Safety of Nuclear Reactors

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Accident (1/2)

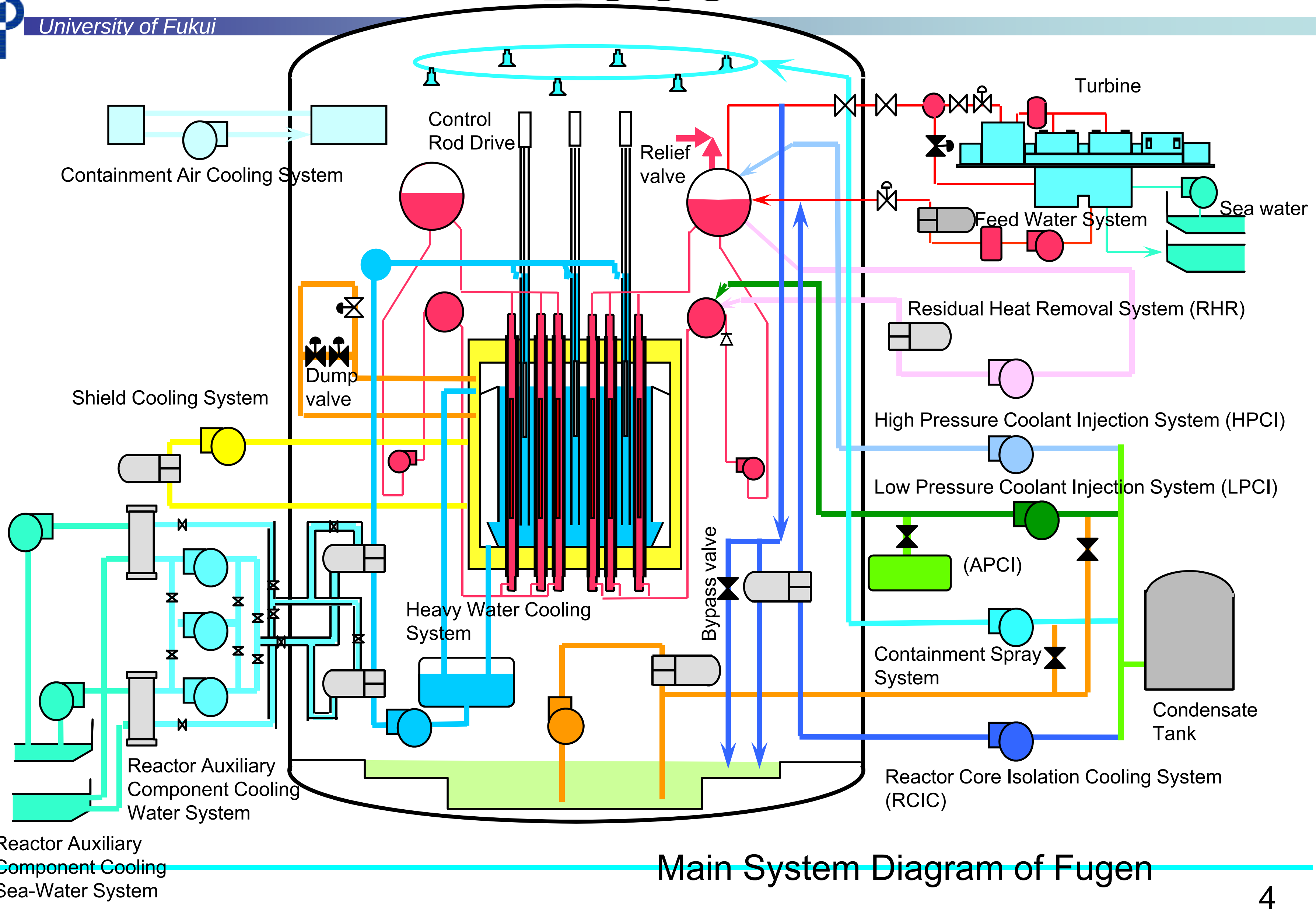
- Design Basis Accident: **DBA**
- Assumption of simultaneous double ended break

- 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

Accident (2/2)

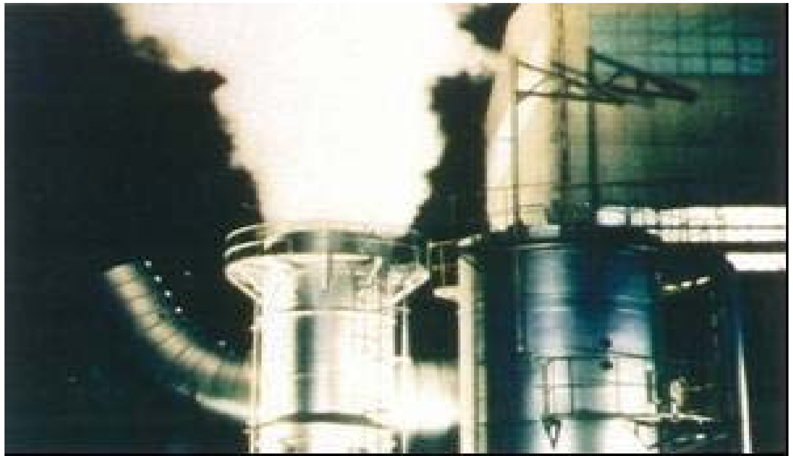
- Computer codes are used to evaluate temperature behavior of fuel bundle.
- Computer codes should be validated.
- Blow-down and ECC injection tests have been conducted using mock-ups.
- **RELAP5/mod3** and **TRAC** code are developed and validated.

