Integrating Sudo-Kaminaga Correlation to the Safety Analysis Code PARET-ANL

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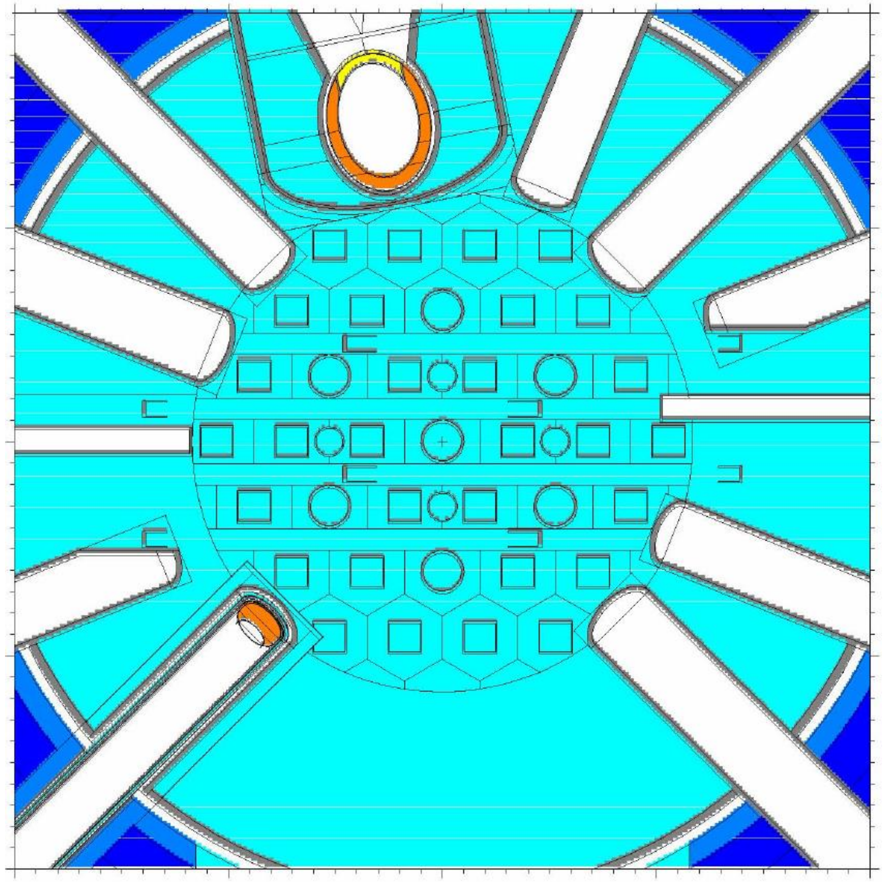
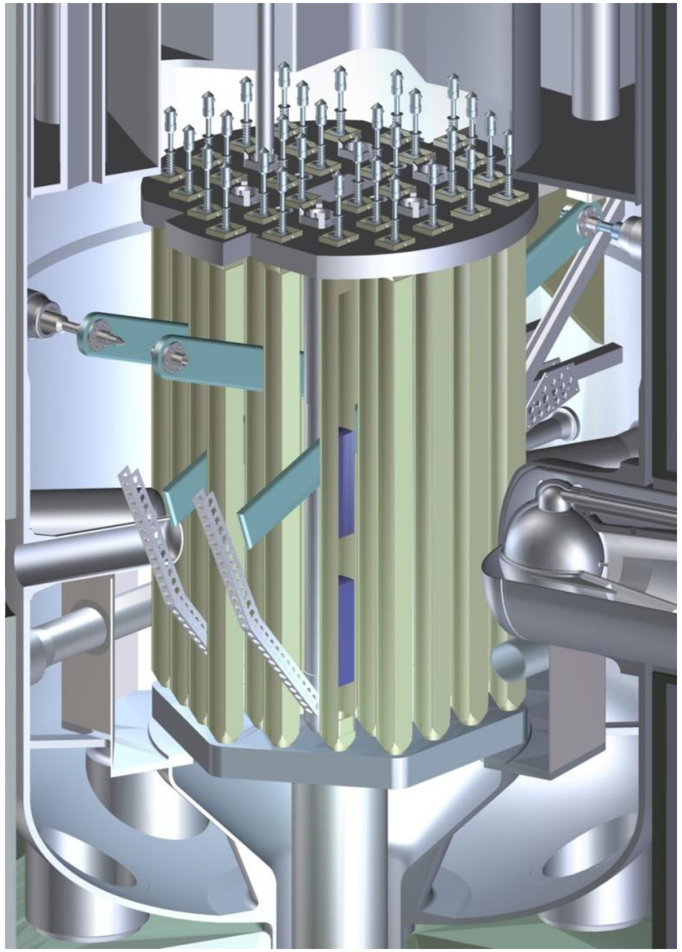
Status of the NBSR

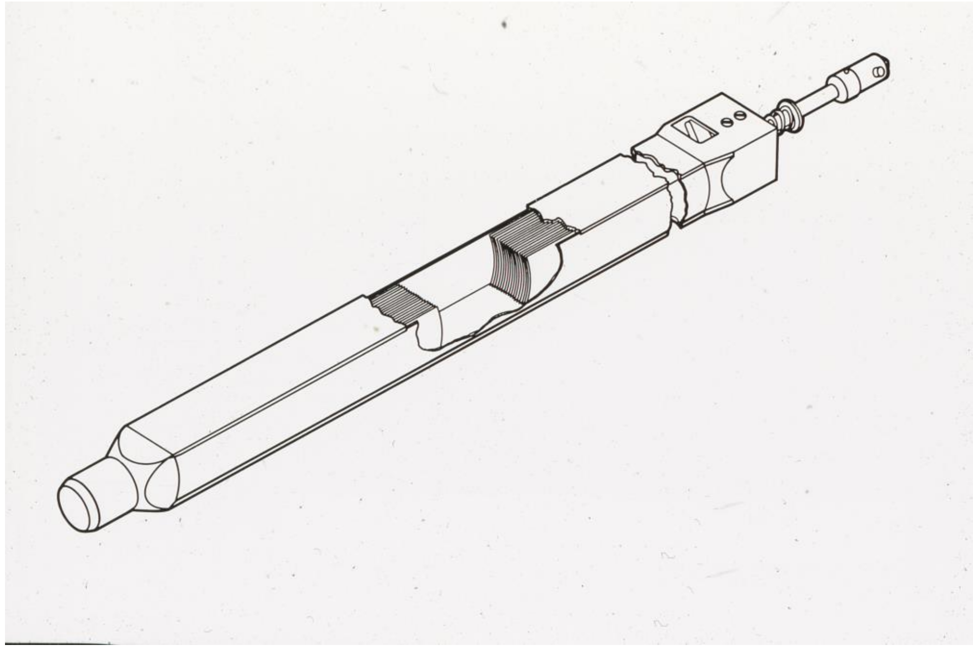
- The lifetime of the National Bureau of Standards Reactor (NBSR) will be coming to an end sometime in the next few decades.
- NBSR Main Characteristics:
  - High-Enrichment Uranium (HEU) fuel: 93 wt%
  - U_3O_8 + Al
  - <u>Vertically</u> Split Fuel Element
  - Full Power: 20 MW
  - D_2O Coolant , Moderator, Reflector



