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# RESEARCH AND DEVELOPMENT OF SODIUM COMPONENTS AND SODIUM TEST RIGS FOR FAST BREEDER REACTORS IN FUJI ELECTRIC CO., LTD.

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## I. INTRODUCTION

Sodium cooled fast breeder reactors look like most promising in future reactors because the improved economy of nuclear power generation and the effective utilization of fertile material. However, they have many problems to which proven nuclear power technology could not give an immediate answer, such as high reliable operation of the plant equipment in liquid sodium and/or sodium vapor, reactor safety, sodium/water reaction, and so on.

Under such conditions, research and development is now being prosecuted aggressively on a national scale in Europe and America. In Japan also, design and development is progressing in the fields of experimental fast reactors and prototype fast reactors as a national project under Power Reactor and Nuclear Fuel Development Corporation. JEFR called " JOYO " (thermal output : 100 MW) is expected to reach criticality in 1973 and JPFR called " Monju " (electrical output : 300 MW) in 1978.

In 1967, Fuji Electric started its activities on sodium reactor technology, and research and development of sodium equipment was begun under the auspices of the Government. Since that time, research and development has been actively performed on various devices and already three sodium test facilities have been built in the company. These three test facilties should be suitable after some future improvements for research and development of models to be performed by individual makers. In this article, an outline of the facilities will be given and the status and future plans of the development will be reported.

## II. SODIUM TEST FACILITY

The sodium test facilities elected in Fuji Electric can be divided into two types : that which is suitable for tests with little flow rate, and that adequate for tests with large flow rate of the sodium. The facilities for testing small parts and for testing the fuel handling grippers belong to the former type and that for testing and development of a fuel channel failure detector belongs to the latter. Since the above mentioned three types of facilities are referred to a " mini-loop ", " 1 in. loop " and " 3 in. loop " respectively in Fuji Electric, these names will be used here for convenience. All three facilities has been operating with no troubles.

### 1. Mini-Loop

The mini-loop is intended mainly for testing in high temperature liquid sodium and in atmospheres containing sodium vapor, the operation of part of mechanisms such as the control rod drive mechanism and fuel handling machine. The design, construction and operation of the mini-loop provided in itself a wide experience applicable to the design and manufacturing other sodium systems and component. Since the mini-loop was completed in July, 1968, several minor changes were added and the mini-loop reached its present states. A flow sheet of the present mini-loop is shown in *Fig. 1* and general view is given in *Fig. 2*.

Since the mini-loop was started, it has already operated for more than 3,000 hours with sodium in circulation. After the tests mentioned above were completed, tests were performed on the characteristics of an AC Faraday type electromagnetic pump, calibration of a small permanent magnet flowmeter, and the bearings of a mechanical pump.

The main specifications of the mini-loop are given in *Table 1*. Other points of interest about this equipment include the DC Faraday type circulation pump, the overflow tank for a test vessel and axial tempreture distribution control by sleeve temperature control.

### 2. One in. Loop

Considerable experience and basic sodium technology were obtained from the development of the mini-loop, but it becomes necessary to develop and prove sodium equipments which could be used in the experimental and prototype fast reactors. For this reason, larger test facility was needed since the mini-loop was not sufficient. The main aim of this facility was research and development of the fuel handling grippers for the experimental fast reactor. Experience obtained in operation of the mini-loop was re-

