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IN-PILE LOOP OWL-2 AND IRRADIATION TESTS DONE WITH IT

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The OWL-2 which was built in the JMTR as the biggest water loop in Japan has been operating for irradiation service since February 1972. The desired objective of the OWL-2, contributing to the development of various nuclear fuels and materials for the light water power reactor and to reactor engineering, has been so fully achieved that the OWL-2 is planned to be dismantled.

After the dismantling, a loop, needed for the research and development of the breeding blanket for the fusion reactor, is going to be installed in place of the OWL-2 as a part of the JMTR Modification Program.

This paper deals with the history of the OWL-2 with an emphasis on the technical affairs taken into consideration when designing the OWL-2, the irradiation tests, development of the turbine flowmeter, results of the surveillance test of the material of the in-reactor tube, the knowledge gained in the course of the investigation into the cause of transgranular stress corrosion cracking (TGSCC) which developed in the wall of the in-reactor tube, and countermeasures taken to prevent TGSCC from recurring.

Keywords: JMTR, OWL-2, In-pile Water Loop, Surveillance Test, In-reactor Tube, SCC

インパイル・ループOWL－2と照射試験

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OWL－2は，我が国最大規模のインパイル・ループとして1972年2月JMTRに設置されて以来，各種の動力炉用燃料・材料試料の照射試験及び炉工学的試験に使用されてきたが，所期の目的を達成したため廃止する計画である。

廃止後には，JMTR改造計画の一環として核融合炉用増殖ブランケットの試験研究を進めて行くうえで必要な新ループの設置を予定している。

本報告は，インパイル・ループの設計上考慮した点を中心に，廃止計画に至るまでの経緯と照射試験，タービン型流量計の開発，炉内管構造材のサーベイランステストの結果及び炉内管に発生したTGSCCとその防止対策などについてまとめたものである。

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1. Introduction

The Japan Materials Testing Reactor (JMTR), located at the Japan Atomic Energy Research Institute in Oarai, Ibaraki Prefecture, is a tank-type reactor rated at 50 MW_t which is moderated and cooled using light water. Currently, in the JMTR, the two in-reactor loops are attached to the reactor. These are the Oarai Water Loop 2 (OWL-2) and Oarai Gas Loop 1 (OGL-1). The OGL-1 is intended to be used for conducting irradiation tests on coated particle fuels and heat resistant materials in an operating environment similar to that of the high temperature gas cooled reactor.

The OWL-2, completed in February 1972 to help develop the technology necessary for the domestic production of power reactors, has been utilized for irradiation testing of nuclear fuels and materials for various power reactors and for tests related to nuclear engineering.

As a part of the JMTR Capability Improvement Program, irradiation loops, necessary for R/D to improve the light water reactor and for the technological development of the breeding blanket intended for use in a fusion reactor, are planned to be built in place of the OWL-2 after it has been removed. By replacing the OWL-2 with the planned loops, a more effective utilization of the JMTR will be realized. All of the desired results have been obtained through use of the OWL-2, and its service has come to an end.

Figures 1.1 and 1.2 show a core arrangement of the JMTR.

2. OWL-2

2.1 Description of the OWL-2

The design work of the OWL-2 began in 1968 and was based on the experience accumulated through the design, fabrication, construction and operation of the previously built loops - the Oarai Water Loop 0 (out-of-reactor testing facility, OWL-0) and the Oarai Water Loop 1 (OWL-1).

After permission to modify the facility was obtained from the Science and Technology Agency (STA) in January, 1969, the manufacturing of the OWL-2 began, and the operation of the OWL-2 for the purpose of irradiation started in June, 1972 after being subjected to the final pre-operation inspection in February, 1972.

1. Introduction

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In March, 1983, transgranular stress corrosion cracking (TGSCC) developed in the wall at the top region of the OWL-2 in-reactor tube, forcing the in-reactor tube to be replaced with a new tube which was specifically fabricated to prevent the occurrence of such cracking; and the operation of the OWL-2 resumed in November, 1986.

The OWL-2, a high temperature, high pressure loop, having a cooling circuit independent of that of the reactor, is intended for conducting the following experiments:

(1) Irradiation experiments on various nuclear fuels and materials for the PWR, BWR, and ATR.

(2) Compatibility tests of the above fuels and materials.

(3) Corrosion tests of the fuels and materials exposed to the coolant.

(4) Irradiation tests of fuel specimens at the desired rate of heat generation or at the critical heat flux in a high-temperature and high-pressure environment.

(5) Thermohydraulic tests of the fuel assemblies.

(6) Nuclear fuel damage tests.

Table 2.1 gives a history of the OWL-2 and Table 2.2 presents engineering data of the OWL-2.

2.2 Structure of the OWL-2

As Fig. 2.1 shows, the OWL-2 chiefly consists of the primary loop, secondary loop, make-up system, auxiliary system, instrumentation, safety system, and shielded plant room.

(1) Primary Loop

The primary loop is composed of the in-reactor portion and the out-of-reactor portion. The in-reactor section, disposed inside the reactor pressure vessel, is to irradiate the specimens with the neutrons generated in the reactor core. The out-of-reactor region, being connected to the in-reactor portion, is a closed circuit to allow the coolant to circulate through the in-reactor and out-of-reactor regions.

a. In-reactor Portion

As Fig. 2.2 shows, the in-reactor section is made up of the in-reactor tube, accommodator, a stabilizer around the in-reactor tube in the core, the gears supporting the in-reactor tube, and the outlet and the inlet branch tubes connecting the in-reactor portion and out-of-reactor portion.

The in-reactor tube, a double-walled, straight, once-through type, consisting of the pressure tube and outer tube is 6 inches (165.2 mm) in nominal outer diameter and about 11 m long. It is vertically mounted through the upper lid of the reactor pressure vessel, through the reactor core at the grid location K.L-3.4, and through the bottom of the reactor pressure vessel.

In order to provide a thermal insulating barrier between the coolant (at about 280°C) in the loop primary circuit and the reactor primary cooling water (50°C max.), the circular space between the pressure tube and the outer tube is filled with nitrogen gas having a low thermal conductivity, at a pressure of 0.2 kg/cm^2G.

The difference in the axial lengths of the pressure tube and the outer tube, caused by the difference in temperatures of these tubes, is accommodated by the pressure tight bellows attached at the bottom of the in-reactor tube.

A specimen that is to be irradiated is attached to the lower end of a hanger rod, which is hung from the top-closure of the in-reactor tube, and then inserted through the top-opening of the in-reactor tube down to the in-reactor test section situated at an elevation opposite to the reactor core. The dimensions of the test section of the in-reactor tube were determined to be $117.8^{+0.3}_{-0}$ mm in inner diameter and 1,100 mm in effective axial length, taking into consideration the size of the ATR fuel assembly which was irradiated in the OWL-2. Operational data, like temperatures of the specimen being irradiated, are collected through the measurement lines run through the seal mechanism in the top closure of the in-reactor tube. Figure 2.3 shows a detailed structure of the OWL-2 in-reactor tube and Figure 2.4 a specimen assembly for irradiation.

Major specifications of the in-reactor tube are as follows:

Design pressure: 8.7 MPa (88 kg/cm^2G)

Design temperature: 300°C

Standards applied are as follows:

Engineering Standards for Structures of Facilities of Power Plants (MITI Notice No.501)

ASME Boiler and Pressure Vessel Code, Section VIII, 1968

Material chiefly used:

Austenitic Stainless steel (denote SUS 316 or SUS NG below)

b. Out-of-reactor Portion

The out-of-reactor portion comprises the loop pump system, heat removal system, boiler system, condenser system and the purification system (Fig. 2.1). The primary coolant is subcooled in a range from 10 to 30°C and is discharged from the outlets of the two circulation pumps. The flow rate is regulated by a flow control valve and the coolant proceeds upwards in the in-reactor tube, removing the heat generated from the specimen, before coming back to the inlets of the pumps to complete a closed circuit.

The heat sources for the primary coolant include the nuclear heat from the specimen being irradiated, heat produced in the circulation pumps, the immersion heaters mounted in the boilers, and the strap-on type electric heaters attached to the outer surface of the piping of the primary circuit and next to the inlet of the in-reactor tube.

Using the flow control valves VC-193 and VC-194, the temperature of the coolant in the heat removal system is maintained at 200°C. These valves regulate the ratio of the amount of coolant which passes through the primary cooler to that of the remaining coolant that flows through without being cooled.

The temperature of the coolant in the loop pump system is controlled by introducing a portion of the 200°C coolant from the heat removal system through the mixer 103 so as to maintain the temperature of the coolant at the inlet of the in-reactor tube identical at a pre-set value that is decided beforehand as one of the operation parameters.

It is reasonable to assume that the temperature of the coolant at the inlet is about the same as of the operating temperature, since the difference in the temperatures of the coolants at the inlet and around the specimen being irradiated in the in-reactor tube is less than 1°C.

The pressurization in the primary loop is achieved by using the steam generated in the boilers. The pressure in the primary circuit is regulated by condensing the steam coming into the condenser from the boilers using a fraction of the 200°C coolant diverted from the heat removal system and sprayed into the condenser. The coolant flow path, as shown in Fig. 2.5, is altered according to the operating mode, either the pressurized mode or the boiling mode.

For the pressurized operating mode, the coolant, after leaving the outlet of the in-reactor tube, returns to the inlet of the circulation pumps without passing through the separator. The valve on the separator circuit is closed. For the boiling operating mode, the coolant, turned into a two-phase flow at the test section of the in-reactor tube (steam quality 20% max.), is allowed to pass through the separator where the water-steam mixture is separated into steam and saturated water. After being separated, the steam proceeds to the condenser to be reduced to water together with the steam coming from the boiler using a portion of the 200°C coolant coming from the heat removal circuit. This water is brought back to the inlets of the circulation pumps. The flow rate of the primary coolant in the loop pump system can be measured with two kinds of flowmeters (orifice and turbine types) installed in the circuit next to the outlets of the circulation pumps (a detailed description of the turbine-type flowmeter is given in Paragraph 3.1.)

Since insoluble corrosion products generated in the primary circuit cause the quality of the coolant to deteriorate, about 1.5% of the coolant is shunt through the purification system consisting of ion exchanger columns to maintain the quality of the coolant at the desired level.

Major specifications of the out-of-reactor section are as follows:

Design pressure: 11.4 MPa (115 kg/cm^2G)

Design temperature: 320°C

Coolant flow rate: 1100 kg/min max

Water quality: conductivity; less than 0.36μS/cm, pH; about 7, dissolved oxygen concentration; less than 50 ppb

Standards applied: The First Class Vessel Construction Standard

Chief material used: Type SUS 316 stainless steel

(2) Secondary Loop

The secondary loop, consisting of the circulation pumps, surge tank, cooler and ion exchanger columns disposed in the form of a closed circuit, removes the heat generated in the primary loop, dissipating it to the atmosphere through the Utility Cooling Loop System (UCL).

After passing through the outlets of the two parallel secondary circulation pumps, the secondary coolant flow rate is regulated at 700 kg/min by the flow control valve VC-281 and measured by the orifice-type flowmeter. The heat removed from the coolant circulating in the secondary loop is transferred to the UCL through the secondary cooler, allowing the coolant to be controlled at 100°C.

The secondary loop operating pressure is kept at around 1.9 MPa (18 kg/cm^2G). A pressure sensor in the surge tank regulates the pressure in an on-off manner when the pressure reaches a specific high or low point, the electric heater are automatically signaled to turn off or on.

To maintain the quality of the water at the desired level, about 1.5%

of the secondary coolant is continuously passed through the purification system consisting of the ion-exchange columns in the same fashion as the purification system on the primary loop.

Major secondary loop specification:

Design pressure: 3.0 MPa (30 kg/cm^2G)

Design temperature: 220°C

Maximum coolant flow rate: 1,000 kg/min

Coolant quality: electrical conductivity; less than 0.36μS/cm, pH; about 7

Standards applied: First Class Pressure Vessel Construction Standard

Chief material used: Type SUS 316 stainless steel

(3) Auxiliary System (Make-up, Drain, and Sampling Systems)

The make-up system supplies the demineralized water to fill both the primary loop and secondary loop initially and to make-up water lost from the systems during the operation period. The drain tank is for receiving the drainage from the primary loop. The sampling system was installed on the primary loop to sample the coolant from the system in the sampling box. All the sampling lines from the essential portions of the circuit are kept in the sampling box.

