Assessment of AECL Computer Codes for ACR Applications

Nik Popov, Manager
Reactor, Safety & Licensing

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TECHNOLOGIES INC.

Outline

- Overview of key computer codes for ACR safety analysis
- ACR key design features
- Overview of the code assessment objectives and process
- Overview of code assessment progress
- Summary assessment findings for key ACR computer codes
- Conclusions

Presentation Objectives

- Provide an overview of key computer codes for ACR safety analysis
- Provide an overview of the code assessment objectives, process and progress
- Provide an overview of the codes assessment for application to ACR-700
  - incremental with respect to CANDU 6
- Provide a summary assessment findings for key ACR computer codes

Key Computer Codes for ACR Safety Analysis

- **Thermal-Hydraulics**
  - CATHENA
  - NUCIRC
  - MODTURC-CLAS
- **Physics**
  - WIMS
  - RFSP
  - DRAGON
- **Fuel**
  - CATHENA
  - ELOCA
  - ELESTRES
- **Fuel Channel**
  - CATHENA
  - TUBRUPT
- **Fission Product Transport**
  - SOURCE
  - SOPHAEROS
  - SMART
- **Containment**
  - GOTHIC
- **Dose**
  - ADDAM
- **Severe Core Damage Accidents**
  - MAAP-CANDU

ACR Design & Computer Codes – Proven Technology - 1

- **ACR is based on proven CANDU technology with over 30 CANDU units operating or under construction or commissioning**
  - **Over 300 years of CANDU reactor operation**
- **ACR computer codes have proven technology base that is supported by decades of research and development**

ACR Design & Computer Codes – Proven Technology - 2

- The ACR design is fundamentally equivalent to the proven CANDU design in terms of the overall safety system and safety-related system configuration and function
  - Most key design features of ACR are identical to the current fleet of operating CANDUs (e.g., horizontal fuel channels, CANDU fuel bundles, etc.)
  - 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 applicable to the ACR
    - Minor modifications, experimental database extensions, and incremental confirmatory validation, where appropriate

ACR Key Design Changes & ACR codes

ACR Design Feature	Impact
Light-water cooled Reactor Cooling System (RCS)	No impact
Reactor Cooling System pressure at 13 Mpa	Fluid properties extended
CANFLEX 43-element fuel bundle	Codes modified to address fuel bundle design change
SEU fuel with Dysprosium central fuel element	Physics libraries extended
Higher fuel burnup	Code extensions for higher burnup completed
Ticker pressure tube	No impact
Smaller channel lattice pitch	No impact
Oval control rods and SORs	No impact

