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Investigation of VVER-1000 rod ejection accident according to the Phenomenon Identification and Ranking Tables for PWR

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ABSTRACT

Resulting from a mechanical failure of a control rod system the reactor coolant pressure ejects the control rod assembly from the core. The control rod ejection accident belongs to the category of reactivity initiated accident (RIA) and is defined as a design basis accident. The U.S. Nuclear Regulatory Commission (U.S. NRC) has established the Phenomenon Identification and Ranking Tables (PIRT) for rod ejection accident in pressurized water reactor (PWR) which will be used in this study.

The present work is focused on selected physical parameters which can be calculated by coupling two best estimate codes for VVER-1000/V-320. Namely, TRACE V5 p4 was used to develop a thermal-hydraulics model and PARCS 3.2 for establishing a detailed neutronics model of a reactor core. Both models were executed in a merged version TRACE V5 p4.

The analysis was focused on evaluation of the following parameters according to PIRT: relative power in the reactor core and rejected fuel assembly, inserted reactivity, moderator properties (density and temperature), and inner/outer temperature of the heat structure. Nominal power of 100% and the control rod ejection time of 0.1 s were considered. The inserted reactivity has reached 0.45$ and the maximum power in the reactor core has risen to 179.7%.

1. Introduction

Control rods consist of neutron absorber materials such as B, Ag, Cd, In or Hf. These elements lower the reactivity when the rods are partially or fully inserted into the core. Unintentional withdrawal of control rods resulting from a mechanical failure of the drive mechanism leads to an uncontrolled rapid insertion of positive reactivity due to the decreasing neutron absorption. As a result, power excursion with large localized relative power increase in the reactor core occurs (RIA, 2010).

Simulations and understandings of such accidents are key objectives for safety authorities around the world. In 2001, the U.S. NRC has established a Phenomena Identification and Ranking Tables (PIRT) panel in order to define and evaluate the phenomena occurring during rod ejection accident for pressurized water reactors (PWRs) and boiling water reactors (BWRs) containing high-burnup fuel. Main goal of this document is to present the most relevant parameters of fuel and reactor core, during accident conditions, which are important to treat in a plant safety analysis. The PIRT methodology represents a systematic way of collecting knowledge and experiences from experts on a specific rod ejection accident and ranking the importance of the phenomena in order to meet some decision-making objectives for other researchers and engineers.

Members of the PIRT commission were composed of international and U.S. experts who are involved in the relevant research fields. Detail description of PIRT can be found elsewhere (Boyacket al, 2001).

Using experimental data and calculation codes, research institutions around the world may examine adequacy and applicability of the PIRT results (Boyacket al, 2001). In the present work we have selected TRACE/PARCS codes as a computational tool which is under the development of the U.S. NRC research program CAMP (Code Application and Maintenance Program).

It is not possible to calculate all phenomena defined in the PIRT by the codes used here. For this reason, only the most relevant parameters were investigated in this paper. As a computational model we have used a VVER-1000/V-320 type of reactor with hexagonal lattice geometry.

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Nomenclature

VVER	Water-Water Power Reactor
PWR	Pressurizer Water Reactor
RIA	Reactivity Initiated Accident
PIRT	Phenomena Identification and Ranking Tables
PARCS	Purdue Advance Reactor Core Simulator
TRACE	TRAC/RELAP Advanced Computational Engine
SNAP	Symbolic Nuclear Analysis Package
CAMP	Code Application and Maintenance Program
U.S. NRC	U.S. Nuclear Regulatory Commission
SG	Steam Generator
RPV	Reactor Pressure Vessel
ECCS	Emergency Core Cooling System
FA	Fuel Assembly
HS	Heat Structure
XS	Cross-Section
T-H	Thermal-Hydraulics
IAEA	International Atomic Energy Agency

2. Codes and coupling procedure

Currently, many various computational codes exist which can perform wide range of accident simulations and safety analyses for different types of reactors. Some advanced calculations require coupling of two physically distinguished codes, one for neutron kinetics and another for thermo-hydraulic calculations in order to assess complexity of accident scenarios. Especially for transient analyses where large insertion of reactivity occurs such as rod ejection accident.

Most of nuclear technology suppliers or research institutions developed their own calculation codes but not all of them can be employed by other users due to some limitations. Thus, two U.S. codes were selected for this study. They are based on the implementation agreement between U.S. NRC and the State Office for Nuclear Safety of the Czech Republic and also because they are available at the Czech Technical University in Prague.

