3-D Core Design of the TRU-Incinerating Thorium RBWR Using Accident Tolerant Cladding

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Abstract. This study investigates the safety of the full core optimal design for the RBWR-TR – a reduced moderation BWR with a high transuranic (TRU) consumption rate. This design is a variant of the Hitachi RBWR-TB2, which arranges its fuel in a hexagonal lattice, axially segregates seed and blanket regions, and fits within an ABWR pressure vessel and is capable of unlimited TRU recycling as do fast reactors. The RBWR-TR eliminates the internal axial blanket, eliminates absorbers from the upper reflector, and uses thorium for the fertile fuel. Both designs are initially presented with Zircaloy-2 cladding.

The neutron spectrum in the RBWR-TR is softer than in the RBWR-TB2, which results in a lower cladding fast neutron fluence; however, the peak fluence of fast neutrons (E > 0.1 MeV) the cladding is exposed to exceeds the bounds of Zircaloy-2 at accident scenarios, limiting its material feasibility and affecting the reactor safety. The constraining phenomena which are enhanced by the high fast neutron fluence include accelerated corrosion, accelerated embrittlement rates, and hydrogen pickup. Alternative cladding materials to Zr-based alloys are being investigated for accident-tolerant scenarios. These include stainless steel based materials, which are not limited by hydrogen pickup phenomena. Since these alternative claddings have larger absorption cross sections than Zr-based alloys, the impact on the achievable discharge burnup and other key neutronics parameters is assessed. The design variables used in the parametric studies include: the cladding material, cladding size, gap between cladding and fuel, and fuel-to-moderator volume ratio. The changes of the void feedback, cycle length, burnup, shutdown margin, and critical power ratio to variation in each of the design variables are calculated to determine their impact on the design. The design presented in this paper does not exceed material bounds at same burnup value of the original design. By increasing the gap and reducing the cladding dimension we were able to meet all design constraints. However, due to significant changes in gap and cladding dimensions, the RBWR cores require further intensive studies related to swelling accommodation and pellet-cladding interaction effects.

Key Words: TRU transmutation, reduced moderation BWR, thorium, FeCrAl

1. Introduction

The RBWR-TR [1] core design is based on the Hitachi RBWR-TB2 [2], a reduced-moderation BWR that employs axial seed and blanket segregation for continuous burning of LWR transuranic waste (TRU). The discharge fuel from the RBWR-TR is recycled, and a mixture of natural thorium and reprocessed LWR TRU is added to maintain the fuel inventory. The RBWR-TR differs from the RBWR-TB2 in that it uses thorium rather than depleted uranium as the fertile component of the makeup fuel and of the axial blankets, and it eliminates the internal blanket while elongating the seed region and the outer blankets.

Reduced-moderation BWR core concepts, referred to by Hitachi as the Resource-renewable BWR (RBWR), were initially pursued by Hitachi [2] in an attempt to design hard spectrum BWRs to provide missions traditionally assigned to liquid metal cooled reactors – fuel sustainability (RBWR-AC) or TRU transmutation with unlimited recycling (RBWR-TB2)

[3,4]. As the RBWR-TB2 and RBWR-TR use water coolant, although of low density, their spectrum is softer than that of a TRU-burning sodium fast reactor (SFR) as the ABR [5] but harder than spectrum of a typical BWR, as shown in Figure 1. Figure 2 shows the spectra of neutrons inducing fission; more than half of the fissions in the RBWR-TR are induced by neutrons between 1 eV and 0.1 MeV.

Compared to the reference conversion ratio (CR)=0.5 metal-fuelled ABR, the RBWR-TR burns slightly more TRU from LWR UNF per unit of electricity generated but, similarly to the RBWR- TB2, has roughly one third of the discharge burnup, power density, and specific power. It requires a larger reprocessing capacity, but can operate in longer cycles with a comparable reactivity swing. Overall, the fuel cycle cost will be greater for the RBWR-TR than for the ABR, but the capital cost of the RBWRs is expected to be lower than the ABRs. Also relative to the reference ABR and RBWR-TB2 designs, the fuel discharged from the RBWR-TR core contains significantly less fissile Pu and significantly more ^{238}Pu per kg of spent fuel. However, the RBWR-TR also discharges significant amount of U, over 60% of which is fissile.

