Journal of NUCLEAR SCIENCE and TECHNOLOGY, Vol. 39, No. 11, p. 1169–1181 (November 2002)

Conceptual Design of a Modular Island Core Fast Breeder Reactor "RAPID-M"

Mitsuru KAMBE*

Central Research Institute of Electric Power Industry (CRIEPI), 2-11-1 Iwadokita, Komae-shi, Tokyo 201-8511

(Received April 30, 2002 and accepted September 10, 2002)

A metal fueled modular island core sodium cooled fast breeder reactor concept RAPID-M to improve reactor performance and proliferation resistance and to accommodate various power requirements has been demonstrated. The essential feature of the RAPID-M concept is that the reactor core consists of integrated fuel assemblies (IFAs) instead of conventional fuel subassemblies. The RAPID concept enables quick and simplified refueling by replacing IFAs in which all the core and blanket fuel elements are comprised. In this paper, the 600 MWe RAPID-M design consists of 7 IFAs is presented. Significant reactor mass savings and the improvement of inherent safety features are discussed. Plant dynamics analyses using the multi-point reactor kinetics equations to accommodate the modular core configuration demonstrated a favorable transient response in case of unprotected transient over power (UTOP).

KEYWORDS: FBR type reactors, performance, sodium cooled reactor, modular island core, inherent safety, integrated fuel assembly, plant dynamics, reactor kinetics, sodium void worth, RAPID refueling, lithium expansion module

I. Introduction

In conventional fast reactors, the fuel handling mechanism is essential for handling fuel subassemblies of the reactor core. The existence of fuel handling mechanisms, together with the rotating plug(s) of the closure head, considerably affects the reactor structure design which is quite complicated compared to the light water reactors. Because of these circumstances, the author proposed a 150 MWt–60 MWe metal fueled fast breeder reactor (FBR) concept RAPID[1)] (Refueling by All Pins Integrated Design) to eliminate conventional fuel handling systems and to provide significant advantages in safety, plant performance and economic aspects. The essential feature of the RAPID concept is that the reactor core consist of an integrated fuel assembly (IFA) instead of conventional fuel subassemblies. The RAPID concept enables quick and simplified refueling 2 months after reactor shutdown by replacing the IFA in which all the core and blanket fuel pins (fuel elements) are comprised. Refueling requires only 10 days. The reactor can be operated without refueling for up to five years. The RAPID refueling concept possesses high resistance to state-supported removal of plutonium for nuclear weapons production. The power level of RAPID is, however, restricted to up to 60 MWe because of criticality safety, decay heat removal and transportation of the IFA. In order to overcome this difficulty and to meet greater power requirements, a modular island core FBR concept RAPID-M by using several IFAs has been proposed.[2)]

The modular island core concept is defined as placing several cores (modular islands) in the same reactor vessel; each module being separated from the others by one or several non fissile subassembly rows (isolating zones). Several attempts have been made to make use of this concept to achieve significant sodium void worth reduction, while maintaining the other major core parameters. Commisariat à l'Energie Atomique (CEA) attempted this type of design,[3)] but modular cores have a very sensitive power distribution to small local perturbations when the level of neutronic coupling between the modulus is low. To solve this problem, the optimized modular cores obtained were fully decoupled using a specific isolating zone. In comparison with the reference large cores, their sodium void worth was reduced to almost zero from about 5 dollars, but their burnup swing was increased, the cycle length was reduced by a factor 2, and the core radius was increased about 35%.

The neutronic coupling of the fast reactor core is generally strong in comparison with the light water reactor core, therefore fewer control rods are sufficient to control the fast reactor core. However the increase in the volume of either the single core or a modular island core and, correspondingly, in its radius, has shown that the radial flux distribution is no longer characterized by the fundamental mode alone, and higher mode components contribute largely to this distribution. This effect is shown, for example, by the interaction of the control rod assemblies. In case of Super Phenix core (1,200 MWe), greater deformation of the flux distribution due to control rods movement is presented.[4)] Such higher sensitivity to local perturbations caused reactivity variation of more than factor 10 of the reactivity worth of one control rod. While in case of Phenix core (250 MWe), this value was only a 25%.

In this paper, a 1,500 MWt–600 MWe design by 7 IFAs is presented to demonstrate the advantages of the combination of the RAPID refueling concept and modular island core design. Considerable reactor block mass saving, core performance, safety features and moderate sensitivity to local perturbations are discussed. Various power ranges from approximately 200 to 1,200 MWe could be achieved by adjusting the number of IFAs.

Potential uses for the RAPID-M are in power plants in urban areas of industrialized nations to relieve the peak load, and for developing countries where remote regions cannot be conveniently connected to the main grid and where it is economical to provide local generation capacity. Even if the

*Corresponding author, Tel. +81-3-3480-2111, Fax. +81-3-3480-2493, E-mail: kambe@criepi.denken.or.jp

power level required to the plants in different countries varies, a common IFA can be adopted, accordingly all these IFAs could be processed in the same reprocessing plant. Therefore, significant improvement in proliferation resistance could be assured by the RAPID concept combined with an international network of the IFA fuel cycle, so long as only reprocessing plant(s) constructed in nations which undertake nonproliferation measures and accept international safeguards are included.[1]

II. Reactor Concept

The reactor structure of the RAPID-M can be realized by both the loop and pool type configurations. **Figure 1** shows an example of a loop type configuration with sodium inlet and outlet temperature of 380 and 530°C, respectively. The reactor is basically a sodium cooled primary-intermediate-steam configuration, however, an elimination of the intermediate system could be adopted. The difference from the conventional reactor is the 7 IFAs placed on the core support platform.

1. Integrated Fuel Assembly

A schematic drawing of the integrated fuel assembly (IFA) is shown in **Fig. 2**. Each IFA consists of approximately 10,000 fuel pins (fuel elements) with a pitch to diameter ratio of 1.2. Similarly to the conventional fuel subassemblies, all these elements are combined by a core support grid and several spacer grids, and are assembled into a fuel cartridge of 1.7 m in diameter and 4 m long, instead of the conventional hexagonal duct. Refueling can be carried out by removing the fuel cartridge. For fuel handling operations, the fuel cartridge is provided

