ACR Technology Base

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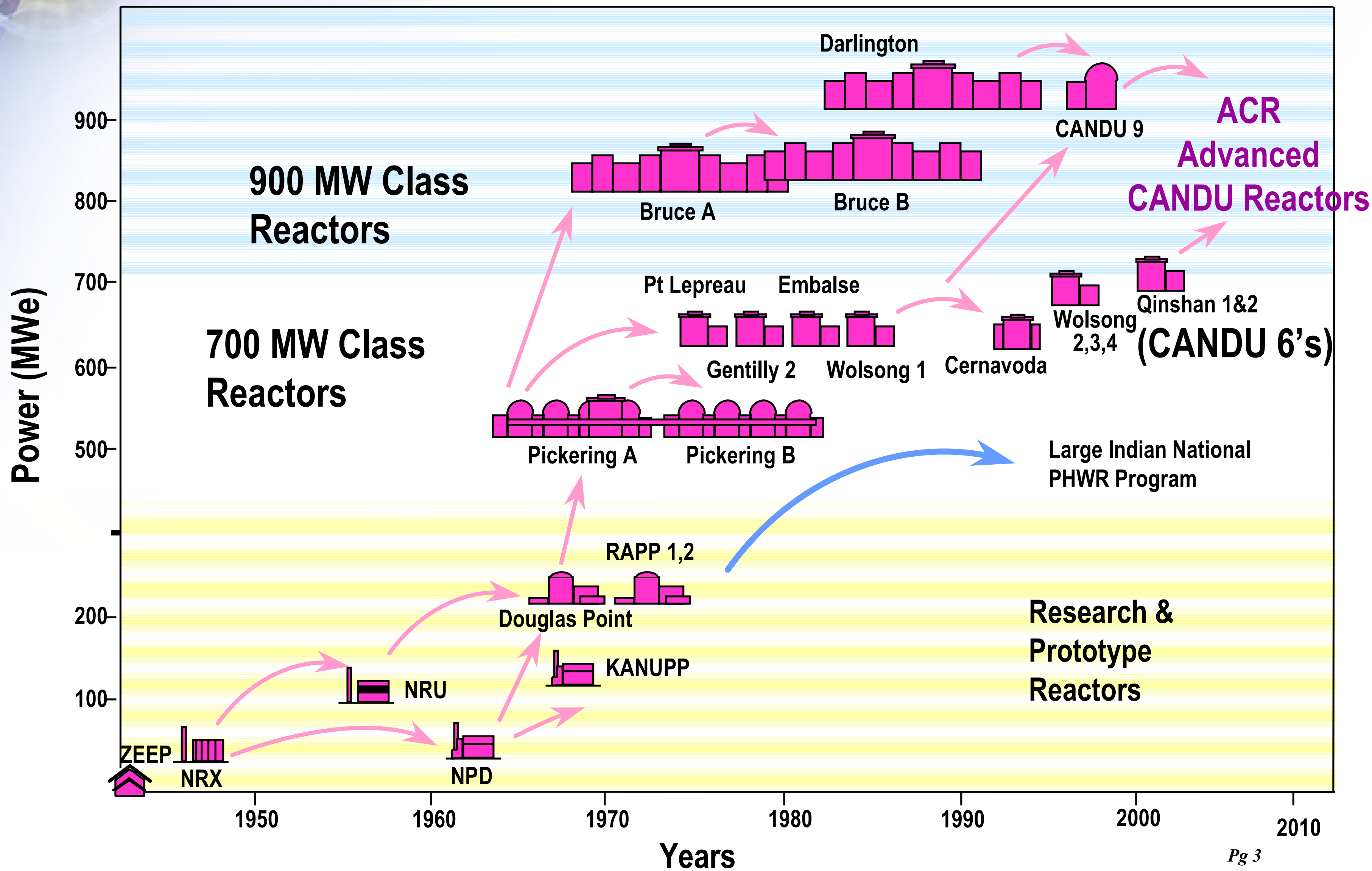
ACR Technology Base

- The ACR technology is based on the fleet of CANDU reactors in operation around the world.
- The ACR is an evolutionary extension of the CANDU design
  - The basic design features and technologies are common with the current CANDUs
  - The knowledge base embodied in the current CANDUs is directly applicable to the ACR design

CANDU Genealogy

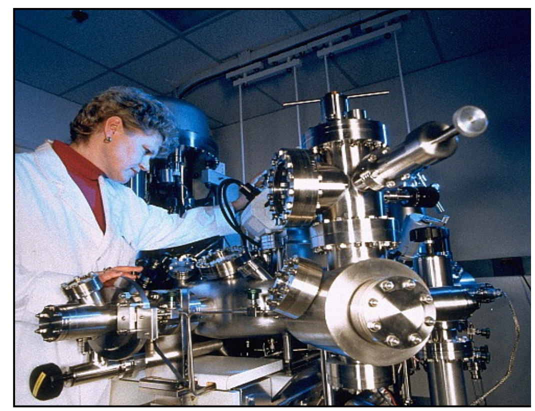


CANDU Technology History

- First sustained fission reaction outside USA achieved in 1945 in "ZEEP" (Zero Energy Experimental Pile)
- Large research reactors:
  - 1947 NRX (National Research EXperimental)
  - 1957 NRU (National Research Universal)
- NRX and NRU key facilities in development of CANDU design and other nuclear technology
- CANDU development emerged from the Chalk River Laboratories in early 1960s
- NRU continues to be used by AECL for CANDU fuel and materials testing and for advanced materials research

Chalk River Laboratories

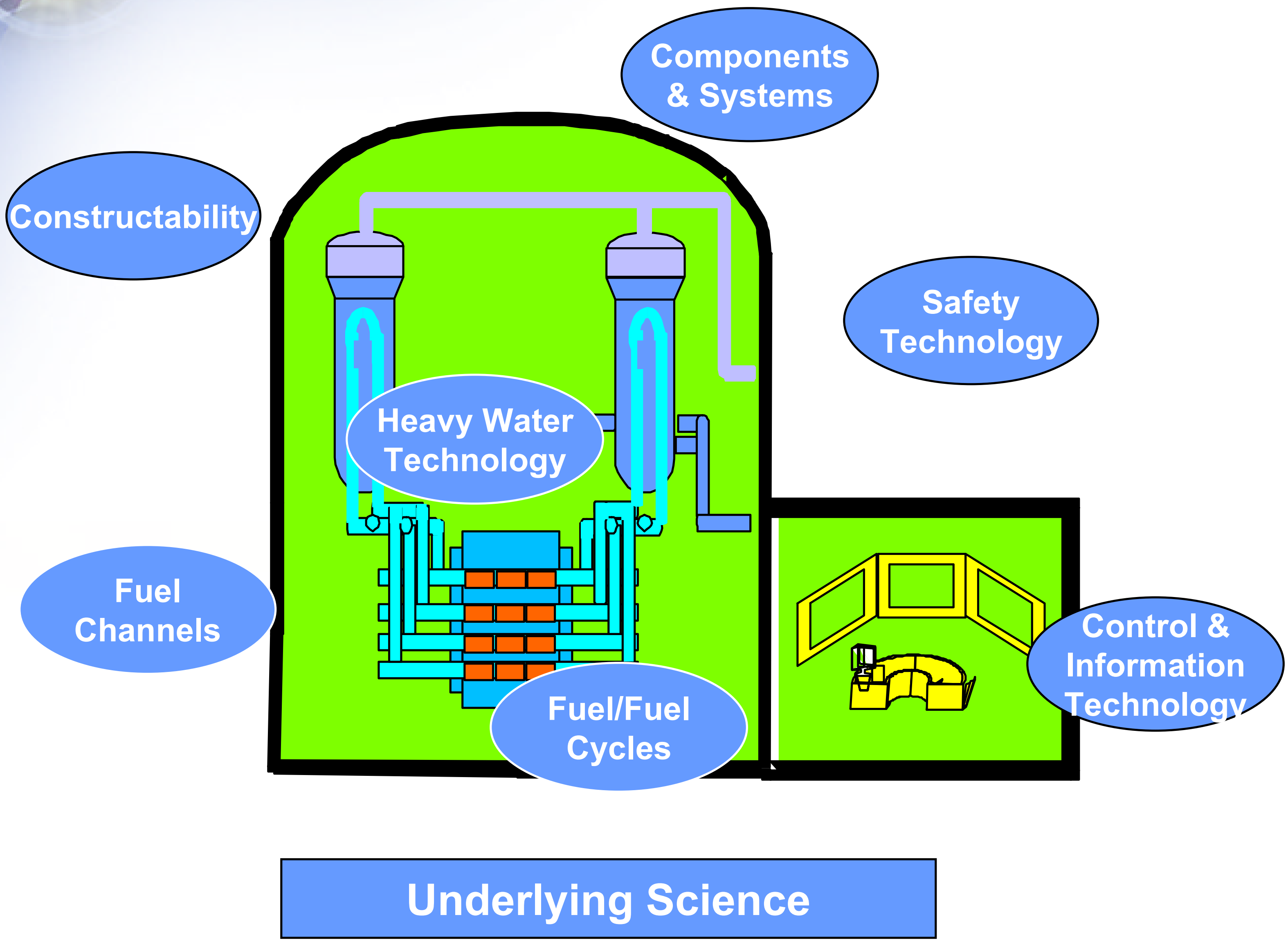
- The CANDU technology base is maintained and advanced primarily through the resources of the Chalk River Laboratory (CRL)
  - ~ 2000 scientists, technologists and support staff
  - Integrated nuclear laboratory with major research and testing facilities
    - NRU research reactor
    - CANDU test fuel manufacturing capability
    - Thermalhydraulic test loops
    - Materials test loops and laboratories
    - Safety and severe accident phenomena test facilities
      - Model development and validation



CRL Mandate

- Support CANDU Technology
  - Maintain safety, design & licensing basis
  - Support operating nuclear facilities
  - International technology cooperation (IAEA, NEA, etc.)
  - Decommissioning and waste management
- Support AECL Commercial Activities
  - New products & services development
  - Research reactor fuel fabrication
  - Isotope production/isotope facility operations (MDS Nordion)
  - Commercial R&D

CANDU Development Technologies

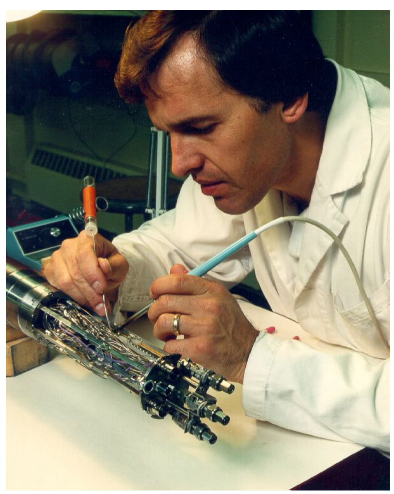


Channel Reactor Technology

- **The channel core design is a distinct CANDU development advantage**
  - **The reactor core output can be scaled by changing the number of channels (and calandria volume) without changing the single channel design or the overall core safety characteristics**
  - **The small channel size permits full-scale testing of important core components (fuel bundles, channel components)**

NRU Reactor – A Major Research Tool

- 200 MW_{th} fuel-channel research reactor
  - CANDU fuel development
  - CANDU materials development
  - Safety experiments
    - Fission product release from fuel
    - Fuel behavior under severe accident conditions



ACR R&D Requirements

- The ACR is an evolutionary extension of the current CANDU design.
- The design is well supported by the existing CANDU database and validated codes
- The anticipated R&D program is not exploratory:
  - Additional data will extend established databases for existing codes and models.
  - Results will improve confidence in the design and validate margins.

