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ACR-700 ANTICIPATORY R&D PROGRAM

ACR-700

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1. INTRODUCTION

This document describes the Anticipatory R&D Program that AECL plans to carry out to provide data to support the licensing of the ACR-700™* design.

The ACR-700 is an advanced reactor design based on a change from heavy-water-cooled natural uranium fuel to light-water-cooled slightly-enriched uranium fuel, while retaining heavy-water moderation. The ACR-700 requires different components in several systems in order to accommodate these changes in fuel and coolant. As well, the ACR-700 design will adopt improved systems and components, and improved design and construction methods to meet market requirements for a low capital cost product.

The ACR design quality assurance program includes requirements to verify the safety of the design through appropriate combinations of analysis and testing. Since the ACR-700 design is an evolution of the operating CANDU®† reactors, there is a large technology base that provides for verification of those design features that are common. Where possible, the ACR-700 design incorporates systems and components that have already been demonstrated in existing plants. For new ACR-700 systems or components that have a significant impact on safety, an anticipatory R&D program has been established to provide assurance to licensing authorities and customers that new design features are effectively proven.

This document describes the anticipatory research and development activities that will be carried out to support the safety design of an ACR-700 plant. The R&D activities and requirements were established based on a review of the new features in the ACR design [1] and the impact of these features on safety systems and key safety analysis tools. Revisions to this program during the basic engineering phase may arise as a result of refinements in the design, interim research results or input from other sources such as the design feedback process and regulatory feedback.

The R&D program addresses five design verification objectives:

1. Qualification of safety analysis tools.
2. Fuel Design Verification.
3. Safety System Performance Verification.
4. Fuel Channel Performance Verification.
5. Verification of Other Components and Systems.

Completion of the work in this report is sufficient to verify those codes, systems and components important for safety. However, ongoing R&D is planned to extend the database supporting the performance of the design, particularly in studies addressing aging related phenomena where longer-term testing is valuable.

The R&D described in this document is focussed on those design verification activities that are necessary to support the safety of the ACR-700 design. Additional R&D will be carried out in

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* ACR-700™ (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).

support of the development of the ACR-700 and in support of opportunities to improve the overall performance and economics of the ACR-700 plant. In addition research and development activities will take place after scheduled completion of the design verification work to extend the supporting database for ACR-700 materials and components, to extend validation of design margins, and to address design support requirements for non-critical plant improvement features.

2. SAFETY ANALYSIS TOOL QUALIFICATION

The safety analysis of the ACR design will be performed using a set of qualified safety analysis tools (computer codes) that have been validated over their range of application [2]. Since the ACR is an evolutionary adaptation of the current CANDU design, the safety analysis tools that have been validated for the CANDU application are generally applicable to the ACR design as well. However, there are advances in the ACR design that change the relative importance of some phenomena for certain event sequences or that impose additional requirements on the standard CANDU analysis tools.

Table 2-1 lists the key safety analysis tools and their discipline. All of the codes have been assessed for their applicability to ACR safety analysis. For several codes, a need for significant extensions to the validation base has been identified. These codes include:

- CATHENA
- NUCIRC
- WIMS/RFSP/DRAGON
- ELESTRES
- ELOCA
- TUBRUPT
- MODTURC_CLAS.

The planned R&D program to address the validation requirements for these codes is outlined in the following sections.

Table 2-1
ACR Safety Analysis Tools

Discipline	Computer Code
REACTOR PHYSICS	
Lattice Physics	WIMS-IST*
	DRAGON-IST
Core Physics	RFSP-IST
SYSTEM THERMALHYDRAULICS	
Primary Circuit Transient Thermalhydraulics	CATHENA
Primary Circuit Steady-State Thermalhydraulics	NUCIRC
CONTAINMENT BEHAVIOUR	
Containment Thermalhydraulics & H_2 Behaviour	GOTHIC-IST
MODERATOR BEHAVIOUR	
Moderator Circulation	MODTURC-CLAS-IST
Moderator Behaviour for In-Core Break	TUBRUPT-IST
FUEL BEHAVIOUR	
Fuel Initial Conditions	ELESTRES-IST
Fuel Transient Behaviour	ELOCA-IST

* The suffix IST stands for Industry Standard Toolset, and represents a code, which has been validated and verified in a joint collaborative effort by the Canadian nuclear industry (AECL, OPG, NBPower, Bruce Power and Hydro Quebec).

Discipline	Computer Code
FISSION PRODUCT BEHAVIOUR	
Fission Product Release from Fuel	SOURCE-IST
Fission Product Transport in Primary System	SOPHAEROS
Fission Product Transport in Containment	SMART-IST
DOSE CONSEQUENCE ASSESSMENT	
Atmospheric Dispersion	ADDAM-IST
SEVERE ACCIDENT BEHAVIOUR	
Severe Core Damage During Severe Accidents	MAAP4-CANDU

2.1 CATHENA

The CATHENA code is used in CANDU safety analyses 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 verification was identified for the water properties databases in the code for light water at the higher ACR coolant operating temperature and pressure. This has been completed.

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 2-2. The labelling of the phenomena in Table 2-2 matches the identification of the phenomena in the ACR Technical Basis Document for safety analysis [3].

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 measure:

- System depressurization and coolant voiding rates,
- Condensation heat transfer,
- Nucleate boiling, and
- Convection heat transfer.

In addition to providing data for phenomena validation, the RD-14M facility [4] will be used to validate CATHENA modelling of ACR Emergency Core Cooling System performance in scaled loss-of-coolant accident tests.

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.

Table 2-2
Incremental CATHENA Validation

Phenomenon	Validation Extension for ACR
TH1: Break Discharge Characteristics	Extend validation to include higher pressures using existing database and RD-14M LOCA tests (completed).
TH2: Coolant Voiding	Extend validation to include higher pressures using existing database and RD-14M LOCA tests (completed).
TH4: Level Swell and Void Hold-up	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 be extended to include higher pressures. RD-17 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-17 tests planned.
TH10: Condensation Heat Transfer	Validation to be extended to include higher pressures. RD-17 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 using numeric tests.
TH18: Fuel Channel Deformation FC18: Pressure Tube Deformation or Failure FC19: Calandria Tube Deformation or Failure	Existing validation considered adequate until data from tests with prototype ACR pressure tubes and calandria tubes is available.

2.1.1 Planned Experimental Programs

Eight groups of phenomena are identified in Table 2-2 as requiring incremental validation. Existing data has been identified to complete validation for two of these phenomena (break discharge and coolant voiding). A numerical test is available to complete validation of one phenomenon (radiative heat transfer). Data is being sought to extend the validation of one phenomenon (level swell). Test results are required to extend the validation of the remaining four phenomena (convective heat transfer, nucleate boiling heat transfer, condensation heat transfer and fuel channel deformation). To aid in the planning for these tests, suggested test parameters are discussed in Sections 2.1.1 to 2.1.3 below.

A series of RD-14M tests on break discharges to extend the database to the ACR coolant conditions has already been completed. Combined with the existing database, this information will support validation of CATHENA for the first two phenomena of Table 2-2.

While not strictly required to support CATHENA validation for ACR analyses, two additional sets of thermalhydraulic tests have been identified to support the ACR development program. For completeness, these test programs are also briefly described in Sections 2.1.4 and 2.1.5.

2.1.2 RD-17 Heat Transfer Tests

These planned tests will provide data to extend validation of liquid convection and nucleate boiling heat transfer at higher pressures. The tests will be conducted at ACR-representative pressures.

A set of steady-state tests at ACR coolant and secondary side conditions are planned for early in 2004. The tests will include different power levels, different flows and different pressures. The tests will be conducted in the RD-17 facility located at Whiteshell Laboratories. This is facility consists of a single electrically-heated fuel element simulator contained in a flow tube. It is capable of operating at ACR pressures, temperatures and scaled mass fluxes.

The measured parameters of greatest interest in these tests are those related to fuel element simulator-to-coolant heat transfer, such as coolant and sheath temperatures, coolant pressures, flows and void fractions, and heated section power.

2.1.3 RD-14M Very Small Break Tests

As part of the existing CATHENA MOD-3.5c/Rev 0 validation, three very small-break blowdown tests were conducted in RD-14M. The 3-mm break size used in these tests produced very slow depressurization and voiding. In these tests, void produced in the heated sections and outlet feeders was condensed in the steam generators. Thus these tests provided useful data for validation of nucleate boiling and condensation heat transfer at pressures between 6 and 8 MPa.

Validation of the nucleate boiling and condensation phenomena at higher pressures is required to support application of CATHENA to ACR analyses. A set of similar tests will be conducted, with initial conditions that are more representative of ACR operating conditions.

Three tests have been conducted with RD-14M in the full 5-heated-section-per-pass configuration. The recently completed RD-14M/ACR LOCA tests included two tests with a 3.0-mm break located at the inlet header. Due to the smaller volume and lower total power input to the modified facility, the required period of sustained high pressure, with boiling in the heated sections and condensation in the steam generators, was not produced in two ACR tests. For the single-heated-section-per-pass ACR configuration, a break size smaller than 3 mm is required.

One or two tests using a smaller break size (probably about 1 mm) will be carried out. In these tests, the objective will be to create an operating state where steam is produced in the heated test sections via nucleate boiling, and is condensed in the steam generators, while the pressure is sustained at as high a value as possible (at least 11 MPa). Pre-test simulations to determine the required break size will be a key part of the planning for these experiments.

The primary measured parameters such as coolant and sheath temperatures, coolant pressures, flows and void fractions, and heated section power will provide direct validation of nucleate boiling heat transfer. Coolant pressure will be used as an indirect validation of condensation heat transfer.

2.1.4 Fuel Channel Deformation Tests

The design of the ACR has increased margins with respect to fuel and pressure tube maximum temperatures for large loss-of-coolant accidents (LOCA) due to the elimination of a power pulse associated with coolant voiding. As a result, preliminary safety analyses predict that there will be no significant fuel channel deformation for design basis accidents and particularly large LOCA events. However, an R&D program is planned to extend the existing database on fuel channel deformation and heat transfer under high-temperature transient conditions to address the changes in the ACR design, including changes to the pressure tube and calandria tube geometry and use of the CANFLEX®* fuel bundle.

Several experimental programs are planned to examine the anticipated fuel channel behaviour for LOCA conditions. The tests would also look at fuel channel deformation and heat transfer characteristics for beyond design basis events (LOCA plus loss of emergency cooling) in support of severe accident assessment and mitigation (see Appendix A). These tests would include investigations of pressure tube ballooning and possible rupture under coolant boil-off conditions, subcooling requirements to maintain calandria tube (and therefore fuel channel) integrity after pressure tube contact, and the effect of local hot spots (for example due to bearing pad contact or the dropping of molten metal onto the pressure tube inner surface).

The results of similar tests using the current CANDU fuel channel design have been the basis of validation of CATHENA MOD-3.5c for the fuel channel deformation phenomenon. The proposed tests will provide data to support the validation of CATHENA MOD-3.5d for events that could lead to deformation of the ACR fuel channel. Results from the proposed tests program are not likely to be available before FY 2004/05. These results will be reviewed as they become available, and used where applicable and as necessary to extend the validation of CATHENA for the fuel channel deformation phenomenon.

