Safety Design Concept and related Safety R&D in Korea

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Outline

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II System Analysis Code Development

III Safety Analysis

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I Safety Design Approach

I.1 Safety Philosophy

❖ Design Defense-in-Depth (DID)* is implemented by

– Engineered SSCs, which constitute a set of radio nuclide transport barriers

– Engineered Safety Features (ESFs) to protect the integrity of these barriers

- Reactivity control
- Decay heat removal
- Radioactivity confinement

❖ Scenario Defense-in-Depth* is defined in terms of a scenario framework

– Prevention of abnormal operation and failures (Design Simplicity & Robustness)

– Control of abnormal operation and detection failures (Accident Prevention)

– Control accidents (Redundancy & Diversity of ESFs, Accident Protection)

– Control of severe conditions (Accident Mitigation)

– Mitigation of radiological consequences (Emergency Planning or SA Termination)

❖ Deterministic approach harmonized by Probabilistic approach

– Single failure criteria

– CDF quantification

– Probabilistic decision making

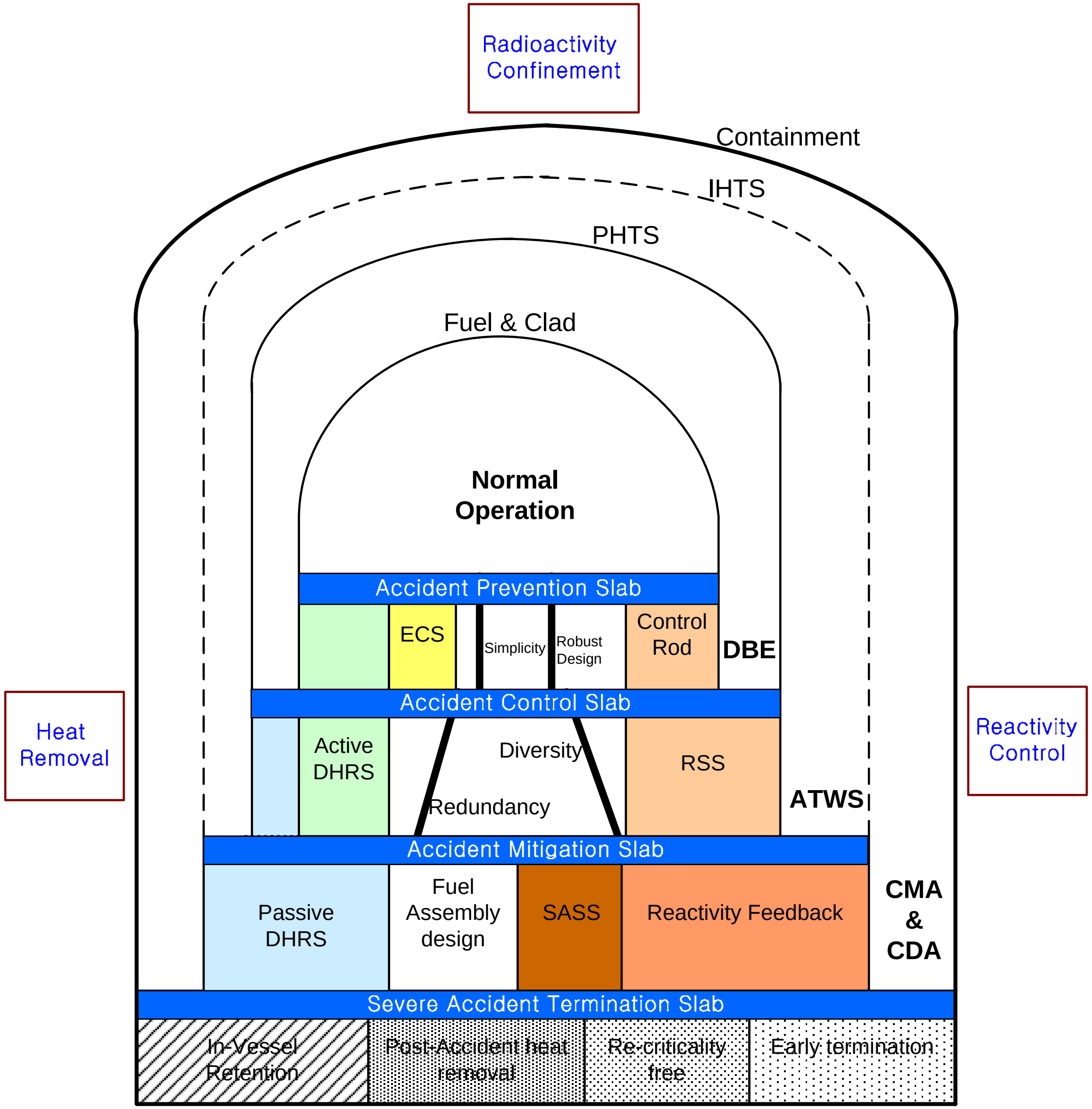
* Defined by K.N. Fleming, 2002

I.2 Implementation of DID

❖ DID concept & Structure implemented in KALIMER design

❖ 4-levels of safety slabs

❖ Normal and abnormal conditions can be divided with LCOs



I.3 Safety Goals

❖ The general target is to satisfy the safety goals for the Generation IV nuclear systems

❖ Gen IV nuclear energy systems will excel in safety and reliability

– Assure the safety and reliability equivalent at least to those of an advanced thermal reactor which is acceptable by the public

❖ Gen IV nuclear energy systems will have a very low likelihood and degree of reactor core damage

– CDF < 10^{-6}/RY