Technology Base Overview

- Physics
- Fuel
- Fuel Channels
- Safety
  - Source Term
  - Thermalhydraulics
  - Containment
  - Severe Accidents
  - Code Qualification
  - Shutdown Systems
- Advanced Components

Physics Technology Base

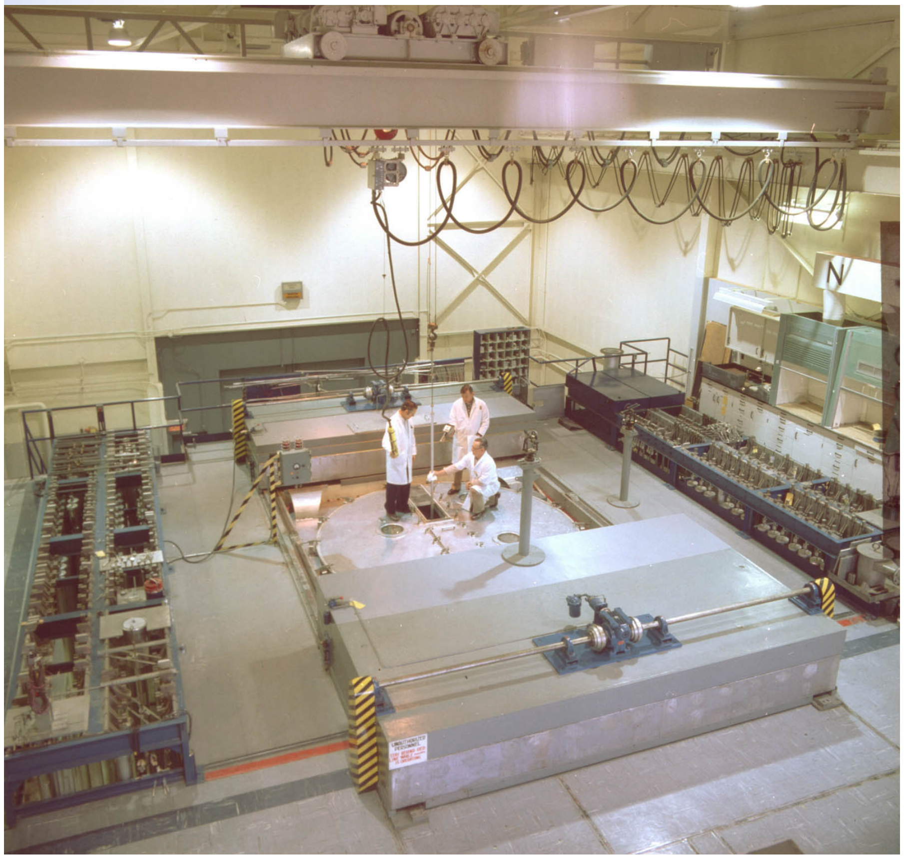
- The ACR physics technology is based on the validated CANDU codes:
  - Part of Industry Standard Toolset (IST)
  - Theoretically rigorous treatment
  - Good accuracy and general applicability
- Validation involved
  - Comparison against ZED-2 measurements
  - Comparison against power reactor measurements
  - Extrapolation / scaling using MCNP

ACR Physics Codes

- ACR will use the current validated CANDU nuclear physics codes:
  - WIMS – multi-group, 2-D transport lattice code
  - RFSP – 2-energy group diffusion code
  - DRAGON – multi-group, 3-D transport
  - MCNP – Monte Carlo code
- The validation will be extended to cover the smaller core lattice pitch, light water coolant and SEU fuel.

ACR Physics Code Validation

The physics codes will be validated for ACR application primarily using the ZED-2 Critical Facility.

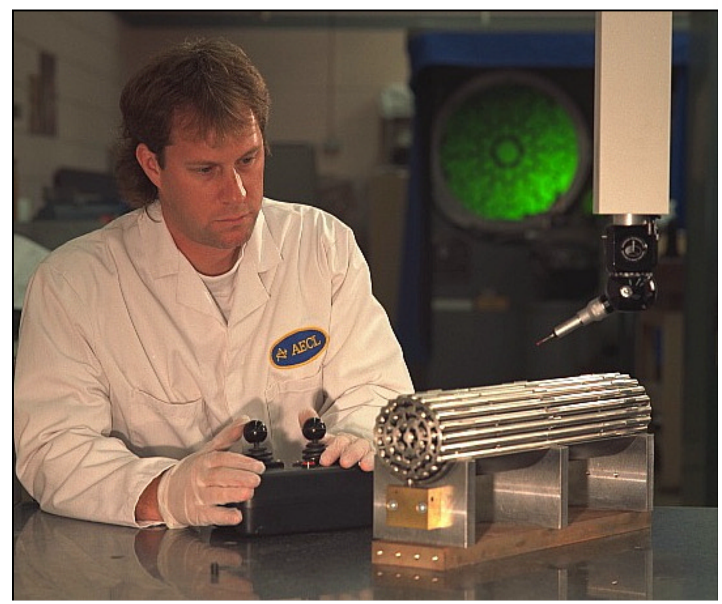


- ZED-2 is a zero power critical facility:
  - core with 55 channels each containing 5 bundles
  - driver fuel selected to match ACR lattice requirements
  - heavy water moderator
  - adjustable lattice

Fuel Technology Base

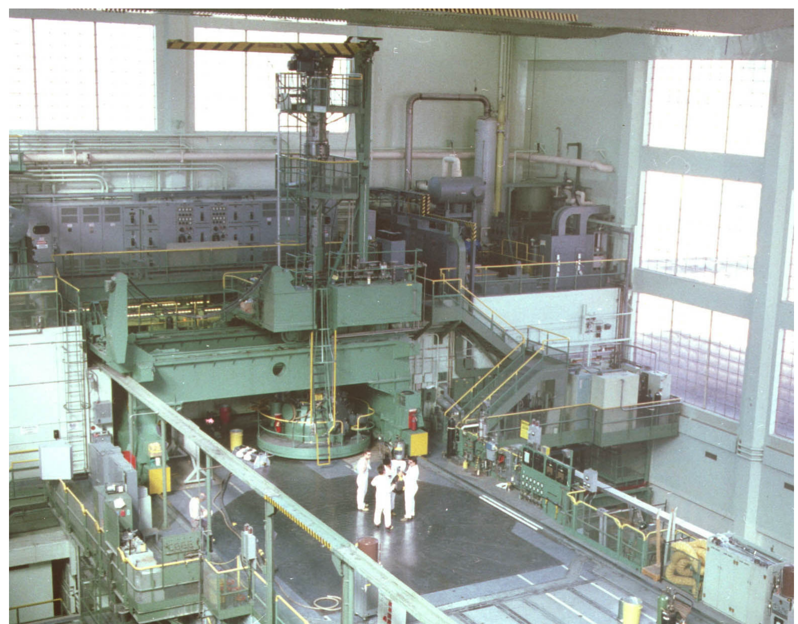
- CANFLEX fuel is based on the simple fuel bundle used in all CANDU reactors.
  - Advanced fuel bundle design for larger critical heat flux margins and lower element power ratings.
  - Natural uranium CANFLEX fully developed and tested for application in current CANDUs.

- ACR CANFLEX
  - Uses SEU to extend fuel burnup and enable light water coolant.



ACR Fuel Qualification Program

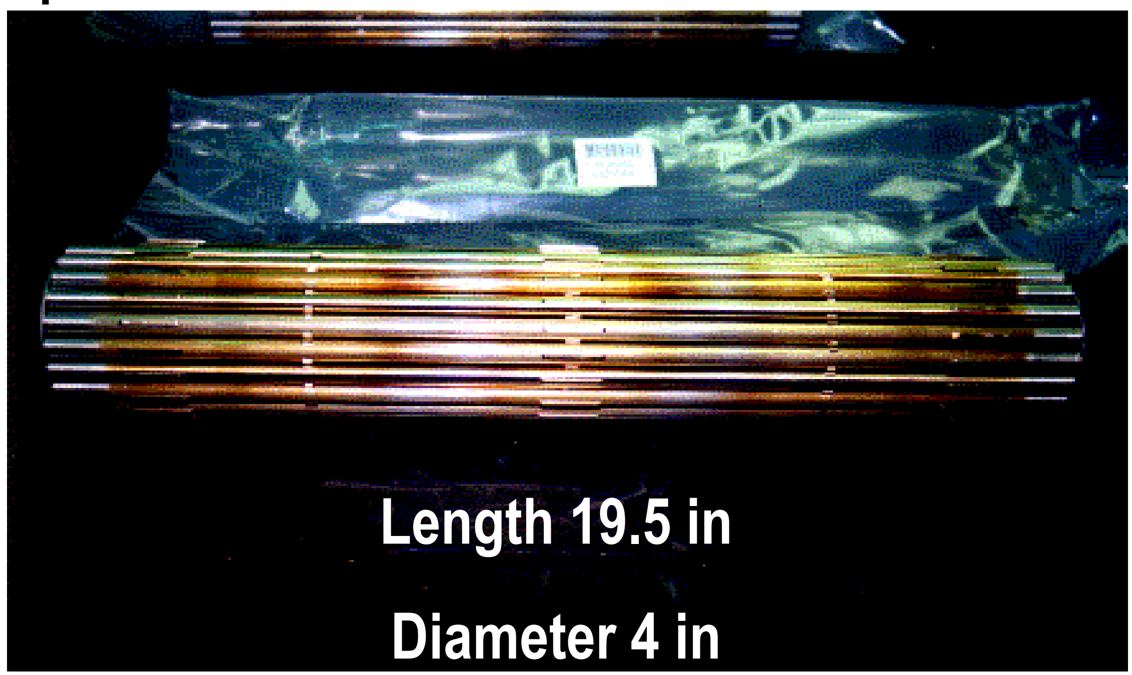
- Out-reactor: endurance and mechanical integrity
- In-reactor: high burnup and power ramp performance testing in NRU
- Thermalhydraulic performance and correlation development