The mid-plane of the NBSR core

Neutron Beam Split-core Reactor

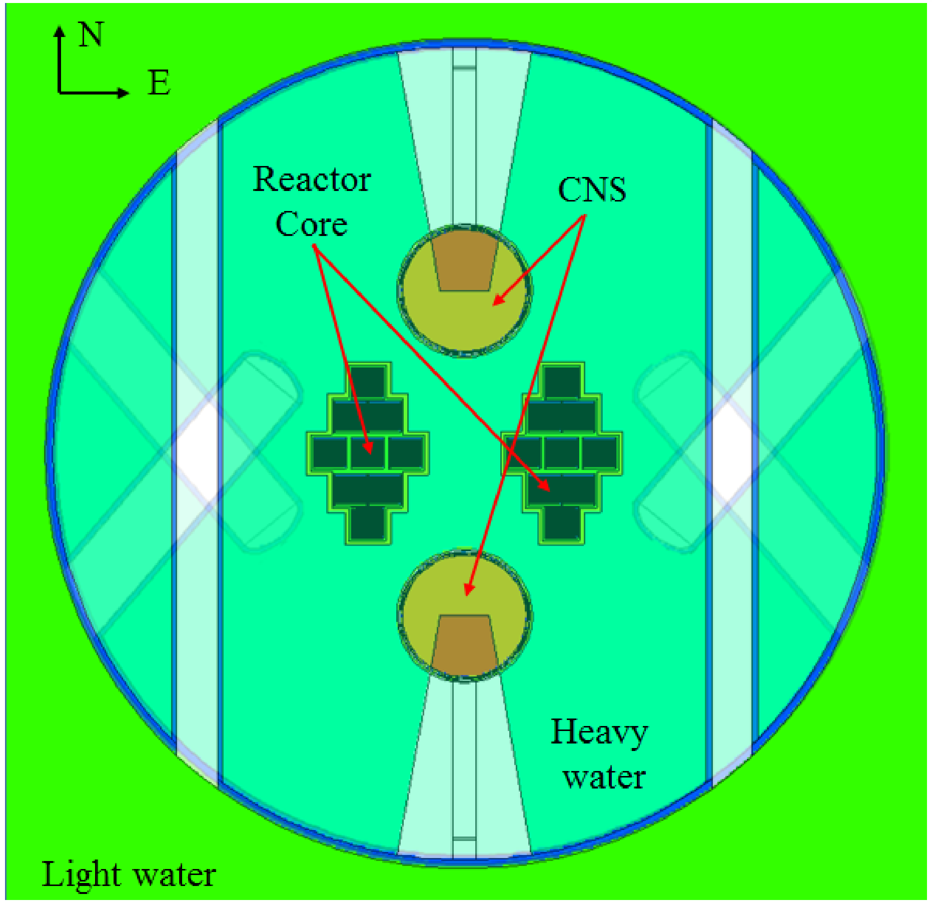




NBSR Fuel Element

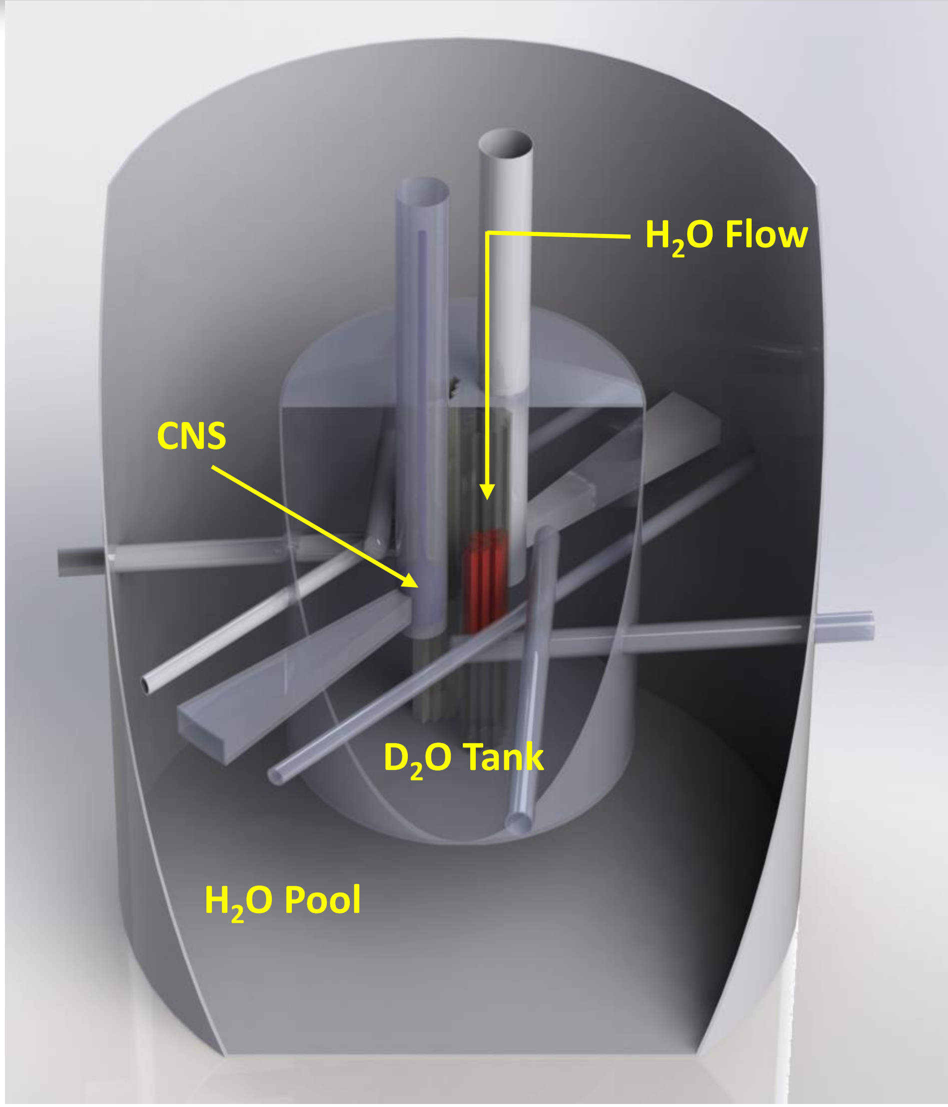
Overview of the Conceptual New Reactor

- Reactor Core Characteristics:
  - Low-Enrichment Uranium (LEU) fuel: 19.75 wt%
  - U_3Si_2 + Al
  - Horizontally Split Core
  - **Two** Cold Neutron Sources (CNS)
  - Full Power: 20 MW
  - Operation Cycle: 30 days
  - H_2O Coolant, Moderator