2.1.5 CANFLEX Fuel Bundle Flow Regime Tests

CANFLEX fuel bundle flow regime tests are not required to support validation of CATHENA for application to ACR analyses, but are required to support the modelling of phase separation in a CANFLEX bundle. Initially, air/water tests will be conducted, at atmospheric pressure. If these tests reveal a significant change in the flow stratification criteria, more sophisticated tests can be carried out.

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* CANFLEX® is a registered trademark of AECL and the Korea Atomic Energy Research Institute (KAERI).

2.1.6 ACR ECC Tests in RD-14M

In existing CANDU plants, Emergency Core Coolant (ECC) is injected into all headers following a LOCA. Simulations of postulated large LOCAs indicate that, with this all-header injection design, significant delays could occur before the reactor fuel channels are refilled, in particular in the unbroken core pass (downstream of an outlet header break or upstream of an inlet header break).

For the ACR, the emergency cooling injection system has been modified with the addition of a large outlet header interconnect pipe (Figure 1). This design feature removes the large elevation change required to pass water flow over the steam generator tubes, allowing flow to cross easily from the unbroken to broken core pass. With the outlet headers interconnected, emergency coolant injection is provided only to the inlet headers. Preliminary CATHENA simulations of postulated LOCAs indicate that this design change results in an improvement in refill effectiveness.

These proposed tests are not required to support phenomenological validation of CATHENA for application to ACR analyses, as they provide no new data on the phenomena identified as requiring additional, ACR-specific validation. However, the tests do contribute to verification of the overall applicability of CATHENA to ACR analyses and will constitute a systems-level validation of the code performance.

The modification to the RD-14M facility for these tests will be carried out based on a scaling analysis of the test configuration to size the ECC supply (pumped injection is proposed for the tests rather than the independent accumulator tanks in the ACR design) and the outlet header interconnect pipe.

2.2 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 also 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 physics. 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 bundle and 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 as discussed in Section 3.3.

The thermalhydraulic margins for the ACR fuel will be very high at the start of reactor life. At that time the internal cross-section of the pressure tube is at its minimum and the coolant flow through the fuel bundles is at its maximum. During its operating life, the pressure tube diameter will increase slightly due to thermal and irradiation induced deformation (see Section 5.4). This deformation will permit an increasing fraction of the coolant to flow to by-pass the fuel bundles and decrease the thermalhydraulic margin. The tests will be conducted to measure data for thermalhydraulic behaviour at both beginning and end of the pressure tube lifetime.

2.3 WIMS/RFSP/DRAGON

The WIMS, RFSP and DRAGON codes are used to analyse the nuclear physics of the ACR core and reactivity control devices. While the nuclear physics of both ACR and CANDU cores is similar, 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 2-3 lists the planned activities to extend the validation of the WIMS/RFSP/DRAGON physics code suite. All of these tests will be carried out in the ZED-2 zero power critical lattice facility at Chalk River Laboratories. Two basic types of experiments are performed in ZED-2. Flux maps involve a whole core of test fuel, in either square of hexagonal lattices at room temperature, and yield spatial neutron flux profiles. Simple analytical data fitting is used to extract the whole core buckling. By contrast, in a substitution experiment, the tests fuel is limited to up to seven channels in the centre of a hexagonal lattice surrounded by reference (or driver) fuel that is usually different from the test fuel. Post-processing by specialized software is required to extract the whole core buckling, but the technique can be used for fuel temperature dependent measurements and when it is not possible to have a whole core load of the test fuel. In either case, the resulting whole core bucklings are used for code validation.

A series of preliminary measurements have already been completed in the ZED-2 facility using readily available reference lattice and substitution fuels. A full core-load of reference lattice fuel is currently being fabricated, with the maximum enrichment possible for use in ZED-2, in order to carry out a more extension series of validation measurements to a higher degree of accuracy. The completed and planned tests are described briefly in the following sections.

Table 2-3
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 coolant 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 coolant 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[1] fuel in H_2O and air-cooled tight lattices
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 an SEU CANFLEX bundle at both room and elevated temperatures in H_2O and in 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 H_2O and air coolant

[1] MOX refers to mixed-oxide substitution fuel bundles that will contain both Pu and U to simulate irradiated ACR fuel.

Phenomenon	Validation Extension	Purpose
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

2.3.1 Preliminary ZED-2 Measurements (Completed)

2.3.1.1 Buckling Measurements using 28-Element NU Fuel

Buckling measurements have been performed using the standard ZED-2 28-element natural-uranium (NU) UO_2 assemblies. These measurements were performed using H_2O and air coolant, representing zero void and 100 percent void, respectively. Measurements were performed by flux-map at three hexagonal lattice spacings, 20 cm, 21.59 cm and 22.86 cm. The measurements were repeated using the substitution method so that the substitution-derived bucklings could be compared to the flux-map-derived values. These data expand the substitution-method validation into the regime of ACR-type lattices.

2.3.1.2 Substitution Experiments using 37-Element LVRF

Substitution measurements have been performed using seven CANDU-type pressure tube/calandria tube (PT/CT) assemblies containing 37-element Low Void Reactivity Fuel (LVRF) containing dysprosium. The reference lattice for these measurements comprised of ZEEP (ZEro power Experimental Pile) rods arranged at a 21.59-cm hexagonal spacing. Three coolant types were used in the LVRF test assemblies—H_2O, D_2O and air—to represent void fractions of 0%, 87% (nominal) and 100%. These measurements provide information on coolant void reactivity (CVR) for slightly enriched uranium (SEU) fuel containing dysprosium.

2.3.1.3 Fine-Structure Experiments using 37-Element LVRF

Fine-structure reaction rate measurements have been performed using seven 37-element LVRF test assemblies. For these measurements a special demountable bundle with removable elements was loaded with neutron activation foils.

2.3.2 ACR Lattice ZED-2 Measurements (Planned)

This section describes the planned series of physics code qualification measurements that will be performed in the ZED-2 facility using ACR-type lattices that incorporate 0.95 wt.% ^{235}U fuel. A full core-load of this reference lattice fuel is currently being fabricated at the Chalk River Laboratory.

2.3.2.1 Hexagonal Lattice Measurements

Flux-map buckling measurements using hexagonal lattices will be performed using assemblies containing 0.95 wt% SEU fuel with three coolants—H_2O, D_2O and air. The bundles will be inserted into the existing ZED-2 aluminum PT/CT assemblies for these tests. Measurements will be performed at three or more lattice spacings spanning the equivalent ACR pitch corresponding to a moderator-to-fuel ratio of 7.1.

2.3.2.2 Square-Lattice Measurements

The flux-map buckling measurements will be repeated using square lattices at various pitches spanning the equivalent ACR pitch. Copper neutron activation foils will be positioned across the lattice and out into the heavy water-reflector. These activation data will demonstrate how the thermal neutron flux peaks in the reflector on coolant voiding an ACR-type lattice.

2.3.2.3 Substitution Measurements

Substitution measurements will be performed using seven ACR-type PT/CT assemblies containing CANFLEX LVRF. The LVRF bundles will contain SEU in the outer 42 elements and NU plus dysprosium in the centre element. The reference lattice fuel for these measurements will comprise 0.95% SEU in the existing ZED-2 aluminum PT/CT assemblies. Three coolant types should be used in the test assemblies—H_2O, D_2O and air.

The substitution measurements will be repeated using available mixed oxide (MOX) bundles containing plutonium to simulate irradiated ACR fuel.

2.3.2.4 Reactivity Coefficient Measurements

These measurements will provide data to verify the ability of reactor-physics cell codes to calculate various temperature and density reactivity coefficients for ACR (e.g., high-temperature coolant void reactivity, fuel-temperature coefficient, fuel/coolant temperature reactivity, moderator temperature coefficient, etc.)

High temperature experiments (up to 300°C) will be performed using CANFLEX LVRF in existing ZED-2 hot channels to measure the temperature effect on fuel-temperature reactivity and coolant void reactivity. The measurements will be repeated using MOX bundles representing simulated irradiated ACR fuel.

Flux-map buckling measurements will be performed at two or more moderator temperatures in the range of 10°C to 40°C (nominal) using a uniform lattice of 0.95% SEU fuel at the ACR spacing at three coolant conditions—H_2O, D_2O and air.

2.3.2.5 Moderator Poison Experiments

Flux-map experiments will be performed with 0.95% SEU fuel in a lattice arranged at spacing representative of an ACR lattice with various boron and/or gadolinium concentrations in the moderator. Measurements will be performed using both H_2O and air coolant and with a moderator temperature range of about 10°C to 40°C. These measurements will provide data on the ability of reactor-physics cell codes and core codes to calculate the reactivity effect of poison in the moderator of an ACR-type lattice.

Substitution experiments using MOX bundles simulating irradiated ACR fuel will be performed with clean moderator (at room temperature) and with the moderator poisoned (at room temperature).

2.3.2.6 Downgraded Moderator Experiments

Two or more flux-map experiments will be performed with 0.95% SEU fuel in a lattice arranged at representative ACR spacing with moderator purities ranging between 99.8 wt% D_2O and 99.1 wt% D_2O using H_2O and air coolant. The isotopic purities are the nominal operating range for the moderator in the ZED-2 facility and the tests will be scheduled to take advantage of the rate of downgrading that occurs as a result of normal facility operation. These measurements will provide data to verify the ability of reactor-physics cell codes to calculate the reactivity effect of downgraded moderator/reflector in an ACR-type lattice.

2.3.2.7 Fine-Structure Flux Distribution

Fine-structure measurements in a CANFLEX LVRF demountable bundle positioned in the centre of a hexagonal lattice will be performed. The measurements will use three coolant conditions (H_2O, D_2O, and air) and will be performed at room temperature. Seven ACR-type assemblies will occupy the seven centre positions in a 0.95% SEU driver lattice. The measurements will be repeated using available MOX fuel to simulate irradiated SEU fuel. These measurements will provide data to verify the ability of WIMS-IST to calculate detailed reaction rates and neutron distributions in ACR-type lattice cells.

2.3.2.8 Checkerboard Lattice Measurements

Measurements will be performed with various coolant conditions in a checkerboard region corresponding to normal operation and various loss-of-coolant accident (LOCA) scenarios. A square lattice (7.1 moderator-to-fuel ratio) arranged as a checkerboard of CANFLEX LVRF and MOX fuel will be studied. An outer region of 0.95% SEU fuel rods will drive the checkerboard region. These measurements will provide data to verify the ability of RFSP/WIMS-IST to calculate the spatial-flux distribution for a highly heterogeneous ACR-type lattice. These measurements are similar to benchmark measurements that have been performed for the current CANDU lattice pitch.

2.3.2.9 Control Device Measurements

An ACR in-core absorber (see Section 4.1) will be inserted into the square checkerboard lattice described in Section 2.3.2.8 and fine-structure plus flux-mapping experiments will be performed at room temperature.

A rod-drop experiment will be performed using a square SEU lattice at a spacing representative of the ACR core. These measurements will determine the component of the delayed neutron fraction due to photoneutrons for ACR and measure the reactivity worth of an ACR-type shut-off rod.

