AECL EACL

Safety Analysis Computer Code Qualification: Status and Plan

ACR

108-03510-225-001
Revision 0

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1. INTRODUCTION

AECL is developing an advanced reactor design, designated the Advanced CANDU Reactor™ (ACR™)* based on an evolutionary extension of the existing CANDU 6 design. AECL will use a limited number of analytic, scientific and design computer codes (or tools) to develop the ACR design, to perform safety analyses of postulated accident events and to assess the design against specific safety acceptance criteria. The use of these computer codes will follow the quality assurance requirements set out in the Quality Assurance Manual for the ACR project and associated procedures [1]. A combination of computer codes developed for the design and analysis of the CANDU®** reactor and third-party software tools, with more generic application, will be used to support the ACR development program.

The codes used in the ACR design and safety analysis will be qualified for their application. In most cases the codes are already qualified for use in ACR applications. In some cases, the existing analytical codes may need incremental validation for new applications that are specific to the ACR configuration or design parameters. In a very few instances, modifications may be required for some codes to address new features of the ACR design. Any modification to an existing computer code or extension to the validation basis will meet the requirements of AECL's Software Quality Assurance Program as set out in the Quality Assurance Manual for Analytic, Scientific and Design Computer Programs and associated procedures [2].

This document describes the qualification status and plans for the primary safety analysis computer codes for use during the Basic Engineering Phase of the ACR-700 development project. The qualification will be conducted consistent with AECL's quality assurance requirements. The highest priority and the highest level of qualification are assigned to those codes whose use would have significant safety implications or which would entail the greatest project risk.

This document does not address the status of secondary computer analysis codes, or those codes that may be used for the following purposes:

- interpretation of R&D program experimental results,
- informal design concept assessment (where such assessments will be superseded by formal assessment using the codes addressed in this document),
- environmental impact analyses (for reactor operation or waste management),
- real-time plant information management and control.

The detailed qualification plans for each computer code are beyond the scope of this document and are contained in code-specific documents.

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* ACR™ (Advanced CANDU Reactor™) is a trademark of Atomic Energy of Canada Limited (AECL).

** CANDU® (CANada Deuterium Uranium) is a registered trademark of Atomic Energy of Canada Limited (AECL).

1.1 AECL Software Quality Assurance Program

The AECL Software Quality Assurance Program complies with the CSA Standard N286.7-99 [3] that applies to the design, development, maintenance, modification, procurement and use of computer programs to perform or support:

a) design and analysis of safety-related equipment, systems, structures and components,
b) deterministic and probabilistic safety analyses and reliability studies,
c) reactor physics and fuel management calculations, and
d) transfer of data between computer programs or pre- or post-processing calculations associated with (a), (b) and (c) above.

This program does not apply to computer programs such as:

a) computer-assisted design and drafting system (CADDS),
b) “real time” computer programs for control and monitoring systems,
c) commercially available and widely used general-purpose office automation computer programs such as word processors, spreadsheets, databases, and graphics computer programs (Programmed applications are, however, included within the scope of this manual.),
d) compilers, interpreters, and operating systems, including run-time libraries, and
e) computer programs used for training or user evaluation or other purposes for which the results will not be used in safety-related design or licensing of nuclear power plants.

The documentation supporting the use and maintenance of a computer program is required to include:

a) Programmer’s Manual
b) Computer Program Abstract
c) Theory Manual
d) User’s Manual
e) Validation Report/Manual
f) Version Tracking Record, and
g) Verification reports for test cases.

This documentation may be contained in one or more physical documents, as appropriate.

A requirement of the Software Quality Assurance program is that Qualification Plans be prepared which list the requirements against which existing (legacy) computer codes are verified if such computer programs are used in performing substantially new safety or licensing analysis.

The Qualification Plan specifies actions/plans/schedule to qualify a code. This plan will include:

a) identification of the extent to which the computer program conforms to the design and development requirements from the AECL Software Quality Assurance Manual,
b) justification for non-conformances with the same design and development requirements,
c) definition of additional verification needs and the verification activities to be performed, and
d) a schedule over which verification activities will be performed.

A computer code is fully qualified for use within a specific range of application when a completed set of quality assurance documentation is available, the coding has been verified and validation reports are available covering the range of application.

A computer code that is not fully qualified may be used under circumstances where the impact of such use is judged to entail acceptable project risk, or where the there are no significant safety implications.

The ACR product development entails concurrent engineering design, research, development and testing. The R&D program includes the tasks required to qualify the computer codes to meet the needs of the engineering design program as the product matures. As such, the code qualification program is proceeding in parallel with the design effort. The qualification effort is prioritized to focus on the most important computer codes used in both design and safety analysis. In addition, priority has been given to completing the most important validation extension work in 2004.

Because the ACR design is an extension of the current CANDU design, no major surprises are anticipated from the validation extensions. A priority focus on the validation extension of key codes is intended to ensure that uncertainties in code predictions are minimized. In the event that the results from validation exercises lead to further requirements for code improvements, key safety analyses will be repeated.

In cases where safety analysis is performed with a computer code for which qualification is incomplete, confirmatory analyses will be performed when the specific computer code is fully qualified.

2. SAFETY CODE QUALIFICATION PROGRAM

The primary codes used in the safety analysis of the ACR-700 design must meet the highest level of quality assurance and will be fully qualified. Because the ACR design is functionally similar to the CANDU 6 design, the suite of safety analysis tools available for the CANDU 6 is generally applicable to the ACR design as well. There is no requirement to develop new codes to address unique ACR design features or accident sequences. The qualification of an existing computer code for application to the ACR design consists of three elements:

1. Assessment of the applicability of the code and the adequacy of the existing validation basis.
2. Modification of the code, if necessary, to address specific ACR design features or parameters.
3. Additional validation of the code, if necessary, either to extend the validation basis, or to align the validation range with the applicability range of the computer code.

