March 25, 2003

ORGANIZATION: ATOMIC ENERGY OF CANADA LIMITED (AECL)

SUBJECT: SUMMARY OF MEETING HELD ON FEBRUARY 5, AND 6, 2003, TO DISCUSS ACR-700 THERMAL HYDRAULIC ISSUES AND THE PROBABILISTIC RISK ASSESSMENT (PRA) METHODOLOGY USED BY AECL

The Nuclear Regulatory Commission (NRC) hosted a public meeting with Atomic Energy of Canada Limited (AECL) on February 5 and 6, 2003, at the U.S. Nuclear Regulatory Commission (NRC) Headquarters to discuss Advanced CANDU Reactor (ACR)-700 thermal hydraulic issues and the probabilistic risk assessment (PRA) methodology used by AECL. A list of attendees from the thermal hydraulic portion of the meeting is provided as Enclosure 1. A separate breakout session was held to discuss the ACR-700 PRA. A list of meeting attendees for the PRA portion of the meeting is provided as Enclosure 2. 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 along with a series of Open Literature papers provided during the meeting by AECL may be accessed through the ADAMS system under Accession No. ML030800383. 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.

Highlights from Thermal Hydraulic Portion of the Meeting

This meeting was the third in 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 objective of the meeting was to present the Canadian Algorithm for Thermalhydraulic Network Analysis (CATHENA) code, its validation methodology, and to discuss fuel channel thermal hydraulics and moderator thermal hydraulics.

On the first day, an overview of the ACR systems was provided and the unique features of the ACR design were discussed. Above the core, the ACR design is very similar to pressurized water reactors (PWR). However, below the core, the ACR has very unique design features such as horizontal fuel channels, pressure tube, calandria tube, short 43 element fuel bundles, individual fuel channels supplied by feeder pipes, and end fittings to allow for on-power fueling, a passive moderator heat sink, etc. AECL developed the CATHENA code for the thermal hydraulic analysis of CANDU reactor coolant systems during postulated events and it is being applied to ACR analysis. CATHENA evolved from a previous code ATHENA (Algorithm for Thermal Hydraulic Network Analysis) which was derived from RELAP-5. CATHENA is a two-fluid non-equilibrium model with two velocities, two temperatures, and two pressures. The gas phase can include up to four noncondensable components and it includes the properties of water and heavy water. The modeling assumptions were discussed and the components of the CATHENA model were described. The CATHENA Solid Heat Transfer Model (GENHTP) is used to model conduction within solid components and all heat transfer between solid components and the thermal hydraulic fluids; for example, radial conduction within multi-region fuel, radial conduction within pipe walls, fuel pin-to-fluid heat transfer, and pipe wall-to-fluid heat transfer. Axial conduction is not included since refilling is predominantly across and not along the fuel channel. In response to staff questions, AECL stated that unlike PWRs, axial conduction is not a dominant modeling parameter for CANDU reactors and it is, therefore, not included in GENHTP. The GENHTP model was described and discussed in detail. AECL clarified that the model is capable of tracking the level of the surface as the volume of the water in a channel changes. In response to staff questions, AECL noted that CATHENA normally models the fuel as a heat source. CATHENA is linked to ELOCA, a single-element fuel transient thermal mechanical code, when a more detailed model of the fuel is desired. The meeting continued with a discussion of the uncertainty analysis. The impact of the uncertainty in the correlations coefficients in CATHENA can be assessed by modifying the parameters through a defined uncertainty. The new value of the correlation coefficient can be defined through an internally defined standard deviation and offset or bias. The user can define the multiplier or bias to define the new value of the correlation coefficient. AECL agreed to expand on this discussion at a future meeting in order to clarify questions raised by the staff; such as accounting for uncertainty within the code, the number of parameters that can be varied, and the internally defined standard deviation and bias, etc.

AECL noted that the Integral Test Facility (RD-14M) simulator does not account for the pressure drop based on alignment of fuel bundles. The RD-14M facility was modified in 2001 to run at ACR pressure and temperature and the headers and below header piping was modified to the RD-14M/ACR configuration. A comprehensive database of integral thermal hydraulics experiments and a wide range of test types including loss-of-coolant accident (LOCA), natural circulation, loss-of-flow, transition to shutdown cooling, and flow stability exists for CANDU reactors.

The expected release date for CATHENA MOD 3.5d is December 2003. The present validation of MOD 3.5c thermal hydraulic phenomena will be extended to include scenarios and conditions relevant to ACR. There are 21 relevant phenomena for the ACR design, each validated separately.

On the second day, the ACR thermal hydraulic workshop continued with a discussion on fuel channel thermal hydraulics and moderator thermal hydraulics. To clarify questions from the staff on the issue of fuel bundle misalignment, AECL stated that no effort is made, nor needed, to manage fuel bundle alignment in a CANDU core. Once in the reactor, subsequent fuel bundle rotation is precluded by the axial forces on the bundle string imposed by the hydraulic forces applied by the coolant flowing through the channel. It was noted that critical heat flux (CHF) correlations are developed from data with aligned bundles, which results in conservatively lower CHF. These conservative data are used to develop correlations that are used in accident analysis. AECL noted that there are plans to conduct endurance testing on full size fuel channels. Flow induced vibration is not observed in CANDU 6 and not expected in ACR.