SEU Fuel Technology

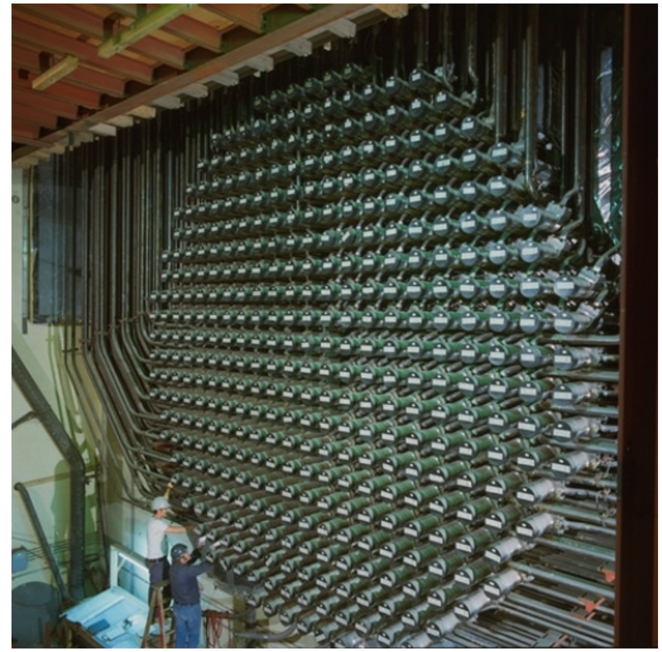
AECL has extensive experience in irradiation of SEU CANDU fuel bundles.

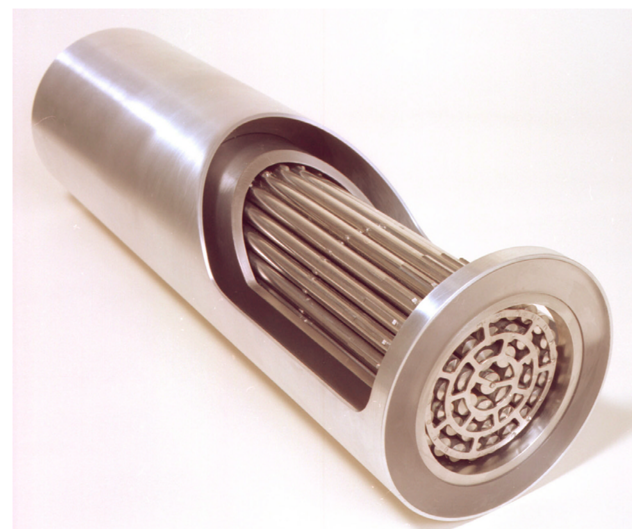
- **All tests in NRU have used enriched fuel to obtain desired powers and burnups.**



Fuel Channel Technology

- CANDU has several hundred reactor operating years of experience with the performance of fuel channels and fuel channel materials.
- The ACR fuel channel operating temperatures and pressures are slightly greater than those where the experience is greatest.
- The R&D program will provide the data necessary to extend predictive models of fuel channel materials performance.





Fuel Channel Technology Basis

- Fuel Channel Properties
  - Extensive knowledge of Zr-2.5%Nb material
  - Specifications for chemistry, metallurgical properties and manufacturing process based on operating reactors and research reactor studies
- Fuel Channel Performance
  - Deformation model
  - Corrosion/hydrogen ingress model
  - Models backed by data from tubes in operating reactors, and samples irradiated in research reactors

ACR Fuel Channel R&D

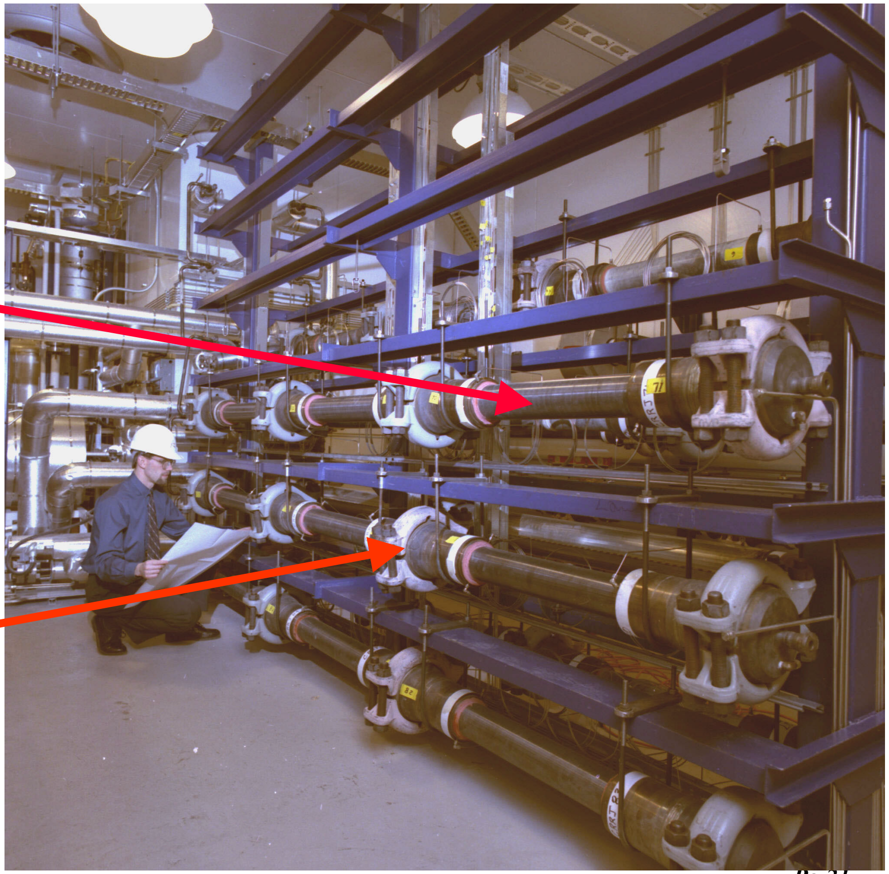
- The R&D program addresses the impact of new component specifications and coolant conditions.
- Qualification of manufacturing process
  - Thicker pressure tube
  - Thicker calandria tube
- Extension of deformation model
  - Extend database (including tests on intermediate thickness pressure tube (5.2 mm))
- Extension of corrosion and hydrogen uptake model
- Extension of materials properties database

Corrosion Test Loop

Measurement of hydrogen uptake at rolled joints

Pressure Tube Section

End-Fitting



Safety Technology Base

- The ACR design is fundamentally equivalent to the proven CANDU 6 design in terms of the overall safety system and safety-related system configuration and function.

  ➔ The key phenomena associated with safety analyses are common with the current CANDUs.

  ➔ The analysis tools and methodologies used for safety analysis of current plants are generally applicable to the ACR.

  ➔ The experience base from the current CANDUs is applicable to the ACR.

Technology Base Overview

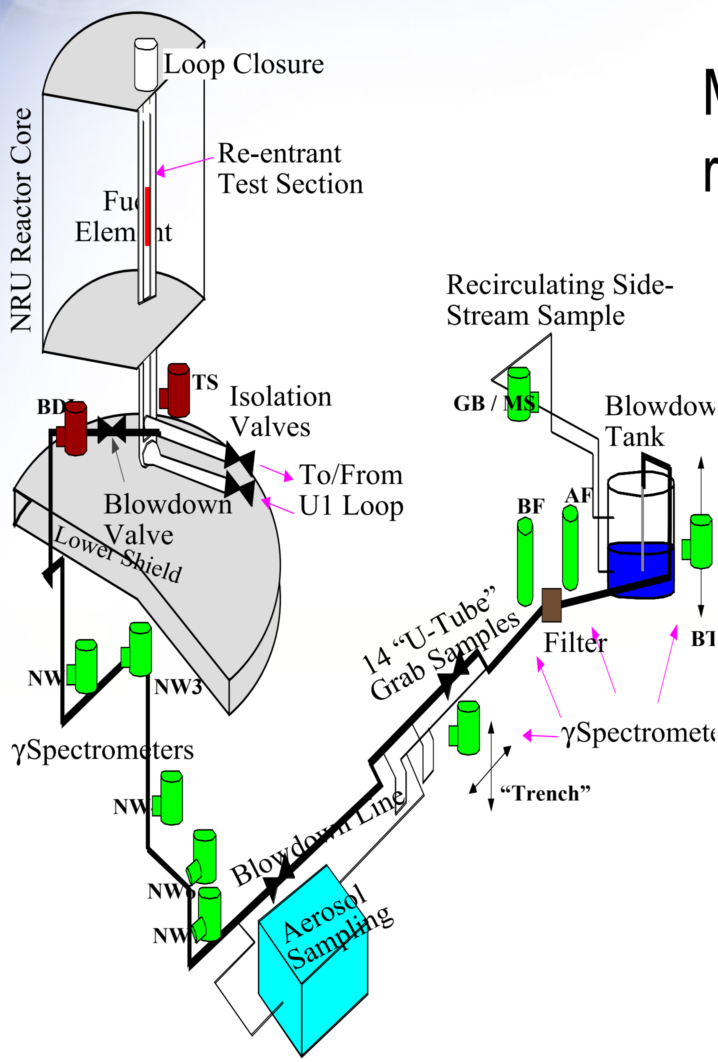
- Physics
- Fuel
- Fuel Channels
- Safety
  - Source Term
  - Thermalhydraulics
  - Containment
  - Severe Accidents
  - Code Qualification
  - Shutdown Systems
- Advanced Components

CANDU Source Term

- The CANDU source term for accidents is determined by transient calculation using accident boundary conditions and AECL fission product release tests.
- The source term from containment is backed by an iodine model supported by integrated iodine behavior tests.

Radioiodine Test Facility (RTF)

Blowdown Test Facility (BTF) in NRU



Measurement of CANDU fuel fission product release under severe accident conditions.

