ACR Safety Analysis Code Validation

Dave Wren

Manager, ACR Research and Development

Presented to US Nuclear Regulatory Commission

Washington DC

May 15-16, 2003

AECL
TECHNOLOGIES INC.

Outline

- Outline of the ACR Code Validation Requirements
- ACR Code Validation Test Program (for key codes)
  - CATHENA
  - NUCIRC
  - WIMS / RFSP / DRAGON
  - ELESTRES
  - ELOCA
  - TUBRUPT
  - MODTURC_CLAS

Presentation Objectives

- Provide a background of the validation requirements and process for AECL computer codes
- Summarize the areas where incremental validation for codes used in ACR analysis is required
- Provide summary of the validation status for the key ACR safety analysis codes
- Outline specific experimental tests that are needed to expand the validation database for ACR conditions
- Provide a brief summary of the schedule for completion of planned experiments and validation tasks

Code Validation for ACR Application

- The computer codes to be used in the safety analysis of the ACR will be qualified following the requirements AECL's Software Quality Assurance Manual and processes
- ACR will use the validated CANDU Industry Standard Toolset (IST) for safety analyses
  - Most of the computer codes used by the nuclear industry in Canada for reactor safety analyses are shared by the industry partners
- Limited modifications to the IST codes required to address new ACR analysis requirements
  - Assessment of code applicability to ACR-700 application performed systematically and consistently for each code
- In requisite areas the validation base of the codes will be extended to cover a new range of application for the ACR

ACR Code Validation Program

- Overall validation requirements established by Technical Basis Document (TBD)
- Validation Matrices establish for database for key phenomena
- R&D program provides extension to the database, if required, for ACR code application

Key ACR Safety Analysis Codes

- CATHENA – transient thermal-hydraulics
- NUCIRC
- WIMS/RFSP/DRAGON
- ELESTRES
- ELOCA
- TUBRUPT
- MODTURC

CATHENA Validation Status - 1

- Validation Requirements Established by Technical Basis Document
- Validation Matrix Documents:
  - Phenomena covered by existing Thermal-Hydraulics and Fuel and Fuel Channel Validation Matrix documents apply to ACR, but range of parameters differ slightly in some cases
  - Data needed to extend the range of applicability to ACR conditions
  - Validation Matrix Documents being updated

CATHENA Validation Status - 2

- All relevant phenomena examined for applicability of existing validation to to ACR
- 15 phenomena require no further validation
- Validation extension identified for remaining 8 phenomena
- Qualification Plan outlines activities required to address these gaps, including, where necessary, requirements for new experimental data

CATHENA Validation Status – 3

Phenomenon	Validation Extension for ACR
TH1: Break Discharge Characteristics	Extend validation to include higher pressures. Existing Edwards Blowdown tests and RD-14M LOCA tests completed.
TH2: Coolant Voiding	Extend validation to include higher pressures. Existing Edwards Blowdown tests and RD-14M LOCA tests completed.
TH4: Level Swell and Void Holdup	Existing validation includes pressures up to 7.3 MPa. Validation is considered adequate, but could be extended to include higher pressures. Literature search for additional data planned.

CATHENA Validation Status - 4

Phenomenon	Validation Extension for ACR
TH7: Convective Heat Transfer FC13: Cladding-to-Coolant and Coolant-to-Pressure Tube Heat Transfer	Validation to extended to include higher pressures. RD-14M tests planned.
TH8: Nucleate Boiling FC13: Cladding-to-Coolant and Coolant-to-Pressure Tube Heat Transfer	Validation to be extended to include higher pressures. RD-14M tests planned.
TH10: Condensation Heat Transfer	Validation to be extended to include higher pressures. RD-14M tests planned.

