Current Status of Severe Accident Analysis for Korean Sodium-cooled Fast Reactor

International Workshop on
"Prevention and Mitigation of Severe Accidents in Sodium-cooled Fast Reactors"

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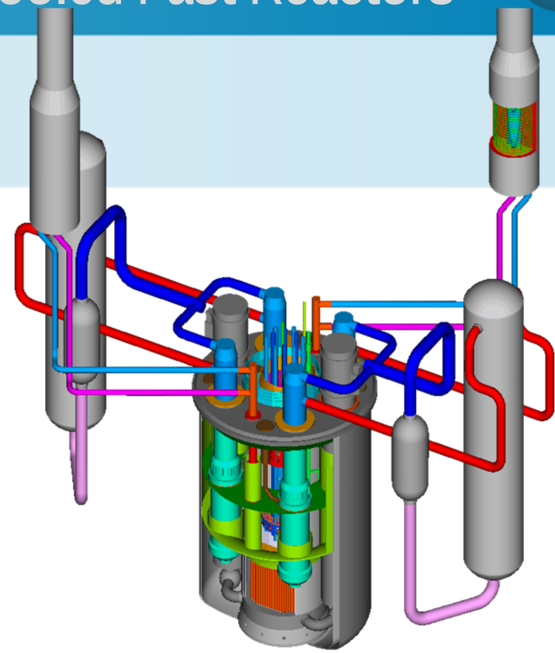
Seok Hun Kang

KAERI
Korea Atomic Energy Research Institute



Outline

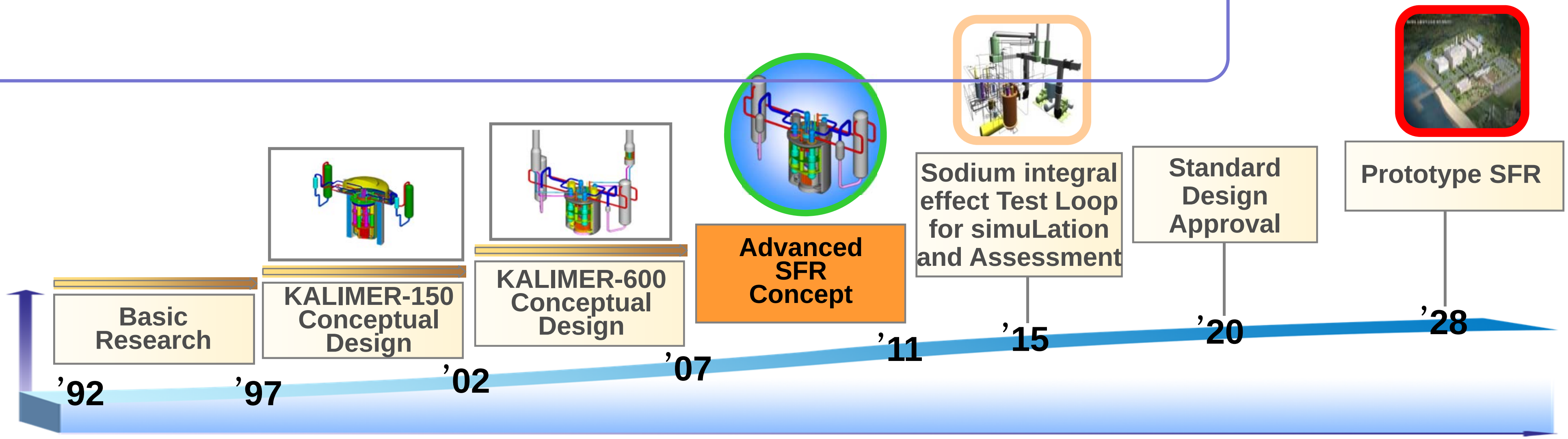
SFR Technology Development Status

- The design concept of KALIMER, Korea Advance Liquid Metal Reactor, has been developed under the national R&D program since 1992.
- A systematic SFR development program was launched in 1997.
  - The conceptual design of KALIMER-150 was developed through the collaboration with General Electric in 2002.
- The development of a mid-sized KALIMER-600 concept was followed, based on the KALIMER-150 experience.
  - The conceptual design was completed in 2007
- Now, projects are under going to develop a prototype SFR which is scheduled to be constructed by 2028.

◆ Proliferation resistant core without blankets

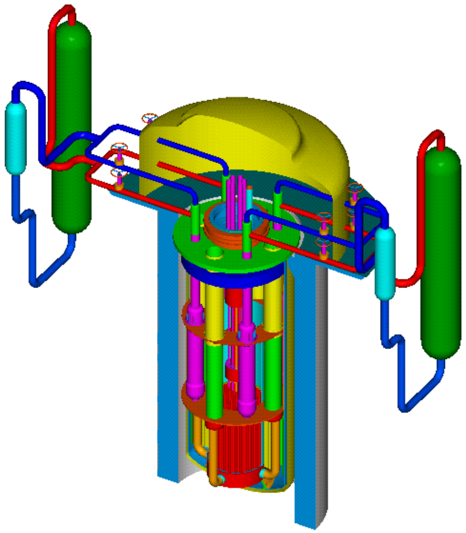
◆ MA bearing metal fuel

◆ Enhanced safety with passive systems



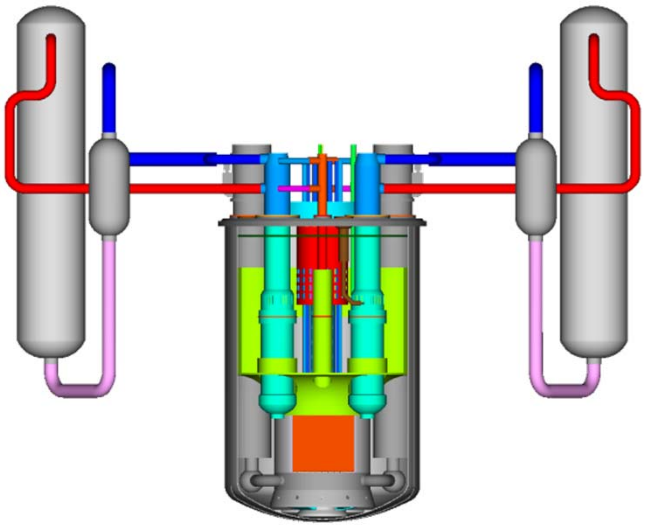
Conceptually designed KALIMERs

KALIMER-150



- 150 MWe, Pool-type Reactor
- Fuel : U-Zr -> U-TRU-Zr
- Core I/O Temp. : 365/510℃
- DHR System : PDRC
- 2-loop IHTS/SGS
- Net Efficiency : 38.8%

KALIMER-600



- 600 MWe, Pool-type Reactor
- Fuel : U-Zr -> U-TRU-Zr
- Core I/O Temp. : 365/510℃
- DHR System : PDRC/ADRC
- 2-loop IHTS/SGS
- Net Efficiency : 38.8%