2.1. TRACE code

The TRACE code (TRAC/RELAP Advanced Computational Engine) represents an advanced computational tool for simulation of design basis accidents. The U.S. NRC developed this best-estimate reactor systems code for analyzing transient and steady-state thermal-hydraulics behavior in PWR and BWR reactors. It is the product of a long-term effort to combine the capabilities of the U.S. NRC's four main systems codes TRAC-P, TRAC-B, RELAP5, and RAMONA into one modernized computational tool (TRACE, 2014). The version used in this analysis is TRACE v5 patch 4 released in 2014.

The TRACE code can model a reactor pressure vessel, steam generator and loop components in order to perform 3D heat-transfer analyses in the primary and secondary circuits (TRACE, 2014).

TRACE code is not appropriate for transients in which there are large asymmetries in the reactor core power such as would occur during a control rod ejection transient, unless it is used in conjunction with a PARCS spatial kinetics module. In TRACE stand-alone, neutronics is evaluated by a point reactor kinetics model with reactivity feedback, and the spatially local neutronic response associated with the ejection of a single control rod cannot be modeled. One-dimensional or three-dimensional reactor kinetics capabilities are possible through coupling the PARCS with TRACE (TRACE, 2014).

2.2. PARCS code

PARCS is a three-dimensional (3D) reactor core simulator which solves steady-state and time-dependent, multi-group neutron diffusion and low order transport equations in orthogonal and non-orthogonal geometries. The main calculation features of PARCS include the ability to perform eigenvalue calculations, transient (kinetics) calculations, Xenon transient calculations, decay heat calculations, pin power calculations and adjoin calculations, which have been extended to include not just Light Water Reactors, but also the Pressurized Heavy Water and High Temperature Gas Reactors (Downar et al., 2012). The main limitation for VVER reactors is that the pin power calculation is not available in the hexagonal geometry but all other functions of the PARCS code are available.

The PARCS code is coupled with the thermal-hydraulics system code TRACE. TRACE provides temperature and flow field information to PARCS via few group cross-sections (Downar et al., 2012). The cross-section data must be specified as a part of the PARCS input and the user may choose any code capable of nodal cross-section generation. There are several codes such as HELIOS, SCALE (TRITON or POLARIS), CASMO, and SERPENT that are suitable for this purpose. In our work, all cross-sections data were calculated using SERPENT2 code developed by the research company VTT in Finland.

PARCS is also available as a stand-alone code for performing calculations which do not require coupling with TRACE. Before coupling with TRACE, the rod ejection calculations were performed by a stand-alone PARCS code in order to estimate the functionality and accuracy of the neutronics model.

2.3. Coupling TRACE/PARCS

The basic principle of the coupling procedure is a connection between neutronics node in PARCS and thermal-hydraulics (T-H) node in TRACE. All fuel assemblies and reflectors modeled in the neutronics model must be coupled with a corresponding T-H node in TRACE. This can be a difficult process if the nodalization of both codes is different.

The process of coupling is as follows:

- TRACE calculates boron concentration, new coolant/fuel properties such as moderator temperature, liquid/vapor density, void fraction, centerline and the surface temperature of heat structures. All mentioned data are based on power distribution calculated by PARCS, and PARCS is used as a heat source for heat conduction. (TRACE, 2014).
- PARCS updates macroscopic cross-sections data using the local node conditions based on the coolant and fuel properties received from TRACE. PARCS computes 3-D neutron flux and sends node-wise power distribution to TRACE. Graphical representation of the coupling data transfer is shown in Fig. 1, (Downar et al., 2012).

Mapping the T-H nodes and neutronic nodes is provided by an interface file called MAPTAB. In order to minimize the user effort to prepare the MAPTAB file, automatic mapping schemes were developed for the coupled TRACE/PARCS code that accepts data from code input files to generate the mapping information internally. This functionality is provided by the Symbolic Nuclear Analysis Package (SNAP) software (SNAP, 2012). Unfortunately, the automatic mapping function is not available for hexagonal geometry. In our case, a simple program was written in the MATLAB code which created appropriate MAPTAB file.

Coupling of TRACE and PARCS code for VVER-1000 purposes was verified and validated against the Nuclear Energy Agency (NEA), Organization for Economic Co-operation and Development (OECD) benchmark exercises (Kolev et al., 2010).

Such work was performed by Ivanov et al. (2006) for Exercise 1 and 2 of the VVER Coolant Transient Benchmark Phase 2.