Analysis of the Versatile Test Reactor Using RELAP5-3D

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Outline

- Scope
- VTR main characteristics
- Suitability of RELAP5-3D for SFR simulation
- RELAP5-3D VTR Modeling & Simulation
  - Preliminary Safety Analysis for Protected/Unprotected Transients
- Conclusions

Scope

- US-DOE established the Versatile Test Reactor (VTR) program in February 2017
  - Reports indicated an **existing gap** between current **fast neutron irradiation capabilities** in US and needs of different stakeholders
  - 3-years R&D effort to enable an informed DOE decision on the further development of a fast neutron source
  - INL is leading the R&D program and it is the **proposed hosting site**
  - INL expanding its technical capabilities on SFR technology leveraging its staff expertise on experiments designs, reactors operation, codes development and assessment, nuclear safety analyses

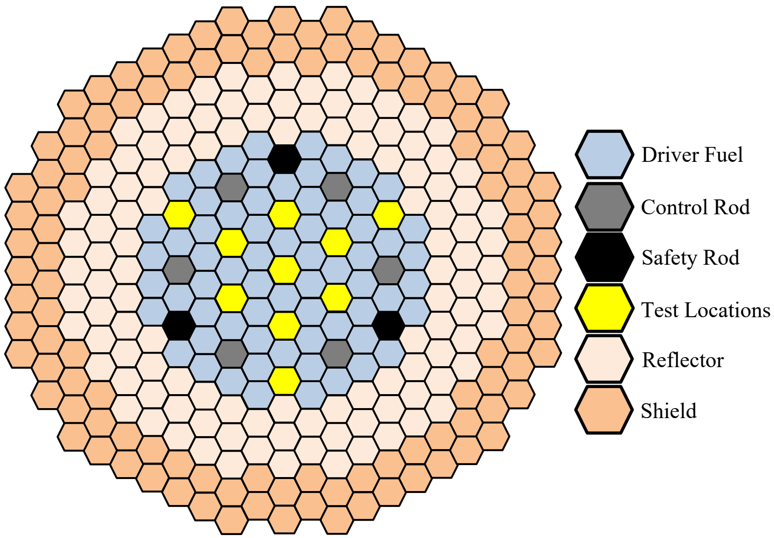
Scope

- Reference VTR thermal-hydraulic (TH) design and model being developed by Argonne National Lab (ANL) using SAS4A/SASSY code
- INL developed an independent RELAP5-3D model
  - leveraging INL expertise in RELAP5-3D code development, modeling & simulation (M&S)
  - quicker learning process on SFR technology
- Following best practices outlined in IAEA SSG-2, " Deterministic Safety Analysis for Nuclear Power Plant"
  - "*The operating organization shall ensure that **an independent verification of the safety assessment is performed by individuals or groups separate from those carrying out the design, before the design is submitted to the regulatory body**". Additional independent analyses of selected aspects may also be carried out by or on behalf of the regulatory body."*

VTR main characteristics

- Remark: following information was developed before the selection of GE-Hitachi (GEH) Nuclear Energy PRISM technology by VTR program
- Conceptual core main characteristics

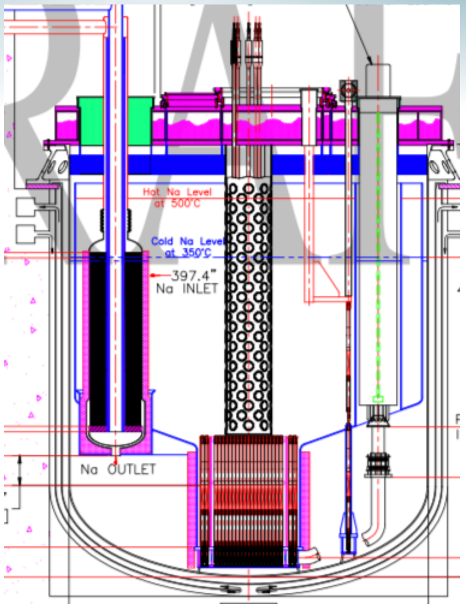
Parameters	Value
Core Power (MWth)	300
Peak Fast Neutron Flux (n/cm2 s)	>4.0×10^{15}
Number of Fuel Assemblies	66
Number of Radial Reflector Assemblies	114
Number of Shield Reflector Assemblies	114
Assembly Length (m)	3.53
Control Rods (Control + Safety)	6 + 3
Assembly pitch (cm)	12
Fuel Height (cm)	80
Plenum Height (cm)	80



