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CONCEPTUAL DESIGN OF LARGE SCALE TEST FACILITY (LSTF) OF ROSA-IV PROGRAM FOR PWR SMALL BREAK LOCA INTEGRAL EXPERIMENT

December 1981

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Conceptual Design of Large Scale Test Facility (LSTF) of ROSA-IV Program for PWR Small Break LOCA Integral Experiment

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There has been an extensive reorientation of the light-water reactor (LWR) safety research program since the TMI-2 accident and the emphasis is now on small break loss-of-coolant accidents (SBLOCAs) and transients. The Japan Atomic Energy Research Institute (JAERI) has instituted the Rig of Safety Assessment Number 4 (ROSA-IV) program for the purpose of studying SBLOCAs and transients. JAERI is constructing the Two-Phase Test Facility (TPTF) for the separate effect tests and going to construct the Large Scale Test Facility (LSTF) for the system effect tests.

The design philosophy and the primary specifications of the LSTF system are described in this report.

Keywords : ROSA-IV, LSTF, PWR, LOCA, Small-Break LOCA, Two-Phase Flow, Heat Transfer, Transients, System Effect Test Conceptual Design, Specifications

* On leave from Ishikawajima-Harima Heavy Industries Co., Ltd.

** On leave from EG&G Idaho, Inc.

PWRの小破断LOCA総合実験装置ROSA-Ⅳ LSTFの概念設計

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（1981年11月25日受理）

TMI-2号炉事故を契機として軽水炉の安全性研究計画が見直され，小口径配管破断冷却材喪失事故（SBLOCA）および異常過渡事象に関する研究の重要性が指摘された。これを受けて，日本原子力研究所ではROSA-Ⅳ計画を開始し，現在分離効果実験用のTPTF（Two-Phase Test Facility）の建設ならびにシステム効果実験用のLSTF（Large Scale Test Facility）の設計を行っている。本報告書はLSTFの設計方針とその主要仕様を紹介したものである。

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1. Introduction

A nuclear power plant accident that occured at Unit 2 of the Three Mile Island power plant (TMI-2)[1), 2)] in U.S.A. on March 28, 1979 was initiated by the closure of all the inlet and outlet valves on the condensate polisher line tripping the feedwater pumps. The valve closure was due to a failure of the valve control system which was caused by water from the condensate polisher flowing through the station air system and into the instrument air system. The emergency feedwater pumps started immediately after the main feedwater pump trip, however, the emergency feedwater was not supplied to the steam generators because of closure of valves at the outlet of the emergency feedwater pumps. The loss-of-feedwater accident became more serious with a small break loss-of-coolant accident (SBA) resulting from the stuck-open relief valve on the pressurizer and the inadequate operator action reducing the high pressure coolant injection (HPCI) and make-up flow rates.

The accident indicates that a minor transient can be developed to a serious accident by transients by equipment failures and/or inadequate operator actions.

There has been an extensive reorientation of the light-water-reactor (LWR) safety research program since the TMI-2 accident and the emphasis is now on small breaks and transients[3)].

Recent examinations of SBAs in large pressurized water reactors (LPWRs) have indicated a need for increased understanding of the physical phenomena and various plant parameters which affect the performance of LPWRs during a SBA. The Japan Atomic Energy Research Institute (JAERI) has instituted the Rig of Safety Assessment Number 4 (ROSA IV) Program for the purpose of studying these phenomena and parameters and of developing computer models for analyzing and predicting the results of SBAs in LPWRs. As part of the ROSA IV program, JAERI is constructing a Large Scale Integral Test Facility (LSTF) to provide experimental data for the study of SBA behavior and for computer model assessment and verification. Results from LSTF will be used in combination with those from the ROSA IV separate effects tests to be performed at the Two-Phase Test Facility (TPTF) to develop and verify a SBA computer code model.

2. Scope and Objectives

2.1 Scope

2.1.1 ROSA IV Large Scale Test Facility (LSTF)

The ROSA IV LSTF is a large scale, integral test facility for the study of overall system behavior during a Small Break Accidnet (SBA) or anticipated transients. LSTF is designed to model explicitly the major components of the reactor, reactor coolant and emergency core cooling systems. LSTF will also model, to the extent required, those other plant systems which affect SBA performance by their interaction with the reactor coolant and emergency core cooling systems, e.g., the feedwater, condensate and main steam systems. LSTF will also provide the service systems required for proper facility operation, e.g., component cooling water, instrument air, etc. As well as a process and experimental instrumentation and data acquisition system to measure and record test parameters, facility control and process information, and facility test data.

The LSTF simulates 3423 MWt PWR with a 17 × 17 fuel bundle design. The overall scale factor for LSTF will be 1/48 full-size (volumetric scaling) with component elevations maintained at full-scale (1/1 elevation scaling) to the maximum extent practicable. Core power is set to provide the same power input per unit volume as the full-scale plant for decay core powers below approximately 14% of full power. The loop design for LSTF will be two symmetric loops, each 2/48 full-scale volumetric scaling, and 207 mm of piping diameter. Break size will be restricted to medium break loss-of-coolant accident sizes, i.e., less than 10% break size.

Systems to be provided as part of the LSTF are:

(1) Core Simulator

(2) Reactor Vessel and Internals

(3) Reactor Coolant System

  a. System Piping
  b. Reactor Coolant Pumps
  c. Pressurizer
  d. Steam Generator

(4) Emergency Core Cooling (ECC) and Residual Heat Removal Systems

a. High Pressure Injection System
b. Accumulators
c. Low Pressure Injection System
d. Residual Heat Removal System

(5) Feedwater, Condensate and Steam System

a. Feedwater and Condensate Systems
b. Emergency Feedwater System
c. Main Steam System

(6) Simulated Containment System (Break Flow Measurement Tank)

(7) Process Controls and Instrumentation System

a. Core
b. Reactor Coolant System
c. ECC and RHR Systems
d. Feedwater and Condensate
e. Steam System
f. Containment System (Break Flow Measurement System)
g. Electrical System

(8) Experimental Instrumentation System

(9) Data Acquisition System

(10) Facility Service Systems

a. Electrical
b. Cooling Water
c. Facility Makeup and Purification
d. Instrument Air
e. Building

Functional requirements for each system and the components thereof are discussed in detail in Section 5.

2.1.2 Analysis Program

In addition to demonstrating the performance of LPWRs during SBAs and/or anticipated transients the ROSA IV Program will develop and verify a computer code which can accurately model that performance. The computer code is expected to be a two-temperature two-velocity (2T2V) code. The basis for which will be the RELAP-5 computer code. The results of separate effects testing both as part of ROSA IV, i.e., from the Two-Phase Test Facility (TPTF) and test results from other sources will be used to modify the basic RELAP 5 code as required to provide mathematical correlations and analytical models of the physical phenomena peculiar to long term transients in LPWRs. The resulting analytical model will be assessed and verified for accuracy by comparison to test results obtained at ROSA IV LSTF and other integral SBA test facilities (LOFT, Semiscale, PKL, etc.)

2.2 Objectives

The purpose of tests to be performed at LSTF is to provide large scale test data on the transient performance of PWRs under SBA and transient conditions and on effectiveness of emergency safeguards systems and procedures under such conditions. The tests will also provide experimental data on two-phase fluid flow in PWRs. Specifically, LSTF will be used to:

(1) Study the effectiveness of the ECCS under SBA and plant transient conditions. Both standard and potential alternate ECCS will be evaluated.

(2) Study the effectiveness of secondary side cooling via the steam generators under SBA and plant transient conditions.

(3) Examine the nature of forced and natural circulation cooling in PWRs in various flow regimes and cooling modes and in transition from one flow regime or mode of cooling to another.

(4) Examine the effect of break size and location on system behavior.

(5) Study the effects of non-condensible gases on system behavior during a SBA or plant transient.

(6) Investigate alternate design systems and/or procedures whcih are being considered to improve system performance during SBA and/or plant transient.

(7) Provide test data with which to develop/verify the SBA analytical model being developed in connection with the ROSA IV Program.

3. Experiment Description

A. TMI Simulation Test Series

The primary objective of the TMI simulation tests is to characterize the LSTF vis-a-vis a LPWR. The following TMI accidnet simulation tests will be performed for the above objective.

(1) Time Sequence Simulation Test of TMI Accidnet

A TMI-accident simulation test will be done according to the real time simulations of the operations and the events during the accident.

(2) Trip Condition Simulation Test of TMI Accident

A TMI-accident simulation test will be done according to the trip conditions of the TMI-2 plant.

(3) TMI Accidnet Simulation Test for Westinghouse Plant

A TMI-accident simulation test will be conducted according to the trip conditions of a Westinghouse LPWR.

B. Anticipated Transient Tests

These tests are to be a direct, continuous simulation of a PWR anticipated transient from the time following scram when core decay heat reaches 14% of full power through to completion of plant recovery. Transients to be investigated are:

(1) Loss of Heat Sink -- e.g., as a result of steam generator dryout, inadvertent steam generator isolation, loss of RHR system.

(2) Overcooling of Steam Generator -- e.g., as a result of a main steam or feed line rupture or sudden injection of cold feedwater.

(3) Loss of Electrical Power -- e.g., loss of AC and total loss of power.

C. Two-Phase Flow Phenomena Tests

These tests are to be a series of specialized experiments to examine the overall system behavior of a PWR under specific, pre-determined fluid conditions. Series of steady-state and/or relatively short transient tests are expected. These tests shall supplement/verify data obtained during separate effects testing. They shall also provide data for developing and verifying analytical models to be used in the 2T2V computer code which is being developed. Test series to be performed include:

(1) Forced circulation with reactor coolant inventory and core power level as parameters.

(2) Natural circulation testing with reactor coolant inventory, core power level, non-condensibles, etc., as parameters.

Testing will cover the range of system inventories from full capacity to partial core uncovering and investigate the various modes of forced and natural circulation, i.e., single-phase, two-phase circulating, and two-phase reflux cooling.

D. Small Break Accident Tests

These tests are to be a direct continuous simulation of a PWR small break LOCA from the time following scram when core decay heat reaches about 14% of full power through to completion of plant recovery, i.e., reactor coolant system inventory stable, the core covered and cooling via the decay heat removal system. Parsmeters to be investgated are break size, break location and ECCS conditions.

E. Alternate ECCS Tests

TMI type accident and other SBA tests will be performed with alternate ECCS to develop to universal methods for core-cooling during the accident and plant recovery after the accident.

4. Design Philosophy

4.1 General Design Requirements

4.1.1 Facility Sizing Basis

LSTF is an experimental test facility designed to model a full height primary region of an LPWR. The reference PWR for LSTF is a 1100 MWe (3423 MWt) PWR with 50,952 fuel pins arranged in 17 × 17 square lattices. The scale factor for LSTF is 1/48. Scaling of LSTF is accomplished as follows:

a. Elevations are preserved, i.e., the scaling ratio is 1/1. Preserving correct elevations is important to LSTF, since gravity strongly influences PWR long-term transient behavior, for instance, natural circulation.

b. Volumes are scaled by the facility scale factor of 1/48.

c. Flow Area in the pressure vessel and steam generators are scaled by the facility scale factor of 1/48 and 1/24, respectively. But the flow area of primary loop, i.e., hot-leg and cold-leg, is determined from the conservation of the volume scaling and the Strouhal number[4] so that the flow regime transition can be simulated.

d. Core Power is scaled by the facility scale factor of 1/48 so that the power input per unit volume in the core region is the same as for the reference PWR. Note, for full power operation, the scaled power of the core would be 71 MW. However, heater rod power supply is limited to 10 MW. Hence, proper core power scaling can only be attained for simulator core power starting at about 14% full power.

e. Fuel Assembly dimensions, i.e., fuel rod diameter, pitch and length, guide thimble diameter pitch and length, and ratio of number of fuel rods to number of guide thimbles, are the same as for the 17 × 17 fuel assembly of the reference PWR in order to preserve the heat transfer characteristics of the core. The total number of rods is scaled by the facility scale factor and is 1080 heated and 104 unheated rods.

f. Design Pressures for the LSTF fluid systems will be at least the same as those for their counterparts in the reference PWR.

g. Fluid Flow ΔPs of major components, e.g., pumps, pressure vessel and steam generators will be the same as in the reference PWR.

h. Flow Capacities for LSTF systems are scaled by the facility scale factor to preserve fluid mass flux.

4.2 Reactor System Design Requirements

4.2.1 Test Vessel

a. The vessel shall be sized to accommodate a core of 1080 electrically heated and 104 unheated rods representing full length 17 × 17 fuel bundles.

b. Wall thickness shall be based on a pressure rating of 17.26 MPa.

c. Two hot leg nozzles shall be provided. The elevation of the top of the hot leg inside diameter shall be the same as for the reference PWR.

d. Two cold leg nozzles shall be provided. The elevation of the cold leg centerline shall be the same as those of the vessel hot legs.

4.2.2 Vessel Internals

a. Core Barrel

1) A simulated core barrel shall be provided.

2) The core barrel is to be full length.

3) Provisions shall be made to simulate core barrel vent valves. External piping may be used for this purpose.

b. Lower Plenum

The design of the lower plenum shall maintain, to the extent practicable, the volume and flow resistances typial of the Reference PWR.

c. Downcomer

1) The downcomer shall be full length.

2) The flow area of the downcomer shall be designed including the

bypass flow area in the core barrel.

3) Provisions shall be made to simulate the bypass leakage between the top of the downcomer and the upper plenum around the hot leg nozzles.

d. Upper Plenum

1) The design of the upper plenum shall include a set of scaled internal structures typical of a PWR.

2) The upper plenum is to be full height and provide proper elevations of the upper core support plate and top plate of the internals package with respect to the elevation of the top of the hot legs.

3) The upper plenum volume shall be scaled by the facility scale factor.

4.2.3 Core

a. The core shall consist of a cylindrical configuration of heater rods simulating 17 × 17 array fuel including unheated rods.

b. Flow paths at the upper end of the core shall be geometrically similar to PWR end boxes and upper core support plate. In the remainder of the core, flow paths and geometry shall reasonably simulate the fuel for both single and two-phase flows.

c. The core shall be designed to provide axial power profiles typical of a PWR. Radial profile shall be adjustable to allow variation of this profile as a test parameter.

d. Power to the core shall be controllable to vary power distribution to achieve the experimental requirements.

e. Heater rod design shall simulate the heat capacity of fuel rods to the degree practicable. MgO shall be used as the insulator in fuel rods.

4.3 Reactor Coolant System Design Requirements

4.3.1 Loop Piping and Configuration

a. Two identical reactor coolant loops each representating two loops of the Reference PWR shall be provided.

b. Hot leg, cold leg and pump suction IDs shall be 207 mm.

c. Piping wall thickness shall be based on pressure rating and experimental operating cycle requirements.

4.3.2 Reactor Coolant Pumps

a. Reactor coolant pumps shall be provided with head-flow characteristics similar to that for the Reference PWR with flow scaled.

b. To the maximum extent practicable, the reactor coolant pumps shall be geometrically similar to those of the Reference PWR.

c. Provisions shall be made to ensure pump coast-down characteristics similar to that for the Reference PWR reactor coolant pumps.

4.3.3 Steam Generators

a. The steam generators shall be designed in accordance with the facility scaling requirements.

b. The steam generator tubes shall be full size. Average length and height shall be the same as for the Reference PWR. To the extent practicable, tube height distribution shall be the same as the Reference PWR. As a minimum, 3 different height tubes shall be provided. The tube wall thickness shall be 2.6 mm for the instrumentation requirement, while it is 1.25 mm for the Reference PWR.

c. Provisions shall be made to model steam generator tube rupture events.

4.3.4 Pressurizer

a. The pressurizer shall be scaled in accordance with the facility scaling requirements. Height to diameter ratio H/D shall be conserved.

b. The pressurizer shall be equipped with heaters and spray for control of temperature and pressure typical of the Reference PWR. Heater capacity shall be designed to provide power-to-pressurizer volume ratio typical of the Reference PWR. Spray capacity shall be 4.3 m^3/h in accordance with the scaling requirements.

c. The pressurizer surge line shall be typical of the Reference PWR. Provisions shall be made to permit connecting the surge line to either hot leg or the reactor vessel upper head.

d. The normal source of pressurizer spray shall be typical of the Reference PWR, e.g., the reactor coolant pump discharge portion of the cold leg.

e. Vent lines shall be provided between the reactor vessel head and the pressurizer vapor region to permit testing alternate means of controlling the reactor coolant system during small break and plant transient conditions.

f. Provisions shall be made for connecting the simulated reactor coolant safety and relief valves to the pressurizer vapor region in a manner typical of the Reference PWR.

4.3.5 Safety and Relief Valves

a. ROSA IV LSTF shall simulate the safety and relief valves of the Reference PWR. Flow rates shall be scaled per Section 4.1.1 above. Provisions shall be made to permit increasing or decreasing the pressure relieving capacity of the safety and relief valves, e.g., to simulate one of three safety valves failed in the closed position.

b. Nominal opening and closing setpoints shall by typical of the Reference PWR. Specifically, nominal opening setpoints shall be 17.2 MPa and 16.2 MPa for safety and relief simulation valves respectively. Corresponding nominal closing setpoints shall be 16.5 MPa and 15.9 MPa. Provisions shall be made to permit changing these setpoints.

c. Elevations, orientations, and piping connecting with the pressurizer shall be typical of the Reference PWR. Provisions shall also be made for alterate locations of safety and/or relief valves in the reactor coolant system including the reactor vessel upper head.

4.4 Emergency Core Cooling and Residual Heat Removal Systems

4.4.1 High Pressure Injection System (HPIS)

a. A high pressure injection system shall be provided. This system shall be capable of simulating flow rate as a function of system pressure typical of the Reference PWR. Flow rate shall be controllable to simulate degraded and enhanced ECCS.

b. The HPIS shall have setpoints typical of the Reference PWR. In addition, the HPIS setpoints shall be changeable. Specifically, provisions shall be made to permit the delay time of HPIS actuation and variation of HPIS setpoints with respect to the magnitudes of the monitored system properties, e.g., system pressure, system temperature and hot-leg subcooling.

c. Injection points for the HPIS system shall include:

   o Both cold legs
   o Both hot legs
   o Lower Plenum
   o Upper Plenum

   The design of the HPIS shall permit different combinations of injection points as well as switching between injection points during a test.

d. The HPIS shall be designed to permit testing of alternate HPIS designs including a high head/high flow system designed to remove at least 2% full power decay heat via the safety and relief valves.

4.4.2 Low Pressure Injection System (LPIS)

a. A low pressure injection system shall be provided. This system shall be capable of simulating flow rate as a function of system pressure typical of the Reference PWR. Flow rates shall be controllable to simulate degraded ECCS conditions.

b. The LPIS shall have setpoints typical of the Reference PWR. In addition, the LPIS setpoints shall be changeable both with respect to magnitude of the monitored variable and with respect to the monitored variables themselves.

c. Injection points for the LPIS shall include:

- o Both cold legs
- o Both hot legs
- o Lower plenum
- o Upper plenum

4.4.3 Accumulator Injection

a. An accumulator injection system shall be provided. This system shall be capable of simulating the water flow typical of the reference plant accumulator systems. Flow rate shall be controllable to simulate degraded ECCS conditions.

b. Accumulator piping and check valve arrangement shall be typical of the Reference PWR.

c. Injection points for the accumulators shall include:

- o Both cold legs
- o Both hot legs
- o Lower Plenum
- o Upper Plenum

d. Provisions shall be made to simulate the injection of non-condensibles into the reactor coolant system that occurs at the end of accumulator water injection.

e. Provisions shall be made to control the temperature of the accumulators and piping over their operating range.

4.4.4 Residual Heat Removal Systems (RHR)

a. A residual heat removal system shall be provided whcih is capable of simulating the head-flow characteristics of the RHR of the Reference RWR. In addition, provisions shall be made to permit variation of RHR setpoints with respect to the magnitudes of the monitored system properties.

b. Connection of the RHR inlet and outlet to the reactor coolant system shall be at locations typical of the Reference RWR.

c. The RHR shall be designed to permit testing of alternate RHR designs including a high pressure, high temperature RHR capable of removing 2% core decay heat at operating temperature and corresponding saturation pressure, i.e., approximately 589 K and 11.1 MPa.

4.5 Secondary Side Coolant System

4.5.1 Steam Generator Secondary Side

a. Steam Generators shall be designed in accordance with the facility scaling requirements.

b. To the extent practicable, provisions shall be made to ensure steam exit quality and recirculation ratio in the steam qenerators typical of the Reference PWR.

4.5.2 Steam System

a. Main Steam

1) Provisions shall be made to simulate the transient steam demand on the steam generators by the main steam system including the demand and changes in demand caused by the main steam isolation valves (MSIVs) and turbine bypass. Flow rates shall be scaled in accordance with the facility scaling factors.

2) To the extent practicable, main steam piping from the steam generators to the main steam safety and relief valves shall be typical of that for the Reference PWR and shall be designed per scaling factor.

3) Provisions shall be made for modeling a main steam line rupture accident.

b. Safety and Relief Valves

1) To the extent practicable, provisions shall be made to simulate the main steam safety and relief valves typical of the Reference PWR.

2) Simulated opening and closing setpoints for the safety and relief valves shall by typical of those for the Reference PWR.

Capability shall be provided to vary these setpoints and also to model stuck open/failed closed valves.

