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Safety Researches

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University of Fukui

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Accident (1/2)

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- Design Basis Accident: **DBA**
- Assumption of simultaneous double ended break

- Installation of Engineered Safety Features
  Emergency Core Cooling System: **ECCS**
  Accumulated Pressurized Coolant Injection System: APCI
  Low Pressure Coolant Injection System: LPCI
  High Pressure Coolant Injection System: HPCI

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Accident (2/2)

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- Computer codes are used to evaluate temperature behavior of fuel bundle.
- Computer codes should be validated.
- Blow-down and ECC injection tests have been conducted using mock-ups.
- **RELAP5/mod3** and **TRAC** code are developed and validated.

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ECCS

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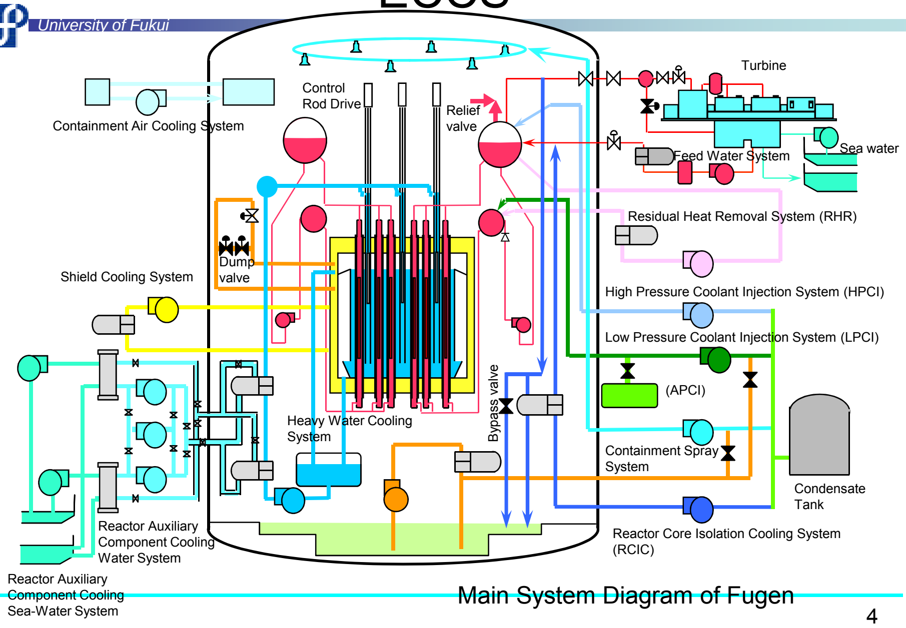

Main System Diagram of Fugen

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Blow-down experiment

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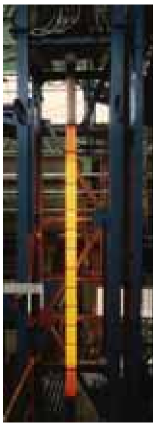

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6MW ATR Safety Experimental Facility

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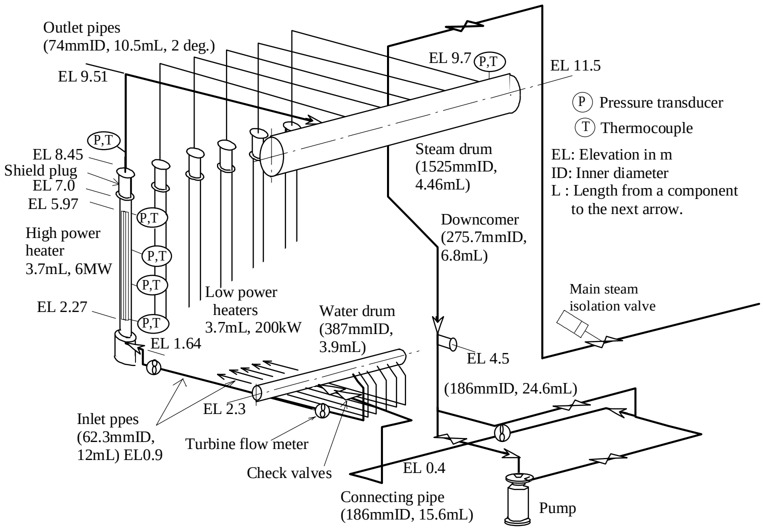

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Water level behavior after a main steam pipe break

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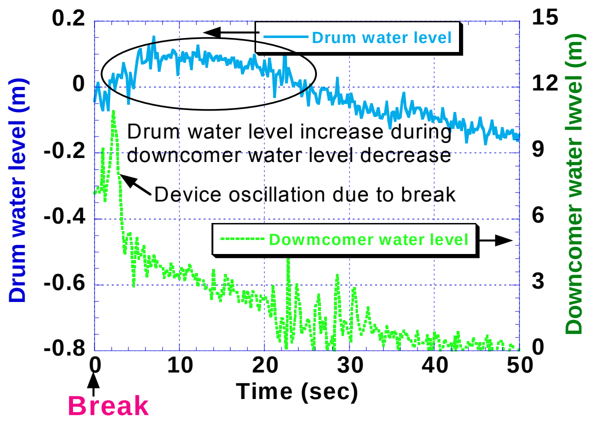

100 mm break at main steam pipe

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Simulated fuel bundle

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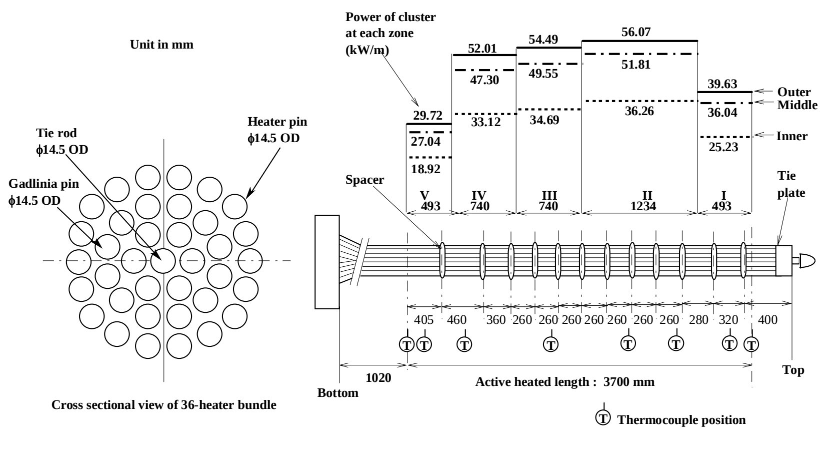