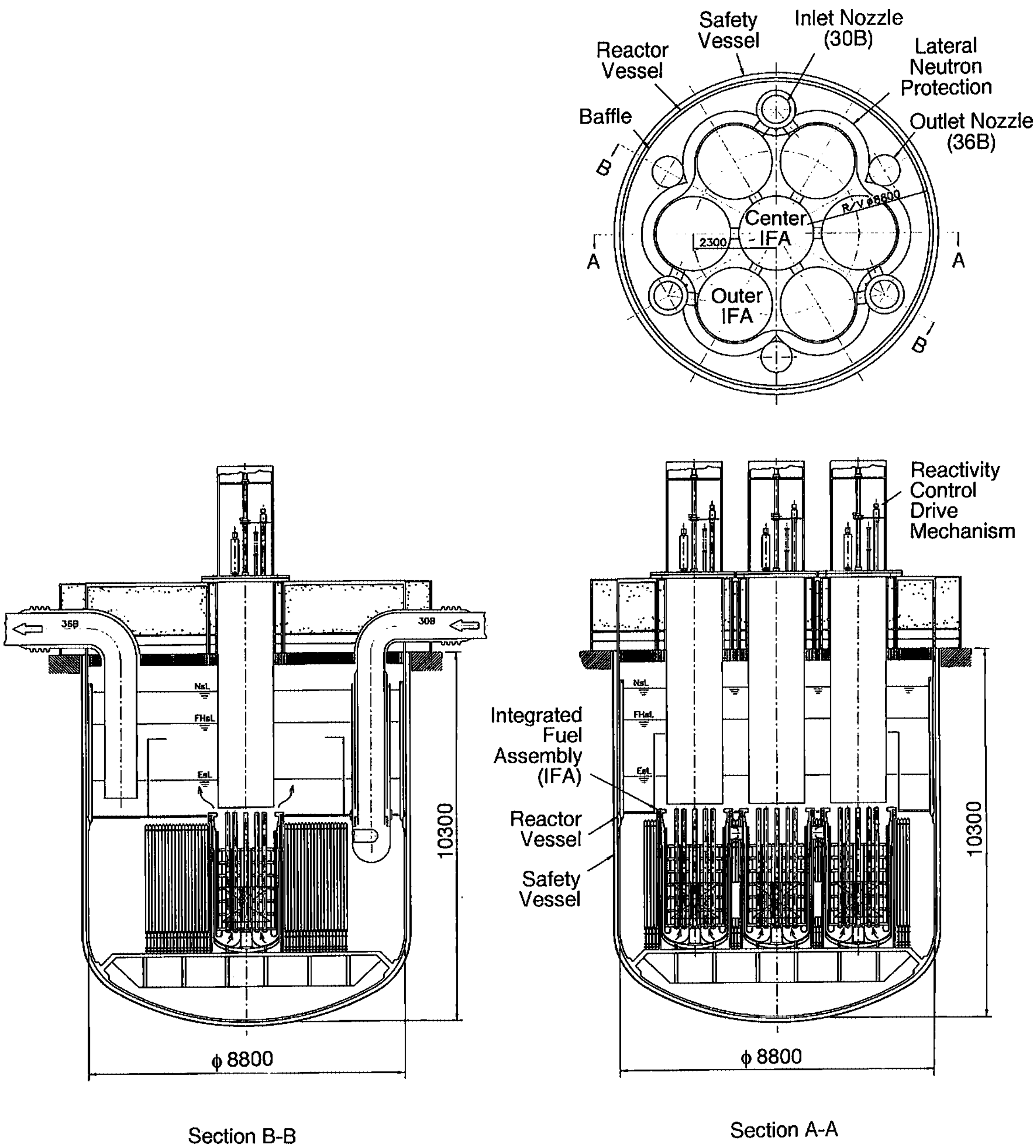


Fig. 1 Reactor structure of RAPID-M

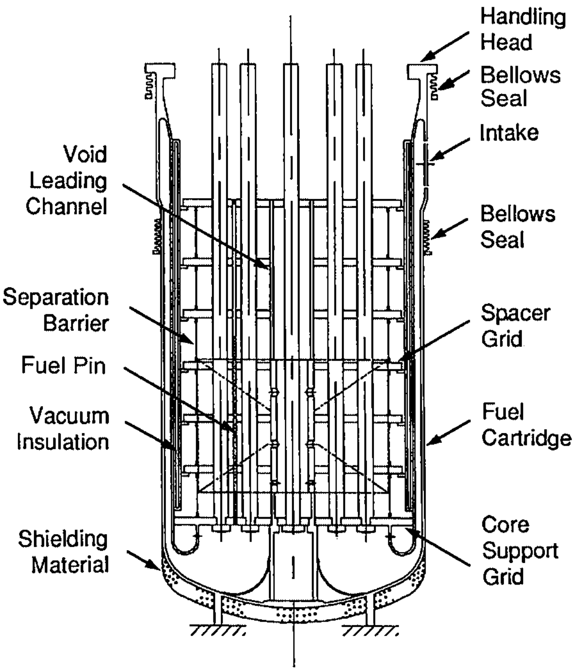


Fig. 2 The integrated fuel assembly

with a handling head on its top. No conventional hexagonal duct is adopted. The flowrate of the active core and blanket pin bundles can be adjusted independently by the separation barrier. Primary sodium is pumped into the annular space between fuel pin bundle and the fuel cartridge. The core support grid, 1.5 m in diameter and 60 mm thick, supported by its center hole and by the periphery, was designed by finite element method analysis. The most attractive features of the IFA is simple bowing behavior and resistance to seismic vibration because no cold clearance between subassemblies exists. In addition no risks of bundle to duct interaction (BDI) and duct to duct interaction (DDI) are expected. Therefore factory fabrication of the IFA associated with high standards of quality assurance excludes a common mode failure of fuel pins.

Based on the operational experience with experimental fast reactor JOYO, no fuel pin failure occurred yet in spite of the irradiation of more than 50,000 fuel pins.[5)] One of the advantages of the metal fuel is its enhanced integrity of the fuel cladding because of the existence of bond sodium between the fuel pellet and fuel cladding. In this design, a sufficient gap between the fuel pellet and fuel cladding is available without loosing thermal conductance between them. A sufficient gap would assure the cladding integrity *vs.* fuel pellet swelling. Therefore, fuel pin failure is most likely expected to be less than 1. In case of a fuel pin failure situation, the reactor operation can be continued because of good compatibility of metallic fuel with sodium coolant. Fuel failure detection (FFD) can be done using conventional devices. Fuel failure detection and localization (FFDL) of the RAPID-M is quite easier than the conventional fast reactors using ordinary subassemblies, because we must only to distinguish a failed IFA among 7 IFAs instead of localizing a failed subassembly among several hundreds of them. This design also fits the reprocessing requirement because total amount of disposed structures in case of the IFA is not so much as that of hexagonal ducts.

These concepts were selected because they best meet the safety and plant performance requirements. (As shown in the introduction, our requirements are to eliminate conventional fuel handling systems and to provide significant advantages in safety, plant performance and economic aspects.) Key design features included in the IFA are summarized as follows:

(1) Improvement of the breeding and burnup performance and reduced reactivity change by an augmented fuel volume fraction.
(2) Reduction of the whole core sodium void reactivity worth.
(3) Enhanced negative radial thermal expansion reactivity coefficient of the grids (*i.e.*, 2.5 times higher absolute value than conventional core) because no cold clearance between subassemblies exists.
(4) Reduced risk of flow blockage. The IFA has spacer grids similar to those of conventional fuel subassemblies. However, the IFA has only a separation barrier instead of many hexagonal ducts. Therefore the IFA has a definite advantage to avoid the flow blockage of a whole subassembly.
(5) Low pressure drop of core bundle because no conventional hexagonal duct exists.
(6) Elimination of conventional fuel handling system enables quick and simplified refueling and substantial mass savings of reactor structure.
(7) Easy breakdown for reprocessing. It is obvious that only one fuel cartridge of the IFA would make the breakdown of fuel pins/spacer grids easier than they would be in approximately 100 separate hexagonal ducts.
(8) Definite proliferation resistance (refer to 3.5).

A 1,500 MWt–600 MWe RAPID-M design by 7 IFAs can achieve all above advantages. In addition the RAPID-M design has other advantages:

- Low risk of void propagation into the adjacent IFAs in case one of the IFAs is voided due to UTOP or local blockage of one of the IFAs.
- Seven IFAs can provide power 10 times as much as that of RAPID which consists of a single IFA, because of the improvement in nuclear economics due to the nuclear coupling adjustment.

2. Coolant Distribution

In this particular reactor concept, the core support platform has neither inlet nozzles nor diagrid structure. The inlet nozzles are connected to the 6 outer IFAs by connecting tubes, as shown in **Fig. 3**. Similarly, the outer IFAs are connected to the center IFA. The flow resistance of the connecting tubes are adjusted to achieve the appropriate coolant distribution according to the power distribution of the center and outer IFAs.

