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ACR-1000™ PROJECT - LICENSING OPPORTUNITIES AND CHALLENGES

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Abstract. Atomic Energy of Canada Limited (AECL) has developed the Advanced CANDU Reactor™ [1] 1000 (ACR-1000™) as an evolutionary advancement of the current CANDU 6 reactor. The ACR-1000 design has evolved from AECL's in-depth knowledge of CANDU™ systems, components, and materials, as well as the experience and feedback received from owners and operators of CANDU plants. The ACR design retains the proven strengths and features of CANDU reactors, while incorporating innovations and state-of-the-art technology. It also features major improvements in economics, inherent safety characteristics, and performance, while retaining the proven benefits of the CANDU family of nuclear power plants.

To ensure that the ACR design is compliant with Canadian and international requirements, regulatory pre-project reviews of the ACR-1000 (and ACR-700™ [1] with lower output) were conducted early in the design work. The regulatory feedback from these pre-project regulatory reviews helped AECL to better understand regulatory expectations in Canada, US and the UK, and to make further advancements and improvements in the ACR design to meet the Canadian and international regulatory requirements.

This paper provides an overview of the key design features of the ACR-1000 reactor design, and summary of the pre-project reviews by those above-mentioned regulatory bodies, demonstrating opportunities and challenges in licensing process of and pointing to the importance of efficient vendor-regulator interaction.

1. INTRODUCTION

Atomic Energy of Canada Limited (AECL), a vendor of nuclear power plants, has designed a two-unit Advanced CANDU™ [1] Reactor (ACR-1000™ [1]) nuclear power plant, each unit with a gross electrical output of 1165 Megawatts electrical. The ACR-1000 design is largely based on the design concepts and the reactor and process system designs of current CANDU™ [1] plants, although there are some important differences between the ACR-1000 design and existing CANDU technologies.

The Canadian nuclear reactor design evolution that has reached today's stage represented by the advanced design of the ACR-1000, has a long history dating back to the early 1950s. In this regard, Canada is in a unique situation, shared only by a very few other countries, where original nuclear power technology has been invented and further developed.

With the Canadian nuclear technology development, in parallel, the development of Canadian regulations and licensing processes took place. This latter development was carried out by the Atomic Energy Control Board (AECB), established in 1946. The AECB mandate, originally focused on security, and later extended to include nuclear safety, was focused on regulating the nuclear sector, establishing health and safety regulations, and also played an instrumental role in forming the International Atomic Energy Agency (IAEA). The AECB, which preceded the Canadian Nuclear Safety Commission (CNSC), established in 2000, licensed all CANDU power reactors in Canada and provided assistance to the overseas regulatory authorities in licensing of CANDU reactors in Korea, Argentina, China and Romania.

The regulatory framework in Canada has been changing to improve regulatory efficiency and effectiveness. The CNSC has developed its regulatory expectations for new nuclear power plants in Canada that were used to develop the ACR design. As a reactor vendor, AECL has always been fully aware of and understood Canadian regulatory requirements and expectations, and international standards such as IAEA Safety Standards. AECL also actively participates in CNSC public consultation process to provide valuable feedback as a vendor whenever the CNSC develops new regulatory standards or guides. At the same time, AECL performs self-assessments to ensure compliance with applicable Canadian and international standards for use in designing a new build in Canada.

As there is no legal framework in Canada for design certification of nuclear power plant (NPP), AECL initiated a pre-project regulatory review of the ACR-1000 reactor design by the CNSC to confirm compliance with regulatory requirements and also incorporate regulatory feedback in the design process to minimize project risks in obtaining future construction and operating licences for NPPs in Canada.

AECL has a long experience and involvement in CANDU reactor design, construction, refurbishment, maintenance and licensing in Canada and internationally. Currently, there are 33 CANDU reactors in-service worldwide with accumulated operating experience of more than 450 reactor years. The CANDU 6 fleet of 11 reactors operating worldwide has achieved 89% lifetime performance as of December 2008. AECL has taken the opportunity and advantage from the experience in design and operation of the CANDU fleet of reactors in introducing a number of design and constructions advancements.

Figure 1 shows a detailed view of the ACR-1000 heat transport system. Figure 2 shows the reactor calandria situated in the reactor vault. Figure 3 shows the ACR-1000 containment and surrounding building along with the four quadrant design approach. Figure 4 shows a view of the ACR-1000 fuel bundle.

2. ACR-1000 DESIGN OVERVIEW

2.1. Safety Design Approach

Consistent with the overall safety concept of defence-in-depth, the design of the ACR-1000 plant aims to, as far as practicable, to prevent, and reduce challenges to the integrity of physical barriers; maintain integrity of any barriers when and if challenged; and obviate failure of a barrier as a consequence of the failure of another barrier. The objectives of this approach are to provide adequate means to maintain the plant in a normal operational state, to ensure proper short-term response immediately following an initiating event, to facilitate the management of the plant in and following any Design Basis Accident (DBA), and in certain defined accident conditions, beyond the DBAs.

