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## NUCLEAR DESIGN METHOD OF

## Gd-LOADED-MOX-FUELED DEMONSTRATION ATR

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### ABSTRACT

The advanced thermal reactor (ATR) is a heavy-water-moderated boiling-water-cooled pressure-tube-type reactor, which possesses favorable characteristics in the utilization of plutonium. A demonstration ATR uses uranium-plutonium mixed-oxide (MOX) as a standard fuel material in order to take the advantage of ATR. The demonstration ATR has adopted a gadolinium-poisoned and axially multi-enriched fuel-assembly in order to flatten radial and axial power distributions. As a result, core performance improvements have been achieved, such as an increase in the average power per fuel assembly. The accuracy of the method applied to the nuclear design of the demonstration ATR has been confirmed by comparing calculated results with experimental data of the deuterium critical assembly (DCA) in Oarai Engineering Center of Power Reactor and Nuclear Fuel Development Corporation and operation data of FUGEN.

### 1. INTRODUCTION

The advanced thermal reactor (ATR) is a heavy-water-moderated boiling-water-cooled pressure-tube-type reactor, which possesses favorable characteristics in the utilization of plutonium because of a well-thermalized neutron spectrum by the heavy-water-moderator. The nuclear design of the demonstration ATR has been conducted on the basis of the experience and technologies of the prototype reactor FUGEN and R&Ds for ATRs<sup>1</sup>, as well as experiences of LWRs.

In order to take the advantage of ATR in the utilization of plutonium, as a standard fuel material, the demonstration ATR uses uranium-plutonium mixed-oxide (MOX) on which a lot of operation and burnup data has been accumulated in FUGEN. It is planned that in reloading cycles, a gadolinium-poisoned fuel assembly (Gd-loaded MOX fuel assembly), in which a few Gd<sub>2</sub>O<sub>3</sub>-mixed uranium-oxide fuel rods are loaded among MOX fuel rods, is used in order to reduce power-mismatches between fresh and burned fuel assemblies.

The nuclear design method has been developed by improving the calculation procedure and the codes adopted for the

design and the core-management of FUGEN to get higher accuracy and shorter calculation time.

Accuracy of this method has been confirmed by comparing calculated results with experimental data of the deuterium critical assembly (DCA) in Oarai Engineering Center of Power Reactor and Nuclear Fuel Development Corporation and operation data of FUGEN.

### 2. NUCLEAR DESIGN

Fig. 2-1 shows a schematic view of the core of the demonstration ATR. The core consists of pressure tube assemblies, in which MOX fuel assemblies are loaded, and a calandria tank containing heavy water. The coolant (light water) flows upwardly in the pressure tube assemblies. The fuel assembly is a 36-fuel-rod cluster arranged in three concentric layers, i.e., inner, middle, and outer layers, as shown in Fig. 2-2.

Design features of the demonstration ATR are summarized in comparison with FUGEN in Table 2-1. In the demonstration ATR, the average channel power, 26% higher than that of FUGEN, is attained with a relatively compact reactor tank (calandria tank) by increasing the number of fuel-rods per fuel assembly: from 28 in FUGEN to 36 in the demonstration ATR, and by some design improvements for flattening power distribution, such as the use of Gd-loaded-MOX. The core power is 606MWe, or about 3.7 times that of FUGEN, while the number of fuel assemblies is 616, or about 2.8 times that of FUGEN. The period of a single operation cycle is planned to be 12 months using 4-batch refueling, and the average burnup of discharged fuels in reloading cycles is expected to reach about 31 GWd/t.

The core configuration is outlined in Fig. 2-3. Besides the use of Gd-loaded-MOX, the following design improvements have been made to flatten power distribution<sup>2</sup>.