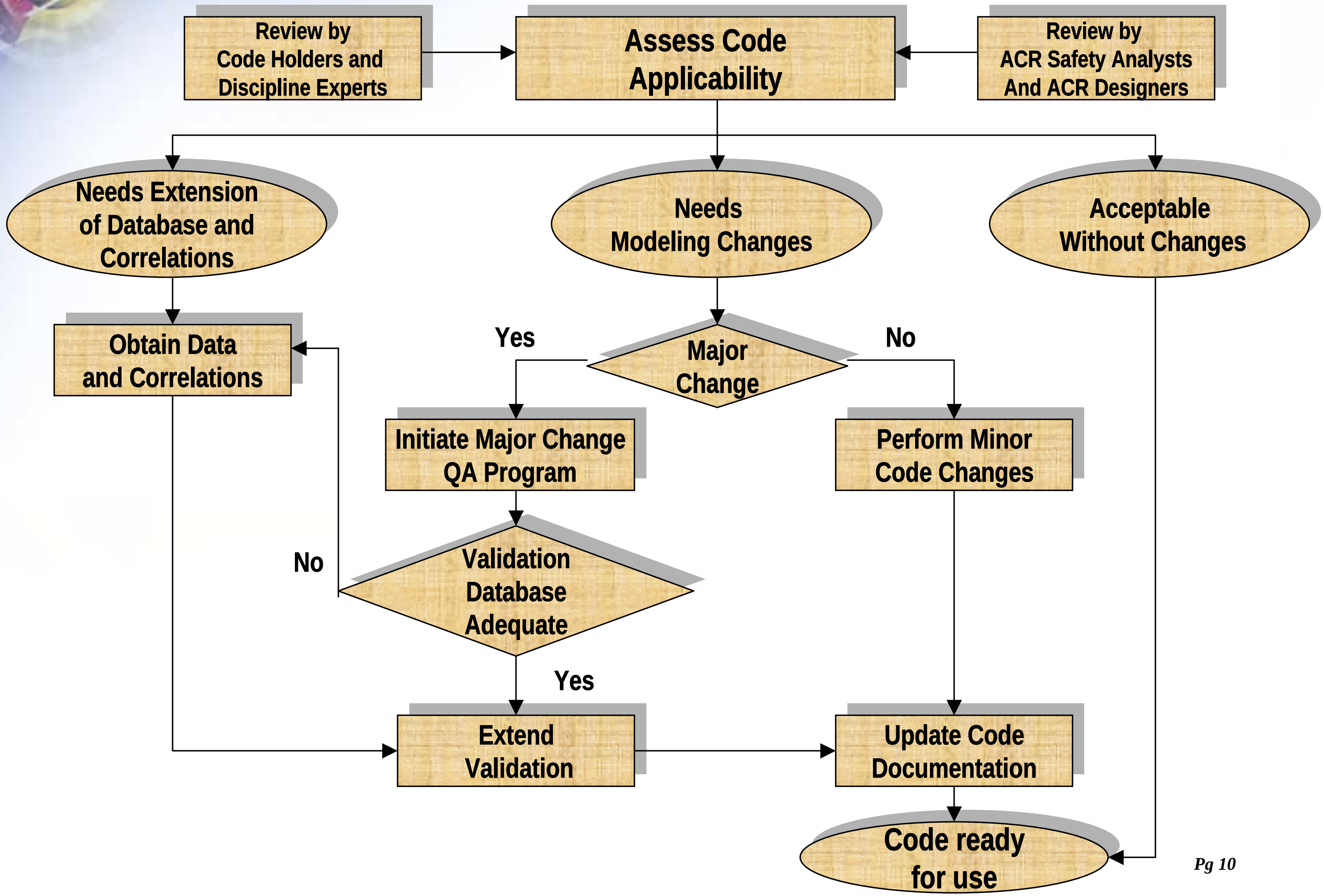
Code Assessment Objectives

- Perform assessment of code modules affected by the ACR design changes
- Perform assessment of the code upgrades to cover specific ACR geometry and conditions
- Perform assessment of potential extensions of the experimental database to cover ACR conditions
- Assess the impact of code changes on the code validation
- Prepare a plan for experimental work to support code changes and incremental validation tasks
- Prepare a plan of code upgrades, verification and incremental validation tasks for each ACR code
- Document the overall applicability and adequacy of computer codes for ACR application

Code Assessment Process - 1

- **Comprehensive and systematic computer code assessment process performed to ensure that AECL codes are applicable to ACR geometry and conditions**
- **Elements of the computer code assessment and upgrade**
  - **Perform codes assessment for ACR applications**
    - **Peer review by code holders and discipline experts**
    - **Peer review by ACR safety analysts, including input from ACR projects designers**
    - **Document the assessment findings**
  - **Determine the depth of code changes**
    - **Minor adjustments and modifications**
      - **Perform code verification**
    - **Significant code changes without experimental database enhancements**
      - **Perform verification and incremental validation**
    - **Significant code changes with experimental database enhancements**
      - **Perform verification and incremental validation**
  - **Update code documentation**

Code Assessment Process - 2



Code Assessment Summary - 1

Code	Application	ACR Assessment
CATHENA	Transient 2-fluid two-phase thermal-hydraulics analysis	Assessment completed. Modifications and incremental validation required
NUCIRC	Steady-state thermal-hydraulics design and analysis	Assessment completed. Modifications and incremental validation required
MODTURC_CLAS	3-D steady-state thermal-hydraulic analysis of the moderator in calandria	Assessment completed. Incremental validation required
TUBRUPT	Thermal-hydraulic transient in moderator with channel rupture	Assessment in progress. Code modifications and validation required
GOTHIC	3-D gas mixing in containment (including combustion)	Assessment completed. Applicable for ACR-700 analysis

Code Assessment Summary - 2

Code	Application	ACR Assessment
WIMS	Lattice cell reactor analysis	Preliminary assessment completed. Physics library update and incremental validation required
RFSP	Full core fuel management analysis	Preliminary assessment completed. Incremental validation required
DRAGON	Lattice cell reactor analysis	Preliminary assessment in progress. Incremental validation required
ADDAM	Atmospheric radionuclide dispersion	Assessment in progress. Validation in progress
MAAP4-CANDU	Severe core damage analysis	Assessment in progress. Validation plan prepared, validation in progress

Code Assessment Summary - 3

Code	Application	ACR Assessment
ELOCA	Fuel behavior under accident conditions	Assessment completed. Modifications and incremental validation required
ELESTRES	Fuel and fission product behavior under operating conditions	Assessment completed. Modifications and incremental validation required
SOURCE	Fission product inventory in fuel	Assessment completed. No ACR-related modifications and validation required
SOPHAEROS	Fission product transport in the reactor coolant system under accident conditions	Assessment completed. Applicable for ACR-700 analysis
SMART	Fission product behavior and transport in containment under accident conditions	Assessment completed. Applicable for ACR-700 analysis

Assessment of CATHENA - 1

- **Brief code description**
  - **Canadian Algorithm for THErmal-hydraulics Network Analysis (CATHENA)**
  - **One-dimensional, two-fluid system thermal-hydraulics code**
  - **Non-equilibrium model (2-velocities, 2-temperatures 2-pressures) plus non-condensables**
  - **Flow regime dependent constitutive relations coupled with the two-phase model**
  - **Semi-implicit finite difference solution method**
  - **Full network, user defined by input file**
  - **Automated time-step control algorithm**
  - **Multiple heat transfer surfaces per thermal-hydraulic node**
  - **Modeling of heat transfer in pin bundles for stratified flow regime**
  - **Default set of heat transfer correlations for entire boiling curve**
  - **Built-in temperature dependent property tables**
  - **Variety of component models available: generalized tank, valves, discharge break, pump, etc.**

