POLICY ISSUE
(Notation Vote)

March 3, 1995 SECY-95-050

FOR: The Commissioners

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: CHANGE IN PLANS FOR DESIGN CERTIFICATION REVIEW OF THE CANDU 3U REACTOR

PURPOSE:

To inform the Commission of a change in the plans of the Atomic Energy of Canada Limited Technologies, Inc., (AECLT) to submit a revised safety analysis report (SAR) for final design approval (FDA) and design certification under 10 CFR Part 52 for the CANDU 3U design, and to request approval for modification of the previously proposed review plan.

CATEGORY:

This paper covers a significant schedule and resource change in staff activities.

BACKGROUND:

The CANDU 3U is a 450 MWe, heavy water-moderated and cooled, pressure tube reactor developed by Atomic Energy of Canada, Ltd. (AECL). AECL developed the design from previous CANDU reactors, most notably the CANDU 6, a 600 MWe

NOTE: TO BE MADE PUBLICLY AVAILABLE WHEN THE FINAL SRM IS MADE AVAILABLE

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design. There are 25 CANDU reactors in operation and 19 under construction around the world. CANDU reactors have operated for over 175 effective full power years.

In December of 1988, a U.S. company, AECLT, the U.S. representative of AECL, was created as the preapplicant for the CANDU 3U licensing effort in this country. In a letter to the U.S. Nuclear Regulatory Commission (NRC) on May 25, 1989, AECLT informed the NRC of its intent to seek design certification of the CANDU 3 under the provisions of 10 CFR Part 52.

On April 8, 1993, the staff submitted SECY-93-092, "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and their Relationship to Current Regulatory Requirements." The staff described five policy issues associated with the CANDU 3 design: accident evaluation, source term, containment performance, positive void reactivity, and design of the control room and remote shutdown area. On March 24, 1994, the staff issued SECY-94-079, "Schedule and Resource Estimates for CANDU 3 Design Certification Review." In this paper, the staff estimated the resources and confirmatory research required to review the CANDU 3 for design certification. The staff estimated 105 full-time equivalents (FTE) and $2.2 million for the Office of Nuclear Reactor Regulation (NRR) to complete a 54-month review schedule starting in fiscal year (FY) 96, and 23 FTE and $18 million in confirmatory research.

On September 30, 1994, AECLT submitted an application for FDA and design certification under 10 CFR Part 52 for the CANDU 3U design. The staff completed an acceptance review of the application and sent AECLT a letter on its findings on December 15, 1994 (Attachment 1). The staff informed AECLT that a docket number had been assigned to the application to facilitate public access to the correspondence and review information, but the staff did not intend to develop a review schedule until the updated safety analysis report (SAR) and schedules for all outstanding information had been submitted. The staff assigned docket number STN 52-005 and requested a response in 30 days to include a schedule for submitting the missing items.

On January 19, 1995, AECLT sent the Commission a letter in which it stated that the response would be delayed until no later than February 8, 1995, pending a review of work planning and scheduling.

On January 30, 1995, the staff met with AECLT to discuss the CANDU status (see viewgraphs, Attachment 2). AECLT again expressed concern regarding the proposed fees for the CANDU review. AECLT proposed changing the timing of some of the major NRC milestones for the review effort to start in April 1997. They also proposed a different allocation of staff resources for the review. Under their proposal, the FTE expenditures would be low in the beginning and at the end, but would grow more rapidly and to a higher peak than staff had proposed in SECY-94-079. In response, the staff described how reviews are conducted and how staff time is scheduled. The staff also stated that the fee issues would be decided by the Commission and that a paper was being prepared for the Commission on this subject (SECY-95-035, "Reassessment of Fee Billing

Practices and Fee Policy for Office of Nuclear Regulatory Research (RES) Activities Associated with Design Certification (DC) Applications").

On February 2, 1995, AECLT responded to the staff's December 15, 1994, letter, by submitting a schedule for updating the SAR and submitting other required information (Attachment 3). AECLT proposed to update the SAR in stages with the final update to be submitted in January 1997 and proposed for NRC to start the design certification review in April 1997. In July 1998, AECLT would submit other required information such as the level III probabilistic safety assessment (PSA); inspection, test, analysis, and acceptance criteria (ITAAC); design acceptance criteria; technical specifications; test programs; and severe accident mitigation design alternative (SAMDA). AECLT proposed for the staff to issue the final CANDU safety evaluation report in October 1999, 30 months after the staff starts the review. AECLT also proposed that the NRC only do limited work on certain generic issues and policy issues over the next 2-plus years. During this period, AECLT indicated that it will limit funding for NRC review costs to $3 million over that time period. AECLT stated that it could not go forward with the review if the NRC costs were to approach $50 million.