2.1.1. Defense in Depth

The ACR-1000 design has evolved from a proven CANDU line of products that has always used the defence-in-depth principle as a basis for design. In addition, the ACR-1000 design includes inherent, passive, engineered and administrative safety features, and incorporates the five major classic physical barriers to the release of radioactive materials to the environment, i.e., the fuel matrix, the fuel sheath, the heat transport system (HTS), containment, and the exclusion zone.

As a part of the inherent safety features, the moderator system in all CANDU designs represents a key additional heat sink, providing another means of core cooling and maintaining the barriers to the release of radioactive materials to the containment. Also, even following in-core Loss Of Coolant Accidents

(LOCAs) in which fission products are released from a channel, the fission products must pass through the moderator water, where the majority are retained.

This classic concept of defence-in-depth for physical systems is also extended and applied to management activities, including organizational, safety, behavioural, or design-related, thus in effect providing an additional safety management barrier. By ensuring that all safety-related activities are subject to overlapping provisions, even if a failure occurs, it is detected and compensated for or corrected by appropriate measures. Application of the concept of defence-in-depth in the ACR-1000 design provides a series of levels of defence (physical barriers, quadrant separation, safety management, inherent features, equipment, and procedures) aimed at preventing accidents and ensuring appropriate protection in the event that prevention fails, and ensuring not only low probabilities of occurrence but utilizing the reactor's redundant and diverse safety features as shown below.

2.1.2. Independence and Separation

Physical and functional separation of systems important to safety performing the same safety function provides independence to ensure that common cause events, such as fires, flooding, earthquakes, etc., and functional interconnections do not impair performance of the required safety functions. To address important common cause events both physical and functional separation are implemented.

ACR-1000 safety and support systems are designed in conformance with the philosophy and safety objective of physical and functional separation for Structures, Systems and Components (SSCs) important to safety required by References [1] and [2]. The separation philosophy includes separation of safety systems from process and control systems, separation between safety systems, and separation of redundant SSCs important to safety.

The four fundamental nuclear safety functions are generally provided by at least two totally redundant systems or subsystems, and the trip signals to actuate these systems are provided by four redundant instrumentation channels that feed redundant actuation logic circuitry. This is done to ensure high reliability in the execution of these essential safety functions. Independence must be ensured between redundant systems and between redundant parts of a system.

Independence is implemented in the ACR-1000 design between redundant systems and between redundant divisions and components within those systems. The "Four Quadrant (4Q) Separation Philosophy" consists of four separate areas or "quadrants" of systems important to safety, and the associated quadruplicated instrumentation channels and power supply divisions for safety system instrumentation. The loss of one quadrant due to a common mode event still allows continued safe operation with the remaining three quadrants intact (4x50% capacity approach used as required). This approach provides an advantage with regard to on-line maintenance and helps achieve high capacity factors. The ACR-1000 quadrant layout is provided in Figure 3.

2.1.3. Safety Systems

The ACR-1000 *safety* systems are those designed to shut down the reactor, remove decay heat, and limit the radioactivity release subsequent to the failure of normally operating process systems. These consist of the Shutdown System 1 (SDS1), Shutdown System 2 (SDS2), Emergency Core Cooling (ECC) system, Emergency Feed Water (EFW) system, and Containment System. Safety support systems are those that provide services needed for proper operation of the safety systems (e.g., electrical power, cooling water, and instrument air).

Shutdown System 1 (SDS1) features a mechanical rod design inserting into the low pressure moderator tank (so there is no potential for rod ejection) and quickly terminates reactor power operation and brings the reactor into a safe shutdown condition by dropping shut-off rods into the reactor core. Reactor operation is terminated when a certain neutronic or process parameter enters an unacceptable range, when any two of the four SDS1 trip channels are tripped by any parameter signal.

Shutdown System 2 (SDS2) provides a second diverse and independent chemical method of quickly terminating reactor power operation by injecting a strong neutron-absorbing solution (gadolinium nitrate) into the moderator when any two of the four SDS2 trip channels are tripped by any parameter signal.

Emergency Core Cooling System (ECC) is designed to supply emergency cooling water to the reactor core to cool the reactor fuel in the event of a LOCA. The design basis accidents are LOCA events where ECC is required to fill and maintain the HTS inventory, and remove decay heat from the fuel. The ECC function is accomplished by two sub-systems:

- The emergency coolant injection (ECI) system, which immediately injects high-pressure coolant into the HTS after a LOCA.
- The Long Term Cooling (LTC) system provides long-term injection including coolant recovery after a LOCA. The LTC system is also used for LTC after reactor shutdown following other accidents, as well as to allow for routine maintenance activities.

Emergency Feedwater System (EFW) heat removal function is accomplished providing cooling water to the secondary side of the steam generators to enable the steam generators to remove the decay heat to the ultimate heat sink on a loss of normal feedwater supply (main feedwater and start-up feedwater).

Containment System basic function is to provide a continuous, pressure-retaining envelope around the reactor core and HTS. Following an accident, the containment system minimizes release of resultant radioactive materials to the external environment below regulatory limits.