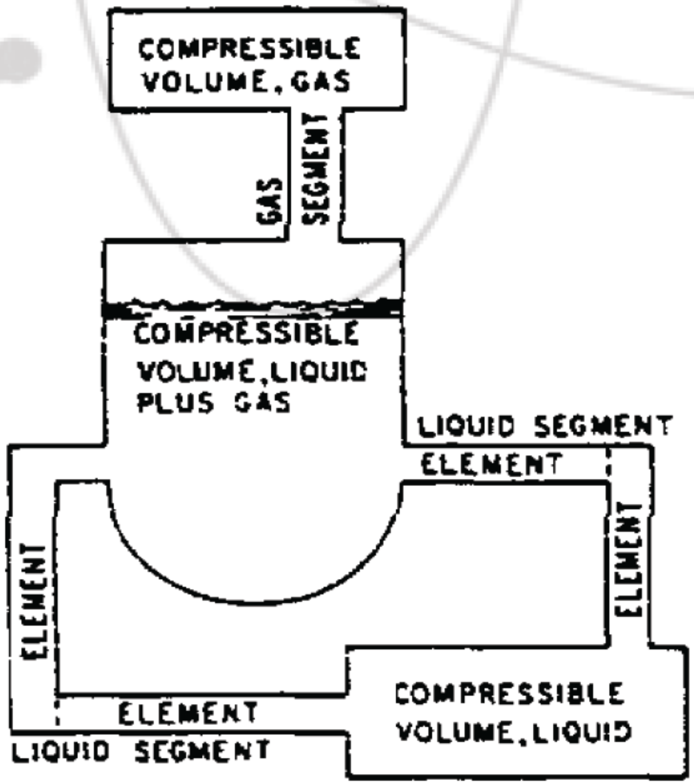
ATWS analysis using SAS4A/SASSYS-1

Objectives

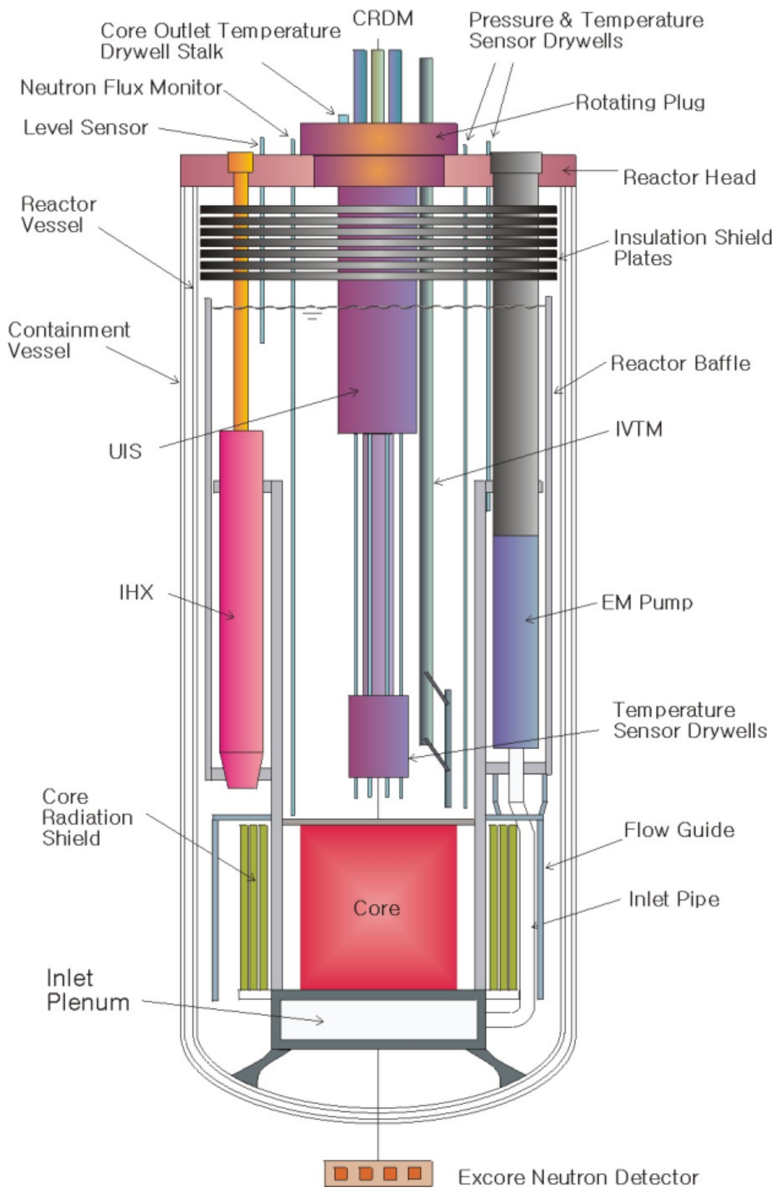
- The ATWS analysis was conducted to examine the tendency of metal fuel to act as a fuse, to avoid an energetic accident sequence that challenges system integrity.

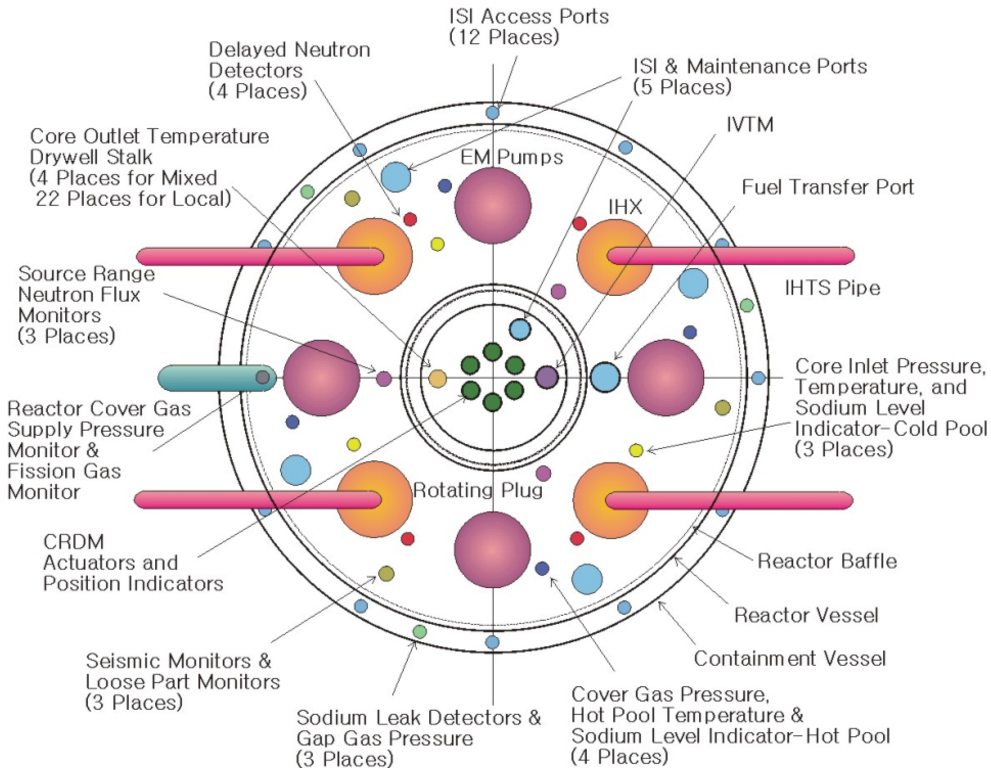
Description

- The SAS4A/SASSYS-1 computer code developed at ANL in the IFR program for transient analysis of liquid metal cooled reactors was used.
- The core models provide the capability to analyze the initial phase of core disruptive accidents through coolant heatup, boiling, fuel element failure, fuel melting and relocation.

