Computer Codes and Validation Adequacy

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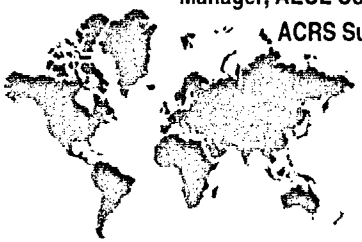
ACRS Subcommittee on Future Plant Designs
Washington D.C.
January 13, 2004



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Outline

- AECL Computer Code Software Quality Assurance (SQA) Program
- Validation Methodology
- Industry Standard Toolset and Key ACR Computer Codes
- Experimental Data for Thermal Hydraulics Validation
- Examples of CATHENA Validation

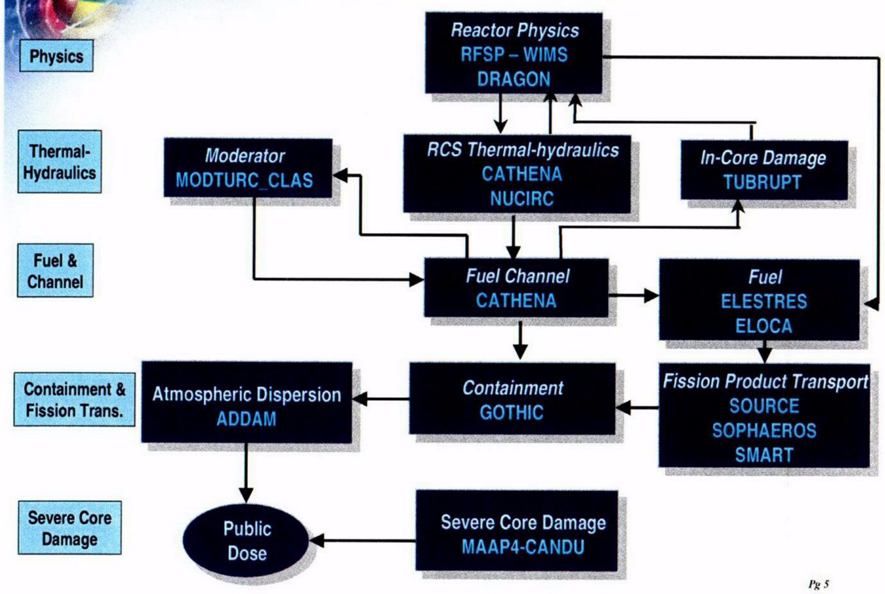
Computer Program Software Quality Assurance (SQA)

- Code Development and Qualification are conducted according to pre-defined QA procedures:
  - The Canadian Standards Association (CSA) published "Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants", N286.7-99 in March 1999
  - AECL published 00-01913-QAM-003, "Quality Assurance Manual for Analytical Scientific and Design Computer Programs in September 1999, and revised the document in March 2001
- Compliance is verified through internal, 3rd-party and regulatory audits

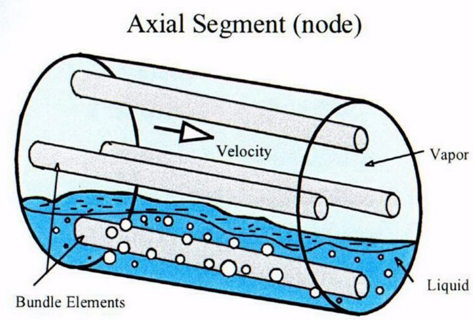
Industry Standard Toolset (IST)

- Formal qualification of safety and licensing codes was recognized as requiring significant investment, and resulting in redundancies and inconsistencies if undertaken separately
- Canadian utilities and AECL worked together to qualify a standard set of computer programs (IST)
  - Agreed to common processes to meet CSA-N286.7-99
  - Shared effort on code development, qualification and support

Key ACR Computer Codes

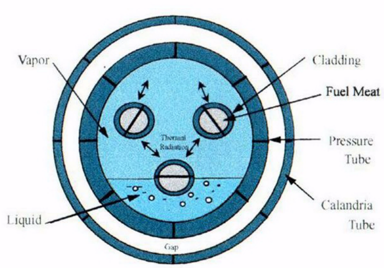


CATHENA THERMALHYDRAULIC MODEL



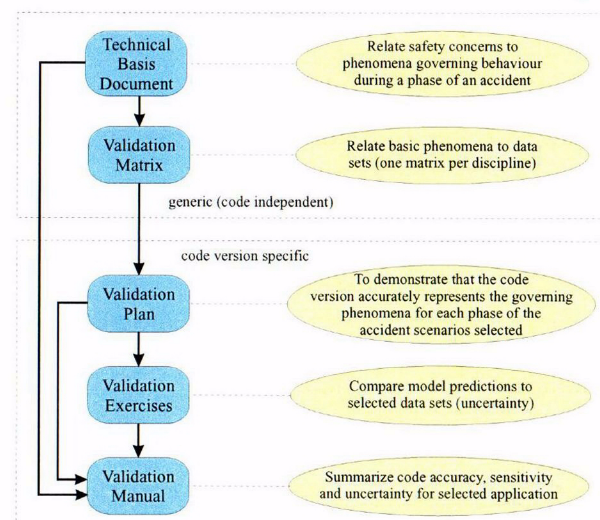
- Non-equilibrium model
  - 2-velocities,
  - 2-temperatures
  - 2-pressures
  - plus noncondensables
- Flow regime dependent constitutive relations couple two-phase model
- CATHENA "interfaces" to other codes:
  - i.e, Fuel Behavior, Plant Control, Physics

CATHENA's Solid Heat Transfer Model



- Multiple surfaces per thermal hydraulic node
- Radial and circumferential conduction modeled
- Models heat transfer within bundles subjected to stratified flow
- Radiation heat transfer calculated
- Built-in temperature dependent material property tables
- Models deformed geometry and pressure / calandria tube contact

Code Validation Methodology



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Technical Basis Document (TBD)

- For a given accident category, the TBD identifies:
  - The key safety concerns
  - The expected phenomena governing the behavior that evolves with time during identifiable phases of an accident
- The TBD establishes a relationship between technical disciplines, the safety concerns associated with a phase of an accident, the governing physical phenomena, and the relevant validation matrices
- Example:
  - Early in a LOCA, "Break discharge characteristics and critical flow" is a primary (high ranking of importance) phenomenon
  - During ECC injection, "Quench/rewet characteristics" becomes a primary phenomenon

Validation Matrices

- Identify and describe phenomena relevant to a discipline
- Rank the phenomena according to their importance in accident phases (consistent with PIRT-like process)
- Identify data sets and cross-reference to phenomena
  - Separate effects experiments, integral and/or scaled experiments, analytical solutions, inter-code comparisons
  - Includes CANDU-specific and otherwise

Thermal Hydraulics Phenomena, (first 10 of 23)