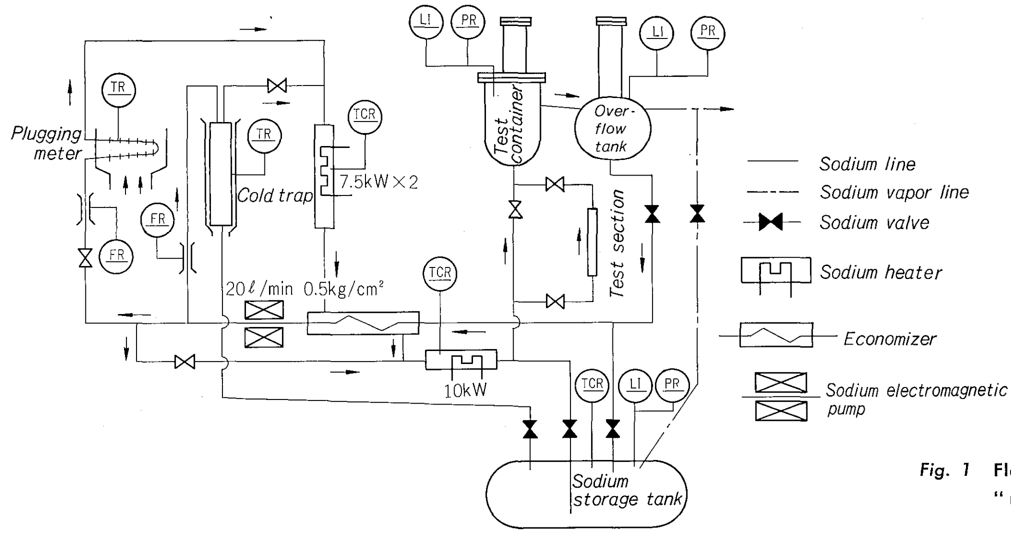


*Fig. 1* **Flow sheet of " mini-loop "**

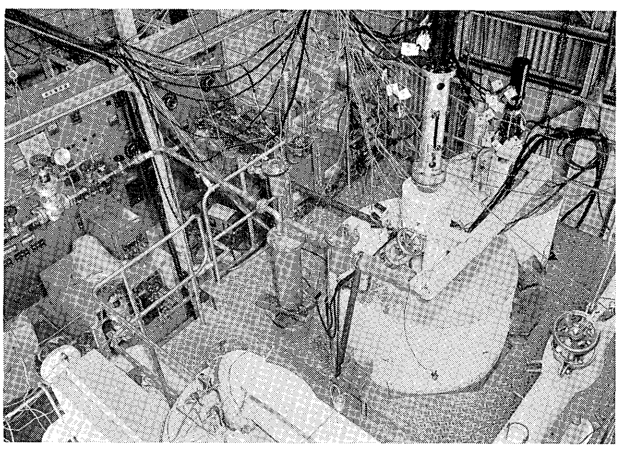

*Fig. 2* General wiew of " mini-loop "

***Table 1*** **Main specifications of " mini-loop "**

| | |
|---|---|
| Test container sodium temperature | Maximum 600°C |
| Sodium flowrate | Maximum 50 $l$/min |
| Piping dimension | Nominal diameter 1 in. |
| Amount of sodium | (Total amount stored) 300 kg |
| | (Amount in circulation) about 200 kg |
| Total capacity of heaters | about 100 kW |
| Sodium purification capacity | about 10 ppm |
| Cover gas | High purity (99.994%) Ar gas |

flected in the design and manufacture of this 1 in. loop, as well as the 3 in. loop, in such points as the addition of safety devices such as various types of interlocks and protective relays, selection, and terminal insulation design of preheater and installation methods.

After completion of the 1 in. line in April, 1970, tests were conducted on the reliability of the fuel and pot grippers, and pot cooling. The 1 in. loop has now operated for more than 2,500 hours. The flow sheet is shown in the right half of *Fig. 3* and a general view in *Fig. 4*.

Except for the size, the 1 in. loop is similar to the mini-loop in construction. Differences include the vertical cylindrical type test container with dimensions of 765 mm inner dia. × 2,500 mm, the maximum liquid level being held to about 2 m not by the overflow tank system but by overflow into the dump tank, and a change from series to parallel connections between the test containers and sodium purification line for the electromagnetic pumps.

This facility has two main features: 1) the 1 in. loop has been made into a single function in order to avoid handling difficulties and drops in test efficiency (the test for sodium flow requirements are performed in the 3 in. loop), and 2) there is a sufficient gap (approx. 4.5 m) above the top part of the test container so that there is little loss due to size of the scale factor and tests on relatively long objects are possible. The main specifications of the 1 in. loop are given in *Table 2*.