In the modular island core configuration, an asymmetric neutron flux distribution is expected in the IFAs placed around a center IFA. Accordingly the core outlet temperature of such cores should also be asymmetric. However the fuel cartridges of the outer cores of the RAPID-M will have no asymmetric temperature distribution, because they are always cooled by cold sodium pumped into the annular space between the fuel pin bundle and the fuel cartridge. Therefore, the asymmetric

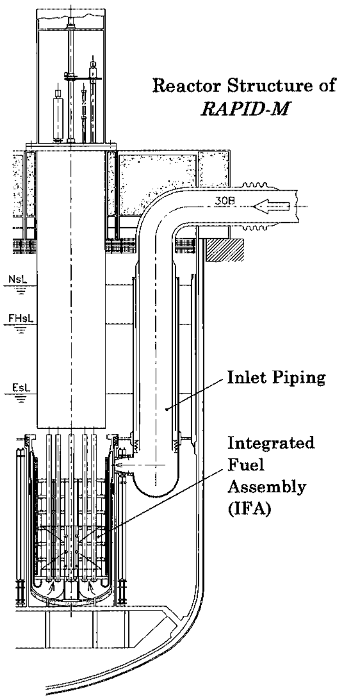


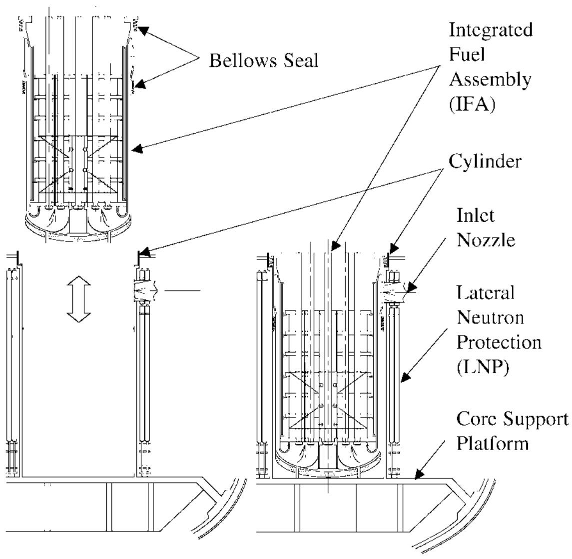
Fig. 3 Detail of the inlet piping/IFA connection

flux distribution mentioned above will cause no problems to the structural integrity of the IFA.

3. Reactor Closure Head

Neither a conventional fuel handling mechanism nor rotating plugs is required in the reactor closure head of the RAPID-M. The reactor closure head has only 7 holes, in which the upper core structures (UCSs) are installed. The 7 IFAs are installed in the cylinder, which is fixed on the core support platform, as shown in **Fig. 4**. Refueling can be carried out by removing the UCS, and the IFA can then be removed through the holes of the reactor closure head.

An attempt has been made to achieve a strong neutron coupling by placing the IFAs as close together as possible. In view of this, the distance between the IFAs is determined by the restriction of the closure head structure. The distance between the centerlines of the adjacent cores was determined to be 2,300 mm. The hole diameter of each closure head is 2,050 mm, therefore the minimum distance between the holes is 250 mm, with a closure head height of 2,500 mm, which is a sufficient space to place the radial support beam to withstand the weight of the closure head itself, as well as the 7 UCSs, even in case of seismic conditions. The weight of each UCS is only 5 t.



(a) IFA removed (b) IFA installed

Fig. 4 Detail of the inlet nozzle, LNP and IFA

4. Reactor Vessel

The reactor structure of the RAPID-M can be realized by both the loop and pool type configurations. Figure 1 shows a loop type configuration and a reactor vessel of 8.8 m diameter and 10.3 m depth. **Table 1** demonstrates the comparison of the reactor vessel size with respect to the conventional reactor concept. A substantial reactor block mass savings of a 60% reduction over comparable sodium cooled FBR systems can be estimated in case of a loop type configuration. Not only its compactness, but also simple reactor closure head (refer to 2.3) and core support platform (refer to 2.5) will have definite advantage of construction cost reduction per installed kW.

The reactor vessel diameter is reduced because of the elimination of the conventional fuel handling mechanism and rotating plugs. Reduction of the reactor vessel height is achieved due to the coolant level requirements in the reactor vessel. During refueling, the decay heat removal of the spent fuel is assured independent of the coolant level in the reactor vessel, because the coolant is always immersed in the fuel cartridge. Therefore, the normal system level (*i.e.* the reactor vessel coolant level during normal operation) can be determined by the expected leakage volume of the coolant from the reactor vessel or the primary circuits. In case of pool type configuration, a compact reactor vessel is also possible, however, the primary pump and the intermediate heat exchanger length dominate the reactor vessel height.

Reduction of the reactor vessel diameter is effective to minimize the reactor vessel stress at the vicinity of the coolant free surface where an abrupt axial temperature distribution causes intensive hoop stress. On the other hand, reduction of the reactor vessel height will enhance an eigenvalue of the reactor structure, therefore the reactor vessel thickness can be reduced. This is again effective to reduce the reactor vessel stress at the vicinity of the coolant free surface. Therefore compact reactor structure of the RAPID-M will have an advantage in view of seismic and high temperature structural design, and will fit to various power requirements up to 1,200 MWe.

5. Reactor Internals

The reactor internals consist of the core support platform on which the cylinder and lateral neutron protection (LNP) are arranged, as shown in Fig. 4. Each IFA is installed in a cylinder that is fixed to the core support platform and surrounded by the LNP. The cylinder has a role to hold the IFA

Table 1 Comparison of the reactor vessel size

600 MWe FBR	Loop type		Pool type	
	Conventional design	RAPID-M	Conventional design	RAPID-M
Diameter (m)	10.4	8.8	13.0	10.0
Height (m)	16.0	10.3	16.0	11.0

in a proper location. It has a sufficient strength to withstand the horizontal seismic load of the IFA. Another role of the cylinder is to establish the flow path of the primary coolant. When the IFA is installed, the cylinder, together with two bellows seals of the IFA, can assure the high-pressure plenum boundary automatically without any intervention on the reactor internals. Therefore the primary coolant comes from the inlet nozzle can be supplied to the IFA. While a few percentage of the primary coolant would leak from the small holes provided on the cylinder. This leakage is necessary to cool the reactor vessel wall. The leaked sodium would enter into a gap between the reactor vessel and thermal liner from its bottom and flows up to its top. The LNP consists of stainless steel rods, however, the LNP rods between the IFAs at the active core level were eliminated to achieve a strong neutron coupling.

The core support platform has neither inlet nozzles nor a diagrid structure. One of the advantages of this concept is enhanced integrity of the core support structure. From a general design standpoint, the integrity of the core support structure is of utmost importance. Thus the most important in-service inspection (ISI) requirement is validation of the core support structure which is entirely immersed in sodium and therefore presents a difficult, and currently unresolved problem. The RAPID concept makes it easier by adopting a combination of replaceable core support grid and permanent core support platform. Advantage has been taken of simple core support platform instead of conventional core support structure; the plain surface of the simple core support platform would make ISI easier than it would be with more complex inlet nozzles and a diagrid structure.[1)]

III. RAPID Refueling Concept

1. Refueling Concept

Refueling is conducted every 5 years. The fuel handling is achieved by the RAPID refueling concept[1)] combined with a movable inerted cell (MIC) and an on-site storage cask (OSSC) as illustrated in **Fig. 5**. The maximum availability of the operating floor should be emphasized. This is an open vessel hot cell refueling concept. Just prior to the refueling, the MIC is placed in its stand-by position, and the upper internal structure is removed by the crane. In case of refueling, the MIC is placed above the reactor and a sodium filled OSSC is set beneath the overhang of the MIC. Then the sodium filled IFA in which all the absorber rods are comprised is removed by an in-cell crane from the reactor and transferred into the OSSC *via* the MIC. After receiving the IFA, the OSSC is equipped with a closure lid on which a heat pipe radiator is attached for decay heat removal. All fuel handling operations are accomplished by conventional remote handling techniques. Refueling can be carried out 2 months after reactor shutdown by which time decay heat of the core is 180 kW. A sufficient period for refueling operation is 2 days. Plant availability can be significantly improved by this refueling concept.

2. Decay Heat Removal during Refueling

In this particular refueling concept, decay heat removal of the IFA in case of loss of power to the in-cell crane must be confirmed. The following two cases are anticipated.

(1) The IFA locked up in the movable inerted cell.
(2) The IFA locked up in the reactor closure head.