ID No.	Phenomenon	Large LOCA	LOCA/ LOECC	Small LOCA	LOF	LOR	Loss of Feed-water	Steam Line Break
TH1	Break Discharge Characteristics and Critical Flow	✓	✓	✓			✓	✓
TH2	Coolant Voiding	✓	✓	✓	×	×		
TH3	Phase Separation	✓	✓	✓	✓		✓	✓
TH4	Level Swell and Void Holdup	×	×	✓				✓
TH5	Pump Characteristics (Single & 2-Phase)	✓	✓	✓	✓			✓
TH6	Thermal Conduction	✓	✓	✓	×	×		
TH7	Convective Heat Transfer	✓	✓	✓	✓	✓	✓	✓
TH8	Nucleate Boiling			✓	✓			
TH9	CHF & Post Dryout Heat Transfer	×	×	✓	✓	✓		
TH10	Condensation Heat Transfer	✓		✓	✓		✓	✓

✓ primary phenomena   × secondary phenomena

Test Data for Thermal Hydraulics Phenomena (sample)

ID#	Data Set Name	TH2 Coolant Voiding	TH3 Phase Separat.	TH6 Thermal Conduct.	TH7 Convect. Heat Tran.	TH16 Flow Oscillation
SE1	Edwards Pipe Blowdown	■				
SE5	Marviken Bottom Blowdown	○				
SE13	PT/CT Contact Heat Transfer Tests			■		
SE21	CWIT Flow Stratification Tests		■		■	
CO1	End Fitting Characterization Tests	○	○	■		
INT9	RD-14 Natural Circulation Tests		○		■	■
INT14	Station Transients		■			■
NUM6	Radial Conduction Test			■		

■ Suitable for direct validation   ○ Suitable for indirect validation

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Experimental Data Base

- CANDU System Makes Use of International Data Sets:
  - Edwards Pipe Blowdown (Break Discharge)
  - Marviken Blowdown Tests (Break Discharge)
  - Dartmouth Air/Water Flooding in Straight Pipe (Counter Current Flow)
  - GE Large Vessel Blowdown Tests (Level Swell)
  - Christensen Power Void Tests (Coolant Voiding)
  - *..... and others*

Experimental Data Base – CANDU Specific

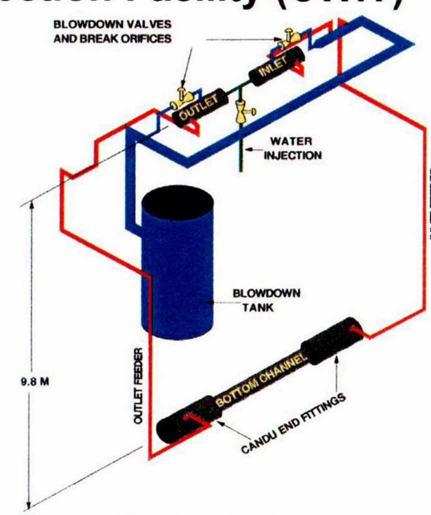
- Can by subdivided into:
  - Small Scale Experiments
  - Component Experiments
  - Integral Experiments
  - CANDU Plant Transients
- The majority of existing data (supporting current CANDUs) can be used for validation of the ACR
- Where "gaps" exist (i.e., higher pressure and temperatures of the ACR), new experiments have been completed and others have been planned
- Small Scale Experiments, Examples:
  - Flooding – downstream of an elbow (relevant to feeder)
  - Pressure Tube / Calandria Tube Heat Transfer Experiments
  - Horizontal Tube Rewetting / Refilling Experiments
  - Pressure Tube Circumferential Temperature Distribution

Experimental Data Base – CANDU Specific

- Full-Scale Component Experiments:
  - Feeder Refilling, Cold Water Injection Test Facility
  - Channel Stratification Studies, Cold Water Injection Test Facility
  - Header Studies, Large Scale Header Facility
  - Header Studies, Header Visualization Facility
  - Pump Characterization, CANDU Pump Facility
  - End Fitting Studies, End Fitting Characterization Facility

Cold Water Injection Facility (CWIT)

- Full-scale heated fuel channel with simulated fuel string
- CANDU representative feeders and End Fittings
- Designed to investigate feeder/channel refill performance, as well as flow stratification within CANDU bundle



CWIT Test Facility

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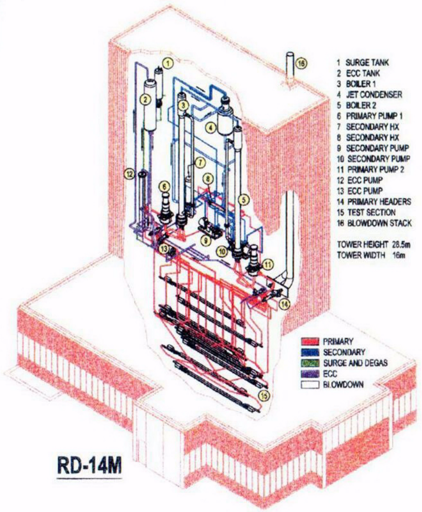
RD-14M Integral Test Facility

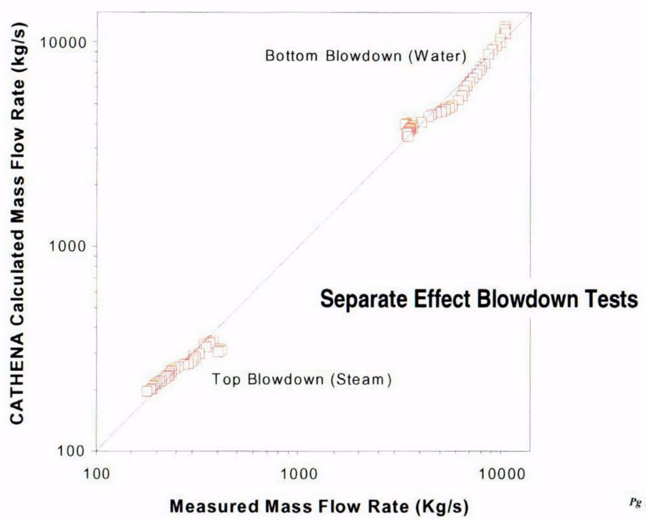


- Full elevation changes between major components and full linear dimensions
- Reactor typical heat- and mass-transfer rates
- Ten full length electrically heated channels (maximum of 11 MW)
- Simulation of all primary-side components - channels, end-fittings, feeders, headers, and steam generators
- Full pressure and temperature conditions (current CANDUs and ACR)