4.5.3 Main Feedwater System

a. Provision shall be made to simulate transient feedwater flow to the steam generators typical of transients possible at the Reference PWR in accordance with the scaling factor.

b. Provisions shall be made to vary feedwater temperature as is typical for the Reference PWR.

c. Provisions shall be made to model a main feed line rupture.

4.5.4 Emergency Feedwater System

a. Provisions shall be made to simulate auxiliary/emergency feedwater flow to the steam generators typical of the Reference PWR in accordance with the scaling factor.

b. Setpoints for the emergency feedwater system shall be typical of those for the Reference PWR and shall be adjustable.

c. Scaled flow rate as a function of steam generator pressure shall be typical of the Reference PWR.

d. Emergency feedwater temperatures shall be controllable over the range expected in the Reference PWR.

4.6 Simulated Containment System

4.6.1 A simulated containment system shall be provided to collect and contain the effluent from the simulated pipe break.

4.6.2 The simulated containment system shall provide a back pressure for the break sufficiently low to ensure break flow is always choked.

4.7 Facility Services Systems

4.7.1 Cooling Water

a. Cooling water shall be provided as necessary to support other facility systems, e.g., heat exchangers in the RHR.

b. Cooling water shall be provided as necessary to support components within the test facility, e.g., pump seals and heater rod power supply.

c. Cooling water shall be provided as necessary to cool instrumentation, e.g., drag discs, gamma densitometers, water sampling stations, etc.

4.7.2 Water Chemistry Control

A water chemistry control system shall be provided to deaerate, demineralize, and chemically buffer as required, water to be used in the facility.

4.7.3 Control Air System

A control air system shall be provided to deliver clean, dry instrument quality air at constant pressure as required to operate pneumatic control valves, air motors and other pneumatic controls in the facility and to provide cooling air to instrumentation and/or electronic signal conditioners adjacent to the test facility.

4.7.4 Building Services

Typical building services shall be provided as required including lighting, heating, ventilation, and air conditioning, plumbing, and communications.

4.8 Process Instrumentation and Controls Systems

4.8.1 Heater Power Supply Controls

a. A control system for the heater rod power supply shall be provided to simulate the power decay of the fuel rods for a typical PWR.

b. The heater power supply control system shall be programmable so that a variety of decay curves representing the expected power decay for the transient of interest can to be obtained.

c. The heater power supply control system shall be equipped with an automatic shutoff to ensure heater rod tmeperature does not exceed the heater rod design limit.

4.8.2 Valve Control System

a. A system to control the automatic valves in the facility to simulate the valve control system and possible inadvertent valve actuation expected for the Refference PWR shall be provided.

b. The system shall be programmable with predetermined setpoints based on process and/or experimental instrumentation setpoints which shall be adjustable.

c. Valves to be included in this system are:

1) Simulated break line valve.

2) Simulated feedwater block valves.

3) Simulated main steam isolation valves.

4) Simulated main steam safety and relief valves.

5) Simulated pressurizer safety and relief valves.

6) ECCS valves.

4.8.3 Flow Control System

a. A system to control the flowrates of various facility system to simulate the flow control system in the Reference PWR shall be provided.

b. The flow control system shall be programmable with predetermined flowrates as functions of facility parameters measured by either process or experimental instrumentation. Flowrate functions shall be changeable.

c. Systems for which flow control is required include:

1) Reactor coolant system, especially pump coastdown.

2) Emergency core cooling system.

3) RHR

4) Feedwater system.

5) Emergency feedwater system.

6) Main steam.

7) Non-condensibles injection system.

4.8.4 Trip Simulation System

a. A control system shall be provided to simulate the trips, interlocks, and other automatic functions of the Reference PWR Engineered Safeguards Systems.

b. Trips and interlocks to be included in the system are:

1) Reactor Scram (for heaters).

2) Pressurizer heater controls.

3) Reactor coolant pump trips.

4) Steam generator isolation system.

5) ECCS initiation system.

6) Simulated turbine and turbine bypass trips.

7) Emergency feedwater system initiation.

4.8.5 Interface with Experimental Instrument System

a. The experimental instrumentation system shall provide data as required for input to the control system to conduct the desired experiments.

b. The automatic control systems shall provide information to the experimental instrument data acquisition system so that a chronology of events can be recorded along with experimental instrumentation data.

4.8.6 Control Room

a. A control room shall be provided into which are assembled all of the process and experimental instrumentation readouts, alarms, controls, etc., required to conduct the desired experiments and to acquire and record the test data.

5. Experimental Facility

5.1 General System Arrangement

The general system arrangement for the ROSA-IV Large Scale Test Facility (LSTF) is shown in Fig. 5.1 ~ 5.3. LSTF has two symmetric coolant loops, each designed to represent two loops of a PWR. Each loop includes loop piping, a reactor coolant pump, steam generator, accumulator (ACC cold) and ECCS systems designed to simulate the corresponding components in the reference plant. The pressure vessel (PV), pressurizer (PR) and residual heat removal (RHR) system are also designed to simulate systems of the reference PWR. The secondary coolant system however shall not completely simulate a PWR. The main steam isolation valve, steam generator secondary, safety and relief valves, main and emergency feedwater systems, and turbine bypass valves were designed to respond in a manner representative of these systems in the reference PWR. The LSTF condenser, however, will be of the jet condenser type and therefore not be representative of a PWR. Details of the system design for the various components are given below.

5.2 Pressure Vessel

The pressure vessel design for the LSTF includes the internal structure found in a PWR. These are inlet annular, annular downcomer, lower plenum, lower core plate, core barrel, electrically heated core, upper core plate, upper plenum, upper core support plate and upper head. Also included are internal structures in the upper plenum and upper head to represent upper core support structures. These are illustrated in Fig. 5.4 ~ 5.5. The design values for some of the pressure vessel components are listed in Table 5.1. The core bypass flow area is not simulated in LSTF. Instead the flow area has been added to the downcomer gap to simulate the hydraulic behavior in a PWR satisfactorily.

5.3 Simulated Fuel Assembly

The LSTF simulated fuel assembly consists of 1080 electrically heated rods and 104 unheated rods arranged in sixteen 7 × 7 rod bundles and eight corner bundles.

The fuel rods and bundle assembly are designed to have the same rod dimensions and bundle geometry as the reference PWR. These are heated length of 3660 mm and rod outside diameter of 9.5 mm. The rod pitch is 12.6 mm. The thimble tubes and instrument tubes of a PWR fuel assembly are simulated in the LSTF fuel assembly. The simulated fuel assembly is shown in Fig. 5.6 ~ 5.7. Fig. 5.6 also shows the detailed design of an individual electrically heated rod. The rod design consists of a nichrome heater, surrounded by magnesium-oxide insulation and enclosed in an inconel sheath. Table 5.2 gives the design parameters. The rods have an axial peaking factor of 1.50 and rod axial power distribution based on a nine step chopped cosine curve. Two types of rods are provided in the fuel assembly simulated in order to represent radially peaked core power profiles. These include 360 high heat flux rods (14.0 kW/rod) and 720 low heat flux rods (9.27 kW/rod).The radial core power distribution of a PWR, however, includes high, medium, and low heat flux regions. These regions will be simulated in the LSTF core through the use of a SCR core power control system which will allow the LSTF high heat flux rods to simulate the PWR high heat flux rods and the LSTF low heat flux rods to simulate the PWR medium and low heat flux rods. A total of 53 rods in the core are instrumented, 16 high heat flux rods and 37 low heat flux rods. In order to maintain the 1/48 system scale factor, the core power required to simulate full power would be 71.3 MW. However, because of power supply limitations a maximum core power of only 10 MW is available, hence proper core power simulation is only possible at power rate below 14% of full power. The grid spacers are supplied in the simulated fuel assembly to set and maintain the rod pitch during experiments. There is a flow area reduction of approximately 22% at the grid spacer locations. Figure 5.7 shows the grid spacer elevations with the elevation from the bottom of the core.

5.4 Primary Coolant System

The LSTF primary coolant system contains components designed to simulate the primary components of a PWR including loop piping, steam generator, reactor coolant pump and pressurizer. The loop piping length (L) and diameter (D) were chosen to conserve the volume scaling ratio and to provide the same $L/\sqrt{D}$ ratio as the reference PWR. This scaling method was chosen to preserve the timing at which important two-phase flow phenomena occurs such as the transition to stratified flow. The resulting lengths and diameters for the design values of LSTF are shown in Table 5.5.

Since each loop of LSTF represents 2 loops of a PWR the steam generator volume scale factor is 1/24. The steam generators are of the U-tube type, each containing 141 full length U-tubes. The tubes have an ID = 19.6 mm and OD = 25.4 mm. Compared to a PWR the tube wall thickness is increased for safety purposes in the experimental facility. As a result the tube outer surface heat transfer area is scaled by 1/20 instead 1/24. The main design values for the steam generator primary side are summarized in Table 5.5. Because of the core power limitations, the LSTF reactor coolant pumps are designed to be representative of a PWR pump only below 14% of normal full flow rate. The LSTF pressurizer geometry is a scaled down version of an actual PWR pressurizer. Pressurizer dimensions are chosen so the total volume would be scaled 1/48, conserving the height to diameter ratio and the steady state liquid level in the PWR. The pressurizer is equipped with two types of heaters; (1) a 7 kW proportional system and (2) two 25 kW on/off systems. The pressurizer is capable of being connected to the hot leg or the vessel upper head. Because of the importance of gravity effects on many of the phenomena to be investigated in LSTF, elevations of system components in the PWR are maintained in the LSTF design. These are shown in Table 5.5.

5.5 Blowdown System

The LSTF blowdown system consists of the break unit, break flow measuring tank and water recirculation pump. The break unit is shown in Fig. 5.12 and is designed to be adaptable to all the different break location points (see Fig. 5.13) which include cold leg, hot leg, pressurizer, steam generator (simulated tube rupture) and pump suction leg. For the hot and cold leg break locations, the break unit can be used to investigate pipe breaks at the top, bottom and middle of the pipe. The break flow mesuring tank (see Fig. 5.14) is designed to intergral break flow. As the break flow is discharged from the system into the tank it is condensed by cold water. The change in the tank level indicates the integral break flow. Tank level change is measured by a differential pressure transmitter.

5.6 Emergency Core Cooling System

The LSTF emergency core cooling system (ECCS) includes high pressure injection system (HPIS), accumulators (cold and hot), low pressure injection system (LPIS), and residual heat removal (RHR) system. Flow diagrams for these systems are shown on Figures 5.15 ~ 5.19.

The system is designed so the injection point of the HPIS is changeable during a test. This is the only system with this capability. The high pressure injection pump (P_H) and the safety injection pump (P_J) are of the positive displacement type with a stroke adjustable flow control device. The injection flow rate is automatically controlled by the programmable controller according to the simulated head-flow curve of the actual PWR pump. The accumulator flow rate is adjusted with the use of different orifices in the surge line to change the flow resistance. Accumulator (cold) simulates the accumulator of an actual PWR. Accumulator (hot) injects hot water into the loop. This allows the effect of ECC subcooling to be investigated during the tests. Table 5.6 shows the design values for the LSTF accumulator systems. The low pressure injection pump (P_L) is of the centrifugal type. The head-flow characteristic of the LPIS pump in a PWR is simulated by the LSTF pump. The LPIS pump is also used as the RHR pump. In the RHR mode a constant flow rate is maintained by an analog controller. The liquid temperature of the return line is also maintained at a constant value by the analog temperature controller. The RHR system heat exchanger is shown in Fig. 5.22. Figures 5.23 and 5.24 show the head-flow characteristics of the LSTF P_J, P_H and P_L pumps.

5.7 Secondary Coolant System

The secondary coolant system basic flow diagram and heat and mass balance is shown in Fig. 5.25.

5.7.1 System Function

Steam generated in the S.G. secondary side flows to the jet condenser and is mixed with cold spray which condenses it into saturated water. The saturated water is cooled with cooling tower No. 1 to the proper subcooled temperature. The subcooled water then flows to the feedwater pump. A part of the feedwater flows to cooling tower No. 2 and is cooled to the spray water temperature. The secondary coolant system pressure is controlled by the spray water control valve. The S.G. secondary liquid level is controlled by the feedwater control valve. Emergency feedwater is supplied from the RWST tank.

5.8 Auxiliary Systems

5.8.1 Cooling Water System

The cooling water system consists of cooling water tank (CWT), cooling water pump, piping, valves, and instrumentation. Cooling water is provided as necessary to cool the RHR heat exchanger, pump seal and others.

5.8.2 Water Chemistry Contral System

The water chemistry control system consists of water purifier, pump and piping. This system supplies purified water to the primary and secondary loops.

5.8.3 N_2 Supply System

The N_2 supply system supplies N_2 gas to accumulators and N_2 injection points to simulate the effect of non condensible. Pressurized N_2 gas is supplied by N_2 gas container.

TABLE 5.1 DESIGN PARAMETER OF PRESSURE VESSEL

DESIGN PRESSURE	17.26 MPa
DESIGN TEMPERATURE	616.2 K
MATERIAL	
VESSEL	STAINLESS CLAD CARBON STEEL
INTERNAL STRUCTURE	STAINLESS

TABLE 5.2 COMPONENT SCALED DIMENSIONS OF LSTF

COMPONENT		PWR	LSTF	SCALE
PRESSURE VESSEL				
VESSEL INSIDE DIAMETER	(mm)	4394	640	1/ 6.87
VESSEL THICKNESS	(mm)	216		
CORE BARREL OUTSIDE DIAMETER	(mm)	3874	520	1/ 7.45
DOWNCOMER LENGTH	(mm)	6066	6066	1/ 1
DOWNCOMER GAP	(mm)	260	60	1/ 4.33
DOWNCOMER FLOW AREA	(m^2)	3.38	0.109	1/31.0
LOWER PLENUM FLUID VOLUME (PV INSIDE VOLUME)	(m^3)	29.6	0.62	1/48
UPPER PLENUM FLUID VOLUME (NOT INCLUDE UPPER HEAD VOLUME)	(m^3)	28.4	0.60	1/48
UPPER HEAD FLUID VOLUME	(m^3)	24.6	0.51	1/48

TABLE 5.3 DESIGN PARAMETER OF FUEL ASSEMBLY

CORE POWER		10 MW
DETAILS OF FUEL RODS		
HIGH HEAT FLUX ROD		14.0 kW/ROD
W/O INSTRUMENTATION		344 RODS
WITH INSTRUMENTATION		16 RODS
LOW HEAT FLUX ROD		9.27 kW/ROD
W/O INSTRUMENTATION		683 RODS
WITH INSTRUMENTATION		37 RODS
NON HEATING ROD		
W/O INSTRUMENTATION		85 RODS
WITH INSTRUMENTATION		19 RODS
AXIAL PEAKING FACTOR		1.50
POWER DISTRIBUTION		COS CURVE 9STEPS
MATERIAL		
HEATER	NCH-1 EQ.	
INSULATION	MgO EQ.	
SHEATH	INCONEL EQ.	
DESIGN PRESSURE		17.26 MPa
DESIGN TEMPERATURE		1173.2 K

TABLE 5.4 COMPONENT SCALED DIMENSIONS OF LSTF

COMPONENT		PWR	LSTF	SCALE
FUEL (HEATER ROD) ASSEMBLY				
NUMBER OF BUNDLES		193	24	
ROD ARRAY		17 × 17	7 × 7	
ROD HEATED LENGTH	(mm)	3660	3660	1/ 1
ROD PITCH	(mm)	12.6	12.6	1/ 1
FUEL ROD OUTSIDE DIAMETER	(mm)	9.5	9.5	1/ 1
THIMBLE TUBE DIAMETER	(mm)	12.24	12.2	1/ 1
INSTRUMENT TUBE DIAMETER	(mm)	12.24	12.2	1/ 1
NUMBER OF HEATER RODS		50952	1080	1/47.2
NUMBER OF NON-HEATING RODS		4825	104	1/46.4
CORE FLOW AREA (WITHOUT SPACER LOCATION)	(m^2)	4.75	0.107	1/44.4
CORE FLOW AREA (WITH SPACER LOCATION)	(m^2)	3.70		
CORE FLUID VOLUME	(m^3)	17.5	0.392	1/44.6

TABLE 5.5 COMPONENT SCALED DIMENSIONS OF LSTF

COMPONENT		PWR	LSTF	SCALE
PRIMARY LOOP (SAME 2 LOOPS)				
HOT LEG INSIDE DIAMETER	(mm)	736.6	207	
HOT LEG LENGTH	(mm)	7040	3730	$L/\sqrt{D}$ SIMULATED
CROSSOVER LEG INSIDE DIAMETER	(mm)	787.4	207	
LENGTH	(mm)	10280	5270	
COLD LEG INSIDE DIAMETER	(mm)	698.5	207	
COLD LEG LENNGTH	(mm)	8400	4570	
PRESSURIZER				
VESSEL INSIDE DIAMETER	(mm)	2100	600	1/ 3.5
VESSEL HEIGHT	(mm)	15500	4200	1/ 3.69
TOTAL VOLUME	(m^3)	51	1.1	1/48
FLUID VOLUME	(m^3)	31	0.65	1/48
ELEVATION				
BOTTOM OF HEATER BUNDLE	(mm)	0	0	
TOP OF HEATER BUNDLE	(mm)	3660	3660	1/1
TOP OF DOWNCOMER	(mm)	4849	4849	1/1
BOTTOM OF DOWNCOMER	(mm)	- 1217	-1217	1/1
CENTER OF COLD LEG	(mm)	5198		
TOP OF COLD LEG INSIDE DIAMETER (CROSS OVER LEG)	(mm)	5548	5548	1/1
CENTER OF LOOP SEAL LOWER END	(mm)	2056		
BOTTOM OF LOOP SEAL LOWER END	(mm)	1662	1662	1/1
CENTER OF HOT LEG	(mm)	5198		
TOP OF HOT LEG INSIDE DIAMETER	(mm)	5567	5567	1/1
BOTTOM OF UPPER CORE PLATE	(mm)	3957	3957	1/1
TOP OF LOWER CORE PLATE	(mm)	- 108	- 108	1/1
BOTTOM OF TUBE SHEET OF STEAM GENERATOR	(mm)	7308	7308	1/1
PLENUM LOWER END OF STEAM GENERATOR	(mm)	5713	5713	1/1
TOP OF TUBES OF STEAM GENERATOR (avg)	(mm)	17953	17953	1/1

TABLE 5.6 DESIGN PARAMETER OF ACCUMULATOR

DESIGN PRESSURE	11.38 MPa
DESIGN TEMPERATURE	423.2 K (COLD)
	493.2 K (HOT)
MATERIAL	
VESSEL	STAINLESS CLAD CARBON STEEL

TABLE 5.7 COMPONENT SCALED DIMENSIONS OF LSTF

COMPONENT	PWR	LSTF	SCALE
ACCUMULATOR (COLD AND HOT)			
VESSEL INSIDE DIAMETR (mm)	3500	950	1/3.68
VESSEL HEIGHT (mm)	5280	6600	1.25
TOTAL VOLUME (m^3)	38.2	4.8	1/7.96
LIQUID VOLUME (m^3)	26.9	3.38	1/7.96

TABLE 5.8 DESIGN PARAMETER OF RHR HEAT EXCHANGER

DESIGN PRESSURE	17.26 MPa (PRIMARY)
	0.67 MPa (SECONDARY)
DESIGN TEMPERATURE	416.2 K (PRIMARY)
	333.2 K (SECONDARY)
MATERIAL	
SHELL	STAINLESS CLAD CARBON STEEL
TUBE	SUS 316 EQ.

TABLE 5.9 COMPONENT SCALED DIMESIONS OF LSTF

COMPONENT	PWR	LSTF	SCALE
RHR HEAT EXCHANGER			
NUMBER OF TUBES/1PASS	568	24	1/23.7
TOTAL U TUBE LENGTH (mm)	8600	8600	1/1
TUBE OUTSIDE DIAMETER(mm)	19.0		
TUBE INSIDE DIAMETER (mm)	16.6		
TUBE WALL THICKNESS (mm)	1.2		
TUBE PITCH (mm)	28.5	28.5	1/1
TUBE ARRAY	Δ	Δ	
HEAT TRANSFER AREA (m^2) (OUTER SURFACE)	590	25	1/23.6

TABLE 5.10 DESIGN PARAMETER OF STEAM GNERATOR

DESIGN PRESSURE	17.26 MPa	(PRIMARY)
	8.27 MPa	(SECONDARY)
DESIGN TEMPERATURE	616.2 K	(PRIMARY)
	571.2 K	(SECONDARY)
MATERIAL		
SHELL	STAINLESS CLAD CARBON STEEL	
TUBE	SUS 316 EQ.	

