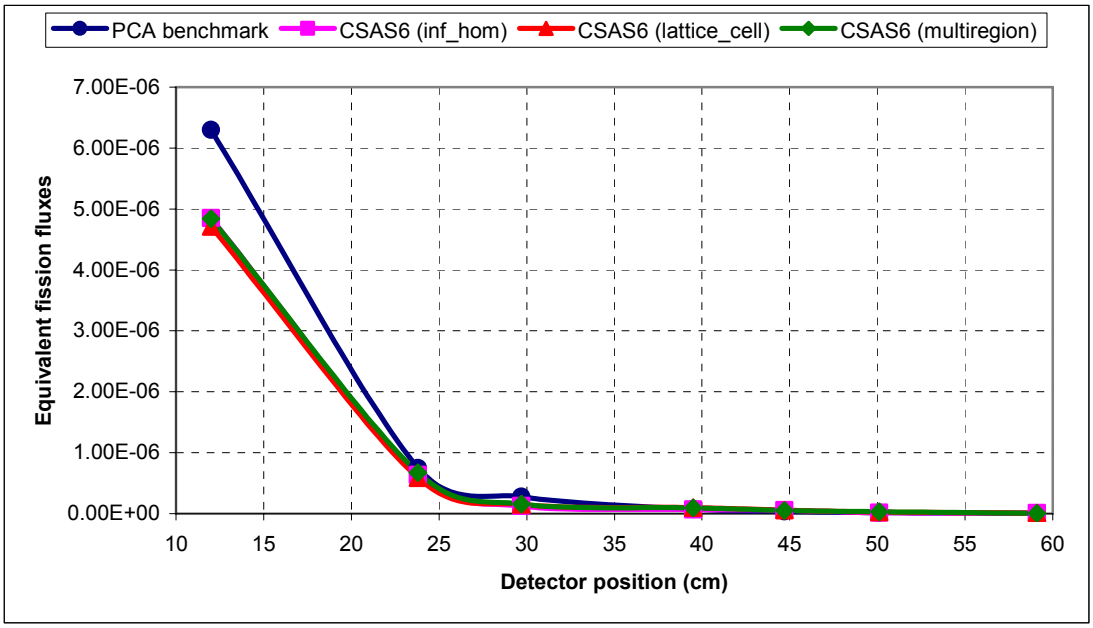

Figure 3: PCA benchmark equivalent fission fluxes versus detector position obtained by experiment and three different unit cell models calculations

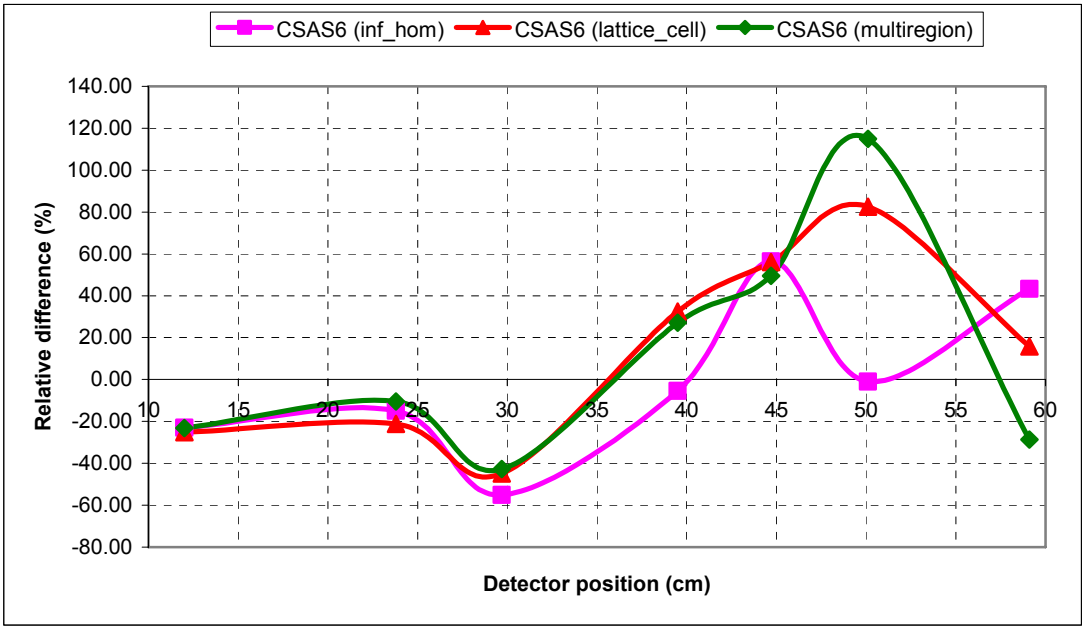

Figure 4**:** Relative differences of calculated and measured PCA benchmark equivalent fission fluxes versus detector position for three different unit cell models

5 CONCLUSIONS

The analysis of the PCA benchmark has been performed using the SCALE code package. The calculational model of the PCA benchmark within CSAS6 sequence of SCALE has been determined. The results of calculations using the established model have been compared with PCA benchmark data.

A good agreement of the calculated and measured equivalent fission fluxes has been obtained. No systematic decrease of agreements between calculations and measurements with increasing distance of detector from the PCA benchmark facility core was observed for any of unit cell models.

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