- (1) The core is radially divided into two regions; Pu-enrichment in the outer region is higher than in the inner region.
- (2) The core is axially divided into three regions, i.e., upper, middle, and lower; Pu-enrichment is higher in the upper and lower regions than in the middle region. Moreover Gd-den-

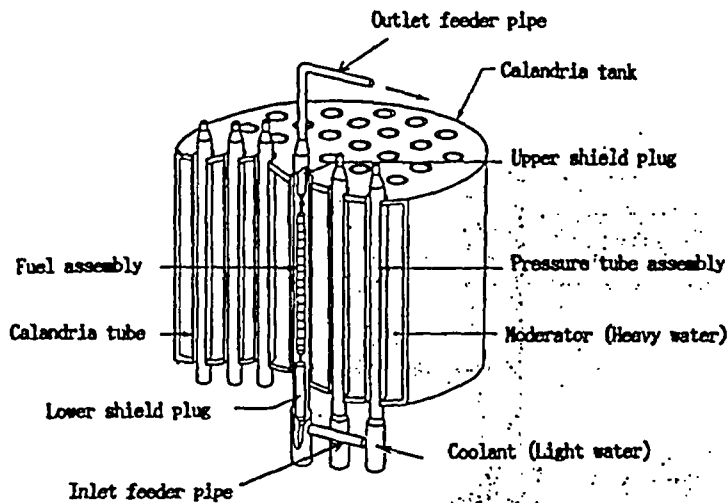


Fig.2-1 Schematic view of the core

Vertical view

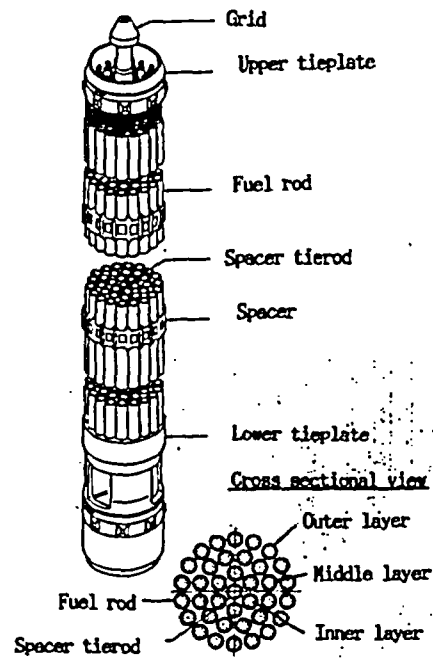


Fig.2-2 View of the fuel assembly(36-rod cluster type)

Table 2-1 Design features of Demonstration ATR and FUGEN

Parameters	Demonstration ATR	FUGEN
<b>Power</b>		
Thermal Power MW	1930	557
Electric Power MW	606	165
Gross efficiency %	31	30
<b>Fuel assembly</b>		
No. of rods/assembly	36	28
Pellet diameter mm	12.4	14.4
Pellet material	MOX Gd-Poisoned;UO <sub>2</sub> (reloading)	MOX/UO <sub>2</sub>
Cladding diameter mm	14.50	16.46
Average enrichment (fissile) initial/reloading wt%	2.6/3.3	1.4/2.0
Average burnup initial/reloading MWd/t	20,000/31,000	10,000/17,000
Max. linear heat rating kW/m (100%power)	49.2	57.4
<b>Reactor</b>		
Effective core height m	3.70	3.70
Effective core diameter m	6.72	4.05
Number of fuel assemblies	616	224

sity is higher in the middle region than in the upper and lower regions.

**Radial power distribution**

The radial power distribution is globally flattened by the design improvement (1) shown above, but the power-mismatches induced by a reactivity gap between fresh and burned fuel assemblies cannot be reduced by this method, so Gd is loaded as a burnable poison in the fuel assembly, as shown in Fig. 2-3. In this fuel assembly, Gd-density is fixed relatively low (Gd:0.7wt% ~ 1.4wt%) in respect to balancing between the effect of reducing power-mismatches with the necessity for burn out during the 12-month period. The number of Gd-poisoned rods per fuel assembly is four in the outer core region in comparison with three in the inner core region, in order to keep the

reactivity reduction effect nearly equal to that in the inner core region under the higher enrichment.

The reduction rate of the radial peaking factor by adoption of the Gd-loaded-fuel is expected to be about 10%, as shown in Fig. 2-4.

