Accident Analysis Design Basis and Beyond Design Basis

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Quadripartite Working Group Meeting
October 9-10, 2008
Paris, France

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Design Basis Accident Analysis

- **Licensees must show for design basis accidents that the Emergency Core Cooling System (ECCS) and other systems function effectively for design basis events selected to bound anticipated (frequency >~10^{-4} /yr) events**
  - **Limiting event is typically a large break in the coolant system**
  - **Analysis is mainly thermalhydraulics though the issues concern materials and structures**
    - **Core can be shutdown**
    - **Core maintains geometry and is coolable**

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Plant Analyses

- Most plants licensed under prescriptive, conservative requirements for thermalhydraulic analyses
  - Peak temperature on fuel cladding < 2200 F
  - < 17% clad oxidation locally
  - < 1% overall clad oxidation
- Conservative thermalhydraulic analyses can limit power uprates especially for PWRs
  - NRC does allow “best estimate” plant analyses with identification and propagation of uncertainties in the calculations

NRC Thermalhydraulic Analyses

- Used as independent “check” on licensee analyses
- In the past, NRC maintained thermalhydraulic codes for various plant types
  - Ramona
  - RELAP5
  - TRAC-P, TRAC-B
- NRC is consolidating its thermalhydraulic analyses into the TRACE code which is coupled to the PARCS neutronics code.
  - TRAC RELAP Advanced Computational Engine

TRACE Development

- Distributed network of developers
- NRC maintains code configuration
- Model developers
  - NRC (~5 developers)
  - National laboratories
  - Universities
  - Engineering consulting firms

TRACE

- Component-oriented reactor systems analysis code to analyze design basis accidents
  - Large break loss-of-coolant accidents
  - Small break loss-of-coolant accidents
  - Transients
- Two-fluids for gas-liquid flow
  - Plus non-condensable gas and dissolved solute (boric acid)
- Modular by component for LWRs

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CLOSURE Relationships

- **TRACE is semi-empirical and requires experimental data for the closure of field equations:**
  - Equation of state
  - Wall drag
  - Interfacial drag
  - Wall heat transfer
  - Interfacial heat transfer
- **TRACE can be validated only for specific ranges and specific applications**

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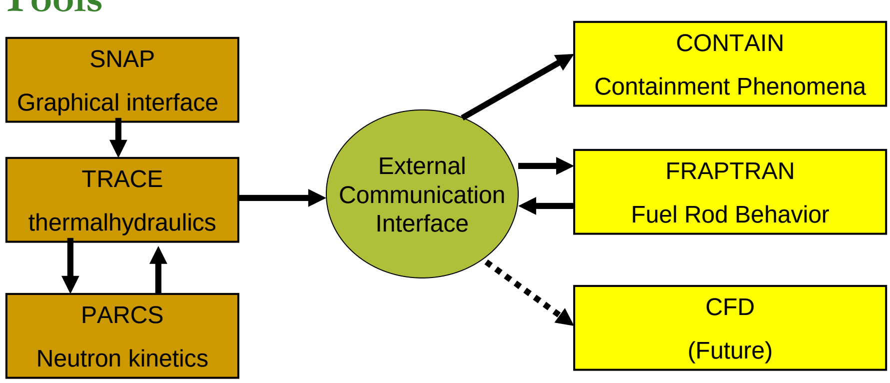
TRACE Couples to Other Accident Analysis Tools



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TRACE

- Used for existing reactors
  - At least as good as specialized codes used in the past
- Being used for advanced LWRs
  - AP1000
  - ESBWR
  - ACR-700 (suspended)
  - EPR started
  - APWR
  - ABWR

TRACE Assessment

- 400 experiments in 35 facilities
- Separate Effects Tests
  - Dartmouth CCFL
  - FRIGG – 36 rods
  - RBHT – 45 rods
  - THETIS – 61 rods
  - THTF – 64 rods
  - Dehbi Condensation
  - Marviken
  - UCB-Kuhn
  - FLECHT – 161 rods
  - G-2 336 rods
  - UPTF full scale
- Small scale and Larger Scale integral tests

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TRACE Assessment

- Small scale and larger scale integral tests (Volume scaling)
  - Semi-scale 1:1600
  - FIX-II 1:777
  - FIST 1:624
  - SPES 1:430
  - PUMA 1:400
  - APEX 1:192
  - BETHSY 1:100
  - LOFT 1:50
  - ROSA 1:48
  - CCTF 1:25
  - SCTF 1:25