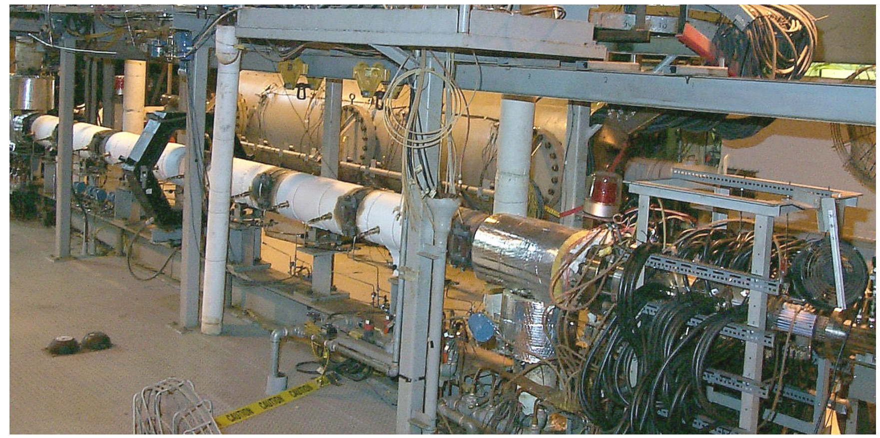
Thermalhydraulic Codes

- CANDU safety analysis uses three major thermalhydraulic codes:
  - NUCIRC – steady-state applications
  - CATHENA – transient and accident applications
  - MODTURC_CLAS – moderator applications

NUCIRC

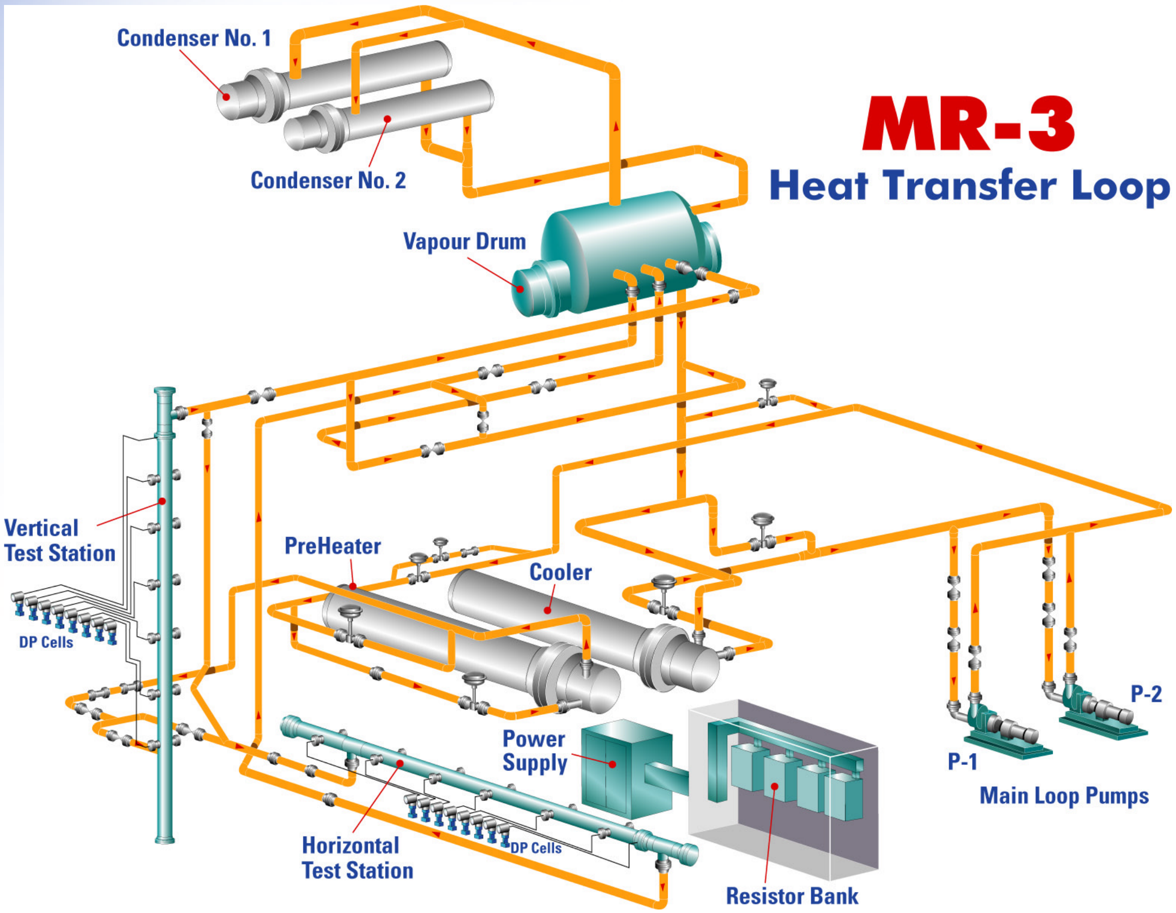
- **NUCIRC**
  - One-dimensional flow two-phase for steady-state applications
  - Evaluate critical channel powers (CCPs) for each channel based on reference system-flow conditions with critical heat flux (CHF) and pressure drop correlations
- **Test facilities**
  - Full-scale high pressure steam-water loop at Stern Laboratories
  - Full-scale low pressure Freon loop at CRL
  - Small-scale steam-water and Freon loops for fundamental and separate-effect studies using simple test sections or bundle sub-assemblies

NUCIRC Development



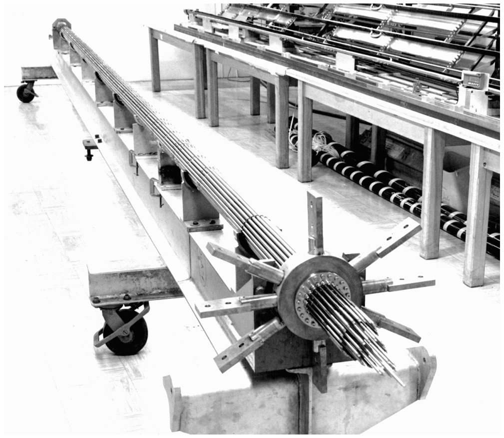
- Separate effect tests in Freon loops
- Critical heat flux measurements in water loop (coolant temperature and pressure, full-length fuel string simulator)

CRL Freon Test Facility



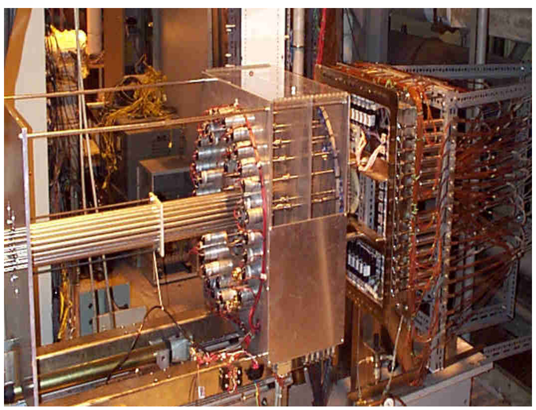
Full-scale CANDU fuel channel for thermalhydraulic tests

Test Bundle Simulator

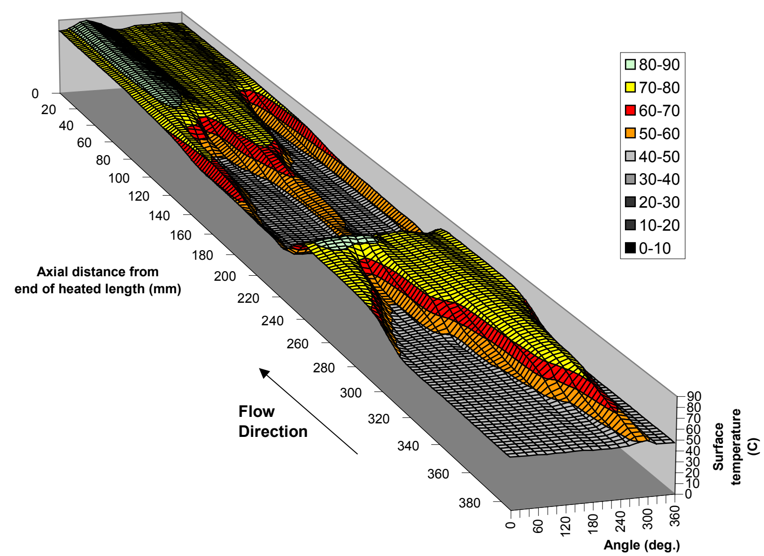


Advanced Instrumentation

- Dry-out and fuel cladding temperatures measured with sliding thermocouple assemblies
  - Cover almost the entire fuel cladding areas in the bundle at locations of interest
  - Ensure the initial dry-out occurrence has been detected
  - Provide 3-dimensional representations of cladding temperatures
- Sliding probes for pressure drop characterizations