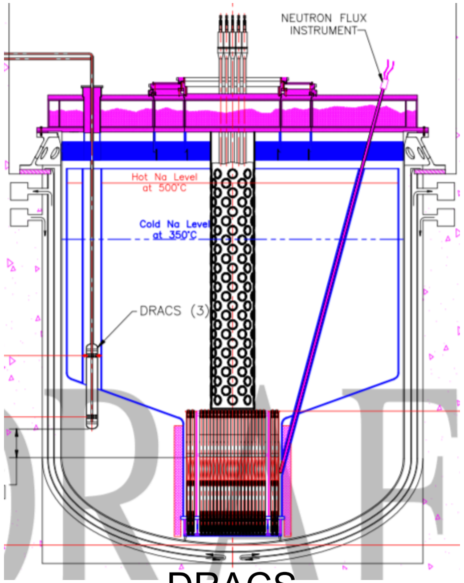
VTR Core Layout

VTR main characteristics

- Use of metallic fuel (U-20Pu-10Zr, 5% enriched U)
- Reactivity control by two independent systems
  - Primary system: 6 CR
  - Secondary system: 3 safety CR
- Pool type configuration
  - Primary Sodium Mass flow: 1566 Kg/s
  - 2 MCP, 2 IHX
  - Sodium Core Inlet/Outlet temp: 350/500 C
- Secondary side IHX dumping heat into atmosphere via air blowers
- Direct Reactor Auxiliary Cooling System (DRACS) remove heat from cold pool
  - 3 Na-K Natural Draft Heat Exchangers (NDHX)



VTR layout



DRACS

VTR main characteristics

- Safety Limits:
  - Limits for Normal Operation (metallic fuel):
    - 650 ºC for fuel clad
    - 1,121 ºC for U-20Pu-10Zr
    - q’ < 450 W/cm
  - DBA FOMs:
    - Peak Internal Cladding Temperature (< 650 ºC)
    - Bulk Coolant temperatures for Hot/Cold Pools (< 650 ºC)
    - Subcriticality
  - BDBA FOMs:
    - Peak Internal Cladding Temperature (< 788 ºC)
    - Fuel Centerline Temperature (< 1119 ºC)
    - Long-term Structural Materials Temperature (< 704 ºC)

VTR main characteristics

- Demonstrate the “inherent safety” of SFR w/ Metal Fuel (EBR-II experience) for BDBA → decrease the probability of severe accidents
  - **Inherent Shutdown**: use of negative reactivity feedbacks (fuel, core, CR expansions) generated by high temperature transients
  - **Passive shutdown heat removal**: use of Natural Circulation for removing decay heat

Suitability of RELAP5-3D for VTR M&S

- RELAP5-3D can model Liquid Metal circuits → Liquid Metal properties files available, Na, Na/K, Lead, Lead/Bismuth
- Dedicated correlation for core thermal heat exchange → Westinghouse/Cheng-Todreas correlation for wire-wrapped hexagonal rod bundles
- Suitability of RELAP5-3D for SFR simulations detailed in: C.B. Davis, “Applicability of RELAP5-3D for Thermal-hydraulic Analyses of a Sodium-Cooled Actinide Burner Test Reactor”, INL/EXT-06-11518 (2006)
- Several peer-reviewed journal articles showed successful application of RELAP5-3D code for simulating SFR sub-channels and plant transients (e.g., see IAEA EBR-II CRP)

Suitability of RELAP5-3D for VTR M&S

- Special RELAP5-3D modeling techniques for VTR
  - Use of Control Variables + Servo-valve for modelling wire-wrapped rod bundle friction losses in the laminar/transition region
  - VTR Fuel Pitch/Diameter (P/D) = 1.18 → Westinghouse correlation OK for heat transfer
  - Use of Control variables for modeling several reactivity feedbacks (e.g., Core axial and radial expansion, CR and Vessel expansion)
- Remark: scenarios involving sodium boiling and disruptive core events cannot be simulated using RELAP5-3D → use SAS4A/SASSY & other SA codes