**Axial power distribution**

The axial power distribution is flattened by the design improvement (2) shown above. The ratio in Pu-enrichment of the upper and the lower region to the middle region is about 1.2, and the ratio in length of each axial region is 1:3:1 (upper:middle:lower).

The reduction rate of the axial peaking factor by adoption of the axially multi-enriched fuel is expected to be about 7%, as shown in Fig. 2-5.

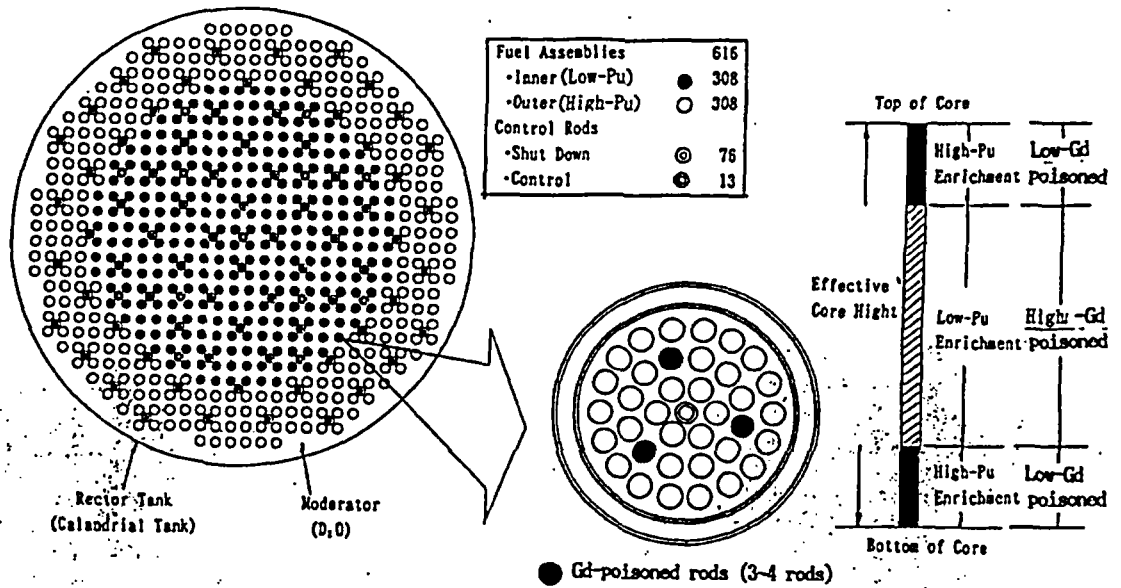


Fig.2-3 Core Configuration of Demonstration ATR

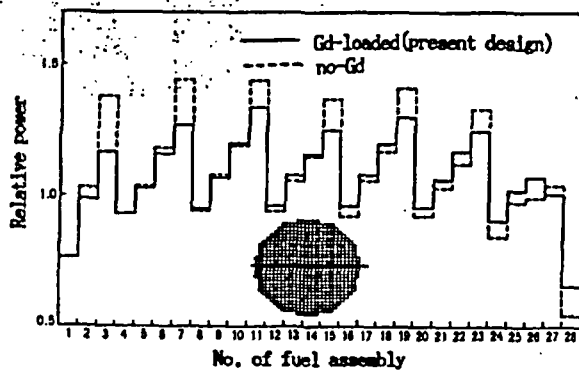


Fig.2-4 Radial power distribution of Gd-loaded core and no-Gd core

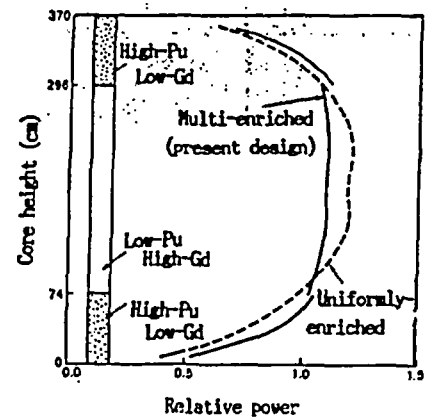


Fig.2-5 Axial power distribution of axially multi-enriched core and axially uniformly-enriched core