Sliding thermocouple Controller

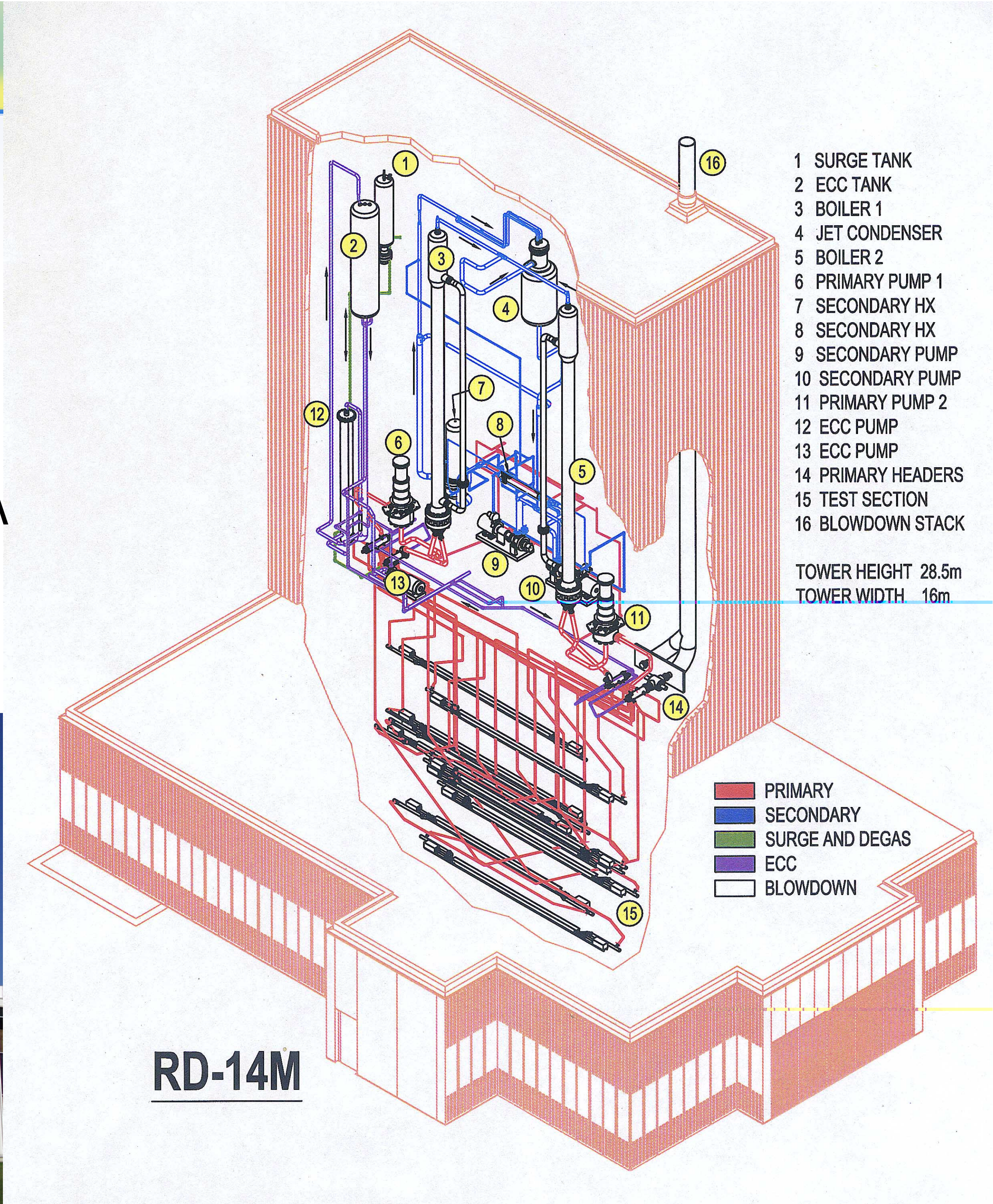


CATHENA

- CATHENA
  - One-dimensional non-equilibrium two-fluid flow for transient and accident applications
- Validation on a phenomenon-by-phenomenon basis.
- Data selected in validation process includes results from numerical tests, separate effects tests, component tests and integral tests.

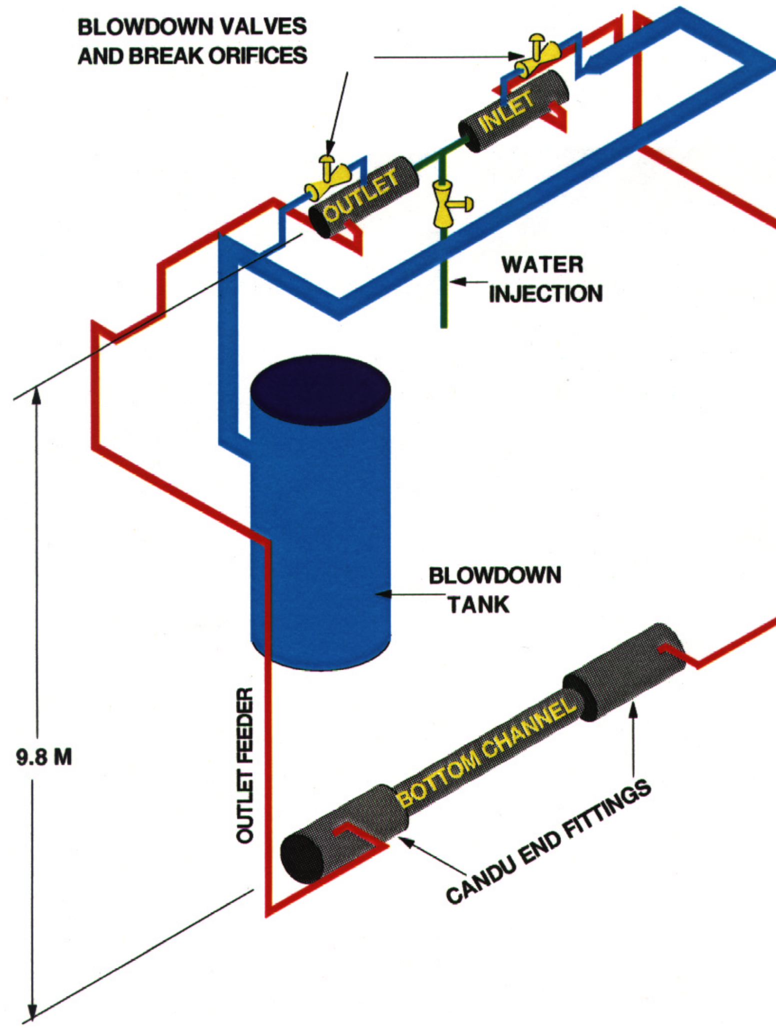
RD-14M

Full elevation scaled CANDU loop for CATHENA validation



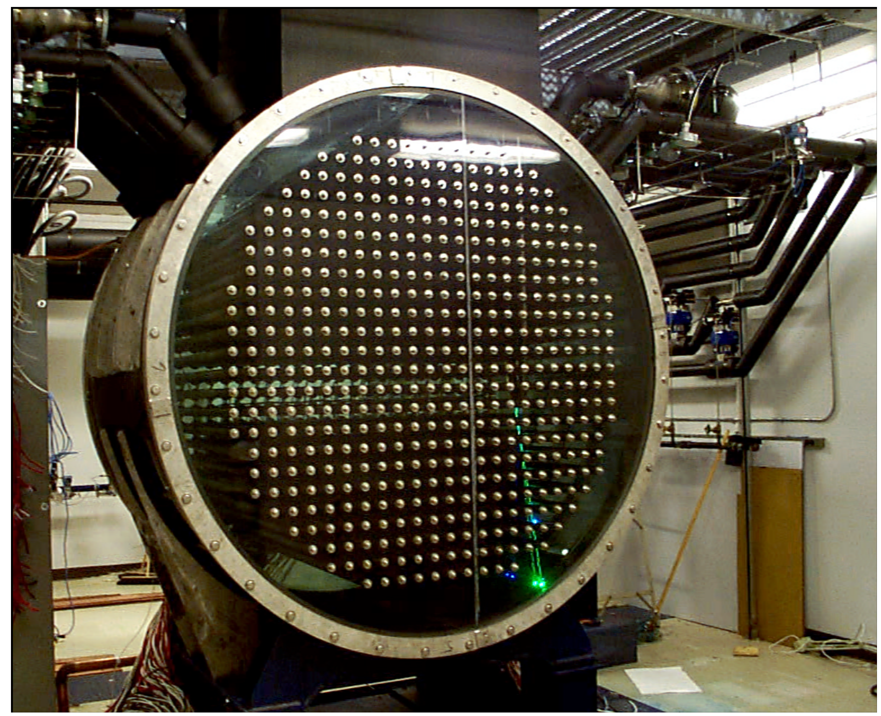
Cold Water Injection Test Facility

- Facility to test blowdown and channel refill performance during LOCA
  - Full-scale heated fuel channel & fuel string
  - Representative feeders and end-fittings

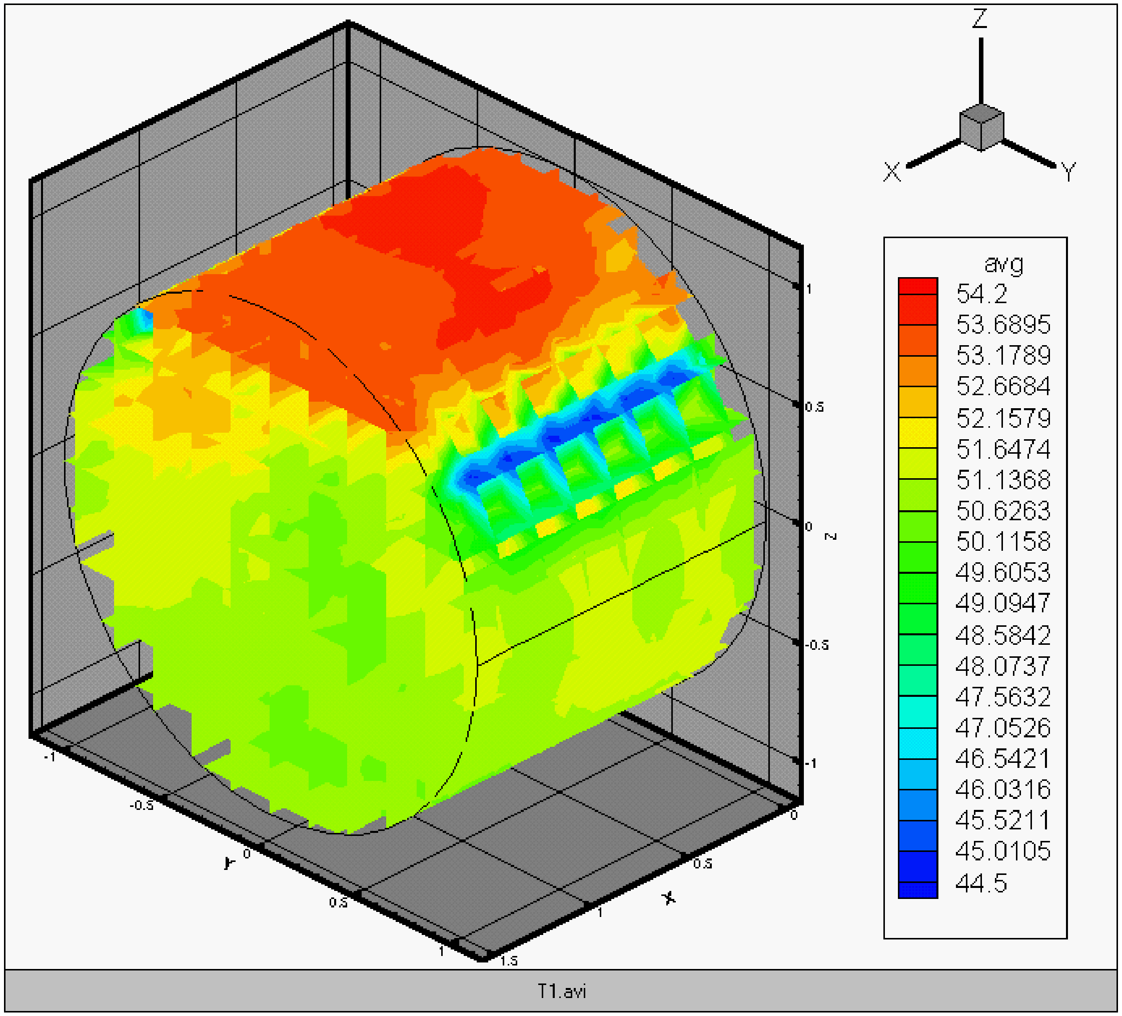


MODTURC_CLAS

- MODTURC_CLAS
  - A 3-D single phase computational fluid dynamics computer code (MODTURC_CLAS) is used to predict moderator flow and temperature distribution.
- MODTURC_CLAS Validation based on tests in the Moderator Test Facility (MTF)
  - MTF provides three-dimensional velocity and temperature distributions in moderator geometry.
  - 1/4 scale calandria used to validate CANDU 9 design.
  - 1/3 scale calandria will be used to validate ACR design.



