ACR
Safety Analysis Methodology and Computer Codes

Overview

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AECL
TECHNOLOGIES INC.

Meeting Agenda

- **Day 1**
  - Introduction – NRC – 15 min
  - Introduction – V. Langman – 15 min
  - Overview –
    N. Popov – 60 min
  - Break – 15 min
  - ACR Sequence of Key Events –
    Z. Bilanovic – 90 min
  - Public comment – 15 min
  - Lunch – 60 min
  - Safety Analysis Methodology –
    Z. Bilanovic – 60 min
  - Break – 15 min
  - Overview of ACR Computer Codes –
    Z. Bilanovic – 90 min
  - Public comment – 15 min
- **Day 2**
  - Technical Basis Document and Validation Matrices for ACR Application –
    D. Wright – 90 min
  - Break – 15 min
  - Assessment of AECL Computer Code for ACR Application –
    N. Popov – 90 min
  - Public comment - 15 mm
  - Lunch – 60 min
  - QA of ACR Computer Codes –
    D. Richards – 60 min
  - Break – 15 min
  - Validation of ACR Computer Codes –
    D. Wren – 60 min
  - Public comment – 15 min
  - Concluding Remarks – 15 min

Outline

- Objectives
- ACR Sequence of Key Events
- Safety Analysis Methodology
- Overview of ACR Computer Codes
- Technical Basis Document and Validation Matrices for ACR Application
- Assessment of AECL Computer Code for ACR Application
- QA of ACR Computer Codes
- Validation of ACR Computer Codes

Presentation Objectives - 1

- Provide an overview of the ACR Safety Analysis Methodology and ACR Computer Codes
  - Part of a series of ACR familiarization sessions to US NRC, each aimed at specific topic covering the ACR design, safety analysis and R&D support
  - This session is focused on the safety analysis methodology and tools for design basis events

Presentation Objectives - 2

- Define ACR accident scenarios and provide an introduction of the sequence of key events for key accident scenarios
- Provide an overview of the ACR key computer codes and the linkage of these codes in specific ACR safety analysis
- Provide outline of the key principles used in the ACR safety analysis approach, bounding assumptions, and acceptance criteria (figure of merit)
- Provide the ACR approach in identifying and ranking the key phenomena that occur and govern design basis events, and provide documentation outline

Presentation Objectives - 3

- Provide an overview of the assessment of applicability of the AECL computer codes for ACR application
  - Assessment with respect to the ACR design and conditions (incremental to CANDU 6 applications)
- Provide an overview of the AECL QA process with respect to computer code development, validation, configuration management and use in ACR applications
- Provide an overview of the incremental and confirmatory validation for ACR
  - Experimental database extensions

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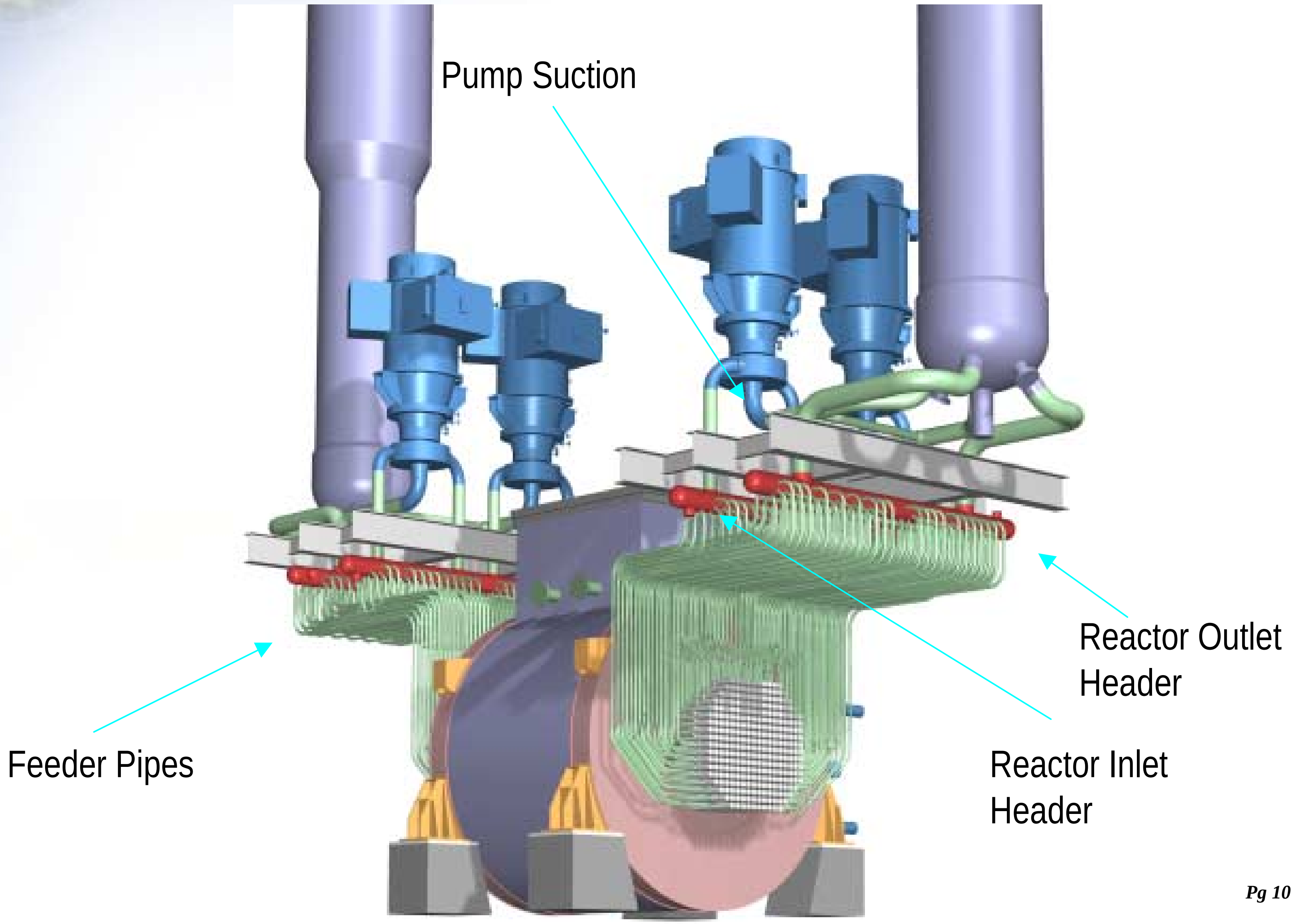
Examples of Design Basis Events