The containment system includes the steel-lined, prestressed concrete Reactor Building (RB) containment structure, main and auxiliary airlocks, containment cooling spray for pressure suppression, and a containment isolation system consisting of valves or dampers in the ventilation ducts and certain process lines penetrating the containment envelope. The design ensures a low leakage rate and provides a pressure-retaining boundary for all DBAs causing high pressure and/or temperature inside containment. The containment system automatically closes all penetrations open to the RB atmosphere when an increase in containment pressure or radioactivity level is detected. Measurements of containment pressure and radioactivity are quadruplicated and the system is actuated using two-out-of-four logic.

2.1.4. Safety Support Systems

Safety support systems provide services needed for proper operation of the safety systems for the ACR-1000 plant.

The ACR-1000 design includes a Reserve Water System (RWS) with a reserve water tank (RWT), located at a high elevation in the RB to provide an emergency source of water by gravity feed to the steam generators (back-up to EFW), containment cooling spray, calandria vessel, reactor vault, and HTS, if required.

The Electrical Power Systems (EPS) supply all electrical power needed to perform safety functions under transient and accident conditions and non-safety functions for Normal Operation (NO). The essential (safety support) portions of the systems are seismically qualified and consist of redundant divisions of standby generators, batteries, and distribution to the safety loads.

The Essential Cooling Water System (ECW) system circulates demineralized cooling water to systems important to safety. The ECW system is seismically qualified and is comprised of four separated closed loops. All four loops operate during NO.

The Essential Service Water System (ESW) disposes heat from the ECW system to the ultimate heat sink. The ESW system is seismically qualified and comprised of four separated open loops. All four loops operate during NO.

The Compressed Air System (CAS) provides service air, instrument air, and breathing air to different safety systems and power production systems in the plant.

The Chilled Water System (CWS) supplies water to air conditioning and miscellaneous equipment, and provides sufficient cooling capacity during NO. The system includes two separate systems; one shared system (between two reactor units) serving loads during normal plant power production function, and one unitized system serving loads credited for safety support function.

2.2. Plant Design Features

In line with the safety design principles outlined in the above sections, the ACR-1000 design features major improvements in economics, inherent safety characteristics, and performance, while retaining the proven benefits of the CANDU family of nuclear power plants.

The major nuclear systems, which comprise each Nuclear Steam Plant (NSP) portion, are located in the RB and Reactor Auxiliary Building (RAB). These systems include, but are not limited to, the following (noting major changes from existing CANDU practice):

- The reactor assembly, consisting of 520 channels in a reduced square lattice pitch, with larger-diameter calandria tubes (CTs) than current CANDU designs, contained within a calandria vessel. Figure 3 presents the ACR-1000 reactor assembly.
- The moderator system with a reduced volume of D2O as compared to CANDU 6 on a per MWe output basis.
- The HTS with light water coolant operating at higher temperatures and pressures than current CANDU designs, in a two-loop, figure-of-eight configuration with four steam generators, four HTS pumps, four reactor outlet headers, and four reactor inlet headers. CANDU 6 reactors operate with heavy water as the primary coolant in the HTS.
- The fuel handling system, which consists of two fuelling machines, each mounted on a fuelling machine bridge and columns, located at both faces of the reactor to allow for on-line refuelling.
- The main steam supply system, with higher pressure and temperature conditions than the current CANDU designs, for improved turbine cycle efficiency.
- Safety systems, specifically, two shutdown systems, the emergency core cooling (ECC) system, the emergency feedwater (EFW) system (defined as the emergency heat removal system), and the containment system, and associated safety support systems.

The Balance-of-Plant (BOP) consists of the Turbine Building (TB), the steam turbine, the generator and condenser, the feedwater heating system with associated auxiliaries, and electrical equipment. The BOP also includes the plant cooling and service water systems, water treatment facilities, auxiliary steam facilities, service and breathing air systems, BOP pumphouses and optional cooling towers, main switchyard, and other systems, equipment, and components credited for the plant power production function in the ACR-1000 two-unit plant.

2.2.1. Fuel Channels and Calandria Assembly

The reactor assembly consists of 520 horizontally-aligned fuel channels arranged in a square pitch consistent and typical for all CANDU designs. The fuel channels are mounted in a calandria vessel containing the D_2O moderator. Each fuel channel assembly consists of a pressure tube (PT), two end fittings, and associated hardware. The PTs contain the LEU fuel and the high-pressure light water coolant. Individual calandria tubes (CTs) surround each individual PT. The end fittings are out of core extensions of the PT, and extend out of the end shields past the feeder cabinets. The end fittings provide connections to the fuelling machine head for on-line refuelling and to the feeder pipes.

The calandria vessel, shown in Figure 2 has end shields located at both ends. They are filled with shielding balls and water to provide shielding. The fuel channels are located by adjustable positioning assemblies on the two end shields and are connected by individual feeder pipes to the HTS. The calandria vessel is enclosed in a concrete vault (reactor vault) filled with light water for shielding (Figure 2). The reactor vault is closed at the top by the reactivity mechanisms deck. Both the moderator in the calandria vessel and light water in the reactor vault provide additional heat sink capability for beyond design basis accidents (BDBAs).