TABLE 5.11 COMPONENT SCALED DIMENSIONS OF LSTF

COMPONENT		PWR	LSTF	SCALE
STEAM GENERATOR (SAME 2 S.Gs)				
NUMBER OF TUBES		3382	141	1/24
TUBE LENGTH (AVERAGE)	(m)	20.24	20.24	1/1
TUBE OUTSIDE DIAMETER	(mm)	22.23	25.4	
TUBE INSIDE DIAMETER	(mm)	19.7	19.6	1/1
TUBE WALL THICKNESS	(mm)	1.27	2.9	
HEAT TRANSFER AREA (OUTER SURFACE OF TUBE)	(m^2)	4784	199	1/24
INLET PLENUM VOLUME	(m^3)	4.25	0.18	1/24
OUTLET PLENUM VOLUME	(m^3)	4.25	0.18	1/24
PRIMARY SIDE VOLUME	(m^3)	29.36	1.22	1/24
SECONDARY SIDE VOLUME	(m^3)	157.33	6.56	1/24

TABLE 5.12 DESIGN PARAMETER OF JET CONDENSER

DESIGN PRESSURE	8.27 MPa
DESIGN TEMPERATURE	571.2 K
MATERIAL	
VESSEL	STAINLESS CLAD CARBON STEEL
TOTAL VOLUME	10 M^3

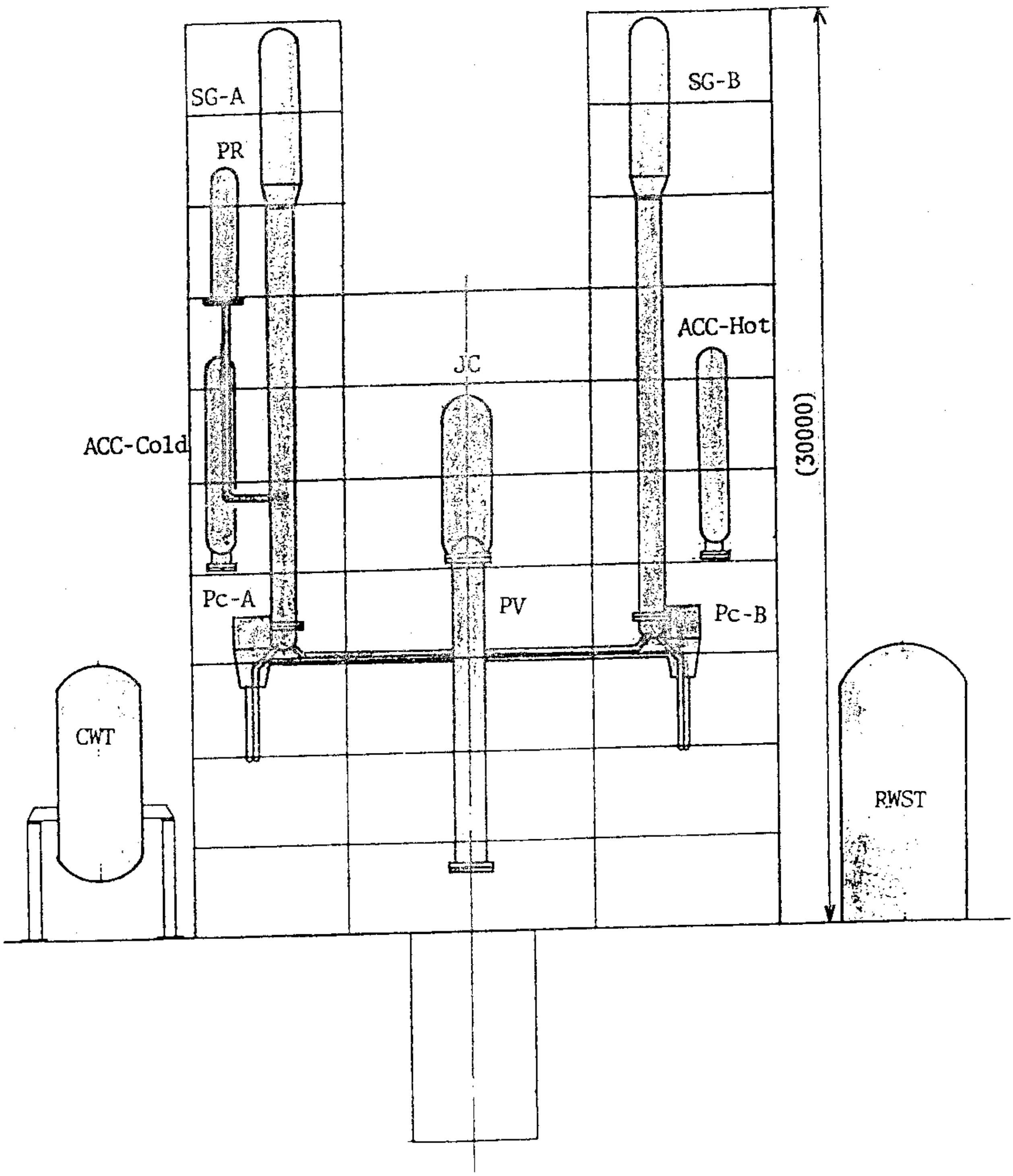

FIG.5.1 GENERAL ARRANGEMENT (ELEVATION)

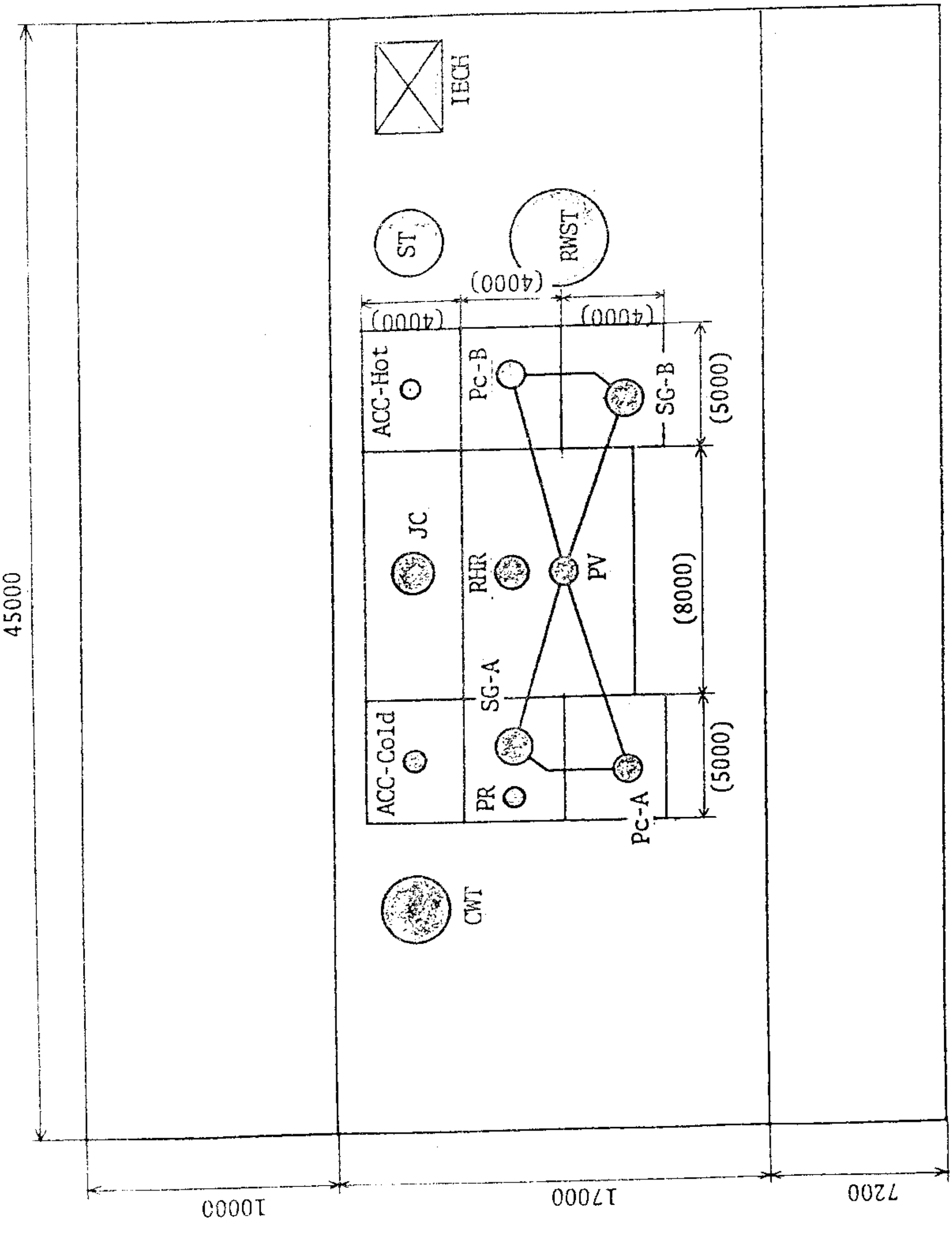

FIG.5.2 GENERAL ARRANGEMENT (PLAN)