ECCS

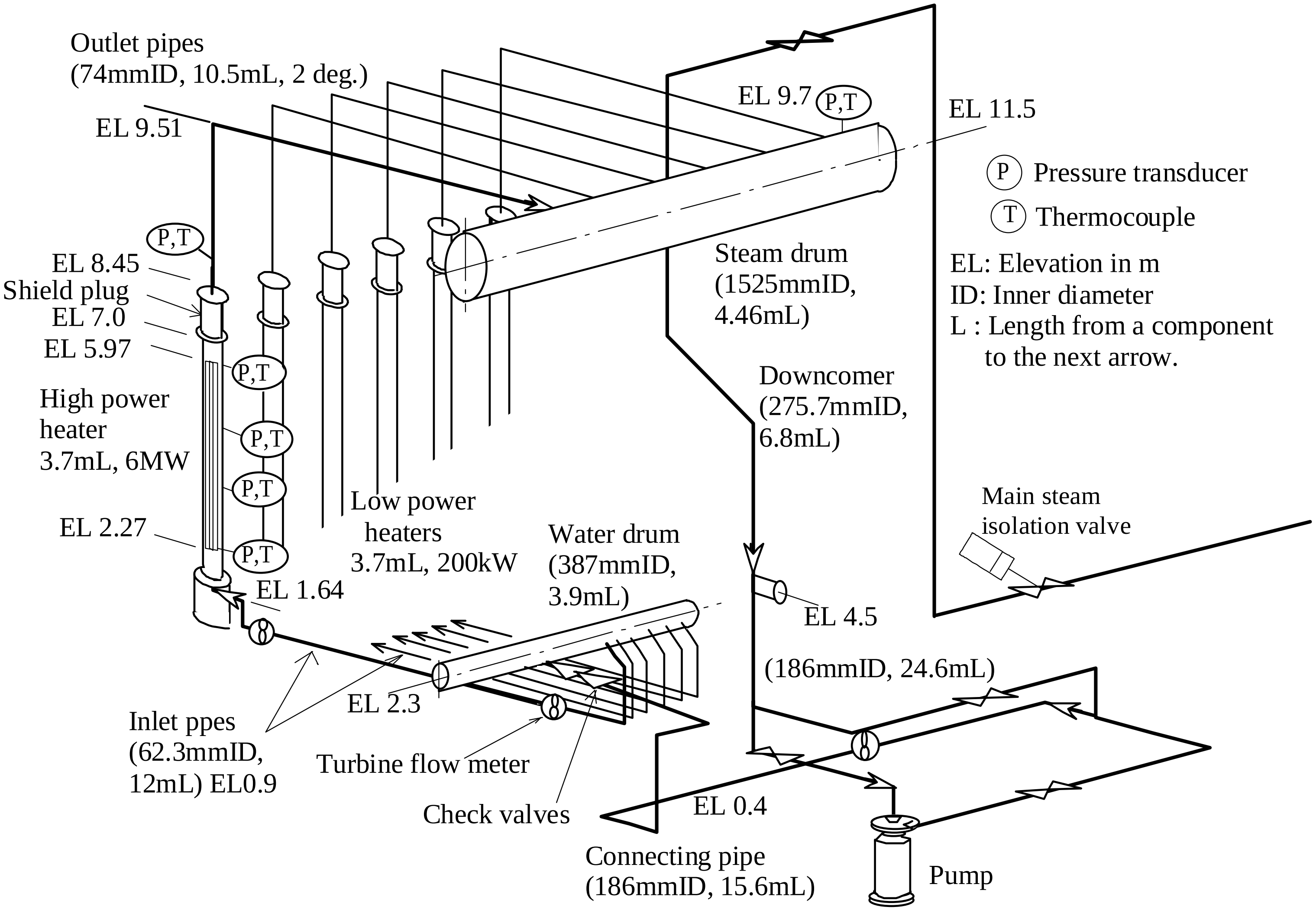


Main System Diagram of Fugen

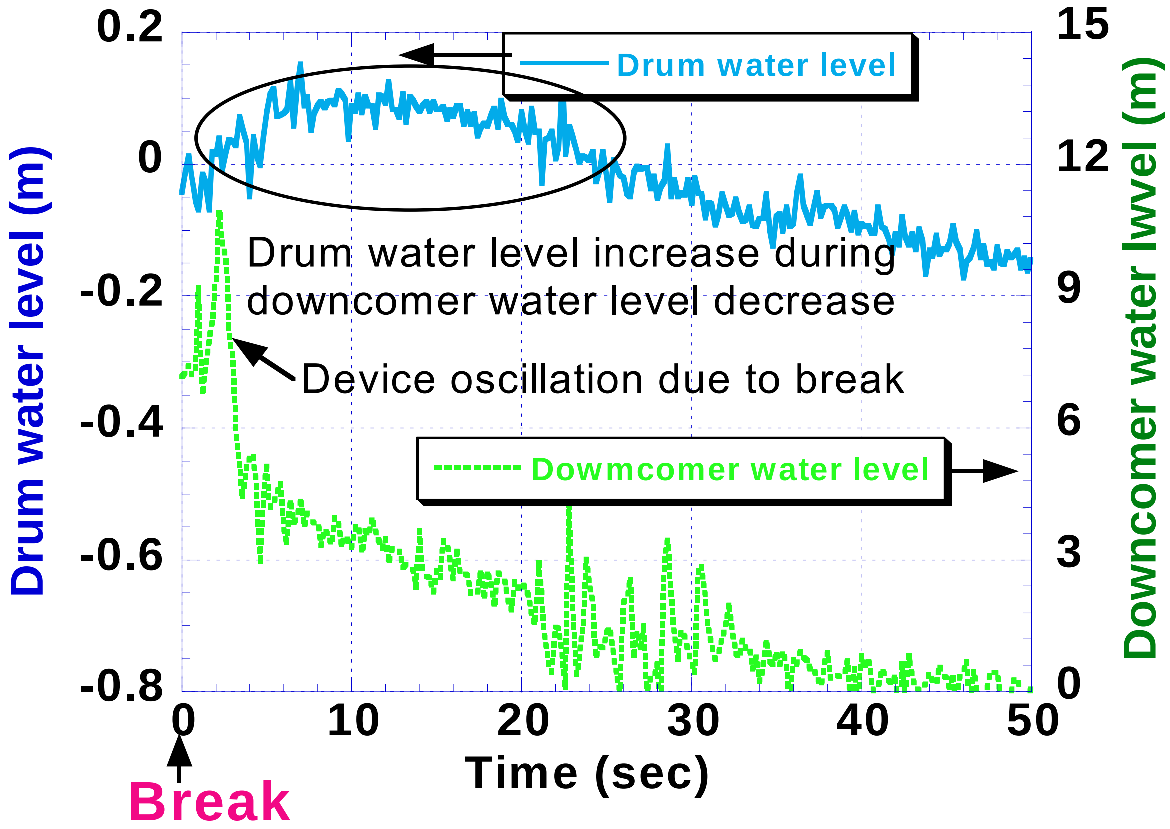
Blow-down experiment



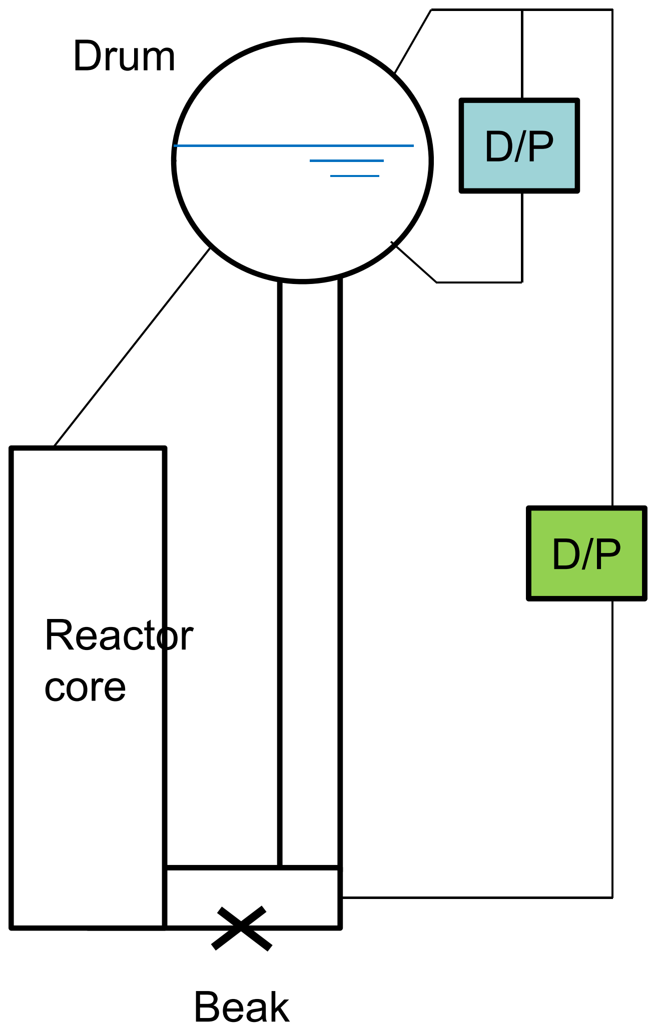
6MW ATR Safety Experimental Facility



Water level behavior after a main steam pipe break



100 mm break at main steam pipe



Simulated fuel bundle

Local peaking is high for the outer rods due to the neutronic characteristics

Location of maximum axial peaking

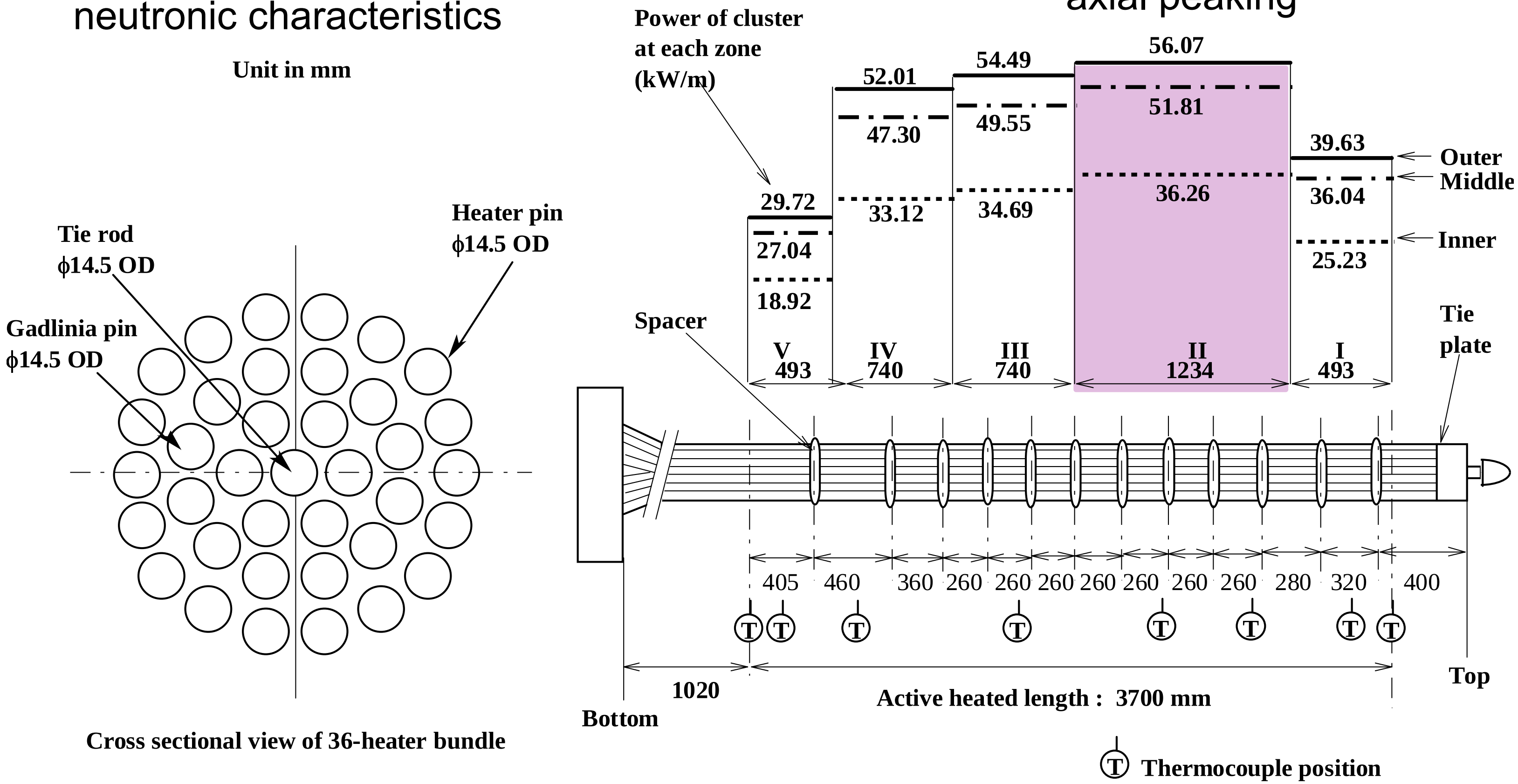


Fig. 7 Power distribution of 36-rod high power heater

Thermocouple positions

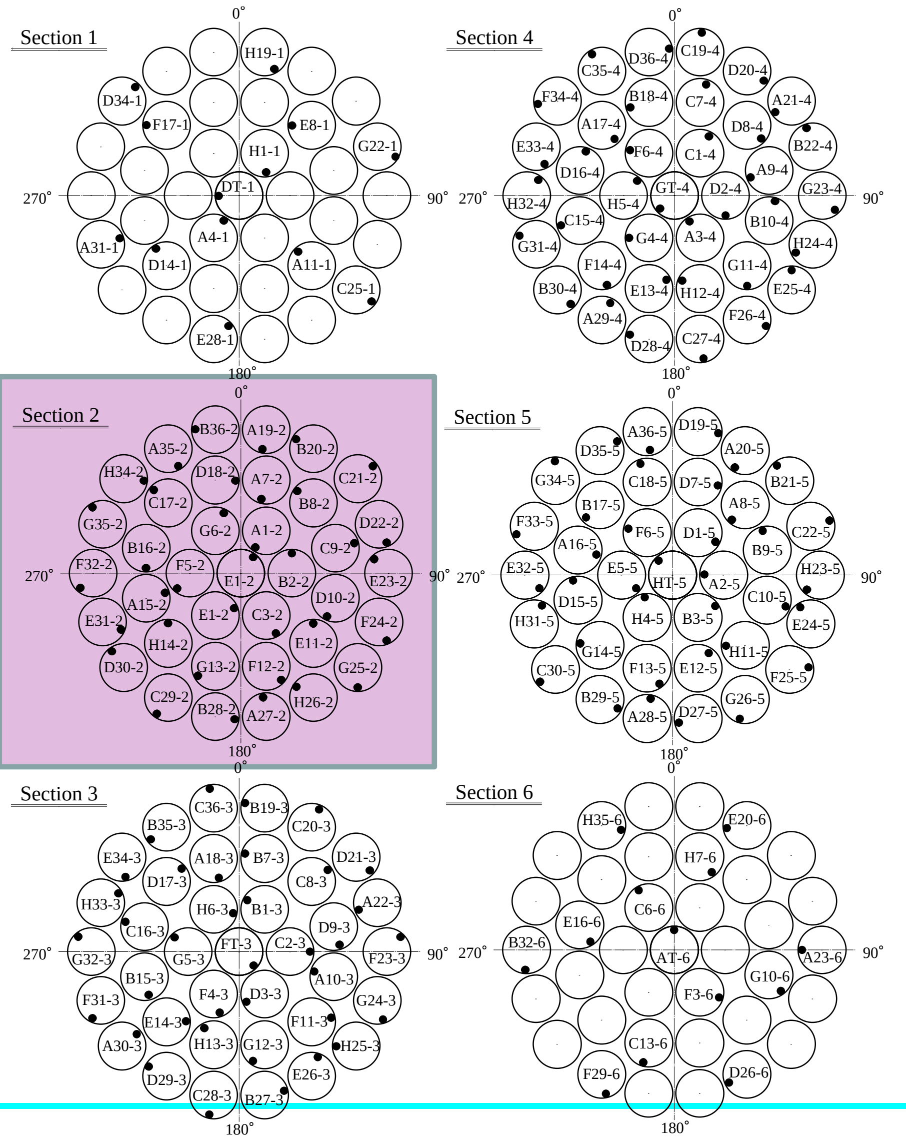


Fig. 8 Thermocouple positions on high power heater rods

Cladding temperature measured in a same cross section of heater bundle

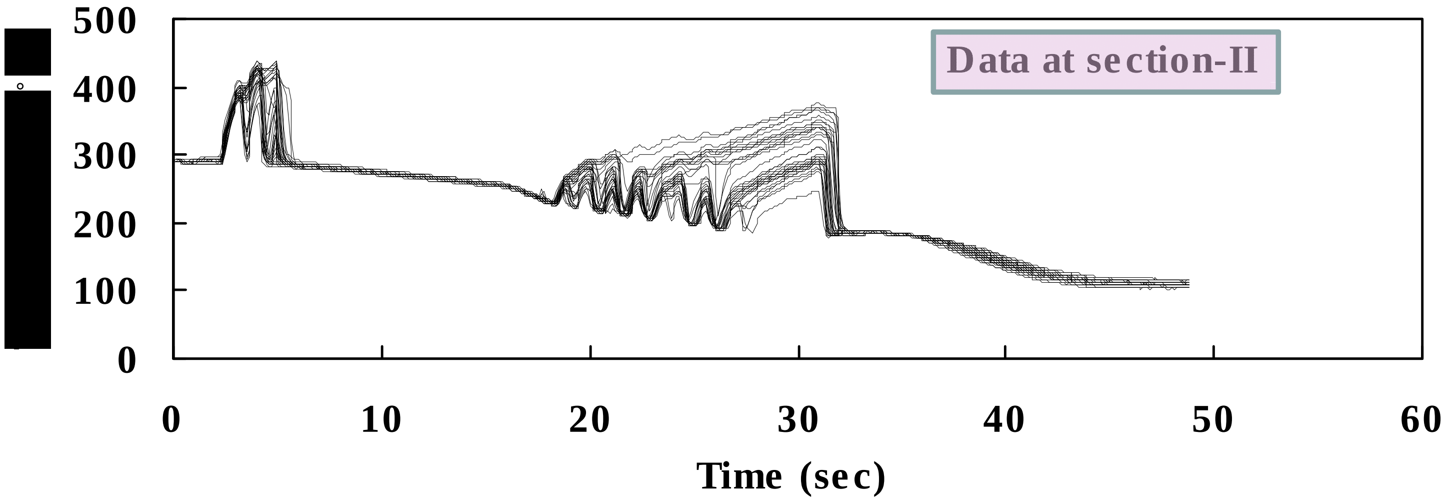


Fig. 14 Experimental cladding temperature for 150 mm downcomer break

Calculation model of pipe break experiment

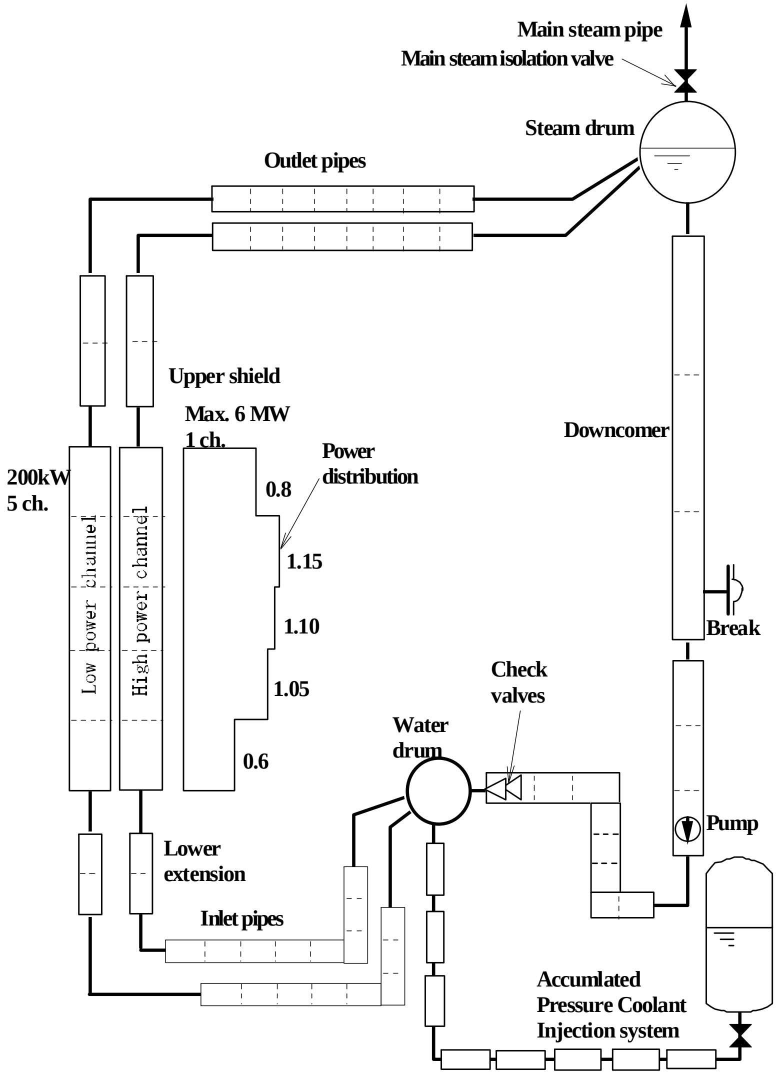
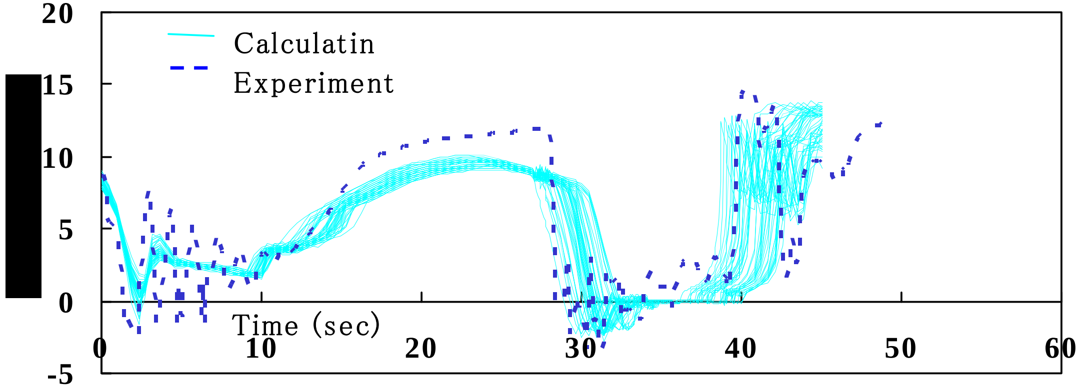


Fig. 9 Nodalization scheme for ATR Safety Experiment Loop

Comparison between experimental result and simulation



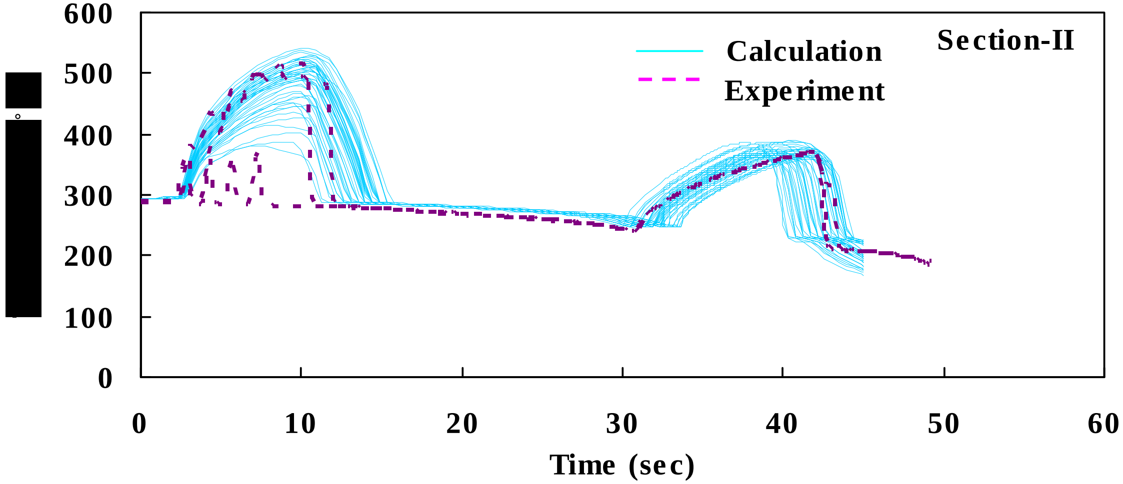
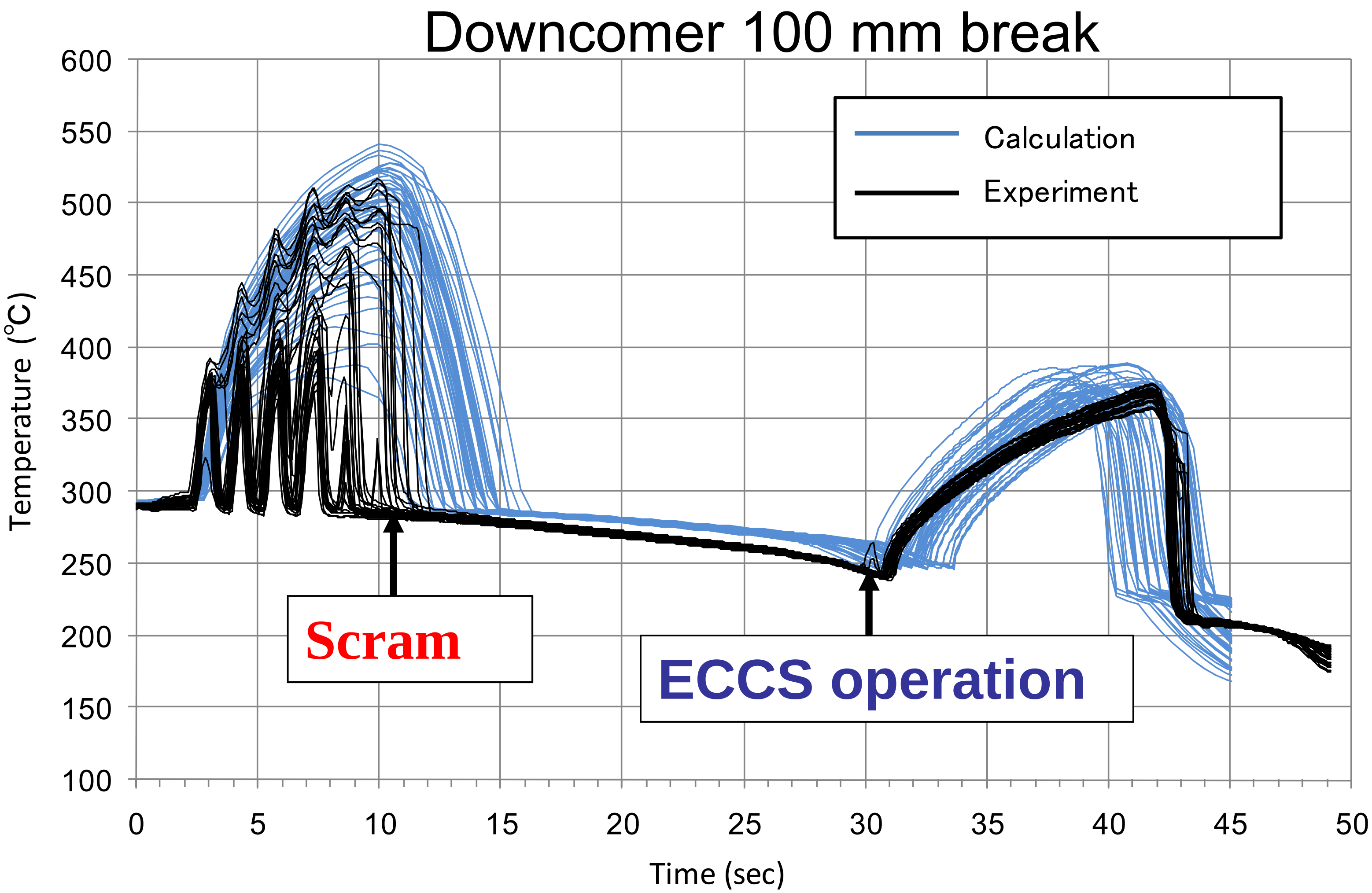
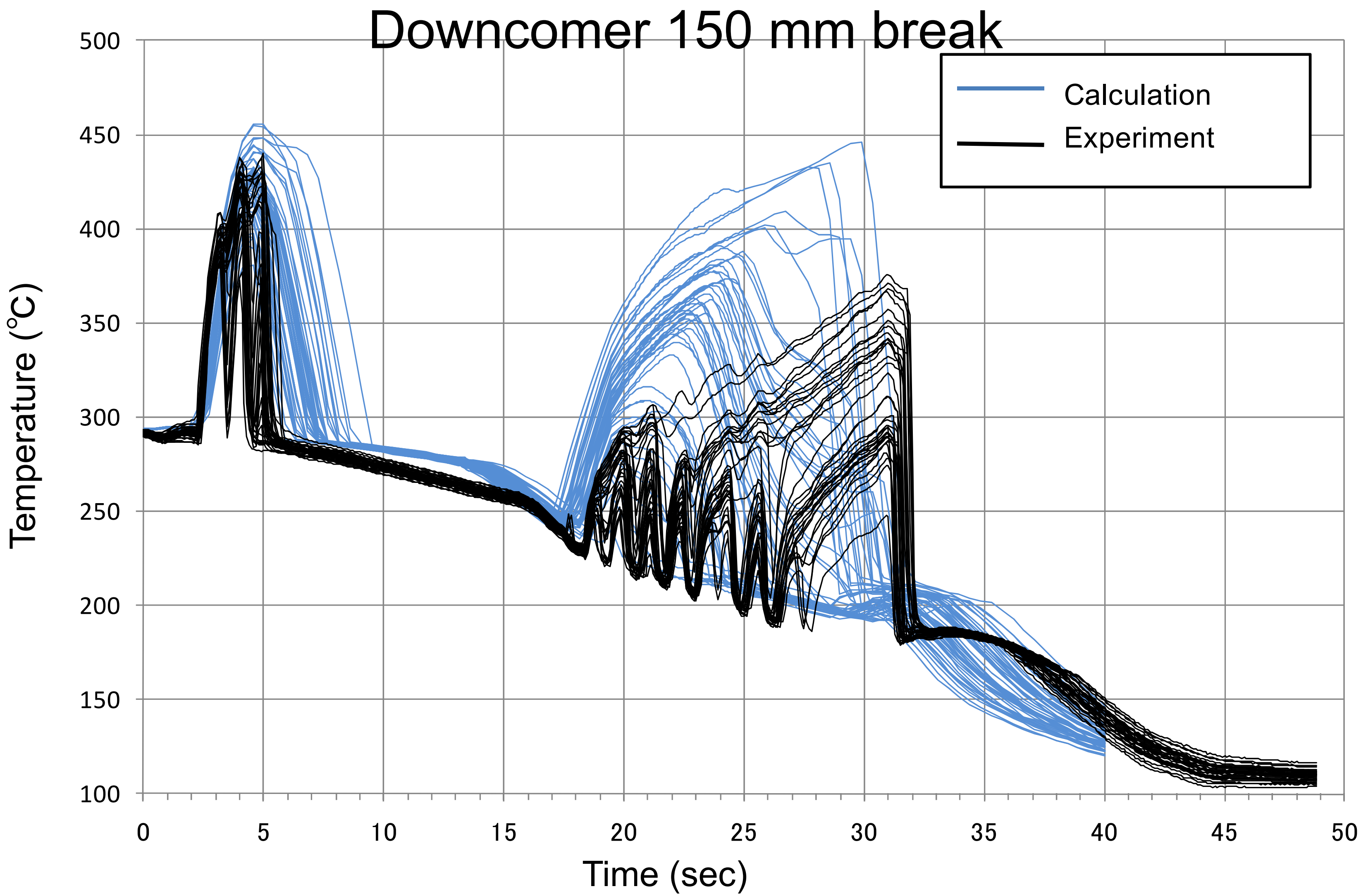


Fig.16 Behavior of cladding temperature after 100 mm downcomer break

Improvement of blow-down analysis by applying statistical method



Improvement of blow-down analysis by applying statistical method



H. Mochizuki, A Validation of ATR LOCA Thermal-hydraulic Code with a Statistical Approach, Journal of Nuclear Science and Technology, **37**, 8 (2000), pp.697-709.

Application of stochastic method to FBR analysis

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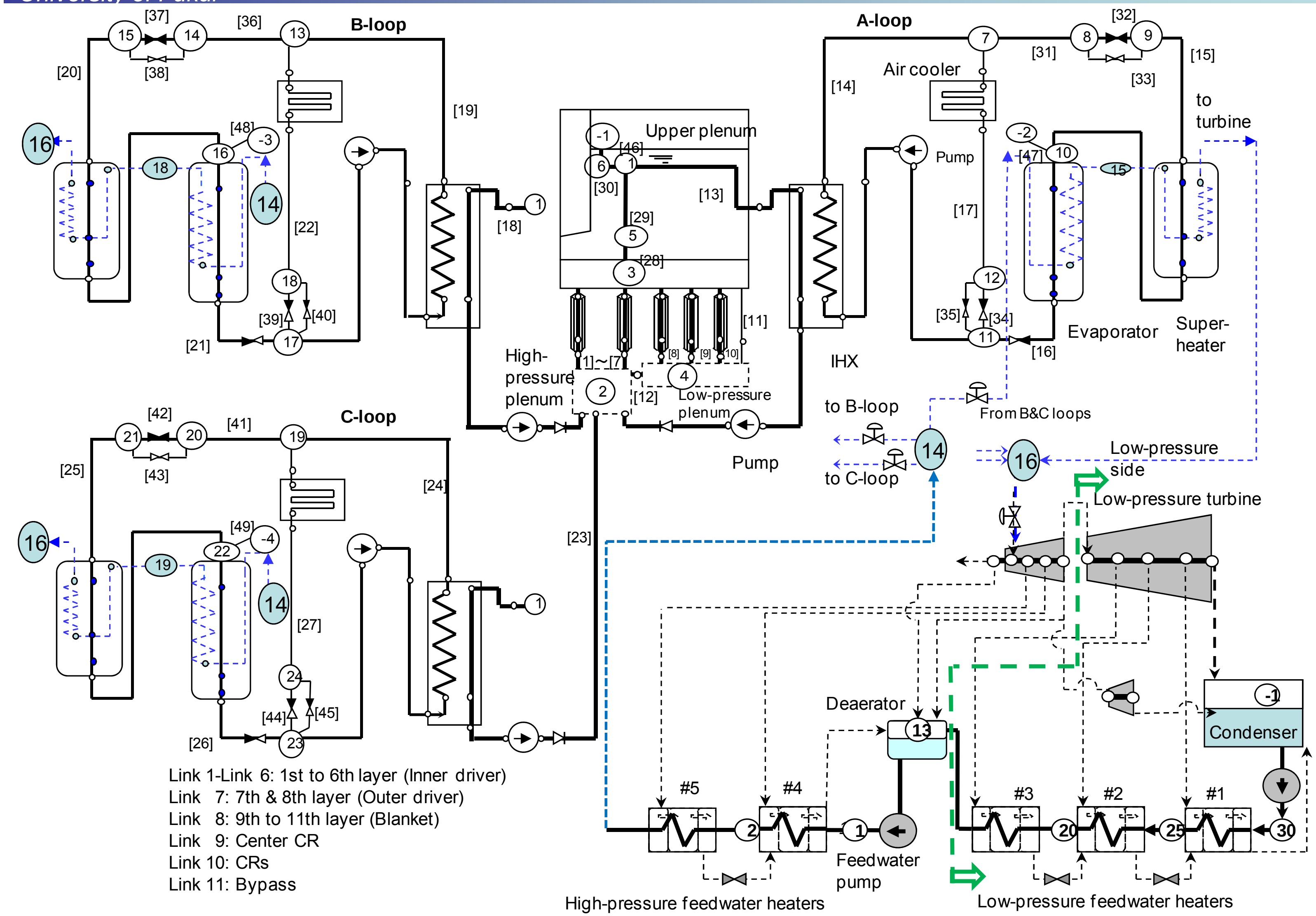
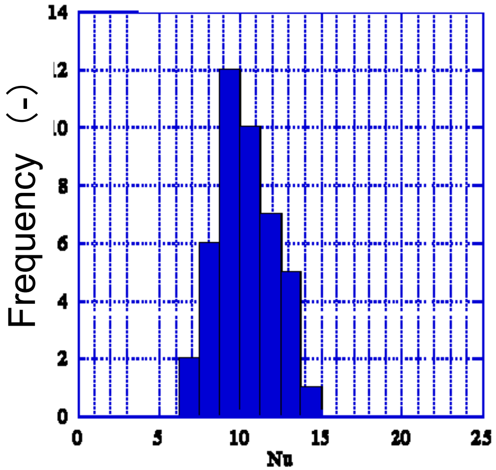


Link 1-Link 6: 1st to 6th layer (Inner driver)
Link 7: 7th & 8th layer (Outer driver)
Link 8: 9th to 11th layer (Blanket)
Link 9: Center CR
Link 10: CRs
Link 11: Bypass

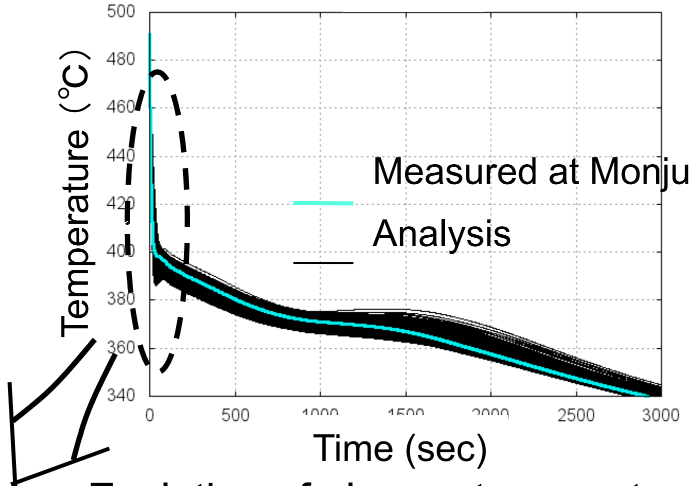
H. Mochizuki, Development of the plant dynamics analysis code NETFLOW++, Nuclear Engineering and Design, **240**, (2010), pp. 577-587.

