UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

March 1, 2004

MEMORANDUM TO: ACRS MEMBERS

FROM: Med El-Zeftawy, Senior Staff Engineer ***/RA/***
ACRS

SUBJECT: CERTIFICATION OF THE MINUTES FOR THE MEETING OF THE ACRS SUBCOMMITTEE ON FUTURE PLANT DESIGNS, JANUARY 13, 2004 - ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued on January 28, 2004, have been certified as the official record of the proceedings for that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc: J. Larkins
H. Larson
S. Duraiswamy
ACRS Staff Engineers

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

January 28, 2004

MEMORANDUM TO: Dr. Thomas Kress, Chairman
Future Plant Designs Subcommittee

FROM: Med El-Zeftawy, Senior Staff Engineer ***/RA/***
ACRS

SUBJECT: WORKING COPY OF THE MINUTES FOR THE MEETING OF THE ACRS SUBCOMMITTEE ON FUTURE PLANT DESIGNS, JANUARY 13, 2004-- ROCKVILLE, MARYLAND

A working copy of the minutes of the subject meeting is attached for your review. Please review and comment on them at your earliest convenience. Copies are being provided to each ACRS Member who attended the meeting for information and/or review.

Attachment: As Stated

cc: ACRS Members
J. Larkins
S. Bahadur
H. Larson

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

MEMORANDUM TO: Med El-Zeftawy, Senior Staff Engineer
ACRS

FROM: Thomas S. Kress, Chairman
Future Plant Designs Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES FOR THE MEETING OF THE ACRS SUBCOMMITTEE ON FUTURE PLANT DESIGNS, JANUARY 13, 2004–ROCKVILLE, MARYLAND

I do hereby certify that, to the best of my knowledge and belief, the minutes of the subject meeting on January 13, 2004, are an accurate record of the proceeding for that meeting.

/RA/ **2/4/04**

Thomas S. Kress Date
Subcommittee Chairman

CERTIFIED BY: T. Kress
On: February 4, 2004

Issued on: January 28, 2004

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
FUTURE PLANT DESIGNS SUBCOMMITTEE
MEETING MINUTES- JANUARY 13, 2004
ROCKVILLE, MARYLAND

INTRODUCTION

The ACRS Subcommittee on Future Plant Designs met on January 13, 2004, at 11545 Rockville Pike, Rockville, Maryland, in Room T-2B3. The purpose of this meeting was to discuss the ACR-700 design features and the related pre-application reviews. The entire meeting was open to public attendance. Med El-Zeftawy was the cognizant ACRS staff engineer and the Designated Federal Official for this meeting. The Subcommittee has received no written comments, or requests for time to make oral statements from any members of the public regarding this meeting. The meeting was convened at 8:30 a.m and adjourned at 6:10 p.m.

ATTENDEES

ACRS

T. Kress, Chairman
G. Apostolakis, Member
P. Ford, Member
G. Leitch, Member
G. Wallis, Member
V. Ransom, Member
S. Rosen, Member
W. Shack, Member
J. Sieber, Member

NRC (principal speakers)

L. Dudes, NRR
J. Lyons, NRR
B. Sosa, NRR
E. Sullivan, NRR
J. Rosenthal, RES
W. Jensen, NRR
D. Carlson, RES
S. Jones, NRR
P. Sekerak, NRR
M. Stutzke, NRR
J. Kim, NRR
A. Attard

AECL

J. Polcyn
S. Yu
V. Langman
R. Ion
V. Snell
N. Popov
P. Chan
P. Boczar
M. Leger
J. Millard
D. Richards
A. Stretch
R. Jaithy

CNSC

P. Akhtar
M. El-Hawary
V. Tang
G. McDougal
Y. Zeng
C. Corriev
A. Delja
G. Rzeutkulie
G. Schwarz
A. Ibrahim
H. Khouaja

A complete list of attendees is in the ACRS Office file and will be made available upon request. The presentation slides and handouts used during the meeting are attached to the Office copy of these minutes.

