Neutronic Analysis of Accident-Tolerant Control Rod in APR-1400 using STREAM/RAST-K Code System

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1. Introduction

During a Loss-of-Coolant Accident (LOCA), reactor core temperatures can exceed 1200 ℃. Such extreme conditions can induce chemical interactions and interdiffusion between reactor structural materials, resulting in unintended phenomena. Although most reactor materials have high melting points, certain material combinations can undergo eutectic reactions at significantly lower temperatures, leading to premature failure of critical components.

Conventional control rod materials in the pressurized water reactors (PWRs), typically composed of B_4C and Ag-In-Cd enclosed in a stainless-steel cladding, are particularly vulnerable under extreme conditions. For example, B4C has a high melting point of 2350 ℃, however forms a eutectic with Iron (Fe) in the cladding at only 1150 ℃. This temperature is lower than the threshold at which zirconium alloy undergoes rapid oxidation (~1200 ℃). It indicates that the control rods may degrade or collapse before significant core damage occurs in severe accidents [1].

To address these safety concerns, the Accident-Tolerant Control Rod (ATCR) concept has been developed as part of ongoing efforts to enhance reactor safety following the Fukushima Daiichi accident [2]. ATCR is designed to improve thermal stability and accident tolerance by incorporating ceramic-based neutron absorbers with higher melting points and greater resistance to eutectic interactions. Candidate materials include Europium (Eu), Dysprosium (Dy), etc., with high thermal neutron absorption cross-section. By minimizing the risk of material degradation and maintaining reactivity control in extreme conditions, ATCR contributes to the advancement of safer and more reliable reactor designs**.**

This paper presents a neutronic analysis of ATCR in APR-1400 core using a two-step core analysis code system, STREAM/RAST-K. The ATCR materials are evaluated for control rod worth, shutdown margin, decay heat, and gamma dose, and their performance is compared with conventional control rod materials. The results show that ATCR can maintain effective reactivity control while enhancing accident tolerance.

The control rod ejection accident analysis in APR-1400 will be conducted further to assess the performance of ATCR in accident scenarios.

2. Computational Methods

2.1 Computational Code

STREAM is a high-fidelity lattice physics code that employs the Method of Characteristics (MOC) for neutron transport calculation. It generates homogenized group constants for core simulations in RAST-K. Additionally, STREAM includes the STREAM-SNF module, which enables radiation source term analysis such as decay heat and gamma emission calculations from ATCR materials.

RAST-K is a state-of-the-art nodal diffusion code that utilizes a non-linear scheme based on multi-group Coarse Mesh Finite Difference (CMFD) acceleration with 3-D Unified Nodal Method (UNM). It is capable of performing both steady-state and transient core calculations.

In this paper, STREAM/RAST-K code system is employed to analyze ATCR-loaded fuel assemblies and the core configuration in APR-1400 reactor. This code system has been extensively validated against various commercial PWR benchmarks and incorporates essential calculation modules for accurate ATCR analysis. As a result, it enables precise evaluation of neutron behavior and reactivity effects in the core, allowing a reliable comparison with conventional control rod systems.

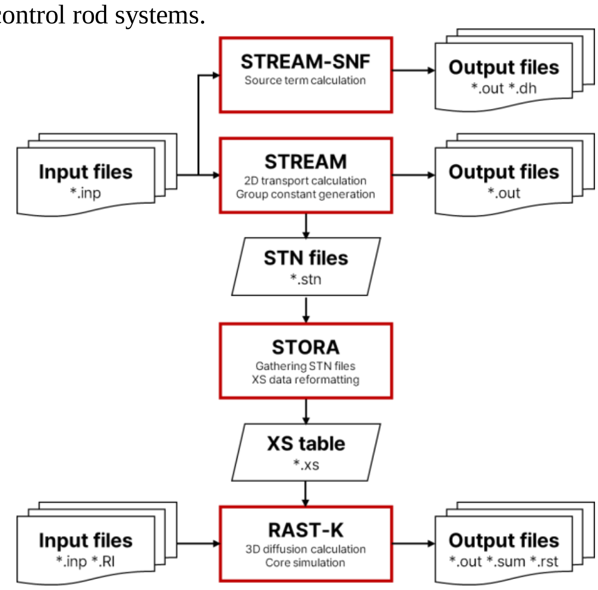

Fig. 1. Flow chart of STREAM/RAST-K code system and STREAM-SNF

2.2 Resonance Treatment in STREAM

A key feature of STREAM is its ability to accurately model resonance self-shielding effects using the Pin-Based Pointwise Energy Slowing Down Method (PSM) [3]. Accurate resonance self-shielding treatment is necessary for the neutronic analysis of ATCR, as they contain strong neutron-absorbing nuclides with significant resonance behavior. Fig. 2 shows the neutron absorption cross-section of major isotopes of ATCR materials from ENDF/B-VII.1.

