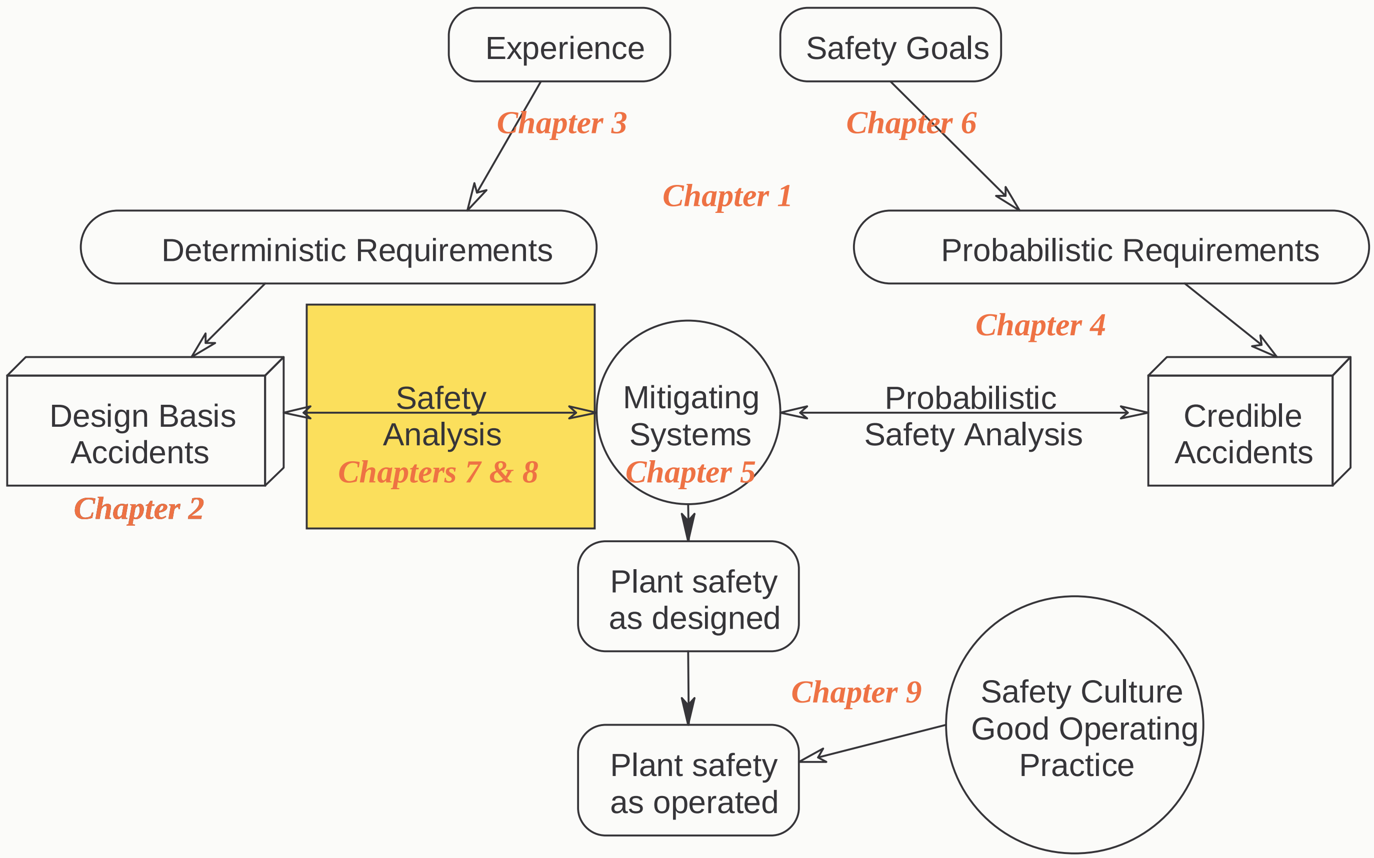
Lecture 9 - Accident Analysis

Dr. V.G. Snell

Nuclear Reactor Safety Course

McMaster University

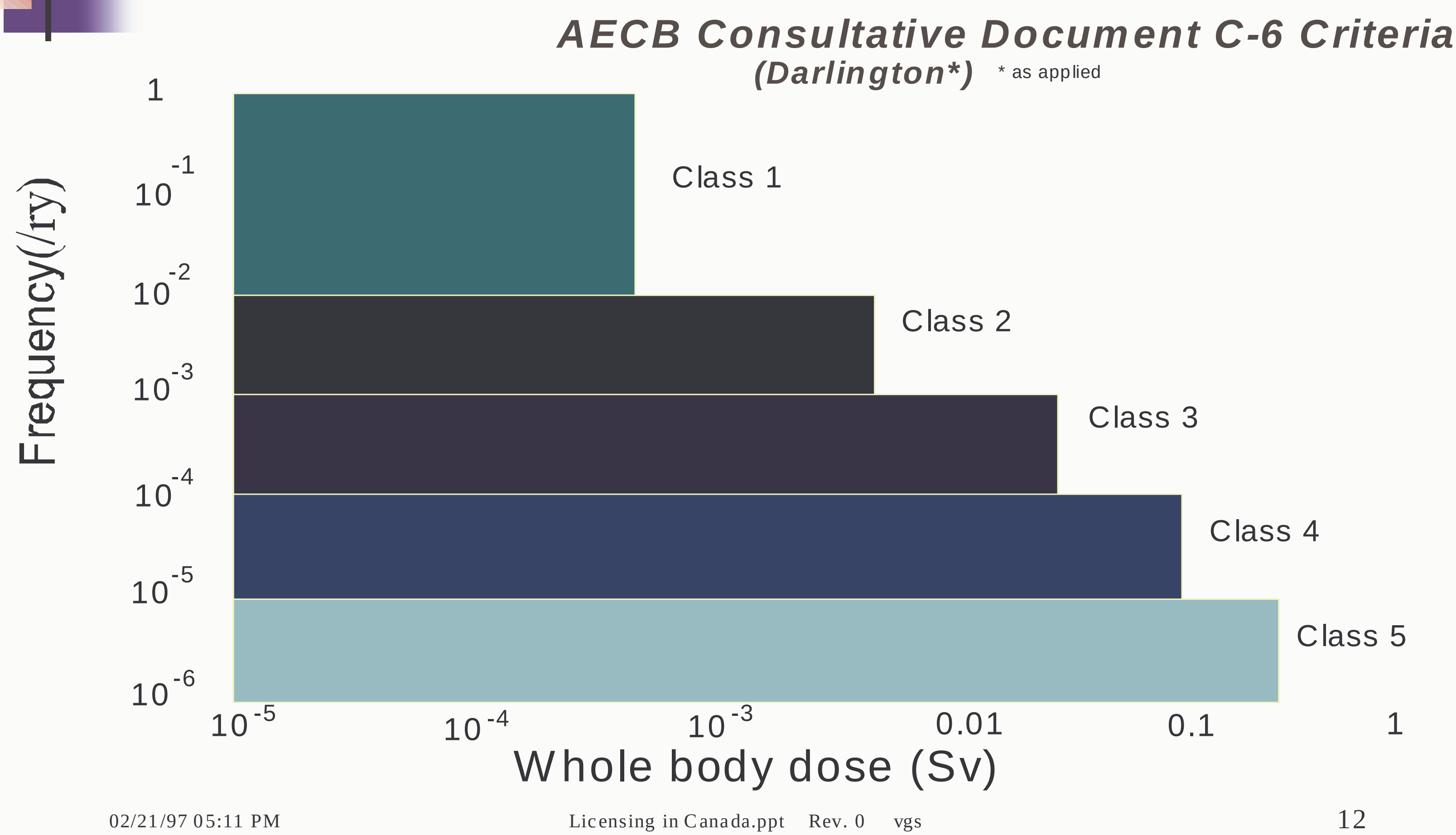
Where We Are


Experience
Safety Goals
Chapter 3
Chapter 6
Chapter 1
Deterministic Requirements
Probabilistic Requirements
Chapter 4
Design Basis Accidents
Chapter 2
Safety Analysis
Chapters 7 & 8
Mitigating Systems
Chapter 5
Probabilistic Safety Analysis
Credible Accidents
Plant safety as designed
Chapter 9
Safety Culture Good Operating Practice
Plant safety as operated

Objective

- How are data & assumptions chosen?
- How do we ensure that the answer is pessimistic?
  - Is this good?
- Details specific to CANDU – methodology is general
- "Think Negatively"!

Selection of Events by Pseudo-Frequency



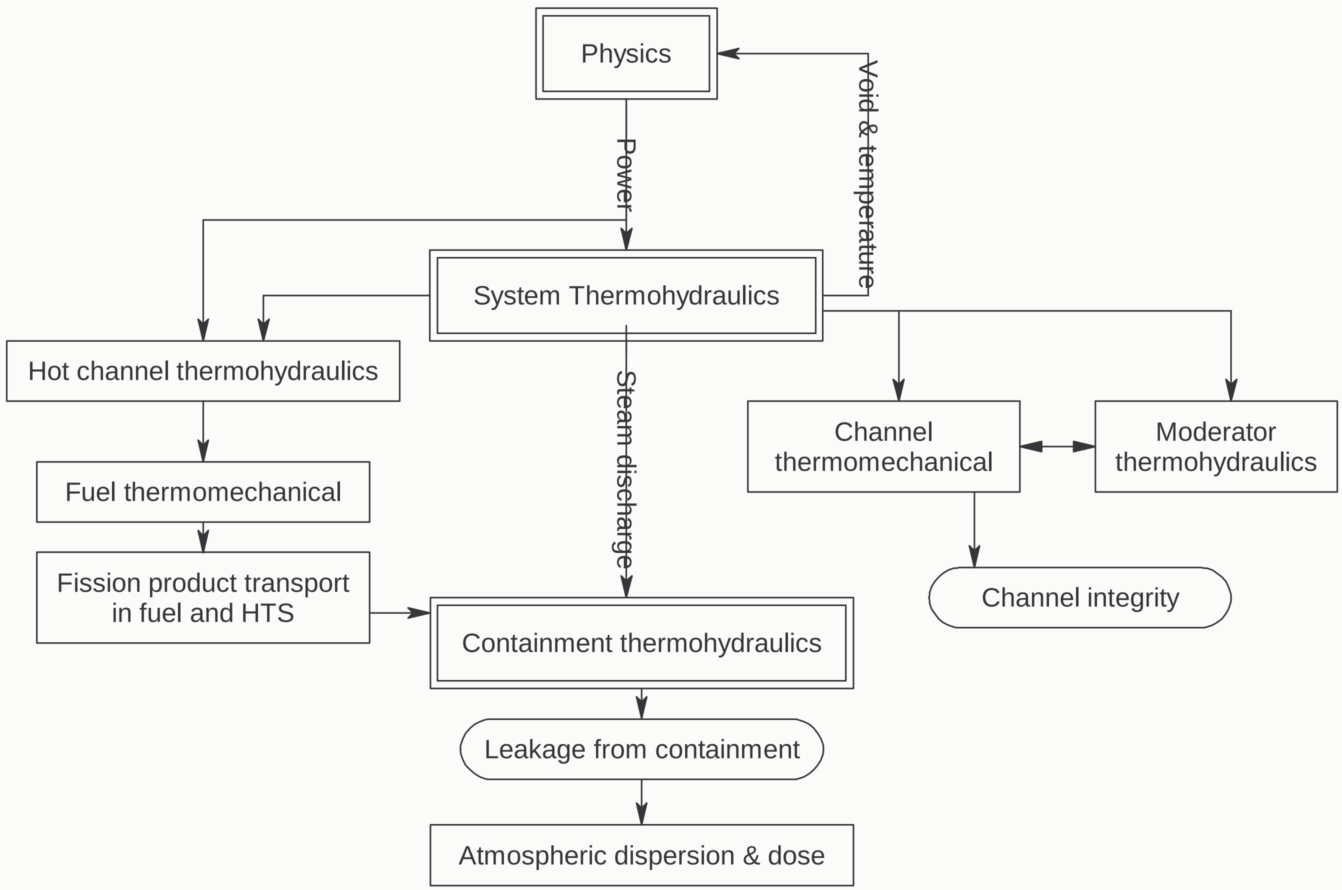
Selection of Events by Phenomena - 1

1. **Reactivity Accidents**
   - Bulk Loss of Reactivity Control
   - Loss of Reactivity Control from Distorted Flux Shapes
   - Inadvertent Criticality
2. **Decrease of Reactor Coolant Flow**
   - Loss of Class IV Power
   - Partial Loss of Class IV Power
   - Single Pump Trip or Seizure
3. **Increase of Reactor Coolant Pressure**
   - Loss of Primary Pressure and Inventory Control (increase)
4. **Decrease of Reactor Coolant Inventory**
   - Large Heat Transport System LOCA
   - Small Heat Transport System LOCA
     - Single Channel Events
     - Single Steam Generator Tube Rupture
     - Multiple Steam Generator Tube Rupture
   - Loss of Primary Pressure and Inventory Control (decrease)

Selection of Events by Phenomena - 2

5. **Increase of Secondary Side Pressure**
   - Loss of Secondary Side Pressure Control (increase)
6. **Loss of Secondary Side Heat Removal**
   - Main Steam Line Break
   - Feedwater Line Break
   - Loss of Feedwater Pumps
   - Spurious Closure of Feedwater Valves
   - Loss of Secondary Side Pressure Control (decrease)
   - Loss of Shutdown Heat Sink
7. **Moderator & Shield Cooling System Failures**
   - Pipe Break
   - Loss of Forced Circulation
   - Loss of Heat Removal
8. **Fuel Handling Accidents**
   - Fuelling Machine On-Reactor
   - Fuelling Machine Off-Reactor