Application of stochastic method to FBR

Pe number: 1000

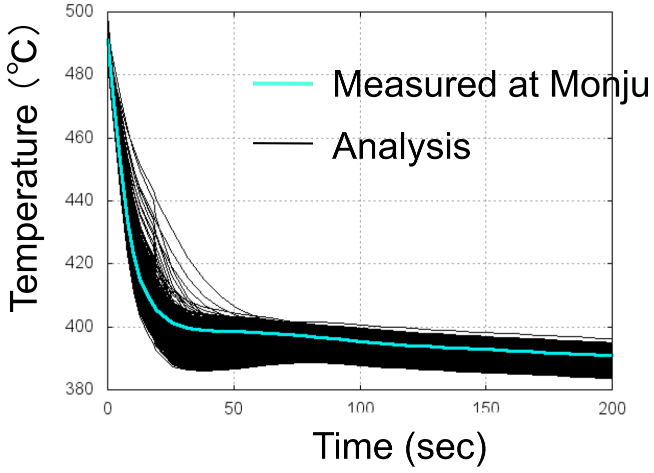


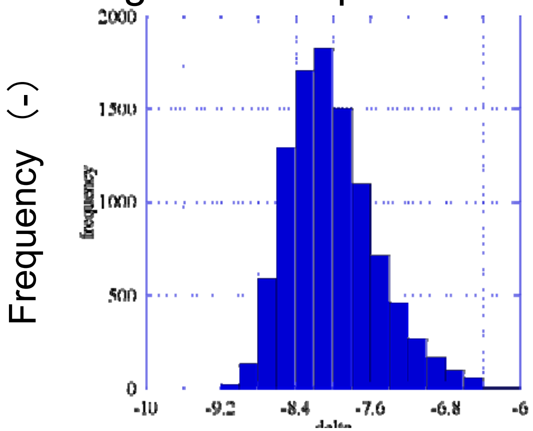
Statistics if measured Nu number



Evolution of plenum temperature during turbine trip

Method:
Plant parameters are investigated by 10,000 trials of the Monte-Carlo calculation for 43 factors which can affect on the plenum temperature.





Cool-down velocity of plenum temperature (°C/s)

Result:
The measured temperature transient has been included in the group of calculated curves. Most non-safety side value could be evaluated taking into account various statistical errors.

[1] N. Kikuchi and H. Mochizuki, Application of statistical method for FBR plant transient computation, FR'13, Paris (2013-March)

Severe accident

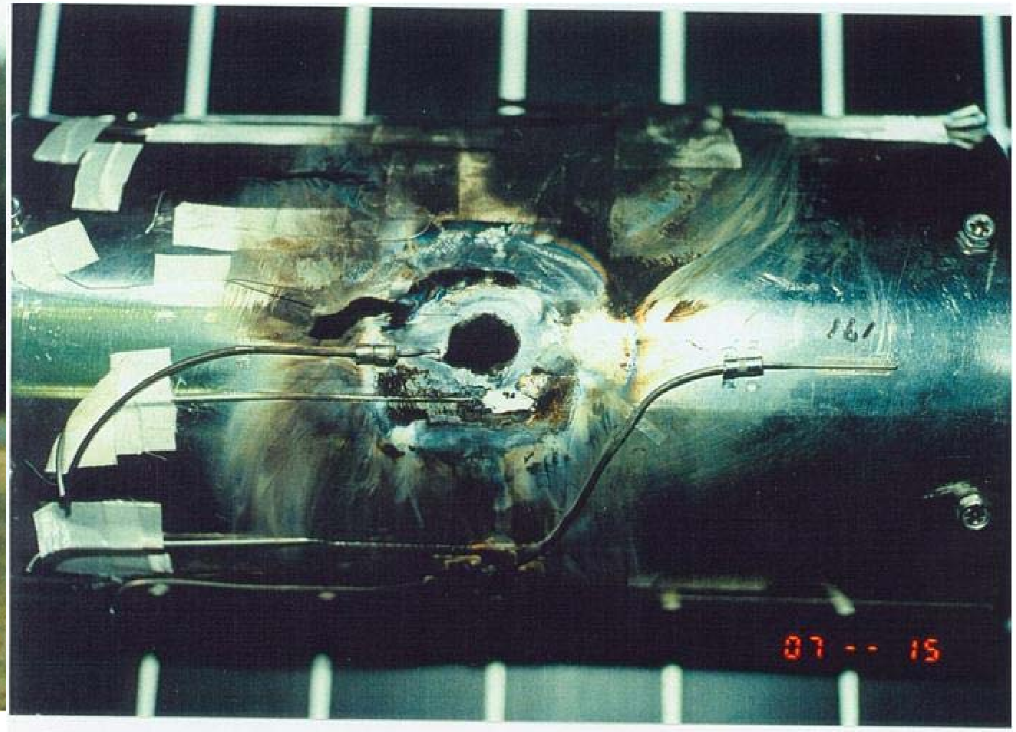
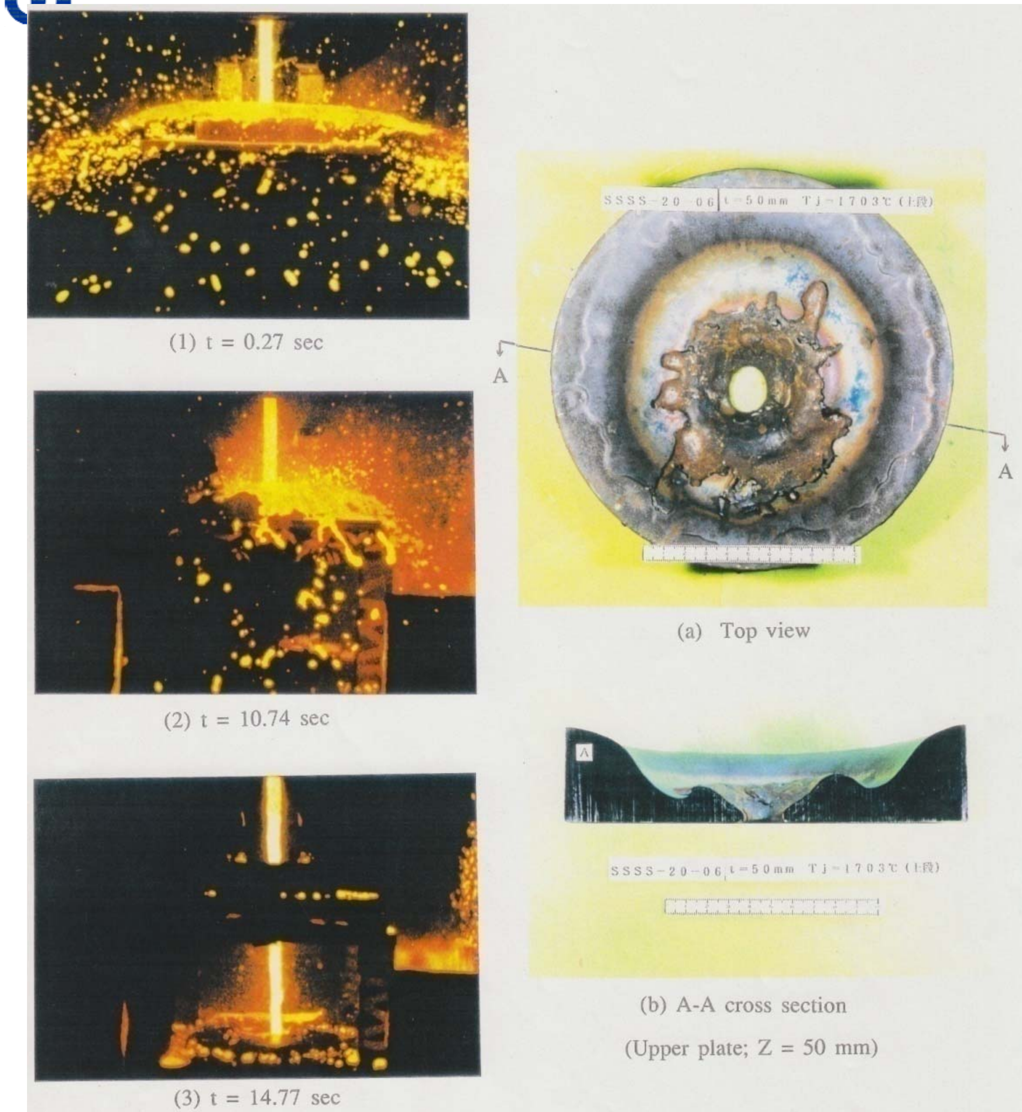


Photo. Erosion experiment of Zr-2.5%Nb pressure tube by molten metal

Heat transfer of melted fuel to material



(1) t = 0.27 sec

(2) t = 10.74 sec

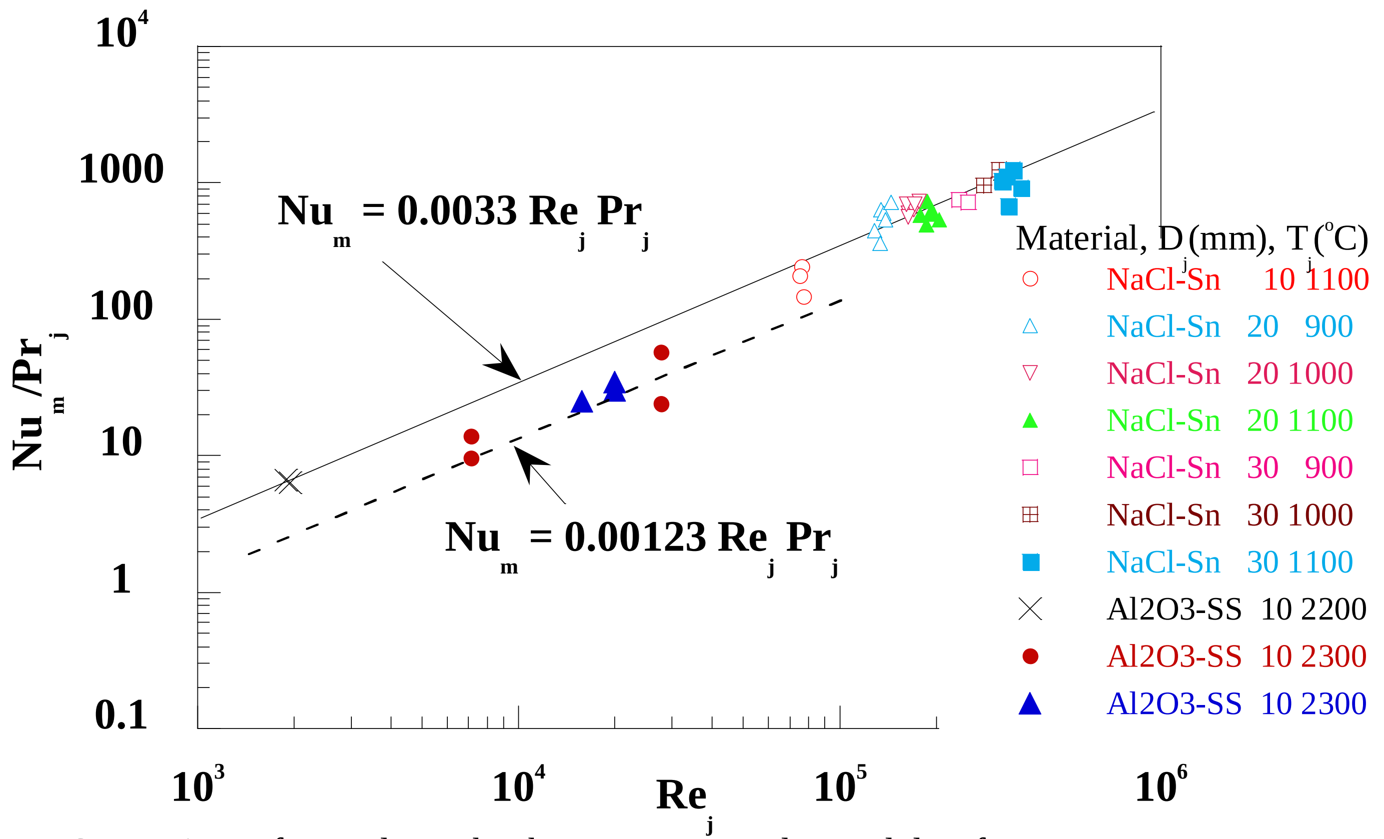
(3) t = 14.77 sec

(a) Top view

(b) A-A cross section

(Upper plate; Z = 50 mm)

Heat transfer between melted jet and materials

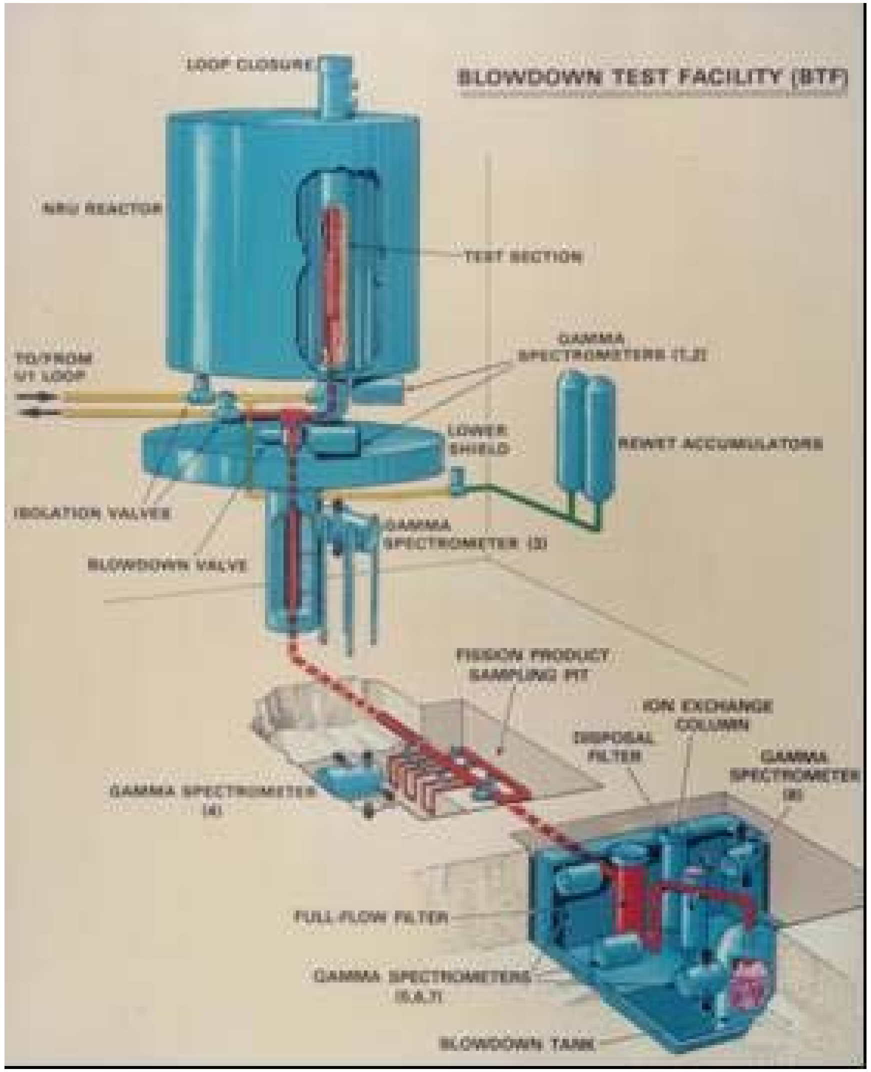


Comparison of Nusselt number between present data and data from Saito et al.1) and Mochizuki2).

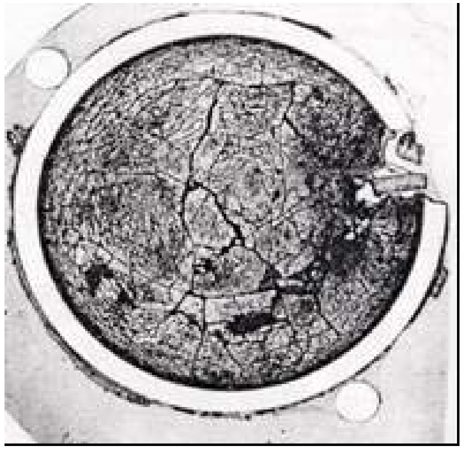
1)Saito, et al., Nuclear Engineering and Design, 132 (1991)

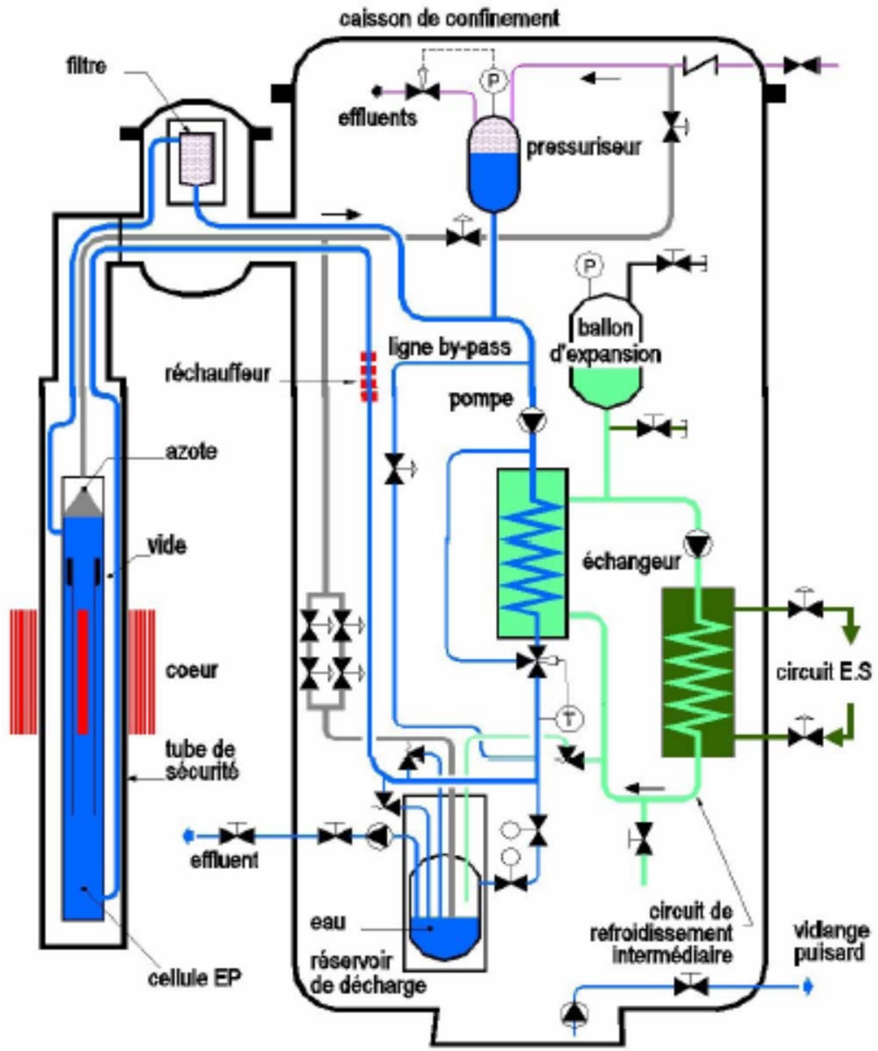
2)Mochizuki, Accident Management and Simulation Symposium, Jackson Hole, (1997).

