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ARTICLE

Safety and proliferation resistance of small-sized sodium-cooled fast reactors with passive shutdown devices

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This report proposes a long-life small-sized fast reactor with passive shutdown devices that insert negative reactivity during an anticipated transient without scram (ATWS). By adopting metal fuel for the core, the available core lifetime was found to be longer than that of the oxide fuel. The core achieved its core life without refueling for over 10 years. The power coefficient was evaluated for the core during operation periods and was found to be negative; therefore, the core had negative feedback during operation. In addition, the required reactivity insertion for devices under ATWS was evaluated. By replacing approximately 10% of the inner core assembly with device assemblies, it was found that the core could be shut down during ATWS. The differences in fuel management and transportation strategy between small-sized modular reactors and conventional large reactors were considered in terms of nuclear safety, security, and non-proliferation.

Keywords: small modular reactor; sodium-cooled fast reactor; transportable reactor; non-refueling operation; passive shutdown device

1. Introduction

Small-sized modular reactors (SMRs) are increasingly being considered a socially acceptable technology option owing to their high inherent safety, economic efficiency, and ease of introduction due to the adoption of innovative technology. In the present research, the transportable reactor system is assembled in the factory, transported to/from the operation site, and long-life onsite operation without refueling. It is one of the SMRs concepts possible to simplify on-site equipment, and maintenance, to prevent mis-operation in fuel treatment, and to minimize fuel accessibility to enhance nuclear safety, security, and safeguards/non-proliferation features (3Ss) [1].

Sodium-cooled fast reactors (SFR) are characterized by superior neutron economy, core compactness, high burnup with trans-uranium (TRU) fuel, and passive safety for decay heat removal. Metal fuels, which have a higher heavy metal density than conventional oxide fuels, are expected to achieve better neutron economy and a longer operation period. Fast reactor cores are not in their most reactive configuration under normal operating conditions, and therefore, they have possibility to result in positive reactivity changes when assuming coolant boiling, cladding discharge, and fuel concentration, as stated in [2]. Hence, multiple measures have been introduced in SFR design such as core catchers and inner duct structure in the driver fuel assembly FAIDUS [3]. Recently, the passive shutdown device (device) is being proposed and studied, which inserts negative reactivity during ATWS accidents and enhances the “versatility” and “robustness” of large SFRs [2].

Therefore, a feasibility study was performed for a long-life SFR core with metal fuel for the transportable reactor concept. The negative reactivity requirement for controlling ATWS accidents was also investigated during the development of the device. Finally, challenges concerning the nuclear safety and non-proliferation of the transportable reactor concept are discussed.

2. FR core design for long term operation

2.1. Basic core design and calculation method

Core specifications of the reference reactor design are shown in **Table 1**. The design of coolant and reactor diameter is based on the developed small and medium-sized SFRs, Prototype Fast Breeder Reactor Monju with mixed oxide (MOX) fuel in the core regions and depleted uranium (DU) in the blanket regions [4]. The calculation model was created from the core layout as a two-dimensional (RZ) model, as shown in **Figure 1**, in each fuel assembly component composed of 89 calculation zones, divided into 56×49 mesh intervals, and 15 material regions. In this study, the SLAROM-UF code [5] was used to prepare an effective cross-section library based on JENDL-4.0, and the CITATION code [6] was used for the reactivity, burnup, and power distribution calculations based on the RZ diffusion theory. The composition of transuranic (TRU) elements is derived from the 50 year-cooled

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Table 1. Specification of SFR in this study.