CATHENA Validation Status - 5

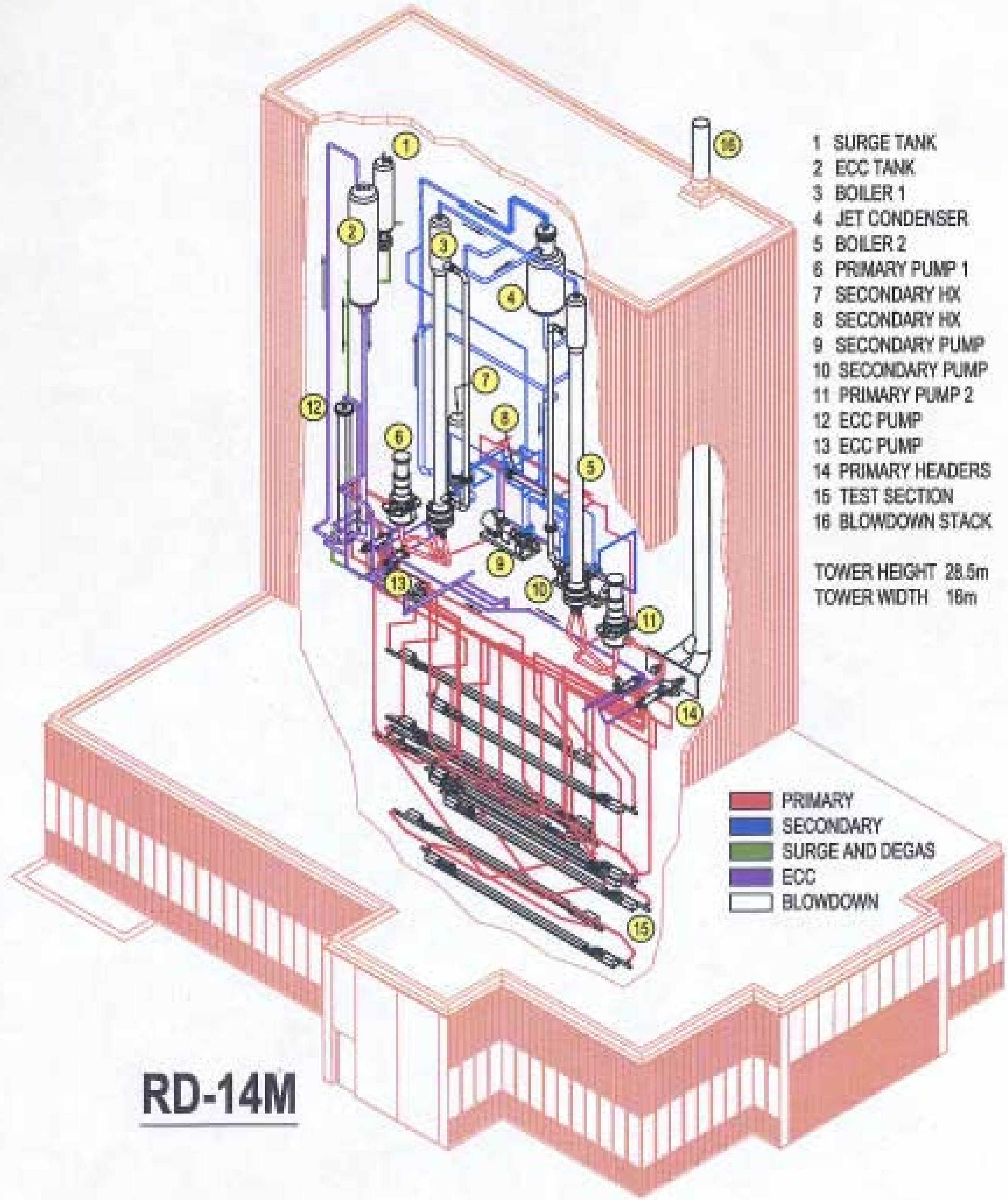
Phenomenon	Validation Extension for ACR
TH11: Radiative Heat Transfer FC21: Element-to-Pressure Tube Radiative Heat Transfer	Validation extended to include radiation between fuel pins of different diameters. Numerical test.
TH18: Fuel Channel Deformation FC18: Pressure Tube Deformation or Failure FC19: Calandria Tube Deformation or Failure	Existing validation considered adequate until data from tests with ACR PT and CT become available.

RD-14M Test Program

- Break discharge and coolant voiding
- ECC performance validation
  - Scaling analysis and loop reconfiguration for ACR-specific tests in progress
    - Scaling analysis for CANDU 6 application completed
  - LOCA + ECC tests
- Heat transfer at ACR pressures
    - Condensation
    - Nucleate boiling
    - Convection

RD-14M

Full elevation scaled CANDU loop for CATHENA validation



Key ACR Safety Analysis Codes

- CATHENA
- NUCIRC – fuel channel thermal-hydraulics
- WIMS/RFSP/DRAGON
- ELESTRES
- ELOCA
- TUBRUPT
- MODTURC

NUCIRC Code Development

- Development and Verification:
  - Enhancements to CHF and fuel channel pressure drop models for ACR conditions (up to 20 MPa)
    - Design Description and Requirements Document
    - Model enhancements installed, verified, and documented
  - Enhancements to Light Water Fluid Properties (up to 20 MPa)
    - installed, verified and documented
  - Enhancements to property tables (sub-cooled properties)