Assessment of CATHENA - 2

- Code general application
  - Multi-purpose reactor cooling system thermal-hydraulics and thermo-mechanical analysis
    - Full reactor cooling system network transient analysis
    - Reactor fuel channel transient analysis
    - Secondary side transient analysis
    - Reactor fuel bundle transient analysis
    - Fuel channel thermo-mechanical transient analysis
    - ECCS system operation analysis
    - Auxiliary system thermal-hydraulics transient analysis
  - CATHENA has the capability to be coupled with other codes (LEPCON, RFSP, ELOCA, etc.)

Assessment of CATHENA - 3

- **Code assessment for ACR application**
  - The applicability of the computer code CATHENA MOD-3.5d/Rev 0 reviewed for application to ACR safety and licensing analyses of system thermal-hydraulics transients
  - CATHENA 3.5c Rev 0 document set is adequate for ACR design-assist application of the code
  - Code Abstract, and User’s Manuals for CATHENA MOD-3.5d/Rev 0 completed and in final review
  - Theory Manual Manual for CATHENA MOD-3.5d/Rev 0 under preparation
  - No major code development required beyond implementation of the thermal-hydraulics models for CANFLEX 43-element bundle
  - Incremental validation planned and in progress:
    - Eight system thermal-hydraulic phenomena identified for further incremental validation for ACR conditions
    - Four fuel channel phenomena identified for further incremental validation for ACR conditions

