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JOYO MODIFICATION PROGRAM FOR DEMONSTRATION TESTS OF FBR INNOVATIVE TECHNOLOGY DEVELOPMENT

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ABSTRACT

A planning is under way at PNC to modify the experimental fast reactor JOYO. The project is called "Mark-III (MK-III) program". The purpose of MK-III is to expand the function of JOYO, and to make it possible to receive demonstration tests of new or high level technologies for the FBR development.

The MK-III program consists of two main modifications; a modification for improvement to highly efficient irradiation facility and a modification for demonstration test of new technologies and concepts that have a high potential to reduce FBR plant construction cost, to elevate plant reliability and to improve plant safety.

These modifications are scheduled to start in 1991.

1. Introduction

The experimental fast reactor JOYO was constructed as the first sodium cooled fast neutron reactor in 1974 and has made significant contributions to the development of FBR in Japan.

JOYO, first of all, provided the technical experience required for the design of the prototype FBR MONJU (280MWe) through operation of the Mark-I core (50~75MWt) which was designed as a miniature of a standard FBR core with a blanket, operated from 1977 to 1981. Secondly, the JOYO plant served as an irradiation facility to develop FBR fuels and materials needed for advanced reactors, by modifying the core configuration from the MK-I core to the MK-II core (100MWt), replacing blankets with reflectors and using higher power fuel assemblies, to increase the fast neutron flux, in 1982.

A planning is under way for a third phase of JOYO, which will develop the innovative technologies required for FBR commercialization. This plan is named "MK-III program" and consists of two main modifications.

The first is an improvement in irradiation capability, to conduct R&D on high performance and high burn-up fuels of commercial FBRs. Modifications to the reactor core, the heat transport system and the fuel handling system are necessary to realize this improvement. As a result, JOYO is expected to increase its reactor thermal rating from 100MWt to about 140MWt.

Another mission is to develop new technologies and concepts that have a high potential to reduce FBR plant construction cost, to elevate plant reliability and to improve plant safety. To achieve these objectives, many programs such as testings of newly developed materials, a demonstration testing of secondary sodium loop elimination including development of a bellows expansion joint and a steam generator with double-walled tubes and so on are under development.

The plan of the MK-III program is scheduled to start in 1991.

2. Description of JOYO

2.1 Outline of JOYO and its operational history

JOYO is a fast experimental reactor which uses plutonium-uranium mixed oxide and sodium as the fuel and the coolant, respectively. The project began in 1964 with preliminary design, followed by R&D of sodium technology. Site construction began at O-arai, Ibaraki-ken on January 1970, and the reactor attained its first criticality on 24th April 1977 with breeder core (MK-I). Conversion from the breeder core to the irradiation core (MK-II) was successfully completed by handling about 600 subassemblies in 10 months on schedule as planned and criticality with the MK-II core was attained in November 1982. The complete operating history of JOYO is illustrated in Fig.1.

Both 50MWt and 75MWt power level were achieved with the MK-I breeder core. The MK-I core operation covered the period from initial criticality in 1977 through 1981. There were 260 reactor startups during MK-I operation resulting from many kinds of reactor tests such as low power reactor physics tests, reactor dynamics tests, power ascent tests and transient tests. The maximum burn-up attained on a MK-I driver fuel assembly was 40,500 MWd/t which was close to the design limit of 42,000MWd/t.

The MK-I operation was completed at the end of 1981 with a scram to natural circulation test from 75MWt. The cooling system of JOYO consists of two main cooling systems and an auxiliary cooling system which removes decay heat of the core if heat removal by the main cooling systems were impossible (Fig. 2). Each of the main cooling systems and the auxiliary cooling system, then consists of the primary system, which transmits the heat from the core to the intermediate heat exchanger, and the secondary system,which transmits the heat from the intermediate heat exchanger (IHX) to the air-cooled dump heat exchanger (DHX), in normal operation.

Replacement of the whole core components from MK-I to MK-II was conducted during the year 1982. The comparison of core configuration of MK-I and MK-II is shown in Fig. 3. Their main core parameters are listed on Table 1. The MK-II core has the characteristics of higher fast neutron flux to enable accelerated irradiation of fuels and materials, higher core power density with advanced fuel subassemblies of higher linear heat rate. To obtain such features, the MK-II core is composed of the core fuel assemblies surrounded by the stainless steel reflectors,and is operated at 100MWt rated power.

The 100MWt power ascent program consisted of low power core characterization tests and high power tests. As for the low power tests, reactivity coefficient, loop pressure drop and core flow rate distribution were measured, and the performance of the reactivity control systems and the core cooling capability were confirmed prior to power ascent. Power ascent was conducted in a step-by-step manner with 25MWt power increments. At each power step, safety, stability and heat balance of the cooling systems together with its control margin were confirmed.

At the full power operation, the reactor output temperature reached 500°C with 370 °C at the inlet. The highest sodium temperature at the fuel subassembly outlet was 554 °C, which agrees with the prediction based on the measured flow rate. Also, it was confirmed that all four DHXs were able to remove the heat with an air flow margin of 20%.

Twelve normal operation cycles at 100MWt had finished by 1986, where one normal cycle consisted of 45 days operation. Maximum burn-up of the fuel pins had reached about 48,000MWd/t. Until now, JOYO has experienced neither fuel pin failure nor serious trouble on the components of the reactor system.

In order to perform the irradiation test more efficiently and also to utilize the driver fuels more effectively, the core has been modified since the thirteenth duty cycle operation which started at the beginning of September, 1987, by using modified driver fuels named J2 fuel with new license.