OPENING REMARKS BY THE SUBCOMMITTEE CHAIRMAN

Dr. Thomas S. Kress, Future Plant Designs Subcommittee Chairman, convened the meeting at 8:30 a.m. He stated that the purpose of this meeting is to discuss with representatives of the NRC staff and the Atomic Energy of Canada, Limited (AECL), the advanced CANDU reactor (ACR-700) design features and the related pre-application reviews. The Subcommittee will gather information, analyze relevant issues and facts, and formulate positions and actions, as appropriate, for deliberation by the full Committee.

AECL PRESENTATION

The ACR-700 is an advanced CANDU (Canada Deuterium Uranium) design that utilizes horizontal fuel channels passing through a heavy-water moderator tank. As with other CANDU designs, the ACR-700 will be refueled during power operation. Other features of the reactor system, coolant pumps, U-tube steam generators, and pressurizer are similar to pressurized water reactor designs in the United States.

The ACR-700 will have features that make it significantly different from operating CANDU reactors. The ACR-700 utilizes light water as coolant within the fuel channels, whereas operating CANDU reactors utilize heavy water. The ACR-700 will be designed to have a negative void reactivity coefficient so that if boiling occurs within the fuel channels, the reactor power will decrease. The negative void coefficient for ACR-700 will be achieved by using slightly enriched uranium fuel elements rather than the natural uranium fuel used in operating CANDU reactors. The reactor core will be smaller than operating CANDU reactors with fewer fuel channels.

Mr. Stephen Yu, Program Manager- Atomic Energy of Canada, Limited (AECL) Technologies, Inc., briefed the Subcommittee regarding the ACR-700 design overview. He stated that the ACR-700 is an evolutionary extension of the CANDU 6 plant, which has ten units in operation on four continents, and one unit currently under construction. The current operating CANDU reactors utilizes natural uranium fuel, heavy water as coolant, and heavy water as moderator. The ACR-700 design relaxes the constraint of natural uranium fuel and utilizes slightly enriched uranium (SEU) fuel, use light water coolant, reduced core size and reduced amount of heavy water moderator to increase reactor coolant system pressure and thermal efficiency.

In refueling, two fuel bundles are introduced into the core through the upstream end of the fuel channel, and two spent fuel bundles are withdrawn from the opposite end of the channel. Each CANFLEX fuel bundle contains 43 fuel rods. Forty two rods contain approximately 2.1 wt% U^{235} while the central rod is natural uranium plus a small percentage of dysprosium (burnable neutron absorber). Fuel burn-up is projected to be in the range of 20,000 Mwd/T. Typically PWR fuel burn-up is 50,000 Mwd/T.

The pressure tube for the ACR-700 fuel channel design is thicker to reduce stresses during normal operation. Only the outlet end of the pressure tube experiences temperatures greater

than in CANDU 6. The calandria tube is Zr-4 and has materials properties equivalent to Zr-2, and it is thicker to withstand spontaneous pressure tube failure.

The major components of the reactor cooling system are arranged with one steam generator and two pumps at each end of the core. Reactor coolant flows through half of the fuel channels and feeders to an outlet header, through one steam generator and two pumps. It returns to the core through an inlet header to the other half of the fuel channels in which flow is in the opposite direction. The use of SEU fuel allows flexibility in choice of coolant void reactivity coefficient (CVR). The CVR was chosen small to increase operating margins related to safety. This also results into negative power coefficient and more stable control.

The on-power fueling is carried out remotely by two fueling machines which act in tandem at opposite ends of the reactor core. On-power fueling is used to keep the reactor critical and controls the global power distribution in the reactor core and the fuel discharge burn-up.

The ACR-700 containment is a steel-lined containment similar to the conventional PWR. It has coolers for heat removal, and passive autocatalytic hydrogen recombiners for core damage accidents. For severe core damage accidents, there are control system and two independent shut-down systems, each is capable of safely shutting down the reactor. For the ACR-700 design, the severe core damage sequence would result only from multiple failures such as LOCA plus loss of moderator cooling plus loss of reserve water tank make-up to the moderator. In severe core damage sequence, core geometry is lost but shield tank water can delay progression or contain debris within calandria. AECL representatives believe that severe core damage can be delayed for hours due to passive boil-off of moderator and shield tank inventory.