- Class 1: Events of Moderate Frequency
  – Total Loss of Class IV Power (Station normal AC power supply)
- Class 2: Infrequent Events
  – Small LOCA
  – End Fitting Failure
- Class 3: Limiting Events
  – Pressure Tube/Calandria Tube Rupture
  – Large LOCA
  – Main Steam Line Break (Inside Containment)

Total Loss Of Class IV Power

Event and System Response	Time (s)
RCS pumps begin rundown Feedwater pumps begin rundown Steam flow to turbine ramped down Loss of feed pumps to RCS Pressurizer heaters fail off	0.0
RRS stepback on RCS high pressure (not credited)	~3
SDS1 RCS high pressure trip SDS1 low flow trip	~3.5
SDS2 RCS high pressure trip	~5.5

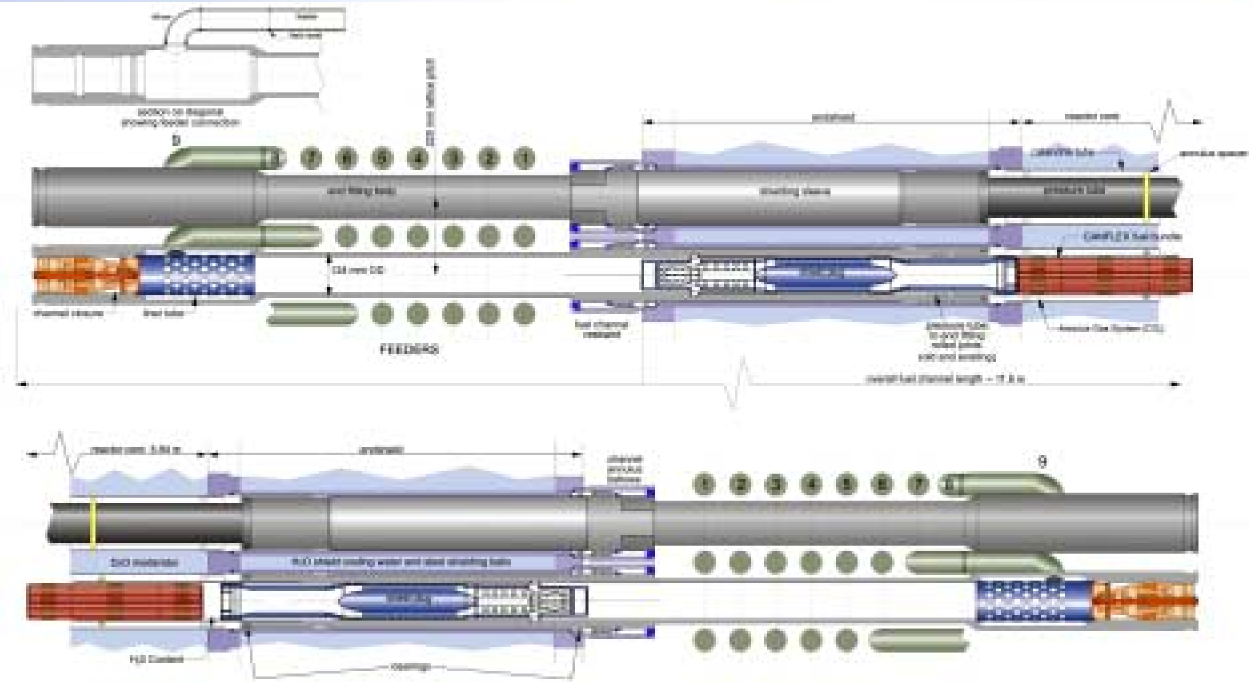