❖ Gen IV nuclear energy systems will eliminate the need for offsite emergency response

– Minimized LERF < 10^{-7}/RY, containment integrity

– Termination of severe accidents

I.4 Safety Design Concept

❖ Inherent safety features

– Use of sodium as a coolant, metallic fueled core with fast spectrum provides superior heat transfer and inherent reactivity feedback characteristics

– These inherent features imbedded in design are the basis for achieving severe accident termination

- Sodium void coefficient
- In-pin fuel motion and relocation

❖ Passive Engineered Safety Features

– Passive DHRS provides superior reliability in accident control and mitigation

– The Self-Actuated Shutdown System is effective in control of severe conditions

❖ Reliable Active ESFs

– Redundancy and Diversity in Reactor Shutdown System and DHRS increase the reliability of ESFs

II Development of System Analysis Code

II.1 Background

- ❖ The KAERI is developing a system analysis code, MARS-LMR, for SFR application
  - – This code will be used as a basic tool in the design and analysis of future SFR systems in Korea
- ❖ The KAERI is concentrating on the verification and validation of the code models using available data
  - – The data on natural circulation and ATWS condition from EBR-II reactor have been evaluated with the MARS-LMR
- ❖ The validation with data from CEA launched PHENIX end-of-life (EOL) test is undergoing
  - – The KAERI joined Phenix EOL program to evaluate the capability and limitation of the MARS-LMR code

II.2 MARS-LMR Code

❖ The MARS-LMR is a liquid metal version of MARS code evolved from RELAP5/MOD3

– The models of equation of state (EOS), core pressure drop, heat transfer for SFR system have been reinforced for a sodium system

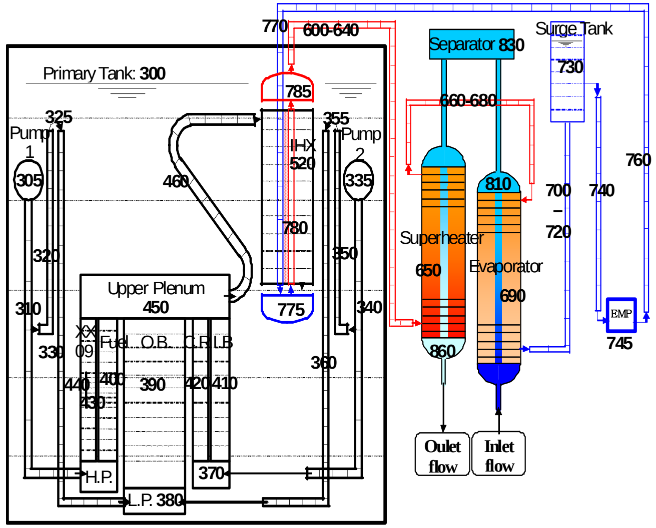
❖ The applicability of the code to a small SFR system has been evaluated with EBR-II data

