Safety of Nuclear Reactors

Professor H. MOCHIZUKI
Nuclear Reactor Thermal Hydraulics Division
Research Institute of Nuclear Engineering
University of Fukui
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.
Main System Diagram of Fugen

- Containment Air Cooling System
- Shield Cooling System
- Reactor Auxiliary Component Cooling Water System
- Reactor Core Isolation Cooling System (RCIC)
- Containment Spray System (APCI)
- High Pressure Coolant Injection System (HPCI)
- Low Pressure Coolant Injection System (LPCI)
- Residual Heat Removal System (RHR)
- Control Rod Drive
- Relief valve
- Dump valve
- Feed Water System
- Bypass valve
- Condensate Tank
- Heavy Water Cooling System
- Turbine
- Sea water
- Reactor Auxiliary Component Cooling Sea-Water System
Blow-down experiment
6MW ATR Safety Experimental Facility

Outlet pipes
(74mmID, 10.5mL, 2 deg.)

EL 8.45
Shield plug
EL 7.0
EL 5.97

High power heater
3.7mL, 6MW

EL 2.27

Inlet pipes
(62.3mmID, 12mL) EL0.9

Low power heaters
3.7mL, 200kW

EL 2.3

Turbine flow meter

Check valves

Water drum
(387mmID, 3.9mL)

EL 1.64

Main steam isolation valve

Steam drum
(1525mmID, 4.46mL)

Downcomer
(275.7mmID, 6.8mL)

Connecting pipe
(186mmID, 24.6mL)

EL 0.4

Pump

Check valves

EL 9.7

P,T

Pressure transducer

P,T

Thermocouple

EL: Elevation in m
ID: Inner diameter
L: Length from a component to the next arrow.

Water drum
(387mmID, 3.9mL)

Inlet pipes
(62.3mmID, 12mL) EL0.9

High power heater
3.7mL, 6MW

EL 2.27

Ground wire

Main steam isolation valve

Steam drum
(1525mmID, 4.46mL)

Downcomer
(275.7mmID, 6.8mL)

Connecting pipe
(186mmID, 24.6mL)

Pump

Check valves

EL 9.7

P,T

Pressure transducer

P,T

Thermocouple

EL: Elevation in m
ID: Inner diameter
L: Length from a component to the next arrow.
Water level behavior after a main steam pipe break

Drum water level increase during downcomer water level decrease

Device oscillation due to break

100 mm break at main steam pipe
Simulated fuel bundle

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

Unit in mm

Cross sectional view of 36-heater bundle

Power distribution of 36-rod high power heater

Location of maximum axial peaking

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

Downcomer 100 mm break

Calculation
Experiment

Scram
ECCS operation
Improvement of blow-down analysis by applying statistical method

Application of stochastic method to FBR analysis

Application of stochastic method to FBR

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

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.

Severe accident

Photo. Erosion experiment of Zr-2.5%Nb pressure tube by molten metal
Heat transfer of melted fuel to material

(1) $t = 0.27\, \text{sec}$

(2) $t = 10.74\, \text{sec}$

(3) $t = 14.77\, \text{sec}$

(a) Top view

(b) A-A cross section

(Upper plate; $Z = 50\, \text{mm}$)
Heat transfer between melted jet and materials

Comparison of Nusselt number between present data and data from Saito et al.1) and Mochizuki2).
Fuel melt experiment using BTF in Canada
Fuel melt experiment using CABRI
# Source term analysis codes

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

Containment spray

In case of containment bypass

Containment recirculation system

Stack

Filter Blower

Water flow

Gas flow

Steam release pool

(13) Air

(11) Containment spray

(12) Annulus

(10)

(9)

(8)

(7)

(6)

(5)

(4)

(3)

(2)

(1)
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 °C  
**Outlet coolant temp.**: 284 °C  
**Fuel**: 1.8%UO₂  
**Number of fuel channels**: 1,693
Above the Core of Ignarina NPP
Core and Re-fueling Machine
Control Room
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 Channel

- Effective core region
- S.S. (Stainless Steel)
- Diffusion welding
- Zr-2.5%Nb (Zirconium-2.5% Niobium)
- Electron beam welding (EBW)
- Roll region
- Spacers
- Fuel assembly
- Connecting rod
- Position: -0.018m
- Materials: S.S. (80), S.S. (72), Zr-2.5%Nb
- Welding: EBW, Diffusion
Heat Removal by Moderation

Heat generated in graphite blocks is removed by coolant.