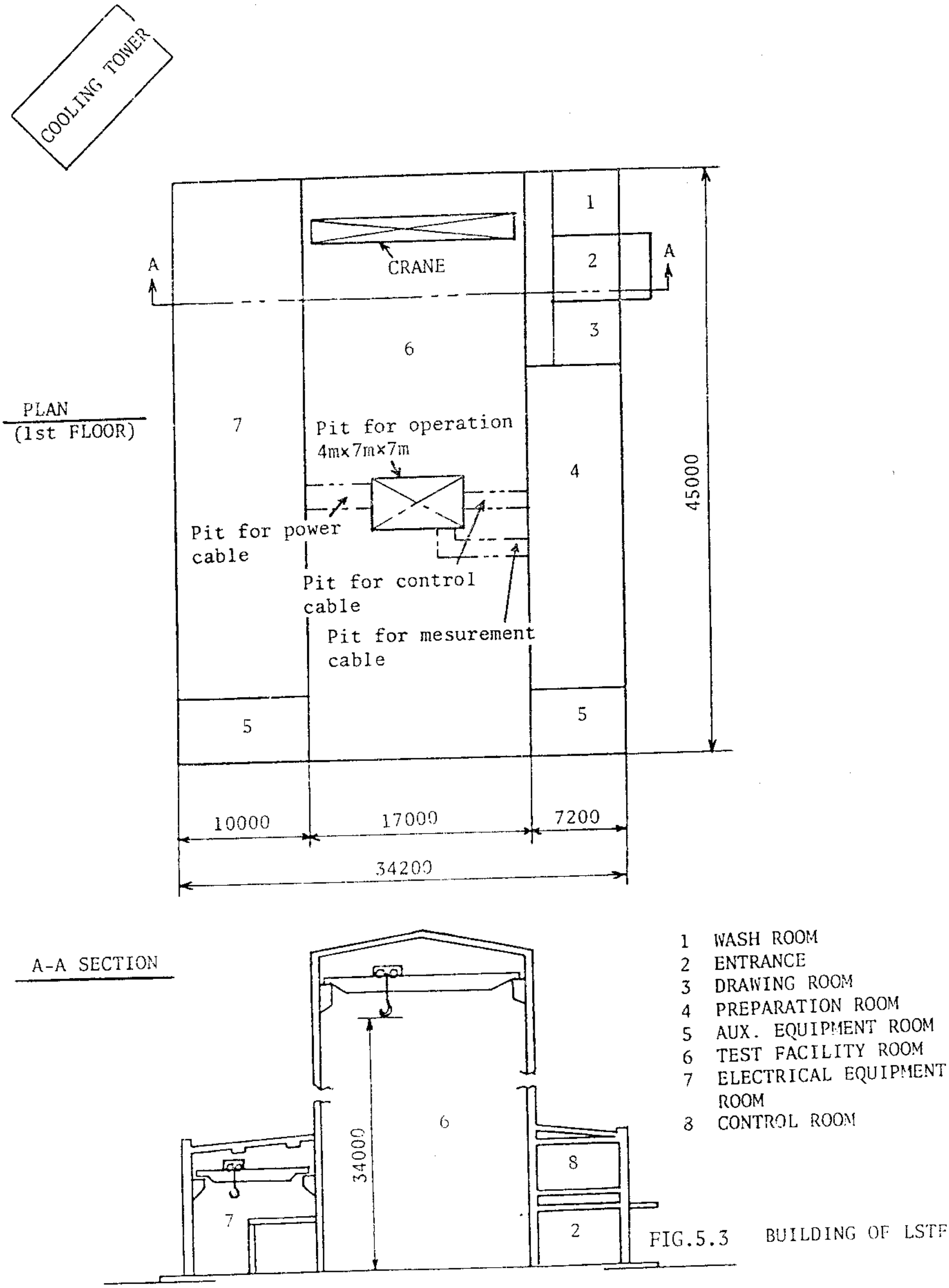

FIG.5.3 BUILDING OF LSTF

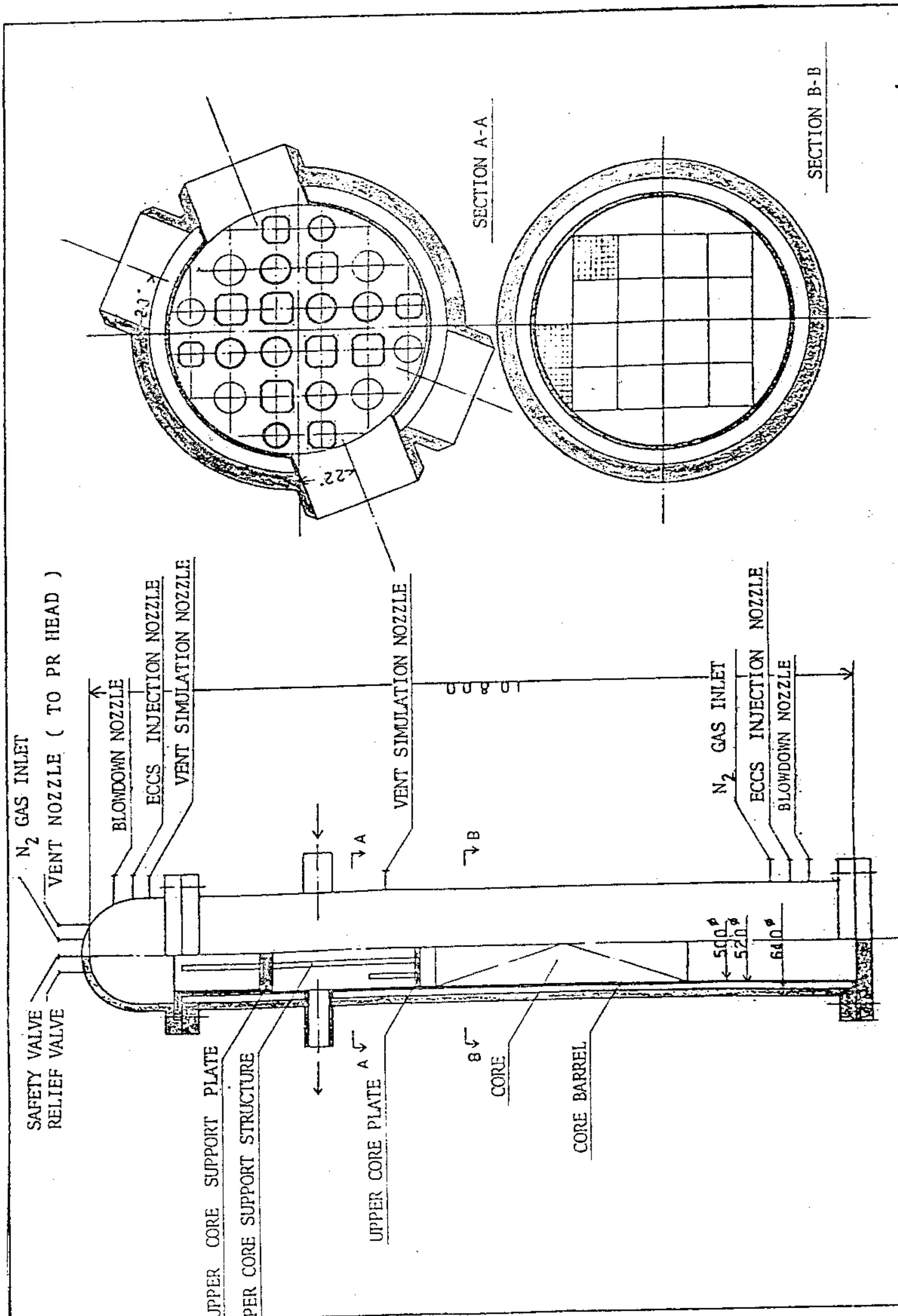

FIG.5.4 PRESSURE VESSEL 1/2

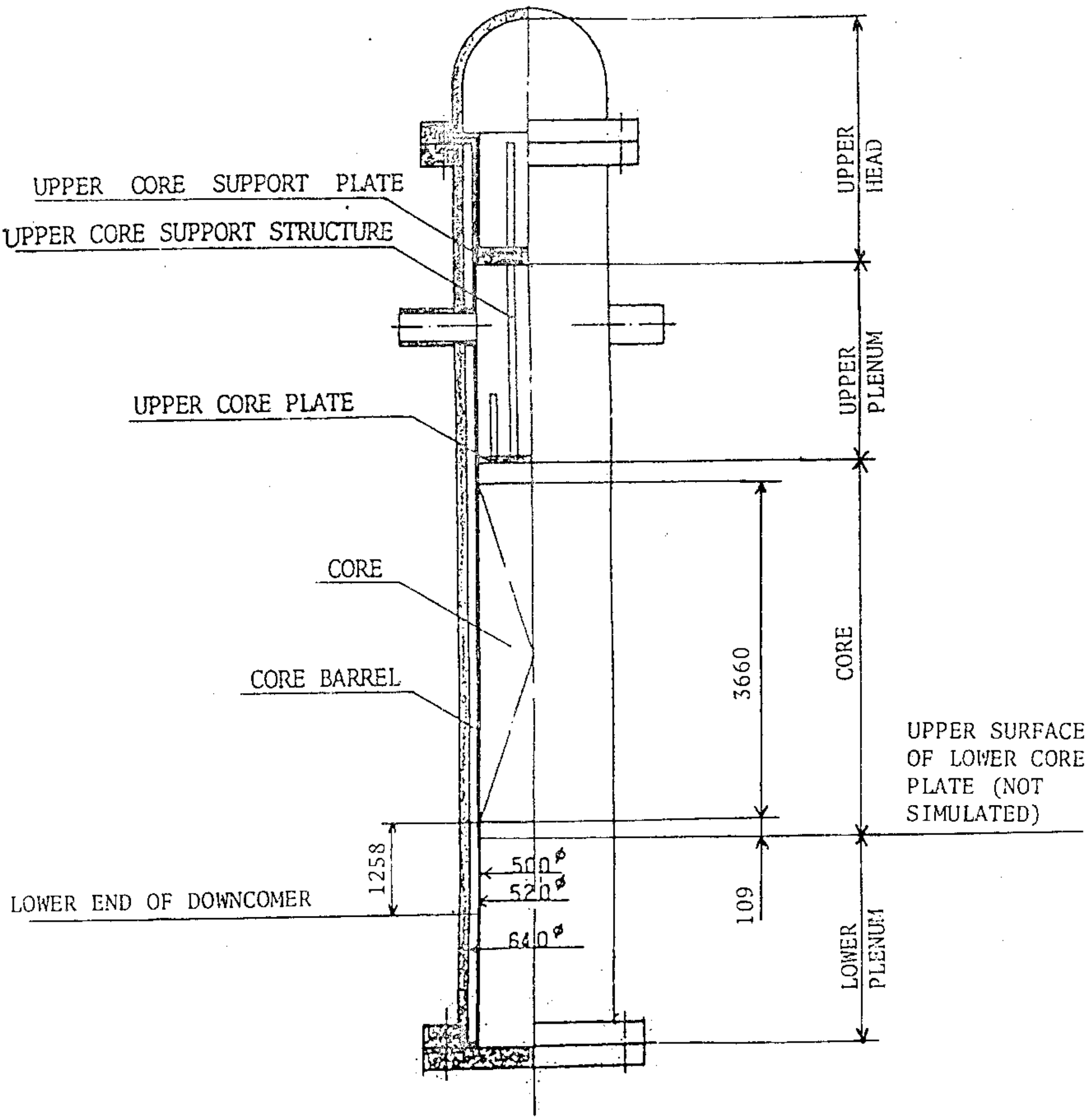

FIG.5.5 PRESSURE VESSEL 2/2

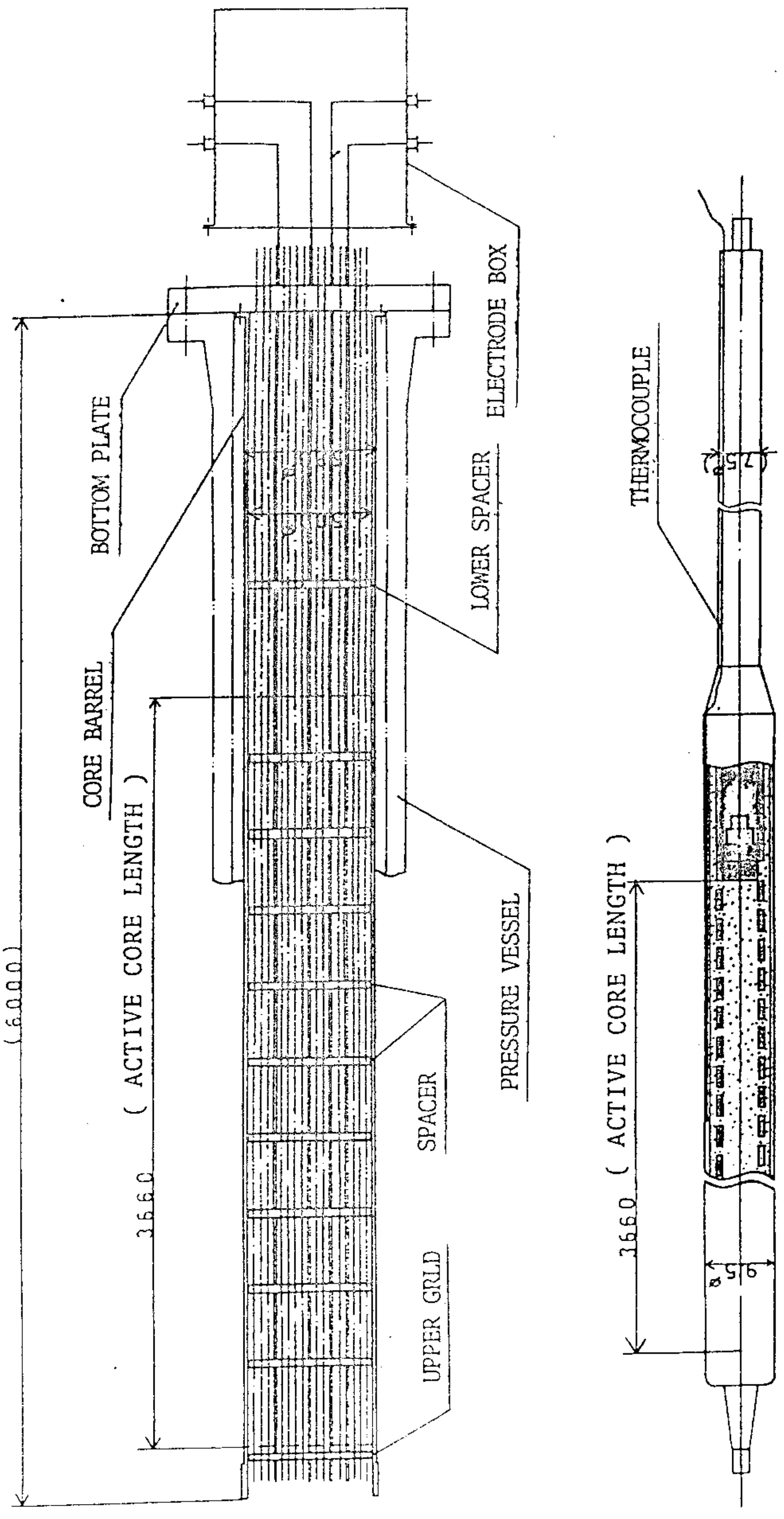

FIG.5.6 SIMULATED FUEL ASSEMBLY

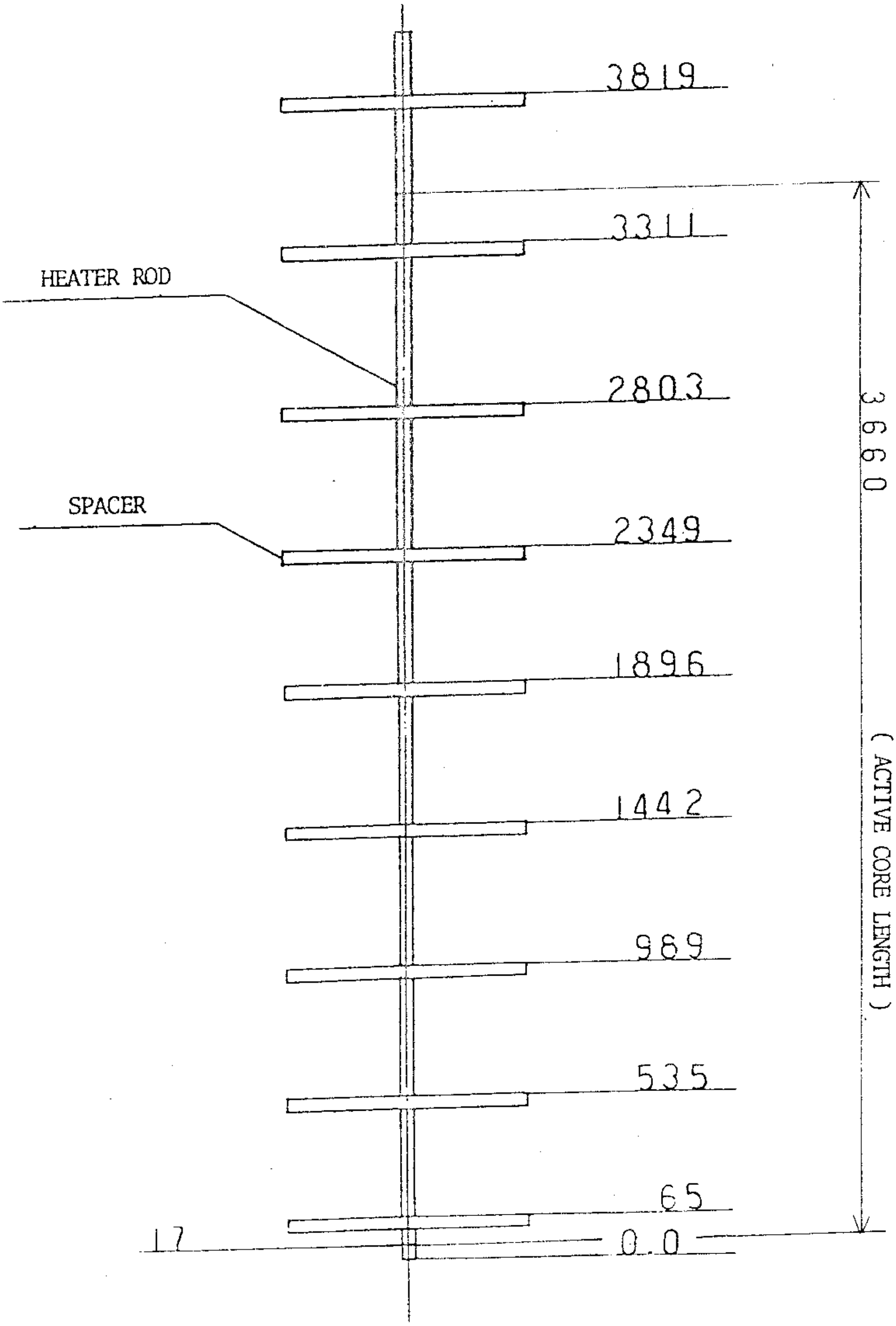

FIG.5.7 ELEVATION OF SPACER

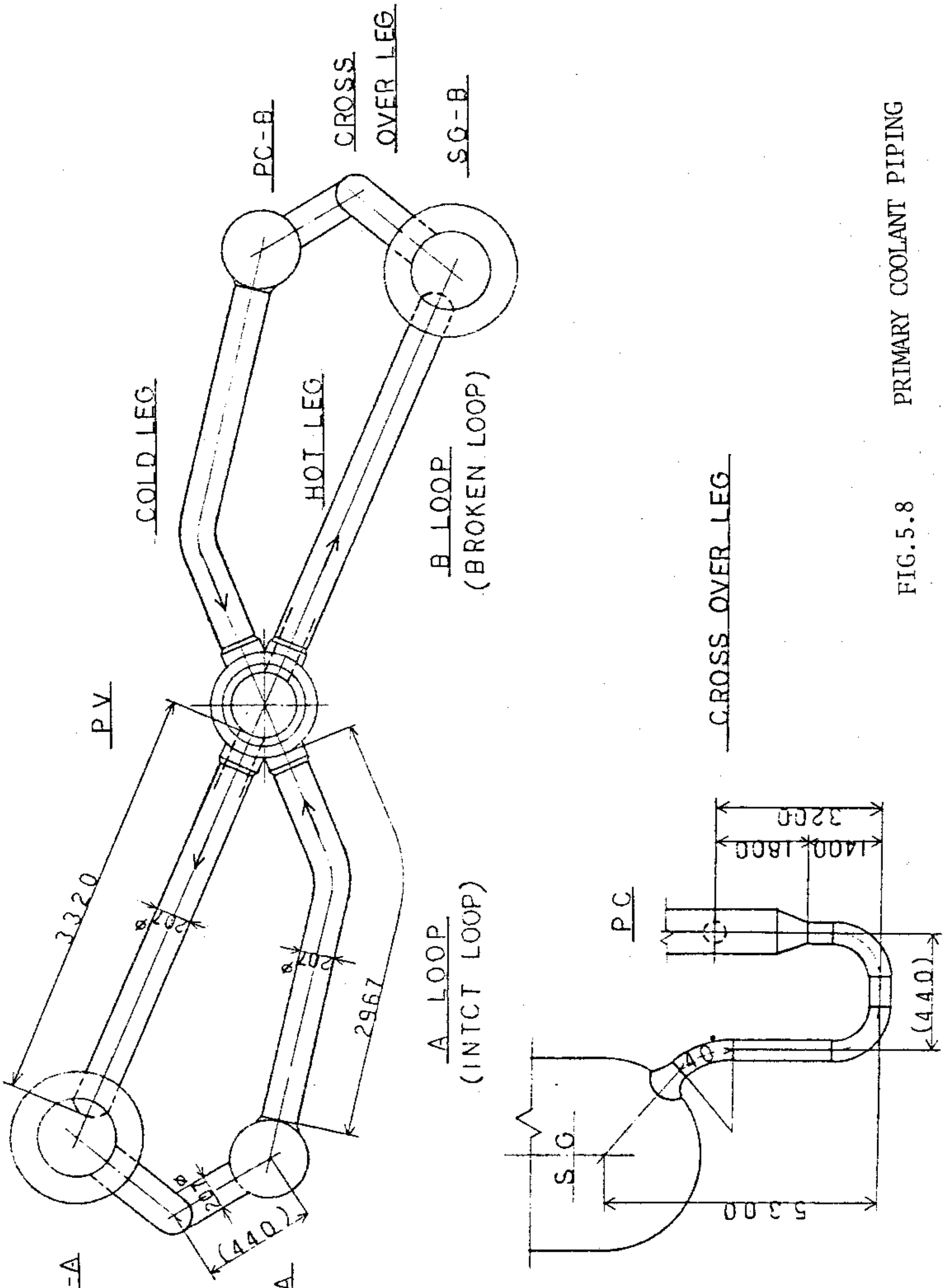

FIG.5.8 PRIMARY COOLANT PIPING

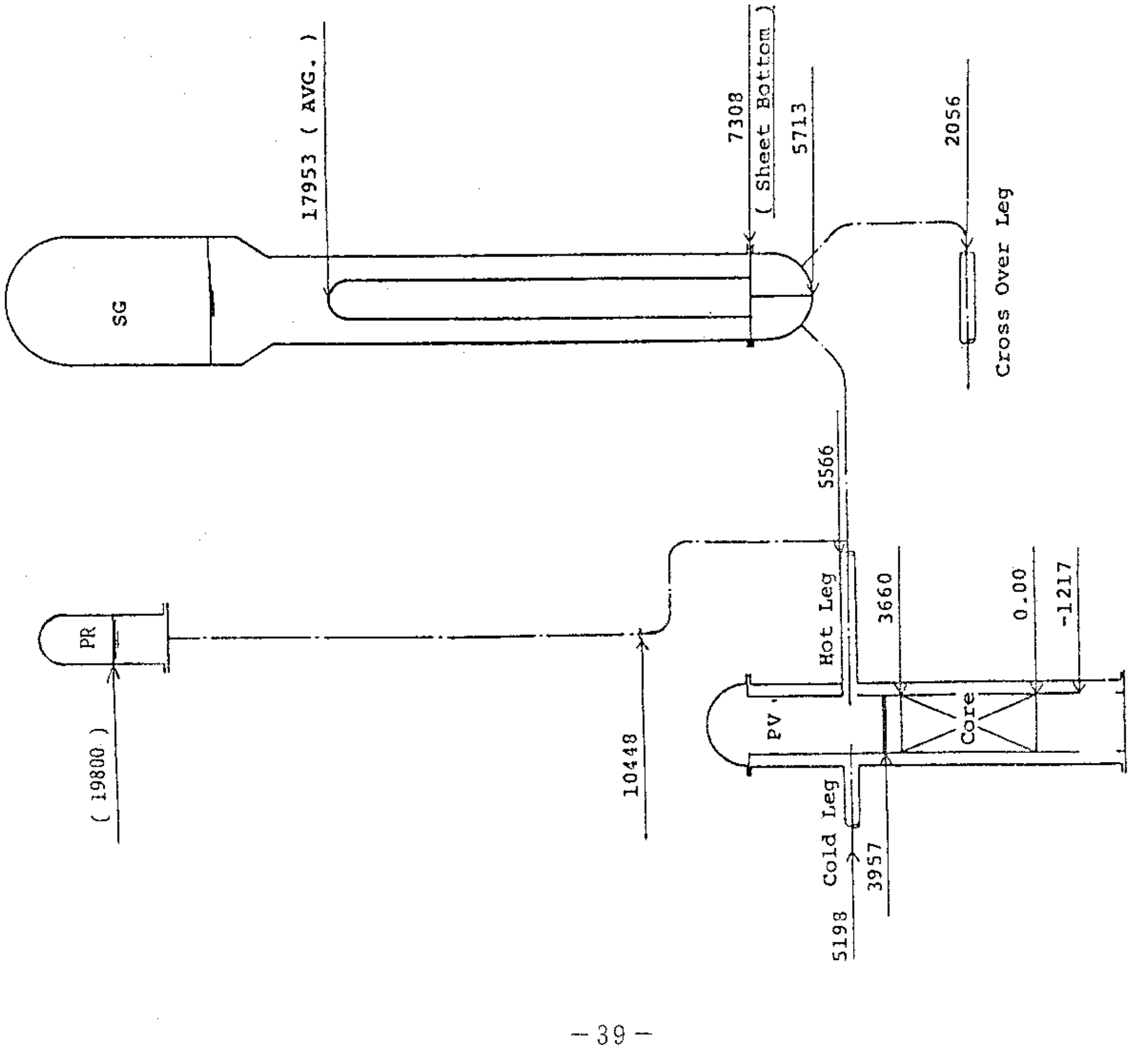