Comparison of the specification of the conventional J1 driver fuels and the newly licensed J2 fuels is shown in Table 2. ^{235}U enrichment of J2 fuel is increased from 12$^{w}/_{o}$ to 18$^{w}/_{o}$ to get higher reactivity. Special feature of J2 fuel is that the fissile Pu content $((^{239}Pu + ^{241}Pu) / (Pu + U))$ is specified, in contrast with the Pu oxide content $(PuO_2 / (PuO_2 + UO_2))$ which was adopted in the specification of J1 fuel. The reason to introduce the new specification is to maintain constant nuclear property of driver fuels against the changes in fissile Pu content of reprocessed Pu.

The number of J2 fuel in the core has been gradually increased as shown in Table 3. The operation period of a cycle was, then, gradually extended from 45 days corresponding to the number of J2 fuel in the core. The average burn-up of the core was also increased at the same time.

Consequently, 70 days operation was attained at the fifteenth duty cycle which finished its operation in May, 1988, where the maximum burn-up of a driver fuel reached over 70,000MWd/t.

2.2 Irradiation facility of JOYO

JOYO provides some kinds of instrumented test assembly, together with various uninstrumented irradiation subassemblies (UNIS).

UNISs have the same outer shape as the core driver fuel subassembly. Thus, they are able to be transferred through the fuel handling facility of the reactor into and out of the core. In addition, they can be exchanged for driver fuel subassemblies, namely, they are able to be irradiated at any position in the core. The UNISs provide certain monitors which inform of typical irradiation conditions through post-irradiation examinations. The UNISs which are utilized at present are classified as type A, B and C, as shown in Fig. 4.

The instrumented test assembly (INTA) has been developed as a test rig capable of monitoring the behavior of fuels and materials during irradiation period. The INTAs provide some on-line instruments which inform of many irradiation conditions and describe the behavior of fuels and materials.

The informations from these instruments are logged continuously during the irradiation, i.e., during the reactor operation. The signals are sent to the data acquisition system by the instrument leads which are penetrating the boundary of the reactor vessel through its upper internals. Thus, the INTAs are to be very long test rigs which are loaded from the top of the reactor vessel and occupy the long space extending from the above of the reactor vessel to the reactor core.

The schematic diagram of INTA is shown in Fig. 5.

3. Outline of MK-III program

3.1 Background

Approximately for forty years after EBR-I's first criticality, in which the concept of FBR was demonstrated and gave impetus to the prospect of a long term reliance on nuclear fuel, the FBR technologies have been steadily developed in each of the countries in which FBRs have been or are being built.

Also in Japan, the R&D for FBR is being proceeded, aiming at the commercialization around 2030 through construction of several FBRs with a step-by-step improvement of technologies and economies. The prototype FBR MONJU which stands between the experimental reactor and the demonstration reactor in the development program of FBR and aims to attain technological advancement and economic prospect towards the establishment of commercial viability of future FBR plants is being built.

In general, the FBR development is in the planning stage to realize a demonstration plant with technologies and economies which is comparable with those of commercial LWR.

In this respect, it is requested to clarify a long-term strategy for FBR development, based on the result of R&D including the experience of plant design, construction, operation and maintenance and further innovative technologies or new concepts for FBR plant system.

As mentioned above, the target of the development for commercialization is expected to attain before 2030 by reducing the construction cost and the fuel cycle cost, namely, by reducing the power generating cost of FBR less than that of LWR, with the same level of safety and maintenability.

From this point of view, the following items are requested to test or demonstrate;

(1) Steady and unsteady irradiation test for the development of high performance and high burn-up fuel.

(2) Evaluation of inherent safety features based on ATWS related test.

(3) Demonstration test of new system concept and components.

3.2 Objectives

JOYO modification program is planned for following utilization; (Fig. 6, Fig. 7 and Fig. 8)

(1) Utilization as a highly efficient irradiation facility

Irradiation capability of JOYO is to be improved by core modification and power up rating. Fast neutron fluence per a year is to be increased to approximately $9 \times 10^{22} n/cm^2$ which is equivalent to twice of present value, in order to demonstrate FBR fuel burn-up of 150~200GWd/t (average in assembly) by about 2000, at which year the determination of basic specifications of commercial plant seems to be scheduled.

(2) Utilization as a demonstration facility of new system concept and components

The innovative plant systems, components and new material such as Self Actuated Shut Down System, Expansion Joint (Bellows) System, Component Integral System,Secondary Sodium Loop Elimination System, In-Core Anomaly Diagnostic System, Advanced Sodium Pump, Structural and Shielding Materials, etc., are to be developed based on demonstration tests in JOYO by taking necessary modifications.

3.3 Modification to a highly efficient irradiation facility

The planned modification is to increase the fast neutron flux by 30%~40% and to increase the plant availability by 50%.

It means that the irradiation time for the necessary fuel burn-up is expected to reduce by a half.

The modifications on reactor core, heat transport system and fuel handling system are mainly necessary to realize this improvement.

3.3.1 Core Configuration

- Expansion of core fuel zone ; The maximum number of core fuel subassemblies are increased from 67 to 85 and the two zoned core is designed for flattening power distribution.

  The core configuration and main parameters are shown in Fig. 3 and Table 1,respectively and the modification procedure of the core configuration from MK-II to MK-III is shown in Fig. 9.
- Change of control rod allocation ; Two control rods are moved to the fifth row from the third row.
- Application of radial shielding assemblies ; Corresponding with the increase of fast neutron flux, the most outside stainless steel reflectors are replaced with the B_4C neutron shielding assemblies which are to provide neutron and gamma shielding for the reactor structure materials.
- Active core length ; The active core length is decreased to 48cm from 55cm in order to get a higher neutron flux.

