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ABSTRACT

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Nuclear power generation in Japan is moving forward on the long-term development programme of nuclear power from the LWR to the FBR, essentially in the same way as in other advanced nuclear countries. In this development programme the unique HWR is also included; it can use plutonium produced in LWRs together with depleted uranium before the introduction of commercial FBRs. This report describes the status of the FBR and HWR development project being carried out by the Power Reactor and Nuclear Fuel Development Corporation (PNC) based upon the Long-Term Programme on Research, Development and Utilization of Nuclear Energy in Japan. Operational experience and technical results are shown for the experimental fast reactor JOYO (100 MW(th)), which reached initial criticality in 1977. The status of the 280 MW(e) prototype reactor MONJU, under construction as of 1982, is described. The conceptual design of the subsequent 1000 MW(e) demonstration plant is outlined, as is additional future planning. Research and development results, mainly carried out at Oarai Engineering Center of PNC, are shown. The 165 MW(e) prototype FUGEN is a heavy-water-moderated, boiling-light-water-cooled, pressure-tube-type reactor which uses plutonium mixed-oxide fuel. This report describes the relationship of the fuel cycle to the HWR in Japan and also discusses the operational experience of the prototype FUGEN, which has operated since 1979. Also described is the design of the 600 MW(e) demonstration plant and the programme of related research and development.

1. INTRODUCTION

In a statement made in 1980 at the Venice summit the development and introduction of petroleum substitute energy was mentioned as a common solution in advanced countries to the problem of the current world situation of unstable petroleum energy supplies. It was also agreed at the Ottawa summit in 1981 that this development should be pursued further. Because of the importance of nuclear energy as a petroleum substitute energy source, in a cabinet decision made in Japan on target supplies of the petroleum substitute energies in November 1980 the target for nuclear power was set at 51 - 53 GW in 1990, which ranks second to coal.

In the studies of the future policy of positive development of nuclear power generation as the nucleus of petroleum substitute energy, the following developments occurred from 1980 to 1981: revision of the long-range programme of atomic energy research, development and utilization by the Japan Atomic Energy Commission (AEC); check and review on the demonstration advanced thermal reactor (ATR) by the Subcommittee on Evaluation and Study of the Demonstration ATR; discussion by the Advisory Committee for Energy. (The target for nuclear power was revised to 46 GW by the Advisory Committee for Energy in June 1982.)

Japan is highly dependent on foreign countries for nuclear fuel resources, so to promote the effective use of nuclear fuel and thereby increase energy security, it is important to promote the development of advanced power reactors which have higher utilization efficiency of nuclear fuel than the existing LWRs [1].

2. DEVELOPMENT OF THE FAST BREEDER REACTOR

2.1 Experimental fast breeder reactor JOYO

In the Power Reactor and Nuclear Fuel Development Corporation (PNC) Oarai Engineering Center, the experimental fast reactor JOYO started cycle operation with the MK-I core at 50 MW(th) in September 1978. Subsequently JOYO began the second phase of cycle operation with the MK-I core at 75 MW(th) in February 1980. Six cycles operating at 75 MW(th) were completed in December 1981. The JOYO master schedule is shown in Fig. 1 [2].

During cycle operation with the MK-I core, many tests and measurements (such as thermal-hydraulic tests, physics tests, and shielding tests including corrosion product measurement and plant parameter measurements) were conducted to clarify and to confirm the characteristics of the JOYO plant. During regular inspection in this period, over 100 inspections and examinations (such as the rotating plug, control rod drive mechanism and fuel loading/unloading machine) were conducted.

Experience in operation, inspection and maintenance and the data obtained through cycle operation have been used to secure the safety of JOYO itself. Moreover, these results are also reflected in the design of the prototype fast breeder reactor (FBR) MONJU, its analysis and evaluation, and planning for its construction and operation. Table I shows the operational experience of JOYO [3].

Now JOYO is shifting to the irradiation of the MK-II core at a power level of 100 MW(th) for irradiation of fast reactor fuels and structural materials, which is the second stage of the programme. (The MK-II core is scheduled to achieve criticality in November 1982 and to reach the 100 MW(th) power level in the spring of 1983.) The main parameters of the JOYO MK-II core are shown in Table II. The irradiation rig assembling facility adjacent to JOYO was completed in April 1981. With this facility, assembly and inspection of special irradiated fuel assemblies and irradiated rigs are performed. Metallographic tests and material property tests of fuels irradiated in JOYO and other reactors are being performed at the irradiated fuel testing facility in the Oarai Engineering Center. These tests have confirmed the integrity of fuels used in JOYO and are also providing the data necessary for safety analysis of fuels.

