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Thermal-Hydraulic Analysis Code Development and Application to Passive Safety Reactor at JAERI

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1. Safety Assessment

(1) EM Code and BE Code

EM (Evaluation Model) Code
Assure Conservativeness of calculated Results
Adoption of Conservative Assumption
(Ex. ; inhibition of rewetting of fuel rod after DNB occurrence)
Applied to Safety Evaluation
Example......WREM

BE (Best Estimate) Code
Reproduce Actual Phenomena
Elimination of Conservative Assumption
Applied to Prediction of Actual Behavior and Evaluation of Conservativeness of EM Code
Example.......TRAC-PF1, RELAP5, etc.

(2) Design Basis Events (PWR)

Gudeline for safety evaluation specifys the events to be analyzed.

Abnormal Transients

- Abnormal Changes in Reactivity or Power Distribution (4 transients)
  - (Withdrawal of Control Rod, Boron Dilution, etc)
- Abnormal Changes in Heat Generation or Heat Removal (7 transients)
  - (Partial Loss of Primary Coolant Flow, Loss of Off-Site Power, etc)
- Abnormal Changes in Primary Pressure or Coolant Inventory (3 transients)
  - (Depressurization of Primary System, Errorneous Actuation of ECCS)

Accident

- Loss of Primary Coolant or Degradation of Core Coolability (5 transients)
  - (Loss of Coolant Accident, Loss of Primary Flow, etc)
- Rapid Reactivity Insertion or Power Increase (1 transient)
  - (Control Rod Ejection)
- Abnormal Release of Radioactive Materials to Environment (5 transients)
  - (SG Tube Rupture, LOCA, etc)
- Abnormal Change in Containment Pressure, Atmosphere (2 transients)
  - (LOCA, Flammable Gas Generation)

(3)Assessment Criteria

Abnormal Transient

- -Minimum DNBR should be less than allowable limit
- -Fuel cladding should not be mechanically damaged
- -Fuel enthalpy should be less than allowable limit
- -Pressure in the primary coolant boundary should be less than 1.1 times of to maximum operating pressure

Accident

- -The reactor core should not be led to serious damage and can be cooled sufficiently
  - Fuel cladding temperature should be less than 1200 deg C
  - Oxidized thickness of fuel cladding should be less than 15% of initial thickness
- -Fuel enthalpy should be less than the specific limit
- -Pressure in the primary coolant boundary should be less that 1.2 times of the maximum operating pressure
- -Pressure inside the containment should be less than the maximum operating pressure
- -Risk by serious radiation exposure should not be given to the neighboring public

(4) Requirements for LOCA/ECCS Assessment

1. Break
   Position, Configuration, Area, Discharge Flow, Spectrum

2. Flow Behavior
   Governing Eq., Two-Phase Flow, Multi-Dimensionality or Parallel Channel, Pressure Drop, Pump Behavior, ECC Water Injection behavior, Non-condensable, Deformation of Flow Path, Initial Flow Conditions

3. Actuation of ECCS
   Actuation Signal, Electric Power Supply and Other Systems, Component Failure, Operator Action, Containment Pessure, Cooling Effict by ECCS

4. Heat Genaration
   Power and Power Distribution before LOCA, Core Power during LOCA, Stored Energy in Fuel, Decay Heat, Gamma Smearing, Metal-Water Reaction

5. Fuel Behavior
   Thermal Property, Suface Heat Transfer, Cladding Deformation, Oxidation of Cladding

6. Computer Program
   Organization, Input Data, Modeling of Reactor System, Sensitivity Analysis

2. Introduction and Development of LOCA Analysis Code

(1) *Introduction and Modification of Audit Calculation Code*

WREM Code System was introduced from USNRC
WREM-J2 was developed by addition of models (CCFL, etc.) to WREM and supplied to NUPEC/JINS for audit calculation in Japan

(2) Development of Own Code System

Japanese own code systems were developed at JAERI for PWR and BWR and published to OECD/NEA for public use and NUPEC/JINS
Technological basis for thermal-hydraulic analysis and code development have been accumulated.

(3) Introduction and Modification of BE Codes by perticipation to International Cooporation Programs