Maximum graphite temperature is 720°C at rated power.
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://insep.ansc.gov/2000/)
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

Thermal Power (MW)

- 3500
- 3000
- 2500
- 2000
- 1500
- 1000
- 500
- 0

Scheduled power level for experiment

Power excursion

20-30% of rated power

- 30MW
- 200MW

Time (sec/min/hour/day)
Time Chart Presented by USSR

- Recirculation flowrate
- Feedwater flowrate
- DS pressure
- Power
- Safety rod insertion
Result in the Past Analysis (1/2)


- Requirement from the Nuclear Safety Committee in Japan

![Graph showing various parameters over time: Feed water, Water level, Drum pressure, Recirculation flow rate, Neutron flux.](Image)
Result in the Past Analysis  (2/2)

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

Power at 200 MW

Power just before the accident was twice as large as the report. Why???

Power at 48,000 MW

Timing of peak was different. Why???

Information Reported by the USSR

Calculated by EUREKA-2
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

• Positive scram

P.S.W. Chan and A.R. Daster

Andriushchenko, N.N. et al.,

Fig. 1. Effect of node size on axial flux distribution. In this figure, B = B₄C absorber and D = graphite displacer.
Trigger of the Accident (cont.)

- Graphite displacer
- Scram rod (24 rods) inserted by AZ-5 button
- Negative reactivity
- Positive reactivity
- Graphite block
- Water Column

Fuel 2 × 3.5m

Dimensions:
- 8.0
- 1.0
- 5.0
- 1.5
Simulation from 1:19:00 to First Peak

Data acquired by SKALA

- Flowrate (m³/s)
- P (MPa)
- Flowrate (calc.)
- Pressure (calc.)
- DS water level (mm)
- Feed water flowrate (kg/s)
- Reactor power (calc.)
- DS water level (calc.)

Flowrate (m³/s), Pressure (MPa)

DS water level (mm), Feedwater flowrate (kg/s)

Time (sec)

Close stop valve: Turbine trip

Push AZ-5 button

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

- Thermal equilibrium steam quality, $x$ (-)
  - Top
  - Center
- Time (sec)
- Push AZ-5 button
- Two-phase
- Water
- Turbine trip
- RCP trip

University of Fukui
Void Characteristic

Pressure 7MPa

\[ \Delta \alpha \] Void fraction increase

\[ \Delta \alpha_1 \] Void fraction increase

\[ \Delta \alpha_2 \] Void fraction increase
Nuclear Characteristics

Doppler

\[ \frac{\Delta k}{k} / ^\circ\text{C} \]

0 500 1000 1500 2000 2500

T (°C)

\[ -1.8 \times 10^{-5} \]

\[ -1.6 \times 10^{-5} \]

\[ -1.4 \times 10^{-5} \]

\[ -1.2 \times 10^{-5} \]

\[ -1.0 \times 10^{-5} \]

\[ -8 \times 10^{-6} \]

Void

\[ \frac{\Delta k}{k} / \text{Void} \]

0 2 4 6 8 10 20 40 60 80 100

Void fraction (%)

\[ 0.0001 \]

\[ 0.0002 \]

\[ 0.0003 \]

\[ 0.0004 \]

\[ 0.0005 \]

Doppler Void
Peak Power and its Reactivity

Power reported by USSR (MW)

-3 -2 -1 0 1 2 Time (sec)

Power (MW)

0 5 10^4 1.5 10^5 2 10^5 2.5 10^5 3 10^5 3.5 10^5

270 275 280 285 290

Total

Scram (input)

Doppler

Void

Reactivity ($

0 1 2 3

-1 -2 -3

270 275 280 285 290

Time (sec)

1:23:30

Push AZ-5 button
Relationship between Peak Power and Peak Positive Reactivity

![Graph showing the relationship between peak positive reactivity and peak power values.](image-url)

- Peak positive reactivity ($$\gamma$$) is plotted against the peak value of first power peak multiples full power.
- The graph displays a trend indicating a positive correlation between the two variables.
Just after the Accident
Control Room and Corium beneath the Core
Before earthquake (14:46) & After Tsunami (15:45) at Fukushima-1 on 11 Mar. 2011

Before the earthquake

Area damaged by the Tsunami

After the Tsunami

Seawater pumps
Intake of seawater
Heavy-oil tanks

出典：http://www.digitalglobe.com/
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

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.

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

Analytical Model of Whole Heat Transport Systems

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: CR
Link 11: Bypass

Station Blackout Event of “Monju”
Station Blackout at 1000 sec.