소듐냉각고속로 vs. 경수로 안전특성

2015. 10. 28

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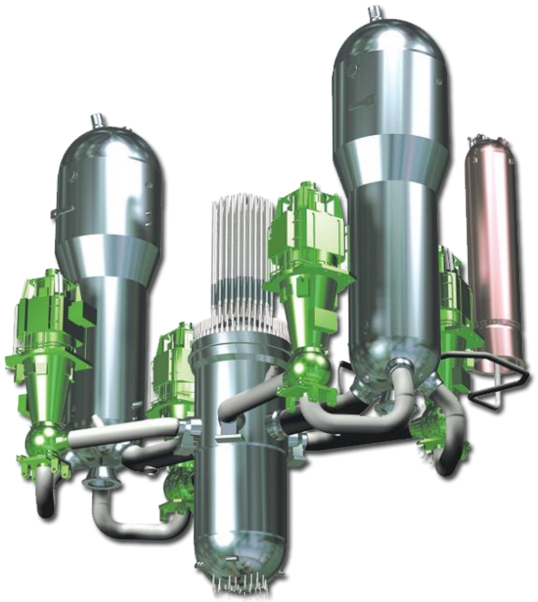
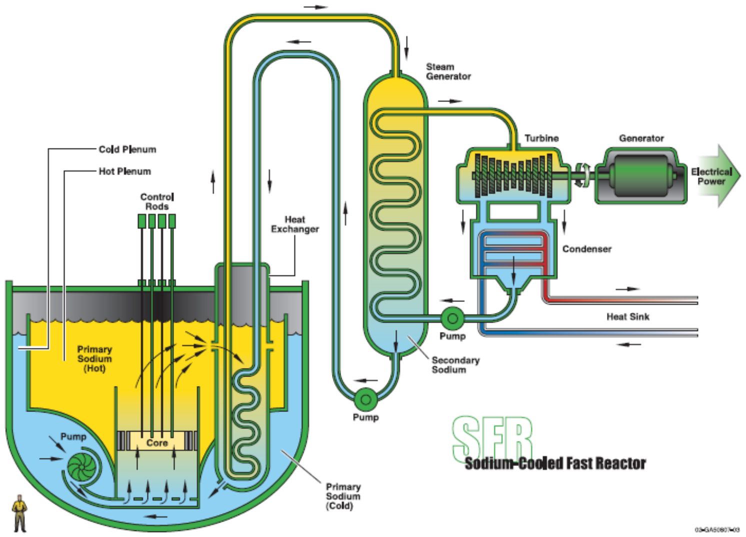
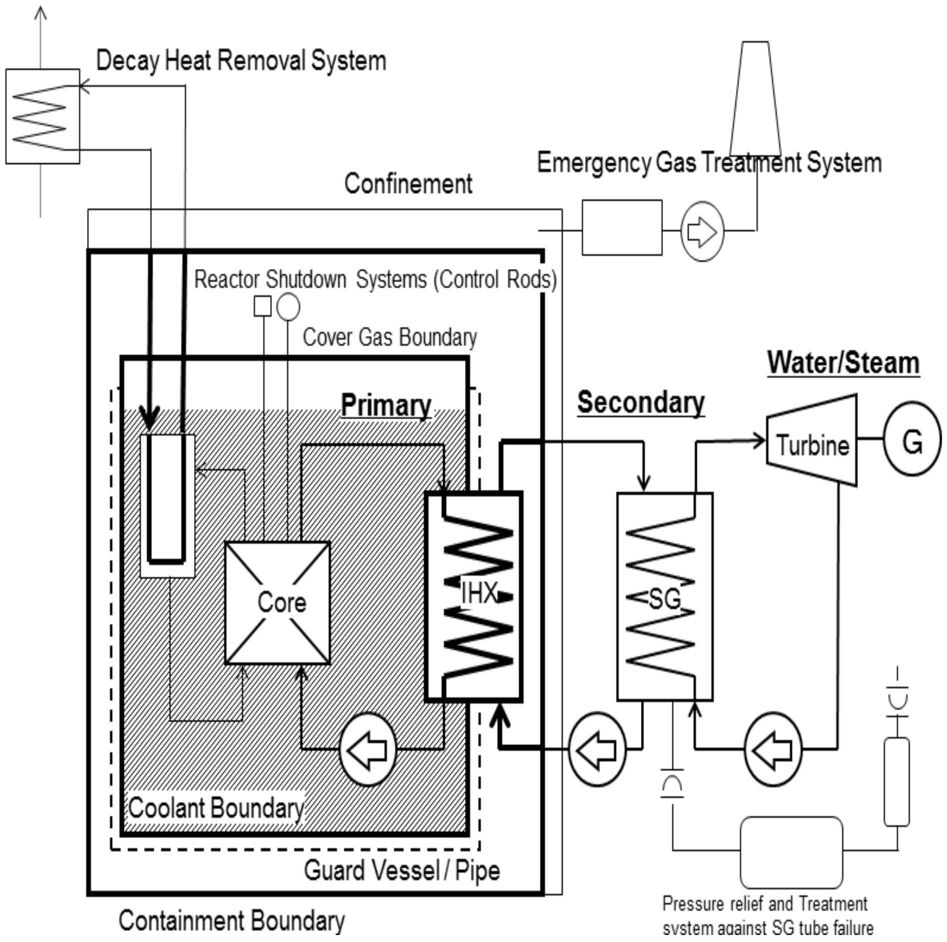
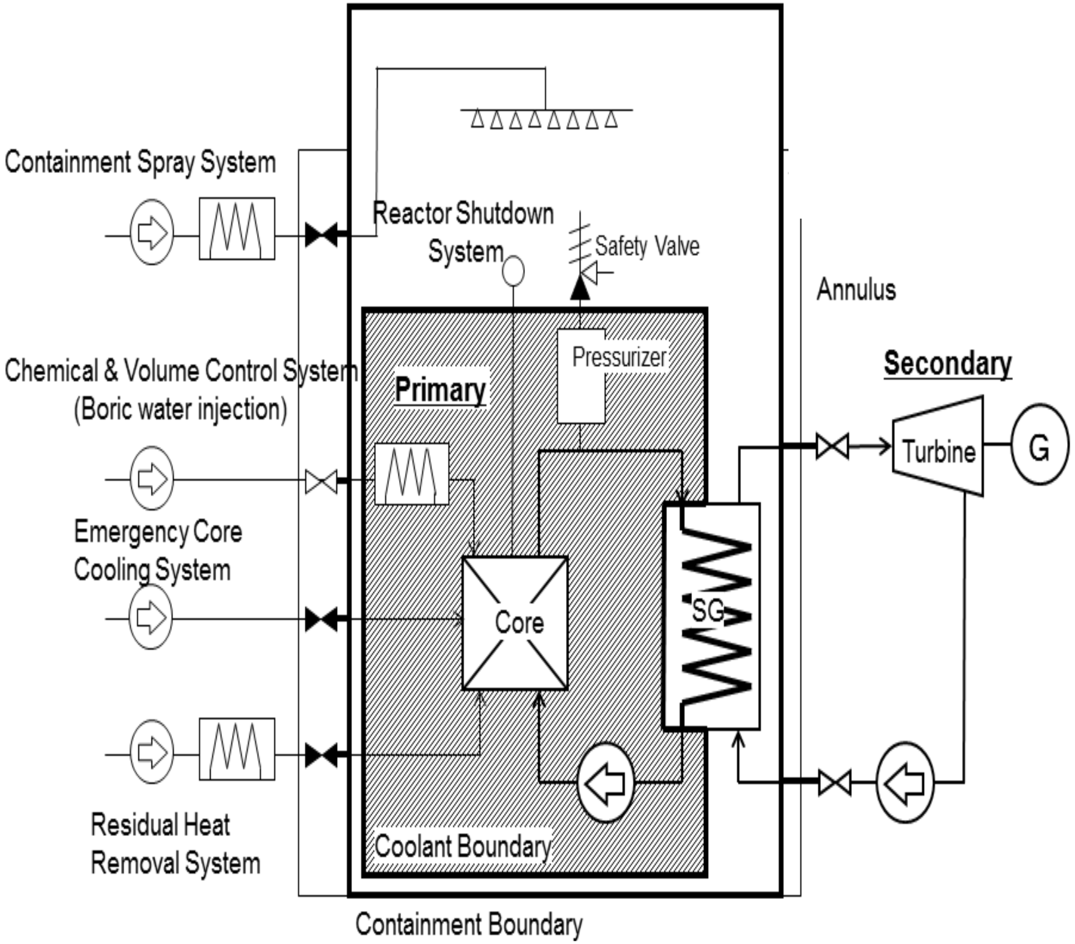