## 3. Three in. Loop

The 3 in. loop was developed for the development and testing of fuel channel failure detection and location system for fast reactors. It was commissioned by Power Reactor and Nuclear Fuel Development Corporation and was completed at the end of 1970. The major differences between this loop and the two loops described previously are as follows:

1) The amount of sodium in circulation is larger.
2) Highly accurate calibration of the flowmeter is possible by the volumetric method and the bootstrap method.
3) For detection tests of failures in the fuel channels, seven fuel subassemblies at almost the same pitch as in the actual reactor can be inserted in the top part of the model in the test container.
4) The sodium flowrates to the central fuel subassembly and the six peripheral subassemblies can

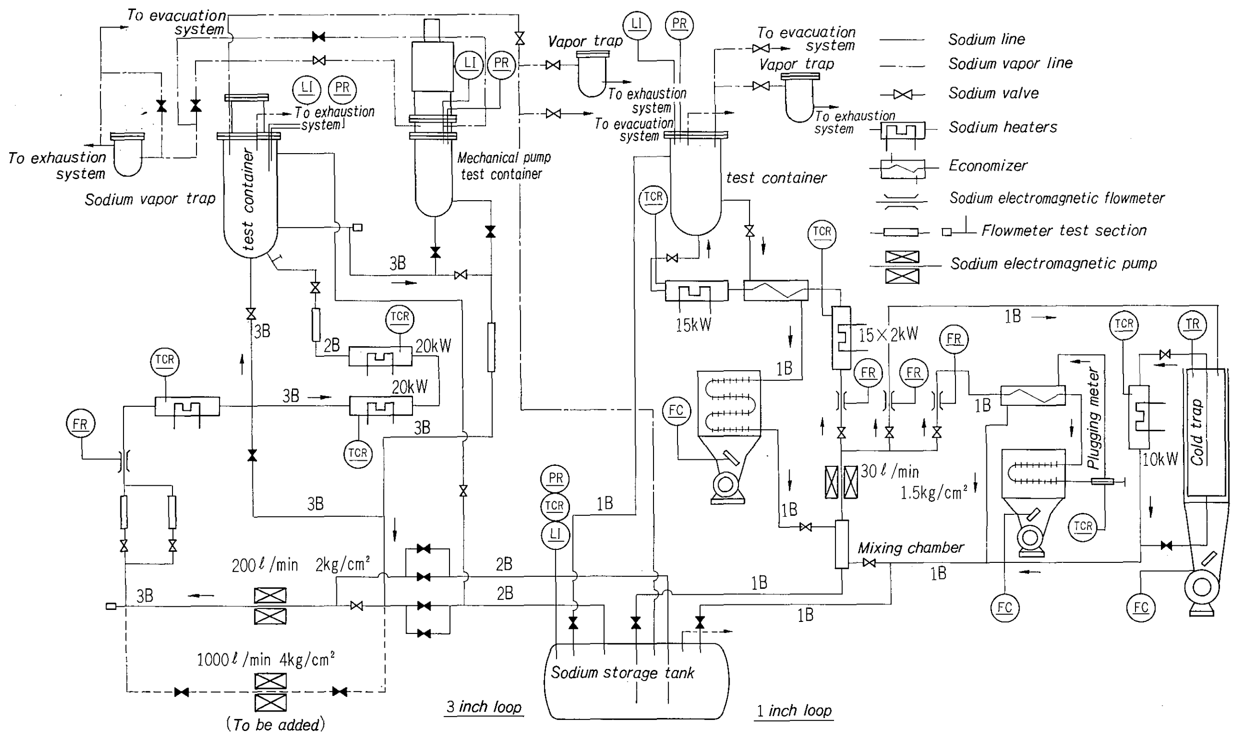


*Fig. 3* Flow sheet of " 1 in. loop " and " 3 in. loop "