2D/3D Program
ICAP Program
ROSA Program

3. Application to Development of Passive Safety Reactor Concept

(1) Proposed Passive Safety Reactor Concepts at JAERI

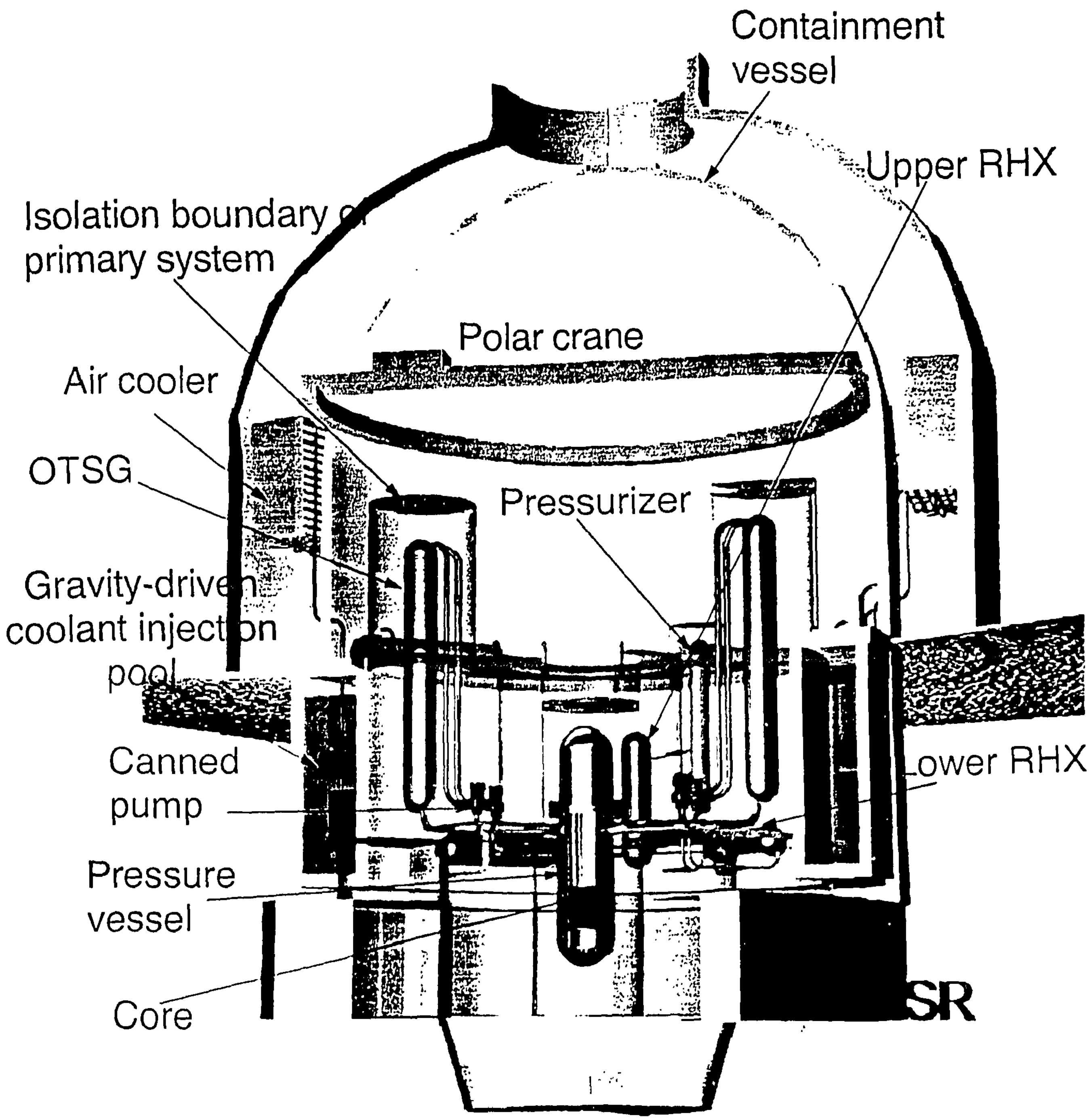
SPWR.... Integral type PWR
MRX.... Integral type marine reactor
JPSR.... Loop type PWR

Adoption of Passive Components

- Passive Cooling System by Natural Circulation
- Passive Coolant Injection System

(2) Development of JPSR Concept

Analysis.....	Confirmation of feasibility of concept
Experiment.....	Understand phenomena