Reactor Cooling System



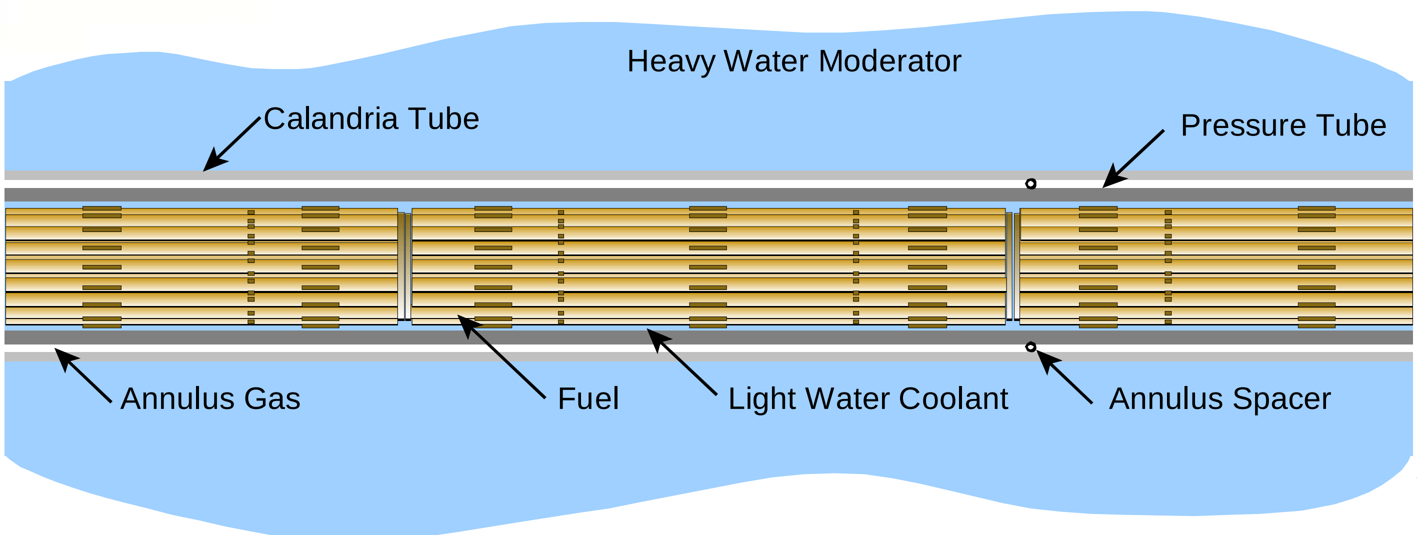
Small Break LOCA

Event and System Response	Time (s)
Break occurs	0.0
High Reactor Building pressure signal ECC Conditioning signal	<45
SDS1 and SDS2 Low RCS flow trip initiated	~45
LOCA signal	~60
Crash cool (automatic depressurization) initiated	~90
ECC rupture discs open	~150
ECC injection stops and long term cooling valved in	~630

ACR Fuel Channel



ACR - Fuel Channel



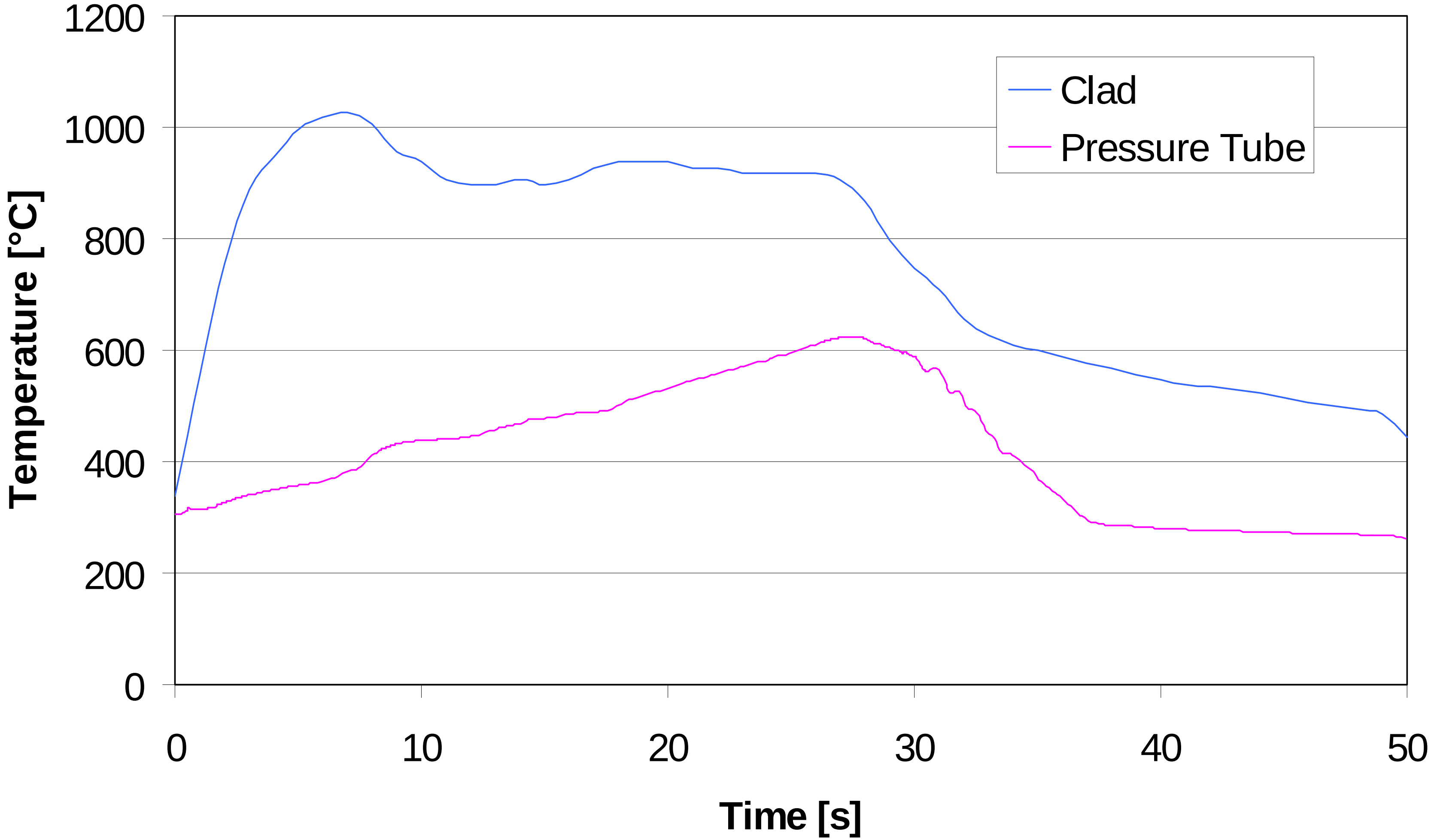
In-Core Breaks

Event and System Response	Time (s)
Break occurs	0.0
Reactor trip	~200
LOCA signal	~230
Crash cool (automatic depressurization) initiated	~260
ECC Initiation	~330