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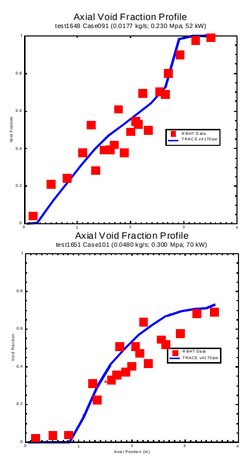
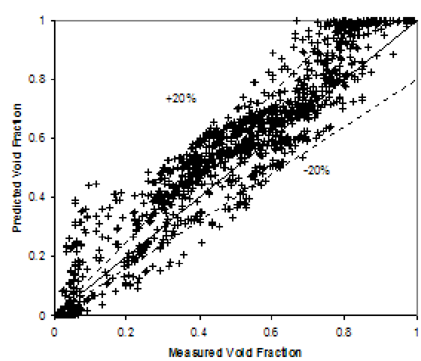
RBHT Mixture Level Swell Tests





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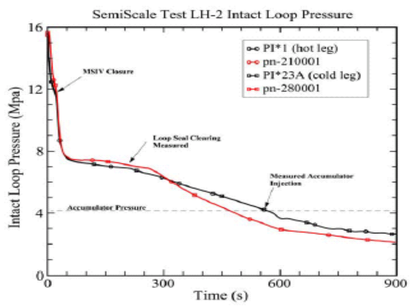
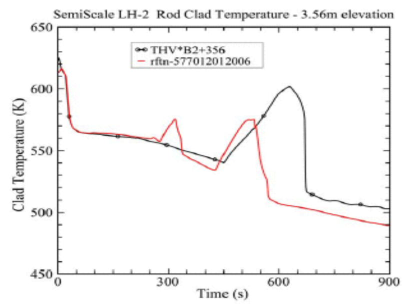
TRACE SBLOCA Assessment

SemiScale Test LH-2 – Scaled 5% Cold Leg Break LOCA

TRACE and Experimental Data Comparison





The TRACE code prediction compared well with data during the first part of the transient. However, after about 250 seconds, the break effluent became mostly steam and depressurization occurred too rapidly. This caused an early accumulator discharge, a reduction in PCT and early core quenching. Test LH-1 showed similar behavior.

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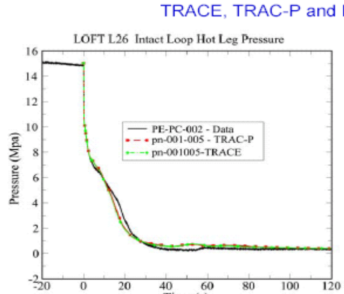
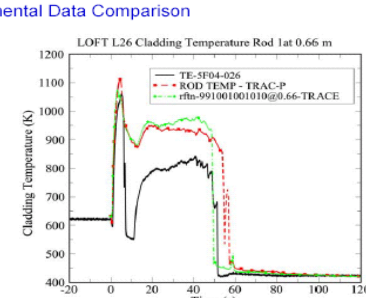
TRACE Preliminary Assessments

Step 1: Demonstrate code consolidation

LOFT Test L2-6 - 200% Cold Leg Break Test.

TRACE, TRAC-P and Experimental Data Comparison





- **Hydraulic parameters generally compare very well with data.**
- **Both Codes yield similar output.**

- **Test Data exhibits an early core rewet not predicted by either code.**
- **Both Codes over-predict the sustained rod heat-up.**

TRACE Assessment

- Assessment is identifying areas for improvement of the code
- Advanced light water reactors do pose challenges that will prompt improvements
  - Passive safety features especially
- Industry expected to continue to press the regulatory envelop for power uprates
  - Suspect that industry already has codes for specialized applications that outstrip TRACE capabilities

Future for Thermalhydraulics

- LWRs focus of regulatory attention for the next 20 years, at least
  - LWR technology will move closer to regulatory limits with ever more sophisticated analyses
- Concern about availability of experimental data adequate to validate TRACE with expanded capabilities
  - Particular concern for full height facilities to test passive emergency core cooling features
- The role of CFD modeling in the future
  - Concern over the reliability of commercial CFD models with unknown "patches" and "tricks" to get convergence
  - Little data for adequate validation of two-phase flow predictions by CFD

Severe Accident Analysis

- **Over a decade following the accident at Three Mile Island, the NRC spent about $0.5 billion to understand reactor accidents that went beyond the design basis**
  - **Did NRC need to include severe accidents in the regulatory framework to provide "…adequate protection of the public health and safety"?**