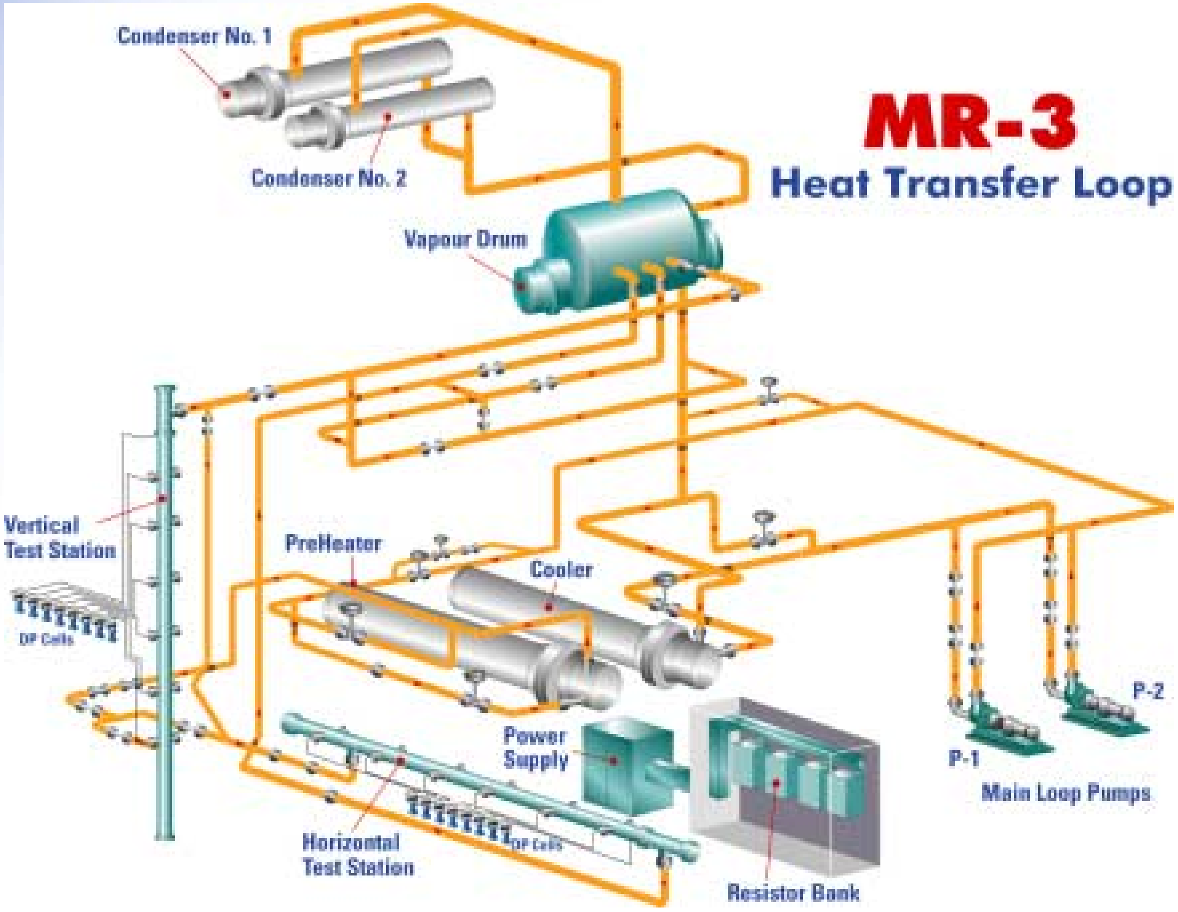
NUCIRC Validation

- Extension to the validation database for thermal-hydraulic correlations required to address ACR coolant conditions and ACR-CANFLEX fuel design
  - Critical heat flux (CHF)/dryout power
  - Fuel channel / header-header pressure drop
- ACR-CANFLEX fuel
  - Inlet-skewed axial power profile
  - Flattened radial power profile
  - Higher bearing pads to improve CHF margin

NUCIRC Validation Tests

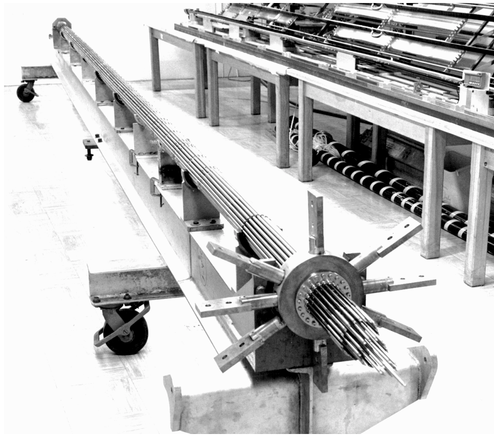
- Test Program
  - Full-scale water CHF tests
    - Correlation database
  - Full-scale Freon CHF and post-dryout tests
    - Variations in radial and axial heat flux profiles
    - Overpower tests