(4) Instrumentation and Control

To facilitate the safe, reliable, simple operation of the OWL-2, all the instrumentation systems were so built that the operator could observe the vital readings indicating the operating conditions of the loop on the various instruments on the control panel. These instrumentation systems were built in accordance with the centralized control concept. These instrumentation systems allow the operator to observe all the key signals transmitted from the various parts of the loop and correct any deviation from the prescribed operating parameters at the control panel.

When a deviation progresses to the alarm level, a warning is sounded and one of the lamps (arranged on the control console in rows) flashes, indicating what is wrong with the loop, leading the operator to take a corrective action to remedy the deviation. If the deviation, despite the operator's effort, progresses further to the next loop-cool-down level, the interlock mechanism is brought into function, automatically cooling down the loop, to secure its safety. The warning which are classified into four levels according to their severity are stated below. The functions of the interlocking system are given in Tables 2.3 and 2.4.

a. Warning (level 1)

A sounding buzzer and flickering lamp notify the operator that a deviation of an operating parameter has progressed to a predetermined level. The flickering lamp tells an operator what the problem is.

b. Cool Down (level 2).

When an operating parameter further deviates up to the level 2 warning, the interlocking cool down system sets in automatically, lowering the temperature of the coolant in a specific sequence, to secure the safety of the OWL-2.

c. Reactor Set Back (level 3)

Further deterioration of the operating conditions puts the "Reactor Set Back" interlocking system into operation, in the same way as the level 2 warning, to lower the power level of the reactor at a rate of a period of 30 sec until the abnormal condition has ceased.

d. Reactor Slow Scram (level 4)

If an operating parameter continues to deviate from the desired value progressing to the pre-set value of "Reactor Slow Scram", the power level of the reactor is instantly reduced to zero. To avoid the unnecessary reduction in the reactor output through either "Reactor Set Back" or "Reactor Slow Scram" owing to a malfunction developed in the instrumentation, both the "Reactor Set Back" and "Reactor Slow Scram" signal channels operate on the two out of three signal basis. Therfore, "Reactor Set Back" and "Reactor Slow Scram" signal channels for shutting down the reactor are triplicated. At least two of the three trip channels must have indications of abnormal conditions, before the reactor is shut down.

The bypass switch enables the instruments and controllers to be replaced with new ones when some defect develops in the devices. This can be done while the reactor is in operation without any alteration in the operating parameters. Figure 2.6 shows the control panel of the OWL-2.

(5) Shielded Plant Room

The radiation level of the out-of-reactor portion of the OWL-2 was so high, due to the nuclides such as Cr-51, Mn-54, Co-58, and Co-60 staying on the inner surface of the primary loop as crud after being activated in the in-reactor tube by bombardment with the fast neutron ($E > 1$ MeV), that the whole primary loop had to be housed in the compartment, called a shielded plant room, built of heavy concrete 1 m thick to reduce the radiation intensity at the outer surface of that compartment to less than 0.1 mSv/h.

To prevent the release of radioactivated nuclides from this concrete compartment, it is made air-tight and the pressure inside the compartment is kept lower than the ordinary atmospheric pressure by 4.8 x 10 Pa (5 mm H_2O).

2.3 OWL-2 Operation

The operation of the OWL-2 is carried out in accordance with the operation schedule of the JMTR. The JMTR is operated, as a general rule, in five operation periods a year. Figure 2.7 presents a typical JMTR operation period which consists of a first half period and a second half period, each lasting 12 days, with a middle shutdown period of 2 days for the refuelling operation. The OWL-2 is started up prior to the startup of the reactor so that the OWL-2 will be in the pressurized operation mode when the reactor is started up. This will eliminate the possibility of the OWL-2 primary coolant boiling in the in-reactor tube due to the nuclear heating caused by the reactor operation. The boiling of the coolant taking place in the in-reactor tube renders unfavorable effects on the nuclear measurement of the reactor. Toward the end of the operation period, the reactor is completely shut down before the cooling-down and depressurization of the OWL-2 begins.

2.4 Irradiation Tests Done in the OWL-2

As shown in Table 2.5, nine specimens were irradiated in the OWL-2 in-reactor tube over a span of 74 operation periods beginning with the operation period No.13 in January 1972, the OWL-2 performance test period, when the dummy fuel rod was loaded in the in-reactor tube. These tests ended with the operation cycle No.86 in April 1989 when the last irradiation specimen, 79LF-14A, was removed from the OWL-2 in-reactor tube. In addition to these nine irradiation specimens, 13 other specimens were loaded in the out-of-reactor test sections located on the primary circuit housed in the shielded plant room. The principal objectives of the irradiation tests conducted in the OWL-2 were to irradiate materials with the aim of investigating their brittle characteristics (knowledge from this test will be used to make a new and better reactor pressure vessel), to irradiate the ATR fuel assembly to see its nuclear and structural behavior, and to irradiate the fuel specimen as a preliminary step before it is subjected to power ramp tests afterwards. (knowledge from these tests well be used to make a new and better reactor vessel)

The out-of-reactor test sections, one located close to the in-let of the in-reactor tube and the other close to the out-let of the tube, were loaded with the specimens of Type SUS 304 and SUS 316 stainless steel and heat-resistant Inconnel to find how and why crud deposition and corrosion occurred.

Table 2.5 gives the contents of the irradiation tests conducted in the in-reactor test section.

Currently the OWL-2 is being operated in the natural convection circulation mode with no specimens loaded either in the in-reactor test section or in the out-of-reactor test sections.

3. Results of the Irradiation Tests

Among the results obtained through irradiation tests conducted in the OWL-2, three representative findings are dealt with below. Table 3.1 gives list of the titles of the reports on the results of the irradiation tests.

3.1 Turbine Flowmeter in the Process Control

(1) A General Description

The OWL-2 was operated in either the pressurized mode or the boiling mode. The primary coolant mass flow rate, varying in a range from 50 to 1,100 kg/min according to the operation mode, needed to be measured accurately by means of a flowmeter since the accuracy of the measurement readily affected the outcome of the irradiation experiments.

In the OWL-2, two kinds of flow meters were installed - a turbine flow meter for measuring the coolant flow rate in the range from 50 to 400 kg/min and an orifice type flowmeter to measure the range from 400 to 1,100 kg/min.

Although its measurement range is limited, the turbine flowmeter was capable of measuring both single phase flow and two phase flow and could be fabricated in a small size. These two features made it suitable to be used as the coolant flow measurement instrument in facilities for studying power the aim of investigating their brittle characteristics (knowledge from this test will be used to make a new and better reactor pressure vessel), to irradiate the ATR fuel assembly to see its nuclear and structural behavior, and to irradiate the fuel specimen as a preliminary step before it is subjected to power ramp tests afterwards. (knowledge from these tests well be used to make a new and better reactor vessel)

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(2) Improvement in Turbine Flowmeter

The turbine flowmeter employed in the OWL-2 consisted of a front support, rear support, shaft, bearing, rotor, a pick-up for collecting the signal generated by the flow and an amplifier (Figs. 3.1 and 3.2). After reaching the flowmeter body, the coolant was streamlined by the front support to rotate the rotor before passing through the rear support and flowing into the piping welded to the body downstream. The pick-up coil mounted around the outer surface of the body collected the electric pulses generated by the rotor which was rotated in proportion to the mass of the coolant passing through the flowmeter.

In the early operation period of the OWL-2, the bearing of the flowmeter broke from time to time. The cause of this problem lay in the fact that different kinds of crud, present in the coolant, came into the rotating portion of the flowmeter and deteriorated the lubrication of the bearing.

To get rid of this problem a modification in the mechanism of the rotating portion was carried out, and different material was used to manufacture this rotor, eliminating any further breakage during the operation. The durability of this rotor was enhanced to such an extent that no other problems occurred.

(3) Design and Manufacture of Turbine Flowmeter for Irradiation Facilities

As a part of the study of fuel safety for light water reactors, an experiment for measuring the temperature of the center of the nuclear fuel was planned using the OWL-1 as the irradiation instrument.

In the preparation stage of this experiment, the development of a in-core turbine flowmeter was a prerequisite in order to measure accurately the flow of the coolant running through the specimen being irradiated in the in-reactor tube of the OWL-1. Utilizing the knowledge gained from the manufacture of the turbine flowmeter which had been used as a component of the process control in the OWL-2, the in-core turbine flowmeter for the fuel safety experiment for light water reactors was developed. It was the first in-core turbine flowmeter ever built in Japan. Figure 3.3 shows an axial sectional view of the in-core turbine flowmeter and Table 3.2 presents engineering data of the process control and the in-core turbine flowmetrs.

3.2 Surveillance Test of the OWL-2 In-reactor Tube Material

(1) A description of the surveillance test

A scarcity of data available around 1971 on the mechanical properties of austenitic stainless steel bombarded with a high fluence of neutrons, when the designing of the OWL-2 in-reactor tube was initiated, forced us to tentatively determine the lifetime of the in-reactor tube in terms of the fast neutron (E>1 MeV) fluence to be 1×10^{21} n/cm^2.

Since the above value appeared to be conservative from the viewpoint of metallurgy, during the design stage of the OWL-2, it was also decided that a surveillance test, using test pieces made from the same material from which the OWL-2 in-reactor tube was made, would be necessary. If the outcome of this test, and other similar tests conducted abroad, could prove that the integrity of the tube would be maintained when subjected to a high fast neutron fluence, the lifetime of the in-reactor tube would be increased beyond our tentatively determined lifespan.

(2) Results of the Surveillance Test

The two test pieces for the surveillance test were made from the same material as the in-reactor tube (Type SUS 316 TP stainless steel) - one for the impact test and the other for the tensile test.

To gain the desired result in the early stage of the test, the test pieces were placed at the location of the reactor core where the fast neutron flux was about 10 times higher than that of the position at which the OWL-2 in-reactor tube was installed.

The test result obtained by examining the test pieces irradiated with fast neutrons (E>1 MeV) up to a fluence of 3.4×10^{21} n/cm^2 showed the following outcomes:

a. The tensile strength increased with an increase in the irradiation fluence, as shown in Fig. 3.4.

b. As Fig. 3.5 shows, the tensile elongation decreased with, contrary to that of the tensile strength, the increase of the amount of the fluence.

The elongation in the irradiated test piece was 40 % at room temperature and was 33 % at a temperature of 285°C. The value at room temperature satisfies the provision in the Japanese Industrial Standards which says that the elongation of Type SUS 316 TP stainless steel must be greater than 35 % at room temperature. However the value of 33 % at 285°C appears acceptable even though no value is mentioned in the Standard, since a material similar to SUS 316 in mechanical properties, exhibits a decrease in elongation with increasing temperature.

C. The impact strength, in the same fashion as the tensile elongation, decreased as the neutron bombardment increased.

Some overseas literature deals with irradiation of materials in which the total fast neutron (E>1 MeV) fluence exceeds 3.4×10^{21} n/cm^2. One literary thesis stated that the elongation that occurred in a type SUS 347 piece of stainless steel irradiated with fast neutrons (E>1 MeV) up to 3×10^{22} n/cm^2 was 35 % at 25°C and 15 % at 315°C. The test piece from the ETR in-reactor tube material (Type SUS 347 stainless steel) irradiated up to a fast neutron (E>1 MeV) fluence of 7.2×10^{21} n/cm^2 showed an elongation of 10.3 %.

(3) Evaluation

The evaluation of the change in mechanical properties that would occur in the OWL-2 in-reactor tube was conducted on the basis of the results of the surveillance test, while taking information from the literature published abroad into consideration, and the conclusion reached was that the lifetime of the OWL-2 in-reactor tube in terms of fast neutron bombardment would be extended from our tentatively specified value of 1×10^{21} n/cm^2 to a total fast neutron (E>1MeV) fluence of 3.4×10^{21} n/cm^2. Based on the above evaluation, the lifetime of the OWL-2 in-reactor tube was shifted from 1×10^{21} n/cm^2 to 3×10^{21} n/cm^2 in respect to the fast neutron (E>1 MeV) bombardment.

3.3 Occurrence of Cracks in the OWL-2 In-Reactor Tube and Countermeasures Taken against Transgranular Stress Corrosion Cracking

(1) An introduction

Cracks which developed in March, 1983 at the top region of the OWL-2 in-reactor tube led to the operations in which the in-reactor tube was removed and examined to identify the cause of the cracks. The investigation into the cause revealed that the pittings generated in the dents on the inner surface of the pressure tube were the origin of TGSCC.

(2) Circumstances under Which Cracks Developed

The region where the cracks originated, as Fig. 3.6 illustrates, is situated where the extension tube for Graylock (type SUS 316 stainless steel) and the T-shaped pressure tube (type SUS 316 TP stainless steel) were welded.