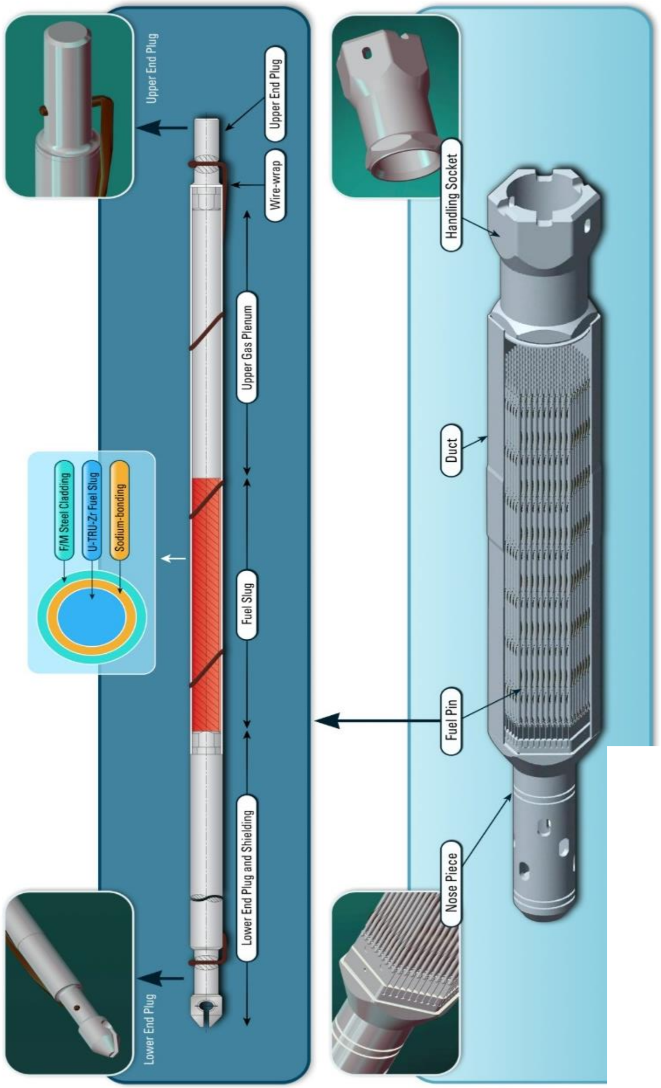
1. SFR vs. 경수로 특성



SFR



PWR



6각형

캔타입

금속연료

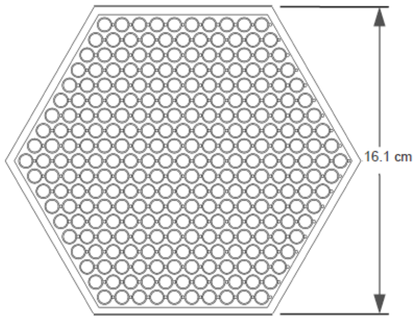
20%농축도

4각형

개방형

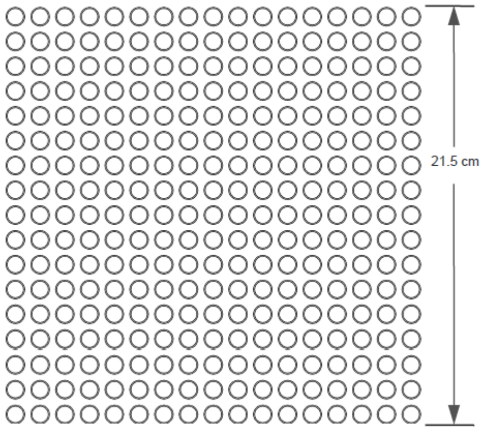
세라믹연료

5%농축도

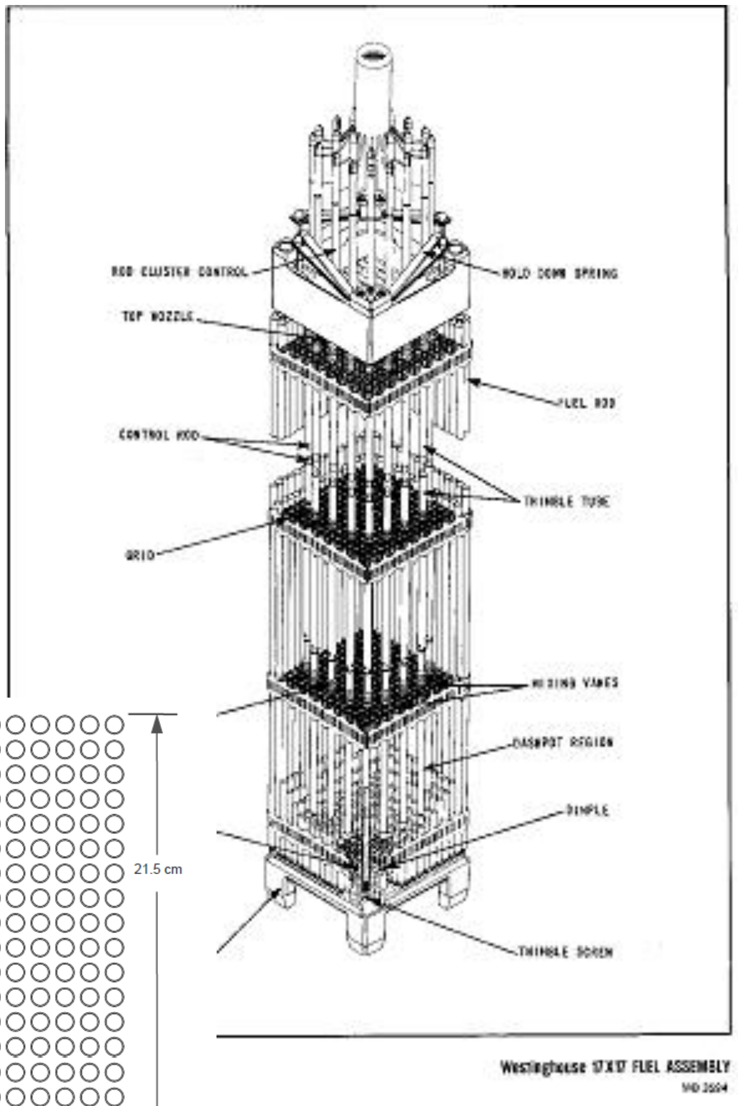


“Typical” SFR Assembly (271 pins)
Pin Diameter = 7.4 mm
Pin Pitch = 8.9 mm

SFR



Typical PWR Assembly (289 pin locations)
Pin Diameter = 9.4 mm
Pin Pitch = 12.5 mm



PWR

- Sodium vs. Water

	Na	Water
Atomic Weight	23.0	18
Melting Point (°C)	97.8	0
Boiling Point (°C)	892	345
Density (kg/m^3)	880	713
Specific Heat (J/kg-K)	1300	5600
Volumetric Heat Capacity (MJ/m^3-K)	1.14	4.0
Thermal Conductivity (W/m-K)	76	0.54
Viscosity (cP)	0.34	0.1
Core ΔT (°C)	155	30
Outlet temperature (°C)	510	330
Margin to boiling (°C)	> 300	15

Thermophysical Properties:	
Excellent Heat Transfer	✓+
Low Vapor Pressure	✓+
High Boiling Point	✓+
Low Melting Point	✓
Material Properties:	
Thermal Stability	✓+
Radiation Stability	✓+
Material Compatibility	✓+
Neutronic Properties:	
Low Neutron Absorption	✓+
Minimal Activation	✓
Negligible Moderation	✓+
Supports Passive Safety	✓+
Cost:	
Initial Inventory	✓+
Make-Up Inventory	✓+
Low Pumping Power	✓+
Hazards:	
Sodium reacts with air and water	