CRL Freon Test Facility



Full-scale CANDU fuel channel for thermal-hydraulic tests

Test Bundle Simulator



Stern Laboratory Water CHF Tests

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

Key ACR Code Validation

- CATHENA
- NUCIRC
- WIMS/RFSP/DRAGON – reactor physics
- ELESTRES
- ELOCA
- TUBRUPT
- MODTURC

Physics Tools

- **WIMS-IST (+ data library with burnable Dy isotopes)**
  – Lattice cell calculations
- **DRAGON-IST** – Device incremental cross-section calculations
- **RFSP-IST** – Core Simulations
- **MCNP** – Benchmark Tool

Physics Code Validation

- Code applicability and qualification requirements have been determined.
- Existing validation base of physics codes adequate to support code application for design
- Validation will be incremental with respect to CANDU 6 (CANDU 6 validation completed as part of the Industry Standard Tool set effort)
  – Validation against existing data sets
  – Validation against experiments in ZED-2
    - Natural uranium fuel in preliminary reference lattices
    - SEU/Dy fuel in preliminary reference lattices
    - ACR fuel in improved reference lattice
  – Inter-code comparisons

ACR Design Factors Affecting Validation

- Light water coolant
- SEU fuel with Dy element
- New lattice geometry
  - Reduced lattice pitch
  - Fuel channel design (CT – PT dimensions)
- Modified reactivity control element designs

Physics Code Development

- ACR-specific data library developed
- WIMS model for ACR lattice prepared
- Adequacy of energy-group representation in core simulations assessed
- MCNP vs WIMS/RFSP comparisons with ACR core model
- Preliminary estimate of WIMS/RFSP core void reactivity bias and uncertainty

Physics Toolset Validation

- **WIMS/DRAGON validation**
  - ZED-2 measurement data
  - Data from other fuel studies
  - Data from other criticality facilities germane to ACR
  - MCNP / WIMS comparisons
- **WIMS/RFSP validation**
  - MCNP comparisons – full or partial core model
  - Modeling of ZED-2 measurements
  - Commissioning physics test data

Validation Test Program

- Preliminary validation tests in ZED-2 using available substitution fuel and reference lattice fuel
  - Measurements with 28-element NU fuel, light water coolant and tight pitches
  - Measurements with 37-element LVRF bundles (SEU + Dy)
- Validation tests in ZED-2 with ACR fuel in ACR fuel channels with and improved reference lattice fuel
  - To reduce the uncertainty in validation a full core of reference fuel for ZED-2 is being manufactured (0.95%U^{235})

ZED-2 Critical Lattice Facility

- fuel substitution tests in reference lattices

ZED-2 Measurements Using Natural Uranium Fuel

Experiment		Purpose	Phenomenon
1	28-element UO_2 flux maps: H_2O Cooled and Voided	Lattice reactivity and void reactivity at reduced lattice spacing (buckling change on voiding) using H_2O and air coolant	PH0, PH1
2	37-element UO_2 substitutions	Base-line data to establish room-temperature buckling for fuel/coolant temperature coefficient measurements for Dy-poisoned SEU fuel and MOX using H_2O and air coolant	PH0
3	37-element UO_2 fuel/coolant temp. Coefficient	Temperature reactivity coefficients (up to 300^0C) for Dy-poisoned SEU fuel and MOX in tight H_2O cooled and voided (CO_2 gas-cooled) lattices	PH2, PH7
4	28-element UO_2 substitutions	Expand the substitution method validation matrix for tight-lattice spacing	

ZED-2 Measurements with SEU Fuel

	Experiment	Purpose	Phenomenon
1	Criticality and coolant void reactivity flux maps	Lattice and void reactivity for full cores of SEU fuel in H_2O and air cooled tight lattices	PH0, PH1
2	Criticality and coolant void reactivity substitutions	Lattice and void reactivity for SEU fuel in H_2O and air cooled tight lattices with expanded calandria tubes	PH0, PH1
3	Fuel/coolant temperature coefficient	Temperature reactivity coefficients (up to 300°) for both SEU and MOX fuel in H_2O and air cooled tight lattices	PH1, PH2 PH7
4	Moderator temperature coefficient by flux map measurements	Moderator temperature coefficient in the range 10° to 40°C for SEU fuel in H_2O and air cooled tight lattices	PH1, PH3 PH4
5	Fine structure flux distribution	Flux distribution in a SEU CANFLEX bundle at both room in H_2O and air cooled tight lattices	PH0, PH1 PH2, PH7 PH14
6	Moderator poison experiments	Boron and/or Gd reactivity effect in ACR type lattices using both H_2O and air coolant	PH1, PH5
7	Benchmark configuration flux distribution	Spatial flux distribution in a heterogeneous ACR-type lattice of SEU and MOX fuel	PH0, PH8 PH14
8	Control device measurements	Absorber device inserted into square uniform or checkerboard lattice of SEU and MOX fuel	PH11
9	Rod drop experiments	Assessment of the contribution of delayed photo-neutrons to the total delayed neutron fraction	PH12