A uniform square lattice of SEU assemblies would then be used for period measurements, one measurement with no absorber in the lattice and one with the absorber in the lattice. These measurements will provide data to verify the ability of DRAGON-IST to calculate reactivity-device cell parameters used in RFSP-IST.

2.4 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 2-4 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 that will be used to validate extensions to ELESTRES.

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 to extend the validation of the ELESTRES code for ACR.

Table 2-4
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

2.5 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 of 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 existing 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.

2.6 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. This will address validation of the code for the design basis accidents associated with spontaneous pressure tube failure.

As part of its ongoing severe accident research program, AECL and the CANDU Owners Group (COG) are carrying out an experimental program to study the pressure transients associated with high-temperature channel debris interaction with the moderator (the Molten Fuel Moderator Interaction (MFMI) program). This phenomenon is associated with two classes of severe single channel accident sequences: inlet feeder stagnation break and severe channel flow blockage (>98.7% of the flow area).

The ongoing MFMI program will provide data on the type of fuel/coolant interaction and the associated pressure transients that will be directly applicable to the ACR design. Additional tests will be carried out using ACR pressure tube sections to demonstrate that the changes in the ACR fuel channel design have minimal impact on the progression and consequences of this type of event.

2.7 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 (Figure 2). 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 in a larger scale vessel within the test 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.

3. FUEL DESIGN VERIFICATION

The ACR-700 will use a 43-element CANFLEX-MkV fuel bundle design. The CANFLEX fuel bundle design has undergone extensive development testing and has been qualified for natural uranium fuel in a CANDU 6 reactor.

The CANFLEX-ACR fuel design uses the latest CANFLEX external bundle design with the internal fuel pellet design optimized to match the ACR core design and the target fuel performance. The ability of the ACR fuel to meet its design requirements will be verified through a comprehensive qualification program. This program covers three performance areas: mechanical performance, irradiation performance and thermalhydraulic performance.

Fuel design verification is achieved through a combination of analysis and testing. The following sections highlight the planned testing that is part of the fuel qualification program. Note that all of the planned tests are related to safety; many tests are planned to verify anticipated improvements in operating performance, and are included here simply for convenience and completeness.

3.1 Fuel Mechanical Performance

The ACR fuel must meet the mechanical design requirements imposed by the interface between the fuel and the reactor coolant system (and fuel channels) and the fuel handling systems. These requirements are to ensure that fuel is not damaged as a result of normal operating and anticipated operational occurrences, and to ensure that the fuel is compatible with the reactor systems/components that it normally encounters and does not cause damage to those components.

There are at least 22 individual design requirements to ensure that there are no systematic fuel failures associated with interfacing systems, and at least four requirements to ensure that there is sufficient allowance in the fuel channel and fuel handling components to protect against damage due to interaction with fuel bundles.

The mechanical performance is verified, in part, through a number of individual test programs conducted out-reactor using non-irradiated fuel. The tests can be divided into two groups: mechanical/materials performance and dimensional compatibility. The test objectives and conditions are outlined in Tables 3.1-1 and 3.1-2.

The mechanical/materials tests will be carried out in a number of test facilities at AECL's Sheridan Park engineering laboratory. These will include an acrylic flow visualization facility that will also be used for fuel channel component development and a test loop that will include a full-scale ACR pressure tube and end-fitting, and will be capable of operation at ACR coolant conditions.

Testing has been initiated for some of the longer-term autoclave tests and the developmental test facilities are currently being assembled. A pilot order of ACR fuel bundles containing only natural uranium is schedule to be fabricated by 2004. These fuel bundles will be used in the out-reactor test program for the tests outlined in Tables 3.1-1 and 3.1-2.

Table 3-1
Fuel Bundle Mechanical/Materials Tests

Test	Objective	Tests Description
Fuel Bundle Strength	Verify adequate strength under axial loads for normal and abnormal support.	Measure loads in a 12-bundle string in a full-scale test loop under normal coolant operating conditions. Tests employ side-stops used to separate fuel bundles during refuelling.
Refuelling Impact	Verify ability to withstand impact loads during refuelling.	Measure fuel bundle distortion for acceleration forces and distances representative of refuelling conditions in flow test facility.
Endurance and Fretting in Axial Flow	Verify that fretting of the fuel bundle in a pressure tube is within design allowances. Verify the integrity of the fuel bundle welds and end-plate with respect to fatigue.	Measure fretting in long-term tests conducted in a full-scale ACR fuel channel test loop under normal coolant operating conditions.
Endurance in Cross-Flow	Verify ability of the fuel bundle to withstand coolant cross-flow forces experienced when the fuel bundle transits the end fitting during refuelling.	Measure fuel bundle integrity in tests conducted in a full-scale fuel channel and end-fitting under ACR coolant flow conditions.
Longitudinal Ridging	Extend the validation of the correlation for longitudinal ridging of the collapsible ACR fuel element cladding.	Measure critical collapse pressures in autoclave tests.
Cladding Corrosion	Verify the validity of correlations for cladding corrosion rates under ACR coolant conditions.	Measure cladding oxidation in autoclave tests under ACR coolant conditions (pH and LiOH level).
Sliding Wear	Verify that the wear rate of the pressure tube and fuel element bearing pads, due to periodic sliding of the fuel bundles during refuelling, is within design allowances.	Cyclic tests in a wear rate facility

Table 3-2
Fuel Bundle Dimensional Tests

Test	Objective	Tests Description
Fuelling Machine Compatibility	Verify that the fuel bundle geometry is compatible with the fuelling machine design.	Tests of fuel handling using a prototype fueling machine coupled to a fuel channel test loop operating at ACR coolant conditions.
Spacer Interlocking	Verify the absence of spacer interlocking when the fuel bundle passes through the reactor.	Measurements of fuel bundle configuration during tests of fuel bundle compatibility with the fuelling machine.
Bent Tube Gauge	Verify the ability of the fuel bundle to traverse the rolled-joint region of the fuel channel and an aged fuel channel (including sag).	A bent tube gauge will be designed to reflect dimensions of the rolled-joint region of the end fitting, the diametral creep and axial sag of an aged fuel channel, and the dimensions of an irradiated fuel bundle. This gauge will be used for fuel bundle acceptance testing.

3.2 Fuel Irradiation Performance

The ACR fuel design is based on the CANDU fuel design used in the operating CANDU 6 reactors plus the CANFLEX fuel design that has been qualified for use in the CANDU 6 reactors. In addition, AECL has a substantial database of irradiation tests on fuel that covers the range of compositions and performance levels required for the ACR fuel.

A limited number of irradiation tests are planned to verify the anticipated irradiation performance of the ACR fuel design. There are two groups of tests: high-power performance envelope tests, and power-ramp tests.

The high-power performance envelope tests will verify the overall ‘health’ of ACR fuel bundles under normal operating conditions. Prototype ACR fuel bundles will be irradiated in the NRU research reactor at power levels typical of ACR operating conditions to burnup levels in excess of the maximum target burnup.

Post-irradiation examination of the prototype fuel bundles will be performed to measure parameters including fission gas release, fuel element deformation (bowing, cladding ridge strain, etc.) and UO_2 grain growth. Data from the post-irradiation examination will be included in the databases used to validate the fuel design codes including ELESTRES (see Section 2.4).

The power-ramp tests will be used to verify the performance of ACR fuel under transient power conditions, including the power envelopes that will occur during normal refuelling operations. The power-ramp tests will be conducted using the full fuel bundle and using ‘demountable’ ACR fuel bundles. These are specially designed test assemblies in which the CANFLEX fuel elements are held in place by removable end-plate assemblies such that the bundle can be disassembled and individual elements can be removed and replaced during an irradiation program (Figure 3). This provides flexibility to test for failure limits on an individual element basis. Post-irradiation examination of the elements used in the power-ramp tests will also provide extra data for the validation of fuel design codes, including ELESTRES.

Irradiation of both prototype ACR fuel bundles and demountable element assemblies is scheduled to begin in the NRU reactor in 2004.

3.3 Fuel Thermalhydraulic Performance

The fuel design must be compatible with the thermalhydraulic design of the reactor coolant system. The performance requirements include:

- The pressure drop across the fuel bundle must be compatible with the reactor coolant system design allowance,
- Adequate thermalhydraulic performance under normal operating conditions,
- Sufficient margin on critical channel power for expected range of axial flux shapes, flows and channel flow areas (subject to increase during channel life due to radial pressure tube creep).

There are three sets of tests planned to obtain data to support correlations for the ACR fuel thermalhydraulic performance: pressure drop, critical heat flux and post-dryout heat transfer. As noted in Section 2.2, data from these tests will be used to validate the correlations used in the thermalhydraulic codes.

Pressure drop tests will be performed in two different facilities. The single-phase pressure drop will be measured in the Freon test facility (MR-3) located at AECL's Chalk River Laboratory using a full-scale channel with 12 prototype ACR fuel bundles. In this facility, the alignment of the fuel bundles can be controlled to measure the impact of bundle orientation on the pressure drop. The resulting correlations will be used in the thermalhydraulics analysis codes (NUCIRC and CATHENA). The two-phase pressure drop will be measured in the Stern Laboratories (Hamilton, Ontario) test loop as part of the water critical heat flux tests. These tests will use an electrically heated full-scale fuel bundle simulator in a full-scale fuel channel with light water coolant at ACR operating conditions. The design of the fuel element simulator is equivalent to all of the fuel bundles and fuel elements in a fully aligned orientation. The Freon tests will provide the data required to determine the impact of bundle miss-alignment.

The critical heat flux (CHF) of the ACR fuel design will also be measured in both Freon and light water. Both sets of tests will be conducted using electrically heated full-scale fuel bundle simulators in full-scale channels in the same facilities used for the pressure drop tests. The water CHF tests will use a fuel element simulator with an axial and radial power profile that is typical of a high-power ACR fuel channel. The fuel element simulators will be instrumented with sliding thermocouples so that the precise location of the critical heat flux within the fuel bundle length and circumference can be determined. The CHF tests will be conducted for a range of ACR coolant flow conditions.

During the pressure tube lifetime, the pressure tube undergoes diametral creep (see Section 5.4). This results in increases in the coolant flow area within the fuel channel. The changes are non-uniform axially because the creep rate depends on the local pressure tube temperature. The increased flow area will be located non-uniformly at the top of the channel, because gravity will locate the fuel bundles at the bottom of the channel. The water CHF tests will be conducted with three different fuel channel internal dimensions to simulate beginning, mid-point and end-of-life radial creep conditions.

A set of CHF tests will also be conducted in the Freon thermalhydraulics test loop. In this facility, the fuel element simulator will have the additional flexibility of variation of the power to the individual fuel elements. This allows testing the sensitivity of the CHF to variations in the radial power distributions within a fuel channel. Like the water CHF fuel simulator, the Freon fuel simulator will be instrumented with numerous sliding thermocouples so that the precise location of the critical heat flux can be determined.