The most severe case is (1) because the decay heat should be removed by radiation and natural convection, while case (2) is not as severe. As well as radiation, natural convection is expected in the annulus between the hole of the closure head and the outer surface of the IFA. Due to the temperature difference between the cover gas space over the reactor sodium surface (200°C) and the hot cell space (less than 30°C), natural convection develops between the annulus of 100 mm width. This is effective for decay heat removal from the outer surface of the IFA.

Decay heat removal capability in case (1) is estimated as follows:

$$Q = Q_{Rad} + Q_{NC} \tag{1}$$

$$Q_{Rad} = \varepsilon\sigma T^4 A \tag{2}$$

$$Q_{NC} = Nu\lambda\Delta TA/L \tag{3}$$

$$Nu = 0.0214(Gr \cdot Pr)^{2/5} \tag{4}$$

$$Gr = g\beta\Delta TL^3/\nu^2, \tag{5}$$

where

Q: Decay heat
Q_{Rad}: Removed heat by radiation
ε: 0.2 (emissivity of the IFA, which is made of austenite steel without oxidation; sodium is removed by evaporation)
$\sigma =$ 5.67×10^{-8} (W/m^2·K^4), the Stefan-Boltzmann constant
T: Absolute temperature of the IFA surface
A: Surface area of the IFA
Q_{NC}: Heat removed by natural convection
Nu: Nusselt number
$\lambda =$ 0.024 (W/m·K), thermal conductivity of the Ar gas
ΔT: Temperature difference between the IFA and environ-

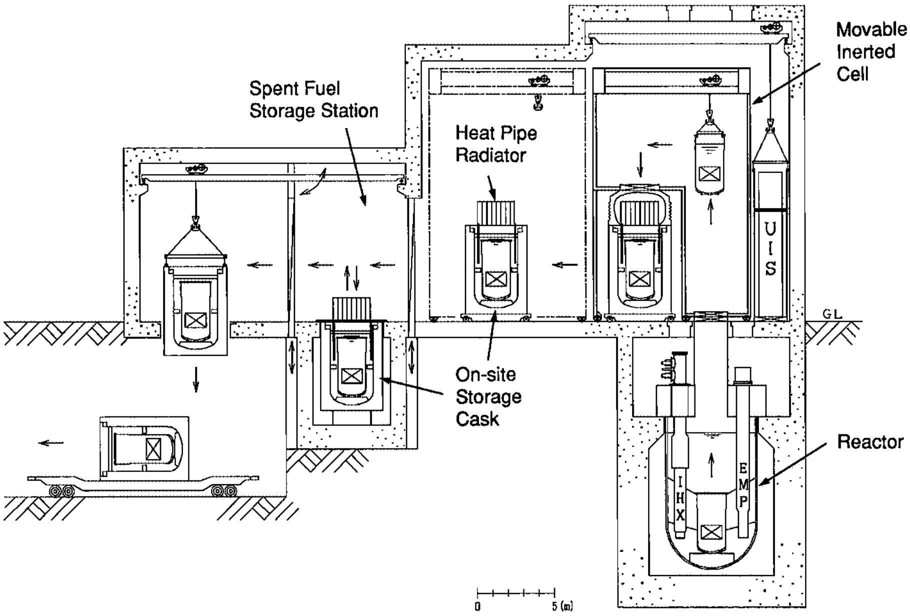


Fig. 5 The RAPID refueling concept

mental Ar gas

$L=$ 3 (m), effective length of the IFA

Gr: Grashof number

$Pr=$ 0.67, Plandtl number of the Ar gas

$G=$ 9.8 (m/s^2), acceleration of the gravity

$\beta=$ 1/273, volumetric expansion rate of the Ar gas

$\nu=$ 0.317×10^{-4} (m^2/s), dynamic viscosity of the Ar gas.

If the IFA surface temperature T is obtained, the temperature difference of the hottest fuel pin center and the IFA surface is approximated by the following equation. This equation is used to estimate the fuel pin temperature distribution. Here the IFA is regarded as a large fuel pin. For the sake of simplicity, the void leading channel in the IFA center is neglected:

$$\Delta T_F = \frac{QF_z}{A}\left(\frac{t_{\mathrm{SUS}}}{\lambda_{\mathrm{SUS}}} + \frac{t_{\mathrm{Na}}}{\lambda_{\mathrm{Na}}} + \frac{r_{\mathrm{core}}}{2\lambda_{\mathrm{core}}}\right), \tag{6}$$

where

ΔT_F: Temperature difference of the hottest fuel pin center and the IFA surface

$Q=$ $Q_{Rad}+Q_{NC}$

$F_z=$ 1.3, radial peaking factor

A: Surface area of the IFA

$t_{\mathrm{SUS}}=$ 45×10^{-3} (m), total thickness of austenite steel of the IFA

$t_{Na}=$ 70×10^{-3} (m), thickness of the sodium layer in the annulus of the IFA

$r_{\mathrm{core}}=$ 750×10^{-3} (m), equivalent radius of the core

$\lambda_{\mathrm{SUS}}=$ 66 (W/m·K), thermal conductivity of austenite steel

$\lambda_{\mathrm{Na}}=$ 20 (W/m·K), thermal conductivity of the sodium

$\lambda_{\mathrm{core}}=$ $0.39\times12+0.45\times66+0.16\times20=38$ (W/m·K), averaged thermal conductivity of the core region.

The averaged thermal conductivity of the core region λ_{core} is obtained on considering the volume fraction of fuel (39%), sodium coolant (45%) and structure (16%).

The result of the calculation on case (1) is shown plotted in **Fig. 6**. The design criteria is to keep the pellet temperature under 750°C to avoid eutectic reaction of the fuel and cladding. To meet this requirement, refueling can be carried out 2 months after reactor shutdown. The other cases to be anticipated are less severe than above case. For example, if the IFA locked up in the closure head, sufficient decay heat removal is attained from the outer surface of the IFA. In cases where the IFA remained in the OSSC without heat pipe radiator, natural circulation of sodium in the OSSC as well as the increased sodium free surface will enhance decay heat removal.

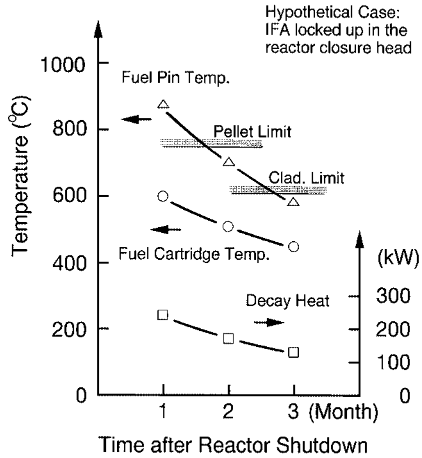


Fig. 6 Calculated temperature of the spent fuel

3. Spent Fuel Storage

The IFA encased in the OSSC is stored in the spent fuel storage station (SFSS) adjacent to the reactor building. At the beginning of the storage, the fuel cladding and pellet temperatures will be limited to 500°C with a maximum decay heat of 180 kW. Dissipation of the decay heat will solidify sodium in the OSSC 3 years from the refueling. Then spent fuel together