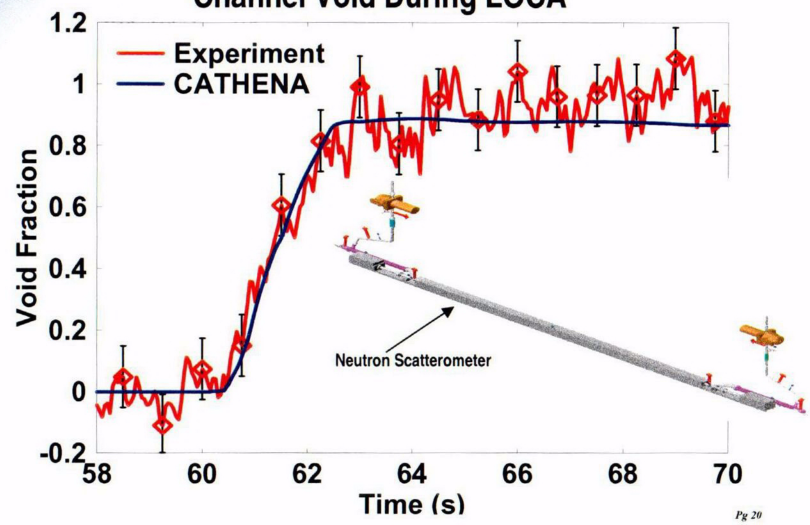
Examples of Validation for CATHENA

- **Component**
  - Marviken tests, discharge characteristics
- **RD-14M**
  - Channel voiding
- **CANDU Plant transient**
  - Single-pump trip

TH1: Break Discharge Characteristics – 3



CATHENA Validation, Example of Prediction of Channel Void During LOCA



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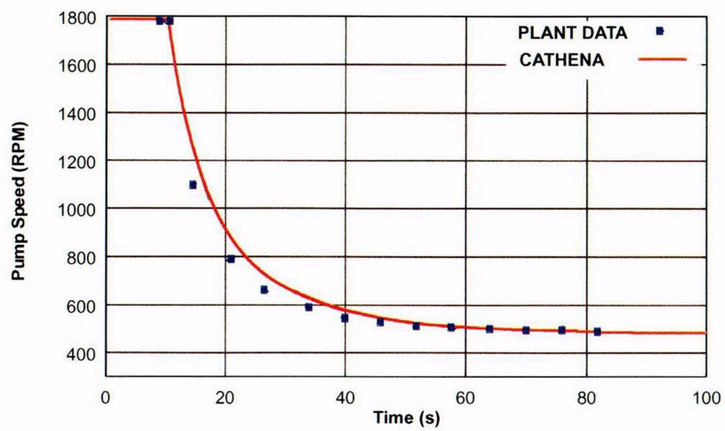
Single Pump Trip in a CANDU 6 Pump Run-down Speed



Conclusion

- ACR analysis codes are developed and qualified under a formal SQA program
- Validation methodology has been demonstrated, using thermal hydraulics as an example, and the CATHENA code
- A wide range of experimental databases is used in the validation process
- Examples of CATHENA validation are provided

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On-Power Fueling

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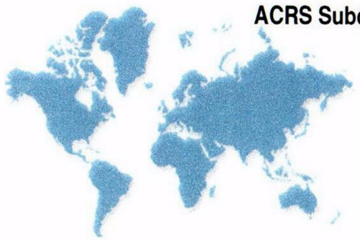
Washington D.C.
January 13, 2004



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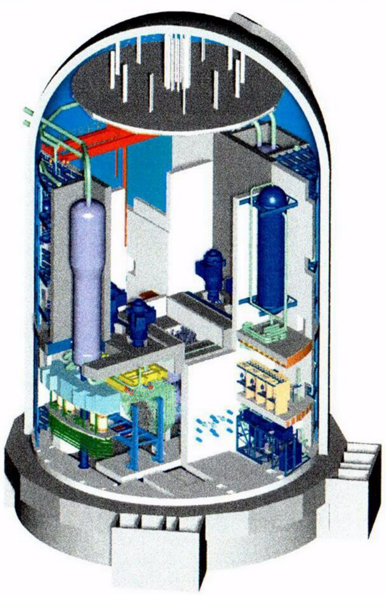
Synopsis

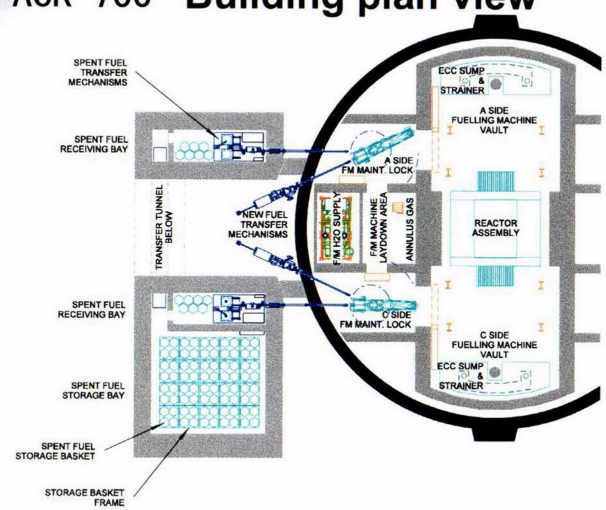
- This presentation discusses the advantages found in the CANDU reactor design from the use of on-power fueling and describes the equipment used
- On-power fueling allows a low core reactivity to be used and provides flexibility in station maintenance outages
- CANDU reactors have been safely and successfully using on-power fueling for 4 decades in 45 reactors
- The ACR design builds on that experience with a new system with improvements in safety, operability and maintainability



ACR Reactor and Fuel Handling Layout

- The ACR 700 reactor is an evolutionary design building on past CANDU designs
- It uses slightly enriched fuel in a 284 channel horizontal pressure tube reactor
- On-power fueling is used with a small constant staff complement

ACR 700 Building plan view



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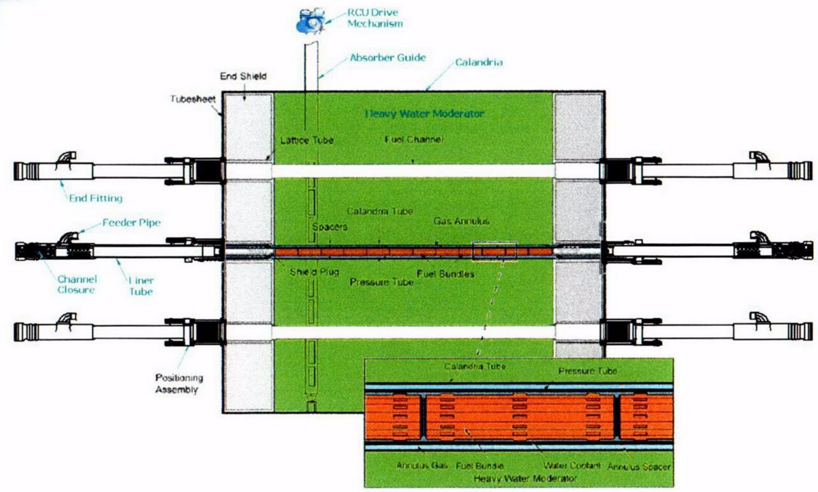
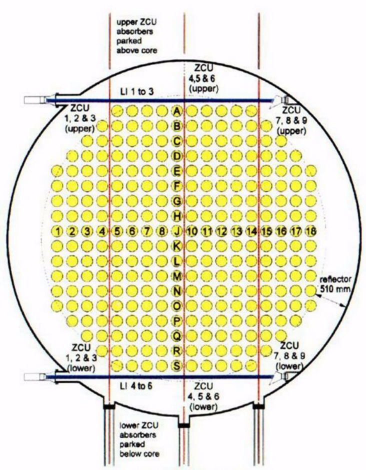
Fueling Scheme

