Available online at www.sciencedirect.com
SCIENCE DIRECT
ELSEVIER
Nuclear Engineering and Design 236 (2006) 657–668
Nuclear Engineering and Design
www.elsevier.com/locate/nucengdes

The PBMR steady-state and coupled kinetics core thermal-hydraulics benchmark test problems

F. Reitsma [a,*], G. Strydom [a], J.B.M. de Haas [b], K. Ivanov [c], B. Tyobeka [c], R. Mphahlele [c], T.J. Downar [d], V. Seker [d], H.D. Gougar [e], D.F. Da Cruz [b], U.E. Sikik [a]

[a] *Reactor Design and Fuel Management Group, PBMR (Pty) Ltd., South Africa*
[b] *Fuel, Actinides & Isotopes Group, NRG, The Netherlands*
[c] *Nuclear Engineering Program, Penn State University, USA*
[d] *Nuclear Engineering Department, Purdue University, USA*
[e] *Fission & Fusion Systems, INL, USA*

Received 6 October 2004; received in revised form 8 December 2005; accepted 8 December 2005

Abstract

In support of the pebble bed modular reactor (PBMR) Verification and Validation (V&V) effort, a set of benchmark test problems has been defined that focus on coupled core neutronics and thermal-hydraulic code-to-code comparisons. The motivation is not only to test the existing methods or codes available for high-temperature gas-cooled reactors (HTGRs), but also to serve as a basis for the development of more accurate and efficient tools to analyse the neutronics and thermal-hydraulic behaviour for design and safety evaluations in future.

The reference design for the PBMR268 benchmark problem is derived from the 268 MW PBMR design with a dynamic central column containing only graphite spheres. Several simplifications were made to the design in order to limit the need for any further approximations when defining code models. During this process, care was taken to ensure that all the important characteristics of the reactor design were preserved. The definition and initial phases of the benchmark were performed under a cooperative research project between NRG, Penn State University (PSU) and PBMR (Pty) Ltd. However, participation has been extended to include Purdue University and INL. All contributions to the benchmark effort were made in-kind by the participating members including the participation in four benchmark meetings over a period of 3 years. Based on the work performed in this benchmark the PBMR 400 MW design with fixed central reflector has been accepted as an OECD benchmark problem and work has already started.

In this paper, the benchmark definition and the different test cases are described in some detail. Phase 1 focuses on steady-state conditions with the purpose of quantifying differences between code systems, models and basic data. It also serves as the basis to establish a common starting condition for the transient cases. In Phase 2, the focus is on performing coupled kinetics/core thermal-hydraulics test problems with a common cross-section and material property sets. The six events selected are described, and examples of some results are included to illustrate the behaviour of the transients. The final results of this work will be published in an NRG report and the focus will move to the OECD 400 MW benchmark problem.
Published by Elsevier B.V.

1. Introduction

The reference design for the PBMR268 benchmark problem is derived from the 268 MW PBMR design with a dynamic central column containing only graphite spheres. The PBMR design applies a continuous reloading scheme where unloaded fuel spheres, which have not reached the target burn-up, are returned to the top of the core. This multi-pass fuel circulation is often referred to as a MEDUL cycle. For a so-called 10-pass equilibrium core used in this definition, this implies that the fuel spheres will, on average, circulate 10 times through the core before being discarded as spent fuel.

In Section 2 the rationale for the creation of the PBMR benchmark is explained, with attention given to the assumptions made. In Section 3 the geometrical description is included, with core dimensions and material region identified that are defined in Sec-

* Corresponding author at: Reactor Design and Fuel Management Group, PBMR Pty Ltd., P.O. Box 9396, Centurion 0046, South Africa. Tel.: +27 12 677 9934; fax: +27 12 663 8797.
E-mail address: Frederik.Reitsma@pdmr.co.za (F. Reitsma).

0029-5493/$ – see front matter. Published by Elsevier B.V.
doi:10.1016/j.nucengdes.2005.12.007

tion 4. Although the data defining all cases (such as the detailed material number densities and cross-sections sets) cannot be included in this paper, the description is adequate to enable some cases to be calculated, and to gain insight into the specification.

The benchmark calculations are performed in two phases. Phase 1 focuses on steady-state conditions with the purpose of quantifying differences between the different code systems, models and basic data. The secondary aim is to establish a common starting condition for the transient cases defined as Phase 2. In Section 5 the steady-state cases are defined, while the transient definitions are given in Section 6. A short description of the codes and methods used to generate the results reported is given in Section 7. The transient cross-section library philosophy and a description of the tools used to generate the data is provided in Section 8. Selected preliminary results are included in Section 9 with the main aim to illustrate the differences in the behaviour of the transient cases. Conclusions and future work is included in Section 10.

