



# Dynamic simulation and control studies of CPR1000 reactor core with advanced MSHIM control strategy

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## ABSTRACT

The advanced mechanical shim (MSHIM) control system and a typical  $\Delta T$  protection system are built on MATLAB/Simulink for the CPR1000 reactor core to verify the feasibility and evaluate the safety margin of the reactor in transient conditions without boron adjustment. The simulation platform consists of a 1D dynamic nodal reactor core model, xenon-iodine dynamics model, power control system, AO control system, over temperature  $\Delta T$  (OTAT) protection system and over power  $\Delta T$  (OPAT) protection system. Based on this platform, three typical operational transients, namely the step increase in load from 90% FP to 100% FP, the 5% FP/min ramp load change from 30% FP to 100% FP and the 12-3-6-3 daily load follow are simulated to study the control effect of the new control system and assess the safety margins during these processes. The simulation results demonstrate that the MSHIM control system has satisfactory regulation capability for the CPR1000 reactor core in these operational transients with no actuation of the protection system and certain safety margins. The results also indicate that the MSHIM control strategy is a good alternative to MODE-G for CPR1000 to improve the performance of the core control system, which could be important references for the further engineering practice.

## 1. Introduction

The CPR1000 is the main type of pressurized water reactor (PWR) of China General Nuclear Power Group (CGNPG) operated for years since it was introduced in 2004 (Pu, 2008). As a Generation II plus PWR, CPR1000 uses the famous MODE-G control strategy to adjust the power level and axial power distribution. MODE-G is an old operation mode developed by Framatome around 1975, for which the core reactivity effects due to the power changes are compensated by control rods and the long term reactivity changes caused by fuel burnup and xenon poisoning are compensated by the boration and dilution of boron acid (Gautier, 1985). MODE-G also is a mature operation mode in French 900 MWe series nuclear power plants, such as Daya Bay nuclear power plant, which allows a more flexible operating range during load variations. One of its benefits is the improved load follow ability that enables the reactor to produce rapid increase in power, while at the same time the waste water generated during the boration and dilution of boron acid in load follow is still large especially at the end of lifetime (EOL). For the sharply increased moderator feedback coefficient and low boron concentration in coolant at EOL, more water is needed to dilute the boron concentration to compensate the power defect due to power variation, which makes the load follow at EOL almost

impossible. According to Framatome, MODE-G only allows the load follow in 80% lifetime of the reactor (Ma and Yao, 2004), which extremely limits the performance of core control system.

Many efforts have been made to overcome this issue and it can be generally solved in three ways at present. The first approach is to maintain the function of boron acid while by reducing the moderator temperature or predicting the boron concentration precisely to compensate the reactivity changes during load variation. Meyer et al. (1978) utilized the moderator temperature reductions to improve the return to power capability and then extended the cycle life time that load follow can be performed. Demonstrations implemented at Tihange Nuclear power plant showed that it is an effective way to improve the range of a rapid power escalation (Meyer et al., 1982). Mathieu and Distexhe (1986) modeled the primary loop, the pressurized circuit and the chemical and volume control system to accurately calculate the boron concentration required in primary coolant that helps to precisely predict the boron makeup. Sohn and Lee (2011) developed a precise boron concentration prediction model to guide the boration and dilution operations during load follow, which enhanced the boron concentration control and reduced the time delay in boration and dilution operations. These two methods essentially are improvement programs of the original control strategy for the issue of waste water is not solved.

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The second way is to totally give up the regulatory role of soluble boron and then controlling the reactivity in reactor core merely by burnable poison and control rods. Sugnet and Yedidia (1989) introduced a new concept PWR core design which has more fuel assemblies and control rod drives than conventional PWRs with boron acid. More gadolinium burnable poisons are also needed in this PWR to suppress the excess reactivity controlled by boron acid before. Thomet (1999), Deffain et al. (1999) and Fiorini et al. (1999) systematically discussed the core design, control strategy and the consequence to the safety system of a soluble boron free French 900 MWe PWR. In their works, the detail fuel and burnable poisons assembly design, loading pattern and the control rod movement were optimized and based on the new reactor core the feasibility of the control system had been studied. This boron free PWR has completely different fuel assemblies, namely the  $19 \times 19$  lattice fuel assembly, and a complicated core control system to minimize the power peak. Boron free PWRs have brand new designs, therefore, it is unsuitable for the traditional Generation II PWRs to upgrade. Morita et al. (1988), Morita and Carlson, 1991 proposed the advanced mechanical shim (MSHIM) operational control strategy at the end of 1980s, which keeps the soluble boron to compensate the reactivity caused by burnup periodically while the power defect and xenon poison are controlled by two independent control banks. Periodical boron adjustment largely reduced the generation of waste water, and mere rod control enables a fast response of rapid power variation. The MSHIM control strategy not only combines the advantages of those two control strategies mentioned above but also avoids the complication of the core control system and little modification is required for the original system, and it has successfully applied in advanced generation III PWR AP1000 (Onoue et al., 2003).

