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Fusion power plant performance analysis using the HERCULES code

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Abstract

The HERCULES code was developed to assist in the safety and environmental analysis tasks of the European Power Plant Conceptual Study (PPCS). Its capabilities, however, extend beyond this kind of assessments. Coupling of the different computational tools and nuclear databases to the same geometry and material compositions, automation of input file generation and post-processing subroutines enable self-consistent calculations and radial–poloidal analyses of several parameters of interest in fusion technology design, optimising time and computer resources. Results include 2D variations of the neutron flux distribution and energy spectra, material activation and related quantities, wall loading, nuclear primary heating, tritium and helium production and radiation damage rates to materials. The capabilities of the neutron transport module of HERCULES are presented here and benchmarked on the PPCS models. Results are consistent with values assumed in the design process or obtained through the more specific PPCS analyses: tritium generation ratios are in the range between 1.12 and 1.36, energy multiplication between 1.16 and 1.39, and average wall loads between 1.90 and 2.54 MWm^{-2}. Issues related to tritium breeding and possible tungsten first wall coating of the near-term models are discussed.

Keywords: Power plant; Neutron transport; Radiation damage; Tritium breeding; PPCS

1. Introduction

The PPCS analysed the economic, safety, environmental and engineering performance of four power plant models based on tokamak burning plasmas, spanning a range of physics assumptions and European blanket and divertor technologies [1]. Compared to previous studies, [2], it prioritised design detail and integration of the different analyses. A fifth model is currently under analysis.

For the safety and environmental assessments several extensive computational codes and nuclear databases are used. In particular, for the neutron transport and activation studies 3D models of each PPCS plant had to be set up. Coupling and integration of MCNP [3], FISPACT [4] and the EAF nuclear libraries

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[5] was achieved by means of the HERCULES code system [6].

The capabilities of HERCULES, however, have applications beyond safety analyses. The neutron transport module has been optimised and tested on the PPCS plant models: presentation of those capabilities, together with a sample of the results obtained for the PPCS plants, is made here.

2. HERCULES neutron transport module

HERCULES couples the computational tools used in neutron transport and material activation analyses, enabling the integration of bounding accident, radioactive waste and nuclear assessments to the same geometry and material compositions, quickly and with minimum effort. The code runs a pre-processing facility where the geometry, neutron source and material compositions are defined, and an MCNP input file is produced for neutron–photon transport analysis. Following MCNP calculations, a post-processing module collates data for every cell in the model. Neutron and photon spectra are extracted alongside tritium generation, energy deposition and FW surface flux tallies. Post-processing produces data such as wall loading, primary heat, tritium breeding ratio (TBR), helium production and radiation damage rates to materials. Neutron spectra results can be automatically exported to FISPACT for the calculation of nuclide inventories, activation and derived quantities, including decay heat, gamma dose rates and clearance indices. Finally, the code can produce a command file for a finite elements package to perform transient thermal analysis of bounding, decay-heat driven accidents.

2.1. Pre-processing

Fig. 1 shows an example poloidal cross-section of the machine geometry. This is parametrically described, [7], following the outermost plasma contour and allowing for the definition of radial layers and poloidal sectors. One sector within one layer defines a cell, each cell being filled with a homogeneous mixture of the materials in the layer. A different material mixture and layering can accommodate the divertor region. The cryostat and central solenoid are also included. The neutron source is a 14.1 MeV-centred,

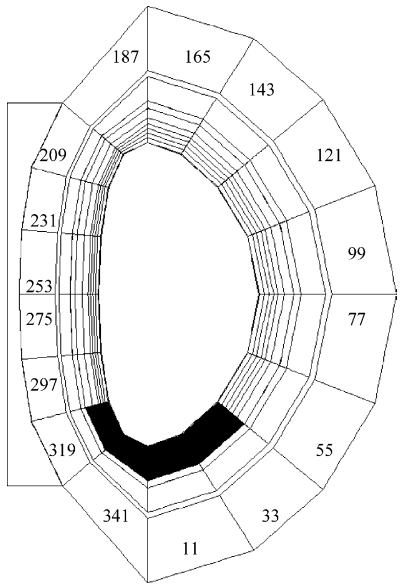

Fig. 1. Example poloidal cross-section of the MCNP models generated with the HERCULES system, showing the poloidal angle segmentation. Dark cells correspond to the divertor.

isotropic, Gaussian energy distribution curve, its geometry being again parametrically specified. The width is controlled by the ion temperature, and the radial intensity by a peaking factor.

