Annals of Nuclear Energy 77 (2015) 172–175

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Benchmark of Neutronics and Thermal-hydraulics Coupled Simulation program NTC on beam interruptions in XADS

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ARTICLE INFO

Article history:
Received 9 September 2014
Received in revised form 12 November 2014
Accepted 13 November 2014

Keywords:
Benchmark
Beam interruption
NTC
Neutronics and Thermal-hydraulics coupling

ABSTRACT

The Neutronics and Thermal-hydraulics Coupled Simulation program (NTC) is developed by FDS Team, which is a code used for transient analysis of advanced reactors. To investigate the capacity and calculation correctness of NTC for transient simulation, a benchmark on beam interruptions in an 80 MWth LBE-cooled and MOX-fuelled experimental accelerator-driven sub-critical system XADS was carried out by NTC. The benchmark on beam interruptions used in this paper was developed by the OECD/NEA Working Party on Scientific Issues in Partitioning and Transmutation (WPPT). The calculation model had the minimum phenomenological and computational complexity which concerned a simple model (single fuel channel thermal-hydraulics) of the average fuel pin corresponding to the BOL fuel condition. This benchmark was designed to investigate the temperature and power responses caused by beam interruption of different durations, which aimed at comparative assessment of NTC and other computation methods. A comparison of NTC and other ten sets of temperature and power was provided, which showed that the results had good agreement.

1. Introduction

Accelerator-driven sub-critical systems (ADSs) are currently under worldwide investigation, aiming at transmuting minor actinides (MA) included in the long-lived nuclear waste. Safety analyses of ADS are mainly performed by codes developed for critical reactors, a point-kinetics model for computing the transient power is often employed, such as SIM-ADS, TRACE, SAS4-ADS and so on. However, Rineiski and Maschek (2005) have shown that the point-kinetics model which assumes time-independent neutron direct and adjoint flux shapes may be inaccurate in the ADS case· Flux shape variations may be significant due to external source related effects even if the material distribution remains almost unchanged, much worse when there was a strong material movement in the system, which may happen during a severe accident. Therefore, ADS safety studies, especially severe accident analysis, require the using of spatial kinetics model for transient analyses.

The Neutronics and Thermal-hydraulics Coupled Simulation program (NTC) employing the spatial kinetics model is developed by FDS Team, which is a code used for transient analysis of advanced reactors, such as Fusion-Driven Subcritical System, Magnetic Confinement Fusion Reactor Blanket and lead- or LBE-cooled fast reactor (Bai, 2007a; Bai et al., 2007b). FDS Team, affiliated with Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences, is an interdisciplinary research team devoting to the research and development of advanced nuclear energy systems.

In this paper, to verify the calculation correctness of NTC code, the beam interruptions of different durations of the 80 MWth LBE-cooled eXperimental Accelerator-Driven sub-critical System (XADS) developed by the OECD/NEA Working Party on Scientific Issues in Partitioning and Transmutation (WPPT) were simulated by NTC. The simulation results were compared with the benchmark results on the same beam interruptions of other ten computational transient analysis codes.

2. Computational code

NTC is a Neutronics and Thermal-hydraulics Coupled code and based on the experience of nuclear design and development of FDS Team (Hu and Wu, 2006; Huang et al., 2009; Qiu et al., 2000; Wu et al., 1999, 2002; Wu et al., 2006, 2007, 2008, 2009a,b), which can be used in reactor transient and severe accidents analyses. The code function structure of NTC is presented in Fig. 1.

The neutronics module of NTC code employs a space- and energy-dependent neutron transport model coupled with a two-dimensional, multi-phase, multi-component fluid-dynamics

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http://dx.doi.org/10.1016/j.anucene.2014.11.017
0306-4549/

thermal-hydraulics module and a heat and mass transfer model of fuel pin and assembly structure. The neutronics calculation provides the power distributions solved by an improved quasi-static method, in which a space- and time-dependent neutron transport equation is factorized into: a shape function that represents the neutron flux distribution but changes slowly with time, and an amplitude function that accounts for the reactor power which changes fast with time. Between the flux-shape time step and the flux-amplitude time step, the reactivity and other kinetics parameters time step must be taken into account because the reactivity and other kinetics parameters are calculated from the neutron flux and macroscopic cross sections and then are used to solve the amplitude equation. The time step size of heat-transfer is set to be the same as the reactivity step, because the reactivity feedback and fuel heat generation are closely related each other. In order to update the component internal energies due to nuclear heating, the fluid dynamic time step is consistent with the flux-amplitude time step. In return, the reactivity are controlled by the fluid dynamics as well, to take into account the influence of change. The fluid-dynamics solution algorithm is based on a time-factorization approach, in which a second-order spatial differencing is employed to reduce numerical diffusion. All those make NTC applicable for the accurate simulations of the transient and severe accidents of many advanced nuclear systems, especially for ADSs in which the neutron flux distributions are anisotropic and non-uniform.