The validation will:

1. demonstrate the capability and credibility of a particular computer code for use in specific design and analysis applications, and
2. quantify the accuracy of the computer code calculations (quantified through comparison with experimental measurements, for use in the determination of uncertainty allowances for conditions/geometries of the intended applications).

All of the codes used in the ACR development project will be assessed and their qualification status determined. Based on the assessment, additional validation and modifications (major or minor), if necessary, will be performed for the codes.

There are three classes of primary codes that will be used in the safety analysis of the ACR-700 design:

1. IST Codes: AECL and the CANDU Owner's Group (COG) have worked together to establish an Industry Standard Toolset (IST). This is a collection of the most important safety analysis codes that have been developed by AECL and others for common application to the CANDU design. The IST codes are fully qualified for use in the analysis of the current CANDU design.
2. AECL CANDU Codes: This class includes codes that AECL has developed for its use as a designer of the CANDU reactor. It also includes versions of the IST codes that have been modified and validated to be applicable to the ACR design configuration. It is AECL's intent to incorporate modifications (if required) into the existing IST codes in such a way as to maintain the qualification of the codes for application to all CANDU reactors, and thereby avoid the creation of a non-IST code version wherever possible.
3. Third-Party Codes: This class includes commercially available third-party computer codes.

The safety analysis codes include the suite of physics codes necessary for the design and analysis of the ACR core, reactor control devices and fuel.

Because of their safety-significance, the qualification of the safety analysis codes that will be used in the development and licensing of the ACR-700 has been given the highest priority in the ACR-700 R&D program.

3. SAFETY CODE ASSESSMENT

The first step in the preparation of a qualification plan is an assessment of the applicability of a code for analysis of the ACR-700 design. This establishes the need, if any, for code modifications and/or further validation of the code. The applicability of a safety analysis code must be assessed against the design of the reactor and the phenomena associated with the potential accident sequences for that design. The ACR-700 reactor design is fundamentally equivalent to the CANDU design in terms of the overall safety systems and safety-related system configuration and function. Most of the key design features of ACR-700 are identical to the current fleet of operating CANDU reactors, including horizontal fuel channels submerged in a large calandria vessel filled with a water moderator operating at near-atmospheric pressure.

The key phenomena associated with potential accident events in the ACR-700 are common with those of the current CANDU reactors. As per the requirements of the AECL Software Quality Assurance program, the phenomena identification and ranking has been completed and documented for the ACR-700 application [4]. The phenomena identification and ranking follows the process established by international practice [5], [6].

A schematic of the process used for code assessment is outlined in Figure 3-1. The assessment of code applicability is performed jointly by a team of specialists, led by the code holder, and including discipline experts, ACR designers and safety analysis specialists [7]. The assessment focuses on the implications of the differences between the ACR design and the standard CANDU design. The assessment covers the important aspects of the computer code structure, models, constitutive correlations, validation database, etc.

The code applicability assessment objectives were to:

i) Identify code modules affected by the ACR design changes,
ii) Identify code upgrades and adjustments required to address ACR design changes,
iii) Perform assessment of potential extensions of the experimental database to cover ACR conditions,
iv) Assess the impact of design changes on the code validation status, and
v) Identify experimental work required to supportf code changes (if necessary).

The assessment process established different requirements for further code development depending on the extent of code changes that are required. For some codes no changes may be required as the phenomena modeled and parameter ranges are common to both ACR and current CANDU designs.

For some codes, changes in the code models or correlations may be required to address ACR design features (central branch of the diagram in Figure 3-1). Depending on the extent, these changes these can be considered as minor or major. Those codes that require major modifications will require most of the time and effort for qualification.

Extensions of the validation for some codes may be required (with or without associated code modifications) if the assessment determines that the existing validation range does not encompass the requirements for ACR safety analyses. This incremental validation may require additions to the validation database. For most of the codes that require changes and/or incremental validation, a documentation update is required.

Initial assessments of the applicability and validation basis of the major safety analysis codes have been performed. The results of these assessments are described in the following sections. Some of the initial assessments were carried out during the ACR-700 design concept development phase during which the reactor product was identified as the Next Generation (NG) CANDU. The NG CANDU design included the same basic design and safety features as the ACR-700 design (since the latter is a more detailed refinement of the NG CANDU concept), so that the key issues of code applicability are generic to both designs. For example, the implications of the use of light water coolant are common to both the NG CANDU and ACR-700 designs.

Finalization of some details in the ACR-700 design could have a minor impact on the code assessment findings (for example the parameter range of interest may be slightly different). The ACR-700 design has now matured to the point where more extensive details are available and documented in the ACR-700 Technical Description [8]. The code assessments will be reviewed and updated, with respect to the latest ACR-700 design information and the Technical Basis Document for ACR safety analyses [9] to ensure that the codes are fully qualified for application to the ACR-700 design. The assessments and associated qualification plans will be further reviewed and revised, if necessary, taking into consideration any changes in the design during the engineering development phase.

The following sections summarize the results of the initial code assessment process. Figure 3-2 shows schematically how the primary safety analysis codes are linked and the linkages are described in Reference [7].