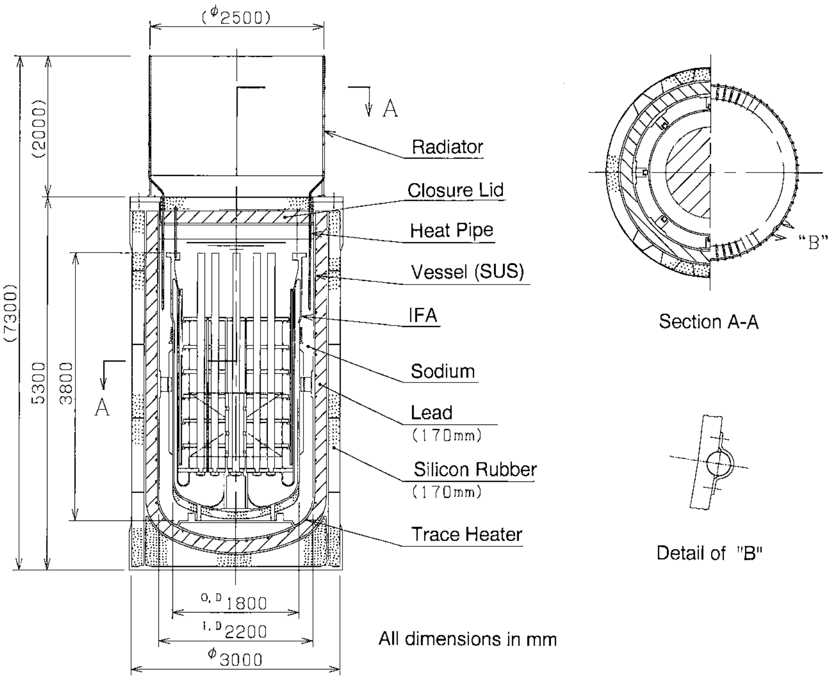


Fig. 7 On-site storage cask

with the OSSC could be transported to the reprocessing plant without any intermediate steps. Detail of the OSSC is shown in **Fig. 7**. The essential features of the OSSC are its passive decay heat removal capabilities performed by the heat pipe radiator and natural circulation air cooling established in the SFSS. Thus the OSSC functions over a full range of expected environments without the use of forced cooling or any supplemental equipment or power source.

4. Safety Considerations on Criticality

One of the most important issues in designing an IFA is its criticality safety on land impact and/or water flooding during fuel handling and transportation. The total plutonium inventory of each IFA is almost 800 kg. It is assumed that a fresh IFA is encased in the OSSC without sodium, while a spent IFA encased in the OSSC with frozen sodium during transportation. In every case, all the control rod followers (12 in total) are provided with poison rods. In addition the void leading channel (VLC: refer to IV.3 and Fig. 2) in the IFA center is also filled with a poison rod. B_4C enrichment of 80 at% is adopted for all the rods. The accident scenarios to be considered are: (1) compaction of a fresh IFA encased in the OSSC on land impact; and (2) compaction of a fresh IFA encased in the OSSC with subsequent water flooding. Compaction of a fresh IFA during manufacturing (*i.e.* an IFA not encased in the OSSC) is less severe than case (1) because no neutron reflection by surrounded lead gamma shield of the OSSC is expected. To ensure the criticality safety during manufacturing, all the control rods should be inserted and temporally locked prior to the fuel pin loading. Compaction of a spent IFA is also less severe case because compaction of the core could be restricted by frozen sodium in the OSSC. Both in cases (1) and (2), compaction mode of side impact (*i.e.* compaction exerted normal to the center axis of the core) with a compaction ratio 0.625 was assumed. This case is quite severe and unlikely to happen because no space of sodium coolant is left, in addition some amount of bonded sodium in the cladding was also removed so that maximum compaction on side impact should be attained. In case of compaction with subsequent water flooding, bonded sodium remained in the cladding is assumed to be replaced by water.

Calculations were performed with the multigroup Monte Carlo code[6] in the exact cylindrical geometry. In the model, neutron reflection by the surrounding lead gamma shield of the OSSC was taken into account. Results obtained were $K_{\rm eff}$=0.9496±0.0017 for case (1), and $K_{\rm eff}$=0.9033±0.0022 for case (2). Subcriticality criteria ($K_{\rm eff}$<0.96) was confirmed in all cases.[1]

5. Proliferation Resistance

For a fuel cycle to be proliferation resistant, unauthorized removal of fissile material must be very difficult and, if such removal is attempted, readily detectable. In view of this, we insist on establishment of international network of the Integrated Fuel Assembly (IFA) fuel cycle for RAPID reactors, combined with an effective safeguards system.[1] This plan possesses definite resistance to state-supported removal of plutonium for nuclear weapons production so long as the following measures could be undertaken:

(1) Highly standardized RAPID reactors with compatible IFA design should be constructed. Construction recommended even in the nation(s) suspicious of state-supported removal of plutonium for nuclear weapons production.
(2) The IFA reprocessing plant(s) should be constructed only in the nation(s) that undertake nonproliferation measures and accept international safeguards. A network of bilateral and multilateral agreements by those

nations should be established.

(3) The reactor power plants should be equipped with the IAEA containment/surveillance devices. They include the followings: reactor power monitors, ultrasonic seals, thermal luminescent dosimeters, closed-circuit television and modular video systems, *etc.*

(4) International network of the IFA shipment from the reactor site in any nations to the reprocessing plant(s) should be guaranteed.

Significant advantages of the IFA fuel cycle/RAPID reactor concept are as follows:

(1) In the current nonproliferation regime, materials accountability is an essential feature of an effective nuclear safeguards program. In the IFA fuel cycle/RAPID refueling concept, however, such a measure is not required for the reactor sites because plutonium would be very difficult to obtain from an IFA without drastic alterations in the reactor installation and/or an OSSC. Such alterations could be easily detectable by satellite surveillance. The essential feature of RAPID refueling concept from the view of proliferation resistance is that the reactor installation does not have fuel handling equipments except for in-cell cranes.

(2) Other measures included in safeguards program are containment and surveillance. The IFA fuel cycle/RAPID refueling concept also satisfies this requirement. Plutonium removal would be difficult because the IFA removed from the reactor will contain liquid sodium and high reactivity fissile materials that will produce high level of decay heat (180 kW), will require handling of the IFA in shielded, inert-atmosphere cell. Furthermore, the reactor installation has neither fuel handling equipment nor sodium removal facility/dismantling equipment. The IFA removed from the reactor should be encased in the OSSC immediately for decay heat removal. The on-site storage cask acts as an additional containment for the IFA during on-site storage and transportation. During transportation, solidified sodium encased in the OSSC also acts as containment. Dimension of the IFA (1.7 m diameter and 4 m long) facilitates thorough surveillance.

(3) Safeguards inspections in the reactor site would be necessary only at the refueling every 5 years, and in spent fuel shipment after 3 years from each refueling.

IV. Reactor Core Design

1. Optimization of the Neutronic Coupling (Design features)

The reactor has 7 cores (7 IFAs) of the same dimensions as that of RAPID. In order to achieve a strong neutron coupling and to obtain the reactor power 10 times as much as that of RAPID, the following design approaches were made.

- The distance between the centerlines of the adjacent cores (2,300 mm) was designed to be as short as possible, while ensuring the structural integrity of the closure head.
- The LNP rods among the IFAs at the active core level were eliminated.
- A radial blanket thickness of only 30 mm was adopted.

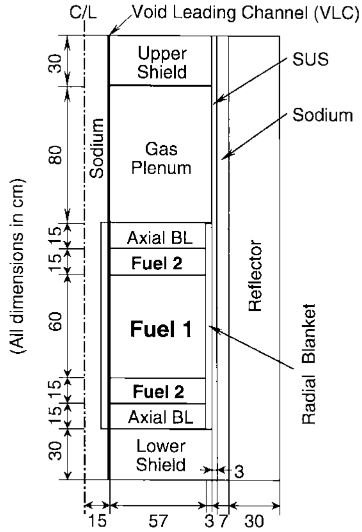


Fig. 8 2-dimensional *R*-*Z* model of one of the cores

The 2-dimensional *R*-*Z* model of each core (IFA) is shown in **Fig. 8**. Each core is a homogeneous design with two regions: a core height of 600 mm for the inner active fuel ("Fuel 1" as shown in Fig. 8), and 150 mm long upper and lower axial active fuel ("Fuel 2" in Fig. 8). The active core region is 1,440 mm diameter with a center channel 300 mm diameter.