2.2.2. Fuel

The CANFLEX-ACR™ [2] fuel (see Figure 4) represents the next evolution in fuel design beyond what is currently used in the Pickering, Bruce, and all of the CANDU 6 reactors. The fuel design is a modified CANFLEX™ [2] type fuel bundle similar to that already demonstrated in the CANDU 6 reactor at Point Lepreau. The fuel consists of 42 elements containing uranium dioxide fuel pellets plus a central element containing burnable poison in a zirconia matrix. The uranium dioxide pellets are made with LEU. The fuel element sheaths are made from zirconium alloy. The 43 elements are assembled between end plates to form a fuel bundle. Each of the 520 channels contains 12 bundles. The fuel enrichment of the reference core is 2.4%, and the average fuel
burnup is 20,000 MWd/t.

2.2.3. Reactor Control Units

The reactivity control units (RCUs) are comprised of the in-reactor sensor and actuation portions of reactor regulating and shutdown systems. RCUs include neutron-flux measuring devices (vertical and horizontal flux detector units, ion chamber units, and fission chamber units), reactivity control devices (zone control units and control absorber units), safety shutdown devices (shutdown units and liquid injection shutdown units), and GSS devices. RCUs are designed to be simple, rugged, highly reliable, and require little maintenance.

Flux detectors are provided in and around the core to measure neutron flux, and reactivity control devices are located in the core to control the nuclear reaction. In-core flux detectors are used to measure the neutron flux in different zones of the core. These are supplemented by fission chamber and ion chamber assemblies mounted in housings on the calandria shell. The signals from the in-core flux detectors are used to adjust the location of the zone control unit assemblies to compensate for changes in power levels and distribution.

Control absorber unit elements penetrate the core vertically. These are normally parked out of the reactor core and are inserted to control the neutron flux level at times when a greater rate or amount of reactivity control is required than can be provided by the zone control units. Slow or long-term reactivity variations are controlled by the addition of a neutron-absorbing poison to the moderator. Control is achieved by varying the concentration of this "neutron-absorbent material" (i.e., gadolinium nitrate and boron) in the moderator.

A long-term guaranteed shutdown state (GSS) can be achieved by the reactor regulating system either by addition of poison to the moderator, or by insertion of dedicated GSS rods into the core. A dedicated system of GSS rods is provided to allow the reactor to be maintained in a GSS without the use of moderator poisons during planned and unplanned maintenance outages. During normal reactor operation, the GSS rods are withdrawn from the core.

2.2.4. Heat Transport System

The HTS (see Figure 1) circulates pressurized light water coolant through the reactor fuel channels to remove heat produced by nuclear fission in the core. The fission heat is carried by the HTS coolant to the steam generators, to produce steam on the secondary side that subsequently drives the turbine generator. The HTS is complemented by auxiliary systems: the pressure and inventory control (P&IC) system, HTS purification system, and HTS pump seal system. The HTS and its auxiliary systems are similar to those in the CANDU 6 design. However, the overall design of these systems has been improved and optimized, based on operational feedback from existing CANDU plants.

The major components of the HTS are the 520 reactor fuel channels and associated feeders, four steam generators, four HTS pumps, four reactor inlet headers, and four reactor outlet headers configured in two figure-of-eight loops with interconnecting piping. Light water coolant is fed to the fuel channels from the inlet headers at each end of the reactor and is returned to the outlet headers at the opposite end of the reactor.

2.2.5. Steam Generator Design

Four identical steam generators with integral preheaters transfer heat from the HTS coolant on the steam generator primary side to raise the temperature of, and boil, feedwater on the secondary side of the steam generator. The steam generator consists of an inverted vertical U-tube bundle installed in a shell. Steam-separating equipment is housed in the upper portion of the shell. A venturi flow restrictor is installed at the outlet nozzle of each steam generator to reduce the pressure inside the RB containment in the event of a main steam line break (MSLB).

2.2.6. Heat Transport System Pump Design

The four HTS pumps are vertical, single-stage centrifugal pumps with single suction and double discharge, each driven by a vertical, totally enclosed, air-to-water cooled squirrel cage induction motor. The motor has built-in inertia to prolong pump rundown on loss of power. A gland seal circuit supplies cooled and filtered water for lubricating and cooling the mechanical seals. A leakage recovery cavity takes the seal leakage to the light water leakage collection system.

2.2.7. Moderator System

Neutrons produced by nuclear fission are moderated by the D2O in the calandria. The D2O moderator is circulated by the moderator pumps through the calandria at a relatively low temperature and low pressure, and cooled by the moderator heat exchangers. The moderator heat exchangers remove the nuclear heat generated in the moderator and the heat transferred to the moderator from the fuel channels. Helium is used as a cover gas over the D2O in the calandria. Chemistry control of the moderator water is maintained by the moderator purification system. The moderator system also acts as a back-up heat sink under certain postulated BDBA conditions.