- CANDU reactors use fueling to keep core reactivity at about 4.5 mk
- Fueling is carried out at a rate of 5.6 bundles per full power day (for daily fueling)
- Each 2 bundle replacement gives about 0.2 mk
- Channels are selected for fueling for overall core balance with a typical fuel residence of about 20 months
- Defected fuel can be removed promptly
- 9 zone controllers provide a total of about 9 mk for spatial control, xenon override and fueling flexibility
- 4 control absorbers provide for power setbacks

Fuel Channel Arrangement

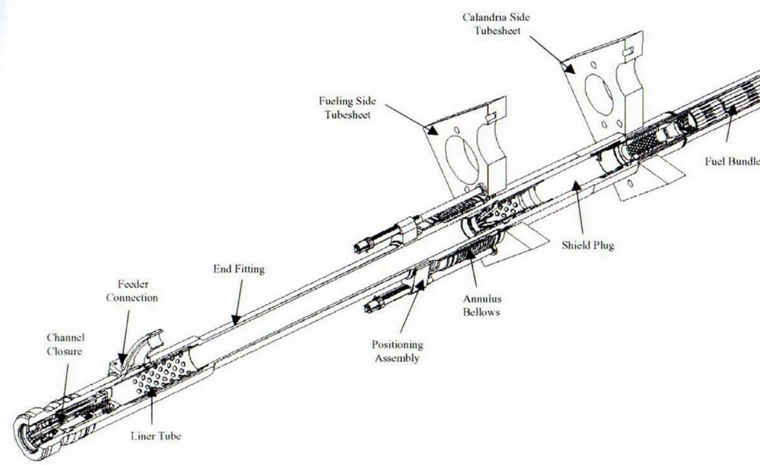


End View of ACR 700



ACR-700 Reactor Core
284 Fuel Channels
220 mm Lattice Pitch

Fuel Channel Assembly



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Fuel Channel Interface

- The fuel channel has:
  - A restraint to react fueling loads and seismic loads
  - A removable shield plug to locate the fuel string
  - Removable closure plugs to provide the pressure boundary
  - An end fitting interface feature to allow the head to latch on, seal and extend the pressure boundary

Fueling

- Each fuel channel contain a string of 12 fuel bundles
- Irradiated fuel is removed from the downstream end and fresh bundles are inserted at the upstream end
- Irradiated fuel is discharged via a fuel port through the containment boundary to a bay in the reactor auxiliary building
- New fuel is supplied via fuel ports also through the containment boundary
- The fueling machine has a movable class 1 pressure vessel that connects to the fuel ports and fuel channels in sequence to move the fuel around

Design Features to Enhance Safety During Fueling

- The principal safety features related to CANDU fueling are well proven and are designed according to recognized standards
- Pressure boundary components are designed to established piping and pressure vessel codes
- Key specialized materials and designs features are governed by Canadian Standards Association codes
- Additional interlocks, mechanical locks and backup systems are incorporated to enhance safety and operability
- Inherent benefit in reduction of reactor coolant system (RCS) activity from defective fuel bundles due to early detection / removal

Design Features to Enhance Safety During Fueling

- Latching snout connection mechanism and additional safety locks to prevent unintentional or unsafe release from a fuel channel to maintain RCS integrity
- Controls and instrumentation that are required to function properly during and following a DBE, LOCA or MSLB are seismically and environmentally qualified
- A seismically and environmentally qualified emergency water system is included to maintain fuel cooling when the fueling machine (FM) is off reactor during and following a DBE, LOCA or MSLB or if the normal system becomes unavailable
- Special stainless steel baskets guarantee sub-criticality of the fuel in all mediums

Barriers to Inadvertent Release of FM from Reactor Channel

- FM snout-to-end-fitting clamping mechanism
- FM snout safety lock – engaged / locked by channel pressure
- Check for integrity of seal between FM snout and channel end fitting prior to removal of channel closure
- FM snout clamp interlocks on pressure and status of snout safety lock
- FM bridge drive/brake safety interlocks
- Limited force of carriage drives
- Check for partial blockage of channel prior to installing channel closures
- Check for integrity of seal between channel closure end fitting prior to unclamping of FM snout

Safety Summary

- Existing CANDU stations have an excellent safety record related to fueling
- CANDU reactors have operated for an accumulated service life approximately 400 years with no accidents
- Fuel handling undesired events showed no major LOCA or major accidental radioactive releases or contamination
- CANDU 6 stations improved on past stations
- CANDU 6s have operated for an accumulated service life >60 years with approximately 43500 fueling cycles
- ACR further builds on the best practices and design features with several safety enhancements further mitigating risk

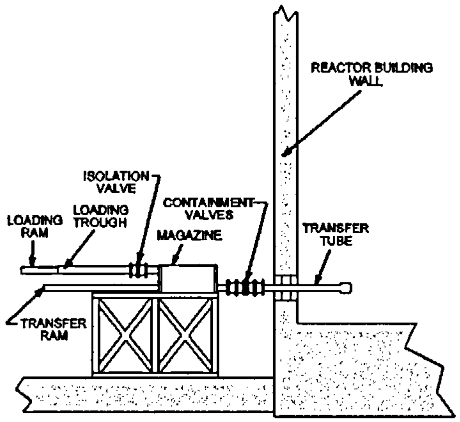
Fueling Equipment

- New fuel storage
- New fuel transfer
- Fueling machine
  - Head
  - Carriage
  - Bridge
  - Catenaries
  - Fluid systems
  - Controls
- Spent fuel transfer
- Spent fuel storage (bay and dry stores)

