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Analysis of Pressurizer Surge Line Flow Effect on TMI-2 Severe Accident Progression

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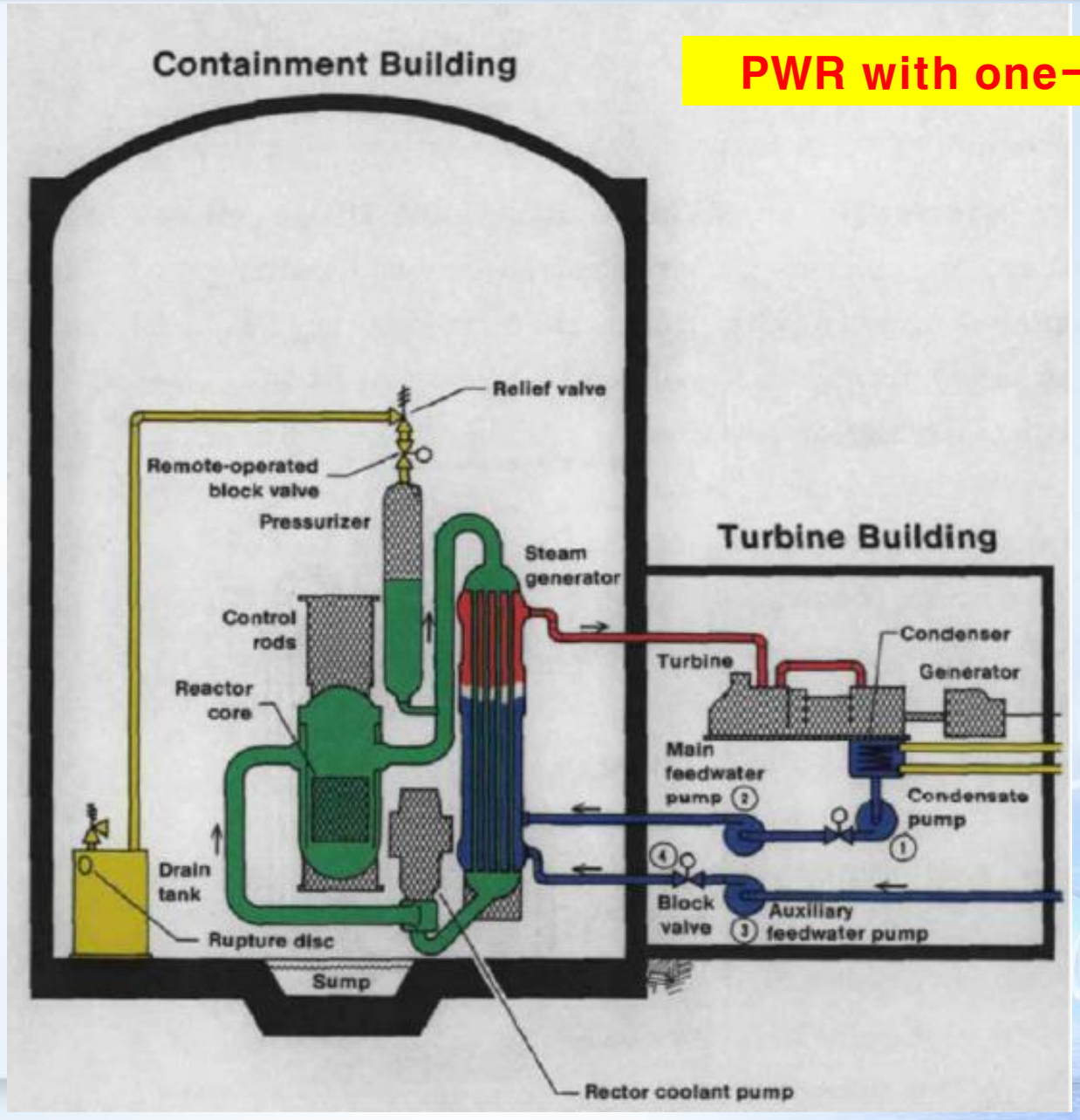
Introduction

❑ TMI-2 원전 중대사고 개요

	TMI-2
원전 소재	미국 펜실바니아
노형	가압경수로(PWR)
설계자	Babcock & Wilcox
출력	906 MWe
상업운전 개시일	1978.12.30
사고발생일	1979.03.28
사고원인	설비 결함과 인적 실수
사망자 및 영향	없음
사고 특이사항	노심용융, 원자로용기 및 격납건물 건전성 유지

