U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

RIC 2007

OECD-NEA ROSA Program

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Introduction: Fundamental Questions

- **What is ROSA?**

  The Rig of Safety Assessment (ROSA) is an experimental program that has been conducted by the Japan Atomic Energy Agency (JAEA), formerly Japan Atomic Energy Research Institute (JAERI), for over 30 years.

- **Why is ROSA unique?**

  Over the years, the ROSA test facility has developed into what is known as the Large Scale Test Facility (LSTF), a full-height, 1/48 volume-scale model of a 4-loop PWR.

- **What has ROSA provided?**

  The program's experiments have played a vital role in clarifying important light water reactor (LWR) thermal-hydraulic phenomena and developing computer codes.

Fundamental Questions Cont.

- **Why is the NRC involved?**

  ROSA continues to be a source of insight into long-standing and emerging thermal-hydraulic safety issues.
  - Passive safety systems in new reactors
  - Accident management (AM) measures in beyond-design-basis situations
  - Computer code validation (TRACE)

The Evolution of ROSA Program

- **ROSA-I ('70-'73):**
  - 1.7m3 volume
  - Two-phase critical flow and vessel blowdown
- **ROSA-II ('74-'77):**
  - PWR integral simulator, 1/400-scale volume and ½-scale height
  - PWR large-break loss- of- Coolant -Accidents (LBLOCAs)
- **ROSA-III ('78-'83):**
  - PWR T/H response to small-break LOCAs (SBLOCA) and abnormal transients - BWR integral simulator, 1/424-scale volume and ½-scale height
  - BWR large and small break LOCAs
- **<u>ROSA-IV ('80-'92):</u>**
  - 1/48-scale volume and full-scale height
    - Reference reactor: Tsuruga-2 3423 MWt Westinghouse-type 4-loop PWR
  - Small-break LOCA experiments
  - Transient experiments
  - Separate-effects experiments
    - Natural circulation
    - Long-term cooling following LOCA
  - OECD/NEA/CSNI International Standard Problem No. 26 (ISP-26)
    - Organization for Economic Cooperation and Development (OECD)/Nuclear Energy Agency (NEA)/Committee on the Safety of Nuclear Installations (CSNI)

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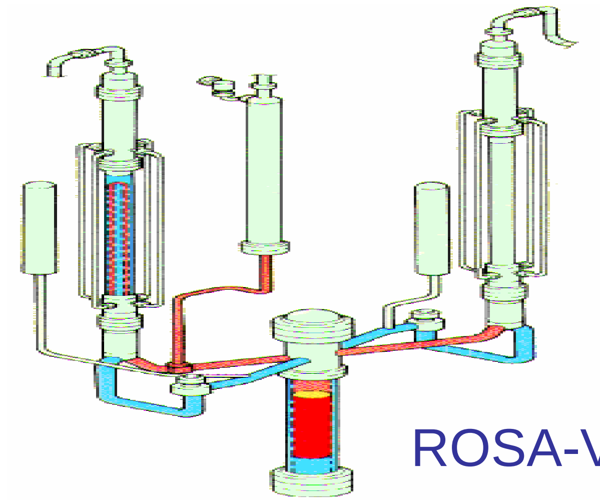
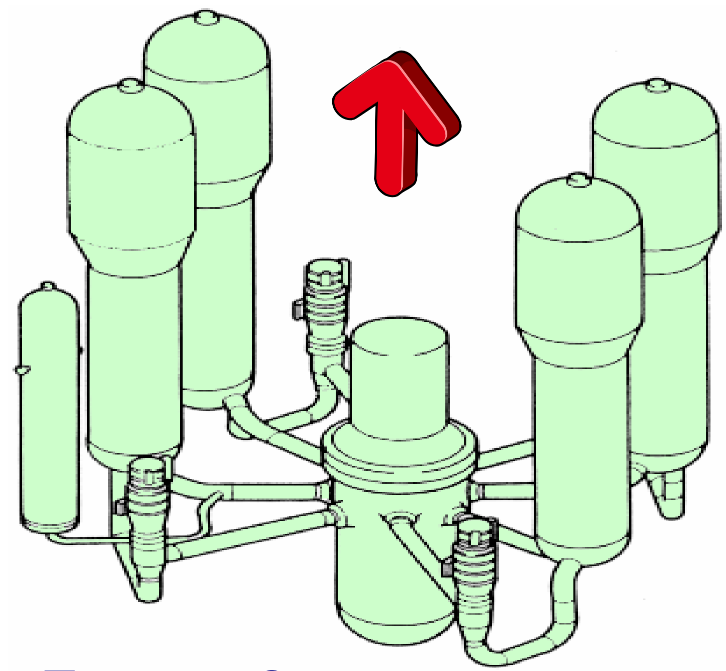
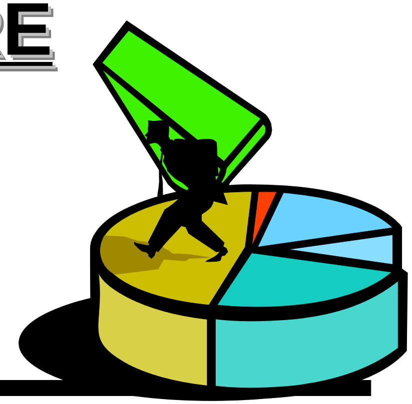
- **<u>ROSA-V ('91-Present):</u>**
  - <u>Objective:</u> AM measures for prevention of severe core damage in beyond-design-basis accident situations.
  - Same height and volume as ROSA-IV
  - LSTF modified to meet new objectives:
    - Core: 10 MW electrical heater with 1588 heater rods
    - ~1600 instruments
  - '92- AP600 with USNRC
  - '98- Passive Containment Cooling System (PCCS) for BWR
  - '05- **OECD ROSA Project**