RELAP5-3D VTR M&S

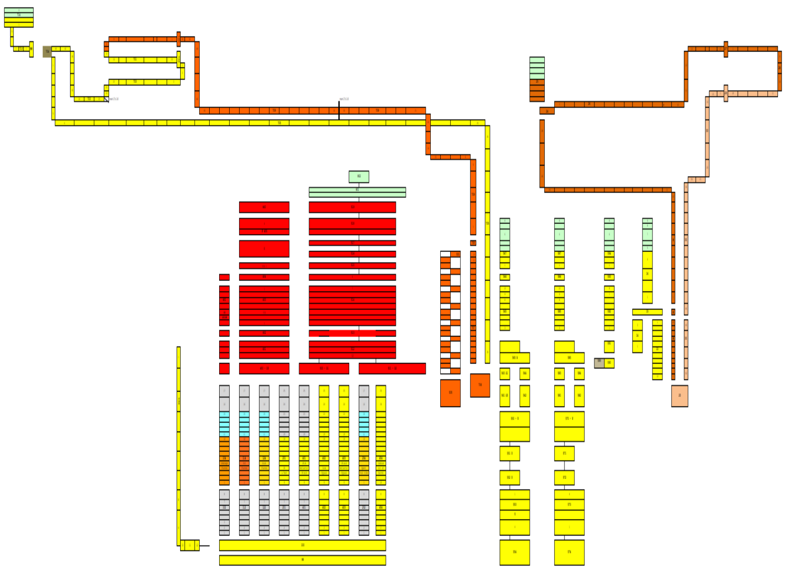
- Criteria for VTR M&S:
  - Preserve relevant elevation, masses, flow areas, flow paths → ref.: ANL and INL design documentation
  - Modeling:
    - relevant components (core S/A, IHX, DRACS, pumps, NDHX, expansion tanks)
    - relevant flow paths (core, bypass, UIS, hot & cold pool, secondary and SHRS flows)
    - Important heat transfer mechanisms (core, core/bypass, IHX, CR and vessel expansion, cold pool – DRACS, heat sink with the atmosphere)
    - 0D Neutron Kinetics feedbacks

RELAP5-3D VTR M&S

- 7 channels core, 3 orifice zones
  - Average Channel (64 S/A) , Hot Channel (1 S/A), Cold Channel (1S/A)
  - CR/Test section channel (25 S/A)
  - Reflector/Shim/Shield channel (240 S/A)
  - Core Bypass (x 66)
  - Reflector Bypass (x 265)
- 2 Fuel materials describing different level of irradiations
- Primary side: Hot/Cold pools + IHX, Main Circulation Pump (MCP), DRACS shell
- Secondary side: IHX + pipelines + secondary MCP + Expansion Tank / DRACS+ pipelines + Expansion Tank + NDHX
- Heat Sink: Atmosphere & Blowers modeled as Boundary Conditions

RELAP5-3D VTR M&S

- RELAP5-3D Model stats
  - Number of Heat Structures: **237** (861 mesh points)
  - Number of Volumes: **656** (675 junctions)
  - **4,800** lines input deck
  - Control and Protection System Logic
  - 0D – Neutron Kinetics (NK)
- Steady State Results consistent with design specifications



VTR RELAP5-3D nodalization

Parameters	Value
Core Power (MWth)	300.0
Core Inlet Temperature (°C)	350.0
Core Outlet Temperature (°C)	500.0
Peak Fuel Temperature (°C)	804.5
Peak Cladding Temperature (°C)	564.0
Peak Coolant Temperature (°C)	532.6
Core Mass Flow Rate (Kg/s)	1566.1
Core Pressure Drop – Total (MPa)	0.53
Primary Pump Head (MPa)	0.563
IHTS Mass Flow Rate – Total (Kg/s)	1558.4
IHX Intermediate Inlet Temperature (°C)	327.9
IHX Intermediate Outlet Temperature (°C)	478.3

Steady State Results

RELAP5-3D VTR M&S

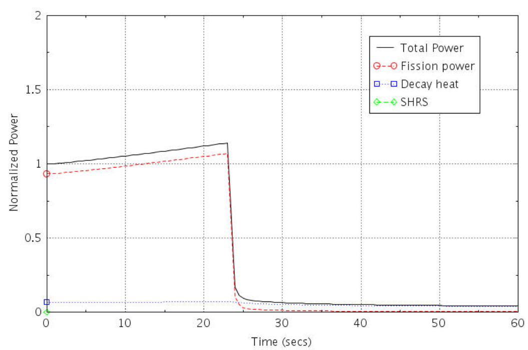
- Implementation of the Control & Protection System
  - Combination of logical trips and control variables
  - Regulate:
    - primary and secondary flows
    - heat sink rate (i.e., blower heat rejection) for secondary cold leg temperature control
    - NDHX heat rejection (i.e., atmosphere temperature) for Natural circulation in DRACS
  - Ref. ANL-VTR4 report for the Protection System actuation logic

Parameter	Threshold	Delay (s)
Reactor Power	115%	0.3
Power-to-Flow Ratio	115%	0.4
Average Core Outlet Temperature	+20°C	2.2
Average Core Inlet Temperature	+20°C	1.8