Large Break LOCA - 1

Event and System Response	Time (s)
Break occurs	0.0
First reactor trip signal Second reactor trip signal	<0.5 <2.0
LOCA signal	<3.0
Crash cool (automatic depressurization) initiated	~35
ECC Initiation	~40
LTC starts	~200

Large Break LOCA - 2



Typical cladding and pressure tube temperature profile during large stagnation break LOCA

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Safety Analysis Methodology - 1

- **Analysis Approach**
  - **Limit of the Operating Envelope (LOE):**

- **The Limit of the Operating Envelope (LOE) is the basis for ACR safety analyses for design basis events**
- **This approach requires that initial and boundary conditions be set to simultaneously conservative, or pessimistic values, taking into account the objective of the analysis**
  - **A particular bounding assumption is linked to a specific safety analysis objective**

Safety Analysis Methodology - 2

- Bounding assumptions include:
  - Initial and boundary conditions (the plant state parameters)
  - Key modeling parameters
- Key modeling parameter selection is based upon:
  - Sensitivity analyses during validation
  - Previous experience
  - Limited sensitivity analyses to ensure the dominant model uncertainties are accounted for

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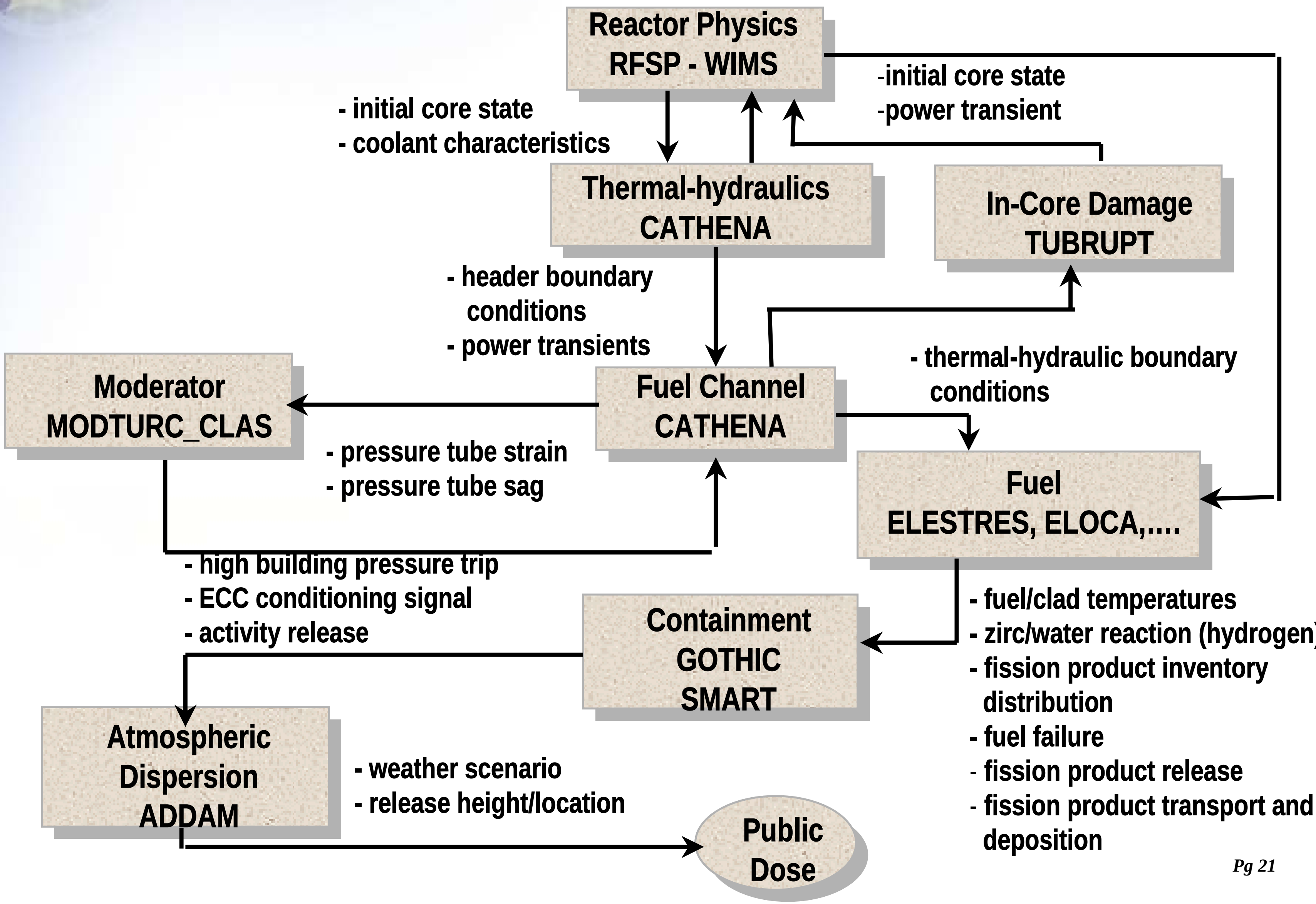
Key Computer Codes for ACR Safety Analysis