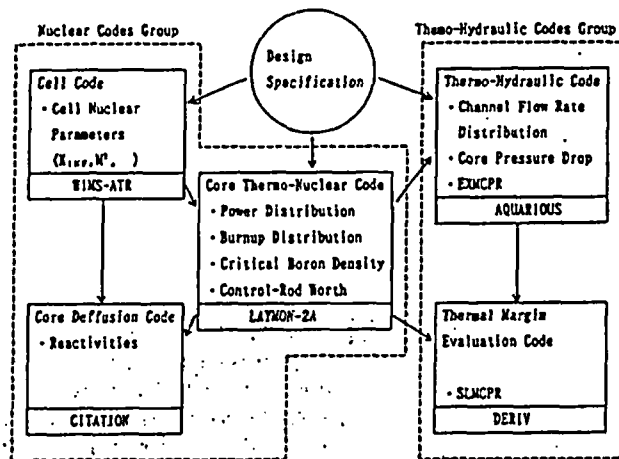


Fig. 3-1 Outline of the code system

### 3. NUCLEAR DESIGN METHOD

The code system for the demonstration ATR core design is outlined in Fig. 3-1. The code system consists of two groups, the nuclear calculation code group in which a neutron transport code for cell calculation, WIMS-ATR, a three-dimensional thermo-nuclear code, LAYMON-2A, and a neutron diffusion code, CITATION, are included, and the thermo-hydraulic calculation code group in which a thermo-hydraulic code, AQUARIUS and a thermal margin evaluation code, DERIV, are included.

The nuclear calculation code group is further divided into two parts: the cell calculation code in which nuclear parameters of a unit fuel lattice cell are calculated, and the core calculation code in which core characteristic such as power distributions and reactivities are calculated with nuclear parameters from the above cell calculation code.

#### 3.1 Cell Calculation Methods Calculation Codes

WIMS-ATR is used as a cell calculation code for the cluster-type fuel assemblies of ATR. The code has been developed for nuclear design of the demonstration ATR and the core management of FUGEN by improving WIMS-D<sup>3</sup>, which was developed in Britain. The modifications of the code were related to library improvements, such as an expansion of temperature dependent data in D<sub>2</sub>O thermal-scattering kernels, and so on<sup>4</sup>.

This code is based on multi-energy group two-dimension transport theory, and calculates cell-averaged nuclear parameters. The cross section library is based on the UKAEA file with some data modified for ATR core design.

In analyzing Gd-loaded-MOX fuel assemblies, some attentions have been paid to accurately deal with strong absorption in the thermal energy region by Gd-poisoned-fuel rods.

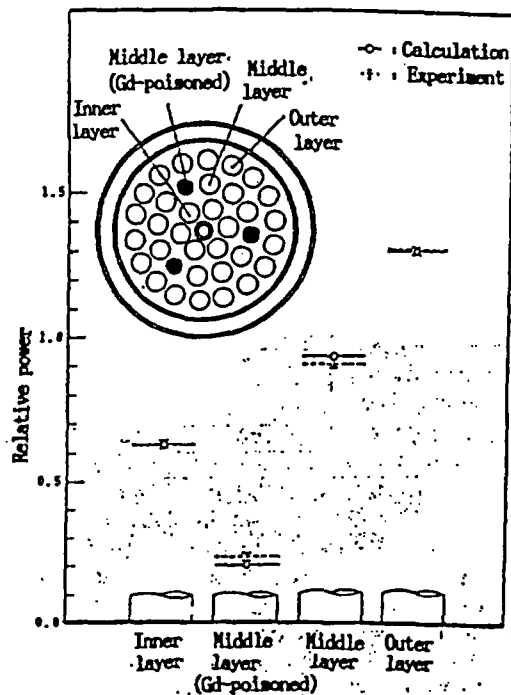


Fig. 3-2 Comparison between calculation and experiment in relative power of the each layer  
(MOX-rod;(3.4,3.4,1.6)w/oPu,Gd-rod;2.7w/oUO<sub>2</sub>,1.0w/oGd<sub>2</sub>O<sub>3</sub>)

- (1) The number of energy groups was increased up to 17 from 11 used in no-Gd fuel assembly analysis, to accurately evaluate the fine structure of the energy spectrum.
- (2) Gd-poisoned-fuel rods were radially divided into five regions to accurately evaluate the neutron flux depression within the rods.
- (3) The burnup step width was shortened to about one fifth that of the no-Gd fuel assembly analysis at the beginning of the burnup calculation, to accurately evaluate the rapid change of Gd density.