Key ACR Safety Analysis Codes

- CATHENA
- NUCIRC
- WIMS/RFSP/DRAGON
- ELESTRES – fuel - fission products during operation
- ELOCA
- TUBRUPT
- MODTURC

ELESTRES

- The applicability of ELESTRES-IST for ACR analysis was assessed and a number of required improvements were identified:
  - Mandated by
    - increased U-235 content
    - increased burnup compared to natural uranium CANDU
    - Dy fuel element
  - Supported by experience from the application of ELESTRES to SEU fuel designs in the mid 1990s

ELESTRES 2.0 – ACR Code Extension

Current interim version	Status
– Flux depression	Completed
– Fission gas diffusivity	Completed
– Grain boundary bubbles	Completed
– Free standing cladding to collapse including degree of circumferential wrap-around	Completed
– Variable heat transfer	Completed
– Cladding material properties	Completed
– Pellet densification	In-progress
– Cladding plasticity	In-progress
– Dy properties	Planned
– Link to externally made fine-element meshes	Planned
– Cladding oxidation	Planned

ELESTRES Validation

- Verification of new code version in progress
- Validation will be conducted against existing database
  - Fuel irradiation in NRU routinely use SEU fuel
  - Data available on Dy fuel from development program for low void reactivity fuel

Key ACR Safety Analysis Codes

- CATHENA
- NUCIRC
- WIMS/RFSP/DRAGON
- ELESTRES
- ELOCA – fuel – fission product release - accidents
- TUBRUPT
- MODTURC

ELOCA-IST

- ELOCA models fuel elements under the rapidly changing coolant and power conditions typical of an accident
- ELOCA-IST calculates:
  - Fuel temperatures
  - Cladding temperatures
  - Internal gas pressure
  - Cladding strain,
  - Cladding failure
  - Cladding oxidation

ELOCA Development

- **ELOCA-IST 2.1 was assessed for application to ACR and three areas for code extension were identified:**
  - **Extension of the thermal properties of UO_2 fuel to the ACR fuel burnup**
    - **Thermal Conductivity**
    - **Thermal Expansion**
    - **Specific Heat Capacity**
  - **Extension of the thermal properties database to include the dysprosium fuel element, and**
  - **Extension of the database on fuel cladding strain**
    - **ACR cladding thickness**

UO_2 Thermal Properties Models

- The models affected are:
  - Thermal Conductivity
  - Thermal Expansion
  - Specific Heat Capacity
- These models have undergone further assessment and changes have been made to accommodate extended burnup and dysprosium-doped fuel
- Autoclave tests are planned to obtain additional thermal properties data

ELOCA Validation Test Program

- Laboratory test program to reduce uncertainties in extrapolation of fuel thermal properties in progress
  - Thermal Conductivity
  - Thermal Expansion
  - Specific Heat Capacity
- Test program to extend the database on cladding strain to include the ACR cladding design (diameter and thickness)

Key ACR Safety Analysis Codes

- **CATHENA**
- **NUCIRC**
- **WIMS/RFSP/DRAGON**
- **ELESTRES**
- **ELOCA**
- **TUBRUPT – in-core damage**
- **MODTURC**

TUBRUPT

- Used to determine the pressure transients within the calandria vessel/shield tank assembly:
  - Spontaneous pressure-tube/calandria-tube rupture
  - Severe flow blockage
  - Feeder stagnation break
- Modeled phenomena include:
  - Flashing coolant hydrodynamic transient in moderator
  - High temperature channel debris interaction with water
  - Ruptured channel projectile formation and impact on the calandria vessel, shutoff rod guide tubes, and on other fuel channels

TUBRUPT Validation

- Assessment of TUBRUPT incremental validation requirements in progress
  - Review of database to pertinent to the geometry and operating conditions of ACR
- Extension of calandria tube rupture database to ACR coolant conditions planned
- Extension to ongoing tests on molten-fuel-moderator interaction planned