FIG.5.9 PRIMARY COOLANT SYSTEM ELEVATION

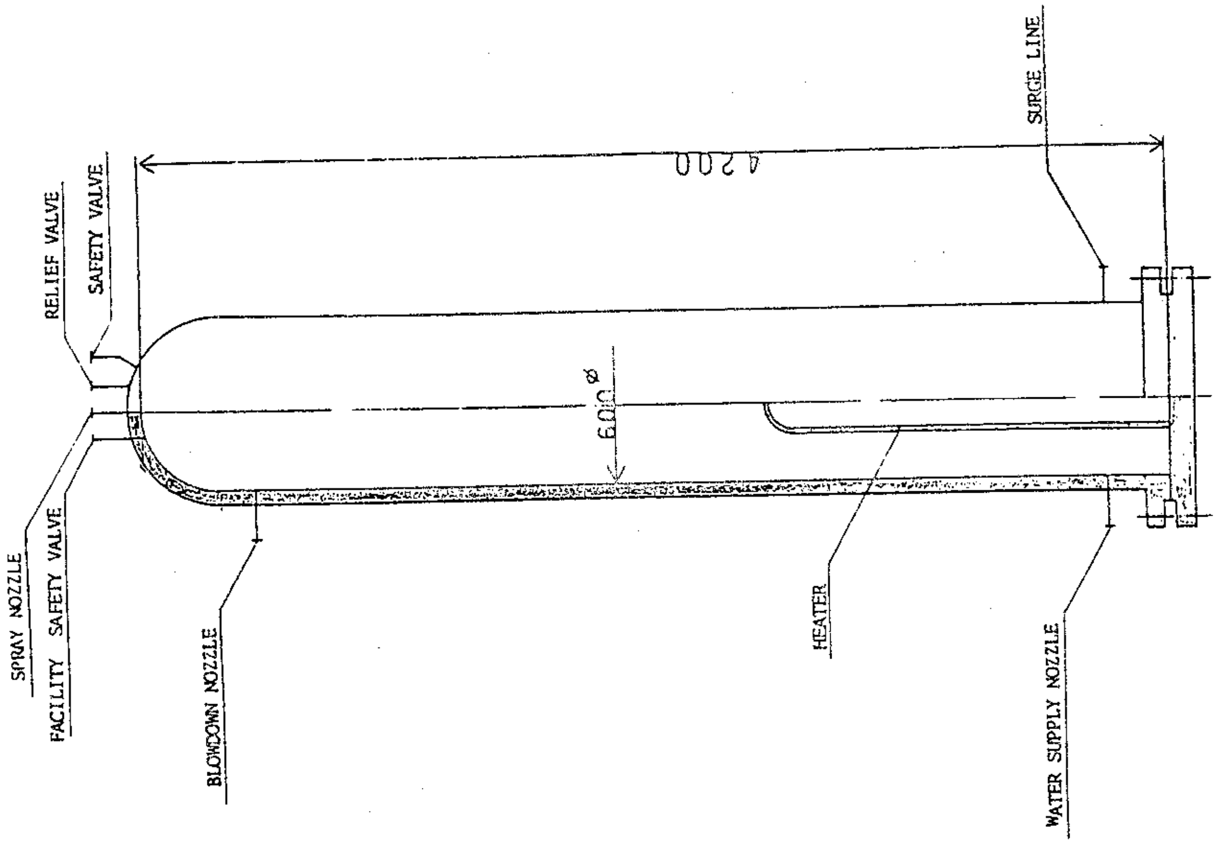

FIG.5.10 PRESSURIZER

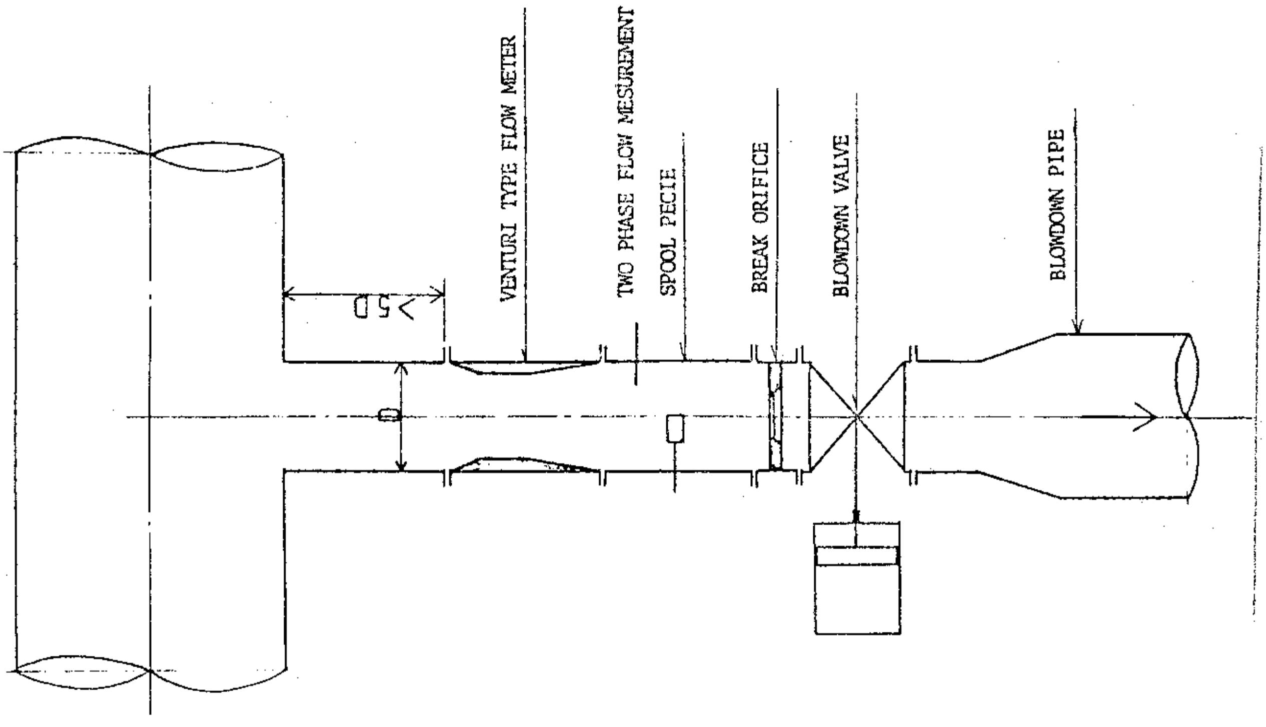

FIG.5.12 BREAK UNIT

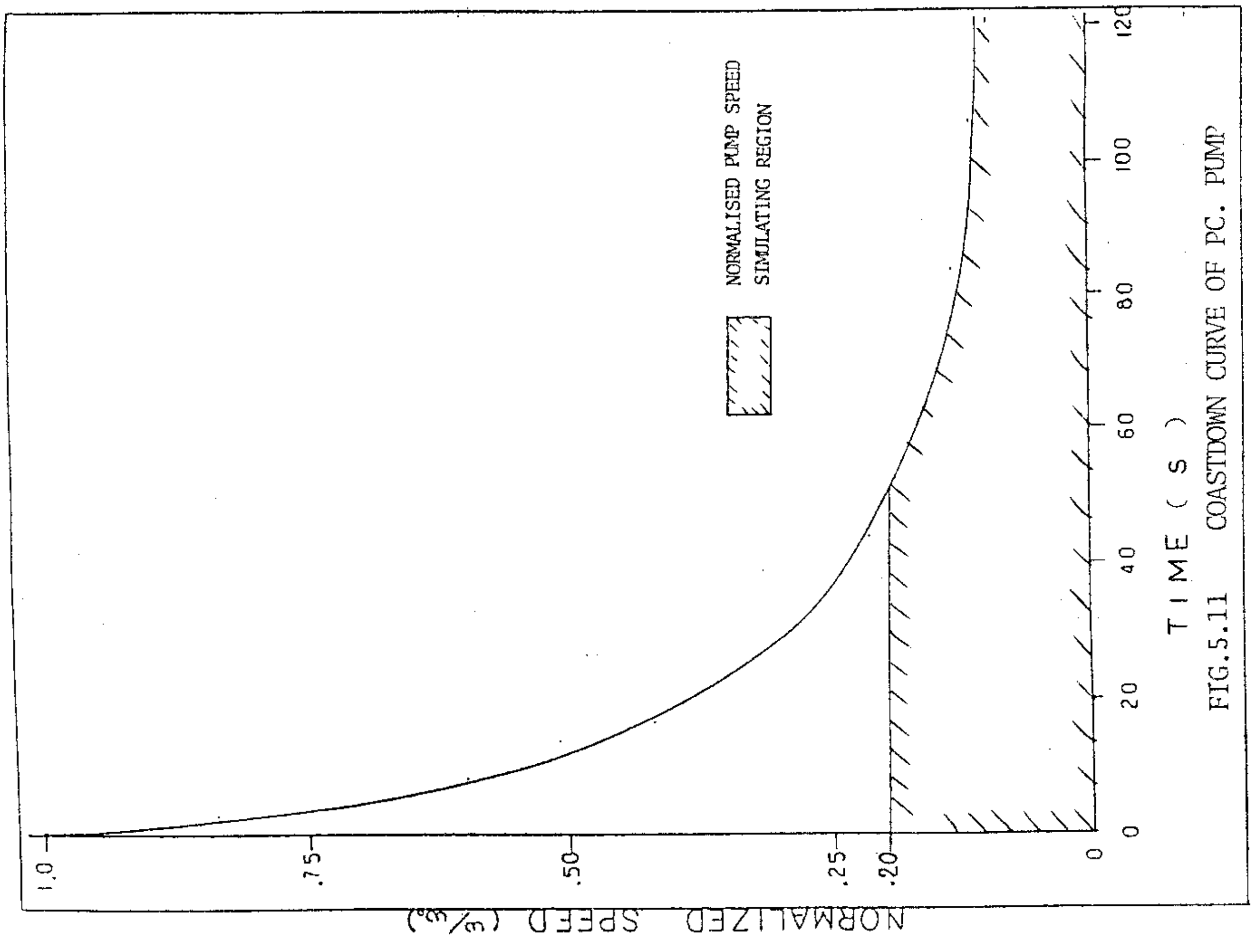

FIG.5.11 COASTDOWN CURVE OF PC. PUMP

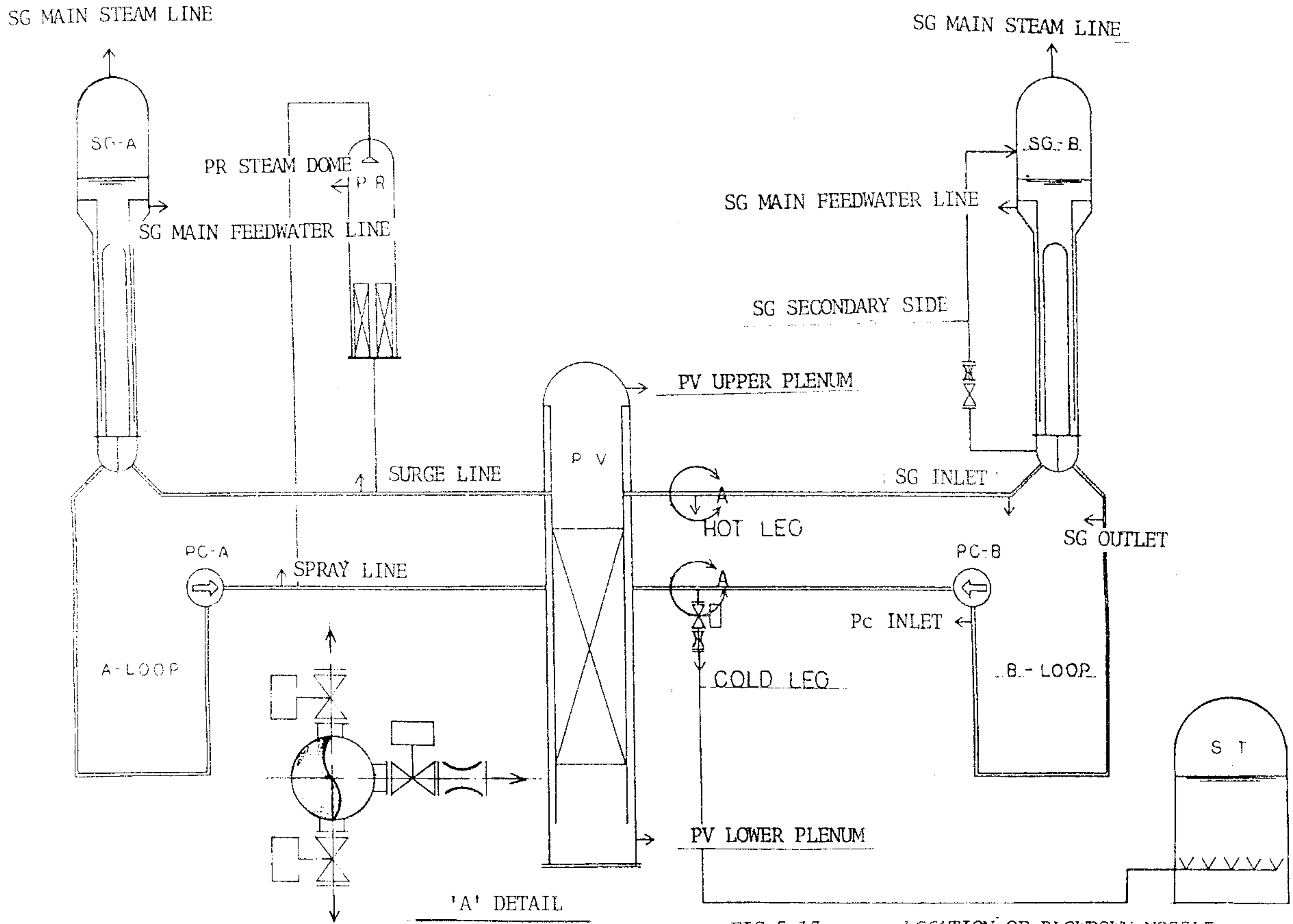

FIG.5.13 LOCATION OF BLOWDOWN NOZZLE

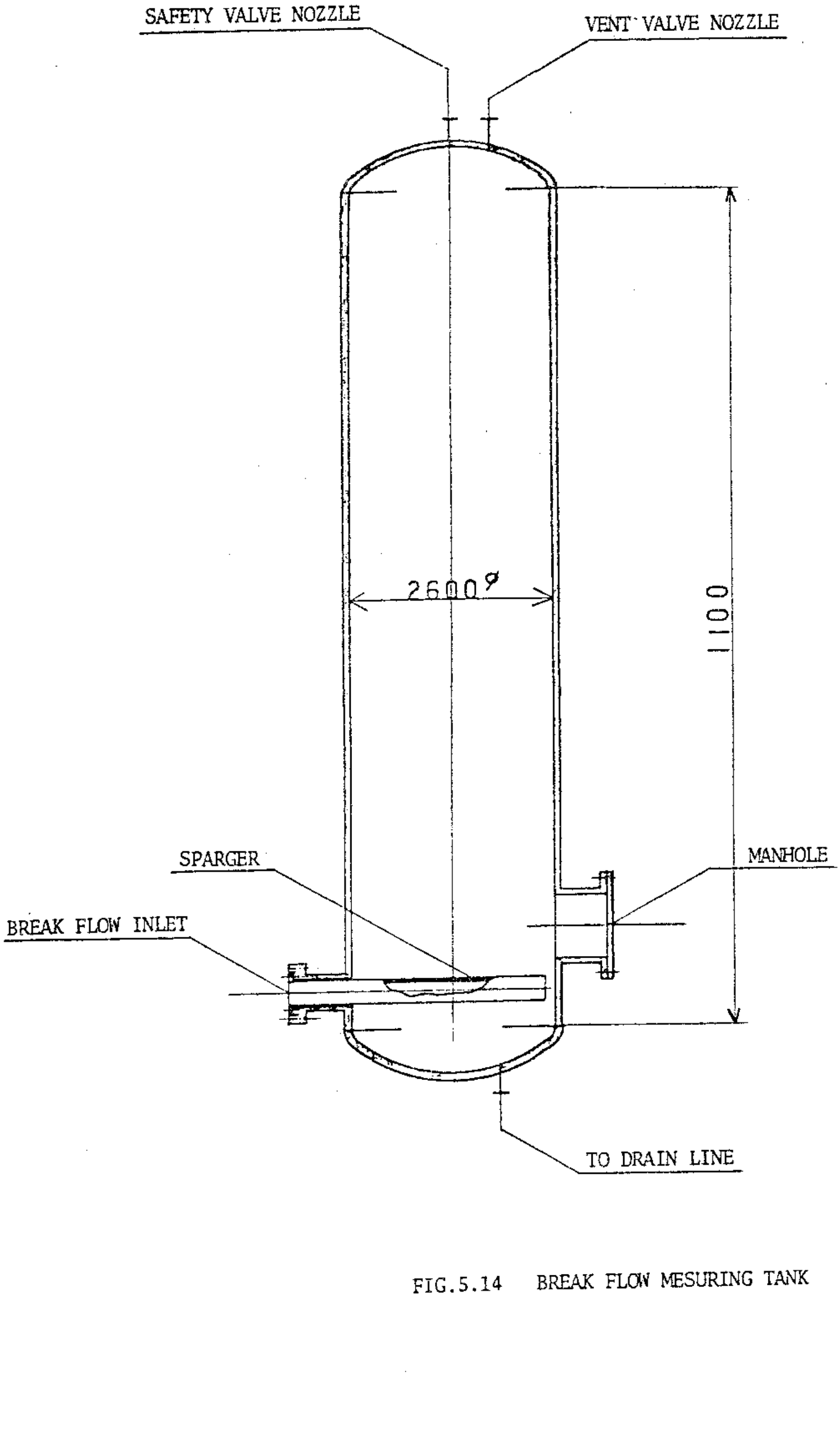

FIG.5.14 BREAK FLOW MESURING TANK

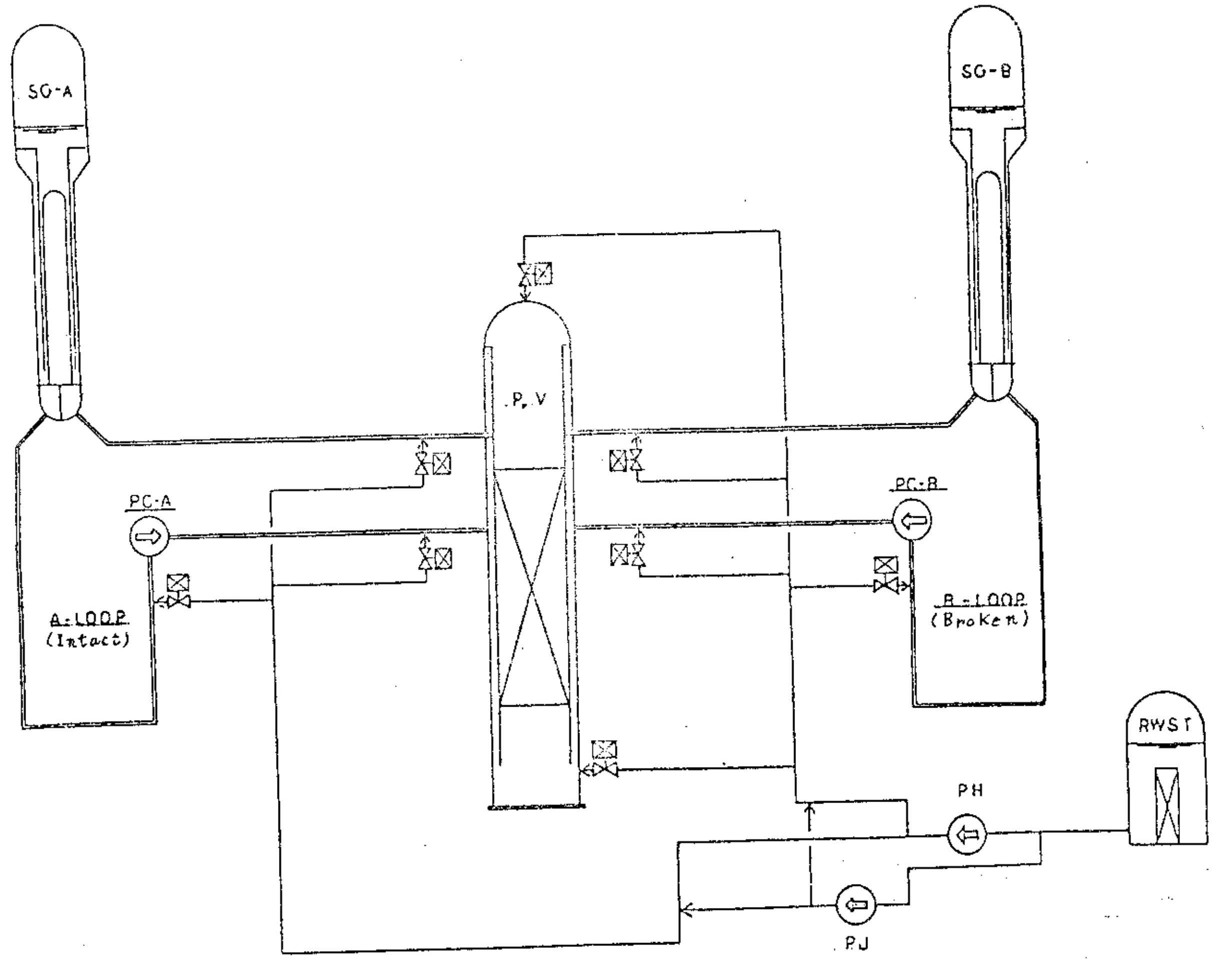

FIG.5.15 HP INJECTION SYSTEM FLOW DIAGRAM