Until the occurrence of the cracks in March, 1983, the OWL-2 had been operated for about 24,000 hours since its first operation for irradiation tests in June, 1972. With a slight difference in both the operating pressure and temperature between the boiling mode operation and the pressurized mode operation, the OWL-2 was operated at a pressure in a range from 7.0 to 7.3 MPa (70 to 73 kg/cm^2G) and at a temperature in a range from 260°C to 285°C, maintaining the quality of the primary coolant (light water) within 0.3 to 0.36 μS/cm in electric conductivity, at about pH 7, and less than 50 ppb in dissolved oxygen concentration.

(3) Investigation into the Cause of the Cracks

a. Manufacturing Process of the In-reactor Tube

The top portion of the in-reactor tube where the crack developed was fabricated of type SUS 316 TP stainless steel (refer to Japanese Industrial Standards G 3459) through the drawing method and was subjected to solution annealing. The top portion was welded through TIG welding process without being subjected to solution annealing and other heat treatments to relieve the stress induced during welding. The soundness of the welded portion of the OWL-2 in-reactor tube was confirmed by non-destructive tests before the in-reactor tube was installed in the reactor.

b. Findings from the Investigation

The investigation into the cause of the cracking was conducted on the sample pieces taken from the area of the in-reactor tube where the cracks developed by means of metallurgical inspection, non-destructive tests and material testing. Table 3.3 gives the methods of the examinations and their results.

A lot of dents were found in the vicinity of the cracks with the naked eye, and optical microscopic examinations revealed that pittings originated from the dents. In particular, the cracks originated from a dent close to the welded portion and extended three-dimensionally both in the axis of the in-reactor tube and the circumference of the tube to such an extent that some of the cracks penetrated completely through the wall of the tube. The observation of the fractures along which the cracks ran with an electron microscope showed that all the cracks were caused by TGSCC.

The portion of the in-reactor tube in which the cracks occurred was located outside of the reactor pressure vessel so that no embrittlement could be induced by the fast neutron ($E > 1$ MeV) bombardment even during the period of the reactor operation. However, stagnation of the coolant was likely to take place owing to both the presence of the finned flow-deflector positioned there and the shape of the top portion of the in-reactor tube. This physical configuration of the top-portion of the in-reactor tube probably did not allow the coolant to circulate to a sufficient degree to reduce the concentration of the dissolved chloride once some chloride ions were carried there by the coolant, creating an environment favorable for inducing pitting on the inner surface of the in-reactor tube.

The pittings which originated in the dents that were produced during the loading operation of the specimen into the in-reactor tube allowed the stress to concentrate at the bottoms of the dents, leading to the development of TGSCC which grew gradually to such an extent that they completely penetrated across the entire wall thickness, being accelerated by both the varied pressure caused by the OWL-2 operation and the residual stress generated by the welding during the fabrication (see Fig. 2.3).

(4) Countermeasure to Prevent TGSCC

Being different from intergranular stress corrosion cracking (IGSCC) commonly found in the austenitic stainless steel used in boiling water reactors, the cracks that developed in the OWL-2 in-reactor tube were of the TGSCC type.

There is a theory that TGSCC in austenitic stainless steel occurs as the various factors like the sensitization of the material, tensile stress, dissolved oxygen concentration, and temperature simultaneously reach a certain level. This general knowledge combined with the findings from the investigation into the cause of the cracks in the in-reactor tube required that the following countermeasures to be taken to prevent further TGSCC.

a. To lower the carbon content as much as possible (carbon content: 0.01-0.015%, SUS 316 NG equivalent) by employing austenitic stainless steel JIS G 3459 SUS 316 TP S-C (1978) as material for the new in-reactor tube.

b. To reduce the residual stress in the in-reactor tube by employing TIG-Torch Heating Stress Improvement (THSI) which was designed to allow less residual stress to take place in a piece of welding work than a conventional welding technique in the welding operation of the top closure extension piece.

c. To continually monitor the quality of the coolant fed into the circuits of the OWL-2 by means of a low level chloride monitor.

d. To lessen the occurrence of dents on the inner surface of the in-reactor tube, modification in the shape of both the flow deflector block and the specimen insertion gear in physical configuration was made.

(5) Conclusion

The new in-reactor tube with practical countermeasures to prevent TGSCC was installed in the JMTR during the 1986 JMTR Summer Annual Inspection Period to resume the irradiation operation of the OWL-2 in November 1986.

4. Summary

The OWL-2 was incorporated with adequate countermeasures to secure its safety by applying the fail-safe philosophy at the core of its design. This was necessary because the incompressible light water coolant of the OWL-2 could explosively change into high-temperature and high-pressure steam, expanding its volume once the pressure boundary of the system fails when being operated. The countermeasures to prevent malfunction and the various interlock mechanisms in the instrumentation system were provided to improve the safety of the OWL-2.

From the viewpoint of the maintenance of the OWL-2, the confirmation of the soundness of the system was carried out by means of conducting a thorough in-service non-destructive inspection of the pipings, which constitute the pressure boundary where thermal stress was expected to take place to a certain extent, in addition to the pre-operational function checks of the instrumentation and the pressure tests of the system prior to every operation period.

These precautions enabled the OWL-2 to be operated satisfactorily, eliminating the possibility of the occurrence of serious malfunctions in the OWL-2, and allowed the OWL-2 to fulfill so completely the originally desired role expected of it at the outset of this program that the execution of the planned dismantling of the OWL-2 was justified.

The dismantling of the OWL-2 is to proceed in two stages, the in-reactor portion stage and the out-of-reactor portion stage. Although experience in the removal of the in-reactor portion from the reactor was gained through the in-reactor tube replacement job performed in 1986 to cope with the leak that developed in the in-reactor pressure tube wall in 1983, scrupulous care has to be exercised in carrying out the planned removal operation of the in-reactor tube of the OWL-2 so that the reactor primary coolant does not leak out of the the reactor excessively in the course of the operation since the in-reactor tube is attached to the reactor through the bottom of the reactor vessel.

Radioactivity of about 4×10^{10} Bq produced in the course of the

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Radioactivity of about 4×10^{10} Bq produced in the course of the irradiation tests conducted in the past about two decades is expected to be deposited on the inner surface of the piping and components comprising the out-of-reactor portion of the OWL-2. The dismantling operation of the out-of-reactor portion is anticipated to last about 1 year, and a few operation periods of the reactor will fall into the period of the dismantling job so that a close study is needed to avoid both the interference of the removal operation on the reactor operation and the possible deterioration of the working conditions in the reactor building.

To cope with the anticipated problem of how to manage and dispose of the radioactive waste that would increase in volume in the future as large-scale facilities at JMTR become obsolete and are removed and replaced, it is necessary to establish a reasonable guideline to follow to dispose of the activated waste.

The authors will be very happy if this report dealing with a general outline of the OWL-2 and a part of the results obtained through the irradiation tests proves to be helpful for the reader to further understand the in-reactor loop.

Acknowledgments

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Sincere cooperation shown in the operation and maintenance of the OWL-2 by personnel of the Radiation Control Division, Radioactive Waste Management Division, and Irradiation Division II, and other Divisions at JMTR deserve author's unbounded gratitude.

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Table 2.1 History of the OWL-2

Jan. '69	Licensing application for installation of OWL-2 was submitted
Mar. '69	Contract for construction of OWL-2 was closed
Apr. '69	Construction of OWL-2 was approved by STA
Apr. '70 - Apr. '72	In-reactor portion construction work
Sept. '71	In-reactor tube installation in the reactor
Oct. '71	Installation of cooler in the shielded plant room
Oct. - Nov. '71	Out-of-reactor portion performance test
Dec. '71	Fabrication of irradiation specimen for in-reactor portion performance test and damaged specimen handling container
Jan. - Feb. '72	OWL-2 performance test (JMTR operation period No.13, with a material specimen loaded in the in-reactor tube)
Feb. '72	Pre-operation licensing test
Mar. - Apr. '72	OWL-2 performance test (JMTR operation period No.14, with a fuel specimen loaded in the in-reactor tube)
Aug. '74	Partial modification of primary circulation pumps
July '79	Replacement of strainers
Mar. '83	Occurrence of leakage in the in-reactor tube
Sept. '84	Withdrawl in-ractor tube with leak out of the reactor
Aug. '86	Installation of the new in-ractor tube

Table 2.2 OWL-2 Engineering Data

In-reactor tube position in reactor core	K.L-3.4
Thermal neutron flux	
Peak	5.4×10^{13} n/cm^2.s
Average	4.1×10^{13} n/cm^2.s
Fast neutron flux ($E > 1$ MeV)	
Peak	5.5×10^{12} n/cm^2.s
Average	4.2×10^{12} n/cm^2.s
Gamma heating (max)	0.5 W/g
Coolant	Light water
Coolant flow rate	1100 kg/min max
Operating temperature	
P-mode operation	270°C
B-mode operation	285°C
Operating pressure	7.3 MPa (73 kg/cm^{2}G)
Operating modes	Boiling water cooling and Pressurized water cooling
Steam quality	20 wt % max
Heat removal capacity	850 kW max
Effective in-reactor test section dimensions	117.8 mm in diameter, 750 mm high

Table 2.3 Functions of OWL-2 Primary Loop Interlock System

Deterioration of operating conditions	Safety function initiated at particular deterioration level														
	Alarm	Strap-on Type Heaters Next to In-reactor Tube Inlet Turned Off	Heater in Boiler	Steam Supper Heater Turned Off	Steam Heater on Boiler Outlet Turned Off	Fully Open flow Control Valve on In-reactor Inlet	Reduce Opening of Spray Control Valve	Close Primary Purification Flow Control Valve	Control Valve for Feeding Water to Boiler	Control Valve for Steam Flow from Boiler	Close In-reactor Tube Thermal Insulating Jacket Isolation Valve	Feed Pumps	Stop Loop Pumps	Reactor Setback	Reactor Slow Scram
Low Flow to Test Section	1	2		2		2								3	4
High Temperature at Test Section Inlet	1	2													
High Temperature at Test Section Outlet	1	2		2		2								3	4
Low Steam Flow	1			2											
High Temperature of Heat Removel Pumps	1												2		
High Temperature of Loop Pumps	1												2		
Low Steam Flow from Boilers	1				2										
High Condenser Pressure	1			2										3	4
Low Condenser Pressure	1	2	2 on	2			2			2 close				3	4
Low Surge Tank Level	1	2		2								2 on		3	4
High Surge Tank Level	1											2 off			
Low Boiler Level	1	2	2 off	2					2 open					3	
High Boiler level	1								2 close						
High Temperature at Primary Purification Circuit Inlet	1							2							
Low Flow of Utility Cooling Loop	1	2		2				2						3	
Low Pressure of Compressed Air	1	2		2										3	4
Low Boiler Pressure	1	2	2 on	2										3	
High Boiler Pressure	1		2 off												
Low Flow in Secondary Loop	1	2		2										3	
Great Difference in Temperatures between Steam and Water in Boiler	1		1 off												
High Pressure in In-reactor Tube Thermal Insulating Jacket	1										1				
High Heater Temperature	1	1	1 off	1	1										
Loop Pump Overload	1												1		
Heat Removal Pump Overload	1												1		
Commercial Power Failure	1	1	1 off	1	1										1
On-site Power Failure	1	1	1 off	1	1										1

Deterioation of operating conditions given in rows bring items in columns into function at a level shown by a number in the frame they meet.

Table 2.4 Functions of OWL-2 Secondary Loop Interlock System

Safety function initiated at particular deterioration level

Deterioration of operating conditions	Alarm	Secondary Surge Tank Heater	Fully Open Flow Control Valve on Secondary Circuit of Secondary Cooler	Close Cubicle Air Supply Valve	Secondary Make-up Pump	Secondary Loop Pumps Turned Off
High Pressure of Secondary Surge Tank	1	1 off				
Low Pressure of Secondary Surge Tank	1	2 on				
High Water Level in Secondary Surge Tank	1				2 off	
Low Water Level in Secondary Surge Tank	1				2 on	
High Temperature of Heater on Secondary Surge Tank	1	1 off				
Secondary Loop Pumps Overload	1					1
High Temperature of Secondary Loop Pumps	1					2
High Temperature of Primary Circuit Outlet of Secondary Cooler	1		2 open			
Small Pressure difference between inside and outside of Cubicle	1			1		
Commercial Power Failure	1	1 off				
On-site Power failure	1	1 off				

Deterioation of operating conditions given in rows bring items in columns into function at a level shown by a number in the frame they meet.