TMI-2 Plant

PWR with one-through SG



Plant Data of TMI-2

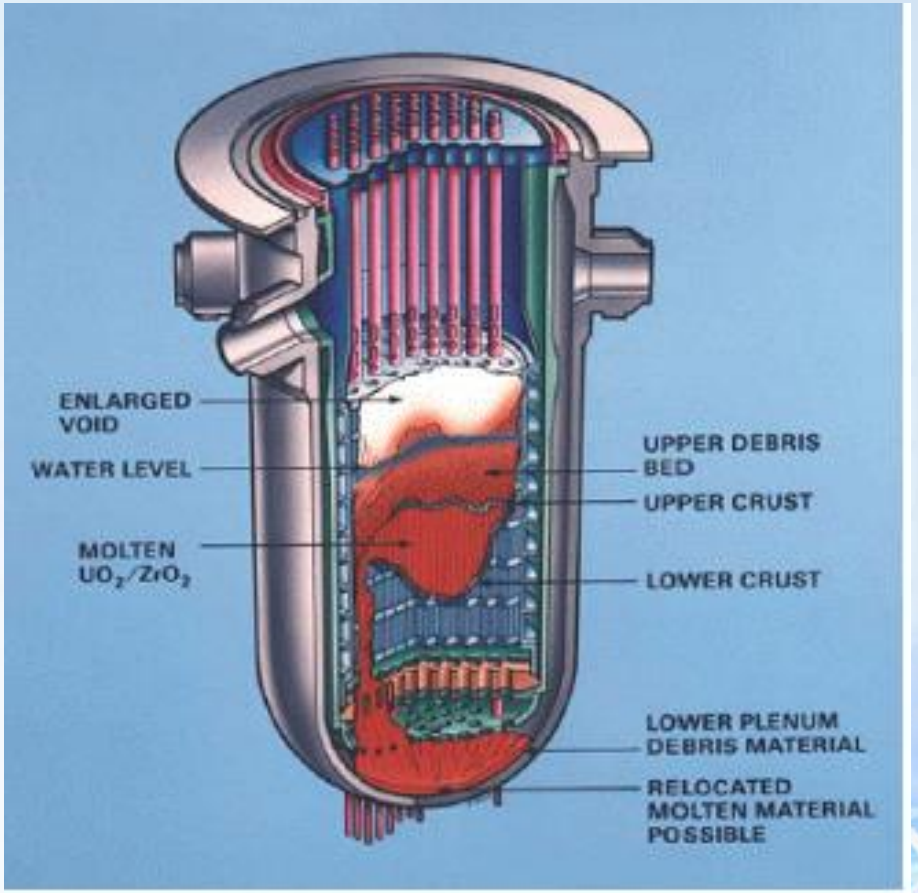
Reactor	Data	Core & Fuel Assemblies	Number
Design Heat Output	2,272 MWt	Fuel Assemblies	177
Vessel Coolant Inlet Temperature	292 ˚C	Fuel Rods per Fuel Assemblies	208
Vessel Coolant Outlet Temperature	320 ˚C		
Core Coolant Outlet Temperature	321 ˚C	Control Rod Guide Tubes per Assembly	16
Average Core Fuel Temperature	649 ˚C		
Core Operating Pressure	15 MPa		

Detailed TMI-2 Main Events

Time (s)	Main Events
0	Turbine and main feedwater pump trip (Total Loss of Feed Water)
3	Pressurizer PORV opening (15.5 MPa)
8	Reactor scram on high pressure signal
13	No Pressurizer PORV closing (15.2 MPa) (SBLOCA)
41	Operation of 1 (of 3) makeup pump 1B
122	HPI operation
278	Stop of HPI
480	Auxiliary feedwater startup
552	Core boiling begins
4,440	Shutdown B-loop RCP (end of phase 1)
6,000	Shutdown A-loop RCP
6,184	Core uncovery
7,742	Cladding oxidation begins (T= 1,000K)
7,719	Cladding failure (T=1,117K)
8,340	Close of the PORV line block valve
9,014	Fuel melting
10,440	Restart one B-loop RCP (end of phase 2)
11,580	Shutdown of the B-loop RCP
12,000	Start of primary system feed and bleed
13,440	Core material slumping (end of phase 3)
18,000	General emergency declared (end of phase 4)

End State in TMI-2

- 노심물질의 약 45 %(62 톤) 용융, 노심에 용융풀과 노심파편층 및 피막층 형성
- 용융물 약 19톤 측면으로 원자로용기 하반구로 재배치
- 운전원의 복구노력으로 원자로용기 파손방지