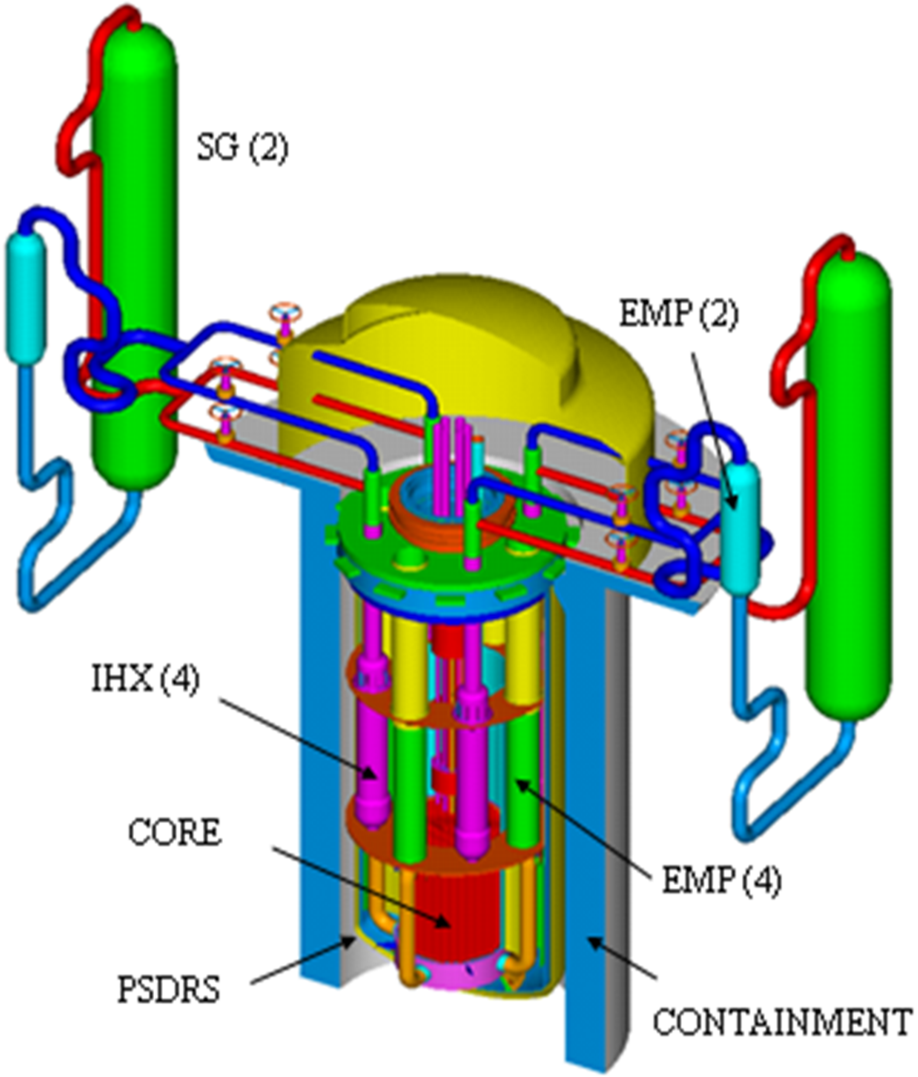


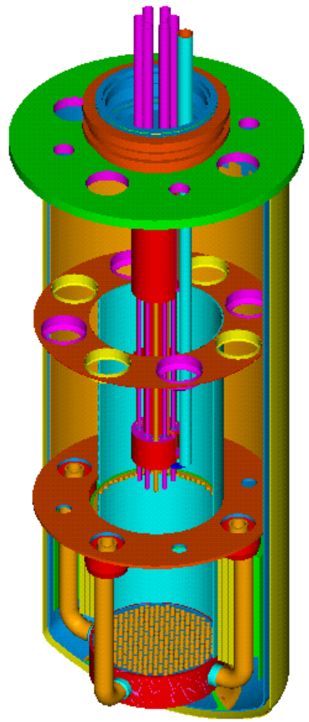
Analysis for KALIMER-150





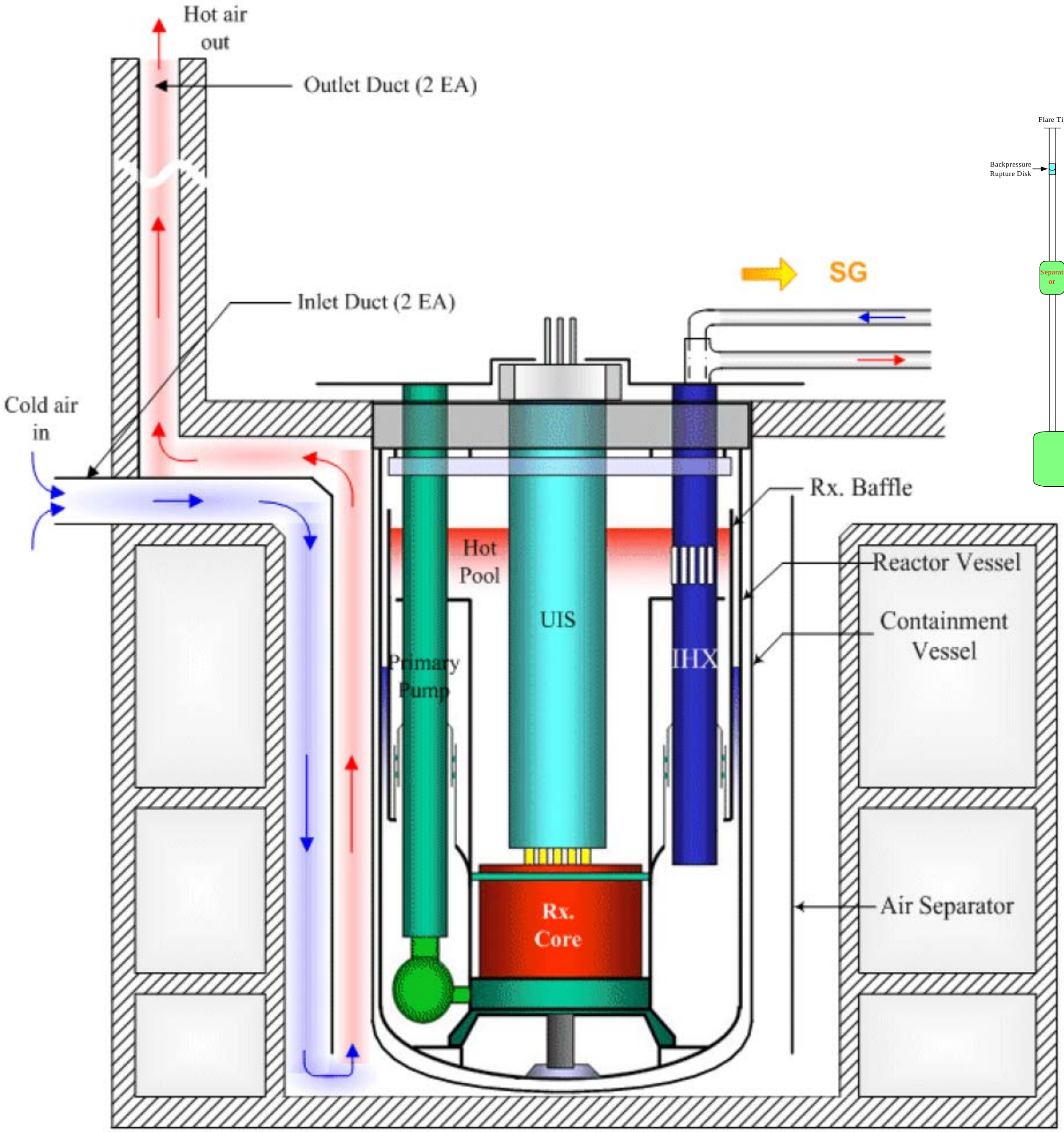
- Similar size in power capacity to the prototype SFR which is in design process and scheduled to be constructed by 2028