2.2 Prototype fast breeder reactor MONJU

For the prototype FBR MONJU [4] (electric power approximately 280 MW) detailed design work is under way to meet the safety review requirements of the Nuclear Safety Committee. This review will include the detailed design of MONJU safety features incorporated as a result of the Three Mile Island accident in the USA.

At the expected site of construction, i.e. Shiraki area, Tsuruga City, Fukui Prefecture, investigation of the environment is continuing. A Report of the Environmental Review for the Prototype FBR has been prepared by the Agency of Natural Resources and Energy. Other concerned government ministries and agencies are reviewing this document.

Preparatory construction work such as improvement of the existing prefectural roads, new construction of access roads and ground preparation of the site is being planned and designed. With the consent of the local government, PNC presented the formal application for permission for construction to the Prime Minister on 10 December 1980. The first safety review of this application by the Nuclear Safety Bureau of the Science and Technology Agency was completed in December 1981.

Estimates for the components and machinery of the MONJU power plant have been requested of Hitachi Ltd., Toshiba Corp., Mitsubishi Heavy Industries Ltd., and Fuji Electric Co., Ltd. Negotiations leading to contract placement are proceeding. Co-operation with the electric power industry is provided through the Japan Atomic Power Company. Table III shows principal design and performance data of MONJU.

2.3 Demonstration fast breeder reactor

On the basis of accumulated technology results and related research and development through design work for JOYO and MONJU, PNC has been proceeding with the design study of a large FBR since the fiscal year 1977 and the power-supplying utility companies have been developing a conceptual design of a demonstration FBR since the fiscal year 1978. These studies have been conducted in mutual co-operation, taking advantage of experience agained.

Both PNC and the power-supplying utility companies are aiming at the development of a loop-type, sodium-cooled FBR with an electric output level of 1000 MW(e) on a commercial scale. Studies are thus being made of several candidate FBR concepts, incorporating fundamental ideas in areas such as optimization of engineering, improvement in safety and reliability and increase in plant factor.

The power-supplying utility companies are making a conceptual design study of the loop-type FBR in principle but at the same time are also investigating the tank-type FBR, particularly the seismic design of components. This approach will result in flexibility in future selection of a particular design type.

The programme for the demonstration FBR (DFBR) in Japan is shown in Fig. 2. Recently, the DFBR co-ordinating committee was established by PNC and the Federation of Electric Power Companies to co-ordinate the various concepts and specifications of the DFBR and to promote the preparatory activities for starting its construction.

2.4 Research and development

Research and development of the FBR are being carried out principally at the Oarai Engineering Center with the co-operation of the Japan Atomic Energy Research Institute (JAERI), universities, manufacturers and research institutes. International co-operation is also active. At present, research and development emphasis is placed on the development of MONJU.

In the field of physics, a partial mock-up experiment on the MONJU core with the fast critical assembly (FCA) was performed by JAERI to confirm the design method for MONJU.

For the purpose of shielding calculations for cavities such as the reactor vessel room with its complex equipment arrangement, shielding research was performed and a cavity penetration calculation code was developed.

In functional tests for MONJU components, mock-up tests of main components such as a refuelling handling machine, a control rod drive mechanism, and a primary mechanical pump were conducted. The test results have been effectively used and will aid detailed design in the future.

For sodium component structural materials, a corrosion test for small sodium leakage from the inlet pipe of the reactor vessel was made. From the results, data could be obtained on corrosion phenomena in piping due to corrosion products from leaking sodium (with temperature, atmosphere, etc. as parameters).

In sodium handling technology, research on decontamination of fast reactor structural materials was done to clarify the corrosion behaviour and the decontamination conditions in 304 stainless steel.

In the development of fuel pins, a titanium coating test was made for the purpose of reducing inner surface corrosion; the results were excellent.

Work on fuel irradiation has included a post-irradiation test of the fuel irradiated in the Rapsodie reactor in France, a post-irradiation test of fuel melting in co-operation with the Department of Energy (DOE) in the United States of America and an irradiation test in the Japan Material Test Reactor (JMTR) and JOYO.