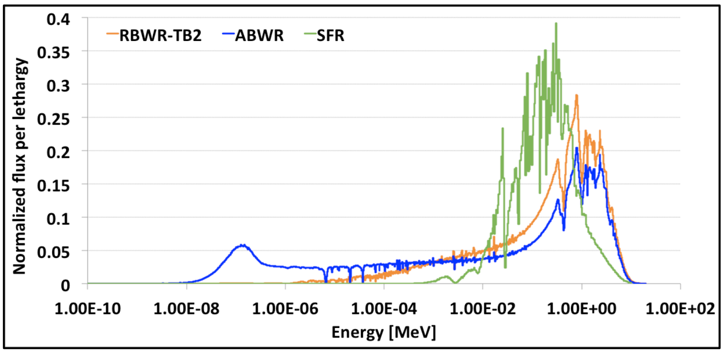

FIG. 1. Flux spectra in the RBWRs against an ABWR and an SFR.

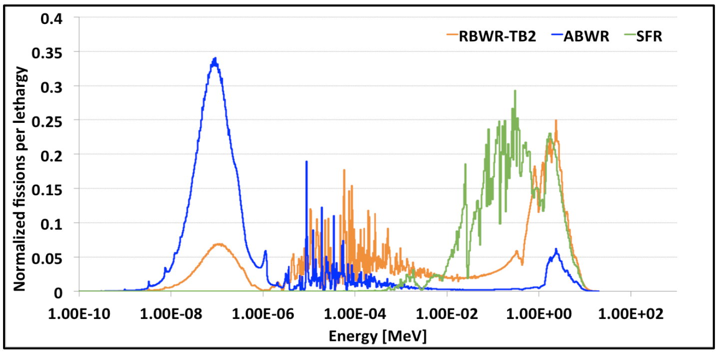

FIG.2. Spectra of neutrons inducing fission in the RBWRs and other reactors.

There were several concerns regarding the RBWR-TB2 core that provided incentive to examine a thorium-based counterpart: uncertainty in the void reactivity feedback, possibly of a margin against critical heat flux being too small, weak neutronic coupling between the two axial seed segments, and insufficient margin for fuel survivability and cladding resistance to high energy neutron flux [6]. These issues largely stemmed from the use of two seeds separated by a large internal blanket. The RBWR-TR design avoids most of these issues by replacing the depleted uranium fertile component with thorium, which permits a single elongated seed to be used [1].

The RBWR-TR has been previously shown to achieve similar transmutation rates and discharge burnups as the RBWR-TB2, while maintaining much higher margin against critical heat flux. It is also needed to maintain sufficient shutdown margin while having a negative void coefficient of reactivity (VCR) [1], and sufficient minimum critical power ratio (MCPR) [7]. In the RBWR-TR, the fuel cladding material is Zircaloy-2, the same as in the RBWR-TB2. Even though the heat flux is reduced with respect to the Hitachi design, Figure 3 shows that the acceptable fast neutron fluence limit is exceeded. This limit is 1.7×10^{26} neutrons/m^2 for fast neutrons of E > 0.1MeV.

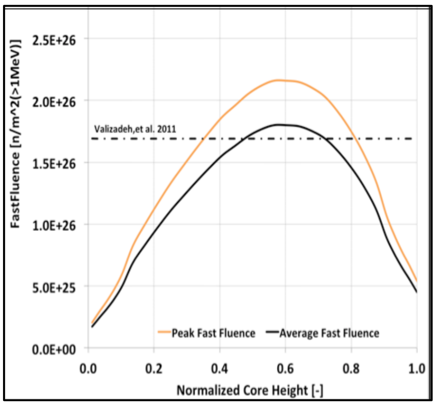

FIG.3. Fast fluences for the RBWR-TR design and the acceptable limits for Zircaloy-2.

This study focuses on determining the safety related parameters for the RBWR-TR core that will keep the fast neutron fluence within margins, and maximize the neutron economy. The possible materials that were examined as alternatives to Zircaloy are presented in Table I. These alternatives were determined from previous fast reactor studies.

TABLE I: POSSIBLE ALTERNATIVE CLADDING MATERIALS FOR THE RBWR-TR CORE.