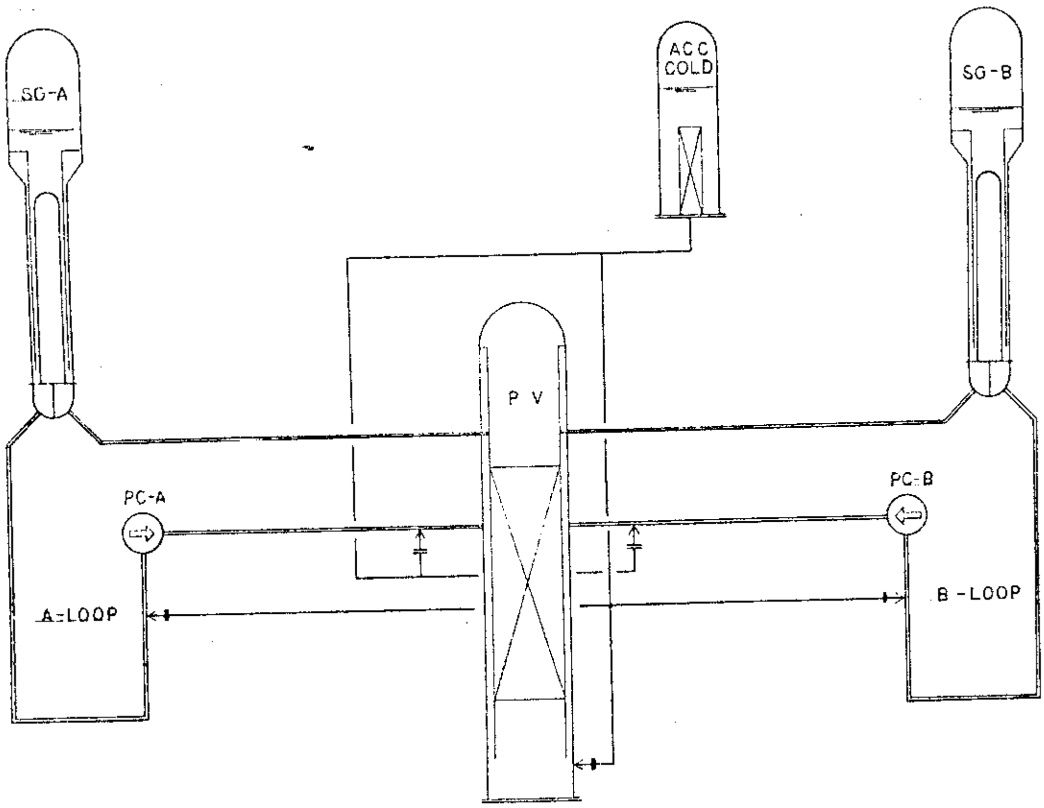

FIG.5.16 ACCUMULATOR (COLD) FLOW DIAGRAM

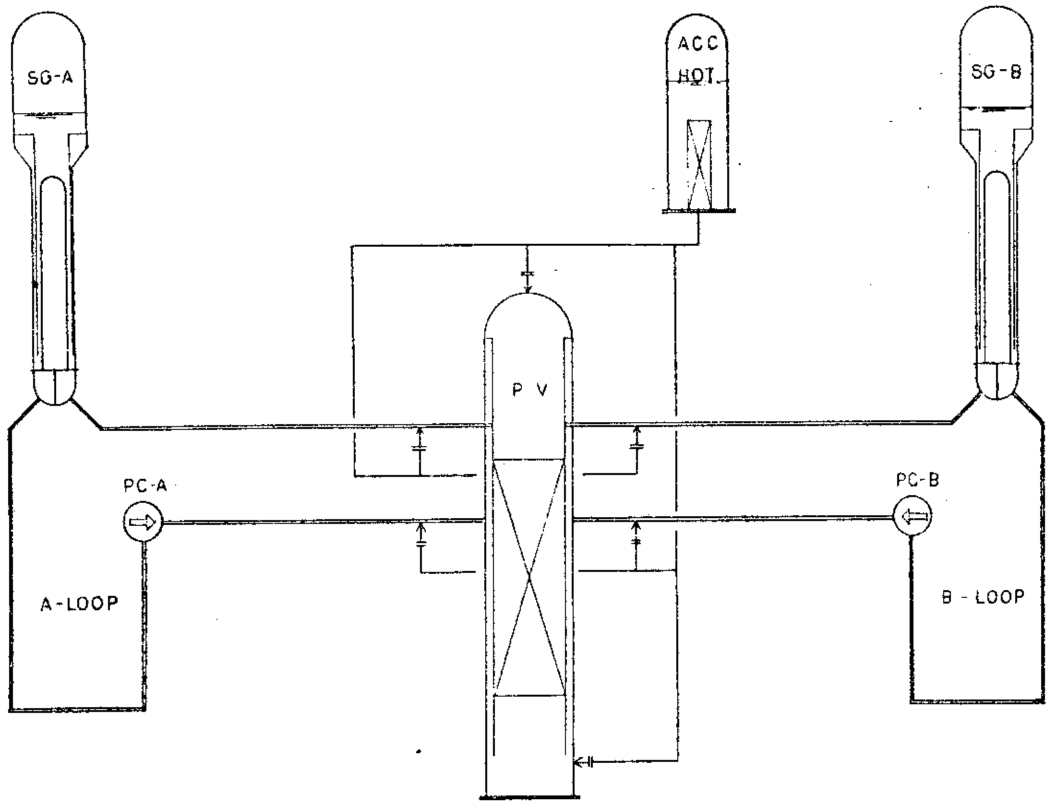

FIG.5.17 ACCUMULATOR (HOT) FLOW DIAGRAM

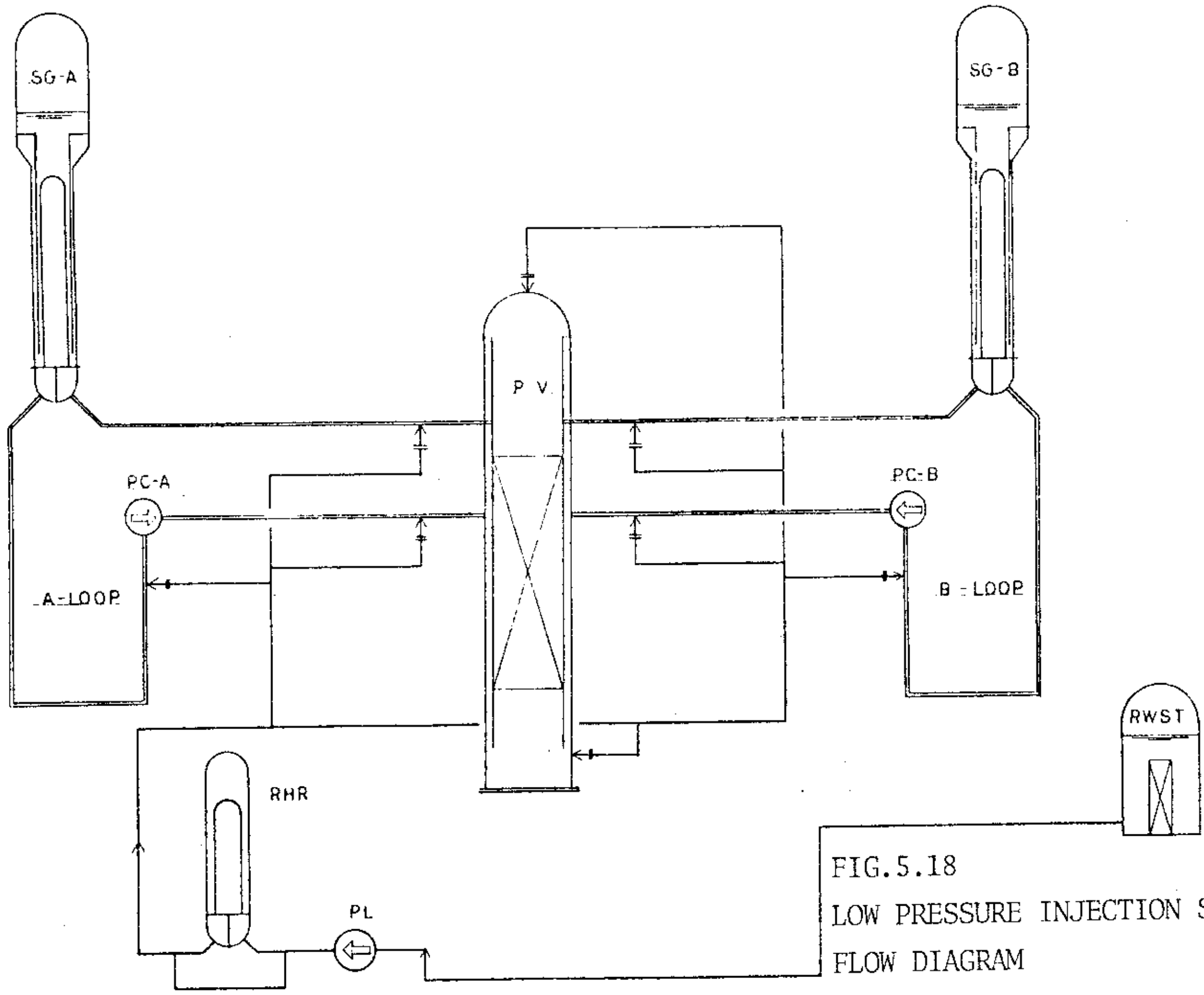

FIG.5.18
LOW PRESSURE INJECTION SYSTEM
FLOW DIAGRAM

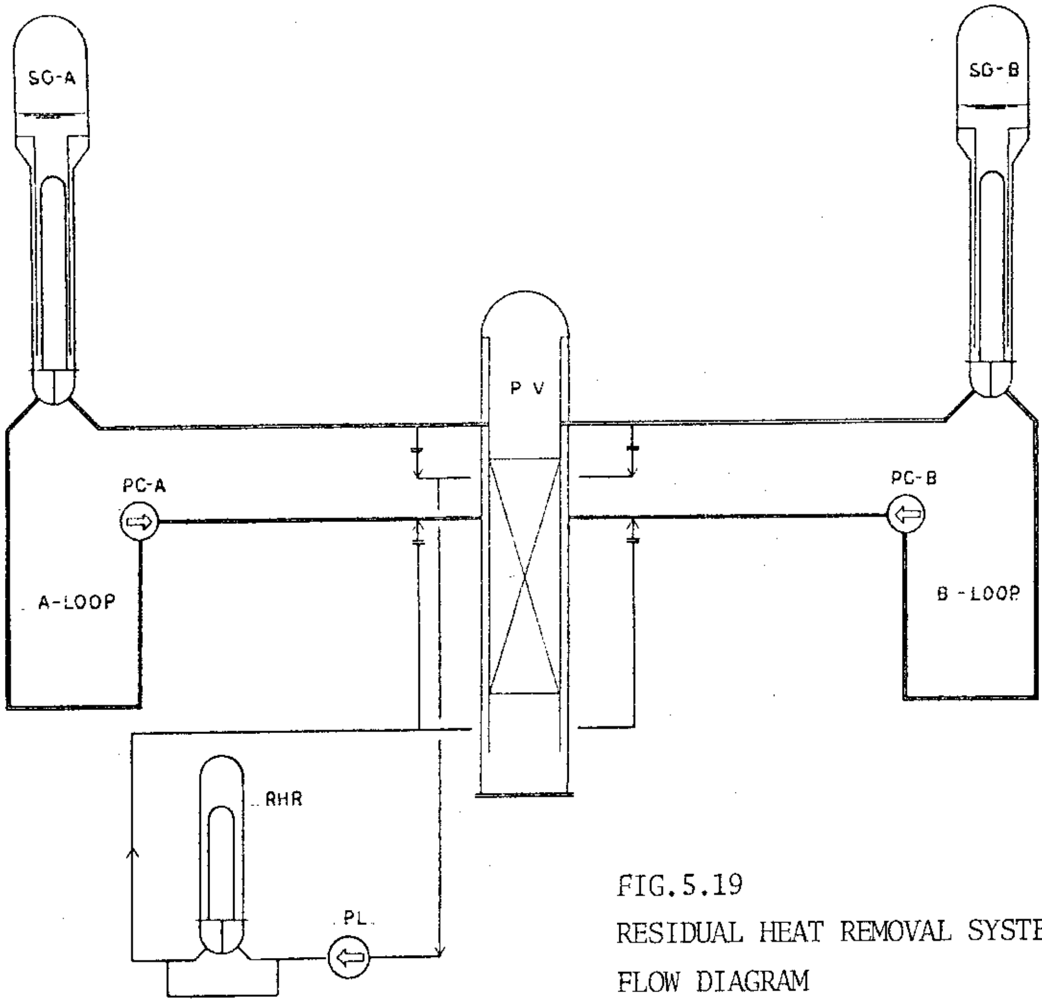

FIG.5.19
RESIDUAL HEAT REMOVAL SYSTEM
FLOW DIAGRAM

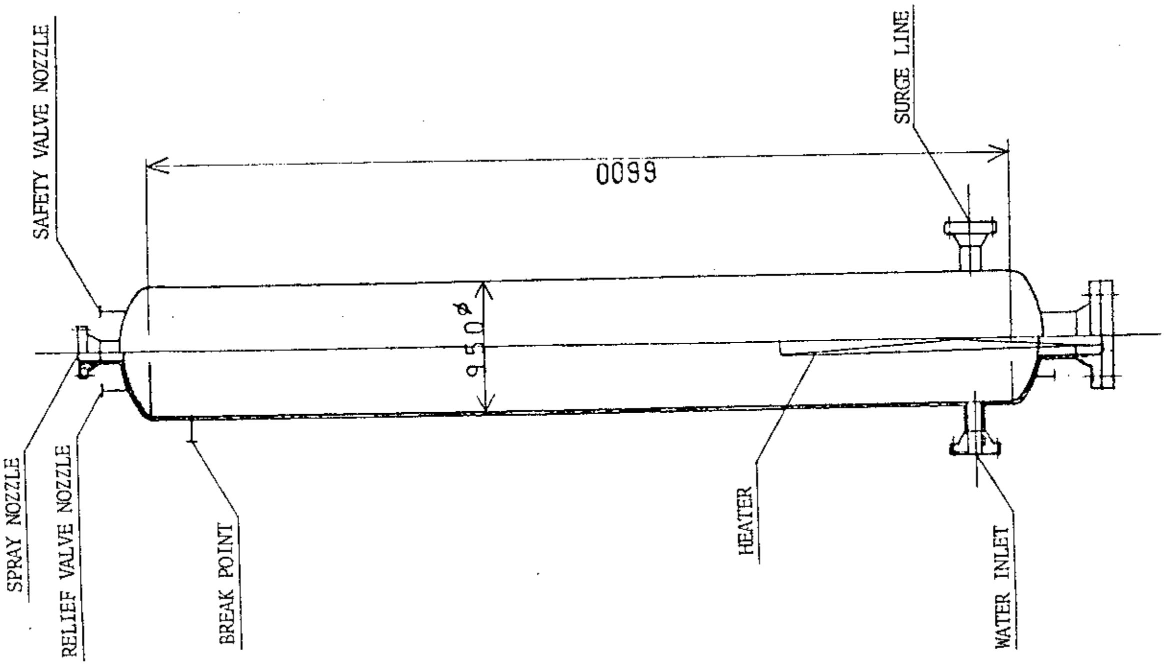

FIG.5.20 ACCUMULATOR (COLD)

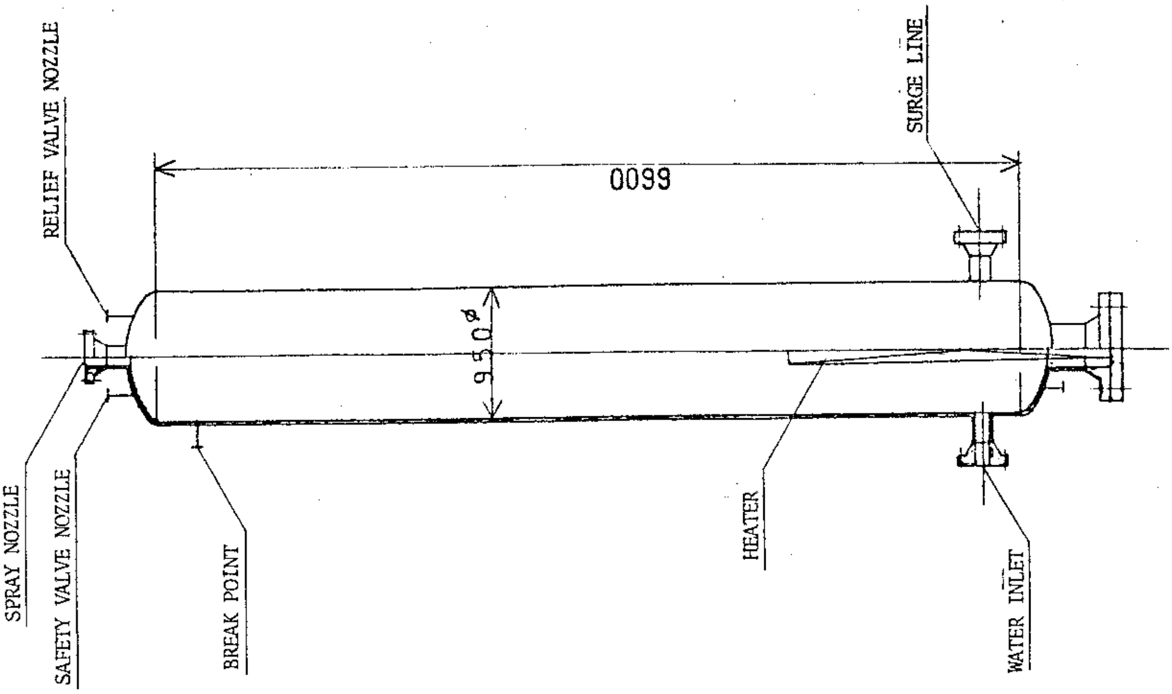

FIG.5.21 ACCUMULATOR (HOT)