In addition to tests to examine thermalhydraulic performance under normal operating conditions, a series of tests will be conducted to measure the post-dryout behaviour of the fuel. The tests will be conducted in Freon to permit high overpower levels to be tested. The use of sliding thermocouples in fuel element simulators will permit determination of cladding temperatures in dry patches following dryout and to determine the size and growth of dry patches. Data from these tests will be used to validate CATHENA predictions of cladding temperatures under accident conditions.

Tests to provide data for pressure drop correlations are scheduled for 2003/04.

The design and fabrication of the fuel element simulators for the thermalhydraulics tests in the MR-3 and Stern Laboratories loops has commenced. Tests in these facilities are scheduled to commence in late 2004.

4. SAFETY SYSTEM PERFORMANCE VERIFICATION

The design of the ACR-700 includes four Special Safety Systems: Shutdown System 1, Shutdown System 2, Emergency Core Cooling and Containment. While all four systems in the ACR-700 are based on the design in the reference CANDU 6 plant, each of the first three systems includes design modifications that require additional verification testing.

4.1 Shutdown System 1

Shutdown System 1 (SDS 1) consists of an array of 20 absorbers that are normally positioned above the core. These absorbers can be released to drop under the influence of gravity into the low-pressure heavy water moderator, guided to locations between the horizontal calandria tubes. The reactivity worth of the absorbers is such that the negative reactivity insertion is sufficiently rapid to fully shut down the core for all anticipated transients.

The design of the Shutdown System 1 is conceptually the same as that used in the reference CANDU 6 design, but the physical design of the drives, absorbers and guide structures is different, to accommodate for the tighter lattice spacing in the ACR core (Figure 4).

The performance requirements of the Shutdown System 1 are not as demanding as those for the reference CANDU 6 design. Because the ACR fuel has a negative coolant void reactivity coefficient, the ACR shutdown system is not required to have as rapid a negative reactivity insertion rate. In addition, an in-core loss-of-coolant event associated with any potential fuel channel failure will displace heavy water with light water and introduce negative reactivity in the core. This reduces the requirements on the shutdown system reactivity depth in the event of damage to some of the absorber guide structures due to the fuel channel failure.

4.1.1 Feasibility Tests

Feasibility tests have been carried out on several prototype shutdown absorber/guide mechanism designs in support of the design of the new absorber components. These tests were conducted using full-scale prototype components in the full-height drop tank facility located at AECL's Sheridan Park engineering laboratory. The tests included a preliminary evaluation of seismic performance and the impact of water cross-flows. All tested design concepts almost met or exceeded the demanding CANDU 6 requirements for drop time performance. This provides substantial confidence that the final ACR shutoff system design will exceed the ACR performance requirements.

In addition to the changes in the mechanical design of the shutoff system absorbers, the design of the shutoff unit drive mechanism and electromagnetic clutch will be updated. This is required to accommodate changes in the available supplies of some drive mechanism components and to fit in to the tighter deck space. A series of feasibility tests will be conducted in 2003 to support the selection of new drive mechanism components.

4.1.2 Verification Tests

The design ACR SDS 1 shutoff units will be fully verified. Shut-off units will be manufactured to the ACR design in 2004 and test in 2005. These tests will be conducted on full-scale shutoff system units manufactured to formal design specifications. The tests will use the full-scale drop facility at Sheridan Park. The tests will verify that the absorbers will drop freely under conditions of normal moderator circulation and under transient conditions, including flows associated with action of Shutdown System 2.

The neutronic performance of the shutoff units will be verified in conjunction with physics code validation tests that are planned to be executed in the ZED-2 critical facility (see Section 2.3.2.9).

The reactivity depth of the shutoff rods will be verified experimentally during the low power critical commissioning phase of the reactor construction. At that time, the total negative reactivity shutdown depth of all the rods will be verified. The speed and effectiveness of SDS 1 will also be verified during the commissioning phase. In these tests, 18 rods (all but the 2 most effective rods) will be allowed to fall into the reactor core following a trip signal and the neutron flux will be measured at pre-determined locations in the reactor core. The results will be compared to simulation results of the same event.

4.2 Shutdown System 2

Shutdown System 2 (SDS 2) consists of a liquid poison injection system that is capable of rapidly injecting a gadolinium nitrate solution into the heavy water moderator, thereby rapidly shutting the reactor down and maintaining it in safe shutdown. The liquid poison is introduced into the moderator from a series of jet nozzles located along tubes that traverse the core horizontally. The ACR design is conceptually the same as that in the reference CANDU 6 reactor. The only significant difference is that in the ACR design the injection tubes are relocated from interstitial sites in the reactor lattice to sites at the top and bottom of the reactor lattice (Figure 5). This relocation places the initial poison injection in the reflector region of the core where the thermal flux peaks. It also has the benefit of eliminating any possibility of contact between the poison injection tubes and the fuel channels in the tighter ACR lattice (associated with fuel channel sag during their lifetime).

Because the ACR core is smaller that the CANDU 6 core, the relocation of the injection tubes to the periphery of the core is not expected to have a profound impact on the performance of the liquid injection system. The developed jet length of the CANDU 6 injection nozzle design (to be preserved in the ACR design) is such that the ACR injection tubes will still provide nearly equivalent core coverage during the injection phase. As is the case for SDS 1, the performance requirements for negative reactivity insertion time for SDS 2 are less demanding than for the reference CANDU design, so that small differences in jet dispersion areas will not be important.

4.2.1 Verification Tests

The design performance of the ACR poison injection system will be verified in a series of tests planned for 2005. These tests will be conducted at AECL's Sheridan Park engineering laboratory and will use a single, full-scale injection tube with injection into a water tank containing a simulated lattice of ACR calandria tubes. These tests will verify the development of the initial jet dispersion into the ACR lattice configuration.

A second series of tests will be conducted using the Moderator Test Facility (located at AECL's Chalk River Laboratory). This facility will include a 1/3 scale ACR core (calandria and electrically heated fuel channels). Scaled liquid poison injection tubes will be included in this facility and tests will be conducted to obtain data on the rate of dispersion of the poison throughout the core. The tests will also verify predictions of the impact of the moderator flow patterns on poison dispersion.

The reactivity depth of the poison injection will be verified experimentally during the low power critical commissioning phase of the reactor construction. At that time, the total negative reactivity shutdown depth will be verified.

4.3 Emergency Core Cooling System

The ACR-700 Emergency Core Cooling (ECC) System design is conceptually the same as that of the reference CANDU 6 design. Both designs include a short-term high-pressure injection system and a long-term, low-pressure recirculating heat removal system. The high-pressure injection is supplied from water tanks using compressed gas as a driving force. The low-pressure recirculation flow is supplied by pumps (located outside of the containment building). This system collects water from sumps in the containment building and passes it through heat exchangers before returning it to the reactor coolant system. Both high-pressure and low-pressure flows are injected into the reactor coolant system through piping connected to the headers located at the ends of the reactor core.

There are a number of significant detailed design differences between the ACR-700 ECC system and the CANDU 6 system including:

- The ACR design replaces valves with passive one-way rupture discs on the interface between the high-pressure injection system and the reactor coolant system.
- The ACR design provides for emergency coolant injection only into the inlet headers (both inlet and outlet headers for CANDU 6) (See Figure 1).
- The ACR design includes a valved interconnect line between outlet headers.
- The ACR design includes an initial, gravity fed, water supply to the containment sumps from an elevated reserve water tank (to ensure emergency coolant pump suction head).

4.3.1 ECC Rupture Discs

One-way rupture discs are provided on the lines connected to the reactor inlet headers to isolate the reactor coolant system from the ECC system (Figure 6). The one-way rupture discs can withstand the high differential pressure that is normally present in the reverse direction (reactor coolant to emergency coolant), but will open at a differential pressure in the forward direction (emergency coolant to reactor coolant).

The on-way rupture disc design was developed for the CANDU 9 reactor design and was subjected to extensive development and qualification testing. The ACR design application of this component is within the qualification range. There may be a requirement for incremental testing to verify the design of the sealing elements in the disc holder assembly, depending on the results of analysis of the temperatures that the disc assembly will experience during normal operation. If required, these tests will be conducted in 2004.

4.3.2 ECC Performance

The performance of the ECC system design is verified by analysis using the transient thermalhydraulics code CATHENA (see Section 2.1). This code has been validated for application to safety analysis of the CANDU design, including the ECC system performance. The validation has been carried out on a phenomenological basis so that the code can be applied to different designs within the range of validated conditions and parameters. For the ACR the validation will be extended to include the new features of the ACR design (inlet header injection and outlet header interconnect line) as discussed in Section 2.1.6.

4.4 Containment

The Containment system does not include any ACR-700 design-specific features at the system level. However, there may be requirements for environmental qualification testing of individual containment system components (e.g., cable penetrations). These are not included in the R&D program to support the Basic Engineering Program for the ACR-700 design as they are usually certified by the manufacturer when purchased to specifications.

The Containment system will include passive hydrogen recombiners to provide for long-term removal of hydrogen, if necessary. These have been developed and qualified by AECL to meet generic CANDU Containment system requirements.

5. FUEL CHANNEL DESIGN VERIFICATION

The design of the ACR fuel channel is conceptually the same as the design of the fuel channel in the reference CANDU 6 reactor and the design is backed by the established technology base for the fuel channel [5]. The ACR design incorporates a number of differences (mostly dimensional) to accommodate design requirements for higher coolant pressure and temperature, and a tighter lattice pitch. This leads to a requirement for limited incremental R&D to verify the ACR design.

Again, not all R&D activities described here are related to safety; many tests are planned to verify anticipated improvements in operating performance or lifetime, and are included here simply for convenience and completeness.

5.1 Pressure Tube Performance

The ACR-700 fuel channel is based on the use of a cold-worked Zr-2.5Nb pressure tube. This is the standard material that has been successfully used in the current generation of CANDU reactors. For the ACR design, the pressure tube thickness has been increased from 4.2 mm to 6.5 mm to accommodate the increase in reactor coolant pressure.

In the ACR-700 design the coolant temperature across the channel increases from 278°C to 325°C in comparison with the coolant temperature increase across a CANDU 6 fuel channel from 266°C to 310°C.

The ACR pressure tubes will experience aging associated with thermal and radiation induced changes and the resulting impacts on such mechanical characteristics as mechanical properties, deformation, corrosion and hydrogen uptake (Sections 5.3 and 5.4).

5.2 Pressure Tube Fabrication

The pressure tubes for the ACR will be fabricated using the same extrusion and drawing process that has been successfully qualified for the current CANDU plants. To verify the fabrication process for the thicker ACR pressure tubes, three lots of prototype pressure tubes will be manufactured to the ACR design specifications. Samples from these tubes will be measured to verify their metallurgical properties and conformance to design requirements. Samples from these tubes will also be used in the materials performance tests described below.

The prototype ACR pressure tubes will be used in other qualification test programs as required.

The first set of prototype tubes was fabricated in 2002. The second set of prototype tubes is scheduled for fabrication in late 2003 and the third set of prototype tubes will be manufactured in 2004.