3.1 IST Safety Code Assessment

The Industry Standard Toolset is the group of primary analysis tools that is used to perform the safety analyses required for licensing. The IST codes have been validated for application to the existing CANDU reactor designs [10], [11], [12]. Table 3.1-1 lists the IST codes that will be used in the ACR safety analysis and their assessment status. For those codes where no modifications or only minor modifications have been determined to be necessary, the IST version of the code will be used for ACR analysis. For the few codes where major modifications are required, notably ELESTRES, an ACR-specific code version may be created. AECL intends to maintain the IST status and common applicability of the IST codes where possible and subject to the agreement of the IST partners.

The planned work for those codes where substantial modifications or incremental validation is required is described in Section 4.

Table 3.1-1
IST Codes for ACR Safety Analysis

Code	Application	ACR Assessment Status	Comment
MODTURC_CLAS-IST	3-D steady-state thermalhydraulics analysis of the moderator	Initial Assessment completed. Incremental validation required.	Incremental validation will address changes in core dimensions.
ELOCA-IST	Fuel behaviour under accident conditions	Initial Assessment completed. Minor modifications and incremental validation required.	Correlations for U-Dy fuel to be added.
ELESTRES-IST	Fuel and fission product behaviour under operating conditions	Initial Assessment completed. Major modifications and incremental validation required.	Modifications will address change to SEU fuel design.
TUBRUPT-IST	Thermalhydraulic transient in the moderator with fuel channel rupture	Initial Assessment in progress. Incremental validation required.	Extension of database for higher channel pressure, and smaller core lattice pitch.
SMART-IST	Fission product behaviour and transport in containment under accident conditions	Initial Assessment completed. Applicable to ACR safety analysis.	ACR containment phenomena common with current CANDUs.
SOURCE-IST	Fission product inventory in fuel	Initial Assessment completed. No ACR-related modifications or validation required.	ACR fuel phenomena common with current CANDUs.
GOTHIC-IST	3-D gas mixing in containment (including combustion)	Initial Assessment completed. Applicable to ACR safety analysis.	ACR containment phenomena common with current CANDUs.
MAAP4-CANDU-IST	Severe accident analysis	Initial Assessment completed. Major modifications required.	Changes to address new core design: geometry and materials/dimensions.

Code	Application	ACR Assessment Status	Comment
WIMS-IST	Lattice cell reactor physics analysis	Initial Assessment completed. Requires physics library update and incremental validation.	Changes to address coolant change and SEU fuel design
RFSP-IST	Full-core fuel management	Initial Assessment completed. Incremental validation required.	Changes to address coolant change and SEU fuel design
DRAGON-IST	Lattice cell reactor physics analysis	Initial Assessment in progress. Incremental validation required.	Changes to address coolant change and SEU fuel design
ADDAM-IST	Atmospheric radionuclide dispersion	Assessment in progress. Validation in progress.	No ACR-specific requirements anticipated.
SOPHAEROS	Fission product transport in the HTS under accident conditions	Initial Assessment completed. Applicable to ACR safety analysis.	ACR heat transport system phenomena common with current CANDUs.

3.2 AECL Safety Code Assessment

In addition to the safety analysis codes of the Industry Standard Toolset, AECL will use two major AECL-developed codes for the analysis of the ACR-700. These codes apply particularly to the design of heat transport system for both normal operating conditions and for transient analysis under accident conditions.

As noted in Section 3.1, for those IST codes where major modifications are required, an ACR-specific code version may be created. Those resulting codes would, in future, fall into the category of AECL-developed codes, but would retain their IST heritage.

Table 3.2-1
AECL Codes for ACR Safety Analysis

Code	Application	ACR Assessment Status	Comment
CATHENA	Transient 2-phase thermalhydraulics analysis	Initial Assessment completed. Minor modification and incremental validation required.	Addition of light water properties for coolant and extension of validation to higher temperatures and pressures.
NUCIRC	Steady-state thermalhydraulics analysis	Initial Assessment completed. Minor modifications and incremental validation required.	Addition of light water properties for coolant and extension of validation to higher temperatures and pressures.

3.3 Third-Party Safety Analysis Codes

The IST tool set (listed in Table 3.1-1) includes a number of safety analysis codes that have been developed by non-CANDU industry suppliers and modified or qualified for application to the CANDU design. These include MODTURC-CLAS, GOTHIC and MAAP4-CANDU. With the exception of the MAAP4-CANDU code, there is no requirement for code modifications for ACR application.

The basic design configuration of the ACR reactor, core geometry and orientation, heat transport system design and containment design are common with current CANDUs. In addition the phenomena applicable to modelling of severe accident progressions in both ACR and current CANDUs are the same. However, major modifications of the MAAP4 code are required to address the specific features of the ACR design that are hard-wired within the code. This includes the availability of a Reserve Water System in the ACR design to mitigate the progression of severe accidents. A contract will be issued to the code supplier, Fauske and Associates (FAI), to modify the MAAP4-CANDU code with the result being a specific ACR code version.

In addition to the primary safety analysis codes, other third-party safety analysis codes will be used to address features of the ACR-700 that are not unique to the CANDU design (e.g., radiation shielding requirements and probabilistic risk assessment methodology) or to provide independent corroboration of the predictions of ACR safety analysis codes. These codes include MCNP (Monte Carlo physics analysis), CAFTA (Computer-Aided Fault Tree Analysis for probabilistic risk assessments) and DOORS (a suite of codes for analysis of radiation physics). For the DOORS codes, the radiation physics libraries will be selected to be applicable to the ACR core physics design.

4. ACR SAFETY ANALYSIS CODE QUALIFICATION PLANS

This section outlines the planned code modifications and incremental validation for the primary safety analysis codes as indicated in Table 3.1-1 and Table 3.2-1. The high-level schedule for the planned qualification work is given in Figures 4-1 and 4-2.