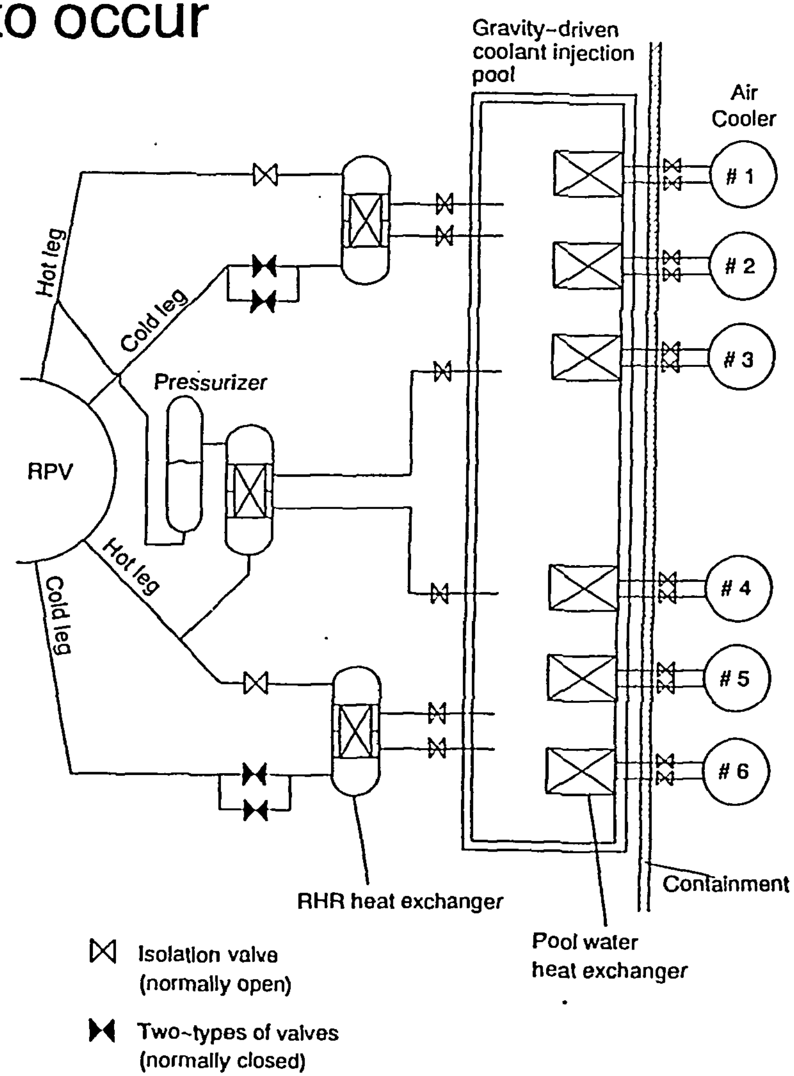


Conceptual design of JPSR

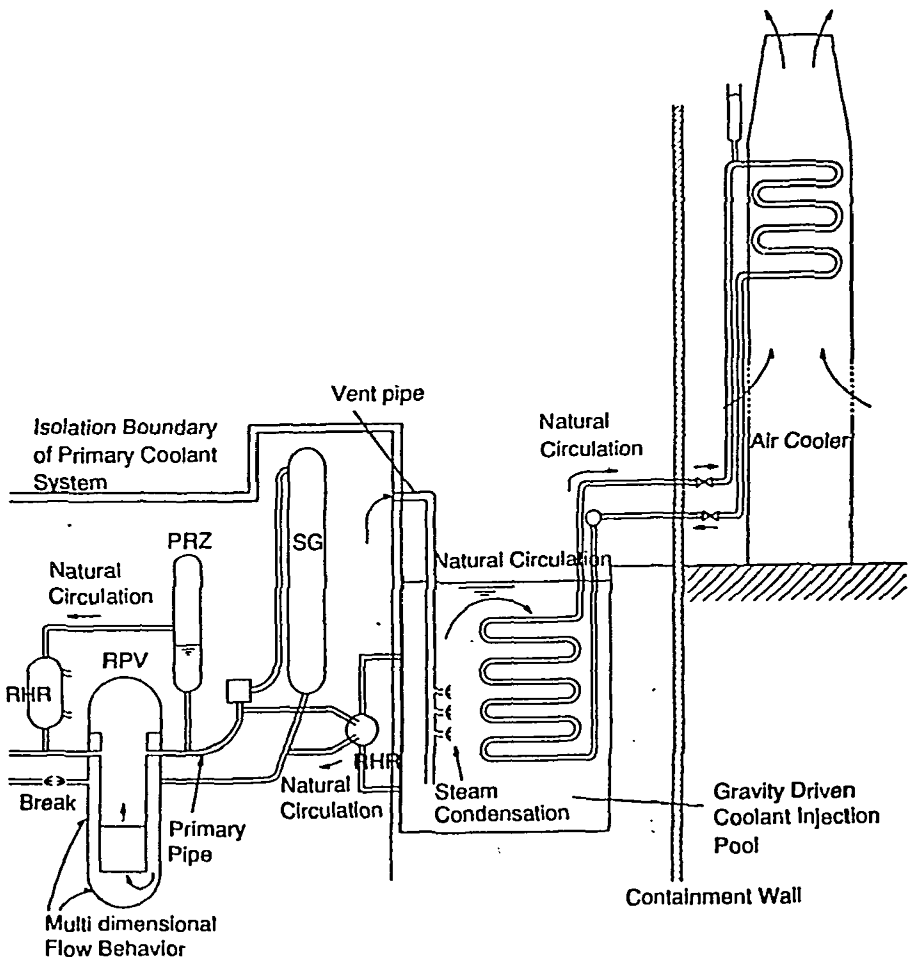
Major design parameters of JP-600

Reactor type	Two-loop PWR
SG type	OTSG
Contro rod drive mechanism(CRDM)	In-vessel CRDM
Primary coolant pump	Canned-motor pump
Thermal reactor power	1853 MWt
Operating pressure	15.6 MPa
Total primary coolant flow rate	32×10^6 kg/h
Core inlet temperature	558 K
Core outlet temperature	598 K
Pressure loss through the core	0.157 MPa
Primary coolant volume	319 m^3
Pressurizer volume (total/steam region)	48 / 30 m^3
Average linear heat generation rate	13.2 kW/m
Number of fuel bundle	145
Number of fuel rod per bundle	264
Outer diameter of fuel rod	9.5 mm
Thickness of clading	0.57 mm
Feal rod arrangement	Square latice of 17x17
Active fuel length	3.66 m
Axial peaking factor (BOEC)	1.2393
Radial peaking factor (BOEC)	1.6159
Coolant density reactivity coefficient (BOEC)	57.96 $/g/cm^3
Coolant temperature reactivity coefficient (BOEC)	-2.44×10^{-4} $/K
Doppler reactivity coefficient (BOEC)	-4.02×10^{-3} $/K
Scram reactivity (BOEC)	-42.18 $