Fig. 7 Power distribution of 36-rod high power heater

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Thermocouple positions

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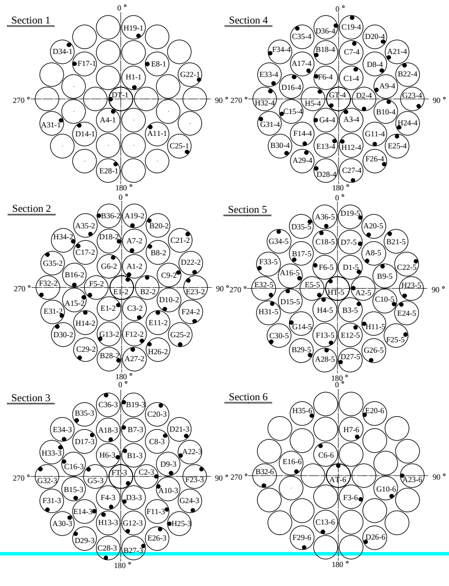

Fig. 8 Thermocouple positions on high power heater rods

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Cladding temperature measured in a same cross section of heater bundle

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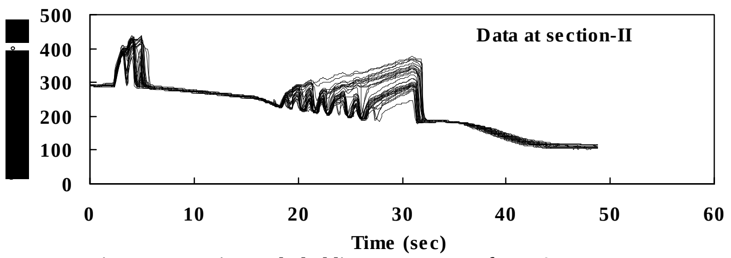

Fig. 14 Experimental cladding temperature for 150 mm downcomer break

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Calculation model of pipe break experiment

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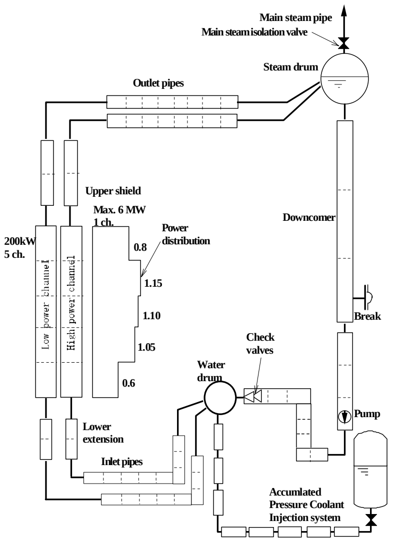

Fig. 9 Nodalization scheme for ATR Safety Experiment Loop

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Comparison between experimental result and simulation

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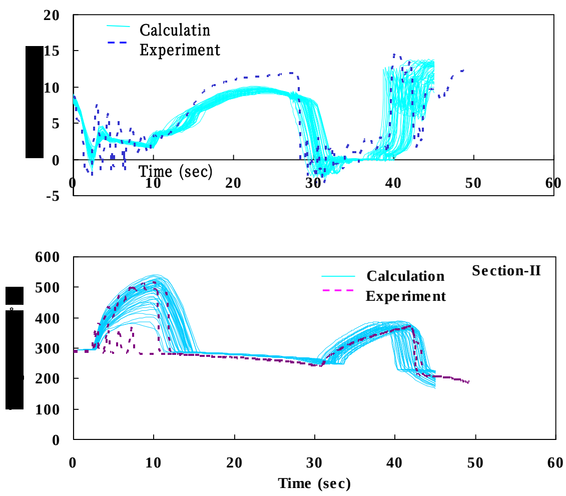

Fig.16 Behavior of cladding temperature after 100 mm downcomer break

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Severe accident

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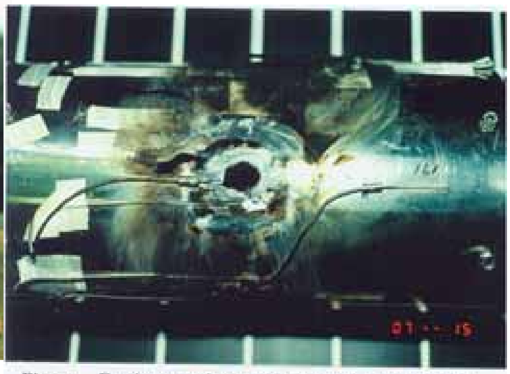

Photo. Erosion experiment of Zr-2.5%Nb pressure tube by molten metal

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Heat transfer of melted fuel to material

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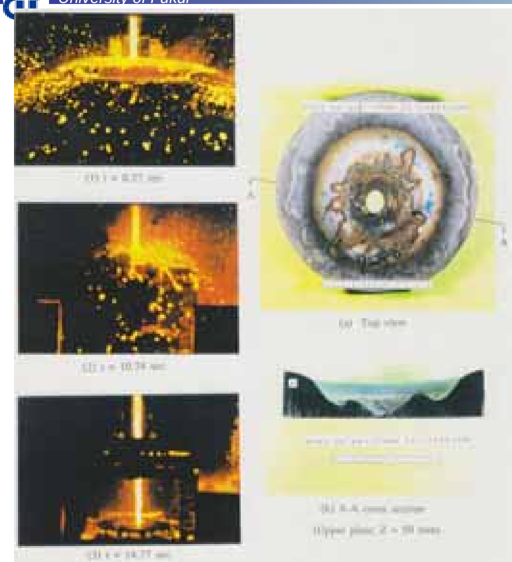