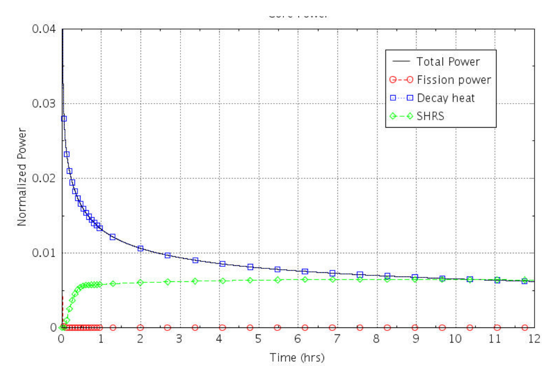
Trigger	Delay (s)	Action
RPS Reactor Trip Signal Sent	0.1	Complete Control Rod Scram
Initiation of Control Rod Scram	0.2	PHTS & IHTS Pumps Trip
Initiation of PHTS & IHTS Pump Trips	2.0	DHX Heat Rejection Terminated
RPS Reactor Trip Signal Sent	10.0	SHRS Air Dampers Open

Transients calculations: PTOP

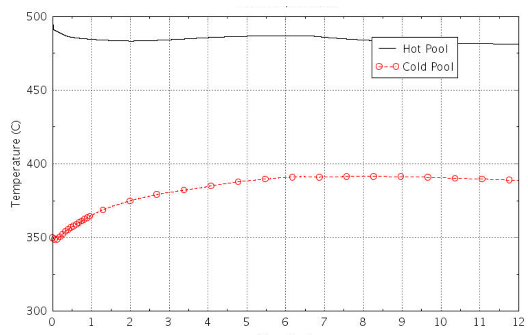
- Protected Transient Overpower (PTOP)
  - Assume CR Reactivity insertion step: 87 c in 175 seconds
  - Reactor scram at t=21 secs after core outlet temperature surges +20 ºC
  - Cooldown after pump trip by DRACS → power removed = decay heat at t=10 hrs
  - Significant safety margins



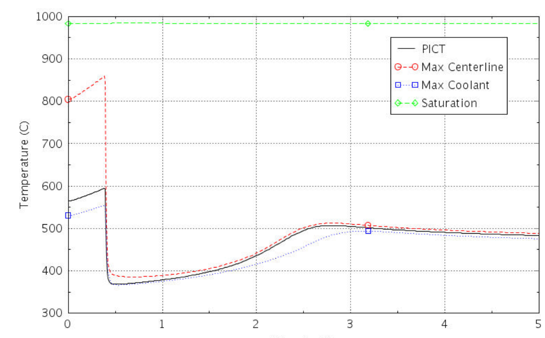
Reactor Power – Short term



Reactor Power – Long term



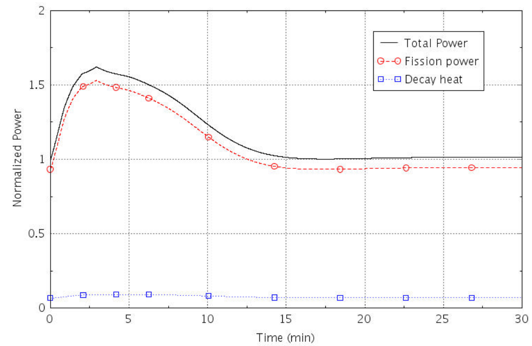
Cold and Hot Pool Temperatures



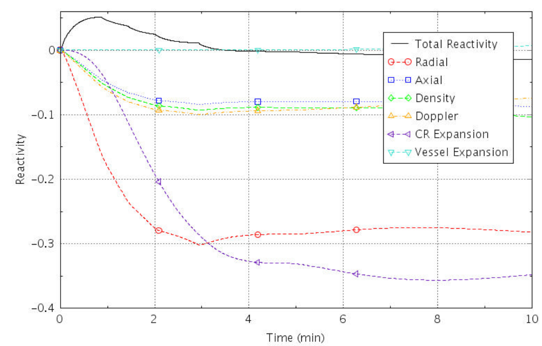
Fuel and Coolant Max Temperatures

Transients calculations: UTOP

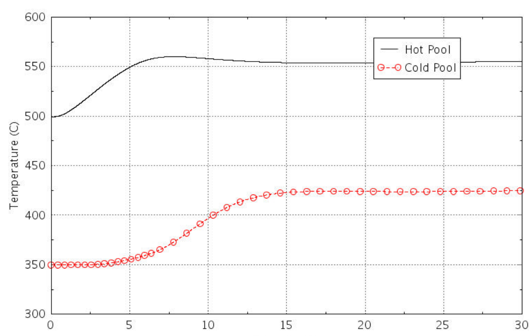
- Un-Protected Transient Overpower (UTOP)
  - Assume CR Reactivity insertion step: 87 c in 175 seconds
  - No Reactor scram, no blower trips → Power surge at 161% P_{nom}
  - Negative reactivity coefficients bring reactor power back to P_{nom} at t=+15 minutes
  - Peak Fuel Centerline Temp = 1074 ºC → safety margin ~ 72 ºC