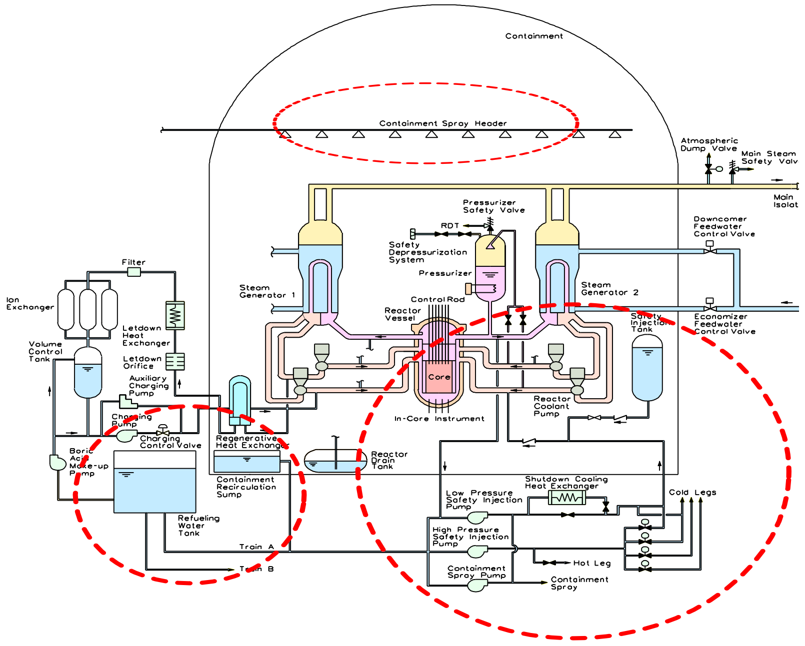
Sodium 냉각재 특성

- **안전성 관련 특성 비교**

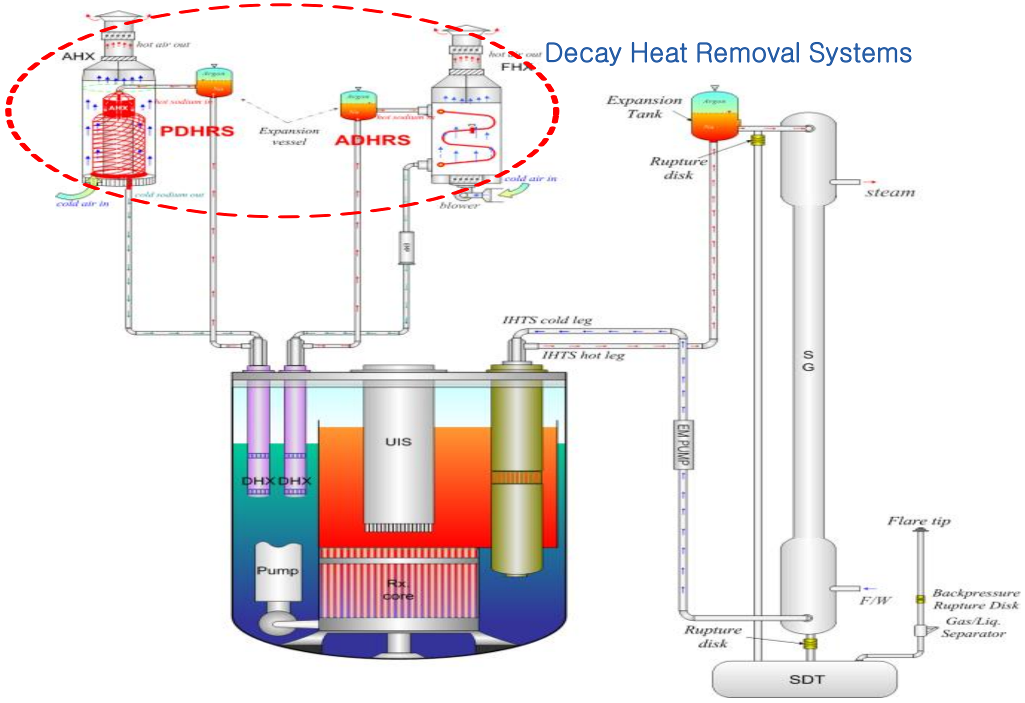
항목	PWR	SFR	
Reactivity	Max. reactivity design with intact configuration	Under max. reactivity with intact configuration	- Recriticality issue
Power density	Low (~100 kW/liter)	High (~300 kW/liter)	
Power Coefficient	Negative (Coolant boiling)	Positive (Coolant boiling)	- Faster power excursion and possible positive PC
System Pressure	High	Low	- LOCA is not a concern - No ECCS
Sub-cooled margin	Low (~20 K))	High (~300 K)	- No boiling
Natural Circulation	-	Better	
Final Heat Sink for decay heat	Water (sea, river)	Air	
Chemical reaction	No	-Sodium fire -Sodium water reaction	

• **안전계통 특성**

안전계통	PWR	SFR	
Shutdown system	Control rod + Chemical shim	Two S/D control rod systems	-Different mechanism
Inventory control	ECCS (SIT+HPSI+LPSI)	Not necessary	- Leak make-up
Decay heat removal	RHRS with water heat sink	DHRS with air heat sink	Natural circulation possible (PDHRS)
Containment	Needs to take LOCA (Max. P ~ 6 bar) Containment cooling and spray systems	Needs to take sodium fire (Max. P ~2 bar) Containment inert system	- Temperature could be higher



Safety Injection, Shutdown Cooling , Containment Spray Systems



Decay Heat Removal Systems

2. 안전분석 특성

- SAR Chapter 15 Events Categories by Type of Events
  - 15.1 Increase in Heat Removal by the Secondary system
  - 15.2 Decrease in Heat Removal by the Secondary system
  - 15.3 Decrease in Reactor Coolant Flow Rate
  - 15.4 Reactivity and Power Distribution Abnormalies
  - 15.5 Increase in RCS Inventory
  - 15.6 Decrease in RCS Inventory
  - 15.7 Radioactive Material Released from a System/Component
  - 15.8 ATWS : BDBA – BE Analysis

- SFR 주요 사고 분석
  - DBA:
    - Loss of Heat Sink
    - Loss of Flow
    - Transient Over Power
    - PHTS Pipe Break
    - RV Sodium Leak
  - DEC:
    - Ultimate TOP (ATWS)
    - Ultimate LOF (LOF+ATWS)
    - Ultimate LOHS (LOHS +ATWS)
  - Sodium Accidents
    - SG wastage
    - Sodium Fire

- 미 CRBR(Clinch River Breeder Reactor) PSAR에 수록된 사고 사례

Anticipated Events

Control Assembly Withdrawal at Startup
Control Assembly Withdrawal at Full Power
Seismic Reactivity Insertion (OBE)
Small Reactivity Insertions
Inadvertent Drop of a Single Control Rod at Full Power
Loss of Off-Site Electrical Power
Spurious Primary Pump Trip
Spurious Intermediate Pump Trip
Inadvertent Closure of One Evaporator or Superheater Module
Turbine Trip
Loss of Normal Feedwater
Inadvertent Actuation of the Sodium/Water Reaction Pressure
Loss of One D.C. System
Loss of Instrument or Valve Air System
IHX Leak
Off-Normal Cover Gas Pressure In PHTS
Off-Normal Cover Gas Pressure In IHTS

Unlikely Events

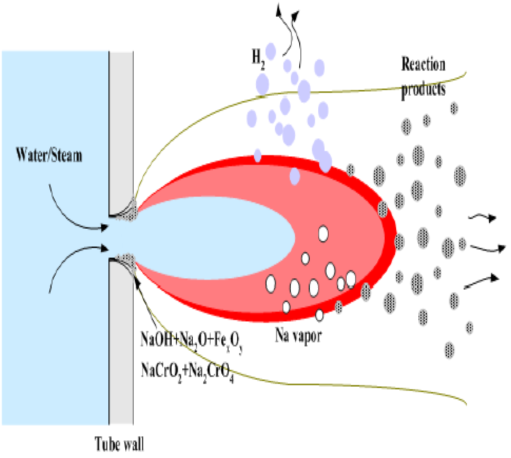
Loss of Hydraulic Holddown
Sudden Core Radial Movement
Maloperation of Reactor Plant Controller
Single Primary Pump Seizure
Single Intermediate Loop Pump Seizure
Small Water to Sodium Leaks In Steam Generator Tubes
Failure of Steam Bypass System
Fuel Assembly Dropped During Refueling
Attempt to Insert a Fuel Assembly Into an Occupied Position
Single Fuel Assembly Cladding Failure In Fuel Handling System
Cover Gas Release During Refueling
Heaviest Crane Load Impacts Reactor Closure Head
Inadvertent Release of Oil Through the Pump Seal (PHTS)
Inadvertent Release of Oil Through the Pump Seal (IHTS)
Generator Breaker Failure to Open at Turbine Trip
Rupture in the RAPS Cold Box
Liquid Radwaste System Failure
Failure In the EVST NaK System
Leakage from Sodium Cold Traps
Rupture In RAPS Noble Gas Storage Vessel Cell
Rupture in the CAPS Cold Box

