AECL
TECHNOLOGIES INC.

February 17, 2003

Our File: 108US-013210-021-001
Your File: Project No. 722

U.S. Nuclear Regulatory Commission,
Document Control Desk,
Washington, D.C. 20555

Attention: Ms. B. Sosa
Project Manager, ACR

Re: **Proprietary & Public Versions – ACR R&D Plan Overview Documentation in support of the ACR Pre-Application Review**

Pursuant to the NRC's request for the proprietary and public versions of the ACR R&D Plan Overview documentation please find enclosed one copy of each version in support of the ACR Pre-Application Review.

The information contained in Enclosure 1 (AECL Report 108-01200-430-003 Revision 0, "ACR R&D Plan for Basic Engineering Support", June 2002) contains proprietary information of the type that AECL normally maintains in confidence and withholds from public disclosure. The report has been handled and classified as proprietary to AECL as cited in the affidavit provided in Attachment 1. Therefore, it is requested that the AECL proprietary document provided as Enclosure 1, be handled by the USNRC on a confidential basis and be withheld, in its entirety, from public disclosure in accordance with the provisions of 10CFR2.790 and 9.17.

The information contained in Enclosure 2 (AECL Report 108-01200-430-004 Revision 0, "ACR R&D Plan Overview", February 2003) contains information that may be made available to the public.

If you have any questions on this letter and/or the enclosed information please contact the undersigned at (905) 823-9060 extension 6543.

Yours sincerely,

V. J. Snell for

Vince J. Langman
ACR Licensing Manager

/Attachments:
1. AECL Proprietary Information Affidavit

D070

901 15th Street, NW, Suite 440, Washington, DC, 20005-2301 USA, Tel. 202-354-3863, Fax 202-318-2363

AECL
TECHNOLOGIES INC.

/Enclosures:

1. Proprietary version of ACR R&D Plan Overview, AECL Report 108-01200-430-003 Revision 0, "ACR R&D Plan for Basic Engineering Support", June 2002.

2. Public version of ACR R&D Plan, AECL Report 108-01200-430-004 Revision 0, "ACR R&D Plan Overview", February 2003.

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AECL
TECHNOLOGIES INC.

April 1, 2003

Our File: 108US-013210-021-001
Your File: Project No. 722

Ms. Belkys Sosa
ACR Project Manager
US NRC – M/S 0-4D9A
11555 Rockville Pike
Rockville, MD 20852-2738
U.S.A.

Dear Ms. Sosa,

Re: **AECL Copyright Notice**

During the current pre-application review for the ACR AECL Technologies will be transmitting documents to the NRC that bear an AECL copyright notice. The NRC is permitted to make the number of copies of these reports which is necessary for NRC's internal use in connection with the ACR pre-application review, subject to the requirements of 10CFR 2.790 regarding restrictions on public disclosure to the extent that such information has been identified as proprietary by AECL, copyright protection not withstanding. For the non-proprietary versions of these reports, where they exist, the NRC is permitted to make additional copies (i.e., beyond those necessary for its internal use) in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington DC and in local public document rooms as may be required by NRC regulations. All copies made by the NRC of any AECL documents must include any copyright and/or proprietary notices that are present in the copies sent to the NRC.

If you have any questions with regards to this letter please contact the undersigned at (905) 823-9060 extension 6543.

Yours sincerely,

Vince Lang

Vince J. Langman
ACR Licensing Manager

ATTACHMENT 1

APPLICATION FOR THE NUCLEAR REGULATORY COMMISSION'S WITHHOLDING FROM PUBLIC DISCLOSURE OF PROPRIETARY AECL REPORTS

10 C.F.R. § 2.790
AFFIDAVIT OF KEN HEDGES

I, Ken Hedges, Vice-President, AECL Technologies Inc., do hereby affirm and state:

1. I am the Vice-President, Technology for AECL Technologies Inc., and have been delegated the function of reviewing the proprietary information sought to be withheld from public disclosure, and am authorized to apply for its withholding on behalf of AECL Technologies Inc.

2. In the attached letter V. Langman to B. Sosa, "Proprietary & Public Versions – ACR R&D Plan Overview Documentation in support of the ACR Pre-Application Review", dated February 17, 2003, and Enclosure 1 to that letter, AECL Technologies Inc. is providing information in support of the Nuclear Regulatory Commission's (NRC) pre-application review of the Advanced CANDU Reactor (ACR). The document included in Enclosure 1 constitute proprietary commercial information that should be held in confidence by NRC pursuant to 10 CFR §§ 2.790(a)(4) and 9.17(a)(4), because of one, or more, of the following reasons:

   i. This information is confidential and has been held in confidence by AECL, which is the parent company of AECL Technologies Inc. The information is contained in AECL reports or other documents that are normally held in confidence in accordance with AECL's procedures for the protection of information. The reports or other documents are part of AECL's comprehensive safety and technology base for the CANDU design, and their commercial value extends beyond the original development costs, which in themselves are considerable.

   ii. The information is contained in CANDU Owners Group Inc. (COG) reports that are held in confidence by both AECL and the Canadian nuclear utilities that participate in research and development programs via COG. There is a rational basis for holding the reports in confidence since the information contains sensitive technical and/or commercial information relating to the supporting research, design and/or operation of CANDU reactors. Also, COG reports are only distributed to participants in COG research and development programs. These participants expend significant amounts of money to fund the COG research and development programs, which produce the information described in these reports. Additionally, public disclosure by the NRC of the information contained in COG reports, which are supplied in confidence by COG to AECL, could jeopardize the future availability of such information to AECL. AECL is contractually obligated to COG and to other participants in COG programs to maintain the confidentiality of such reports. AECL relies, in part, on COG reports to improve the safety, operability and maintainability of the ACR, and to help develop and recommend improvements to enhance the safety, operability and maintainability of existing CANDU plants. COG would be reluctant to provide such information to AECL, and could move to restrict AECL Technologies' ability to provide such reports to the NRC, if there was a possibility that the NRC might make the information publicly available, after being supplied to the NRC by AECL Technologies Inc. AECL would suffer harm to its commercial business and competitive position if it did not have access to these reports and was unable to improve existing and future designs. Further, other participants in COG research and development programs would be reluctant to enter into such programs in which AECL was a participant; those participants enter into and fund such programs with the exception that the results will remain confidential to COG and program participants; if there is a possibility that information generated in such programs would become publicly available through AECL Technologies' provision of COG reports to the NRC. For the same reason, disclosure of such reports by the NRC would also hinder the ability of the NRC to receive similar reports in the future from AECL Technologies, since COG would likely withhold such reports from AECL.