Accident Analysis Flow Chart



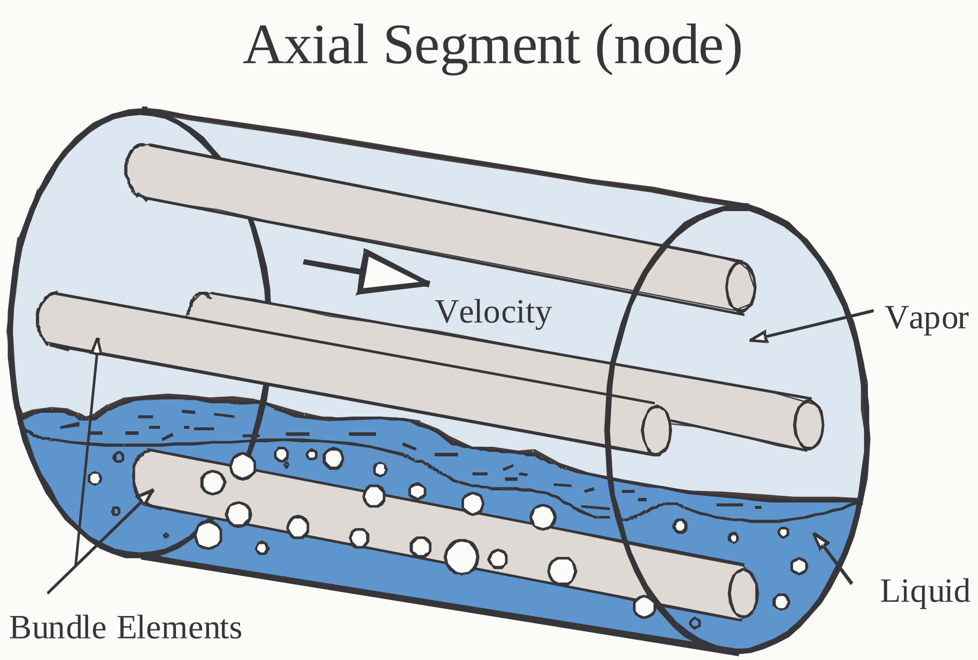
Models - 1

- Reactor physics
  - Transient 3D
- System thermohydraulics
  - Transient 2- or 3-fluid, 1D, non-equilibrium, network
- Fuel thermo-mechanical
  - Initial - strain, fuel-to-sheath heat transfer coefficient, fission gas release, temperatures
  - Transient - fuel sheath strain, beryllium braze penetration, sheath embrittlement, athermal strain, and excessive fuel energy content

Acceptance Criteria

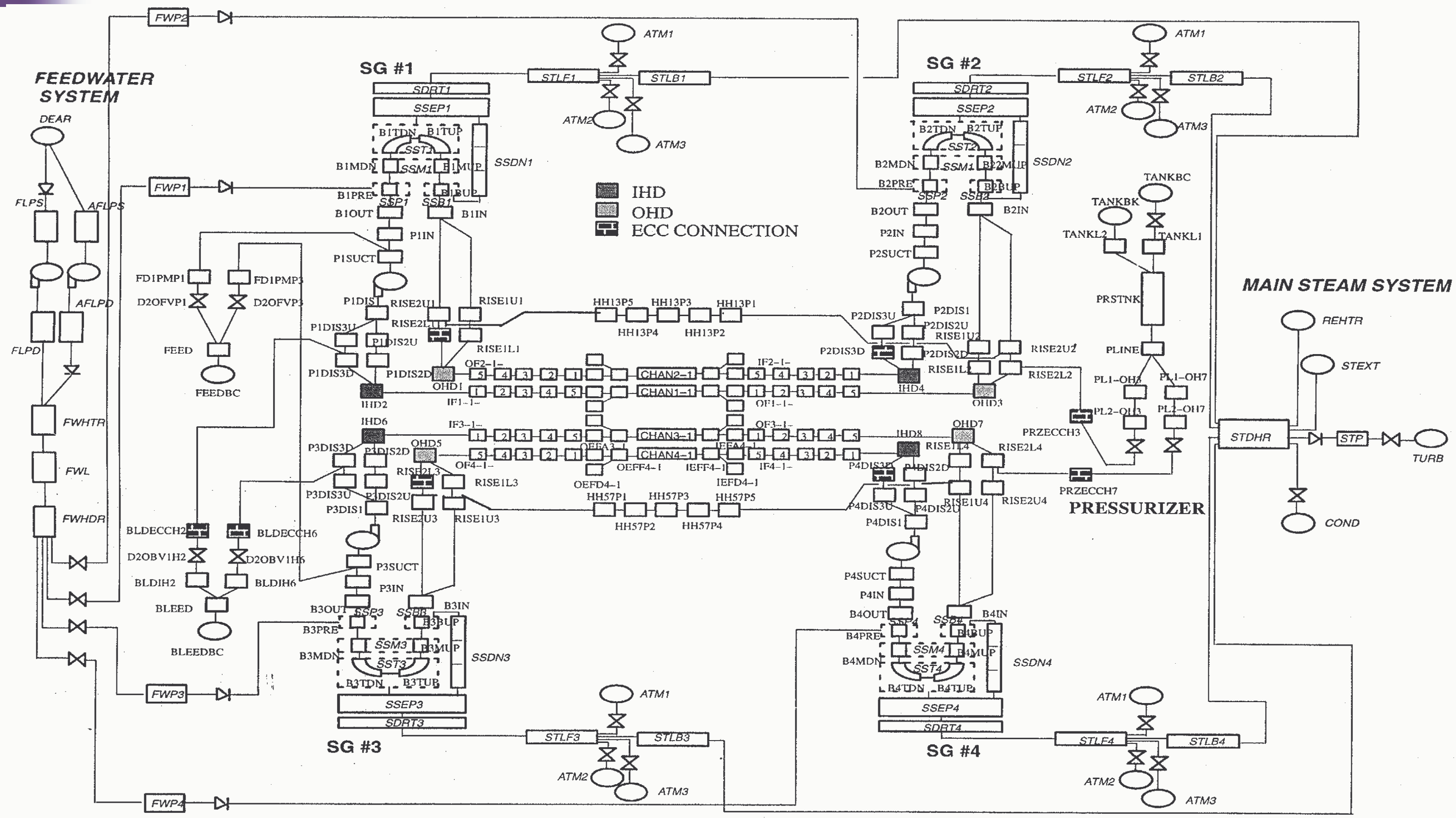
- Public dose as per siting guide or C-6 or RD-337
- Designers choose secondary targets
- Shutdown is special:
  - *Each* shutdown system must be independently effective
  - *Two* diverse trips on each system for each accident for operating plants
    - RD-337 drops this requirement for direct trips

Thermohydraulic Model



- **Non-equilibrium model**
  - **2-velocities,**
  - **2-temperatures**
  - **2-pressures**
  - **plus noncondensables**
- **Flow regime dependent constitutive relations couple two-phase model**
- **Interfaces to other codes:**
  - **Fuel Behaviour**
  - **Plant Control**
  - **Physics**

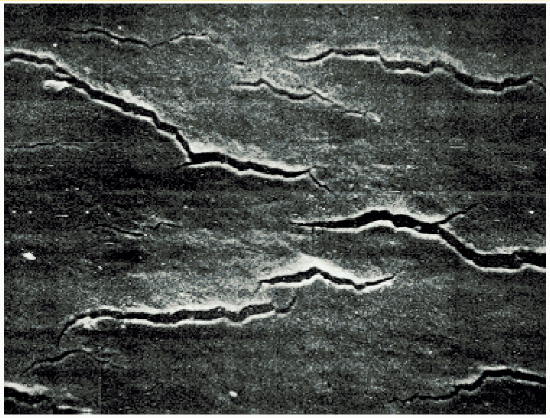
Typical HTS Nodalization



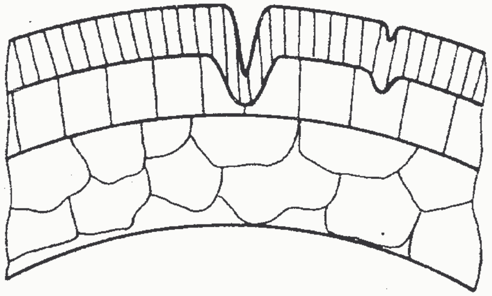
Fuel Failure Mechanisms

- no excessive straining- 5% strain less than 1000°C

- no-oxide cracking- 2% strain greater than 1000°C



- no beryllium-braze penetration
- no oxygen embrittlement
- no fuel melting

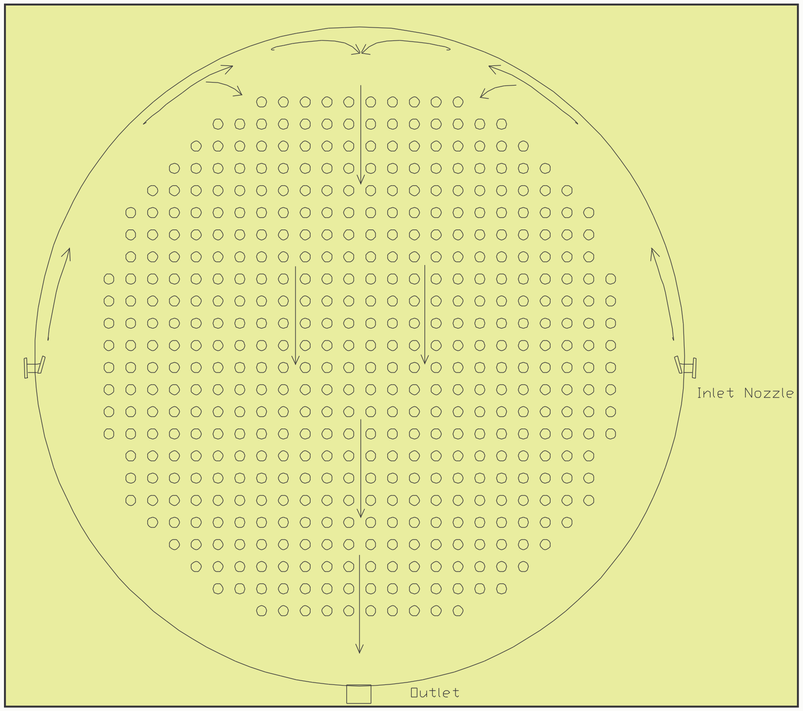


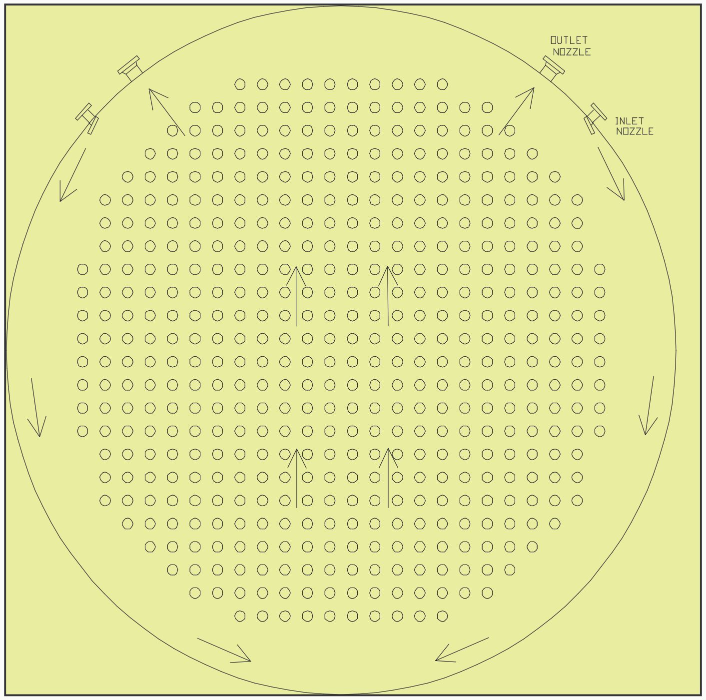
Models - 2

- Pressure-tube thermo-mechanical
- Moderator temperature & flow
  - 3D, steady state & transient
- Fission product transport
  - Within HTS; at break; within containment
- Containment thermohydraulics
  - 1D & 3D multi-fluid transient
- Atmospheric dispersion and dose
  - Gaussian plume; ICRP-60