TMI-2 사고의 최종 모습

Research Needs & Objective

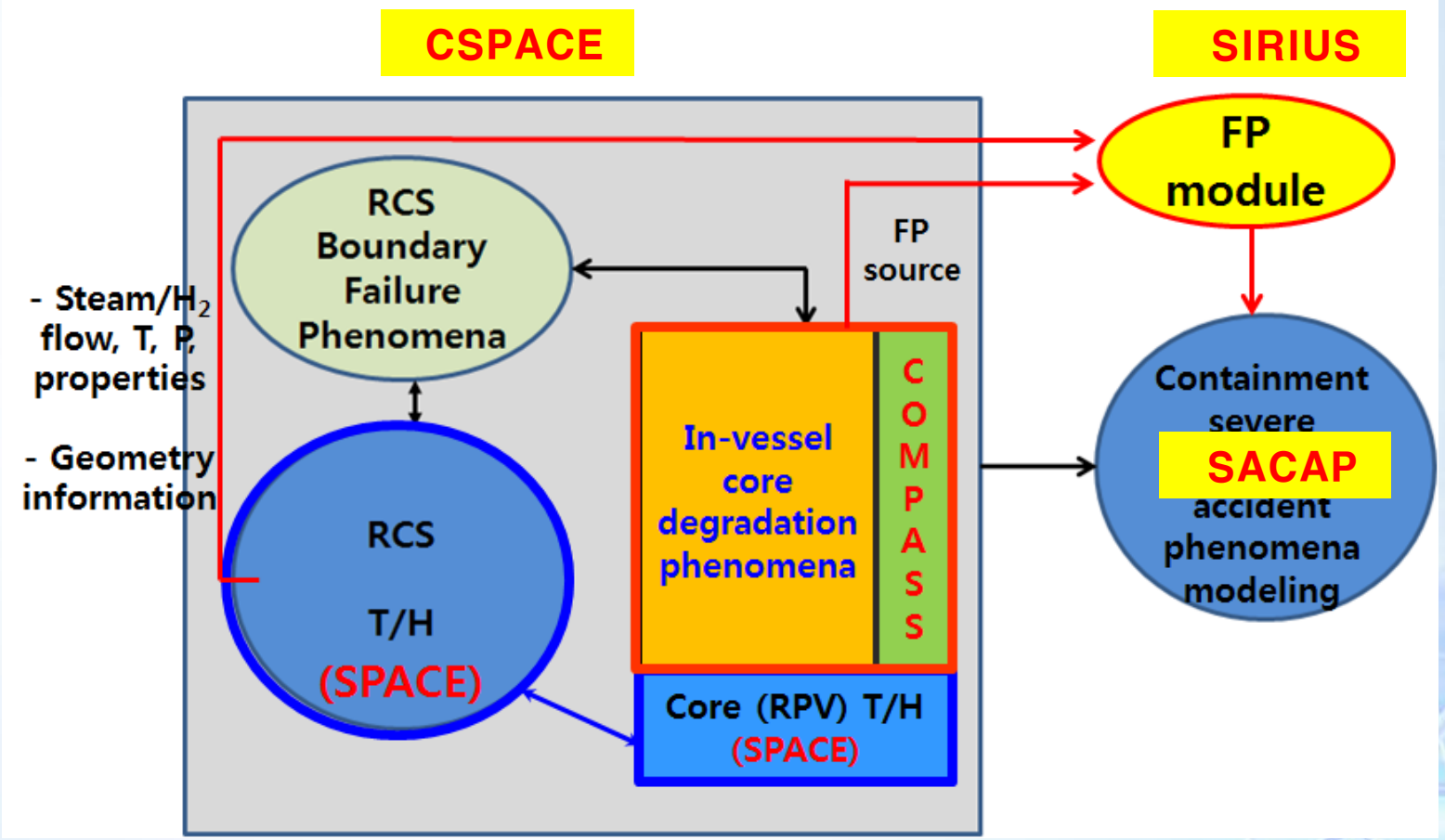
- The surge In the TMI-2 severe accident, the break location was the PORV (Pilot Oprated Relief Valve), which was located top of the pressurizer.

- A flow path through the the surge line between the hot leg and the pressurizaer was generated and the water level of pressurizer was very high, which resulted in the melt progression in the core.

- The surge line modeling effect on the core melt progression in the TMI-2 severe accident was analyzed using the CINEMA computer code.