iii. This information is being transmitted to the NRC in confidence.

iv. This information is generally not available in public sources and could not be gathered readily from other publicly available information.

v. Public disclosure of this information would create substantial harm to the competitive position of AECL by disclosing sensitive commercial information about the design and/or operation of CANDU reactors and/or the ACR to other parties whose commercial interests may be adverse to those of AECL. Also, the information contained in these reports has been developed at significant cost to AECL (the parent company of AECL Technologies).

AECL

TECHNOLOGIES INC.

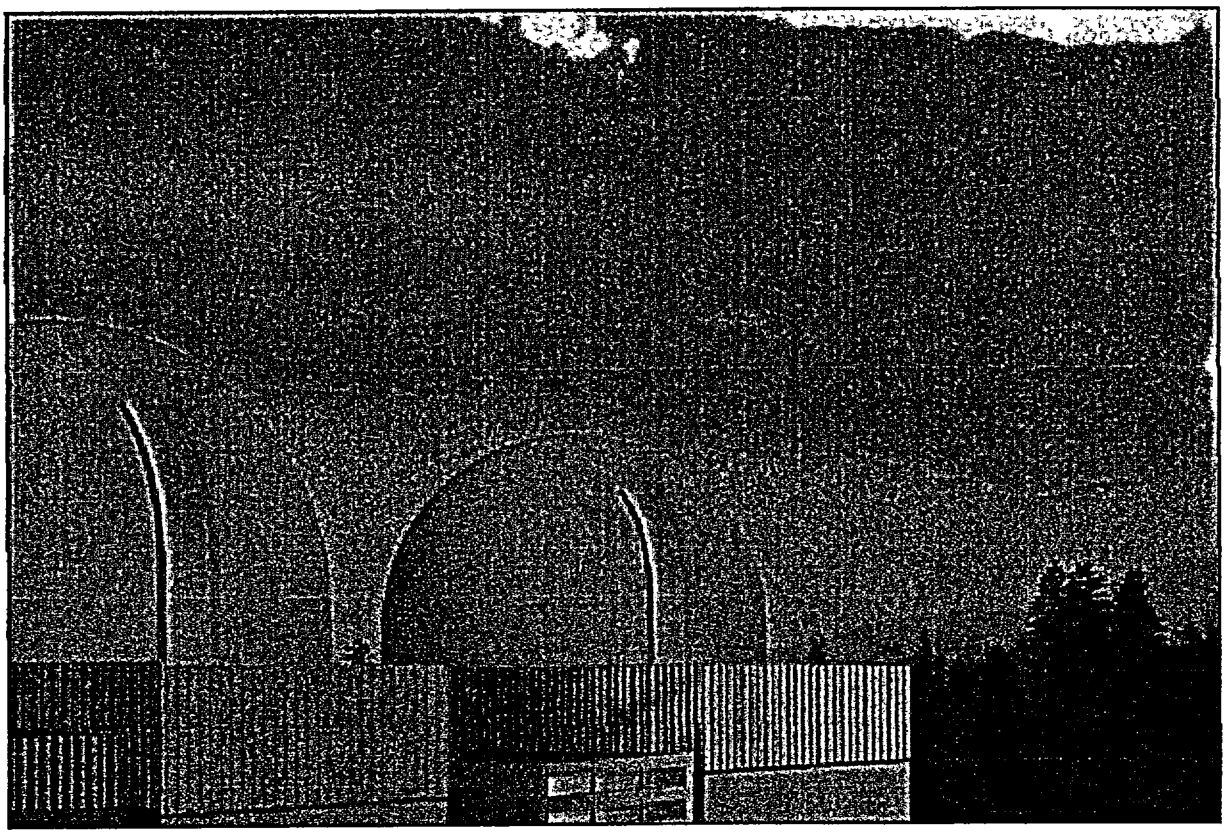
3. Accordingly, AECL Technologies Inc. requests that the information provided in Enclosure 1 be withheld from public disclosure pursuant to the policy reflected in §§ 2.790(a)(4) and 9.17(a)(4).

Ken Hedges, Vice-President, AECL Technologies Inc.

Subscribed and sworn before me on this 18 day of Feb, 2003.

Notary Public

ACR R&D Plan Overview
- Non-Proprietary Version-



Enclosure 2 for letter V. Langman to B. Sosa
of February 17, 2003

AECL
TECHNOLOGIES INC.

AECL EACL

ACR R&D Plan Overview

ACR

108-01200-430-004
Revision 0

2003 February

Février 2003

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Mississauga, Ontario
Canada L5K 1B2

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AECL EACL

ACR R&D PLAN OVERVIEW

ACR

108-01200-430-004

Revision 0

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Approved by
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2003/02/03 2003/02/03

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Release and Revision History / Liste des documents et des révisions

0939B Rev. 13

Document Details / Détails sur le document

Title / Titre	Total no. of pages / Nbre total de pages
ACR R&D Plan Overview	

Release and Revision History / Liste des documents et des révisions

Release Document No./N°	Release Document Date	Revision Révision No./N°	Revision Date	Purpose of Release; Details of Rev./Amendement / Objet du document; détails des rév. ou des modif.	Prepared by / Rédigé par	Reviewed by / Examiné par	Approved by / Approuvé par
1		0	2003/02/03	Issued to NCR for Licensing Review.	D. Wren	J.W. Love	S.K.W. Yu

DCS/RMS Input / Données SCD ou SGD

Rel. Proj. / Proj. conn.	Project / Projet	SI	Section	Serial / Série	Sheet / Feuille No. / N°	Of / De	Unit No (s) / Tranche n°
	108	01200	430	004	1	1	

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1. INTRODUCTION

This document describes the high-level R&D Program Plan to support the basic engineering design program for the development of a next generation CANDU nuclear power reactor.