  - D_2O Reflector, Moderator

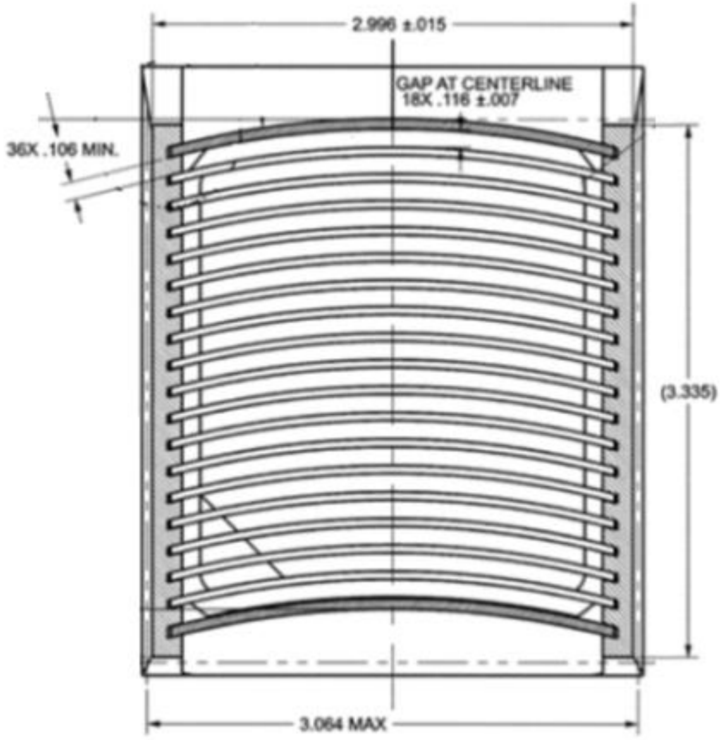


The mid-plane of the split core reactor

3-D View of the Horizontally Split Core Design



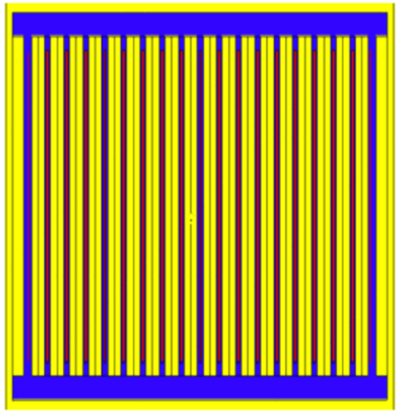
Parameter Changes in FE for New Reactor



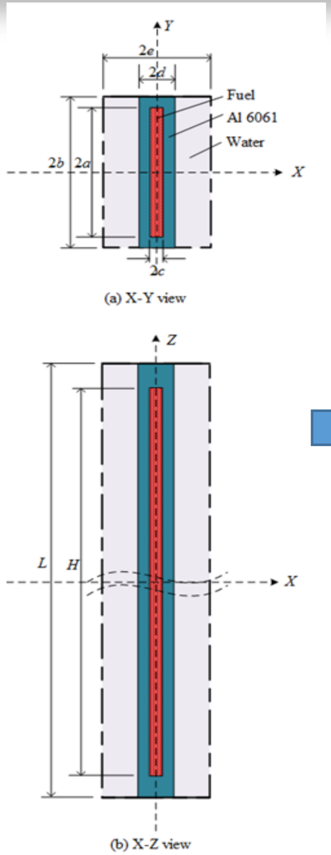
Cross Sectional View of the Fuel Element

Parameters	NBSR	New Core
Number of fuel elements	30	**18**
Half thickness of fuel meat	0.0254 cm	**0.033 cm**
Length of fuel meat	55.88 cm	**60 cm**