Mr. Yu stated that the ACR-700 design will have reduction in outage due to system and equipment monitoring and readily accessible on-line data. The design and operation are connected via reliability centered maintenance that proven to be effective from application to existing CANDU reactors. Reduction in operator error is also expected due to improved control room design, comprehensive plant status via large screen display, and improved alarm recognition system.

Mr. Vince Langman, AECL, briefed the Subcommittee regarding the ACR-700 pre-application scope, rationale and expectations. Phase 1 (June 2002 to August 2003) of the pre-application review established CANDU specific focus topics and technical familiarization of the design. Phase 2 (September 2003 to September 2004) will include further technical meetings with the NRC on focus topics, participate in NRC Phenomena Identification and Ranking Tables (PIRTs) meetings, and respond to NRC staff request for additional information (RAI).

AECL's rationale for the ACR-700 pre-application review is to focus the review effort on the CANDU specific aspects of the ACR-700 design that are not easily addressed by the current NRC regulations. AECL also plans to deal with focus topics that are inherent to the ACR-700 design (i.e., design aspects that cannot and should not be changed), and have prohibitively large monetary or schedule impacts if proposed AECL positions will not be accepted by the NRC.

The focus topics for pre-application review are:

1. Class 1 pressure boundary design (key)
2. Design basis accidents and acceptance criteria
3. Computer codes and validation adequacy (key)
4. Severe accident definition and adequacy of supporting research and development
5. Design philosophy and safety-related systems
6. Canadian design codes and standards
7. Distributed control systems and safety critical software
8. On-power fueling- including fuel design (key)
9. Confirmation of negative void reactivity (key)
10. Preparation for Standard Design Certification Docketing
11. ACR-700 PRA methodology
12. ACR-700 technology base

Currently AECL expects the NRC to identify whether there are any impediments to licensing the ACR-700 in the U.S., and to identify success paths for any unresolved pre-application focus topics. In addition, the Canadian Nuclear Safety Commission (CNSC) is also reviewing the ACR-700 design. This parallel ongoing licensing reviews offer an excellent opportunity for regulatory synergy between the NRC and the CNSC. AECL expects to integrate the extensive licensing experience of the NRC and CNSC to develop a common North American technical basis for licensing the ACR-700 design in both U.S. and Canada.

Mr. Marc Leger, AECL, described the pressure boundary features of the ACR-700 design. The basic layout of the reactor coolant system allows dissipation of heat through natural circulation, if all electrical power to the reactor coolant pumps fails. There are 284 horizontal pressure tube made from an alloy of Zirconium and 2.5 wt% Niobium in which trace elements, particularly chlorine, copper and iron, are closely controlled. The material is recognized by the Canadian Standards Association (CSA) and standards have been developed for it which are consistent with the intent of the ASME codes.

The pressure tube material is at reactor coolant system temperature, with spacers to keep the outside surface of the tube at a distance of 17mm from the calandria. The annulus between the two provides thermal insulation. If the pressure tube sags into contact with the calandria, a cold spot occurs on the pressure tube, at which, over a period of time, zirconium hydride can form and cause material cracking.

AECL desired outcome is for the NRC staff to accept the principal design features of the pressure boundary (i.e., the use of Zr-2.5 wt% Nb pressure tubes, rolled joints, closure plugs, 403 SS end fittings, and fueling machines as components of a Class 1 pressure boundary). Mr. Leger stated that the CANDU fuel channels are a proven technology licensed in five jurisdictions including Korea. He also added that there is an extensive technology base supporting the ACR-700 design.

Mr. David Richards, AECL, outlined the computer codes and validation adequacy for the ACR-700 design. He stated that the ACR-700 analysis codes are developed and qualified under a formal software quality assurance (SQA) program. The code development and qualification are conducted according to pre-defined QA procedures such as CSA N286.7-99, "Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants" , and AECL-QAM-003, "Quality Assurance Manual for Analytical Scientific and Design Computer Programs" published in September 1999.

The Canadian utilities and AECL worked together to qualify a standard set of computer programs, industry standard toolset (IST) and agreed to common processes to meet CSA-N286.7-99. Validation matrices were developed to identify and describe phenomena relevant to a discipline, rank the phenomena according to their importance in accident phases, and to identify data sets and cross-reference to phenomena. Validation methodology has been demonstrated, using thermal-hydraulics as an example, and the CATHENA code. A wide range of experimental database is used in the validation process.