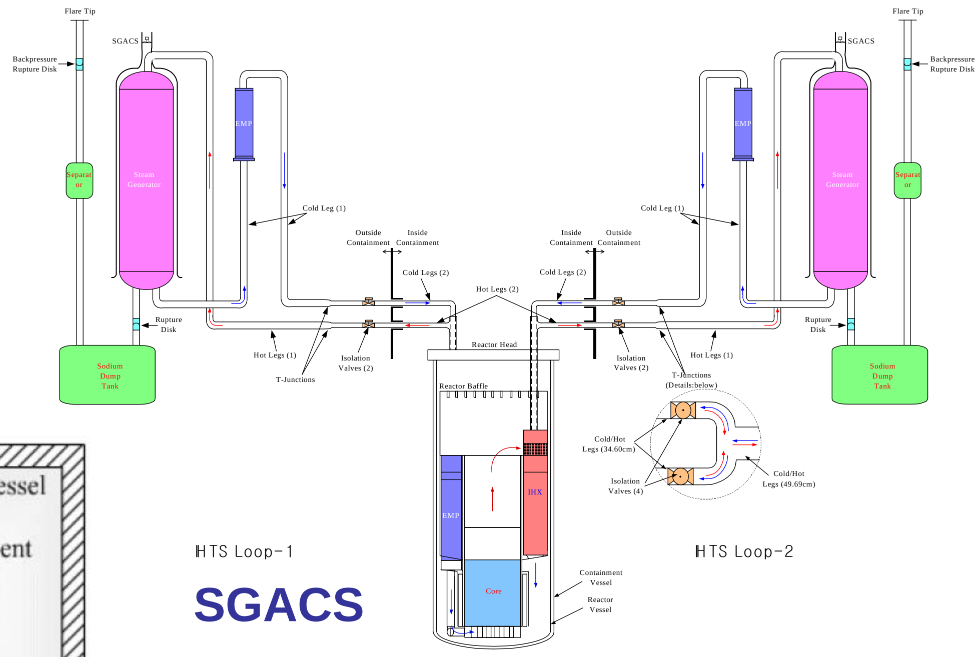




- Flow rate: 2143kg/s
- Core I/O Temp. : 365/510℃
- Core outlet P : 0.101 MPa
- 4 IHXs, 4 EMPs
- Cover gas: He
- Annular space filled gas: Ar

RHRS





SGACS

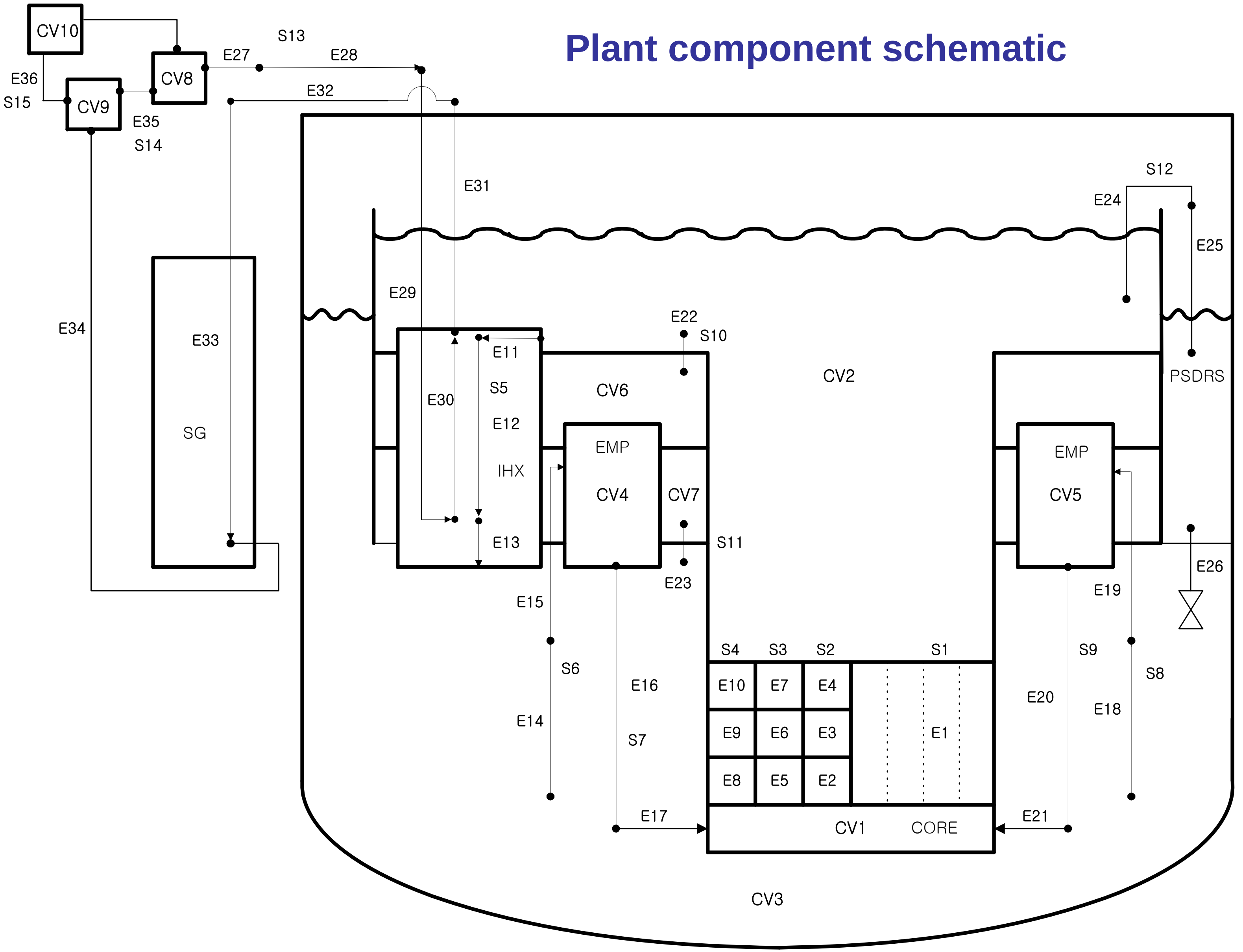
(Steam Generator Auxiliary Cooling System)

PSDRS

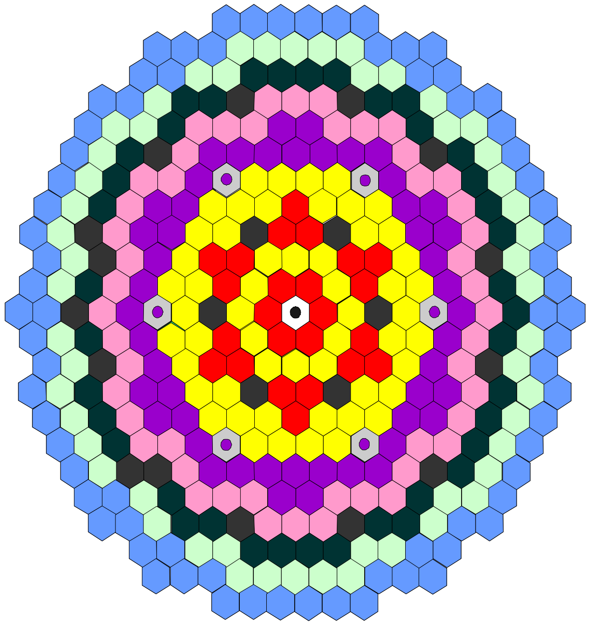
(Passive Safety Decay heat Removal System)