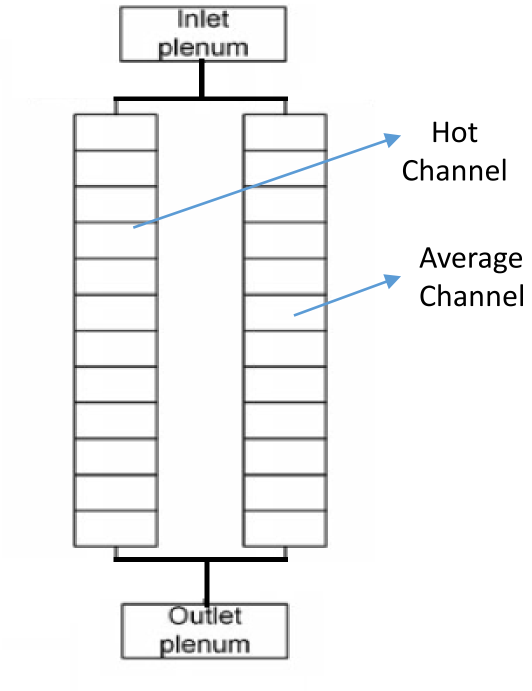
Transient Safety Analysis Modeling



Fuel Element



A T/H Channel definition



PARET-ANL Two-Channel Model

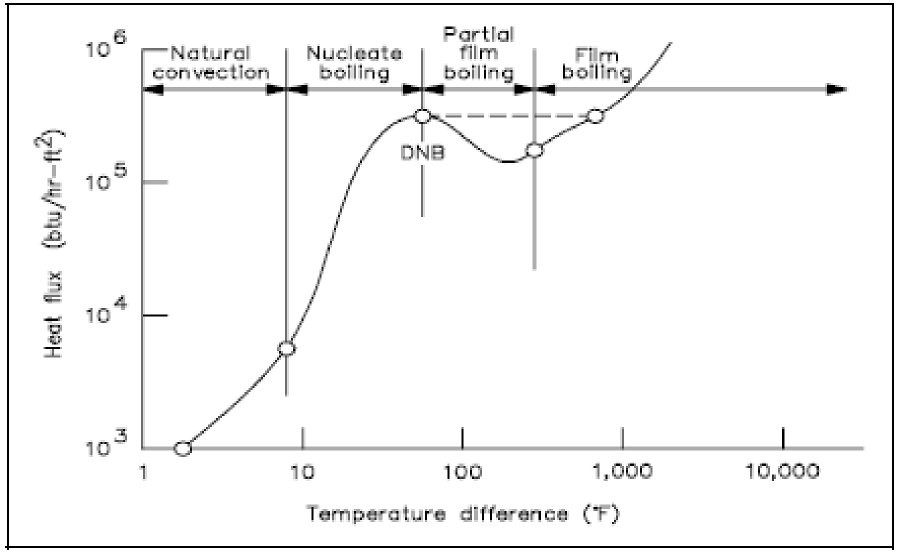
Introduction to PARET-ANL Code

- **P**rogram for the **A**nalysis of **RE**actor **T**ransients developed by Argonne National Laboratory (ANL)
- Intended primarily for the safety analysis of test and research reactors that use plate-type (flat) fuel elements
- Based on an evaluation of the coupled thermal, hydrodynamic, and nuclear effects of the core (point-kinetics model)
- Program calculates Critical Heat Flux Ratio (CHFR) using the <u>Mirshak</u> Correlation.

Critical Heat Flux & Onset of Flow Instability

Critical Heat Flux (CHF)

- The thermal limiting condition where a phase change occurs during heating which decreases efficiency of heat transfer causing localized overheating of the heating surface.

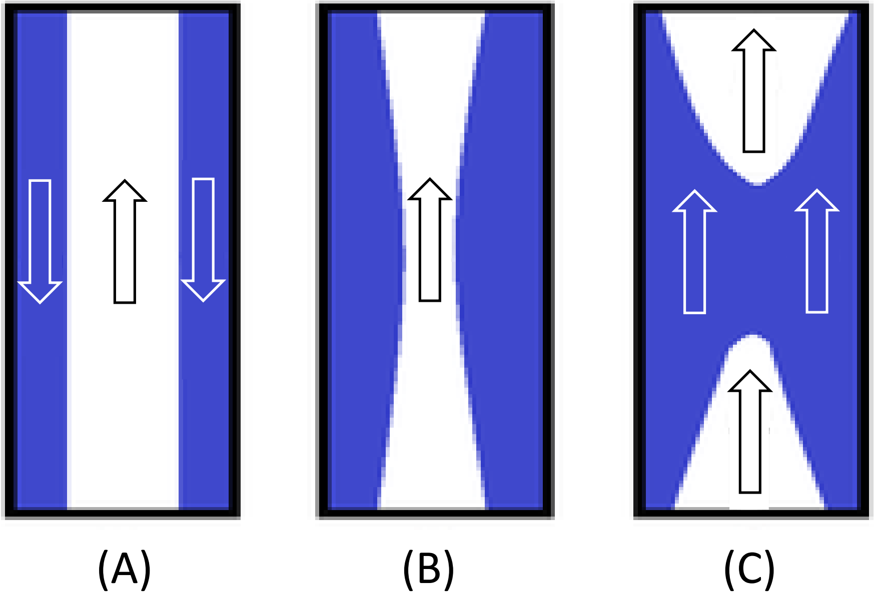


Thermal Limits Criteria

Probability Level	CHFR	OFIR
90%	1.301	1.310
95%	1.391	1.403
99.9%	1.778	1.828

Onset of Flow Instability (OFI)

- Excursive flow instability due to the onset of net vapor generation in the coolant channel.



Main Objectives

- Upgrade the critical heat flux ratio (CHFR) calculations with the PARET-ANL output by using the Sudo-Kaminaga correlation

- Determine the safety margins for various transient cases based on the critical heat flux ratio (CHFR)

Mirshak Correlation (1959')

Mirshak-Durant-Towell

$$q''_{CHF} = 1.51(1 + 0.1198\,U)(1 + 0.00914\,\Delta T_{sub,0})(1 + 0.19P)$$

Where

$\Delta T_{sub,0}$ is the subcooling at the channel exit (^{o}C)

U is the coolant velocity (m/s)

P is the absolute pressure at the channel exit (bar)

- $CHFR = \frac{q''_{CHF}}{q''_{model}}$

Sudo-Kaminaga Correlation (1993')