The ACR-700™* is an advanced reactor design that is an evolutionary departure from the current CANDU designs. The ACR-700 design is 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 significantly 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.

In addition to reaching an aggressive capital cost target, an operating ACR-700 plant will deliver electricity at a competitive price. This means that it will incorporate innovative features that will reduce the operating, maintenance and administration (OM&A) costs of the plant, compared to the current CANDU designs. Achieving this goal will require design features that will contribute to effective plant life management and the maintenance of a high capacity factor.

Power reactor customers expect a proven product. Regulators demand validation of safety design features, and the public expects designs with improved safety and reduced environmental impacts. Where possible, these requirements will be met for ACR-700 through the use of systems and components that have already been demonstrated in existing plants. For new ACR-700 systems or components, the R&D program will satisfy design verification requirements and provide assurance to licensing authorities and customers that new design features are effectively proven.

This document describes the planned research and development activities that will be carried out to support the design of an ACR-700 plant. The requirements were established based on input from research experts and designers during the conceptual design stage. Revisions to this plan will be based on additional requirements that are identified during the basic design phase of the product development, the results of previous and on-going R&D activities, and input from other sources such as the design feedback process and regulatory feedback.

The R&D program has five main goals:

1. Completion of fuel channel design verification testing.
2. Completion of component and equipment development and testing.
3. Qualification of the fuel handling system design.
4. Qualification of the fuel design.
5. Completion of safety verification and validation activities to support a preliminary safety analysis report.

The fuel handling system is considered separately from the other components and equipment because it alone represents a major activity with multiple sub-activities related to the various

* ACR-700™ (Advanced CANDU Reactor™) is a trademark of Atomic Energy of Canada Limited (AECL).

components of the fuel handling system and the interfacing of the fuel handling system with the fuel channel design and heat transport system characteristics.

The R&D Program is divided up into five sub-programs that address each of the key goals listed above. Within each subprogram there are a number of major tasks that are planned. The durations in each sub-program are very long owing to the number and complexity of the tasks required. Additional factors are the time required to manufacture critical components for tests and the extended time required for irradiation testing of key components.

The R&D program to support the development of the ACR design will not terminate with the completion of the sub-programs. Ongoing R&D is planned to extend the database supporting the design, particularly in studies addressing aging related phenomena where longer-term testing is valuable. Some of the planned longer term R&D activities are identified in the more detailed schedules discussed below.

The R&D plan is integrated with the basic engineering plan for the ACR-700. In particular, the R&D to evaluate and qualify specific component or equipment designs or design features follows on from the production of appropriate engineering design requirements and preliminary design drawings and the definition of component verification specifications. R&D which is carried out in conjunction with conceptual engineering design to support concept design decisions does not require the same level of supporting engineering documentation.

This plan covers the activities required to support the engineering design and preparation of a product delivery project. Additional research and development activities will take place after scheduled completion of plan 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. FUEL CHANNEL PROGRAM

The use of light water coolant and a higher coolant temperature necessitate an evolution in the fuel channel design for the ACR-700 reactor. As a result, the fuel channel of the ACR-700 is substantially different from the fuel channel design of the reference CANDU 6 plants. However, the fuel channel is based on the proven CANDU design features using well-tested materials and the ACR-700 design is an extension of the current design configuration. The comprehensive R&D program to support the fuel channel design is aimed primarily at verifying that the new fuel channel design features can achieve the design lifetime performance targets.

The program objectives are described below.

2.1 Pressure Tube

The central element of the ACR-700 fuel channel continues to be a Zr-2.5%Nb pressure tube for which AECL has amassed a large knowledge base. The ACR-700 coolant temperature and pressure will be more demanding than the CANDU 6 coolant parameters. To compensate for these conditions, the thickness of the ACR-700 pressure tube has been increased. This results in stresses within the tube, under normal operating conditions, that are within the range of experience for the current pressure tube design.

The design of the ACR-700 pressure tubes will be based on extrapolation of the CANDU 6 pressure tube predictive design equations to ACR-700 operating conditions. The objective of the pressure tube R&D program is to generate sufficient data to make it possible to determine the uncertainties in the extrapolations.

The lifetime for a CANDU pressure tube is limited by two ageing phenomena, deformation (radial creep and axial elongation) and corrosion (with associated hydrogen uptake). To minimize the rates of these two ageing processes, the ACR-700 pressure tube design will take advantage of the accumulated experience and knowledge obtained from pressure tubes removed from operating plants and ongoing materials research. In particular, the ACR-700 pressure tubes will be fabricated using Zr-2.5%Nb with optimized microchemistry specifications and the best established manufacturing processes.

To provide the material for testing, a number of prototype ACR-700 fuel channels will be manufactured to the optimized materials specifications.

AECL has established a deformation equation to predict the rate of creep and growth in its current pressure tubes, and has a research program in place to examine the behaviour of an intermediate thickness Zr-2.5%Nb pressure tube. The ACR-700 fuel channel R&D program will obtain data on representative samples of ACR-700 fuel channel material to extend the validation database for the deformation equation to the ACR-700 temperature conditions. Since the deformation process is driven by a combination of thermal creep and radiation-induced creep, experiments will be conducted in the high-flux OSIRIS research reactor under temperature-controlled conditions. These irradiations will be complemented by other studies at lower fluxes under temperature-controlled conditions. The test program is schedule to begin in 2002/03, and will continue for an extended period of time to accumulate deformation data. Sufficient data will be available by the end of 2005/06 to provide confidence that the pressure tube will meet its lifetime performance target with respect to deformation.