3.3.2 Fuel design

The geometrical specifications of driver fuel is the same as current Mark-II fuel. However, at the evaluation of maximum allowable linear heat rate (more than 500W/cm), the growth effect of central void in fresh fuel pellets during power ascent is to be considered.

3.3.3 Heat transport system

As the result of the core design to improve fast neutron flux, JOYO is expected to increase its reactor thermal rating from 100MWt to about 140MWt.

For the increase of thermal rating, the temperature difference between core inlet and outlet is to be changed from 130°C to 150°C, coolant flow rate is increased by about 20% and the heat transfer area of DHX is increased by 40%.

3.3.4 Improvement of plant availability

- Improvement of fuel handling system ; For the transfer rotor which transfers the fuel between inside and outside of containment vessel, another function as a storage facility of spent fuels (about 15 subassemblies per one refueling) is to be added to decrease the refueling time by about 30%. The comparison of refueling time of MK-II and MK-III is shown in Fig. 10.
- Reduction of annual inspection period ; The period of annual inspection is decreased to about two months from 4~5 months by utilization of remote inspection devices.
- Improvement of irradiation technology ; The loading time of INTA is decreased to about two days from one month by the utilization of in-

sodium connector which eliminates the cutting of instrumental lead in the core region.

3.4 Development programs of new technologies and concepts

(1) New materials testing

The possible candidate materials of structure, shielding and fuel are planned to test.

(2) Self-actuated shutdown system (SASS)

A self-actuated shutdown system with a temperature sensitive electromagnet, which is expected to operate by temperature increase of the coolant without any control of the existing protection systems, is scheduled to be installed for the demonstration test.

(3) Demonstration test of bellows expansion joint

The demonstration test of a bellows expansion joint is planned in the secondary system of JOYO at the time of the modification of the heat transport system.

(4) Development of in-core anomaly diagnostic system

The demonstration test of an in-core neutron flux monitor and acoustic monitor and the functional tests of in-core anomaly diagnostic system are planned to improve FBR plant safety.

(5) Development of fuel failure diagnostic system

Demonstration tests are planned at the MK-III core on a fast failed fuel detection (FFD) & location system (FFDL) and an on-line plant contamination monitor, which improve operational reliability and optimize plant design and safety logics.

(6) Development of automatic reactor operation system

The demonstration test of the automatic reactor operation system is planned for the reduction of operator's load and the improvement of operational reliability.

(7) Development of new concept for the heat transport system

The demonstration test program on secondary sodium loop elimination system which has a high potential to reduce FBR plant construction cost (down 10~15%) is under development. In order to realize this concept, the technical feasibility studies are under way for the development of highly reliable steam generator and the refinement of safety logics. A 70MW steam generator of double-walled tube type is planned to be installed in the primary heat transport system. The layout of secondary sodium loop elimination system is shown in Fig. 11.

3.5 Procedure and schedule of modification

The modification to highly efficient irradiation facility is to start with a licensing work on changing the location of control rods and the expansion of core size from 67 of fuel assemblies to 85 in 1991.

The modifications of the fuel handling system and the heat transport system are planned to complete by 1993 and 1996, respectively.

The demonstration tests of new technologies and concepts is planned to proceed step by step from 1993 to 1999.

The demonstration test of the secondary sodium loop elimination system including development of a bellows expansion joint and a steam generator with double-walled tubes is planned to start with the preparations for the establishment of safety logics and the development of highly reliable steam generator.

4. Conclusion

The modification planning is under way for the third phase of FBR development utilizing JOYO, which will develop the innovative technologies required for FBR commercialization.

JOYO has many re-licensing experiences such as modification to the MK-II core, change of the MK-II driver fuel specifications and fission products release test and so on in the past.

The MK-III program will be also proceeded step by step, along with the licensing process, the reassessment of current plant conditions and the results of related R&Ds.

Table 1 Main Core Parameters of JOYO

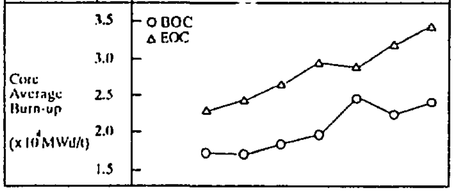
Items \ Core (Fuel)		MK-I		MK-II	MK-III
		First	Second		
Reactor Output	MWt	50	75	100	140
Primary Coolant Flow Rate	t/h	2,200	2,200	2,200	2,750
Reactor Inlet Temperature	°C	370	370	370	350
Reactor Outlet Temperature	°C	435	470	500	500
Core Stack Length	cm	60	60	55	48
Core Volume (max.)	ℓ	294	304	250	235
Linear Heat Rate (max.)	W/cm	210	320	400	500
Fuel Pin Diameter	mm	6.3	6.3	5.5	5.5
$PuO_2/(PuO_2+UO_2)$	w/o	18	18	~30	~30
^{235}U Enrichment	w/o	23	23	~12(J1) ~18(J2)	~12(Inner Core) ~20(Outer Core)
Neutron Flux (max.)	$n/cm^2 \cdot sec$	2.1×10^{15}	3.0×10^{15}	4.2×10^{15}	5.5×10^{15}
Neutron Flux (Core av.)	$n/cm^2 \cdot sec$	1.2×10^{15}	1.9×10^{15}	3.1×10^{15}	3.8×10^{15}
Max. Excess Reactivity	%Δk/k	~4.5	~4.5	~5.5	T.B.D.
Control Rod Worth	%Δk/k	Safety Rod ~5.6 Regurating Rod ~2.8	Safety Rod ~5.6 Regurating Rod ~2.8	~9	T.B.D.
Max. Burn-up(pin av.)	MWd/t	25,000	42,000	75,000	T.B.D.
Operation Cycle		45 days Operation 15 days Outage	45 days Operation 15 days Outage	70 days Operation 23 days Outage	T.B.D.