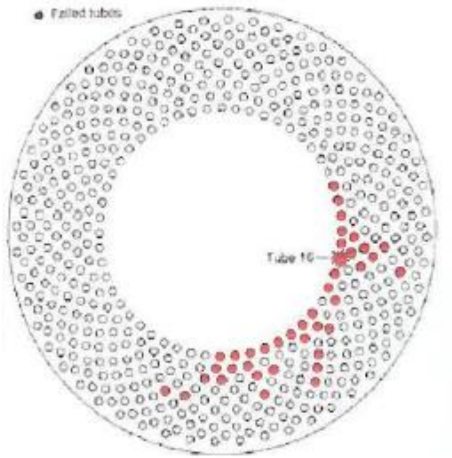
- **Sodium Accidents**

✓ $Na + H_2O \rightarrow NaOH + \frac{1}{2} H_2 + 162 \ kJ/mole$

✓ 특성

- 폭발적인 발열반응 (반응영역온도 : **1200~1400℃**)
- 부식성 반응생성물
- 수소발생





Sodium-water reaction experience at PFR, 1987

- SGTR and wastage
  - 증기발생기 튜브 파손으로 증기누출 시 소듐-물 반응으로 연쇄적인 튜브파손 발생

- Sodium fire
  - 소듐 누출 시 공기와 반응으로 화재 발생
  - Sodium – concrete interaction

  - 소듐의 누출 지역의 콘크리트 포함된 수분 및 탄화물 등과 반응하여 콘크리트 침식 및 화재 발생

- **Major Limiting Accidents, Acceptance Criteria, Safety Analysis**

- **PWR**
  - LOCA, MSLB, LOF, LOHS
  - Acceptance Criteria
    - DNBR & Local Power Density Limits
    - PCT, Max. Cladding Oxidation, Max. H2 Generation
    - RCS Pressure Boundary
    - Containment Pressure

- **SFR**
  - TOP, LOF, LOHS
  - Acceptance Criteria
    - Fuel melting temp (metal fuel: ~1300 K)
    - Clad melting temp (~1500 K)
    - Coolant boiling temp (1165 K)
    - Clad Cumulative Damage Function < 0.05

- **Accident Phenomena and safety analysis**
  - PWR : complex – cannot avoid two phase conditions and phase transition
  - SFR : simple – can maintain single phase during transients

3. 중대사고 현상

- **Severe Accidents in PWRs**

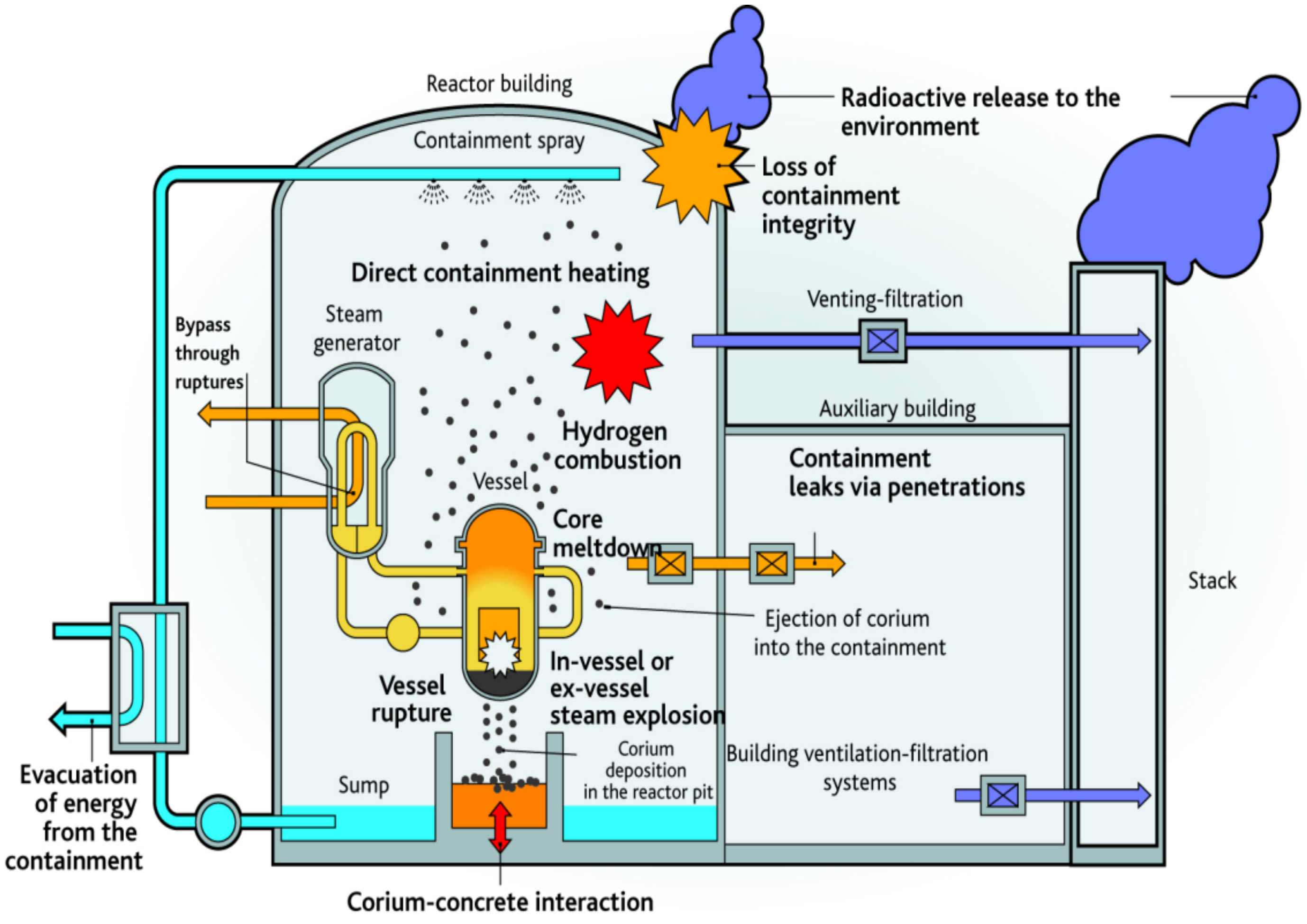
In-Vessel Phenomena

- Core Uncovery and Heatup
- Cladding Oxidation
- Core Melting and Relocation
- Heat Transfer to Lower Head
- Reactor Vessel Failure