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Severe Accident Analysis

- **Wide ranging research program**
  - **Fuel degradation**
  - **Fission product release and transport**
  - **Loads on containment**
  - **Aerosol behavior and engineered safety features**
  - **Containment response**
- **Concluded that existing licensing and regulation did provide adequate protection**
  - **Research results basis of Level III PRA of five representative plants**
- **Recognized that understanding of severe accidents was very incomplete**

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Severe Accident Strategy

- Continue to participate in severe accident research on a lower priority basis in competition with other research needs
  - Pioneering program had done the “easy” stuff
  - Continuing severe accident research very expensive
  - Cooperative research with other countries
  - Support regulatory processes
- Develop a computational vehicle for the preservation of knowledge gained and to be gained in severe accident research
  - MELCOR computer code

Beyond Design Basis Accident Analysis

- The only regulated aspect of severe accidents is hydrogen generation
  - Inerted containments for Mark-I and Mark-II BWRs
  - Hydrogen igniters for ice condenser containment PWRs and Mark-III BWRs
- Severe accident source terms from fuel degradation used for defense-in-depth assessment of containment features in siting analysis
  - Source term from severe fuel degradation to containment
  - Licensees show natural and engineered mitigation sufficient to limit doses at site boundary (< 25 rem TEDE) and control room (< 5 rem TEDE)
- Severe accident analysis used in Level II and Level III risk analyses

Severe Accident Analysis

- NRC routinely analyzes accident source terms for sitting decisions
  - MELCOR
  - RADTRAD
- Licensees typically use the MAAP code or other specialized analysis method
  - More conservative of the results becomes part of the licensing basis

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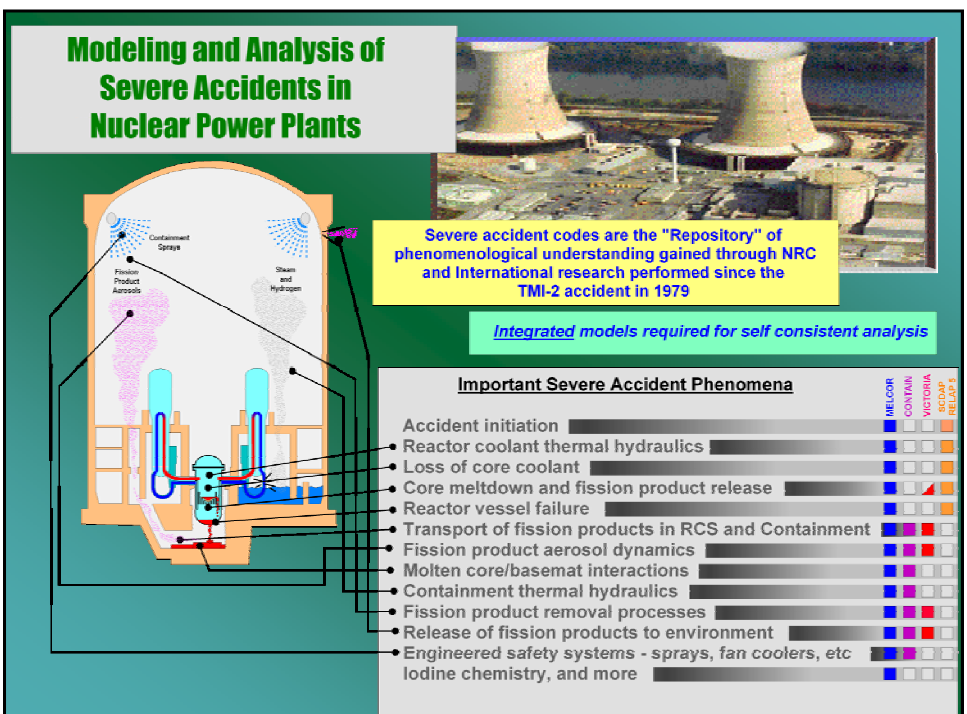
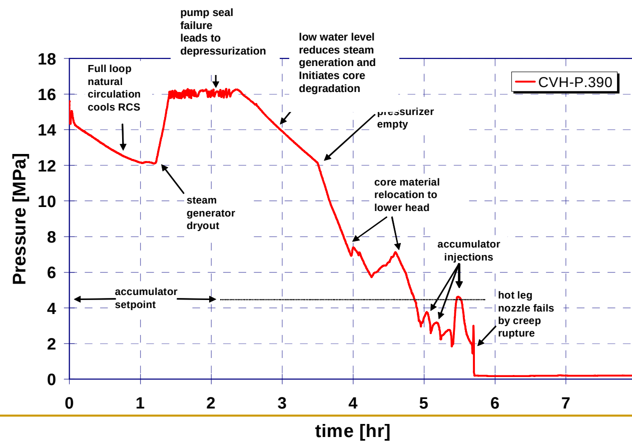