Fuel melt experiment using BTF in Canada



Fuel melt experiment using CABRI

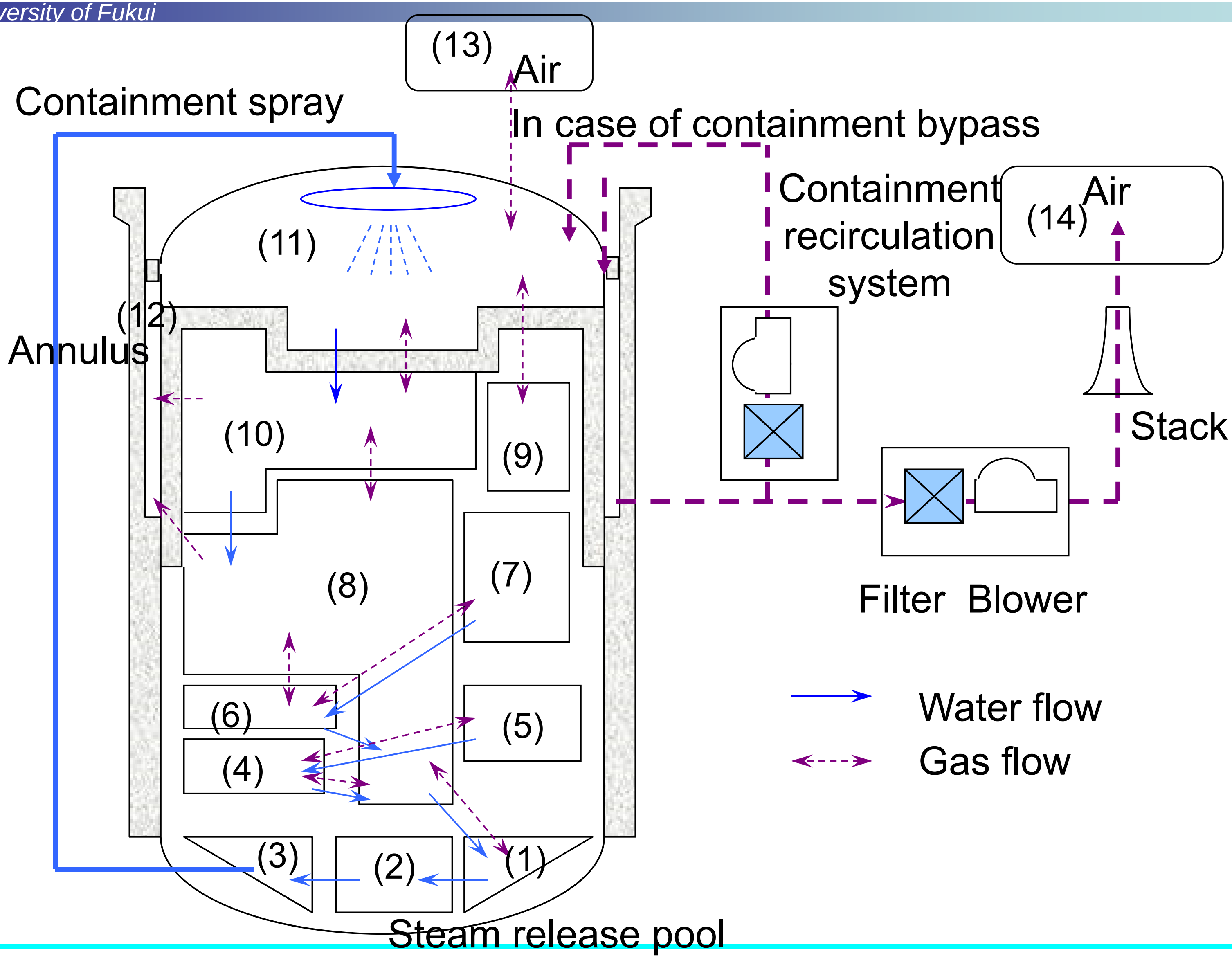




Source term analysis codes

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

CONATIN code



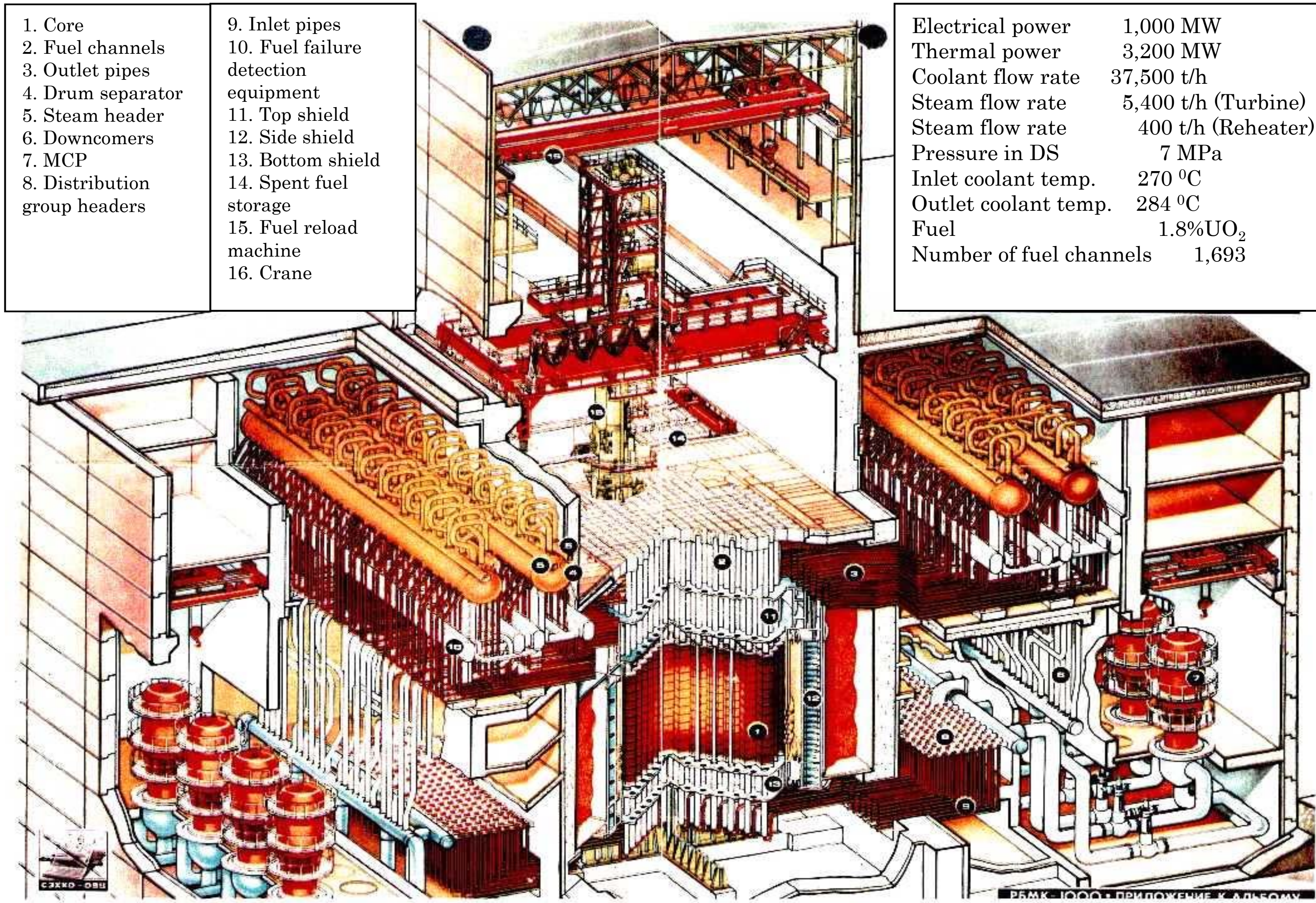
Fluid- structure interaction analysis during hydrogen detonation

Analysis of Chernobyl Accident
- Investigation of Root Cause -

Schematic of Chernobyl NPP

1. Core
2. Fuel channels
3. Outlet pipes
4. Drum separator
5. Steam header
6. Downcomers
7. MCP
8. Distribution group headers
9. Inlet pipes
10. Fuel failure detection equipment
11. Top shield
12. Side shield
13. Bottom shield
14. Spent fuel storage
15. Fuel reload machine
16. Crane

Electrical power	1,000 MW
Thermal power	3,200 MW
Coolant flow rate	37,500 t/h
Steam flow rate	5,400 t/h (Turbine)
Steam flow rate	400 t/h (Reheater)
Pressure in DS	7 MPa
Inlet coolant temp.	270 ^{0}C
Outlet coolant temp.	284 ^{0}C
Fuel	1.8%UO_2
Number of fuel channels	1,693



Elevation Plan

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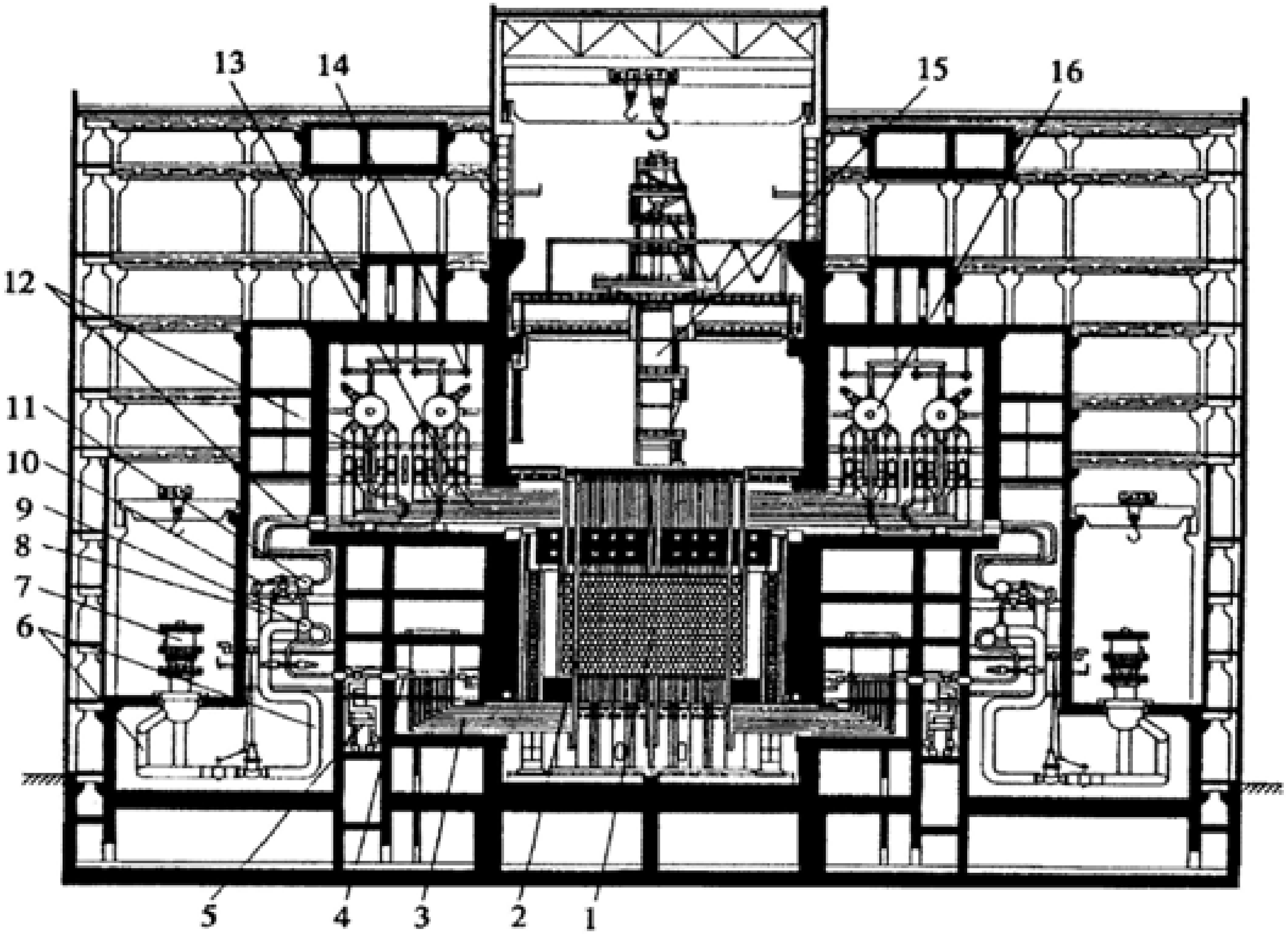
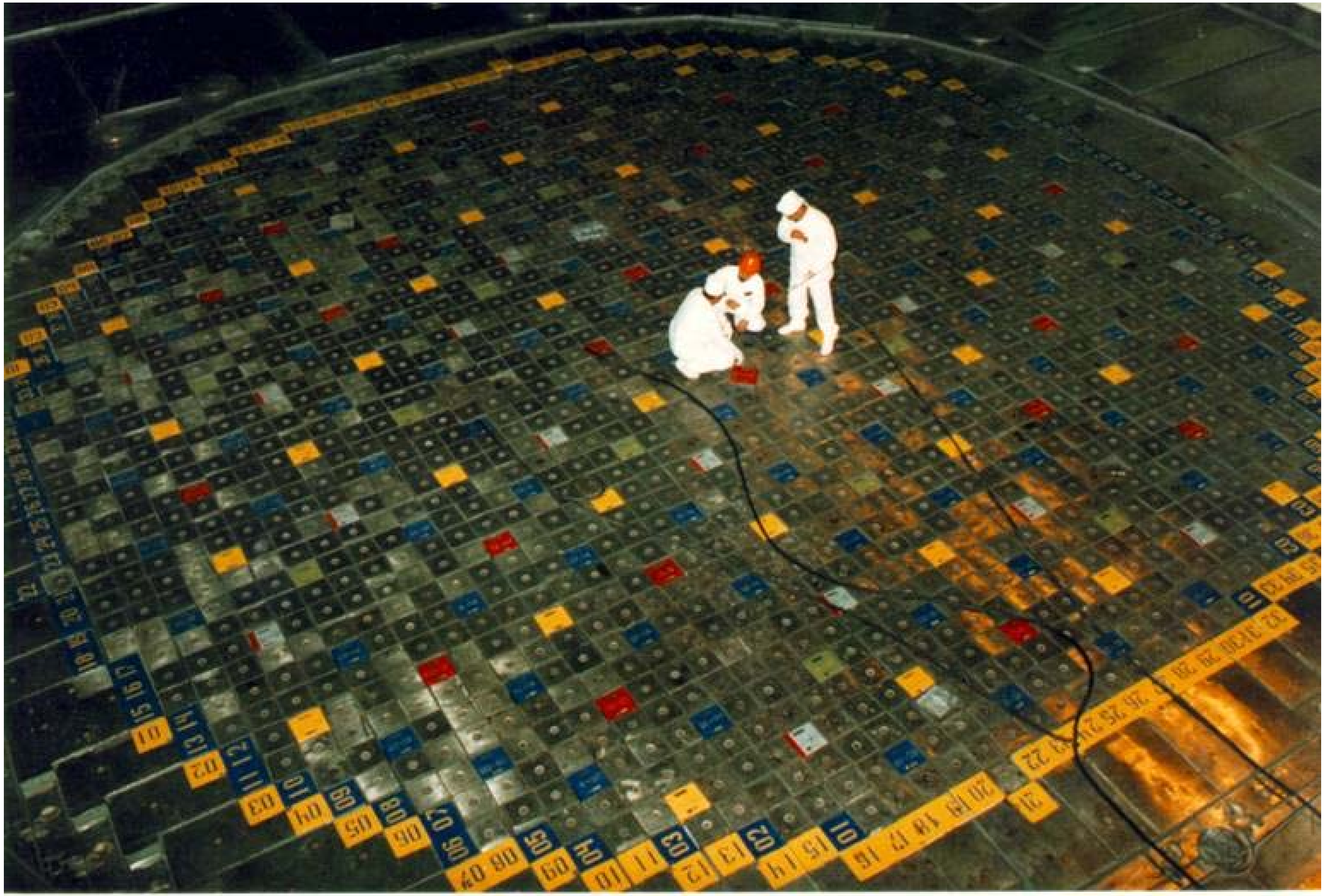


Above the Core of Ignarina NPP

University of Fukui



Core and Re-fueling Machine

Control Room

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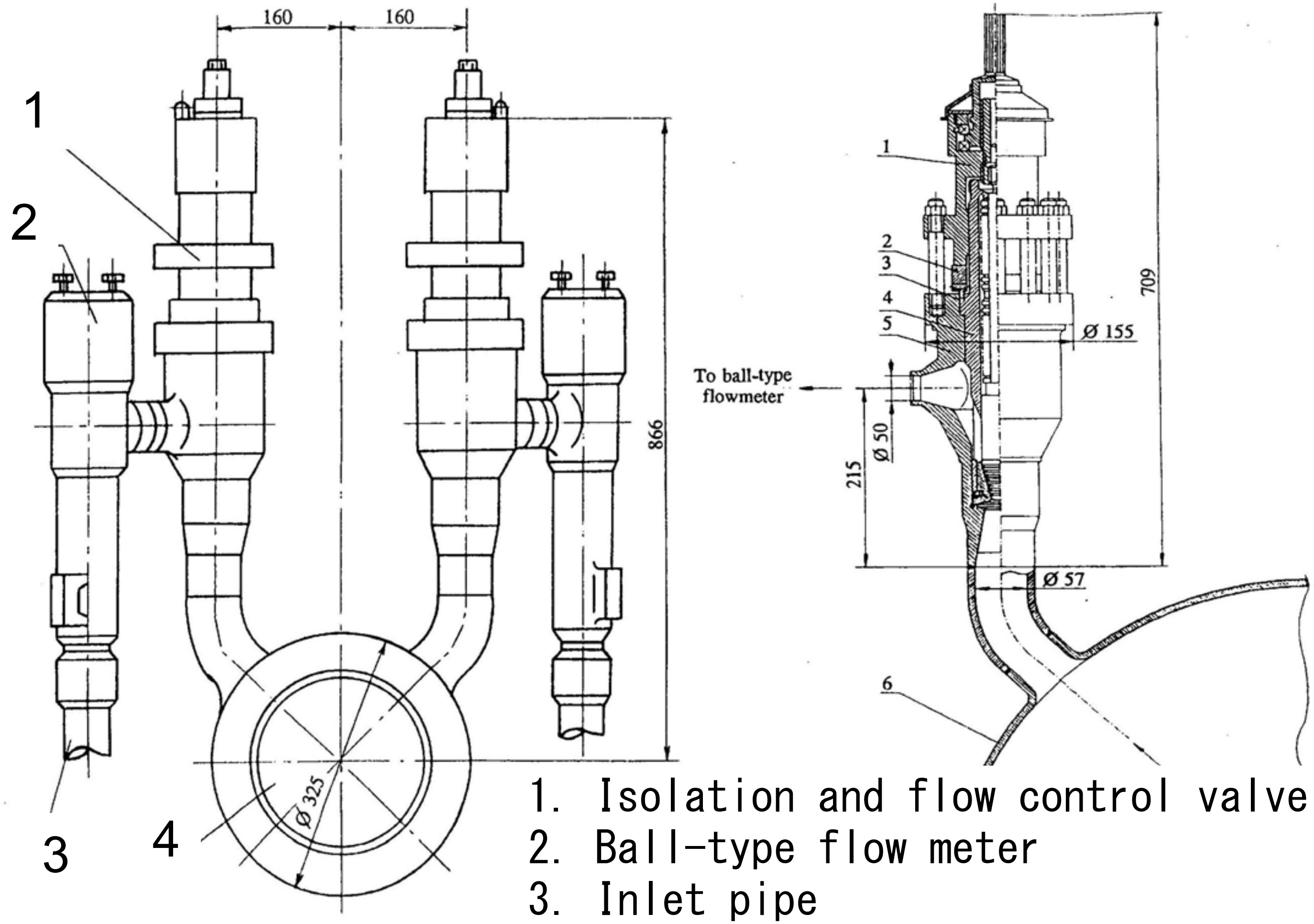
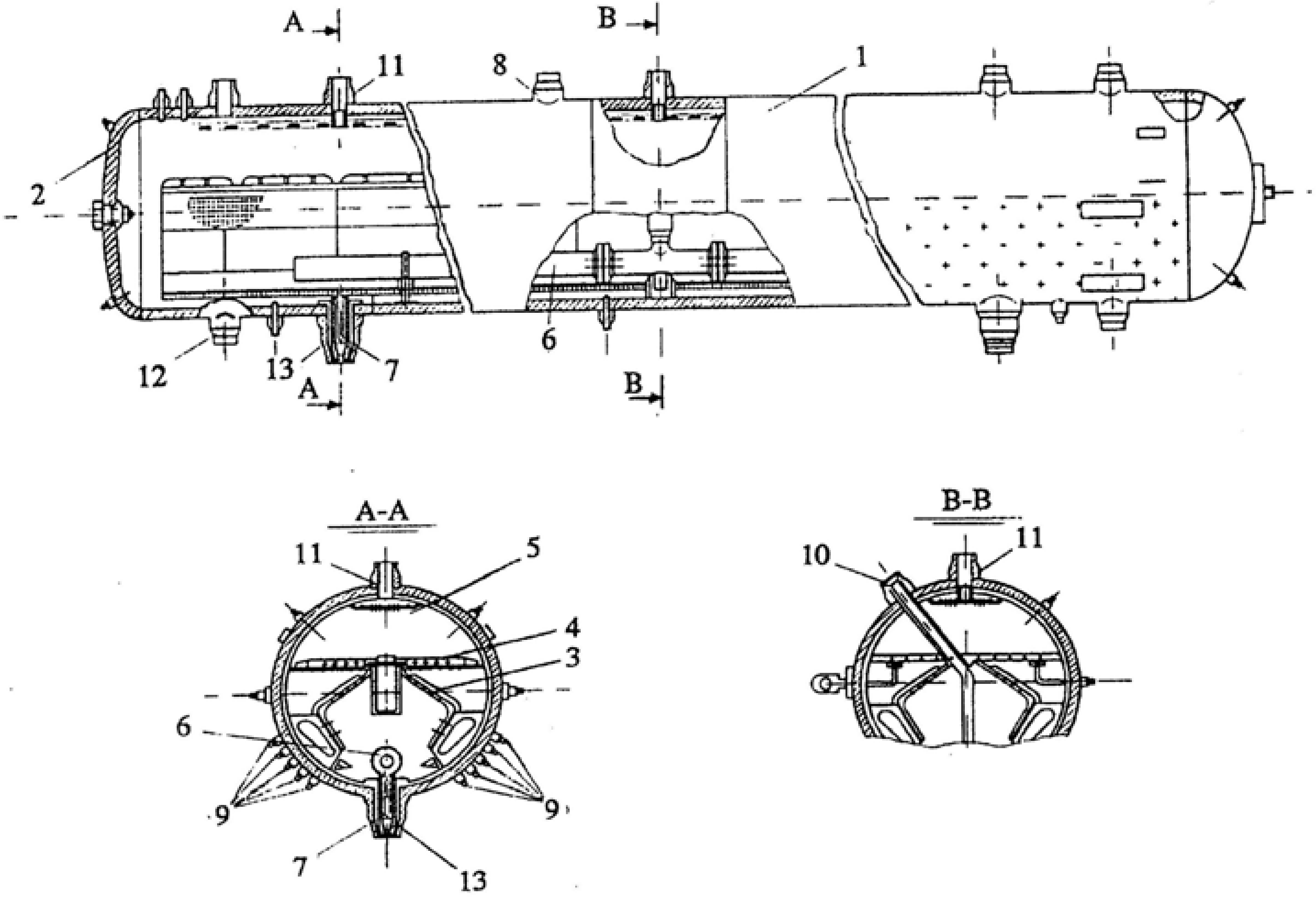
Configuration of inlet valve