DISCUSSION:

The proposed schedule to complete the SAR and proposal to review only generic licensing issues would substantially delay the staff from the schedule and resource estimates in SECY-94-079. Further, the proposed plan for reviewing selected issues and constraining the level of effort departs from normal practices and could result in significant resource and schedule inefficiencies.

In response to the AECLT request, NRC staff plans the following actions:

1. Hold discussions with AECLT on the issues listed in Enclosure 2 to their letter of February 2, 1995, to better understand the type and scope of review they propose and to determine which areas should be considered highest priority for review by NRC staff, using appropriate contractual assistance.

2. Assign NRR staff to work in the area or areas of first priority.

3. Inform AECLT when we have spent 90 percent of the available resources. This notification will state that we will cease work as soon as the resource limit is reached, unless AECLT wishes to provide additional support.

4. Cease all NRC activities in areas not directly related to the requested review areas, including NRR contract activities and design-specific Office of Nuclear Regulatory Research (RES) activities, including the development and use of independent audit codes.

5. If AECLT requests detailed schedules and cost estimates for the tasks undertaken, we would charge the costs of developing the estimates to the applicant. Such costs are likely to be significant because information may be needed from many technical staff.

6. Begin design certification review of the SAR after AECLT completes and submits all necessary revisions. According to their letter, the revisions and submissions will be completed by January 1997, which will delay NRC from beginning a full-scale review by 15 months from October 1995 as stated in SECY-94-079. Therefore, delays in all planned review activities, supporting research activities and potential staff resource shortages would significantly delay completing the FDA. However, the staff cannot develop a detailed review schedule until July 1998, when all of the required information, including the completed PSA, ITAAC, technical specifications, tests programs, and SAMDA, has been submitted.

7. Send AECLT revised cost and schedule estimates when the generic licensing issue review has been completed. Completing certain generic reviews before the design certification review may result in some limited savings in specific areas of the design certification process. However, the inefficiencies of conducting a resource limited review effort prior to the design certification review, which may include starting and stopping work as resources expire, will likely result in some net increase in the cost estimate.

8. Continue to hold periodic senior management meetings with the Atomic Energy Control Board of Canada. However, defer any planned expenditures of resources, such as the planned exchange of personnel until the start of the design certification review.

RESOURCES:

The FY 1996-1997 NRC budget contains resources in accordance with the review plan proposed in SECY-94-079. Resources to conduct the design certification review for the CANDU 3U design in response to AECLT's revised proposal will be addressed during the FY 1996-2000 Internal Program/Budget Review process.

RECOMMENDATION:

Approve the staff's plan for reviewing CANDU 3U issues, this plan includes: deferring the design certification review, doing a limited review of selected issues identified by AECLT within its resource constraints, cessation of contract activities and RES efforts unrelated to specific AECLT requests, and give AECLT revised estimates for the design certification review when the staff is ready to begin this effort.

NOTE:

On March 1, 1995, the staff was told by AECLT that there are indications that the Board of Directors of AECL has placed the CANDU design certification on hold. AECLT is presently seeking clarification of the implications of the Board action. AECLT also indicated that they may be visiting individual Commissioners sometime next week, when they have further information. Should AECLT wish to pursue continuation of NRC's review, staff recommends that the proposed plan outlined in this SECY be adopted. Attachment 4 is a proposed letter the staff plans to send AECLT if they indicate they wish to proceed with limited review.

James M. Taylor
Executive Director
for Operations

Attachments:
1. Staff Acceptance Review Letter
2. AECLT Viewgraphs from Jan. 30, 1995 Meeting
3. Letter from AECLT (Feb. 2, 1995)
4. Draft Letter to AECLT

Commissioners' comments or consent should be provided directly to the Office of the Secretary by COB Friday, March 17, 1995.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Friday, March 10, 1995, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

December 15, 1994

Mr. A. D. Hink, President
AECL Technologies, Inc.
9210 Corporate Boulevard
Suite 410
Rockville, Maryland 20850

Dear Mr. Hink:

SUBJECT: RESULTS OF THE ACCEPTANCE REVIEW FOR AECL TECHNOLOGIES' APPLICATION FOR FINAL DESIGN APPROVAL AND DESIGN CERTIFICATION FOR THE CANDU 3U DESIGN

In a letter dated September 30, 1994, AECL Technologies (AECLT) submitted its application for final design approval (FDA) and standard design certification (DC) under Part 52 of Title 10 of the Code of Federal Regulations (10 CFR) for the CANDU 3U design. The contents of the application were provided in the form of a CANDU 3U Safety Analysis Report (SAR) consistent with the format in Regulatory Guide 1.70 and the Standard Review Plan. The application letter acknowledged that certain information required by 10 CFR 52.47 was missing from the application. The missing information was identified as the inspections, tests, analyses, and acceptance criteria (ITAAC), technical specifications, severe accident mitigative design alternatives (SAMDA), and the failure modes and effects analyses (FMEA). It should also be noted that the required Level II and Level III probabilistic risk assessments (PRAs) and design acceptance criteria (DAC), if any, were not included as part of the CANDU 3U application.