Nodalization

Plant component schematic



Core modeling



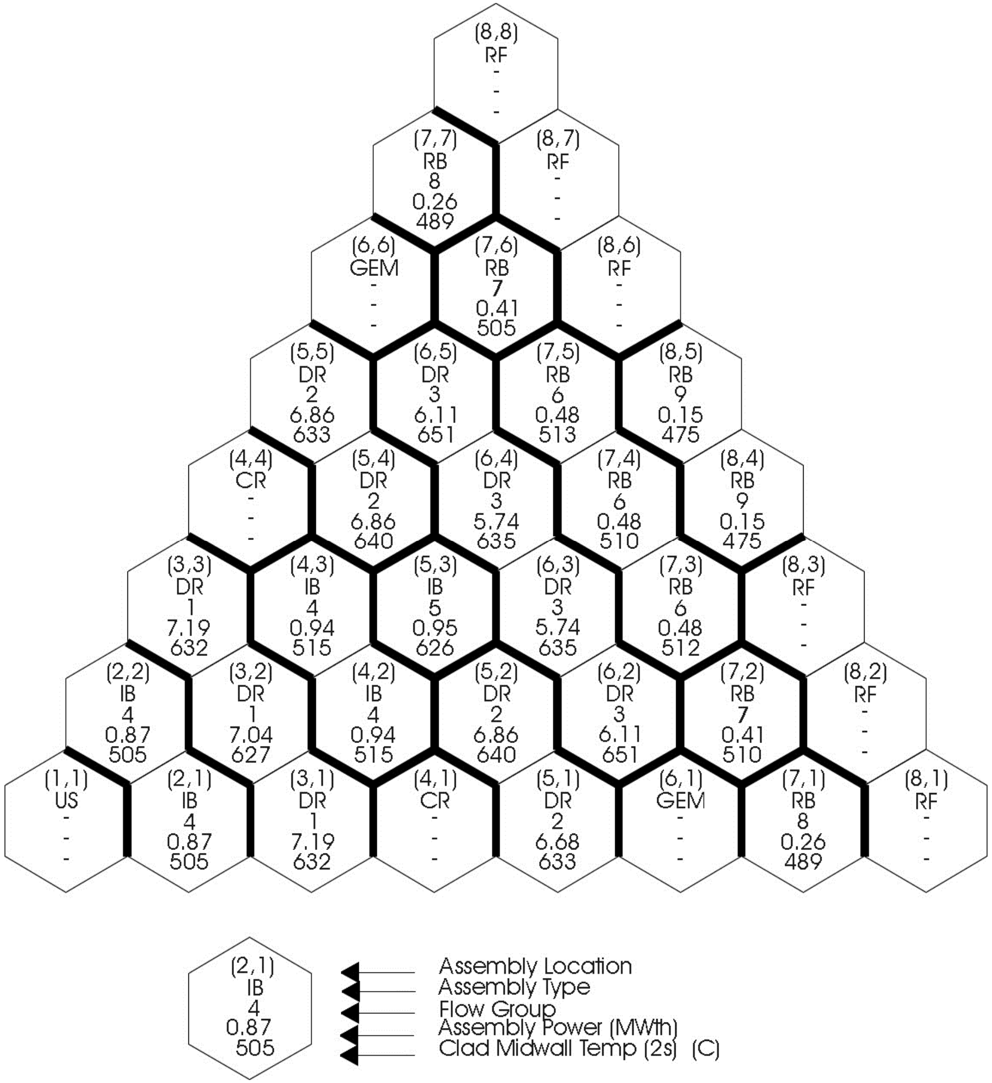
Driver Fuel	54
Internal Blanket	24
Radial Blanket	48
Control Rod	6
USS	1
GEM	6
Reflector	48
B_4C Shield	54
IVS	54
Shield	72
Total	367

[Reactor core loading pattern]

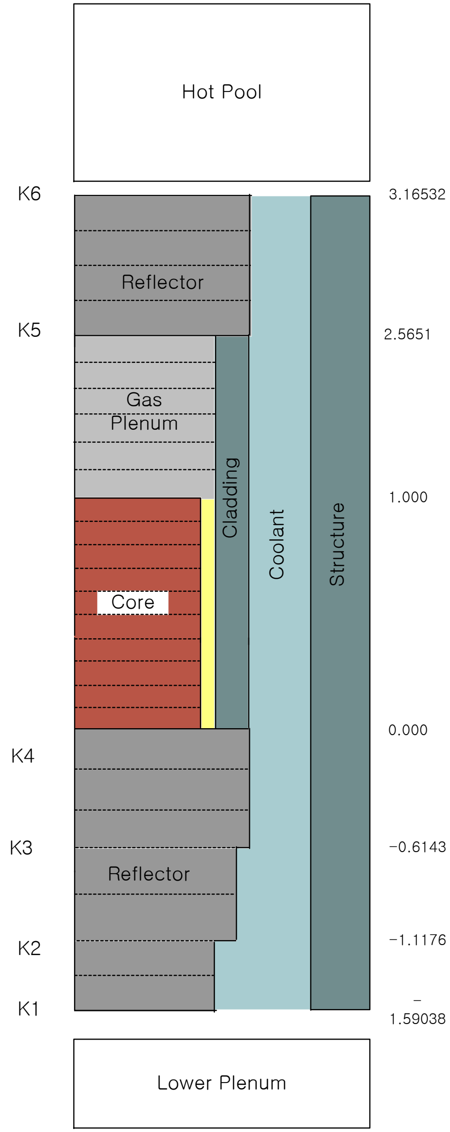
Core configuration	heterogeneous
Metallic fuel design	U-TRU-10%Zr
No. of assembly	367
Enrichment, %	30
Breeding ratio	1.05
Active core height, mm	1,200
Axial blanket	No
Structural material	HT9
Specific Power of fuel, W/kg	60,000
Core diameter, mm	3,447
Fuel / blanket pins per assembly	271 / 127
Driver / blanket assemblies	48 / 18
Reactor core I / O temp., C^o	368 / 512
Burnup of driver fuel, GJ	75,504,753
Prompt neutron lifetime, s	2×10^{-7}
Delayed neutron fraction	0.035
Fuel mass, kg	9,976

Layout of core assignment

(1/6 assembly)