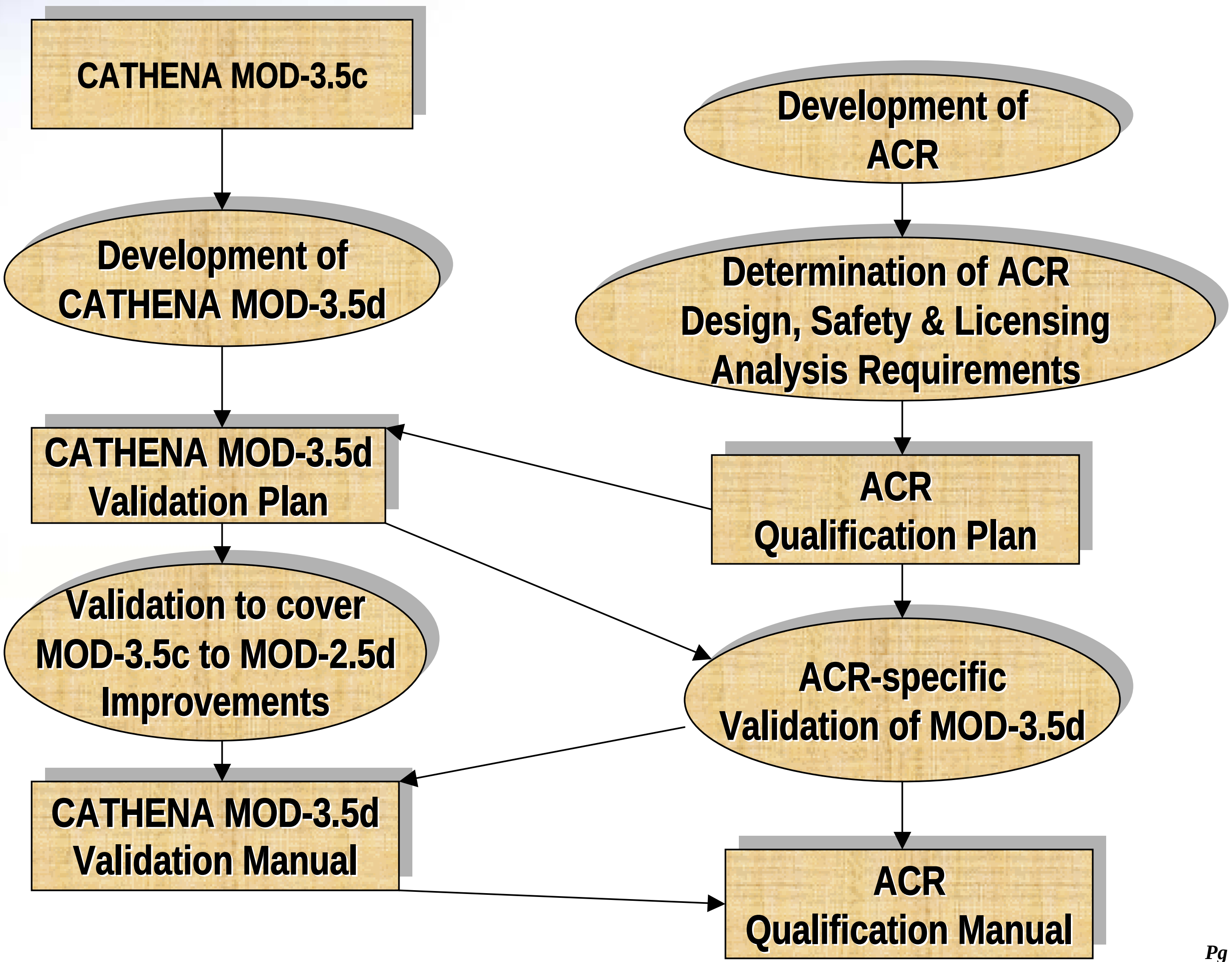
Assessment of CATHENA - 4

- **Incremental validation of thermal-hydraulics phenomena**
  - **TH1 – Break Discharge Characteristics**
  - **TH2 – Coolant Voiding**
  - **TH4 – Level Swell and Void Holdup**
  - **TH7 – Convective Heat Transfer**
  - **TH8 – Nucleate Boiling**
  - **TH10 – Condensation Heat Transfer**
  - **TH11 – Radiation Heat Transfer**
  - **TH18 – Fuel Channel Deformation**

Assessment of CATHENA - 5

- **Incremental validation of thermo-mechanical phenomena**
  - **FC13 – Cladding-to-coolant and Coolant-to-Pressure Tube Heat Transfer**
  - **FC18 – Pressure Tube Deformation Failure**
  - **FC19 – Calandria Tube Deformation Failure**
  - **FC21 – Element-to-Pressure Tube Radiation Heat Transfer**

Assessment of CATHENA - 6



Assessment of NUCIRC - 1

- **Brief code description**
  - **NUclear Heat Transport CIRCuit Analysis (NUCIRC)**
  - **1-D cross-sectional averaged steady-state code**
  - **Full reactor cooling system model (full core including inlet and outlet feeders and channels, headers, pumps, boilers, etc.)**
  - **Predicts pressure, temperature, flow and quality for steady-state conditions of the reactor cooling system**

Assessment of NUCIRC - 2

- **Code general application**
  - Detailed calculations of pressure, channel flow, temperature and void fraction (quality) calculations within the reactor cooling system
  - Critical Channel Power (CCP) calculations at both fuel dryout and fuel melting conditions for any required number of channels
  - Critical Channel Flow (CCF) calculations at fuel dryout for specified input power levels under reduced flow conditions for any required number of channels

Assessment of NUCIRC - 3

- Code general application (cont'd)
  - Feeder pipe sizing calculations which meet established thermal-hydraulic criteria
  - The effect of single- and double-ended fuelling on channel and reactor cooling system operating conditions
  - Provides assessment of plant aging issues
  - Provides boundary conditions and input data for other simulation codes (CATHENA, RFSP,THIRST,etc.)

Assessment of NUCIRC - 4

- Code assessment for ACR application
  - Enhancements to CHF and fuel channel pressure drop models for ACR conditions (up to 20 MPa)
  - Enhancements to Light Water Fluid Properties (up to 20 MPa)
  - Enhancements to property tables (sub-cooled properties)
  - Extension to the validation database for thermal hydraulic correlations required to address ACR coolant conditions and ACR-CANFLEX fuel design
  - ACR-CANFLEX fuel validation
    - Inlet-skewed axial power profile
    - Flattened radial power profile
    - Higher bearing pads to improve CHF margin

Assessment of NUCIRC - 4

Task Description	Status
Implementation of Pressure Drop and CHF models for ACR Applications	Completed August 2002
Implementation of CANFLEX Fuel Bundle CHF Correlations for ACR Applications	Completed March 2003
Calculation of Light Water Fluid Properties	Completed March 2003
Implementation of a Radial Dryout Patch Model	In progress
Implementation of Subcooled Properties for Light Water	In progress
Summary Report of Model Enhancements for the Release of NUCIRC-MOD2.003	In progress
Implementation of CANFLEX CHF Look-up Tables	In progress

Assessment of MODTURC_CLAS - 1

- **Brief code description**
  - MODTURC_CLAS - MODerator TURbulent Circulation Co-Located Advanced Solution (MODTUR_CLAS)
  - Consists of coupling CANDU moderator related specific modules with the general purpose CFD code CFX-TASCflow 2.9, a commercial CFD code owned by AEAT ESL (AEA Technology Engineering Software, Ltd)
  - Porous media approximation - volume-based porosity & distributed hydraulic resistance
  - Boussinesq approximation of buoyancy in momentum equation
  - Two-equation k-epsilon model of turbulence
  - Code capabilities include
    - Calculation of pressure losses in the calandria tube array
    - Modeling of the buoyancy term in the momentum equation
    - Calculation of the volumetric heat load distribution in the calandria vessel from steady-state and transient neutronic power and radioactive decay distributions
    - Calculation of moderator subcooling from temperature distribution
    - Simulation of the moderator temperature control system
    - Modeling of the moderator heat exchangers and associated control valves
    - The setting up of transient boundary conditions (inlet/outlet mass flows, transient poison concentration and other scalar inlet/outlet conditions, and restart capability)