Material	**Pros**	**Cons**
FeCrAl	• Resistance to high oxidizing media • High yield strength, excellent toughness • Easy manufacturability	• Large absorption cross sections
HT9	• Resistance to high oxidizing media • High yield strength, excellent toughness	• Low and high temperature irradiation embrittlement • Low fracture resistance after neutron irradiations

The steel alloys are favorable for low hydrogen pick up and resistance to highly oxidizing materials. The downside is either the large absorption, which can lead to neutronic penalties, or early stage of development and manufacturability. In this study, the steel-based cladding is chosen – FeCrAl, in particular. HT9 characteristics are presented for future studies as another steel based cladding alternative to Zircaloy.

2. Methodology

The analysis is done at two levels of the core design: the single assembly unit cell and the full core. The MocDown/MCNP methodology is used to determine the equilibrium cycle of the single assembly and the Serpent/PARCS/PATHS coupled core simulator/thermo-hydraulic system code methodology for RBWR-TR full core analysis.

2.1. Single assembly unit cell

MocDown [8] was used with MCNP5 [9] in order to define the single unit cell which is used in the search of an equilibrium fuel composition for each different cladding material that has the radiation damage constraint satisfied. MocDown is coupled with MCNP5, which provides flux and cross sections to the main code, and with ORIGEN2.2 [10], which performs transmutation and returns number densities to MCNP5. MocDown is also coupled with PATHS [11], a thermo-fluid calculator, which ensures a thermal-hydraulically self-consistent solution. MocDown uses an accelerated convergence scheme in which it runs several transmutation-only cycles between fully coupled cycles. The final equilibrium state is that state in which the batch-averaged reactivity between successive transport cycles is less than 100 pcm.

The unit cell modeled by MocDown is shown in Figure 4. The MocDown analysis is done for the initial design that uses Zircaloy-2 for the reference cladding of 0.06 cm and then for FeCrAl as a function of the cladding thickness [1]. Criticality could be reached either by reducing the cycle length or by reducing the cladding absorption. The reduction in the absorption is achieved by reducing the cladding thickness, which may require increasing the size of the gas gap.

2.2. Full core analysis

In order to perform full core simulation, the cross sections (XS) are first generated for the range of envisioned conditions (fuel temperatures, moderator density, burnup, and control rod position) using SerpentXS [13] to run depletion calculations for each state of the reactor. For each different state of the core, a three dimensional, single unit cell transport calculation with periodic boundary conditions is performed. The unit cell modeled by Serpent2 version 2.1.17 [12] for the XS generation is presented in Figure 4.

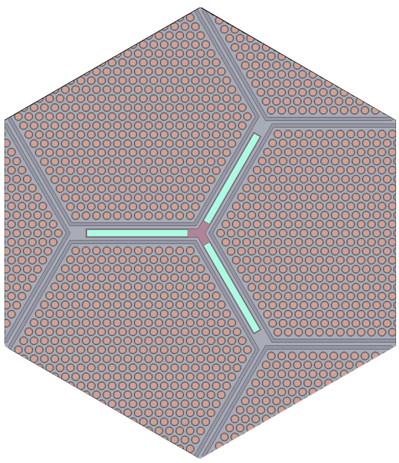

FIG.4. Cross-sectional view of the RBWR-TR unit cell, as modeled in MocDown and Serpent.

Once the cross-sections are processed and converted into PMAXS files by GenPMAXS [14], they are used to perform full core simulations with PARCS/PATHS [16]. The T-H feedbacks are included by coupling PARCS [15] with PATHS [11], which couples each assembly with a T-H channel.

The equilibrium cycle characteristics are calculated with PARCS/PATHS for the RBWR-TR design. This is an iterative process that consists of depleting the full core and then shuffling the fuel bundles until the maximum local burnup difference between fuel recycles is less than 0.1 GWd/t. The 720 fuel assemblies are arranged in a four-batch fuel loading. The shuffling and the control rod movements are adapted to the new core designs, and consist of 16 steps throughout the depletion. The burnup has to be large enough to reduce the cost of the fuel cycle and to satisfy the constraint that the reactor must have sufficiently long cycles – desirably at least one year.