ROSA-V


Tsuruga-2

OECD PROGRAM STRUCTURE

Organizations of host country, JAEA + Japan Nuclear Energy Safety (JNES), share operating costs with program sponsors



1. Czech Rep. – UJV
2. Belgium - AVN
3. Finland – STUK
4. France
   - IRSN
   - CEA
   - FRAMATOME
   - EDF/SEPTEN
5. Germany – GRS
6. Hungary – KFKI
7. Japan
   - JNES
   - JAEA
8. Rep. Korea – KAERI
9. Spain – CSN
10. Switzerland – PSI
11. UK – HSE/NII
12. USA - NRC

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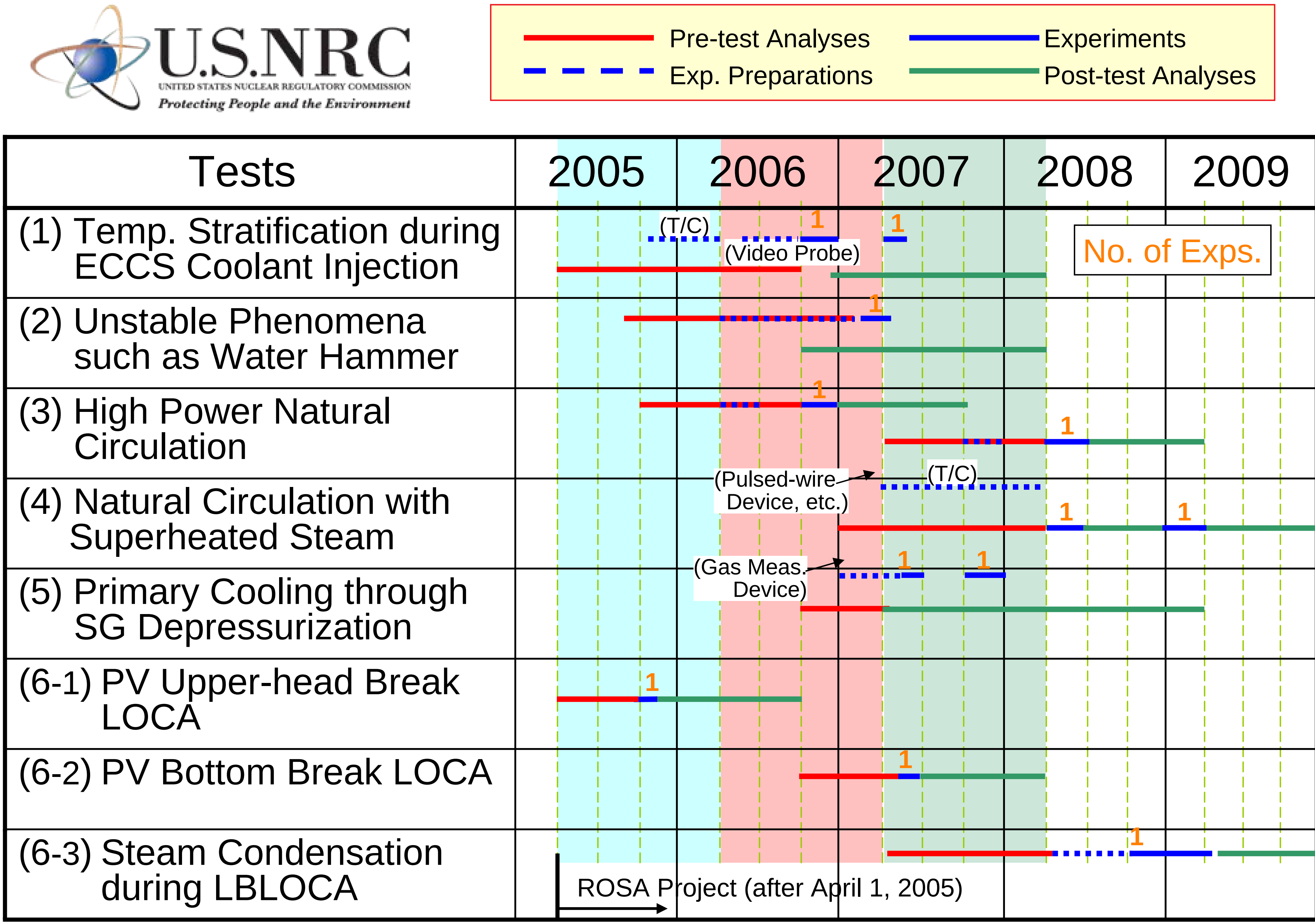