Moderator

Existing Reactors | **ACR**

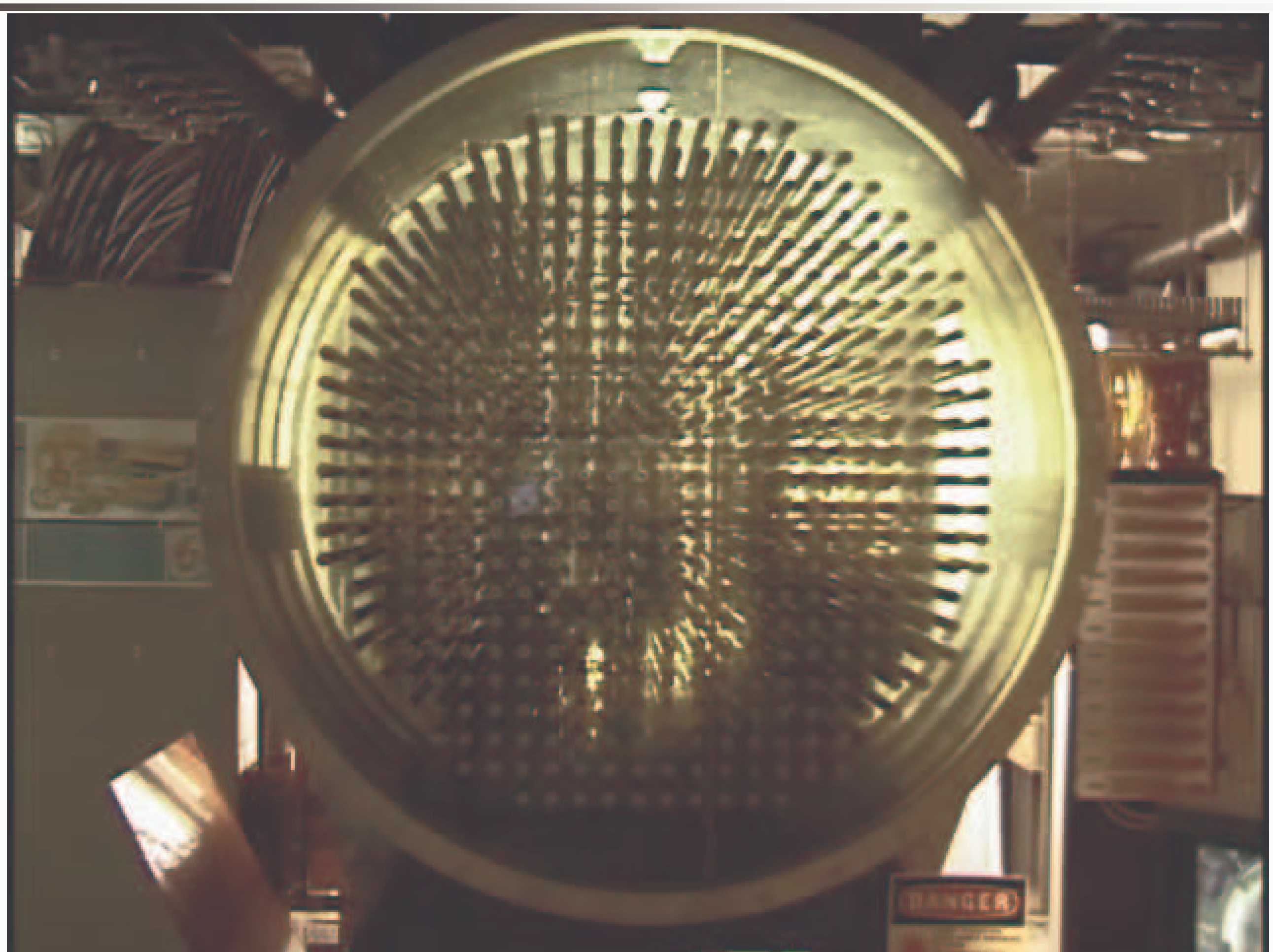




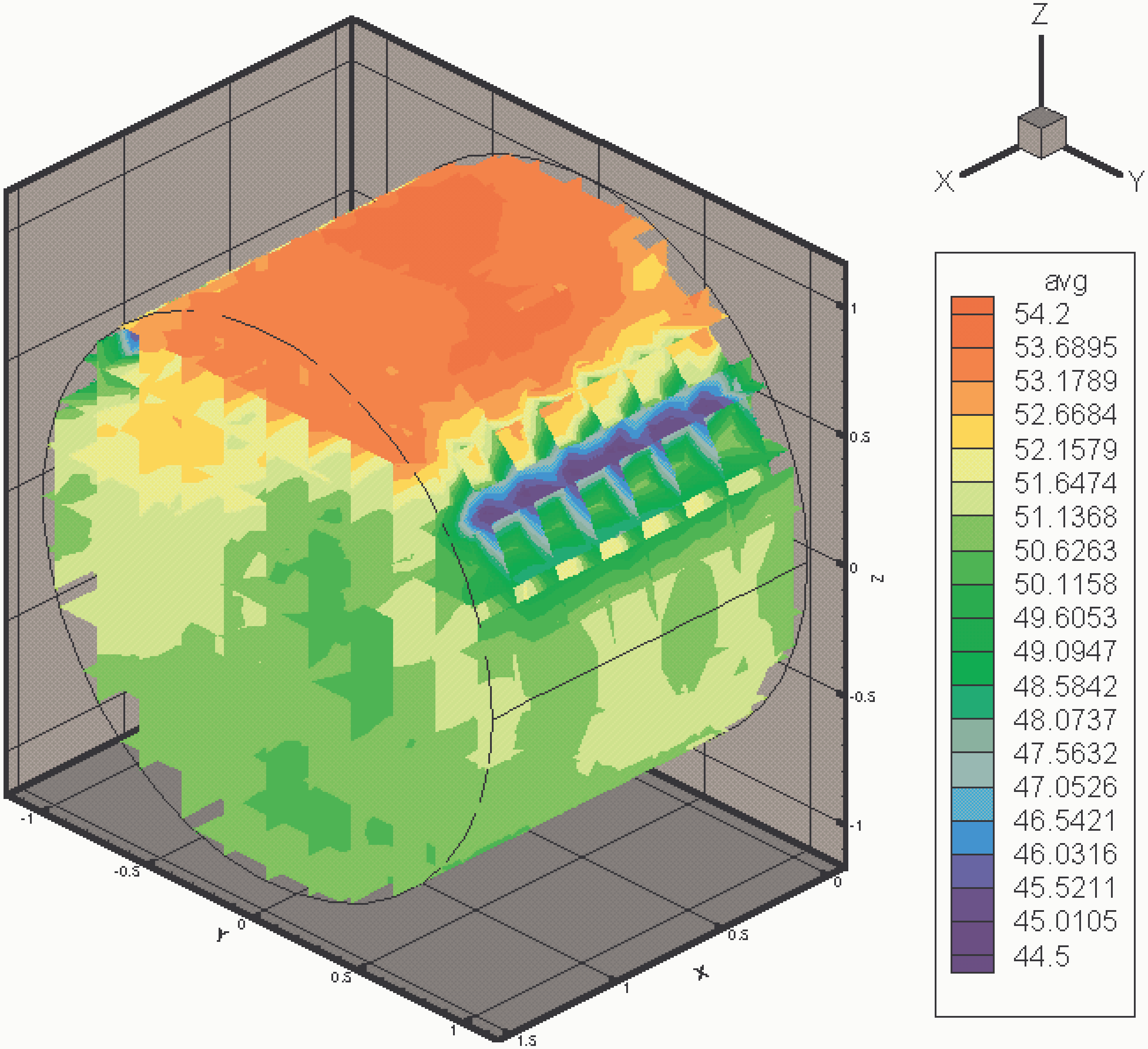
Momentum & buoyancy forces opposing – higher temperatures

Momentum & buoyancy forces same direction – lower temperatures

Moderator Flow - ACR



Moderator



Z
X
Y
avg
54.2
53.6895
53.1789
52.6684
52.1579
51.6474
51.1368
50.6263
50.1158
49.6053
49.0947
48.5842
48.0737
47.5632
47.0526
46.5421
46.0316
45.5211
45.0105
44.5

Initial Conditions - 1

Parameter	Conservative Direction	Rationale
Reactor thermal power	High	Minimize time to use up cooling water inventory, minimize margins to critical heat flux, etc.
Reactor regulating system	Normal operation or inactive, whichever is worse; setback is generally not credited unless it tends to "blind" the trip	Choose so as to delay reactor trip
Radionuclide operating load in the HTS	Highest permissible operating iodine burden (and associated noble gases) and end-of-life tritium concentration	Maximize radionuclide release from station and public dose

Initial Conditions - 2

Steam generators	Clean & fouled cases	Reduce reactor trip effectiveness
Steam generator tube leak rate	Maximum permitted during operation, plus assessment of any consequential effects due to the accident	Increase radioactivity release
Pressure tube creep	Largest value expected	Reduce margins to critical heat flux and increase void reactivity
HTS flow	Low	Reduce margins to critical heat flux
HTS Instrumented channel flow	High	Reduce low flow trip effectiveness

Initial Conditions - 3

Coolant void reactivity coefficient	High;	Maximize overpower transient;
	Low	Delay HTS high pressure trip
Fuel loading	Equilibrium;	Maximize fuel temperatures, radioactivity releases;
	Fresh	Maximize overpower transient
Shutdown system	Backup trip on less effective shutdown system using the last of three instrumentation channels to trip	Delay shutdown system effectiveness
SDS2 injection nozzles	Most effective nozzle unavailable	Reduce shutdown system reactivity depth

Initial Conditions - 4

SDS1 shutoff rods	Two most effective rods unavailable	Reduce shutdown system reactivity ‘bite’ and depth
Maximum channel/bundle power	High	Maximize fuel & sheath temperature
Reactor decay power	High	Minimize time to use up cooling water inventory
Initial flux tilt	High	Maximize fuel & sheath temperature
Moderator initial local maximum subcooling	Low	Minimize margin to critical heat flux on calandria tube

Initial Conditions - 5

Number of operating containment air coolers and other heat sinks	Low;	Maximize containment pressure;
	High	Delay high pressure trip and maximize likelihood of hydrogen combustion
Number of containment dousing spray headers	Low (typically 4 out of 6);	Maximize short-term containment pressure;
	High	Maximize long-term containment pressure and leak-rate, maximize likelihood of long-term hydrogen combustion
Containment leak rate	High (typically 2x to 10x design leak rate);	Maximize public dose;
	Low	Maximize containment pressure

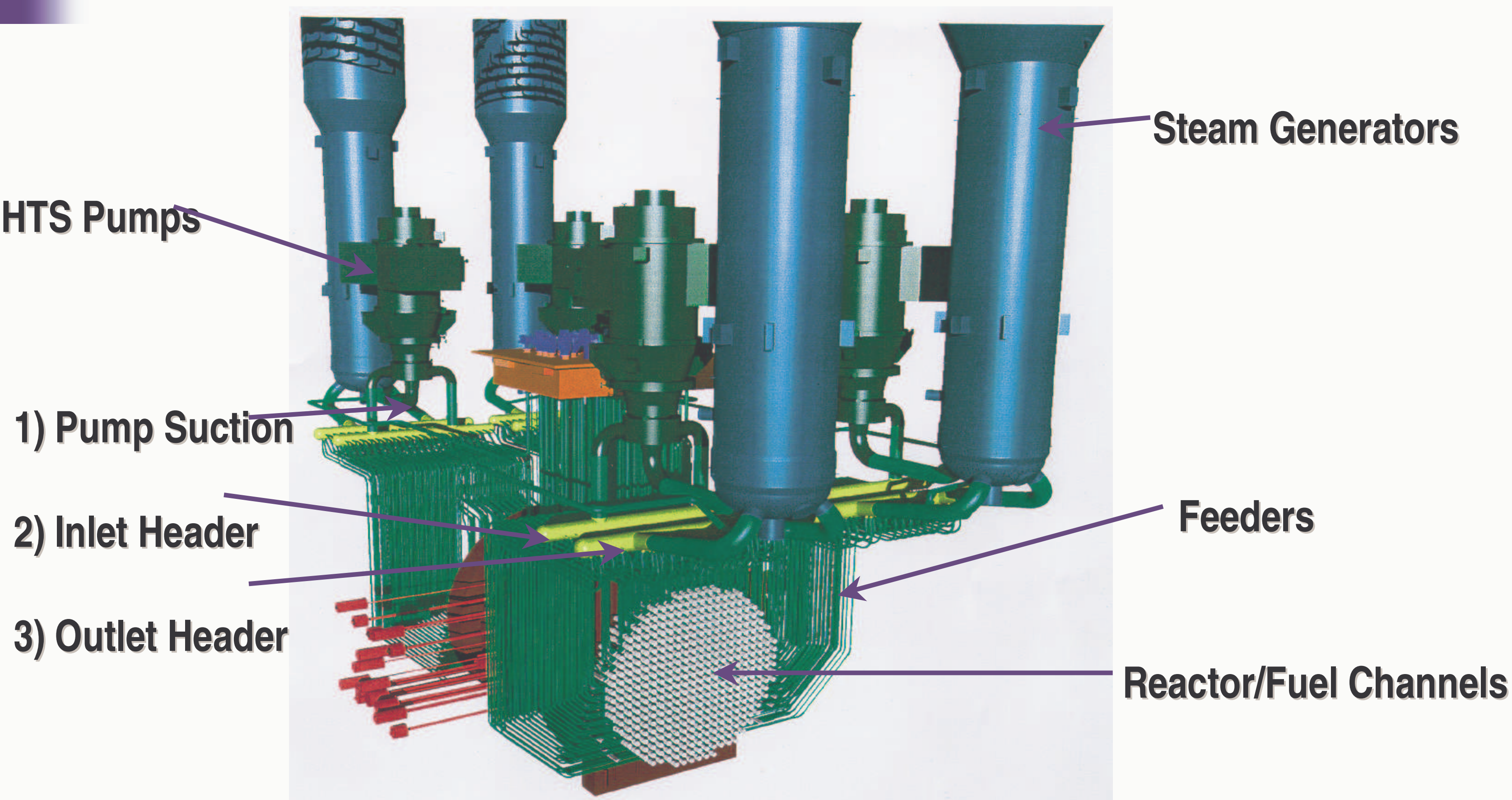
Initial Conditions - 6

Containment bypass leakage	Pre-existing steam generator tube leak	Maximize public dose
Weather	Least dispersive weather occurring >10% of the time	Maximize public dose
Operator Actions	Not credited before 15 minutes after a clear indication of the event, for actions that can be done from the control room; and not credited before 30 minutes, for actions that must be done “in the field”	Ensure adequate time for diagnosis

Large LOCA – Initiating Event

- Instantaneous break up to 2X area of largest pipe
- Large pipes all above core
- RIH, ROH, PSH

Break Locations



Sequence of Events - 1

- Large break occurs discharging steam to containment
- Coolant voids, reactivity increases
- Reactor power increases
- Reactor is shutdown on a neutronic trip
- HTS flow decreases fastest in the core pass downstream of the break
- Power pulse & fuel dryout result in an increase in fuel temperature