2. Benchmark philosophy and assumptions

Some benchmark problem definitions and experimental facilities do of course exist for high-temperature reactors (HTRs), including a few existing for pebble-bed reactors. This includes, amongst others, the Proteus pebble-bed critical experiments (IAEA, in press) and ASTRA facility (Ponomarev-Stepnoi et al., 2003; Reitsma and Naidoo, 2003) as examples of critical assemblies, the HTR-10 reactor in operation at Institute of Nuclear Energy Technology, Tsinghua University, Beijing, China (Xu, 2002), reactors that operated in Germany in the past such as the AVR (Arbeitsgemeinschaft Versuchsreaktor GmbH—a 15 MW pebble-bed experimental reactor built at the FZJ site, Jülich and operated from December 1967 till 1988), and several code-to-code comparisons performed as part of the IAEA CRP-5 (Co-ordinated Research Project (CRP) on "Evaluation of HTGR Performance") and similar programmes. Some transient experiments were also performed at the AVR and simulated by TINTE (Gerwin et al., 1989). In the design of the HTR-Modul the ZIRKUS and KIND codes were used and development has continued at IKE at the University of Stuttgart (Rademer et al., 2004). All of these contributed to the benchmarking and V&V of coupled neutronics/core thermal-hydraulics tools used in pebble-bed reactor designs. The question thus remains where the current benchmark definition and code-to-code comparison effort can make a contribution.

The focus of the current benchmark test cases is on establishing well-defined transient benchmark cases. Although some work on the benchmarking of core simulation methods (steady-state) has been performed on PBMR designs (Tyobeka et al., 2003; van Heek et al., 2001), the focus of the current work is on developing coupled kinetics/core thermal-hydraulics test problems that include both fast (reactivity insertions) and slow (thermal heat up due to decay heat) transients. In order to achieve this, several simplifications were made to the design specification so that the need for any further approximations is limited. It also meant that number densities and macroscopic cross-sections are given for reference equilibrium core cases. Furthermore, a multi-dimensional cross-section library and interpolation routines are supplied to participants as the basis for all transient studies. Although this requires source code changes, it circumvents differences due to different sources of cross-section data and preparation. This is perhaps the feature that distinguishes this effort the most from others. This approach is not new and has being applied to power water reactor (PWR) and boiling water reactor (BWR) transient benchmark definitions (Ivanov et al., 1999; Solis et al., 2001).

During the simplification process, care has been taken to ensure that all the important characteristics of the reactor design are preserved. The simplifications make the core design essentially two-dimensional (r, z). Flow channels within the pebble bed have been simplified to be parallel and at equal speed while the dynamic central column and mixing zone widths were defined to be constant over the total axial height. This implies flattening of the pebble-bed's upper surface and the removal of the bottom cone and de-fuel channel that results in a flat bottom reflector. Control rods in the side reflector are modelled as a cylindrical skirt (also referred to as a grey curtain) with a given B10 concentration. A multi-pass fuel circulation (MEDUL with 10 passes) and vertical pebble flow are assumed.

Thermal-hydraulic simplifications include the specification of stagnant helium between the barrel and reactor pressure vessel (RPV) and stagnant air between the RPV and heat sink (outer boundary). The coolant flow is restricted to upwards flow from the inlet below the core within a porous ring in the reflector and downwards flow through the pebble bed to the outlet plenum. No reflector cooling or leakage paths were defined.

Other simplifications include the assumption that all heat sources (from fission) will be deposited locally, i.e. in the fuel, and that no other heat sources exist outside the core (for example neutron absorption in the control rod region). Simplifications are also made in the material thermal properties, in as far as constant values are employed or specific correlations are employed.

3. Geometrical description

The reactor geometrical layout and dimensions are summarized in Table 1 and Fig. 1.

4. Materials, properties and cross-sections

Pebble fuel spheres with low enriched (8%) uranium-oxide triso-coated particles and with a loading of 9 g per fuel sphere are used in the benchmark definition. A pebble packing fraction of 0.61 is assumed. Only one void region, being the 25 cm between the top of the pebble bed and the bottom of the top reflector, is present. In the diffusion calculation, directional dependent diffusion coefficients are used to represent the neutron streaming effects. A factor is multiplied to the diffusion coefficient for the r and z direction, 0.1 for r and 0.5 for the z direction. The radius of the region is taken as the diffusion coefficient. The fuel and material definitions are given in Table 2.

The equivalent B-10 concentration for the control rods, modelled as a gray curtain, is given in Table 3. The control rods are inserted into the side reflector to a level of 200 cm below the core

Download English Version:

https://daneshyari.com/en/article/299494

Download Persian Version:

https://daneshyari.com/article/299494

Daneshyari.com