#### Verification of the Codes

The applicability of WIMS-ATR to the nuclear design of the demonstration ATR with Gd-loaded-MOX fuel assemblies has been verified using DCA as part of the "Verification Test of Advanced Power Reactors (ATR Verification Test)"<sup>5</sup> sponsored by the Ministry of International Trade and Industry.

Fig. 3-2 shows an example of comparison of power distribution in a 36-rod cluster. This cluster is demonstration ATR-type, in which three fuel rods containing gadolinium are loaded among MOX fuel rods. As shown in Fig. 3-2, the calculated results and the experimental data agreed well in Gd-poisoned rods, the outer layer on which the peak of local power is presented, and the other layers. Also, Fig. 3-3 shows comparisons of the power peaking factor in several kinds of fuel assemblies with different Pu-enrichments and Gd-densities. The calculational results and the experimental data agreed well within 5%.

No	Gd loaded 36-rod fuel			Symbol
	MOX-rod	Gd-rod #2		
	Pu-fissile [w/o] *1	U-235 [w/o]	Gd <sub>2</sub> O <sub>3</sub> [w/o]	
1	3.40/1.20	no Gd-rod		◆
2	3.40/1.60	2.7	0.0	■
3	3.40/1.60	2.7	1.0	▲
4	3.14/1.48	2.5	3.0	▼
5	3.78/1.78	3.0	0.5	●
6	3.78/1.78	3.0	1.0	★

\*1 Inner, Middle/Outer-layer

\*2 3rods in Middle layer

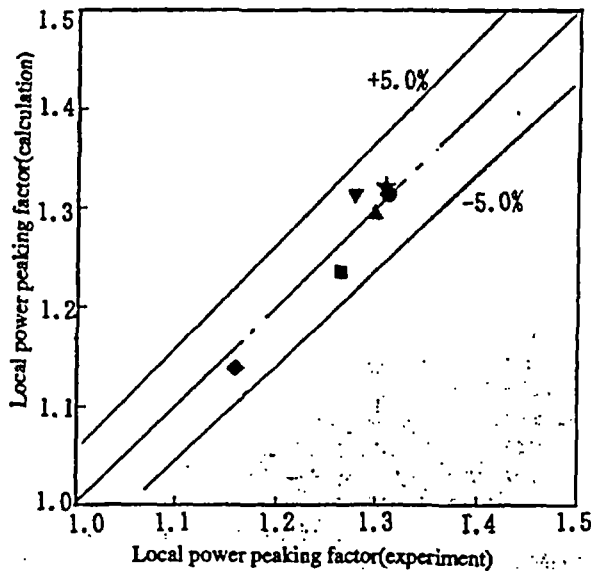


Fig. 3-3 Comparison between calculation and experiment in local power peaking factor of Gd-loaded MOX fuel assemblies

### 3.2 Core Calculation Methods Calculation Codes

The three-dimensional thermo-nuclear code, LAYMON-2A, has been developed for calculation of nuclear characteristics such as power distribution and for core-burnup planning.

This code is based on the LAYMON code developed for FUGEN. The improvements of the code in comparison with LAYMON were related to the neutron diffusion model, the functions that deal with the Gd-loaded and axially multi-enriched fuel assembly, and so on.

As the neutron diffusion model, it uses a one-energy-group, three-dimensional, modified coarse-mesh method <sup>6</sup>. In this model, the same radial mesh division is employed as in an ordinary coarse-mesh finite difference method (FDM), i.e., one

square mesh is used for each fuel lattice on the radial (XY) plane; as for the axial mesh division, 15 meshes are used in the core region. By introducing a modification factor based on the modified coarse-mesh method in the finite difference-type diffusion equation, however, almost the same calculation accuracy as that of a fine mesh FDM with three times as many mesh points is attained within the calculation time of the coarse-mesh FDM <sup>7</sup>.