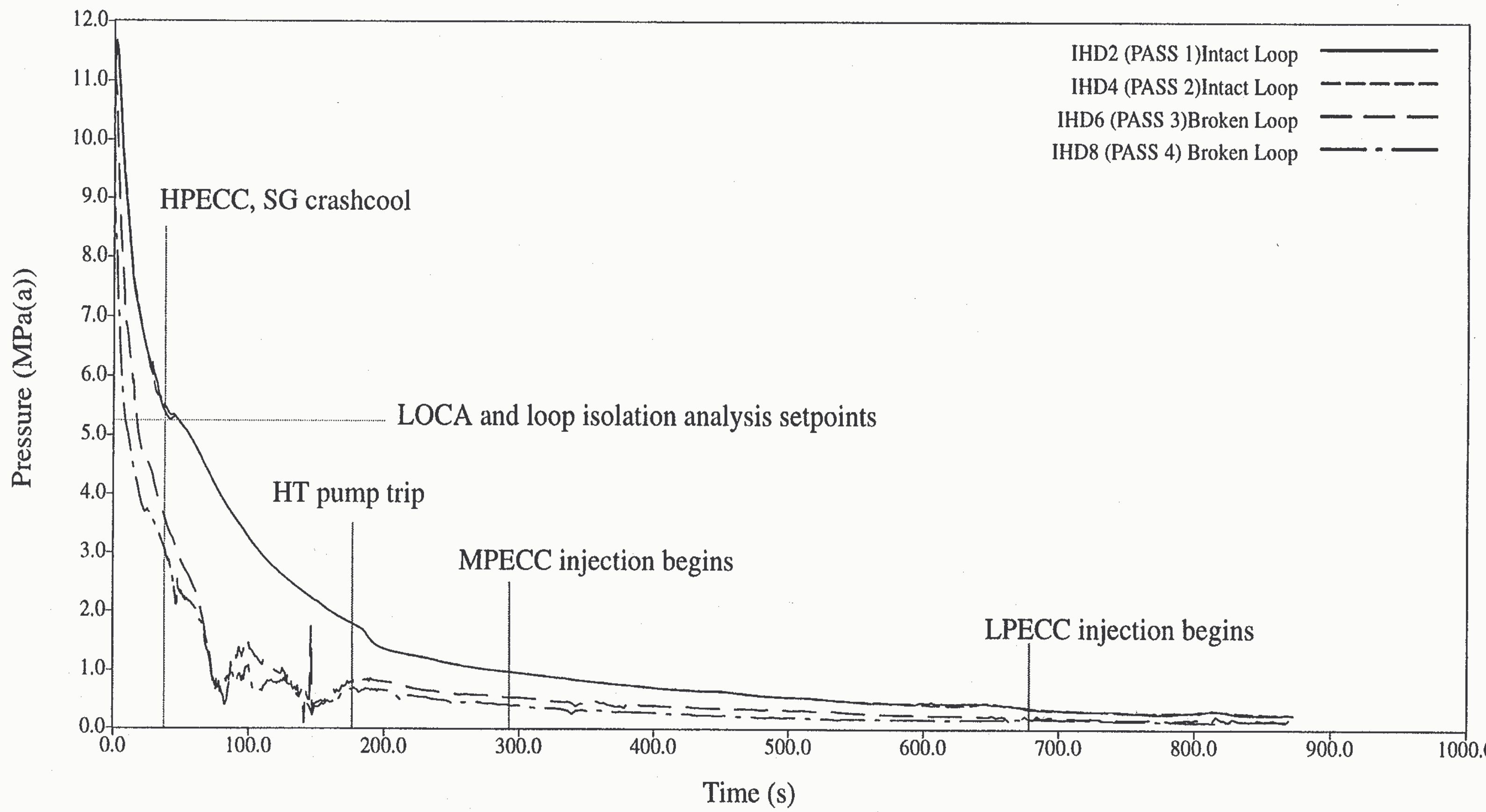
But for ACR…

- On large LOCA, gross reactivity initially increases due to “checkerboard voiding”, then decreases slowly
- Power increases, then decreases slowly
- Reactor trips on process parameter e.g. low flow
- Fuel temperature increases due to loss of heat removal and redistribution of stored heat

Sequence of Events - 2

- HTS pressure reduces to the ECC activation setpoint & ECC is activated; two loops (where relevant) are isolated, crash cool-down
- Containment pressure rises, building isolates, dousing sprays turn pressure over
- Some fuel fails, fission products released to containment, some small leakage
- Loops refill, fuel temperature falls
- Long-term ECC recirculation

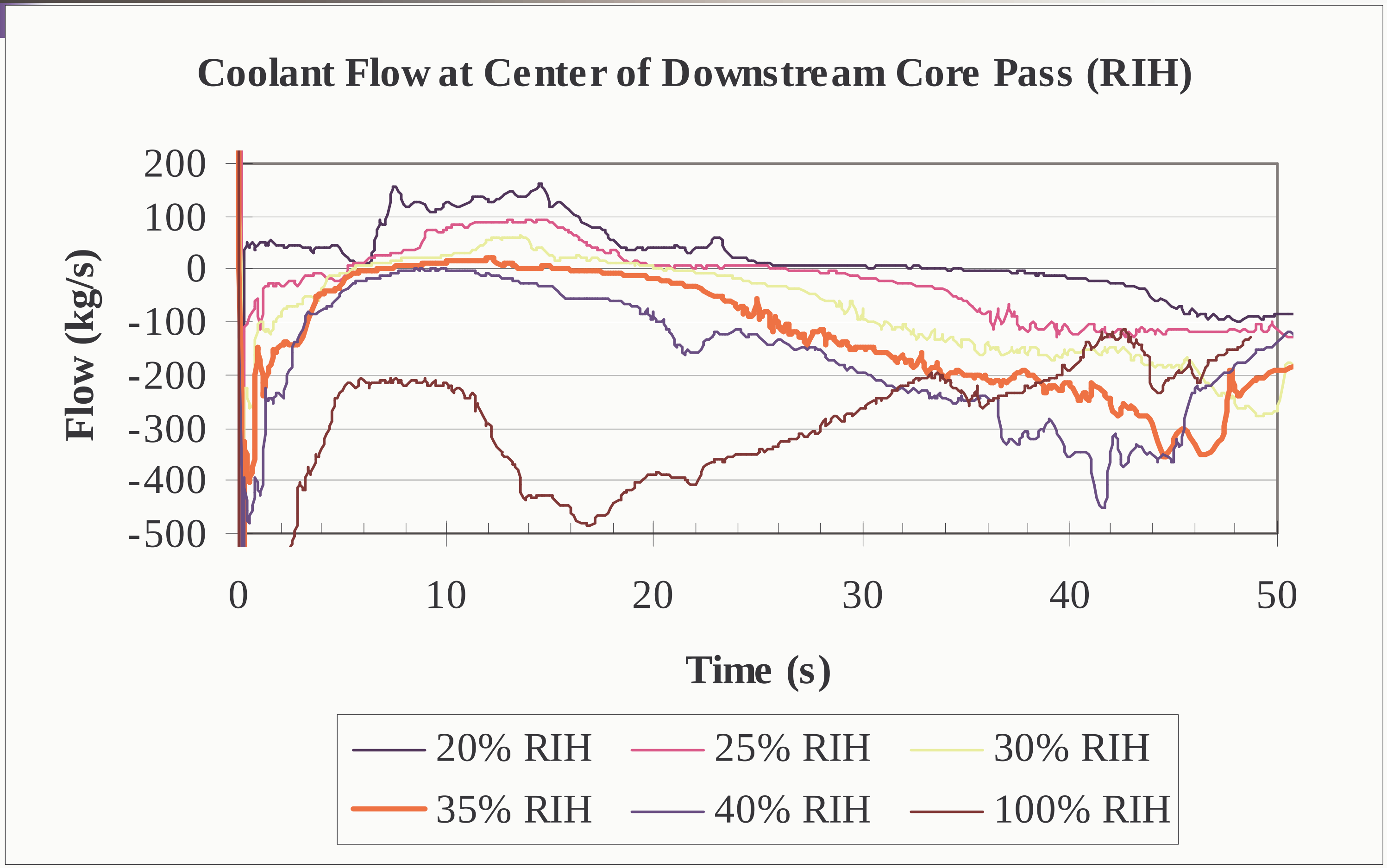
Time Scale of Large LOCA



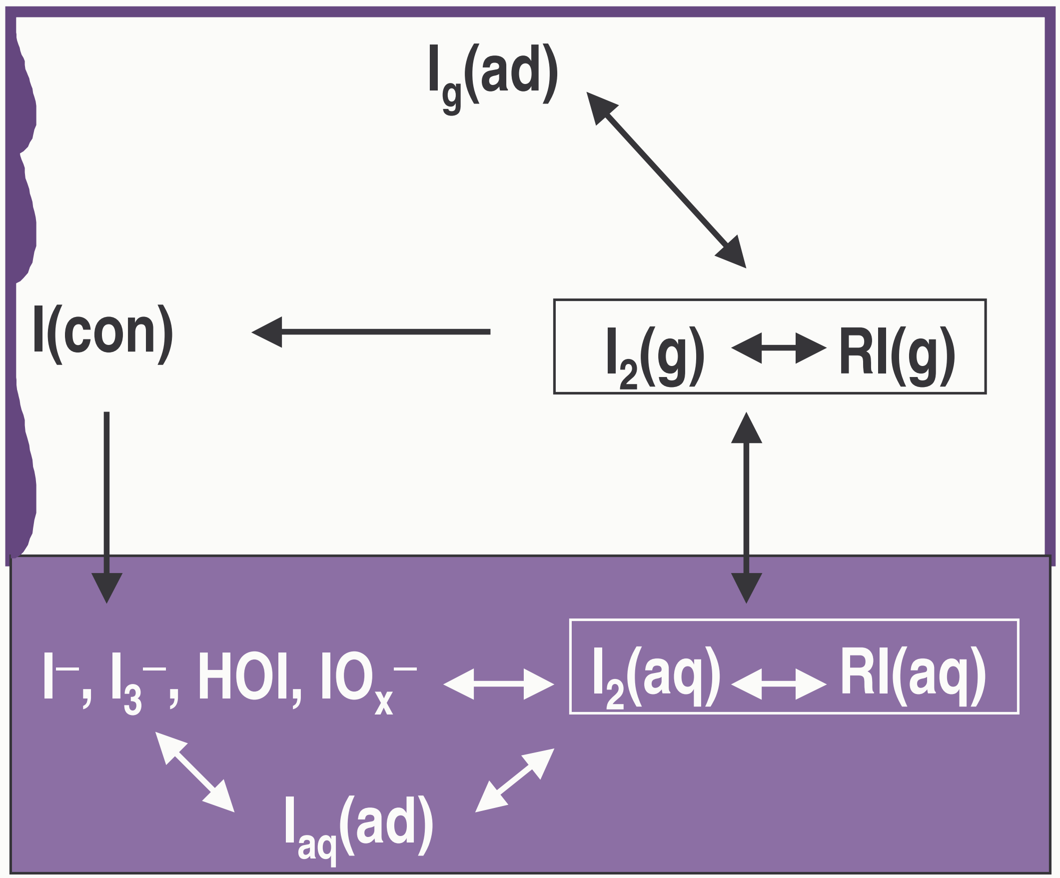
Safety Aspects

- Jet forces & pipe whip
- Flow decrease downstream of break
- Power increase & neutronic trip
- Fuel heatup and sheath strain
- Pressure-tube heatup, strain to contact with calandria tube
- Heat transfer to moderator
- Containment pressure increase
- Leakage of radioisotopes

Stagnation Break



Iodine Transport



Acceptance Criteria - 1

- Dose to the most exposed individual in the critical group is below relevant limit
  - Class 3 for C-6, single failure for siting guide, DBA for RD-337
- Pipe whip is limited so that:
  - no impairment of either of the shutdown systems below their minimal allowable performance standards
  - no break induced in the piping of the other HTS loop
  - no shearing off of large numbers of feeder pipes
  - no damage to the containment boundary
  - no break induced in ECC piping

Acceptance Criteria - 2

- Channel geometry must remain coolable
  - amount of fuel sheath oxidation must not embrittle the sheaths on rewet
  - amount of sheath strain must be limited so that coolant can flow through the channel.
- Channel integrity is maintained.
  - no fuel melting
  - no sheath melting
  - no constrained axial expansion of the fuel string

Acceptance Criteria - 3

- If the pressure tube strains or sags
  - the pressure tube does not fail prior to contacting the calandria tube ($\varepsilon < 100\%$)
  - the calandria tube remains intact after pressure tube contact (no prolonged film boiling)
- Pressure within containment is below design pressure.
- Pressure within containment compartments does not cause internal structural failures.

Three Classification Schemes

- Siting guide
  - Single process failure
  - Dual failure (process system failure + safety system failure)
- C-6 Rev. 0
  - 5 accident classes
- RD-337
  - AOO, DBA, BDBA – Severe Accidents

RD-337

- ***Anticipated Operational Occurrence (AOO)*****—**a deviation from normal operation that is expected to occur once or several times during the operating lifetime of the NPP but which, in view of the appropriate design provisions, does not cause any significant damage to items important to safety, nor lead to accident conditions.;

- ***Design Basis Accident (DBA)*****—**accident conditions for which an NPP is designed according to established design criteria, and for which damage to the fuel and the release of radioactive material are kept within regulated limits;and

- ***Beyond Design Basis Accident (BDBA)*****—**accident conditions less frequent and more severe than a design basis accident. A BDBA may or may not involve core degradation.