	Item	Oxide (Monju-like)	Metal
	Power	$500MW_{th}(196MW_{e})$	$500MW_{th}(196MW_{e})$
	Na Temperature(°C)(inlet/outlet)	397/529[5]	397/529
	Reactor diameter(cm)	330[5]	330
	Core height(cm)	93[5]	93,120
	Irradiation time(day)	750(1 batch)	5475(1 batch)
Fuel	TRU composition(wt%)	$^{238}Pu/^{239}Pu/^{240}Pu/^{241}Pu/^{242}Pu/^{241}Am$=1.6/56.4/24.0/1.0/6.8/10.1	
	$^{235}U/^{238}U$ (wt%)	0.2/99.8	
Core	Material form	(U,TRU)-Oxide	U-TRU-10w%Zr
	Smear density(%)	85[5] (Pellet TD ratio)	75 (Smeared)
	Volume ratio(fuel/structure/coolant)	35.5/24.7/39.8	51.3/19.9/28.8
	Diameter of fuel pellet/slug(cm)	0.54	0.48/0.96
Radial Blanket (RDBL)	Material form	UO_2	U-10wt%Zr/U-TRU-10wt%Zr
	Smear density(%)	93[5] (Pellet TD ratio)	85 (Smeared)
	Volume ratio(fuel/structure/coolant)	46.5/19.8/33.7	50.7/20.5/28.8
	Diameter of fuel pellet/slug(cm)	1.04	1.01/1.02
Axial Blanket (AXBL)	Material form	UO_2	U-10wt%Zr/U-TRU-10wt%Zr
	Smear density(%)	93[5] (Pellet TD ratio)	85 (Smeared)
	Volume ratio(fuel/structure/coolant)	35.5/24.7/39.8	51.3/19.9/28.8
	Diameter of fuel pellet/slug(cm)	0.54	0.51/1.03

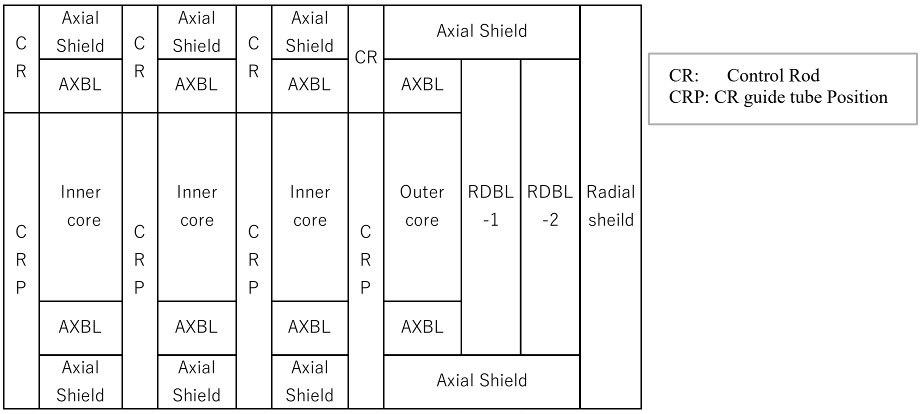


Figure 1. CITATION RZ calculation model in the Reference core.

separated Pu after reprocessing spent fuel irradiated in a PWR (44GWd/ tHM) [7]. Temperature evaluation was performed for the fuel, cladding, and coolant in the core based on the power density distribution results from CITATION, combined with the 1-dimensional radial heat conduction equation within the representative single channel.

2.2. Long life operation

Burnup reactivity calculations were performed to confirm the available reactor operation period. As a design target, the available operation period was set to more than 10 years without refueling, and the burnup reactivity change was within 7% of the main control rod worth. The results are shown in **Figure 2**, with the TRU enrichments for the five core designs listed in **Table 2**. First, based on the calculation using MOX fuel with a small diameter of 0.54 cm and core height 93 cm shown as "Oxide_169pins" of Case 1 in Table 2 and Figure 2, the available operation period was confirmed to be within 3 years at most. Because TRU enrichment may not be realized due to the limitation of reactivity, a higher heavy metal loading is needed to increase the operation period without dramatically changing the core power and reactor size. Therefore, the following cases were examined: Case 2 using metal fuel with a higher heavy metal density and thermal conductivity with core height 93 cm shown as "Metal_169pins", Case 3 increasing the diameter of the pin to the same extent as the radial blanket (**Figure 3**) with core height 93 cm shown as "Metal_61pins", Case 4 increasing the core height from 93 cm to 120 cm shown as "Metal_61pins_120cm", Case 5 changing RDBL and AXBL fuel composition from U-Zr metal fuel to U-Pu-Zr fuel with 5wt% reactor-grade Pu enriched and U-Am-Zr metal fuel with 2wt.% Am dope, shown as