4.1 CATHENA

The CATHENA code is used in CANDU safety analysis to model transient thermalhydraulics in the heat transport system under accident conditions. The assessment of the CATHENA code determined that all of the phenomena modelled by the code are applicable to the ACR design and that there are no gaps or omissions. A need for minor code modification was identified to extend the water properties databases in the code to include light water at the higher ACR coolant operating temperature and pressure.

Examination of the 23 separate phenomena modelled in the code determined that incremental validation was required for eight phenomena. These requirements are outlined in Table 4.1-1. The labelling of the phenomena in Table 4.1-1 matches the identification of the phenomena in the ACR Technical Basis Document for safety analysis.

The primary tool for extension of the validation database for CATHENA will be the RD-14M test Facility located at AECL's Whiteshell Laboratories. This facility includes a full-elevation scaled ACR heat transport system loop. It will be used to generate experimental data that will be used to validate the code in the following areas:

- Break discharge and coolant voiding rates,
- Condensation heat transfer,
- Nucleate boiling, and
- Convection heat transfer.

The extension of the validation for CATHENA is largely required to deal with extension of the existing validation databases to the higher temperature and pressure of the coolant under normal operating conditions. Since the temperature increase is modest (from 310°C to 325°C at the outlet header) it is not anticipated that there will be a need to change the correlations used for the heat transfer phenomena in CATHENA, with the exception of minor changes associated with the geometry of the ACR fuel bundle. In the latter case, the CATHENA code will incorporate the critical heat flux correlation updates that will be validated for the NUCIRC code (see Section 4.2).

In addition to providing data for phenomena validation, the RD-14M facility will be used to validate CATHENA modelling of ACR Emergency Core Cooling System performance in scaled loss-of-coolant accident tests. This is a priority validation extension that is planned for completion in early 2004.

Extension of the database for fuel channel deformation (or failure) and heat transfer rates under deformed channel conditions will be obtained in separate effects tests in the high-temperature heat transfer test facility at AECL's Chalk River Laboratories. This validation extension is required for analyses of events that are beyond design basis, where the channel geometry is subject to change and there can be physical contact between channel components.

Table 4.1-1
Incremental CATHENA Validation

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.
TH7: Convective Heat Transfer FC13: Sheath-to-Coolant and Coolant-to-Pressure Tube Heat Transfer	Validation to extended to include higher pressures. RD-14M tests planned.
TH8: Nucleate Boiling FC13: Sheath-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.
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 planned.
TH18: Fuel Channel Deformation FC18: Pressure Tube Deformation or Failure FC19: Calandria Tube Deformation or Failure	Existing validation considered adequate until data from planned tests with prototype ACR pressure tubes and calandria tubes is available.

4.2 NUCIRC

NUCIRC is the steady-state thermalhydraulics code used for CANDU heat transport system design analysis. Assessment of the code determined that a number of code modifications were required to enhance its performance and extend its applicability to light water coolant at the higher ACR operating temperatures and pressures. These code improvements are being implemented in the code on a prioritized basis to support the design development.

There is a need to modify some of the thermalhydraulic correlations in the code to address the ACR coolant conditions and the ACR-CANFLEX fuel design and core design. The ACR coolant conditions require and extension of the existing CANFLEX thermalhydraulic correlations to higher coolant temperatures and pressures. The ACR-CANFLEX fuel design with SEU fuel leads to channel power profiles that are different from those of natural uranium CANFLEX fuel in the current CANDU design. Notably, the ACR fuel channel will have fuel bundles with a flatter radial power profile and with an axial power profile with the peak power skewed towards the channel inlet.

To extend the validation of the thermalhydraulic correlations, tests will be conducted using electrically heated full-length, full-scale fuel bundle simulators. Tests will be conducted in AECL’s MR-3 test facility using Freon as a coolant to examine variations in radial and axial power profiles and to measure post-dry out and overpower limits. Tests will be conducted in the Stern Laboratories thermalhydraulics test facility to measure critical heat flux at full ACR operating pressure and temperature. Like the previous tests that AECL has conducted to validate the critical heat flux correlations for natural uranium CANFLEX fuel, these tests will be conducted with light-water coolant.

4.3 WIMS/RFSP/DRAGON

The WIMS, RFSP and DRAGON codes are used to analyse the nuclear physics of the ACR core and reactivity control devices. The physical design and the nuclear physics of both ACR and CANDU cores is similar and the base validation for the physics code suite is applicable and the same set of physical phenomena are modelled. However, there are substantive changes in the fuel design (SEU fuel with a dysprosium poison element for ACR) and neutron moderation (light water coolant and tighter lattice pitch for ACR) that impact on the validation basis for the application of these codes to ACR safety analysis.

The WIMS/RFSP/DRAGON code suite is applicable to analysis of the ACR design, but the associated nuclear cross-section data library has been updated from that used for the current CANDU design. This update addresses the changes in the ACR core physics and neutron energy spectrum that arise from the use of higher burnup SEU fuel, a tighter lattice pitch and a change in moderation with the use of light water coolant, compared to conventional CANDUs.