For cladding materials, a creep test was made on a 316 stainless steel cladding tube prepared by adjustment in micro-constituents such as titanium within the standards. The results confirmed that creep strength could be substantially increased as compared with ordinary 316 stainless steel.

In research and development in the high-temperature structural design area, analysis methods important for design analysis of a fast reactor operated at high temperature such as the MONJU and studies on the application of inelastic analysis are continuing.

In tests of structural material, the work performed includes mechanical strength tests and sodium compatibility tests for base metal, welded joint and forged material of stainless steel and low-alloy steel. The data obtained are reflected in the guide for design and manufacturing standards.

Seismic structure tests involving vibration of 29 fuel assemblies in series were made using full-size mock-up fuel assemblies. The results verified the analysis method used for MONJU fuel assemblies under earthquake conditions. Also, a seismic test was conducted using a 1/4 scale plastic model of the reactor vessel; the results showed the adequacy of the mathematical model for analysis.

To study failure behaviour of fuel pins and the behaviour after failure at the time of a hypothetical accident, an in-reactor reactivity accident simulation test with a single fuel pin is being conducted in the CABRI reactor in France in a joint experiment by Japan, France, the Federal Republic of Germany, the United Kingdom and the United States. Also, a large-scale, in-reactor safety test to examine a localized accident and the fuel behaviour afterwards using 7 - 19 pin bundles has been conducted with the ETR and TREAT reactors in the United States under co-operative agreements between Japan and the United States.

In the development of a prototype steam generator (SG), performance tests using prototype generators (No. 1 SG and No. 2 SG) were carried out. After the 3400 hour performance tests of No. 1 SG were finished in April 1975, it was disassembled for inspection. The No. 2 SG was constructed and the performance tests began in January 1976 (the accumulated operating times of No. 2 SG are 9500 hours for the evaporator and 3300 hours for the superheater). Through evaluation of experimental data and comparison with analysis, the heat transfer and flow dynamics design method for MONJU steam generators was completed.

Research and development for solving problems with materials, design and fabrication, operation and control, water leak detection and maintenance and repair were conducted and good results were obtained. A practical feasibility test of these research and development works was also executed using the No. 2 SG evaporator under the assumption of the occurrence of a water leak. No significant deterioration was observed in No. 1 or No. 2 SG.

After the feasibility test was completed in October 1980, an endurance test of No. 2 SG was started again with a target of another 10,000 hours of operation.

3. FUGEN HWR

FUGEN is a 165 MW(e) prototype of a heavy-water-moderated, boiling-light-water-cooled reactor. This project was begun in October 1967 with the support of the government, utilities, manufacturers, and research organizations. Much research and development work was carried out and FUGEN has been in commercial operation since 20 March 1979, generating over 2 million MW(e)·h of electric power.

In addition, design work on a 600 MW(e) FUGEN HWR demonstration plant has been under way since 1973, and associated development work is being performed to support the design.

The check and review of the 600 MW(e) demonstration plant programme was made by an ad hoc committee organized by the Japan AEC. On 4 August 1981, the ad hoc committee submitted its report to the AEC after having held discussions and evalutions for 16 months. The committee recommended the construction of a 600 MW(e) FUGEN HWR demonstration plant with appropriate support of the government.

3.1 Role of the FUGEN HWR in Japan [5]

The FUGEN HWR possesses excellent specific characteristics for using plutonium:

(1) Plutonium mixed-oxide fuel could be loaded in the full core, and the amount of natural uranium required would be only 1/2 to 1/3 that of LWRs.

(2) If, as expected, the nuclear characteristics are not greatly affected by the isotopic composition of the plutonium, then the FUGEN HWR could be easily operated with little consideration other than keeping the fissile content (^{235}U and fissile plutonium) of fuels constant.

(3) Plutonium could make the coolant void reactivity more negative, which would increase reactor stability and

safety.

In Japan 24 LWRs, 17 GW(e) in total, are now in operation, and mainly LWRs will be built in the near future. Plutonium will then be stored in the form of LWR spent fuel or plutonium extracted by reprocessing.