– Three shutdown heat removal tests (SHRT) 17, 39, and 45 have been simulated

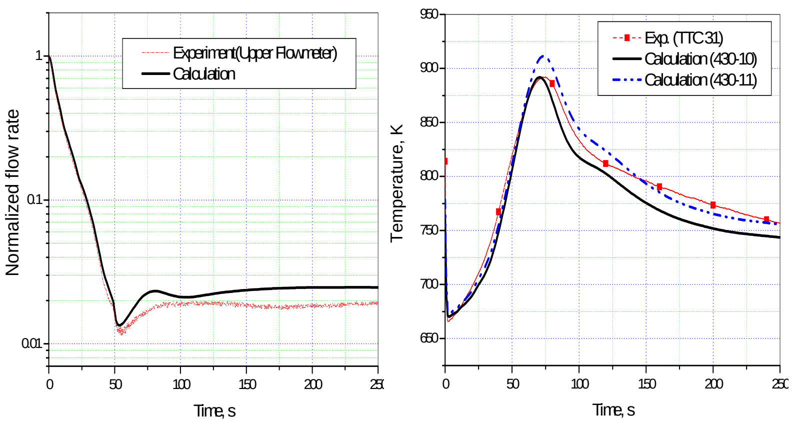
– Simulated results for the temperature and flow rate agreed well with the experimental data