(3) Passive Heat Removal System and Phenomena considered to occur



Passive Heat Removal Systems



Phenomena considered to occur

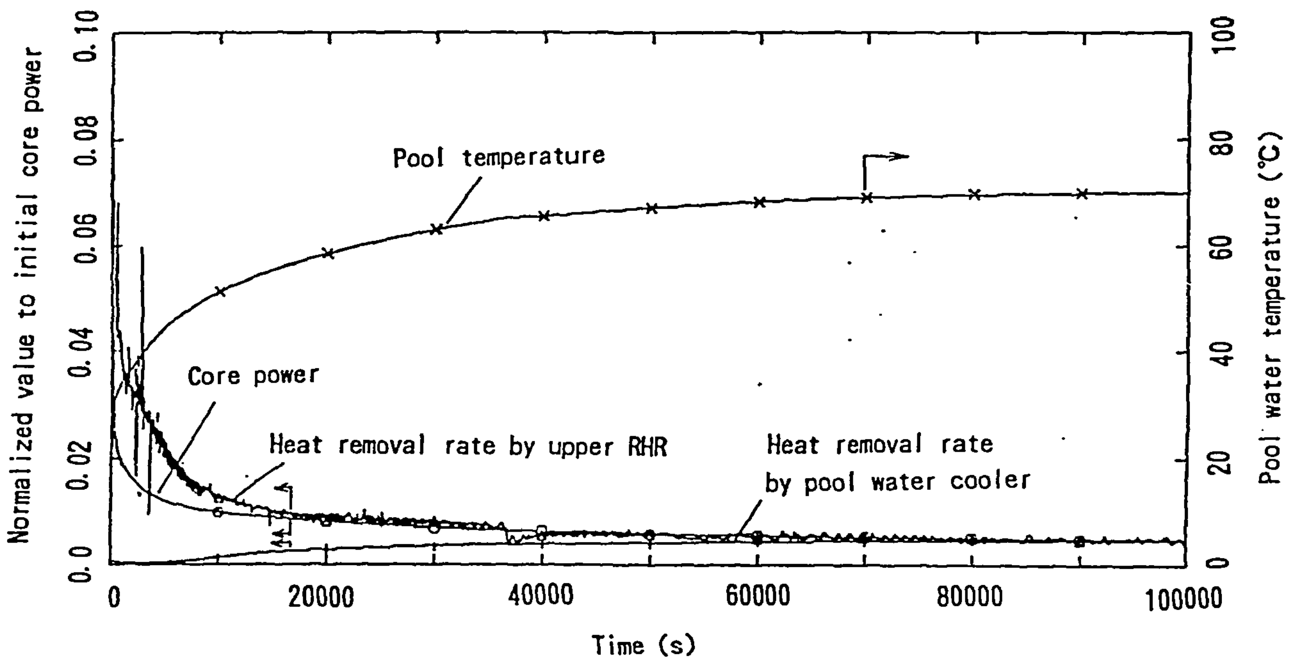
(5) Example of Analysis (2)