Void fraction distribution for thermo-nuclear coupling calculation is obtained from the node-averaged quality which is calculated by integrating the power from core inlet to the concerned node point.

This code has several core critical search functions in order to prepare operation planning easily, so we can do the simulation analysis such as a B-10 change in moderator (D<sub>2</sub>O) with core burnup. Also, it can calculate MCPR (Minimum Critical Power Ratio) based on the CHF (Critical Heat Flux) correlation developed from experimental data of 14MW Heat Transfer Loop in Oarai Engineering Center of Power Reactor and Nuclear Fuel Development Corporation.

On the other hand, the neutron diffusion calculation code, CITATION <sup>8</sup>, is used to calculate core reactivities such as void feedback reactivity. In this calculation, we use a three-energy group, fine-mesh model.

### Verification of the Codes

As LAYMON-2A and CITATION are the main codes for nuclear characteristics calculation of the demonstration ATR, the codes have been verified by extensive comparison between calculational results and data obtained through operation of FUGEN and the experimental data of DCA.

The calculation error by LAYMON-2A in power peaking factors of the power distributions of FUGEN and DCA has been shown to be within 3% in radial peaking factors, and almost the same accuracy in axial peaking factors.

Fig. 3-4 and Fig. 3-5 show an example of evaluated results of power distribution of FUGEN. These are comparisons between calculated results and operational data of radial and axial power distributions at the cycle when the demonstration ATR-type, 36-rods fuel assemblies (no-Gd) were loaded in the center of the core. As shown in Fig. 3-4, the calculated results and the operation data agreed well in 36-rods fuel assemblies on which the peak of power was presented, and the other fuel assemblies. And Fig. 3-5 shows good agreement in the axial power distribution between the calculated results and the operation data.

Also, Fig. 3-7 shows an evaluated results of radial power distribution of a DCA core. In this core, a demonstration ATR-type, Gd-loaded, 36-rod fuel assembly was placed in the center of the core, as shown in Fig. 3-6. The calculated results and the experimental data agreed well in the Gd-loaded fuel assembly and the other fuel assemblies.

The calculation errors by CITATION in coolant void reactivity coefficients and power reactivity coefficients of FUGEN have been shown to be within  $\pm 2.0 \times 10^{-3} \Delta k/k/\%$  void and  $\pm 20\%$  (relative error), respectively.

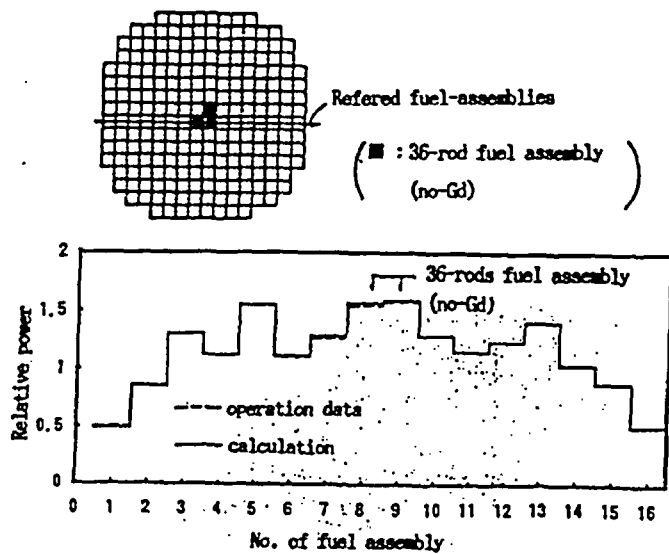


Fig.3-4 Comparison between calculation and operation data in one example of radial power distribution of FUGEN