CINEMA Development in Korea

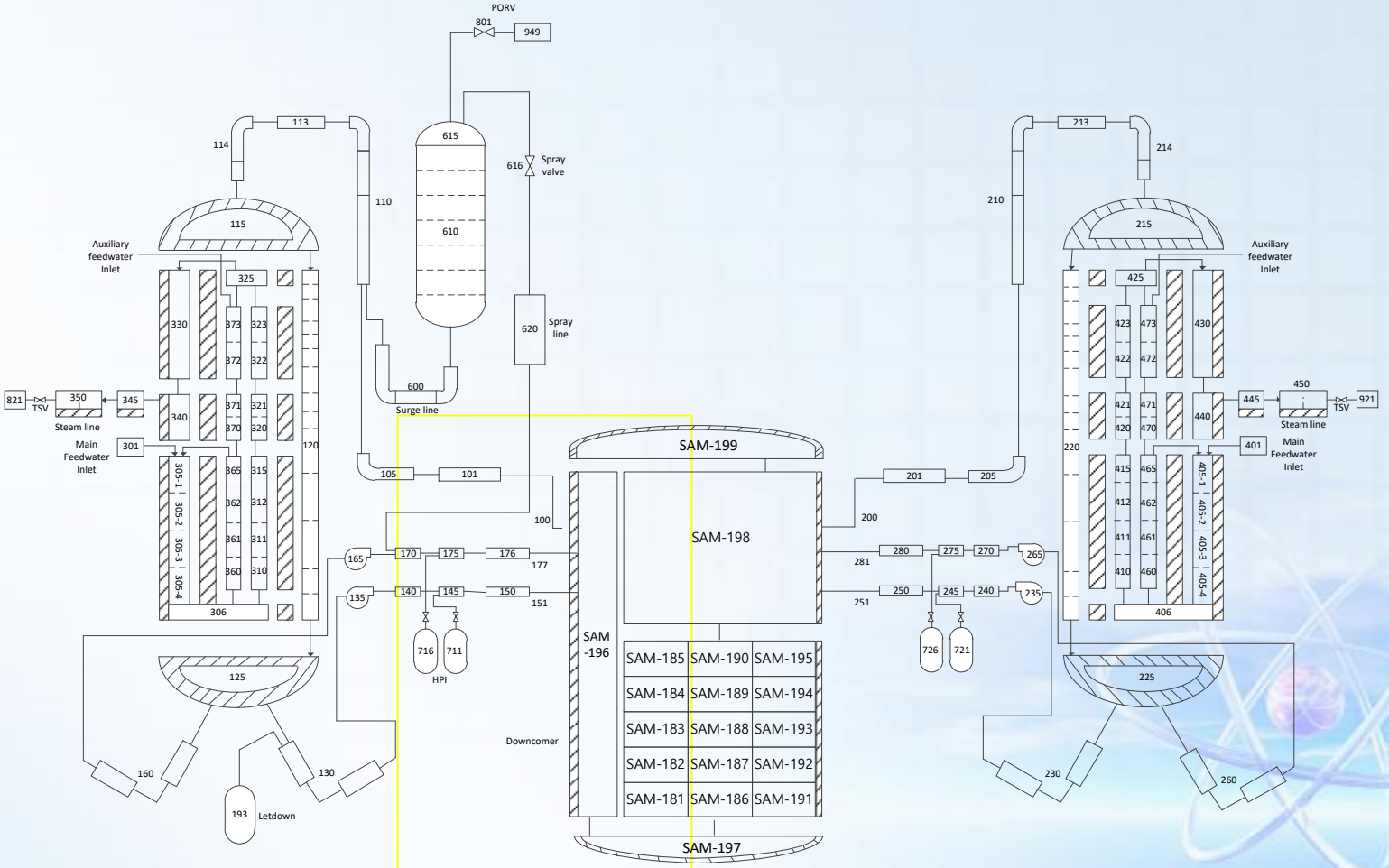
- An integrated severe accident analysis computer code (CINEMA) has been developed by the collaboration in Korea.