Table 2 Main Parameters of MK-II and MK-III Driver Fuel

Items \ Core (Fuel)		MK-II		MK-III	
		J1	J2	Inner Core	Outer Core
Cladding Outer Diameter	(mm)	5.5	5.5	5.5	5.5
Cladding Inner Dia.	(mm)	4.8	4.8	4.8	4.8
Fuel Pellet Outer Dia.	(mm)	4.63	4.63	4.63	4.63
Fuel Pellet Form		Solid	Solid	Solid	Solid
Fuel Pellet Density	(%T.D.)	93	94	94	94
^{235}U Enrichment	(wt%)	~12	~18	~12	~20
Pu Fissile Content	(wt%)	22	20	~20	~20
Core Height	(cm)	55		48	48
No. of Fuel Pins	(#)	127		127	127
No. of Core Fuel Assemblies	(#)	~67		~25	~60

Table 3 Extension of Operation Period and Core Average Burn-up

Operation Cycle No.		12	13	14	15	16	17	18
Operation Days		45	55	60	70	32	70	70
Component of Fuel Subassemblies	J1	65	60	53	42	39	29	21
	J2	0	5	12	22	24	33	41



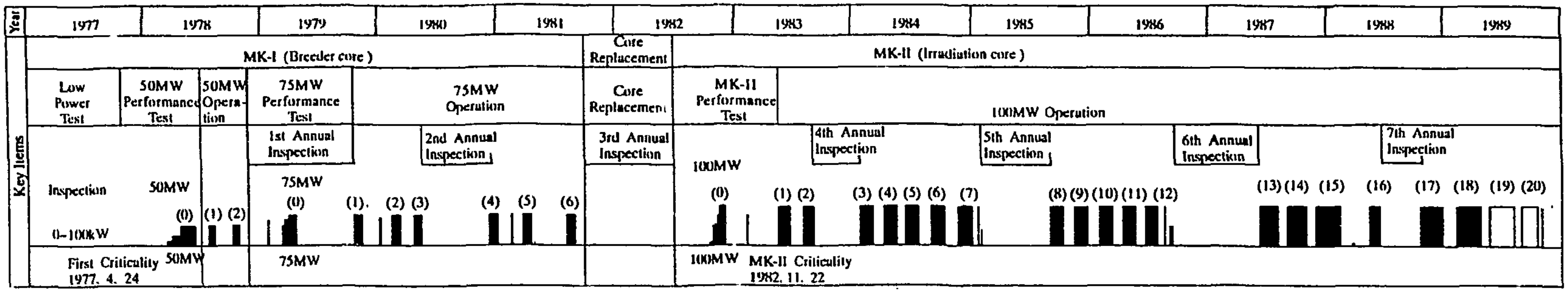
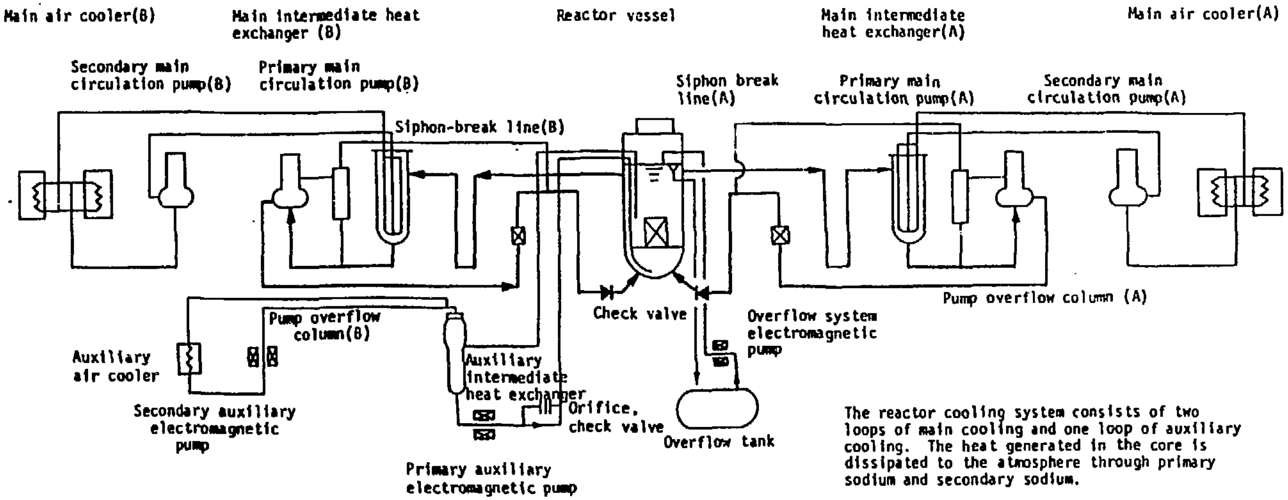


Fig. 1 Experimental Fast Reactor JOYO Operational History



The reactor cooling system consists of two loops of main cooling and one loop of auxiliary cooling. The heat generated in the core is dissipated to the atmosphere through primary sodium and secondary sodium.