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PROGRAM PROCEDURES

Each experiment generally involves the following tasks:

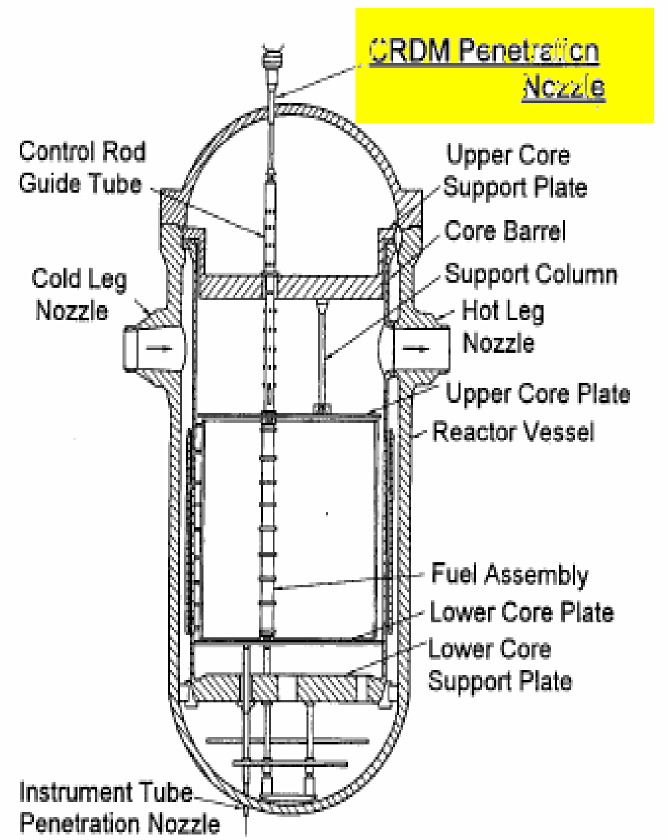
- Pre-test analyses by JAEA to define experimental conditions in reference to reactor simulation analyses
- Installation of instrumentation and/or facility modifications
- Integral or separate-effect experiments with data processing
- Post-test analyses by Participants including JAEA and discussions with the Program Review Group (PRG).