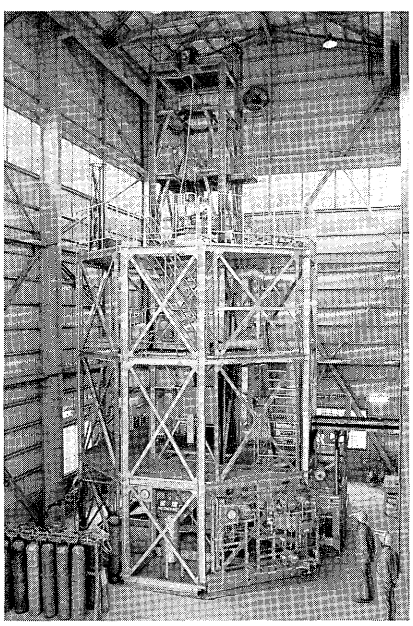

*Fig. 4* General view of " 1 in. loop "

be adjusted separately.

5) The sodium temperature is quasi-static and that of the central fuel subassembly and those of the six peripheal subassemblies can be given independently.

6) The flow to the central subassembly is to be gas/sodium two phase flow due to the injection of Ar gas (planned for 1972).

7) The test container contains a port for liquid level meter tests.

8) The sodium storage tank can be used in common

*Table 2* Main specificasions of " 1 in. loop "

| | |
|---|---|
| Test container sodium temperature | Maximum 550°C |
| Sodium flowrate | Maximum 30 *l*/min |
| Piping dimension | 1 in. |
| Amount of sodium | Storage in dump tank 3,000 kg<br>(Amount in circulation) About 1,000 kg |
| Total capacity of heaters | About 200 kW |
| Sodium purification capacity | About 10 ppm |
| Cover gas | High purity (99,994%) Ar gas |

with the 1 in. loop and there are no independent purification lines.

This facility will also be available for small scale heat transfer flow tests and mass transfer tests with future improvement. The construction of the 3 in. loop is shown in the left part of the flow sheet in *Fig. 3* and a general view is given in *Fig. 5*.

After the facility was set up, tests on permanent magnet flowsensors, eddy-current probe-type flowsensors for fuel channel monitoring and thermocouples at the outlet of fuel subassemblies entrusted by Power Reactor and Nuclear Fuel Corporation, the transit time method using cross-correlation of temperature fluctuations by Tokyo University, and performance tests for mechanical pumps and AC Faraday type electromagnetic pumps were performed using this equipment. The total operating time has already exceeded 1,500 hours. The main specification of the 3 in. loop are shown in *Table 3*.

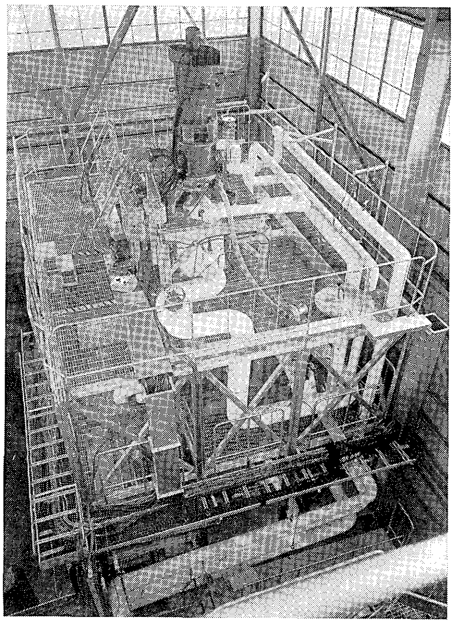

Fig. 5 General view of " 3 in. loop "