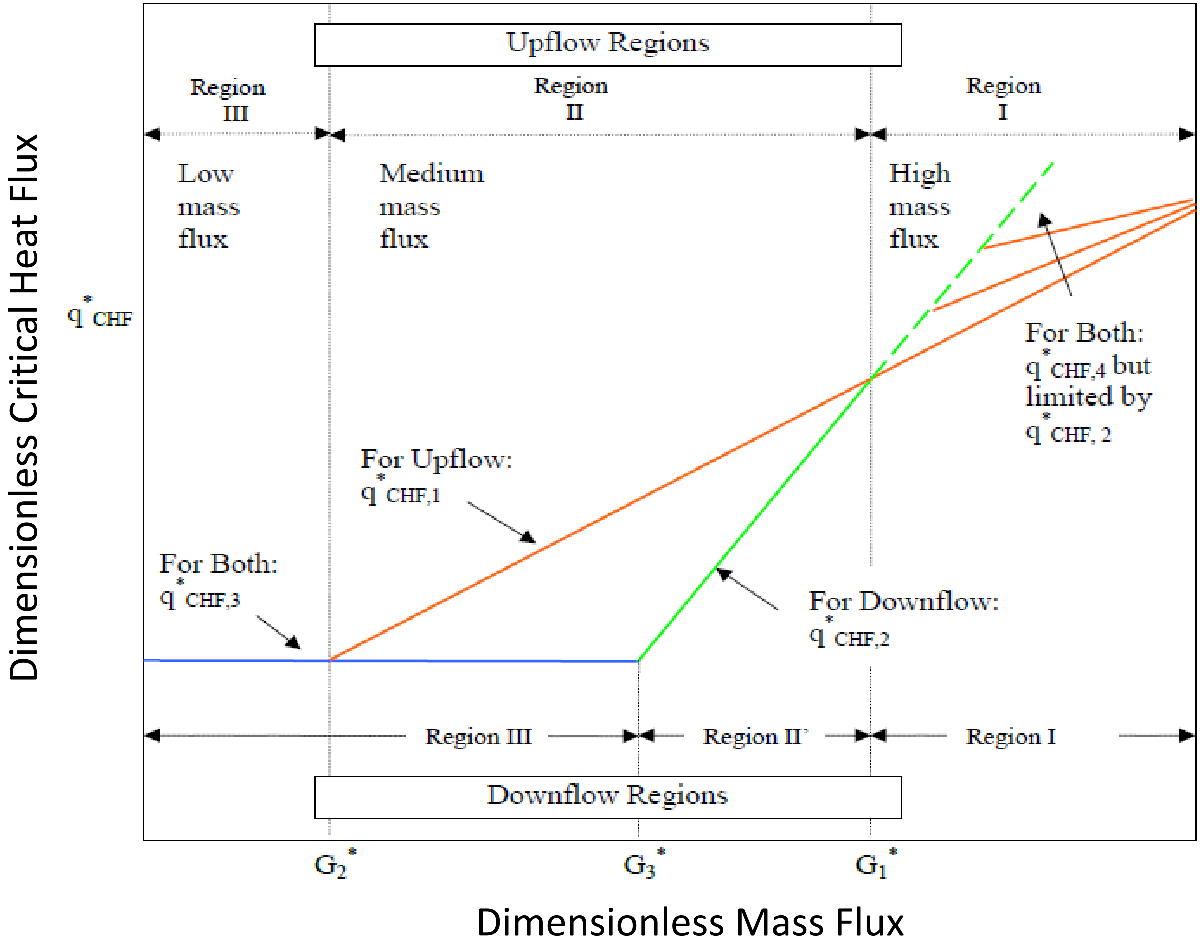
- Developed specifically to calculate CHFR for plate-type fuels and vertical rectangular flow channels of research reactors
- Considers effects:
  - Pressure (0.1-4 MPa)
  - Inlet sub-cooling (1-213 K)
  - Outlet sub-cooling (0-74 K) and quality (0-1)
  - Mass flux (-25,800-6250 kg/m^2-s)
  - Channel configuration
  - Flow direction

- $q^*_{CHF,1} = 0.005|G^*|^{0.611}$
- $q^*_{CHF,2} = \frac{A}{A_H}|G^*|\Delta T^*_{sub,in}$
- $q^*_{CHF,3} = 0.7\frac{A}{A_H}\frac{\sqrt{\frac{W}{\lambda}}}{\left(1+\left(\frac{\rho_g}{\rho_l}\right)^{\frac{1}{4}}\right)^2}(1.0 + 3.0\Delta T^*_{sub,in})$
- $q^*_{CHF,4} = 0.005|G^*|^{0.611}\left(\frac{5000}{|G^*|}\Delta T^*_{sub,0}\right)$

- $q''_{CHF} = q^*_{CHF}h_{fg}\sqrt{\lambda(\rho_l - \rho_g)\rho_g g}$
- $CHFR = \frac{q''_{CHF}}{q''_{model}}$

Sudo-Kaminaga Correlation (Cont.)

Sudo-Kaminaga Correlation Scheme



- S-K correlation takes account of mass flux, inlet and outlet sub-cooling, flow direction, pressure and channel configuration.
- The mass flux boundaries for each region are calculated based on flow fluid properties and flow channel conditions.
- CHF correlations for each region are determined in terms of dimensionless parameters.
- The CHFR is evaluated as

$$\mathrm{CHFR} = \frac{q''_{\mathrm{CHF}}}{q''_{\mathrm{Model}}}.$$

Why Sudo-Kaminaga Correlation?

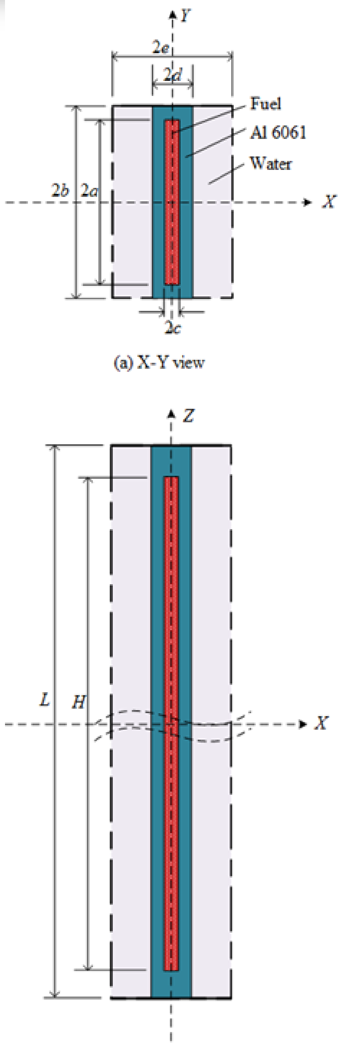
- Mirshak correlation was used in previous NBSR SAR (2004').

- Sudo-Kaminaga correlation represents an improvement due to the enhanced geometric similarity and increased range of applicability that are more prototypic of the actual operation conditions for the NBSR and an overall approach that is more mechanistic.

- Sudo-Kaminaga correlation is used in updated NBSR SAR (2010') and judged most appropriate for the flow conditions in the reactor as well (Justified by safety analysis experts in BNL nuclear reactor analysis team).

- The new reactor possesses great geometric and physics similarities to the NBSR.