RD-337 Dose Acceptance Criteria

Dose Acceptance Criteria	
AOOs	DBAs
0.5 mSv	20.0 mSv

Summary of Acceptance Criteria

Event Frequency (occ/yr)	C-6 (Darlington)		Siting Guide (BA/BB/PN/PL/G2)		RD-310/RD-337	
	Event Classes	Dose Limits (mSv)	Categories	Dose Limits (mSv)	Class es	Dose Limits (mSv)
> 10^{-2}	Class 1	0.5	Single Failure	5	AOO	0.5
10^{-2} to 10^{-3}	Class 2	5	Single Failure	5	DBA	20
10^{-3} to 10^{-4}	Class 3	30	Single Failure	5		
10^{-4} to 10^{-5}	Class 4	100	Dual Failure	250		
10^{-5} to 10^{-7}	Class 5	250	Dual Failure	250	BDBA	---

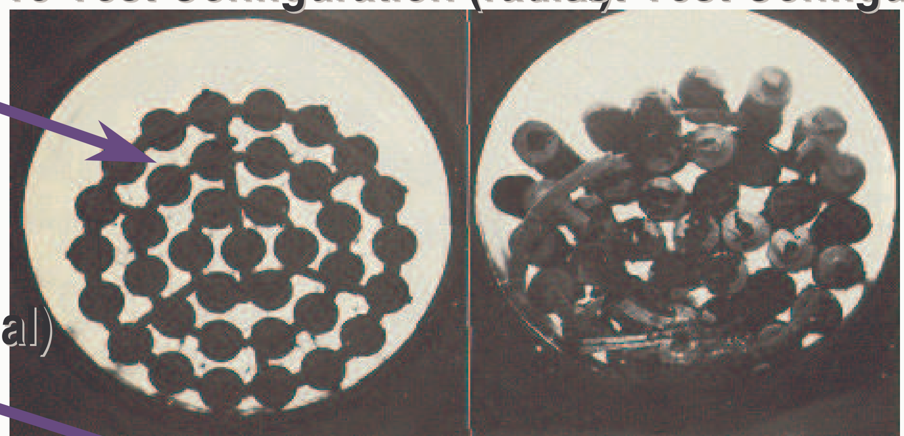
Event Combinations - ECC

- ECC Impairments
  - Failure of injection
  - Failure of crash cooldown
  - Failure of loop isolation
- Moderator required as a heat sink
- Low steam flow to channel
  - Metal water reaction
  - Hydrogen production & transport
- Cooling of broken loop

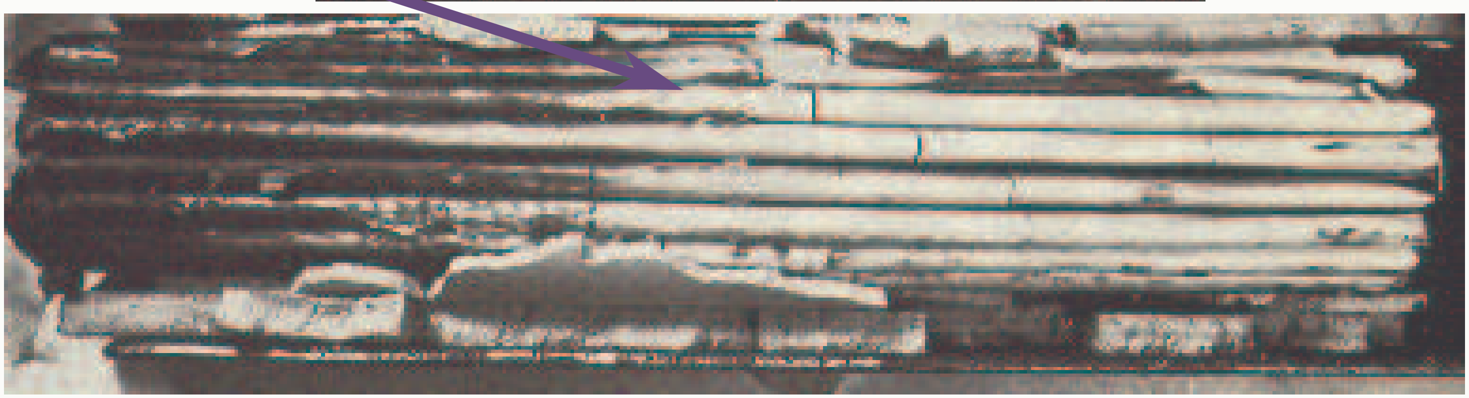
Fuel in LOCA + LOECC

Pre-Test Configuration (radial) Post-Test Configuration (radial)

37-
ELEMENT
BUNDLE



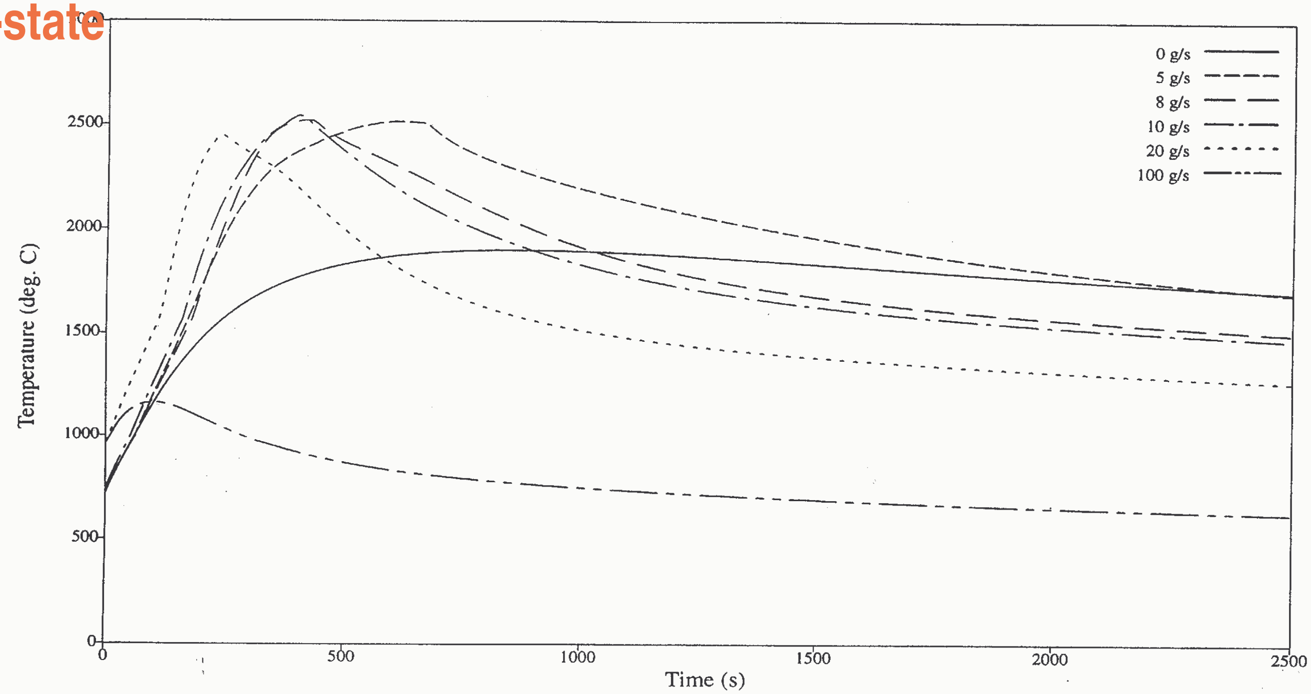
Post-Test
Configuration (axial)



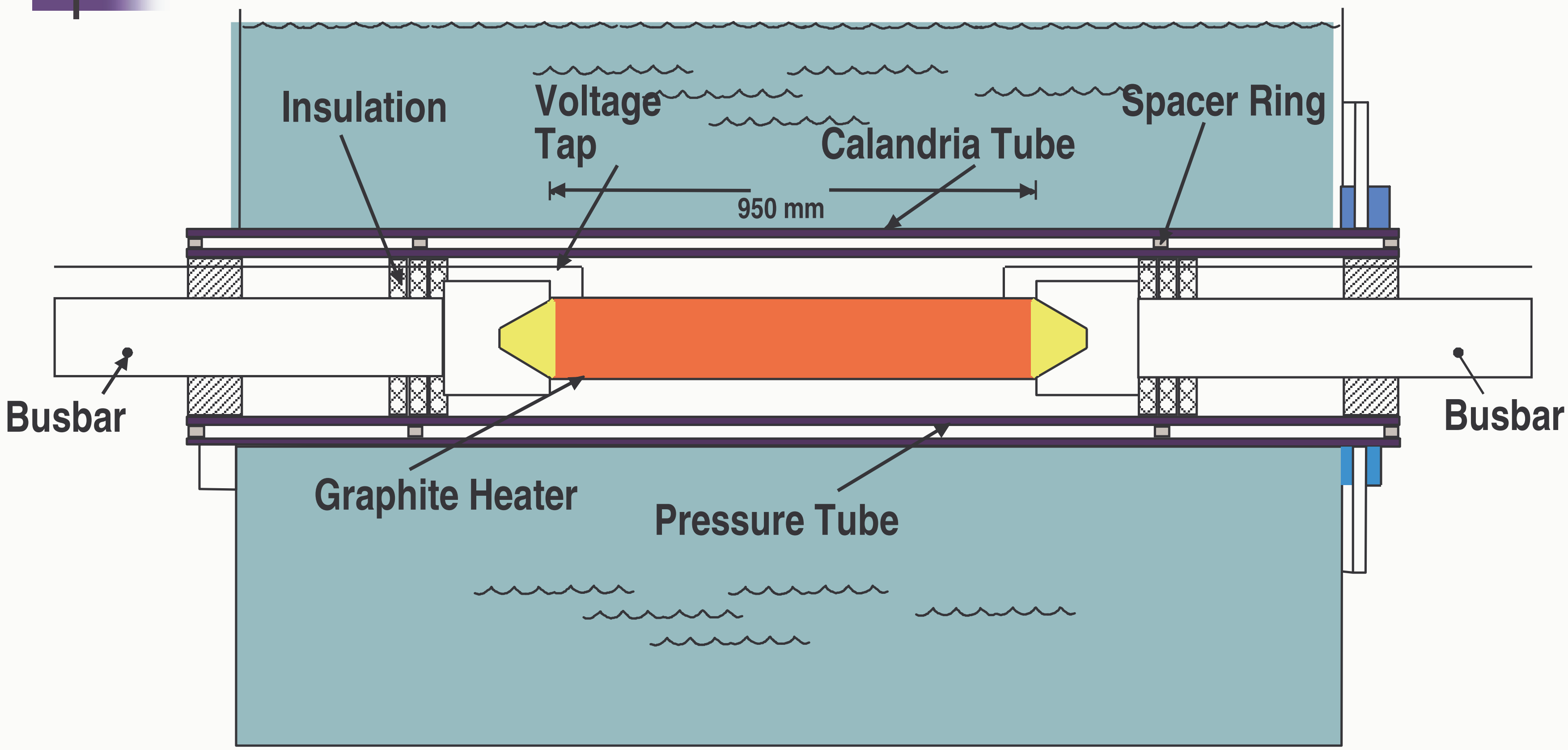
Sensitivity to Steam Flow

3 STAGES:

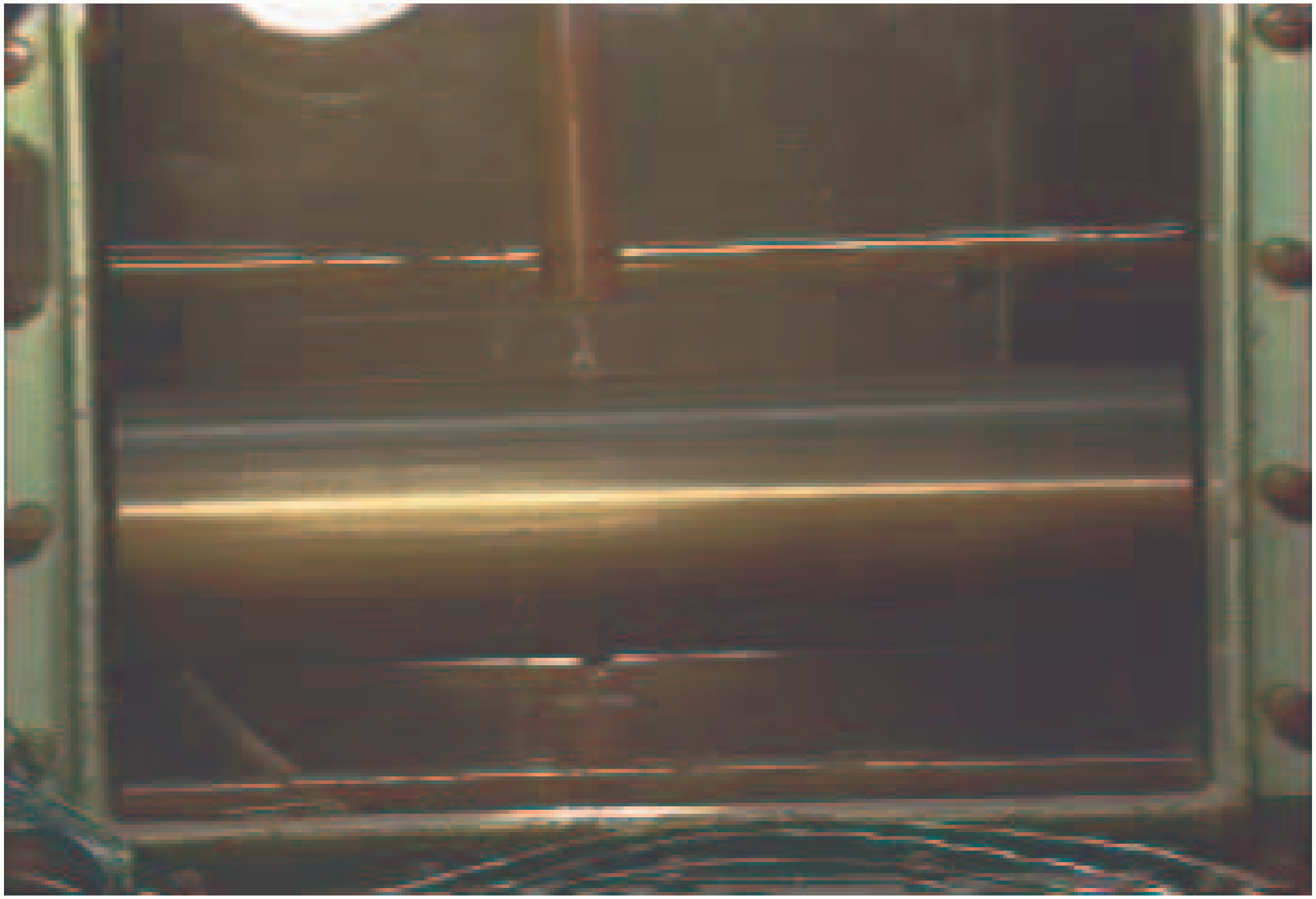
1) Initial heatup 2) Exothermic steam-Zircaloy reaction 3) Cool down to steady-state



