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Count Rate Analysis of the Source-range Detector for M310

Jiandi Guo[1], *Zhouyu* Liu[1], *Qingming* He[1*], *Danbin* Huang[2], *Ziwei* Lin[2], *Liangzhi* Cao[1], *Hongchun* Wu[1]

[1]School of Nuclear Science and Technology, Xi'an Jiaotong University, Xi'an, Shaanxi 710049, China
[2]Fujian Fuqing Nuclear Power Co., Ltd, China

Abstract. This paper analyses the count rate of the ex-core source-range detector for Unit 4, M310 Pressurized Water Reactor (PWR), at Fujian Fuqing Nuclear Power Co., Ltd (FQNPC). The analysis encompasses power states during Cycle 1 startup and Cycle 3 fuel unloading. The analytical process involves analysing the count rate of the source-range detector, simulating secondary neutron source intensity and calculating nuclide information and neutron source intensity of irradiated fuel assemblies, etc. The maximum error between the theoretically calculated and measured count rate of the source-range detector is found to be 8.318%, which confirms the accuracy and validity of the count rate analysis method applied to the ex-core source-range detector for M310.

1 Introduction

During the fuel loading and startup process of a nuclear reactor, in order to ensure criticality safety, the reactor core should be under the supervision of ex-core source-range detectors throughout the process[1]. Due to the very low neutron fluence rate in the core, neutrons cannot be effectively detected by ex-core source-range detectors. Implementing neutron sources enables the ex-core source-range detector to maintain a higher initial count rate when the reactor operates in above states. With the exception of the Pressurized Water Reactor (PWR) designed by Russia, all PWRs designed by China, France, or the United States incorporate external neutron sources. The primary neutron source (252Cf) is employed during Cycle 1 startup, while the second neutron source (Sb-Be), irradiated by the in-core neutrons, is utilized for reload cycles startup[2].

The design life of the secondary neutron source is about 15 years. At the end of its lifespan, the cladding's strength will significantly diminish, rendering it more susceptible to damage from external forces. This increase in vulnerability considerably heightens operational and inspection risks. In addition, the secondary neutron source will persistently generate tritium in the reactor, resulting in environmental pollution. Currently, numerous PWR nuclear power plants (NPP) in the United States and France, initially designed with external sources for startup, have transitioned to employing passive startup mechanisms in reload cycles.

* Corresponding author: qingming_he@xjtu.edu.cn

Compared with the full loading state in the next cycle, the reactivity without the boron and control rods of the shutdown state in the previous cycle is much smaller. This is why it is safer to do experiments during the shutdown state in the previous cycle. During the Cycle 3 overhaul of M310, efforts were made to minimize the influence of secondary neutron source components on the detectors. This was achieved by placing the lattice containing secondary neutron source rods at a significant distance from the detectors positioned at the B11 and P11 within the M310 core. This placement aimed to assess whether the irradiated fuel assembly triggers the minimum count rate requirements of the source-range detector. If the experiments prove that the count rate of the source-range detector meets the minimum limit, it can be speculated that the limit can also be met during Cycle 4 startup without a secondary neutron source. But speculation is not enough; theoretical calculations are needed to confirm it. Hence, accurate theoretical models capable of simulating the intensity of secondary neutron sources, nuclide information and neutron source intensity of irradiated fuel assemblies, and the count rate of ex-core source-range detectors are crucial for designing rational fuel loading and unloading procedures.

Prior research predominantly concentrated on simulating the count rate of the source-range detector during fuel loading using the Monte Carlo method[3], or the subcritical power state using deterministic methods[4], specifically for Cycle 1 of PWR. However, there are few studies on the simulation of source-range detector in the reload cycles during fuel loading and unloading process. To calculate the power state of a PWR core with partially filled fuel, Monte Carlo (MC) method is employed to simulate the count rate of the ex-core source-range detector. Additionally, the deterministic method is utilized for conducting the PWR core burnup calculation.