Mr. J. Millard, AECL, described the on-power fueling focus topic. He discussed the advantages found in the CANDU reactor design from the use of on-power fueling. He stated that on-power fueling allows a low core reactivity to be used and provides flexibility in station maintenance outages. CANDU reactor shave been safely and successfully using on-power fueling for four decades in 45 reactors. The ACR-700 design builds on that experience with a new system with improvements in safety, operability and maintainability.

To refuel, two fueling machines lock on to either end of the same fuel channel. One of the fueling machines carries two new fuel bundles in its magazine. Each fuel channel contain a string of 12 fuel bundles. The irradiated fuel is removed from the downstream end and fresh bundles are inserted at the upstream end. The irradiated fuel is discharged via a fuel port through the containment boundary to a bay in the reactor auxiliary building. New fuel is supplied via fuel ports also through the containment boundary. The fueling machine has a movable class 1 pressure vessel that connects to the fuel ports and fuel channels in sequence to move the fuel around.

Fuel that becomes defective during operation is detectable, and can be removed by the fueling machines with the reactor at power. The computer controlled and automated on-power refueling ensure an optimum fuel usage. This helps to minimize the amount of radioactive material that could be in the reactor coolant system. Defense-in-depth principles and multiple barriers are used to ensure safety and reliability of the ACR-700 reactor.

For focus topic # 9, "negative void reactivity", and focus topic # 11, "PRA methodology", Mr. Langman, AECL, indicated that the desired outcome would be for the NRC to accept that

the ACR-700 has a negative void reactivity, and the PRA methodology as sufficient for the purpose of licensing the ACR-700 in the U.S.

NRC STAFF PRESENTATION

Ms. Belkys Sosa, NRR, outlined the NRC planned pre-application review process. Ms. Sosa stated that the approach and criteria to be aplied in the review of the ACR-700 design are in some cases different from those applied to conventional LWRs because of the unique features and design characteristics of the ACR-700. The review will identify where new staff positions, regulations and regulatory guidance is needed to address the unique characteristics of the design; such as pressure tubes and fueling machine as reactor coolant system components, and on-power fueling.

In the application of existing regulations and guidelines, the staff may need to interpret the guidance developed for LWRs for application to non-LWR concepts and issues under review. The approach is directed toward ensuring an equivalent level of safety as that of current-generation LWRs.

It is expected that ACR-700 pre-application review will accomplished in two phases. Phase 1 is a series of familiarization meetings designed to provide the staff with a general overview of the ACR-700. Phase 1 has been completed on July 31, 2003, and the review is currently in Phase 2 (expected completion date is September 30, 2004). The objective of Phase 2 is to provide more specific and detailed information regarding the design and to facilitate the staff's review of the focus topics. AECL has requested that priority be given to the following focus topics during Phase 2 of the review:

Focus Topic #1- Class 1 pressure boundary design

Focus Topic #3- Computer codes and validation adequacy

Focus Topic #8- On-power fueling

Focus Topic #9- Confirmation of negative void reactivity

In addition, the staff is conducting PIRT meetings to identify key phenomena and processes that are important to understanding plant behavior under normal and accident conditions. The NRC staff plans to issue a Safety Assessment Report (SAR) by September 2004 and forward a draft SAR to the ACRS in July 2004.

Mr. Edmund Sullivan, NRR, outlined the staff's process to review Class 1 pressure boundary design (PBD) for the ACR-700 design. Currently, the staff plans to develop understanding of differences between the ACR-700 and plants already operating or reviewed and identify where are existing regulations that may not be met by the ACR-700 design. Potential issues include basis for fatigue design curves, governing creep equations, sagging of pressure tubes and hydride blister formation, effect of large number of bent pipes, erosion, corrosion, effect of irradiation damage, design of rolled joints, Canadian design standards and inspection codes, and code classification of components. Other issues could include leak-before-break approach, on-power fueling as an extension of the Class 1 PBD, design of transport mechanism in Class 1

components support structure, and component material behavior under severe accident conditions.