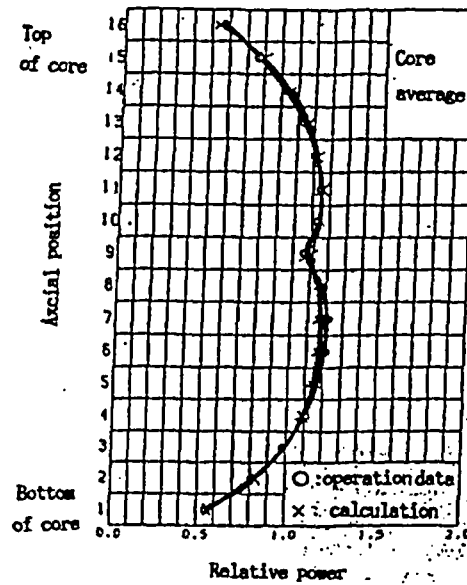


Fig.3-5 Comparison between calculation and operation data in one example of axial power distribution of FUGEN

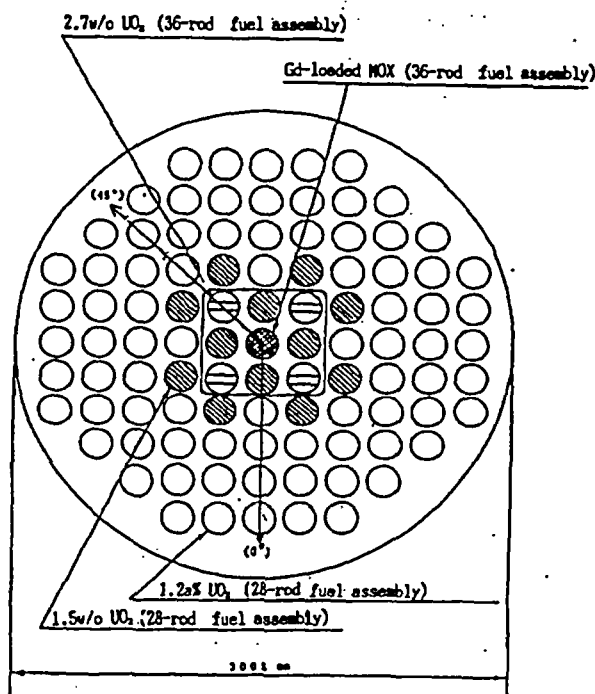


Fig.3-6 Loading pattern of fuel assemblies of DCA

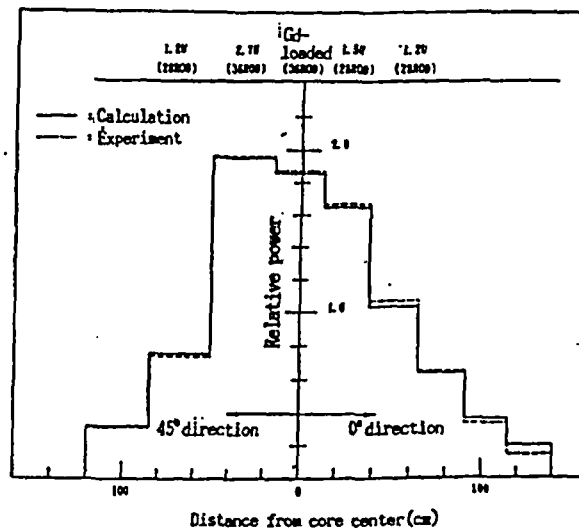


Fig.3-7 Comparison between calculation and experiment in radial power distribution of DCA  
 (Gd-loaded MOX :  
 MOX-rod;(3.78,3.78,1.78),w/oPuf,  
 Gd-rod;3.0w/oUO<sub>2</sub>,0.5w/oGd<sub>2</sub>O<sub>3</sub>)

#### 4. CONCLUDING REMARKS

The nuclear design method for the demonstration ATR has been developed and its accuracy has been evaluated on the basis of FUGEN and DCA data. It shows good applicability of the nuclear design method to the evaluation of the nuclear characteristics of the demonstration ATR with Gd-loaded-MOX fuel assemblies.

At present, lead test assemblies of the demonstration ATR-type Gd-loaded-MOX are being irradiated in FUGEN. It is planned to continue the evaluation of operation data attained during the irradiation in FUGEN for verification of the nuclear design codes of the demonstration ATR.

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