The subsequent sections of this paper are organized as follows: Section 2 presents the analysis methods employed for evaluating the count rate of the ex-core source-range detector for M310. Theoretical calculation models and results are detailed in Section 3. Finally, Section 4 provides a summary of the work conducted.

2 Theoretical analysing method

The theoretical analysing method of count rate analysis of source-range detector of M310 by NECP-MCX[5] and Bamboo-C[6] is shown in Fig. 1. Firstly, a refined 3D full-core M310 model is established, utilizing NECP-MCX for the neutron transport criticality calculation to validate the accuracy of the M310 model. Subsequently, the thermal neutron fluence rate within the sensitive area of the source-range detector is determined by solving the fixed source problem of neutron transport during Cycle 1 startup for M310. This calculated value, combined with measured data, is then employed to calibrate the thermal neutron sensitivity coefficient of the detector. Thirdly, during Cycle 3 fuel unloading of the M310, NECP-MCX is utilized to simulate the intensity of secondary neutron source, while Bamboo-C is employed to calculate the full-core burnup to get nuclide information and neutron source intensity of irradiated fuel assemblies. The results of the third step will serve as input for NECP-MCX, combined with the calibrated thermal neutron sensitivity coefficient, to calculate the count rate of the source-range detector during Cycle 3 fuel unloading of M310.

2.1 M310 PWR

M310 PWR at FQNPC is designed by French Framatome. The core is loaded with 157 fuel markings in Fig. 2. Additionally, control rod loading in the M310 core are also identified in Fig. 2.

The M310 core contains two primary neutron source assemblies and two secondary neutron source assemblies. The position of the primary neutron source and the secondary

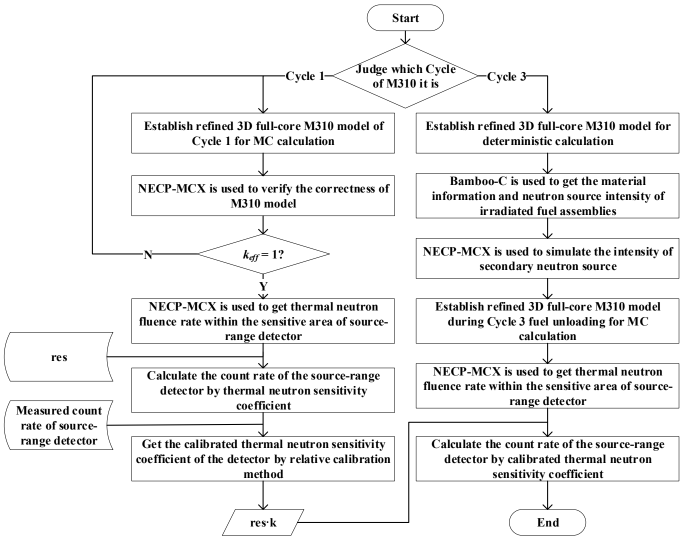


Fig. 1. The flow chart of count rate analysis of source-range detector of M310.

neutron source assemblies in the core is shown in Fig. 3, while the location of the primary neutron source and the secondary neutron source rods in the assembly is shown in Fig. 4.

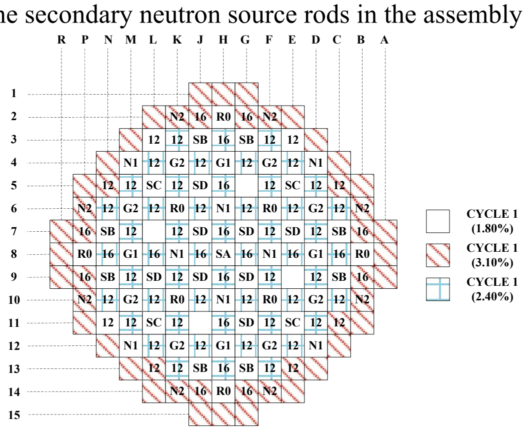


Fig. 2. Fuel assembly, control rod and poison rod loading in Cycle 1.