Axial zones of core channel

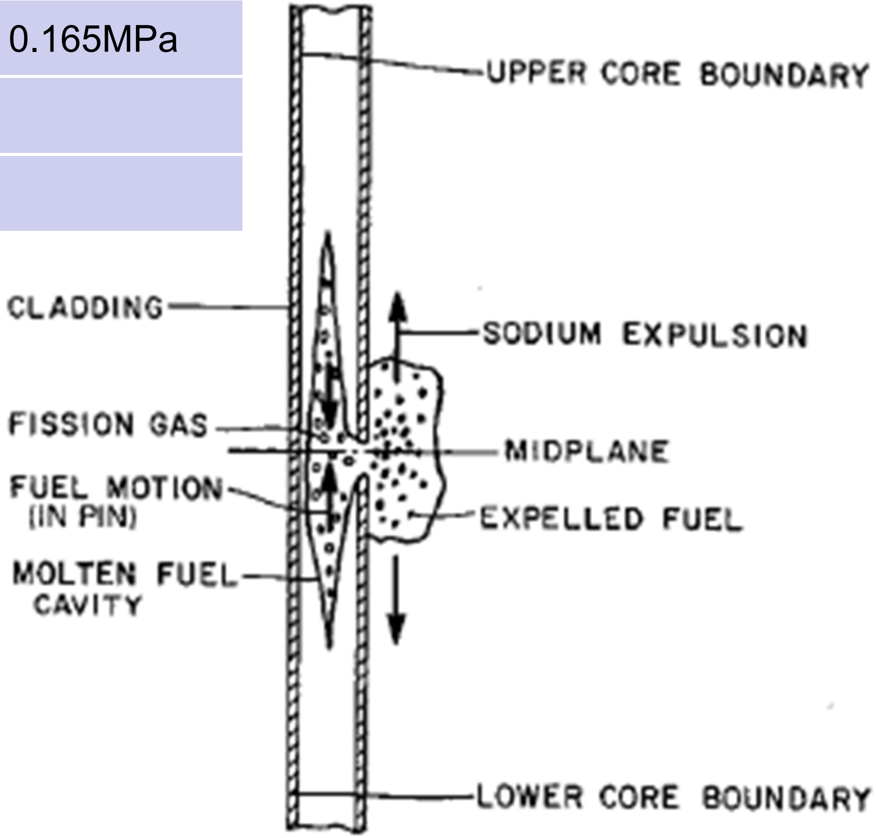


- Single pin geometry
- Axial zones: fuel, sodium bond, gas plenum, reflector

Safety criteria

[KAERI/TR-3370/2007]

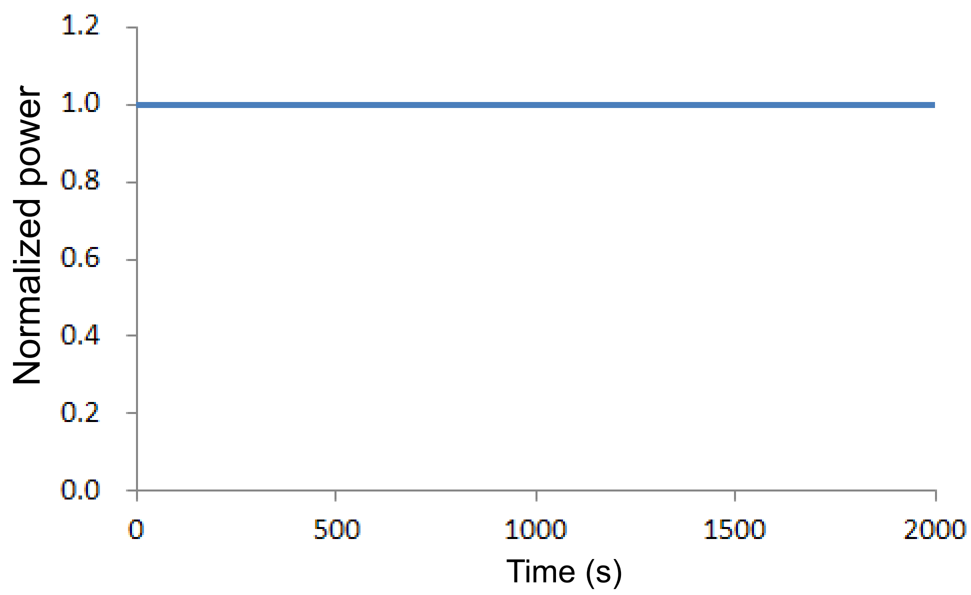
	limit	criteria
Reactor structure	760℃	✓ maintain its structural integrity ✓ average core outlet coolant temperature
Coolant	940℃	✓ sodium boiling during natural circulation, 0.165MPa
Cladding	700℃	✓ eutectic point of HT9
Fuel	955℃	✓ fuel melting



Analysis results

Calculation at full power steady state condition

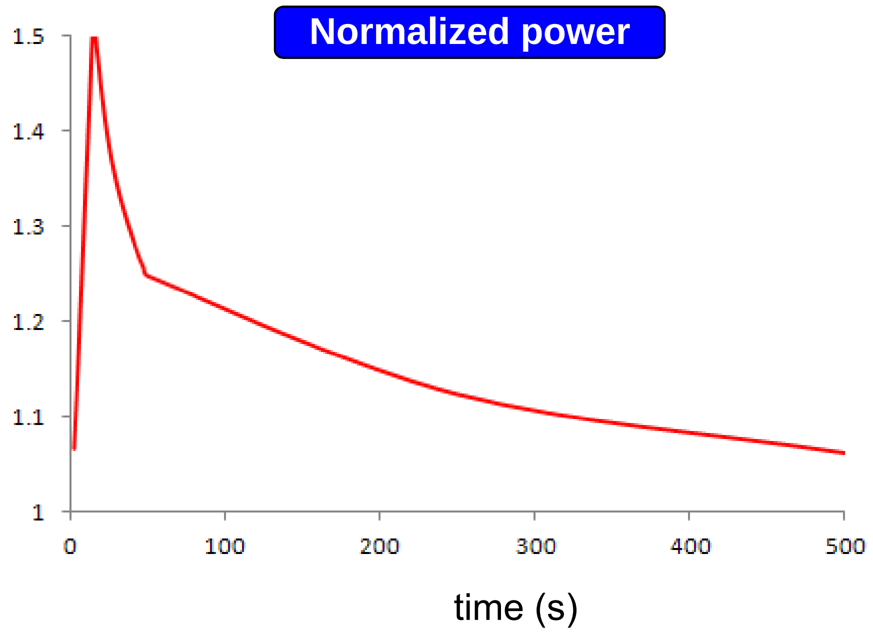
Plant parameters	Design	SAS4A/SASSYS-1
core power, MW_t	392	392
primary flow rate, kg/s	2143	2143
core inlet temp., ℃	386	385
core outlet temp. ℃	530	530
cover gas pressure, Pa	10133	10133
cold pool level, m	10.6	10.7
hot pool level, m	15.6	15.7
pump head, m	83.6	83.6



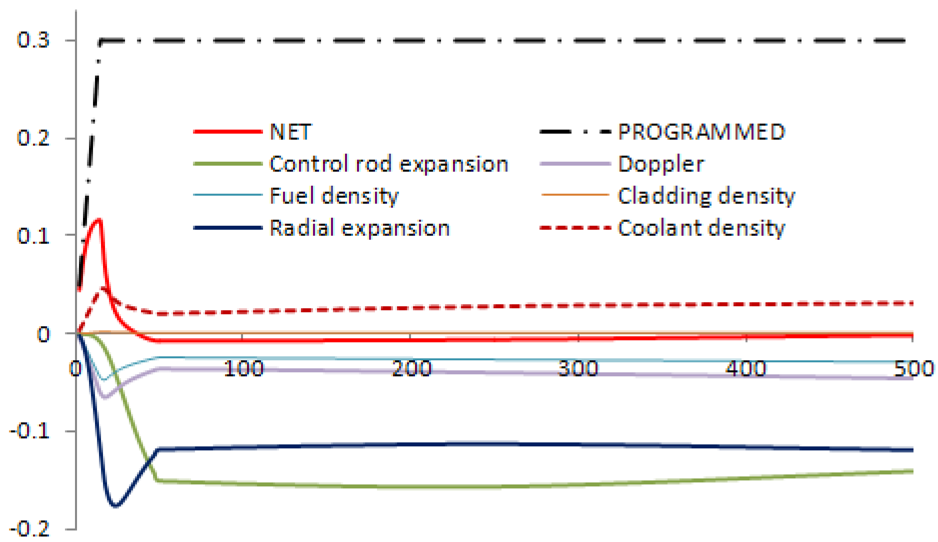
✓ Code predicted well the steady state conditions of the plant design.