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Heat transfer between melted jet and materials

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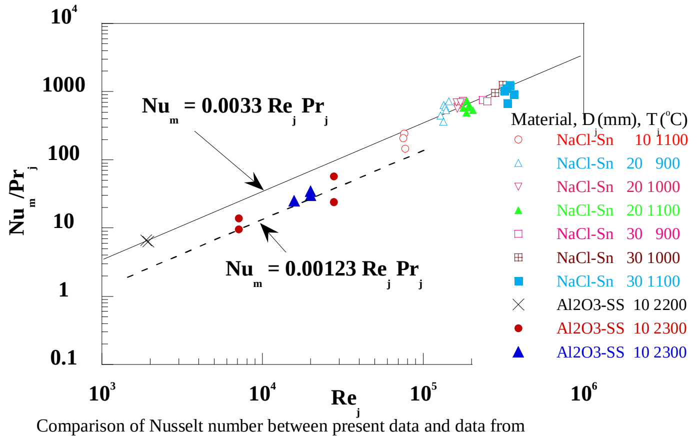

Comparison of Nusselt number between present data and data from Saito et al.1) and Mochizuki2).

1)Saito, et al., Nuclear Engineering and Design, 132 (1991)

2)Mochizuki, Accident Management and Simulation Symposium, Jackson Hole, (1997).

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Fuel melt experiment using BTF in Canada

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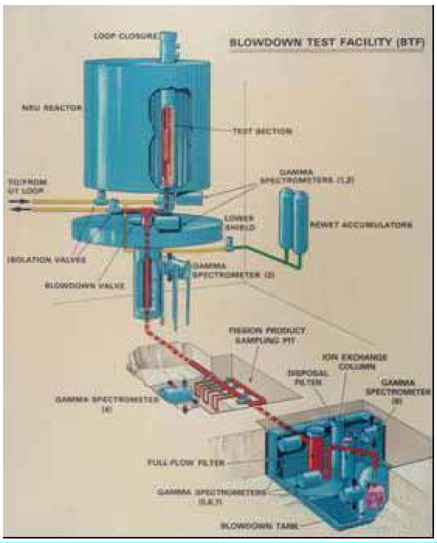

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Fuel melt experiment using CABRI

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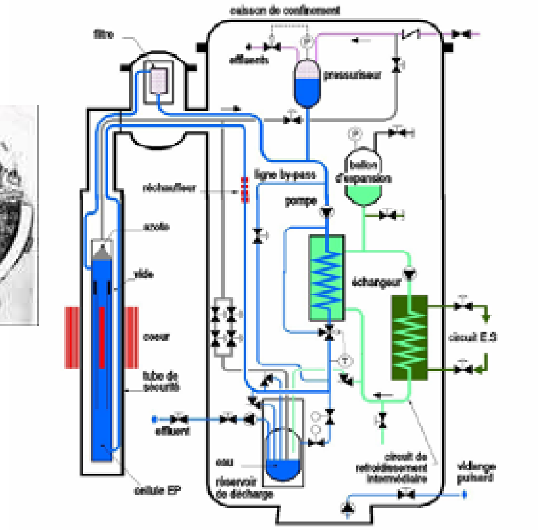

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Source term analysis codes

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General codes	NRC codes	ORIGEN-2, MARCH-2, MERGE, CORSOR, TRAP-MELT, CORCON, VANESA, NAUA-4, SPARC, ICEDF
	IDCOR codes	MAAP, FPRAT, RETAIN
	NRC code (2nd Gen.)	MELCOR
Precise analysis codes	Core melt	SCDAP, ELOCA, MELPROG, SIMMER
	Debris-concrete reaction	CORCON
	Hydrogen burning	HECTOR, CSQ Sandia, HMS BURN
	FP discharge	FASTGRASS, VICTORIA
	FP behavior in heat transport system	TRAP-MELT
	FP discharge during debris-concrete reaction	VANESA
	FP behavior in containment	CONTAIN, NAUA, QUICK, MAROS, CORRAL-II

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CONATIN code

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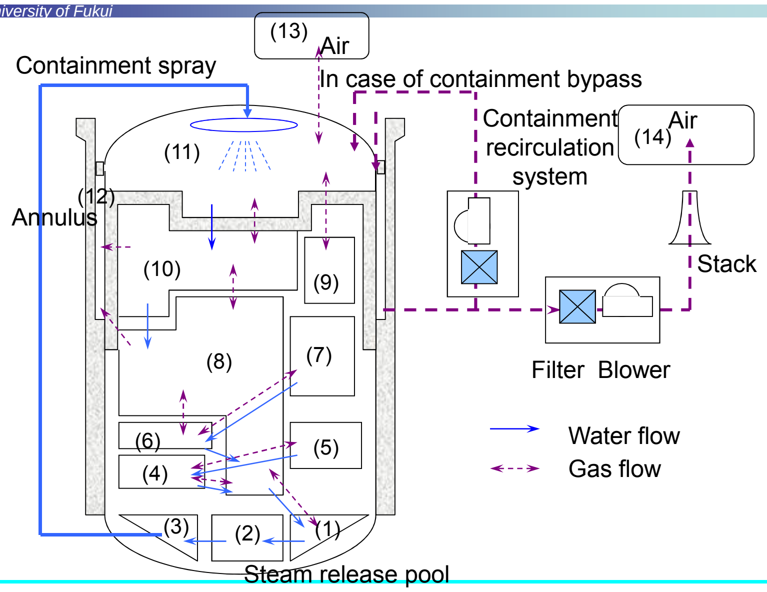

Fluid- structure interaction analysis during hydrogen detonation

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Analysis of Chernobyl Accident
- Investigation of Root Cause -