Table 4.3-1 lists the planned activities to extend the validation of the WIMS/RFSP/DRAGON physics code suite. The organization of Table 4.3-1 is different from Table 4.1-1 and shows the links between planned physics experiments and the groups of phenomena that they will address. All of these tests will be carried out in the ZED-2 zero power critical lattice facility at Chalk River Laboratories. The tests will include substitution measurements using both ACR fuel and other fuel designs in reference lattices that provide a match with the ACR core physics conditions (within the capabilities of the ZED-2 facility). The test program is in progress and the

first components of the tests to investigate tight lattice assemblies using existing reference lattice fuel have been completed. A full core load of 0.95% ^{235}U is now being fabricated for future tests. Tests with this reference fuel will provide validation data that will support tighter ranges of uncertainty on physics code results.

Included in the planned work are tests to using prototype ACR reactivity control device elements to assessment of the contribution of delayed photo-neutrons to the total delayed neutron fraction.

In addition data from the ZED-2 tests, other available reactor lattice data will be used to extend the physics code validation. Potential sources of such data include the Japanese FUGEN project (DCA measurements), the UK Steam Generating Heavy Water Reactor (SGHWR) (DIMPLE measurements), the Italian CIRENE project (ECO Reactor measurements) and the Savannah River Laboratory Heavy Water Organic Cooled Reactor (HWOCR) project (Process Development Pile (PDP) measurements).

The goal of the validation extension for ACR is to reduce the uncertainties in the predictions of the reactor core physics. Substantial margin is included in the ACR fuel design and core design to address the level of uncertainties associated with the current validation base. Initial validation exercises completed by 2004 will confirm the applicability of the physics tools within the large margins. Subsequent validation work will reduce the uncertainties.

Table 4.3-1
Incremental WIMS/RFSP/DRAGON Validation

Phenomenon	Validation Extension	Purpose
PH0: Lattice pitch, fuel channel and fuel bundle design, core configuration and operating parameters PH1: Coolant-Density-Change Induced Reactivity	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:	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
PH1: PH2: Coolant-Temperature-Change Induced Reactivity PH7: Fuel-Temperature-Change Induced Reactivity	Fuel/coolant temperature coefficient	Temperature reactivity coefficients (up to 300°) for both SEU and MOX fuel in H_2O and air-cooled tight lattices

Phenomenon	Validation Extension	Purpose
PH1: PH3: Moderator-Density-Change Induced Reactivity PH4: Moderator-Temperature-Change Induced Reactivity	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
PH0: PH1: PH2: PH7: PH14: Flux and Power Distribution in Space and Time	Fine structure flux distribution	Flux distribution in a SEU CANFLEX bundle at both room in H_2O and air-cooled tight lattices
PH1: PH5: Moderator-Poison-Concentration-Change Induced Reactivity	Moderator poison experiments	Boron and/or Gd reactivity effect in ACR type lattices using both H2O and air coolant
PH0: PH8: Fuel-Isotopic-Composition-Change Induced Reactivity PH14:	Benchmark configuration flux distribution	Spatial flux distribution in a heterogeneous ACR-type lattice of SEU and MOX fuel
PH11: Device-Movement Induced Reactivity	Control device measurements	Absorber device inserted into square uniform or checkerboard lattice of SEU and MOX fuel
PH12: Prompt/Delayed Neutron Kinetics	Rod drop experiments	Assessment of the contribution of delayed photo-neutrons to the total delayed neutron fraction

4.4 ELESTRES

The ELESTRES code is used model the fuel behaviour under normal operating conditions. Assessment of the code applicability to the ACR determined that major modifications to the code were required. These were mandated by the increased ^{235}U content on the fuel (~2.1%), the increased burnup of the fuel compare to natural uranium CANDU fuel (~ three times the burnup), and the inclusion of a central 'poison' fuel element containing dysprosium.

Table 4.4-1 lists the major modifications planned to extend the applicability of the ELESTRES code to the ACR design. Most of the planned work has been completed or is in progress. Verification of the new code is proceeding as the code modules are developed.

There are no additional experiments planned to support extension of the validation of ELESTRES code as the existing database has been assessed to be sufficient. Fuel research and development programs at AECL have centred on irradiation of development fuel in the NRU research reactor. The test fuel used in these programs is routinely enriched to achieve the desired power levels and burnup, so there is a substantial database available on SEU fuel irradiation.

AECL has an ongoing program to develop low void reactivity fuel for use in the current CANDU reactors. This fuel design includes elements containing dysprosium at concentrations greater than those planned for the ACR fuel. Data from this program will be used in the validation of the extended ELESTRES code for ACR.

Table 4.4-1
ELESTRES Code Modification for ACR Application

ELESTRES Code Feature	Extension Status
Flux depression	Completed
Fission gas diffusivity	Completed
Grain boundary bubbles	Completed
Free-standing cladding collapse including degree of circumferential wrap-around	Completed
Cladding material properties	Completed
Pellet densification	In Progress
Cladding plasticity	In Progress
Dysprosium properties	Planned
Link to external fine-element meshes	Planned
Cladding oxidation	Planned

4.5 ELOCA

The ELOCA code models the behaviour of fuel under the rapidly changing coolant and power conditions associated with an accident. The assessment of the ELOCA code for ACR application determined that minor modifications to the code were required to extend the thermal properties data for $UO_{2,}$ used in the code, to address the extended ACR fuel burnup. This includes extension to range of data for the thermal conductivity, thermal expansion and specific heat capacity. In addition, the thermal properties database needed to be extended to include the properties of the dysprosium-doped fuel element.

The UO_2 properties database will be extended using available sources of data. Autoclave experiments are in progress to obtain the additional data required for extension of the dysprosium-doped fuel.

The thickness of the fuel cladding in the ACR fuel has been designed to accommodate the ACR coolant conditions (higher temperature and pressure than operating CANDUs). As a result, there is a need to extend the database on cladding strain to include the ACR cladding design values. An experimental program is in progress to extend previous experiments to provide the requisite data for code validation.