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

Drum Separator



Configuration of Fuel Channnel

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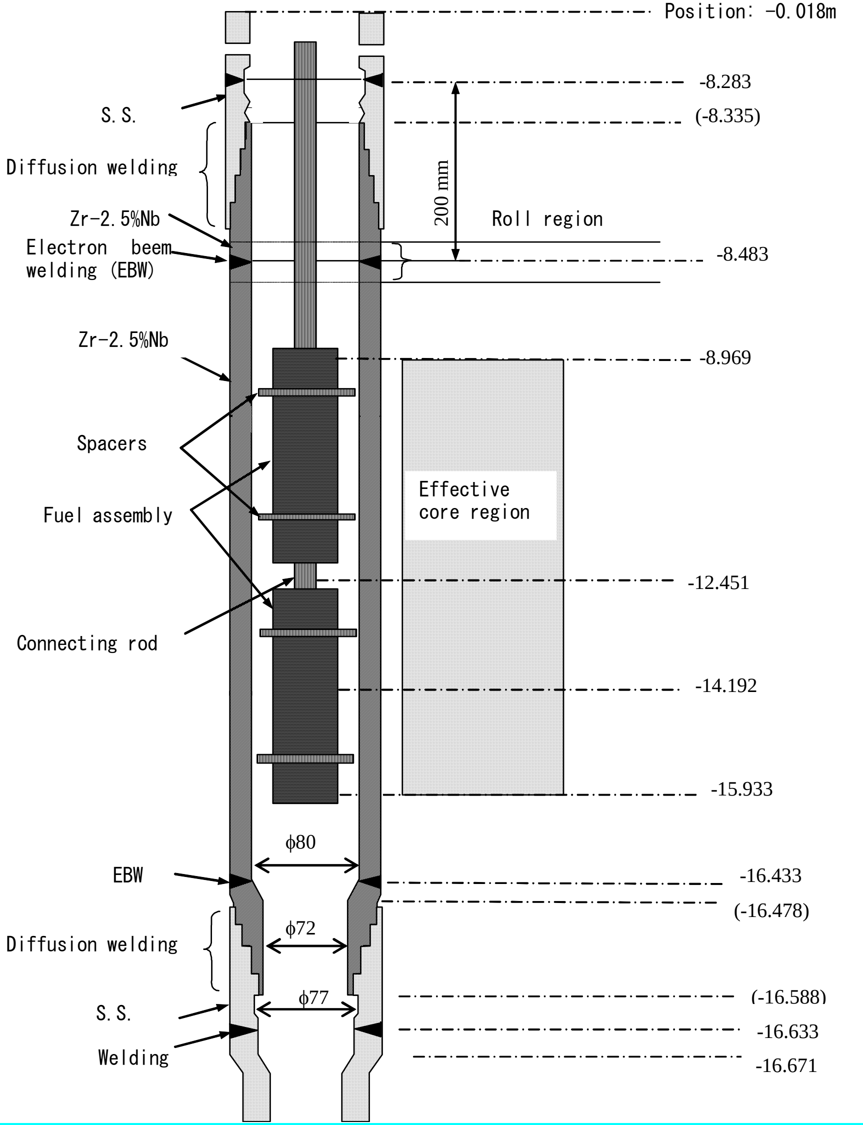
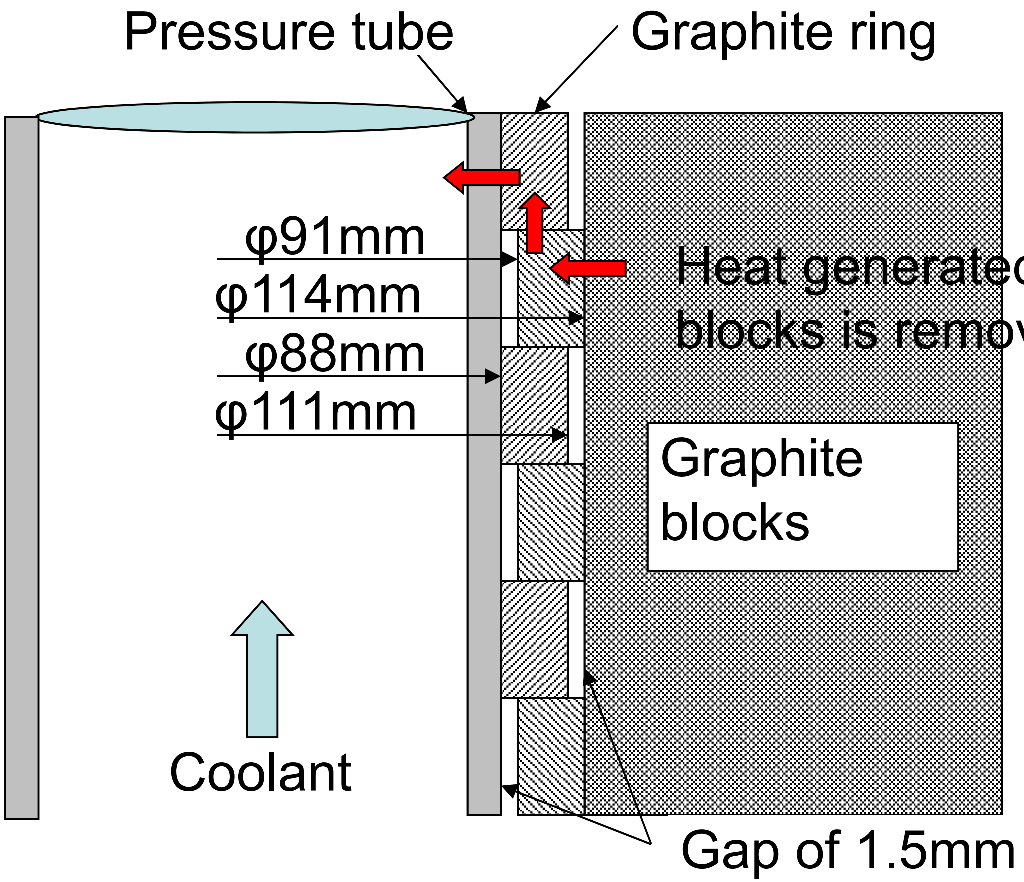


Heat Removal by Moderation



Maximum graphite temperature is 720℃ at rated power

RBMK & VVER

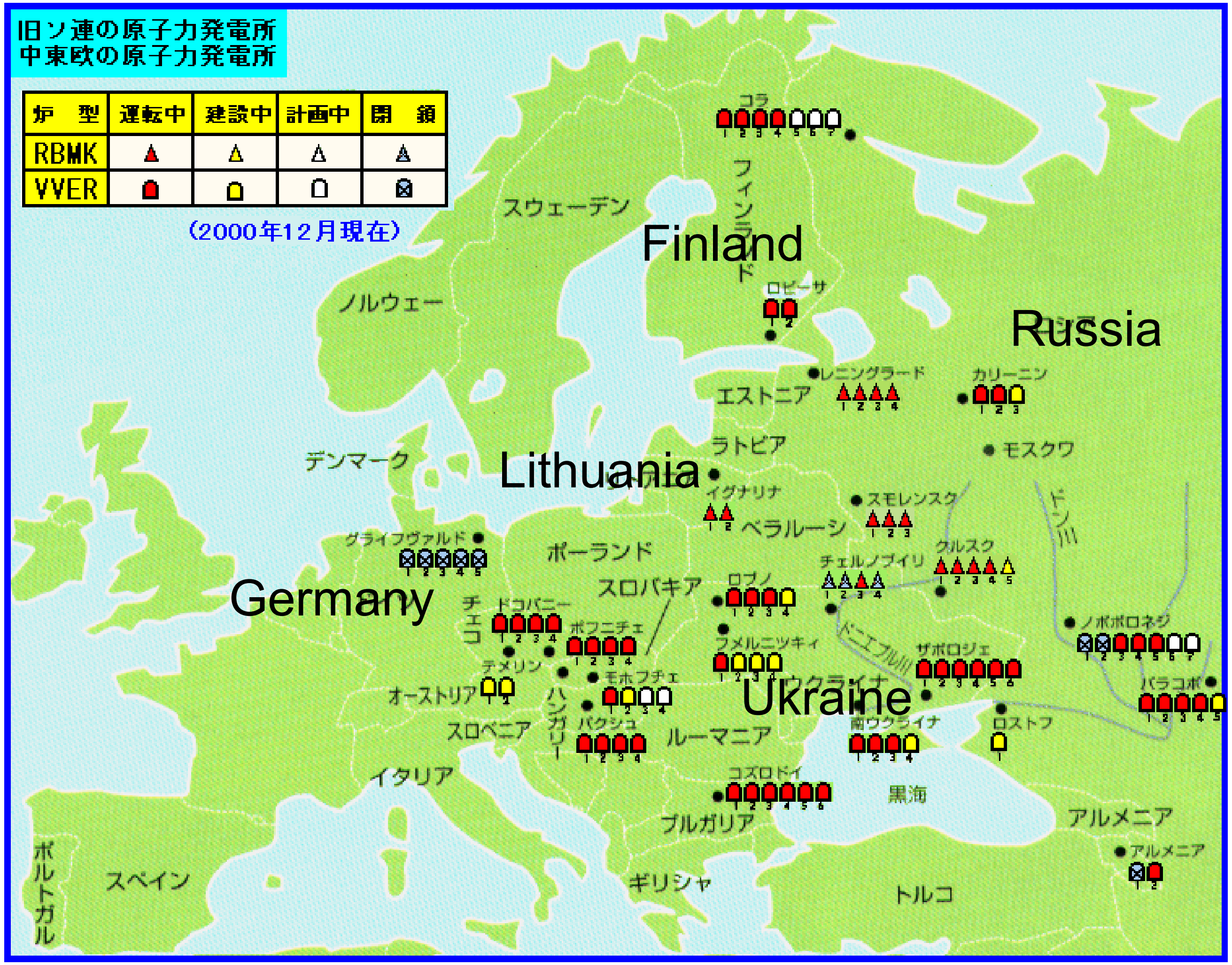


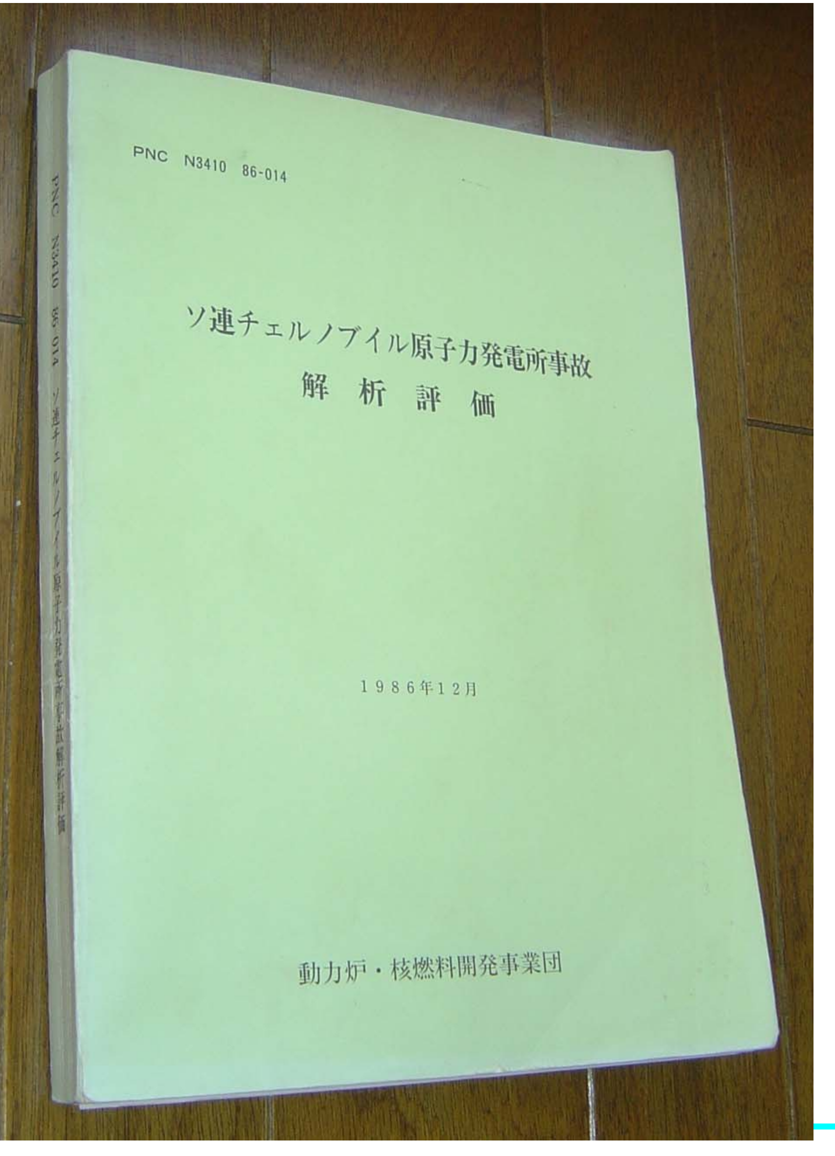
図２　ソ連型炉の所在地

［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/)

Objective of the Experiment

- 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.

Report in Dec. 1986



Trend of the Reactor Power

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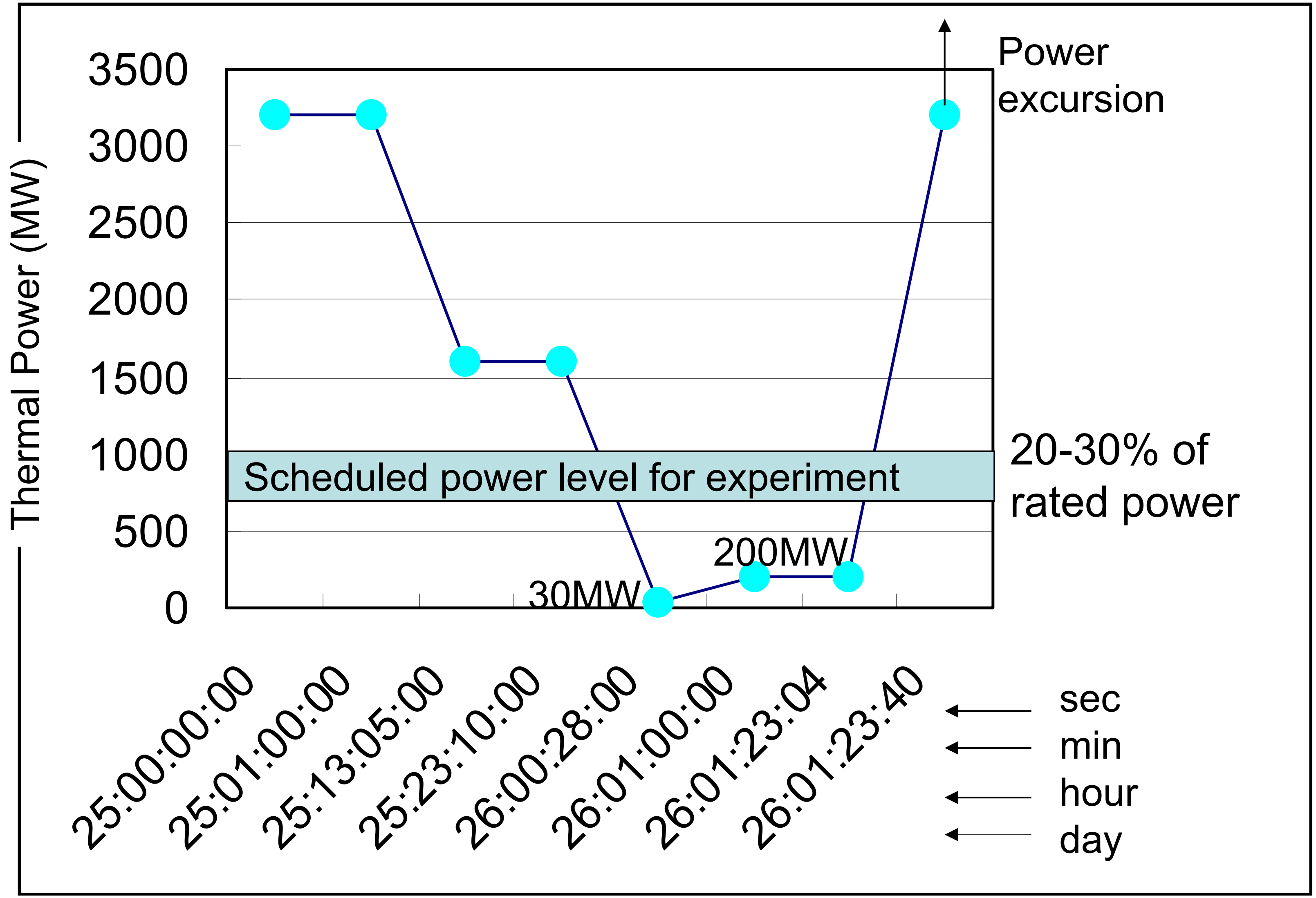
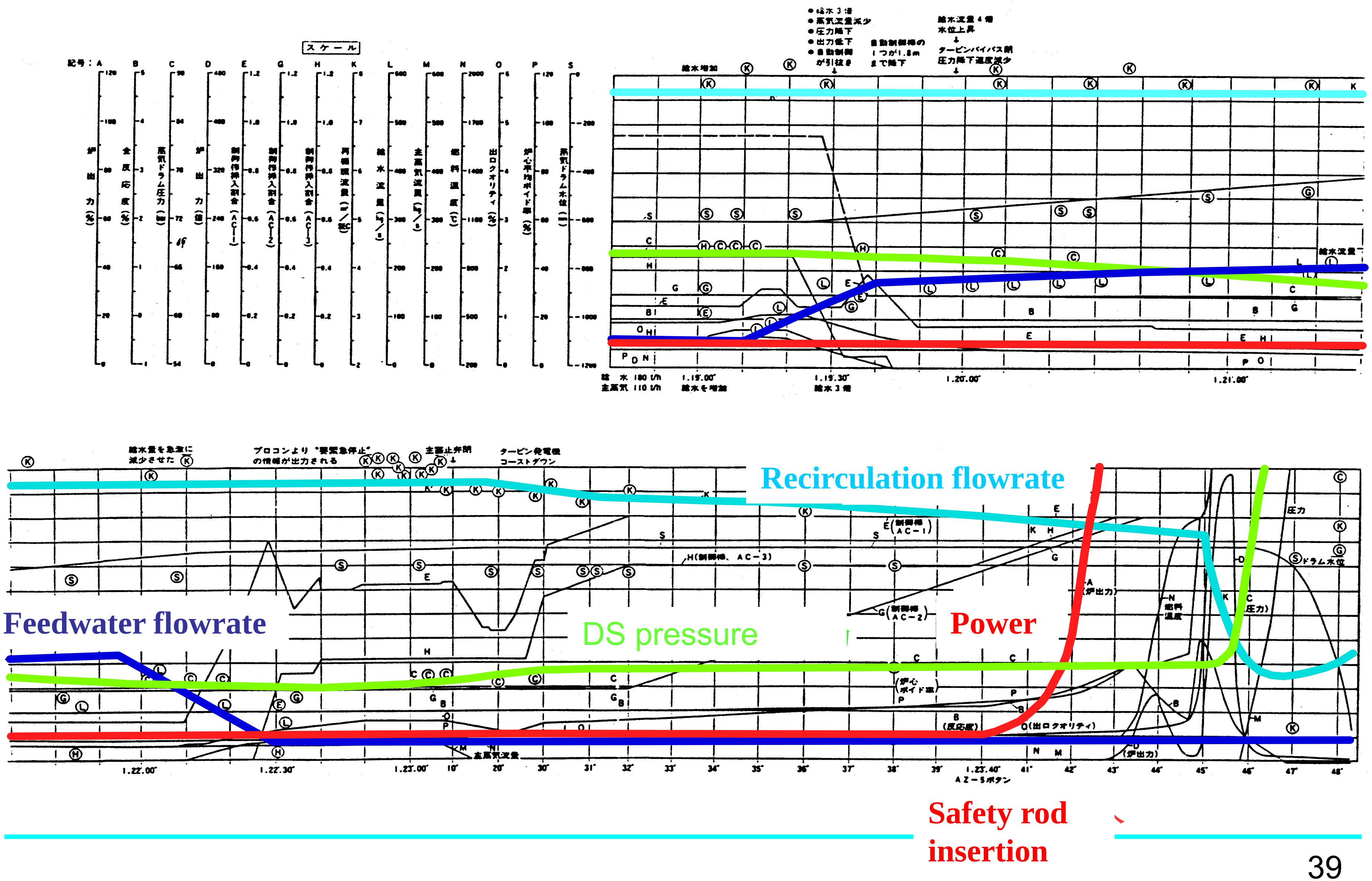


Time Chart Presented by USSR



Result in the Past Analysis (1/2)

