June 12, 2003

ORGANIZATION: ATOMIC ENERGY OF CANADA LIMITED (AECL)

SUBJECT: SUMMARY OF MEETING HELD ON MAY 15-16, 2003, TO DISCUSS ACR SAFETY ANALYSIS METHODOLOGY AND COMPUTER CODES

The Nuclear Regulatory Commission (NRC) hosted a public meeting with Atomic Energy of Canada Limited (AECL) on May 15-16, 2003, at the NRC Headquarters to discuss the Advanced CANDU Reactor (ACR-700) safety analysis methodology and computer codes. A list of meeting attendees is provided as Enclosure 1.

This meeting was part of a series of technical workshops planned during the ACR pre-application phase with the purpose of familiarizing the staff with the CANDU design. The main objectives of the meeting were to present the ACR safety analysis methodology and computer codes for design basis events, define ACR accident scenarios and the sequences of key events and assess the applicability of AECL computer codes for ACR application.

An overview of the safety analysis approach based on the limit of the operating envelope (LOE) for design basis events (DBEs) was provided. The following three classes of design basis events were defined and examples of each class were provided:

- Class 1: Events of Moderate Frequency (Loss of Class IV Power)
  - Incidents which may occur during a calendar year for a particular plant

- Class 2: Infrequent Events (Small LOCA, End Fitting Failure)
  - Incidents which may occur during the lifetime for a particular plant

- Class 3: Limiting Events (Large LOCA, Pressure Tube/Calandria Tube Rupture, Main Steam Line Break)
  - Faults that are not expected to occur but are postulated because of their potentially significant consequences

The detailed outlines of representative sequences in each class were presented and AECL concluded that the ACR design can accommodate the DBEs due to (1) two independent shutdown systems to ensure fast and reliable reactor trip; (2) emergency core cooling with emergency coolant injection system; (3) steam generator heat sink ensured by diverse means of supplying backup feedwater; and (4) indefinite heat sink provided by long term cooling system.

The staff asked why there were no events describing partial or total scram failures. AECL responded that this approach was based on the DBEs that may not result in a partial or total scram failure.

The staff also requested AECL to provide the acceptance criteria for examples of the events. The acceptance criteria for DBEs were presented by AECL during the March 27, 2003, meeting. AECL agreed to add the acceptance criteria section originally presented during that meeting at the end of the planned presentation.

The meeting continued with a presentation on the key computer codes for ACR safety analysis. The detailed descriptions of inputs required and outputs produced by the following computer codes with corresponding safety analyses were presented:

- Thermal-Hydraulics: CATHENA, NUCIRC, MODTURC-CLAS
- Physics: WIMS, RFSP,DRAGON
- Fuel: CATHENA, ELOCA, ELESTRES
- Fuel Channel: CATHENA, TUBRUPT
- Fission Product Transport: SOURCE, SOPHAEROS, SMART
- Containment: GOTHIC
- Dose: ADDAM
- Sever Core Damage Accidents: MAAP-CANDU

The staff questioned how those different computer codes were interfacing with each other. AECL responded that some codes are automatically linked to each other but the outputs of most computer codes were manually transferred as inputs to other codes.

Although the staff has received the CATHENA code for thermal-hydraulics analyses, a question was raised by the staff as to when AECL plans to submit other computer codes used on the ACR safety analysis. AECL expects to submit physics and thermal-hydraulic codes for review during the pre-application phase.

When some questions concerning the applicability of AECL computer codes for the ACR and codes validation process were raised, AECL presented the quality assurance (QA) process with respect to computer code development, validation, configuration management and use in ACR application. AECL stated that Technical Basis Document (TBD) and Validation Matrix documents are the high-level documents that guide the code validation process. Subsequently, the incremental and confirmatory code validations based on experimental database extensions for the ACR will be performed.

The next ACR familiarization meeting is scheduled for June 4-5, 2003 at the Whiteshell Laboratories in Winnipeg, Manitoba. This will be a closed meeting and the NRC staff plans to attend the technical presentations on description, scaling criteria and experiment results of the RD-14M thermal hydraulics followed by a tour of the on-site test facilities.

For additional details on the material covered in these presentations please refer to the Agencywide Documents Access and Management System (ADAMS). This system provides text and image files of NRC's public documents. The presentations mentioned above may be accessed through the ADAMS system under Accession No. ML031420448. If you do not have access to ADAMS or if there are problems in accessing the handouts located in ADAMS, contact the NRC Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr@nrc.gov.

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Project No. 722

Enclosures: As stated

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ADAMS ACCESSION NUMBER: ML031420448-Meeting Handouts
ML031540528-Meeting Summary
ML031540517-Pkg.

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Distribution for May 15-16, 2003, Meeting Summary dated June 12, 2003

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ACR-700 Safety Analysis Methodology and Computer Codes
May 15-16, 2003 Auditorium 9:00am - 4:00pm

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ACR-700 Safety Analysis Methodology and Computer Codes
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