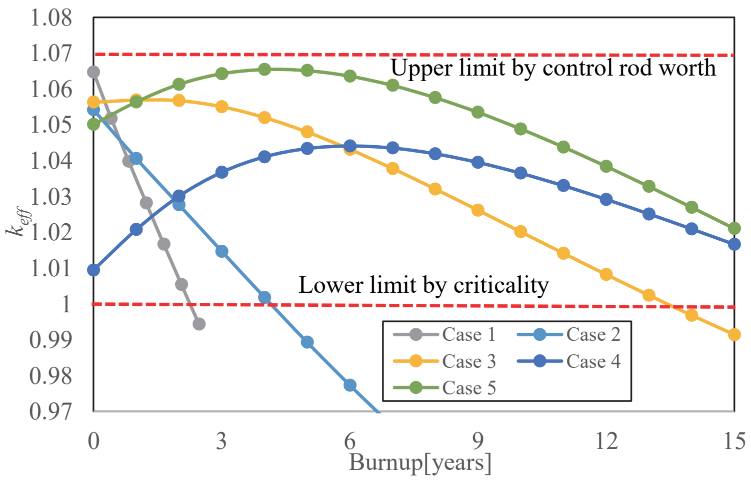


Figure 2. k_{eff} with Burnup for five designs.

Table 2. TRU enrichment for five core designs.

Fuel type_pins /subassembly	Case 1	Case 2	Case 3	Case 4	Case 5
	Oxide_169pins	Metal_169pins	Metal_61pins	Metal_61pins_120cm	Blanket_change
inner	27.3	19.5	14.8	12.5	13.6
outer	40.6	27.3	22.7	20.3	20.2
AXBL	0	0	0	0	2.0(Am)
RDBL	0	0	0	0	5.0

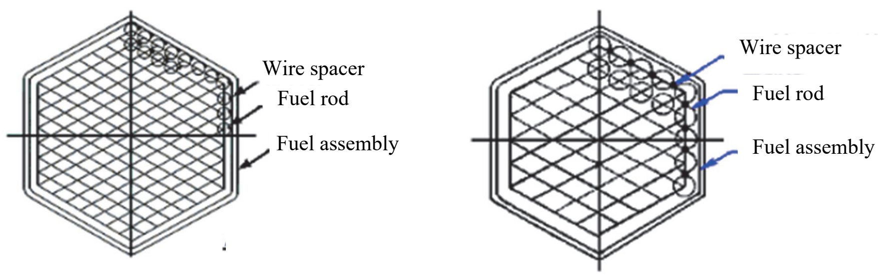


Figure 3. Fuel assembly cross section (Left: 169pins, Right: 61pins) [9]

"Blanket_change."

In both cases 3 and 4, the available reactor operation periods were evaluated as being greater than 10 or 15 years, respectively, with less burnup reactivity change owing to the higher Pu conversion. Owing to the long operation periods, the power density distribution changed with a large difference between the beginning of cycle (BOC) and end of cycle (EOC). This had a significant impact of the outlet coolant temperature in the radial blanket, and the difference between BOC and EOC was approximately 135 K. To address this problem, case 5 examined Pu enriched fuel in the RDBL to compensate for the power change during operation. The number of RDBL fuel assemblies was reduced from 3 layers to 1 layer by replacing with reflector. The results are; the outlet coolant temperature in the radial blanket, and the difference between BOC and EOC was only 60K. As for AXBL, though it was found to contribute to keeping the fuel assembly power stable during the operation, high purity of ^{239}Pu was generated, called weapon grade. Therefore, in case 5, 2% Am was also added [8] to the axial blanket to denature the generated Pu by even-mass number Pu isotopes (Figure 2, case 5). The calculation conditions for Case 5 were used in the section 2.3 and followings.

2.3. Isothermal coefficients and power coefficients

The core characteristics are required to calculate the isothermal coefficients from the Doppler constant and the axial and radial expansion coefficients. The Doppler constants were obtained from the fuel, structural material, and coolant. The elements of reactivity change owing to the thermal expansion of the core and blanket are (1) fuel pellet, (2) cladding, (3) duct, (4) coolant, and (5) core support plate expansion [9]. The calculation of the power coefficient is based on the isothermal coefficient from the thermal expansion and the Doppler constant. The equation for the power coefficient is expressed below [10];

Table 3. Isothermal coefficient (BOC, PEAK, EOC).