Ex-Vessel Phenomena

- Direct Containment Heating(DCH)
- Fuel-Coolant Interaction(FCI)
- Molten Core- Concrete Interaction(MCCI)
- Hydrogen Combustion

Fission Product Release and Transport

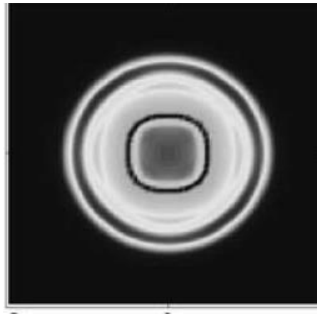


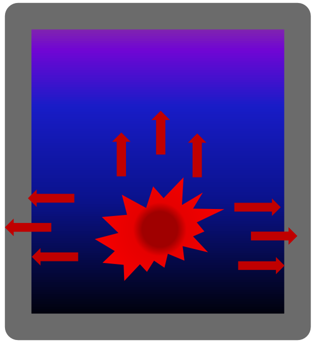
- Containment integrity is of a primary concern

- **Severe Accidents in SFRs**

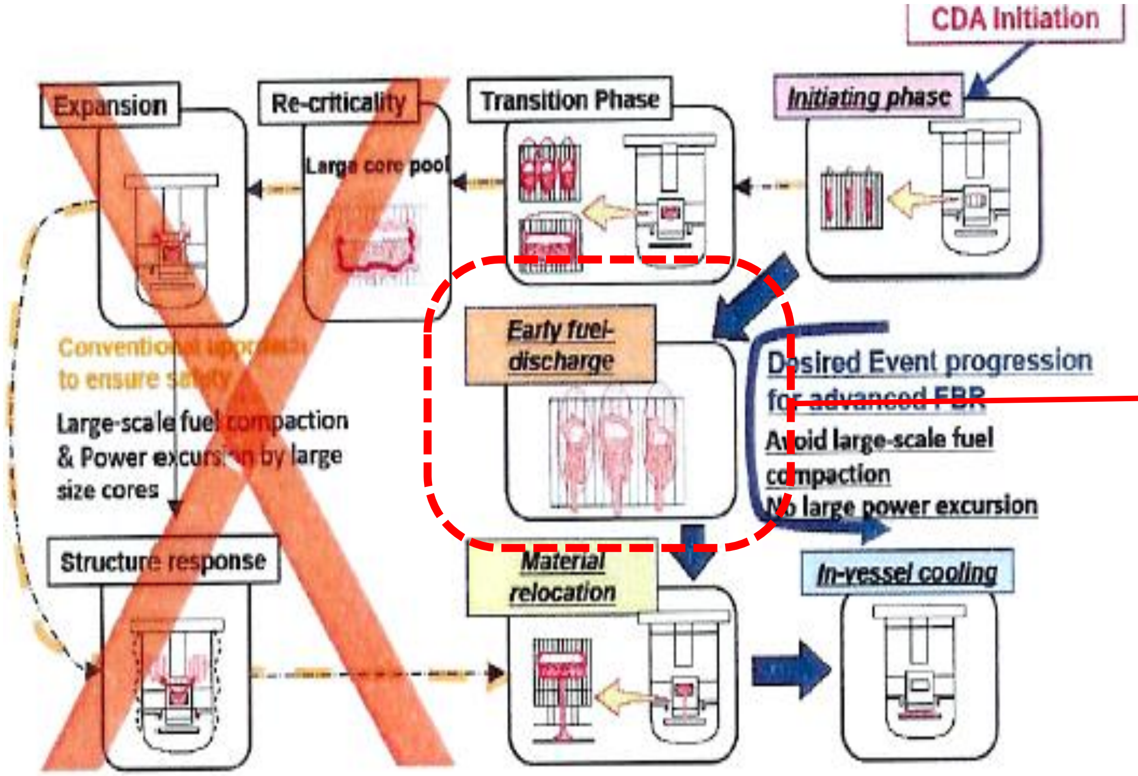
  - Core Disruptive Accident – bounding approach