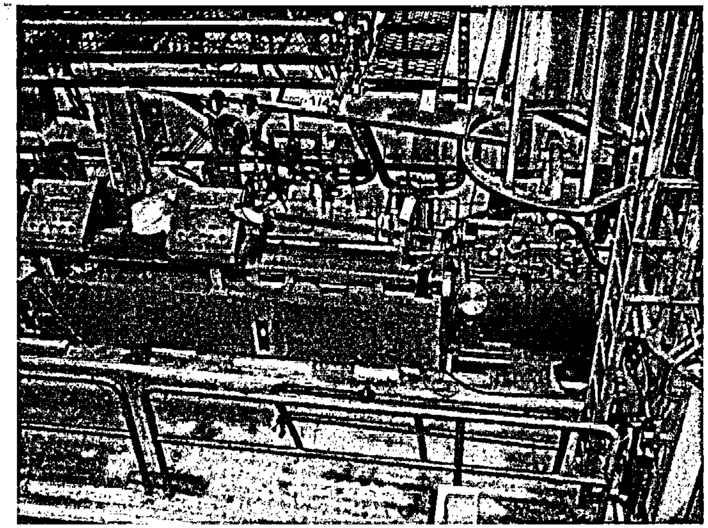
New Fuel Transfer

- New fuel is stored in a secure storage area and transported to the transfer room as required
- All fuel is stored with features to prevent inadvertent criticality
- With the containment valves closed, fuel is inspected and then loaded in to the transfer mechanism magazine
- With the isolation valve closed, the containment valves are opened so the fuel can be transferred into the fueling machine head

New Fuel Transfer



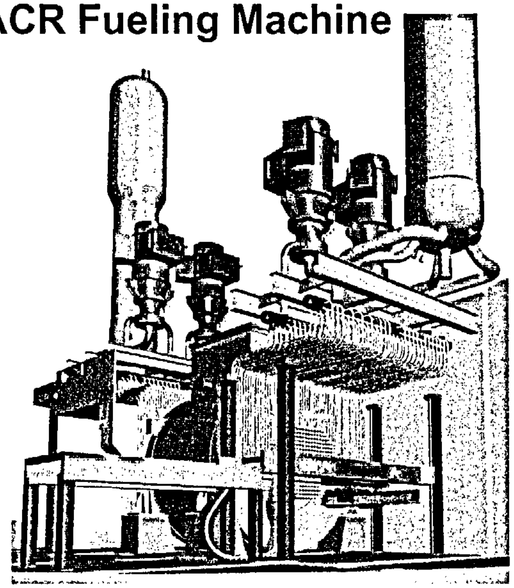
CANDU 6 New Fuel Transfer Mechanism



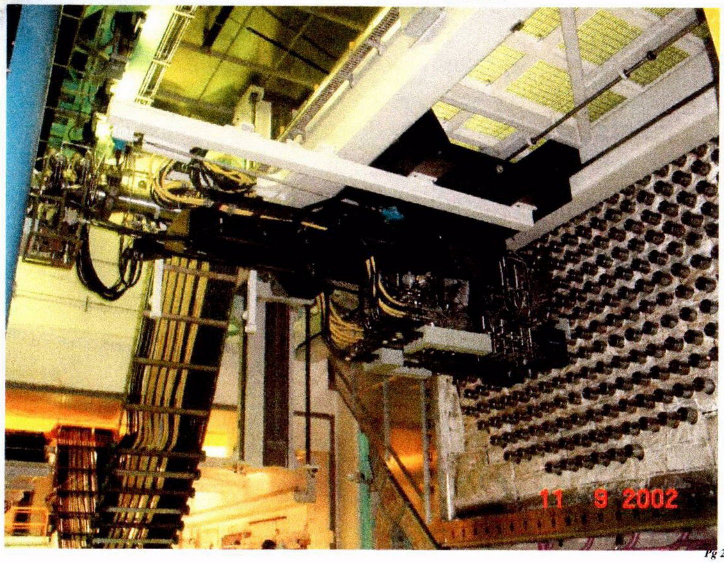
Fueling Machine

- The CANDU fueling machine consists of:
  - A head encased in a class 1 pressure vessel with a snout to connect it to fuel channels and ports, separators to separate one bundle from the next, a magazine to hold fuel and hardware, and a latching ram to move fuel and hardware
  - A carriage to hold the head providing axes to align and push the head on to ports / channels
  - A bridge to lift the head and carriage around the reactor vault and support inspection and maintenance
  - A catenary system to take power and fluid to the head and connect them back to an accessible area
  - A control system with viewing and safety interlocks to allow remote control
  - A process system to provide pressure control and cooling

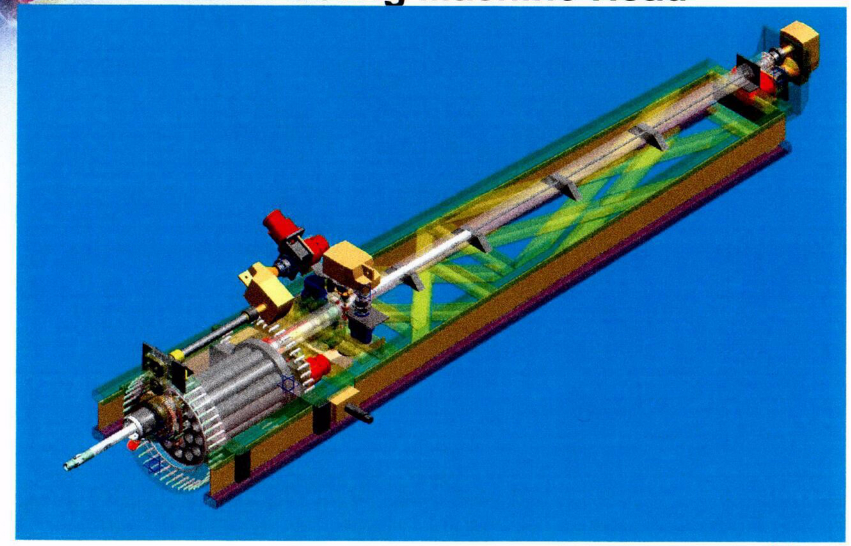
ACR Fueling Machine



CANDU 6 FM in Vault



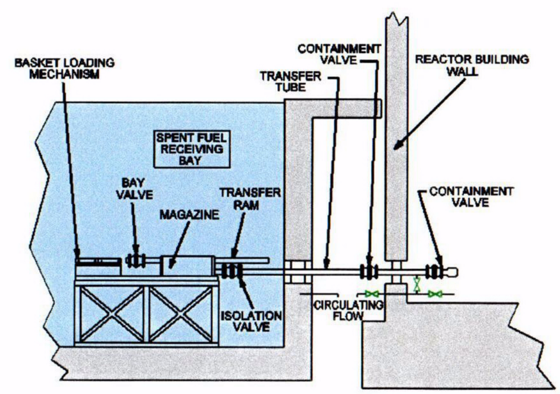
ACR Fueling Machine Head



Spent Fuel

- A spent fuel port in the fueling machine maintenance lock allows the fueling machine to latch on and extend its pressure boundary
- Fuel is then inserted into the port and transferred through to the fuel bay
- In the bay the fuel is transferred into baskets which are first stored in a buffer zone and then moved into the main bay area
- After fuel decay heat is significantly reduced, fuel is typically loaded into dry storage vaults