5.3 Corrosion and Hydrogen Uptake

During normal operation, the Zr-2.5Nb pressure tube is subject to corrosion and the formation of a ZrO_2 layer on the internal surface of the tube. The corrosion rate depends on the coolant conditions and the pressure tube temperature. During the corrosion process, a small fraction of hydrogen release from the water is absorbed into the pressure tube body. The rate of hydrogen uptake is important because, at sufficiently high concentrations it can lead to the precipitation of zirconium hydrides that can promote delayed hydride cracking of the pressure tube material

(Section 5.5). AECL has established predictive tools to assess the rate of pressure tube corrosion and hydrogen uptake based on an extensive database obtained from measures of pressure tubes in operating CANDU reactors and tests conducted in research facilities. The existing database includes measurements on Zr-2.5Nb samples at temperatures up to 325°C.

Based on the results of development tests and laboratory studies, the specifications for the minor constituents of the Zr-2.5Nb alloy for the ACR pressure tubes have been more narrowly established, while remaining within the permissible range for the material. It is expected that the tighter specifications will reduce the range of variability observed in hydrogen uptake rates in operating reactors and will also lead to a substantial decrease in the average rate of hydrogen uptake.

A test program has been established to extend the database for the corrosion and hydrogen uptake models to fully cover the range of ACR operating conditions. This test program will use samples obtained from the prototype ACR pressure tubes to obtain data to verify the anticipated benefits of the tighter chemical specifications.

Irradiation of pressure tube coupons was initiated in the Halden research reactor in late 2002. A selection of these coupons will be examined annually to obtain data on the corrosion rate and hydrogen uptake rate. The tests are being carried out at temperatures that span the maximum ACR fuel channel temperature. The tests are being conducted with a heavy water coolant to provide increased measurement accuracy of hydrogen uptake and for consistent comparison with the existing database obtained for conventional CANDU materials. There is no significant isotope effect (D vs. H) that affects the uptake rate.

5.4 Deformation

Neutron irradiation at reactor operating temperatures induces irradiation creep (slow deformation cause by applied stress) and irradiation growth (deformation in the absence of applied stress) of zirconium components (pressure tube and calandria tube). Creep and growth contribute to the elongation of the pressure tubes and to their diametral expansion. The calandria tubes experience little change in length of diameter because they operate at low stress and at low temperature, and growth rates of the calandria tube material are very low. The pressure tubes, which operate at high temperatures and high stresses, exhibit creep and growth over the channel lifetime.

AECL has developed models and tools to predict the rate of deformation of cold-worked Zr-2.5Nb pressure tubes based on an extensive database of measurements of tubes from operating reactors and tests conducted in research irradiation facilities. The ACR development program has established an experimental program to extend the existing database to the higher temperatures experienced at the outlet end of the ACR fuel channel.

The ACR deformation program consists of two main components. Irradiation tests are planned in the NRU reactor at Chalk River Laboratory and the higher flux OSIRIS reactor in France. The test conditions will include temperatures in excess of the maximum ACR coolant temperature. The two sets of tests will provide data for both the thermal and irradiation component of the deformation and extend the models to ACR conditions. Irradiation of the test specimens is scheduled to commence in 2004. Data on deformation rates will be obtained from annual measurement campaigns. It is anticipated that data obtained after the first few years of irradiation will provide a basis for deformation predictions at higher temperatures, but tests will continue until sufficient data has been accumulated to fully support the extended models.

5.5 Leak-Before-Break

Although pressure tube rupture is an assumed accident for the purposes of licensing and the design of safety systems, avoidance of rupture is a key objective of fuel channel and interfacing system design. For ACR, such rupture avoidance is based upon a number of factors related to design, pressure tube manufacturing standards, assembly requirements, operating and maintenance standards and inspections and Fitness-for-Service Assessments. Many of these factors are now covered by CAN/CSA Standards in the N285 series.

In addition to design to reduce the potential for tube ruptures, a further defence-in-depth is applied. This is the use of the leak detection capability of the annulus gas system and the application of operating procedures that will result in the shutdown and depressurization of the reactor before a leaking crack can develop into a tube rupture. The demonstration by analysis that this system and the associated operating procedures for the leak event will be effective in preventing pressure tube rupture is termed leak-before-break (LBB) (see Reference 4).

The analysis of leak-before-break involves the demonstration of crack stability for all the conditions of a growing crack during the complete time period from the time at which the crack first leaks to the time at which the reactor is cold and depressurized. This is carried out using a sequence-of-events analysis. The sequence-of-events analysis calculates, for each time period, an upper bound of the crack size (length) and compares that crack size with the crack size that could potentially be unstable for the channel conditions of temperature and pressure if the material toughness were at the lower bound of observed material toughness. The inputs required for the analysis include data on the delayed hydride cracking (DHC) velocity and the stress intensity factors (K_{IH}) require to initiate delayed hydride cracking and the leak rates associated with developing cracks.

Based on conservative extrapolations of the existing database plus improvements in the leak detection capabilities of the annulus gas system, preliminary analysis of the ACR design shows increased margins for leak-before-break detection in the ACR. However, an R&D program has been established to extend the database to include tests on prototype ACR pressure tube material at higher temperatures.

Laboratory experiments on the delayed hydride cracking velocity and K_{IH} have been initiated and preliminary results are available on hydrided ACR pressure tube material at temperatures above 310°C. These results have shown that extrapolations of the lower temperature data are conservative (the delayed hydride cracking velocity decreases at the higher temperatures).

The thicker ACR pressure tube will lead to changes in the leak rate as a function of crack development. An existing test facility at Chalk River Laboratory is being upgraded to measure leak rates in cracks in prototype ACR pressure tubes at the ACR coolant conditions. Tests are planned for 2004 and data from these tests will be used in the ACR leak-before-break analysis.

5.6 Pressure Tube/End-Fitting Rolled Joint

In the fuel channel, the pressure tube is connected to the end fittings at both ends using a mechanical rolled joint. This joint is created by roll expanding the Zr-2.5Nb pressure tube into a series of grooves in the 403 stainless steel end fitting. The roll-expanded joints have been qualified to provide a strong, leak-tight connection in operating CANDU reactors.

In the ACR design, the pressure tube is thicker (6.5 mm vs. 4.2 mm) than the reference CANDU pressure tube, and the space available in the rolled joint region is smaller than that in the reference CANDU design to accommodate the smaller lattice pitch. As a consequence of these differences, a program has been established to qualify the procedures for producing a rolled joint for the ACR design. This program is in support of design optimization and verification of procedures for production use, and includes both test and analysis components.

5.6.1 Feasibility Tests

A series of feasibility tests are in progress to assist in the design of the pressure tube joint to the end fitting and to optimize the rolling procedure. The tests combined with engineering analysis will verify the detailed design of the ACR end fitting in the rolled joint region and establish the rolling procedures to achieve the design requirements for the joint performance (strength, leak tightness and residual stresses). Initial testing is in progress on rolled joints using sections of prototype ACR pressure tubes.

5.6.2 Qualification Tests

A series of tests will be carried out to qualify the detailed design and procedures for fabricating the pressure tube rolled joint. During qualification tests, joints will be fabricated using fabrication variables from the extremes of the preferred ranges defined during the development program. The joints will be tested to verify that acceptable rolled joint performance is achieved under specified production conditions. The tooling and procedures used during production fabrication will be used to fabricate these joints.

5.7 Channel Closure

The function of the ACR channel closure is to seal the end of the fuel channel assembly and to provide on-power access for refuelling by the fuelling machine. During fuel changing, it is necessary to remove the channel closure. This is accomplished by the fueling machine, which connects onto the end fitting, removes the channel closure and stores it in its magazine. Following the fuelling operations, the channel closure is re-installed in the end fitting, and then the fuelling machine is disconnected. These refuelling operations are carried out at reactor coolant system and fuelling machine temperatures and pressures.

The basis of the ACR channel closure design is a flexible metallic element that makes a face seal against a shoulder of a groove in the end fitting. The seal element is supported and retained by the body of the channel closure, which is held in position by jaws engaging a groove in the end fitting.

A feasibility test program is in progress to evaluate options for the closure design and to perform preliminary verification tests. Once the design has been selected, a comprehensive series of development tests, combined with analysis, are planned to verify the performance of the closure. The planned developmental tests will address the following performance requirements:

- Debris tolerance
- Installation tolerances

- Endurance
- Cycle testing
- Safety lock testing
- Leak testing

The channel closure design will be formally verified in conjunction with the verification of the fuelling machine design. The closure will be installed in the prototype end fitting attached to a test loop operating at ACR coolant conditions. The test loop will be located at AECL's Sheridan Park engineering laboratory.

For the formal verification tests, the prototype channel closure will be fabricated to final engineering drawings and specifications. The verification tests will include all of the performance tests outlined above. In addition, the testing will verify that the closure meets all interfacing system requirements associated with the fuel channel and the fuelling machine (Section 6).

5.8 Channel End Fitting

The fuel channel is terminated at both ends by stainless steel end fittings that are joined to the pressure tube by rolled joints (Section 5.6). The inlet and outlet end fittings are joined to the reactor coolant system with their respective inlet and outlet feeder pipes. The design of the end fitting interfaces with the design of the channel closure and rolled joint, and the end fitting design will be verified in conjunction with analysis and testing of those components.

In addition to the pressure boundary components, the fuel channel includes internal components that influence the thermalhydraulic characteristics of the heat transport system and interface with the fuel handling system. The R&D program for these components includes a design development phase, which is now in progress, and a verification phase. Priority in the development phase has been given to those components that will reside within the fuel channel (the liner tube and the shield plug). The shield plug is normally located inside the end fitting and supports the fuel column during normal operation (the shield plug is removed by the fueling machine during refuelling), and the liner tube is located inside the end fitting at the feeder connection site and contains perforations to pass the coolant from the feeder pipe to the fuel channel. An acrylic, flow visualization test facility has been assembled to measure the low-temperature hydraulic characteristics of prototype designs of these components. Design assist testing will be carried out in 2003/04 to finalize the design of these components to meet requirements for pressure drop and other flow parameters. Subsequently a high-pressure facility will be constructed for tests to verify the performance of these components at full coolant operating conditions.

Testing of the internal channel components is linked to testing of the fuel performance (Table 3-1) because the channel components influence the thermalhydraulic forces experience by the fuel. Hence some of the planned fuel performance testing will be carried out in the facilities used for the fuel channel component design development.

6. OTHER COMPONENTS

The ACR design includes a number of components that differ from the reference CANDU design and which require design verification (beyond those associated with the safety systems described above). The design of these components does not have a major impact on the safety analyses for design basis accidents, and the planned R&D to support their development and design verification is not discussed here.

One of the most significant of these components is the calandria tube (Sections 6.1 and 6.2). Other important components are those associated with the fuel handling systems and, in particular, the fuelling machine head and the channel components that interface with the fuelling machine head. These components/assemblies include the fuelling machine snout (which provides the interface between the fuelling machine and the fuel channel) and the fuelling machine ram (which manipulates the fuel and channel components).

The design of the components/assemblies of the fueling machine will be tested individually. Because some of the fuel handling components must interface with the fuel channel components (especially the fueling machine ram), development and verification testing of the fuel handling components will be linked with development and verification testing of those fuel channel components and will use the same test facilities.

