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Sensitivity analysis to a RELAP5 nodalization developed for a typical TRIGA research reactor

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ABSTRACT

The main aim of this work is to identify how much the code results are affected by the code user in the choice of, for example, the number of thermal hydraulic channels in a nuclear reactor nodalization. To perform this, two essential modifications were made on a previously validated nodalization for analysis of steady-state and forced recirculation off transient in the IPR-R1 TRIGA research reactor. Experimental data were taken as reference to compare the behavior of the reactor for two different types of modeling. The results highlight the necessity of sensitivity analysis to obtain the ideal modeling to simulate a specific system.

1. Introduction

The user of a thermal hydraulic system code has a very large number of available basic elements (single volumes, pipes, branches, junctions, heat structures, pumps, etc.) to develop a detailed reactor nodalization. The model can reproduce a specific part or the whole system to be simulated. However, as there is not a fixed rule to perform the nodalization, a large responsibility is passed to the user of the code in order to develop an adequate model scheme which makes best use of the various modules and the prediction capabilities of the specific code (Petruzzi and D'Auria, 2008; D'Auria and Galassi, 1998).

The influence of the user on calculations for thermal hydraulic codes is clearly evident in the relatively wide variation in results from different organizations and code users participating in international standard problem (ISP) exercises. Although some of the user-to-user variation is due in part to the use of different computer codes, a substantial variation is also observed when different users apply the same codes (Adorni, 2007). Particularly, in the RELAP5 code, a physical system consisting of flow paths, volumes, areas, etc., is simulated by building a network of volumes connected with junctions. Therefore, the transformation of the physical system to a system of volumes and junctions is an approximate process (US NRC, 2001). In spite of the substantial progress over the past two decades in the development of more accurate and more user tolerant computer codes for accident analysis, the user can still have a significant effect on the quality of the analyses.

Sensitivity analysis including systematic variations in code input variables or modeling parameters, must be used to help identify the relevant parameters necessary for an accident analysis by ranking the influence of accident phenomena or to bound the overall results of the analysis. Results of experiments can also be used to identify important parameters.

The main aim of this work is to identify how much the code results are affected by code user choices. To perform this, two modifications were made on a previous validated nodalization for analysis of steady-state and forced recirculation off transient in the IPR-R1 TRIGA research reactor (Reis et al., 2010). The modifications include:

(1) variation in the number of the thermal hydraulic (TH) channels in the core (from 13 in the original nodalization to 91 in the modified nodalization) and
(2) insertion of cross-flow model in several core channels of the new nodalization.

The original nodalization has been identified with the name 13-THC, as a reference to the 13 TH channels present in the core of the original modeling and the new one, the modified nodalization, is referenced as 91-THC. The comparison between the results obtained with both nodalizations is presented here. The code used

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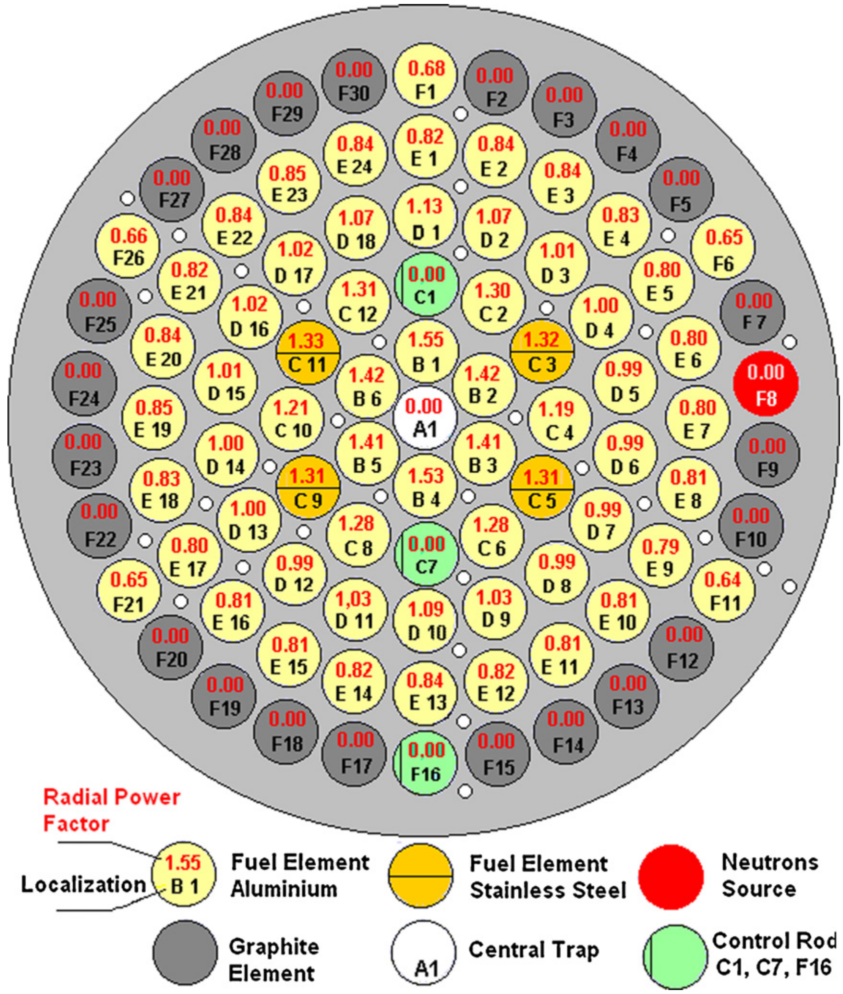