- 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

Interface of ACR Computer Codes

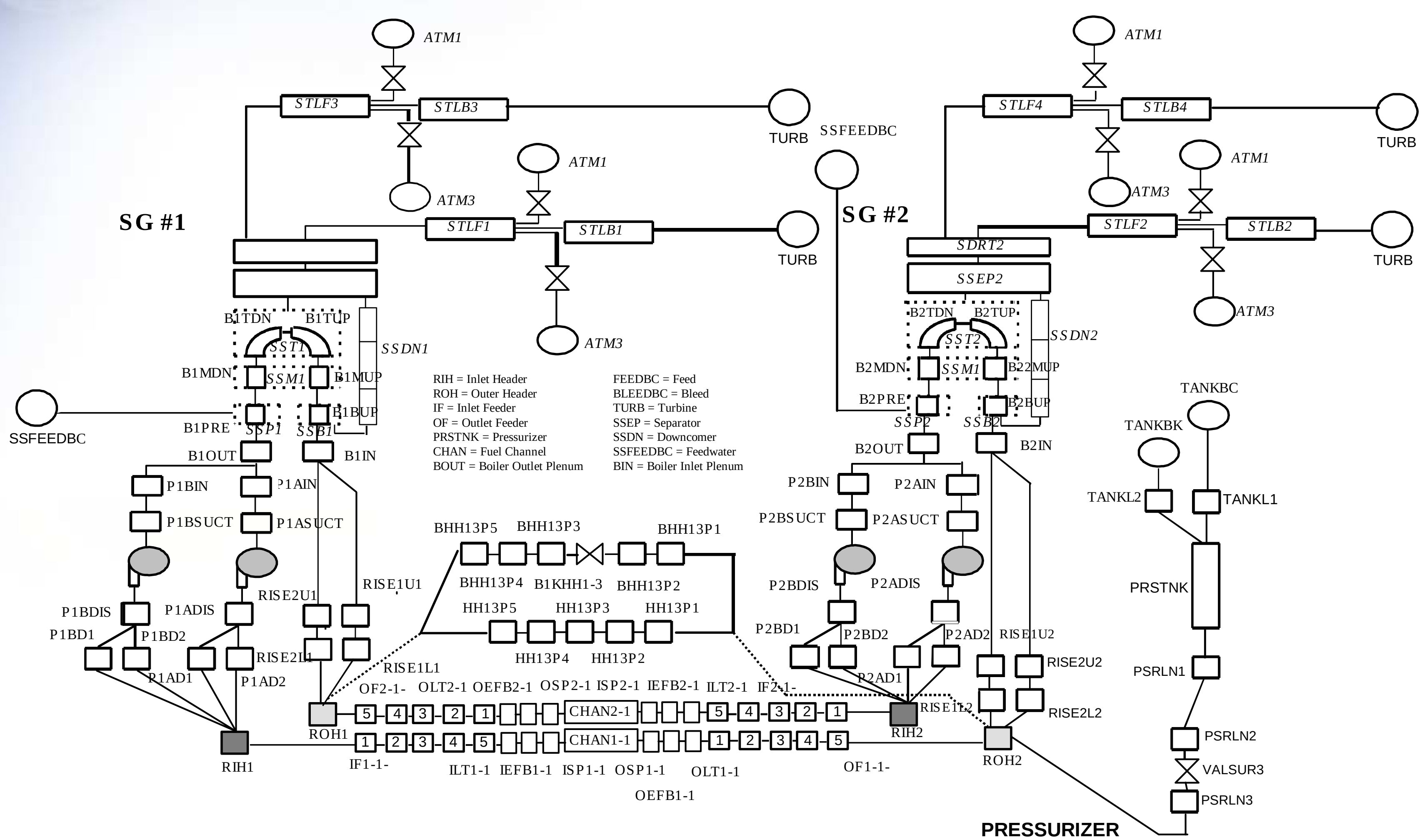


Thermal-Hydraulic Analysis Tools - 1

CATHENA

- Non-equilibrium two fluid system thermal-hydraulics code
- Full network defined by user in the input files
- Has D_2O and H_2O properties
- Generalized heat transfer package:
  - Multiple surfaces per thermal-hydraulic node
  - Models heat transfer in and between fuel pins
  - Has built-in temperature dependent properties (has options for user input properties)