Moderator Thermal Analysis



Containment

- ACR containment behavior will be modeled using the validated CANDU analysis tools – no new phenomena.
- Technology basis
  - Large-scale gas mixing facility
  - Large-scale combustion test facilities
  - Radioiodine Test Facility
- Modelling:
  - GOTHIC, with addition of CANDU-specific models for hydrogen behavior, used to model containment thermalhydraulics and hydrogen transport
  - LIRIC/IMOD – iodine behavior models

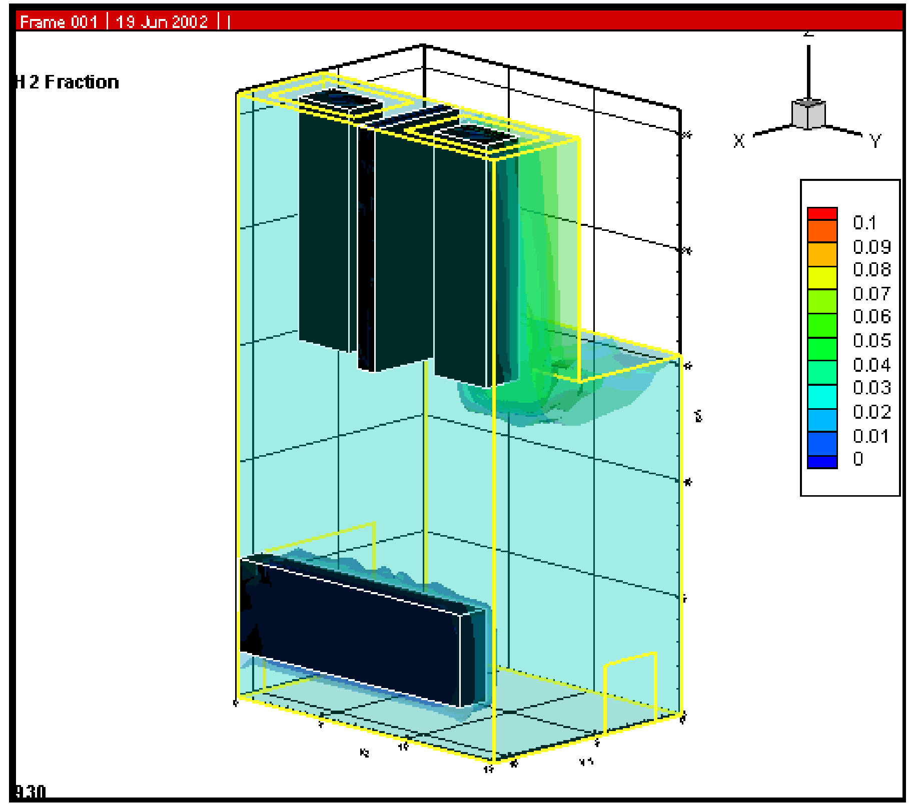
Hydrogen Combustion

Large-Scale Vented Combustion Facility used to establish deflagration model.

Diffusion Flame Facility

GOTHIC Validation

Large-Scale Gas Mixing Facility used to validate GOTHIC code for containment.



Predicted hydrogen distribution in containment during a large LOCA (header break).

IODINE Model Validation

LIRIC

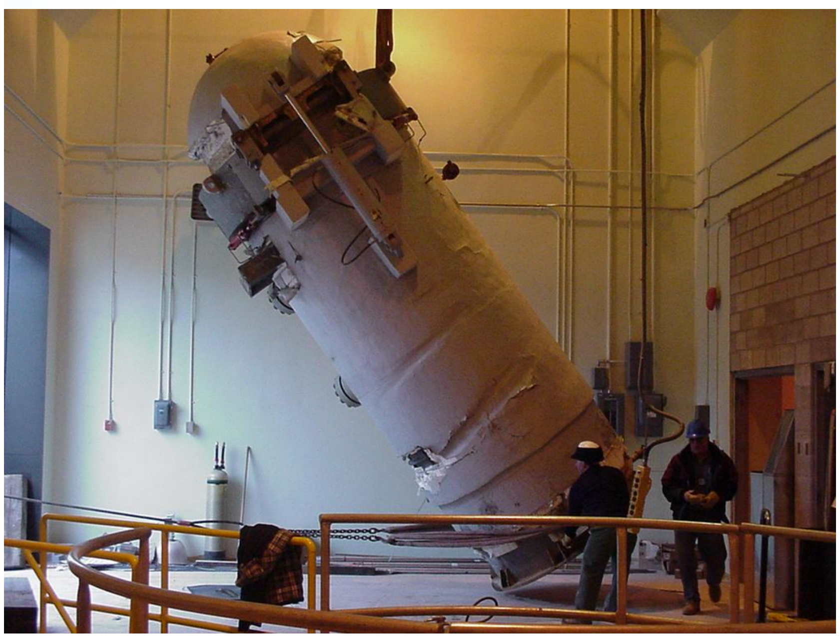
- **A comprehensive mechanistic model, based on our extensive knowledge of relevant chemical reactions and mass transport.**
- **Performs well when tested against bench-scale and Radioiodine Test Facility tests carried out over a wide range of conditions.**
- **Benchmark code used as a reference for production code, IMOD.**

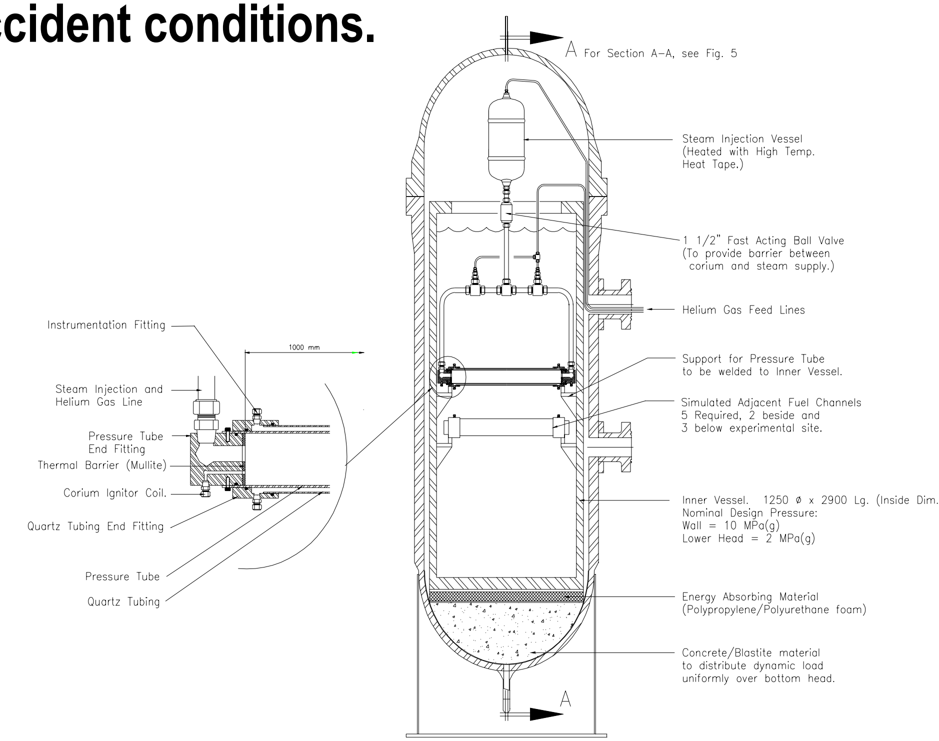
IMOD

- **Reduced reaction set based on extensive LIRIC analysis and simulations of various RTF tests.**
- **A smaller and simpler model, but maintains many of the capabilities of LIRIC.**

Molten-Fuel Moderator-Interaction Tests (MFMI)

- Instantaneous and complete flow blockage of a single channel has been assumed in the CANDU safety analysis.
- A test program is in place to confirm the dominant mechanism of interaction between molten fuel (ejected at operating pressure) and the moderator under CANDU accident conditions.





ARRANGEMENT

Severe Accident Categories

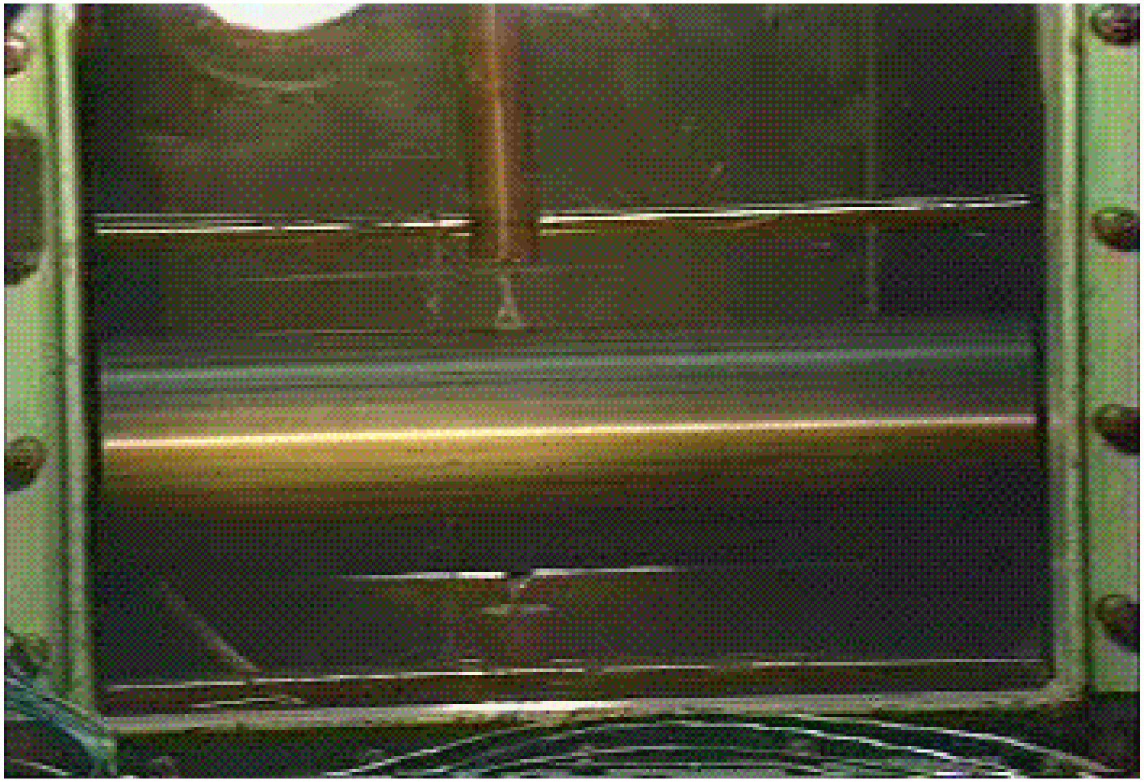
There are two categories of severe accidents for CANDUs:

- **Moderator available to remove decay heat – even with no coolant in the channels (e.g., LOCA + LOECC)**
  - **No loss of core geometry**
  - **No fuel melting**
- **Moderator heat removal unavailable (e.g., LOCA + LOECC + loss of moderator cooling)**
  - **Slow boil-off (many hours) and core collapse**

Moderator Heat Sink Availability

Fuel channel integrity and the availability of the moderator heat sink for severe core under-cooling conditions is supported by experiments.