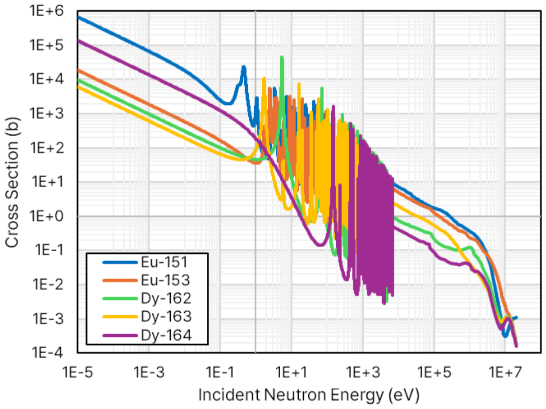

Fig. 2. Neutron absorption cross-section of major isotopes of ATCR candidate materials

Equivalence Theory (ET) has been widely used due to its computational efficiency. However, it has limitations in handling resonance interference effects and spatial self-shielding, which lead to an overestimation of the absorption cross-section.

PSM explicitly solves the neutron slowing-down equation on a fine energy grid within each pin-cell, eliminating the need for pre-tabulated cross-section data. This approach accounts for resonance scattering and interference effects, significantly improving accuracy in heterogeneous systems. PSM also captures spatial self-shielding effects, which are crucial for ATCR absorbers that contain multiple isotopes with overlapping resonance regions. It ensures reliable control rod worth predictions by providing a more physically accurate representation of resonance behavior.

STREAM allows the use of both ET and PSM for fuel assembly calculations. Table 1 presents k_{inf} values calculated using different resonance treatment methods. The calculations were performed on a 2-dimensional fuel assembly model loaded with ATCR materials (Fig. 3). The results from each method are compared to the reference value obtained from the Monte Carlo code, MCS.

To assess the impact of resonance treatment, B_4C was analyzed as a reference material. The k_{inf} difference between ET and PSM is only 60 pcm due to the absence of strong resonance absorption in ^{10}B. In the case of ATCR materials, the k_{inf} results calculated using ET for Dy-based and DyHf-based are approximately 1130 pcm and 1390 pcm lower, respectively, than the reference k_{inf} obtained from MCS. In contrast, the k_{inf} obtained using PSM shows good agreement with the reference, with differences within 30 pcm.

The results indicate that ET tends to overestimate the neutron absorption cross-section for ATCR materials. Therefore, PSM is considered a more accurate and suitable method for resonance treatment in ATCR analysis.

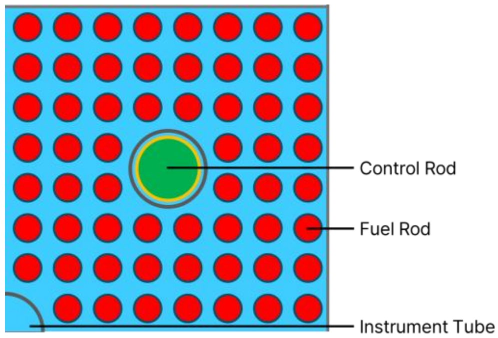

Fig. 3. Quarter fuel assembly model loaded with control rods

Table 1. k_{inf} of B_4C and ATCR loaded fuel assembly from MCS and STREAM (ET, PSM)

Materials	MCS	k_{inf} (ET)	k_{inf} (PSM)
B_4C	-	0.81664	0.81724
Dy-based	0.85028 ±0.00025	0.83901	0.85044
DyHf-based	0.85170 ±0.00025	0.83780	0.85198

3. Results and Disscussion

3.1 Control Rod Worth

Control rod worth is defined as the negative reactivity added when all regulating banks are fully inserted into the core. The worth calculation was performed under Beginning-of-Cycle (BOC) Hot Zero Power (HZP) conditions for the reference core, APR-1400. The core loading pattern and control rod placement (Fig. 4 and 5) are set as specified in the design certification document of APR-1400 [4].