*Table 3* **Main specificasions of " 3 in. loop "**

| | |
|---|---|
| Test container sodium temperature | Maximum 600°C |
| Sodium flowrate | Maximum 350 *l*/min<br>(Planned increase to maximum 1,000 *l*/min in 1972) |
| Sodium filling rate | Maximum about 100 *l*/min |
| Sodium discharging rate | Maximum about 200 *l*/min |
| Piping dimension | Nominal diamete 3 in |
| Amount of sodium | (Amount in circulation) about 1,000 kg |
| Total capacity of heaters | About 130 kW |
| Cover gas | High purity (99.994%) Ar gas |

## III. DEVELOPMENT OF SODIUM DEVICES AND TECHNIQUES

In other outline of the test facilities given in section II, certain items of the research and development carried out by Fuji Electric were touched upon and it can be surmised from these brief remarks that the main points in the development of sodium devices and techniques for fast reactors have been the fuel handling system, sodium instruments and sodium pump. The following sections will desdescribe the main points of what has been achieved as well as future plans.

### 1. Fuel Handling System

1) Test on small parts

Research and development in Fuji Electric started with tests on small parts using the mini-loop. These tests were of two types: (1) seal tests using packings and seal gas, and (2) tests on metal sliding surfaces including the ball-latch mechanism and the pin and jaws latch mechanism. In the former tests, sodium deposition on seal faces and packing damage were investigated and in the latter tests, changes after tests on sliding faces up to a maximum of 600°C including continuous tests of more than 2,000 times at 500°C, and surface treatment were investigated. The tests results were all good and they provided valuable data for subsequent design and manufacturing of sodium equipment.

2) Test on the fuel handling mechanism for the experimental fast reactor

The gripper tests have already been completed for the fuel handling mechanism of the experimental fast reactor. The models of the fuel handling gripper for use inside the reactor vessel and the fuel unloading machine gripper for fuel transfer between the interior and exterior of the reactor vessel were fabricated (refer to *Fig. 6*). Operation tests, test for eccentric gripping by estimated displacement due to swelling, and overload tests with estimated seizing and galling were performed in air. After operation characteristics were checked by these tests, the grippers were attached to the test container of the 1 in. loop and tests in sodium under actual model conditions were performed. These results of these tests confirmed the suitability of the shape, surface treatment, and gap clearances of the pins.

Tests concerning the pot cooling and door valves were also performed in the 1 in. loop. The pot contains new fuel or spent fuel which is being transferred in or out of the reactor vessel respectively and it must be able to remove decay heat when the spent fuel is taken out. An electric heater was placed in the model pot in order to simulate the decay heat and comparative investigations of cooling effects were performed. The door valves are attached on the top of the reactor and in the exit port of the fuel storage rack and are used to prevent leakage of cover gas and radiation from the reactor vessel. They were tested together with the pot cooling device.

Reliability tests in sodium on the full-scale model of the fuel handling mechanism in Oarai Engineering Center of Power Reactor and Nuclear Fuel Development Corporation and gripper cleaning test are also

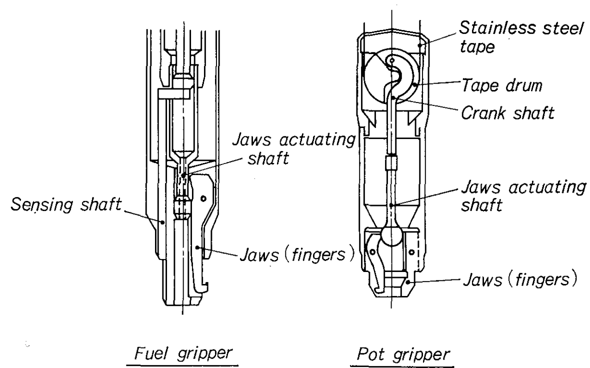


Fig. 6 Construction of fuel gripper and pot gripper

to be performed. The former is to use a model based on the results of research and development on the gripper mentioned previously and the performance tests in air as well as part of the tests in sodium have already been completed. The latter test are aimed at the development of an effective cleaning method for sodium which adheres to the grippers during fuel handling. The test is now being performed in Fuji Electric.