Mr. Jack Rosenthal, RES, outlined the ACR-700 PIRT review. The objective is to develop initial PIRTs for neutronics, severe accidents and thermal hydraulics. The purpose of this research effort which began in September 2003, is to develop infrastructure to support the forthcoming design certification effort.

For the neutronics PIRT, specified scenario is the large break of an inlet or outlet header, voiding all fuel channels within 1 to 3 seconds. PIRT tables are organized according to the three main elements of CVR calculation (operating conditions, lattice physics, and core simulation) to address fundamental physics as well as safety analysis methods. The knowledge level of each phenomenon's impact on CVR is assessed as known, partially known, or unknown.

For the severe accident PIRT, specified scenarios are single channel event either critical break in a single feeder pipe or a flow blockage; or whole core event initiated by LOCA or station blackout. The figure of merit for single channel is potential for damage progression to lower neighboring channels.

For the thermal hydraulics PIRT, specified scenario is a critical break, defined as the break size leading to early flow stagnation in the core. Figure of merit is fuel time-temperature history and the event is divided into two phases: blowdown and reflood. Phenomena within each component are identified and ranked by importance and by state of knowledge, using a scale of high, medium, or low. The PIRT report is due in May 2004.

Mr. Walton Jensen, NRR, described the NRC review of computer codes and validation adequacy. The NRC staff objectives include scoping review to determine code strengths and weaknesses and if experimental verification is needed; identify any "show stoppers" that would prevent the codes from being used for the ACR-700 safety analysis; and identify any regulatory or policy issues that will need to be resolved.

The regulatory Standards to be used for the thermal hydraulics codes include draft Regulatory Guide DG-1120, "Transient and Accident Analysis Methods", and draft Standard Review Plan (SRP) Section 15.0.2, "Review of Analytical Computer Codes." Documents to be submitted by AECL include CATHENA theoretical manuals.

Mr. Anthony Attard, NRR, stated that for the neutronics codes, the documents to be reviewed include code theory manuals for RFSP, "neutron diffusion code for 3D power distribution and burnup"; WIMS, "2D lattice physics code to generate fuel neutron cross sections for RFSP"; and DRAGON,"3D lattice code to generate cross section data of control devices for RFSP". The regulatory standards to be used include draft Regulatory Guide DG-1120, and SRP section 15.0.2.

Mr. Donald Carlson, RES, described the NRC confirmatory analysis of the ACR-700 coolant void reactivity (CVR). He stated that the void reactivity is key to evaluating the design in relation to GDC-11, Reactor Inherent Protection. Void reactivity effects can significantly impact the progression of analyzed transients and accidents. Evaluation of bias and uncertainty in the

calculated CVR predictions (i.e., validation) will figure prominently in staff conclusion. Validation of computed CVR predictions will be based on ACR-specific benchmark measurements in AECL's ZED-2 critical facility.

Currently, the staff is concerned that when code calculations predict a small negative CVR, then how confident are the staff regarding the actual CVR, and if it is indeed be negative in view of prediction bias and uncertainty. RES plans to provide related input on status, initial insights, and plans for assessing neutronics validation data for CVR in May 2004. In addition, RES plans to develop independent static calculations of nominal CVR values using detailed models with existing state-of-the-art methods, validation benchmark analysis to evaluate CVR bias and uncertainty, and provide SCALE lattice data and PARCS core models for simulating ACR-700 operations and transients.

Mr. Steven Jones, NRR, briefed the Subcommittee regarding the on-power refueling issue. He stated that the on-power refueling is not previously licensed in the U.S. Currently, the NRC staff is reviewing regulatory issues such as possible exemption from existing regulations or rulemaking; policy issues such as new criteria for evaluation of design and new classes of design-basis events; and technical issues such as new methods of review or analysis for fuel handling accidents. The staff plans to develop regulatory and policy framework to support design certification. Mr. Patrick Sekerak, NRR, indicated that the staff is currently reviewing AECL report , "The Technology of On-Power Fueling." The staff is comparing the content of such report to the design certification 10 CFR 52.47 regulations. Policy issues could be acceptance criteria, and CSA Standards as proposed alternatives to 10 CFR 50.55a.