Under these conditions, the FUGEN HWR coupled with LWRs is considered in Japan to improve our national energy security by using plutonium and depleted uranium extracted from spent fuels of LWRs and thereby reducing the demands for natural uranium and uranium enrichment work. At the same time, the FUGEN HWR could adjust the plutonium stock by selecting fuel, plutonium or uranium, depending upon conditions. Such fuel utilization (shown in Fig. 3) will boost the establishment of a plutonium utilization technology.

As the spent fuel of the FUGEN HWR may contain less than 0.2% of ^{235}U, uranium recycling would not be considered, which could reduce the problems caused by the accumulation of ^{236}U that may be encountered in uranium recycling.

3.2 Operating experience of FUGEN [6]

FUGEN is designed to generate 165 MW(e); its main design data are shown in Table IV.

Some problems were encountered during commissioning and operation, but except for stress corrosion cracking in the stainless steel pipes they were minor and have been overcome.

Stress corrosion cracking was found in the stainless steel pipes of the residual heat removal system and emergency core cooling system during the scheduled shutdown in November 1980. Replacement of the 304 stainless steel pipes with 316L stainless steel was completed in August 1981.

FUGEN is now in full-power operation, giving good reactor performance and characteristics.

One hundred and sixty-eight mixed-oxide fuel (MOX) assemblies were loaded and 92 have been withdrawn. The maximum burnup of MOX fuel assemblies reached 11,400 MW·d/t by the end of March 1982, and none has yet ruptured.

In the early days of the FUGEN project, full-scale development facilities were built at the Oarai Engineering Center - the deuterium critical assembly (DCA), the 14 MW heat transfer loop (HTL) and a full-scale reactor safety development facility. Tests have been carried out to support reactor safety, core design and reactor operation and to verify the endurance of fuel assemblies and pressure tube assemblies.

In parallel with such work, manufacturers have carried out in full-scale mock-up tests the development of major components such as the refuelling machine, the reactor proper, the pressure tube assemblies, the fuel assemblies, etc.

Much work has been performed on the development of MOX fuel assemblies, both their integrity and fabrication. Irradiation tests of MOX fuel assemblies were done in the SGHWR and Halden facilities.

3.3 600 MW(e) demonstration plant

The 600 MW(e) demonstration plant is designed mainly to use MOX fuel, considering the role of the FUGEN HWR [7]. The design principles of FUGEN are being followed for the demonstration plant design in such important systems and components as the emergency core cooling system, pressure tube assembly, and control rod mechanism. Some modifications, however, have been made as a result of experience obtained in FUGEN and LWRs (see Fig. 4). Some examples are as follows.

(1) Plutonium content in MOX fuel is designed to give almost the same burnup as that of HWRs.

(2) A 36 rod cluster is to be used instead of a 28 rod cluster.

(3) The heavy-water dump system is eliminated and the liquid-poison injection system is adopted, thereby reducing the diameter of the calandria.

4. CONCLUSIONS

The PNC has proceeded with the development of advanced power reactors in co-operation with the power-supplying utility companies, manufacturers and research organizations. Advanced power reactors should form the core of nuclear power generation, which plays an important role in Japan because of the current energy situation.

The experimental fast reactor JOYO has successfully achieved its first stage development objectives, terminating its final cycle of operation at the end of 1981 at 75 MW(th) with a breeding core (MK-I). It is moving towards the second stage of the irradiation programme (MK-II) with a rated power of 100 MW(th). The development of the prototype FBR MONJU has gone foward, and the safety licensing examination to obtain the permission for MONJU construction and to obtain the consent of the local government for construction of the prototype FBR MONJU is proceeding. Conceptual designs of the demonstration FBR are being conducted.

The prototype ATR FUGEN has achieved adequate results in its design, construction and operation with the generation of 2 million MW(e)·h of electricity. Accordingly, the technology of the heavy-water-moderated, boiling-light-water-cooled reactors is being demonstrated.

In the development of the demonstration ATR, design work has been in progress since 1973 and associated development work is now in progress. The check and review of the programme was made by an ad hoc committee organized by the Japan AEC. The committee submitted its report to the Japan AEC last August; it was recommended to build the 600 MW(e) demonstration ATR plant.

The PNC is making strenuous efforts not only in the power reactor development area but also in the establishment of the nuclear fuel cycle. These efforts are based on Japan's own technology and involve international co-operation.