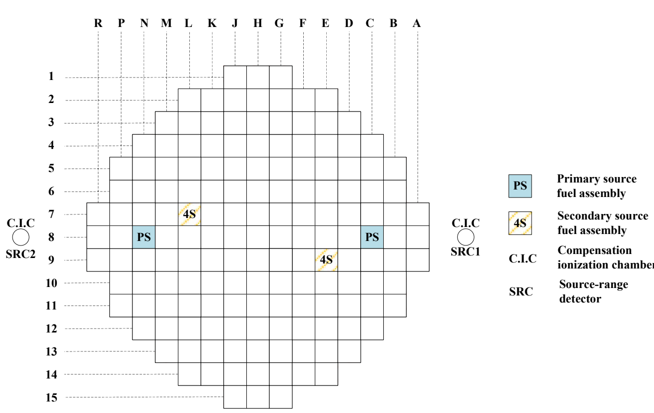


Fig. 3. The position of the primary neutron source and the secondary neutron source assemblies in the core of Cycle 1.

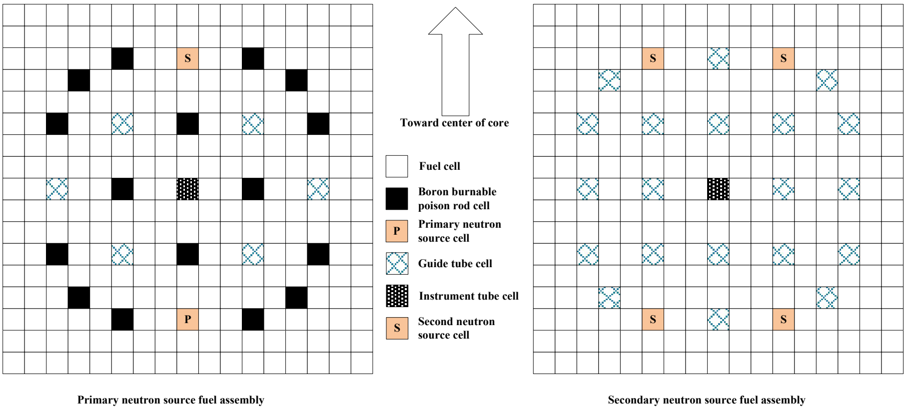


Fig. 4. The primary neutron source and the secondary neutron source rod locations in the assembly.

2.2 Simulation of the primary neutron source

The primary neutron source in the axial direction is about 4cm high and located in the lower 1/4 of the core active area. During startup, the intensities of the two primary neutron sources are 2.62E+08n/s and 2.64E+08n/s respectively. The expression of the neutron emission spectrum of ^{252}Cf is:

$$f(v) = \sqrt{\left(\frac{m}{2\pi kT}\right)^3} 4\pi v^2 e^{-\frac{mv^2}{2kT}} \tag{1}$$

where k is Boltzmann's constant, chosen as 1, T is the nuclear temperature, chosen as 1.43MeV, mv^2 is the neutron energy, and $f(v)$ is the probability on the left side of the equation.

2.3 Source-range detector modelling

In M310, the position of the compensation ionization chamber and source-range detector are illustrated along the radial direction in Fig. 3. Axially, the central portion of the sensitive area of the source-range detector is aligned with the primary neutron source. Table 1 shows the modelling parameters detailed of the source-range detector.

Table 1. Modelling parameters of the source-range detector.

Parameters		**ZJ1520**
Mechanical dimensions	Maximum outer diameter of tube	50mm
	Sensitive zone length	575mm
Materials	Cathode	Ti
	Anode	Au-W
	Insulator	Al_2O_3
Sensitive materials	Ingredients	Abundance>92% ^{10}B
	Mass thickness	0.6~0.8mg/cm^2

2.4 Calculation method for count rate of the source-range detector

In practical engineering applications, the theoretical calculation formula of the count rate of the source-range detector is:

$$R_d = \overline{\phi_d} \cdot res \tag{2}$$

where R_d is the calculated count rate of the detector (cps), $\overline{\phi_d}$ is the calculated thermal neutron fluence rate at the sensitive area of the detector (n • cm^{-2} • s^{-1}), res is thermal neutron sensitivity coefficient of the detector (cps • n^{-1} • cm^{2} • s), with the value set at 8.