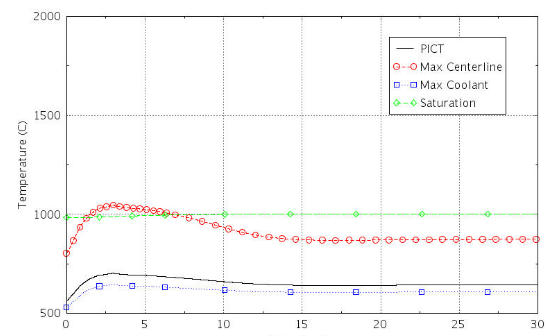
Reactor Power



Reactivity Trend



Cold and Hot Pool Temperatures

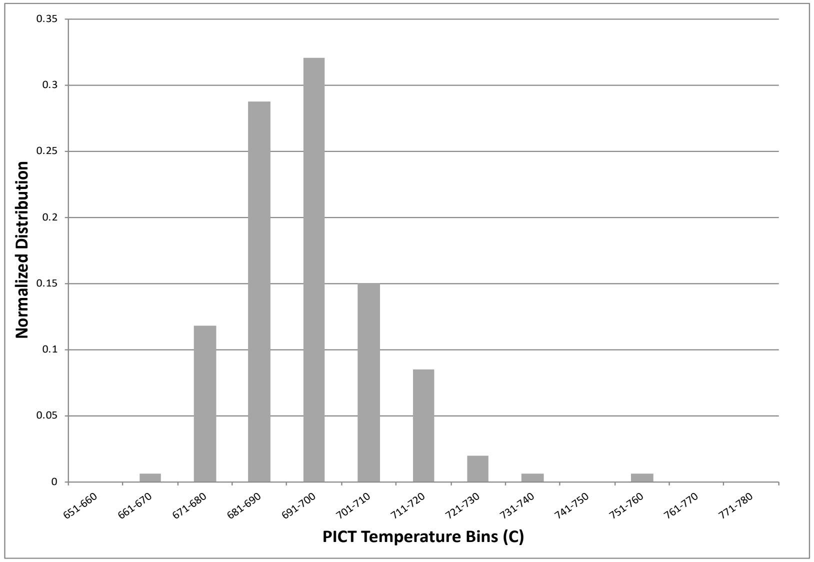


Fuel and Coolant Max Temperatures

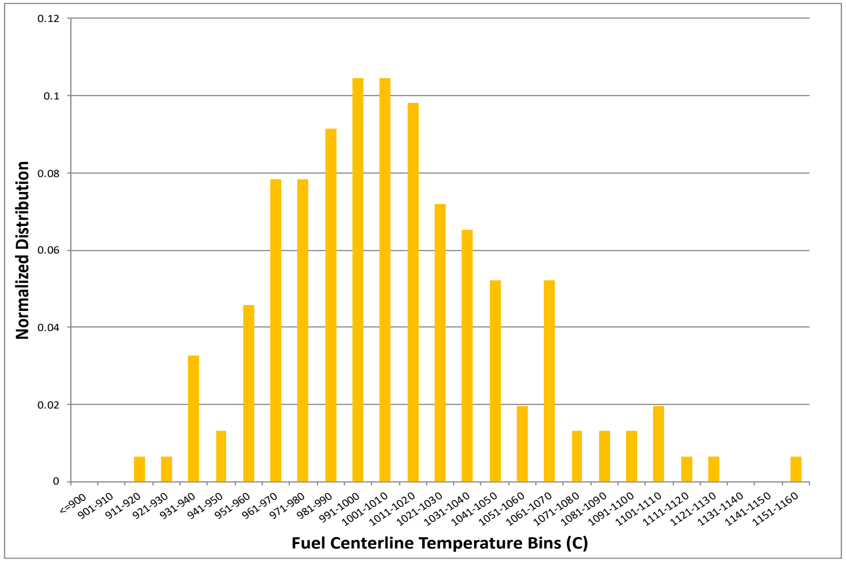
Transients calculations: UTOP

- Monte Carlo calculations for a better understanding of UTOP safety margins → Use of RELAP5-3D/RAVEN
  - 5 input parameters, with normal distribution
  - 153 runs → get 95%/95% confidence/probability values
  - PICT = 716 ºC and Peak Fuel Centerline temp = 1107 ºC