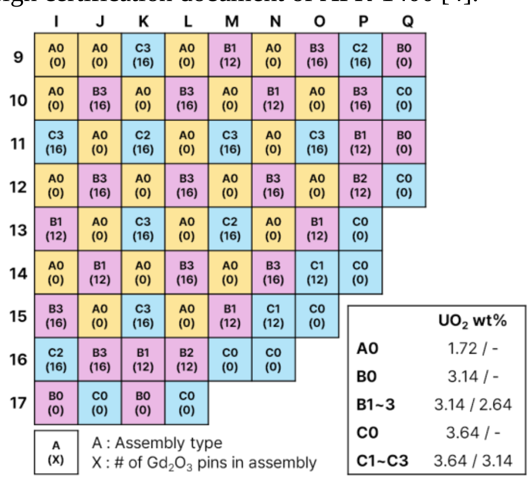

Fig. 4. Loading pattern of APR-1400 quarter core

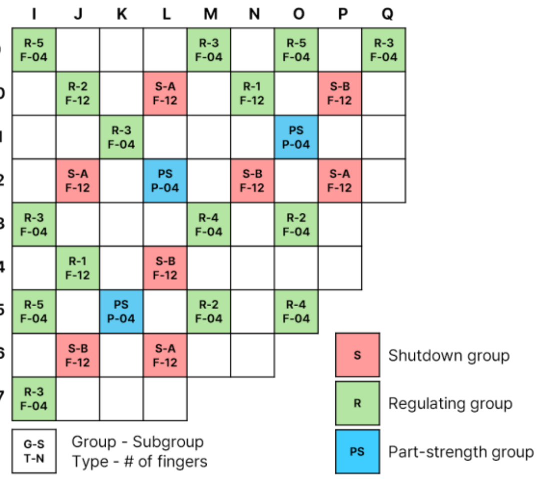

Fig. 5. Control rod arrangement of APR-1400 quarter core

Table 2 presents the total worth of the regulating banks, while Fig. 6 shows the worth as a function of insertion depth of the control rod in the core. The calculated worth for Eu-based, Dy-based ATCR materials and B_4C are 3291, 2852, and 3226 pcm, respectively. These results demonstrate that Eu-based ATCR has a higher worth than conventional control rod, confirming its superior effectiveness in reactivity control.

Table 2. Control rod worth results of ATCR and B_4C

Composition	CRW [pcm]	Diff. [pcm]
B_4C	3226	-
Eu-based	3291	65
Dy-based	2852	-374

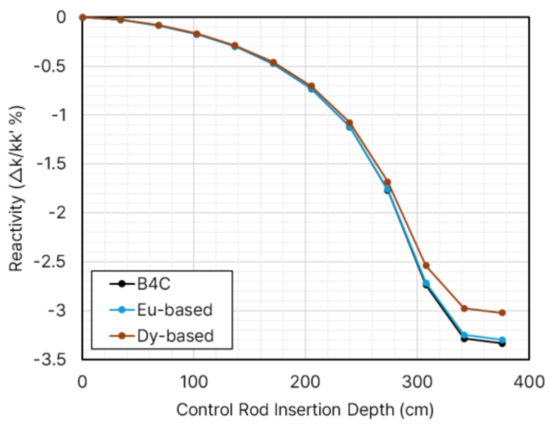

Fig. 6. Control rod worth according to rod insertion depth

3.2 Shutdown Margin

To assess the applicability of ATCR in APR-1400, the shutdown margin (SDM) is calculated for each material under End-of-Cycle (EOC) conditions, where the Moderator Temperature Coefficient (MTC) is most negative. This condition represents the highest reactivity requirement for achieving a safe shutdown.

SDM is typically calculated by subtracting the sum of power defect and control rod insertion allowance (RIA) from SCRAM1 worth. It refers to the total negative reactivity when all the control rods are fully inserted into the core, except for the highest-worth rod. In this calculation, the stuck rod is set to the shutdown bank group B at the (N,14) position in Fig. 4. All the calculated SCRAM worth values are conservatively reduced by 5% to account for code uncertainty. This ensures that the evaluation reflects a more realistic safety margin under potential modeling inaccuracies. Power defect is the positive reactivity due to the change of the core state from Hot Full Power (HFP) to HZP. RIA value at HFP condition is referenced from a reactor type similar to APR-1400 [5].

Table 3 presents the SDM values for each ATCR material. According to the design requirements of APR-1400, SDM must exceed 5500 pcm. The results show that all ATCR materials satisfy this criterion; the SDM values of Eu-based and Dy-based are 7560 and 6118 pcm, respectively. These results demonstrate the feasibility of ATCR as an alternative control rod design, confirming that it can provide adequate shutdown capability in APR-1400 while ensuring sufficient reactivity control and improved accident tolerance.