2.2.8. Fuel Handling System

The fuel handling system is used to fuel the reactor on demand for the purpose of controlling the reactivity distribution and reactor power distribution in the long term. The fuel handling system stores and handles fuel, from the arrival of new fuel to the storage of spent fuel. The fuel handling system is divided into new fuel handling and storage, fuel changing, and spent fuel handling and storage.

Fuel changing is performed on-power and remotely, using two fuelling machines. One fuelling machine is connected to each end of the fuel channel being fuelled. A two-bundle shift scheme is used for the reference core. In the two-bundle shift operation, two new fuel bundles are inserted at the inlet end of the channel and two spent fuel bundles are removed from the outlet end of the fuel channel.

Fuel is cooled by the fuel handling system once it is removed from the fuel channel, and it remains in the fuel handling system until it is reinserted in the channel or discharged into the spent fuel bay. The normal refuelling sequence does not require reinsertion. The spent fuel bundles remain fully submerged in water while being transferred from the fuelling machine in the RB, to the spent fuel reception bay in the RAB. The spent fuel bay cooling and purification system removes the decay heat generated by the fuel, removes the suspended activation products, and controls the water chemistry.

3. ACR-1000 PRE-PROJECT REGULATORY REVIEWS

AECL has initiated several regulatory pre-project reviews during the development of the Advanced CANDU reactor (both ACR-700™ [3] and ACR-1000). These regulatory reviews were conducted with the following objectives:

- Identify any potential regulatory issues early in the design, so there is time to address them before project commitment;
- Mitigate and reduce licensing risk, and ensure cost and schedule of the project is acceptable before a project is committed; and
- Ensure good understanding, implementation and compliance with the regulatory requirements expected of new NPPs in Canada, and in foreign jurisdictions, which are higher compared to the past regulatory practice.

To achieve the above objectives, AECL has engaged the US NRC in pre-application review of the ACR-700 (2002-2004), the UK NII in pre-project design assessment of ACR-1000 (2007-2008), and the Canadian Nuclear Safety Commission (CNSC) on ACR-700 (2003-2006) and ACR-1000 (2008-2009). The process, scope and key findings of the above pre-project reviews are covered in the following sections.

3.1. USNRCReview of ACR-700

The US NRC has a formal process for Design Certification of new reactor designs. Its policy encourages early discussions (prior to such an application) between the NRC and potential applicants, such as utilities and reactor designers. These discussions provide applicants with licensing guidance and help them to identify and resolve potential licensing issues early in the licensing process. Designers have the opportunity to ask the US NRC to review specific areas in advance, in a process called "Pre-Application Review". In June 2002, AECL initiated the pre-application review of the ACR-700 design for compliance of a pressure-tube heavy-water-moderated reactor with US requirements. The scope encompassed the assessment of the following specific focus topics:

- Class 1 Pressure Boundary Design;
- Design-Basis Accidents and Acceptance Criteria;
- Computer Codes and Validation Adequacy;
- Severe Accident Definition and Adequacy of Supporting Research and Development;
- Canadian Design Codes and Quality Assurance Standards;
- Distributed Control Systems and Safety Critical Software;
- On-Power Fuelling;
- Confirmation of Negative Void Reactivity;
- ACR Probabilistic Risk Assessment Methodology; and
- CANFLEX Fuel Design.

The NRC Staff did a thorough review that was documented in the "Pre-Application Safety Assessment Report (PASAR) related to the Advanced CANDU Reactor 700 MWe" issued in October 2004.

The US NRC regulations have been developed and written exclusively for LWRs. Therefore, the potential policy and regulatory issues identified by the US NRC Staff that required a resolution during the licensing process were mostly related to the differences in the ACR-700 design compared to the LWR designs. Some of those differences include: the multi-channel core, use of pressure tubes and pressure-tube materials, heat transport system pressure boundary at the rolled joint between the pressure tube and the end fitting, or the connection of a fuelling machine(s) to the heat transport system as an extension of the reactor coolant pressure boundary.

The issues raised in the PASAR are not considered major, and these could be resolved during the licensing process. A number of these were specific to the ACR-700 design, some of CANDU generic nature. AECL prepared a strategy to address US NRC comments and suggestions, and these strategies for the most important issues are already implemented in the ACR-1000 design, and followed-up in the Canadian pre-project design review process.

The US NRC Staff s summary position on the potential for success of a Design Certification application for the ACR-700 concludes the PASAR, as follows: "Notwithstanding, based on the information provided, the Staff believes at this time that AECL will ultimately be able to satisfactorily address these policy, regulatory, and technical issues during the design certification review."