Assessment of MODTURC_CLAS - 2

- **Code general application**
  - Used in nuclear safety analysis to predict velocity, temperature and/or poison concentration distributions (corresponding to SDS2 activation)
  - This information is used
    - to determine moderator subcooling availability for the following postulated accident scenarios:
      - large break LOCA's with and without ECC(involving PT/CT contact heat loads to the moderator)
      - loss of moderator circulation
      - loss of moderator cooling
  - to determine the poison distribution corresponding to
    - in-core, single-channel breaks in an over-poisoned guaranteed shutdown state

Assessment of MODTURC_CLAS - 3

ACR Design Feature	Assessment
Flat plate zone-control, shut-off and control absorber units	Assessment completed. No impact on coding (through input deck)
Potentially different location and geometry of inlets and outlets	Assessment completed No impact on coding (through input deck)
Calandria tube array having smaller pitch and larger calandria tube diameters	Assessment completed No impact on coding (through input deck)
Significant fraction of power generated in the vessel shell	Assessment completed No impact on coding (through input deck)

Assessment of TUBRUPT - 1

- **Brief code description**
  - TUBRUPT is a code used to determine the pressure transients within calandria vessel due to the injection of fuel channel content during in-core break accidents
  - Phenomena modeled
    - Flashing coolant hydrodynamic transient in moderator (FC23)
    - High temperature channel debris interaction with water (FC24)
    - Ruptures channel projectile formation and impact on the calandria vessel, shutoff rods guide tubes, and other fuel channels (FC25)

Assessment of TUBRUPT - 2

- Code general applications
  - Estimate the extent of in-core damage due to a single fuel-channel rupture caused by either of the following scenarios
    - Spontaneous pressure tube / calandria tube rupture
    - Severe flow blockage
    - Feeder stagnation break
  - Code calculates the following parameters
    - Moderator pressure transient
    - Damage mapping of
      - Adjacent channels to the broken fuel channel
      - Shut-off rods guide tubes
      - Calandria vessel

Assessment of TUBRUPT - 3

- **Code assessment for ACR application**
  - Review of code validation results completed
  - A set of simulations selected for assessment of code performance and suitability for ACR applications
  - A review of code models in progress to identify any limitations for ACR applications
  - 14 code adjustments / modifications identified so far (some of these are ACR-related)
  - Code incremental validation is planned to address all outstanding validation tasks and ACR-related tasks
  - Code documentation update planned
    - Most of code documentation will be re-issued to address ACR requirements

Assessment of GOTHIC - 1

- **Brief Code Description and Application**
  - Generation Of Thermal-Hydraulic Information for Containment (GOTHIC)
  - Multi-dimensional thermal-hydraulic code specialized for containment analysis
    - capable of lumped-parameter, 1, 2, 3D modeling of free volumes
    - code allows for hybrid modeling of containment volumes, ie, combinations of lumped parameter, 1D, 2D or 3D volumes
  - Conservation equations are solved for 3 fields
    - Steam / gas mixture
    - continuous liquid
    - liquid droplet
  - Thermal non-equilibrium is allowed between phases and unequal phase velocities
    - Steam / water does not have to be at saturation conditions

Assessment of GOTHIC - 2

- **Brief Code Description and Application (cont'd)**
  - Full treatment of momentum transport in multi-dimensional models
  - Optional models for turbulent shear, mass and energy diffusion
  - Hydrogen combustion
  - Engineering model options include
    - pumps, fans, valves, doors, heat exchangers, fan coolers
    - vacuum breakers, spray nozzles
    - coolers, heaters, volumetric fans
    - hydrogen recombiners and ignitors
    - pressure relief valves
  - Trip logic models and control variables are available
  - Modeling of solid structures (thermal conductors) for flat plate (e.g., walls), cylindrical tube, solid rod

Assessment of GOTHIC - 3

- Code assessment for ACR application
  - Assessment completed
  - GOTHIC is generally applicable to ACR safety analysis
  - Fluid property ranges are sufficient to cover expected ACR containment thermal-hydraulics behavior
  - Built-in equipment models are adequate for ACR application (Version 7 has improvements for Local Air Coolers and Recombiners)
  - GOTHIC is suitable for predicting pressure loads resulting from near-flammability limit burns and accelerated sub-sonic deflagrations
  - GOTHIC not suited for containment loads from detonation burns, or for standing flames

Summary of Physics Codes & Phenomena

Phenomenon	Reactor Physics Phenomenon	Primary Code
PH1	Coolant-Density-Change Induced Reactivity	WIMS
PH2	Coolant-Temperature-Change Induced Reactivity	WIMS
PH3	Moderator-Density-Change Induced Reactivity	WIMS
PH4	Moderator-Temperature-Change Induced Reactivity	WIMS
PH5	Moderator-Poison-Concentration-Change Induced	WIMS
PH6	Moderator-Purity-Change Induced Reactivity	WIMS
PH7	Fuel-Temperature-Change Induced Reactivity	WIMS
PH8	Fuel-Isotopic-Composition-Change Induced Reactivity	WIMS / RFSP
PH9	Refuelling-Induced Reactivity	RFSP
PH10	Fuel-String-Relocation Induced Reactivity (not relevant for ACR)	RFSP
PH11	Device-Movement Induced Reactivity	DRAGON / RFSP
PH12	Prompt/Delayed Neutron Kinetics	WIMS/RFSP
PH13	Flux-Detector Response	RFSP
PH14	Flux and Power Distribution in Space and Time	WIMS / RFSP
PH15	Lattice-Geometry-Distortion Reactivity Effects	WIMS / RFSP
PH16	Coolant-Purity-Change Induced Reactivity (not relevant for ACR)	WIMS
PH17	Core Physics Response to Moderator Level Change	WIMS