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Schematic of Chernobyl NPP

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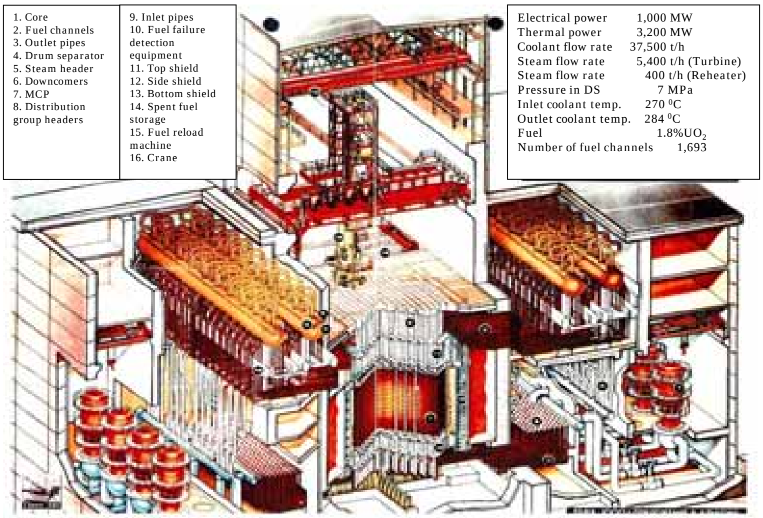

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Elevation Plan

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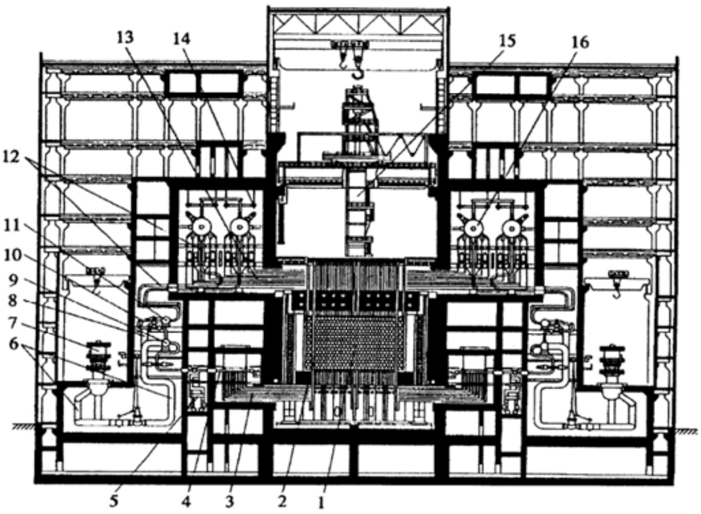

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Above the Core of Ignarina NPP

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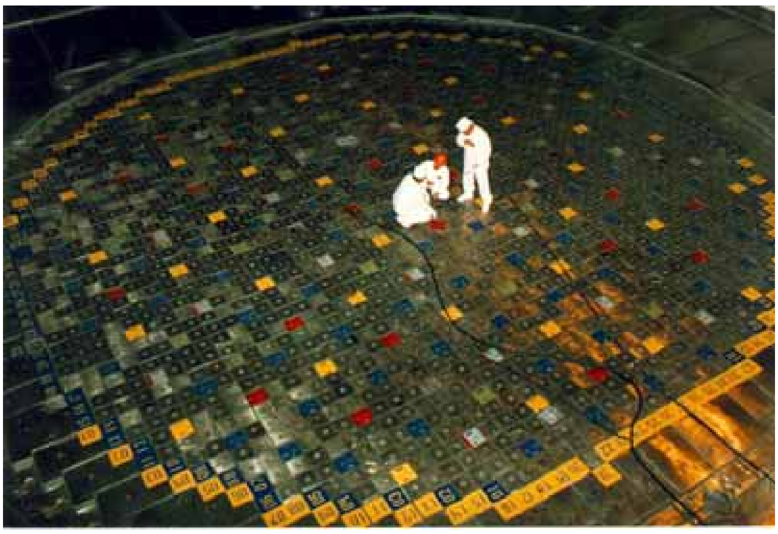

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Core and Re-fueling Machine

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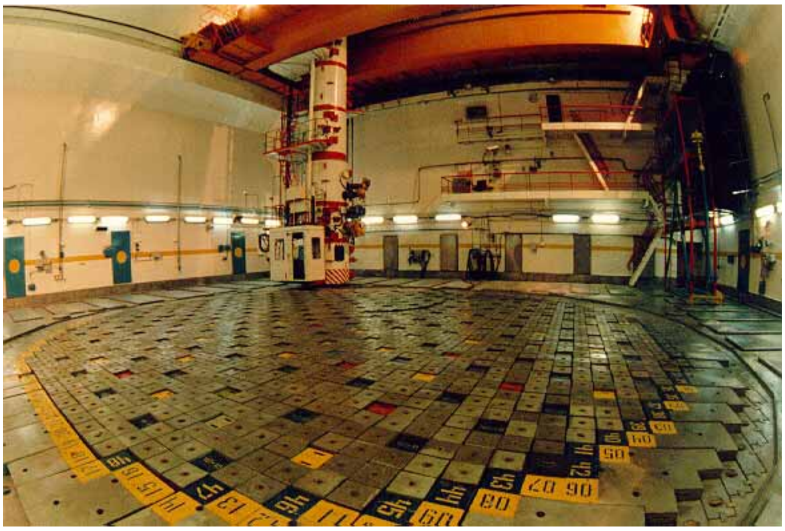

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Control Room

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Configuration of inlet valve

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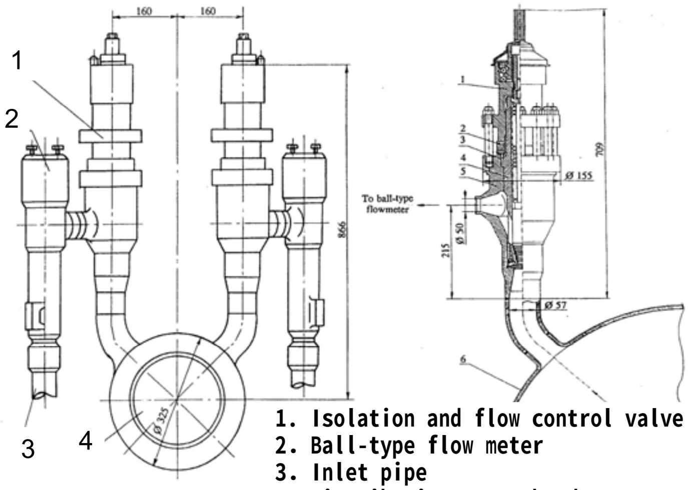