Parameters	Sigma
Doppler Coefficient	20%
CR Driveline Expansion Coefficient	20%
Coolant Density Coefficient	20%
Radial Expansion Coefficient	20%
Hot Channel Conductivity	10%



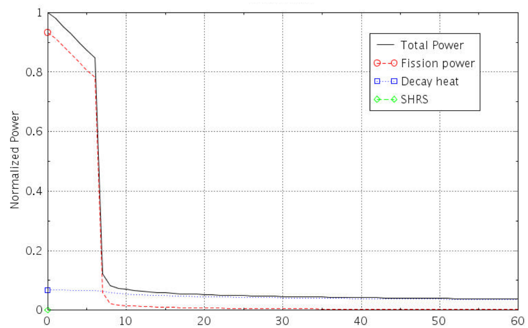
PICT Distribution



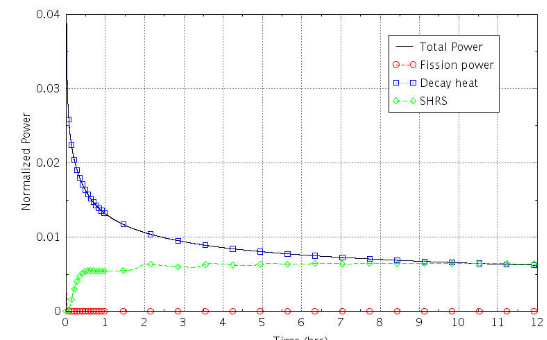
Fuel Centerline Temperature Distribution

Transients calculations: PSBO

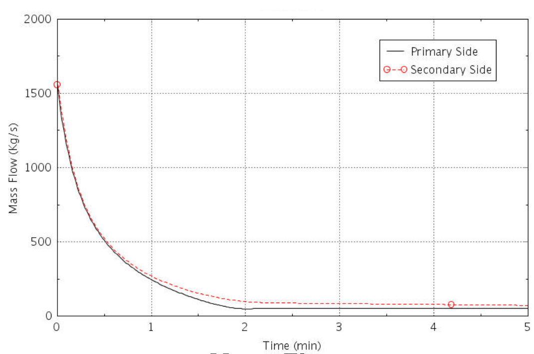
- Protected Station Blackout (PSBO)
  - Assume loss of all AC power → pumps trip and failsafe opening of DRACS HX
  - Reactor scram when power-to-flow ratio > 115 P_{nom}
  - Cooldown after pump trip by DRACS → power removed = decay heat at t=10 hrs
  - Significant safety margins