The system is flexible and provides the user with full control over the geometry and neutron source parameters, number of layers and sectors, material definitions, elemental compositions and MCNP input file options. Natural isotopic abundance is assumed throughout: ^{6}Li enrichment is allowed only in the tritium breeder. The MCNP input file thus generated, which can be modified at will and/or run elsewhere, contains all the necessary information for the run, including the user-specified energy bin format for the tallies and an automatically developed importance map. Additional definitions can be made of dummy materials (structural, coolant, breeder, multiplier and armour), used for determining primary heating, material damage and tritium generation distributions.

2.2. Post-processing

The MCNP output file is post-processed to extract information that enables subsequent analyses and to determine several crucial parameters for in-vessel

component design and operational performance. Post-processed information is presented in a comprehensive manner: a file is produced containing neutron and photon spectra for every cell, providing a complete map of the radial and poloidal distribution of neutron and photon flux and enabling subsequent FISPACT activation analyses that constitute the basis of safety and environmental assessments. A second file maps the remaining information:

(a) Neutron wall load poloidal distribution.
(b) Neutron and photon heating for every cell. Average values per cell, as well as heating induced individually on the structural, armour, breeder, multiplier and coolant materials, can be calculated.
(c) ^{6}Li, ^{7}Li and total tritium production per cell, and total breeding ratio. A clear map of the tritium production in the blanket is produced, helping optimise port size and positioning around the vessel, and indicating the possibility of varying the level of ^{6}Li enrichment in the breeder.
(d) Neutron dose and dose rate values (dpa/year), defining the radiation damage the various in-vessel materials sustain during operation. Dpa response functions are taken from the RFL libraries, [8].
(e) Subsequent analysis of FISPACT output provides helium production rates (appm/year), and He/dpa ratios which, together with dose rate and especially for the structural and armour materials, determine the irradiation creep and swelling behaviour.

All these results, defining micro-structural damage and influencing the thermo-mechanical properties of the materials, assist in the analysis of the fusion power plant performance by dictating operational limits, such as maximum lifetime (dpa and He production), coolant temperature and pressure requirements (nuclear heating, He/dpa ratios) or allowable port size and blanket segmentation (TBR). Estimation of these leads to determination of plant availability, maintenance schemes and efficiency of the conversion system.

3. PPCS power plants and computational models

PPCS plants explored a range of physics and technology assumptions spanning from near-term, ITER-like to advanced scenarios. The first four were named A–D following an order of increasing extrapolation from current expertise, and differ substantially in plasma parameters, electrical output, blanket and divertor technology from the models that formed the basis of earlier European studies. For all of them, system analyses produced self-consistent plant parameter sets with optimal economic performance, initially aiming to a fixed production of 1.5 GWe although the final outcome differed slightly [9]. Table 1 presents a summary description of these, and a comparison with ITER. A fifth concept, plant model AB, is a further near-term design currently under study [10].

Table 1
Design parameters of ITER and PPCS plant models, [9]

Parameter	ITER	PMA	PMB	PM-AB	PMC	PMD
Unit size (GWe)	–	1.55	1.33	1.5	1.45	1.53
Blanket concept	–	WCLL	HCPB	HCLL	DCLL	SCLL
T breeder/neutron multiplier	–	LiPb	Li_4SiO_4/Be	LiPb	LiPb	LiPb
^{6}Li enrichment	–	90%	30%	90%	90%	90%
Coolant	–	H_2O	He	He	LiPb/He	LiPb
Structural material	–	RAFM	RAFM	RAFM	RAFM	SiC
Plant efficiency	–	0.31	0.36	0.35	0.42	0.60
Fusion power (GW)	0.50	5.00	3.60	4.24	3.41	2.46
Major radius (m)	6.2	9.8	8.6	9.6	7.5	6.1
Aspect ratio	3.1	3.0	3.0	3.0	3.0	3.0
Elongation	1.7	1.9	1.9	1.9	2.1	2.1
Triangularity	0.33	0.4	0.4	0.4	0.7	0.7
Avg. plasma temperature (keV)	20	22	20	21.5	16	12