1. Isolation and flow control valve
2. Ball-type flow meter
3. Inlet pipe
4. Distribution group header

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Drum Separator

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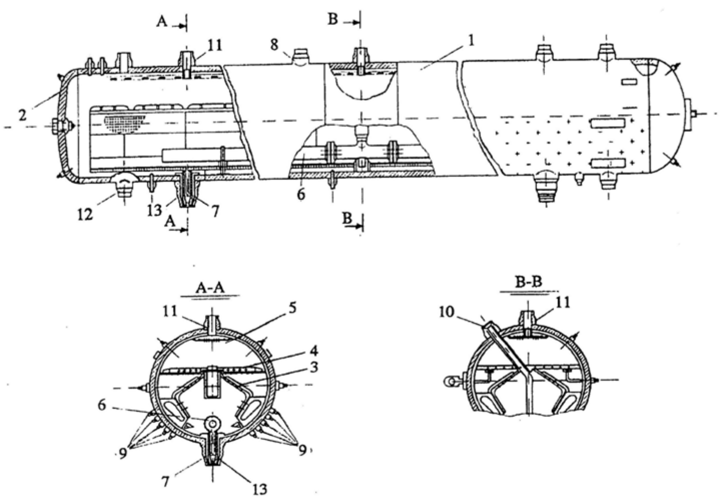

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Configuration of Fuel Channnel

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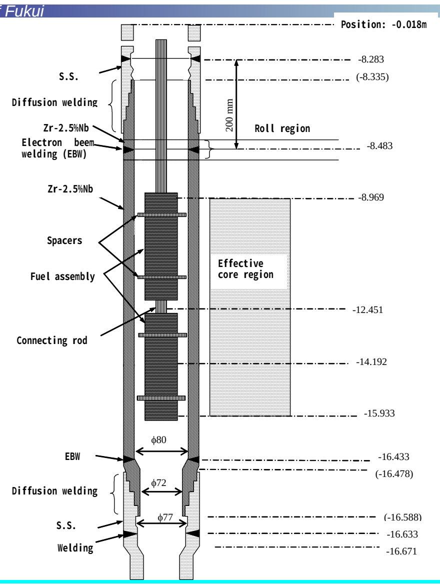

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Heat Removal by Moderation

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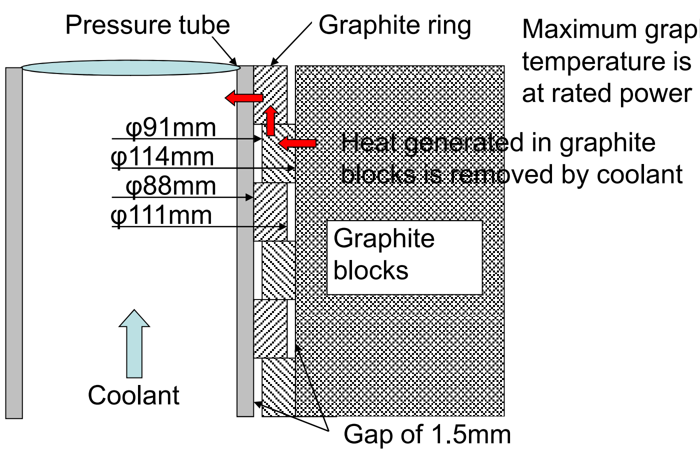

Maximum graphite temperature is 720℃ at rated power

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RBMK & VVER

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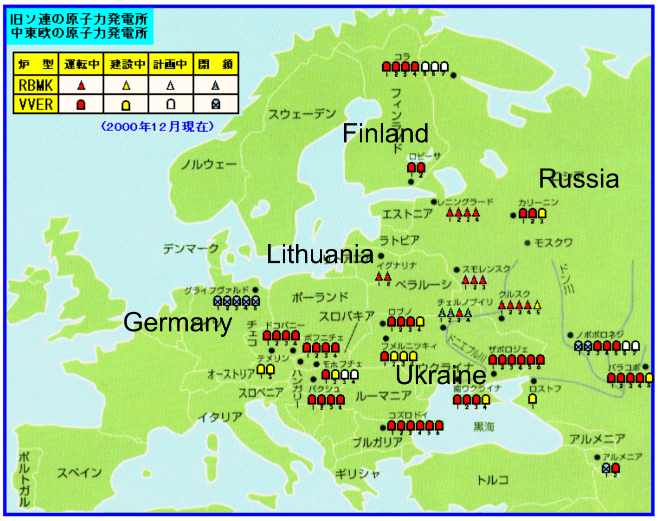

図2 ソ連型炉の所在地

[出典] (1) 資源エネルギー庁原子力広報推進室(編): 見直される旧ソ連の原子力発電、ロシア東欧貿易会、p.1
(2) 国際原子力安全計画(http://insp.pnl.gov:2080/)

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Objective of the Experiment

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- Power generation after the reactor scram for several tens of seconds in order to supply power to main components.
- There is enough amount of vapor in drum separators to generate electricity.
- But they closed the isolation valve.
- They tried to generate power by the inertia of the turbine system.

