AECL
TECHNOLOGIES INC.

May 28, 2004

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

Attention: Ms. B. Sosa
Project Manager, ACR

References:

1. E-mail J. Kim to R. Ion, "Questions on the Design Basis Accidents & Severe Accidents", February 11, 2004.
2. E-mail R. Ion to J. Kim and J. Lee, "Proposed Accident Source Term for ACR-700 (advanced copy)", May 27, 2004.

Re: **Proposed Accident Source Term for ACR-700**

In the April 6-7, 2004, meeting between AECL and the NRC on Design Basis Accidents and Severe Accidents, AECL has made presentations to cover a set NRC questions that have been sent to AECL with Reference 1. While the NRC's questions have been addressed during the April 6-7 meeting, AECL has been requested to prepare a position paper on the proposed accident source term for ACR-700. This paper is provided in attachment 1 to this letter.

An advanced copy of this paper was sent to you via electronic mail on May 27, 2004 (Reference 2).

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

Yours sincerely,

Vince Lang

Vince J. Langman
ACR Licensing Manager

/Attachment:

1. Proposed Accident Source Term for ACR-700

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Attachment 1:
(Letter V. Langman to B. Sosa, "Proposed Accident Source Term for ACR-700", May 28, 2004)

Proposed Accident Source Term for ACR-700

1.0 Purpose

At the April 6-7, 2004 meeting between AECL and NRC on the subject of Design Basis Accidents, AECL made a series of presentations describing the methodology for calculating source terms from containment for dose analysis purposes. In particular AECL presented the sequence of events and some results for a large LOCA with loss of emergency core cooling (LOCA/LOECC). For ACR-700, the LOCA/LOECC source term is proposed by AECL in place of the alternate source term for LWRs described in US NRC Regulatory Guide 1.183 (Reference 1) for siting and licensing. LOCA/LOECC is the Maximum Credible Accident that results in a large fission product release from the fuel.

The purpose of this report is to summarize the technical rationale for proposing this scenario-specific mechanistic source term rather than the one in RG 1.183, which was derived from LOCA/LOECC accidents in LWRs and BWRs.

2.0 ACR-700 Design Features

The ACR-700, like other CANDU reactors, is a pressure tube reactor with multiple, separated fuel channels surrounded by a cool, low pressure heavy water moderator, contained within a calandria vessel rather than a LWR core contained within a reactor vessel. The following description of the ACR-700 focuses on design features that are significant in terms of the response of the core to a large LOCA/LOECC.

2.1 Core and Reactor Coolant System Design

The ACR-700 design is based on the use of modular horizontal fuel channels surrounded by a heavy water moderator, the same feature as in all CANDU reactors. The ACR-700 core consists of 292 individual fuel channels each containing 12 short (19-inch long) fuel bundles placed end-to-end in the channel. Each fuel bundle consists of 43 individual fuel pins arranged in three concentric rings surrounding a center pin. In an equilibrium core, the powers of the 3504 fuel bundles range from about 100 kW to about 900 kW. The core average element linear power is on the order of 27 kW/m. The distribution of fission products in the fuel bundles is largely a function of the bundle design and operating conditions, which are unique to the ACR-700 design.

A characteristic of the CANDU design is that fuel bundles are not directly surrounded by more fuel bundles, but by two tubes and a cool water moderator, and that no fuel rod is more than about 4 inches from the moderator.

Each of the 292 channels in an ACR-700 has an inlet feeder connecting it to one of the two reactor inlet headers, and an outlet feeder connecting the channel to one of the two reactor outlet headers. Each header has 146 feeder connections. Light water coolant is pumped into the inlet headers and the coolant flows through each of the feeders into the inlet end-fitting of the individual channels. Coolant then flows axially through the fuel bundle string, removing heat generated in the fuel bundles. Coolant exits through each channel's outlet end-fitting to the outlet feeder. Coolant from each individual channel mixes again in the reactor outlet header. Steam generators that are connected to the reactor outlet headers remove the heat from the coolant. The coolant exiting the steam generators is pumped into the inlet header in the opposite core pass. The arrangement of the core into segregated channels each with individual feeder pipes is a unique feature that affects the progression of accidents, including LOCA/LOECC accidents. Additionally, the end-fittings and feeders provide a considerable surface area for fission product deposition after their release from failed fuel elements in the channel.

An individual fuel channel consists of a horizontal pressure tube which contains the 12 fuel bundles, through which the light water coolant flows. Each pressure tube is surrounded by a calandria tube. A gas flows in the annulus between the pressure tubes and calandria tubes, which allows for leak detection.

The array of 292 channels is contained in a large vessel called the calandria vessel. The calandria vessel contains the heavy-water moderator. Reactivity measurement and control devices are located in the calandria vessel in the spaces between fuel channels. The moderator temperature and pressure are controlled at specific setpoints. The presence of a large volume of moderator water in the vicinity of the core provides an additional passive heat sink for LOCA/LOECC accidents, which is a significant difference from pressure vessel reactors.