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TABLE I. OPERATIONAL EXPERIENCE OF JOYO (MK-I CORE)

Accumulated operation time	12,968 h
Accumulated heat generation	673,330 MW·h
Maximum fuel burnup achieved	40,500 MW·d/t
Number of startups	260 (including critical test)
Number of fuel assemblies handled	≈ 150
Number of annual inspections	2

TABLE II. MAIN PARAMETERS OF JOYO (MK-II CORE)

Reactor output	100 MW(th)
Primary coolant flow rate	2200 t/h
Reactor inlet temperature	370°C
Reactor outlet temperature	500°C
Number of core fuel assemblies (including fuel test rig)	67
Type of irradiation rig	
Fuel test: non-instrumented	3
instrumented	1
Structural material test	1
Irradiation test conditions	
Linear heat rate	about 600 W/cm
Burnup	about 130,000 MW·d/t
Cladding temperature	about 700°C
Maximum number of fuel irradiation rigs	9
Maximum neutron flux (>0.1 MeV)	3.7×10^5 n/cm^2·s

TABLE III. PRINCIPAL DESIGN AND PERFORMANCE DATA OF MONJU

Reactor type	Sodium-cooled loop-type
Thermal power	714 MW
Electrical power	about 280 MW
Fuel material	PuO_2, UO_2
Average burnup of discharged fuel	80,000 MW·d/t
Cladding material	316 stainless steel
Cladding outside diameter/thickness	6.5/0.47 mm
Permissible cladding temperature	675°C
Breeding ratio (initial/equilibrium)	1.20/1.21
Reactor in/out sodium temperature	397/529°C
Secondary sodium temperature (IHX outlet/IHX inlet)	505/325°C
Steam/water temperature	483/240°C
Number of loops	3
Pump position (primary and secondary loop)	Cold leg
Type of steam generator	Helical coil once-through unit type
Refuelling system	Single rotating plug with fixed arm FHM

TABLE IV. DESIGN DATA OF FUGEN AND THE DEMONSTRATION PLANT

	FUGEN	Demonstration plant
Electric power (MW)	165	600
Thermal power (MW)	557	1,930
Reactor		
Core diameter (mm)	4,053	6,951
Core height (mm)	3,700	3,700
No. of channels	224	648
Calandria diameter (mm)	7,950	9,750
Pressure tube		
Material	Zr-Nb	Zr-Nb
Inside diameter (mm)	117.8	117.8
Fuel		
No. of rods/assembly	28	36
Pellet diameter (mm)	14.4	12.4
Enrichment (Pu_f + ^{235}U)	2.0	2.7
Burnup (average)	17,000	27,000
Control rods		
B_4C	49	76
Stainless steel	0	17
Primary cooling system		
No. of loops	2	2
Recirculation flow (t/h)	7,600	22,600
Steam flow (t/h)	910	3,300
Steam drum pressure ($kg/cm^2(g)$)	68	69