The Pu enrichments were adjusted to minimize both the axial and radial peaking factor of all the cores as well as to reduce the burnup reactivity swing. In the modular island core, there is a greater dependence of the radial peaking factor as a function of the Pu enrichments of each island core (IFA). If all the cores were the same, the power of the center core would be twice as much as the outer cores, so this design was rejected. Finally the followings are the fuel chosen for the cores.

Center core (IFA):	U–12.8Pu–10Zr (Fuel 1),
	U–18.5Pu–10Zr (Fuel 2)
Outer cores (IFAs):	U–13.0Pu–10Zr (Fuel 1),
	U–19.0Pu–10Zr (Fuel 2)

Criticality calculations on considering neutronic coupling among the cores were performed using the neutron transport code THREEDANT with third order of Legendre polynomial expansion and eight Gaussian quadrature points for neutron scattering angle (P_3S_8 approximation) on 2-dimensional *X*-*Y* models of a 1/4 segment. The power density distributions of the beginning-of-life (BOL) cores are shown in **Fig. 9**. Because of asymmetric flux distribution in the outer cores, their power distribution is horizontally asymmetric. A performance breakdown of the reference core is shown in **Table 2**.

2. Burnup Characteristics

Burnup calculations were performed using diffusion code CITATION on 3-dimensional *X*-*Y*-*Z* models. The burnup reactivity swing after 5 years was less than $1.8\%dk/kk'$ which can meet both the safety and control rods requirements. Therefore the reactor can be operated without refueling for up

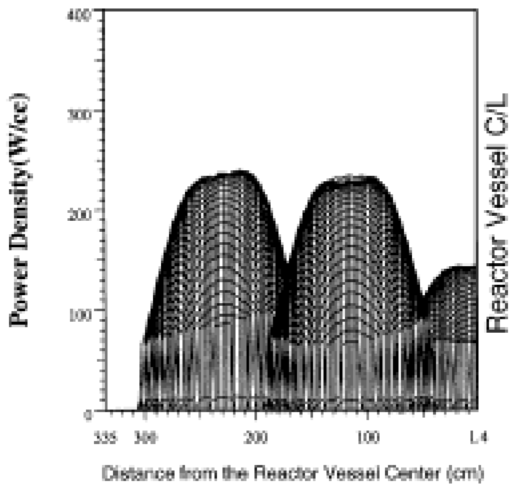


Fig. 9 Power density distribution of the BOL cores

Table 2 Performance breakdown of the reference core

Fuel enrichment (Fuel 1/Fuel 2)	
Center core: U–12.8Pu–10Zr/U–18.5Pu–10Zr	
Outer cores: U–13.0Pu–10Zr/U–19.0Pu–10Zr	
Thermal output	600 MWe
Design lifetime	5 years
Breeding ratio	1.12
Burnup	50,000 MWd/t
Peak linear power	260 W/cm at BOL outer core
Peak power density	240 W/cc at BOL outer core
Sodium void reactivity worth	<1$: One of the core voided ∼4$: All the core voided

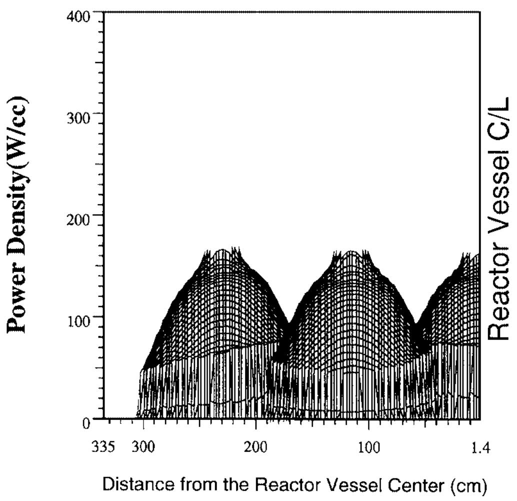


Fig. 10 Power density distribution of the MOL cores

to 5 years. The power density distributions of the middle-of-life (MOL) cores are presented in **Fig. 10**. The peak power density (240 W/cc) and the peak linear power (260 W/cm) were found in the outer BOL cores. Breeding gain of no less than 1.1 and burnup of 50,000 MWd/t have been achieved. The breeding gain and burnup obtained were not so high as those of Super Phenix or MONJU, however, our design policy prefers not only such parameters, but also other important advantages (*i.e.*, long life fuel, quick and simplified refueling, compact reactor structure, proliferation resistance, *etc.*). Result of the burnup calculation is listed in **Table 3**.

3. Sodium Void Reactivity Worth

Considerable attention has been made to reduce a sodium void reactivity worth by placing a void leading channel (VLC)[1] in each core (IFA), as shown in Figs. 2 and 8. In this particular core configuration, the neutron streaming through the VLC plays an important role in the total neutron balance that determines $K_{\rm eff}$. Sodium void worth calculations were made using 3-dimensional MCNP code.[6] A sodium void reactivity value of less than 1$ was confirmed in case that sodium voided in one of the cores (IFAs). One of the advantages of the RAPID-M concept is that void propagation into the adjacent cores (IFAs) could be avoided. Even though sodium voided in all the cores (IFAs) simultaneously, a whole core sodium void reactivity value of approximately 4$ has been estimated. The results are summarized in **Table 4**.

Table 3 Result of burnup calculation

Reactor power (MWth)	1,500	
Number of modulus	7 IFAs	
Design life (years)	5	
Effective multiplication factor: $K_{\rm eff}$		
BOL	1.04276	($\delta\rho$)
After 1 year	1.03981	−0.27%
After 2 years	1.03642	−0.59%
After 3 years	1.03271	−0.93%
Breeding gain		
BOL	1.174	
After 3 years	1.127	
Burnup (MWd/t)	Center core	Outer cores
After 1 year	7,258	10,096
After 2 years	15,262	20,043
After 3 years	24,449	29,762
Maximum linear heat rate (W/cm)	Center core	Outer cores
BOL	161	261
After 1 year	172	207
After 2 years	180	198
After 3 years	187	189

Table 4 Result of sodium void worth calculation

Case	$K_{\rm eff}$	$\delta\rho$
Normal operation	1.03716	—
Center core voided	1.04091	0.35%
2 outer cores voided	1.04655	0.87%
4 outer cores voided	1.05157	1.32%
6 outer cores voided	1.05525	1.65%
All the cores voided	1.05553	1.68%

Definition: Inner core, outer core, axial blanket and VLC are supposed to be voided in the voided IFA.

Table 5 Kinetic parameters

	Center core	Outer cores
β_{eff}	3.937E−03	3.935E−03
Λ	2.480E−09	2.456E−07
λ1	1.30E−02	1.30E−02
λ2	3.14E−02	3.15E−02
λ3	1.36E−01	1.36E−01
λ4	3.46E−01	3.46E−01
λ5	1.38E+00	1.39E+00
λ6	3.87E+00	3.87E+00

V. Reactor Kinetics

1. Kinetics Parameters

The kinetics parameters of the RAPID-M on considering neutronic coupling among the cores (IFAs) were calculated using the neutron transport code THREEDANT in the P_1S_4 approximation on 3-dimensional X-Y-Z models of a 1/8 segment. The results are shown in **Table 5** in which effective delayed neutron fraction β_{eff}, prompt neutron life Λ (s), and decay constant λ (s^{-1}) are listed. No practical difference was perceived between the center and outer cores.

The reactivity parameters were also calculated. In the geometry coefficient calculation, deviation of the distance between the IFAs is taken into account using a 2-dimensional X-Y model of a 7 cores (7 IFAs) model.

2. Multi-point Reactor Kinetics Model

In case of a large core where the coupling between distant regions in the core is weak, there may arise a flux tilt, or the flux distribution may change globally, even if the control rods are moved locally.[4] In view of this, the multi-point reactor kinetic equations proposed by Kobayashi[7] were applied to investigate the behavior of the RAPID-M core.