Table 2.5 The Specimens Irradiated in OWL-2 (1/4)

1. 71LM-10J (material specimen)

Dummy fuel rod

Material: SUS 304

Outside diameter: 16.25 mm

Operating conditions: Operating mode; pressurized water, Pressure at in-reactor tube inlet; 7.3 MPa (73 kg/cm^2G), In-reactor tube inlet temperature; 270°C

Operation period: No.13

Remarks: 71LM-10J was loaded in the OWL-2 as a material specimen to determine the performance of the OWL-2.

2. 71LF-9J (fuel specimen)

10 fuel rods enriched to 2%, 4 fuel rods enriched to 5%, 13 fuel rods enriched to 10% and one dummy fuel rod

Fuel form: UO_2 pellet

Fuel outside diameter: 16.25 mm

Operating conditions: Operating mode; pressurized water and boiling water, In-reactor tube inlet pressure; 7.3 MPa (73 kg/cm^2G), In-reactor tube outlet temperature; 285°C, Steam quality; 8 %

Operation periods: No.14, 15

Maximum heating rate: 400 kW

Maximum linear heating rate: 415 W/cm

Temperature at the center of a fuel: 1600°C

Remarks: 71LF-9J was irradiated in OWL-2 as a fuel specimen to determine the performance of OWL-2

3. 71LM-11A (material specimen)

51 tensile test pieces, 170 impact test pieces, 16 hardness test pieces, 6 corrosion test pieces and 9 WOL test pieces

Material: ASTM-A5533B

Operating Conditions: Operating mode; pressurized water, In-reactor tube inlet pressure; 7.0 MPa (70 kg/cm^2G), In-reactor tube inlet temperature; 270°C

(Continued on the following page)

Table 2.5 The Specimens Irradiated in OWL-2 (Continued) (2/4)

Operation period: No.16

Remarks: 71LM-11A was the same material as intened for reactor vessel.

4. 70LF-8P-I (fuel and material specimens)

Fuel specimen: 18 fuel rods enriched to 2%, 18 fuel rods enriched to 5%, Form; UO_2 pellet Outside diameters; 14.72 mm and 9.74 mm

Material specimen: 48 tensile test piece, material; Zr-2.5Nb.

Operating conditions: Operating mode; pressurized water and boiling water, In-reactor tube inlet pressure; 7.0 MPa (70 kg/cm^2G), In-reactor tube outlet temperature; 285°C

Steam quality: less than 15 %

Operating periods: No.17 and 18

Maximum heating rate: 350 kW

Maximum linear heating rate: 344 W/cm

Temperature at the center of a fuel: 1360°C

Remarks: 70LF-8P-I was the same as No.4 special fuel assembly for ATR (fuel and material).

5. 70LF-8P-II (fuel and material specimen)

Fuel specimen: the same as 70LF-8P-I

Material specimen: 28 tensile test pieces and 20 corrosion test pieces

Operating conditions: the same as for 70LF-8P-I

Operating periods: No.20, 21, 22 and 24

Remarks: Identification Code 70LF-8P-II was used in place of 70LF-8P-I after the material specimens of 70LF-8P-I were replaced with the new ones.

6. 70LF-8P-III (fuel and material specimen)

Fuel specimen: the same as 70LF-8P-I

Material specimens: 2 burst test pieces, 16 tensile test pieces, 6 bending test pieces and 8 corrosion test pieces

(Continued on the following page)

Table 2.5 The Specimns Irradiated in OWL-2 (Continued) (3/4)

Operating conditions: the same as for 70LF-8P-I

Operating periods: No.32, 33 and 34

Remarks: Identification Code 70LF-8P-III was used in place of 70LF-8P-II after the material specimens of 70LF-8P-II were replaced with the new ones.

7. 74LF-12P (fuel and material specimen)

Fuel specimens: 18 fuel rods enriched to 3 %, 18 fuel rods enriched to 5 %, form; UO_2 pellet, outside diameters; 14.72 mm and 9.7 mm

Material specimens: 2 burst test pieces, 16 tensile test pieces, 6 bending test pieces and 8 corrosion test pieces.

Operating conditions: the same as for 70LF-8P-I

Operating periods: No.35, 36, 37 and 43

Maximum heating rate: 400 kW

Maximum linear heating rate: 384 W/cm

Maximum temperature at fuel center: 1650°C

Remarks: 74LF-12P was the same as the 4th special fuel assembly for ATR.

8. 76LM-12J (material specimen)

Specimens: 3 test pieces for instrumentation, 6 irradiation growth test pieces, 27 external pressure creep test pieces and 4 internal pressure creep test pieces.

Operating conditions: operating mode; pressurized water cooling, in-reactor inlet pressure; 7.3 MPa (73 kg/cm^2G), in-reactor inlet temperature; 270°C

Operating periods: No.38, 39, 40, 45, 46, 47, 48, 49, 50, 51, 52

Remarks: 76LM-12J was the same as fuel cladding for light water reactors.

9. 79LF-14A (fuel specimen)

Specimens: 8 fuel rods for BWRs, 18 fuel rods for PWRs and 10 dummy fuel rods

(Continued on the following page)

Table 2.5 The Specimens Irradiated in OWL-2 (Continued) (4/4)

Operating conditions: operating mode; Pressurized water cooling, pressure at in-reactor tube inlet; 7.3 MPa (73 kg/cm^2G), in-reactor tube inlet temperature; 260°C

Operating periods: No.53, 54, 55, 56, 57, 58, 59, 60, 61, 76, 77, 78, 79, 80, 81, 82, 83, 84, 85, 86, 87, 88, 89, 90

Maximum heating rate: 153 kW

Maximum linear heating rate: 226 W/cm

Maximum temperature at fuel center: 980°C

Remarks: after being preliminarily irradiated in the OWL-2, these specimens are subjected to ramp test in the BOCA facility.

Table 3.1 The Reports on Results of Tests Conducted in the OWL-2 (1/2)

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Table 3.1 The Reports on Results of Tests Conducted in the OWL-2 (2/2) (Continued)

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18. Yoshiaki IKESHIMA, et al., "Fabrication of the OWL-2 In-reactor Tube," Available to JAERI personnel only, Nov. 1987.
19. Masanobu USUI, et al., "Investigation of Cruds Deposited on Piping of the Primary System of a High-temperature and High-Pressure Loop - Preliminary Study," Available to JAERI personnel only, May 1988.
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21. Hitachi Co., Ltd., "Post-irradiation Test of Special Fuel for ATR at NFD," PNC ZJ 302 81-14 (1), (2), Sept. 1981.
22. Hitachi co., Ltd., "Post-irradiation Test of Special Fuel for ATR Vol. 1," PNC ZJ 302 84-02 (1), (2), Mar. 1984.
23. Advanced Thermal Reactor Development Department, PNC, "Summary of Post-Irradiation Test of Fuel for ATR at JMTR," Dec. 1977.

Table 3.2 Engineering Data of the Flowmeters for Process and In-core Instrumentation

		Process Use	In-core Use
1. Interior			
Size		1 1/2 inches	3/4 inches
The number of blade		6	4
Flow rate range		40-400 kg/min (at 250°C)	32-60 kg/min (at 45°C)
Temperature range		20-300°C	40-300°C
Pressure		3.0-7.6 MPa (30-77 kg/cm^2G)	71 MPa (71 kg/cm^2G)
Coolant		Demineralized water	Demineralized water
Material	Shaft	Cemented carbide	Cemented carbide
	Bearing	Cemented carbide	Cemented carbide
	Thrust	17-4 pH	
2. Pickup coil			
Structure		Externally mounted design pressure: atmospheric	Submerged in coolant with being watertight and pressure resistant
Signal generation method		Magnet method	Magnet method
Size		—	20 mm long x 17 mm wide x 5.5 mm deep
Magnet material		Alnico	Lanthanet

Table 3.3 Nondestructive Examinations Performed and Their Results (1/2)

1. Appearance Inspection	Items inspected	Results
a. Visual inspection	Deformation and damage.	A few dents found in area around the cracks.
b. Microscopic examination	Existence of pittings.	Large pittings in dents and small ones in region without dents.
c. Dimension measurement	Deformation and decrease in wall thickness.	No deformation, no decrease in thickness.
2. Nondestructive examination		
a. Radiographic examination	Location and size of cracks.	Cracks run in only one direction.
b. Liquid peretrant	The same as the above.	The same as the above.
3. Mechanical tests		
a. Hardness test	Hardness measurement of base material and welded region around the crack.	Heat-affected area slightly harder than other area.
b. Tensile test	Tensile strength and yield strength.	Nearly the same as those recorded in specification sheet.
3. Metallurgical examination		
a. Cross-sectional macroscopic examination	Check area around the crack.	—
b. Cross-sectional microscopic examination	Form of the crack and the degree of sensitization.	Representative transgranular stress corrosion cracking.
c. Ferrite measurement	Destribution of ferrite	No defect
d. Fractgraphy	Fracture patter of cracks.	Fracture with TGSCC characteristics.

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Table 3.3 Nondestructive Examinations Performed and Their Results (2/2)
(Continued)

e. Compositions	Amounts of compositions.	Nearly same as those shown in qualification sheet.
f. Grain size measurement	Grain size.	No defect.
4. Stress measurement		
a. Reaction stress	Extent of reaction stress	Slight.
b. Residual stress	Residual stress produced by welding.	Residual stree was likely to be the cause of the TGSCC.
5. Sensitization measurement with EPR method	Extent of sensitization.	So sensitized to produce TGSCC.