Thermal-Hydraulic Analysis Tools - 2



CATHENA Nodalization diagram of the ACR RCS circuit

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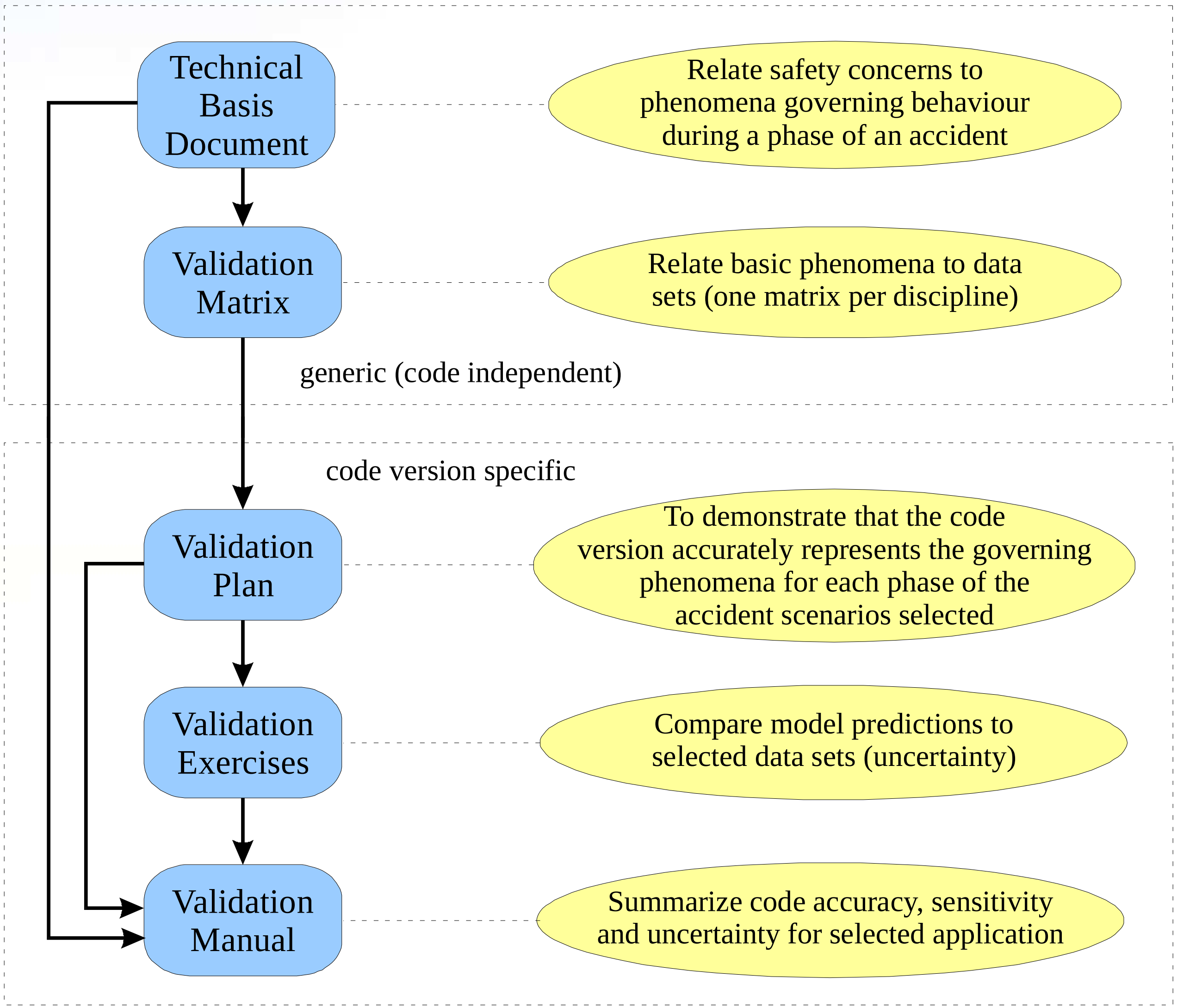
TBD and VMs Overview - 1



TBD and VMs Overview - 2

- Technical Basis Document (TBD) and Validation Matrix (VM) documents are the top-level documents in the code validation process
  - TBD and VMs applicable for operating CANDUs are in use by the Canadian nuclear industry
  - AECL has prepared an ACR-specific TBD, and will prepare ACR-specific VMs over the next six months
- TBD is structured event by event
- A separate section describes each of the accident scenarios
- TBD identifies and ranks the phenomena which play a role in each scenario

TBD and VMs Overview - 3

- Eight VMs, one for each safety analysis discipline:
  - Reactor physics
  - System thermal-hydraulics
  - Fuel & fuel channel thermomechanical
  - Moderator and shield system
  - Fission product release & transport
  - Containment
  - Radiation physics
  - Atmospheric dispersion
- VMs provide a synopsis of each phenomenon
- VMs identify and describe sources of data which can be used to validate the modeling of each phenomenon

TBD Specifics - 1

- ACR TBD is an evolution of the current CANDU TBD
- ACR TBD reflects the ACR design: accident scenarios and phenomena rankings
- No new ACR-specific phenomena have been identified
- The accident scenarios described in the TBD encompass the individual accident sequences in the particular group of events
- Individual accident sequences are identified and discussed
- Each TBD section describes:
  - the safety concerns for the given accident scenario
  - the relevant system behavior
  - the role of the primary physical phenomena which govern the system behavior

TBD Specifics – 2

- **Phenomena designation based on discipline:**
  - **PH: reactor physics**
  - **TH: system thermal-hydraulics**
  - **FC: fuel & fuel channel**
  - **MH: moderator and shield system**
  - **FPR/FPT: fission product release/transport**
  - **C: containment**
  - **RAD: radiation physics**
  - **AD: atmospheric dispersion**

TBD Specifics - 3

- Phenomena identification and ranking process:
  - Team of experts for each discipline
    - Analysts, code developers, code validation analysts, reactor designers
  - Review of safety analysis results, code models
  - Identification of safety concerns
  - Description of system behavior
  - Identification of phenomena
  - Ranking of phenomena based on importance for system behavior and safety concerns
  - Ranking is done conservatively: if any doubt, select the higher ranking
    - Particular attention focused to phenomena for which the impact is not fully understood, or the knowledge base is not fully developed

VM Specifics

- Like the TBD, ACR Validation Matrices (VMs) are an evolution of the current CANDU VMs
  - Eight generic VMs developed by teams of experts from AECL and the Canadian nuclear industry partners
- ACR VMs reflect the ACR design: accident scenarios, phenomena rankings and sources of experimental and other data for validation
- No new ACR-specific phenomena have been identified
- VM main sections:
  - Phenomena rankings, by accident scenario (similar to TBD)
  - Phenomena synopses
  - Sources of Data for Validation