2.2 Safety Systems

Safety systems include shut down systems, emergency core cooling system and containment. These safety systems are similar in function to systems in other water-cooled nuclear power reactors.

2.2.1 Shutdown Systems

There are two independent reactor shut down systems operating on diverse principles. Shut down system 1 is an array of solid absorber rods that are inserted vertically into the moderator in the spaces between fuel channels. Shut down system 2 is liquid poison that is injected directly into the moderator through horizontal nozzles spanning the length of the calandria vessel.

2.2.2 Emergency Core Cooling System

The emergency core cooling system consists of a high-pressure injection stage and a medium pressure long-term cooling stage. The system includes a steam generator crash cooldown function to rapidly depressurize the reactor coolant system to promote rapid continued injection of emergency coolant. The system automatically detects a LOCA by measuring the pressure in the headers and generating an ECC initiation signal if the pressure falls below a setpoint, and other conditioning parameters are met.

There are two sources of water available for the ECC system: the accumulator tanks containing water at relatively high pressure and the reserve water tank containing water at a high elevation. When the ECC signal is generated, some water from the reserve water tank is directed to the reactor building sumps to provide a water source for the LTC.

There are three flow paths for injection of the make-up water into the reactor coolant system: direct injection under pressure from the accumulator tanks, pumped injection from the reactor building sump and passive injection from the high-elevation reserve water tank. In all cases, injection is directed to the two reactor inlet headers. The injection of water from the accumulator tanks is called emergency coolant injection (ECI) and the pumped injection is called long-term cooling (LTC) system.

2.2.3 Containment System

The containment for ACR-700 consists of a concrete, steel-lined containment envelope with a low design leak rate, an isolation system that rapidly closes openings in the containment envelope upon detection of high building pressure or high activity, an energy suppression system to mitigate the pressurization after an accident and an atmospheric control system.

2.3 Reserve Water System

The ACR-700 has an emergency water supply in the reserve water tank located at a high elevation in the reactor building. This reserve water system serves both as a safety support system for design basis accidents and a means of strengthening defence against beyond design basis accidents. Connections are made from this tank to all the major fluid systems in the containment: reactor coolant system, moderator system, shield cooling system, steam generator secondary side and reactor building sump. The connections to the reactor coolant system and to the reactor building sump are automatically opened on either the ECC initiation signal or crash cooldown signal. Flow to these various systems is by gravity.

3.0 Classification

A LOCA/LOECC in ACR-700 is a major accident that is beyond the design basis. LOCA frequency is low because of the high-quality, high-integrity design of the Reactor Coolant System (RCS) piping. The ECC system is designed to be highly reliable and incorporates considerable redundancy in components.

In the ACR-700 design, there is no single active component failure that can result in failure of either of the ECC water sources or any of the three flow paths and there is no single passive component failure that can result in failure of the LTC system in the long term (recirculation) after a LOCA. Therefore all the three systems (ECI, LTC, and reserve water system) are designed to meet the single failure criterion.

Hence, LOCA/LOECC accidents can only result from multiple independent failures. The frequency of such accidents is expected to be similar to LOCA/LOECC in LWRs. In ACR-700, scenarios that could lead to fuel melting would have additional failures beyond LOCA/LOECC, and are not considered to be credible because of their low probability of occurrence.

4.0 LOCA/LOECC Accident Sequence

In a LOCA/LOECC accident sequence, a large diameter pipe in the reactor coolant system (one of the two inlet headers) is postulated to break and discharge coolant into containment. The reactor trips and signals are generated to initiate the emergency core coolant (ECC) system.

The limiting, credible ECC system failures that are analyzed are (1) failure of ECI and LTC but with passive injection from the reserve water tank available until depletion of the water source, and (2) failure of LTC and passive injection from the reserve water tank, but with ECI available. The main difference between the two ECC failure scenarios is the timing of events. Additional failures or combinations of failures of the ECC system are not credible.

For either scenario, without effective addition of cooling water to the reactor coolant system, inventory continues to decrease, and fuel and pressure tube temperatures increase. Fuel heatup, clad oxidation, clad failure and consequent fission product release occurs in most channels.

Eventually, the pressure tubes may become hot enough to deform under the weight of the fuel in the channel. By this time, the reactor coolant system pressure is too low to drive pressure tube creep. Any subsequent heat up of the pressure tubes may result in sagging and contact with the calandria tubes. The portion of a pressure tube, which contacts a cold calandria tube, drops in temperature significantly. Heat is transferred from the calandria tube to the surrounding moderator. The moderator cooling system removes heat from the moderator.

Because of the heat removal via the moderator, LOCA/LOECC in ACR-700 has a well-defined end-state that does not involve core melt. This represents an extra level of defence compared to pressure vessel type reactors. Preserving the channel boundary is further enhanced in ACR-700 because of passive makeup to the RCS and the moderator.