Key ACR Safety Analysis Codes

- CATHENA
- NUCIRC
- WIMS/RFSP/DRAGON
- ELESTRES
- ELOCA
- TUBRUPT
- MODTURC – moderator thermal-hydraulics

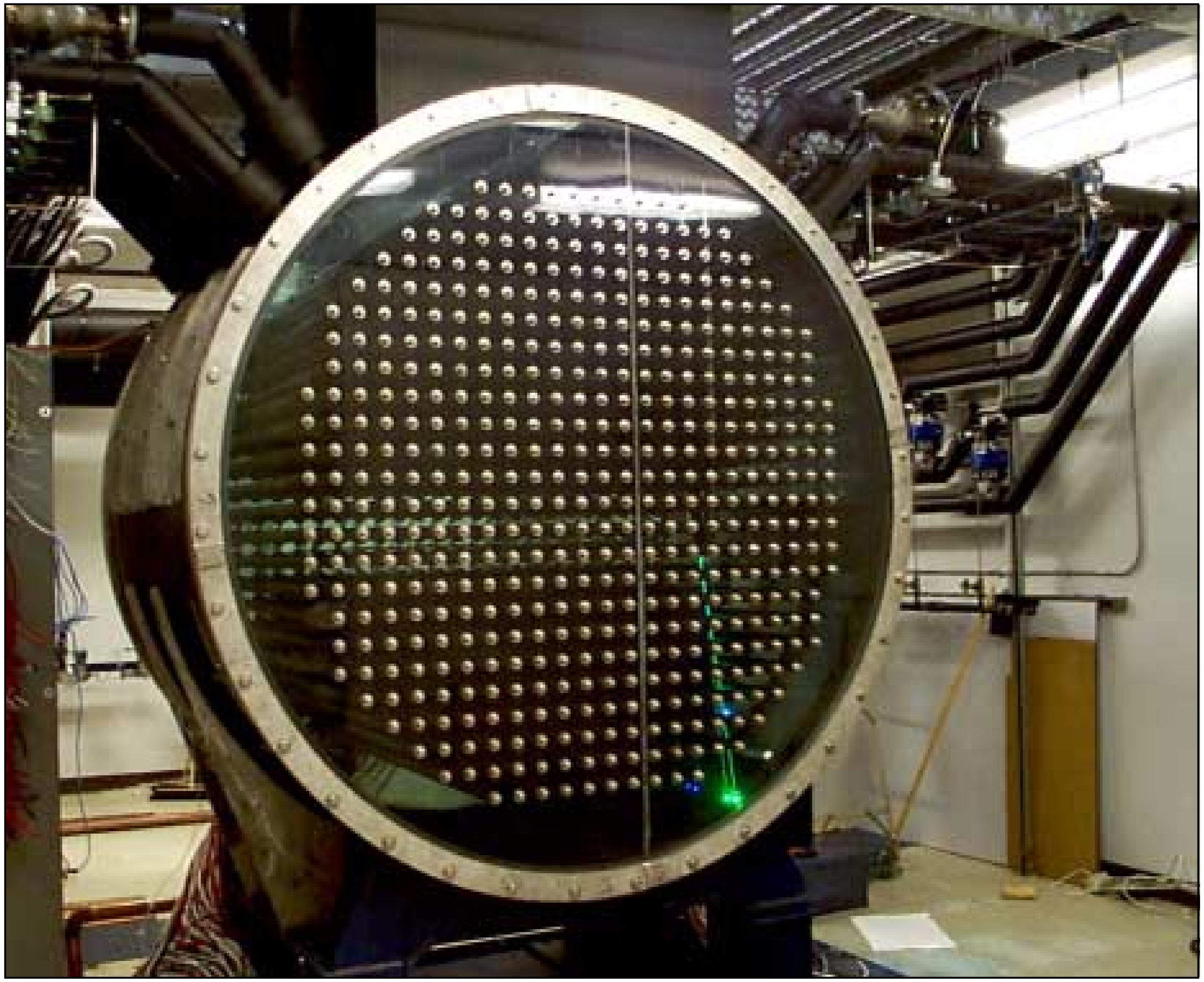
MODTURC_CLAS

- **A 3-D single phase computational fluid dynamics computer code (MODTURC_CLAS) is used to predict moderator flow and temperature distribution within the calandria vessel**
- **Validated for CANDU application using data from integral-effects tests in the Moderator Test Facility at CRL (tests at ¼ scale)**
  - **three-dimensional velocity and temperature measurements**