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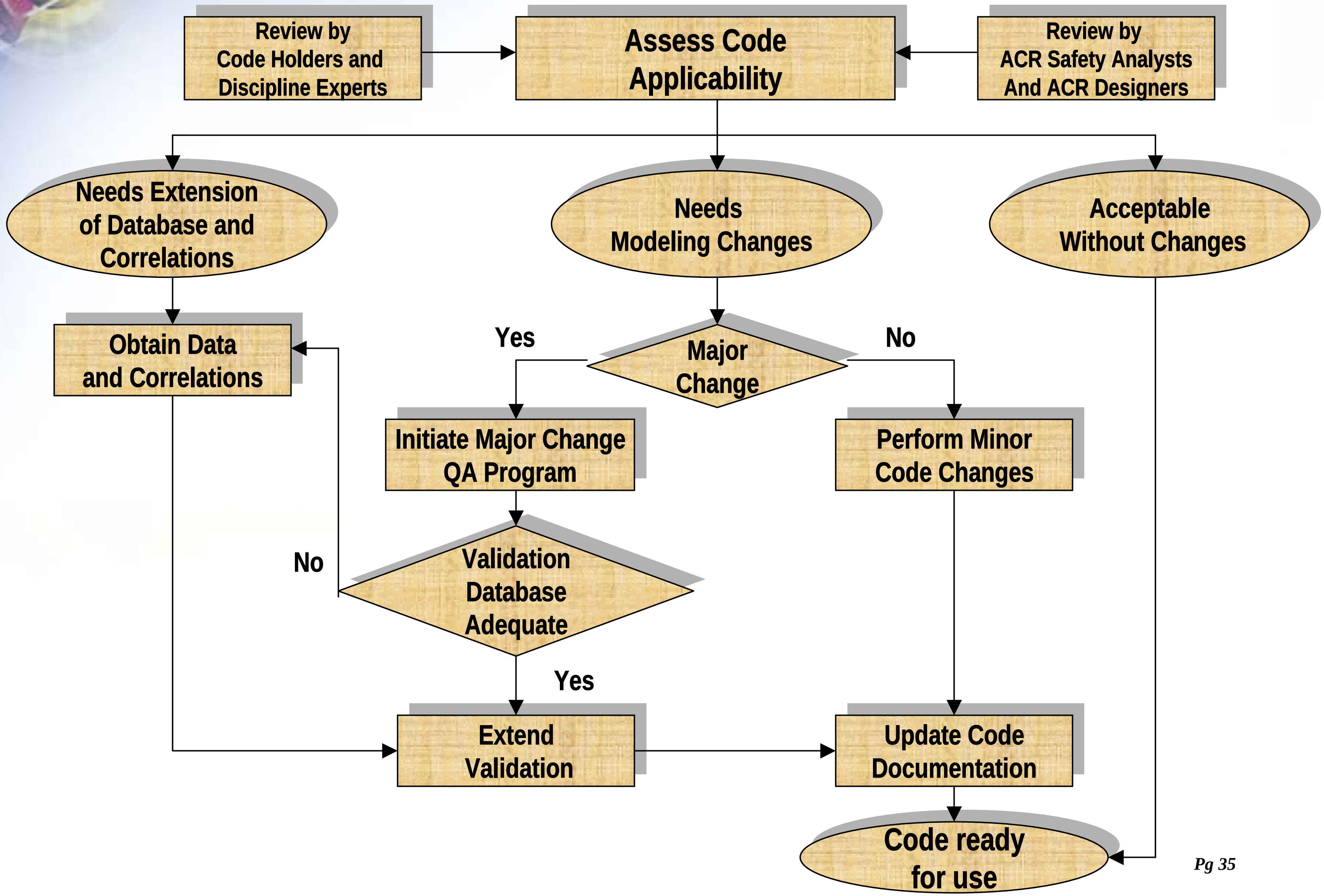
Code Assessment Objectives

- 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**
- **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**
  - **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