Spent Fuel Transfer



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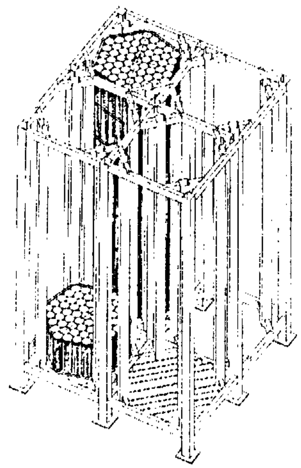
Fuel Storage Baskets and Racks

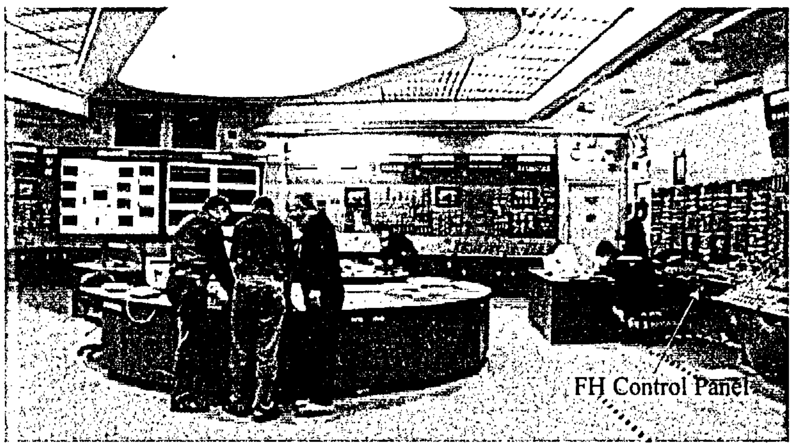


- Spent fuel stored in a hexagonal shaped basket
- Baskets stacked in rectangular, seismically qualified frames
- Basket fabricated of SS tubes which guarantee sub-criticality
- Ample space for convection induced flow

Fuel Handling Operations

- Refueling operations require previous approval from the Senior Nuclear Operator
- The Senior Nuclear Operator approval is required at critical steps:
  - when clamping to a reactor channel
  - removing channel closures
  - moving / transferring fuel
  - manual operation, etc.
- The Fuel Handling Panel Operator Training Program is very rigorous and extensive (18 months)

CANDU 6 Fuel Handling Control Panel



Fuel Handling Operations

- Fuel Handling Operations follow many practices to guard against errors:
  - Never start fueling operations without full equipment redundancy available
  - Never perform operations on-reactor that not been tested and rehearsed and without approval
  - High emphasis on human performance and qualification. Intensive training programs and refreshing courses
- Automatic computer control with extensive software checks and software interlocks
- Independent and separate fuel handling (FH) hardware interlock system

Fuel Handling Maintenance

- All major equipment is accessible during reactor full-power operation
- Fuel Handling controls, drives and indications are provided with 100% redundancy
- Predictive/Preventative Maintenance Program System – PMS
- Fuel Handling Maintenance Shop area and tools/facilities are provided
- ACR-700 requires no unit scheduled maintenance outages for fueling purposes

Conclusions

- On-Power Refueling capability of ACR reactors completely eliminates unit outages built around fueling requirements
- On-Power Refueling provide flexibility and gives operations with improved safety margins
  - Low core reactivity
  - Prompt removal of defected fuel
  - Outage flexibility
- Computer controlled and automated On-Power Refueling ensure an optimum fuel usage
- Defense-in-Depth principles and multiple barriers used ensure safety and reliability of ACR reactors

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Confirmation of Negative Coolant Void Reactivity (ACR Physics Design)

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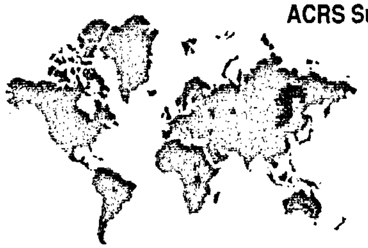
ACRS Subcommittee on Future Plant Designs
Washington D.C.
January 13, 2004



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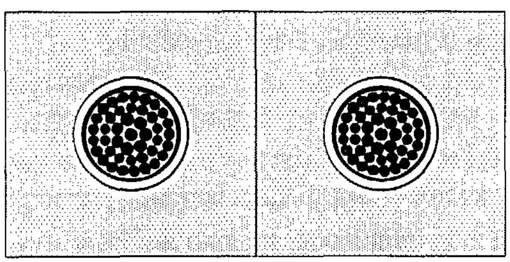
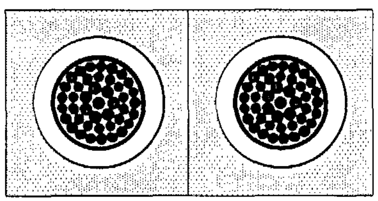
Outline

- ACR achieves a slightly negative Coolant Void Reactivity (CVR) by manipulating upon voiding
  - Changes in Spatial Flux Shape
  - Changes in Neutron Spectrum
- Confirmation of Negative Coolant Void Reactivity in ACR by:
  - Comparisons of CVR calculated by AECL's computer codes (WIMS, RFSP, DRAGON) with other international codes such as MCNP, HELIOS, DONJON, NESTLE
  - Experimental verification of negative CVR in AECL's ZED-2 Reactor at Chalk River Laboratories (CRL)

Comparison of NU CANDU and ACR Lattices



NU CANDU
Lattice Pitch 28.6 cm
Moderator/Fuel Ratio
= 16.4



ACR
Lattice Pitch 22 cm
Moderator/Fuel Ratio
= 7.1

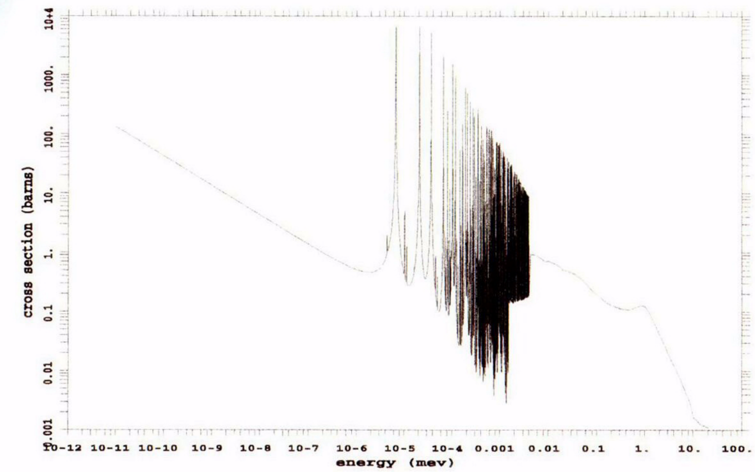
Physics Innovations to Achieve Slightly Negative CVR in ACR