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Report in Dec. 1986

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Trend of the Reactor Power

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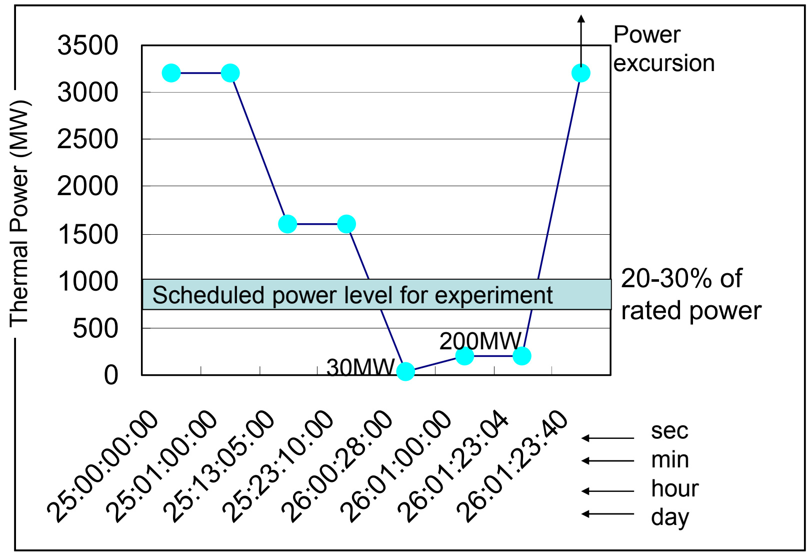

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Time Chart Presented by USSR

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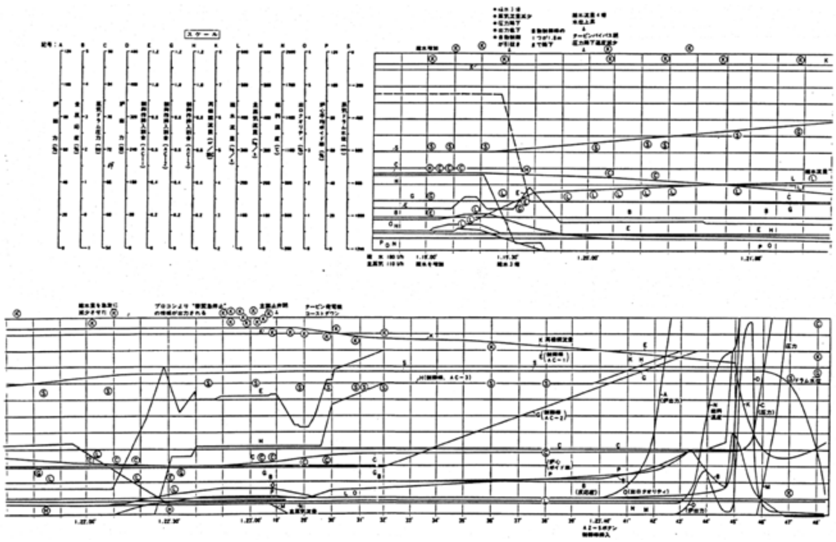

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Result in the Past Analysis (1/2)

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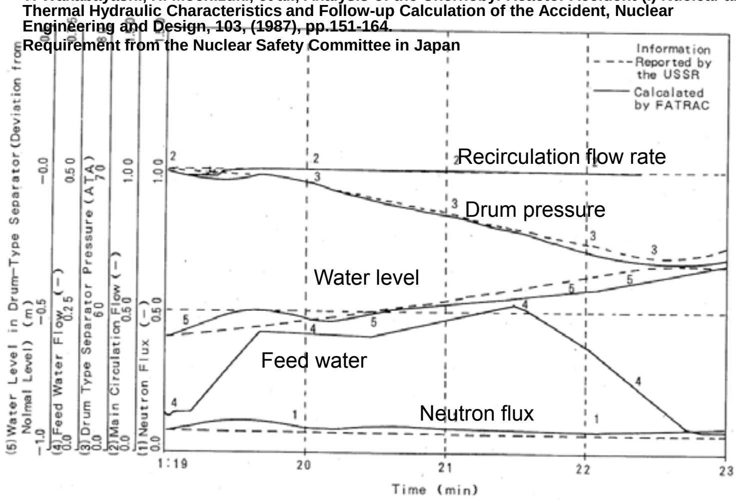
- T. Wakabayashi, H. Mochizuki, et al., Analysis of the Chernobyl Reactor Accident (I) Nuclear and Thermal Hydraulic Characteristics and Follow-up Calculation of the Accident, J. Atomic Energy Society of Japan, 28, 12 (1986), pp.1153-1164.
- T. Wakabayashi, H. Mochizuki, et al., Analysis of the Chernobyl Reactor Accident (I) Nuclear and Thermal Hydraulic Characteristics and Follow-up Calculation of the Accident, Nuclear Engineering and Design, 103, (1987), pp.151-164.
- Requirement from the Nuclear Safety Committee in Japan



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Result in the Past Analysis (2/2)

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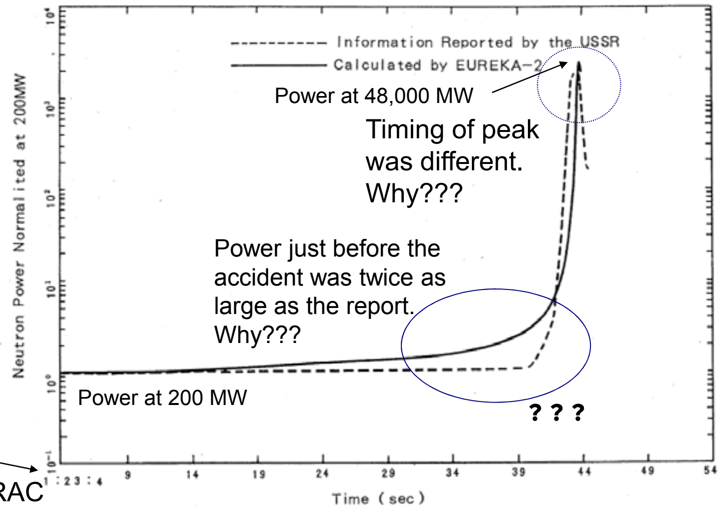

Result of FATRAC code is transferred, and initial steady calculation was conducted.