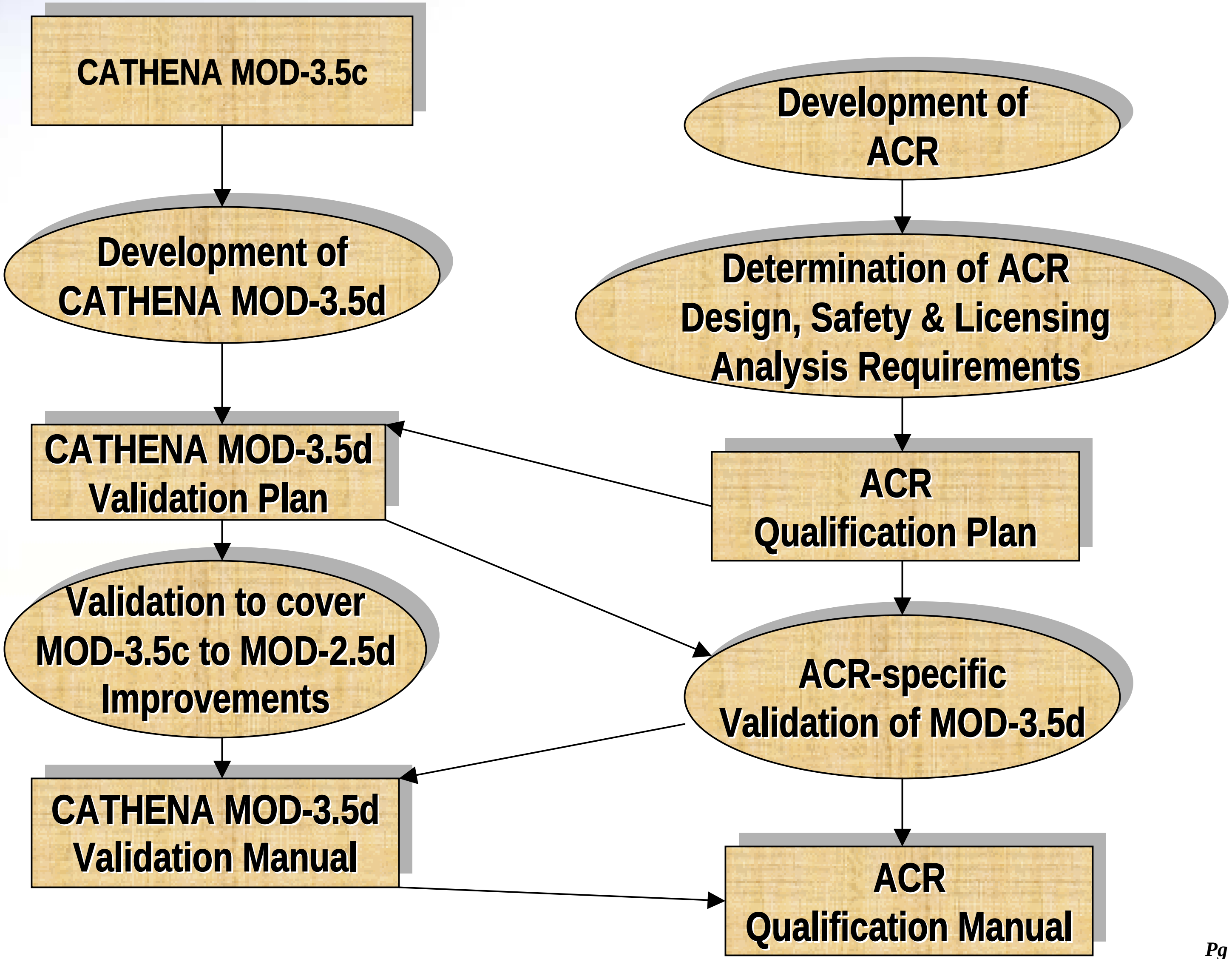
Code Assessment Summary - 2

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



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- **QA of ACR Computer Codes**
- Validation of ACR Computer Codes

Background

- 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 2001 March to address comments internal to AECL as well as external (Canadian Nuclear Safety Commission (CNSC))

Elements of AECL's SQA Program

- Responsibilities
- Requirements for Computer Programs
- Computer Program Design and Development
- Acquisition of Analytical, Scientific and Design Computer Programs
- Configuration Management
- Change Control
- Validation
- Use of Computer Programs
- Documentation
- Verification Processes

Responsibilities

- **AECL is responsible to ensure SQA activities are performed in accordance with the N286.7-99 Standard**
  - **This is accomplished through adherence to the AECL Manual and specific AECL Procedures**
  - **Verification of compliance is established through audits (Internal as well as third-party External Audits (e.g, CNSC, Clients))**
  - **Activities are monitored by the office of AECL's Chief Quality Officer (Dr. A.M.M. Aly) who reports directly to the President and CEO of AECL (Mr. Robert Van Adel)**

“Legacy” Computer Programs

- Are those developed prior to 1999 (prior to promulgation of the Standard, N286.7-99)
  - Qualification Plan must be generated
  - Qualification Report is generated that verifies the Qualification Plan has been completed
  - All further development must conform to the Software Development Cycle:
    - Problem Definition
    - Development Plan
    - Theoretical Background
    - Requirements Specifications
    - Design
    - Coding

Configuration Management

- Configuration components include:
  - source code
  - operating system, compiler, library functions, object modules, executable code, and instructions used with the compiler and linker
  - computer program documents
- Each configuration is uniquely identified
- Any change to one or more components constitutes a new configuration

Use of Computer Programs

- **AECL ensures the proper use of Computer Programs by ensuring that:**
  - Computer programs are validated for the intended use
  - Only those physical states are analyzed that are within the documented range of the computer program's applicability
  - The input data is verified to ensure that it adequately represents the physical system or process analyzed
  - The derivations and sources of input data are documented in a form that facilitates independent review
  - The configuration of the computer program and the input data are identified so that results can be reproduced
  - The results produced by the computer program are reviewed to confirm that they are reasonable
  - User qualifications are specified and the necessary training is provided to minimize the effect of user dependency

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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 (with limited modifications to the IST codes required to address ACR analysis requirements)
- In requisite areas the validation base of the codes will be extended to cover a new range of application for the ACR
- Overall validation requirements established by Technical Basis Document (TBD) and Validation Matrices (VM)
- R&D program provides extension to the database, if required, for ACR code application

CATHENA Validation

- Validation Requirements Established by Technical Basis Document
- Validation Matrix Documents being updated
  - Data needed to extend the range of applicability to ACR conditions
- 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 prepared to address these gaps, including, where necessary, requirements for new experimental data

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

Fuel Toolset Validation

- ELESTRES
  - 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
- ELOCA
  - 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)

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)

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

Summary

- ACR is based on proven CANDU technology with over 30 CANDU units operating or under construction and commissioning
  - Over 300 years of CANDU reactor operation
- ACR computer codes have proven technology base that is supported by decades of R&D
- The ACR design is fundamentally equivalent to the proven CANDU design (overall safety system and safety-related system configuration and function)
  - Most key design features of ACR are identical to the currently operating CANDUs
  - The key phenomena associated with safety analyses are common with the current CANDUs
  - The analysis tools and methodologies used for safety analysis of current CANDU plants are applicable to ACR-700
    - Minor modifications, experimental database extensions, and incremental confirmatory validation, where appropriate

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