		BOC	PEAK	EOC
Expansion **$\Delta k/kk'/\Delta T$**	Fuel	-3.9E-06	-3.6E-06	-3.8E-06
	Cladding	2.2E-06	2.3E-06	2.8E-06
	Wrapper Tube	6.8E-07	7.0E-07	8.2E-07
	Coolant	6.5E-06	7.0E-06	8.4E-06
Doppler Const. **($T\Delta k/\Delta T$)**	Fuel	-3.5E-03	-2.8E-03	-2.3E-03
	Cladding	-3.4E-04	-3.2E-04	-3.4E-04
	Coolant	-1.1E-06	0.0E+00	1.1E-06
$\Delta k/kk'/\Delta T$	Support plate	-9.4E-06	-9.0E-06	-9.2E-06

Table 4. Power coefficient (BOC, PEAK, EOC).

	BOC	PEAK	EOC
$\Delta k/kk'/MW$	-1.4E-06	-1.0E-06	-4.6E-07

$$\Delta\rho_T = \alpha_{iso}\Delta T\ , \qquad \alpha_P = \frac{\Delta\rho_T}{\Delta P} \tag{1}$$

α_P: Power coefficient（$\Delta k/kk'$/MW),
α_{iso}: Temperature reactivity coefficient of each part of the core ($\Delta k/kk'$/°C)
ΔT_i: Temperature changes in various parts of the core due to changes in the reactor power (°C)
ΔP: Reactor output change (MW).

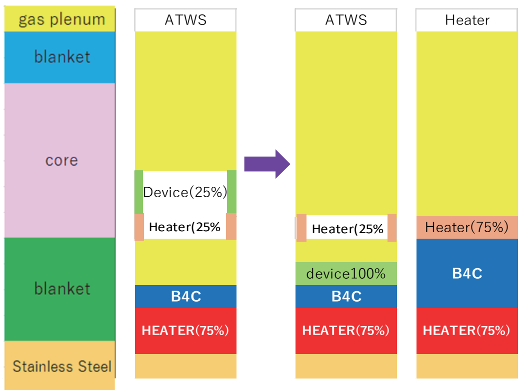
The isothermal coefficient results are listed in **Table 3**. The PEAK shown in Table 3 is at 4 years of burnup, where the maximum k_{eff} peak value was observed, as shown in Figure 2. The positive isothermal coefficients were evaluated in terms of structural expansion and coolant density. As the burnup increased up to 150GWd/t maximally at EOC, the absolute value of the coolant temperature coefficient increased because (1) the neutron absorption decreased because of the enormous amount of fission products accumulation (2) the fission reaction increased by fissionable nuclides of ^{240}Pu, ^{243}Am, and ^{244}Cm, and (3) the number of neutrons produced by a single fission reaction increased.

The power coefficients are presented in **Table 4**. Those values are used in reactivity balance estimations to ULOF and UTOP with conversion factor ($\beta_{eff} = 0.0038$). The longer the burnup, the smaller is the absolute value of the power coefficient. Finally, negative feedback effects were confirmed for the entire core, even after 15 years of operation.

3. Passive shutdown device

3.1. Device models and characteristics

The proposed devices, as shown in **Figure 4**, contain fuel material that is solid at core temperature conditions under normal reactor operation but liquefies with increase in core temperature during an ATWS accident [2]. During an accident, a large negative reactivity can be applied to the core by moving (relocating) the liquefied fuel to a region of low reactivity in the device pins by simple physics alone. In this study, reactivity insertion by ATWS accidents was estimated, and the requirements for the external negative reactivity feedback expected by these devices were evaluated.



(a) Driver pins (b) Device pins (c) Heater pins

Figure 4. Device model.

The device subassembly consists of two types of pins; 91 devise pins shown as “device”, and 36 heater pins shown as “heater” in Figure 4. The fuel smear density is shown as 25% and 75% in Figure 4. The device fuel selected in this study was a metallic alloy (15wt.%Pu-U-10at.%Fe). This alloy has the characteristics applicable to fast reactor fuel and a relatively low melting point to the U-Pu-Zr alloys. The solidus temperature is 703°C and liquidus temperature is 867°C. As fuel swelling is caused by fuel irradiation, The device structure was designed [11] with hollow fuel slugs, which facilitate the amount of liquefied fuel during operation. During device operation liquefied fuel flows downward along hollow wall of fuel slugs. Above the solidus temperature, the molten fuel becomes a mixture of solidus and liquidus fuel. Temperature 747°C at the fraction of liquidus of 60% is selected as threshold temperature which enables the liquefied fuel relocate downward by gravity because the viscosity of the mixture decreases sharply and significantly above this temperature. Fuels used in heater positions were ternary metal fuel alloys in forms of 45wt.%Pu-U-10wt%Zr for lower heater

Table 5. Average temperature in UTOP situation.