- **CINEMA: Code for INtegrated severe accidEnt Management Analysis**

CINEMA Nodalization for TMI-2

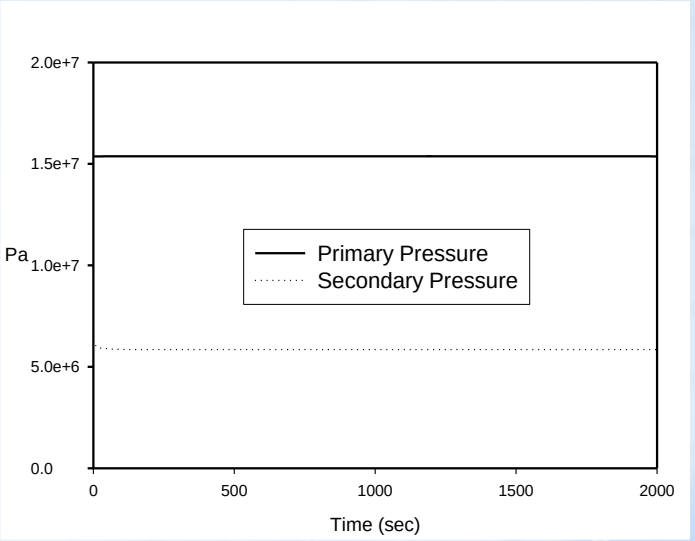
- Using TMI-2 Design Data
- Based on SCDAP/RELAP5 Input