In this method, the coupling coefficient between any regions can be exactly calculated by dividing the core into appropriate subregions, and the relation of the flux tilt and the strength of the coupling between regions can be available. The neutronic characteristics of the modular island core depend on the degree of decoupling between the modules (*i.e.* the cores). The coupling coefficients k_{mn} are defined by Eq. (7), and are given by the following parameters:

- Effective multiplication factor K_{eff} and the neutron flux ϕ of all the cores (7 IFAs).
- Intensity of the neutron source $S(r)$ and importance G_m in each core (IFA).

The coupling coefficients k_{mn} denote the probability of the fission in the region m due to a neutron generated in the region n:

$$k_{mn} = \frac{\sum_g \int_{V_n} \chi_g G_m(r, g) s(r) dr}{\int_{V_n} s(r) dr}, \tag{7}$$

where

G_m: Importance function
$S(r)$: Intensity of the neutron source
V_n: Space volume for region n
g: Energy group number
χ_g: Fission spectrum for group G.

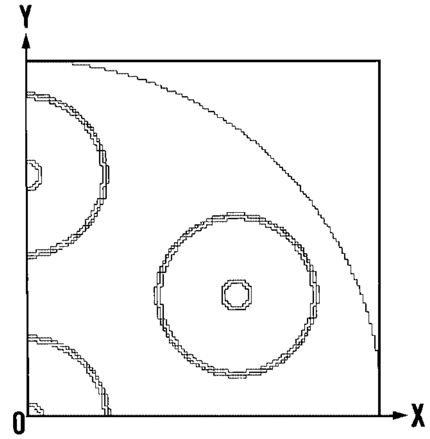


Fig. 11 1/4 sector of X-Y-Z model for the forward caluclation

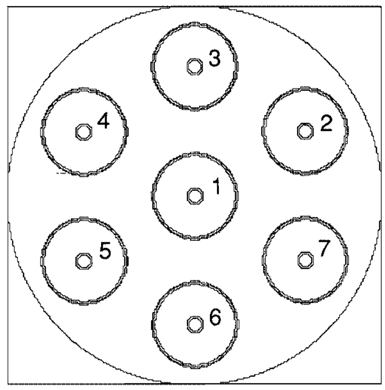


Fig. 12 Whole core model

A 1/4 sector of the X-Y-Z model for the forward calculation is shown in **Fig. 11**. The adjoint calculation, however, requires a whole core model as shown in **Fig. 12** to estimate the importance distributions of all the combinations of 7 cores (IFAs). Calculations were performed using neutron transport code in the P_3 approximation on 2-dimensional X-Y model. Such an approximation on 2-dimensional X-Y model is valid to estimate lateral neutron transport which dominates the coupling effect of the cores (IFAs), however, an axial neutron leakage was also taken into account by introducing the buckling factors. The total neutron flux distribution is shown in **Fig. 13** that provides the intensity of the neutron source $S(r)$. The importance distributions in respect to the core 1 (center core) and the core 2 (one of the outer cores) are presented in **Figs. 14** and **15**, respectively. Each figure presents a reasonable distribution. **Figure 16** illustrates the importance distribution at the section X=0. The coupling coefficients can be calculated by above neutron source intensity and importance distributions. The results are listed in **Table 6**.

Table 6 Calculated coupling parameters K_{mn}

mn	1	2	3	4	5	6	7
1	1.056E+00	3.565E−03	3.541E−03	3.565E−03	3.565E−03	3.541E−03	3.565E−03
2	3.269E−03	1.056E+00	3.258E−03	4.026E−05	2.965E−06	4.201E−05	3.226E−03
3	3.246E−03	3.257E−03	1.056E+00	3.257E−03	4.198E−05	2.991E−06	4.198E−05
4	3.269E−03	4.026E−05	3.258E−03	1.056E+00	3.226E−03	4.201E−05	2.965E−06
5	3.269E−03	2.965E−06	4.201E−05	3.226E−03	1.056E+00	3.258E−03	4.026E−05
6	3.246E−03	4.198E−05	2.991E−06	4.198E−05	3.257E−03	1.056E+00	3.257E−03
7	3.269E−03	3.226E−03	4.201E−05	2.965E−06	4.026E−05	3.258E−03	1.056E+00

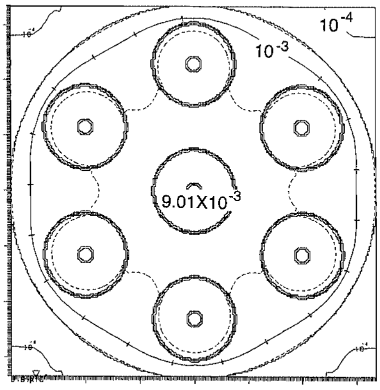


Fig. 13 Total neutron flux distribution

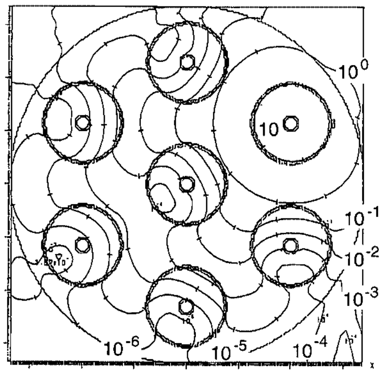


Fig. 15 Importance distribution in respect to one of the outer IFAs

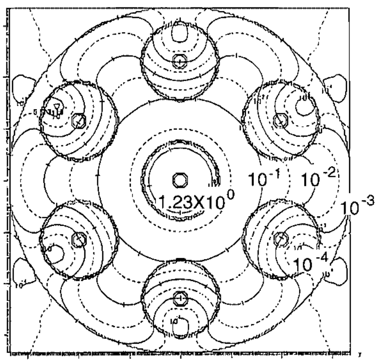


Fig. 14 Importance distribution in respect to the center IFA

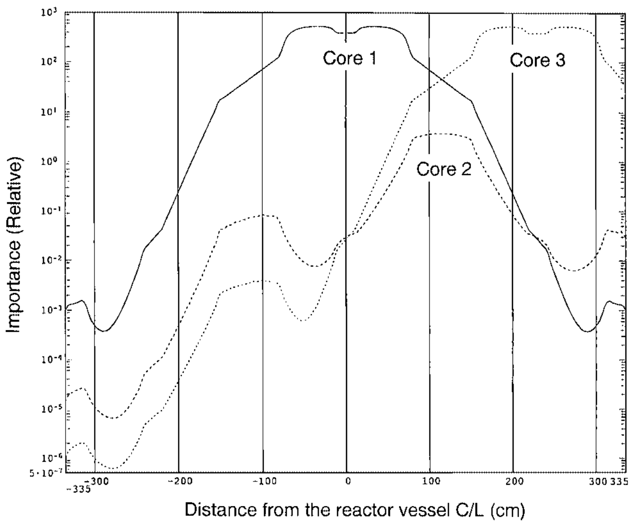


Fig. 16 Importance distribution at section $X=0$

VI. Plant Dynamics Analysis

1. Kinetics Equations

The time dependent coupling coefficients for prompt and delayed neutrons are defined as Eqs. (8) and (9), respectively.

$$k^p_{mn}(t) = \frac{\int_{V_n}\left(\sum_g G_m(r,\ g)\chi^p_g F\phi_g(r,\ t)\right)dr}{\int_{V_n} F\phi_g(r,\ t)dr} \tag{8}$$

$$k^d_{i,mn}(t) = \frac{\int_{V_n}\left(\sum_g G_m(r,\ g)\chi^d_{i,g} C_i(r,\ t)\right)dr}{\int_{V_n} C_i(r,\ t)dr}, \tag{9}$$

where

- $G_m(r,\ g)$: Importance function
- χ^p_g: Prompt fission neutron spectrum
- $F\phi_g(r,\ t)$: Fission rate (fissions/s)
- $\chi^d_{i,g}$: Energy spectrum of the delayed neutrons of the i-th delayed neutron group
- $C_i(r,\ t)$: Density of the delayed neutron precursor of the i-th delayed neutron group.