Assessment of WIMS - 1

- Brief code description
  - General-purpose lattice-cell code
  - Based on collision-probability method
  - Solves fundamental neutron transport in 2D
  - Multi-group with three external microscopic libraries
  - Fuel burnup and isotopic depletion
  - Resonance treatment being refined
  - Update of nuclear data libraries (completed)
    - ENDF/B-V library for CANDU
    - ENDF/B-VI library for CANDU
    - 2 ENDF/B-VI -based libraries for CANDU
    - 3 ENDF/B-VI -based libraries for ACR
      - 89-group, 120-group and 172-group

Assessment of WIMS - 2

- Code general application
  - Provides cell-averaged nuclear data used in operation, safety assessment, and licensing of CANDU
  - Used to model physics phenomena listed in the previous table

Assessment of WIMS - 3

- Code assessment for ACR application
  - Preliminary assessment completed for all ACR design features
    - WIMS generally applicable to ACR conditions
    - Requires improvements to resonance treatment
  - Requires library update (for dysprosium)
    - Update completed
  - ZED-2 experimental database extension planned (will be used for WIMS validation)
  - WIMS incremental validation scheduled
    - Using extended validation database
    - Using comparison to MCNP

Assessment of RFSP - 1

- **Brief code description**
  - Solves the two-group 3D diffusion equations by a finite difference iterative technique using cell-average fluxes
  - Capable of generating nominal power distributions and simulating reactor operations, including refuelling and burnup steps
  - It allows variable mesh spacing
  - At any boundary, it can use extrapolated boundary conditions
  - It is of modular design
  - Reactivity devices and structural materials are represented by incremental cross sections which are added to the fuel cross section of the affected lattice cells.

Assessment of RFSP - 2

- **Code general application**
  - Coupled with the CATHENA code (thermal-hydraulics feedback)
  - During design stage
    - Used to simulate the initial transient from startup to equilibrium
    - Used to investigate the effect of various fuelling rules
    - Used to obtain accurate estimates of maximum powers, discharge burnups, etc.
  - During reactor operation
    - Used to track the reactor operating history and obtain bundle power, channel power, and bundle irradiation histories (obtain 3-D core power and burnup distributions) to
      - select channels for refuelling
      - ensure that channel and bundle powers are kept within specified limits
      - evaluate burnup

Assessment of RFSP - 3

- Code assessment for ACR application
  - Preliminary assessment completed for ACR design features
    - RFSP generally applicable to ACR conditions
  - ZED-2 experimental database extension planned (will be used for RFSP validation)
  - RFSP incremental validation scheduled
    - Using extended validation database
    - Using comparison to MCNP (full or partial core model)
    - Commissioning physics test data

Assessment of DRAGON

- Brief code description
  - Multi-group collision-probability code
- Code general application
  - 3D capability to generate incremental cross sections of reactivity-control devices in CANDU reactors
- Code assessment for ACR application
  - Preliminary assessment in progress
  - No ACR-related changes identified
  - DRAGON validation
    - ZED-2 measurement data to be used
    - Data from other fuel studies
    - Data from other criticality facilities germane to ACR
    - Compare WIMS/DRAGON/RFSP full-core calculations with the MCNP calculations

Assessment of ELOCA - 1

- **Brief code description**
  - ELOCA models fuel elements under the rapidly changing coolant and power conditions typical of an accident
  - ELOCA calculates
    - Fuel temperatures
    - Cladding temperatures
    - Internal gas pressure
    - Cladding strain
    - Cladding failure
    - Cladding oxidation

Assessment of ELOCA - 2

- **Phenomena modeled**
  - Fission and Decay Heating — FC1
  - Diffusion of Heat in Fuel — FC2
  - Fuel-to-cladding Heat Transfer — FC3
  - Fuel-to-End Cap Heat Transfer — FC4
  - Fission Gas Release to Gap and Pressurization — FC5

    (currently steady-state gas release model)
  - Cladding Deformation — FC6
  - Cladding Failure — FC7
  - Fuel Deformation — FC8
  - Cladding Oxidation or Hydriding — FC9
  - Fuel or Cladding Melting and Relocation — FC11

    (relocation not currently modeled)

Assessment of ELOCA - 3

- **Code general application**
  - ELOCA models the thermo-mechanical transient behavior of a CANDU fuel pin during postulated accidents
  - The “steady state” behavior of the fuel pin under normal operating conditions is modelled by ELESTRES and used in ELOCA
  - ELOCA analysis is coupled to CATHENA analysis to provide
    - Thermal-hydraulics boundary conditions from CATHENA to ELOCA
    - Thermal-mechanical behavior of fuel pin from ELOCA to CATHENA