- WIMS lattice simulations indicate CVR can be reduced by reducing the Moderator / Fuel ratio in the lattice cell
- Design Target of Slightly Negative CVR requires reduction of lattice pitch (LP) from current value of 28.6 cm to 20 cm
- Minimum LP = 22 cm required to provide space for feeders between channels
  - Use larger calandria tube (CT) to displace more moderator
  - Add Dy (7.5%) to central NU pin
  - Use 2.1% SEU fuel in remaining 42 fuel pins to achieve average fuel burnup of about 21 MWd/kgU
- Full core LOCA reactivity effect = - 7 mk

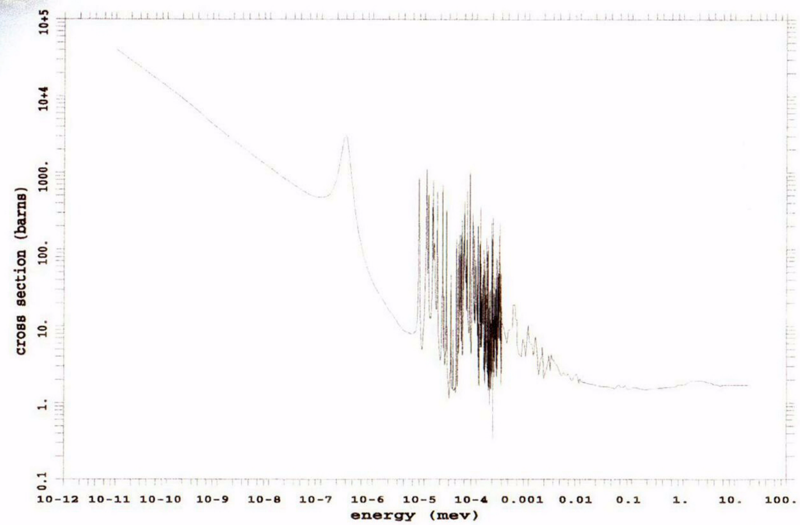
Effect of Coolant Void in ACR

- ACR lattice is under-moderated with normal H_2O coolant
- H_2O acts as both coolant and moderator
- LOCA further reduces moderation from the lattice
- Coolant Void Reactivity (CVR) is a combined effect due to loss of absorption (positive) and loss of moderation (negative) from H_2O
- Major contributors to the negative CVR
  - –Lattice Cell
    - Increase in Resonance Absorption (1 eV to 100 keV) in U238
    - Decrease in Fission (0.3 eV resonance) in Pu239
    - Increase in neutron absorption by Dy in the central pin upon voiding
  - –Increase in Reactor Leakage

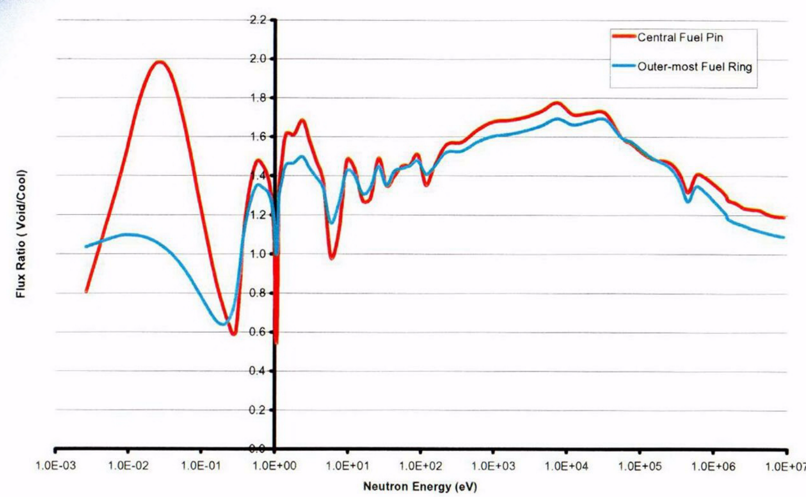
Absorption Cross Section of ^{238}U



Fission Cross Section of ^{239}Pu



Spatial and Spectral Changes in Neutron Flux due to Coolant Voiding in ACR



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Major Contributors to CVR (mk) in ACR

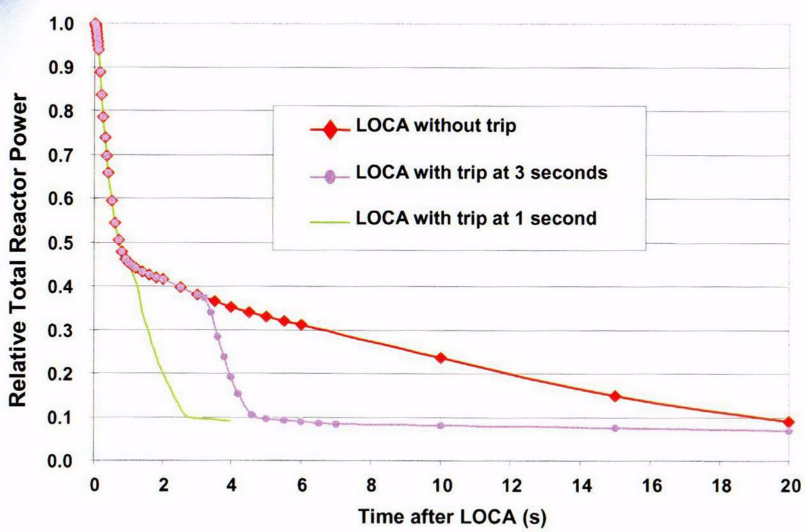
U235	10.1
U238	-15.1
Pu239	-11.5
Pu240	-0.4
Pu241	-1.6
Pu242	-0.1
H1(Coolant)	31.5
Dy	-10.7
Other Nuclides in Lattice	-4.7
Net Lattice Reactivity	**-2.8**
Reactor Leakage	-4.2
Full-Core CVR	**-7.0**

Unique LOCA Features in ACR

- Power in reactor core region drops upon LOCA due to negative void reactivity
- Process trip is sufficiently fast to terminate LOCA
- Rapid rise in thermal neutron flux in the reflector region relative to the core region due to migration and subsequent thermalization of fast neutrons from the core region
- This increase in neutron leakage results in a more negative LOCA reactivity than that predicted by the lattice code

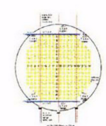
LOCA Power Transient in ACR-700

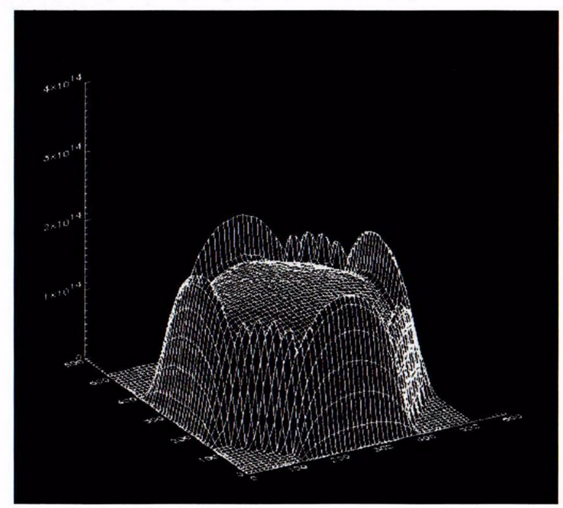
(Nominal Voiding Rate)