High-temperature heat transfer tests with full-scale sections of fuel channels and simulator heaters.



Severe Core Damage Accident Program

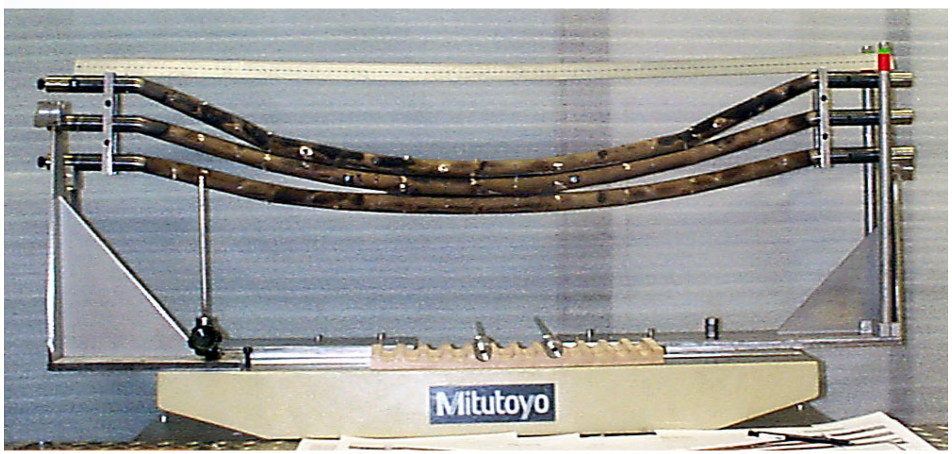
- ACR program is an extension of the current CANDU technology base.
  - Models in MAAP-CANDU will be extended to include the ACR design.
  - Core collapse model
  - Hydrogen combustion and gas mixing models (GOTHIC)
  - High-temperature channel heat transfer
  - Single channel failures
  - Fission product release

Severe Accident Analysis Methodology

- The CANDU nuclear industry has adopted the MAAP code to create the MAAP-CANDU code to model severe core damage accident progression.
- There is an ongoing experimental program to support the CANDU specific models in the MAAP-CANDU code (including reactor core collapse).

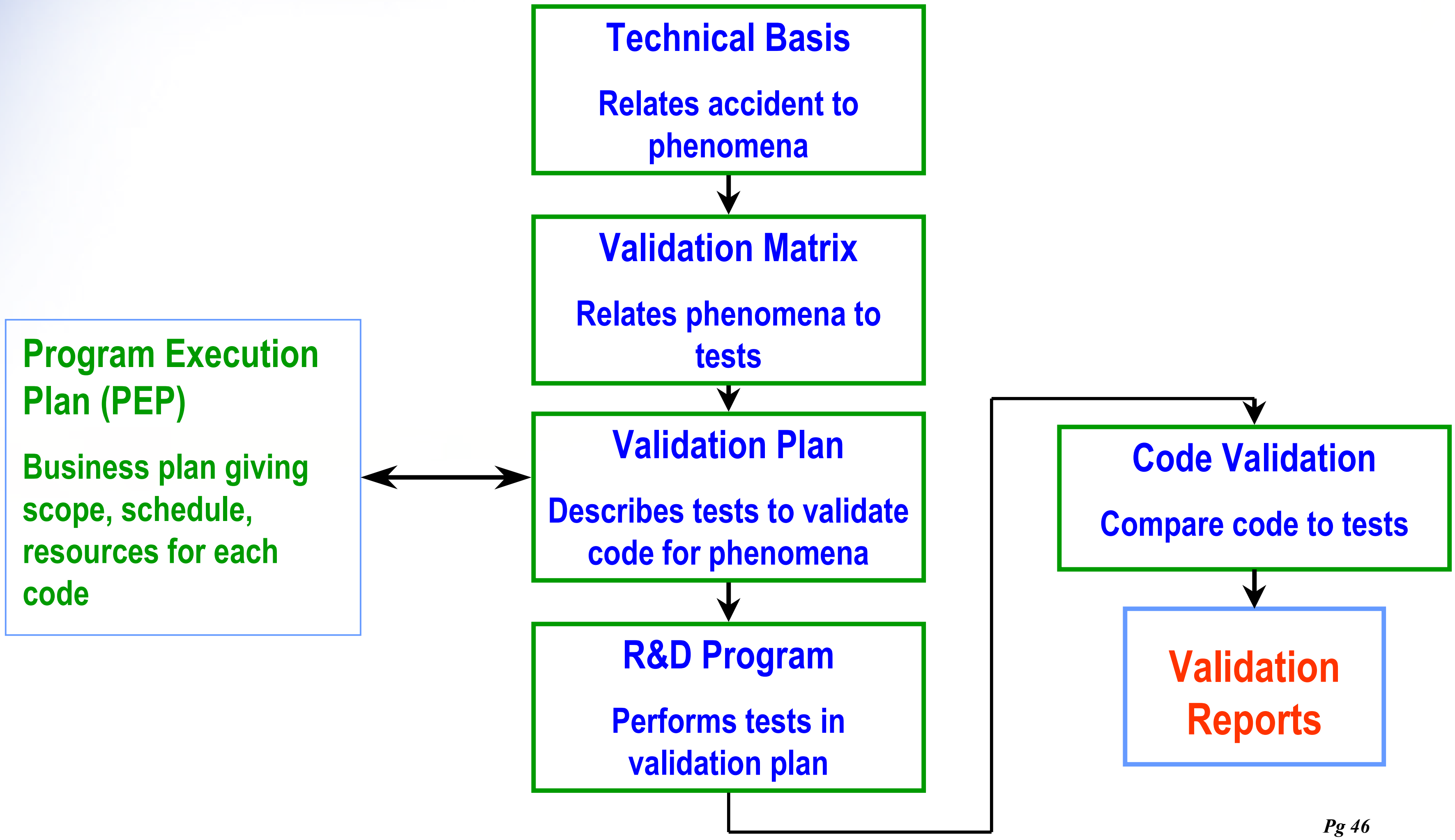
Core Collapse Model Development

- CANDU core collapse during severe accident sequences is slow and graceful.
- A test program is in place to confirm MAAP-CANDU core collapse models.



Tests with scaled arrays of
CANDU fuel channels

Code Validation Uses Standard Methodology



Other Safety Analysis Codes

- Fuel behavior: ELESTRES (normal operating conditions and accidents) and ELOCA
- Fission Product behavior: SOURCE, SOPHAEROS, SMART and ADDAM

Extensive safety R&D provides solid technology base for confidence in safety analysis tools.

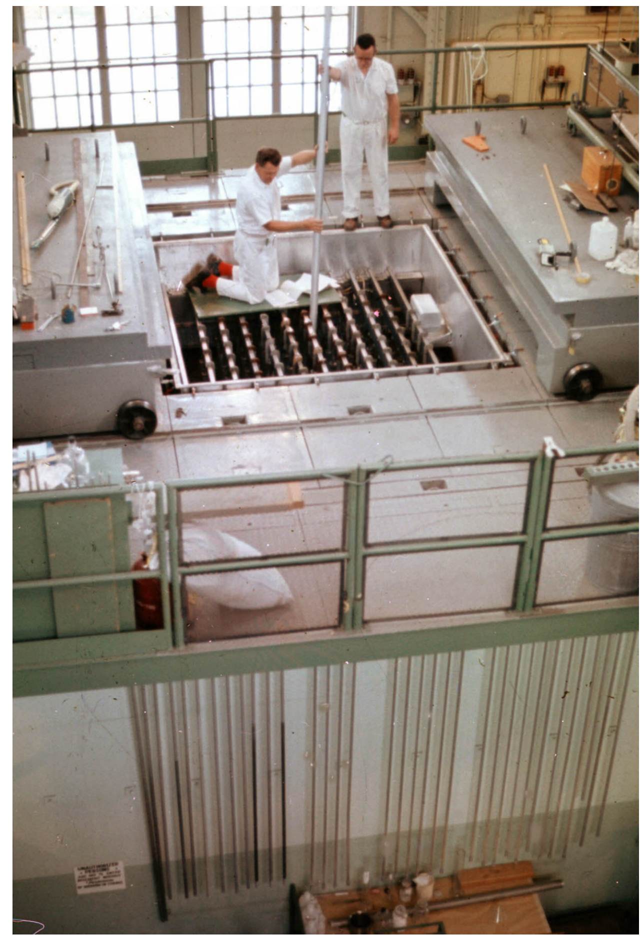
Safety Shutdown System Performance

- SDS 1 is based on the proven CANDU shut-off design.
  - The ACR design performance will be confirmed by testing.
- SDS 2 is based on the proven CANDU liquid poison injection design.
  - The ACR design performance will be confirmed by testing.

SDS 1 Qualification

Full-scale drop tests of prototype shut-off rods.

Tests of rod reactivity worth in ZED-2.