3.2. UK NII Review of ACR-1000

The UK set up a process for regulatory design review prior to a Project Commitment - called a "Generic Design Assessment" (GDA). The relevant UK regulatory agencies involved were the Nuclear Installations Inspectorate (NII), the Office for Civil Nuclear Security (OCNS) (both part of the Health and Safety Executive's (HSE) Nuclear Directorate (ND)) and the Environment Agency. The NII regulates nuclear safety aspects; the OCNS regulates security aspects, while the Environment Agency regulates the environmental aspects of licensing in the UK. The GDA allows the technical assessments of the reactors to be conducted before any specific nuclear site licence assessments are undertaken, thus identifying and resolving any potential regulatory issues before commitments are made to construct the reactors. The GDA conducted by HSE consisted of four steps and will take approximately four years to complete. The Environment Agency's process consisted of two stages, with the first stage being the design review and the second stage covering site licensing and authorizations for various permits. The time frame was similar to the HSE assessment.

- Step 1 (Initiation), which covers preparatory work, was completed by AECL in July 2007.
- In August 2007, AECL submitted the documentation for Step 2 (for NII and OCNS) and for the Preliminary Assessment (Stage 1 for Environment Agency). The aim was to carry out an overview of the fundamental acceptability of the ACR-1000 in the UK regulatory regime. As part of the Step 2 assessment, HSE requested that the International Atomic Energy Agency (IAEA) undertake a technical review of all four Requesting Parties' designs against the relevant IAEA standards. The IAEA high-level review was published in a summary report, which, stated in summary that *"IAEA did not reveal any fundamental safety problems with the ACR-1000".*

The reviews were completed in March 2008 with the publication of interim reports by HSE and EA. In the Step 2 interim report, NII summarized the findings of the Step 2 review on safety and security aspects. The Step 2 examination of AECL's documentation demonstrated that: *"...in summary, NII has not found any safety or security shortfalls that are so serious as to rule out at this stage eventual construction of the ACR-1000 design on licensed sites in the UK".*

3.3. CNSC Review of ACR-1000

The Canadian Nuclear Safety Commission (CNSC) is Canada's sole nuclear regulatory agency operating under the Nuclear Safety and Control Act (NSCA). The CNSC regulates the use of nuclear energy and materials to protect the health, safety and security of Canadians and the environment, and to respect Canada's international commitments on the peaceful use of nuclear energy.

On request from a reactor vendor, the CNSC conducts pre-project design reviews which assess a nuclear power plant design based on a vendor's proposed reactor technology. A pre-project design review is solely intended to provide early feedback on the acceptability of select aspects of a nuclear power plant design based on Canadian regulatory requirements and expectations. This review does not certify a reactor design or involve the issuance of a licence under the NSCA, and it is not required as part of the licensing process for a new nuclear power plant. The conclusions of any design review do not bind or otherwise influence decisions made by the Commission. The CNSC undertakes a more detailed review of a design and safety case when it considers an application for a licence to construct a nuclear power plant at a specific site.

In April 2008, AECL requested the CNSC to perform a pre-project design review of the ACR-1000, which was originally proposed to include two phases (with a potential for a third phase), starting on April 1, 2008, and ending on August 30, 2009.

3.3.1. Design Review Objectives

As with other design reviews conducted by the CNSC, the objectives of this review were to:

- Assess whether the ACR-1000 design as submitted was, at an overall level, compliant with the CNSC regulatory requirements;
- Assess whether the design provisions provided for selected review focus areas met the CNSC's expectations for new nuclear power plants in Canada; and
- Identify, based on the review of the focus areas, whether there were any potential fundamental barriers to licensing the ACR-1000 design in Canada.

A vendor pre-project design review provides an opportunity for the CNSC staff to assess the design prior to any licensing activities, and to identify potential issues for resolution relating to the compliance of a design with regulatory requirements and expectations. Such a review helps increase regulatory certainty and ultimately contributes to public safety.

3.3.2. Design Review Phases

The CNSC pre-project review consisted of two main phases, with a third phase optional to follow-up on selected topics from the first two phases.

- *Phase 1: Assessment of Compliance with Regulatory Requirements.*
  This phase was an overall assessment of the information submitted for 16 review focus areas. Its purpose was to determine whether the design intent in these areas is compliant with the CNSC requirements and meets the CNSC's expectations for the design of new nuclear power plants in Canada. The Phase 1 review of the ACR-1000 design was completed in December 2008, and the CNSC Phase 1 report provided to AECL concluded that, at an overall level, the design intent was compliant with the CNSC's requirements and expectations.
- *Phase 2: Identification of Fundamental Barriers to Licensing.*
  Subsequent to Phase 1, this phase went into further detail in each of the focus areas with the intent of identifying whether there are any potential fundamental barriers to licensing the design in Canada. This phase was completed on August 30, 2009, with CNSC issuing the final review report stating that the ACR-1000 design does not have any fundamental barrier to licensing in Canada.

- *Phase 3: Follow-up on selected topics from Phase 2.*
  Subsequent to Phase 2, the main objective is to review and report on the implementation of AECL's planned activities in a number of focus areas and selected topics which have been reviewed and discussed with the CNSC staff during Phase 2.