❖ Pre-test analysis of natural circulation test of PHENIX EOL is being performed

– One-dimensional thermal-hydraulic behaviors for large pool design are analyzed



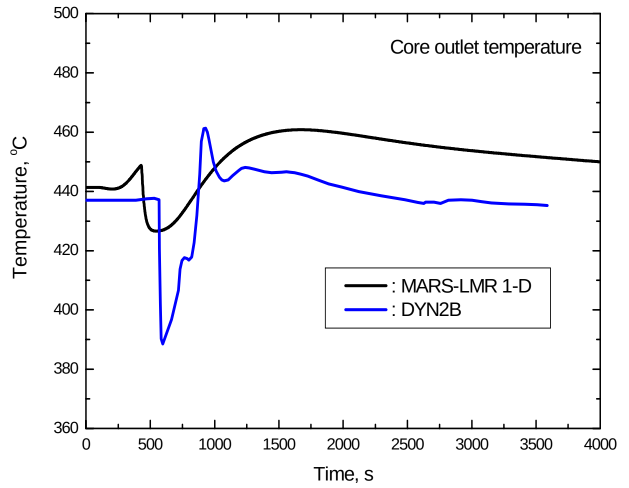
MARS-LMR Nodalization for EBR-II



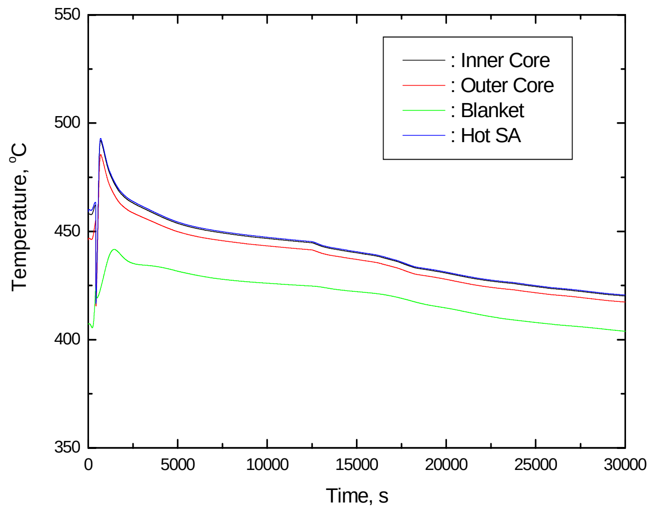
EBR-II SHRT-17 LOF analysis results

II.3 Pretest Analysis of PHENIX NC Test

- ❖ The main purpose of the present study is to evaluate the applicability of the MARS-LMR to a large pool-type reactor
  - – The preliminary calculation of steady-state and transient condition have been completed
- ❖ Higher core outlet temperature is predicted by MARS-LMR than DYN2B
  - – This is caused by the higher reactor power at the moment of reactor scram
- ❖ The trend of temperatures at subassembly outlets are reasonable but slightly over-predicted
  - – The predicted temperatures have similar trend to the measured SA outlet temperatures



Comparison of Core outlet temperature



SA outlet temperature by MARS-LMR

III

Safety Analysis

III.1 Safety Evaluation

❖ DBE analysis : MARS-LMR, MATRA-LMR/FB

– To assure safety margin provided by inherent safety features and ESFs

– DBE scope includes the following categories of events:

- Reactivity events : TOP
- Loss of flow type events : LOF
- Loss of heat removal events : LOHS
- Primary and secondary boundary failure : Vessel leak
- Local faults
- Others : Tube leak

❖ ATWS analysis : SSC-K

– To assure the inherent safety characteristics provided by the reactivity feedbacks imbedded in the design

- UTOP
- ULOF
- ULOHS

III.2.1 DBE Analysis – Scope and Assumptions

❖ Typical DBEs are analyzed by MARS-LMR
  – LOF, TOP, LOHS, Pipe Break, Vessel Leak, SBO

❖ All events are assumed to be occurred at the rated power and flow

❖ Reactor is scrammed by following conditions;
  – High power trip: 111 %
  – High core outlet temperature: 555 °C
  – Low pumping flow rate: 84 %
  – Low hot-pool level: 5 cm below normal level

❖ Decay heat model: ANS-79 (conservative)

❖ Pump trip is assumed to be occurred at 5 seconds after reactor scram

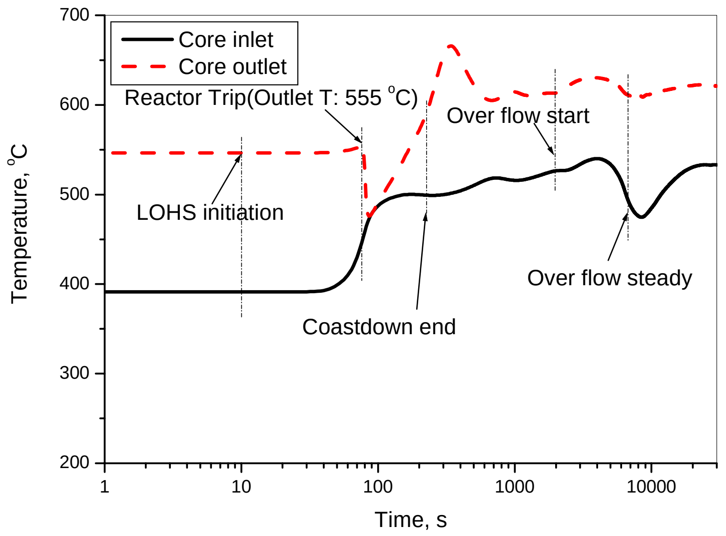
❖ Feedwater line isolation time is the same as pump trip