Heat Removal from Core to Atmosphere under Loss of Heat Sink Event

-> Natural Circulation of Upper RHR

-> Multi-Dimensional Flow in Pool

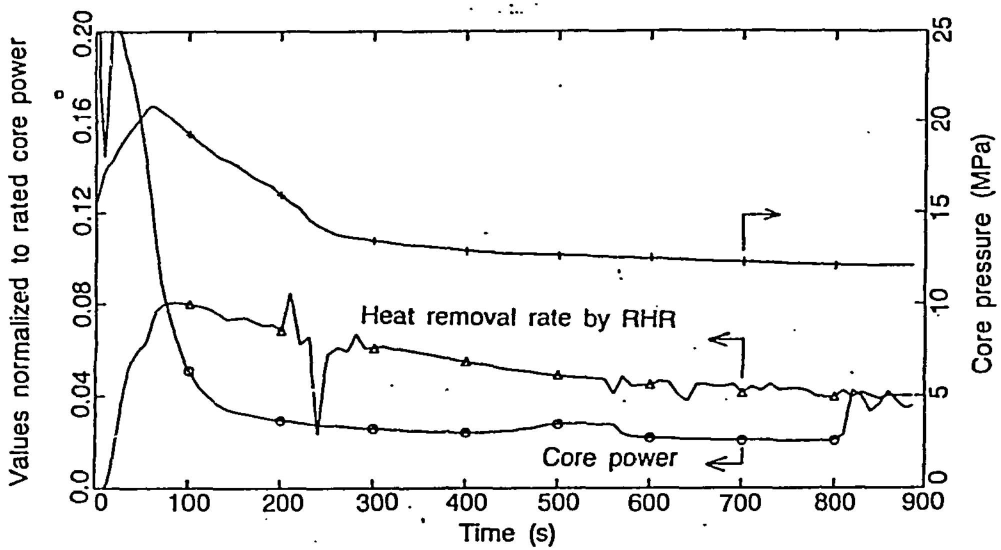
-> Combination of 1D and 3D Components



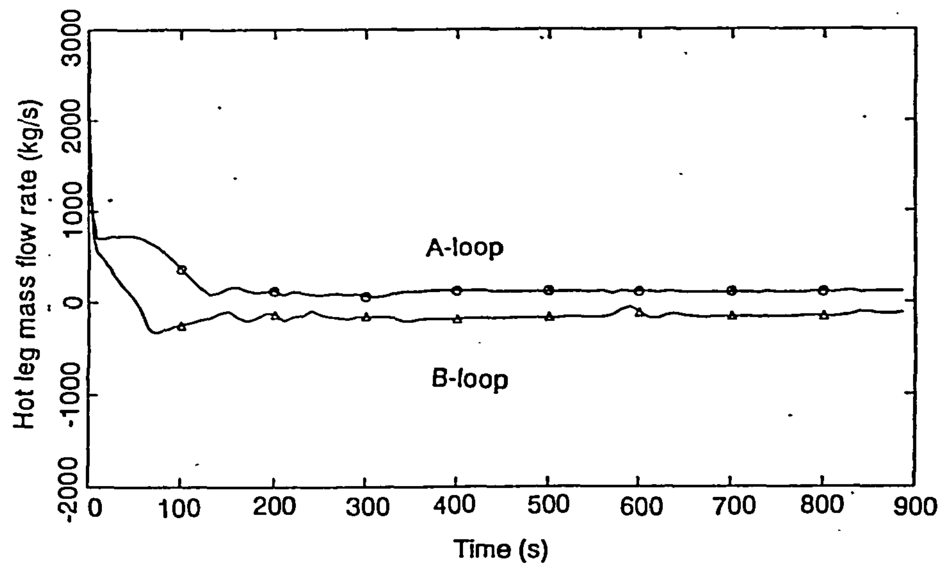
Core power, heat removal rate by RHR and pool cooler and pool temperature

(4) Example of Analysis (1)

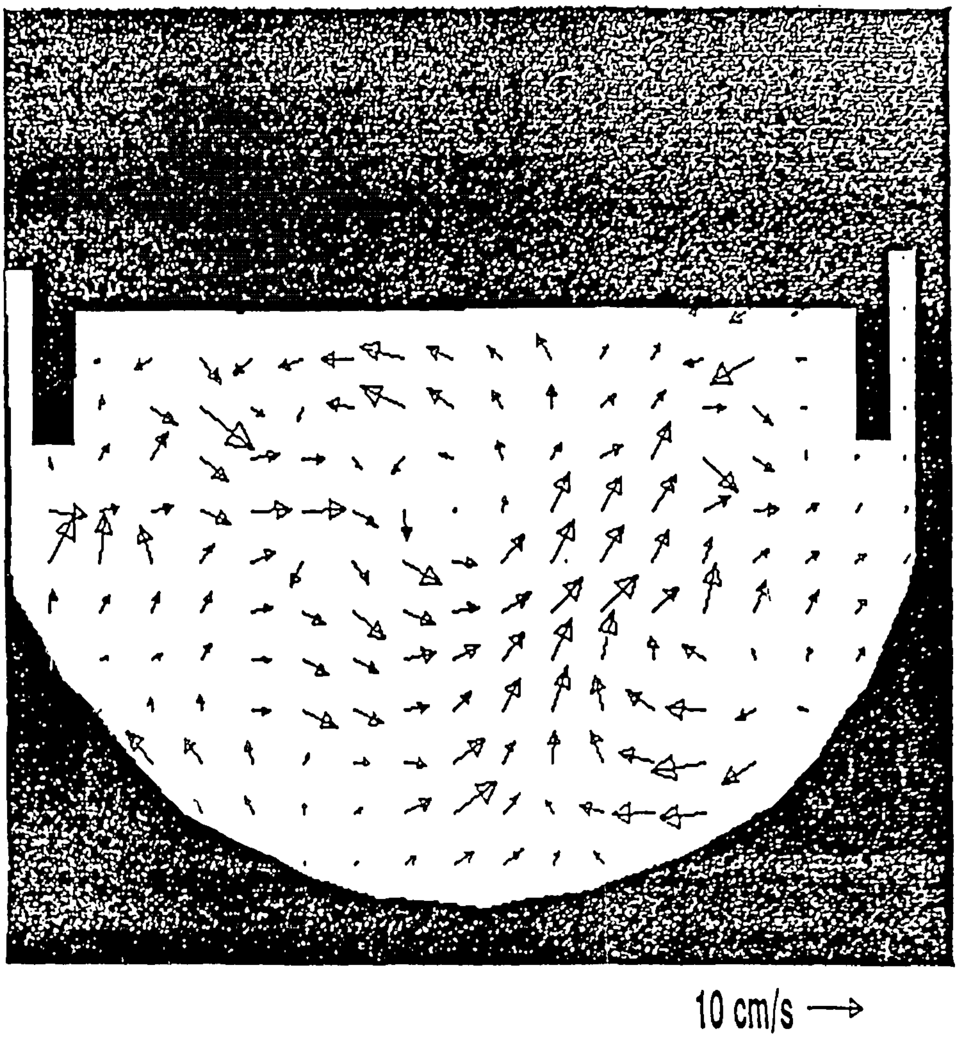
Heat Removal by Upper RHR under Loss of Heat Sink Event
-> Natural Circulation of Upper RHR
-> Multi-Dimensional Flow in Upper Plenum