δT=T(UTOP, *P*/*F*=2.3) -T(normal)(K)	BOC		PEAK		EOC	
	Inner core	Outer core	Inner core	Outer core	Inner core	Outer core
Coolant outlet	160	223	174	194	178	172
Fuel Average	170	209	192	194	195	178

Table 6. Average temperature in ULOF situation.

δT=T(ULOF, *P*/*F*=2.3) -T(normal)(K)	BOC		PEAK		EOC	
	Inner core	Outer core	Inner core	Outer core	Inner core	Outer core
Coolant outlet	160	223	174	194	178	172
Fuel Average	75	103	81	91	87	83

Table 7. Device requirements (Unit cents).

Reactivity	ULOF	UTOP	Zero to hot power	CR	Requirements
BOC	---	13.7	-18.1	-35	-39.4
	-3.9	---	-18.1	---	-22.0
PEAK	---	7.9	-13.1	-35	-40.2
	-8.4	---	-13.1	---	-21.5
EOC	---	0.58	-6.0	-35	-40.4
	-13.5	---	-6.0	---	-19.5

pins with 75% smeared density slug and 22.5wt.%Pu-U-10wt%Zr for upper heater pins with 25% smear density hollow slug, respectively as shown in Figure 4.

For the initial step, three devices were arranged separately in the rows 2,3,4 and 5. Each device subassembly was placed in a patten that avoids adjoining each other. Applying the two-dimensional RZ model (Figure 1), however, power distributions and device fuel relocation worth were calculated. For estimating average device worth in the row-2, the effective multiplication factor ($Keff_0$) was calculated for the model in which six fuel subassemblies were replaced by the device subassemblies for the normal state corresponding to "before operation" as shown the left of (b) in Figure 4. In order to obtain the $Keff_1$ for the relocated state corresponding to "after operation", six relocated device fuel model was placed in the model of the right of (b) in Figure 4. The average device worth per device was given by the equation $(Keff_0\text{-}Keff_1)/(Keff_0 * Keff_1)/6$. Other device worth calculations were made by the same way for the row-3, row-4 and row-5, respectively.

After adjusting material arranges in device fuel pins and heating pins, reactivity worth of fuel device with 40cm length below the core mid-plane was obtained per device; -3.2 cents in row-2, -3.1 cents in row-3, -3.5 cents in row-4, and -3.5 cents in row-5, respectively. Simple summation gives 40 cents for 12 devices in BOC. For the EOC, the device reactivity worth could be stronger, because the inner core region had larger flux levels compared to that in outer core region, reflecting inner core had higher conversion ratio than outer core.

Device fuel temperatures were estimated at BOC and EOC, in considering engineering uncertainties combination of hot-spot-factor (HSF) and over power condition during normal operation of the core as suggested in [11], peak device fuel temperatures were 660°C for BOC and 682°C for EOC, respectively, below the solidus temperature 703°C with adjusted coolant flow rate of the device subassembly. The result confirmed the requirement to the device under rated power operations.

3.2. Estimation of reactivity insertion by ATWS to identify device requirements

P(Power)/*F*(Flow) was used to statically evaluate the requirements of the devices during ATWS accidents by normalizing $P/F \equiv 1$ during the rated operation. The temperatures under ATWS were calculated so that the fuel temperature did not exceed 1100°C, the melting point of metal fuel, and the coolant outlet temperature did not exceed 880°C, the boiling point of sodium. The *P*/*F* ratio during device operation was determined using temperature calculations. The results of the temperature changes during ATWS accidents are shown in **Tables 5** and **6**, along with the results of the evaluation for *P*/*F*= 2.3. Based on these results, the required reactivity of the device during ATWS accidents was evaluated through quasi-static reactivity balance and the results are shown in **Table 7**. External reactivity demand to the devices were identified by estimating the reactivity changes under ATWS conditions. In this study, the reactivity demand against ULOF comprised reactivity changes due to the *P*/*F* status from 2.3 to1.0 and due to hot power *P*/*F*=1 to zero power (hot stand-by) level. Against UTOP, additional control rod withdrawal was included. The magnitude of 35 cents was excerpted from the previous SMR metallic core design [12]. Reactivity demands to achieve reactor passively shutdown under ULOF and UTOP events are summarized