The staff discussed the applicability of Appendix K to horizontal core flow and noted that there could be policy issues to consider for CANDU. The staff suggested that a report on ACR Fuel Channel Thermal Hydraulic, formatted in accordance with Chapter 4 of the standard review plan, should be considered for design certification.

The meeting continued with a presentation on the moderator thermal hydraulics. NRC staff questioned whether design aspects of the ACR core; such as, smaller lattice pitch (since it could result in higher hydraulic resistance) and plate-type reactivity devices (since they could affect local temperatures) are a concern. AECL indicated that these were not expected to be significant effects. The model used by the CANDU industry for safety and licensing assessments of moderator circulation in the calandria vessel is MODTURC_CLAS V2.9-IST (MODerator TURbulent Circulation Co-Located Solution). The model calculates pressure loss through the calandria tube array, the buoyancy term in the momentum equation, the volumetric heat load distribution in the calandria vessel from steady-state and transient neutronic power and radioactive decay distributions, and the moderator subcooling from temperature distribution. The model simulates the moderator temperature control system, the moderator heat exchangers and associated control valves, and it sets up transient boundary conditions. Assessments of moderator circulation in the ACR design indicates minor differences from the proven CANDU 9 design. In order to confirm this, MODTRUC_CLAS will be validated using data from the Moderator Test Facility (MTF) modified to simulate ACR design and operating conditions. The one-third scale ACR test is planned for 2006. Prior to adjourning the meeting, AECL provided the staff a total of 27 Open Literature papers on ACR thermalhydraulics. These reports are available in ADAMS under Accession No. ML030800383.

Highlights from the PRA Portion of the Meeting

AECL provided an overview of the PRA methodology that it has used for various CANDU designs. AECL has not yet developed a detailed PRA for the ACR-700. During the meeting AECL provided the staff with three reports describing additional background information regarding the investigation of severe accident phenomena for the ACR-700. These papers are: (1) “CANDU Response to Loss of All Heat Sinks,” (2) “Modular Accident Analysis Program for CANDU Reactors,” and (3) “Coolability of Severely Degraded CANDU Cores,” AECL-11110, Revised January 1996. The papers are part of the presentation material that can be accessed through ADAMS under Accession No. ML030800383.

During a background discussion of the PRA, AECL mentioned that it jointly developed a CANDU source term using WASH-1400 methodologies with a group in the Netherlands. The staff expressed an interest in this document and AECL mentioned that it was publicly available. The title of the document is “Summary of CANDU 6 Probabilistic Safety Assessment Study Results,” dated 1990, Nuclear Safety, Vol 31, No.2, pp. 202-214.

The staff and AECL exchanged information regarding AECL’s approach to PRA and the staff’s approach to reviewing PRAs as part of a design certification application. The staff expressed a desire to continue with a technical dialogue regarding PRA during the pre-application phase for the ACR-700. The staff and AECL agreed to explore arranging a trip to a CANDU reactor in Canada so that the NRC’s PRA reviewers could become more familiar with the CANDU design. In addition, in a December 18, 2002, letter, AECL provided a detailed deliverable and schedule for focus topics. This letter included a schedule for a familarization meeting with the staff in June 2003 regarding the PRA methodology that will be applied to the ACR. The staff and AECL agreed that the discussion topics for the June 2003 meeting may need to be broadened. The PRA reviewers also expressed an interest in attending a March 27, 2003, meeting with AECL to discuss the ACR safety design philosophy and design basis accidents.

The staff stated that it would be beneficial for AECL to consider the following to prepare for future discussions regarding PRA and severe accidents:

- AECL should review SECY papers that provide guidance for severe accident reviews that were performed for the designs that have been certified including how the use of in-vessel retention was addressed for the AP600. The following SECY papers were suggested for review:

  – SECY 90-016, “Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements”

  – SECY 93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs”

  – SECY-96-128, “Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design”

- AECL should review the safety evaluation reports for designs that have been certified in order to become familiar with how PRA related issues were addressed. For example, how design alternatives were considered, how digital instrumentation and control was modeled and evaluated, and how severe accidents were addressed using a combination of deterministic analyses and a Level 3 PRA.

The staff expressed an interest in learning more about the extent of AECL’s use of international experience to address PRA and severe accident issues. In particular, the staff would like to learn more about the differences between the Modular Accident Analysis Program (MAAP4) code and the CANDU version of the code (MAAP-CANDU) that AECL intends to use for the ACR-700.

The next ACR familiarization meeting is scheduled for March 27, 2003, to discuss ACR safety philosophy, design basis accidents, severe accidents, and acceptance criteria.

/RA/

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Project No. 722

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ACR-700 Thermal Hydraulics Meeting
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ACR-700 PRA Meeting
February 6, 2003 O-04B8 10:30am - 3:45pm

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