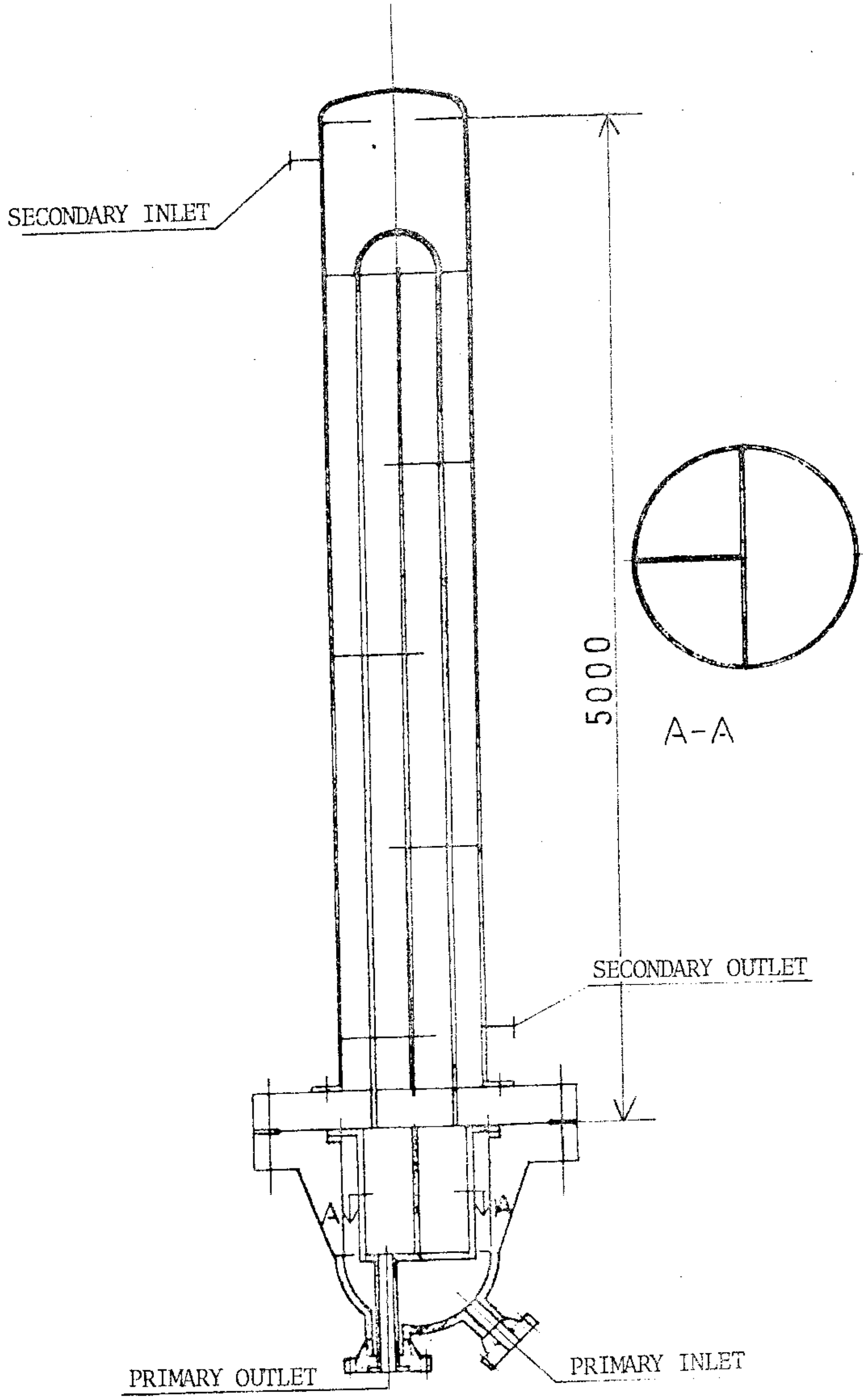

FIG.5.22 RESIDUAL HEAT REMOVAL HEAT EXCHANGER

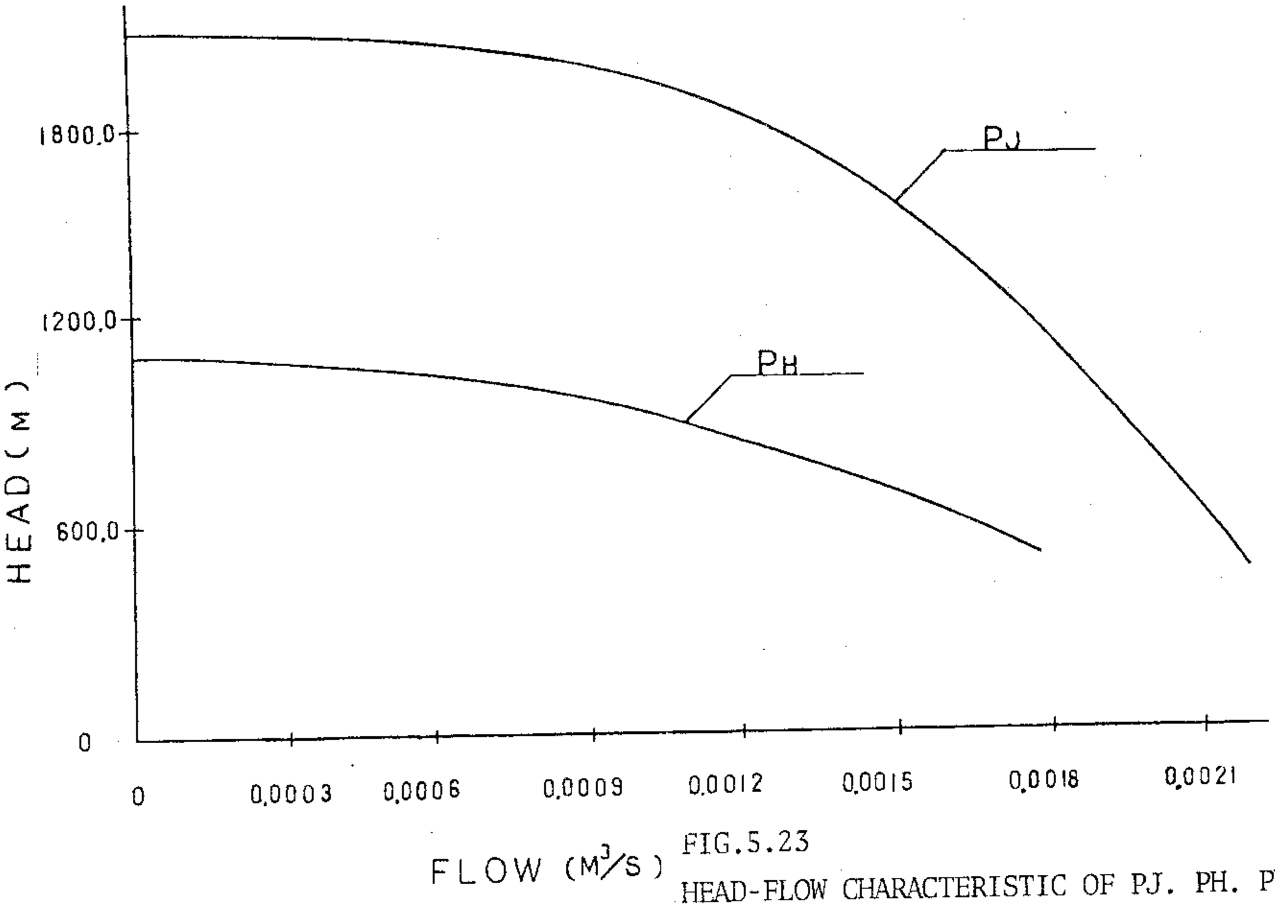

FIG.5.23
HEAD-FLOW CHARACTERISTIC OF PJ. PH. PUMPS

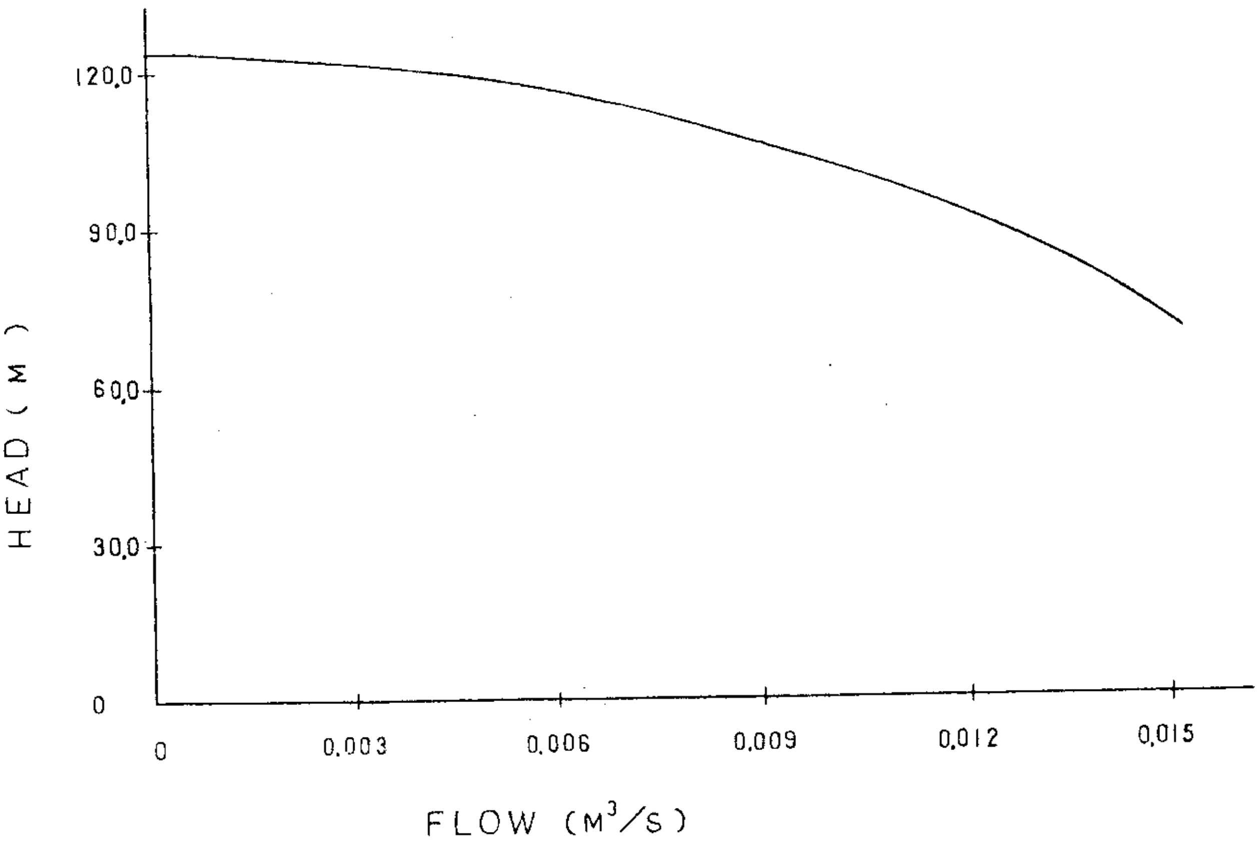

FIG.5.24 HEAD-FLOW CHARACTERISTIC OF PL. PUMP

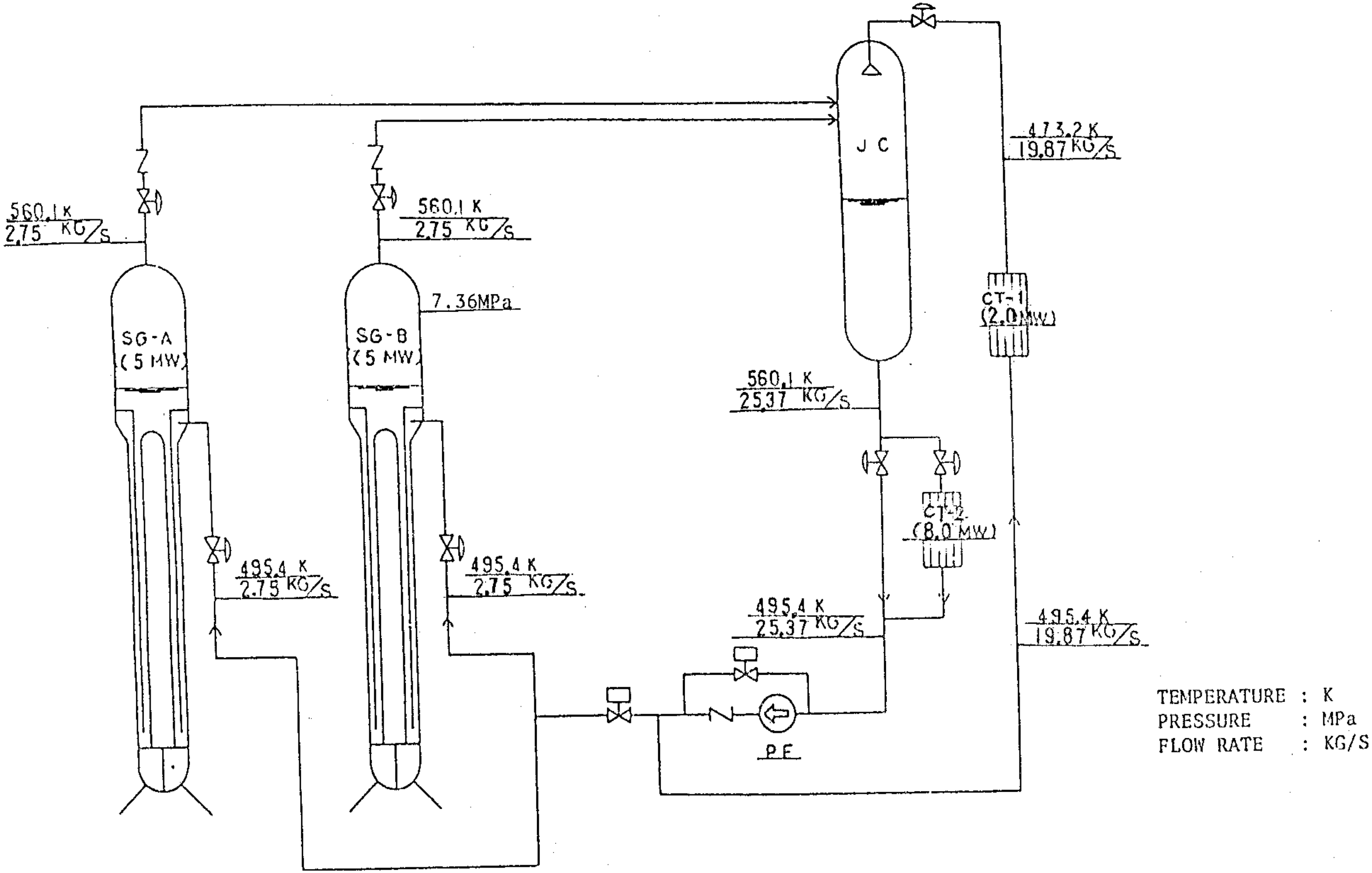

FIG.5.25 HEAT & MASS BALANCE IN SG SECONDARY SIDE

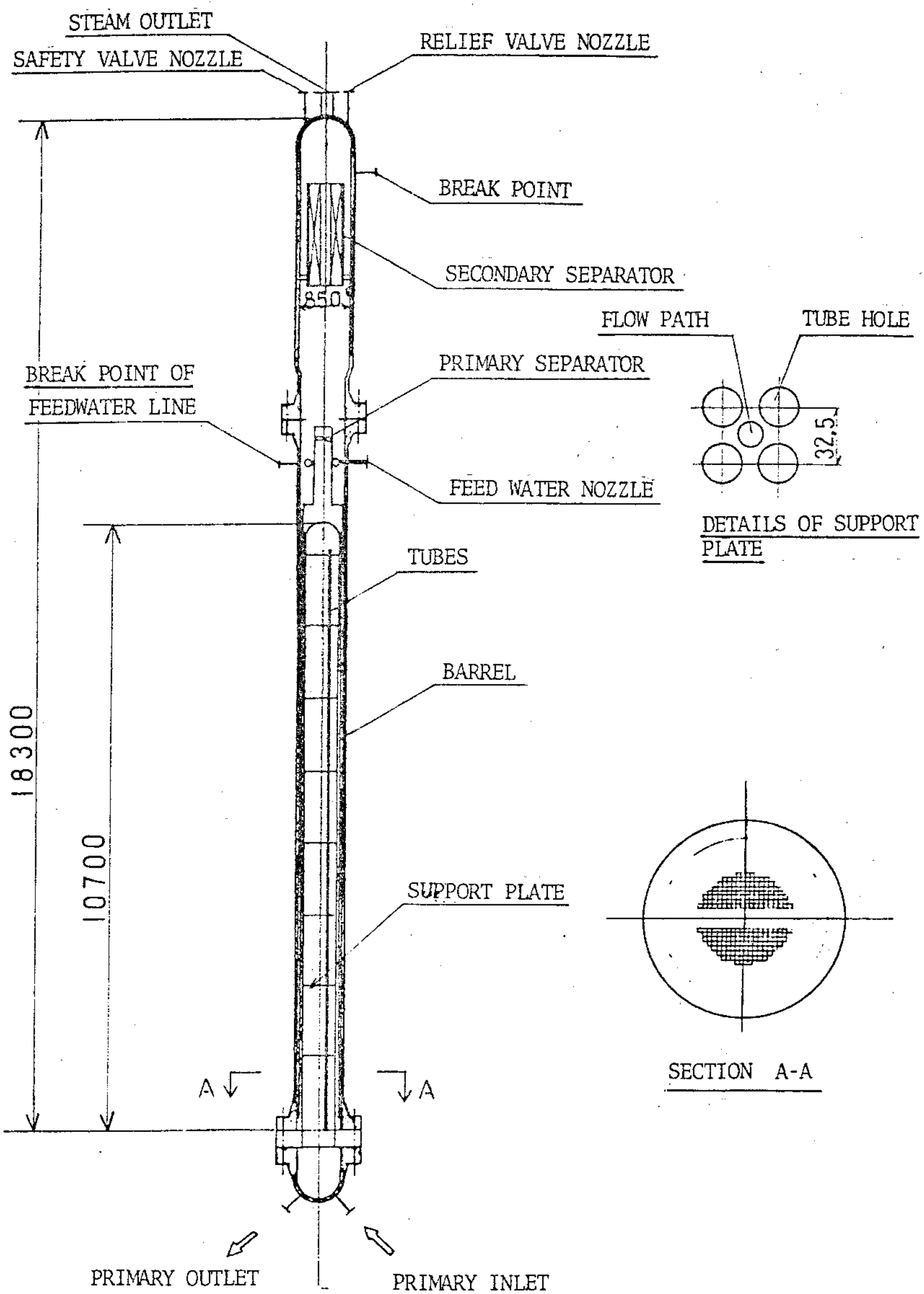

FIG.5.26 STEAM GENERATOR

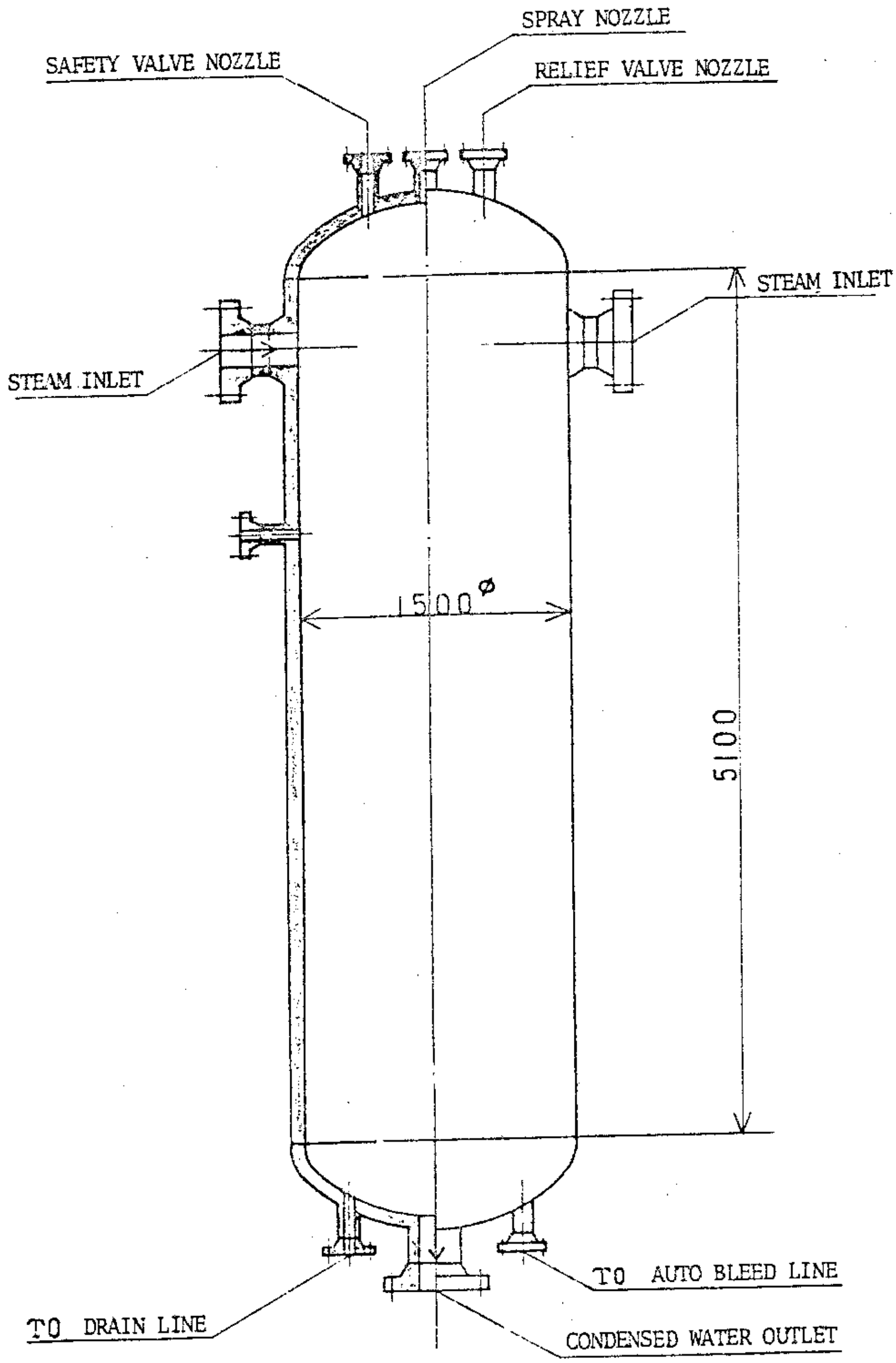

FIG.5.27 JET CONDENSER

6. Instrumentation

6.1 Measurement System

The experimental measurement system of LSTF is being designed to obtain thermo-hydraulic data during the simulated PWR small break LOCE and transient tests. The data obtained from these experiments will contribute to the assessment of an analytical computer code concurrently being developed. The system is similar to the measurement systems of similar test facilities such as CCTF, LOFT and Semiscale.

Instrumentation locations are shown in Fig. 6.1 through Fig. 6.5 and on Table 6.1.

Typical measured parameters in LSTF are pressure, differential pressure, flow rate, electric power, pump speed, fluid and metal temperatures, liquid level, coolant fluid density, on-off type signals and the like.

Pressure and differential pressure transducers are two-wire, direct-current type which convert diaphragm displacement to electric capacitance. The pressure lead pipes are either the standard single, cylindrical pipes used in conjunction with condensate pots or, dual concentric cylinders thus allowing for the circulation of cooling water to prevent flashing of the fluid.

Flow rate is measured by either orifice, venturi or flow-nozzle type flow meters depending on the fluid condition and measurement location.

The temperatures of the fluid, structural materials and fuel rod cladding are measured with Chromel-Alumel thermocouples (CA T/C) of 1.6 mm ϕ, or 0.5 mm ϕ. Ungrounded sheathed thermocouples of 1.0 mm ϕ are used as one of the electrodes in the conductivity liquid level detector described below.

Liquid levels are measured by either differential pressure transducers, of the type described above, or needle type electrical conductivity probes developed in the ROSA-III program. The probe is assembled in the shape of a rod with conductivity sensors distributed along its length to detect the existence of water or vapor at each level.

Electric power for the simulated fuel rods is controlled as a predetermined function of time and is measured by a fast response electric power meter.

Pump speed is measured by a pulse generator integral to the pump. On-off signals such as selected valve positions, decay heat and pump coastdown simulation initiations and the like are sent to a hard-copy device (printer) in order to record the exact signal actuation time.

Fluid density in the pipe is measured by means of gamma-ray densitometers. Preliminary studies indicate three-beam densitometers should be used to determine the flow regime. The gamma source is ^{137}Cs and the detector is a water cooled NaI (Tl) scintillation type.

In addition to the instrumentation described above, the measurement techniques for two-phase (steam-water) conditions are under development and the prototype of a momentum flux detector is being tested in the ROSA-III test program.

6.2 Data Aquisition System (DAS)

The DAS consists of standard devices such as a central processing unit (CPU), process input/output unit, auxiliary memory devices, etc. The data is recorded on magnetic disks (up to 2000 channels) and after being compiled on magnetic tapes is ready for off-line processing by the FACOM M200 system at JAERI. After evaluation, for example by comparing the initial and final pressure values with standard values, the data is reprocessed using the correct conversion factors as determined from the consistency examination.

6.3 Control System

The control system of LSTF is designed to monitor and maintain all operationally important facility parameters, such as pressure, flow rate, fluid temperature and fluid level, and to perform the sequential control required during various experiments. The control system flow diagram of LSTF is shown in Fig. 6.6.

The facility is equipped with a control room which contains the control and monitoring instrumentation necessary for operation of the facility and the DAS.

6.3.1 Pressure Control System

Pressure control is provided for the pressurizer (PR), the accumulators (ACC-Hot and ACC-Cold) and the jet condenser (JC). The PR pressure is designed to be controlled by PR spray, immersion heaters and a relief valve thus simulating the reference PWR control system.

6.3.2 Flow Control System

Flow rate control is provided for the steam generator (SG) feedwater flow, main steamline flow, high pressure injection (HPI) flow and low pressure injection (LPI) flow. The feedwater and main steamline flow controls are designed to simulate the secondary coolant system behavior of the reference PWR (such as turbine bypass and main feed to auxiliary feed change-over). The HPI and LPI flow controls are designed to simulate the reference PWR HPI and LPI pumps by regulating the plunger pump stroke or the flow control valve as a function of primary coolant pressure based on the reference pump performance curves.

6.3.3 Fluid Temperature Control System

Fluid temperature control is provided for the residual heat removal heat exchanger (RHR-Hx) outlet water, SG feedwater and JC spray water. The SG feedwater temperature control system is designed to maintain the water temperature at the desired value by regulating the flow control valve at the cooling tower outlet.

6.3.4 Fluid Level Control System

Fluid level control is provided in the PR, SG secondary, ACC-Hot and ACC-Cold, water storage tank (RWST) and the like. The SG level is controlled by a three-element feedwater controller designed to regulate the feedwater valve by continuously comparing the feedwater flow signal, the water level signal and the pressure compensated steam flow signal. The SG feedwater flow is also able to be controlled independent of level control by control mode selection. Other level controls are designed to maintain the water level at the desired value prior to initiation of the experiment.

6.3.5 Sequential Control System

Sequential control by the computer system of the DAS is provided during various experiments and is composed of timing controls such as core decay heat simulation, pump coastdown simulation, blowdown initiation, ECC injection and main feedwater to auxiliary feedwater change-over.

TABLE 6.1 NOMENCLATURE

SYMBOL LIST FOR EXPERIMENTAL INSTRUMENTATION

T	Fulid Temp.
TW x2	Metal Wall Temp. (Inside wallx1, Outside wallx1)
L	Water Level
P	Absolute Pressure
dP	Differential Pressure
cP	Conduction Probe
cPT	Conduction Probe with T/C (Temp)
F	Flow Rate (Flow nozzle , Orifice or Ventury type)
M	Fluid Velocity (Multi-points pitot tube)
TQ	Shaft Torque
V	Vibration
R	Rotating Speed
W	Electric Power
D	Fluid Density (3-beam gamma densitometer)
2P	Spool Piece for Two-phase Flow Measurement :
	1-Drag disc
	1-Densitometer (Vibrating Vane type)
	1-Absolute pressure
	1-Fluid temp
	1-Pitot tube rake (6-Pitot tube with 3-T/C)
S	Valve Status (open/close)
O	Valve Lift
TV	TV Camera

TABLE 6.1 NOMENCLATURE (CONTINUED)

SYMBOL LIST FOR CONTROL SYSTEM INSTRUMENTATION

◇C	Cascade / Computer Control
△C	Sequential Control
INTLK	Interlock
SWITCH	Signal Change-over
SELECT	Signal Selection
MEAN	Mean Value Calculation
PROG	Control Program
HC	Hand Controller
☐CIA	Digital Controller with Data Indication and Alarm
☐IA	Digital Data Indication with Alarm
☐A	Process Alarm
☐R	Analog Recorder
☐I	Analog Indicator
V	Vibration Detector
R	Revolution Transducer
W	Electric Power Transducer
F	Flow Transmitter
L	Liquid Level Transmitter
P	Pressure Transmitter
T	Thermocouple
	Air Operated Control Valve
AO	Air Operated Solenoid Valve
STP	Pump Motor Starter
HTR	Electric Heater
X	Break Location
☆	ECCS Injection Location