Example calculation of early stages of a station blackout event at a pressurized water reactor calculated with MELCOR



MELCOR Modeling of Core Degradation

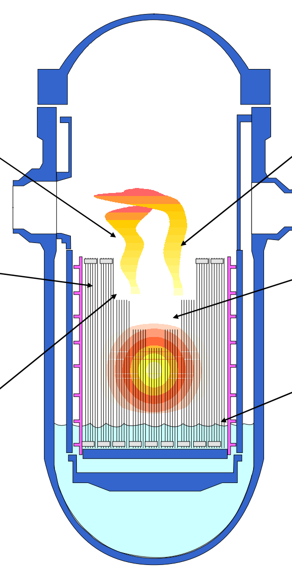
Fission product specie/volatility modified (Cs_2MoO_4) – Phebus Tests – affects RCS deposition

RN Package expanded to allow analysis of FP release from mixed MOX/LEU core

(French VERCORS and RT tests)

Fuel failure criteria expanded via control function – Phebus tests – affects hydrogen generation and melt progression

BWR failure criteria expanded



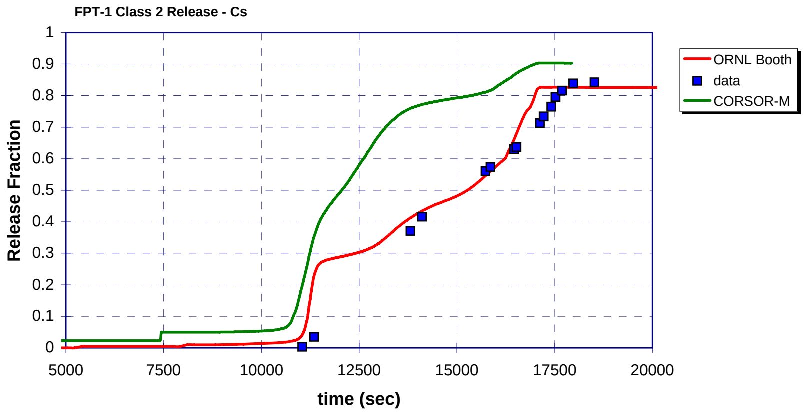
Ag release model added – Phebus Tests – important for iodine chemistry

- Important to agglomeration

B_4C oxidation model added (PWR) – QUENCH Tests, Phebus FPT-3

Quench-reflood modeling – QUENCH tests – quench front not necessarily water level

MELCOR Calculation of Cs Release in the PHÉBUS FPT-1 Test



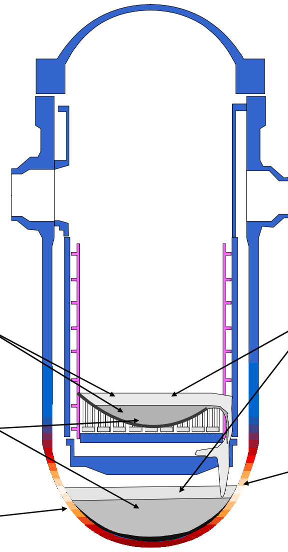
MELCOR Modeling of Late Stage Core Degradation

Molten pool stratification into light and heavy layers in core and lower head (RASPLAV,MASCA)

Fission product partitioning between melt phases (RASPLAV,MASCA)

Natural convection heat transfer in circulating melt pools

Insights from OECD LH program in creep-rupture models for lower head



Core periphery baffle and formers added to COR package – allows TMI-like side wall melt release

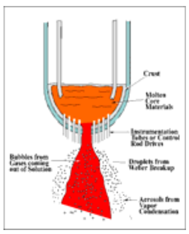
Molten pool / crust model added to core and lower head regions (TMI-2)

Curved lower head description added to COR (BH integration) including 2-D heat transfer

