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Review

On-line method to identify control rod drops in Pressurized Water Reactors

T.J. Souza [a,*], A.S. Martinez [a], J.A.C.C. Medeiros [a], D.A.P. Palma [b], A.C. Gonçalves [a]

[a] UFRJ – Universidade Federal do Rio de Janeiro, PEN/COPPE-UFRJ, Ilha do Fundão, 21941-914 Rio de Janeiro, Brazil
[b] CNEN – Comissão Nacional de Energia Nuclear, R. General Severiano 90, 22290-901 Rio de Janeiro, Brazil

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ABSTRACT

A control rod drop event in PWR reactors leads to an unsafe operating condition. It is important to quickly identify the rod to minimise undesirable effects in such a scenario. The goal of this work is to develop an online method to identify control rod drops in PWR reactors. The method entails the construction of a tool based on ex-core detector responses. It proposes to recognize patterns in the neutron ex-core detectors responses and thus to make an online identification of a control rod drop in the core during the reactor operation. The results of the study, as well as the behaviour of the detector responses demonstrated the feasibility of this method.

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1. Introduction

According to the Final Safety Analysis Report for Angra I (FSAR), the event of a control rod drop is a fault of moderate frequency. In a normal operating reactor, this event could lead to an unsafe operating condition. Therefore, it is important to quickly identify the dropped control rod to minimise undesirable effects. If one or more rod position indicator channel should be out of service, detailed operating instructions should be followed to ensure the alignment of the non-indicated assemblies. These operating instructions require monitoring within a set time sequence that characterises an inability to identify in real-time the control rod dropped by the operating procedures.

PWR reactors operate with the aid of neutron ex-core detectors that monitor the neutron flux, providing signals to indicate the state of the operation, control, and protection of the reactor. In a control rod drop event, there is a variation in the power distribution in the reactor core. Ex-core detectors respond to the distortion in power distribution in the core. Consequently, ex-core detector responses, which were quite similar, change considerably.

* Corresponding author. Tel.: +55 21 2560 4987.
E-mail addresses: tsouza@nuclear.ufrj.br (T.J. Souza), aquilino@lmp.ufrj.br (A.S. Martinez), canedo@lmp.ufrj.br (J.A.C.C. Medeiros), dapalma@cnen.gov.br (D.A.P. Palma), alessandro@nuclear.ufrj.br (A.C. Gonçalves).

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The aim of this work is to develop an online method to identify, in real time, a control rod dropped in the core of a PWR reactor based on the readings of the ex-core detector responses at any time.

1.1. Ex-core detector responses

A mathematical model has been reported in literature (Crump and Lee, 1978) (Seon et al., 2002) to calculate ex-core detector responses. In previous works (Sadde, 2000) a model was used to determine the ex-core detector response in the Angra-1 PWR reactor, based on weighting functions which are determined by means of the neutron transport theory. The work compares the detector responses obtained with the values from Angra-1 and with the figures calculated by the APA (Nguyen and Rathkopf, 1994) system in the same reactor conditions. The results show good alignment.

In this work, the model used to determine the ex-core responses is similar to the one used in the APA system. The model relates detector response with core power distribution for a given reactor configuration. The responses of the four ex-core detectors were thus calculated with the average power ratios for five fuel assemblies in each quadrant.

2. Core description

The core considered in this work is similar to that of the Angra-1 PWR, a Westinghouse reactor, with 121 fuel assemblies and 33 control rods, shown in Fig. 1. The ex-core neutron detectors NE-41, NE-42, NE-43, and NE-44 are located outside the reactor core and are placed in four radial positions in the concrete protection that surrounds the reactor vessel.

3. CNFR code description

The power distributions in the reactor core were generated using the CNFR – Código Nacional de Física de Reatores – (Palma et al., 2013). The CNFR code can simulate the behaviour of PWR reactors Angra-1 and Angra-2 in a steady state, solving neutronic phenomena models, thermal hydraulic and isotopic decay that characteristic to these types of reactors.