3. The benchmark specification

The specification of the calculations was designed to have the minimum phenomenological and computational complexity of the models. Moreover, the benchmark specification defined a computational exercise without any degree of freedom, and obtained a particularly clean comparative assessment of the different computation methods used to evaluate power and temperature transients induced by beam interruptions of different duration.

Transients induced by recovered beam interruptions of varying duration (1 s, 3 s, 6 s, 12 s and a definitive beam trip) in a MOX-fuelled and LBE-cooled, 80 MWth XADS-type system, were proposed as a computational benchmark problem. The benchmark concerned a simple model (single fuel channel thermal-hydraulics coupled with point kinetics) of the average fuel pin. In particular, the simple model geometry refers to the "fresh fuel" assumption corresponding to the BOL fuel condition. For the sake of simplicity, the heat decay power was ignored. The main parameters of benchmark problem are shown in Table 1. The fuel channel geometry, power data, material properties, heat transfer correlations and the point kinetics were in detail provided in the report (D'Angelo and Gabrielli, 2003). The existing results for this benchmark work were calculated by ten codes, which are TIESTE-MINOSSE, SITHER-PKS, EXCURS-M, SASSYS/SAS4A, SIM-ADS, TRAC-M, LOOP2, TRAC MOD, SAS4ADS and DESINUR codes. All of them employ point kinetics.

Table 1
The main parameters of benchmark problem.

Fuel zone height	0.90 m
Inner fuel radius (inner hole radius)	0.9×10^{-3} m
Outer fuel radius	3.57×10^{-3} m
Inner clad radius	3.685×10^{-3} m
Outer clad radius	4.25×10^{-3} m
Coolant flow area	9.89×10^{-5} m^2
Coolant inlet temperature	573.0 K
Fuel pin total power (initial value)	7 203.6 W
Coolant outlet temperature (initial value)	673.0 K

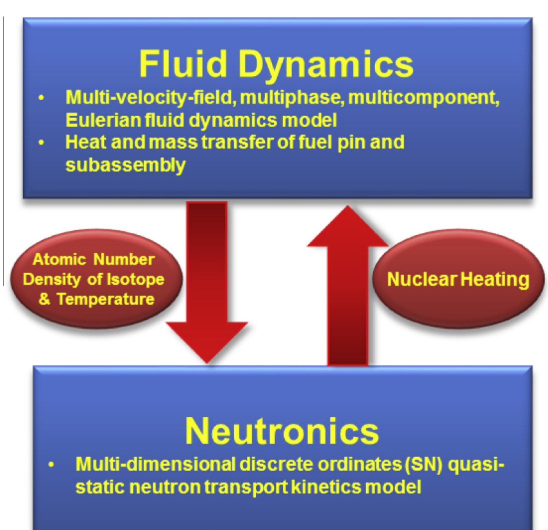

Fig. 1. Code function structure of NTC.

4. Simulation results and discussions

4.1. Initial steady-state results

All the results of benchmark calculations relevant to the beam interruptions from the paper are presented and compared in this paper. The steady-state axial temperature distributions calculated by NTC code and ten sets of results calculated by different participants using various codes in the benchmark are presented in Fig. 2. In Fig. 2, the maximum temperature set is relevant to the fuel centerline axial distribution. Results relevant to the fuel surface, clad surface and coolant temperatures are drawn at progressively lower temperatures in the same figure. The results show that the trends of the temperatures of NTC and other ten codes were the same, and the differences were small. Even the two largest differences on fuel centerline temperature, relevant to the NTC results and to the SAS4ADS results are not significant (~1.8%). The reason for the discrepancies obtained in the calculation of NTC was mainly that the power data was calculated by spatial neutronics rather than a given power value. In any case, the results can be considered to be within acceptable limits.

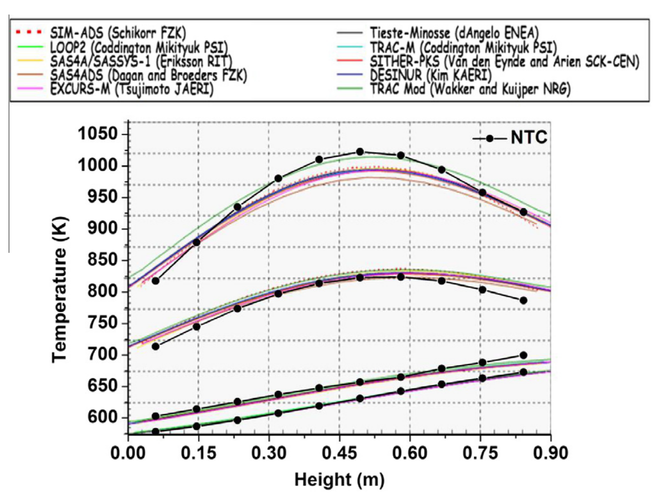

Fig. 2. Initial (steady-state) axial temperature distributions.