Core power, heat removal rate and pressure

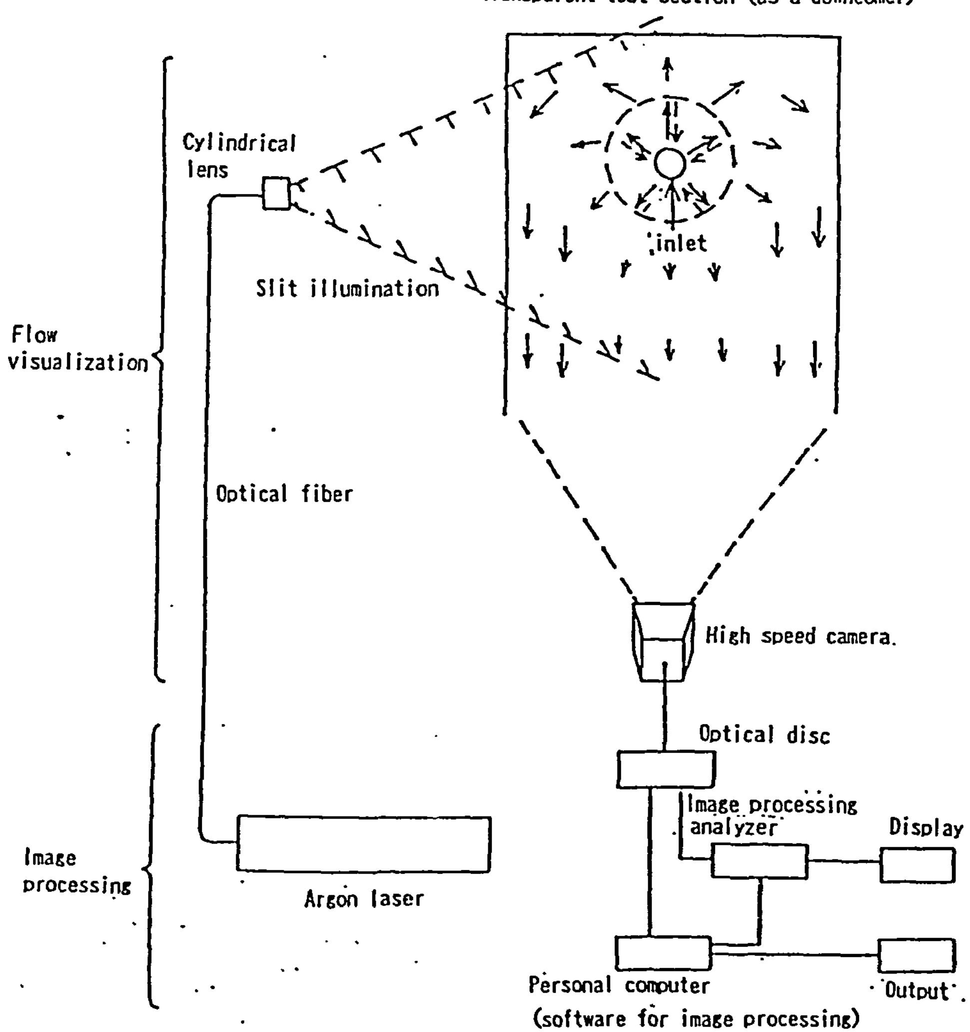


Hot leg mass flow rate

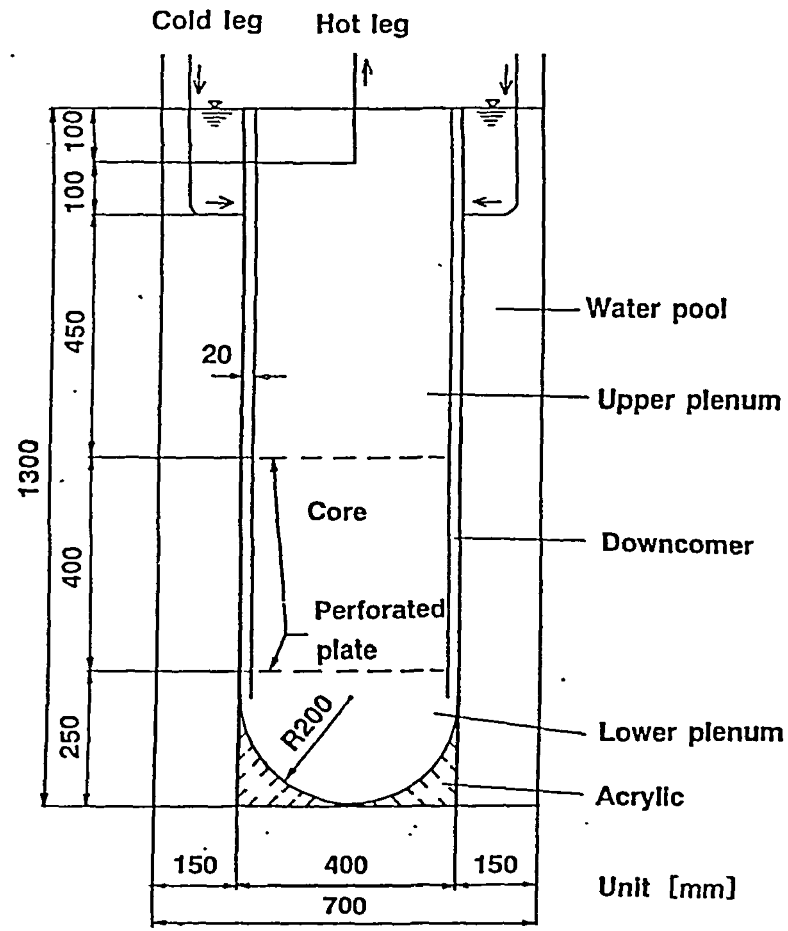


Flow Pattern in Lower Plenum

(6) Experiment for understanding Multi-Dimensional Flow Field in RPV (Flow