    Recriticality에 의한 출력 폭주 에너지 계산 및 기계 구조적 건전성 분석



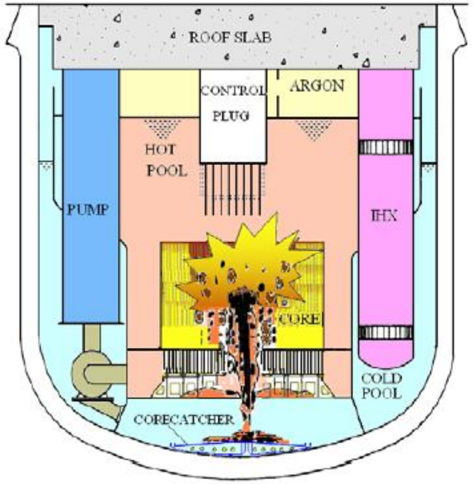


  - SFR 중대사고 진행 과정 및 대응 전략

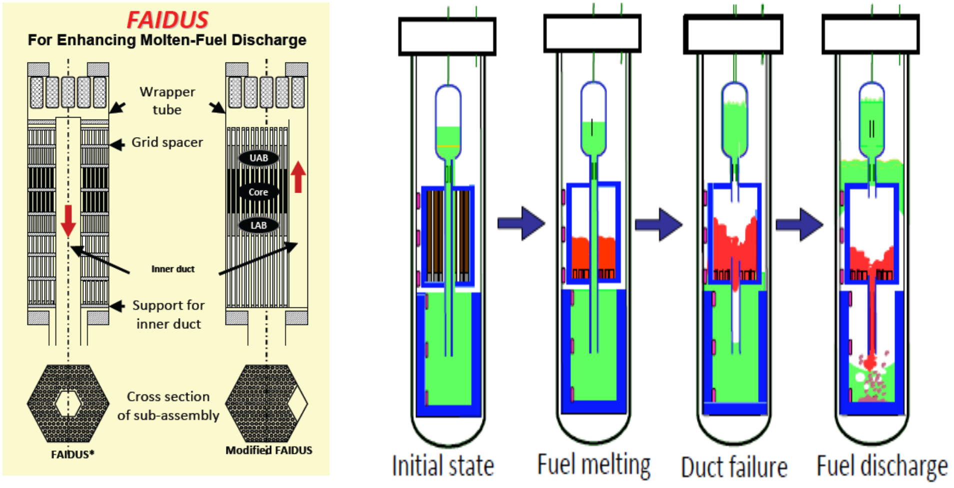


소듐냉각고속로 중대사고
대처전략의 핵심:
핵연료 용융시 최대한 빨리 용융연료를 노외로 배출시켜 재임계를 방지

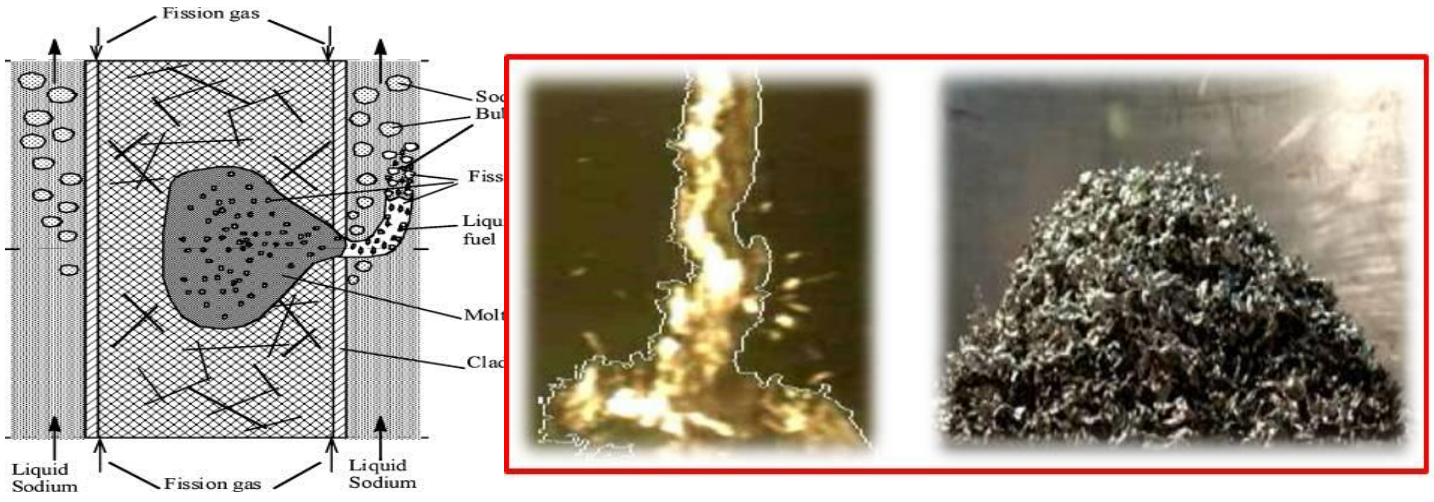
- **SFR 중대사고 대응 전략**
  - PWR 중대사고의 핵심: 격납건물 건전성 확보, 노심용융물 장기 냉각성 보장
  - SFR 중대사고 : PWR 요건 + Re-criticality 방지
  - SFR의 경우 Re-criticality 방지의 설계 반영에 보다 집중
  - SFR 중대사고 대처 전략 – 노심용융 발생 시 신속한 Shutdown 유도