MODTURC_CLAS Validation for ACR

- Validation will be extended to the ACR
  - Moderator Test Facility will be 1/3 of the linear scale of the ACR
    - Larger scale possible because of smaller ACR calandria
- Validation data will be obtained from steady-state tests
  - isothermal, normal operation with two outlet-to-inlet temperature differences and same Archimedes number
  - inlet flow asymmetries at full and ~50% power
  - two stylized transients (large LOCA/LOECC, large LOCA/loss-of-class IV)

Moderator Test Facility

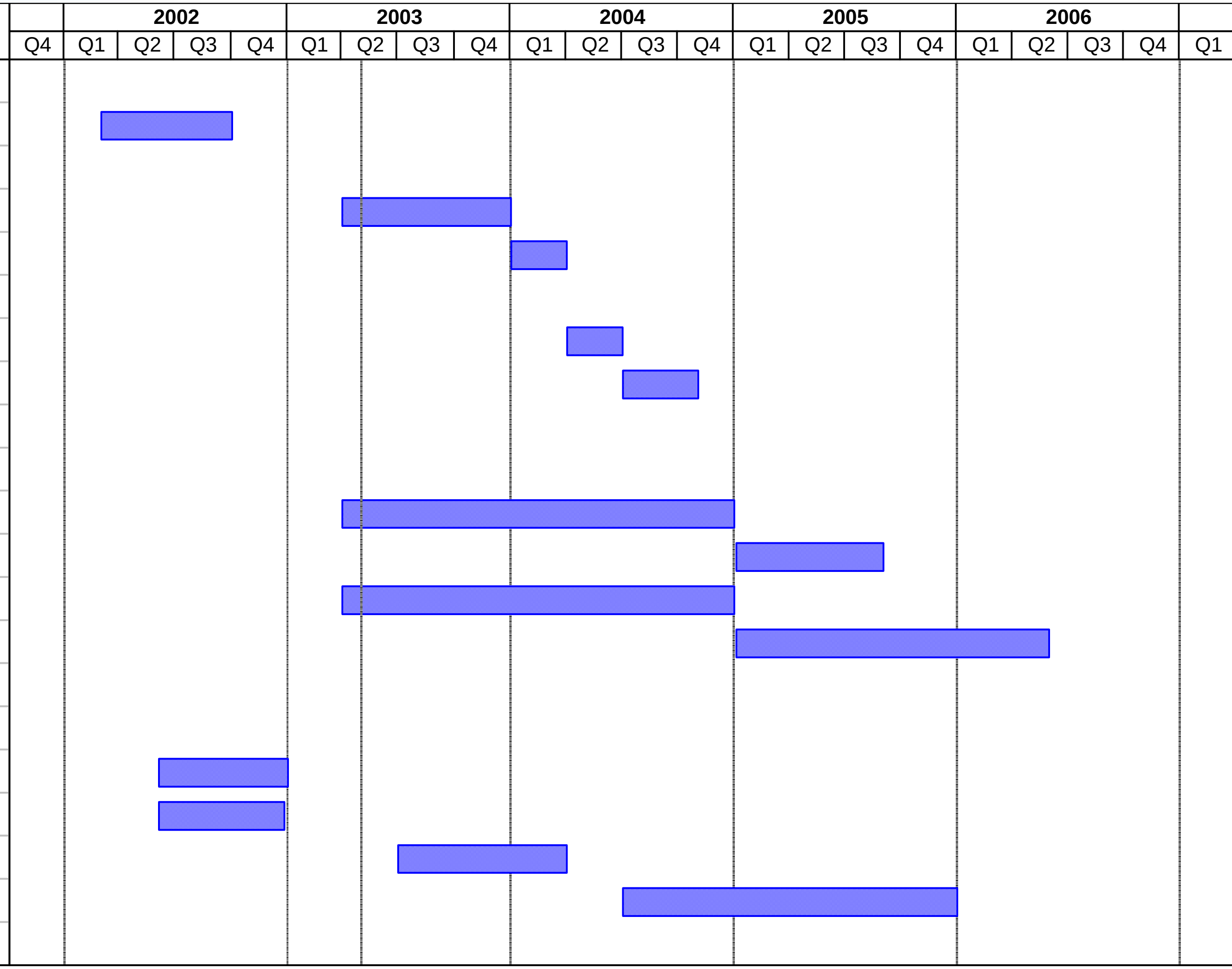


Validation Test Program Schedule

- Tests to obtain the data required to support extension to the validation of key ACR safety analysis codes planned and in progress
- Validation is incremental to reduce uncertainties in code applications