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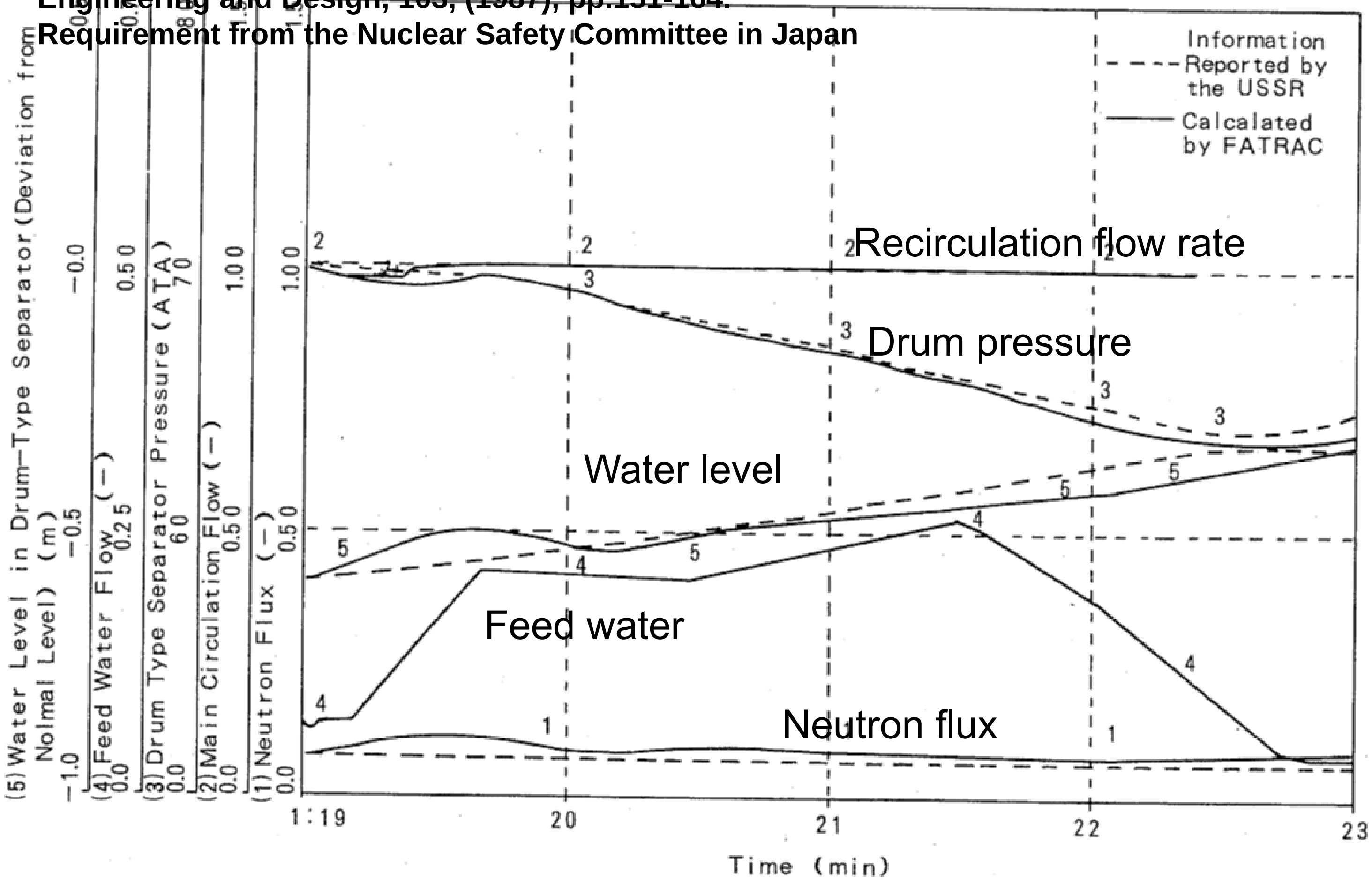
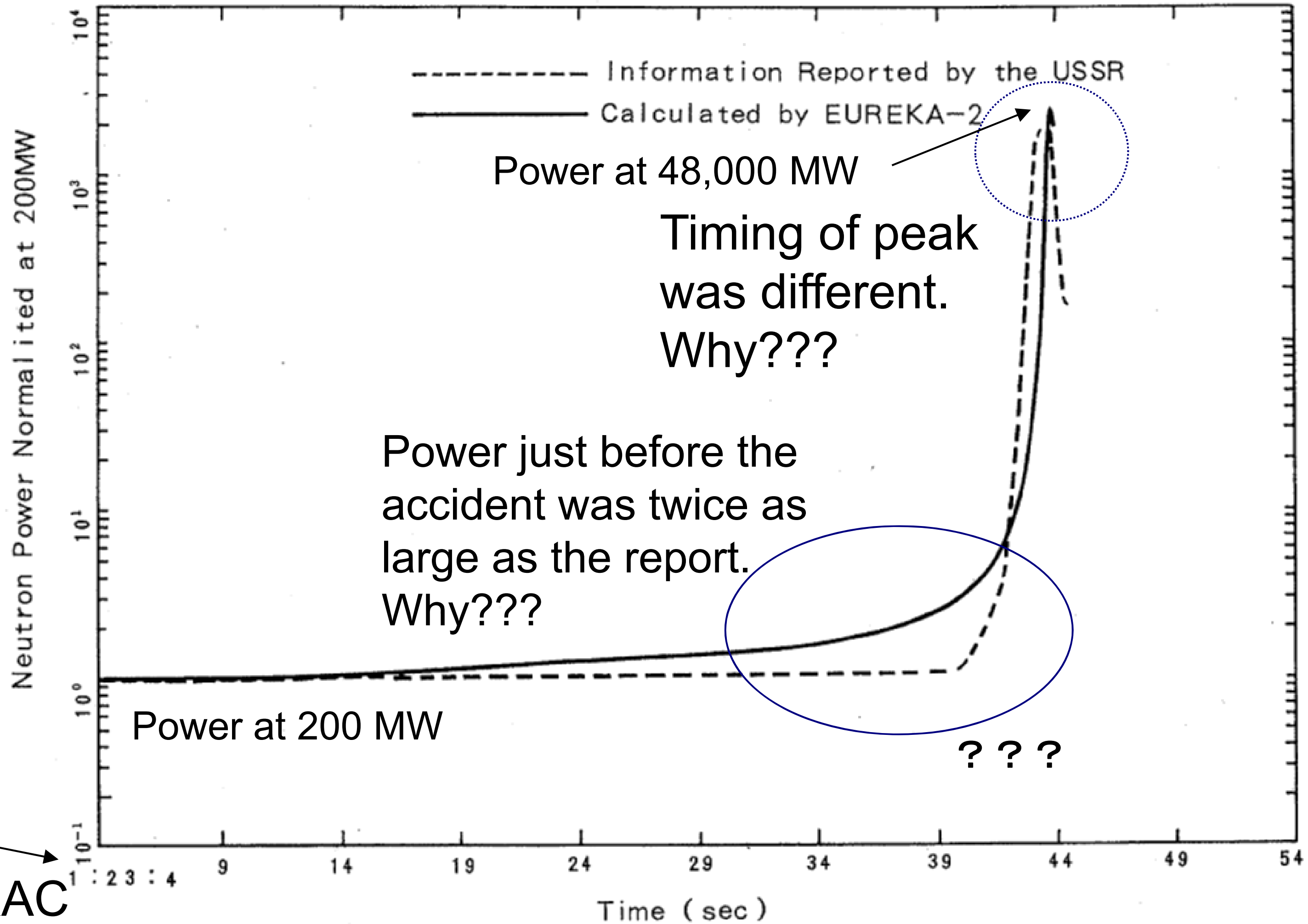
- T. Wakabayashi, H. Mochizuki, et al., Analysis of the Chernobyl Reactor Accident (I) Nuclear and Thermal Hydraulic Characteristics and Follow-up Calculation of the Accident, J. Atomic Energy Society of Japan, 28, 12 (1986), pp.1153-1164.
- T. Wakabayashi, H. Mochizuki, et al., Analysis of the Chernobyl 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



Result in the Past Analysis (2/2)

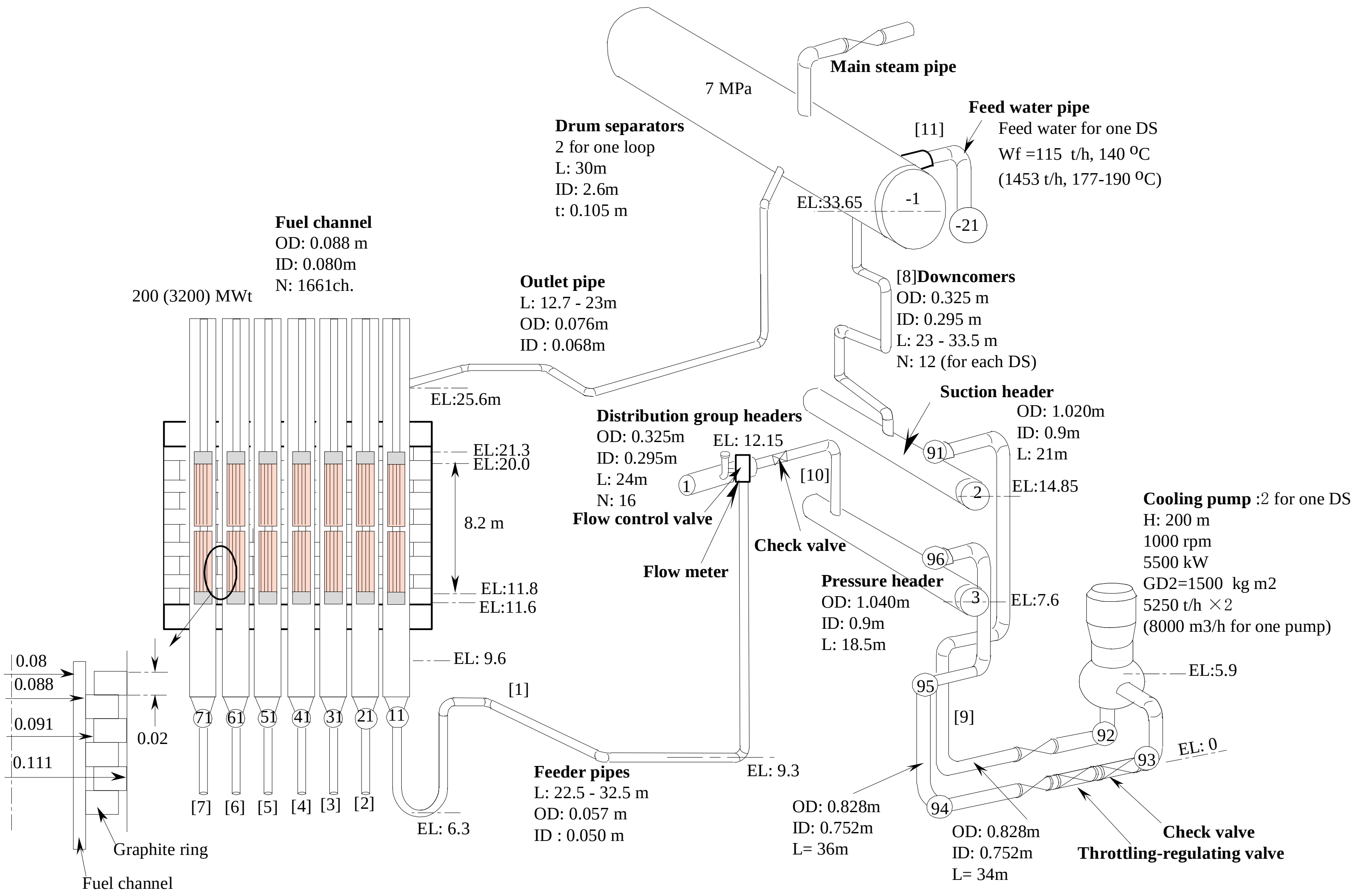


Result of FATRAC code is transferred, and initial steady calculation was conducted.

Possible Trigger of the Accident

- Positive scram due to flaw of scram rods
- Pump cavitation
- Pump coast-down
- Opening of turbine bypass valve (6.96MPa)

Calculation Model by NETFLOW++ Code



Trigger of the Accident

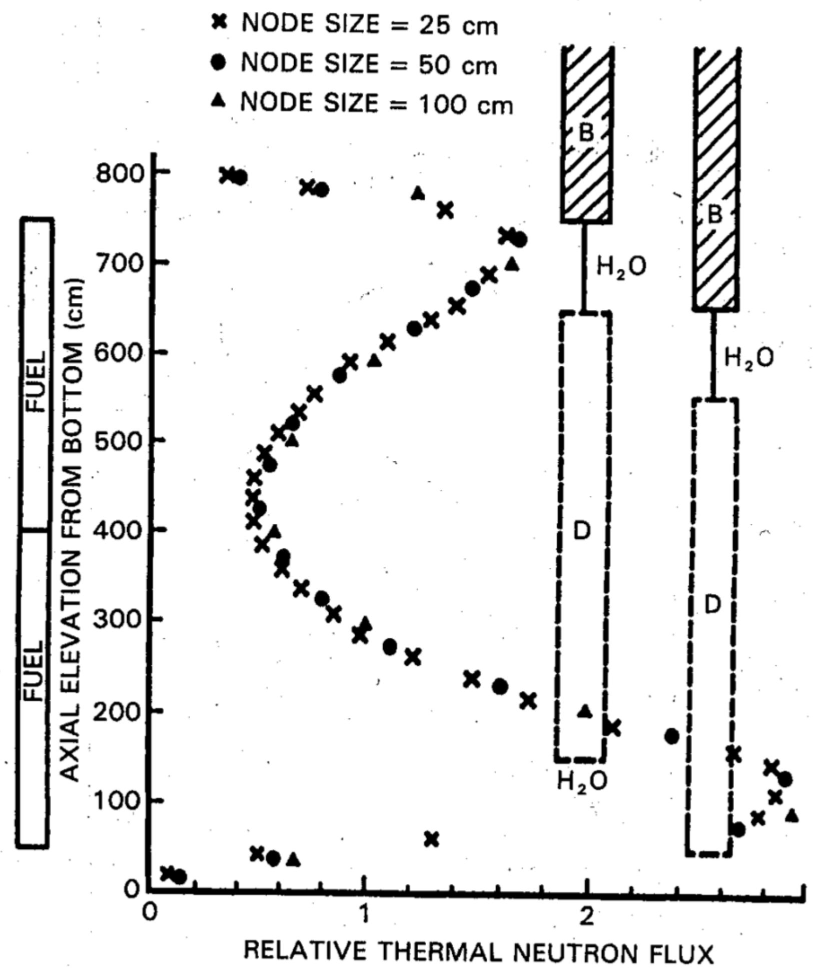


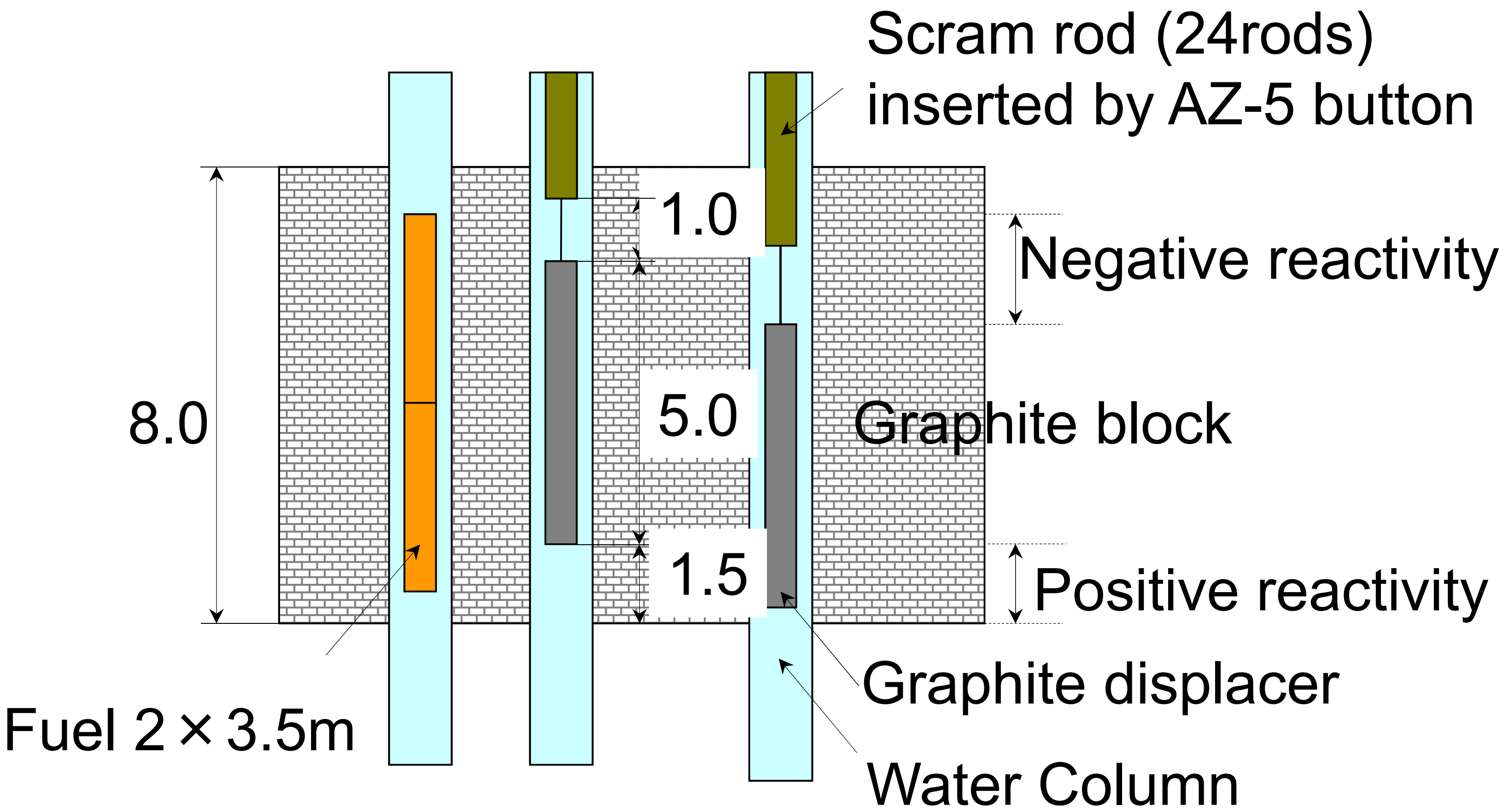
Fig. 1. Effect of node size on axial flux distribution. In this figure, B = B_4C absorber and D = graphite displacer.

- 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.)



Simulation from 1:19:00 to First Peak

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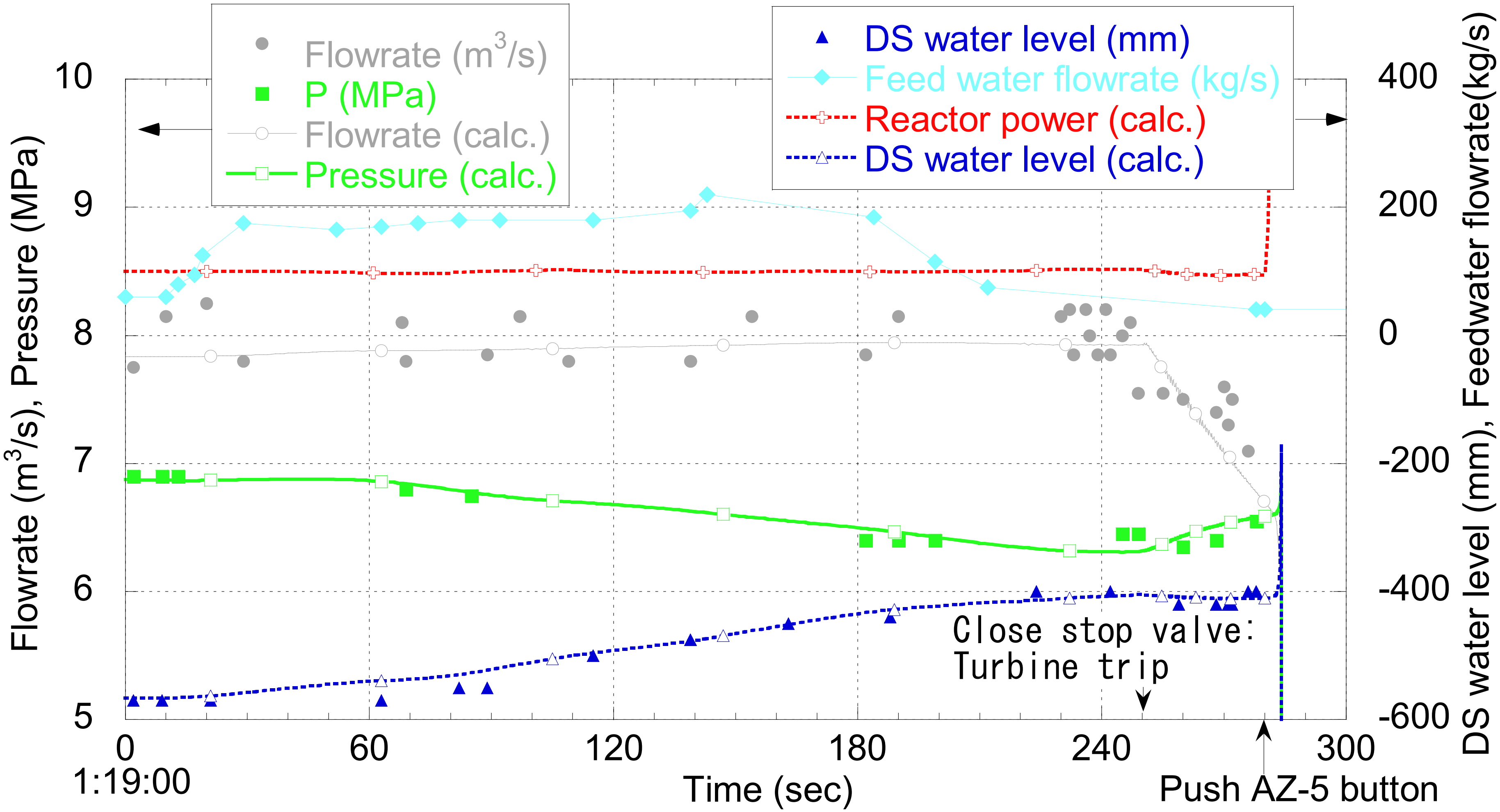
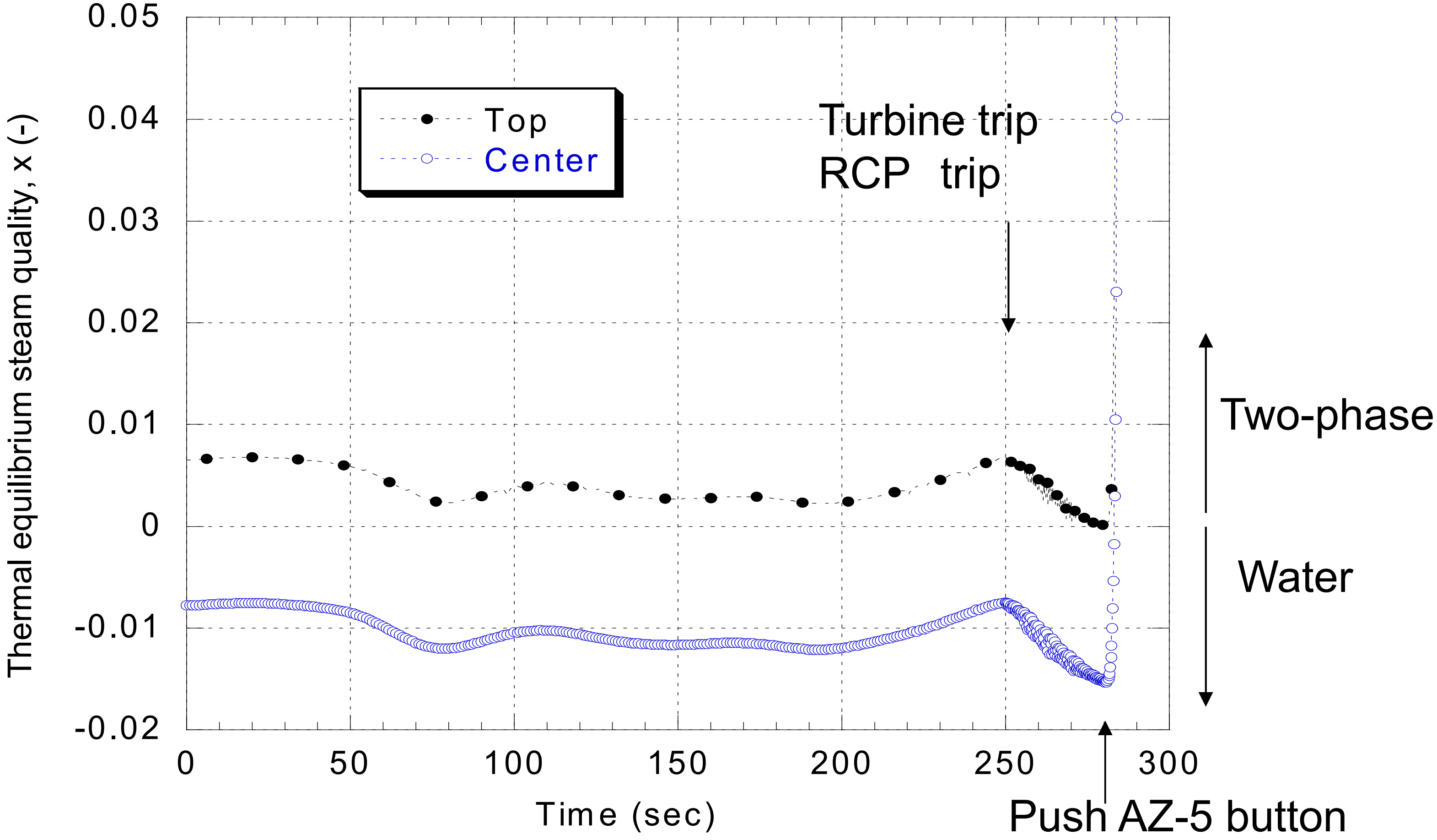
Data acquired by SKALA