The other principle ageing process is corrosion and hydrogen uptake. The Zr-2.5%Nb alloy is subject to a delayed hydride cracking failure mechanism that is a concern only when the hydrogen concentration in the pressure tube exceeds the critical concentration for precipitation of metal hydrides. Since corrosion is very slow for zirconium alloys under reactor coolant conditions, the hydrogen concentration builds up slowly and can be monitored during reactor operation. In addition, AECL has established equations to predict the rate of corrosion and hydrogen uptake for its current pressure tubes. The Fuel Channel R&D program includes a number of different experiments both in-reactor and out-reactor to extend the corrosion equations to ACR-700 conditions.

Experiments will be conducted in the Halden research reactor under temperature and coolant chemistry controlled conditions to measure corrosion and hydrogen uptake rates on prototypical ACR-700 material. The high temperature experiments will build on the existing database of corrosion data that has already been obtained at 325°C on CANDU 6 pressure tube material, and is expected to confirm a reduction in the corrosion rate that is predicted with the use of optimized microchemistry in the pressure tube material specification. The Halden irradiation program on ACR-700 material is scheduled to commence in 2002/03. The in-reactor experiments will ensure that the effects of radiolysis on the coolant chemistry are fully addressed in the corrosion rate prediction.

In addition to the in-reactor studies, a number of out-reactor autoclave tests will be carried out. These will address two issues: the hydrogen/deuterium isotope effect, and corrosion at the pressure tube rolled joint. The current CANDUs operate with heavy water as a coolant and therefore the hydrogen uptake in those reactors is actually a deuterium uptake. Chemically there is no difference between the two isotopes of hydrogen, but separate tests will be conducted to determine what impact, if any, the change to light water coolant will have on the predicted corrosion rates.

Corrosion and hydrogen uptake occur both over the bulk pressure tube inner surface and at the end of the pressure tube that is mechanically rolled into the end-fitting. The rolled joint provides both a mechanical coupling and a leak-proof seal. The temperature of the pressure tube at the rolled joint is much lower than the temperature of the bulk tube, there is an electrochemical coupling between the pressure tube and the end-fitting materials and the tight fit between the material in the joint region provides a site for crevice corrosion. An experimental program is planned for the CTL-1 loop to measure the hydrogen uptake rate at the rolled joint region. This will be supplemented by separate autoclave corrosion tests to look at the electrochemical coupling phenomena for the rolled joint design.

A design requirement for the pressure tube is that its tensile strength shall be greater than a specified limit and that leak-before-break be assured to design life. An experimental program will be carried out to extend the testing range on the materials properties of currently available irradiated pressure tube material to the operating conditions of the ACR reactor and then to verify the applicability of these results by testing unirradiated and irradiated material from prototype ACR pressure tubes. This will include studies of the K_{IH}, fracture toughness and delayed-hydride cracking (DHC) velocity, using samples of prototype ACR pressure tubes. Tests will also be performed to confirm predictions of the leak rate from cracks in the thicker ACR pressure tube.

In addition to the R&D program described above, the ACR-700 pressure tube design will incorporate the latest refinements arising from AECL's ongoing product support program, such as the development of surface preparation techniques to reduce the rate of hydrogen uptake.

2.2 Calandria Tube

The calandria tube in the ACR-700 reactor will be both larger in diameter and thicker than the CANDU 6 calandria tube. The larger calandria tube thickness will necessitate the development and qualification of a manufacturing process for this tube design. This work is included in the R&D plan and will commence during 2002/03 using calandria tube material that has already been obtained.

The aim of the development program will be to establish a low cost manufacturing route that delivers a calandria tube with sufficient strength and toughness to withstand a spontaneous pressure tube rupture for the length of time required to safely shut-down the reactor. The manufacturing qualification will also provide prototype calandria tube material for use in other elements of the R&D program. This includes verification of the process for installation of the calandria tubes in the reactor assembly.

Prototypical CT material will be used in studies to demonstrate the resistance of the CT to failure under accident conditions involving spontaneous pressure tube rupture.

2.3 End-Fitting/Rolled Joint

In the CANDU 6 reactor, the Zr-2.5%Nb pressure tube is connected to two Type 403 stainless steel end-fittings by means of the rolled joint discussed above. To permit the tighter lattice pitch between pressure tubes that is a feature of the ACR design, the end-fitting must be redesigned. AECL has considerable experience with the rolling of joints between dissimilar materials. As part of the verification of this design feature, prototype end-fitting components will be manufactured and the rolling process for the joints between the end-fitting and the pressure tube will be qualified.

2.4 Other Fuel Channel Components

The fuel channel contains several other components. A development and qualification program will be carried out for each of the components as required.

A new spacer to separate the pressure tube from the calandria tube, while accommodating the larger pressure tube to calandria tube gap, will be designed and tested.

A new 'bore seal' channel closure is being developed to provide a high-pressure channel seal with a mating surface on the inside of the end-fitting. Development testing of elements of the conceptual design for the channel closure is in progress and a full qualification test program will be carried out. This will include testing to demonstrate compatibility with new fuel handing equipment.

A liner-tube/flow-diffuser component is required as an element in the end-fitting where the flow from the channel feeders enters and leaves the end-fitting. A second component affecting the fluid flow in the end-fitting is a shield plug that will be latched into position to support the fuel column horizontally. The conceptual designs of these components will be tested in conjunction with qualification of the fuel and fuel handling equipment. The testing program will include acoustic analysis to ensure that there is no possibility of excessive wear from fuel vibration against the pressure tube during operation. Development testing and characterization of the shield plug and flow diffuser conceptual design will begin in 2002/03 in conjunction with flow impact testing of the fuel.

3. COMPONENTS PROGRAM

The Components R&D Program includes all of the development R&D for components and equipment in the ACR design that is not covered by the three major areas (fuel, fuel handling, and fuel channel) or safety. It incorporates the development testing of design concepts (and qualification testing where warranted) of new mechanical components in the ACR-700 design. The major components development work focuses on the new features of the ACR-700 design in the reactor assembly and reactivity control devices located in the reactor assembly. R&D in other areas will be carried out to support the design specifications of major equipment (e.g., steam generators), to confirm design decisions (e.g., acoustics analysis) or to develop components with improved performance. The ACR-700 development will leverage AECL's other nuclear R&D programs in the last area and adopt improved designs or specifications as they become available.