4.6 TUBRUPT

The TUBRUPT code is used to analyse the thermalhydraulic transient within the calandria vessel/shield tank assembly for events that include a pressure tube and associated calandria tube failure. The phenomena modelled by TUBRUPT include the flashing coolant hydrodynamic transient in the moderator and the high-temperature debris interaction with water. The full assessment of the TUBRUPT code incremental validation requirements is in progress, but the initial assessment has determined a need for an extension of the database on the calandria tube

rupture to include ACR coolant conditions. An experimental program is now in progress to extend the existing database on scaled calandria tube rupture tests to the ACR conditions.

AECL has a separate experimental program in progress to study the pressure transients associated with high-temperature channel debris interaction with the moderator. This phenomenon is associated with two classes of single channel accident sequences: inlet feeder stagnation break and severe channel flow blockage.

4.7 MODTURC_CLAS

The MODTURC_CLAS code is a 3-D single-phase computational fluid dynamics code that is used to predict moderator flow and temperature distributions within the calandria vessel. Assessment of the code determined that no code modifications were required for ACR application, but that the validation database should be extended. This code has been validated for CANDU application using data obtained in the Moderator Test Facility at Chalk River Laboratories. The baseline validation was for the CANDU 9 core design with the test facility containing on a ¼ linear scale calandria vessel. To address the tighter lattice pitch of the ACR design, the validation base will be extended to include tests in a new calandria vessel/heater lattice that is 1/3 scale of the ACR design. The ACR validation tests can be conducted at a higher scale within the facility because the ACR core is more compact than the conventional CANDU core.

The modifications required for the new Moderator Test Facility vessel have been designed and the validation tests will be conducted when fabrication is complete.

5. SUMMARY

AECL is developing an advanced reactor design, designated the Advanced CANDU Reactor (ACR) based on an evolutionary extension of the existing CANDU 6 design. AECL will use a limited number of analytic, scientific and design computer codes (or tools) to develop the ACR design, to perform safety analyses of postulated accident events and to assess the design against specific safety acceptance criteria.

The codes used in the ACR design and safety analysis will be qualified for their application. A comprehensive review and assessment of the primary safety analysis tools required to support the ACR design has been initiated and efforts to extend code qualification, where required, are in progress. For most codes there are minor or no modifications required for ACR use and there are, at most, minor requirements to extend the validation databases. For a few codes a need for major code modifications or substantial extensions to the validation databases have been identified. Work is now in progress to address all of these requirements.

Particular attention has been paid to the code qualification requirements for the transient thermalhydraulics code (CATHENA) and the reactor physics code suite (WIMS/RFSP/DRAGON). These codes are applicable to analysis of the ACR design without a need for major modifications. However, there is a need to extend the validation database for some of the phenomenal modelled by the codes. A comprehensive set of experiments is planned and in progress to provide the additional validation data.

All primary ACR safety analysis codes are based on the Industry Standard Toolset codes or existing AECL safety analysis codes that have been verified and validated. Upgrades to these codes and incremental validation to meet ACR requirements will be conducted in compliance with AECL's software quality assurance standards.

6. REFERENCES

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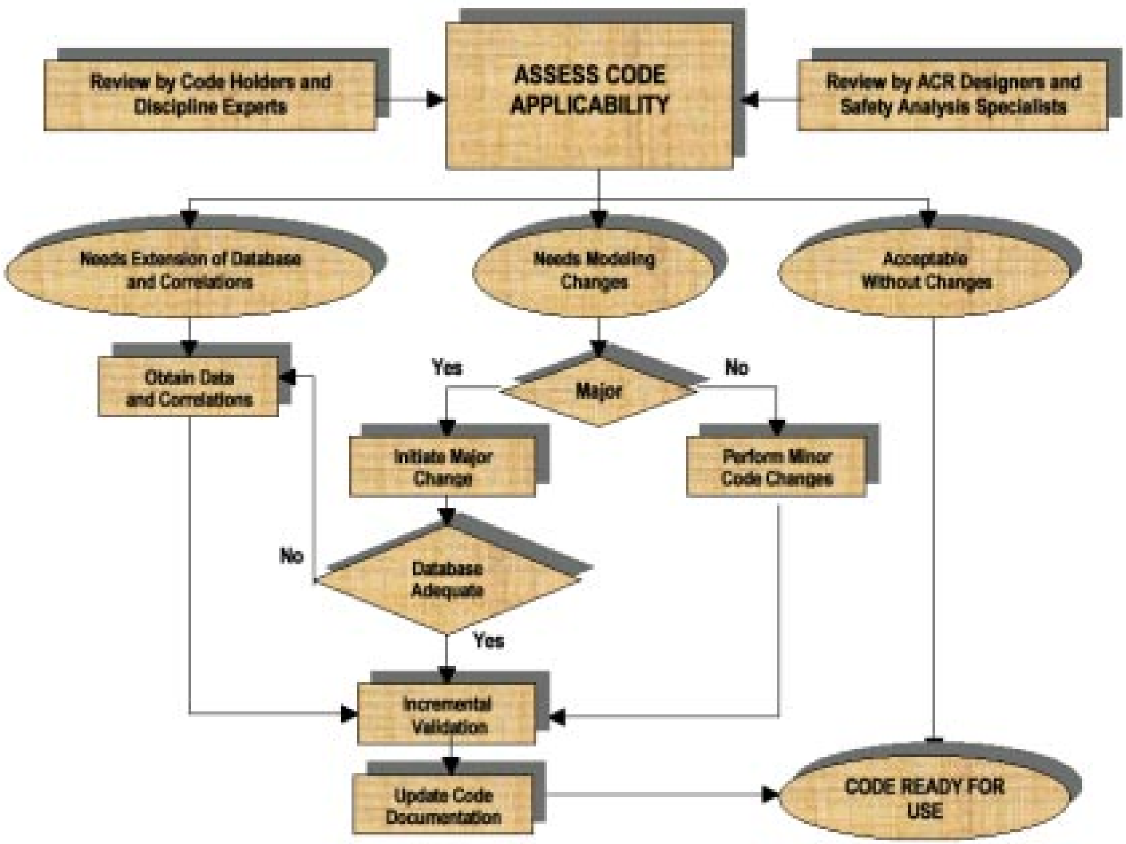