Thermal Flux Profiles in ACR-700 upon Instantaneous LOCA

(click picture to start animation)





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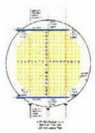
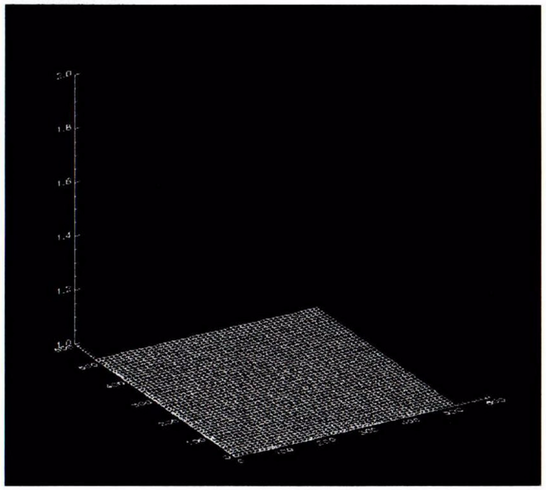
Thermal Flux Ratios in ACR-700 upon Instantaneous LOCA

(click picture to start animation)





ACR Physics Codes

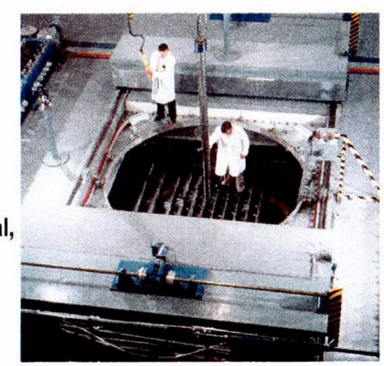
- WIMS
  - lattice-code produces 2-group cell-averaged cross sections for use in RFSP
- RFSP
  - 2-group diffusion method
  - wide range of calculational models: Reactor core design, fuel management, kinetics, xenon-transients
- DRAGON
  - from Ecole Polytechnique, for supercell calculations
    - device "incremental x-sections", added to cell-averaged cross sections in RFSP at device locations
- MCNP
  - extensively used for benchmarking the major physics codes
- DONJON
  - multi-group diffusion code from Ecole Polytechnique for comparison with RFSP

Confirmation of Negative CVR

- WIMS CVR of ACR fuel lattice has been confirmed by comparisons with
  - MCNP
- Negative reactivity due to Full-Core LOCA in ACR calculated by RFSP has been confirmed by comparisons with
  - DONJON
- Full-core MCNP model will be used to simulate LOCA in ACR and to confirm results from WIMS/RFSP simulations

Experimental Confirmation of Negative CVR in ZED-2 Reactor

- Tank-type critical facility, 3.3 m in diameter and depth
  - runs at a few watts
- Flexible facility
  - allows testing of a variety of fuels, different pitches, different coolants: D_2O, H_2O, air (voided)
- D_2O moderated
- Typical lattice arrangement is hexagonal, with 55 channels, each containing 5 bundles
  - can also have square lattice
- 7 "hot sites" can be located in center
  - 10 MPa up to 300 °C



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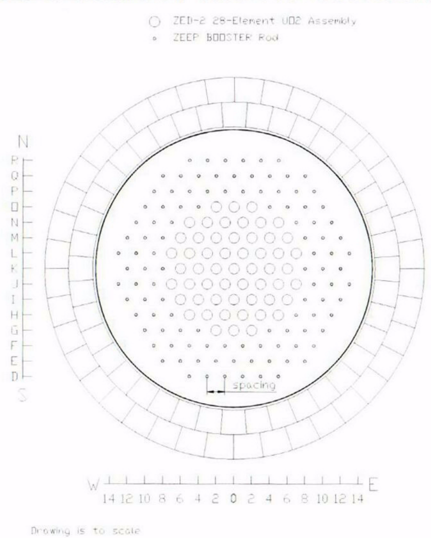
ZED-2 Measurements: General

- Material buckling (reactivity)
  - full core flux maps and substitution experiments
    - substitution method extensively validated
  - reactivity coefficients
    - void reactivity; fuel temperature; coolant temperature and purity; moderator temperature, purity, and poison
- Worth of reactivity devices (shutoff rod, adjuster rod)
- Reactor period measurements (for neutron kinetics)
- Reaction rates in foils
  - U-235, Pu-239, Dy-164, Cu-63, Mn-55, Au-197, In-115, Lu-176
  - reaction-rate ratios are sensitive indicators of the energy spectrum
- Fine lattice cell flux distribution (Cu-63)

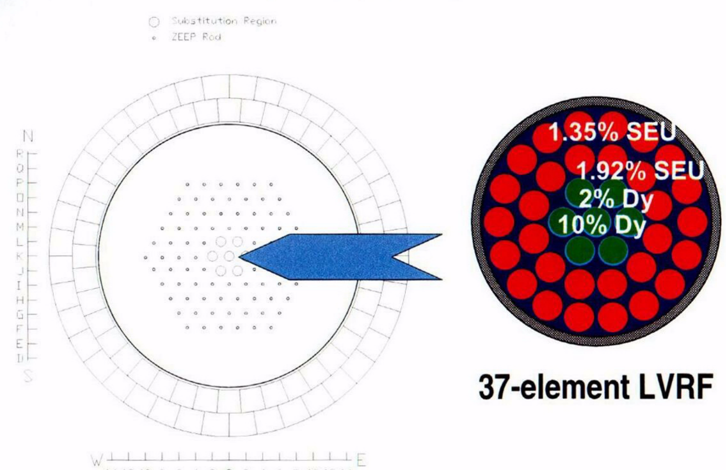
Preliminary ZED-2 Measurements Completed using Existing Fuel

- Buckling measurements using 28-element NU fuel
  - flux-maps, H_2O and air-cooled, hex lattices 20, 21.59, 22.86 cm pitch
  - 21.59 cm hex lattice pitch gives ACR moderator to fuel ratio of 7.1
  - measurements confirmed that CVR decreases when lattice pitch is reduced
- Substitution experiments using 37-element LVRF (Low Void Reactivity Fuel)
  - 7 channels into ZEEP (Zero Energy Experimental Pile) lattice at 21.59-cm hexagonal pitch, with H_2O, D_2O and air
  - Measurements confirmed negative CVR
- Fine-structure experiments using 37-element LVRF
  - a special demountable bundle with removable elements loaded with thin activation foils positioned between fuel pellets
  - activation data will be compared to WIMS predictions

Lattice Used for the 28-element Flux Maps



Physics Measurements of SEU and Dy Fuel in ZED 2



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