Mr. Martin Stutzke, NRR, described the NRC staff review of the PSA for the ACR-700 design. The objectives of the staff's review is to determine that if the AECL PSA methodology will produce a PSA with adequate scope, level of detail, and technical acceptability to satisfy regulatory needs. The staff will develop a schedule and resource estimate for reviewing the PSA submitted with the standard design certification application.

In its review, the staff will use the guidance from the July 8, 1986, "NRC Policy Statement on Regulation of Advanced Power Plants," NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," and 10 CFR Part 52.47(a)(v). The staff will also use Regulatory Guide 1.174, Standard Review Plan (Chapter 19), and Chapter 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities."

Currently, the staff is developing the understanding of how to interpret the core-damage frequency risk acceptance guidelines specified in the Staff Requirements Memorandum on SECY-90-16 with respect to the ACR-700 design. Potential policy issue could be-- should a guideline pertaining to the frequency of accidents that potentially involve a release but no fuel damage (e.g., tritium release) be developed? Mr. James Kim, NRR, stated that the staff plans to submit a safety assessment report (SAR) to the ACRS in July 2004.

General Comments and Observations from the Subcommittee members

- Dr. Kress stated that the AECL's nominal value of coolant void coefficient (CVR) is only slightly negative, and a confirmatory analysis is needed to account for the bias and

uncertainty of CVR predictions. The CVR values are also sensitive to core design and operating conditions.

- Dr. Kress indicated that thermal-hydraulic codes such as CATHENA have to be validated against other codes (e.g., RELAP5) to account for margins and conservatism.

- Dr. Ford expressed concern regarding the ACR-700 materials degradation. He stated that there is not a lot of experience in independently assessing the safety margins associated with the fact that the Zirc-alloy pressure tube around the fuel is now the primary pressure boundary.

- Dr. Ford stated that the complexity of the piping arrangement for the ACR-700 design presents challenges to the adequacy of inspection techniques, in terms of access and probability of defect detection, and periodicity.

- Mr. Rosen requested a copy of the AECL report on “The Technology of On-Power Fueling” and urged the staff to analyze the content of such report.

- Dr. Apostolakis questioned the AECL rationale for the reliability of shutdown system software. AECL believes that the reliability of the safety critical software is demonstrated through “trajectory-based” random testing. Dr. Apostolakis asked for more explanation and a copy of References 5, 6, 7, and 8 mentioned in the Safety Characteristics of the Advanced CANDU Reactor Design ACR-700 report (June 2003).
- Mr. Rosen asked the staff to investigate the issue of sump blockage for the ACR-700 design.

- Dr. Ford expressed concern regarding the proposed component changes from the CANDU design, and stated that there is no way of quantitatively assessing the changes in frequency of component failure and their consequence, so that these issues can be managed or regulated in a risk-informed manner.

- Dr. Wallis indicated that there is a lot of major issues that have to be resolved, but the current schedule does not reflect the allotted time to resolve such issues.

- Mr. Leitch indicated that more information is needed regarding the emergency electric power supply and consideration of safeguards and security.

- Dr. Ransom requested more analysis to ensure covering the core in the severe-accident space.

- Dr. Kress indicated that more information, and possibly more subcommittee meetings, is needed for PRA, materials degradation, severe accidents, core melt progression, source terms, and the frequency-consequence acceptance curves in comparison to the U.S. criteria.

SUBCOMMITTEE ACTION

The Subcommittee Chairman plans to provide a subcommittee report regarding this matter during the February 5-7, 2004 ACRS meeting.

Documents provided to the Subcommittee prior to January 13, 2004

- Safety Characteristics of the Advanced CANDU Reactor Design ACR-700, J.G. Waddington & K.C. Rogers, June 2003.
- Advanced CANDU Reactor (ACR-700) Design–Slides / AECL.
- ACR-700 Technical Description, Rev.0, AECL June 2003 (CD).

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NOTE: Additional details of this meeting can be obtained from a transcript of this meeting available for downloading or viewing on the Internet at "http://www.nrc.gov/ACRSACNW" or can be purchased from Neal R. Gross and Co. –1323 Rhode Island Ave., NW, Washington, DC 20005 (202) 234-4433.