Assessment of ELOCA - 4

- **Code assessment for ACR application**
  - ELOCA assessment for application to ACR completed
  - Three areas identified that require code update
    - 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
    - Extension of the database on fuel cladding strain
  - Laboratory test program for extrapolation of fuel thermal properties in progress to address the above issues (Dysprosium doped fuel)
  - Test program scheduled to extend the database on cladding strain to include the ACR cladding design (diameter and thickness)

Assessment of ELESTRES - 1

- Brief code description
  - ELESTRES computer code models the thermal and mechanical behavior of an individual fuel element during its irradiation life under normal operating conditions
  - ELESTRES is composed of two models
    - one-dimensional fuel performance models
      - thermal model for temperature calculations
      - microstructural model for fission gas and associated calculations
    - two-dimensional stress analysis model
      - used to calculate axisymmetric deformations of the fuel pellet and cladding

Assessment of ELESTRES - 2

- **Code general application**
  - The main ELESTRES calculated parameters
    - fuel temperature, fission gas release, internal pressure, and fuel deformation
  - Additional ELESTRES calculated parameters
    - heat generation and diffusion across the pellet
    - the effect of burnup and enrichment on the radial distribution of neutron flux
    - thermal conductivity as a function of temperature, burnup, porosity/density
    - heat transfer coefficients between the pellet and the cladding
    - microstructural change in the pellet (equiaxed and columnar grain growth)
    - fission product release
    - pellet densification and swelling
    - pellet thermal, elastic, plastic and creep strains in the pellet, and pellet cracking
    - cladding thermal, elastic, plastic and creep deformation
  - Used to generate input data or initial conditions for the ELOCA transient fuel behavior code

Assessment of ELESTRES - 3

- Code assessment for ACR application
  - Assessment completed for ACR applicability
  - A number of improvements were identified
    - Imposed by ACR design features / changes
    - Supported by experience from the application of ELESTRES to SEU fuel designs in the early 1990s
- Code improvements
  - Flux depression, cladding material properties, Dy properties, cladding oxidation, fission gas diffusivity, etc.
- Code validation in progress
  - Verification of new code version in progress
  - Validation will be conducted against existing database

Assessment of SOURCE - 1

- **Brief code description**
  - Developed to calculate fission-product releases from CANDU fuel under CANDU reactor accident conditions
  - Phenomena modeled

Diffusion	FPR2
Grain Boundary Sweeping / Grain Growth	FPR3
Grain Boundary Coalescence / Tunnel Inter-linkage	FPR4
Vapor transport / Columnar Grains	FPR5
Fuel Cracking (Thermal)	FPR6
Gap Transport (Failed Elements)	FPR7
Oxide Formation	FPR9, FPR10, FPR11
Fuel-Zircaloy Interaction	FPR12
Fuel Dissolution by Molten Zircaloy	FPR13
Fuel Melting	FPR14
Fission Product Vaporization	FPR15
Matrix Stripping	FPR16
Temperature Transients	FPR17
Grain Boundary Separation	FPR18
Fission Product Leaching	FPR19

Assessment of SOURCE - 2

Range of Application

Parameter	Range of Application
Fuel Temperature	273 K < T < 3812 K (from just below the freezing point of light water to the boiling point of urania)
Cladding Temperature	273 K < T < 3000 K
Fuel Stoichiometry ($UO_{2\pm X}$)	$1.8 \leq 2\pm X \leq 2.667$ ($UO_{1.8}$ to U_3O_8)
Cladding Condition	Intact, failed during normal operation, or failed during transient (failure time is an input to the code)
Fuel Environment	Steam, hydrogen and inert gases (air)
Burnup	0 < Burnup < 33.3 MWd/kgU
Nuclides Modeled	Up to 150 in the current implementation

Assessment of SOURCE - 3

- Code assessment for ACR application
  - Preliminary assessment completed
  - Fission-product release behavior in ACR fuel under accident conditions is similar to the behavior of current natural uranium fuel in current CANDU reactors
  - SOURCE code is generally applicable to ACR conditions

Assessment of SOPHAEROS - 1

- **Brief code description**
  - SOPHAEROS is used to calculate reactor cooling system fission-product transport and retention under accident conditions
  - The code has been obtained from IRSN in France, and enhanced to address CANDU-specific design features
  - The code employs a pseudo-kinetic thermochemical-equilibrium numerical solution scheme using deviation from the equilibrium state as driving force to calculate fission-product speciation and volatility

Assessment of SOPHAEROS - 2

- Code general applications
  - The code includes the following mechanistic models
    - Fission-product vapor transport
    - Condensation, vaporization and chemisorption
    - Aerosol nucleation, growth, agglomeration, deposition and re-suspension
    - Chemical speciation in reactor cooling system piping and components
  - Calculated parameters / phenomena
    - Time dependence of fission-product releases into containment
    - Deposition of fission products in reactor cooling system components

Assessment of SOPHAEROS - 3

- Code assessment for ACR application
  - Assessment completed
  - The code is generally applicable to ACR safety analysis
  - Primary fission-product phenomena relevant to ACR safety analysis are modeled with the code
  - Pool-scrubbing models need to be further assessed

Assessment of SMART - 1

- Brief code description
  - Simple Model for Activity Removal and Transport (SMART)
  - Calculates radionuclide behavior in containment
  - The code is composed of a set of one-dimensional, partial differential equations that describe the aerosol and fission product behavior
  - An aerosol general dynamics equation is solved to calculate aerosol size distribution as a function of space and time
  - Mass conservation equations are solved to predict fission product concentrations in containment
  - Various sink / source terms are described, such as break, decay activity, discharges through escape paths, mass transfer in the containment free volume, etc.