Test 6-1

1.9% Pressure Vessel Upper-head Small Break LOCA Experiment

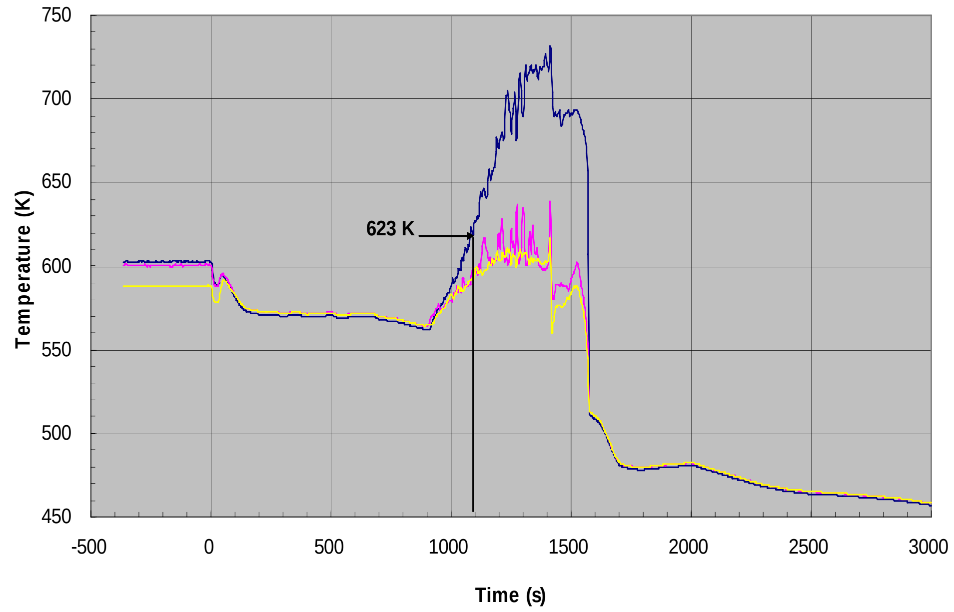
- Impetus: Primary water stress corrosion cracking (PWSCC) in the penetration nozzle of the control-rod drive mechanism (CRDM) could cause a SBLOCA at the vessel head of a PWR.

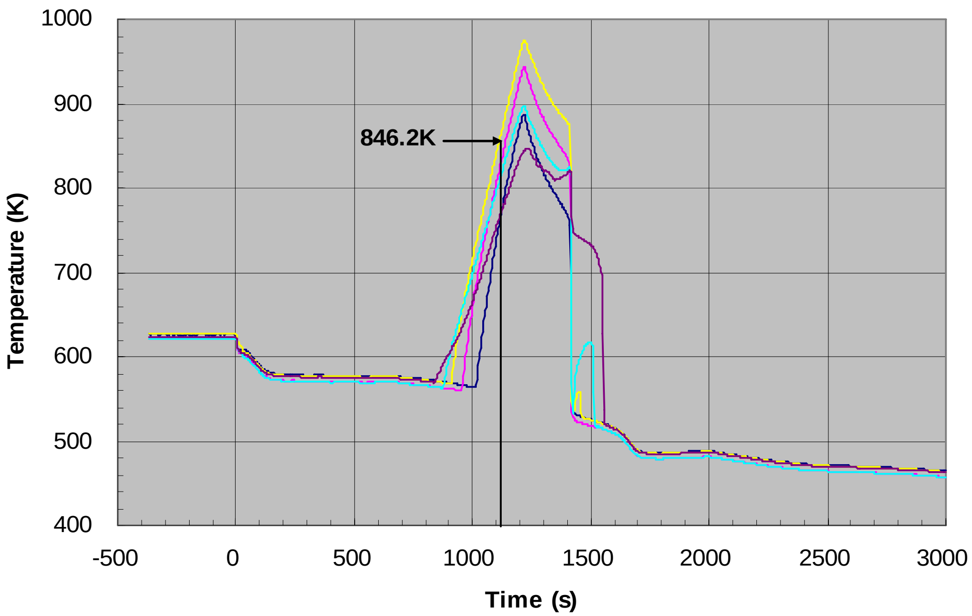
- Experimental Conditions:
  - Break size is 1.9% cold leg equivalent. (Equivalent to the ejection of one whole CRDM)
  - Total failure of high pressure injection (HPI)
  - Loss of off-site power
  - Symmetrical steam generator (SG) secondary-side depressurization as accident management (AM) strategy initiated after core exit temperature reaches 623K