Figure 3-1 Assessment of Computer Codes for ACR Safety Analysis

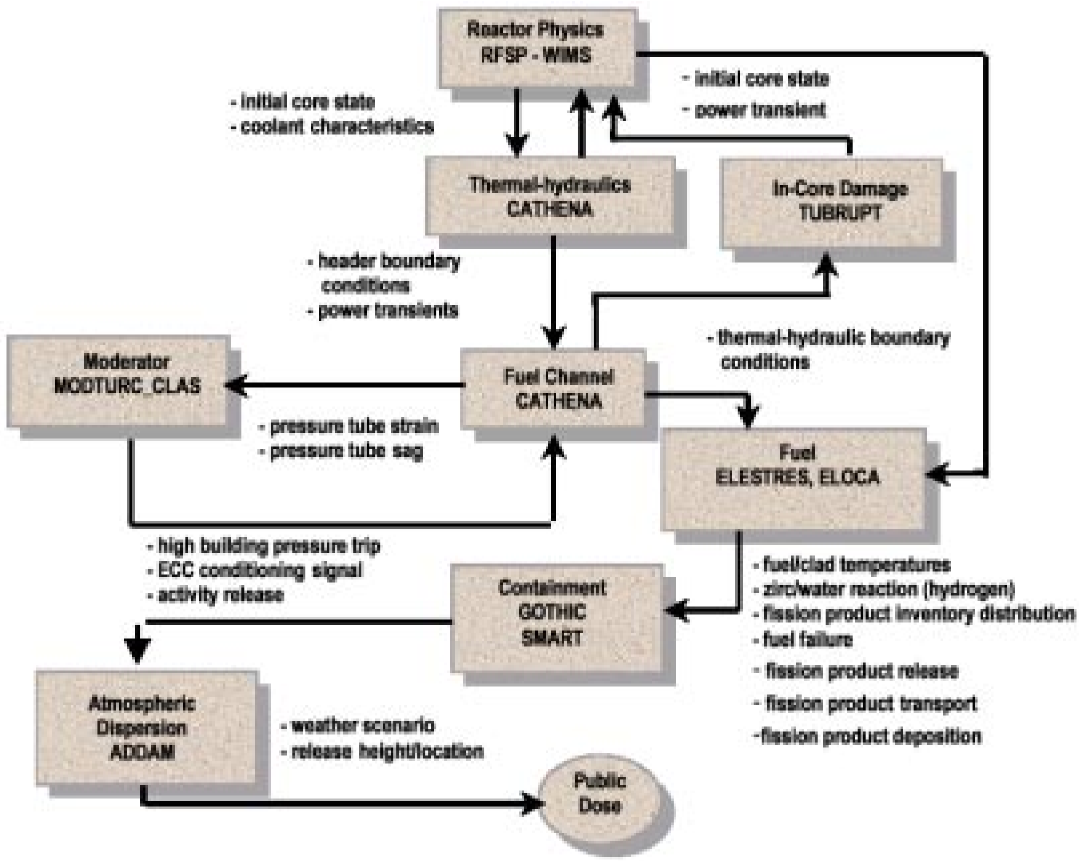

Figure 3-2 Schematic of Primary Computer Codes Used in ACR Safety Analysis

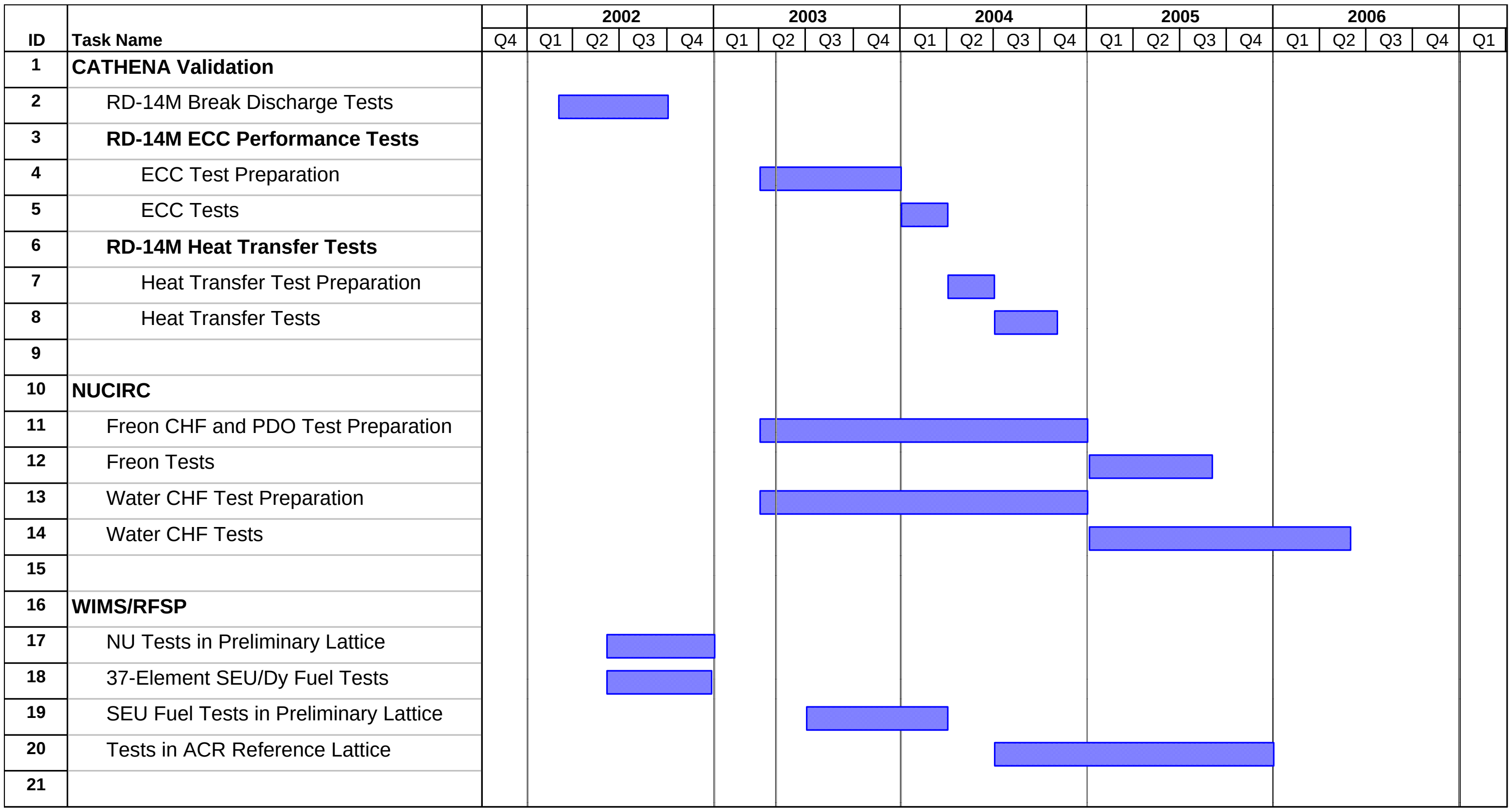