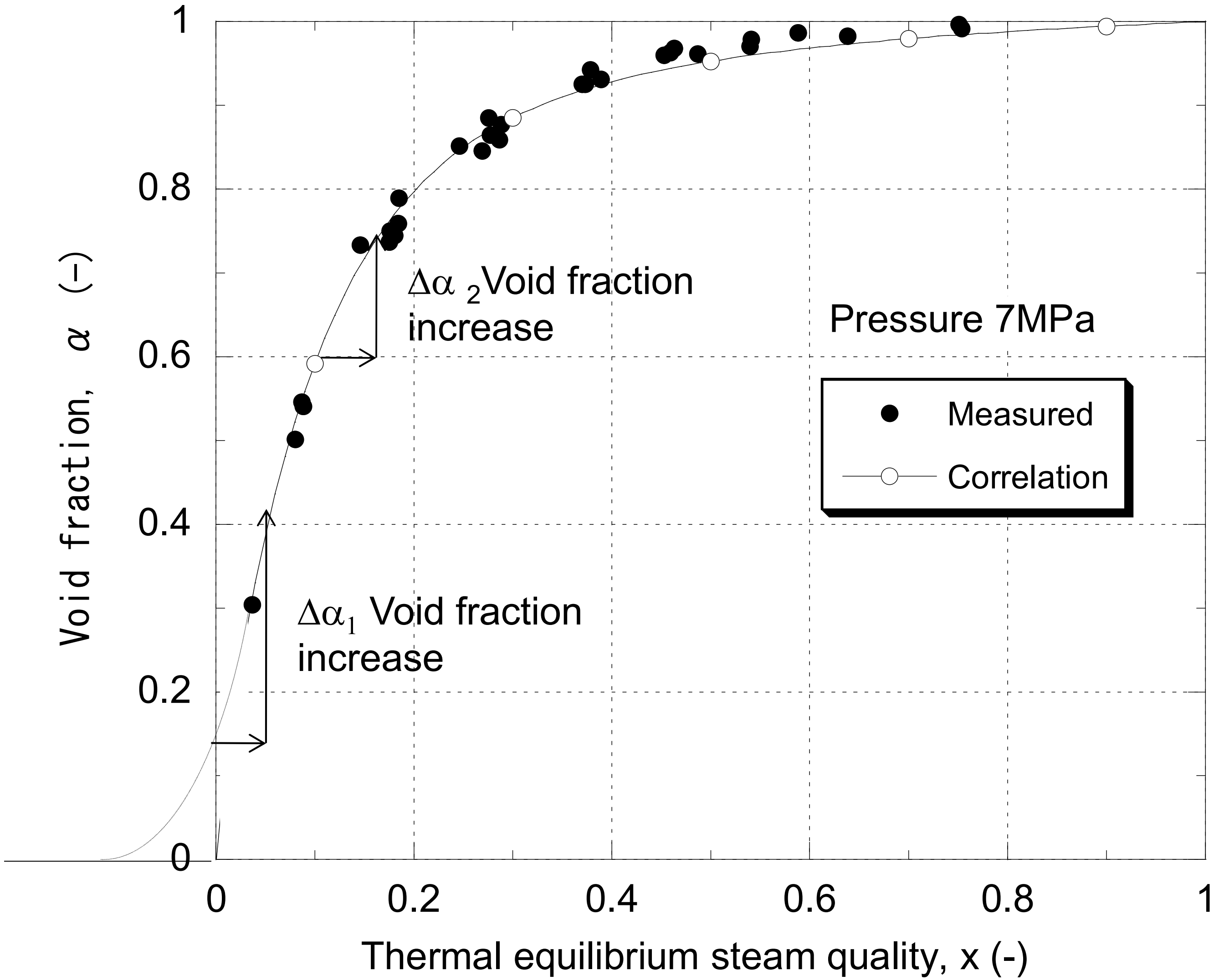


Trend of parameters for one loop from 1:19:00 on 26 April 1986

Behavior of Steam Quality

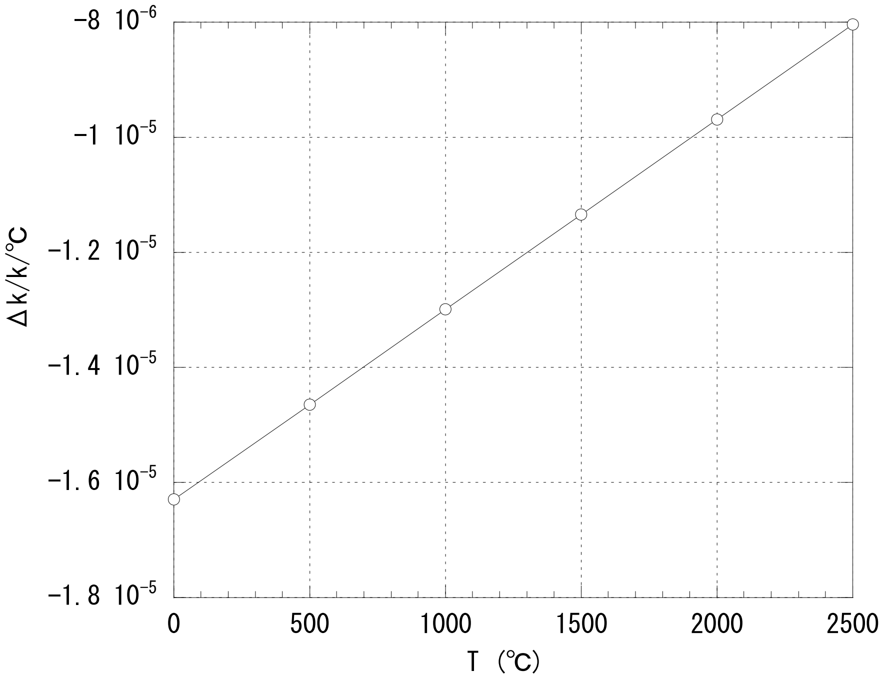


Void Characteristic

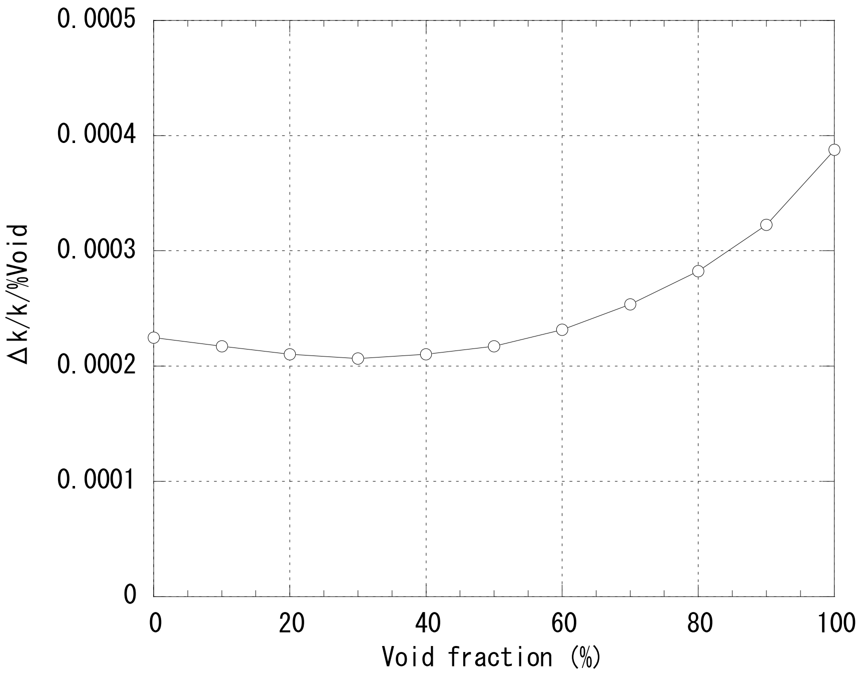


Nuclear Characteristics

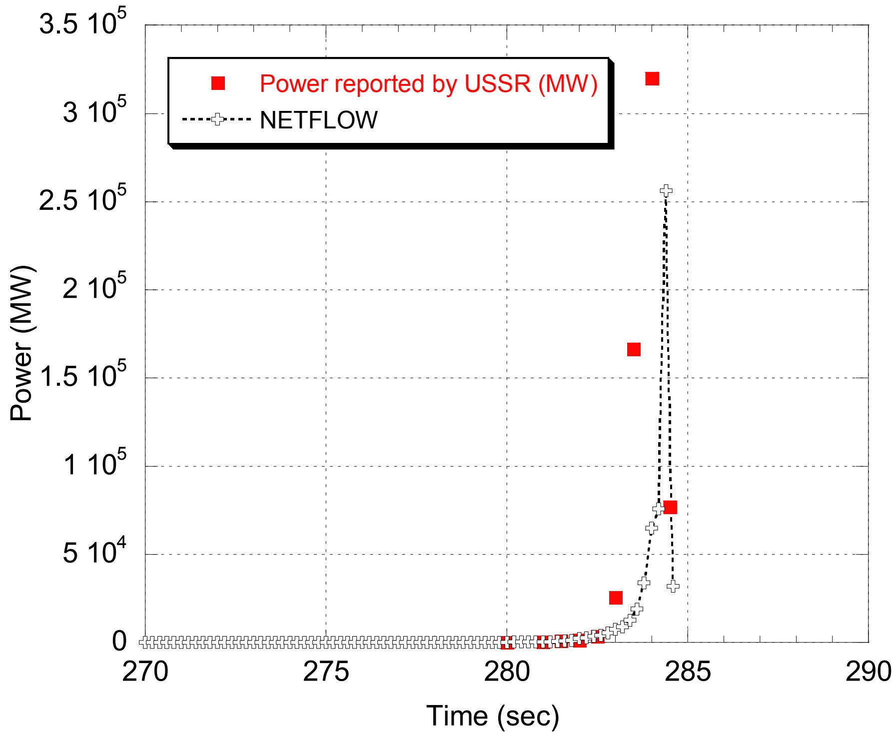
Doppler

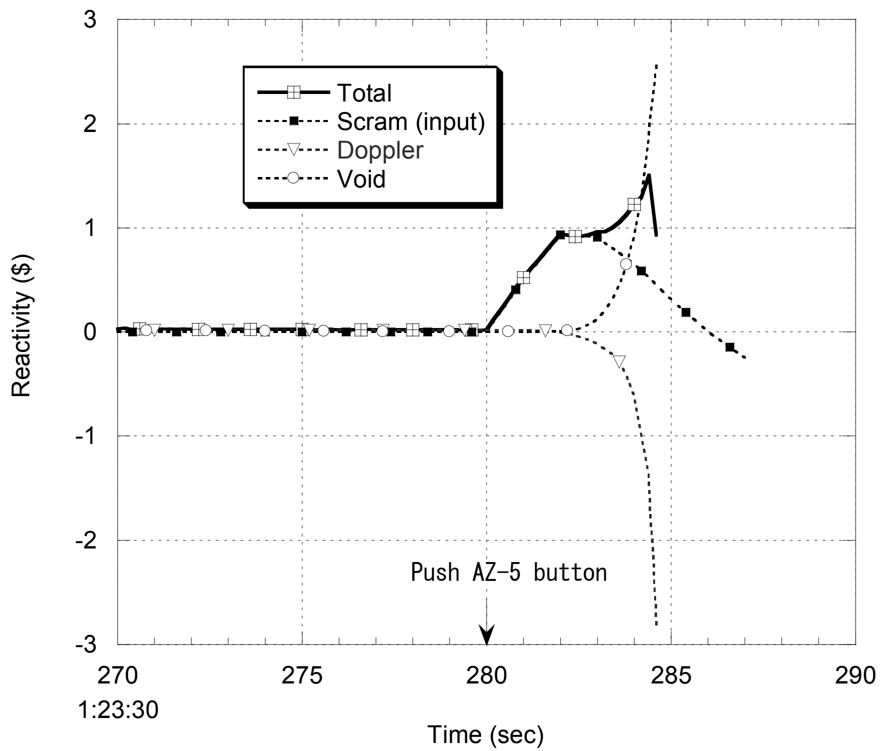


Void

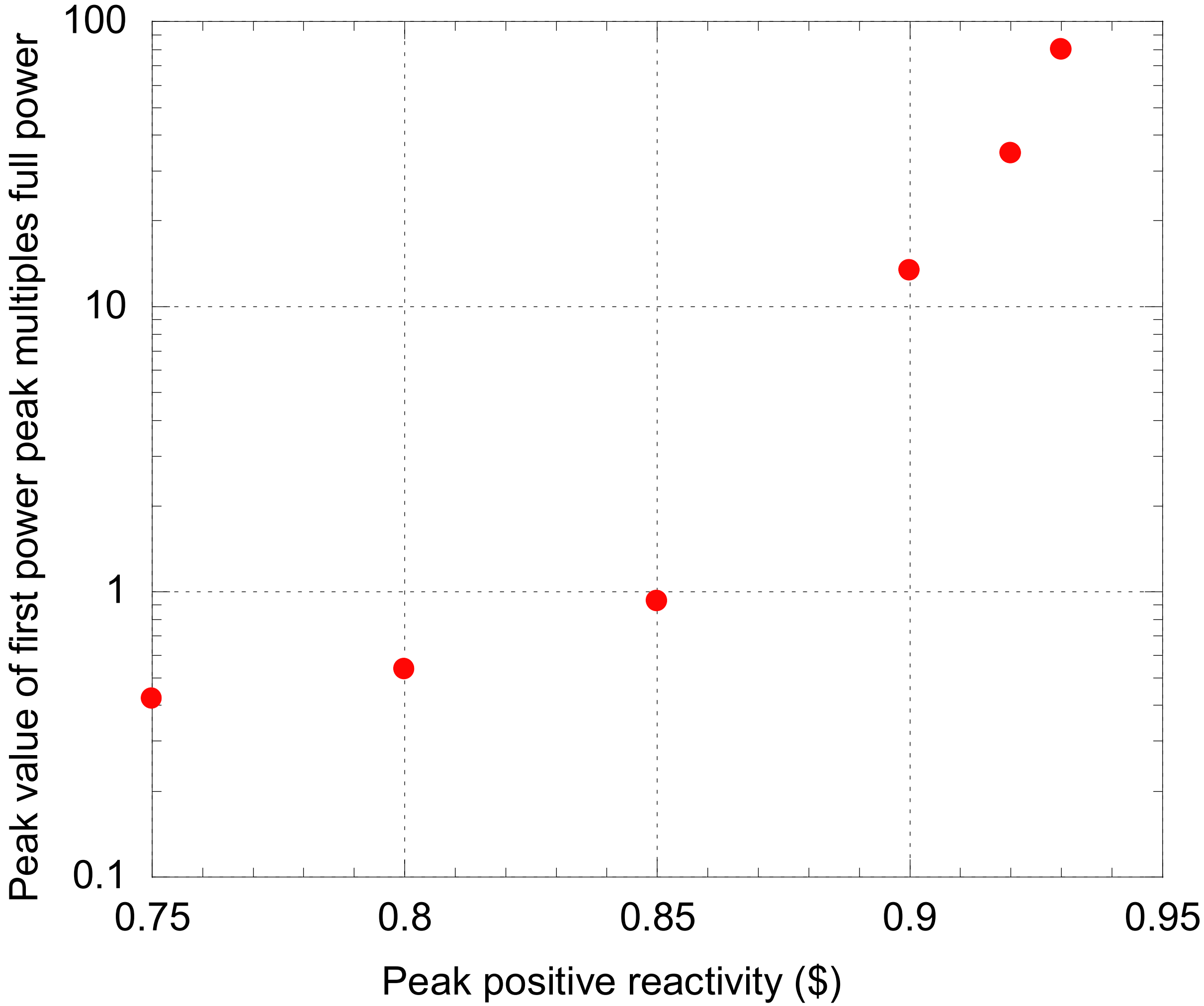


Peak Power and its Reactivity

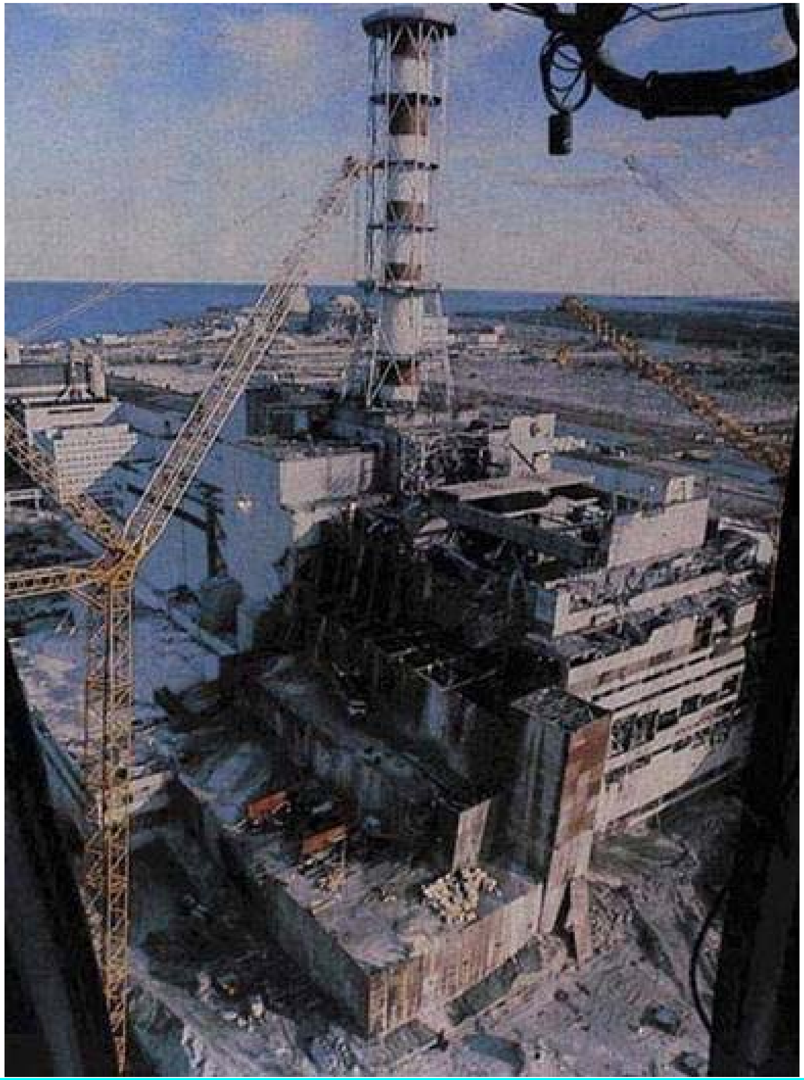




Relationship between Peak Power and Peak Positive Reactivity



Just after the Accident



Control Room and Corium beneath the Core

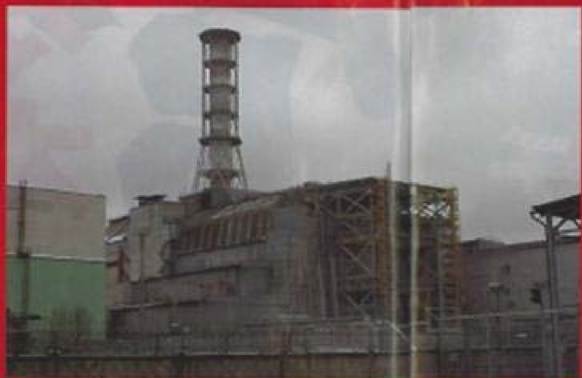
University of Fukui

事故当時、建設中だった冷却水用の冷却塔は、事故で工事が中断、未完成のまま放置されている。ガイドが冷却塔のすぐ近くに放射線測定器を置くと、日本の自然放射線の600倍を超える数値を示した。「4号炉の屋根のアスファルトが、爆発でここまで飛んできた。その残骸がいまも残っている」（ガイド）

爆発した4号炉内部にある炉心の核燃料とコンクリートが溶解してできた、通称「象の足」。事故直後は、人が即死すると言われるほどの放射線を発していた。現在はだいぶ弱まったが、昨年時点で3Sv/hで、1時間浴びると1カ月後の死亡率は50%。［入手写真・撮影日不明］

冷却塔内部はほぼ空洞。機材の残骸が多少あるだけで、床は草に覆われている。すぐそばには、同じく建造途中で放棄された5、6号の2つの原子炉もある。

コンクリートの「石棺」に覆われた4号炉。石棺面の黄色い鉄骨は、屋根の重みで傾いてきた外壁を補強するため、数年前に建設された。石棺から数百メートルの距離で撮影したが、放射線測定器は日本の自然放射線の100倍ほどの数値を表示。「放射線が強いので、撮影は短時間にするように」（ガイド）が、石棺の補強や新しい石棺建設のため、周辺では多くの原発職員や民間の建設作業員が働いている。



❺4号炉を封印した"石棺"の内部。細かい粒子が写っているが、これは「強い放射線の影響でデジカメのセンサーが反応してしまった」（職員）もの。フィルムカメラを使用すると、フィルムが放射線で感光し、全く写真が撮れない状態になるという。［入手写真・07年10月撮影］

❻爆発した4号炉の制御室。事故当時、室内にいた職員も被曝したが、事故時に死亡したのは制御室の外のポンプ室にいた職員1人のみで、遺体は未発見。現在、制御室の制御盤からは機器やスイッチが抜かれ、かろうじて現役時代の面影を残すだけ。［入手写真・07年6月撮影］

原発から半径30キロ以内は立入禁止区域。が、不法に村へ戻ってしまう住人も。ウクライナ政府は96年以降、彼らの居住を認め物資援助などを行っている。バレンティーナさん(75)を訪ねると、近所の森で採れたベリーのジャムや蜂蜜でもてなしてくれた。汚染されていないというが。