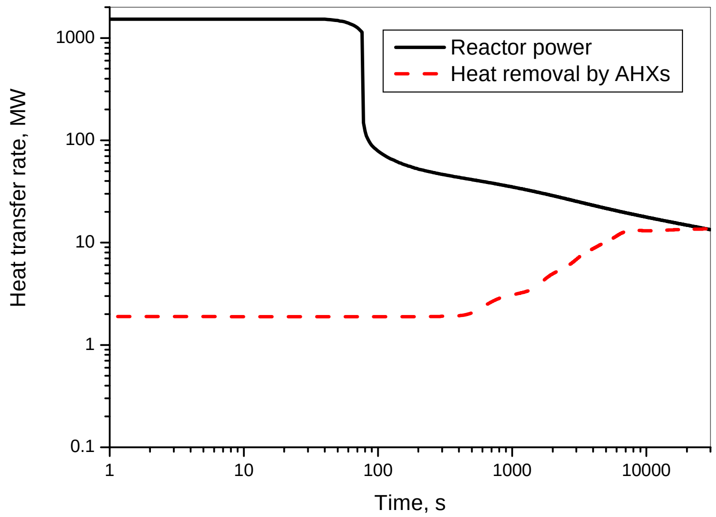
❖ Two independent PDRCs are available

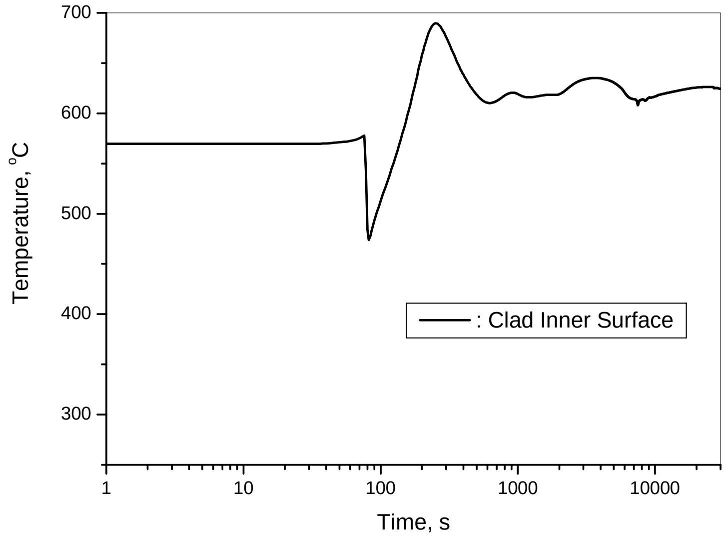
III.2.2 DBE Analysis – LOHS

❖ Event Sequence

- Accident initiation: 10 s (FW isolation)
- Reactor scram by High outlet T: 76.65 s
- PHTS Pump trip: 81.65 s
- SG dryout (IHTS sodium heated): ~20 s
- Pump Coastdown end: ~200 s
- IHTS $T_{cold} \cong T_{hot}$: ~85 s
- PDRC overflow start: ~2000 s
- Overflow quasi-steady: ~6800 s







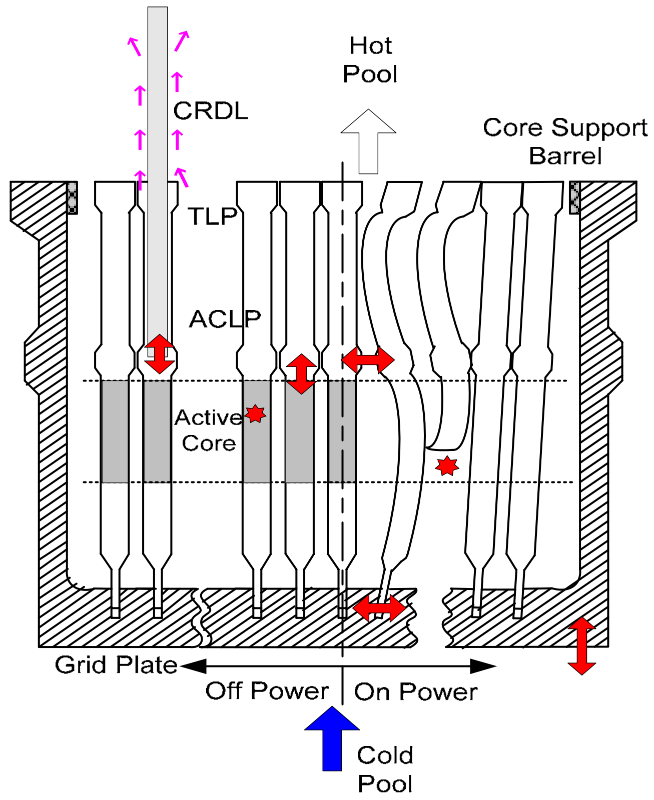
III.3.1 ATWS Analysis – Reactivity Feedback