Test 6-1: Experimental Findings

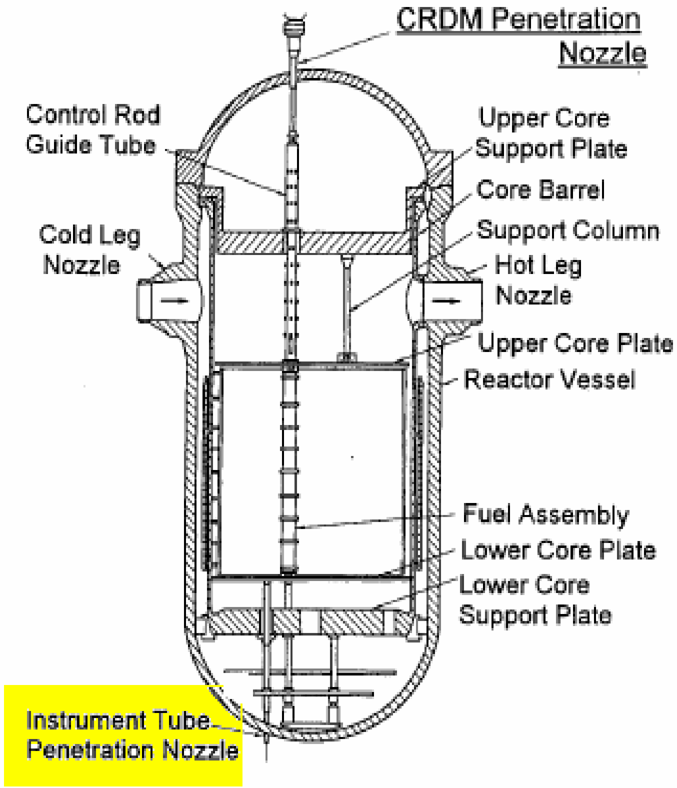
- Core exit thermocouples affected by reflux condensation
  - Core super-heat indication delayed ~200s
- Liquid level in upper-head controlled break flow
- Relatively large break size resulted in fast primary depressurization
- Fuel temperature increased faster than 1-D post-test calculation
  - Analyses require multi-dimensional core
- Effectiveness of AM strongly dependent on timing of core-exit temperature indication
  - Power shut off to electrical rods when temperature reached 958K





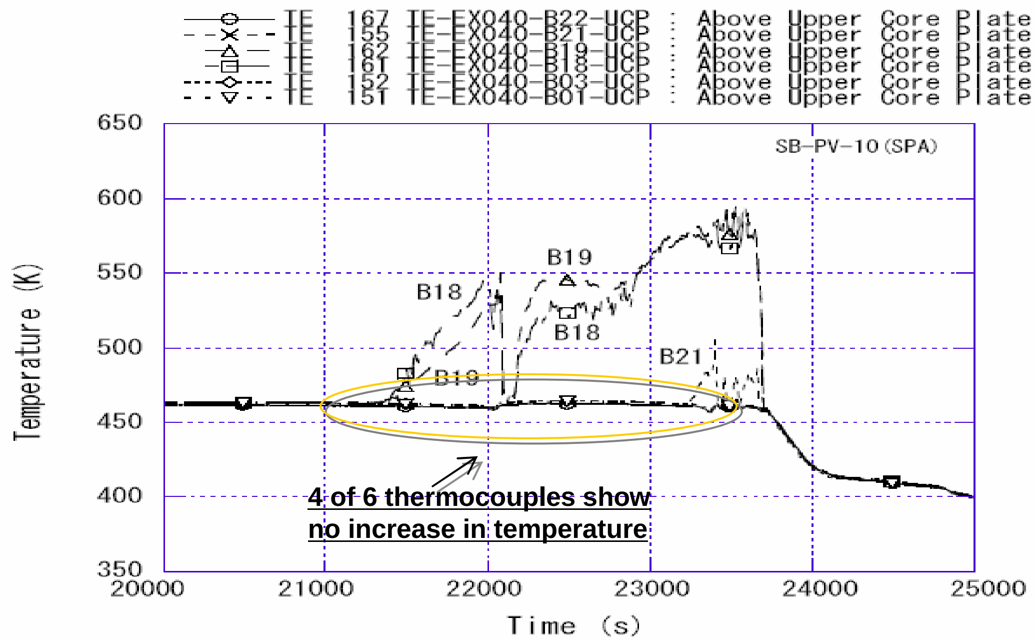
Test 6-2
0.1% Pressure Vessel Bottom Small Break LOCA Experiment