4号炉から数キロの街プリピャチの音楽学校ホール。プリピャチでは事故当時、原発職員やその家族約5万人が生活しており、うち1万4000人が子どもだった。ソ連当局が事故の発表を遅らせたため、多くの住民が被曝。最終的には全住民が強制避難させられ、現在は街が丸ごと廃墟となっている。

石棺前の資料展示館。職員のシエクスチロ氏が模型で事故の状況を解説する。事故は、低出力状態で制御が不安定になるという炉の欠陥に加え、低出力実験の際に出力が低くなりすぎるのを防ぐため、全ての制御棒を抜くなどしたヒューマンエラーもあり、複合的な要因で起こったとされる。

❼石棺内部。光は照明ではなく、全て太陽光。石棺は事故直後、被曝の危険で長時間作業ができず、遠隔操作機器も用いて突貫工事で建設された。そこに老朽化が加わり、もはや密閉を保てていない。屋根の隙間からは雨水も浸入、外壁も一部が傾くなど崩落の危険すらあるため、2012年完成を目標に新たな石棺建設計画が進んでいる。［入手写真・撮影日不明］

Before earthquake (14:46) & After Tsunami (15:45) at Fukushima-1 on 11 Mar. 2011

Before the earthquake →

After the Tsunami

Area damaged by the Tsunami

Seawater pumps

Intake of seawater

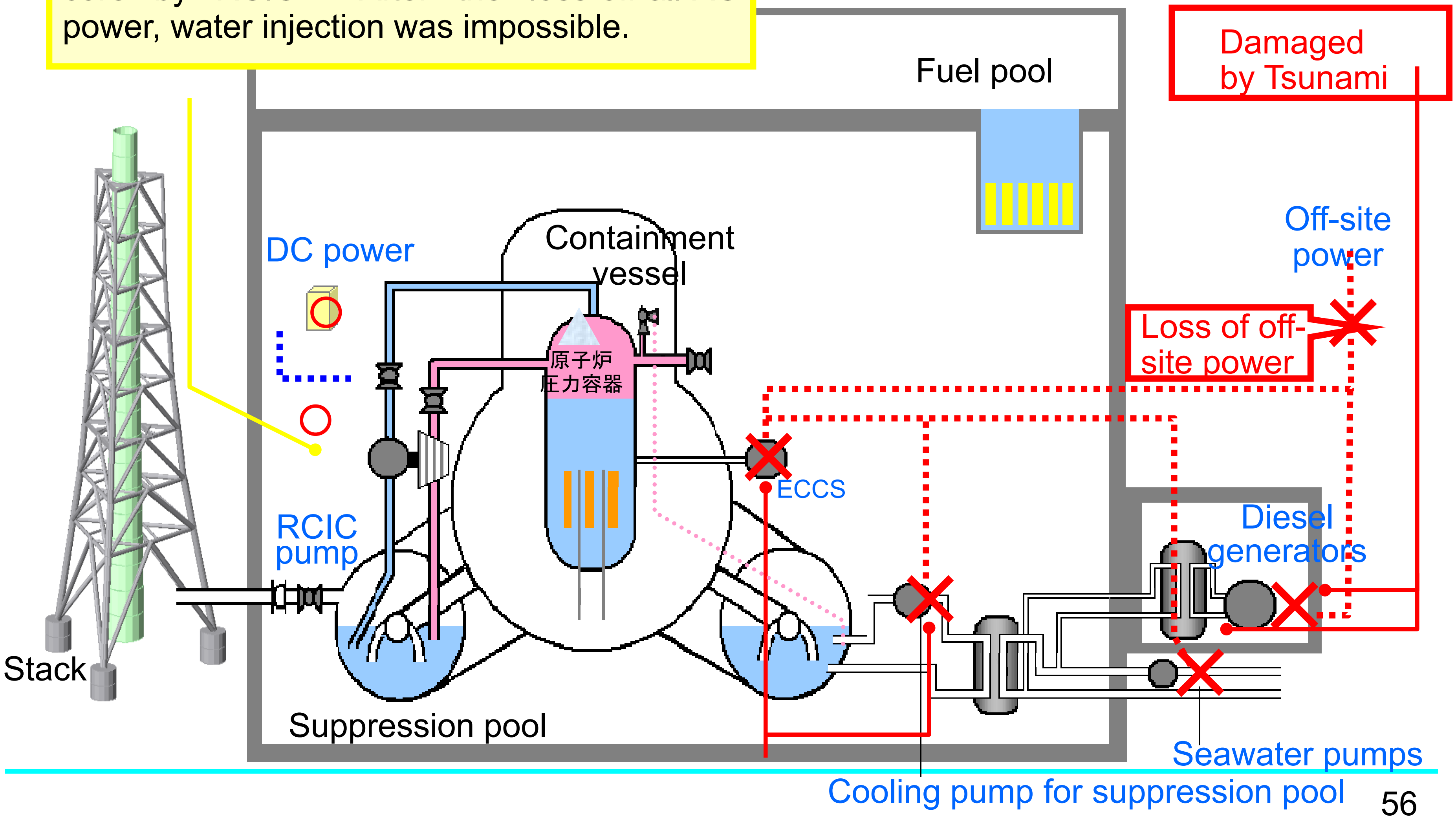
Heavy-oil tanks

出典:http://www.digitalglobe.com/

Severe accident at Fukushima-1

After Tsunami

Operators injected water into the reactor core by RCIC. After the loss-off-all-AC-power, water injection was impossible.



Hand calculation to estimate uncovery

- Assumption: Reactor diameter=4m, water level from top of fuel=4m
- Water inventory ≒ 50t
- Latent heat at 7MPa≒1500kJ/kg
- Initial heat generation rate of Unit-1（460,000kW)
  ≒460,000/0.3 ≒ 1,500,000kW　(70% of heat is released to seawater）
  Decay heat ratio at around 1000 sec.≒ 2%
  Heat generation rate by decay heat≒
  1500000×0.02=30,000kW
- Mass evaporated for 1000 sec. : M(kg)
  M=30,000×1000÷1500
  =20000kg=20t

Real water level and detected water level

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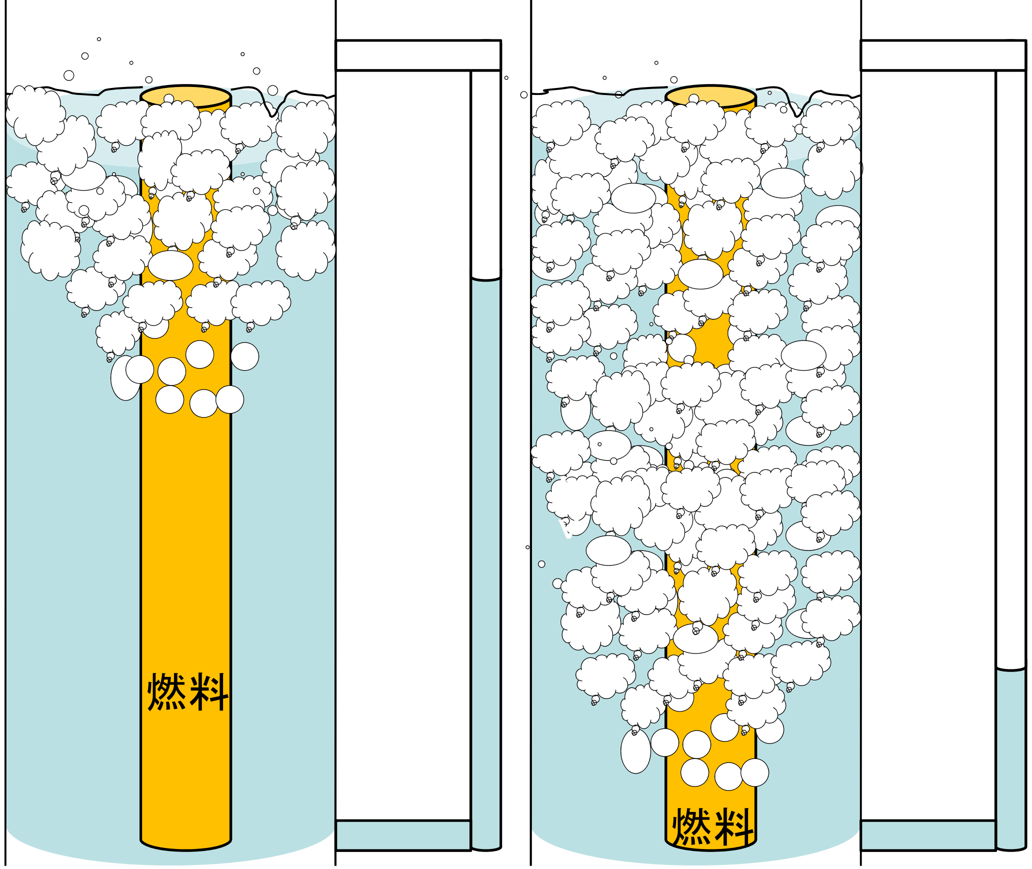
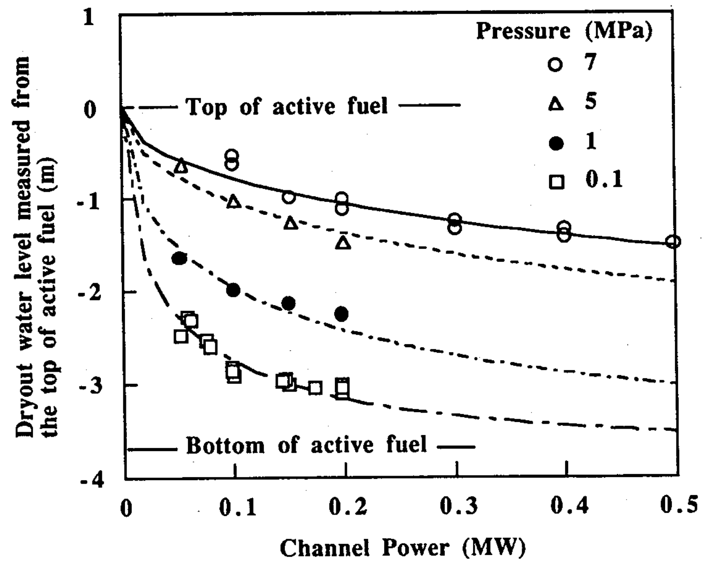
High pressure (Core)
Volume of steam expands 21 times of water at 7 MPa.

Low pressure
(Spent fuel pool)
Volume of steam expands 1600 times of water at 0.1 MPa.

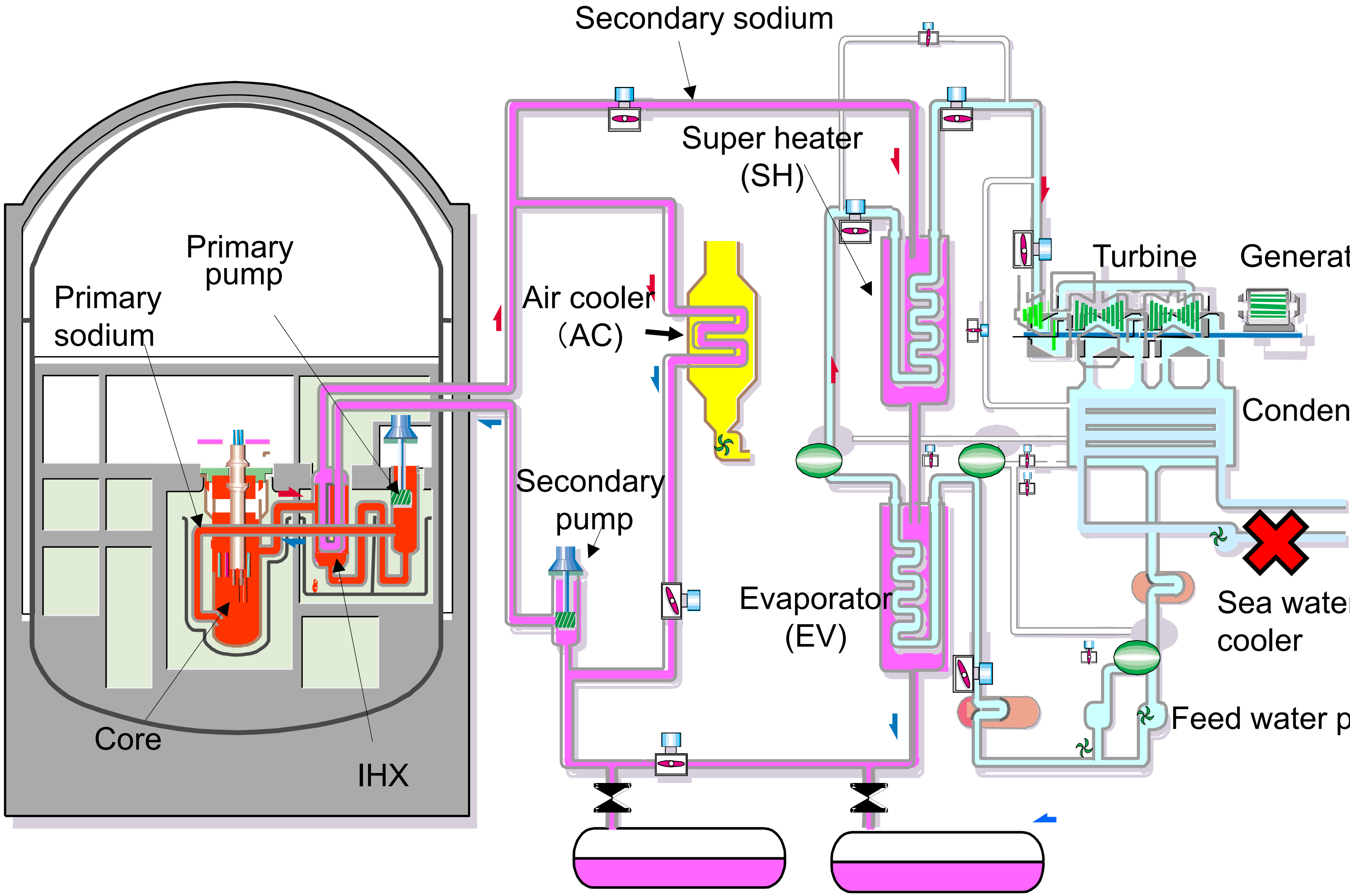


Experiment



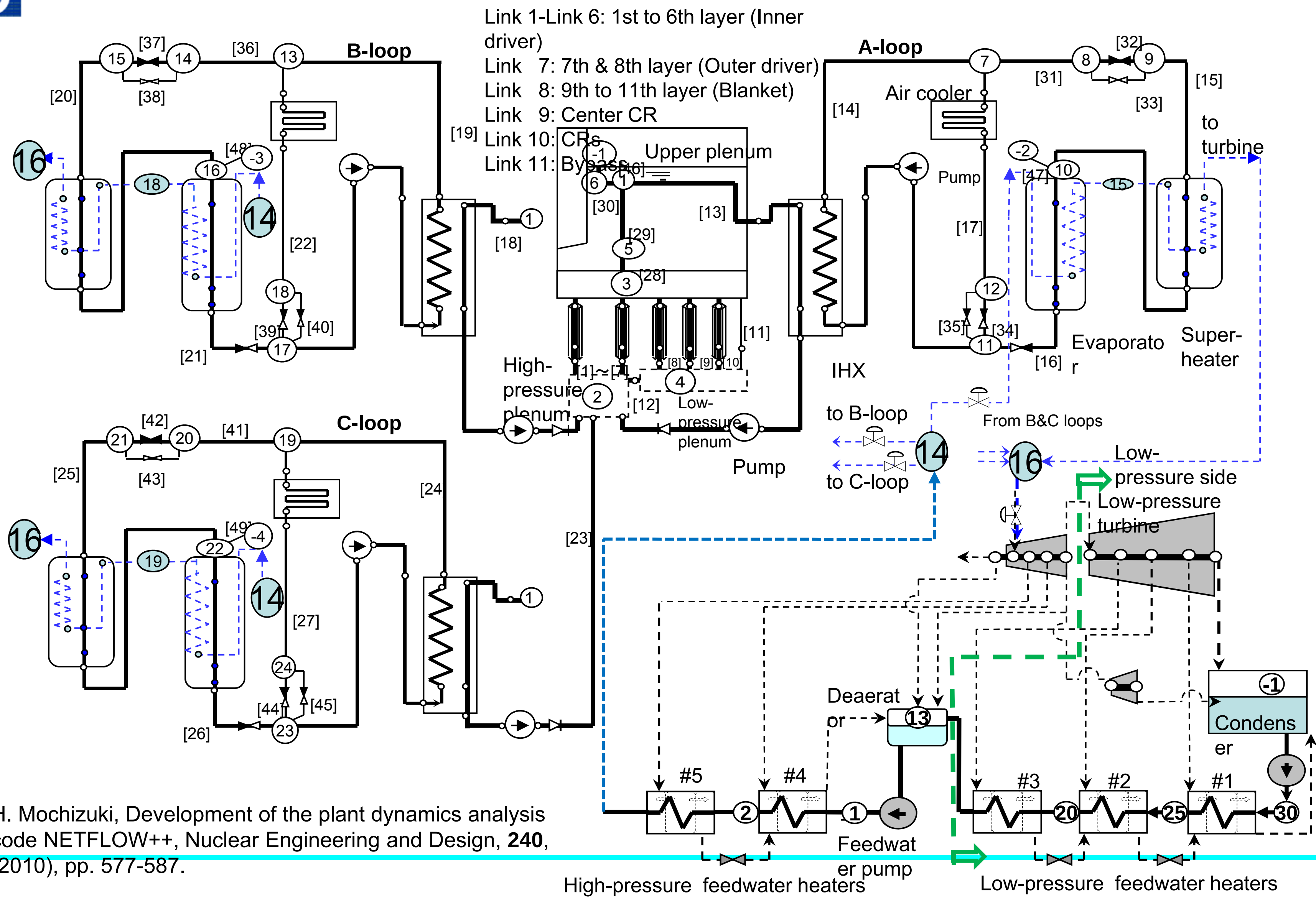
Mochizuki, H. et al., Core coolability of an ATR by heavy water moderator om situation beyo9nd design basis accidents, Nuclear Engineering and Design, 144 (1993), pp293-303.

Natural circulation after the loss of AC power and sea water pump



[1] H. Mochizuki, Plant Behavior of a Fast Breeder Reactor under Loss of AC Power for Long Period, Nuclear Engineering and Design, 245, (2012), pp.19-27.

Analytical Model of Whole Heat Transport Systems



H. Mochizuki, Development of the plant dynamics analysis code NETFLOW++, Nuclear Engineering and Design, **240**, (2010), pp. 577-587.

Station Blackout Event of "Monju"

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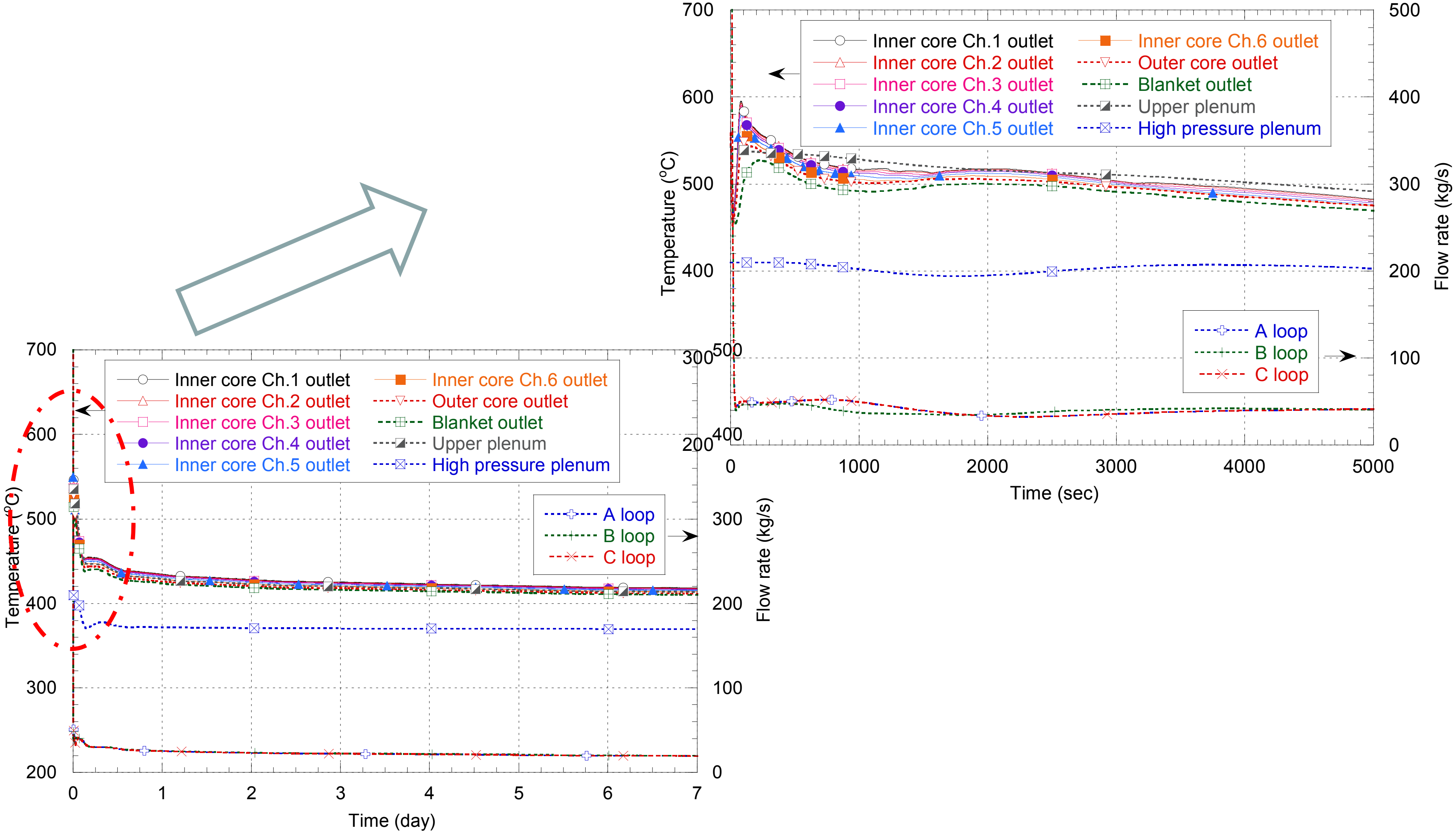

Station Blackout at 1000 sec.

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