Recriticality 가능

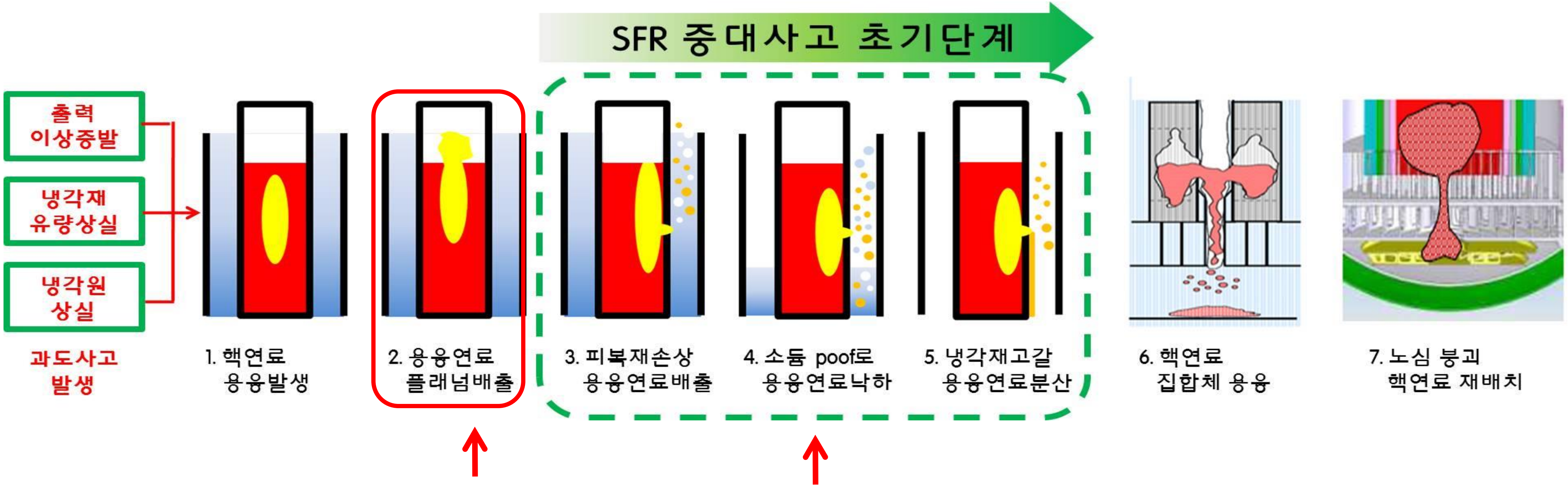


Oxide 핵연료 SFR의 전략



Metal 핵연료 SFR의 전략

- **금속 연료 SFR 중대사고 진행과정**



In-Pin 현상 **Ex-Pin 현상**

용융금속연료의 분산으로 재임계 방지 및 사고종료 유도

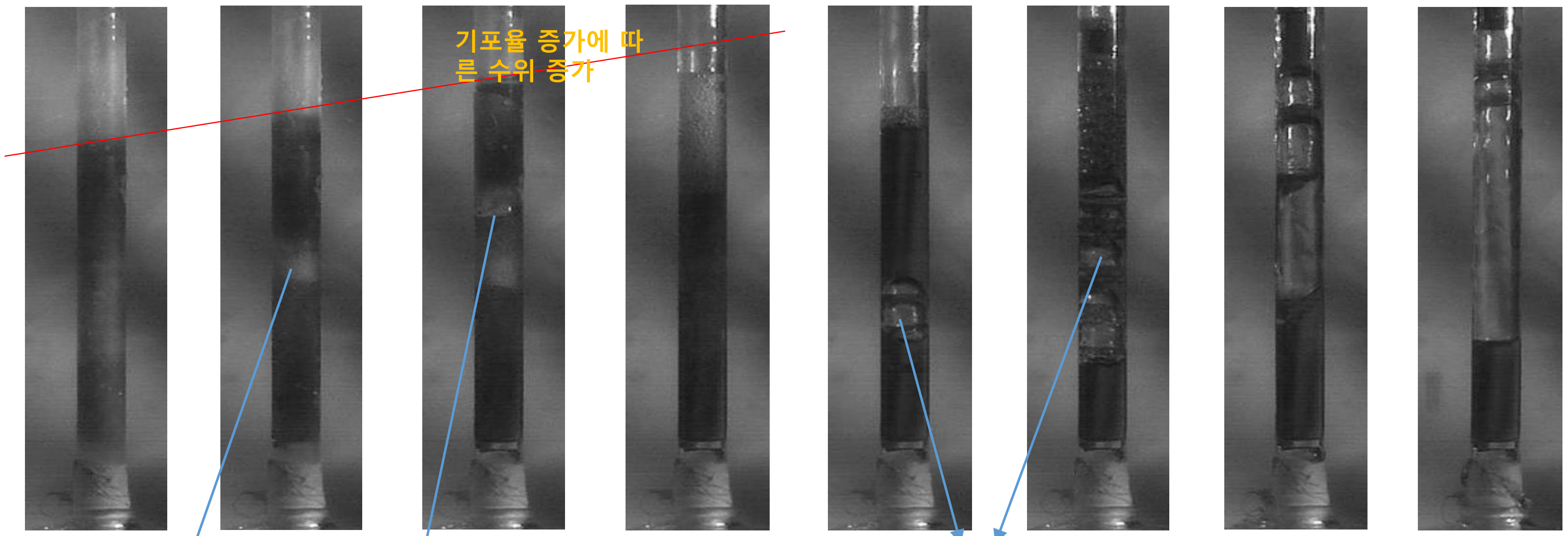
- TREAT* 실험: In-pin (용융연료의 상부 플래넘 이동) 및 Ex-pin (용융연료의 냉각재 혼합 배출) 현상 발견

*Metal Fuel을 사용한 Transient Overpower 실험으로 6회 수행

• 국내 모의실험 결과*

용융 및 용해된 기체 분출 (Bubbly Flow) 비등 (Slug Flow)

In-Pin 현상



용융발생 시작

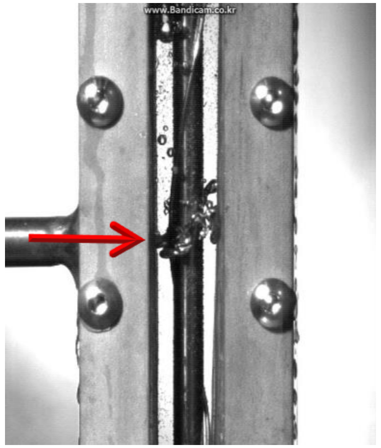
기포들이 모여 Slug 형성 (상부는 아직 고체상)

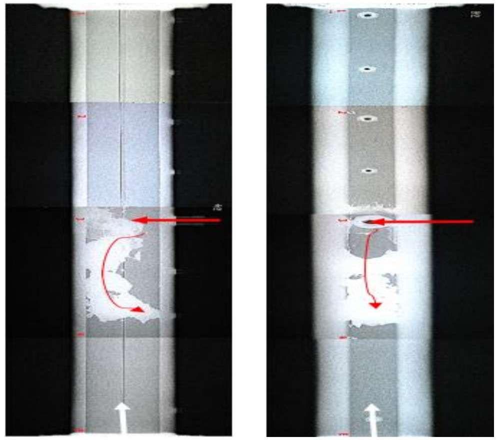
비등에 의해서 액체 내부에서 급격한 기체유입이 발생하며 이는 기포율의 크기 및 기포율을 증가시켜 유동양식을 변화

➡ 실제 IN-PIN 발생시 급격한 온도상승과 기체 분출로 인해 유사한 상황이 발생할 수 있을 것으로 예상

Ex-Pin 현상

용융물 주입구





* SFR 핵연료 용융현상 연구센터 수행 (원자력선진기술연구센터)

4. 결 언

- PWR은 냉각재 공급을 위한 ECCS 와 중대사고 방어를 위한 Robust Containment가 중요
- SFR은 Recriticality 방지를 위한 정지계통과 잔열제거를 위한 DHR이 중요