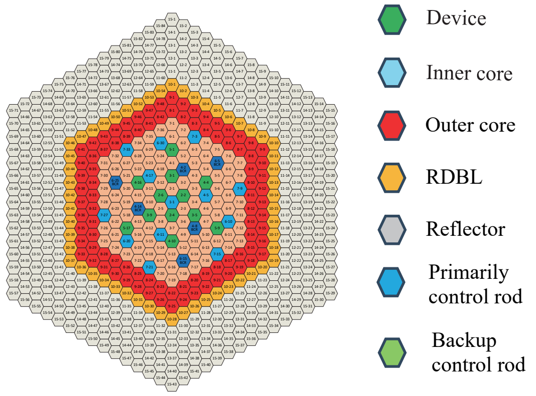


Figure 5. Core model with device installed.

in Table 7. To overcome ULOF, about 22 cents was needed in this core. Against UTOP, about 41 cents was externally required. If the devices operate synchronously, seven devices could lead the reactor into stable cooling situation in ULOF and twelve devices in UTOP situation as well. Considering uncertainty of asynchronous operation of the devices, though, more numbers of device would be required, certain change could be acceptable in the current core design as indicated in the reference [11], and this would be the important future works. As typical core layout is depicted in **Figure 5** with 12 devices loaded in the inner core.

Through the discussions, the device can be effective as passive countermeasure against ULOF and UTOP in sodium-cooled-metal-fueled SMRs.

4. Nuclear safety and non-proliferation challenges in the transportation of nuclear reactors

In this chapter, the challenges of nuclear safety and non-proliferation in the transportation of transportable reactors are discussed. In **Figure 6**, the transportation of the transportable reactors in the present research is shown, in which the reactor system is assembled in the factory, transported to and from the operation site, and then operated for a long time without refuelling. The first challenge is the transport-container issue. For simple and compact transportation of transportable reactors, the containment vessel of the reactor must satisfy the role and requirements of transportation containers of nuclear fuel, including severe performance tests against various external events and radiation biological shielding. This would cause a fundamental design change in the containment vessel that is currently required. Otherwise, an additional transport container would be required to contain the entire reactor, which would be massive and enormous, and the transportability, essential to the "transportable reactors," might be lost. The second challenge is criticality safety. Metal fuels with high TRU content greatly increase the reactivity against submersion, and the heater pins in the device have high enrichment. Water tightness for transport containers is required to prevent criticality accidents as well as sodium water reactions. Other options need to be discussed in future work, such as independent transportation of the coolant material sodium and pouring it onsite and neutron poison installation inside the reactor vessel during transportation.

5. Conclusion

A long-life small-sized SFR with passive shutdown device that insert negative reactivity during the ATWS of ULOF and UTOP conditions was proposed. To evaluate the burnup reactivity and power coefficients, a core lifetime of over 10 years without fuel reloading was achieved by using a metal fuel-loaded core with negative power coefficients during operation. By replacing approximately 10% of the inner core assemblies with the device, core shutdown can be expected during ATWS. Finally, issues regarding the nuclear safety and non-proliferation of transportable reactor concepts were discussed, and the role of containment vessels in nuclear fuel transportation and criticality safety during transportation were addressed.

Acknowledgements

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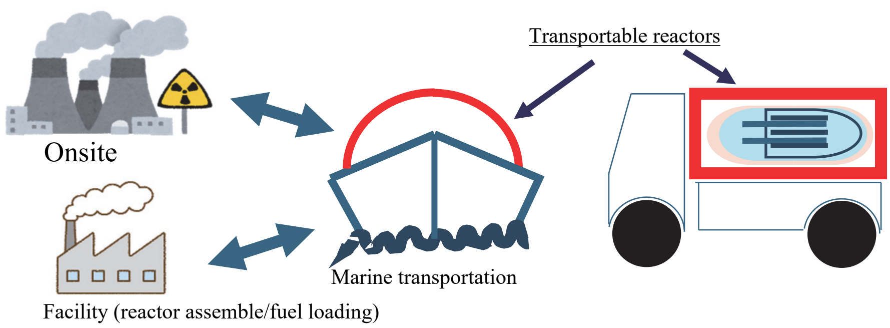

Figure 6. Transportation model for transportable reactors.

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