However, several challenges arise in practical applications and calculations. The simplification of the reactor calculation model diverges from the actual physical model, while accurately simulating the intricate structure of the detector remains challenging, resulting in inherent approximations in the modelling process. Typically, there exists a discrepancy between the provided and actual thermal neutron sensitivity coefficient of the detector once in operation at the nuclear power plant (NPP) after leaving the factory. Given these challenges, to accurately compute and acquire the response of the source-range detector, it becomes essential to initially calibrate the thermal neutron sensitivity coefficient of the detector. This calibration process aims to align the sensitivity coefficient used in the numerical calculation model with the actual PWR core. The implementation is as follows:

$$R_m = \overline{\phi_d} \cdot res \cdot k \tag{3}$$

where R_m is the measured count rate of detector (cps), k is the dimensionless calibration coefficient between calculated and measured value. The calibration thermal neutron sensitivity coefficient $(res \cdot k)$ obtained by the above formula is used in the Cycle 3 fuel unloading process to obtain the true calculated value of the detector response.

The relative calibration method is employed to calibrate the thermal neutron sensitivity coefficient of the source-range detector for M310. The fundamental concept behind this approach involves utilizing the sum of variances between the ratio of the theoretically calculated value to the measured value and 1 for calibration purposes. Employing the principle of least squares, the calibrated thermal neutron sensitivity coefficient is determined when the variance between the two reaches its minimum. This calibrated coefficient is then utilized to calibrate the count rate of the source-range detector during the Cycle 3 fuel unloading. The specific calculation formula is as follows:

$$Y_{min} = \sum_{i=1}^{4} \left(1 - k\frac{R_i}{R_{m,i}}\right)^2 \tag{4}$$

where Y_{min} is the minimum variance between the theoretically calculated and the measured value, i is the serial number of the source-range detector, $R_{m,i}$ is the measured value of the ith detector, R_i is the calculated value of the ith detector, k is the calibration coefficient.

2.5 Secondary neutron source intensity simulation

The secondary neutron source comprises Sb-Be blocks encased within a stainless steel cladding. The neutron generation principle is depicted in Fig. 5. To conduct simulation calculations, the process can be divided into the following three steps:

$$^{123}Sb + n \rightarrow {}^{124}Sb \tag{5}$$

$$^{124}Sb \rightarrow {}^{124}Te + \beta + \gamma \tag{6}$$

$$^{9}Be + \gamma \rightarrow {}^{8}Be + n \tag{7}$$

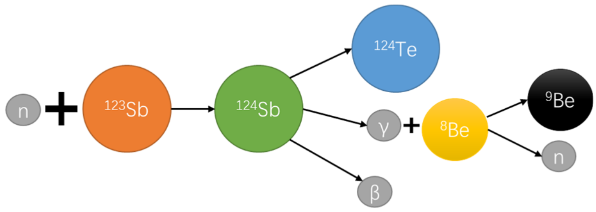


Fig. 5. Neutron generation mechanism of Sb-Be blocks.

2.5.1 Calculation of the activity of ^{124}Sb in the secondary neutron source

Given the irradiation of ^{123}Sb within the secondary neutron source block by a constant neutron fluence rate, the rate of change of the atomic number of ^{124}Sb at time ‘t’ is:

$$\frac{dN}{dt} = N_a\sigma\bar{\phi} - \lambda N \tag{8}$$

where N_a is the atomic amount of ^{123}Sb, σ is microscopic cross section for the reaction between ^{123}Sb and neutrons to generate ^{124}Sb, $\bar{\phi}$ is the irradiation neutron fluence rate, λ is the decay constant of ^{124}Sb, N is the atomic amount of ^{124}Sb. From the Eq. (8), the activity of ^{124}Sb in the secondary neutron source at time ‘t’ is:

$$A(t) = N_a\sigma\bar{\phi} * (1 - e^{-\lambda t}) \tag{9}$$

From Eq. (9), it's evident that with prolonged irradiation of the neutrons in the reactor, ^{124}Sb will eventually reach saturation. Post Cycle 3 shutdown, the decay of ^{124}Sb begins. According to the irradiation time ‘t_1’ of the secondary neutron source during Cycle 3 and the decay time ‘t_2’ during the fuel unloading, ^{124}Sb is approximated to be calculated based on Eq. (10):

$$A^* = N_a\sigma\bar{\phi} * \left(1 - e^{-0.693*\frac{t_1}{60.2}}\right) * e^{-0.693*\frac{t_2}{60.2}} \tag{10}$$

The model of the fuel assembly containing secondary neutron source rods is established and computed. The nuclide information regarding the MC model material is acquired through the burnup calculation results of M310. Solving the critical neutron transport problem results that the volume activity of ^{124}Sb in the Sb-Be block during the Cycle 3 fuel unloading is 7.68E+11Bq/cm^3. Multiplying this volume activity by the actual volume of the block yields the total activity of ^{124}Sb.

2.5.2 Calculation of the γ fluence rate of the Sb-Be block

The minimum energy requires for the ^{9}Be(γ ,xn) reaction is 1.67MeV[7], which means that γ -rays emitted by ^{124}Sb with energies greater than 1.67 MeV are considered effective γ -rays[8]. The total absolute intensity of these effective rays is 53.58%. The source intensity of these effective rays can be determined by multiplying the activity of ^{124}Sb by the absolute intensity of the effective ray. The calculated effective γ source intensity for a fuel assembly amounts to 5.58E+13s^{-1}.

The predominant contribution to the neutron source strength of the Sb-Be core block within a secondary neutron source rod originates from its self-emitted γ -rays[9]. Negligible contributions arise from γ -rays emitted by other secondary neutron source rods and fuel rods within the same assembly. As the source intensity and energy spectrum distribution are similar for each secondary neutron source rod in the assembly, a model is established for a single secondary neutron source rod. Utilizing NECP-MCX to resolve the photon transport fixed-source problem yields the photon fluence rate spectrum of the secondary neutron source rod.

2.5.3 Calculation of neutron source indensity

By employing the effective γ fluence rate spectrum within the secondary neutron source Sb-Be core block, the reaction rate of the ^{9}Be(γ,xn) nuclear reaction can be derived using the microscopic cross-section of the ^{9}Be(γ,xn) nuclear reaction[7]. Multiplying the atomic amount of ^{9}Be by the reaction rate of the ^{9}Be(γ,xn) nuclear reaction yields the neutron source intensity of the secondary neutron source rod. The calculated neutron source intensity for a single secondary neutron source rod amounts to 1.13E+09 n/s.

3 Calculation models and results

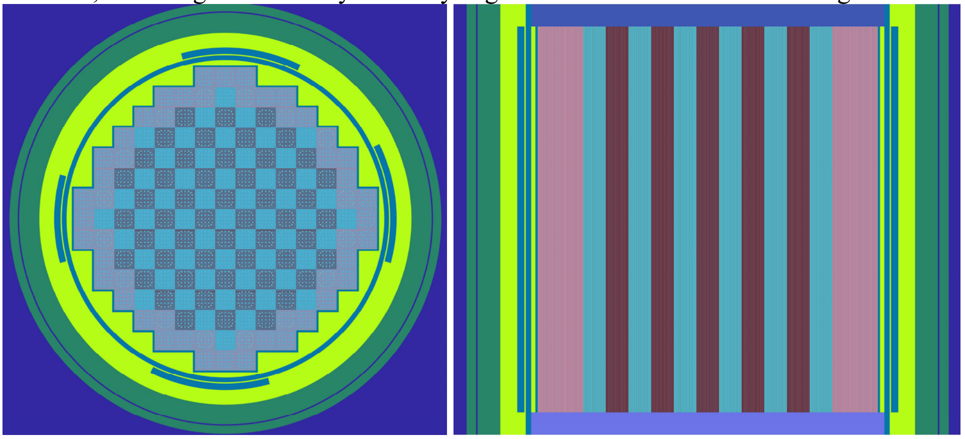
Section 3.1 covers critical state calibration, count rate analysis of the source-range detector during Cycle 1 startup, and the calibration of the thermal neutron sensitivity coefficient of detector. In Section 3.2, the analysing results of the secondary neutron source intensity and burnup of M310 serves as input for simulating the count rate of the source-range detector during Cycle 3 fuel unloading.