CINEMA-CSPACE Nodalization

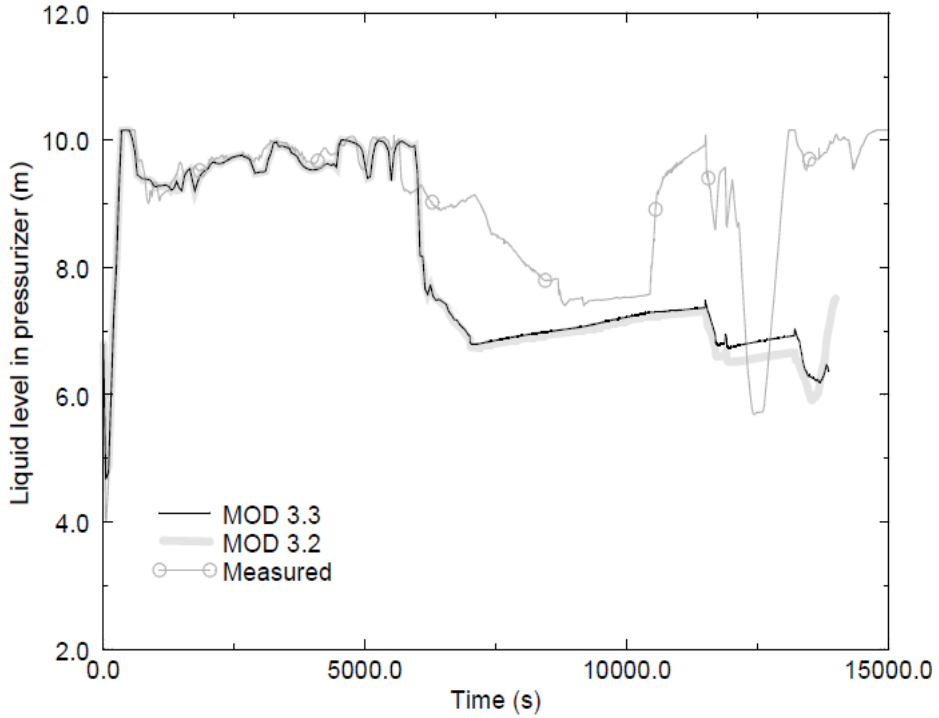
Steady State Results

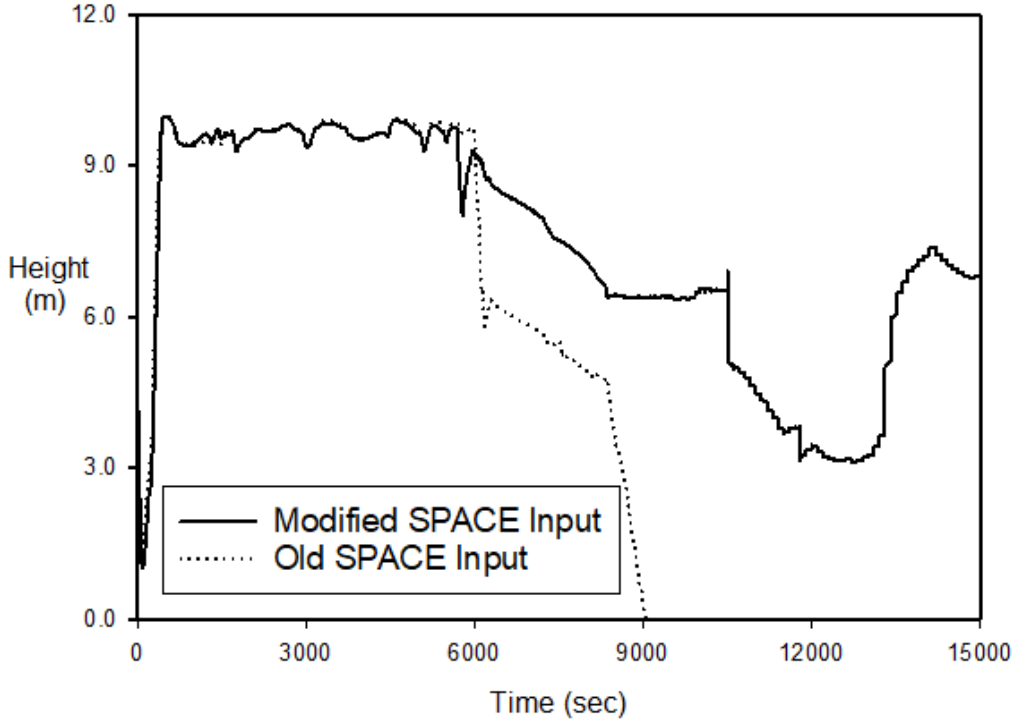
Parameter	Plant Operating Condition	CINEMA Results
Reactor Power (MW)	2700.0	2700
Primary System Pressure (MPa)	15.2	15.3
Cold Leg Temperature 1A (K)	561.0	571.0
Cold Leg Temperature 2A (K)	548.0	571.0
Hot Leg Temperature Loop A (K)	592.0	598.0
Hot Leg Temperature Loop B (K)	592.0	598.0
Feedwater Temperature (K)	513.0	513.0
SG A Pressure (MPa)	7.31	5.85
SG B Pressure (MPa)	7.24	5.85
SG A Steam Temperature (K)	586.0	578.0
SG B Steam Temperature (K)	585.0	579.0



Pressurizer Water Level

❖ SPACE 입력 개선: 가압기 surge 관 Strong CCFL,
PORV의 Critical Flow 모델(New HF 모델)