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Possible Trigger of the Accident

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- Positive scram due to flaw of scram rods
- Pump cavitation
- Pump coast-down

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Calculation Model by NETFLOW++ Code

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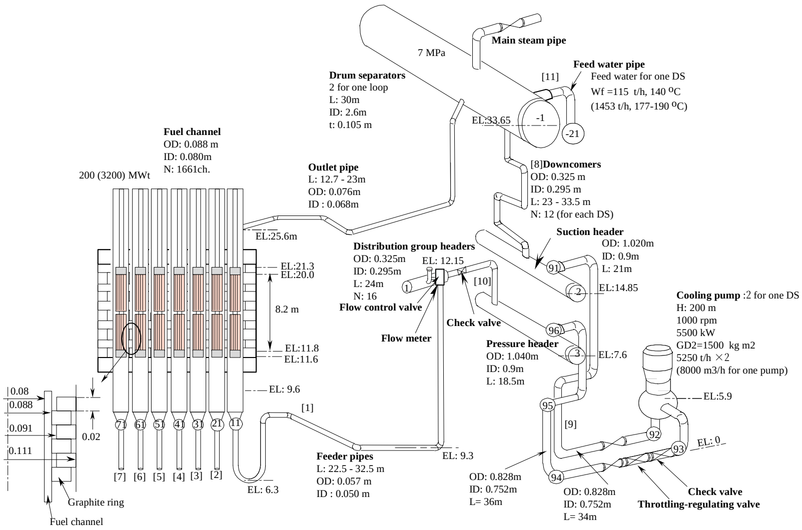

Trigger of the Accident

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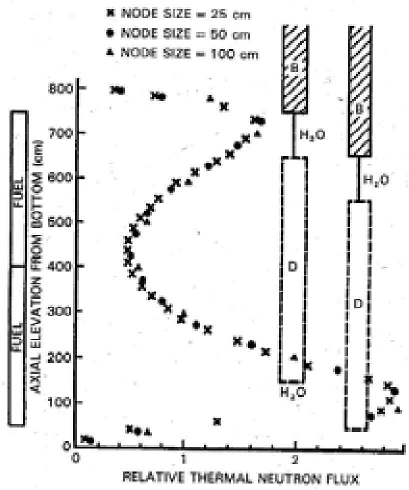


Fig. 1. Effect of node size on axial flux distribution. In this figure, B = B_4C absorber and D = graphite displacer.

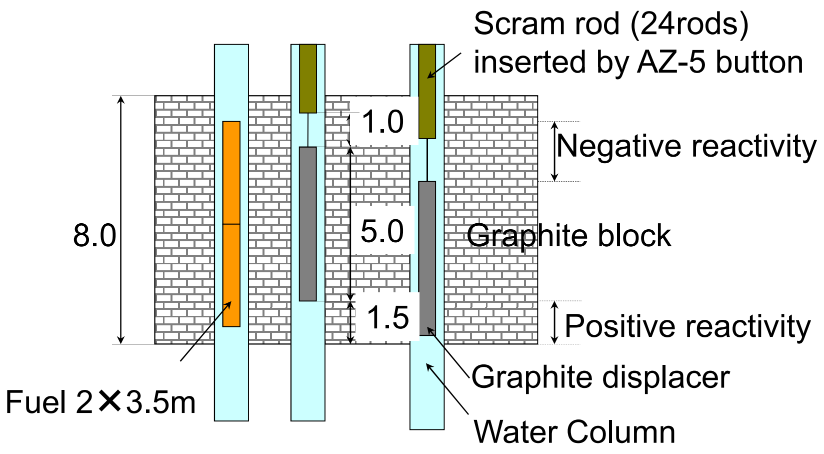
- Positive scram

P.S.W. Chan and A.R. Daster
Nuclear Science and Engineering, 103, 289-293 (1989).

Andriushchenko, N.N. et al.,
Simulation of reactivity and neutron fields change, Int. Conf. of Nuclear Accident and the Future of Energy, Paris, France, (1991).

Trigger of the Accident (cont.)

University of Fukui



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Simulation from 1:19:00 to First Peak

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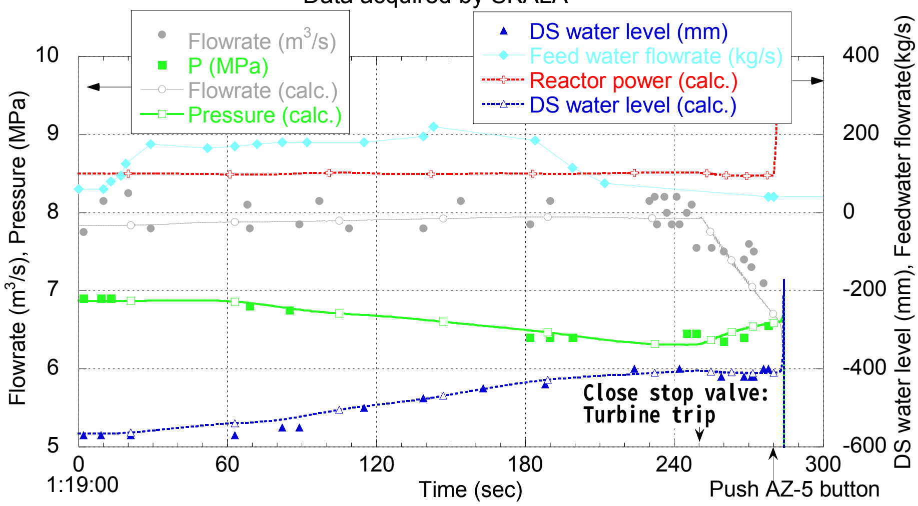
Data acquired by SKALA



Trend of parameters for one loop from 1:19:00 on 26 April 1986

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Behavior of Steam Quality

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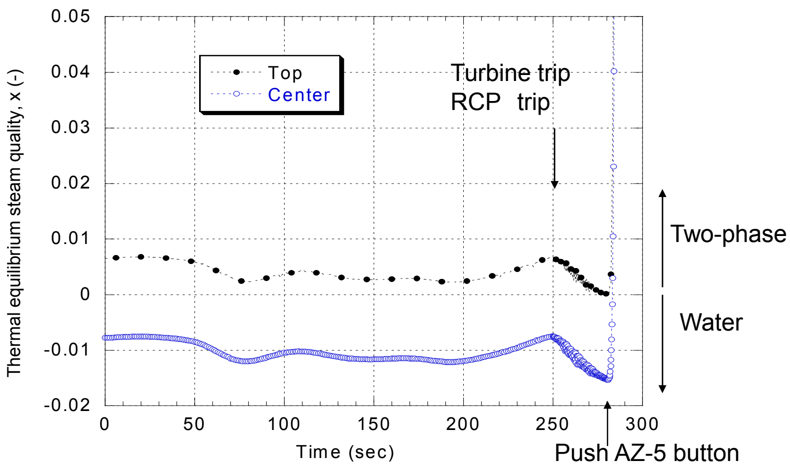