Unprotected Transient Over Power (UTOP)

- Assumption that all the control rods are accidentally removed
- Reactivity insertion rate is set to 2 ₡/s for 15 seconds, the maximum speed of shim motors withdrawing the control rods.

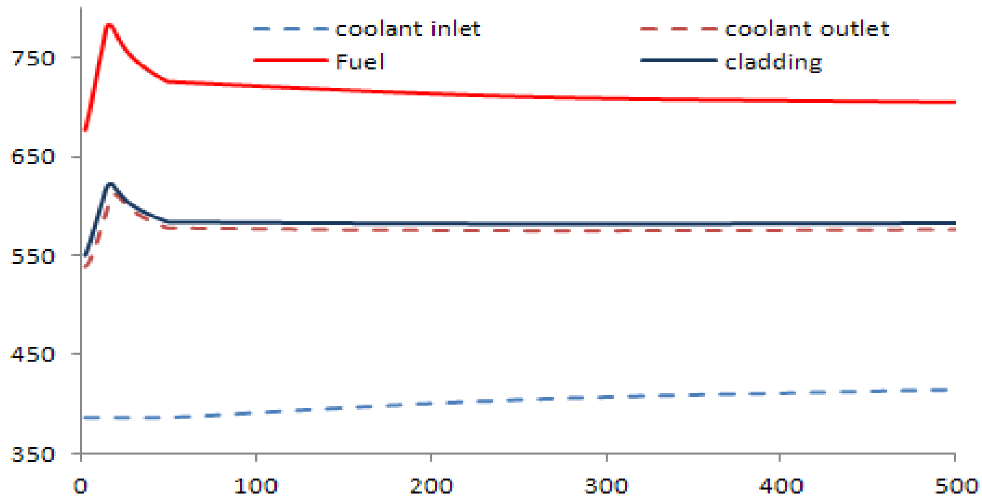


Reactivity, $



time (s)

Temperature, ℃



time (s)

Assessment of CDA (Core Disruptive Accident) Energetics

Objective / Work done

- Analysis was performed for the KALIMER-150 core during super-prompt critical power excursion induced by reactivity insertion.
- Analysis code was developed to investigate energy release during the accident.
- Based on the mathematical formulations developed in the framework of the Bethe-Tait method.

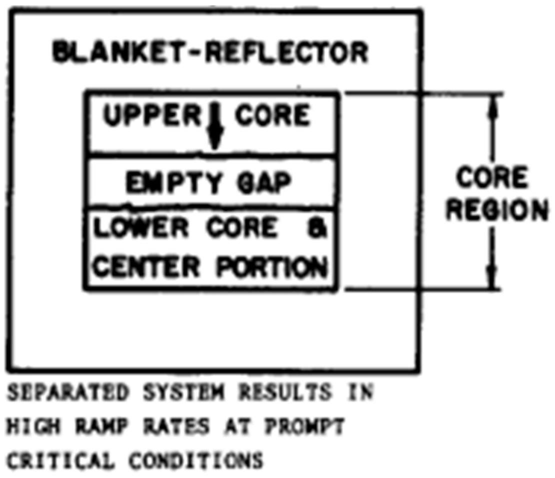
Historical background

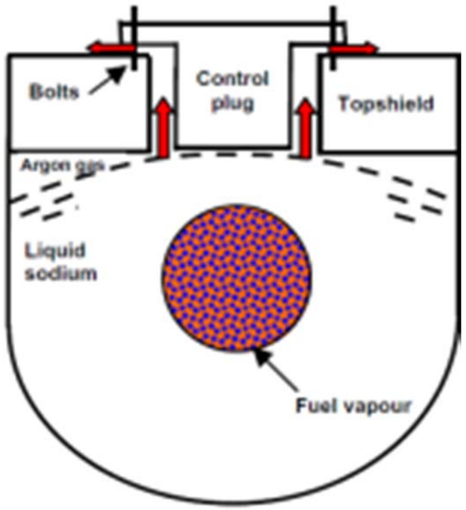
- Analytic method of CDA was originally developed by Bethe and Tait (1958).
- Further elaborated by Jankus (1962), applied to analyze EBR-II, Fermi Reactor.
- The method evolved over 1960s to include Doppler reactivity effect and more realistic equation of state (Nichoson; 1964, Wolfe ; 1963, Hicks ; 1965, Meyer ; 1968).
- Mechanistic approach has been developed since early 1970s to analyze comprehensive phenomena during CDA sequences ; SAS, VENUS, SIMMER.

Mechanism

1. Sodium coolant boils out of the core.
2. Fuel melts.
3. Fuel moves from its normal configuration into one of higher reactivity for some reason.
4. The reactivity increases above prompt critical, causes a power excursion to develop.
5. The power excursion is ultimately terminated by disassembly of the core.

   The forces effecting the disassembly are high pressures produced in the core by the power excursion.



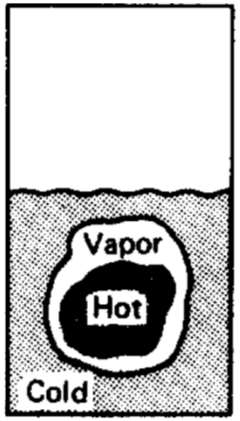


Source: Fauske; 1977, Velusamy; 2011

Formulation: Bethe-Tait assumption

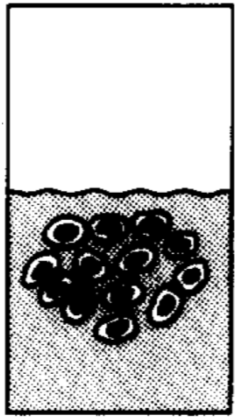
Phase I

- There is void space left in the core.
- Core expansion takes place internally
- Pressure does not become high until there has been sufficient thermal expansion of the molten core to fill the void space.



Phase II

- Pressure rises rapidly.
- Core disassembly occurs quickly.