T/H Channel Parameters and Conditions



(a) X-Y view

(b) X-Z view

Dimensions	Size (cm)
Half width of the fuel meat (a)	3.067
Half width of the fuel plate (b)	3.3325
Half thickness of the fuel meat (c)	0.033
Half thickness of the fuel plate (d)	0.0635
Half pitch of the fuel plates (e)	0.211
Length of the fuel meat (H)	60
Length of the channel (L)	67.28

T/H Conditions	Value
Outlet pressure (kPa)	200
Inlet temperature (°C)	37
Inlet volumetric flow rate (gpm)	8000
Flow area of the channel (cm^2)	1.97
Heated surface area of the channel (cm^2)	736
Wetted perimeter of the channel (cm)	13.63
Hydraulic diameter (cm)	0.58
Press drop along the channel (kPa)	56.04

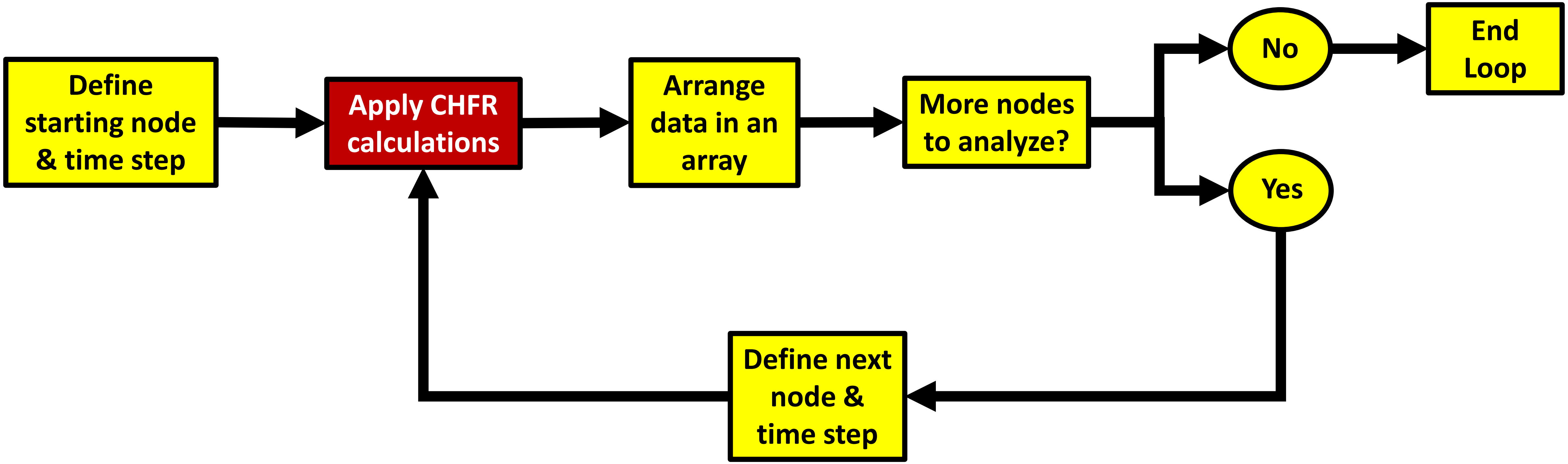
Kinetics Parameters (MCNP-6.1)

The Reactor	End-of-Cycle
k_{eff}	1.00254 ± 0.00012
Prompt neutron generation time (μs)	252.63 ± 2.22
Effective delayed neutron fraction (β_{eff})	0.00718 ± 0.00019

Constants for the Delayed Neutron Precursors

grp #	β_i	λ_i [1/s]	β_i/β
1	0.00025	0.01334	0.03477
2	0.00125	0.03269	0.17385
3	0.00123	0.12071	0.17107
4	0.00285	0.30302	0.39638
5	0.00119	0.85105	0.16551
6	0.00042	2.85707	0.05841
sum	0.00719		

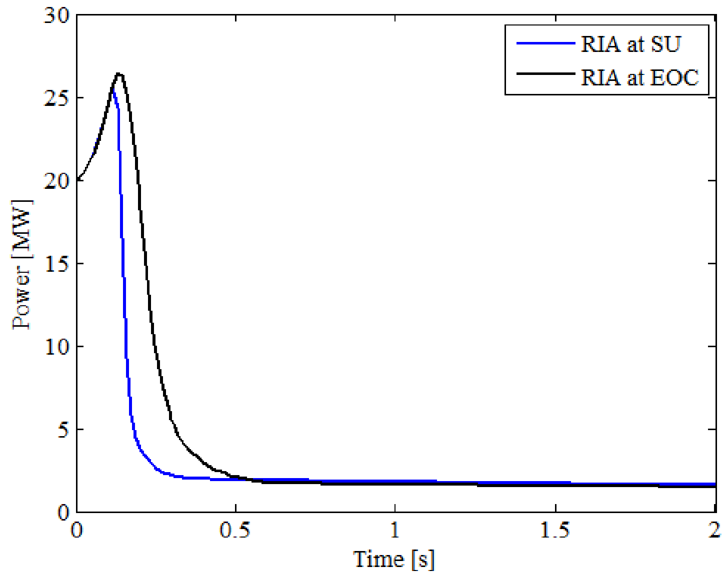
Automation of S-K Correlation Integration



Simulated Transient Cases

Reactivity Insertion Accident (RIA)

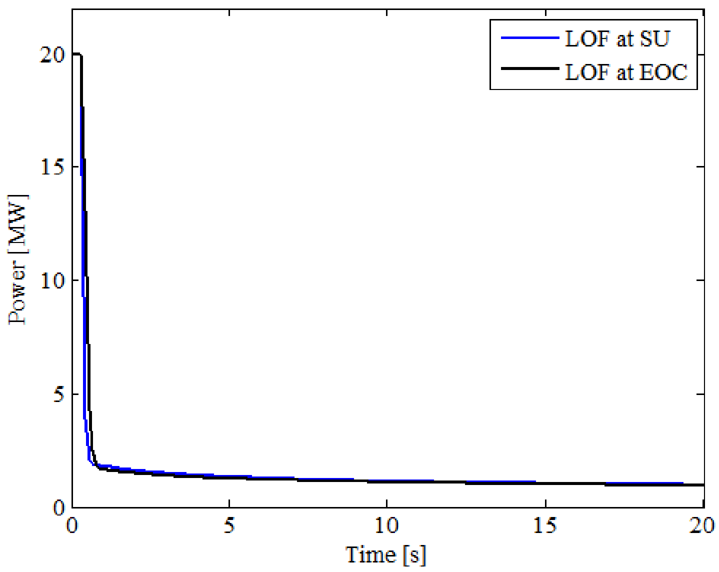
- Positive reactivity insertion in the core that may be caused by experiments removed from the core



The power response during RIA transient.