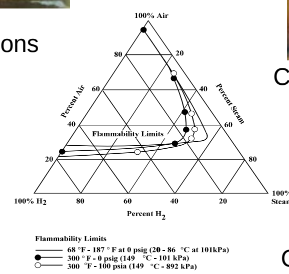
Energetic Phenomena in Containment

Steam Explosions

Core Debris Interactions with Concrete

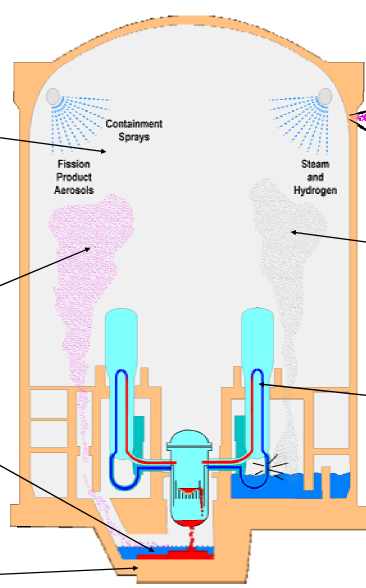


Direct Containment Heating



Hydrogen Combustion

MELCOR Modeling of Containment Phenomena



Control function specification of volume velocities for heat transfer added

Total airborne mass by control volume type added as control function argument

Modifications to MCCI model to explore enhanced cooling by overlying water (ANL ACE and MACE)

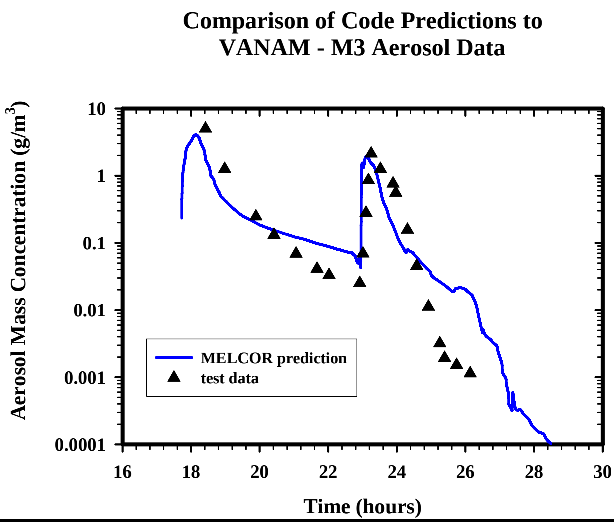
Numerical improvements to CAV package

Fission product flows through flow paths added to plot file (MACCS)

Separate plot variables for hydrogen produced from Zr, SS and B4C added to plot file

Developing improved models for FP retention in steam generators (ARTIST program)

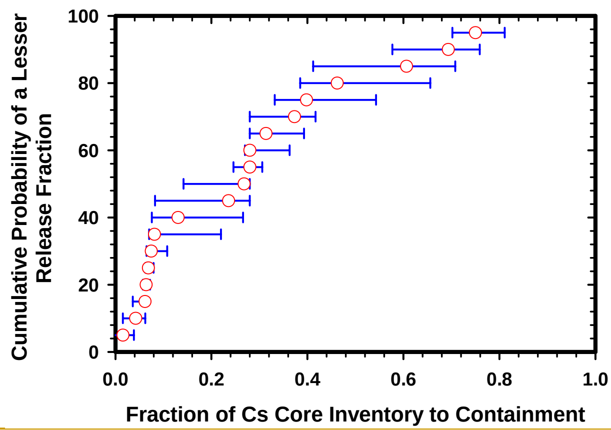
Code Used More for Aerosol Physics than Analysis of Loads on Containments



Uncertainty Analysis

- Growing demand for quantitative uncertainty analysis from phenomenological computer codes
  - Parametric uncertainty
  - Model uncertainty
  - Aleotory uncertainty
- MELCOR has growing capabilities for computing nonparametric distributions for outputs based on Monte Carlo methods
  - Used extensively for regulatory decisions
    - Emergency power for hydrogen igniters
    - Allowable main steam isolation valve leakage
    - Revised source terms for MOX and High Burnup fuels
  - Confidence intervals for outputs; importance ranking of sources of uncertainty

Uncertainty Distribution for Cesium Release Used for Selecting a Representative Accident for Siting Analysis of Defense in Depth



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MELCOR Status

- **MELCOR strength is the code structure**
  - Now available in Fortran 95
  - Major assessment effort in next few years
  - Currently doing definitive TMI calculation
- **MELCOR weakness is chemical transformations of fission products in the RCS and containment**
  - VICTORIA
  - Outcomes of PHEBUS-FP, - IST, BIP, etc
- **MELCOR having a bigger impact on regulatory activities than might be anticipated**

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Future Severe Accident Analysis

- **Gas-cooled reactors**
  - Phenomena identification and ranking completed for fuel and accidents
  - Suspect a new code will be developed
  - MELCOR being used for reactor development
- **Liquid metal-cooled reactors**
  - Advanced burner reactor
  - BRISC code structure effort
  - Source term considerations
  - MELCOR modified to include point kinetics neutronics