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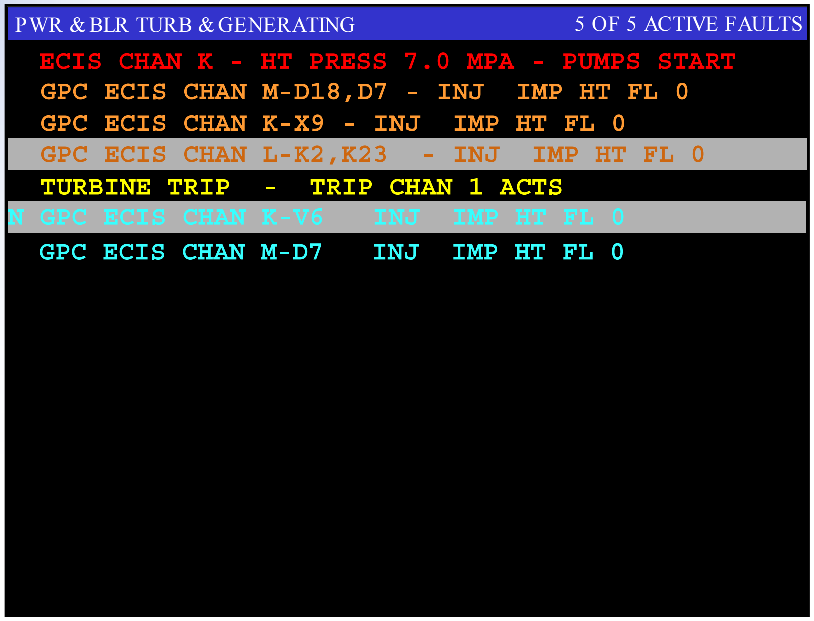
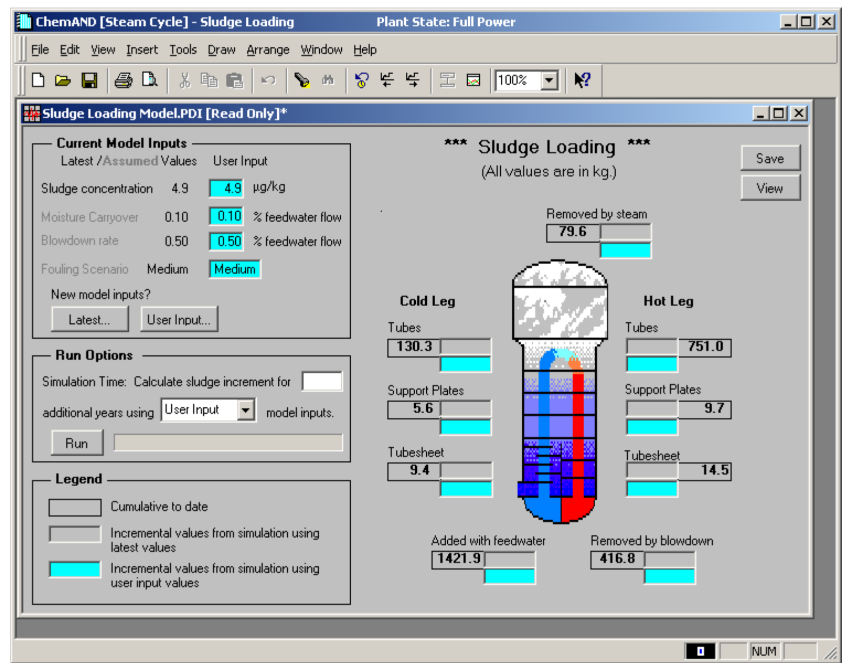
Component Development

- R&D to confirm design decisions, support design specifications or to achieve improved performance:
  - Steam generators
  - Reactor assembly
  - Chemistry control and materials specifications
  - Plant monitoring capabilities
  - Information communication and management technology
  - Non-safety systems improvements
  - Fuel Handling systems
    - Channel closure

Advanced Plant and System Health Monitoring



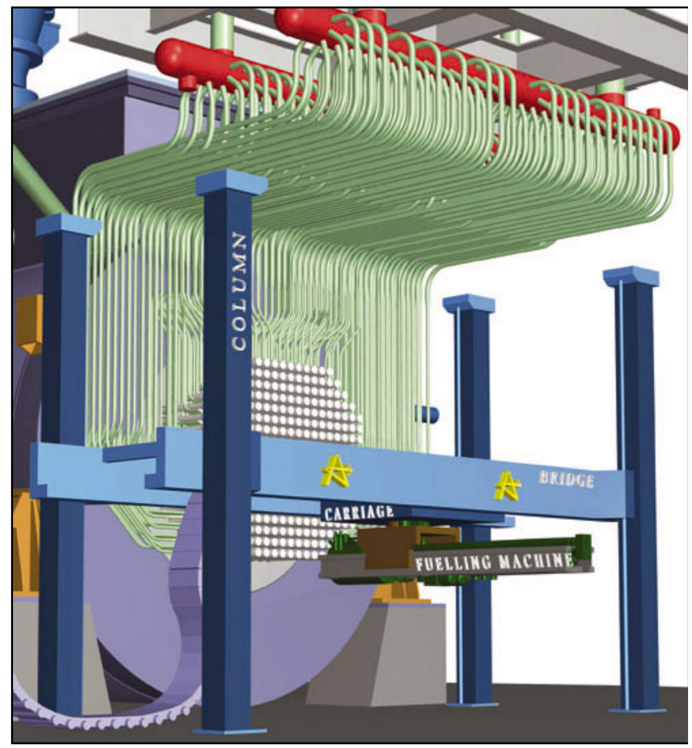
Improved control room technology and alarm monitoring capability



Chemistry and plant performance monitoring and diagnostic capability

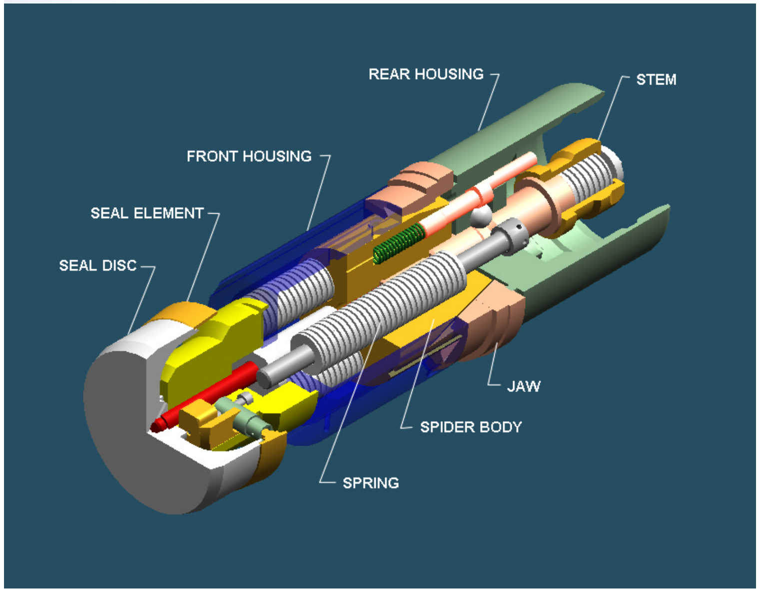
Fuel Handling Development

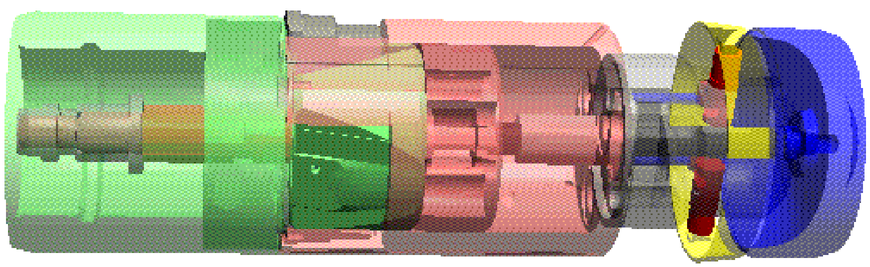
- The ACR on-line refueling systems are based on the proven CANDU concepts.
- The ACR fuel handling components are designed to improve performance and reliability.
- A comprehensive testing program will confirm the design of the key fuel handling system components.



Advanced Bore Seal Closure

Improved channel seal that enables new end-fitting design for a compact core.

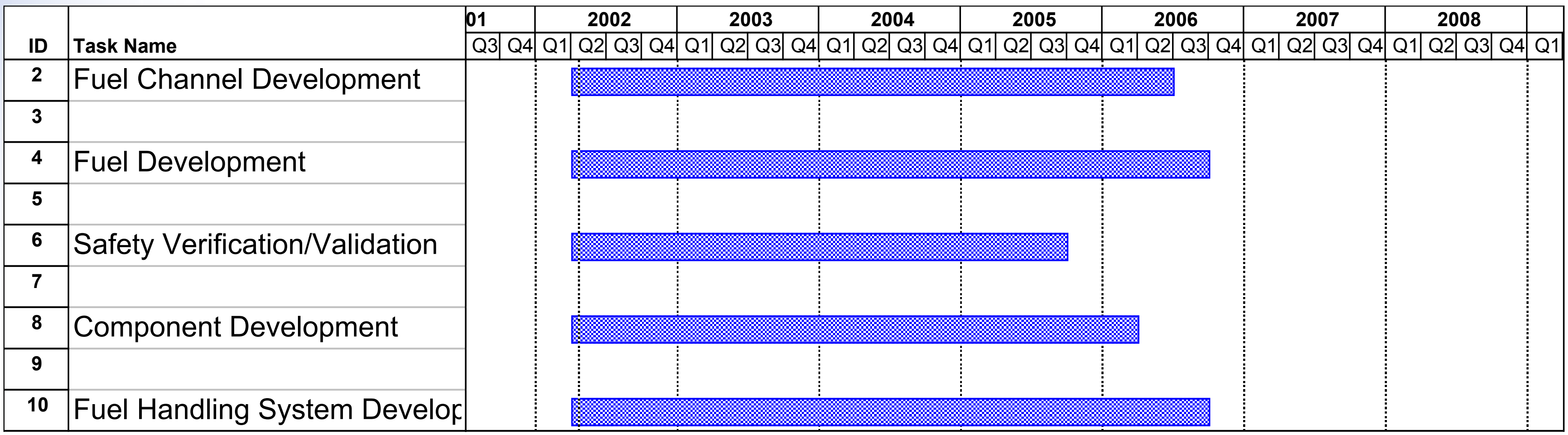




R&D Program Integration

- Integrated with the ACR Basic Engineering Program
  - Supports safety assessments
  - Supports conceptual design decisions
- Integrated with AECL's ongoing Platform R&D program in support of current CANDU plants
  - Programs directed to provide data of common value
  - Development of generic codes

Overall Project R&D Schedule



Summary

- The ACR design is based on the well-established technology base of the CANDU product line.
- The anticipated R&D program to support the development of this evolutionary product is tightly focused.
- The ACR safety analysis will use the validated CANDU analysis tools supported by the results of large-scale tests.
- An integrated R&D program is in place to provide the additional information required to support the ACR design.

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