3.3.3. Design Review Scope

To facilitate the Phase 2 review, AECL submitted documentation including the ACR-1000 Technical Description, ACR-1000 Generic Safety Case Report, and the Safety Design Guides used by the designer and information on the research and development (R&D) being undertaken. Additional information was submitted as requested by CNSC staff in support of the review. In performing the Phase 2 review, CNSC staff aimed to identify: items requiring further information, items requiring further follow-up, issues for which there was clear nonconformance with regulatory expectations, or issues that could lead to potential fundamental barriers.

For the Phase 2 review, CNSC staff retained the same 16 review focus areas as for Phase 1, and added one supplementary focus area - the ACR-1000 R&D program:

- Defense in depth, classification of systems, structures and components (SSCs), dose acceptance criteria
- Reactor core nuclear design
- Means of shutdown
- Fuel design
- Reactor control system
- Emergency core cooling system and emergency feedwater system,
- Containment
- Severe accident prevention and mitigation
- Pressure-boundary design provisions for the primary heat transport system,
- Fire protection
- Radiation protection
- Quality assurance program
- Human factors
- Out-of-core criticality
- Robustness, safeguards, and security
- R&D program
- Safety analysis (deterministic and probabilistic)

3.3.4. Design Review Criteria

For each of the design review focus areas, CNSC staff assessed the submitted documentation against the following requirements and expectations:

- CNSC Nuclear Safety and Control Act and Regulations;
- CNSC Regulatory documents, in particular regulatory document "Design of New Nuclear Power Plants" (RD-337) [2]; and
- Canadian CSA Standards and Codes, and International Standards.

3.3.5. Design Review Findings

CNSC staff acknowledged that, throughout the Phase 2 review, AECL staff was open and transparent in sharing available information, and that they responded diligently to every CNSC request for clarification and additional information.

The Phase 2 review was based mainly on the ACR-1000 Generic Safety Case Report (GSCR) that was produced in June 2008. This was done as part of AECL's Basic Engineering Program Phase in which sufficient engineering and safety analysis of the nuclear power plant was completed on a generic level to be ready for project implementation. A GSCR is a higher level non-site specific precursor to a Preliminary Safety Analysis Report (PSAR) which would be submitted as part of an application for a licence to construct for a specific site. A PSAR contains more complete information on the design to demonstrate that it is ready for construction, and more information on the analysis and R&D program to fully demonstrate the adequacy of the design. Note that AECL prepared a first revision of a generic PSAR for ACR-1000 in September 2009.

For the Phase 2 review, particular attention was given to the focus areas where:

- There are new features in the design. This was to ensure that AECL has performed or has planned the work for testing and analysis to prove the adequacy of the design; and
- RD-337 (Reference [2]) as applied to the design of new nuclear power plants sets expectations higher than or departing from past practice.

During the Phase 2 review, CNSC staff extended the scope of review focus areas to include AECL's R&D program. AECL has provided an overview of the ACR-1000 R&D program in support of the development of the ACR-1000 design, and details for four R&D areas that CNSC staff selected, based on their significance in terms of making a safety case for the ACR-1000 design. The four R&D areas are related to: reactor physics (which was a principal part of the focus review area for reactor core nuclear design), the fuel design, thermal-hydraulics design, and severe accidents to support the containment design.

3.3.6. Design Review Conclusions

In summary, based on the review of the 17 focus areas, the CNSC staff concluded that there are no fundamental barriers to licensing the ACR-1000 design in Canada. It should be noted that this conclusion is subject to the successful completion of AECL's planned activities, in particular those related to R&D.

This overall conclusion was based on the following key CNSC findings:

- AECL has provided sufficient design and analysis information for the purpose of the reviews. At a high level, the design intent is compliant with the CNSC regulatory requirements and meets the expectations for new nuclear power plants in Canada.
- CNSC staff s review of the 17 focus areas did not identify any fundamental barriers to licensing the ACR-1000 design in Canada, subject to the successful and timely completion of outstanding R&D, and the resolution of key findings in the focus areas.
- The overall ACR-1000 R&D program was derived logically from the existing knowledge base and appears to be comprehensive. Also, the R&D program for the four selected topics appears to be adequate. CNSC staff expects all key safety-related R&D to be completed prior to the submission of an application for a licence to construct.
- For any nuclear power plant the commissioning program plays an important role in verifying that systems, structures and components are correctly installed and will function or operate as intended. This is particularly important for those design features which are new or first-of-a-kind. As part of a licence application, CNSC staff would expect a commissioning program to be submitted that is commensurate with industry best practice that would verify to the extent practicable that new features of the ACR-1000 design will function in accordance with their design requirements.

4. CONCLUSIONS

Based on decades of design development and R&D of well-established CANDU line of reactor designs in Canada, AECL developed the ACR-1000, 1165 MWe-class light-water-cooled, heavy-water-moderated pressure-tube reactor. The ACR-1000 design retains the basic, proven, CANDU design features while incorporating innovations and state-of-the-art technologies to ensure fully competitive safety, operation, performance and economics.