Figure 4-1 High-Level Schedule for ACR Safety Analysis Code Validation Extension

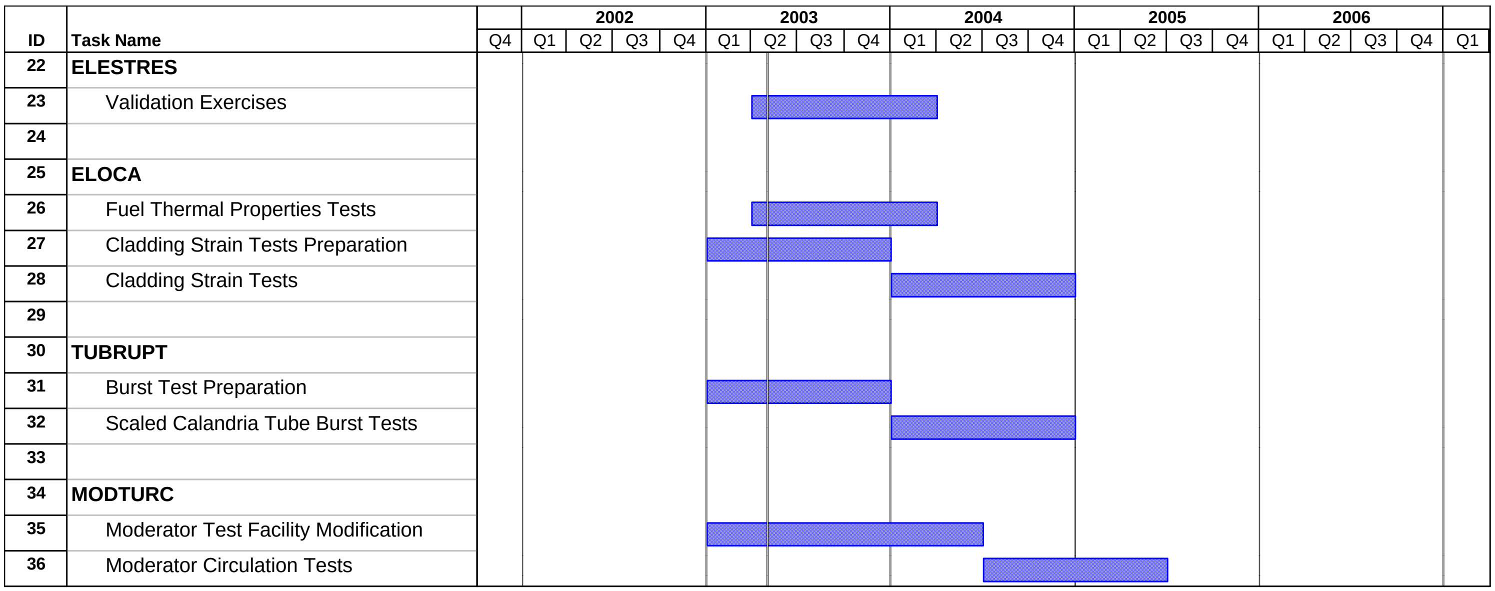

Figure 4-2 High-Level Schedule for ACR Safety Analysis Code Validation Extension - continued