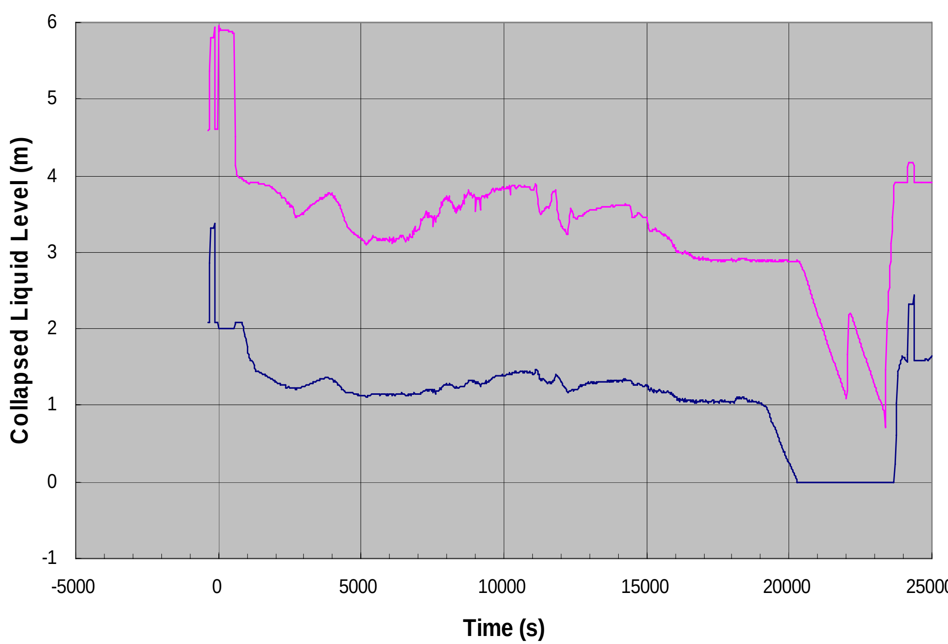
- Impetus: PWSCC around the circumference of two instrument-tube penetration nozzles on the lower-head of a PWR could cause a SBLOCA.
- Experimental conditions :
  - Break size is 0.1% cold leg equivalent. (Equivalent to the ejection of one whole instrument tube)
  - Total failure of HPI
  - Loss of off-site power
  - Asymmetrical SG secondary-side depressurization as AM strategy Initiated 30s after safety injection (SI) signal to achieve 55 K/h cooling in primary system