Fig. 2 Reactor Cooling System

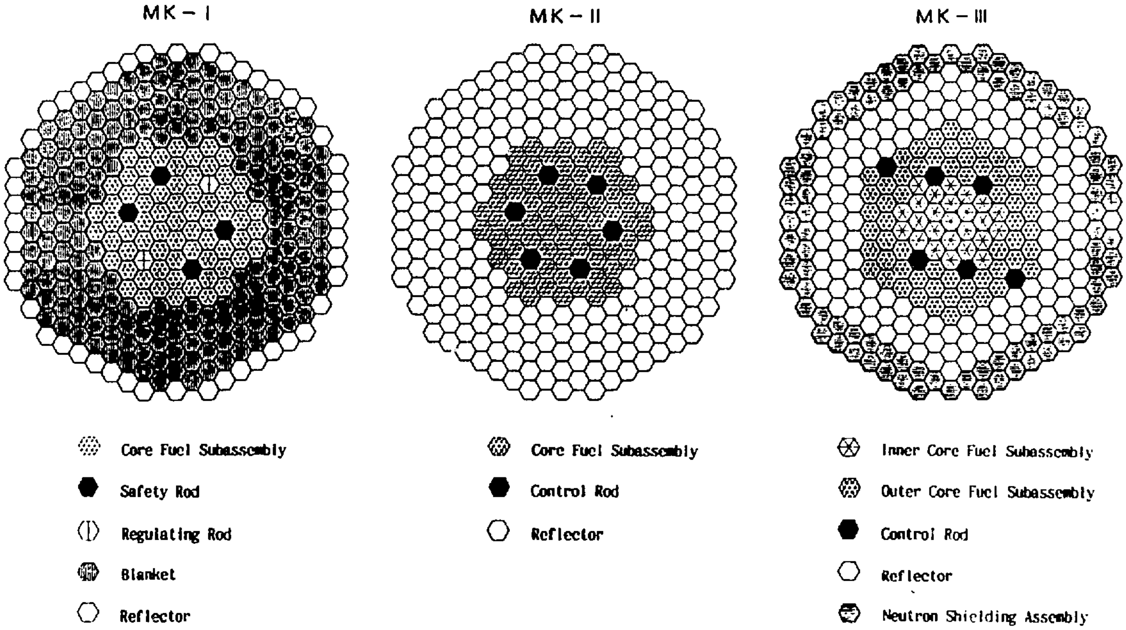


Fig. 3 Core Configuration

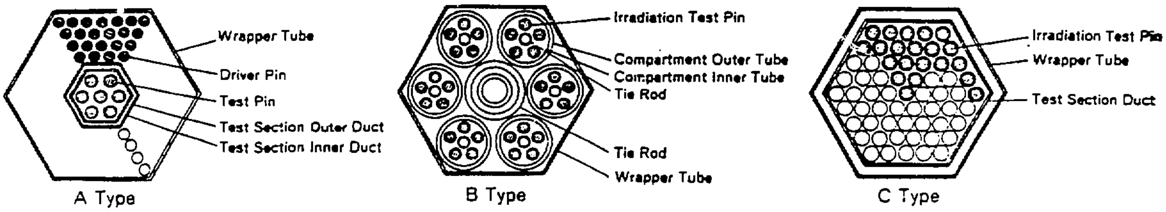


Fig. 4 Uninstrumented Irradiation Subassemblies

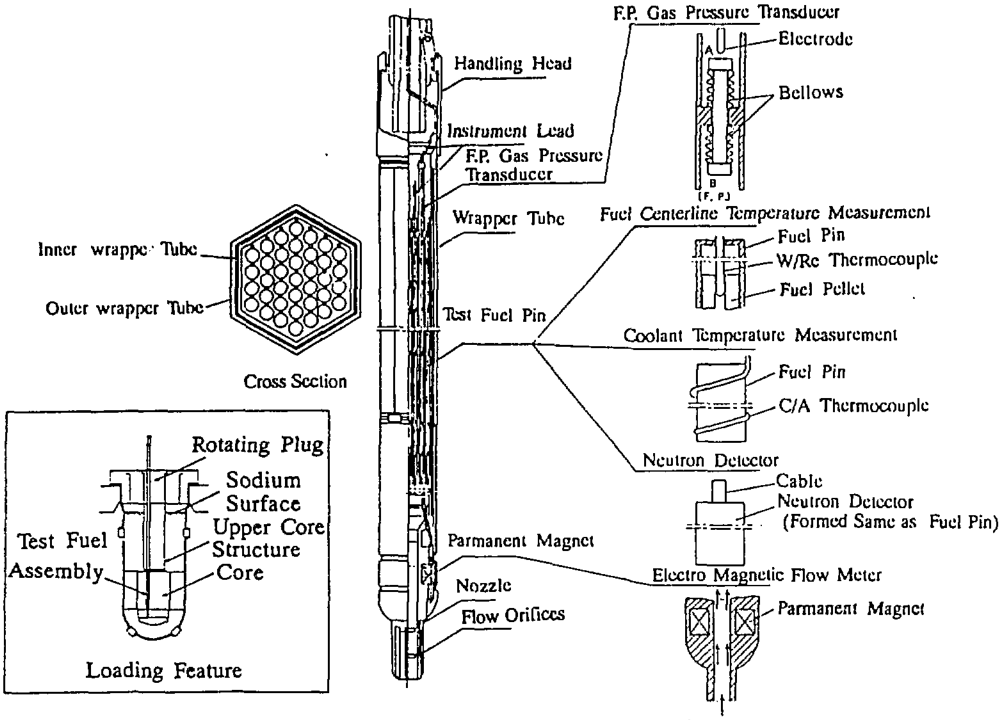
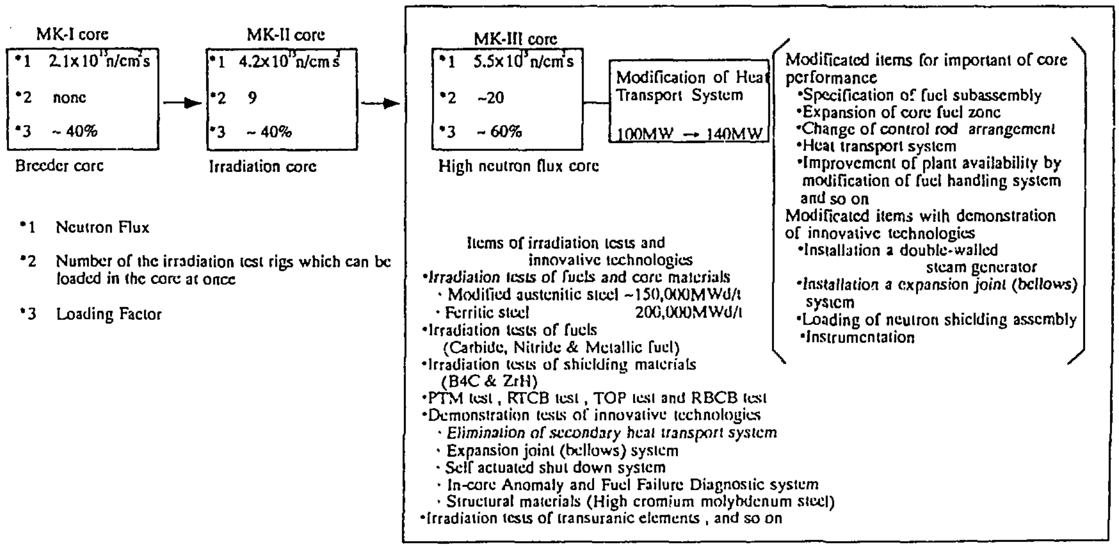