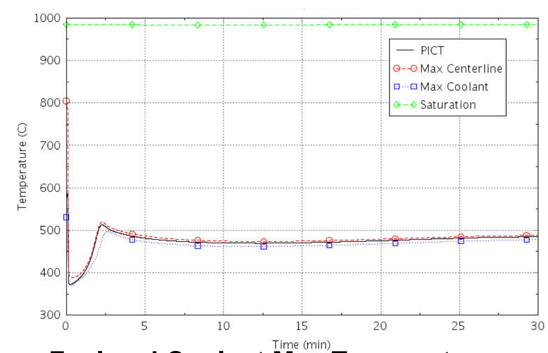
Reactor Power – Short term



Reactor Power – Long term



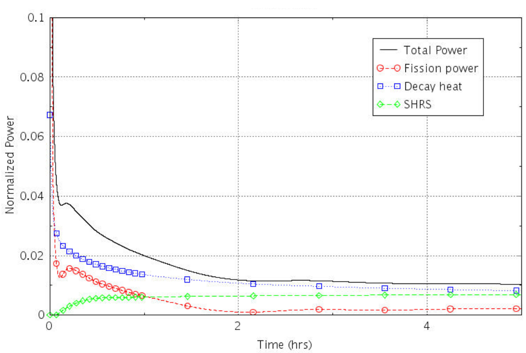
Mass Flows



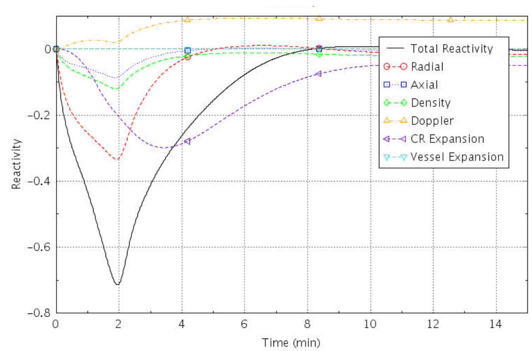
Fuel and Coolant Max Temperatures

Transients calculations: USBO

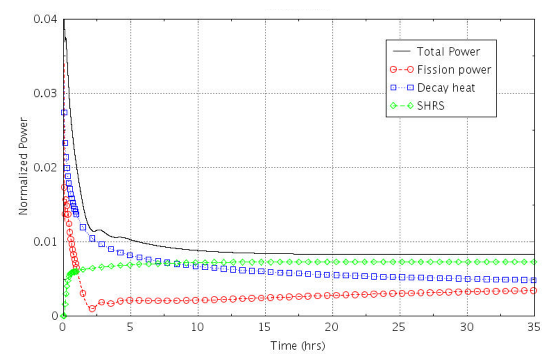
- Un-Protected Station Blackout (USBO)
  - Assume loss of all AC power → pumps trip and failsafe opening of DRACS HX
  - No Reactor scram → temp increase → Power reduced by inherent negative reactivity feedbacks
  - Reactor remains critical (k_{eff} =1.0), DRACS removes total power over long term (>~20 hrs)
  - Pools and Fuel temperatures below safety limits



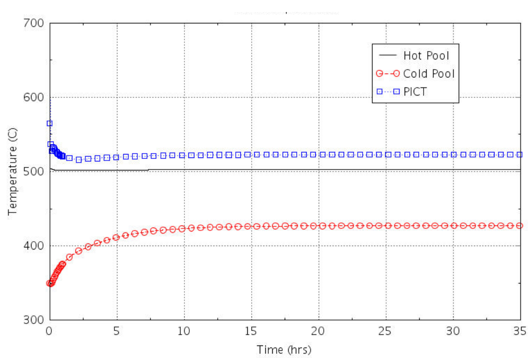
Reactor Power – Short term



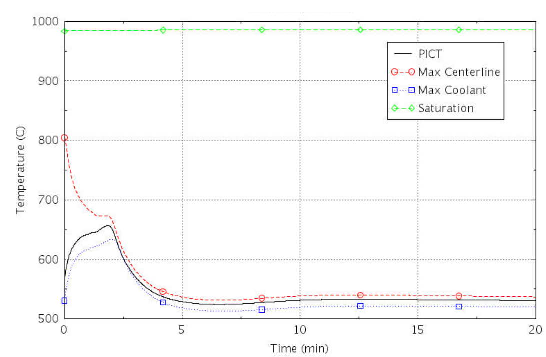
Reactivity Trend



Reactor Power – Long term



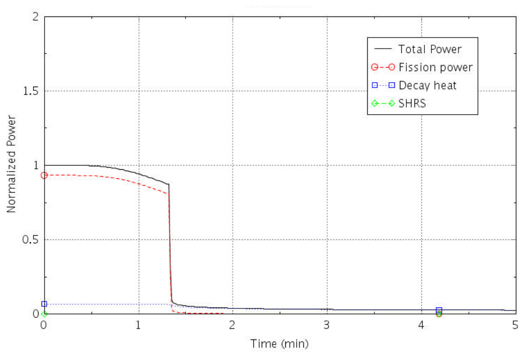
Cold and Hot Pool Temperatures



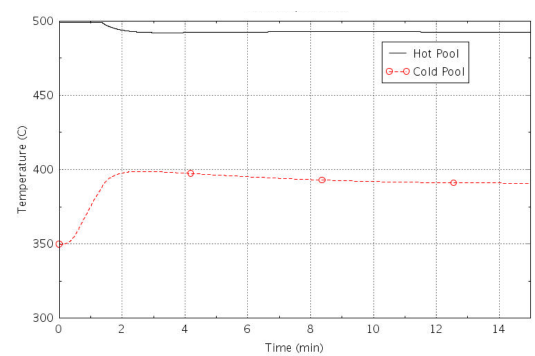
Fuel and Coolant Max Temperatures

Transients calculations: PLOHS

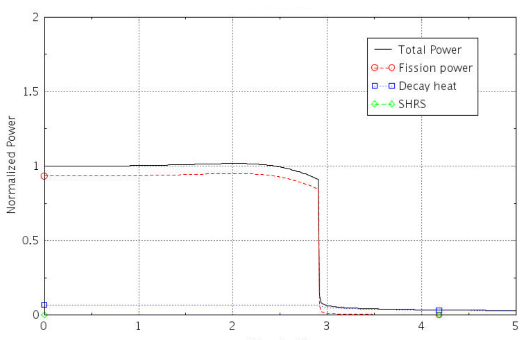
- Protected Loss of Heat Sink (PLOHS)
  - Loss of heat removal from IHXs → High inlet core temperature → power decrease for inherent negative feedbacks → reactor scram for high inlet temperature
  - Scram timing and final cold pool temperature depends by Cold Pool modeling
  - Significant safety margins