The CNFR generates the distribution of the average neutron flux in the core of PWR reactors by solving the neutron diffusion equation in three-dimensional Cartesian Geometry for two energy groups, using a nodal expansion method (NEM) Finnemann et al., 1977.

With the CNFR code it is possible to simulate different configurations for control rod drop and obtain the core power distribution for these cases.

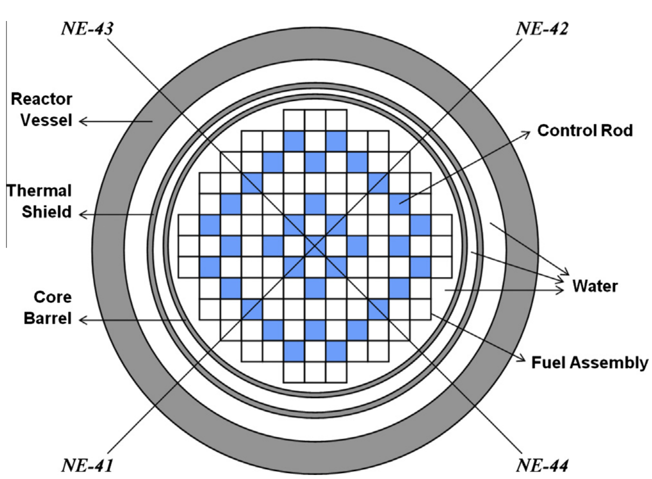

Fig. 1. Reactor layout.

Table 1
Simulated burnup steps.

Burnup steps	Equivalent in days
1	0
2	3
3	20
4	60
5	80
6	120
7	140
8	180
9	200
10	240
11	260
12	280
13	320
14	340
15	363
16	368
17	373
18	380

4. Methodology

In simulations the reactor was kept at full power and all control rods withdrawn until the boron reached 10 ppm. The simulations to obtain the power distribution for a criticality condition were made for 18 burnup steps for a fresh cycle of the reactor. A reference case was simulated without a control rod drop and in all successive cases of control rod drops. In brief, apart from the reference case, 33 simulations were run for each of the 18 burnup steps, generating different configurations of the core, one for each control rod dropped. Table 1 shows the simulated burnup steps and equivalent in days.

4.1. Model to calculate ex-core detector responses

Similarly to the APA system, the methodology used to generate the ex-core responses was based on the average power ratio for the five fuel assemblies, peripheral and closest to the detectors in each quadrant. Fig. 2 shows the fuel assemblies used to calculate the responses. Additionally, Fig. 2 shows the control rod positions, numbered from 1 to 33.

The calculation made for ex-core detectors responses with a specific control rod dropped in a burnup step of the reactor is as follows:

$$\text{NE-41} = \frac{(\bar{P}_{J02} + \bar{P}_{J03} + \bar{P}_{K03} + \bar{P}_{K04} + \bar{P}_{L04})}{5} \quad (1)$$

$$\text{NE-42} = \frac{(\bar{P}_{D11} + \bar{P}_{D12} + \bar{P}_{C11} + \bar{P}_{C10} + \bar{P}_{B10})}{5} \quad (2)$$

$$\text{NE-43} = \frac{(\bar{P}_{D02} + \bar{P}_{D03} + \bar{P}_{C03} + \bar{P}_{C04} + \bar{P}_{B04})}{5} \quad (3)$$

$$\text{NE-44} = \frac{(\bar{P}_{J11} + \bar{P}_{J12} + \bar{P}_{K11} + \bar{P}_{K10} + \bar{P}_{L10})}{5} \quad (4)$$

where $\bar{P}_{ij}$ is defined as:

$$\bar{P}_{ij} = \frac{P_{ij}^{BC}}{P_{ij}}, \quad (5)$$

P_{ij}^{BC} and P_{ij} are respectively the average power of the fuel assembly (i, j) in the case of a control rod drop and average power of the fuel assembly (i, j) to a reference case without the control rod inserted into the core. Values (i) and (j) indicate the position of the fuel elements, as shown in the core map in Fig. 2.