Full verification of the fuelling machine design is planned after completion of the basic engineering phase of the ACR program and will be performed on fuel handling components fabricated to final design specifications. This verification will be carried out at Sheridan Park engineering test facilities that will include a loop with a full-scale ACR pressure tube operating at ACR coolant conditions.

6.1 Calandria Tube

The ACR-700 reactor core contains 284 fuel channels that are located horizontally in the calandria vessel of the reactor assembly. Each fuel channel includes a thin-walled Zircaloy-4 calandria tube that separates the heavy water moderator in the calandria vessel from a gas annulus that surrounds each pressure tube (containing the fuel and reactor coolant). The ACR calandria tube has a large diameter and greater thickness than the reference CANDU 6 calandria tube. As part of the ACR development program, prototype ACR calandria tubes will be manufactured and tested to verify that the manufacturing process selected for the ACR design will meet the design requirements.

The calandria tube is not a component in the reactor coolant system boundary and is not formally designed to meet full pressure vessel requirements. However, it is relatively stronger that current CANDU calandria tubes and is expected to remain intact following a spontaneous pressure tube failure[2]. While a combined pressure tube/calandria tube failure will still be assumed in accident analysis, the stronger calandria tube will reduce the frequency of such an event.

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2 If the calandria tube does not fail, the bellows, that seal the end fitting to the reactor end-shield, may fail leading to a leak from the pressurized annulus. Such a leak will be a small LOCA that is well within the design capabilities of the emergency core cooling system.

During a pressure tube failure event the calandria tube would be exposed to high temperature water at operating system pressure. A test program will be carried out in a calandria tube burst test facility at CRL to measure the delayed creep rupture strength of the prototype ACR calandria tubes as part of the manufacturing process verification. The burst tests will employ short sections of prototype calandria tubes. It is anticipated that the tests will provide sufficient data to support analytic predictions that the time delay before calandria tube failure is sufficient to allow for detection of the pressure tube failure and safe shutdown of the reactor prior to calandria tube failure.

To provide additional confidence in the performance of the calandria tube under accident conditions, a limited number of full-scale tests will be conducted. These tests will use full-scale prototype ACR pressure tubes and calandria tubes in a facility at the Stern Laboratories. Similar tests have been conducted in that facility for reference CANDU fuel channels. The tests are scheduled for 2006 after the fabrication process for full-scale prototype calandria tubes has been verified by the small-scale burst tests

6.2 Calandria Tube/End-Shield Joint

A mechanical joint connects each end of the calandria tubes to the respective end-shield tube-sheet. In the reference CANDU 6 design, the calandria tube is belled outwards slightly at the ends and the calandria tube is rolled into grooves in the end-shield using a stainless-steel insert to create a ‘sandwich’ with the calandria tube material in the centre. The resulting joint must demonstrate sufficient strength with respect to pullout failure and must exhibit sufficient leak tightness with regard to loss of moderator water into the fuel channel annulus.

To accommodate the tighter ACR lattice pitch, the ends of the ACR calandria tube are not belled. The joint between the calandria tube and the end-shield tube sheet is made by directly rolling the calandria tube into the tube-sheet.

6.2.1 Feasibility Tests

A series of feasibility tests are in progress to assist in the design of the calandria tube joint to the end-shield (Figure 7). These tests are being carried out at AECL’s Sheridan Park engineering laboratory. The preliminary tests have demonstrated that a satisfactory rolled joint can be fabricated by direct rolling of the calandria tube into a tube sheet. Further testing is in progress to optimize the design of the joint and the rolling process.

6.2.2 Qualification Tests

A series of tests will be carried out to qualify the detailed design and procedures for fabricating the calandria tube rolled joint. These tests will be performed using prototype calandria tubes that are fabricated to formal design specifications. The tests will be performed when the calandria tubes are available, scheduled for 2005.

7. REFERENCES

[1] ACR-700 Technical Description, 10810-01371-TED-001 R0, June 2003.

[2] N. Popov, D.J. Wren, V.G. Snell, A. White and P.G. Boczar, "Development and Qualification of AECL Computer Codes for ACR Safety Analysis Applications", proceedings of 24th CNS Annual Conference, Toronto, June 2003.

[3] Technical Basis for the Validation of Computer Programs used for Safety Analysis of the ACR-700 Design, 108US-03500-TBD-001 R0, May 2003.

[4] An RD-14M Experiment for the Intercomparison and Validation of Computer Codes for Thermalhydraulic Safety Analyses of Heavy Water Reactors, Atomic Energy of Canada Limited Report, RC-2491, June 2000.

[5] The Technology of CANDU Fuel Channels, 108US-31100-LS-001 R0, August 2003.

[6] Phenomenology for Limited and Severe Core Damage Accidents in an ACR, 108-126810-LS-001, R0, September 2003.

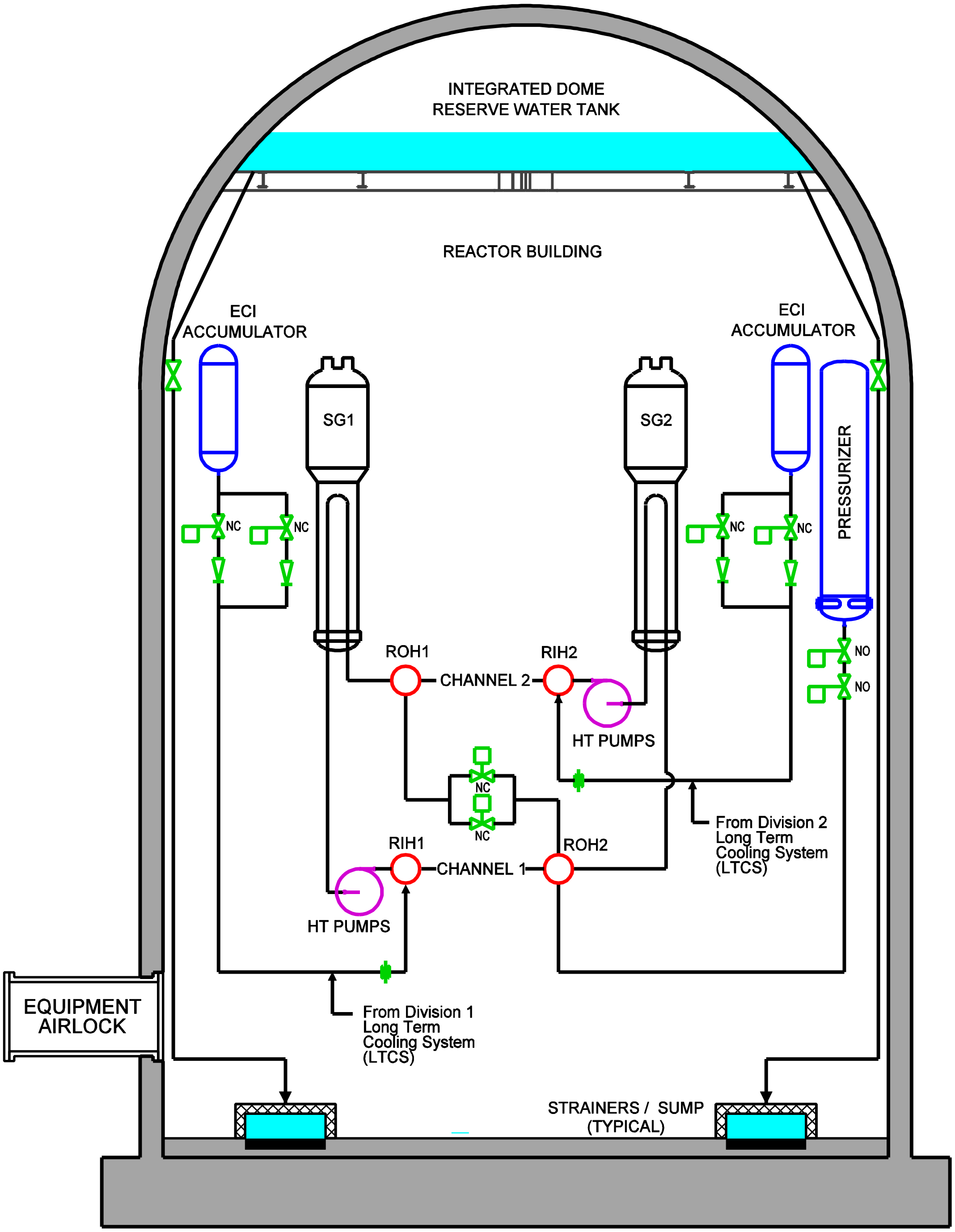


Figure 1 Schematic Diagram of the ACR-700 Emergency Cooling Injection System

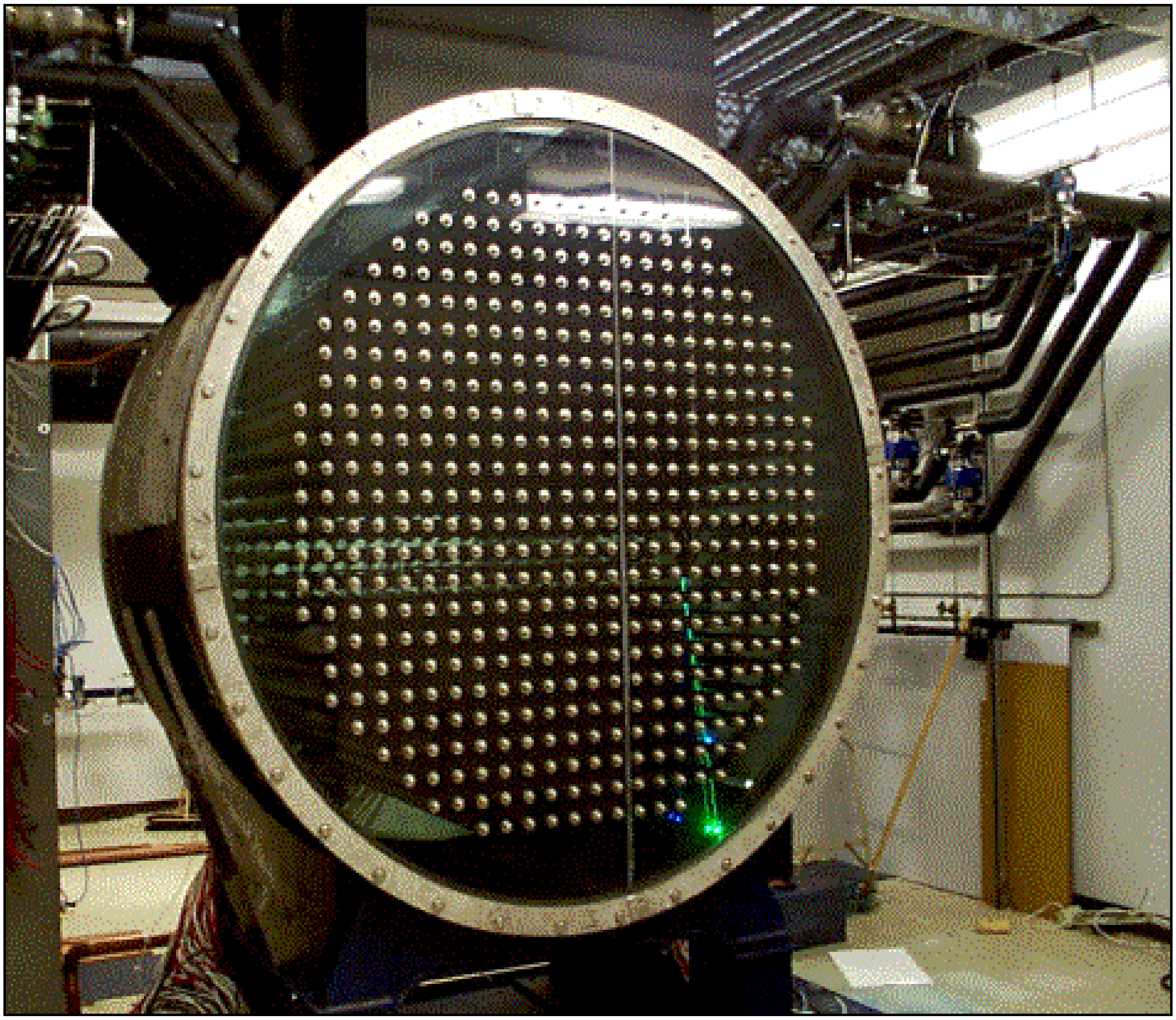


Figure 2 Moderator Test Facility (Scaled to ¼ CANDU 9)