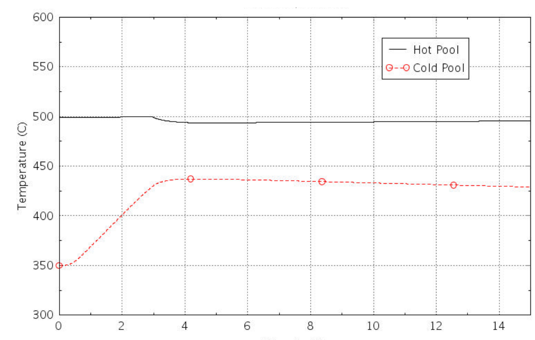
Reactor Power



Pool Temperatures



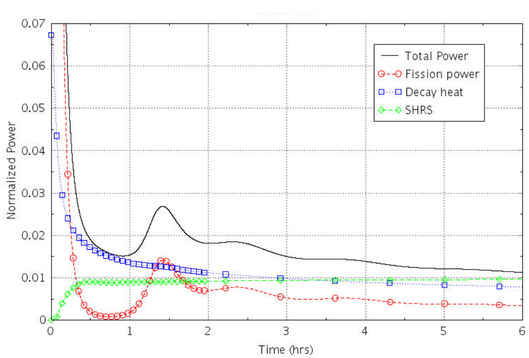
Reactor Power – Modeling Sensitivity



Pool Temperatures – Modeling Sensitivity

Transients calculations: ULOHS

- Un-Protected Loss of Heat Sink (ULOHS)
  - Loss of heat removal from IHXs → high inlet core temperature → power decrease for inherent negative feedbacks → no Reactor scram, MCPs continue to run → Negative reactivity coefficients bring reactor power down to ~ few % P_{nom}
  - Vessel Expansion causes return to criticality (k_{eff} >=1.0) by t = ~1 hr
  - Primary system temps equalize, higher cold pool temp increases power removal by DRACS
  - DRACS remove Decay Heat at t=~5 hrs
  - Pools and Fuel temperatures below safety limits



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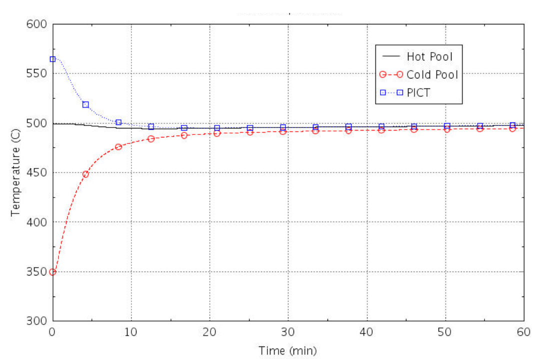
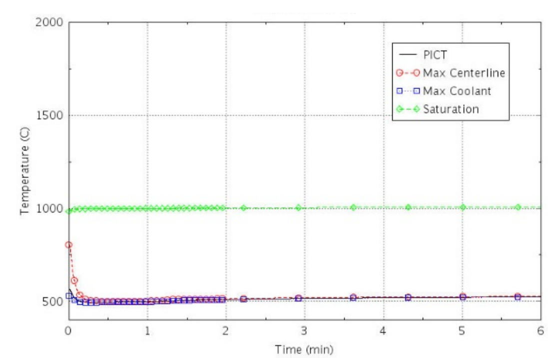


Cold and Hot Pool Temperatures



Fuel and Coolant Max Temperatures

Conclusions

- Technical activities ongoing at INL, for developing necessary technical skills for the VTR hosting site
- State-of-the-art RELAP5-3D TH/0D NK VTR model developed
- RELAP5-3D code flexibility allowed M&S of main reactivity feedbacks and main plant components with an acceptable level of details
- Set of preliminary safety analyses developed → no safety limits violated
- Comparisons with reference code SAS4A/SASSY are satisfactory (not showed in this presentation)