Table 2
Characteristics of the MCNP models of the PPCS plants

	PMA	PMB	PM-AB	PMC	PMD
Number of radial layers	16	13	11	21	19
Number of poloidal sectors	16	16	16	16	16
Divertor sectors (left/right)	2/2	1/1	2/2	2/2	2/2
Cryostat					
Height (m)	24.0	22.0	25.0	24.0	20.0
Radius (m)	19.0	17.0	19.0	18.0	14.0
Thickness (m)	0.1	0.1	0.1	0.1	0.1
Solenoid inner radius (m)	2.4	2.4	2.4	1.5	0.9
Plasma source					
Elongation	1.7	1.7	1.7	1.9	1.9
Triangularity	0.25	0.25	0.27	0.47	0.47
Peaking factor	1.7	1.7	1.7	2.5	2.5
Core ion temperature (keV)	58.0	50.0	53.8	39.9	30.0

The physics of plant models A, B and AB (PMA, PMB and PM-AB) is just above the conservative ITER operational regime. They are based on the water-cooled, stagnant LiPb (WCLL), helium-cooled pebble bed (HCPB) and helium-cooled LiPb (HCLL) blanket concepts, developed from European ITER test module proposals. Reduced activation Eurofer grade steel is the structural material. Plant models C and D (PMC and PMD) are based on progressively more optimistic plasma and materials performance, employing the dual-coolant and a self-cooled LiPb blanket concepts, DCLL and SCLL, both of which use SiC.

The calculations presented here used MCNP models of the plants constructed with the HERCULES system; their main characteristics are shown in Table 2. The radial builds matched the design as closely as reasonably possible, within the constraints of the code [6]; they consisted of FW, blanket, shield (divided into high and low temperature shields, HTS and LTS, where applicable), vacuum vessel (VV) and toroidal field coils (TFC). The shields explored a variety of concepts, tailored to provide appropriate attenuation of the neutron flux. They ranged from conventional water cooled steel (PMA) to LiPb-cooled tungsten carbide (PMD), including a design based on a mixture of steel and zirconium hydride to avoid the use of both beryllium and water coolant in-vessel. Only PM-AB and PMD assumed a millimetric tungsten armour coating the FW surface, although the possibility of introducing this feature in PMA and PMB as well is being investigated [11].

4. Neutron transport analyses and results

Sample calculations were performed with the purpose of testing the HERCULES neutron transport module capabilities. MCNP runs of the four PPCS plant models used nuclear data from the ENDF-VI evaluated library. Neutron (175-group VITAMIN-J format) and photon energy spectra were calculated in all non-void cells of the models, alongside tallies enabling the calculation of neutron wall load, heat deposition and tritium generation. HERCULES was used to post-process the output in the way described previously. FISPACT runs simulated irradiation histories based on the proposed PPCS maintenance scheme [9], assuming a lifetime of 2.5 years for the divertor, 5 years for replaceable in-vessel components (FW, blanket and parts of the shield according to the design) and 25 years for permanent items (the rest). Optimised material compositions in the FISPACT runs included a full set of impurities. The complete output includes time variations of nuclide inventories, activity and other derived quantities, such as biological hazard potentials [12].

A summary of HERCULES post-processing is given in Table 3. Comparison with values of the more elaborated PPCS analyses [9], is possible but should be made with caution because of the different approach and level of detail in the computational models. PM-AB results are provisional, and no specific studies are available. PMA and PMB values are scaled from original assessments at 5500 and 3300 MW fusion power [13]. All HERCULES results are within 7% of PPCS

Table 3
Summary of the HERCULES neutron transport results, and comparison with PPCS values (in brackets)

	PMA	PMB	PM-AB	PMC	PMD
Average neutron wall load (MW/m^2)	2.25 (2.41)	1.99 (2.00)	1.90	2.35 (2.20)	2.54 (2.40)
Total nuclear heating (GW)	4.76 (4.90)	3.99 (3.97)	3.92	3.43 (3.20)	2.32 (2.23)
Energy multiplication	1.19 (1.22)	1.39 (1.38)	1.16	1.26 (1.17)	1.18 (1.17)
TBR	1.13 (1.06)	1.36 (1.12)	1.12	1.19 (1.15)	1.16[a] (1.12)

[a] Blanket value: TBR (divertor) = 0.24 and (total) = 1.40.

results, the exception being PMB total TBR. Poloidal and toroidal in-homogeneity of the real designs introduce gaps (e.g. ports) not taken into account in these calculations, and the relatively greater homogeneity of the radial build in HERCULES is a limiting factor. The usual result is the overestimation of total quantities. In PMB, which has the highest total TBR of all plants, these effects are particularly important (∼21% difference with PPCS value) for they are combined with the blanket model having two more segments than the rest of the plants. Nonetheless, poloidal TBR mapping is proving to be useful in determining design features, such as available port area and poloidal module arrangement [14].