According to the advanced light water reactor utility requirements document (URD) (EPRI, 1999) and European Utilities' Requirements (EUR) (LOKHOV, 2011), the third generation PWRs should not change the boron concentration during load follow in order to reduce the waste water during load variation. CPR1000 is a mature commercial nuclear power plant (NPP) and the fewer changes made on its control system the more stable the system is, so the advanced MSHIM would be a preferred improvement direction for CPR1000. Moreover, the core structure of CPR1000 is similar to AP1000, both of which have 157 fuel assemblies ( $17 \times 17$  lattice design), making the improvement on CPR1000 easier to carry out. Ma and Yao (2004) have conducted the feasibility analysis of load follow without boron adjustment on M310 which is the precursor of CPR1000. Wang et al. (2014a, 2014b) have built a multi-node transfer function model of CPR1000 to analysis the MSHIM control capability under load follow simulation. Zhang et al. (2015) also compared the differences of CPR1000 during load follow under MODE-G and boron free conditions based on a time-dependent one-dimensional diffusion model. But their researches either do not consider the control systems in their simulations, or analysis the safety margins during the process, so it can't reflect the actual result of the new control system.

In this paper, a dynamic simulation and control study platform for CPR1000 has been developed in MATLAB/Simulink. The reactor core model adopts a 1D nodal model which can calculate the axial offset (AO, power difference between the top and bottom halves of the core) of reactor power and the temperature distribution of fuel rod along the axial direction. The advanced MSHIM control system, consisting of AO control subsystem and power control subsystem, is utilized to replace the MODE-G control system of CPR1000, which makes the reactor control far more automatic. In addition, the over temperature  $\Delta T$  (OTAT) protection system and over power  $\Delta T$  (OPAT) protection system are also added to the core control system, bringing the simulations more realistic. Based on the simulation platform, typical 10% full power (FP) step increase in load, 5% FP/min ramp load change and the 12-3-6-3 daily load follow were performed. The safety margins during these operational transients were also evaluated by the  $\Delta T$  protection system.

This paper is organized as follows. Section 2 gives a description of

Table 1

Main design parameters of CPR1000.

| Parameters                            | Value          |
|---------------------------------------|----------------|
| Core thermal power (MW)               | 2895           |
| Fuel assembly number                  | 157            |
| Fuel assembly type                    | $17 \times 17$ |
| Primary loop pressure (MPa)           | 15.5           |
| Coolant inlet temperature (FP) (°C)   | 293            |
| Average Coolant temperature (FP) (°C) | 310            |

the mathematical model of the CPR1000 reactor core. The advanced MSHIM control system, OTAT protection system and OPAT protection system are introduced in section 3. The simulation results and discussions are covered in Section 4. The conclusions are drawn in section 5.

## 2. Model description

The mathematic model of CPR1000 reactor core, including a 1D dynamic nodal neutron kinetics model and the corresponding thermal-hydraulics model, was established. For long time transients and large load variations, the xenon-iodine dynamics model should also be considered. In order to precisely calculate the transient response of coolant temperature, the heat transfer model used between fuel rod and coolant was the famous Mann's model (Kerlin, 1978) which consists of a fuel temperature node and two coolant temperature nodes in each axial section. The design parameters of CPR1000 reactor core are listed in Table 1.

### 2.1. Neutron kinetics model

The CPR1000 reactor core was divided into 16 nodes along the axial direction in this study and the one dimensional two-group time dependent neutron diffusion equation and associated equations for delayed neutron precursors' concentrations of the core can be expressed as

$$\begin{cases} \frac{1}{v_1} \frac{\partial \Phi_1(r, t)}{\partial t} = D_1(r) \nabla^2 \Phi_1(r, t) - \sum_{a1}(r) \Phi_1(r, t) - \sum_{s12}(r) \Phi_1(r, t) \\ \quad + (1 - \beta) \left[ v \sum_{f1}(r) \Phi_1(r, t) + v \sum_{f2}(r) \Phi_2(r, t) \right] \\ \quad + \sum_{k=1}^6 \lambda_k c_k(r, t) \\ \frac{1}{v_2} \frac{\partial \Phi_2(r, t)}{\partial t} = D_2(r) \nabla^2 \Phi_2(r, t) - \sum_{a2}(r) \Phi_2(r, t) + \sum_{s12}(r) \Phi_1(r, t) \\ \frac{\partial c_k(r, t)}{\partial t} = \beta_k \left[ v \sum_{f1}(r) \Phi_1(r, t) + v \sum_{f2}(r) \Phi_2(r, t) \right] - \lambda_k c_k(r, t); \\ k = 1, 2, \dots, 6 \end{cases} \quad (1)$$

where  $\Phi_1(r, t)$  and  $\Phi_2(r, t)$  are neutron flux distribution of fast group and thermal group, and  $c_k(r, t)$  is the density of delayed neutron precursor in family  $k$ , both of which are functions of space and time;  $D_1$ ,  $D_2$ ,  $\sum a1$  and  $\sum a2$  are diffusion coefficients and macroscopic absorption cross sections of fast group and thermal group, respectively;  $\sum s12$  is fast slowing down cross section, and  $v \sum f1$  and  $v \sum f2$  are fast and thermal  $\nu$ -fission cross sections;  $v_1$  and  $v_2$  are neutron velocities of fast group and thermal group;  $\lambda_k$  and  $\beta_k$  are decay constant and fractional yield of delayed neutron precursor family  $k$ ;  $\beta$  is the total fractional yield of delayed neutron precursors per fission.

Integrating all the terms in Eq. (1) over node  $i$  and then the equation becomes

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