The schedule for the Components program is driven by the schedule for the basic engineering design program, to support the evaluation of the concepts selected for development and qualification. The ACR-700 program will use advanced 3-D CADDs design tools to reduce the level of concept testing that is required.

3.1 Reactor Assembly

The ACR-700 design is based on a compact core design and a compact reactor assembly. Cost-effective manufacture of the compact reactor assembly requires optimization of some features of the design. These include the techniques for coupling the calandria tubes to the tube-sheet at the ends of the reactor assembly and the techniques for connecting the lattice tubes in the end-shields. In conjunction with the development of the detailed reactor assembly design, tests will be carried out to verify the design and manufacturing techniques.

This work is scheduled early in the R&D plan in order to confirm some basic reactor assembly conceptual details, to provide input into the fuel channel component design, and to ensure that the design of the reactor assembly is ready for early procurement, since it is close to the critical path for product delivery.

3.2 Reactor Control Devices

There are four sets of reactivity control devices in the ACR-700 design: the Shutdown Units for Shutdown System 1 (SDS1), the zone control units, and the control absorber units for the reactor regulating system, and the liquid injection nozzles for Shutdown System 2 (SDS2). The first three are solid mechanical devices that are vertically inserted into the core from the reactivity mechanism deck and controlled by electric motor drives. The last consists of nozzles located above and below the reactor core that disperse liquid poison into the calandria.

SDS1 and SDS2 are special safety systems and must undergo comprehensive design qualification and verification. The other two systems must undergo a similar level of testing because of their function and importance.

The mechanical design of the first three sets of devices is quite similar. All are solid absorbing elements located within guide tubes. Development work is currently in progress to select a reference conceptual design for the solid devices. The SDS1 system, in particular, will be very similar in concept to the existing SDS1 system in the CANDU 6 design. However, the shape of the solid absorber elements and guide tubes will be modified to mate with the tighter ACR-700 lattice pitch, and the drive units will be updated to fit into the tight reactor deck layout and to replace obsolete parts.

The performance of prototype mechanical reactivity control devices, including the electric motor drives and position units, will be verified by testing.

The SDS2 design will be based on the proven CANDU 6 design. Tests will be conducted on the design for the ACR-700 design configuration to verify the speed of negative reactivity insertion.

3.3 Heat Transport System

The ACR heat transport system (HTS) will operate at higher temperatures than the CANDU 6 design. In addition, the tighter ACR lattice pitch requires a reduction in the diameter of feeder pipes in the reactor face to maintain requisite clearances and this leads to higher coolant flow velocities in those feeder sections than normally experienced in the CANDU 6 design. To address the increased potential for corrosion in the feeders, more corrosion-resistant materials than the standard CANDU 6 feeder material have been identified. The R&D program will provide materials performance assessments and tests to confirm the design specifications. These assessments will include consideration of the role that radiolysis of the coolant may play on flow-assisted-corrosion or cracking of feeder material near the reactor core.

The R&D program tentatively includes a program of in-reactor tests to confirm predictions of radiolysis driven chemistry conditions and oxidation potentials. This, if required, work would supplement ongoing AECL programs in this area. The requirements for ACR-700-specific in-reactor chemistry studies will be evaluated during 2002/03.

The ACR-700 HTS materials will be selected for their corrosion resistance and materials properties. To provide long-term support for predictions of general corrosion rates and local corrosion rates for specific heat transport system features, an ACR-700 Chemistry and Materials Test Loop will be constructed. This loop will be capable of operation at full ACR-700 temperatures, pressures, and fluid velocities. Design of this loop is scheduled to commence in 2002/03.

The operating conditions for the ACR-700 steam generator will be different from those of the current CANDU design. To assist in the design of the heat transport system and to establish the detailed specifications for the steam generator, AECL will use a number of analysis codes (THIRST, VIBIC, CHECWORKS). In addition, selected tests will be carried out to verify the predictions of the codes for the ACR-700 steam generator design and to confirm materials specifications.

3.4 SMART Plant Design

AECL is developing tools to improve the operability and maintainability of current and advanced reactor designs. The ACR development program will adopt these improvements as they become available and will support specific development initiatives that can be completed within the ACR development schedule and that offer substantial reductions in plant operating and maintenance costs. It is anticipated that improved system and chemistry health monitors will be the primary focus of work.

3.5 Control and Information

The area of control and information includes both the electronic hardware required to monitor and operate the plant, and the software and engineering tools that are used to design, construct, commission and operate the plant. The engineering tools are included in this area because of the very strong interaction between the hardware and software required to obtain, manage and interrogate the information required for effective plant operation and management.

Within the context of the continuous evolution and improvement of AECL's engineering design and product delivery tools, the ACR-700 program will support the adoption and implementation of specific tools and engineering processes that can be implemented on a priority basis to reduce the cost and schedule for, and ensure the quality of, a new ACR-700 plant. An early priority for 2002/03, building on AECL's experience in using 3-D CADD tools for overall plant design, will be the adoption of 3-D CADD tools for mechanical component design, linked to advanced manufacturing methods. This will be used in the design and development of new fuel handling system components to reduce plant costs. The overall development target is to implement an advanced suite of engineering and project delivery tools that will facilitate much more efficient and cost effective work processes for engineering, construction, and commissioning. As well, the information these tools produce and manage will be integrated with plant information systems to lower plant operating and maintenance costs.

An area that has been identified for significant cost reduction is the communications infrastructure of the plant. The ACR-700 design will incorporate the use of broad-band communications technology using fibre optic cables and wireless communications, where appropriate, to reduce the cost of the plant equipment and the time required for installation and communication. An assessment of the potential technologies for implementation in the design will be carried out in 2002/03 followed by test programs to verify the applicability of the selected technologies and support the design specifications.