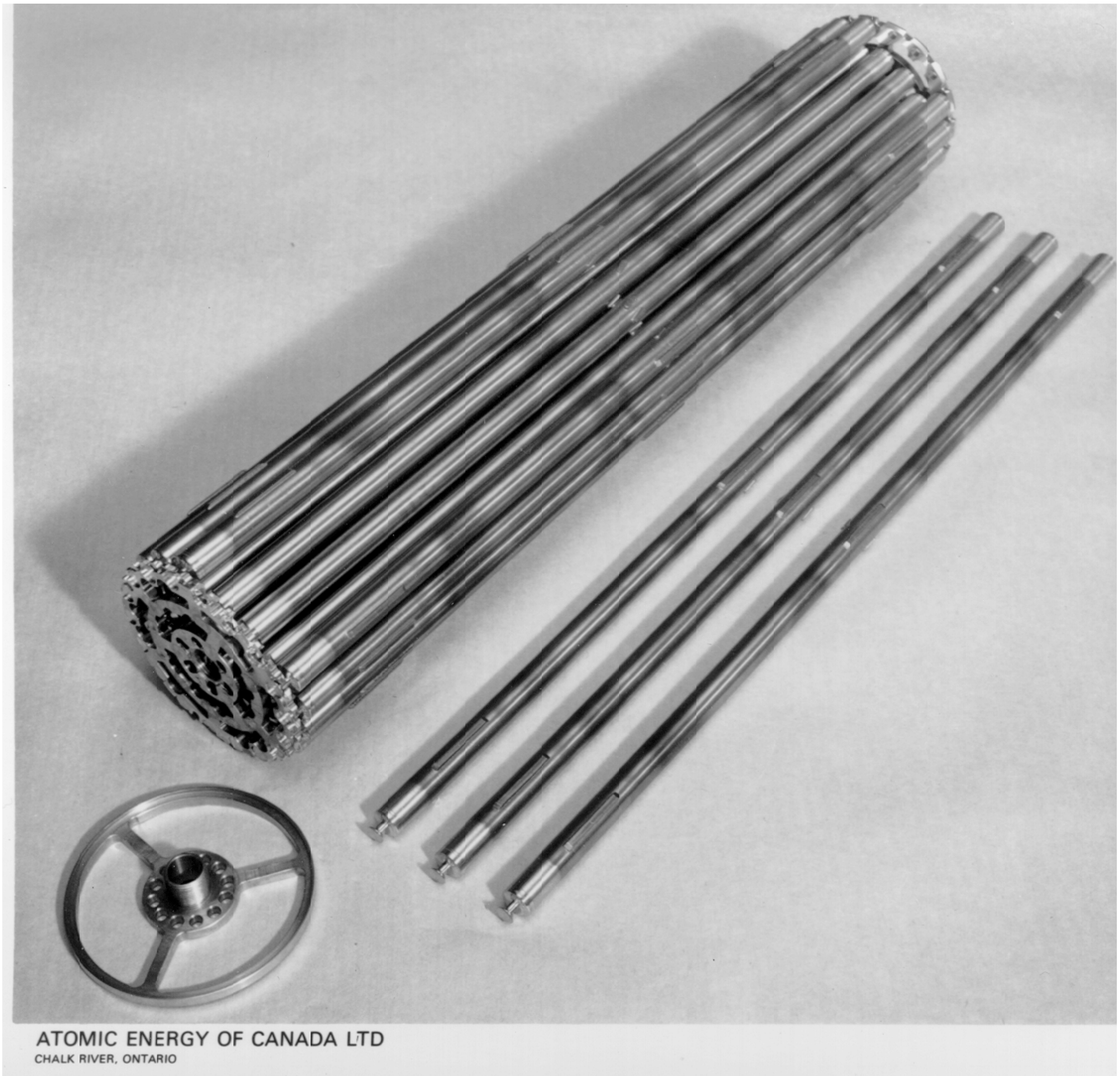
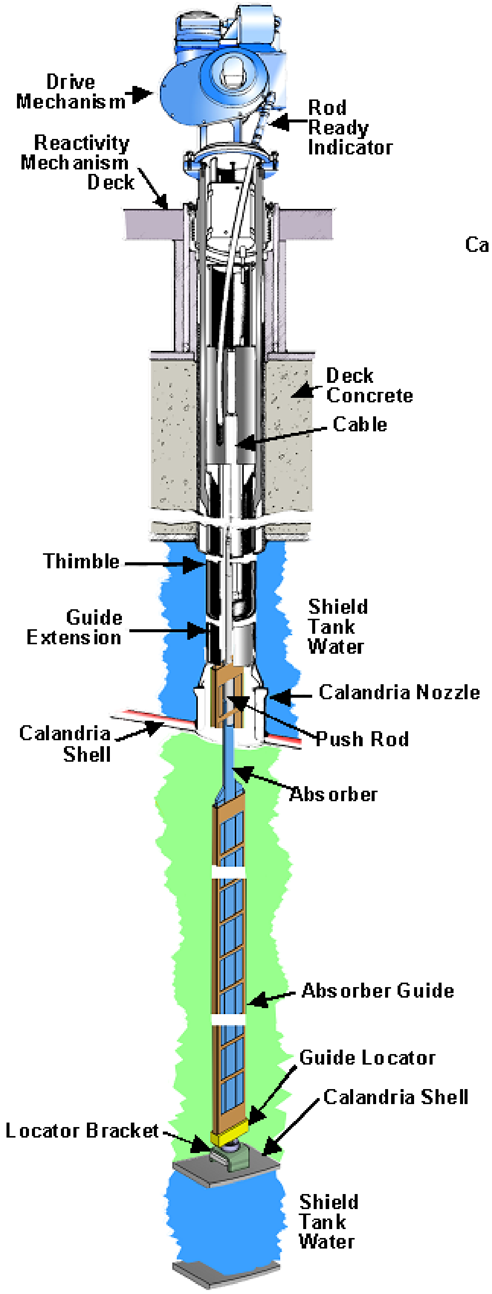


Figure 3 Demountable Fuel Bundle for Element Power Ramp Tests



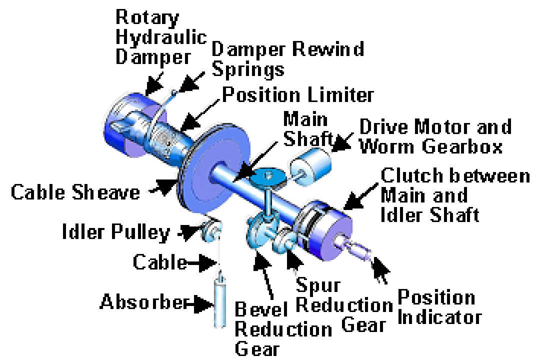


Figure 4 Shutdown System No. 1 Mechanical Shutoff Units

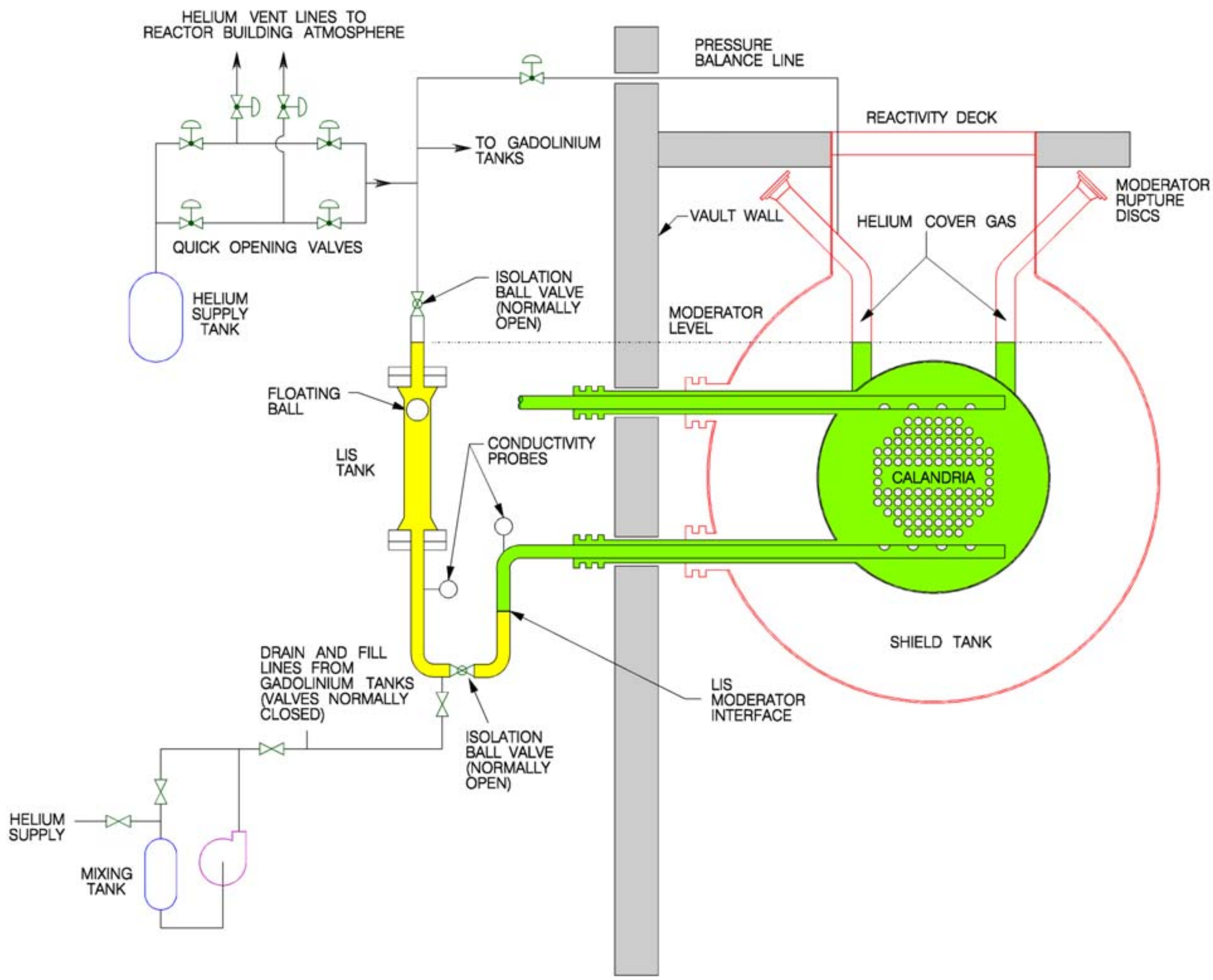


Figure 5 Shutdown System 2 Schematic Diagram

Figure 6 Successfully Opened One-Way Rupture Disc and Rupture Disc Holder (Test Assembly)