Fig. 5 The schematic diagram of Instrumented Test Assembly (INTA-S)


MK-I core

*1 2.1×10^{15} n/cm^{2}s
*2 none
*3 ~ 40%

Breeder core

MK-II core

*1 4.2×10^{15} n/cm^{2}s
*2 9
*3 ~ 40%

Irradiation core

MK-III core

*1 5.5×10^{15} n/cm^{2}s
*2 ~20
*3 ~ 60%

High neutron flux core

Modification of Heat Transport System

100MW → 140MW

Modificated items for important of core performance
- Specification of fuel subassembly
- Expansion of core fuel zone
- Change of control rod arrangement
- Heat transport system
- Improvement of plant availability by modification of fuel handling system and so on

Modificated items with demonstration of innovative technologies
- Installation a double-walled steam generator
- *Installation a expansion joint (bellows)* system
- Loading of neutron shielding assembly
- Instrumentation

Items of irradiation tests and innovative technologies
- *Irradiation tests of fuels and core materials*
  - Modified austenitic steel ~150,000MWd/t
  - Ferritic steel 200,000MWd/t
- Irradiation tests of fuels (Carbide, Nitride & Metallic fuel)
- Irradiation tests of shielding materials (B4C & ZrH)
- PTM test , RTCB test , TOP test and RBCB test
- Demonstration tests of innovative technologies
  - *Elimination of secondary heat transport system*
  - Expansion joint (bellows) system
  - Self actuated shut down system
  - In-core Anomaly and Fuel Failure Diagnostic system
  - Structural materials (High cromium molybdenum steel)
- Irradiation tests of transuranic elements , and so on

*1 Neutron Flux
*2 Number of the irradiation test rigs which can be loaded in the core at once
*3 Loading Factor