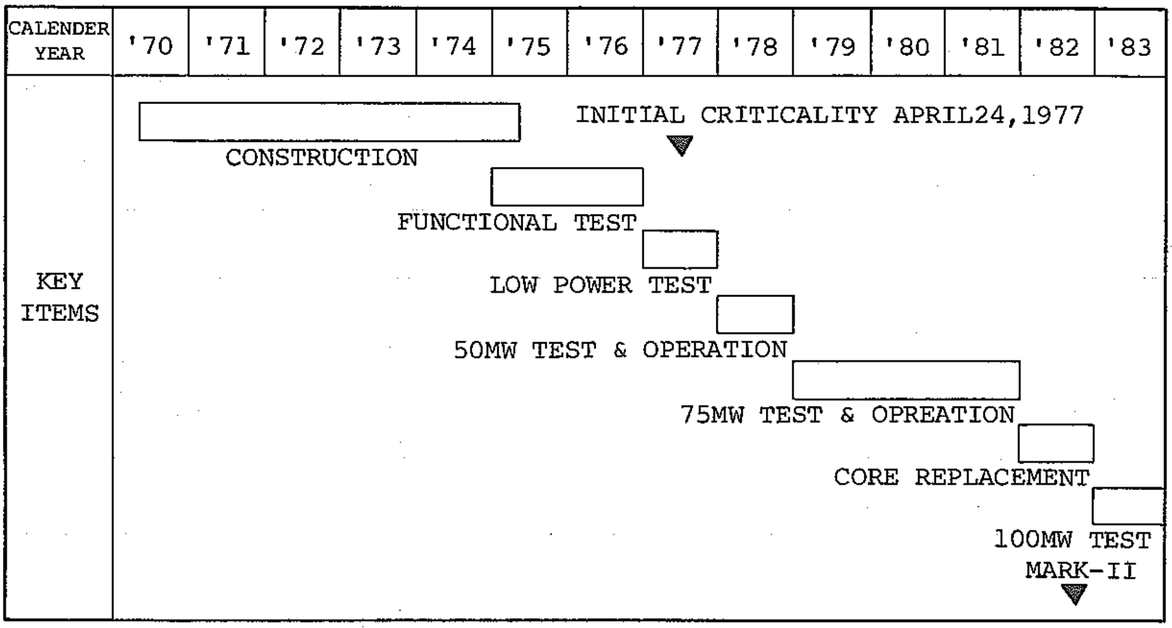

Fig. 1 Master schedule of JOYO

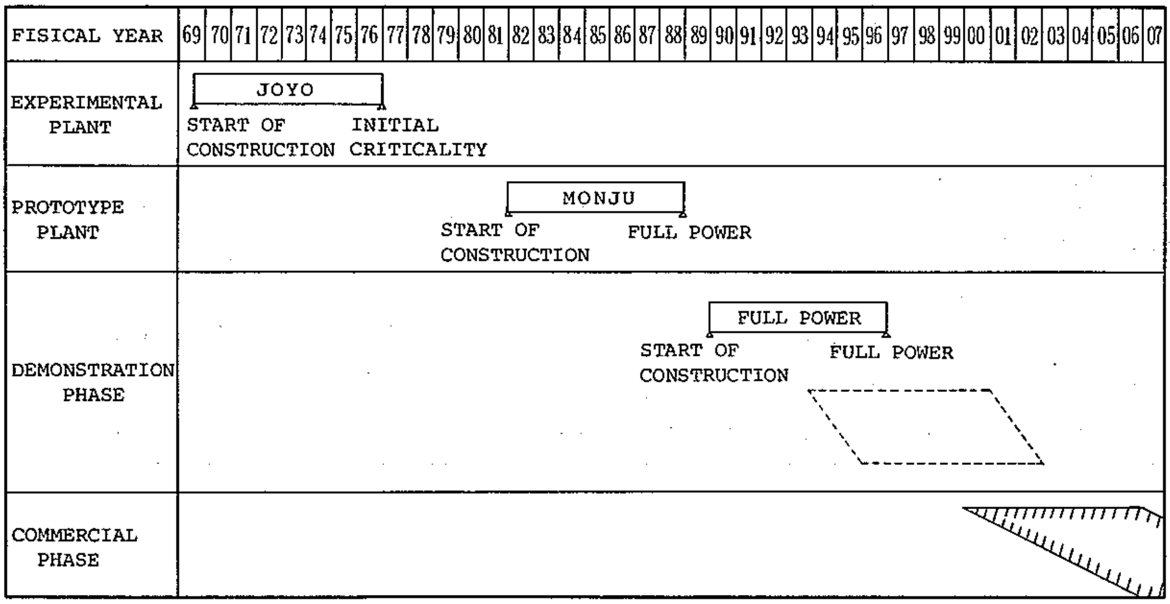

Fig. 2 Fast breeder reactor programme in Japan