Source: Nicholson; 1964, Fauske; 1977

Equations

$$\frac{d^2Q}{dt^2} = \frac{k-1-\beta}{\ell}\frac{dQ}{dt}$$ ~ ***reactor kinetics equation***

$Q(t)$ ~ *generated energy*

β ~ *delayed neutron fraction*

ℓ ~ *prompt neutron live time*

$$k = k_0 + k_i(t) + k_d(t)$$

$k_0 = 1 + \beta$ ~ *initial prompt critical reactivity*

$k_i(t) = \alpha t$ ~ *reactivity insertion*

$k_d = \int \rho(\vec{r},t) u(\vec{r},t) \cdot \nabla D(\vec{r}) dV$ ~ *disassembly reactivity*

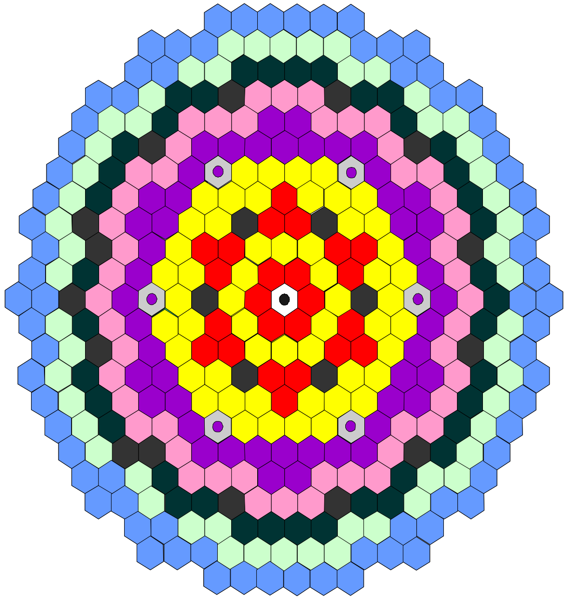
ρ ~ *density of core material*

u ~ *material displacement during disassembly*

D ~ *reactivity change which would occur if a unit volume of core material were to be removed from the core*

Core modeling

Calculation for the KALIMER-150 core



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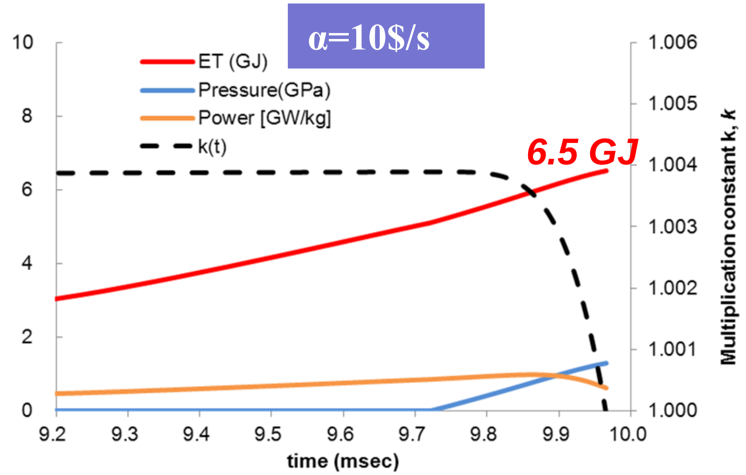
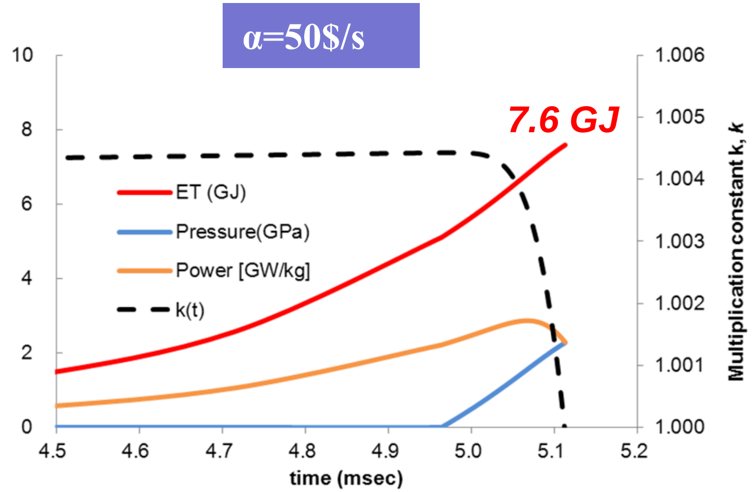
	Total	367

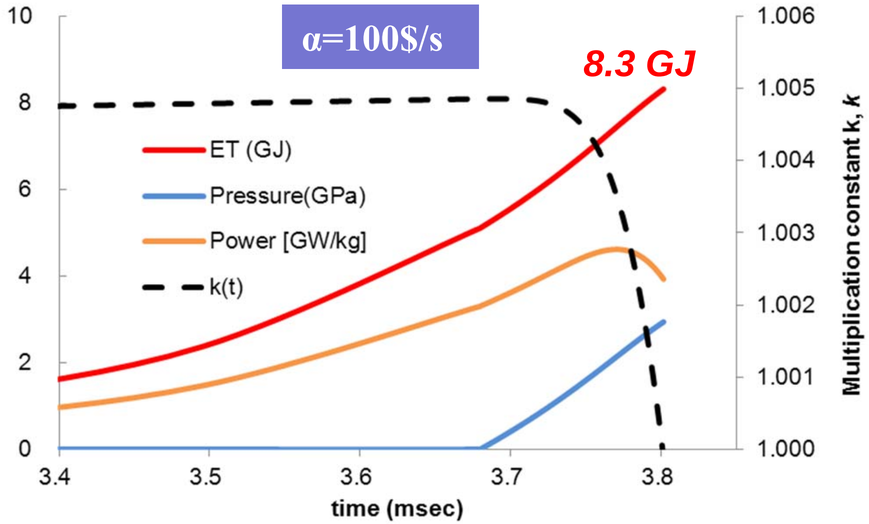
Modeling parameters

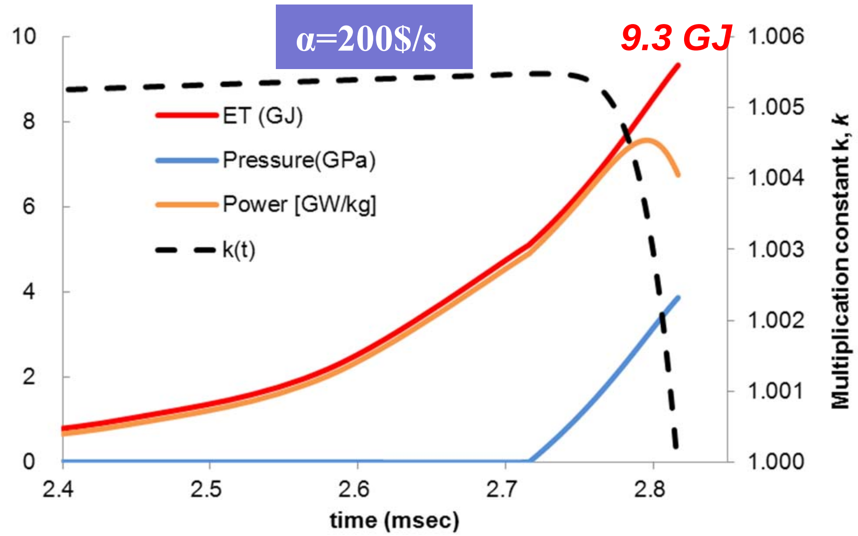
Density of molten liquid core, kg/m^3	7,133
Density of solid fuel, kg/m^3	15,850
Radius of spherical core, m	0.69
Meltdown fuel(whole core), kg	9,976

Calculation results









Summary

- The SFR technology development status was introduced.
- The R&D activities on the severe accident analysis were introduced.
- ATWS analysis using SAS4A/SASSYS-1 code was conducted, and investigated system integrity.
- Now a research plan is scheduled to develop new severe accident simulation code.
- Assessment of CDA energetics using formulated code was conducted.
- Further work will be performed to elaborate the code and develop it to the mechanistic approach tool.