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DYN1D-MSR dynamics code for molten salt reactors

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Abstract

This paper reports about the DYN1D-MSR code development and dynamics studies of the molten salt reactors (MSR) – one of the 'Generation IV International Forum' concepts. In this forum the graphite-moderated channel type MSR based on the previous Oak Ridge National Laboratory research is considered.

The liquid molten salt serves as a fuel and coolant, simultaneously and causes two physical peculiarities: the fission energy is released predominantly directly into the coolant and the delayed neutrons precursors are drifted by the fuel flow. The drift causes the spread of delayed neutrons distribution to the non-core parts of primary circuit and it can lead to a reactivity loss or gain in the case of fuel flow acceleration or deceleration, respectively. Therefore, specific 3D tool based on in house code DYN3D was developed in FZR. The code DYN3D-MSR is based on the solution of two-group neutron diffusion equation by the help of a nodal expansion method and it includes models of delayed neutrons drift and specific MSR heat release distribution.

In this paper the development and verification of 1D version DYN1D-MSR of the code is described. The code has been validated with the experimental data gained from the molten salt reactor experiment performed in the Oak Ridge and after the validation it was applied to several typical transients (overcooling of fuel at the core inlet, reactivity insertion, and the fuel pump trip).

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1. Introduction

The first projects for reactors with liquid fuel go back to the pioneers' time of nuclear energy, when various designs with different fuel-type have been proposed. From those the molten salt reactor (MSR) could be considered as the most developed project (MacPherson, 1985). The research on MSR was performed mainly in the Oak Ridge National Laboratory (ORNL), where the molten salt reactor experiment (MSRE) was realised in the sixties. This experiment has shown that the molten salt technology beside the energy production offers three main advantages:

- excellent neutron economy;
- continuous or in-batch reprocessing;
- inherent safety features.

These three advantages make the MSR attractive also for the present Generation IV International Forum (GIF). The aim of the GIF initiative (DOE, 2003) is the development of technologies that achieve safety performance, waste reduction, and proliferation resistance while providing a nuclear energy option that is economically competitive. The MSR is studied also for spent fuel partitioning and transmutation (P&T) purposes. It is the aim of P&T to separate the different components of spent fuel (partitioning) and to reduce the radiotoxicity, mainly by

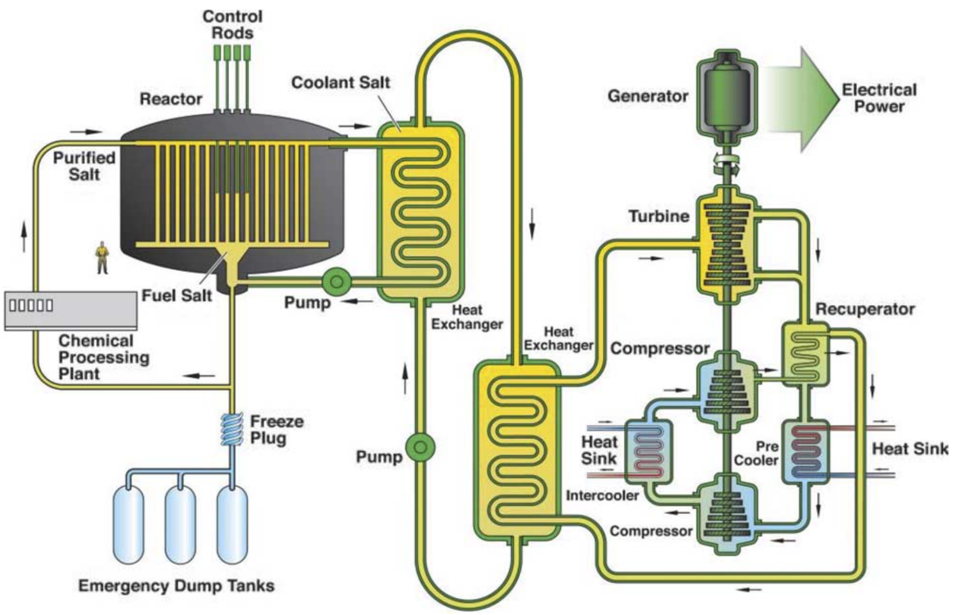

Fig. 1. Scheme of MSR (DOE, 2003).