Fig. 1. Radial relative power distribution. (For interpretation of the references to color in this figure legend, the reader is referred to the web version of the article.)

in this study is the RELAP5 Mod3.3, the same code used in Reis et al. (2010). Experimental data from Veloso (2004) were taken as reference to compare the behavior of the reactor for different types of model. The results demonstrate the necessity for sensitivity analysis to obtain the ideal modeling to simulate a system.

The RELAP5 system code was firstly developed to simulate transient scenarios in power reactors such as PWR and BWR. However, several works have been performed to investigate the applicability of the code to research reactors operating conditions (Antariksawan et al., 2005; Khedr et al., 2005; Marcum et al., 2010).

1.1. IPR-R1 TRIGA – general characteristics

TRIGA reactor is the most widely used research reactor in the world. It has an installed base of over 65 facilities in 24 countries on 5 continents. TRIGA reactors are present in a variety of configurations and capabilities, with steady-state power levels ranging from 20 kW to 16 MW. The current enlarged commercial exploitation of this type of nuclear research reactor has increased the consideration to their corresponding safety issues.

The IPR-R1 is a reactor type TRIGA (Training, Research, Isotope, General Atomic), Mark-I model, manufactured by the General Atomic Company and installed at Nuclear Energy Development Center (CDTN) of Brazilian Nuclear Energy Commission (CNEN), in Belo Horizonte, Brazil. The reactor is housed in a 6.625 m deep pool with 1.92 m of internal diameter and filled with demineralized light water.

The water in the pool has function of cooling, moderator and neutron reflector and it is able to assure an adequate radioactive shielding. The reactor cooling occurs predominantly by natural convection, with the circulation forces governed by the water density differences. The removal of the heat generated from the nuclear fissions is performed pumping the pool water through a heat exchanger. The core has a radial cylindrical configuration with six concentric rings (A, B, C, D, E, F) with 91 channels able to host either fuel rods or other components like control rods, reflectors and irradiator channels. There are 63 fuel elements constituted by a cylindrical metal cladding filled with a homogeneous mixture of zirconium hydride and Uranium 20% enriched in ^{235}U isotope. There are 59 fuel elements covered with aluminum and 4 fuel elements with stainless steel. The main thermal hydraulic and kinetic characteristics of the IPR-R1 core are listed in Reis et al. (2010).

The radial relative power distribution (Fig. 1) was calculated in a preceding work using the WIMSD4C and CITATION codes (Dalle, 2003; Dalle et al., 2002) and also experimental data (Veloso, 2004). The radial factor is defined as the ratio of the average linear power density in the element to the average linear power density in the core. In Fig. 1 is also possible to see the six core concentric rings (A, B, C, D, E, F).

2. Nodalization description

The IPR-R1 original nodalization is represented in a general way in Fig. 2. The reactor pool was modeled using two pipe components,