Fig. 6 Outline of MK-III program

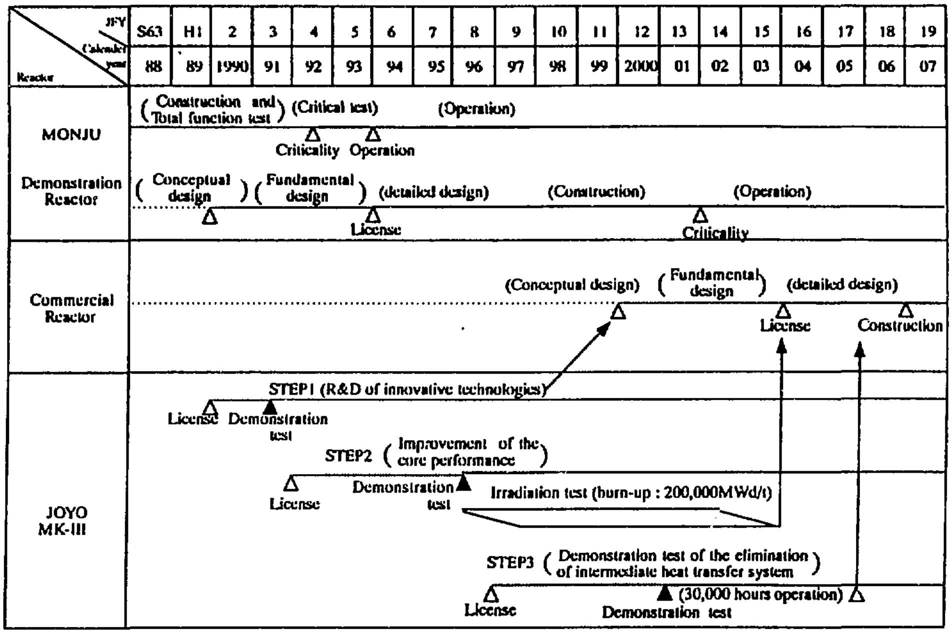


Fig. 7 Schedule of MK-III Program

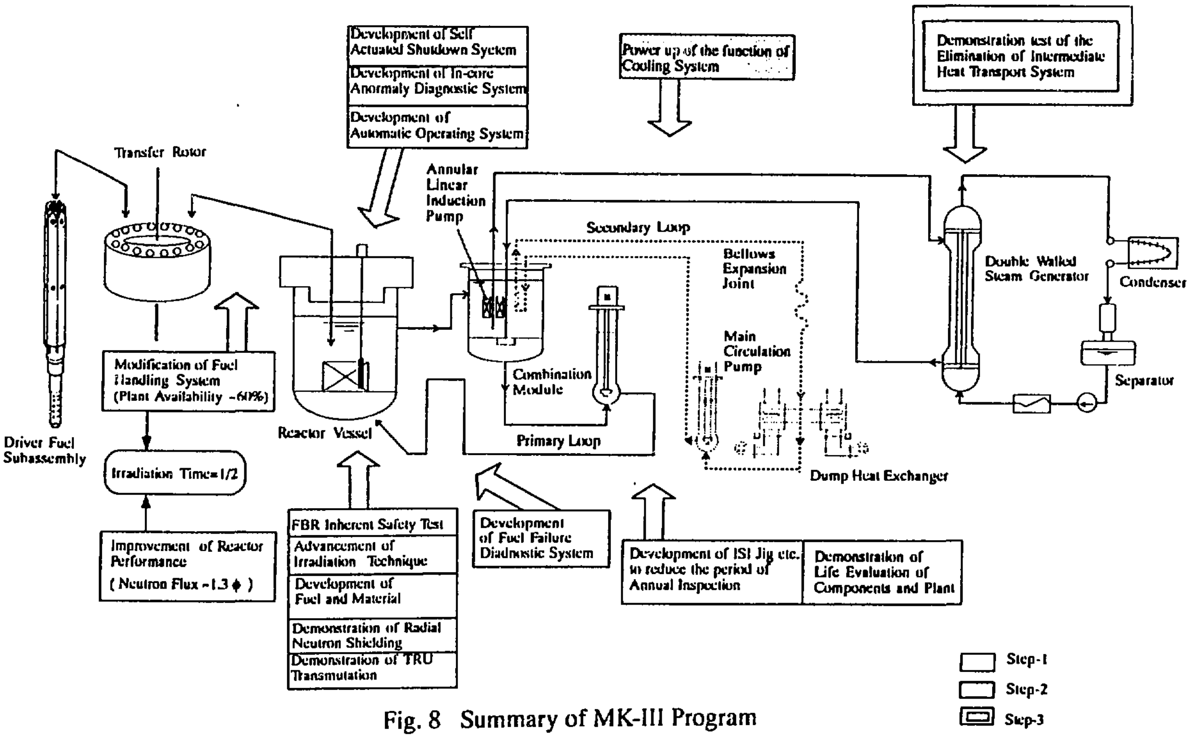
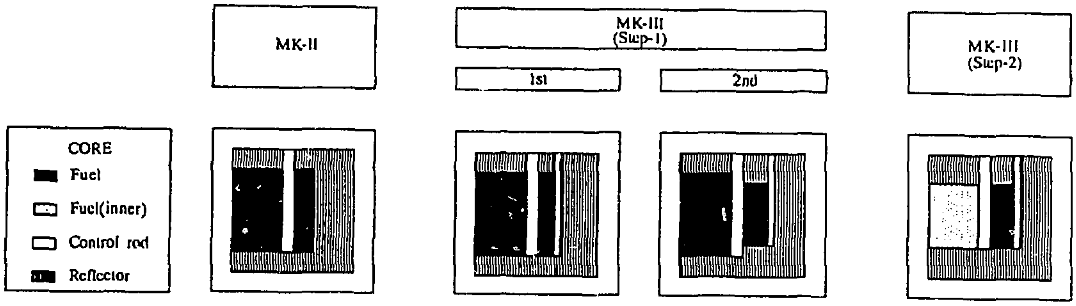
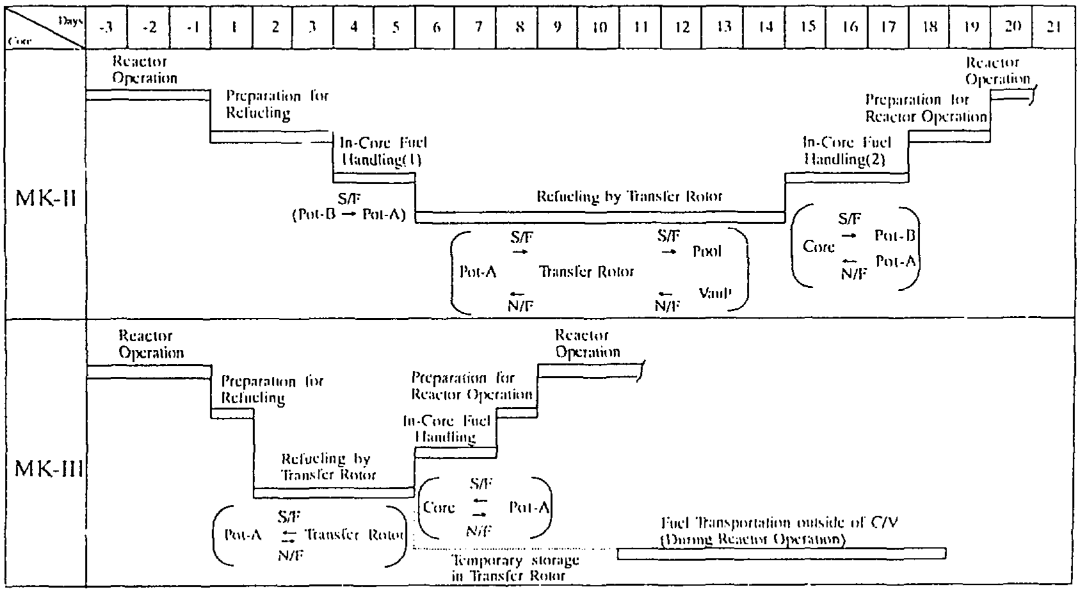