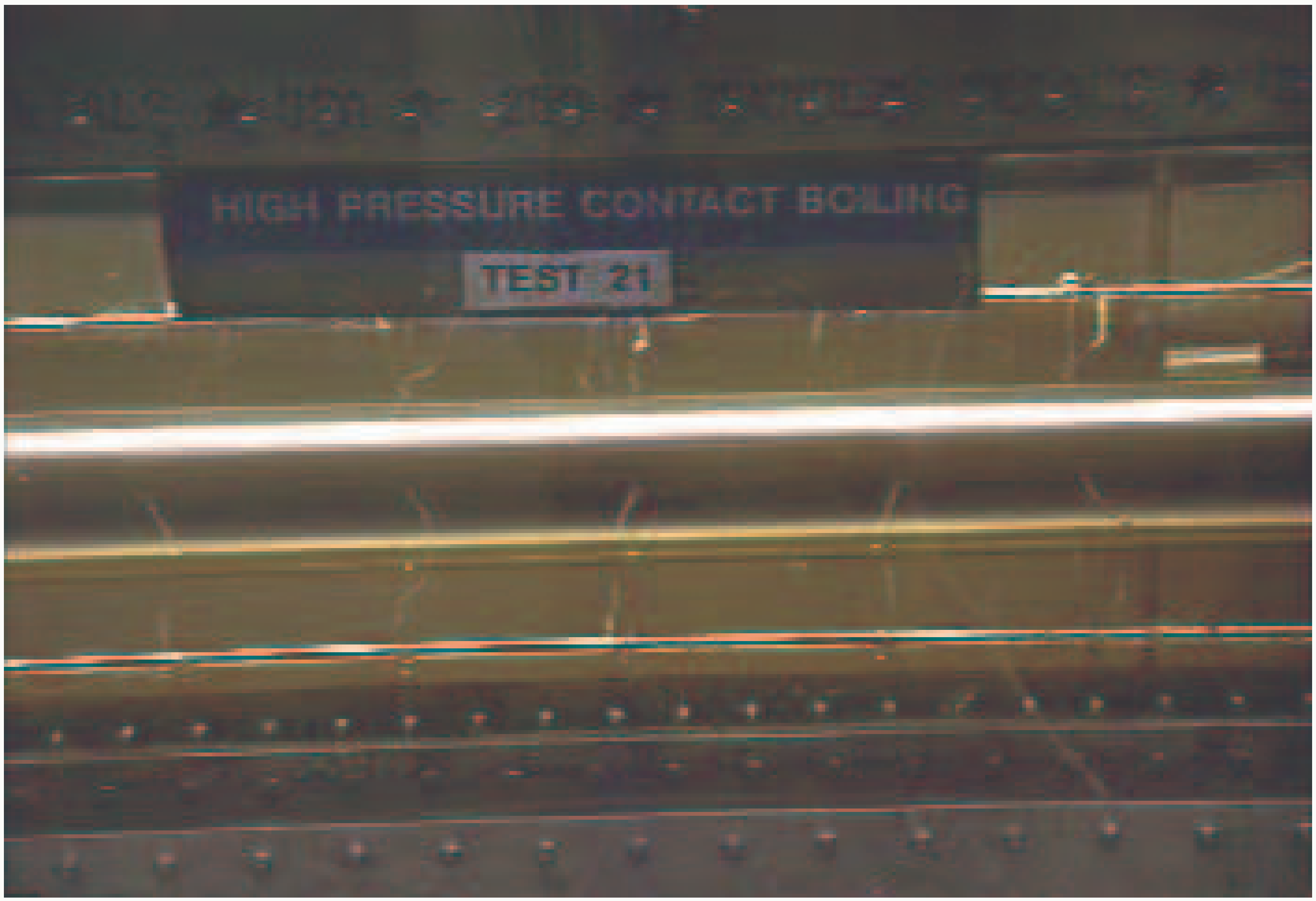
Pressure Tube Ballooning Experiments



Quenching (Nucleate Boiling) After PT/CT Contact

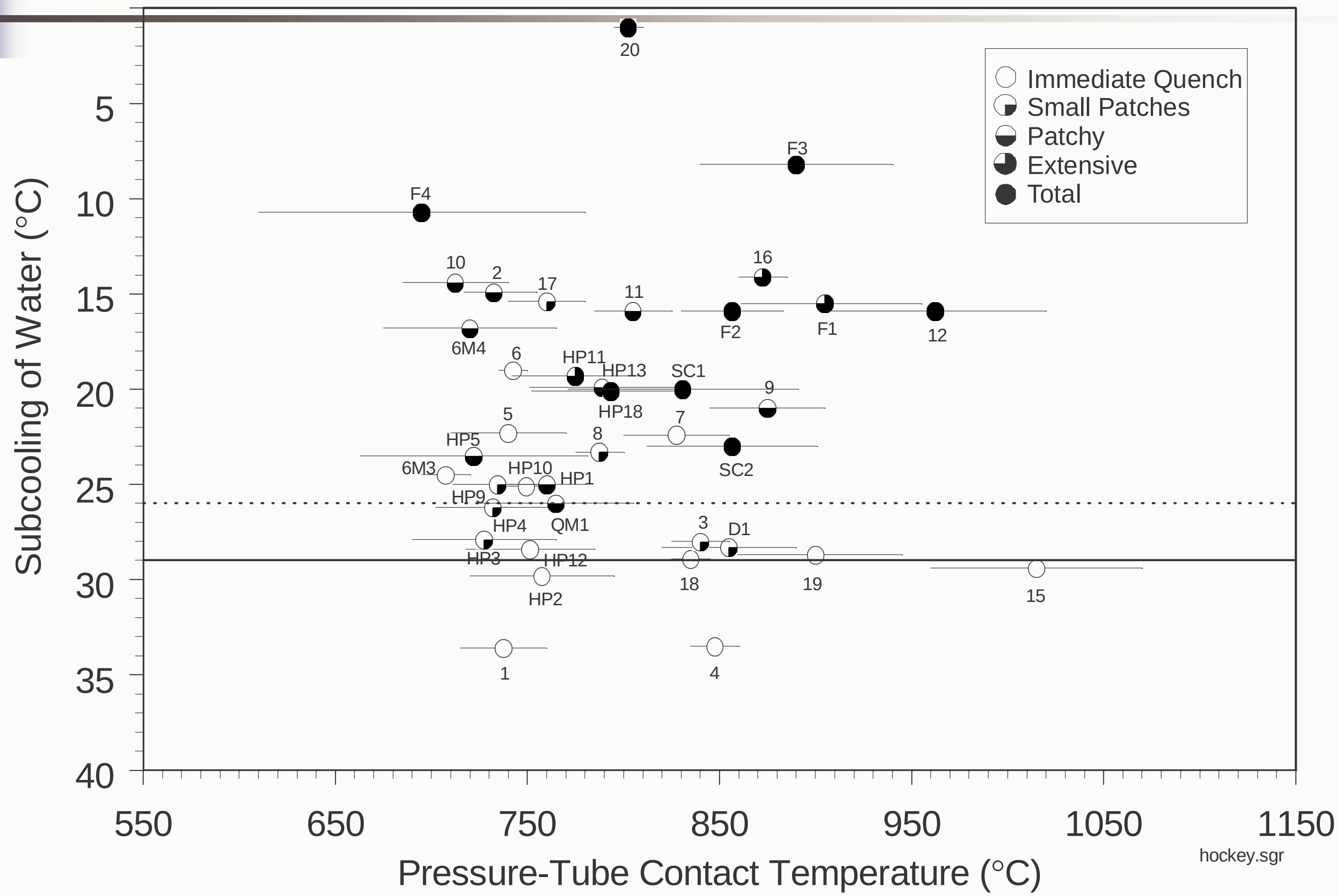


Film Boiling



Effect of CT surface

Subcooled Boiling Map



Safety Requirements – LOECC

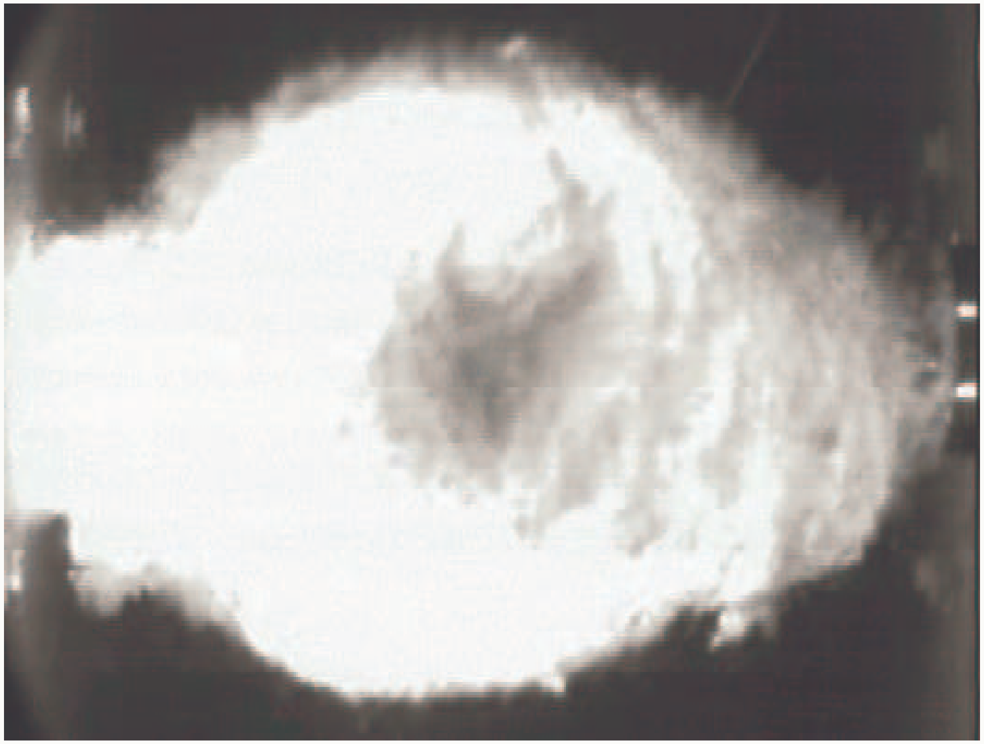
- Dose limits from siting guide and C6, frequency limits from RD-337
- No limit on fuel damage
- Prevent channel failure
  - No fuel melting, adequate moderator subcooling
- No hydrogen detonation or fast flame

Event Combinations - Containment

- Containment impairments
  - loss of air coolers, loss of dousing, open ventilation dampers, deflated airlock door seals, (open airlock doors)
  - partial or total loss of vacuum, failure of the instrumented containment pressure relief valves to open or close, failure of one bank of self-actuating containment pressure relief valves

In-Core Break: Bubble Growth

- Hot-pressurized water is discharged into the cool moderator water
- Coolant flashing occurs
- Steam bubble formation
- Bubble expands/contracts
- Pressurization of surrounding water
- Loading in-core structures
- Short term transient on order of milliseconds

