Project work description

This document describes a content of a project work that should be performed by all students working in teams as a partial fulfillment of the requirements in course SH2702 in Nuclear Reactor Technology. The purpose of the project work is to become acquainted with general features and principles of operation of selected reactor types. The first three tasks of the project are dealing with literature studies, whereas the last three tasks are requiring calculations to determine the basic design parameters of the reactor core.

Task 1 – General design specification of the nuclear power plant with selected reactor type

In this Task a literature study should be performed to find a general description of the reactor core, reactor vessel, primary/secondary loop, balance of plant, etc. The focus should be on the particular features that make the design unique and interesting.

Task 2 – Operational principles of the power plant

The purpose of this Task is to describe how the reactor is operated during start-up, normal operation, and shutdown. Both base load and load-following scenarios should be discussed. The methods for reactivity control should be described. The role and principles of operation of auxiliary systems should be explained. Particular focus should be on these operational principles that make the reactor unique and interesting.

Task 3 – Safety features of the power plant

In this Task it should be explained what are the general principles of reactor safety. Descriptions of the reactor protection system and the engineered safety feature should be included. Major results of reactor safety analysis and some key safety parameters, such as, e.g., core damage frequency, should be mentioned.

Task 4 – Calculation of selected core parameters

In this task it is required to collect key core parameters (this could be done already in the frame of Task 1). If any of the listed below parameters is not available from the open literature, it should be assumed based on the best knowledge and/or engineering judgement. The minimum list of the required parameters is as follow,

- Total core heat output
- Total heat output in fuel pellets (if not known, assume 94.5% of the total core heat output)

- Nominal system pressure
- Total core mass flow rate
- Effective fuel cooling mass flow rate (= total mass flow rate minus bypass flow; if not known, assume that core bypass flow is 5% of the total flow in PWRs and 15% in BWRs)
- Number of fuel assemblies
- Active fuel height
- Channel lateral dimensions in BWRs
- Lattice pitch
- Number of fuel rods per assembly
- Outside fuel rod diameter
- Clad thickness
- Fuel pellet diameter
- Number of spacers, their locations and their local pressure loss coefficients (if not available, assume 6-10 equally-distanced spacers with local loss coefficient of 0.6-0.8 each)
- Clad and channel (BWR) wall roughness

Using the above input data, the following core-averaged thermal-hydraulic characteristics should be calculated and presented as plots (parameter versus axial distance from the core inlet):

- Axial pressure drop distribution
- Axial coolant enthalpy distribution
- Axial coolant temperature distribution
- Axial void fraction distribution (BWRs)
- Flow characteristic of the core, that is a relationship $\Delta p_{core} = f(G_{core})$ – core pressure drop versus mass flux, for core power equal to 0%, 50%, 100% and 150% of the nominal power, with flow varying from 1% to 150% of the core nominal flow.

In these calculations inlet orifices should be applied (via proper local loss coefficient at fuel assembly inlets) to get about 50% of the total pressure drop in the core just at the inlet orifice for BWR and about 25% for PWRs. Such inlet orifices are introduced to stabilize flow through the core and to avoid hydrodynamic instabilities.

If not found in the literature, the spatial core power distribution should be assumed as,

$$q''(r,z) = q_0'' J_0\left(\frac{2.405r}{\tilde{R}}\right)\cos\left(\frac{\pi z}{\tilde{H}}\right), \quad (1)$$

where q_0'' is the heat flux at the core center $r = z = 0$, J_0 is the Bessel function of the first kind and zero order, and $\tilde{R}, \tilde{H}$ are the extrapolated radius and the extrapolated height of the core, respectively. For reflected core, it can be assumed that $R/\tilde{R} = H/\tilde{H} = 5/6$.

Task 5 – Calculation of CHF margins in a hot channel

CHF (Critical Heat Flux) is one of the factors that limit the maximum thermal power in the core. In this Task, the margins to the CHF occurrence should be estimated. For that purpose hot channel should be investigated, assuming that the actual power of the channel is given by Eq. (1). Next, the power in the channel should be gradually increased, keeping all other parameters constant, until CHF condition is achieved. In BWRs the dryout type of CHF should be tested and in PWRs the DNB (Departure from Nucleate Boiling) should be assumed. The following parameters should be found:

$$MDNBR = \frac{\text{minimum critical heat flux in hot channel } q''_{cr}(z)}{\text{local actual heat flux in hot channel } q''(z)} \quad ,$$

for PWRs and,

$$MCPR = \frac{\text{critical power in hot channel } q_{cr}}{\text{actual power in hot channel } q_{ac}} \, ,$$

For BWRs.

The following results for the hot channel should be presented as plots, or as a single parameter, whichever applies:

- Axial pressure drop distribution
- Axial coolant enthalpy distribution
- Axial coolant temperature distribution
- Axial void fraction distribution (BWRs)
- Axial distribution of DNBR (departure from nucleate boiling ratio) and axial location of MDNBR for PWRs
- MCPR for BWRs.

Task 6 – Calculation of the maximum cladding and fuel pellet temperature

The fuel temperature must not exceed the melting temperature of the fuel material at any point in the reactor. Also, the clad temperature should be checked not to exceed the maximum allowed value for the clad material. In this Task these two conditions should be checked. The goal of this Task is to identify the locations of the maximum fuel and clad temperatures (“hot spots”) and the calculation of their values. Calculations should be based on realistic material properties of the clad and fuel. In particular, temperature dependence of the thermal conductivity of clad and fuel materials should be assumed and correlations valid for the actual reactor materials should be used. However, the rod geometry and the fuel structure can be assumed as-fabricated.

Presentation of results

The results of project work should be presented as a written report and during seminars. During the seminars the task results will be presented by all teams. The general requirements and rules for grading of reports and seminar presentations are provided in the Course PM and in the schedule for the project work.

The contents of the literature review tasks (Task 1 to 3) can be freely decided by the teams. However, it is recommended that the presentations of results for the calculation tasks (Tasks 4 to 6) contain the following items:

- Description of input data
- Description of used models and tools
- Presentation of results
- Discussion of the results and conclusions

References and help material

Each team has a freedom to choose the most suitable method to accomplish the tasks. As guidance, the following material is provided:

1. PDF file with most frequently used equations and models in thermal-hydraulic calculation of water-cooled reactors (file TH-DESIGN-EQUATIONS-MODELS.pdf uploaded in Canvas)
2. PDF file with compendium "Thermal-hydraulics in nuclear system" (file THNS_Compendium_2015-01-06.pdf uploaded in Canvas)
3. Scilab (www.scilab.org) scripts with example thermal-hydraulic calculations performed for a core of GE BWR/6 reactor (file ScilabScripts.zip uploaded in Canvas)