3) Research and development of the fuel handling mechanism for the prototype fast reactor

The fuel handling mechanism of the prototype fast reactor differs from that of the experimental fast reactor in that the proposal is to use a pantograph mechanism. There are two types of such mechanism: the fixed distance arm type proposed by Fuji Electric and the variable distance arm type. Under a commission from Power Reactor and Nuclear Fuel Development Corporation, a model of the former system has been designed and constructed by Fuji Electric, and at present, tests are being performed in air. The tests in sodium are to be performed at Oarai Engineering Center after completion of the tests on the full scale model of the fuel handling mechanism for the experimental fast reactor and exchanges of the sodium test container.

## 2. Sodium Instruments

Application of conventional instruments for process industries as sodium instruments is very difficult, and it is necessary to develop almost completely new instruments. The instruments which have been developed up to the present and are now on the market present considerable problems in relation to performance and reliability.

Fuji Electric started development of sodium instruments when the grant for research into the peaceful utilization of nuclear power was received in 1968 and since that time, several types of instruments have been developed. The explanation given below is divided into in-core instruments and process instruments.

1) In-core instruments

In fast reactors, the maximum reactivity occurs at the time of a hypothetical accident such as core compaction after core meltdown and not during operation. When it is assumed that 0.1% of F.P. gas is released on ground level, the prevention of accidents and propagation is very important in large fast reactors of the 1,000 $MW_e$ output class. Dangerous disturbances in the cooling can cause the spread of the accidents and it is essential that automatic protection equipment be used for early detection of such accidents. For this reason, instrumentation should be provided in the fuel subassemblies. Fuji Electric has been developing flowmeters and thermocouples as detectors and the cyclone gas separator as a setting means in places where detection is easy. The flowmeters which are being developed are the flow-through type permanent magnet flowmeter (*Fig. 7*) and the probe-type eddy current flowmeter (*Fig. 8*). A prototype of the former was tested in the mini-loop using the 1968 grant and in early 1971, a model assumed to be attached to the fuel subassemblies of the experimental fast reactor was tested in the 3 in. loop. A test on the high temperature stability of magnetic materials was performed and a model of completely leak tight construction was made (commissioned in both cases by Power Reactor and Nuclear Fuel Development Corporation). Highly significant data have been obtained. For the eddy-current type, models with 13.8 mm and 17.3 mm dia. probes were made. In this flowmeter, two kinds of eddy current are induced in the sodium around the detector: one is proportional to the leakage magnetic field of the excitation coil and the sodium velocity, another induced by the time variation of the magnetic field. Only the former component can be picked up by sensing coils. At the beginning of 1971, the electrical output vs. exciting frequency characteristics of these models were tested in the 3 in. loop. The results were good, as was the sensitivity to gas in the sodium. This meter should be useful as a means of failed fuel detection and location. In the future, simulation experiments on a seven subassembly array are to be performed.