Test 6-2: Experimental Findings

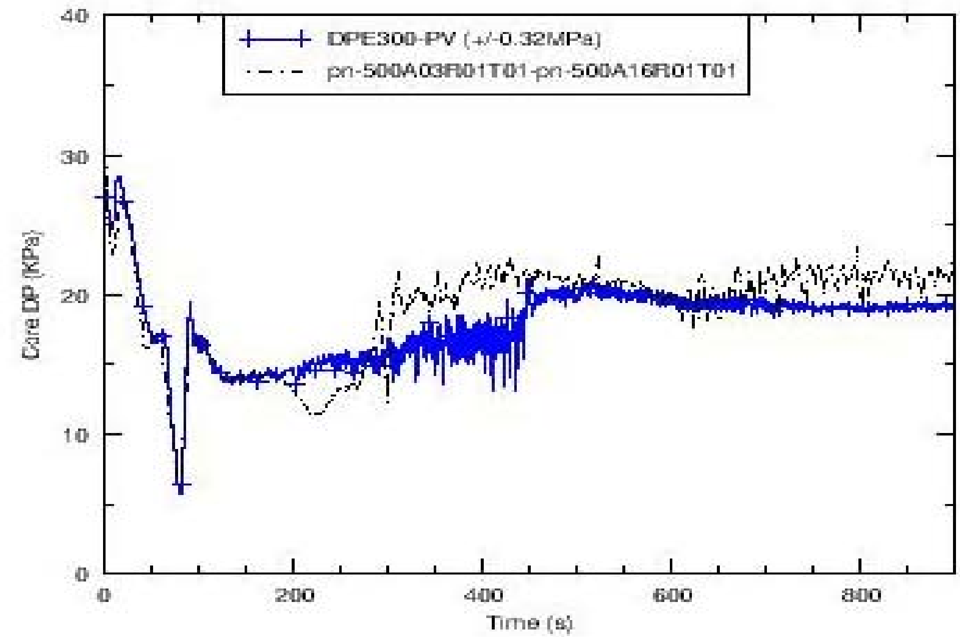
- Core exit thermocouples indicate superheating with asymmetrical distribution after initiation of core uncovery.
  - Thermocouples on loop w/o pressurizer gave no indication of super-heat
- The AM action initiated at 2540s aided in cooling the primary system until gas inflow started from the accumulator injection system.
- The core uncovery began again due to an interruption of the low pressure injection (LPI) system when the primary pressure increased by steam generation following the core quench.

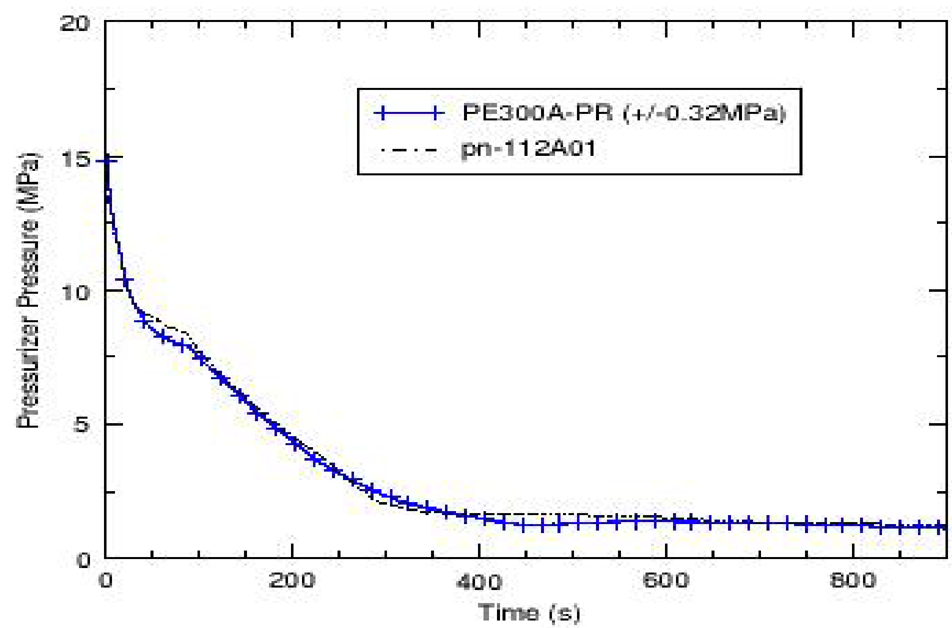




TRACE Assessment

- ROSA-IV data is being used to assess the latest version of TRACE, Version 5.0
- ROSA-V input decks for TRACE are currently being developed to assess the code against the latest experiments





Summary

- ROSA continues to play an important role in the identification, understanding, and modeling of relevant thermal-hydraulic safety issues

- Data generated in current and future ROSA experiments will be used to continually enhance the audit capability of TRACE