2.3. Safety Parameters calculation

The safety parameters and constraints that were considered for the RBWR-TR design, besides the fast neutron fluence, are:

- Displacement per atom (DPA) < 200. The Zircaloy-2 use is limited to ~$2x10^{26}$ neutrons/m^2 (E > 0.1 MeV). For the steel-based cladding, the limit is 200 DPA, which represents the radiation damage constraint of presently verified cladding materials. The accumulated DPA value is calculated using the equation below:

$$\text{DPA}=\eta \frac{\sigma_d}{2E_d} \int dt\varphi$$

- The region-wise effective (spectrum-weighted) one group DPA cross-section -- σ_d, is generated by an FM4 tally of MCNP in units of barns-MeV; the efficiency η is assumed to be 80%; the displacement energy for Fe and Cr is suggested to be 40 eV [17] for steels.
- Negative VCR. The VCR is calculated by setting the coolant flow rate to 85% of the nominal value, and by dividing the change in reactivity and its uncertainty by the change in void fraction. Thus, the VCR is calculated using:

$$\text{VCR} = \frac{\frac{1}{k_{100\%\,\text{flow}}} - \frac{1}{k_{85\%\,\text{flow}}}}{\alpha_{85\%\,\text{flow}} - \alpha_{100\%\,\text{flow}}}$$

- MCPR larger than 1.5 [14], which is recommended for tight lattice fuel designs of the RBWR type cores instead of the traditional 1.3 value for BWR. The calculations of the MCPR are based on the modified-CISE correlations from MIT [7] and the correlation determines the critical quality for dryout for the present design:

$$\text{MCPR} = \text{Power}_{\text{exit}} / \text{Power}_{\text{crit}}$$

  To calculate the MCPR with PARCS, it is needed to determine the boiling length for the most critical cases, by re-running the simulations at the equilibrium cycle at 105% power. At 105% to include a more conservative case for power increase.

- Sufficient shutdown margin, in order to ensure that the reactor can be shut down at all times. The shutdown margin is quantified as the negative reactivity of the subcritical core (i.e. 1/k_(shutdown) -1), calculated at BOEC with fuel temperatures and water densities corresponding to room temperature and fully inserting all control rods except the one with the highest worth.

- Pressure drop through core reactivity kept ≤ 0.3 MPa.

At this point of the research, we do not calculate other safety parameters, such as pellet-cladding interaction or Doppler feedback. Those safety parameters will be part of future research.

3. Results

The first analysis using the new cladding material is done at single assembly level using MocDown. Figure 5 shows that the FeCrAl material satisfies the 200 DPA limit with a large margin.

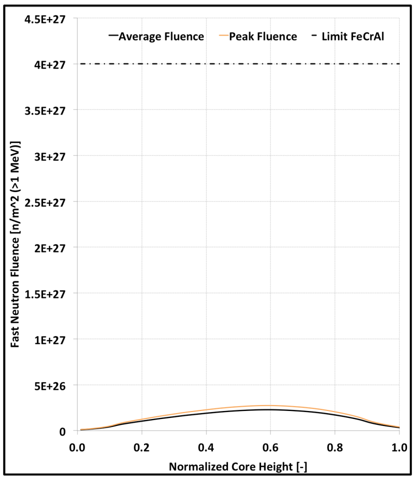

FIG.5. Fast fluences for the RBWR-TR design and the acceptable limits for FeCrAl.

Table II shows how using the same cladding thickness introduces neutronic penalties with respect to the Zircaloy cladding. The penalties are presented as reduced cycle length and discharge burnup.

TABLE II: DEPLETION CYCLE CHARACTERISTICS AND SAFETY PARAMETERS FOR THE REFERENCE ZIRCALOY RBWR-TR DESIGN AND THE FeCrAl DESIGN AT SINGLE ASSEMBLY.

	Units	**Zircaloy**	**FeCrAl**
Cladding thickness	cm	0.06	0.06
Seed length	cm	100	100
Core HM mass	t	86.3	86.3
Cycle length	EFPD	293	258
Discharged BU	GW·d/tHM	50	47
MAX /Accept radiation damage	-	1.38	0.285
Cladding tensile stress (%yield stress)	%	12	7
Cladding hoop stress	%	0.325	0.166

(%yield stress)			
VCR (EOC/BOC)	pcm/%void	2.48/ 2.75	1.21/ 3.31
%Pu239 / %U233	-	0.77	0.76
MCPR	-	1.56	1.59
Max LHGR	W_{th}/cm	229	231.7