Visualization by Particle Tracking Velocimetry)



Experimental apparatus for flow visualization



Test section for lower plenum

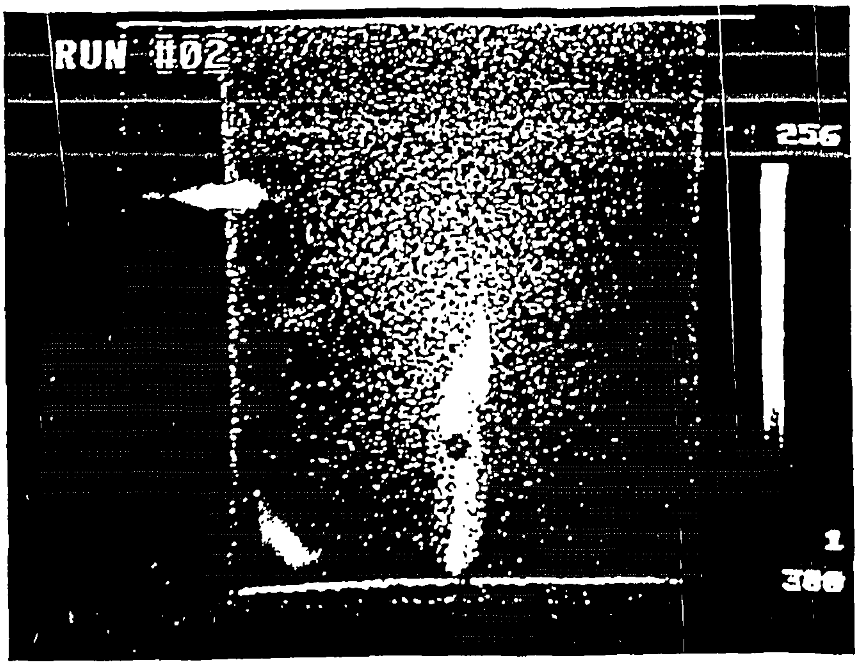
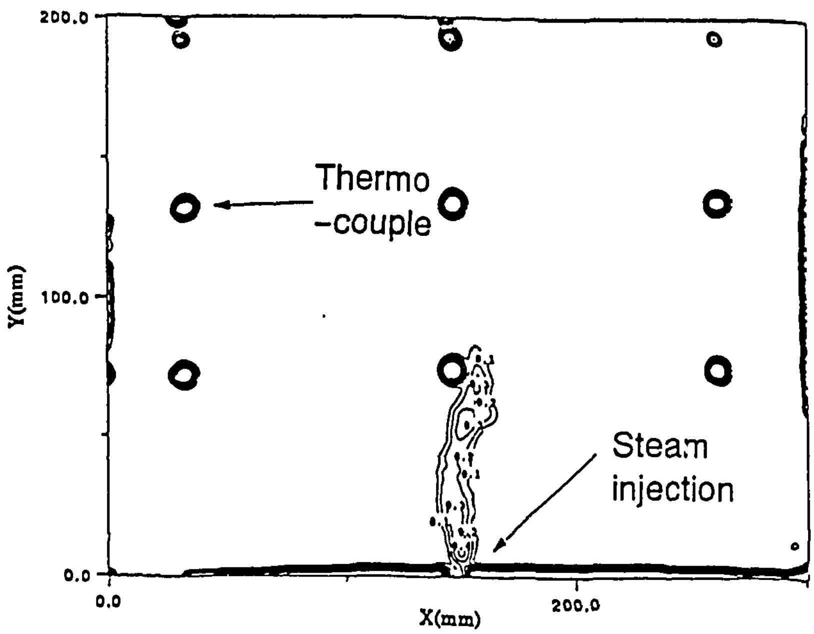
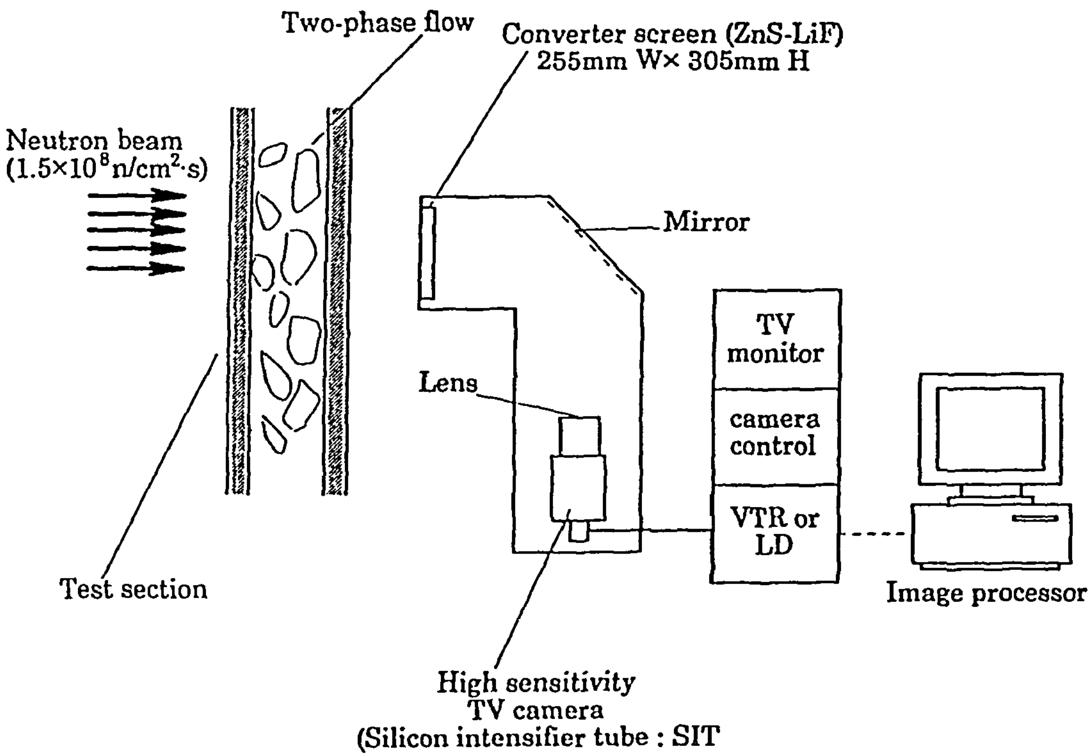