AECL has requested and completed regulatory pre-project reviews from US NRC, UK NII, and recently from CNSC. The CNSC Phase 2 review completed in August 2009 confirmed that the ACR-1000 design complies with applicable Canadian regulatory requirements, and that the design does not have any fundamental barriers to be licensed in Canada.

REFERENCES

[1] "Safety of Nuclear Power Plants: Design Safety Requirements", IAEA Safety Standards Series NS-R-1, September 2000.

[2] "Design of New Nuclear Power Plants", CNSC Regulatory Document RD-337, November 2008.

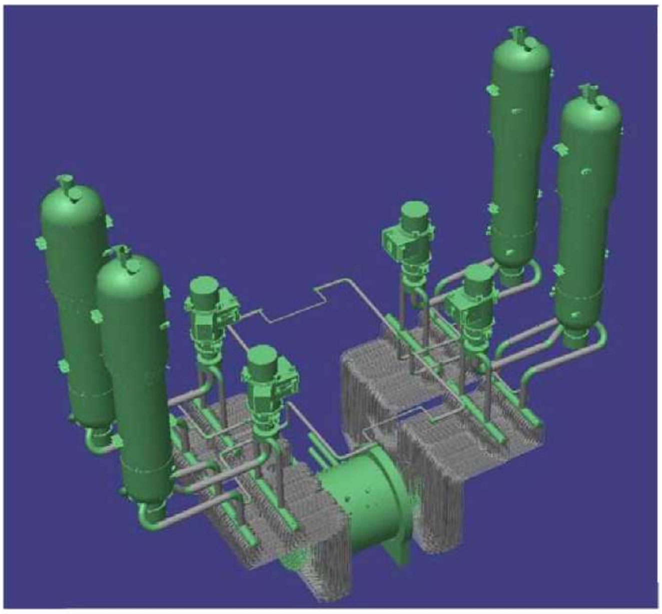

Fig. 1. Schematic of the ACR-1000
Heat Transport System

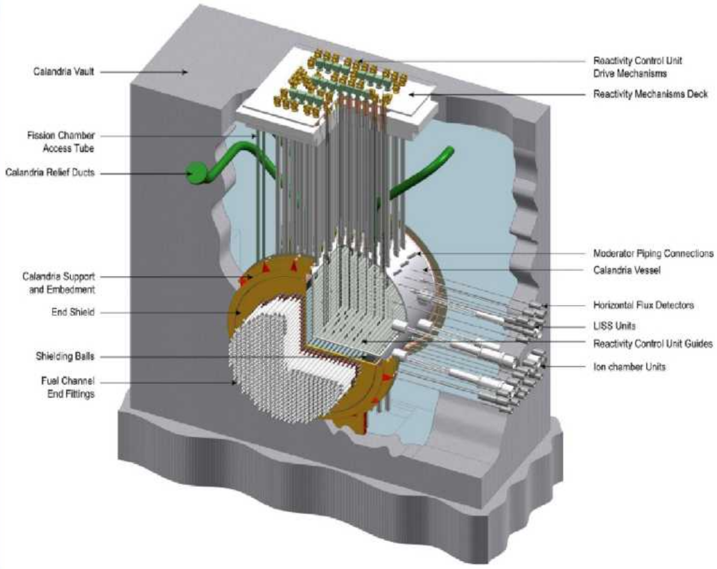

Fig. 2. ACR-1000 Fuel Vault Assembly

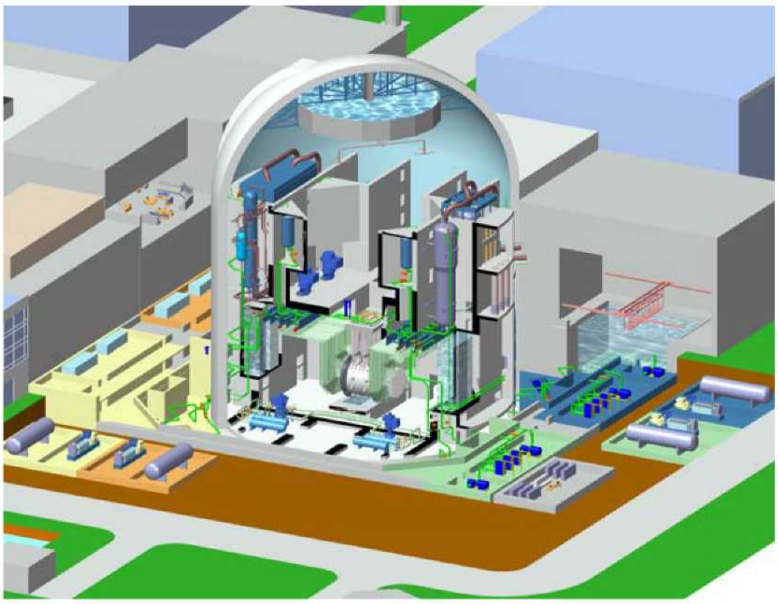

Fig. 3. ACR-1000 Quadrant Layout

Fig. 4. ACR-1000 Fuel Bundle