3.1 Count rate analysis of source-range detector during Cycle 1 startup

In Section 3.1.1, the reactor's effective multiplication factor (k_{eff}) for Cycle 1 of M310 is calculated for criticality calibration. Section 3.1.2 involves selecting various power states during startup and attaining criticality for calculation purposes.

3.1.1 Calibration of critical state

A refined 3D full-core model of M310 is depicted in Fig. 6. The MC calculation results of k_{eff} are presented in Table 2, showcasing an error of 74 ppm between the calculated and measured values. This validates that the MC model aligns with the calculation prerequisites for M310, affirming its suitability for analysing the count rate of the source-range detector.



(a) Radial section view of M310 core model (b) Axial section view of M310 core model

Fig. 6. MC model of M310 Cycle 1.

Table 2. k_{eff} for Cycle 1 of M310.

	MC calculated value	Measured value	Error/pcm
k_{eff}	0.99926	1.00000	-74

3.1.2 Count rate analysis of source-range detector

Two specific power states were chosen for calibrating the thermal neutron sensitivity coefficient: (1) The reactor operated in a thermal shutdown state, with SA, SB, and SC control rods positioned at the top of the reactor at 225 steps, and the remaining rods placed at the bottom of the reactor at 5 steps; (2) Building upon the previous power state, additional adjustments were made by increasing the SD, N1, N2, G1, and G2 control rods to 225 steps, and elevating the R control rods to 100 steps. The MC model, consistent with Fig. 4, is employed, with variations limited to the control rod positions and boron concentration.

As illustrated in Table 3, after relative calibration of the data, the maximum error is only -8.318% between the calculated and measured values of the detector's count rate. This indicates a strong agreement between the theoretical and measured count rates. The calibrated thermal neutron sensitivity coefficient is determined as 9.10 $cps{\cdot}n^{-1}{\cdot}cm^{-1}{\cdot}s^{-1}$, and will be utilized for the calculation of the Cycle 3 fuel unloading process.

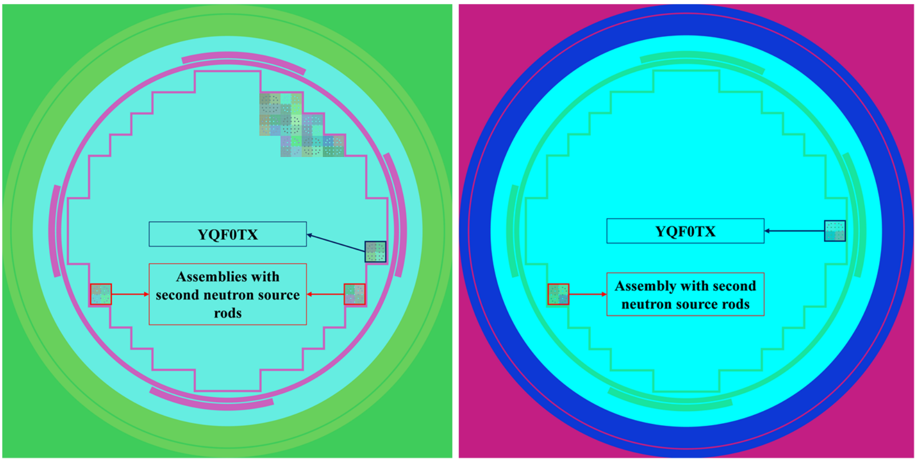
Table 3. The theoretically calculated and the measured value of the count rate of the source-range detector during Cycle 1 startup of M310.