The ACR-700 design will incorporate advanced digital control information and data management/display systems. The R&D program will support the development of the control room design and verify the design's conformance to human factors engineering requirements. Additionally, it will support verification of the suitability of the instrumentation and control system.

Included in the advanced plant information systems, will be advanced system health monitoring capabilities. Implementation of these systems will build on AECL's work to improve the performance capabilities of the current CANDU plants, such as the development of the ChemAND system for plant chemistry monitoring. The ACR-700 R&D program will support the implementation of these new capabilities in the control system, control room design and field instrumentation.

To support human factors considerations in control panel and display concepts, a control centre mock-up will be established for ACR concept development.

The R&D program will also support the development testing of modifications to the in-core instrumentation that may be adopted to accommodate the reactor assembly design. In future years, a development program will be initiated to provide a condition-based maintenance information capability to achieve specific cost reduction targets for the ACR-700 operations and maintenance cost projections.

3.6 Environment, Emissions and Waste Management

The R&D program in this area is intended to support the selection and implementation of technologies in the ACR-700 design that will reduce the potential emissions (both radiological and non-radiological) from the plant under normal operation. As part of this program, advanced models for predicting the potential emissions from the new plant design will be developed. Because of the generic CANDU design aspects of the ACR-700 plant, the ACR-700 will use the validated atmospheric dispersion model that AECL has developed, ADDAM.

The use of light water coolant leads to a sharp reduction in the volume of highly tritiated spent resins that will be generated by the ACR-700 and this in turn provides the opportunity for the introduction of new, cost-effective spent resin management systems. An overall assessment of waste stream management technologies, including spent resin management, has been initiated and will continue in 2002/03. The best candidate technologies arising from this assessment will be selected and tests programs will be carried out in future years to confirm their performance characteristics. This will include an evaluation of the radiological waste forms and their requirements for storage and disposal.

An important element in the licensing of a new plant is meeting the requirements of a successful environmental impact assessment. The requirements in this area depend on the plant design, the site characteristics and emerging acceptance criteria. The ACR-700 R&D program will support AECL's base line efforts to study potential environmental impacts and to develop predictive methods to assess potential impacts. The focus of the ACR-700 contribution to AECL's overall program will be to address site-specific issues that may emerge.

3.7 Constructability

To achieve aggressive cost reduction and schedule targets, the ACR-700 design will take advantage of advances in materials and construction technologies. Where it is determined to be necessary, some R&D will be carried out to establish materials specifications or to develop the application of a particular construction technology.

One area of cost savings and design improvement that has already been identified is the use of high-performance concrete. Work is planned to complete the development of a specification for advanced concrete to reduce the cost and schedule for reactor building construction. Additionally, low-porosity advanced concrete formulations will be examined for potential application as sealants in selected plant locations.

To reduce costs and schedule, the ACR-700 plant will be constructed using a modularisation approach. There are many alternatives available for the implementation of this construction philosophy and module design that are currently being assessed. It is anticipated that some development work will be required to confirm the module design and particularly the use of composite materials or structures. Work plans in this area will be defined and implemented when the module design requirements have been established.

3.8 Other Components

In addition to the major areas for component development described above, there will be a number of other design features that may require R&D support for conceptual design selection and/or design verification. The number such components and the extent of development testing will be determined through on-going design reviews and assessments. One example, under early consideration, is the development of a feeder scanner system to reduce the cost of detection of fuel element failures during reactor operation.

4. FUEL HANDLING SYSTEM

The ACR-700 design is based on the proven CANDU concept of on-line refuelling using fueling machines attached simultaneously at both ends of a fuel channel. However, the ACR-700 will have substantially new detailed designs for fuel handing systems including the fuelling machine (and carriage), the fresh fuel transfer system and the spent fuel transfer system. Changes to the designs of those systems in the CANDU 6 design are required to accommodate the higher temperatures and pressures of the ACR-700 coolant system, and the tighter lattice pitch of the fuel channels. In addition, the designs will be optimized to reduce capital and operating costs and to incorporate feedback from the operating stations. The elimination of tritiated heavy water from the fuel handling systems offers opportunities for simplification and savings. The design effort will take advantage of the development work already completed in support of the development of the CANDU 3 and CANDU 9 designs.

During the course of the basic engineering design phase, the critical components of these systems will be manufactured and tested to confirm the conceptual design and allow some optimization of the detailed design.

During 2002/03, development work is scheduled on the snout seal, fuel separators, fuelling machine ram and in 2003/04 the development work will be extended to include the remaining critical components, including the fuelling machine magazine, the channel homing system and other fuelling machine components. During 2003/04, work on the new water-based spent fuel transfer system will start to confirm extrapolation of elements of past work to the ACR-700 design and to create a system that reliably moves bundles without any shock loadings or transitions into air. Associated with this work will be some development work on a new 'dry store ready' fuel basket system designed to increase seismic capability and decrease handling requirements in the spent fuel bay.

The ACR-700 design will incorporate advanced digital control systems, instrumentation and data management/display systems for the Fuel Handling System. The R&D program will support the development of the control room design and verify the design's conformance to human factors engineering requirements and have a similar operator interface. Additionally the R&D program will support the implementation of these new capabilities in the control system, control room design and field instrumentation.

Following completion of the development testing of the individual sub-systems of the fueling machine, a complete integrated fueling machine head will be manufactured for full-scale verification testing.

5. FUEL PROGRAM

The ACR-700 design will use a slightly enriched uranium oxide fuel in the proven CANFLEX fuel bundle configuration. The novel features of this fuel design with respect to the standard natural uranium CANDU fuel design include:

- Uranium oxide fuel pellets enriched with ^{235}U to levels up to ~ 3%,
- Uranium oxide fuel pellets with designs optimised for fuel burnups to ~20,000 MWd/tU,
- CANFLEX fuel bundles with optimised fuel sheaths,
- CANFLEX fuel bundles with bearing pads designed for compatibility with fuel channel and fueling handling components and optimised for critical heat flux margin, and
- A central fuel element containing dysprosium oxide as a reactivity poison.