- SFR은 PWR에 비해 inherent safety feature가 중요하며, reactivity feedback, natural circulation의 가능성이 양호
- SFR은 저압 운전, 냉각재의 높은 과냉각 조건으로 냉각재 상실 우려가 없고, 냉각재 비등 가능성도 적어 안전계통이 단순, 안전분석의 불확실성도 비교적 작음.
- 중대사고 관점에서 금속연료의 장점이 있으나 용융 시 외부 방출, 용융 고화물의 냉각 가능 형상 유지 가능 등에 대한 증명 필요
- PWR에 비해 운전경험, 실험 자료 등이 충분치 못하여 설계는 안전을 고려하여 상당한 설계여유도를 가지도록 하여야 할 것임
- 항공기충돌, 지진, 침수 등 외부 사건에 대한 설계는 PWR보다 강화되어야 할 것임
- DBA, DEC, 중대사고의 방사선 소스텀에 대한 불확실성 해소 필요
- PWR은 최후 방벽으로 이동형 설비 활용이 가능하나, SFR은 어려워, 비상냉각계통에 보다 높은 신뢰성이 필요.
- 정비 중 소듐 누출 가능성 등 shutdown risk에 대한 분석 필요

참고: Landmark EBR-II Passive Safety Demonstration Test

- Pump coast-down without scram causes reactor core temperatures rise, introducing net negative reactivity reducing reactor power to decay heat level
- Peak coolant temperature reaches the maximum (533°C) in about 70 seconds
  - Coolant saturation temperature at this point is > 900°C, with 366°C as the minimum margin to coolant boiling