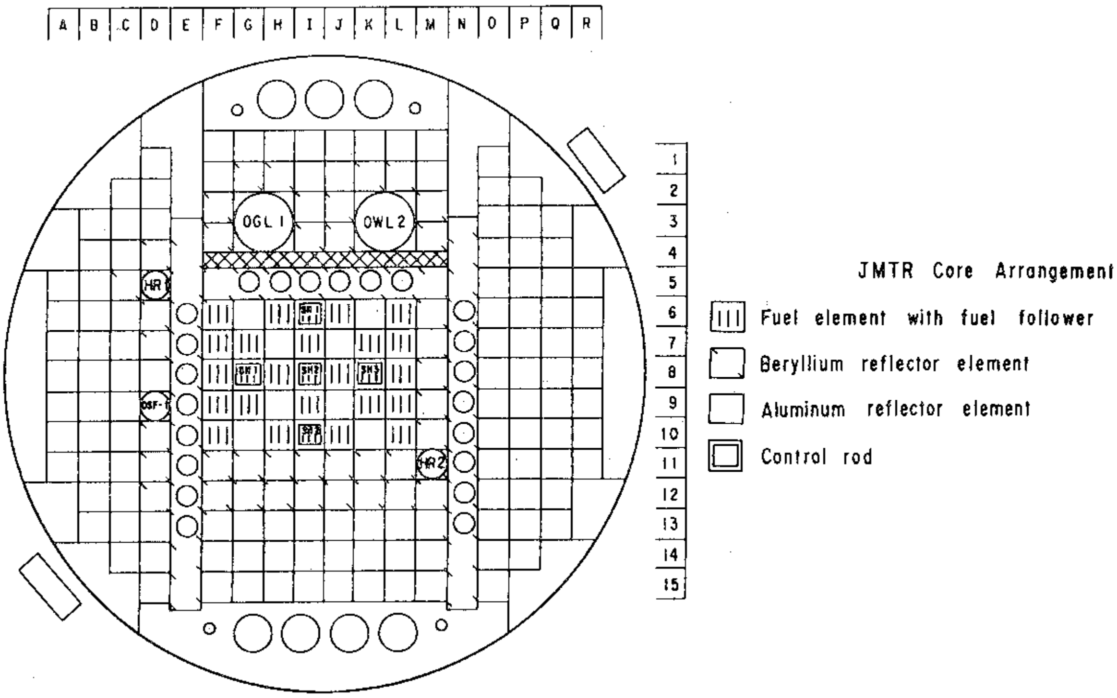


Fig. 1.1 Reactor core arrangement

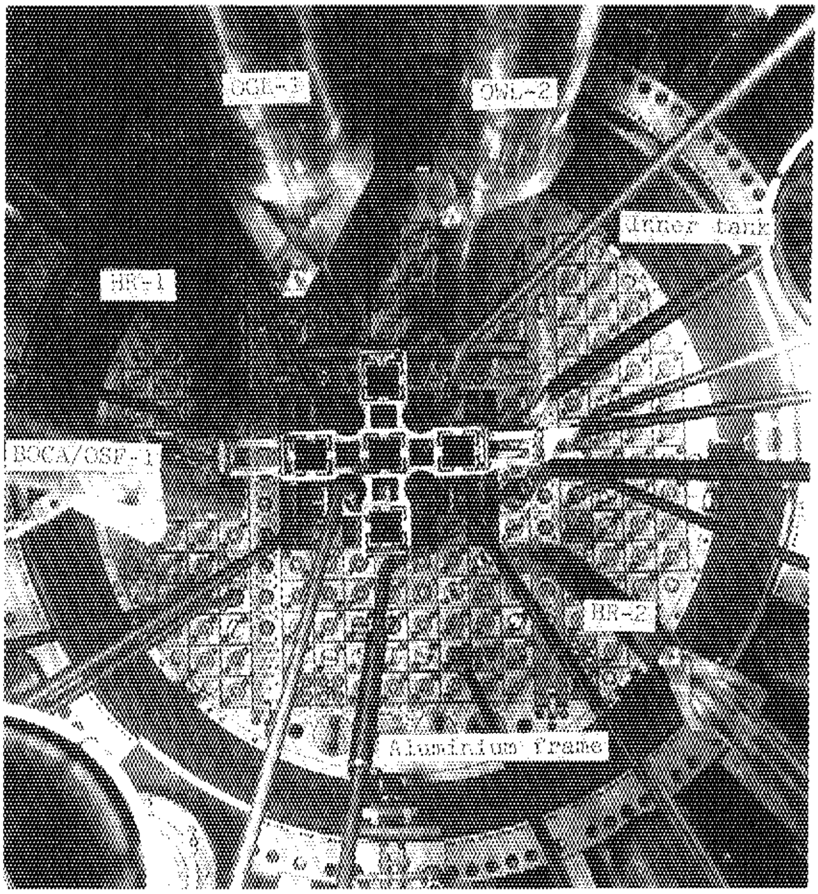


Fig. 1.2 Looking into the core

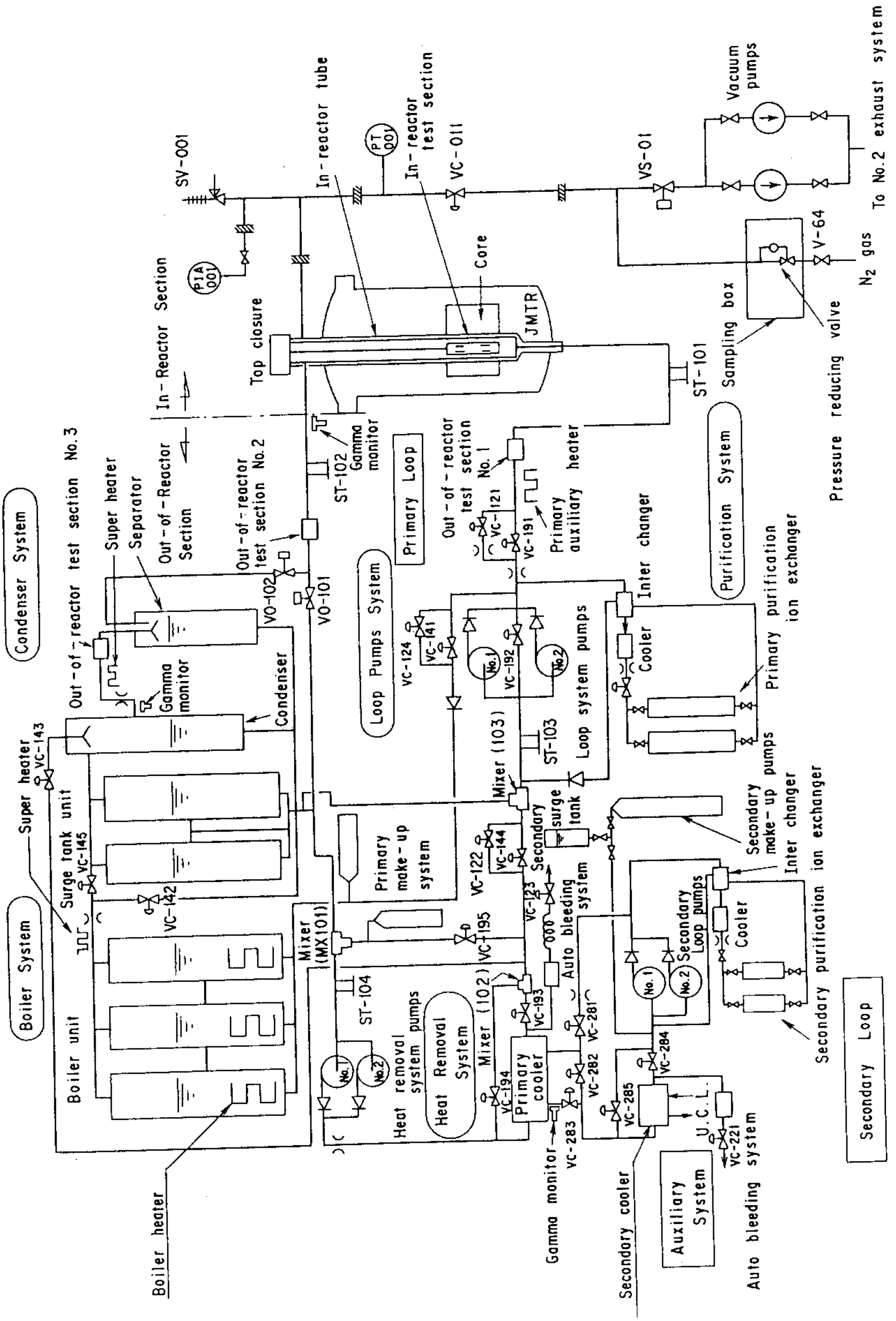


Fig. 2.1 OWL-2 simplified flow diagram

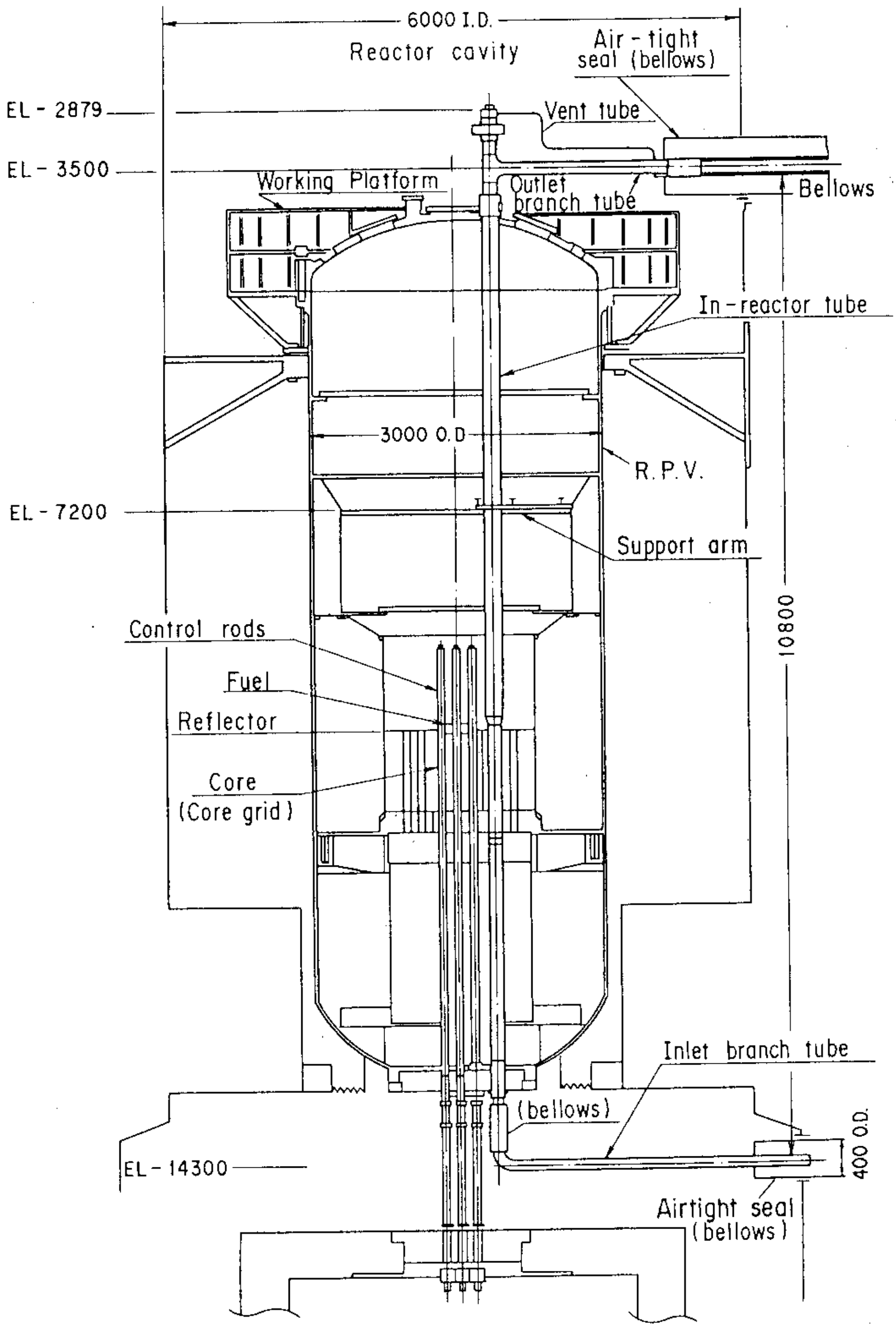


Fig. 2.2 Arrangement OWL-2 in-reactor tube in the reactor

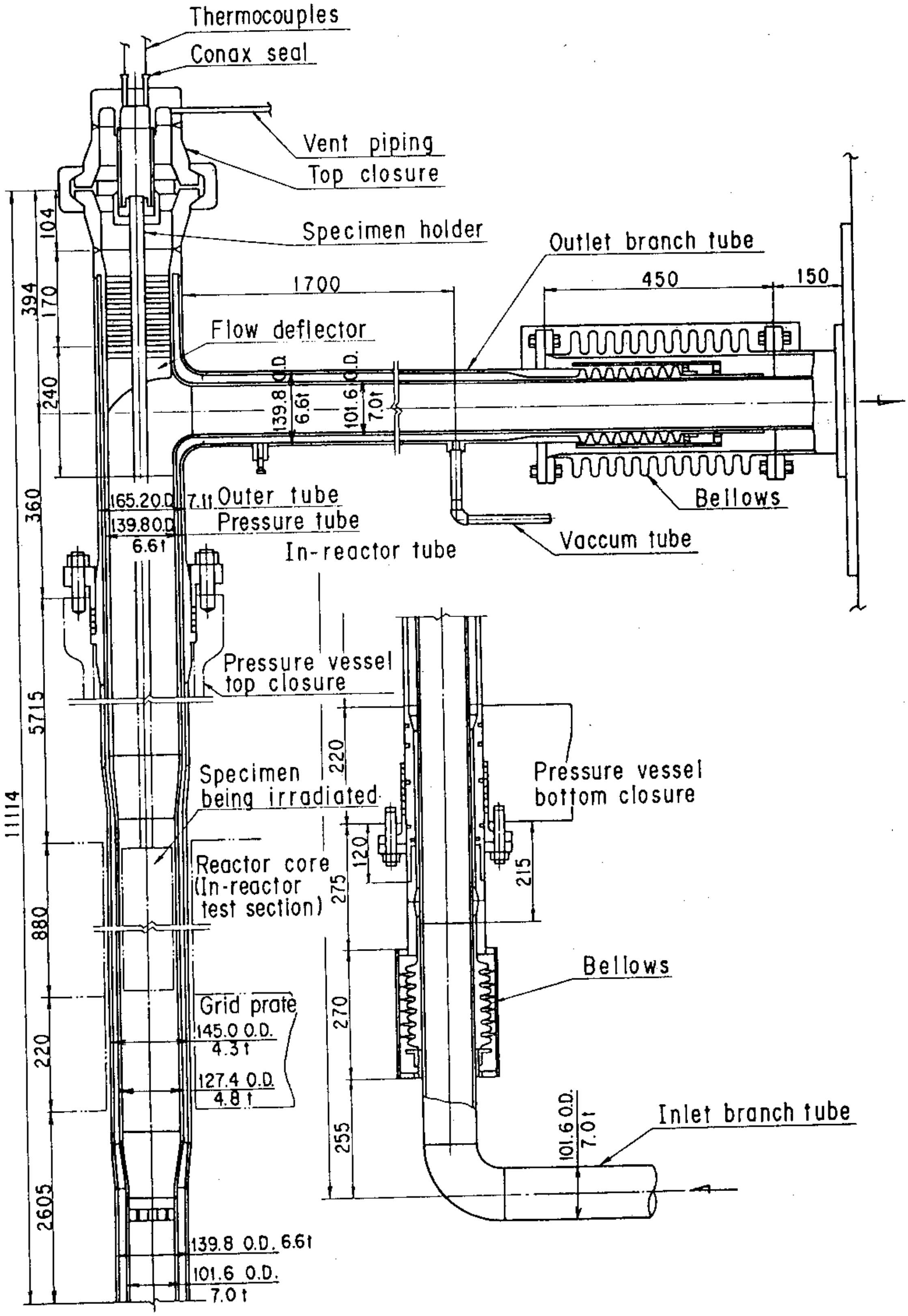
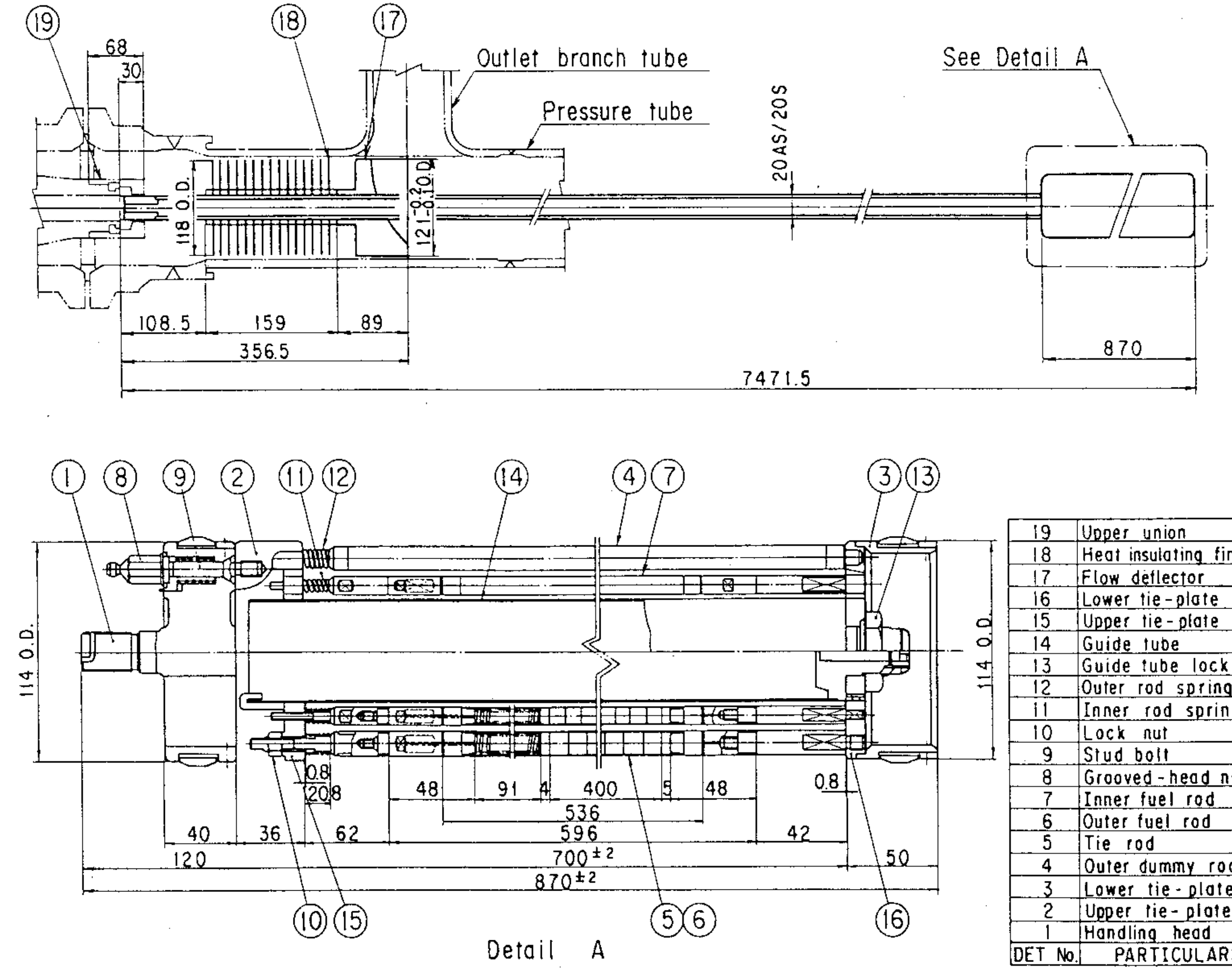
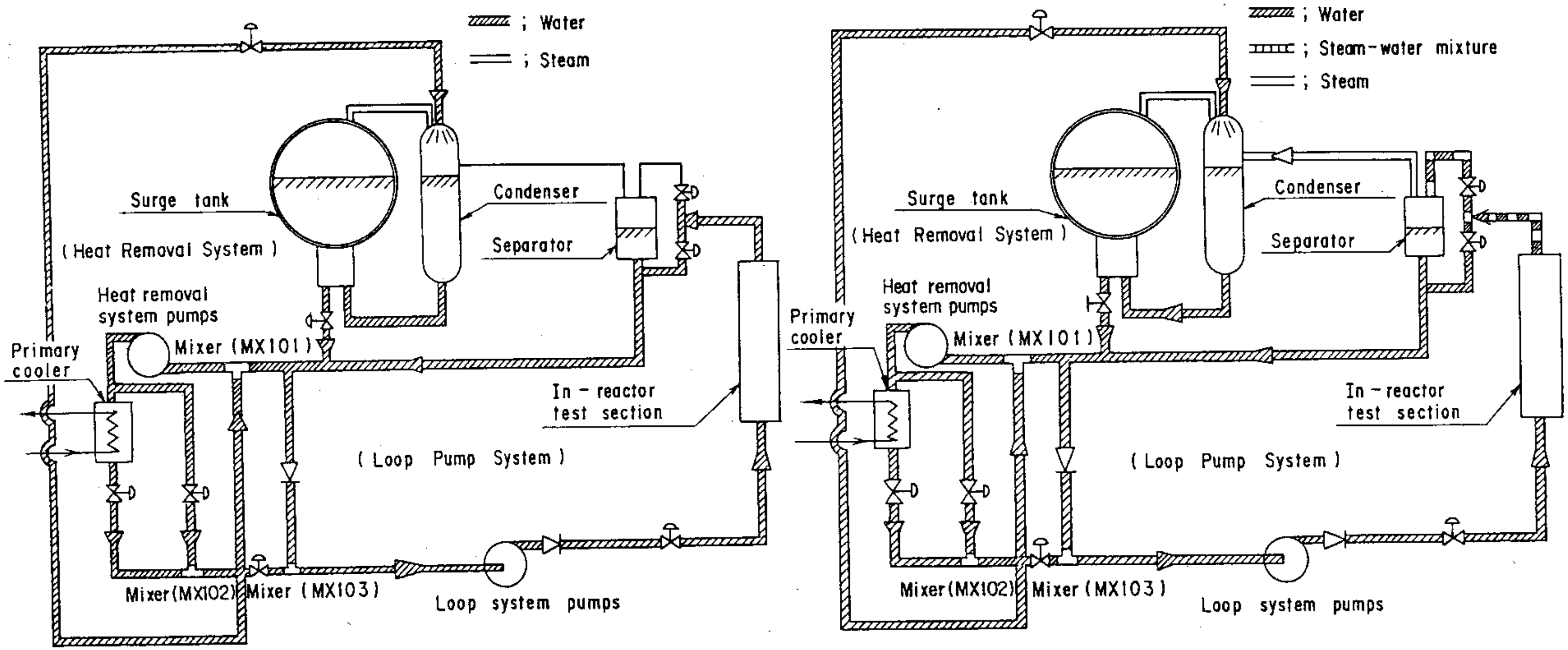


Fig. 2.3 Detailed axial sectional view of the OWL-2 in-reactor tube



DET No.	PARTICULARS	MATERIAL
19	Upper union	SUS 304
18	Heat insulating fin	SUS 304
17	Flow deflector	SUS 304
16	Lower tie-plate	SCS 13
15	Upper tie-plate	SUC 13
14	Guide tube	Zry-2
13	Guide tube lock nut	SUS 304
12	Outer rod spring	Inconel-X
11	Inner rod spring	Inconel-X
10	Lock nut	SUS 304
9	Stud bolt	SUS 316
8	Grooved-head nut	SUS 304
7	Inner fuel rod	
6	Outer fuel rod	
5	Tie rod	
4	Outer dummy rod	
3	Lower tie-plate	
2	Upper tie-plate	SCS 13
1	Handling head	SUS 304