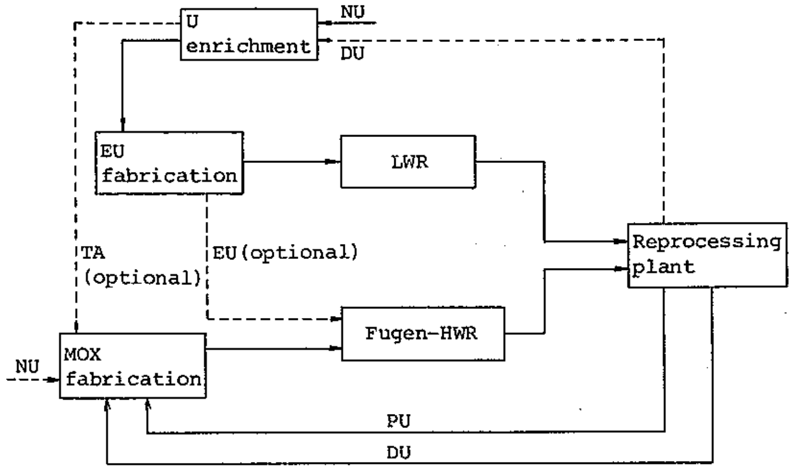

Fig. 3 Fuel utilization in FUGEN HWR

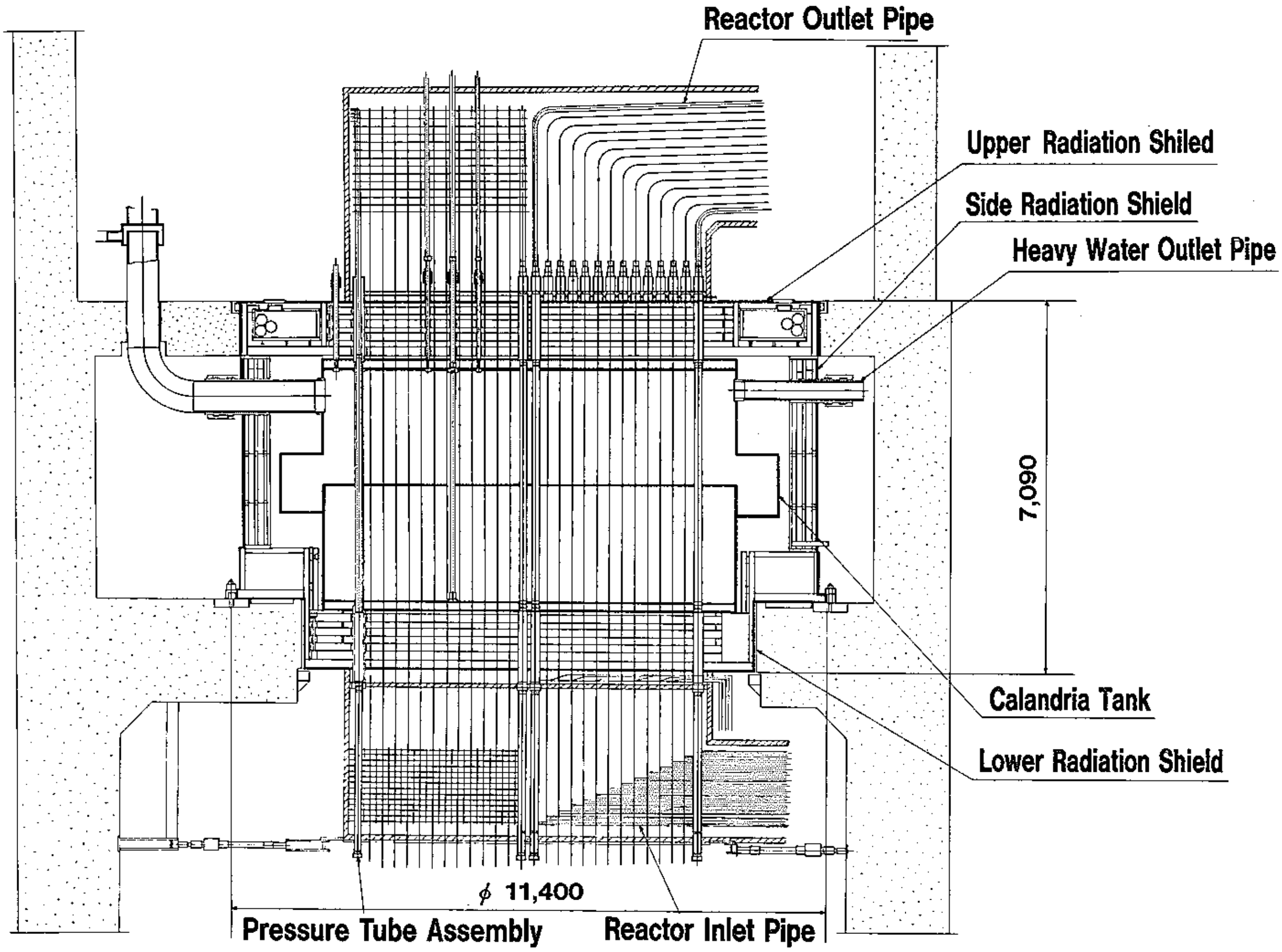

Fig. 4 600 MW(e) demonstration plant