Poloidal variation of the neutron wall load and radial variation at the outboard midplane of the damage rate to the structural material are shown for the four original plant models in Figs. 2 and 3; also midplane radial profiles of the neutron-to-photon heating ratio, helium production rate and helium/dpa ratio in the coolant and structural materials of the WCLL and DCLL blankets are shown (Figs. 4–6). Finally, poloidal variation of the tritium generation at different radial locations in the HCPB blanket is also shown (Fig. 7). As expected, radiation damage rates in the FW follow the same poloidal trend as the wall load, which is useful for scaling this parameter and indicating where the maximum damage to the materials is more likely. Damage rates and helium production have specific limits constraining the useful life of components. A typical figure of ∼150 dpa for structural integrity of steels limits the FW and blanket lifetime depending on the poloidal location. Vacuum vessel helium levels below 1 appm, achieved throughout its lifetime, guarantee reweldability in all cases and for all poloidal locations.

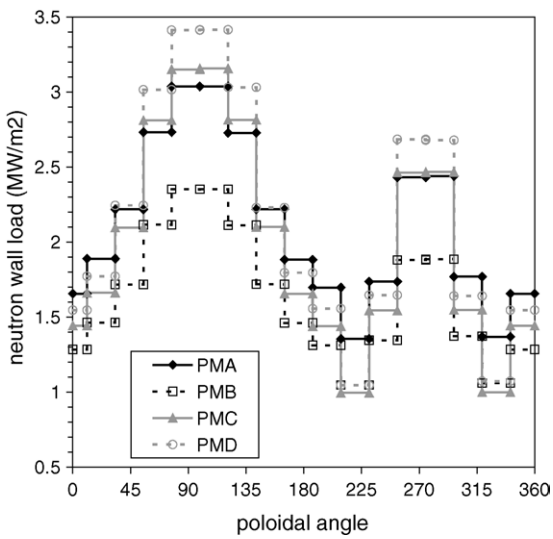

Fig. 2. Poloidal variation of the neutron wall loads of the five PPCS plant models.

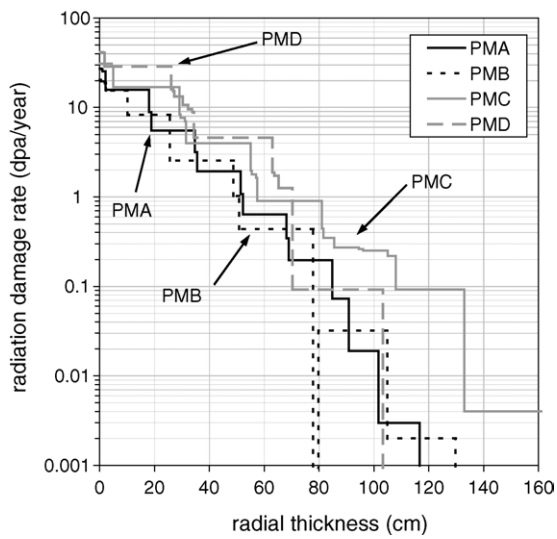

Fig. 3. Radial variation at the outboard midplane of radiation damage rates to the structural material in the five PPCS plant models.

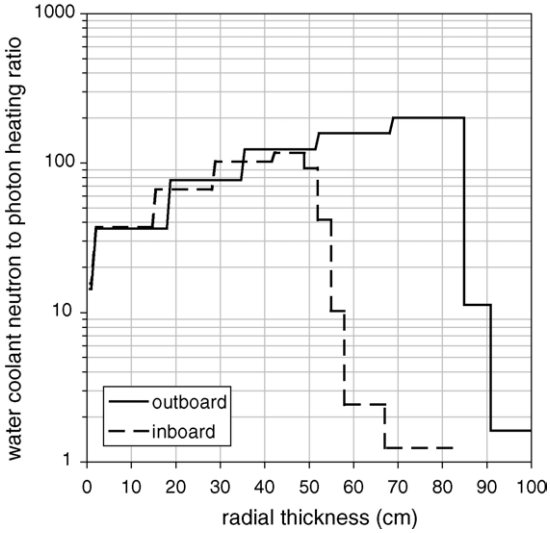

Fig. 4. Radial variation of the neutron to photon heating ratio in the water coolant of PMA WCLL blanket, at the outboard and inboard midplane.