The proposed LOCA/LOECC source term therefore represents a combination of a limiting fault combined with the failure of the main safety system designed to prevent fuel damage, the ECC system. The dose analysis using the LOCA/LOECC source term, which will be presented at the time of the Design Certification application, is a test of the mitigating systems, in the case of ACR-700 the moderator and the containment.

LOCA/LOECC in ACR-700 is the only accident that results in substantial damage to most, if not all of the fuel bundles in the core. Certain low probability accidents that affect a single channel (e.g., severe channel flow blockage) can also result in significant damage to fuel, however the damage is limited to the 12 fuel bundles in the affected channel – fuel bundles in the other 291 channels are well cooled and no fuel clad failures occur. Therefore, releases for these types of accidents are bounded by those for LOCA/LOECC.

5.0 LOCA/LOECC Safety Analysis Methodology and Results

The results of a preliminary analysis of the LOCA/LOECC accident were presented at the April 6-7, 2004, meeting. The maximum fuel temperatures after shutdown predicted by CATHENA are on the order of 1400°C. The maximum fuel cladding temperatures are also on the order of 1400°C. Safety analysis approach and general methodologies used in this analysis are described in References 2 and 3.

The ELESTRES and ELOCA codes are used to estimate the timing of fuel clad failures. Most of the fuel clads in the core are predicted to fail and the gap inventory of fission products is released.

Detailed fuel behavior and fission product release analysis is ongoing; however the total fission product release from fuel for the LOCA/LOECC accident has been estimated using the CORSOR-O correlation. The total release percentage of volatile fission products from fuel was estimated to be on the order of 10% of the core inventory.

Detailed analysis of fission product release and behavior will be performed for the ACR-700. The analysis is supported by a large body of experimental work on fission product release from fuel and fission product behavior after its release from fuel, described in Reference 4. The experimental data is used to validate the computer codes used in the analysis.

The source term into containment will be calculated in terms of detailed isotopic release, timing and magnitude and will be provided at a later date with the Design Certidication application.

The following sections describe the computer tools that will be used for the analysis of fission product release from fuel and into containment.

5.1 Fission Product Release from Fuel

Detailed calculations of fission product release will be performed using the SOURCE IST 2.0 code. SOURCE IST 2.0 is part of the Canadian Nuclear Industry Standard Toolset (IST) code developed and validated to calculate the extent of release of fission products from a fuel element during normal operation and transient (postulated accidents – design basis and limited core damage accidents) conditions. The code calculates the behavior of fission products in the fuel from the beginning of the normal operating conditions irradiation through to the end of the accident.

The SOURCE IST 2.0 code has been validated using the data from a large database of experimental data. The tests for the SOURCE IST 2.0 code validation include a number of laboratory single effects tests (where temperatures and atmospheres are controlled), as well as in-reactor tests both at Canadian facilities (BTF tests) and international facilities (PHEBUS tests).

5.2 Fission Product Transport and Retention

In addition to fission product release mechanisms, transport and retention mechanisms can also be credited for reducing the fission product source term into containment. In the ACR-700 design the pressure tubes, channel end-fittings and feeders provide ample surface areas and temperature gradients for the deposition and retention of fission products. To provide an estimate of the amount and types of species deposited on the ACR-700 RCS piping, the computer code SOPHAEROS will be used.

The SOPHAEROS code was developed by IPSN (Institut de Protection et de Sureté Nucléaire, France) to calculate fission-product (FP) transport and retention under Light Water Reactor (LWR) Reactor Coolant System (RCS) severe accident conditions. As with the fission product release code SOURCE, validation exercises for SOPHAEROS are on-going using laboratory single effects tests as well as in-reactor tests both at Canadian and international facilities.

6.0 Summary

The ACR-700 design has significant design and operational differences from pressure vessel reactors; therefore it is proposed to use a source term that is calculated for the ACR-700 rather than the alternate source term described for LWR and BWR given in Reference 1. The limiting, credible accident scenario for ACR-700 is a LOCA/LOECC scenario, which is a major, beyond design basis accident that results in appreciable release of fission products into containment. Detailed deterministic analysis will be performed for LOCA/LOECC. The detailed isotopic release, timing and magnitude of the source term will be provided at a later date with the Design Certification application. A preliminary analysis using the CORSOR-O correlation for fission product release from fuel suggests that the releases will be on the order of 10% of the total core inventory of volatile fission products.

7.0 References

1. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants", July 2000.
2. Analysis Basis, "Safety Basis for ACR", 108-03600-AB-003, July 2003.
3. Analysis Basis, "Fuel and Fuel Channel Safety Analysis Methodology", 103-03500-AB-004, September 2003.
4. Licensing Submission, "ACR Limited and Severe Core Damage Accidents: Supporting R&D", 108-126810-LS-002, Revision 0, November 2003.