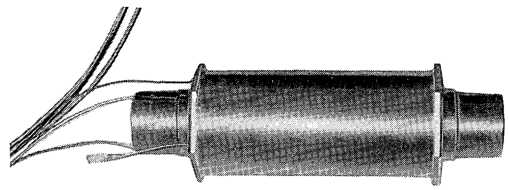

Fig. 7 Permanent magnet flow-through type in-core flowmeter

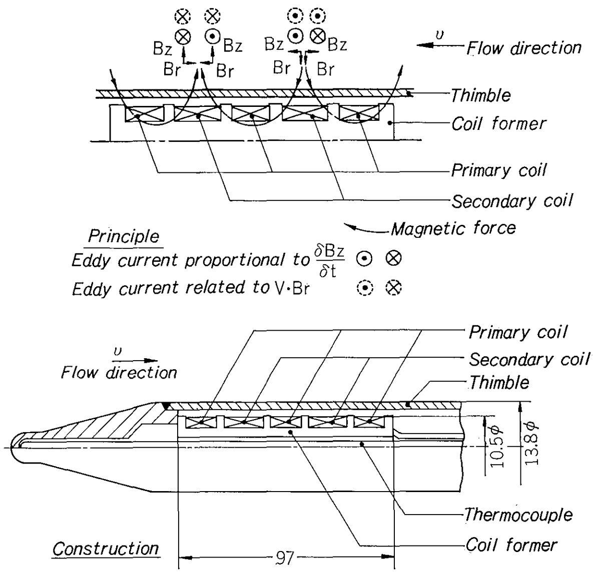


Fig. 8 Principle and construction of eddy-current probe type flowmeter

For the outlet temperature detector, flow reduction in one of the seven fuel subassemblies and the mixing effects with the peripheral subassemblies during an abnormal temperature increase have been confirmed by measurement of temperature distribution, and many highly interesting data has been obtained. In 1971, the flow will be increased, and tests are to be performed under conditions as close to actual condition as possible.

In the case of the cyclone, observations of air/water separation using a water loop and measurements of pressure distribution have been performed. Design data have been obtained.

2) Process instruments

The process instruments which have been developed are the electromagnetic flowmeter because of the necessity of loop construction, and the level meter and oxygen meter because of the many problem points in spite of their great importance. Work on the oxygen meter has been stopped temporarily because of other priorities and manpower problems in Fuji Electric. Many electromagnetic flowmeters have already been delivered to customers and are operating well. For the level meter, development of an improved induction level meter of the English type which at present has a low output sensitivity and a high temperature dependence is now underway both theoretically and practically, and at present, the prospects look good.

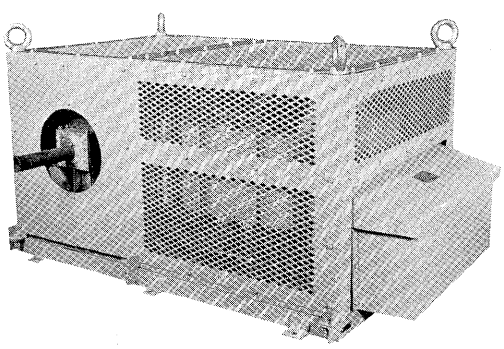

*Fig. 9* 200 $l$/min AC Faraday pump

### 3. Sodium Pumps

Sodium pumps for fast reactors are mechanical type centrifugal pumps for circulation and electromagnetic pumps for use in auxiliary systems. A DC Faraday type electromagnetic pump model has been constructed for use in the mini-loop and development of an AC Faraday type electromagnetic pump, featuring improved long term reliability due to the use of integral construction of electrodes and sodium duct is progressing. As series up 2 kg/cm$^2$ × 200 $l$/min has been completed, lots of units have been delivered and operation results have been good. An outer view of the 200 $l$/min model of the AC Frarday type pumps is shown in *Fig. 9*. A linear induction pump for use with medium flow rate was developed and performance tests were performed.

Model of mechanical seals and a hydrodynamic bearing for sodium, as well as sodium bearing type centrifugal pump have been made by Ebara Manufacturing Co. Ltd. which is a member of the First Atomic Power Industry Group. After tests in a water loop, these models underwent performance and continuous test in sodium in the Fuji 3 in. loop. The results were good and the effects were as expected.

## IV. CONCLUSION

This article has introduced the three sodium test loops set up in Fuji Flectric, as well as an outline of the present state and future plans for development of sodium devices using these loops. Because of the large amount of information to be included in a limited space, only the main points have touched upon but the authors hope that the article gives a picture of the various aspects of the development of sodium devices in Fuji Electric. It is planned to publish more detailed reports concerning this field in later issues.

Although it was not mentioned in the article, a close connection with related techniques such as materials is required in the development of highly reliable sodium devices and research and development are also being performed concerning this. In combination with these techniques, it is expected that the facilities and results mentioned here will be used effectively in the development of better sodium devices.

Finally, the authors wish to thank sincerely all persons at Power Reactor and Nuclear Fuel Development Corporation, and Japan Atomic Energy Research Institute for their cooperation and guidance in this research and development work.