The first to be investigated is reduction in the cladding thicknesses, which corresponds to the decrease in the quantity of absorbing material. The analyzed designs use cladding thicknesses of: 0.03, 0.04 and 0.05 cm. The reactor and safety parameters are calculated by performing full core analysis using the SerpentXS/PARCS/PATHS [16] methodology. In the full core simulation, the same shuffling scheme was used as in the Hitachi-designed RBWR-TB2. Half of the most burned batch was placed in a ring around the periphery of the core, followed by the fresh fuel. The once-burned fuel came next, followed by the other half- batch of the most burned fuel. The twice-burned fuel was placed at the center of the core. The results for the full core analysis are presented in Table III.

TABLE III: DEPLETION CYCLE CHARACTERISTICS AND SAFETY PARAMETERS FOR THE 3 DIFFERENT RBWR-TR DESIGNS.

	Units	**FeCrAl**		
Cladding thickness	cm	0.05	0.04	0.03
Core HM mass	t	86.3	86.3	86.3
Cycle length	EFPD	316.2	318	327
Core pressure drop	MPa	0.2	0.2	0.2
Fuel residence time	EFPF	1264.8	1274	1308
Discharged BU	GW·d/tHM	56.3	56	58.2
Maximum BU	GW·d/tHM	84.8	86	87.6
Cladding tensile stress (%yield stress)	%	13.9	9.7	7.8
Cladding hoop stress (%yield stress)	%	0.339	0.235	0.186
VCR (EOC/BOC)	pcm/%void	-17/-0.18	-15/-8.32	-5.13/-0.53
%Pu239/%U233	-	0.834	0.83	0.824
MCPR	-	1.55	1.58	1.55
Max LHGR	W_{th}/cm	204.64	207.9	212.712
Reactivity swing	%dk	2.8%	2.4%	2.1%
Power coefficient of reactivity	pcm/MW_{th}	-0.8/-0.46	-15.1/-1.1	-1.1/-0.48
Shutdown margin	pcm	-2300	-1234	-2216

Table III shows how the decrease in cladding thickness improves neutronics features. The 0.03 cm cladding thickness provides the longest cycle and largest discharge. However, the small cladding thickness leads to increase in mechanical stress (larger hoop and tensile stress) as shown in Table III. The cladding thickness of 0.05 cm matches the same neutronic parameters as in the reference Zircaloy case, as well as in terms of resisting mechanical stresses.

In addition to the analysis presented in Table III, for the full core model with FeCrAl cladding of 0.05 cm, the batch-averaged linear heat rate profiles at BOC and EOC are shown in Figure 6.1 and Figure 6.2, while Figure 7.1 and Figure 7.2 show the radial power map. The RBWR-TR design has an acceptable low peak linear heat generation rate (LHGR) and an even power distribution over the entire core, which will lead to reduced fission gas release rates and lower plenum pressure. The maximum value of LHGR is 212 W/cm.

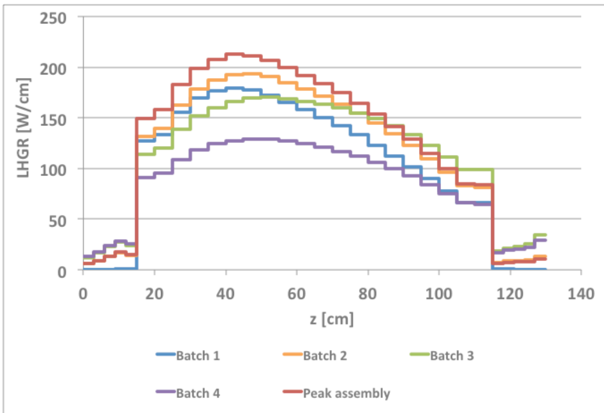

FIG.6.1 LHGR distribution for each of the four batches and the peak assembly value at BOC.

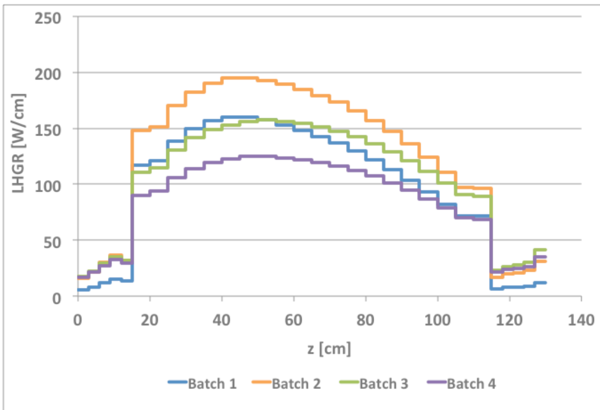

FIG.6.2 LHGR distribution for each of the four batches and the peak assembly value at EOC.