Fig. 2.4 Specimen assembly for irradiation in OWL-2



Coolant path in pressurized operation mode

Coolant path in boiling operation mode

Fig. 2.5 Simplified flow diagrams of the pressurized and boiling operation modes

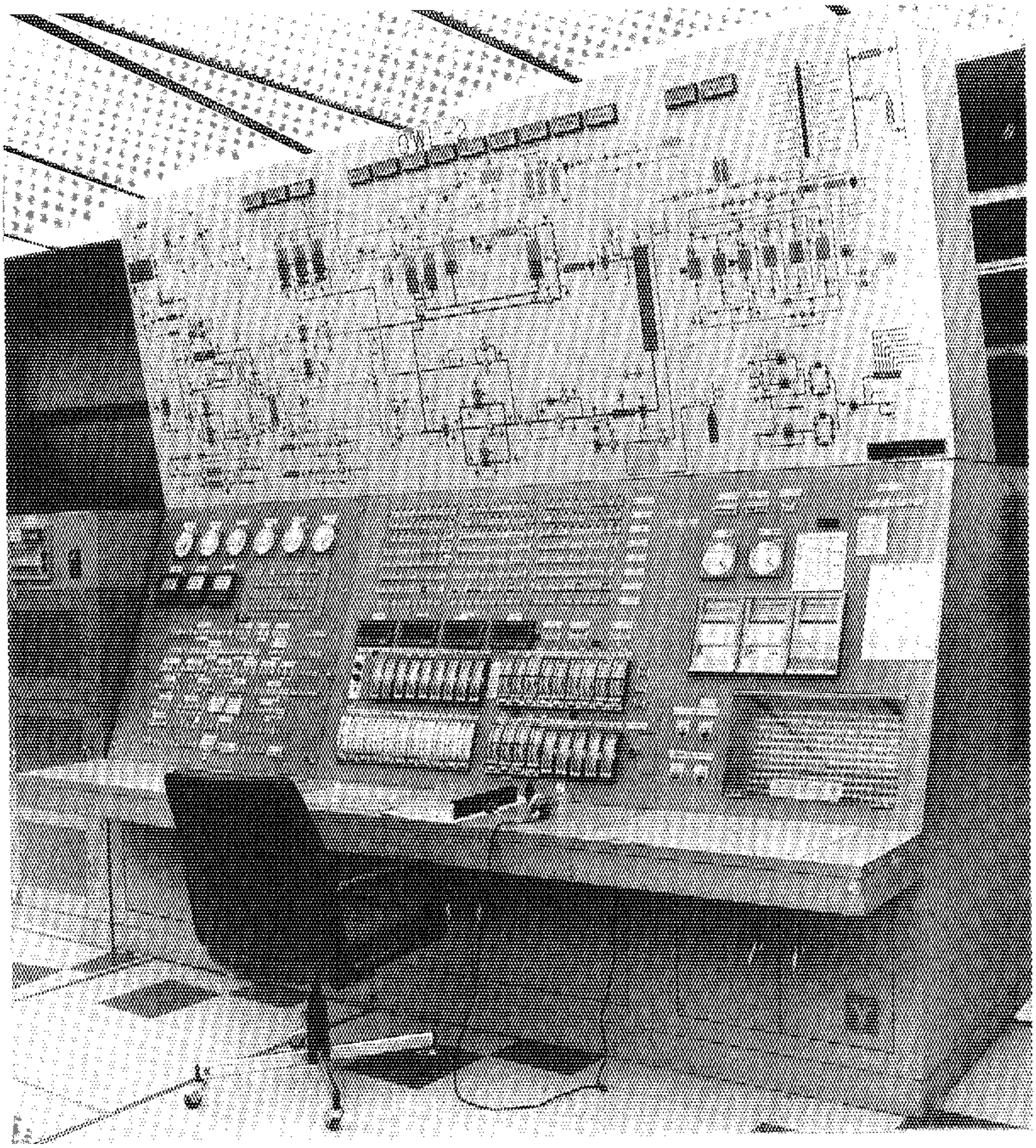


Fig. 2.6 The control panel of OWL-2

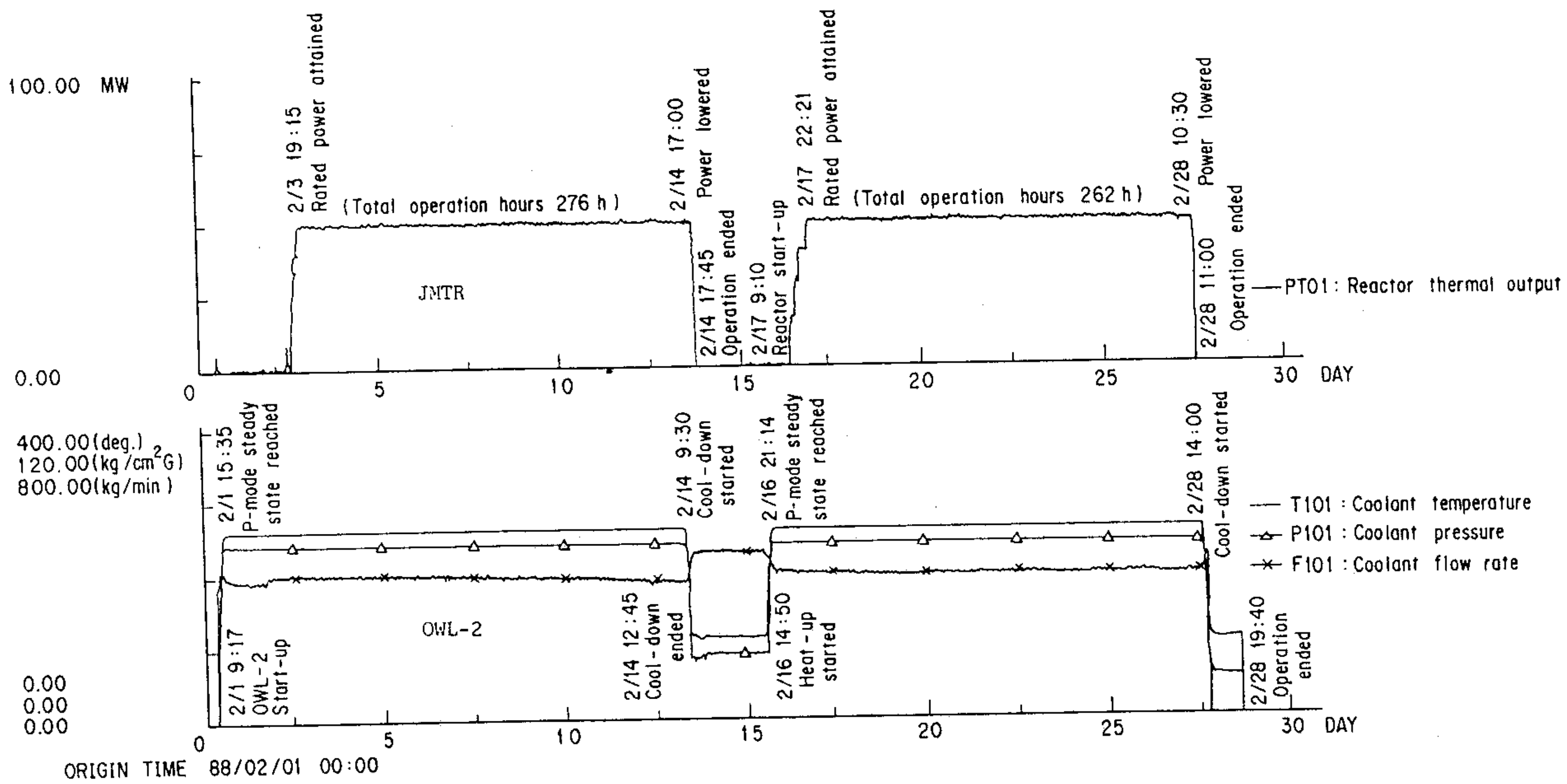


Fig. 2.7 Operation log of No.80 JMTR operation period

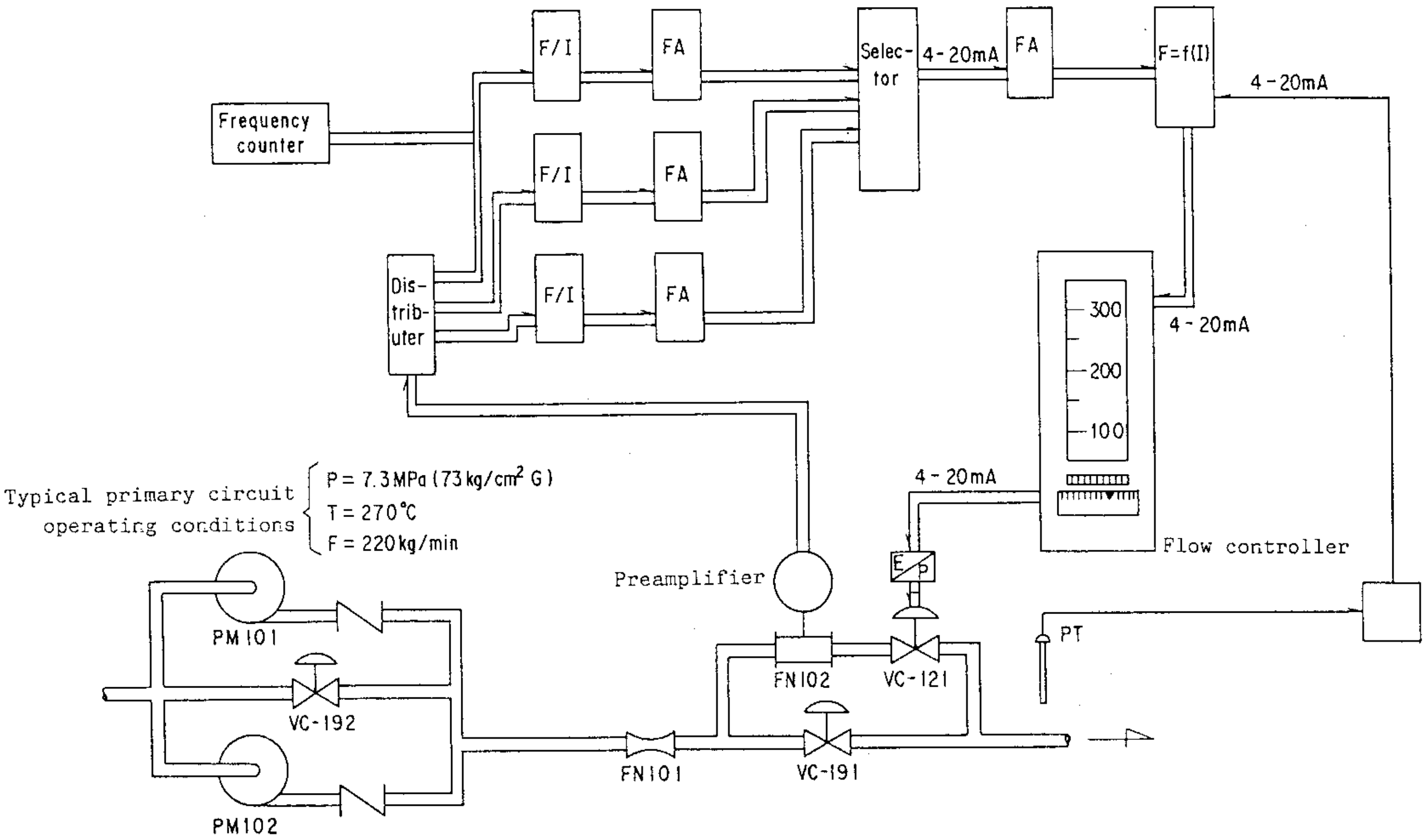


Fig. 3.1 Process control turbine flowmeter organization

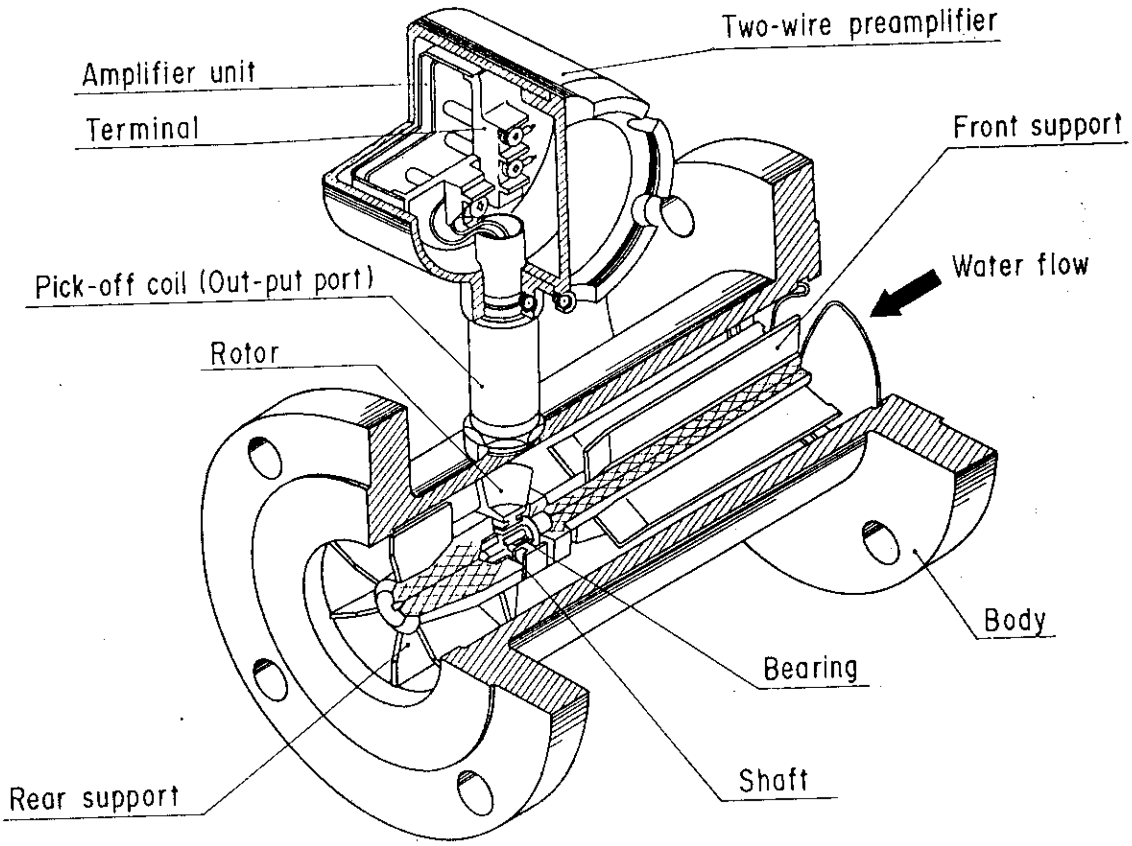