Assessment of SMART - 2

- **Code general application**
  - **Calculates aerosol-fission product behavior within containment under accident conditions and releases to the outside atmosphere**
  - **Provides details about the concentration of individual isotopes present in various parts of the containment**
  - **Provides input to the ADDAM code for calculating dose to public and station staff**

Assessment of SMART - 3

- **Code assessment for ACR application**
  - **Assessment completed**
  - **SMART is generally applicable to ACR safety analysis**
  - **Enhancements of the IMOD 2 code, that calculates in detail iodine behavior in containment, provided improvement of code applicability**
  - **The current code capability includes modeling of Emergency Filtered Air Discharge System for use in multi-unit CANDU stations**

Assessment of ADDAM - 1

- **Brief Code Description and Application**
  - <u>A</u>tmospheric <u>D</u>ispersion and <u>D</u>ose <u>A</u>nalysis <u>M</u>ethod (ADDAM)
  - Used for analysis of hypothetical accident releases of radioactive material to the atmosphere from CANDU stations
  - Conceptual model is based on modeling 15 atmospheric dispersion phenomena (covered in the ADDAM validation matrix)
  - ADDAM calculates concentrations of radioactivity in the air and on the ground and doses to members of the public following an atmospheric release

Assessment of ADDAM - 2

- Code assessment for ACR application
  - Assessment in progress
  - ADDAM is expected to be applicable to ACR safety analysis
  - Validation completed
    - Validation documentation being finalized
    - Validation manual completed

Assessment of MAAP_CANDU - 1

- **Brief code description**
  - Modular Accident Analysis Program (MAAP)
  - MAAP is a family of integrated computer codes designated for Severe Accident Analysis in nuclear plants, used by more than 40 international utilities
  - MAAP_CANDU was developed by Fauske & Associates (FAI)
    - Based on MAAP, widely used by PWR / BWR utilities
    - Enhanced with Ontario Power Generation core heating module intended for PRA Level 2 analysis
  - MAAP_CANDU has models for horizontal CANDU-type fuel channels and CANDU-specific systems, such as Calandria Vessel, Reactor Vault, Reactor Cooling System, Containment systems (dousing), etc.

Assessment of MAAP_CANDU - 2

- **Physical processes modeled in MAPP_CANDU**
  - Thermal-hydraulics processes in reactor cooling system, calandria vessel, reactor vault and shield tank, end-shield, and containment components
  - Core heat-up, melting and disassembly
  - Zr oxidation by steam and hydrogen generation
  - Material creep and possible rupture of reactor cooling system components, calandria vessel ad shield tank walls
  - Ignition of combustible gases
  - Energetic corium-coolant interactions
  - Molten corium-concrete interactions
  - Fission product release, transport and deposition

Assessment of MAAP_CANDU - 3

- Code general application
  - MAAP_CANDU calculates severe accident progression starting from normal operating conditions for a set of plant system faults and initiating events leading to:
    - Reactor cooling system inventory blow-down or/and boil-off
    - Core heat-up and melting
    - Fuel channel failure
    - Core disassembly
    - Calandria vessel failure
    - Shield tank / reactor vault failure
    - Containment failure

Assessment of MAAP_CANDU - 4

- **Code assessment for ACR application**
  - **Assessment in progress**
  - **MAAP is expected to be generally applicable to ACR**
  - **Validation for generic CANDU application in progress**
    - **CANDU-6 plant analyses completed**
    - **There are no known "exact results" available for integrated codes (no "TMI" for CANDU)**
    - **Several systems in MAAP_CANDU are hard-coded and cannot be validated individually**
    - **System response compared with simplified analytical solutions for selected accident sequences**
    - **Validation of separate models against other validated codes performed where possible (GOTHIC)**
    - **MAAP_CANDU validated by FAI using Separate Effect Experiments, Integral Experiments, Industry Experience and detailed analysis for a large number of physical processes**

Summary

- AECL computer codes are based on proven technology and are supported by decades of R&D
- AECL codes assessment is comprehensive and systematic to ensure that all codes are applicable to ACR-700
- Most AECL codes require only minor adjustments to match ACR geometry or conditions
- For codes that will require significant modifications, extension of the experimental database will be performed as applicable, along with code incremental validation
- AECL code documentation is exhaustive and consistent with AECL QA requirements
- Code documentation updates for ACR applications will be conducted as applicable

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