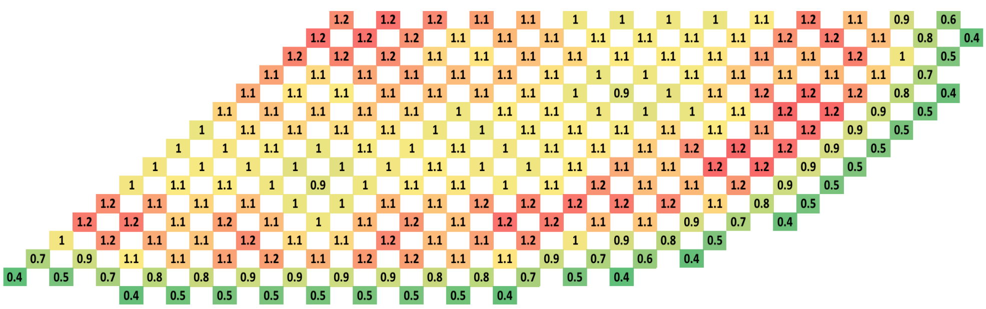

FIG.7.1 Power map for the RBWR-TR using FeCrAl cladding of 0.05 cm at BOC.

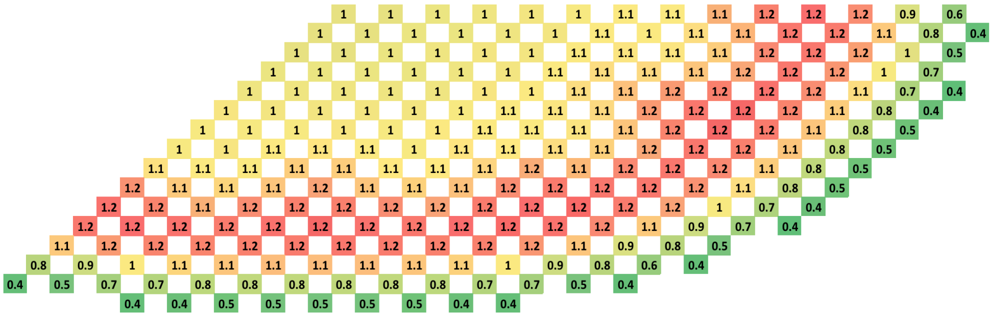

FIG.7.2 Power map for the RBWR-TR core using FeCrAl cladding of 0.05 cm at EOC.

The radial power peaking factor is 1.25 for this case, which is considered acceptable and allows a larger MCPR.

Additional parametric analysis is needed in order to understand possible improvements due to the changes in gap size or due to increase in the HM inventory, including possible implication of the cladding-pellet interactions.

3. Conclusions

In order to avoid Zircaloy-2 cladding failure in a RBWR-TR reactor due to the fast neutrons fluence, we analysed an alternative steel-based material, FeCrAl. The parametric studies that have been performed showed that FeCrAl satisfies the mechanical constraints, the 200 DPA for steel, but introduces large neutronic penalties: lower cycle length and discharge BU. Those limitations can be overcome by reducing the cladding thickness and to maintain the same HM amount at BOEL in the assembly. For the new cladding designs, full core analysis is performed to calculate the safety parameter, such as VCR, MCPR, and SDM. An optimal cladding thickness of 0.05 cm shows a cycle length of 329 days, and an MCPR of 1.5 maintaining a discharge burnup of ~50 MW·d/kgHM. Both, the large shut down margin and the negative VCRs at BOEC and EOEC, prove the feasibility of the proposed design from the safety point of view. However, a more detailed analysis of pellet-clad interaction is needed, due to the FeCrAl cladding thickness reduction as compared with the reference case [1] with Zircaloy-2 cladding. In addition, more parametric studies need to be performed at the neutronic level to study effects of increasing the HM amount and the gap size.

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