❖ **Doppler and Sodium density**
- Detail-meshed reactivity coefficients

❖ **Fuel axial expansion**
- Free expansion before fuel-cladding contact
- Force-balance between fuel & cladding

❖ **Core radial expansion**
- Simple model: only thermal expansion of GP and ACLP
- Subassembly bowing model

❖ **CRDL expansion**
- Coupled with 2-dim. hot pool model
- Consider a reactor vessel expansion



Reactivity Feedback Components

III.3.2 ATWS Analysis Results – ULOF

❖ Assumptions

– All primary pump trips at full power followed by a coastdown

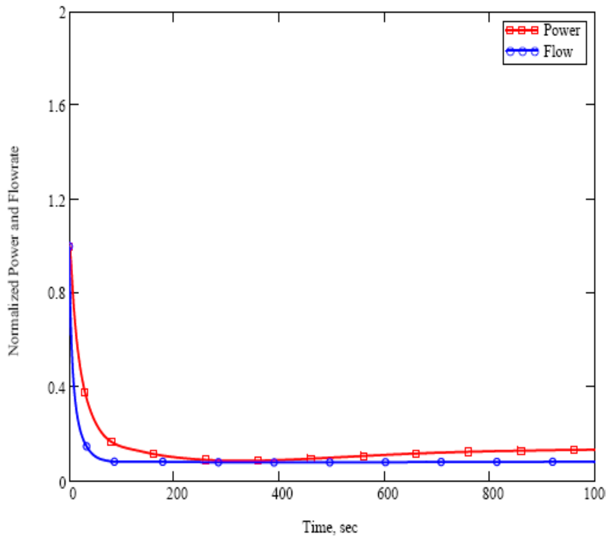
– Normal heat removal path and PDRC are available

❖ Results

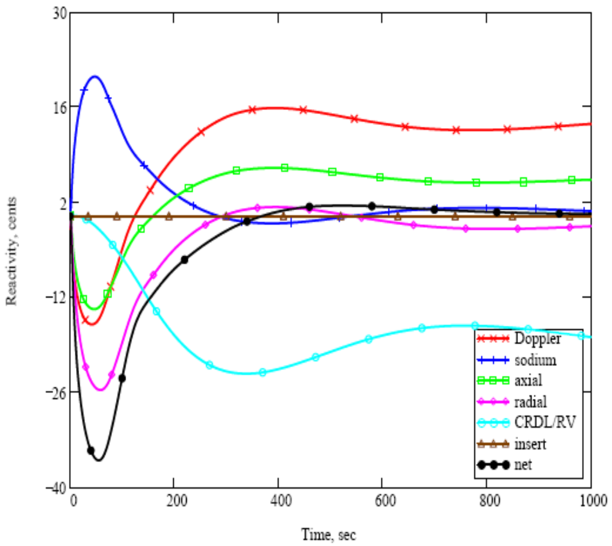
– Peak temperatures calculated

	Peak Fuel Temp., ℃	Peak Clad Temp., ℃	Av. Core Outlet Temp., ℃	Peak Na Temp., ℃
Limit	**1,070**	**700–790 (<0.3hr)**	**650–700 (<5hr) 700-760 (<1hr)**	**Pump on: 1,055 Pump off: 940**
ULOF	797	773 (80s)*	590	769