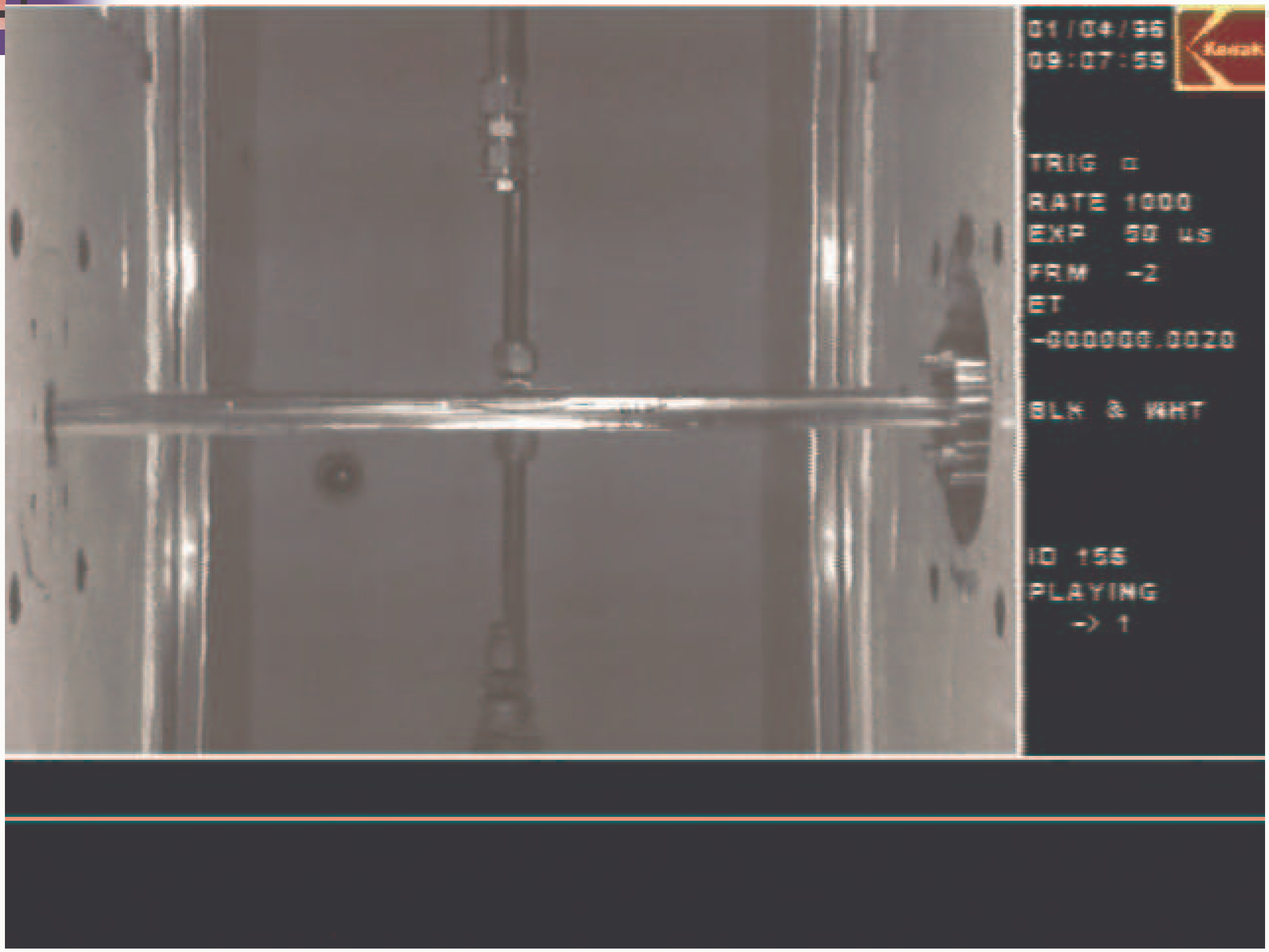


Small-Scale Burst Test Facility

Burst Tests -

- Designed to assess consequences of an in-core FC rupture
- ~ 1/10 scale
- Tests in water with various channel configurations
- Data supports code development and understanding of large-scale tests

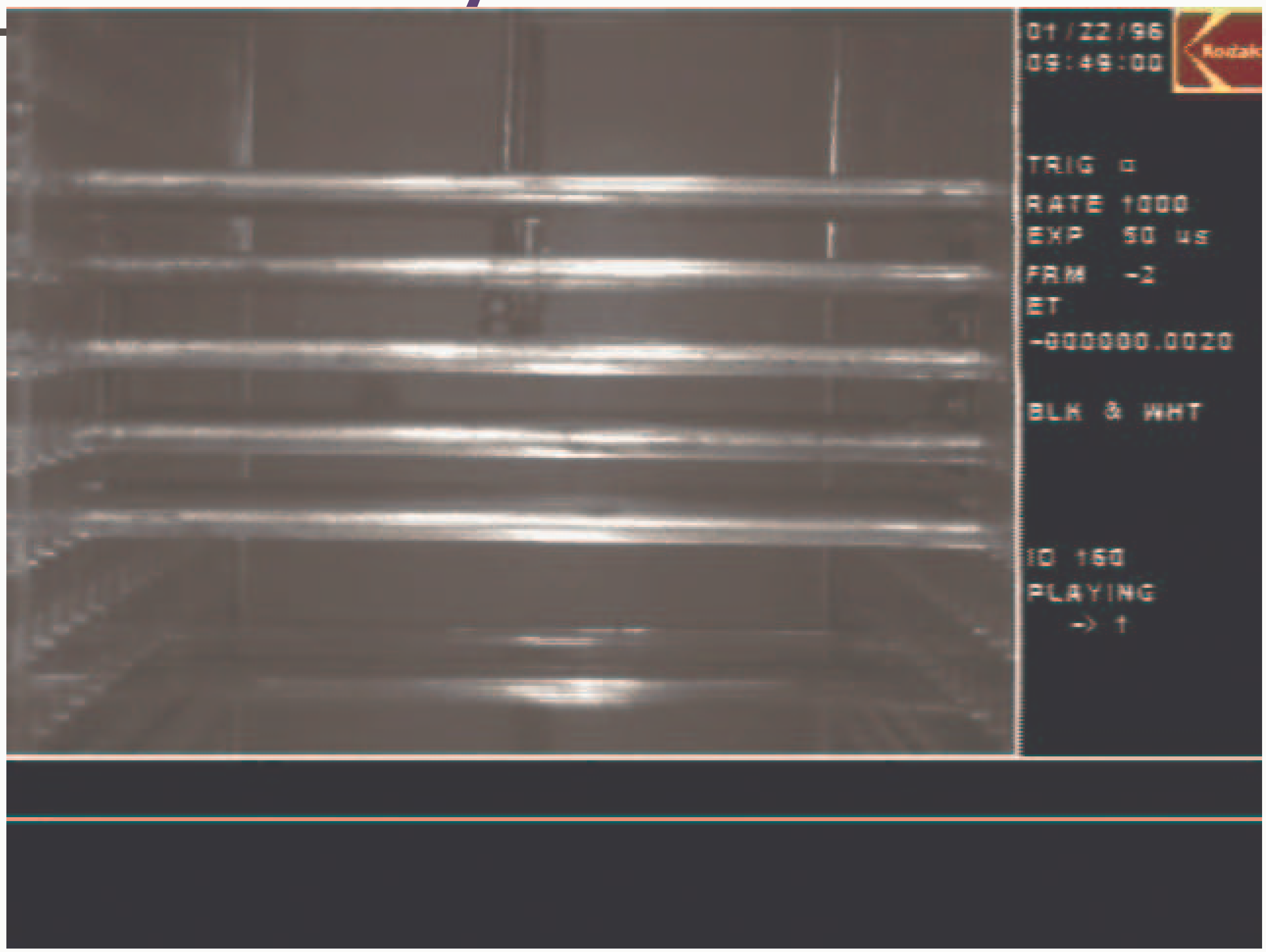
Bubble Dynamics



Burst Tests -

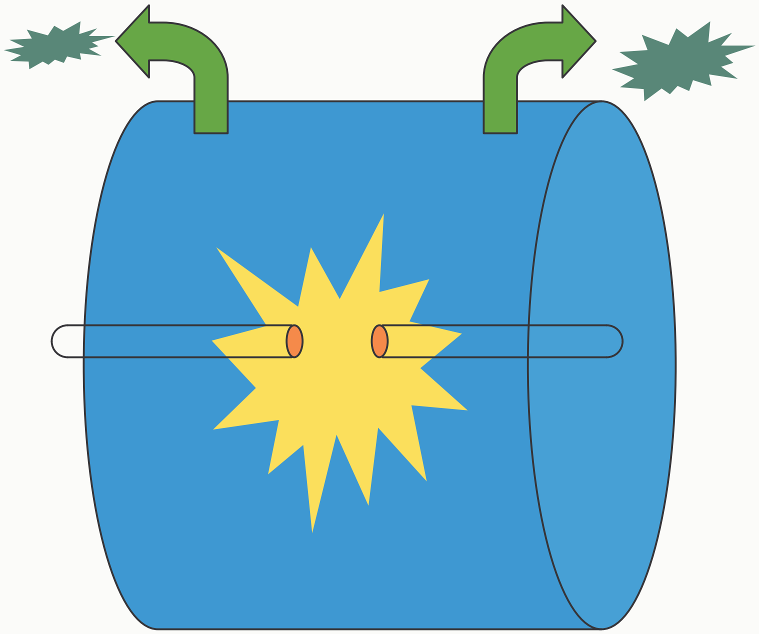
- **~ 1/10 scale**
- **Saturated water ~10 MPa**
- **Tests done with and without neighbouring channels**

5 x 5 Array


01/22/96
09:49:00
Kodak
TRIG
RATE 1000
EXP 50 us
FRM -2
ET
-000000.0020
BLK & WHT
ID 160
PLAYING
-> 1

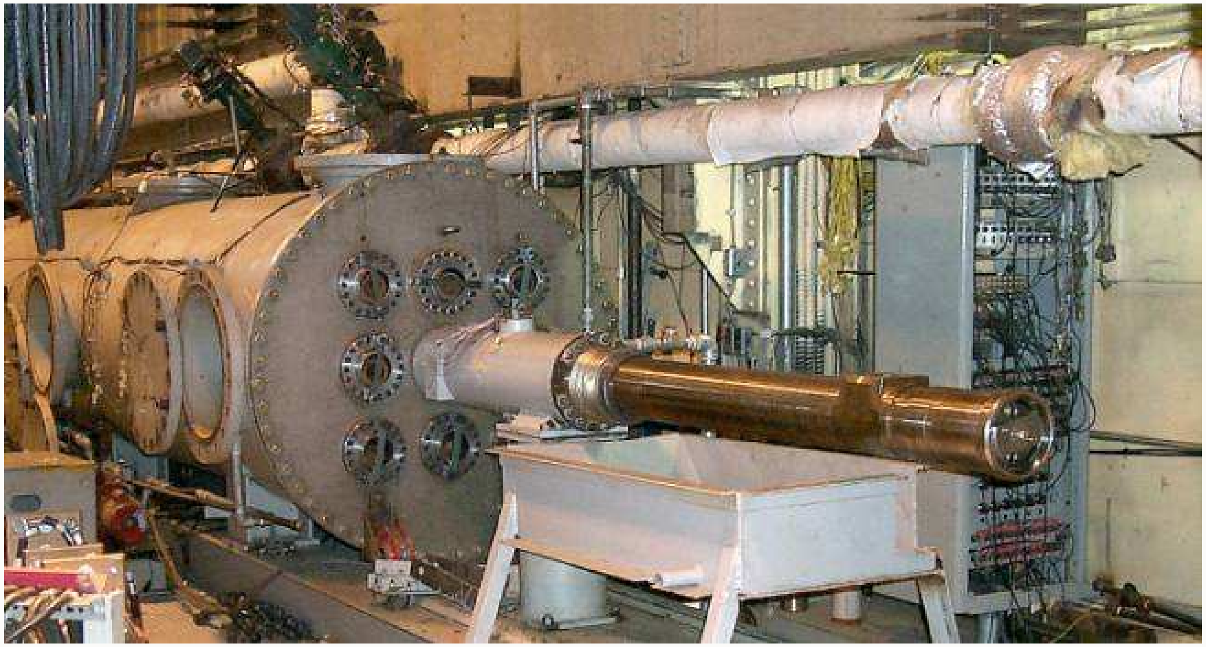
In-Core Break: Safety Issues

- Damage to reactivity mechanisms
- Propagation of the break to other channels
- Calandria overpressure
- Detection signals
- Displacement of moderator poison
- Fission product washout



Full Scale Channel Tests

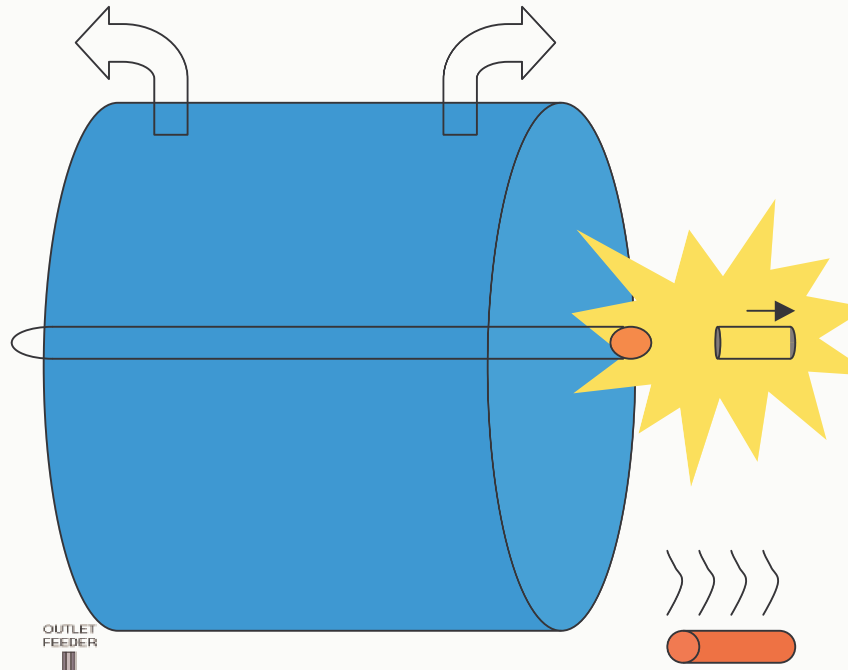
- Full scale Zr-2.5Nb PT in Zr-2 CT burst channel and target channels
- 9 channel array (3x3), burst channel location variable

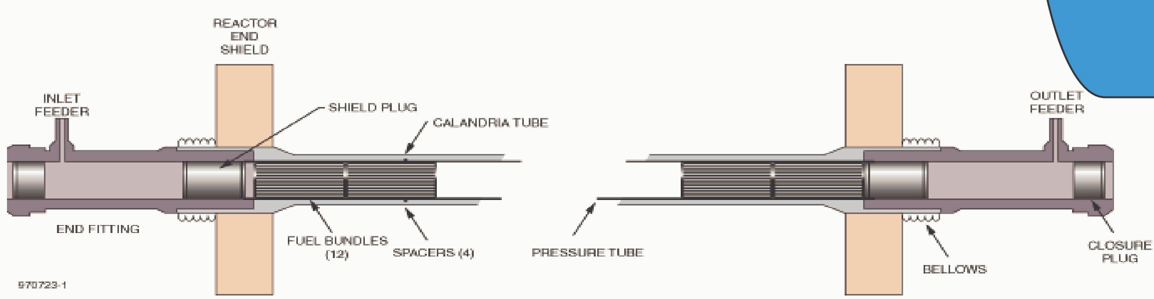


Calandria Tubes Absorb Energy

End-Fitting Ejection

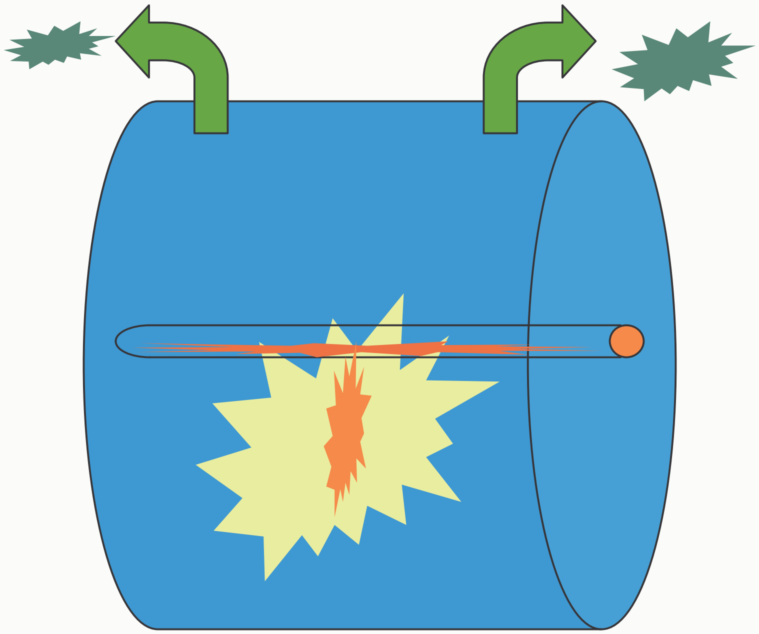
- Fuel bundle ejection into oxidizing atmosphere
- Cooling by water sprays?