In addition to the novel fuel design features, the fuel will be exposed to a normal operating environment that is a departure from the current CANDU conditions. Unique ACR-700 fuel operating features will include:

- Exposure to higher reactor coolant temperatures and pressures,
- Longer in-core residence under new thermalhydraulic flow conditions,
- Power ramping transients associated with refuelling activities.

The ACR-700 fuel is an evolutionary development of the current CANDU fuel design. The CANFLEX fuel bundle has been qualified for use in CANDU reactors with natural uranium oxide fuel. CANDU fuel has been successfully irradiated in power reactors to burnups in excess of 20,000 MWd/tU. AECL has an established program to develop advanced fuel cycles for its current reactor products and the ACR-700 fuel design is based on the knowledge and experience gained in that program. The ACR-700 fuel performance target requires an extension of the proven CANDU fuel performance database and the ACR-700 fuel will be fully qualified.

The R&D program to qualify the ACR fuel design consists of five components:

1. Fuel irradiation tests,
2. Out-reactor fuel performance tests,
3. Fuel thermalhydraulics tests,
4. Fuel design code validation and
5. Core physics and fuel management code validation.

5.1 Fuel Irradiation Tests

The fuel irradiation program will consist of two components. The first component is the irradiation of prototype ACR-700 fuel bundles to burnups in excess of the target level, to demonstrate the performance of the integrated fuel bundle assembly. This irradiation will be conducted in the NRU research reactor. The high-burnup fuel irradiation is scheduled to start during 2002/03.

The second component is a series of irradiation tests to determine the performance limits for the ACR-700 fuel during short-term power ramps. During on-line refueling, the ACR-700 fuel bundles will be exposed to power ramps as a result of fuel movement along the length of the fuel channel in the core and end-flux peaking at the terminal bundle in a fuel string within the core during the fuel movement. This testing program will also be carried out in the NRU research reactor using a demountable CANFLEX fuel bundle. The demountable bundle allows the use of graded enrichments in different elements within the bundle to test for a wide range of power ramp conditions in a limited number of tests. This will maximize the production of data during the R&D program.

The demountable CANFLEX bundle has already been developed and is available for use in the test program.

The detailed specification of the irradiation test matrix will be determined on the basis of calculations of the predicted fuel power histories in the core using the appropriate core physics and fuel management codes. The element power ramping tests are scheduled to commence in 2003/04.

As part of the fuel irradiation program, AECL is also exploring options for additional fuel irradiations at off-shore research reactors. This would increase the database of tests and offers the potential to explore irradiation conditions that cannot be achieved in the NRU reactor.

5.2 Out-Reactor Fuel Performance Tests

Out-reactor tests will carried out to ensure that the ACR-700 CANFLEX fuel bundle will meet all design requirements related to geometry, thermal-mechanical loads and endurance. The development program will include the following tests:

- Cross-flow endurance in an end-fitting,
- Vibration and endurance in a channel,
- Refuelling impact,
- Compatibility with the fuel handling system,
- Spacer interlock prevention and
- Bearing pad sliding wear.

These tests will be conducted using test loops available at AECL's engineering development facilities.

5.3 Fuel Thermalhydraulics Tests

Reactor power operating margins are established based on the thermalhydraulic performance of the fuel. This depends on the details of the fuel design, the fuel power rating and the coolant flow characteristics. The NUCIRC thermalhydraulic code is used to predict the thermalhydraulic behaviour of the CANDU primary heat transport circuit. This code contains correlations that are dependent on the fuel design and operating parameters. While the ACR-700 fuel uses the CANFLEX fuel bundle design, the ACR-700 bundle incorporates minor enhancements to improve the critical heat flux margin of the fuel and to mate the bundle with the fuel channel design and the fuel handling system. The Fuel Development program includes a number of tests to obtain the data required to establish validated correlations for use in the NUCIRC code for ACR-700.

To establish validated critical heat flux correlations, a series of full-scale electrically heated fuel bundle tests will be carried out in water in a full-scale pressure tube. The fuel bundle will be manufactured to simulate the axial and radial heating profiles for a high power channel in an equilibrium core fuel loading.

In addition to the full-scale tests, smaller scale separate effects tests will be carried out in water and Freon loops to ensure good understanding to the results of the full-scale tests.

Separate effects tests will be conducted in small-scale water loops to measure the pressure drop and hydraulic resistance of the ACR-700 CANFLEX fuel bundle to provide the other correlations required for the NUCIRC code.

5.4 Fuel Design Code Validation

The ACR-700 fuel will be designed and optimised based on the extensive operating experience available for natural uranium CANDU fuel and the long history of AECL fuel research and development using the NRU research reactor. AECL has established a suite of design codes to assist in the development of new fuel designs, including higher burnup SEU fuel. To ensure that the fuel is designed to high quality assurance standards, the fuel design codes have been assessed for their applicability to the ACR-700 requirements and any extensions in the code capabilities have been identified.

As part of AECL's comprehensive software quality assurance program, an extension to the validation basis for the fuel design codes will be completed. This will involve formal verification and validation of the applicability of the fuel design codes to the ACR-700 requirements and conditions.

The validation will be carried out using the existing experimental database supplemented by the results of in-reactor and out-reactor tests on prototype ACR-700 fuel bundles and fuel elements, as described above.

5.5 Physics and Fuel Management Code Validation

The neutronic physics of the ACR-700 design is a significant departure from the physics of the natural uranium CANDU reactors with the change to a light water coolant and a slightly enriched uranium oxide fuel. The primary tool used to evaluate the physics of the reactor core is WIMS-IST. This code was originally developed to analyse the Steam Generating Heavy Water Reactor (SGHWR) at Winfrith that had a core design that was substantially equivalent to the ACR-700 core design (heavy water moderator and light water coolant). The subsequent development and validation of WIMS-IST has focussed on the natural uranium CANDU core, and there is a requirement to extend the validation of the code to the new design.