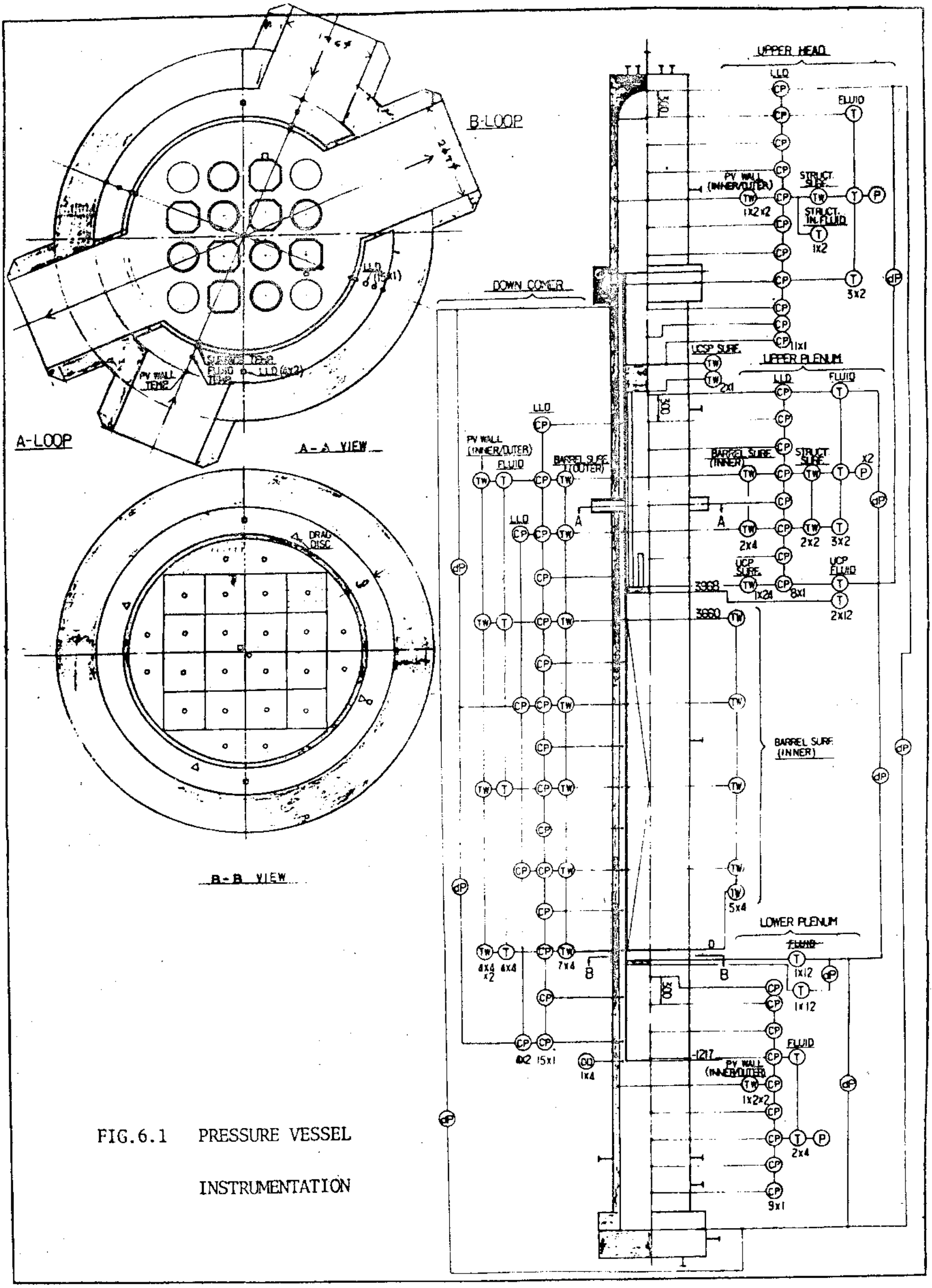

FIG.6.1 PRESSURE VESSEL INSTRUMENTATION

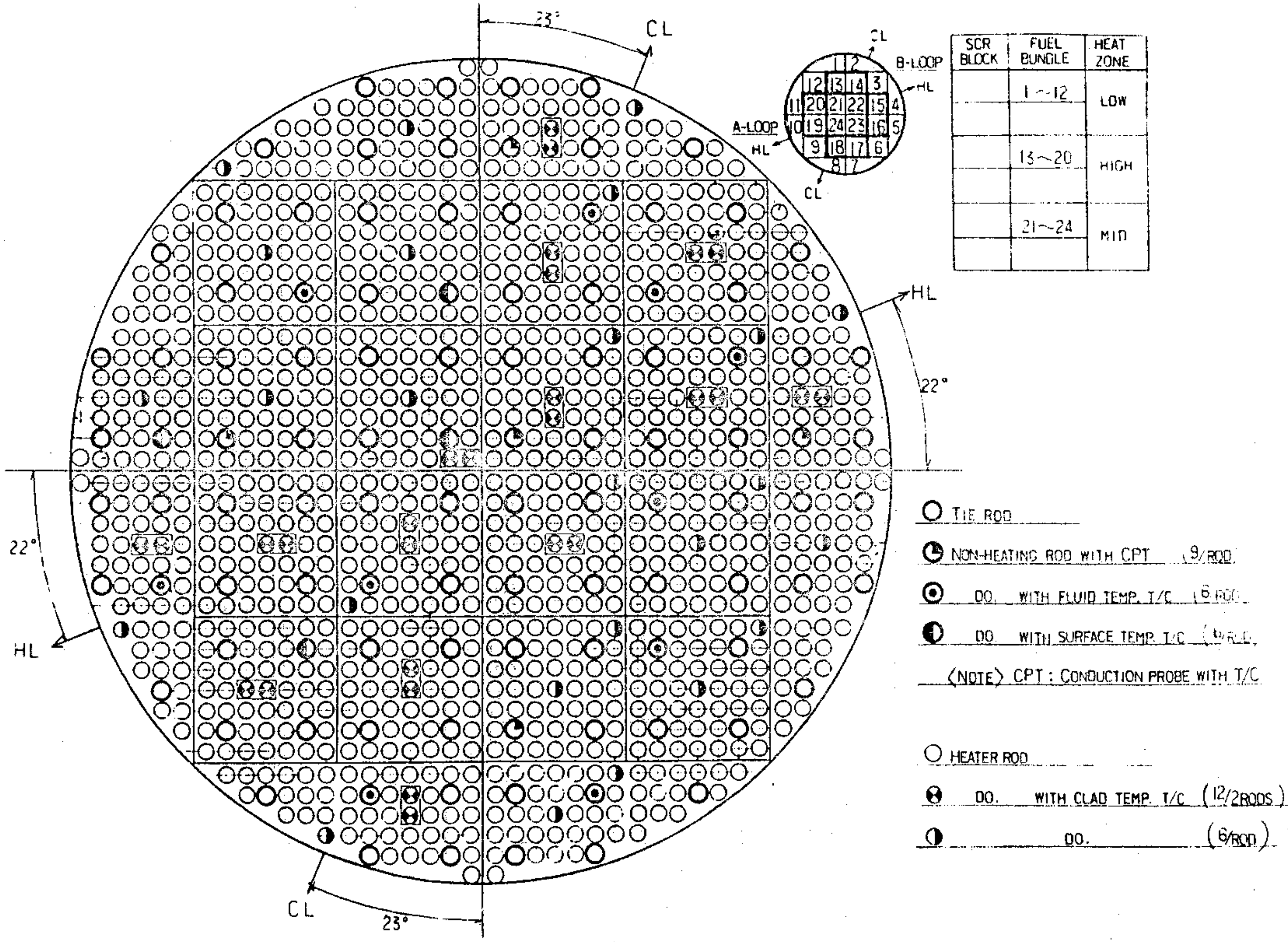

FIG.6.2 IN-CORE INSTRUMENTATION (HORIZONTAL)

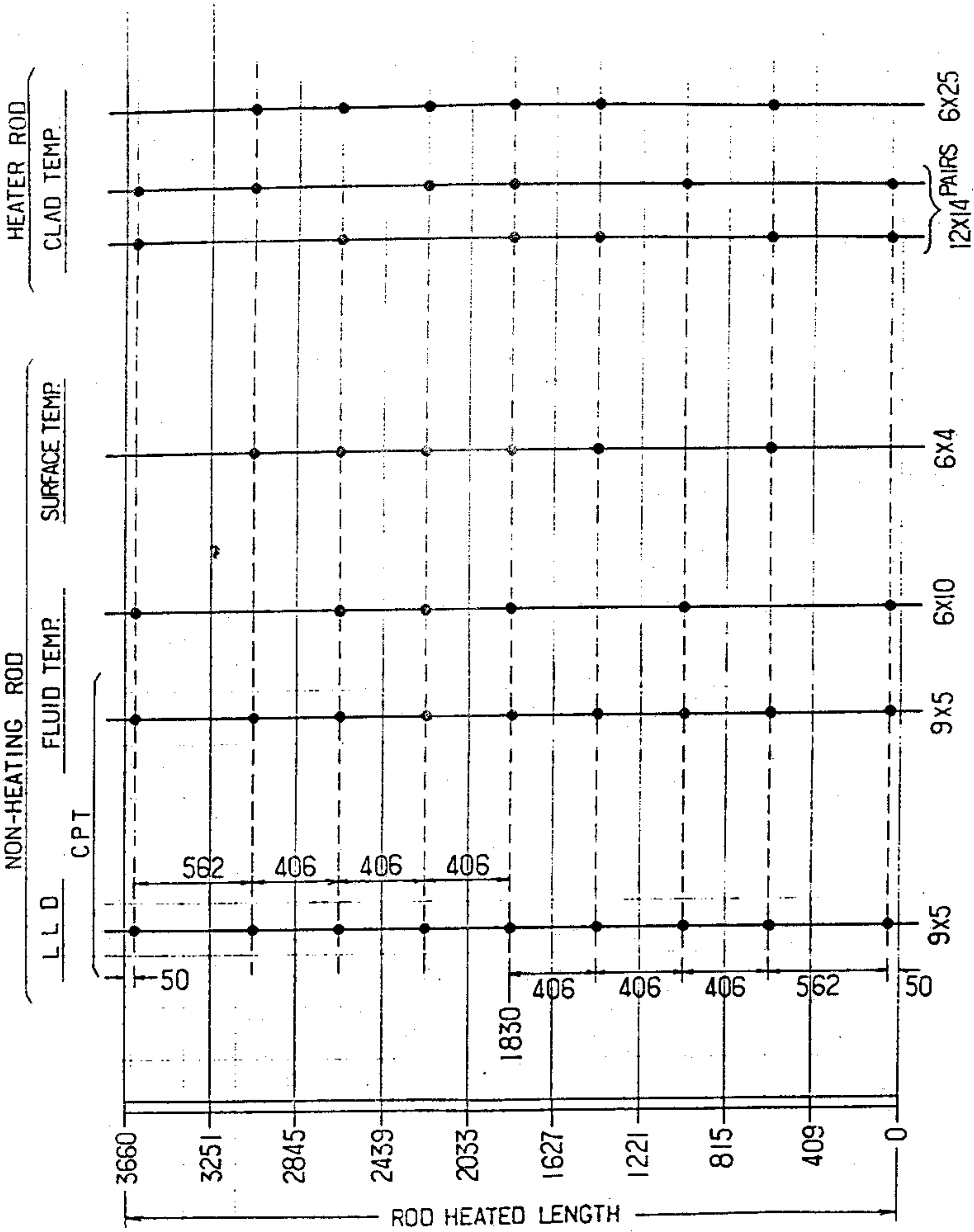

FIG.6.3 IN-CORE INSTRUMENTATION (VERTICAL)

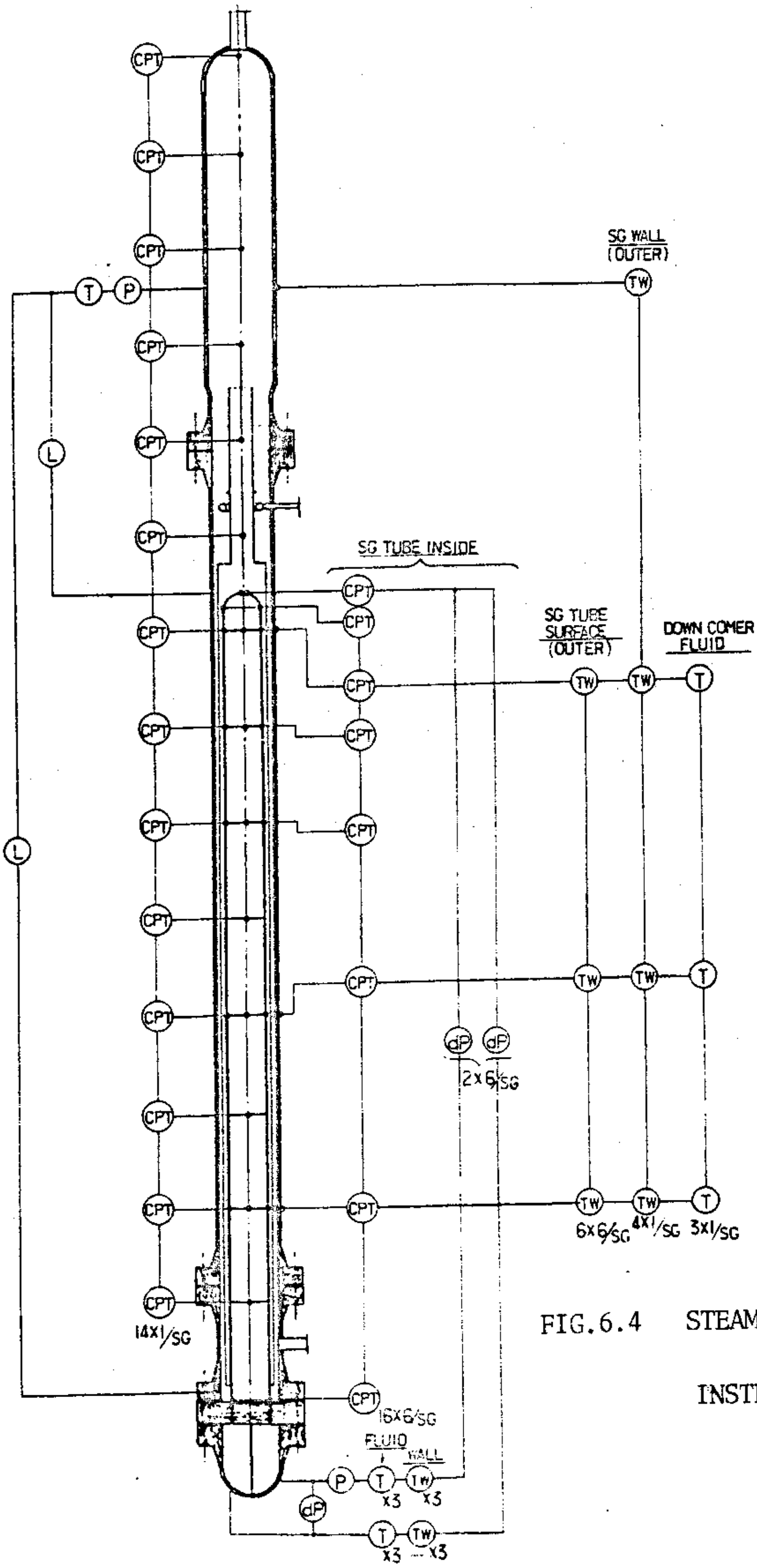

FIG.6.4 STEAM GENERATOR INSTRUMENTATION

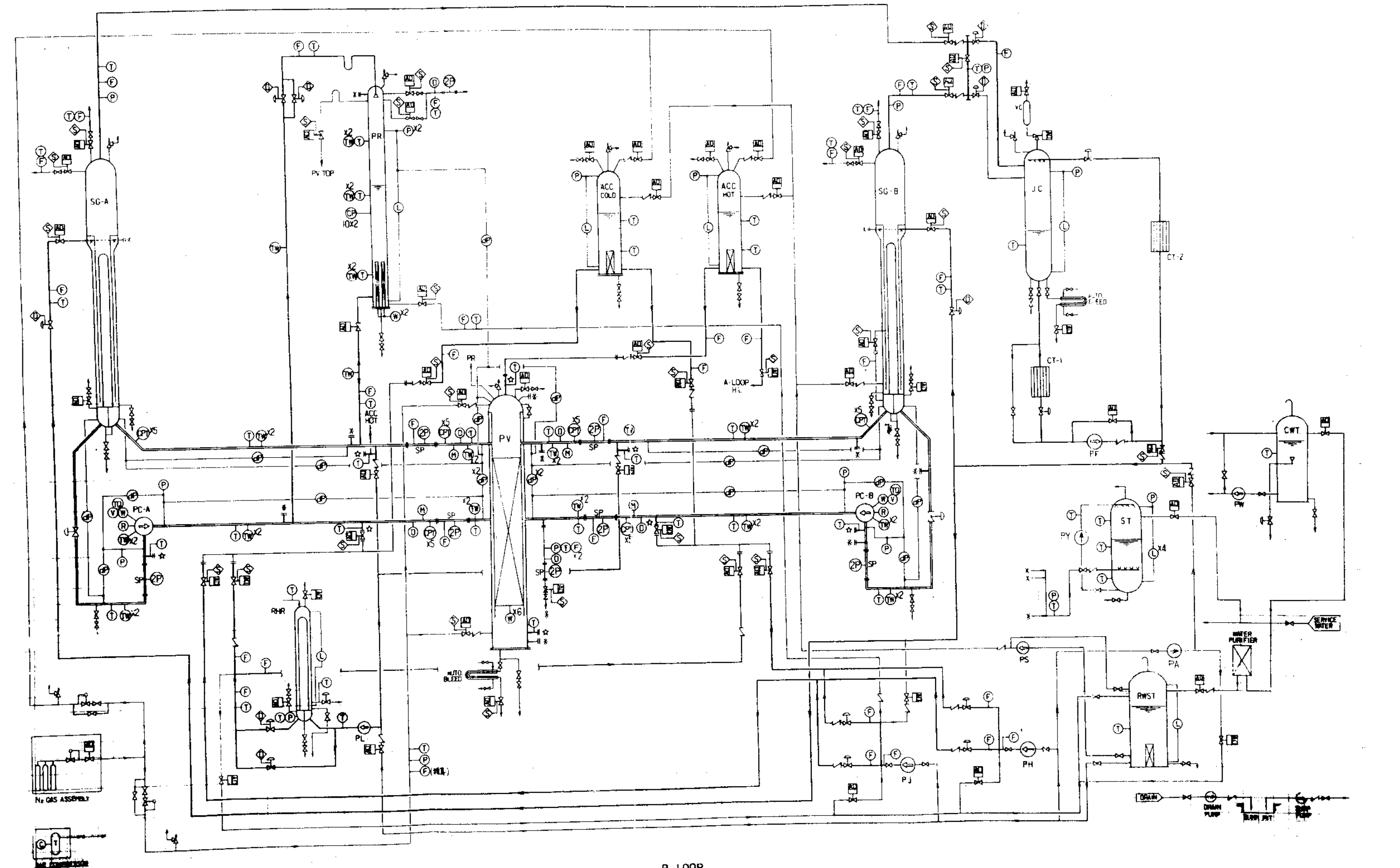

FIG.6.5 EXPERIMENTAL INSTRUMENTATION FLOW DIAGRAM

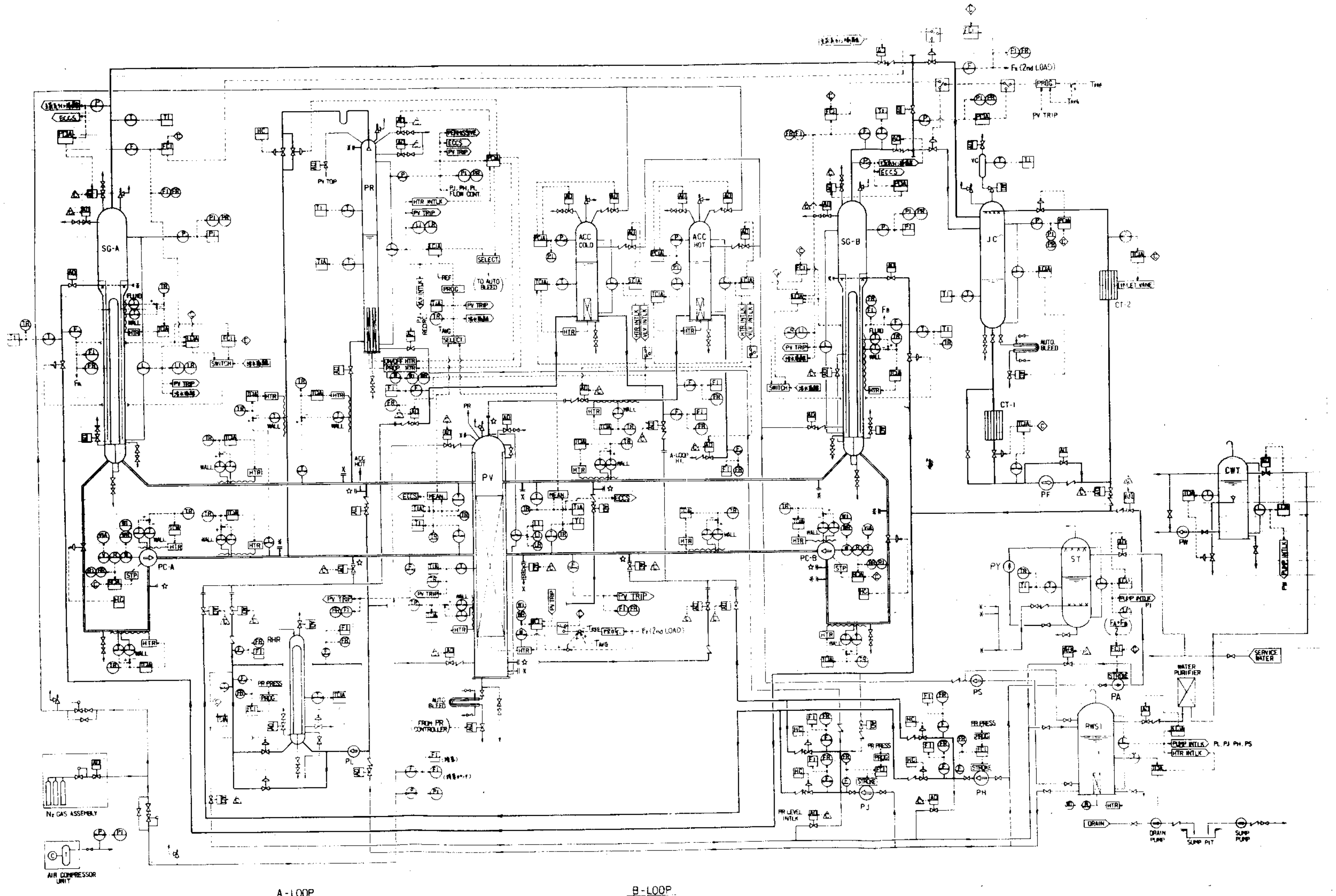

FIG.6.6 CONTROL SYSTEM FLOW DIAGRAM

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