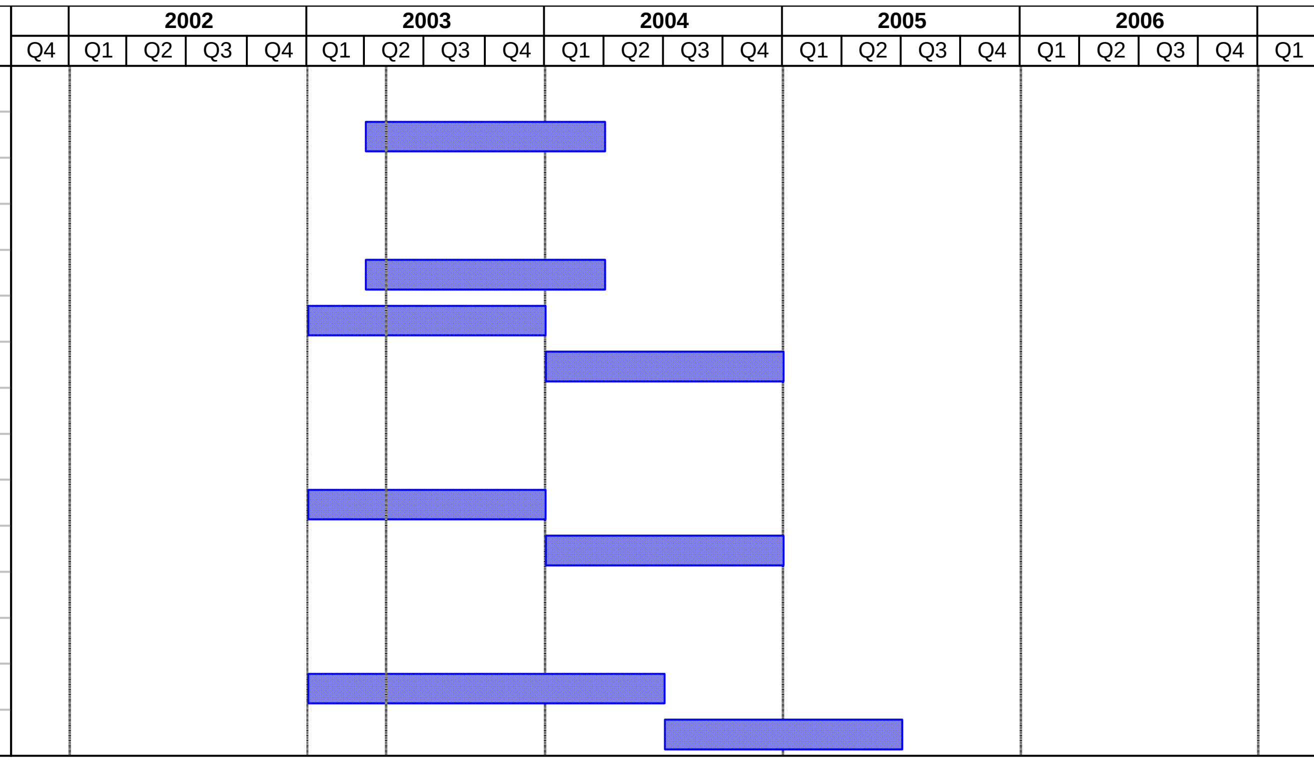
Key ACR Code Validation Test Program - 1

ID	Task Name
1	**CATHENA Validation**
2	RD-14M Break Discharge Tests
3	**RD-14M ECC Performance Tests**
4	ECC Test Preparation
5	ECC Tests
6	**RD-14M Heat Transfer Tests**
7	Heat Transfer Test Preparation
8	Heat Transfer Tests
9	
10	**NUCIRC**
11	Freon CHF and PDO Test Preparation
12	Freon Tests
13	Water CHF Test Preparation
14	Water CHF Tests
15	
16	**WIMS/RFSP**
17	NU Tests in Preliminary Lattice
18	37-Element SEU/Dy Fuel Tests
19	SEU Fuel Tests in Preliminary Lattice
20	Tests in ACR Reference Lattice
21	



Key ACR Code Validation Test Program - 2

ID	Task Name
22	**ELESTRES**
23	Validation Exercises
24	
25	**ELOCA**
26	Fuel Thermal Properties Tests
27	Cladding Strain Tests Preparation
28	Cladding Strain Tests
29	
30	**TUBRUPT**
31	Burst Test Preparation
32	Scaled Calandria Tube Burst Tests
33	
34	**MODTURC**
35	Moderator Test Facility Modification
36	Moderator Circulation Tests



Summary

- Validation extension requirements for key ACR safety analysis codes have been identified
- Test programs have been established to provide requisite additional validation data
- Incremental validation is in progress

AECL

TECHNOLOGIES INC.