Loss of Flow Accident (LOFA)

- A core heat up due to malfunction of the cooling system even if the reactor power is operating at nominal value

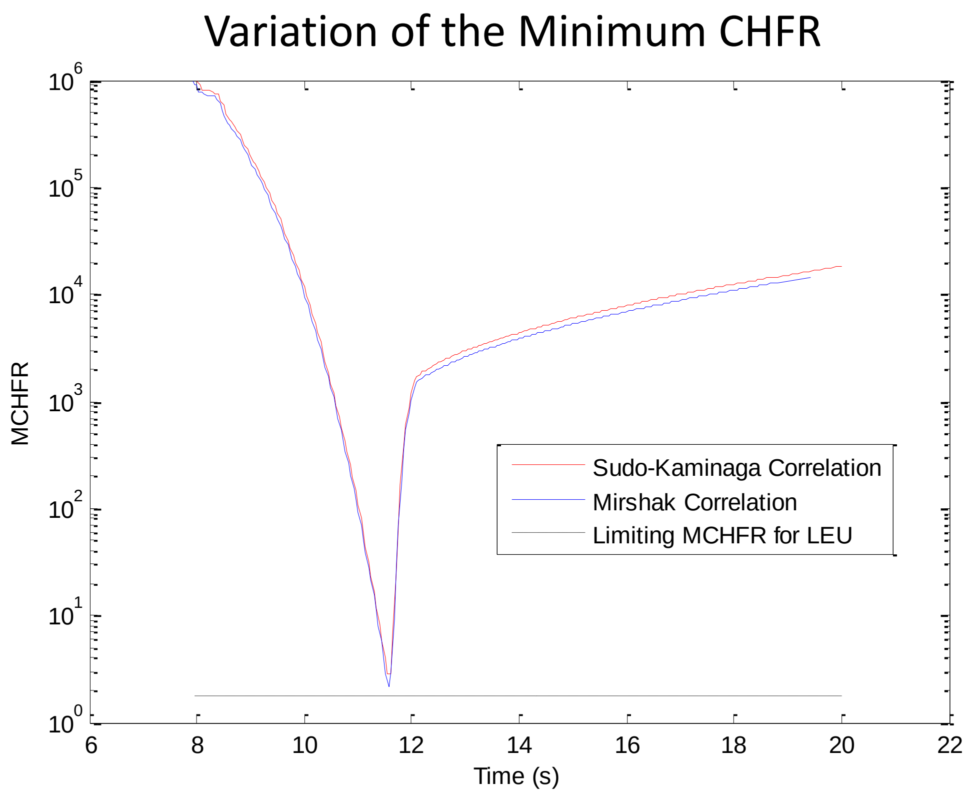


The power response during LOFA transient.

Case 1: Small Reactivity Insertion Accident

<u>Hypothetical Conditions:</u>

- Initiating Power: 2 W
- Reactivity insertion at slow ramp rate: $0.1/s
- The reactor scram occurs at 120% full power (24 MW)
- Time delay constant is 25 ms before control rods start to move
- Control rod moving rate is 1.2 m/s after the scram
- Reactivity feedback coefficients are neglected

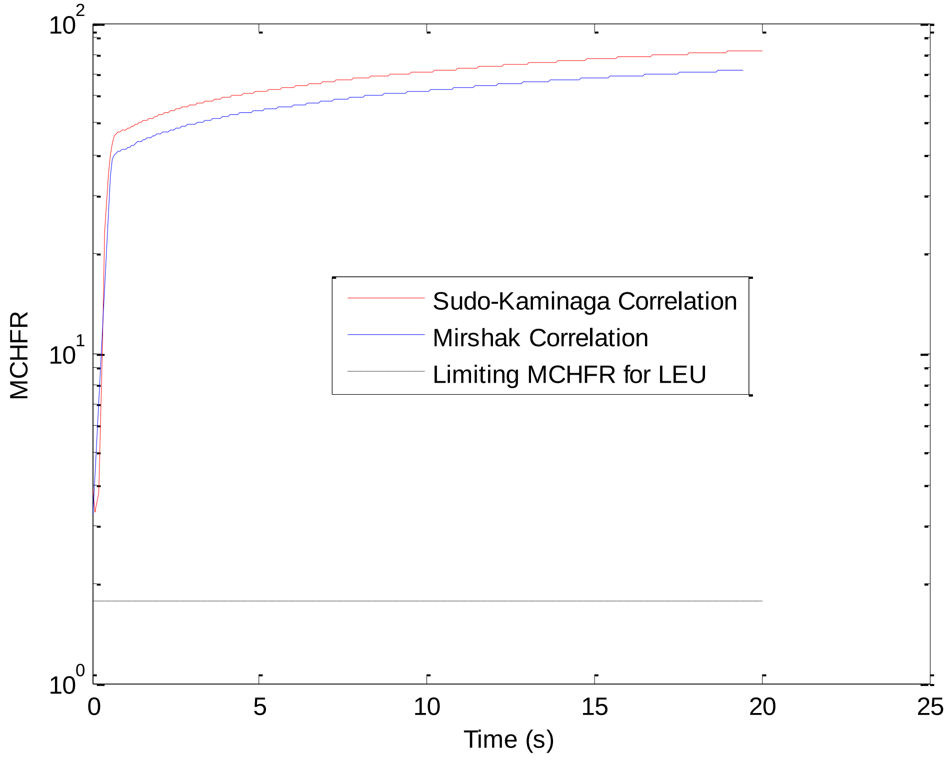


Case 2: Large Reactivity Insertion Accident

Hypothetical Conditions:

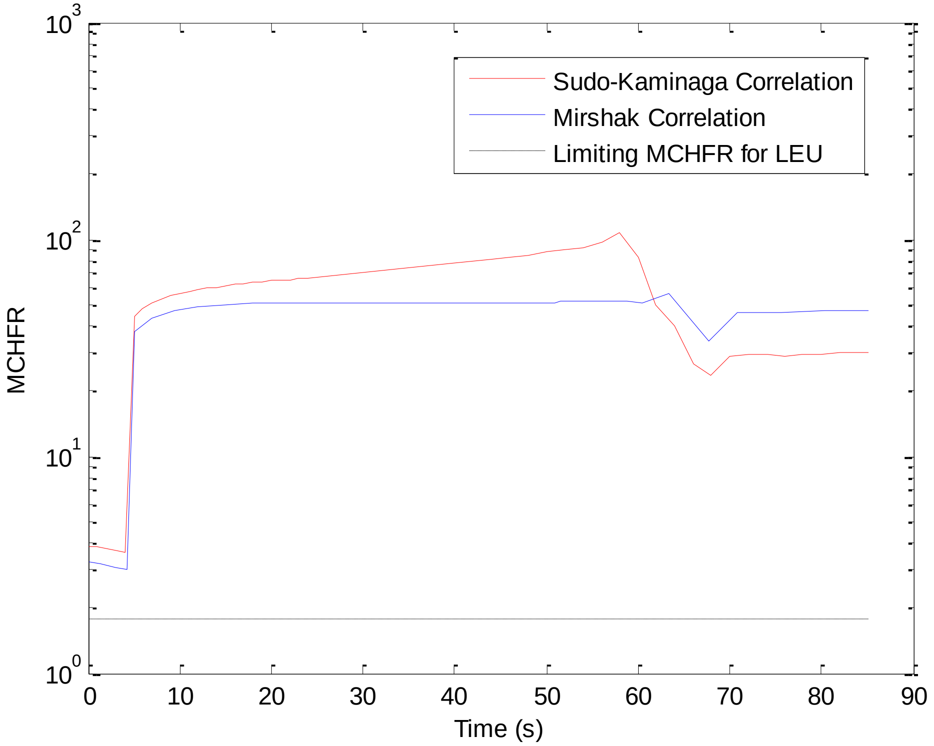
- Initiating Power: 20 MW (full power)
- Reactivity insertion: $1.5 in 0.5 s
- The reactor scram occurs at 120% full power (24 MW)
- Time delay constant is 25 ms before control rods start to move
- Control rod moving rate is 1.2 m/s after the scram
- Reactivity feedback coefficients are neglected

Variation of the Minimum CHFR



Case 3: Slow Loss of Flow Accident

Variation of the Minimum CHFR

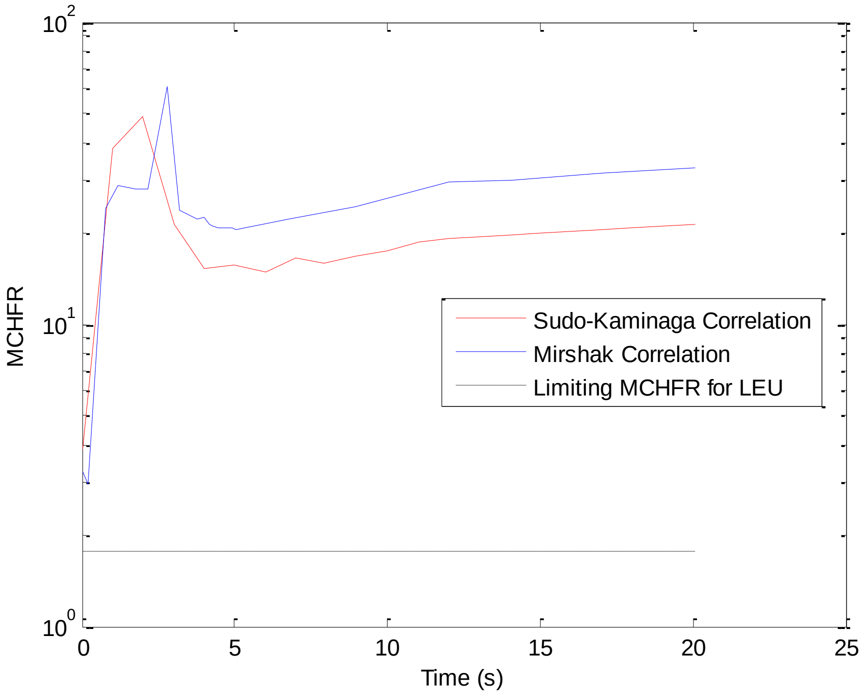


Hypothetical Conditions:

- Initiating Power: 20 MW (full power)
- The flow decay due to pump coast down is modeled as an exponential *exp*(-t/T) decrease with a assumed period T = 25 s.
- The reactor scram occurs when the flow is reduced by 15%.
- Time delay constant is 0.2 s before control rods start to move.
- Control rod moving rate is 1.2 m/s after the scram
- Reactivity feedback coefficients are neglected.

Case 4: Fast Loss of Flow Accident

Variation of the Minimum CHFR



<u>Hypothetical Conditions:</u>

- Initiating Power: 20 MW (full power)
- The flow decay due to pump coast down is modeled as an exponential *exp*(-t/T) decrease with a assumed period T = 1 s.
- The reactor scram occurs when the flow is reduced by 15%.
- Time delay constant is 0.2 s before control rods start to move.
- Control rod moving rate is 1.2 m/s after the scram
- Reactivity feedback coefficients are neglected.

Summary on All Cases

		Sudo-Kaminaga		Mirshak	
#	Case	MCHFR	Times (s)	MCHFR	Times (s)
1	SRIA	2.81	11.55	2.17	11.57
2	LRIA	3.29	0.10	2.50	0.14
3	SLOFA	3.63	4.00	2.99	4.28
4	FLOFA	3.85	0.01	2.80	0.38

All MCHFR values are above 1.778 (Thermal Limit)

Conclusions

- The results predicted by the Sudo-Kaminaga correlation has slightly larger margins than those by the Mirshak correlation for all the tested cases most of the time.
- For the RIA cases, the curve for the Sudo-Kaminaga correlation is slightly above the curve for the Mirshak correlation. For the LOFA cases, the Sudo-Kaminaga correlation starts off being above the other curve until it drops downward and remains below the Mirshak curve until the end of the runtime.
- The minimum CHFR, as calculated from the Sudo-Kaminaga correlation, is clearly above the recommended minimum CHFR, indicating that the fuel in the reactor will be virtually guaranteed (at least 99.9%) to not reach CHF.
- In short, the Sudo-Kaminaga correlation supports that the fuel in the reactor is thermal-hydraulically within safety limits and provides enough margin of safety while having a more compatible geometry compared to the Mirshak correlation.