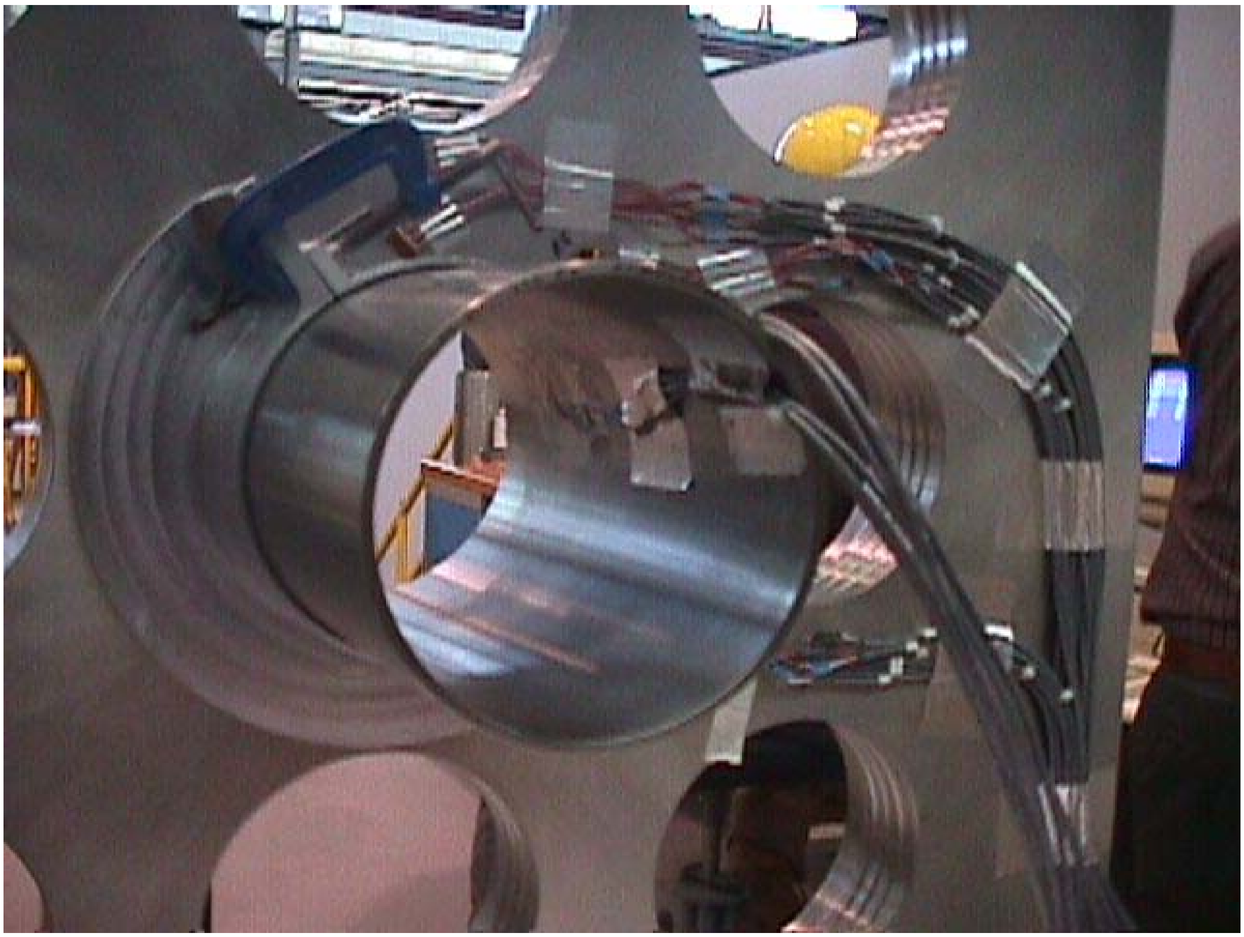


Figure 7 Feasibility Test of Calandria Tube Rolled Joint

Appendix A

Severe Accident R&D Program

A.1 AECL Severe Accident R&D Program

The accident classification for the ACR design recognizes three classes of events: design basis accidents, limited core damage (LCD) accidents and severe core damage (SCD) accidents. The accident classification scheme (discussed in Reference 6) is intended to satisfy domestic licensing requirements (as set by the Canadian Nuclear Safety Commission) while accommodating international licensing requirements (for example, those set by the US Nuclear Regulatory Commission). The classification is complex because the domestic requirements place some of the accidents with significant core damage potential into the Safety Analysis Report (SAR) realm. Internationally, such accidents are typically in the probabilistic safety analysis (PSA) realm.

The LCD accident category, which is more probable than the SCD category, maintains all the fuel in a well-known, coolable geometry that is approximately constant after the initial transient. The SCD accidents involve significant amounts of fuel debris (i.e., corium) outside of the reactor coolant system boundaries. The debris properties, geometry and location change with time, depending on the progression of a severe core damage accident. These changes occur over a long-term time scale and can be affected by operator interventions.

Canadian safety/regulatory philosophy goes beyond the practice in other jurisdictions by including LCD accidents in the Safety Analysis Reports of CANDU nuclear power plants. Including such low probability events, which would be regarded as beyond design basis elsewhere, helps to ensure the robust nature of the design. The SCD accidents fall into the Probabilistic Safety Assessment (PSA) realm.

There are two subcategories of LCD events (severe single channel events and LOCA plus loss of emergency coolant (LOECC) with the moderator heat sink available). A common characteristic is that the accident progression is arrested within multiple, distributed “vessels” (i.e., within the fuel channels), which may be cooled either internally, by water within the channels, or externally, by water surrounding the channels. There is no analogous core damage state in light water reactors (LWRs).

The SCD accidents are similar to “degraded core” accidents in LWRs. As with LWRs, there are two subcategories: in-vessel SCD accidents and ex-vessel SCD accidents. A prerequisite of these accidents is an ECC system failure in conjunction with the alternate moderator heat sink being unavailable. It is important to recognize that while there are similarities in phenomenology, the progression of a severe core damage accident in a CANDU is quite different from that of a LWR. For example, the presence of large volumes of water makes the time scale for core disassembly much longer, and when it does occur, the debris formed is mostly solid compared with the molten material that forms as an LWR core is uncovered. Containment bypass due to steam generator tube failure is unlikely, as a high power channel in a CANDU will fail first. Also, ex-vessel SCDs are highly unlikely as water in containment floods the exterior of the shield tank to a depth of more than 1 m, protecting the integrity of this ultimate barrier.

AECL, in conjunction with the CANDU Owners Group (COG), has supported a substantial R&D development program for many years that addresses both the LCD and SCD events in a CANDU. Extensive experimental and analytical databases exist for the LCD accidents. Computer models are available to simulate these accidents, embedding detailed deterministic models where appropriate. Extensive past analyses for the operating CANDU reactors provide understanding of the interrelationships between the governing processes and phenomena (i.e., what is important to LCD accident consequences).

Virtually all of the available CANDU LCD and SCD data is directly applicable to analysis and modelling of related events in the ACR design as well. From the perspective of a beyond design basis event, involving loss of cooling to the reactor core, both the ACR and the reference CANDU designs are very similar. The physical phenomena are the same. Both designs have effectively equivalent fuel and core configurations, accident sequences and containment designs. Recriticality is, however, a possible phenomenon in ACR, which is not relevant to the reference CANDU.

The phenomenology associated with severe accidents in ACR (and CANDU) designs is described in Reference [6]. This Appendix is provided to give an indication of the R&D database and ongoing R&D program that supports analysis of beyond design basis events in an ACR. There is no attempt in this Appendix to fully describe the database that addresses the severe accident phenomena. Instead, some of the more significant contributors to the database are summarized below.

In support of analysis of LCD events, AECL has accumulated information on safety thermalhydraulics behaviour, fuel behaviour and fission product behaviour.

- Based on over 20 years of research, AECL is the world leader in the modelling of iodine behaviour in the containment. This modelling is directly applicable to evaluation of ACR severe accidents.
- AECL has performed over 20 years of research on hydrogen combustion phenomena in containment, and particularly those relevant to both LCD and SCD severe accidents in CANDU reactors.
- The CATHENA thermalhydraulics code includes the capability to model fuel channel behaviour under conditions of loss-of-coolant plus loss-of-emergency core coolant (LOECC). This includes high-temperature heat transfer from degraded fuel channels (ballooned or slumped). Work to expand this database is ongoing.
- The behaviour of CANDU fuel under LOCA plus LOECC conditions has been studied both in small-scale hot cell tests, and in larger integrated in-reactor tests (using the Blowdown Test Facility).
- Work is in progress to measure the potential interaction between molten fuel and the heavy water moderator for single channel events associated with severe flow reductions and subsequent channel failures.

To support the analysis of SCD events, AECL and COG in association with Fauske and Associates have created the MAAP-CANDU code as a tool for modelling severe core damage accident progression in CANDU reactors. To support this tool, an experimental program is in place to study the core melt progression and the progressive collapse of horizontal fuel channels.

In addition, AECL is a member of an international consortium for tests that are related to the ability of the calandria and/or the shield tank to contain a severely degraded core in the event of a severe accident (RASPLAV).

A.2 ACR Severe Accident R&D Program

To supplement the generic CANDU severe accident database, the ACR development program includes additional R&D activities. Highlights of the planned additional studies are outlined below.

- The MAAP-CANDU code is in the process of being modified by Fauske and Associates to produce a code version that includes the ACR design features, including the Reserve Water System. This is a new system in the ACR (compared to the reference CANDU 6 design) that provides additional mitigation capabilities for severe accidents.
- As discussed in Section 2.1.4, additional experiments are in progress to extend the database on high-temperature heat transfer to include tests with the ACR fuel channel geometry.
- The ongoing generic CANDU experimental program on core degradation and channel collapse for SCD events will include additional tests with geometries scaled to the ACR core dimensions.
- The ongoing generic CANDU experimental program on molten-fuel moderator interaction will include additional tests with ACR fuel channel material and coolant conditions.
- The test program to validate the MODTURC_CLAS code (Section 2.7) will include tests to simulate LOCA + LOECC events. This will verify the ability of the moderator to act as a heat sink for such events and thereby protect against core disassembly.