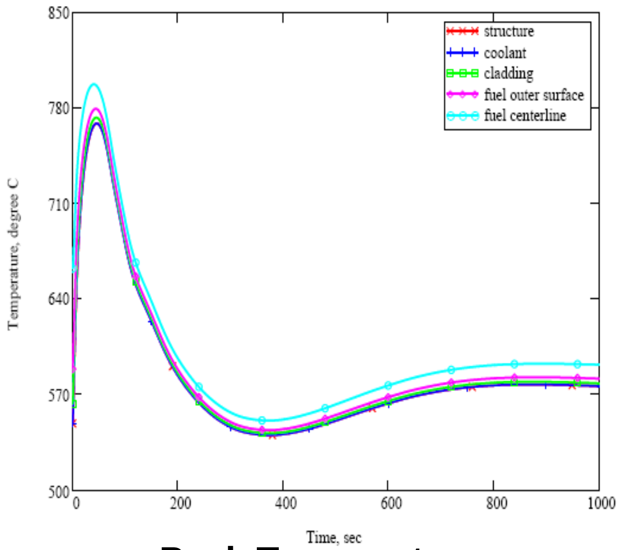
*** holding time**



Normalized Power and Flow



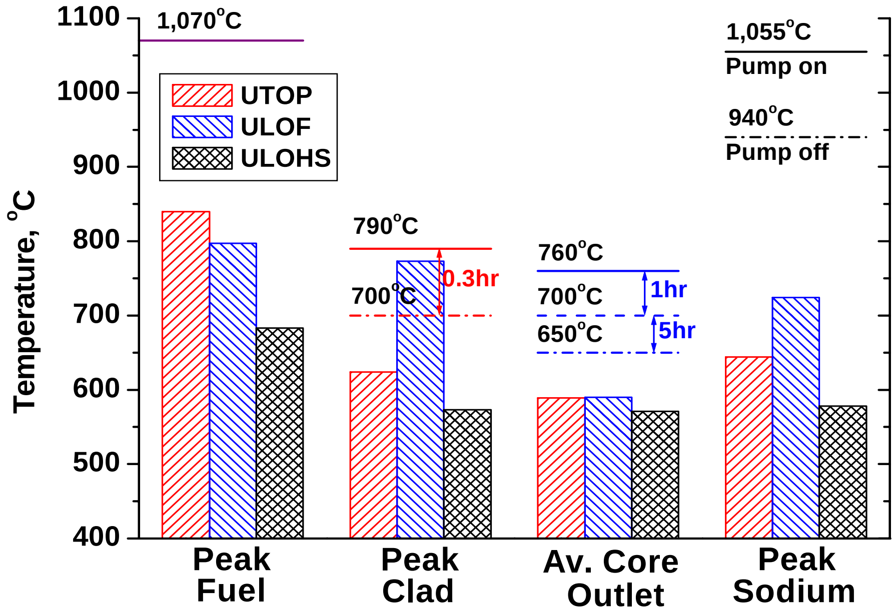
Reactivity Feedbacks



Peak Temperatures

III.3.3 ATWS Analysis Results – Summary

- ❖ KALIMER-600 design has capability to accommodate all the analyzed ATWS events
- ❖ A refined design for control rod stop system is necessary to limit the potential magnitude of the UTOP initiator
- ❖ Self-regulation capability is mainly due to the inherent and passive reactivity feedback mechanisms



	Peak Fuel Temp., ℃	Peak Clad Temp., ℃	Av. Core Outlet Temp., ℃	Peak Na Temp., ℃
Limit	1,070	700–790 (<0.3hr)	650–700 (<5hr) 700–760 (<1hr)	Pump on: 1,055 Pump off: 940
0.4$ UTOP (Pump on)	840	624	589	613
ULOF (Pump off)	797	773 (80 s)*	590	769
ULOHS (Pump off)	683	573	571	571

* holding time

Analysis results for ATWS

IV. Summary

- KAERI is concentrating on the development of key technologies for the implementation of safety design concept.
  - Validation of Passive function of DHRS
  - MA bearing metallic fuel
- The licensing of demonstration SFR will be pursued in the current regulatory framework on the basis of deterministic approach. The safety design will be supported by probabilistic approach.
- A system analysis code for SFR system has been developed. This code will be validated further with available test data worldwide.
- An evaluation methodology for DBEs is under development which aims to perform the safety analysis of demonstration SFR.