Neutron life l_m and delayed neutron fraction β_{mn} were obtained based on the importance function G_m shown above. Then the following rigorous nodal kinetics equations for coupled cores were obtained:

$$l_m(t)\frac{ds_m(t)}{dt} = -s_m(t) + \sum_{n=1}^{N}\left[\left(\frac{1}{k}(1-\beta_{mn}(t))k^p_{mn}(t)\right)s_n(t) + \sum_i k^d_{i,mn}(t)\lambda_i C_{i,n}(t)\right] \tag{10}$$

$$\frac{dC_{i,m}(t)}{dt} = \frac{1}{k}\beta_{i,m}(t)s_m(t) - \lambda_i C_{i,m}(t), \tag{11}$$

where

$$s_m(t) = \int_{V_n} s(r,\ t)dr \tag{12}$$

$$C_{i,m}(t) = \int_{V_n} C_i(r,\ t)dr \tag{13}$$

$$\beta_{i,m}(t) = \frac{\int_{V_n}\beta_i F\phi_g(r,\ t)dr}{\int_{V_n}F\phi_g(r,\ t)dr} \tag{14}$$

l_m: Neutron life
β_{mn}: Delayed neutron fraction
$S_m(t)$: Fission neutrons produced in a unit time in the region V_m
$C_{im}(t)$: Fission neutrons produced in the delayed neutron precursor in the region V_m
N: Number of core regions in a reactor.

In such a way, conventional point kinetics equations can be applied to estimate the kinetic behavior of each core region. In case $N=1$, they are point kinetics equations.

2. System Dynamic Analyses

The system dynamic analyses to identify passive safety features of the RAPID-M have been conducted using multi-point reactor kinetics equations in case of unprotected loss of flow (ULOF) and unprotected transient over power (UTOP).

The major feature of the primary system which dictates the plant safety characteristics in the ULOF transient is the adoption of three main electromagnetic pumps (EMPs) combined with a BABY pump (battery driven bypass EMP). In normal reactor operations, the BABY pump is driven by battery to pump the primary sodium. In case of a loss of power to the main EMPs, the BABY pump is able to keep a minimum core flowrate to ensure the core integrity without any flow coastdown control systems. ULOF is initiated by a loss of power to the main EMPs with failure to scram the reactor, while the BABY pump is driven by battery. The main EMPs' inertia is assumed to provide an initial flow-halving time of 0.1 s. The transient variation in the core flowrate is dependent on the reverse flow resistance of the main EMPs as well as the head-capacity characteristics of the BABY pump. In this reactor concept, each connecting tube, which connects a main EMP and the IFA, was provided with an anti-reverse-flow taper nozzle to establish sufficient core flowrate. A reverse to forward flow resistance ratio of the taper nozzle of 1.35 is assumed.

In general the reactivity feedback coefficients of the EOL core are leased favorable to the ULOF transient response, and

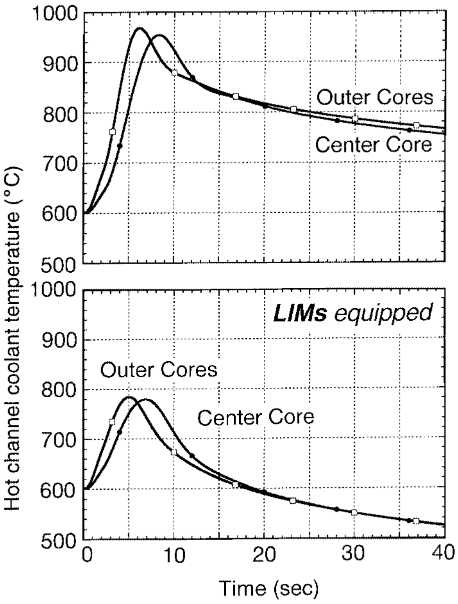


Fig. 17 Results of ULOF analyses

would yield the highest temperature and the smallest margin to coolant boiling. However the analyses have been made on the BOL core because the maximum power density and maximum linear heat rate are found in the BOL outer cores as shown in Fig. 9 and Table 3, respectively. In the analytical model, Doppler feedback, coolant density reactivity feedback and thermal expansion reactivity feedback of each core support grid and spacer grids are taken into account. **Figure 17** shows a result of the ULOF analyses in case that the BABY pump nominal flowrate is 22% of the primary flowrate. A transient peak in the hot channel coolant temperature of the outer cores (outer IFAs) was slightly higher than that of the coolant boiling (*i.e.* 960°C on considering the cover gas pressure and static sodium head exerted on the reactor core). This transient was quite severe than that of conventional fast reactors equipped with the pumps coastdown control systems. Our design policy prefers simpler EMPs (three EMPs and a BABY pump) rather than EMPs with sophisticated flow coastdown control systems or mechanical pumps with flywheels. Another analytical result in case lithium injection modules (LIMs) are equipped in the reactor is also shown in Fig. 17. The LIM[8] is an innovative self-actuated shutdown system using ^6Li as liquid poison. Because of its quick response, the transient peak in the hot channel coolant temperature was well below 800°C in this case.

Passive safety features are also expected in the less stringent UTOP and ULOHS (unprotected loss of heat sink) incidents. UTOP analyses on the beginning-of-life (BOL) core are conducted. The condition of the UTOP is supposed to be a reactivity insertion of maximum 50 ¢ in one of the cores (IFAs), with an insertion rate of 3¢/s by a fault handling of the control rod. **Figures 18** and **19** demonstrate a case of reactivity insertion in the center core (center IFA), and that in one of the outer cores (outer IFAs), respectively. In both cases, no practical disturbance was perceived in any of the cores (IFAs), except for that which had the reactivity inserted. Therefore,

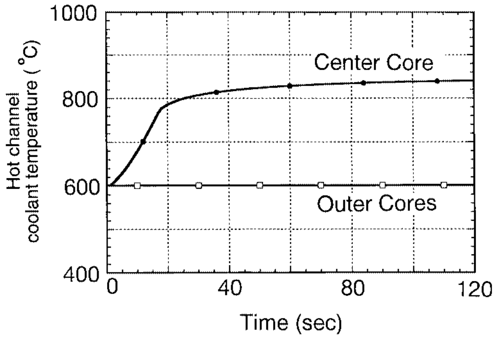


Fig. 18 Result of UTOP analysis (reactivity insertion in the center core)

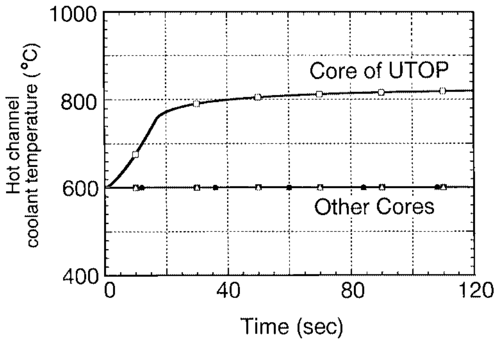


Fig. 19 Result of UTOP analysis (reactivity insertion in one of the outer cores)

UTOP analyses revealed that the RAPID-M modular island core involves a weak sensitivity to any perturbation.

VII. Conclusions

The following results have been demonstrated.

(1) The 600 MWe RAPID-M design has been presented with a reactor consist of 7 cores (IFAs). Breeding gain of no less than 1.1 and burnup of 50,000 MWd/t have been confirmed. The reactor can be operated without refueling for up to 5 years because of excellent breeding performance of the metal fueled core.

(2) The RAPID concept can eliminate conventional fuel handling systems and rotating plug(s). This is effective to reduce the construction cost per installed kW.

(3) The multi-point reactor kinetics equations on considering the coupling effect of 7 cores (IFAs) revealed that these cores involve a weak sensitivity to any perturbation.

Acknowledgments

This work is being performed under the internal funding support of Central Research Institute of Electric Power Industry (CRIEPI). The author wishes to acknowledge the contribution of H. Tsunoda of the Mitsubishi Research Institute for his reactor physics calculations.

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