WIMS-IST validation will be performed using the zero-power critical assembly at CRL, ZED-2. Lattice arrangement experiments and lattice substitution experiments in this critical assembly will be used to validate WIMS-IST predictions for both the reactivity and kinetics of the new ACR-700 core design and also the reactivity change associated with coolant voiding in the event of a loss-of-coolant accident. Experiments in ZED-2 will also be used to verify the reactivity properties of the reactivity control elements for the ACR-700.

To achieve the desired accuracy in the physics code validation, a full core-load of slightly enriched fuel is being manufactured in 2002/03 for the ZED-2 core. The enrichment of this fuel has been selected so as to obtain code validation for the ACR-700 lattice with an uncertainty that ensures substantial margin in safety analyses.

In addition to experimental validation, the WIMS-IST code will be benchmarked against the MCNP Monte-Carlo physics code and other reactor physics tools.

Fuel management is performed using the RFSP physics code. The accuracy of this code for ACR-700 fuel management is under assessment and plans are being developed to modify (if required) and validate this code.

6. SAFETY VERIFICATION PROGRAM

The ACR-700 design will meet regulatory requirements and public acceptance standards with enhanced safety targets and a robust safety analysis case. The starting point for achieving these objectives is the proven performance, reliability and safety of the current CANDU design. From a safety analysis perspective, the ACR-700 design is very similar to the current CANDU design. The ACR-700 safety case will include the same suite of potential accident sequences with the same accident phenomena as the current CANDU design. Hence the CANDU accident analysis methodology, the analysis tools and the database supporting the analysis tools are broadly applicable to the ACR-700 safety assessment.

The Safety R&D program will provide the additional data necessary to extend the application of the existing analytic tools to the ACR-700 operating conditions taking into account safety-related design features. In addition, the Safety Program includes tests of safety-related equipment performance and accident phenomena where they are outside of the range of current CANDU experience.

6.1 Safety Code Validation

The ACR-700 design will use the validated Industry Standard Toolset (IST) of safety analysis codes as a basis for the tools to carry out the safety analyses required for reactor licensing. The IST codes are in the final stages of full validation for their application to the CANDU power reactors. The applicability of these codes to the analysis of the ACR-700 reactor has been assessed. In most cases, the codes are directly applicable to the ACR-700 because of the similarity of the basic design. In selected areas, the validation basis of the codes must be extended to cover the new range of application required by ACR-700 design specific parameters and operating conditions.

A comprehensive R&D program will be carried out to extend the validation of the safety analysis codes for the ACR-700. This effort is already in progress with a priority focus on the codes that have the greatest need for extended validation.

The RD-14 safety thermalhydraulics test loop has been upgraded to enable blowdown testing at the ACR-700 heat transport system design temperature and pressure. The results of tests planned for 2002/03 in the modified loop will extend the validation of the CATHENA safety thermalhydraulics code to cover ACR-700 loss-of-coolant accidents.

The smaller ACR-700 core lattice pitch and the larger calandria diameter require a redesign of the moderator circuit for removal of core heat under both normal and abnormal conditions. AECL has validated the MODTURC_CLAS code for prediction of the heat transfer for the fuel channels to the moderator using a ¼ scale moderator test facility for the current CANDU design. Changes to this facility will be designed in 2002/03 with the scaling required to validate the MODTURC_CLAS code for application to the ACR-700 core.

A test program is also planned to commence in 2002/03 to address the impact of the ACR-700 fuel design on the ELOCA safety code that is used to predict transient fuel behaviour.

Efforts to extend the validation of other, less critical, safety analysis codes is planned to commence in 2003/04. In selected areas, preparations for experiments will commence earlier. For example, the facility used for experiments to validate the TUBRUPT code (used to predict the impact of potential spontaneous fuel channel failure) will be modified to provide ACR-700 conditions in 2002/03.

6.2 Safety-Related Studies

The impact of new features in the ACR-700 design on the predicted behaviour of the reactor under accident conditions will be studied in separate effects tests. The important new safety-significant changes have been assessed and the requirements for studies have been established. The following outlines key planned activities to support the safety analysis.

The performance characteristics of the reactivity control and shut-down systems devices will be verified by experiment. As described in Sections 3.2 and 5.5, the mechanical and neutronic performance characteristics will be tested for both SDS1 and SDS2 components.

The use of SEU fuel with a negative core void reactivity and increased margins in the design of core components (e.g., calandria tube thickness) will decrease the probability and potential consequences of design basis accidents. Studies are planned in selected areas to determine the additional margin that can be credited in predicting accident consequences. Notably, the calandria tube design has been strengthened to withstand a spontaneous pressure tube failure and this predicted capability will be confirmed by testing.

The phenomena and the potential core behaviour in the event of beyond design basis events in the ACR-700 will be very similar to that anticipated for the current CANDU design. However, studies are planned in a number of areas to confirm the predicted behaviour. These include studies of the heat transfer rates for a fuel channel under loss-of-coolant and loss-of-emergency coolant injection and molten core injection into the moderator for a channel blockage event. These studies will be conducted starting in 2003/04 when prototype pressure tubes and calandria tubes are available.

The ACR-700 design will include passive heat removal capabilities. When the design options for these systems have been finalized, tests will carried out, as required, in AECL's passive containment test facility.

7. SUMMARY

A comprehensive R&D program has been established to support the development of the ACR-700 design during the basic engineering design phase. The program addresses the R&D requirements that have been identified with a particular focus on the major new features of the ACR design and those areas where longer-term development and testing is required.

The plan described here is a high-level overview of the significant R&D topics that will be addressed during the development period and is not a comprehensive detailed listing of the individual tests and studies that will be carried out. It is expected that the plan will evolve during the development period as new requirements (or changes to existing requirements) emerge from engineering, regulatory feedback, prospective customer feedback and ongoing operating product feedback.

The ACR-700 design is an extension of the current CANDU design, the ACR-700 R&D program leverages the ongoing current CANDU product support R&D. The program described here focuses on the work directly supported by the ACR-700 development program and does not include all of the activities in the broader AECL R&D program that will support the ACR-700 design.