Flow Blockage

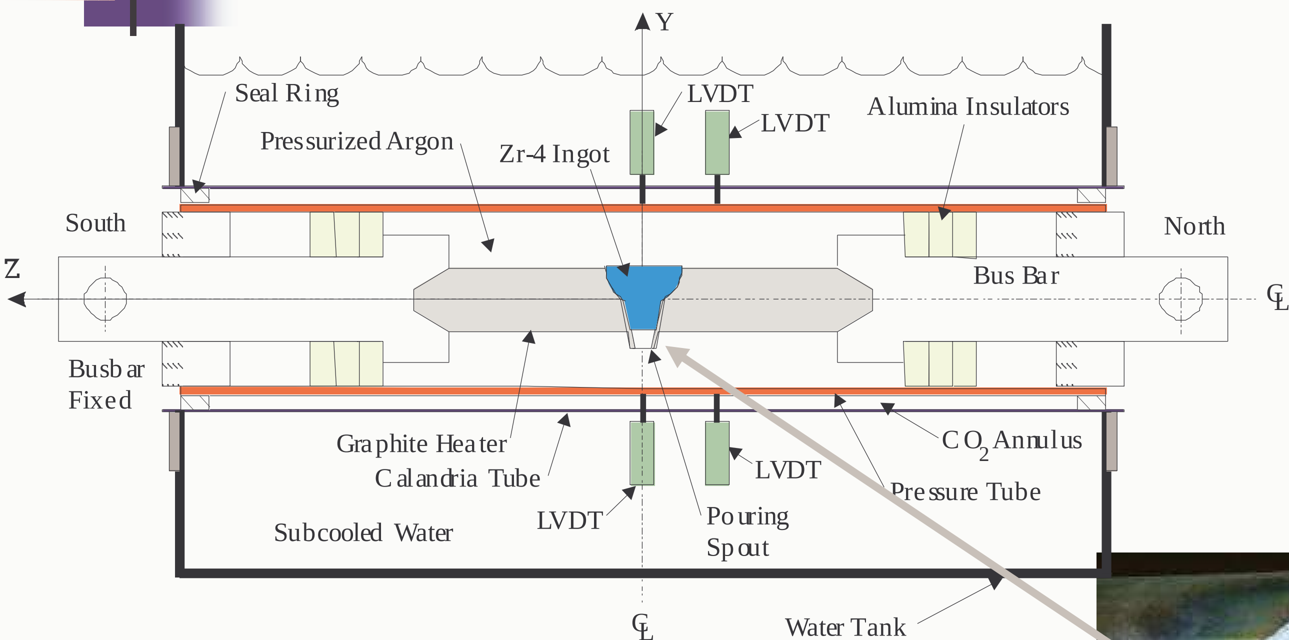
- Molten fuel moderator interaction
- How much melt is present when the channel fails?
- Does the interaction depend on the amount of molten fuel?



Molten Fuel – Channel Interaction

Small quantities of molten material from an overheated bundle have been shown to be sufficient to rupture a fuel channel at high system pressure in existing CANDUs

Flow Blockage Channel Rupture Tests



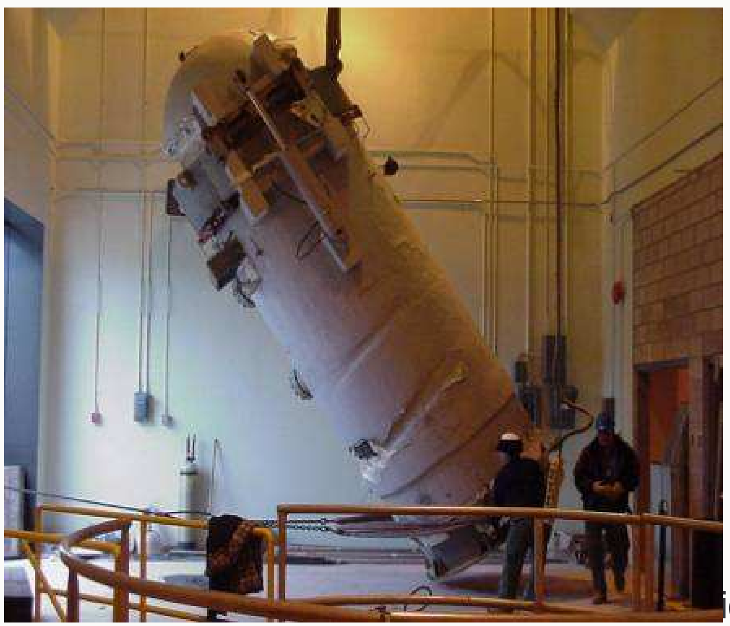
Test Apparatus

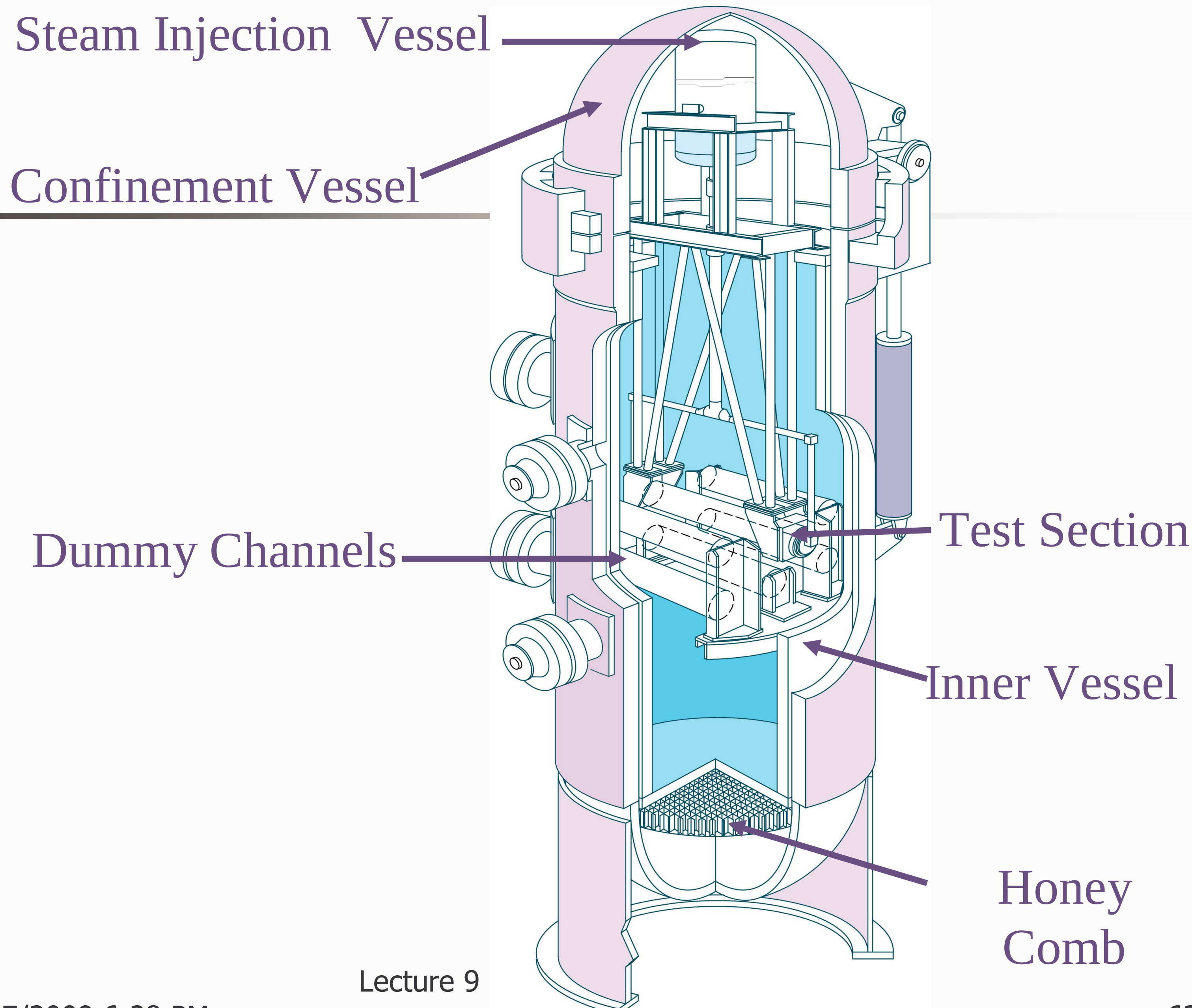
Post-test view of bottom of calandria tube

Flow Blockage Test

Molten Fuel Moderator Interaction

- During a single channel severe flow blockage event, it is postulated that molten material may be generated in the channel, which would subsequently be ejected into the moderator
- A test program has confirmed the dominant mechanism of interaction between molten fuel (ejected at operating pressure) and the moderator




Steam Injection Vessel
Confinement Vessel
Dummy Channels
Test Section
Inner Vessel
Honey Comb

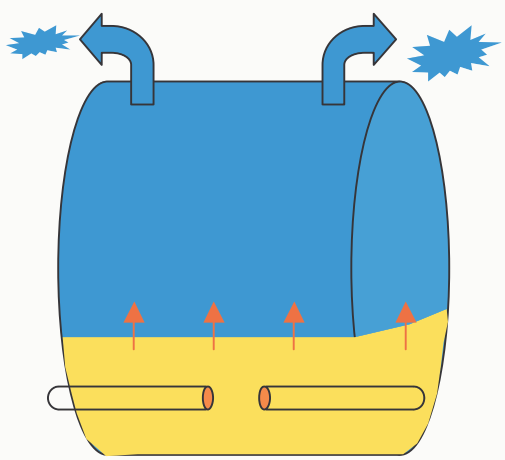
Results

- Completed ejection tests – e.g. for 23kg. test:
  - Melt temperature ~2400°C
  - Melt ejection pressure ~3 MPa
  - Steam injection lines (@10 MPa) opened ~30 ms after PT rupture
  - No "steam explosion" noted
  - Fine fragmentation of melt (<1 mm diameter)

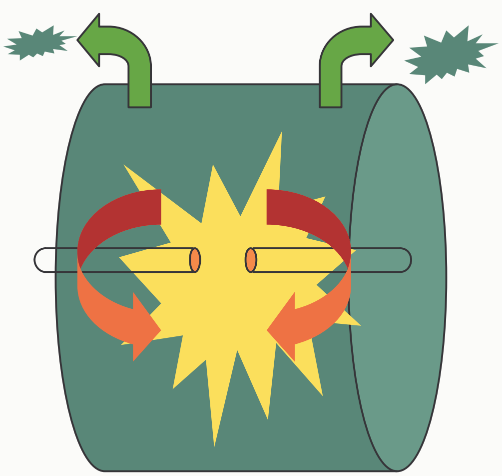
Poisoned Moderator

- Used during startup after a long shutdown
- Displacement of poisoned moderator by coolant
- Mixing modes

Piston model



Perfect Mixing



Reactivity Balance

Parameter	Conservative Direction	Rationale
Initial reactor operating state	Startup after a long shutdown	Maximize reactivity due to decay of neutron absorbers in the fuel
Fuel burnup	Plutonium peak	Maximize fuel reactivity requiring compensation; maximize void reactivity
Moderator poison load	High	Maximize reactivity due to displaced moderator
Coolant isotopic purity	High	Maximize reactivity due to moderator replacement
Failed channel location	Near most effective shutoff rods	Maximize loss of shutoff rod reactivity

ACR

- How would the use of light-water coolant in ACR affect the safety concerns?

In-Class Assignment

Go back again to the ZED-2 reactor and consider a loss of reactivity control caused by an unexpected moderator pump up. Identify as many of the key systems and parameters as you can for this accident; and for each, list the ‘conservative’ assumptions that you would use to ensure your answer (reasonably) overestimates the consequences.