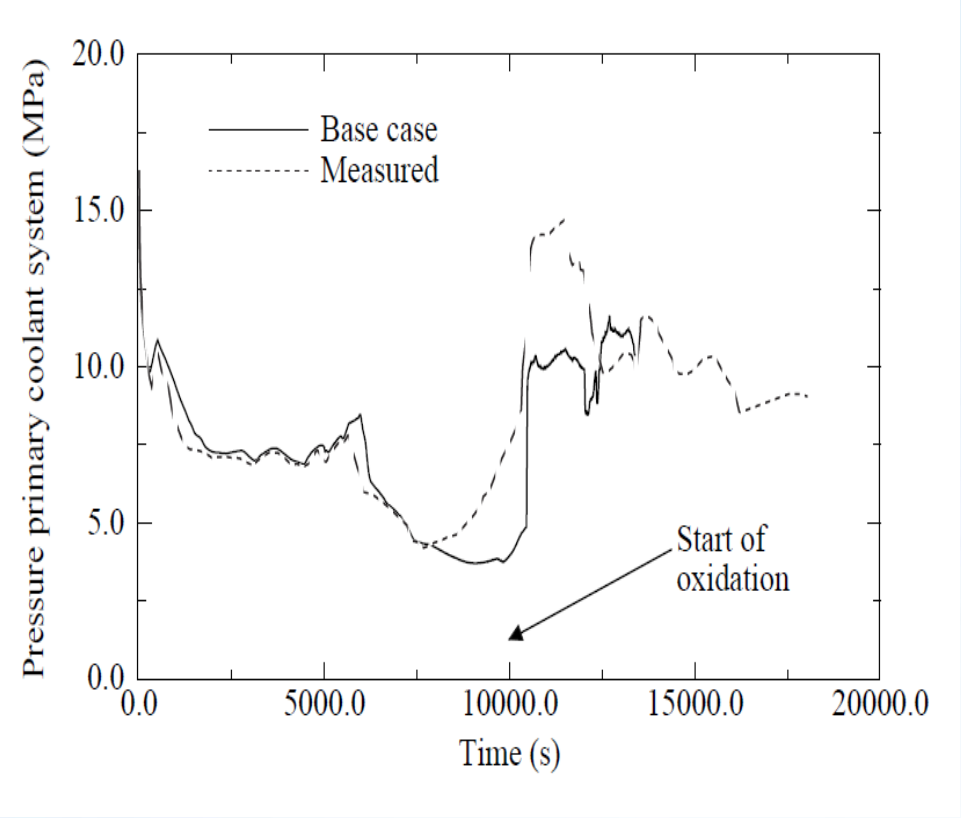


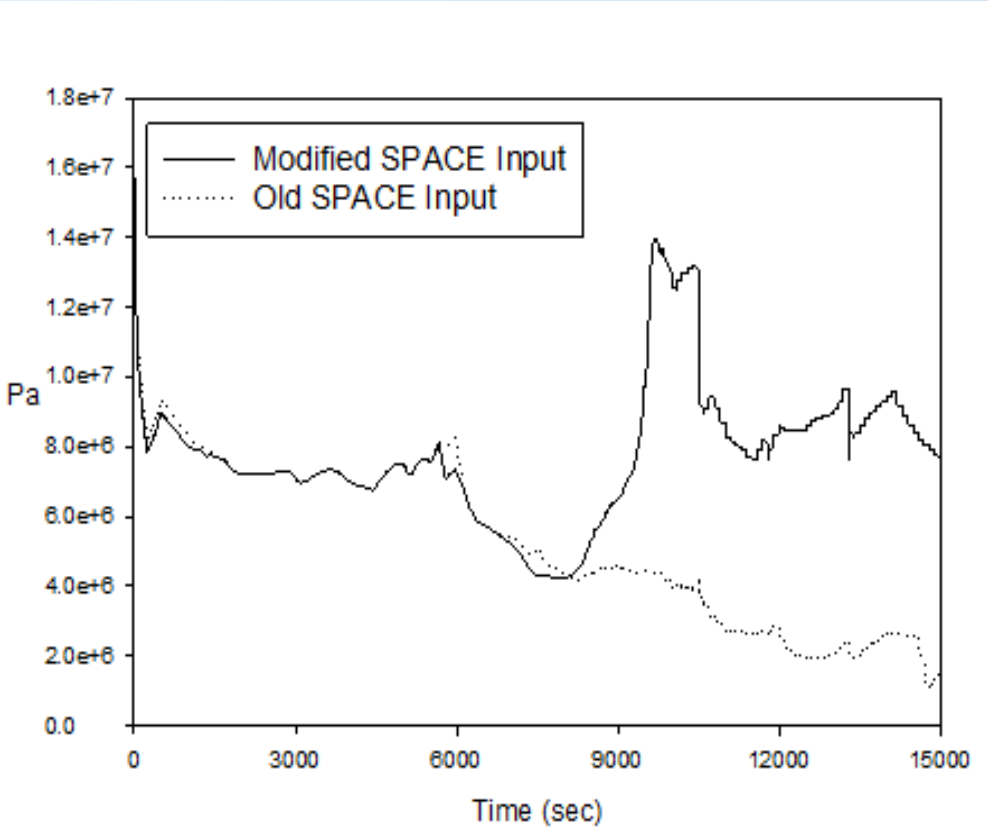


Measured Data and SCDAP/RELAP5 Results

CINEMA Results

Pressurizer Pressure

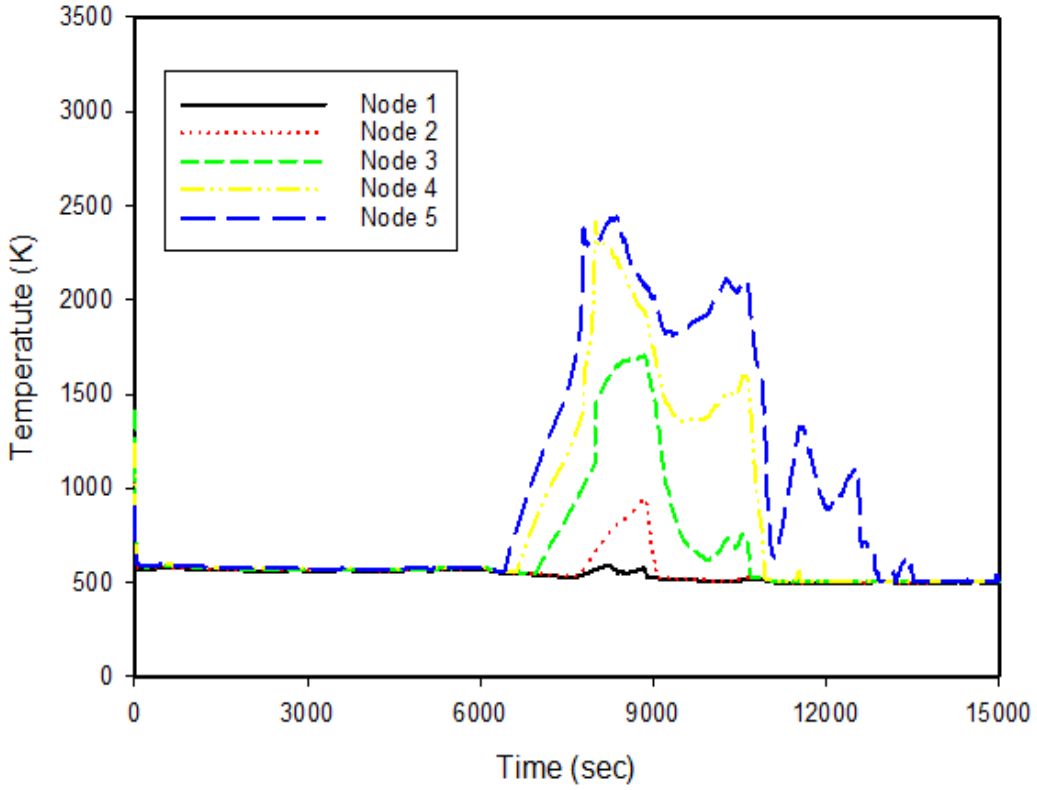


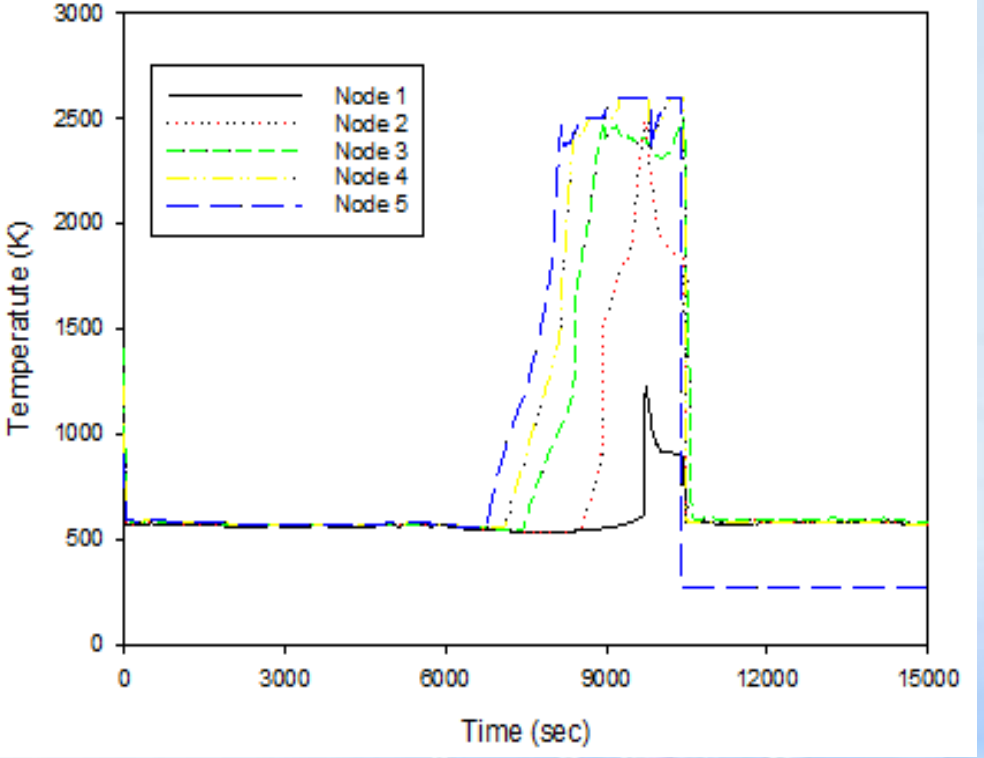


Measured Data and SCDAP/RELAP5 Results

CINEMA Results

Fuel Cladding Temperature





CINEMA Results on No CCFL

CINEMA Results on Strong CCFL in Pressurizer Surge Line

Conclusions

- The surge line flow modeling effect on the core melt progression in the TMI-2 severe accident was analyzed using the CINEMA.
- The CCFL input parameters in CINEMA affect the pressurizer water drain to the core through the pressurizer surge line.
- The CINEMA results on strong CCFL model are very similar to the TMI-2 data in general.
- More CINEMA analysis for a melted fuel relocation and quenching process in the core and lower plenum are necessary to simulate the late phase of the TMI-2 severe accident.

Thank You!

KAERI 중대사고 · 중수로안전연구부
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