Table 3. SDM on EOC HFP condition

	B_4C (pcm)	**Eu-based (pcm)**	**Dy-based (pcm)**
SCRAM worth [A]	10036	10162	8720
Power Defect [B]		2362	
RIA [C]		240	
SDM [D=A-B-C]	7434	7560	6118

3.3 Decay Heat and Gamma Dose Rate

The decay heat and gamma radiation from activated control rod materials are crucial factors in assessing their long-term safety and handling feasibility. Although Eu-based materials demonstrate superior neutron absorption properties, they are also easy to activate under prolonged neutron irradiation, leading to the accumulation of radioactive isotopes with long half-lives. This can result in increased decay heat and gamma radiation levels, which may impose additional safety considerations in spent control rod handling and storage.

To evaluate the radiation characteristics of ATCR materials, decay heat and gamma dose calculations are performed using STREAM-SNF. Each material is assumed to be loaded into APR-1400 C0 type fuel assembly (3.64 wt% ^{235}U) irradiated up to 80 MWd/kgU and subsequently cooled for five years. The resulting decay heat and gamma dose per unit mass are summarized in Table 4 and 5, respectively.

Based on the results, Eu-based ATCR shows decay heat and gamma dose levels comparable to or even exceeding those of UO_2 fuel. After five years of cooling, the decay heat of Eu-based ATCR is 0.35 W/g, which is higher than that of UO_2 (0.22 W/g), and an order of magnitude greater than that of Dy-based ATCR (1.08E-05 W/g). Similarly, the gamma dose for Eu-based material remains 6.45 W/g after five years, drastically higher than UO_2 (7.73E-03 W/g) and Dy-based ATCR (9.06E-06 W/g).

In Eu-based ATCR, the presence of long-lived Eu isotopes such as ^{152}Eu and ^{154}Eu leads to sustained heat generation and high specific activity. Eu-based ATCR, despite its superior reactivity worth, shows post-irradiation characteristics similar to those of spent nuclear fuel, particularly in terms of decay heat and gamma radiation. In contrast, Dy-based ATCR generates lower decay heat and gamma emission levels, although it shows slightly lower control rod worth than Eu-based ATCR.

Therefore, the selection of ATCR materials should consider not only reactivity control performance but also activation characteristics to ensure operational feasibility and safety in commercial reactor applications.

Table 4. Decay heat per unit mass (W/g)

	Burnup (MWd/kgU)				Cooling Time (yr)		
	20	**40**	**60**	**80**	**0.5**	**1**	**5**
UO_2	9.14	8.80	8.90	9.14	0.09	0.07	0.22
Eu-based	0.16	0.30	0.43	0.55	0.47	0.46	0.35
Dy-based	0.17	0.24	0.32	0.40	1.10 E-05	1.09 E-05	1.08 E-05

Table 5. Gamma dose rate per unit mass (W/g)

	Burnup (MWd/kgU)				Cooling Time (yr)		
	20	**40**	**60**	**80**	**0.5**	**1**	**5**
UO_2	4.24	4.02	4.03	4.12	0.03	0.02	7.73 E-03
Eu-based	0.13	0.25	0.36	0.46	0.40	0.39	0.30
Dy-based	0.03	0.04	0.06	0.07	9.19 E-06	9.12 E-06	9.06 E-06

4. Conclusions

The development of ATCR aims to improve both reactivity control performance and safety under severe accident conditions. In this paper, neutronic behavior, control rod worth, shutdown margin, and activation characteristics of ATCR materials are evaluated in APR-1400 reactor. The neutronic analysis is performed using STREAM/RAST-K code system, with source term calculations conducted using STREAM-SNF.

Eu-based ATCR shows the highest control rod worth of 3291 pcm, followed by Dy-based ATCR with 2852 pcm. The calculated SDM values of Eu-based and Dy-based ATCR are 7560 pcm and 6118 pcm, respectively, which satisfy the SDM design requirements of APR-1400 ($\geq$ 5500 pcm). The results show that both ATCR materials have sufficient reactivity to control or shut down the reactor core.

After irradiation to 80 MWd/kgU and five years of cooling, the Eu-based ATCR produces 0.35 W/g of decay heat and 0.30 W/g of gamma dose. Conversely, Dy-based ATCR shows significantly lower decay heat (1.08E-05 W/g) and gamma dose (9.06E-06 W/g).

In summary, while Eu-based ATCR offers superior reactivity performance, its high decay heat and gamma dose rate may present challenges in post-irradiation handling. In contrast, Dy-based ATCR exhibits somewhat lower control rod worth but demonstrates significantly better post-irradiation stability. These results suggest that the design of ATCR should not focus solely on maximizing reactivity control but should also account for long-term stability, including activation characteristics. A balanced approach that considers both neutronic effectiveness and material behavior is therefore essential to develop ATCRs that are not only efficient during normal operation but also manageable under accident conditions.

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