Fig. 8 Summary of MK-III Program



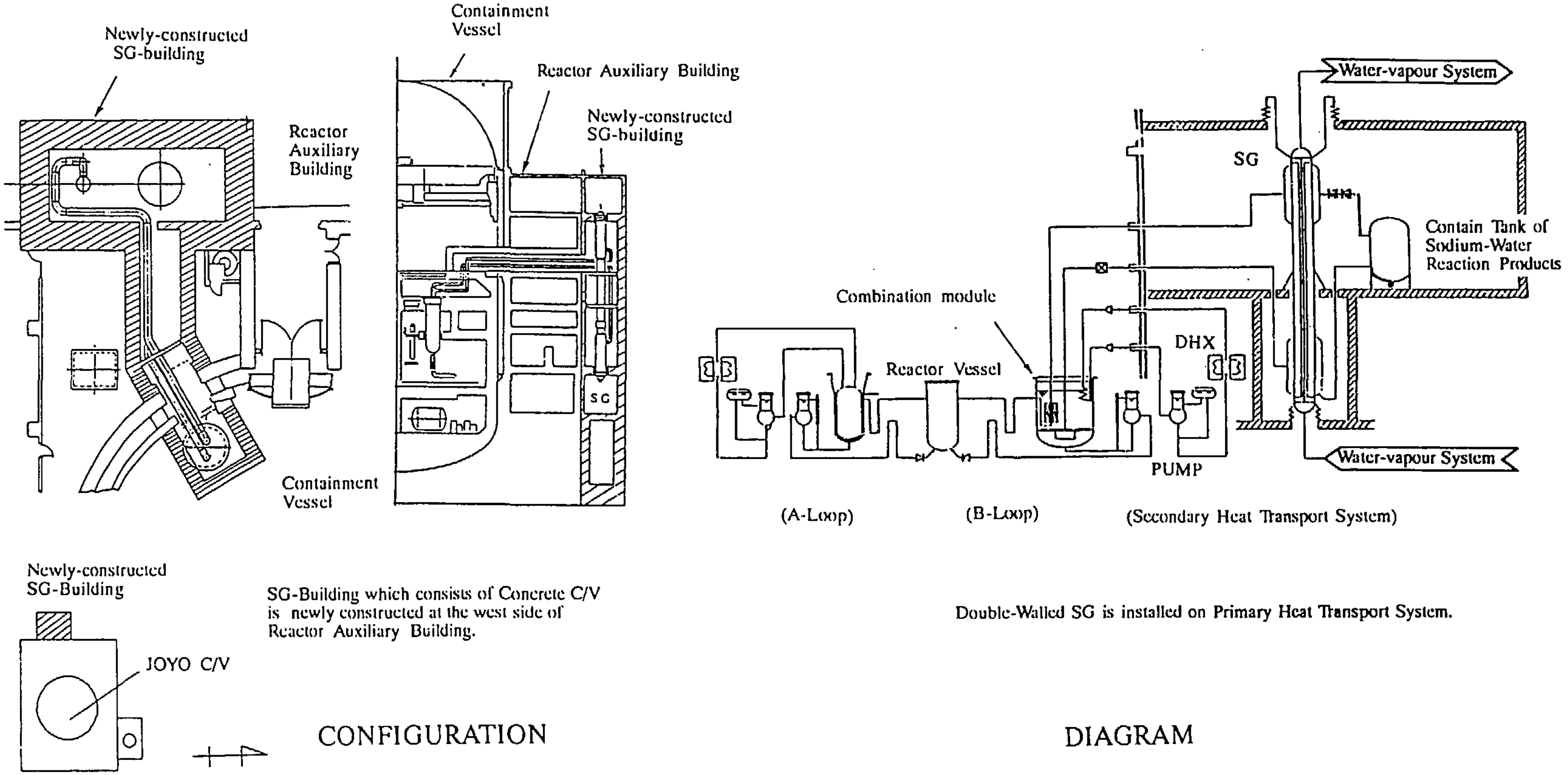
		MK-II	MK-III (Step-1) 1st	MK-III (Step-1) 2nd		MK-III (Step-2)	
Reactor Power(MWt)		100	100	~100		~140	
Neutron Flux		φ	~φ	~φ		~1.3φ	
Maximum Linear Heat Rate(W/cm)		400	~400	~400		(inner) ~500	(outer) ~390
Number of Fuels		67	~67	~85		~85	
Core Stack Length(cm)		55	55	(inner) 55	(outer) 48	48	
Fuel	Pin diameter (mm)	5.5	5.5	5.5	5.5	5.5	5.5
	Number of Pins	127	127	127	127	127	127
	Pellet Density (%T.D.)	94	94	94	94	94	94
Control Rod		6 at 3rd Row	5 at 3rd Row 1 at 5th Row	4 at 3rd Row 2 at 5th Row		4 at 3rd Row 2 at 5th Row	

Fig. 9 Modification Procedure of Core Configuration from MK-II to MK-III



Ref. N/S : New Fuel
S/F : Spent Fuel
Pot-A : Pot for Cooling
Pot-B : Pot for Transport

Fig. 10 Comparison of refueling time of MK-II and MK-III



SG-Building which consists of Concrete C/V is newly constructed at the west side of Reactor Auxiliary Building.

Double-Walled SG is installed on Primary Heat Transport System.

CONFIGURATION

DIAGRAM

Fig. 11 Layout of Secondary Sodium Loop Elimination System