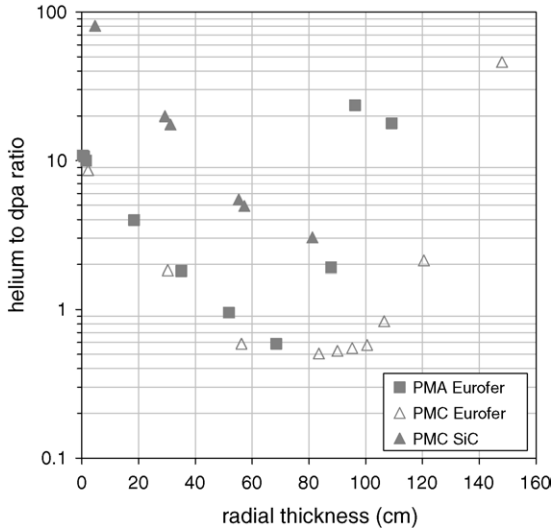

Fig. 6. Outboard midplane radial variation of the helium to dpa ratio in the Eurofer of the WCLL and DCLL blankets, and in the SiC of the latter.

Nuclear heating must be actively removed during operation by the action of the coolant. The distribution of this in the different components of the HERCULES models is in agreement with PPCS results [9]. However, during the investigation of the performance of the near-term models including a tungsten armour [11], nuclear heat deposition in the FW has been found to increase by a factor of nearly 50% for PMA (similar but lower for PMB). This points to the need for fine-tuning the cooling systems accordingly, though acknowledging that

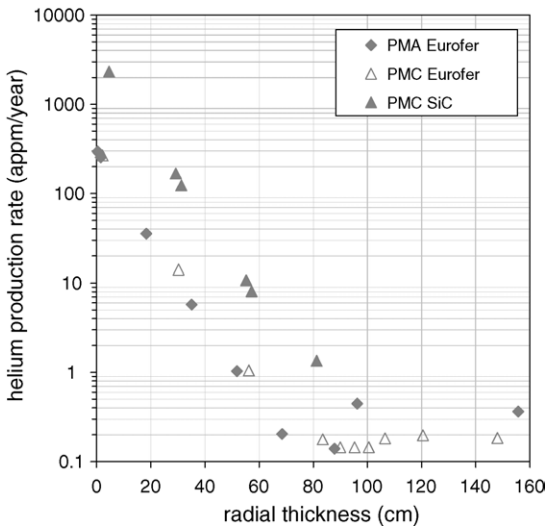

Fig. 5. Outboard midplane radial variation of helium production rate in the Eurofer of the WCLL and DCLL blankets, and in the SiC of the latter.

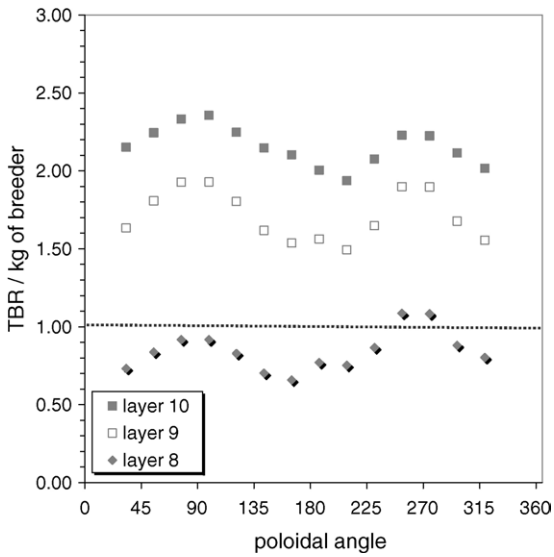

Fig. 7. Poloidal variation of the TBR at different radial locations in the HCPB blanket (production and consumption per kg of breeder).

nuclear heat accounts for only about 20% of the total heat deposition in the FW. Neutron to photon heating ratio in water coolant (Fig. 4) is known to dictate water chemistry requirements and corrosion/dissolution rates of the loop structural material.

5. Conclusions

The capabilities of the neutron transport module of HERCULES have been benchmarked on PPCS plant models. Key engineering parameters and their radial–poloidal variations were obtained, the applications of which range from the design of in-vessel components to assisting in the decision-making of satisfactory geometry, materials and plasma physics choices, overall providing a fast and self-consistent analysis. Sample results presented here are consistent with values obtained through detailed design or more specific PPCS analyses, when accounting for the different degree of detail in the models (i.e. poloidal and/or toroidal variations, such as ports).

HERCULES enables fine-tuning and design optimisation before committing to further, more elaborate and time-consuming analyses. Results could serve as a starting point for conceptual design studies and allow comparison of various schemes based on crucial parameters dictating plant performance.

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