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Void Characteristic

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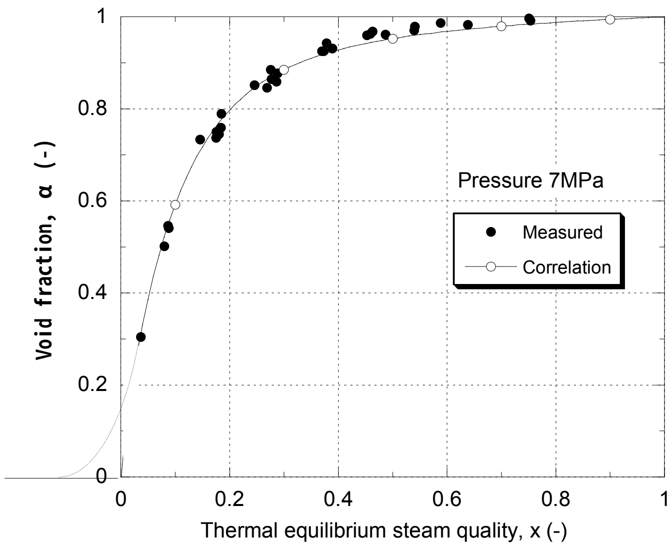

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Nuclear Characteristics

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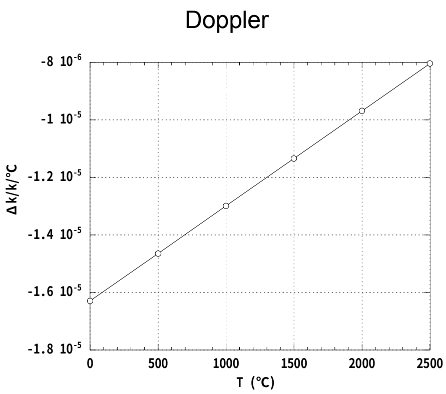

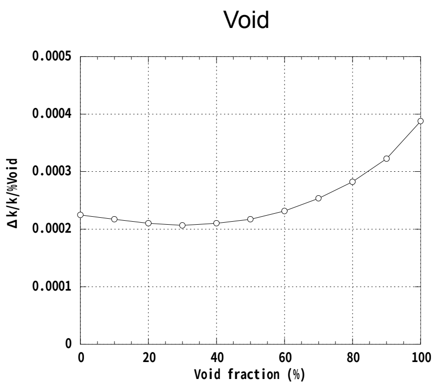

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Peak Power and its Reactivity

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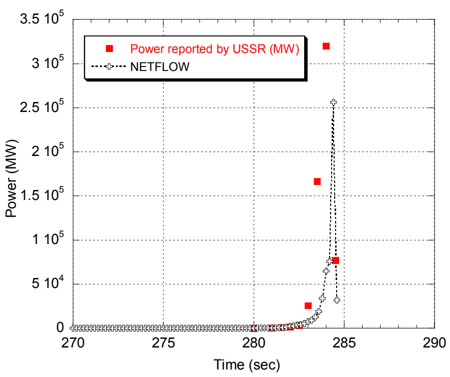

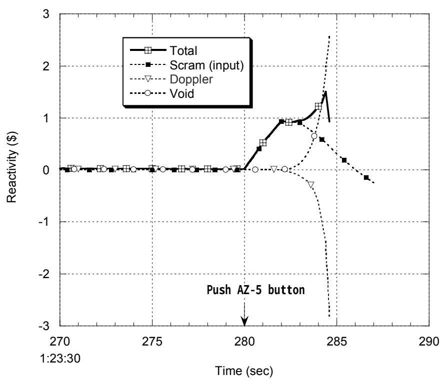

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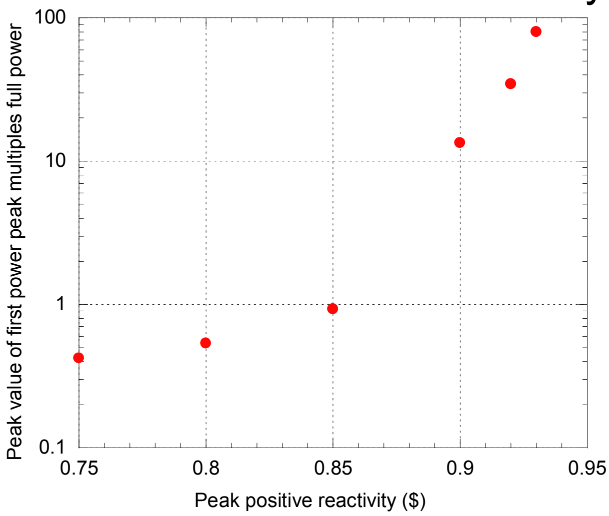
Relationship between Peak Power and Peak Positive Reactivity



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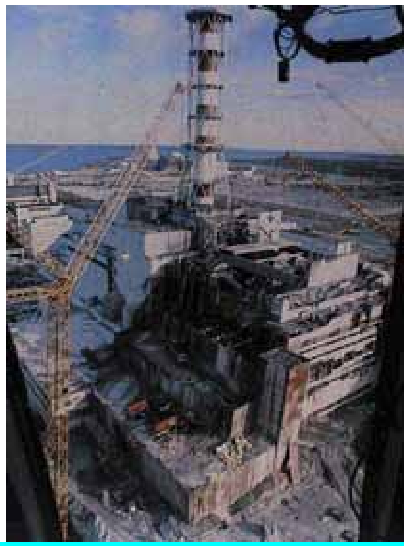
Just after the Accident



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Control Room and Corium beneath the Core

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