State	Detector	Variance of tally (%)	Error (%)	Calibrated thermal neutron sensitivity coefficient
1	SRC1	0.372	8.095	9.10
	SRC2	0.396	5.553	
2	SRC1	0.285	-7.557	
	SRC2	0.297	-8.318	

3.2 Count rate analysis of source-range detector during Cycle 3 fuel unloading

The simulation of the startup process with the primary neutron source cannot fully replicate the actual conditions experienced during the reloading process after replacing the secondary neutron source assembly with irradiated fuel components. This simulation also cannot validate the accuracy of source intensity and energy spectrum calculations for the irradiated fuel components. In accordance with the fuel unloading process of M310 Cycle 3, the irradiated fuel assembly, YQF0TX (with an enrichment of 3.70% and a burnup of approximately 26580MWd/tU), was successively placed in core positions A09 and A08 for measurements of the detector responses.

The nuclear density of the fuel during the Cycle 3 fuel unloading of M310 is computed. The sources for the MC model in M310 fuel unloading comprise three components: the secondary neutron source, the spontaneous fission neutron source, and the neutron source generated through (α ,n) reactions within the irradiated assemblies. Refined 3D models of the entire core are established, as depicted in Fig. 7. The fitting calculated results, employing the calibrated thermal neutron sensitivity coefficient calculated in Section 3.1.2, are illustrated in Table 4.



(a) Fuel assembly loading at step 146a (b) Fuel assembly loading at step 155a

Fig. 7. MC calculation model during Cycle 3 fuel unloading of M310.

Table 4. The theoretically calculated and the measured value of count rate of source-range detector during Cycle 3 fuel unloading of M310.

Step	Detector	Variance of tally (%)	Error (%)	Calibrated thermal neutron sensitivity coefficient
146a	SRC1	1.711	-6.133	9.10
155a	SRC1	0.727	-2.419	

Based on the data from Table 4, the following conclusions can be drawn:

(1) The maximum error between the calculated and the measured value of the detector's count rate is only -6.133%, indicating a strong agreement between the theoretical calculation and the measured count rate.

(2) When positioning the assembly with secondary neutron source rods at P11 and YQF0TX assembly at B11, the influence of the secondary neutron source on the count rate of the source-range detector SRC1 is minimal, and the count rate consistently exceeds 0.5 cps, meeting legal and regulatory requirements.

4 Conclusions and discussions

The count rate analysis method of the source-range detector for M310 employs NECP-MCX and Bamboo-C. Considering the diverse simulated power states and the intricate model complexity during fuel loading and unloading, the MC method is chosen to simulate the count rate of the ex-core source-range detector for M310 PWR. To optimize computational time and resources, the deterministic method is utilized for core burnup calculations.

Based on the count rate simulation results of the ex-core source-range detector, the following conclusions are drawn: (1) The error between all calculated and measured count rates of the source-range detector is within 10% during both Cycle 1 startup and Cycle 3 fuel unloading. (2) Throughout Cycle 3 fuel unloading, despite the assembly with secondary neutron source rods being placed at a distance from the detector SRC1, the measured and calculated count rates exceed 0.5 cps with the irradiated fuel assembly YQF0TX.

The count rate analysis method of the source-range detector is validated, demonstrating its applicability for studying M310 startup without secondary neutron sources. Nevertheless, accurately calculating the count rate of the ex-core source-range detector and comprehensively predicting the passive startup process purely through theoretical means proves to be extremely challenging. To address this, a substantial number of physical experiments and measurements during the unloading of the front cycle are imperative, aligning with the requirements for passive startup. On the one hand, the measured data is used to check the theoretical calculations and determine the design accuracy; on the other hand, it directly contributes to the passive startup of reload cycles by offering crucial verification and reference data.

Acknowledgements

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