Image data of VTR

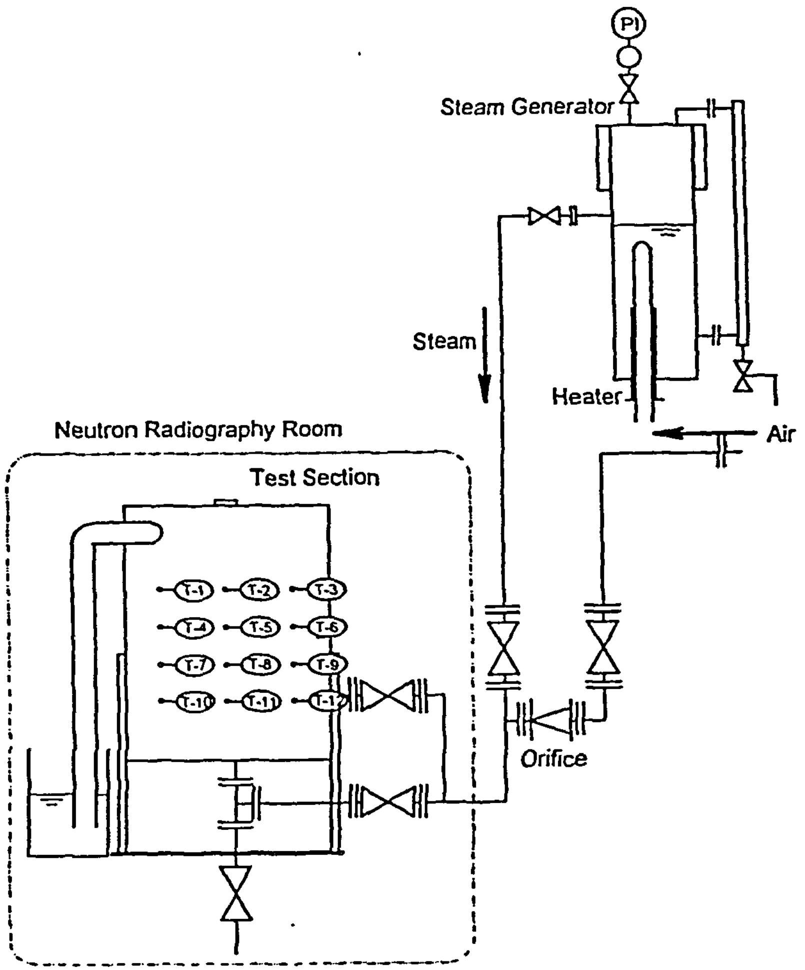


Void fraction distribution

(7) Experiment for Steam Condensation in Water Pool
(Flow Visualization by Neutron Radiography Technique)



Experimental apparatus for NRG experiment



Test section for steam condensation