Fig. 3.2 Internal construction of the process control turbine flowmeter

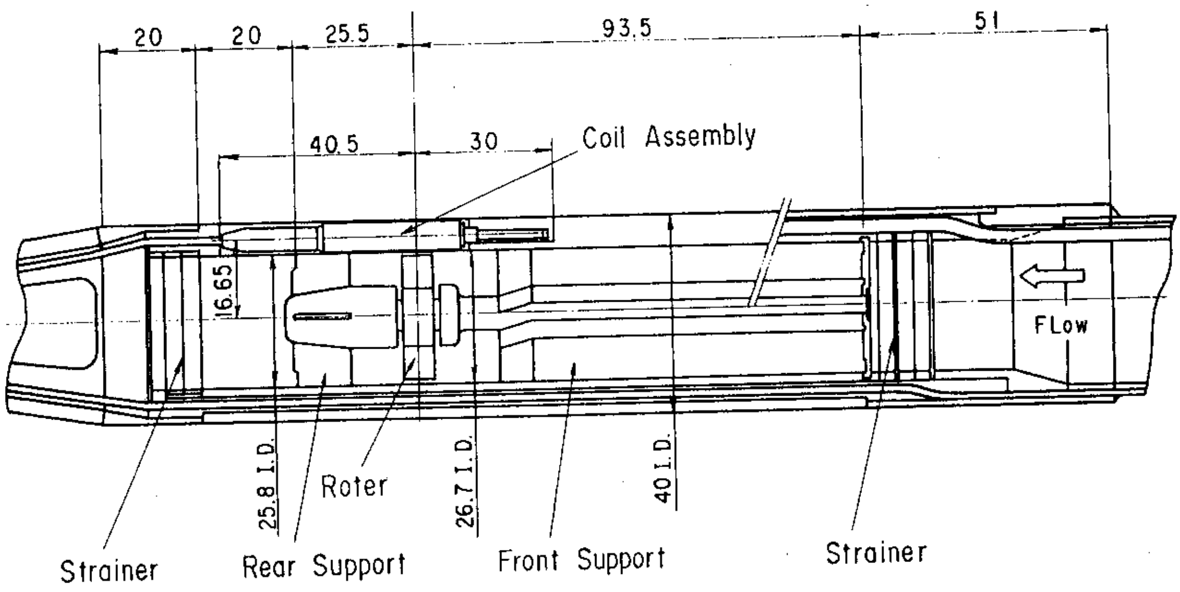


Fig. 3.3 Axial cross-sectional view of the in-core turbine flowmeter

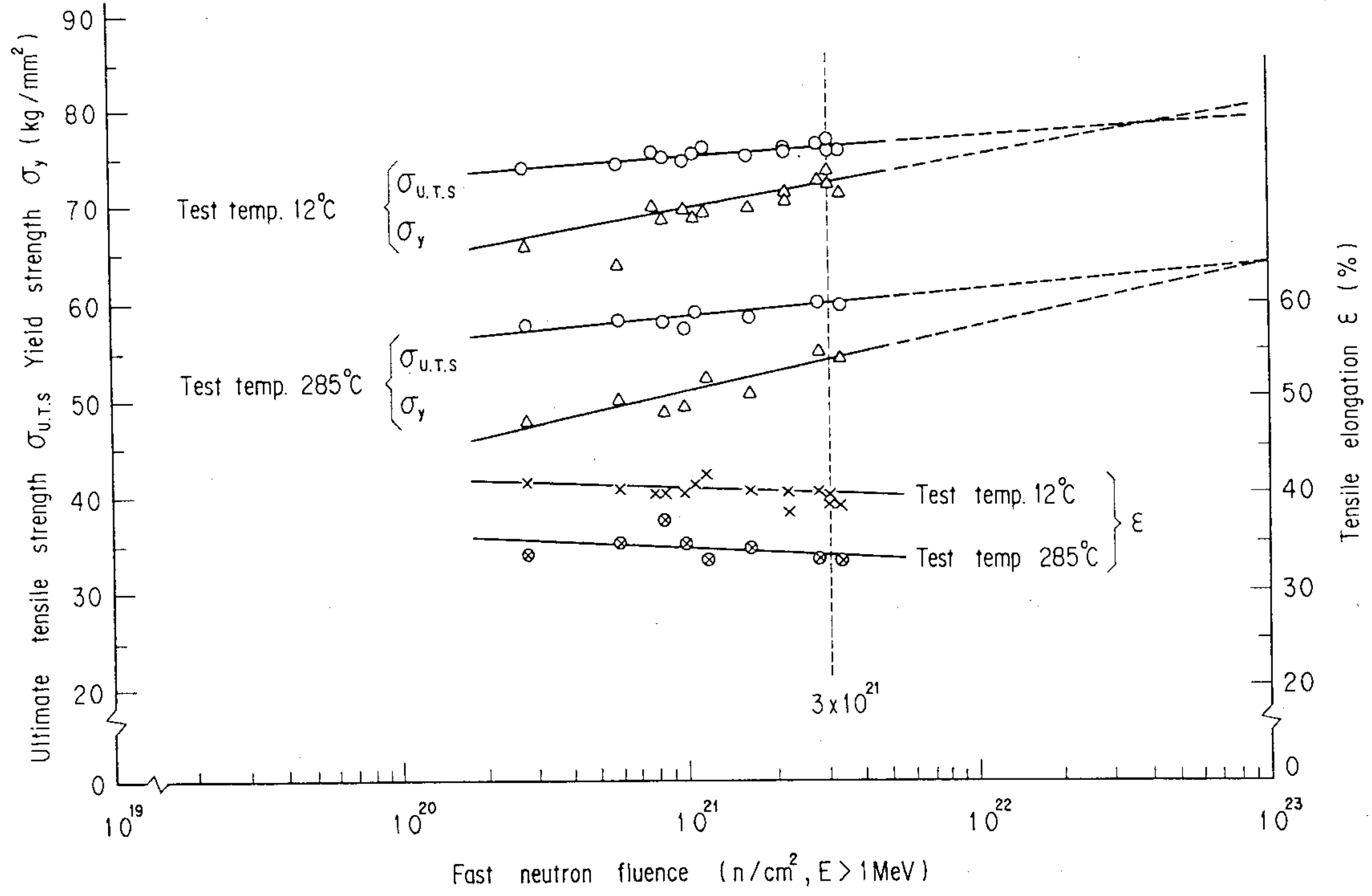


Fig. 3.4 Irradiation effects of fast neutron on mechanical properties

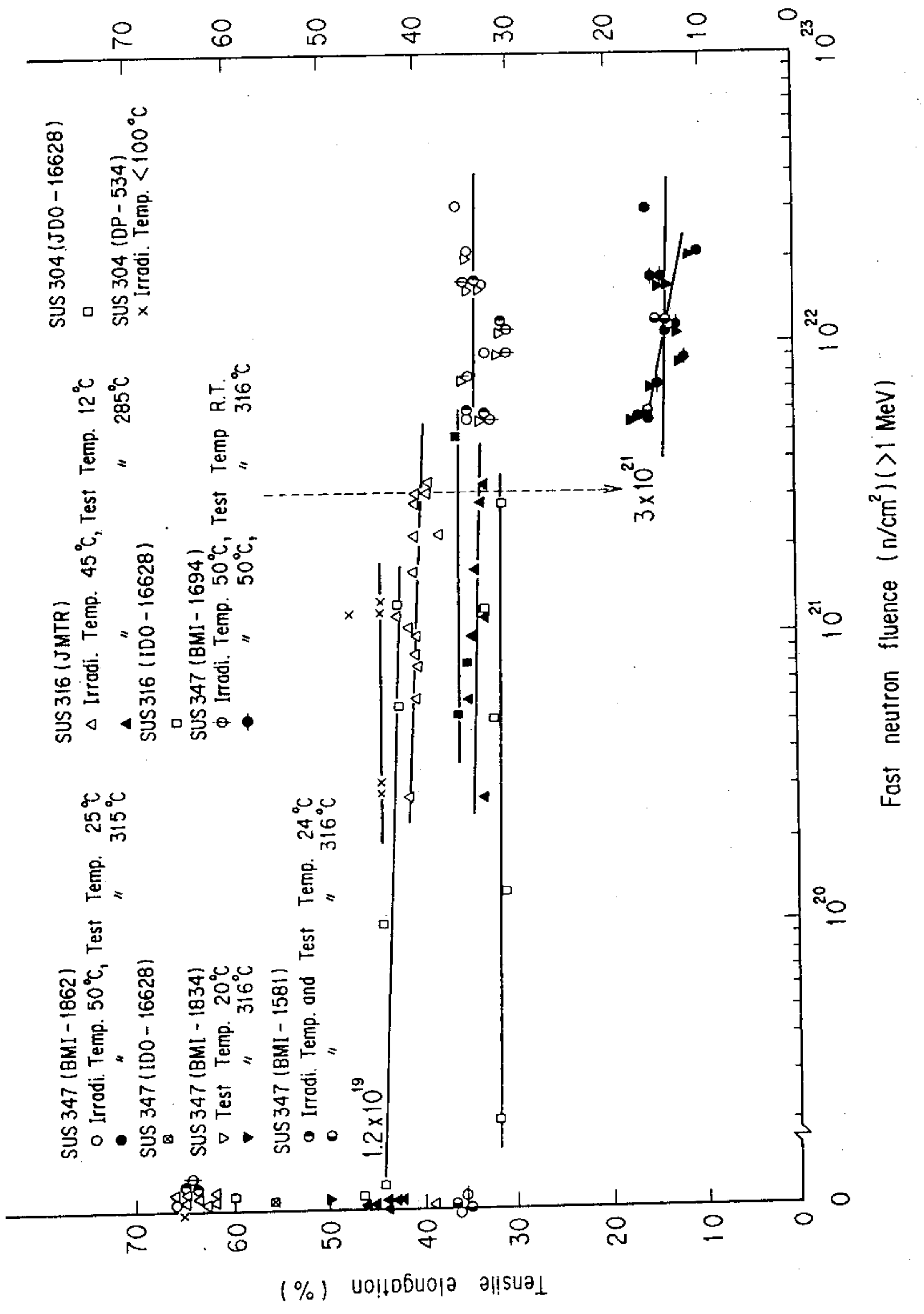


Fig. 3.5 Irradiation effects of fast neutron on tensile elongation

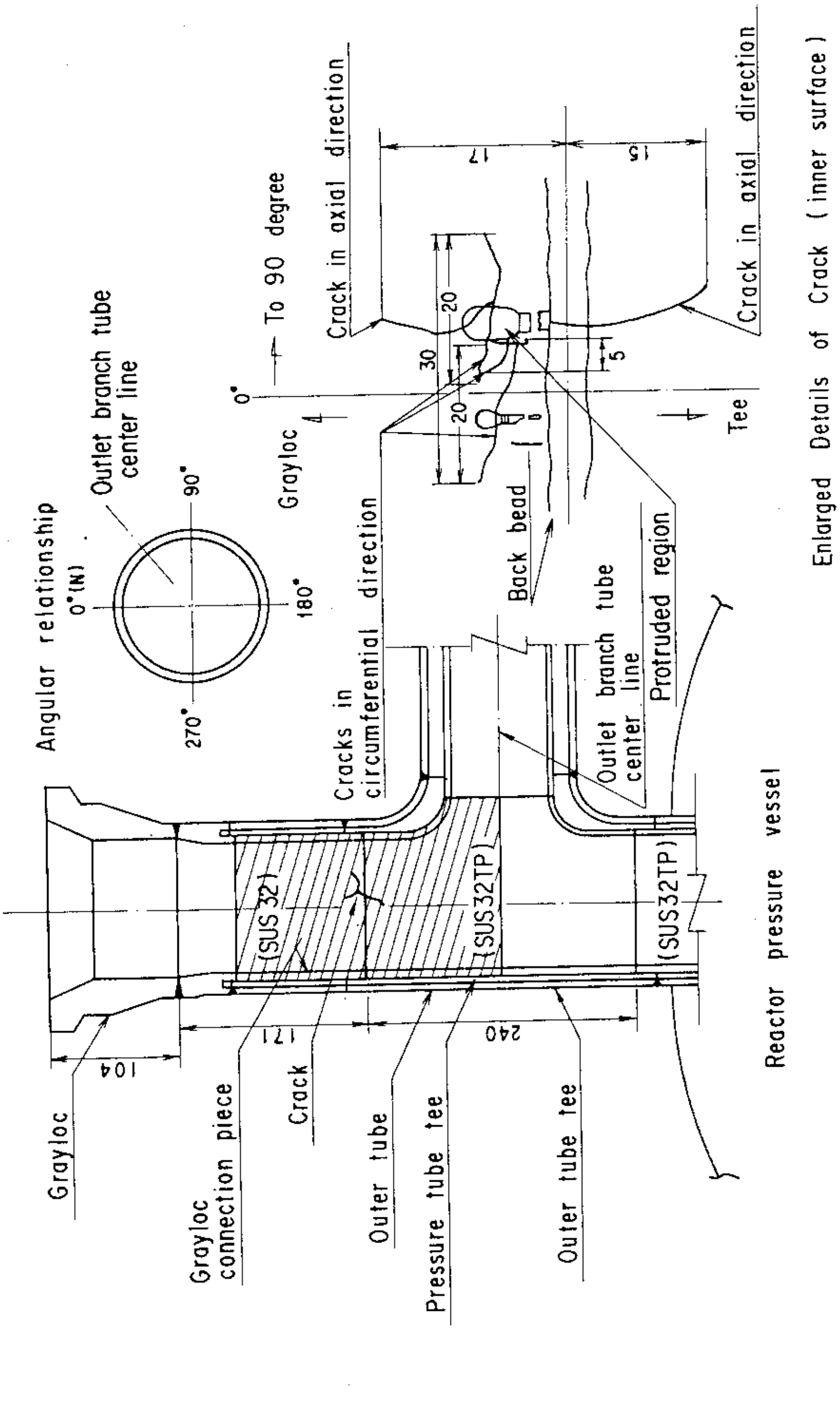


Fig. 3.6 Schematic representation of cracks in the in-reactor tube