In accordance with 10 CFR 2.101, the staff performed an acceptance review to determine if the CANDU 3U was sufficiently complete to enable the staff to carry out the design certification review. The staff has determined that a significant amount of information is either missing, or in a form which would cause the staff to expend a great deal more resources to complete the DC review than previously anticipated. The staff has previously indicated it could proceed with the review in the absence of certain information: ITAAC, technical specifications, and SAMDA. This position was based on the staff's experience that the areas in question would not require a detailed review early in the review process. However, a schedule for submittal of these and other missing items is needed. Submittal of these items will influence the schedule by which the CANDU 3U review is carried out. Early submittal of all information is necessary to assure that staff can develop and maintain an effective review schedule. In that regard, the staff understands that it is your intention to submit a PRA, completed through Level III, in about a year. Furthermore, it should be noted that it is the staff's intent to keep the number of DAC to a minimum.

Attachment 1

NRC also requires a clear identification in the SAR that CANDU 3U meets the applicable U.S. codes and standards and the NRC's General Design Criteria (GDC). While the SAR indicates that many of the Canadian (CSA) codes and standards cited are equivalent to the existing U.S. codes and standards, the equivalence is not explained in sufficient detail to demonstrate that the CSA codes and standards do indeed meet our requirements. Where Canadian standards are necessary because there are no U.S. standards or acceptance criteria, an equivalent level of safety analysis to the GDC should be provided.

Enclosure 1 provides further details related to information needed by the staff to continue the review of the CANDU 3U design. Please be aware that the enclosure does not represent a comprehensive list of deficiencies; it is limited to those found during the limited acceptance review. Other issues will be identified to AECLT through requests for additional information during the review.

The staff has assigned Docket number STN-52-005 to the CANDU 3U application to facilitate public access to correspondence and review information. A copy of the Federal Register notice is enclosed (Enclosure 2) for your information. AECLT should reference this docket number when submitting the requisite 38 updated copies of the CANDU 3U SAR pursuant to 10 CFR 50.4 for the start of the DC review. The staff does not plan to develop a detailed review schedule until the updated SAR and schedules for all outstanding information have been submitted.

In response to your application letter, the staff has considered the resources needed to issue an FDA for the CANDU 3U design. The staff has examined the review process and the full-time equivalent (FTE) staff required for the completion of each of the evolutionary plant DC reviews. While there should be some potential resource savings from the staff experience in conducting DC reviews, it is expected that this will be offset by the potential difficulties inherent in reviewing a non-light water design. The acceptance review confirms that there are a number of significant issues which potentially could require enhanced NRC review efforts. Therefore, the staff still considers the resource and schedule estimates made in the March 24, 1994, Commission paper (SECY-94-079, "Schedule and Resource Estimates for CANDU 3 Design Certification Review") to be appropriate at this time. This estimate included 105 FTE and $2.2 million for the Office of Nuclear Reactor Regulation (NRR), and 23 FTE and $18 million for the Office of Nuclear Regulatory Research (RES). Two additional factors will influence the ultimate schedule and the fees actually billed to AECLT for the DC review: how closely the application complies with the information required by the Standard Review Plan; and the results of the ongoing agency reevaluation of the NRC fee structure related to research needed to support licensing of advanced reactor designs. Until both of these issues are resolved, the staff cannot provide a more complete estimate of the costs and schedule to complete the CANDU 3U review.

In this regard, AECLT should provide within 30 days of the date of this letter a schedule for submittal of the updated CANDU 3U SAR and the other missing information such as the ITAAC, technical specifications, and SAMDA. We understand that unless we hear differently from you, the staff plans to continue its limited work on some key issues such as void reactivity and shutdown system reliability. If you have any question regarding this letter please contact the NRC project manager, Dino C. Scaletti at (301)504-1104.

Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Associate Director
for Advanced Reactors and License Renewal
Office of Nuclear Reactor Regulation

Docket No. 52-005

Enclosures:

1. Request for Additional Information
2. Federal Register Notice

cc w/enclosures:
See next page

ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION

210.1 Classification of Structures, Systems, and Components (SSCs)

a. SAR Section 3.2 and other SAR Sections contain references to (1) a series of Canadian Standards (CAN/CSA N-285 through N-290) that provide requirements for safety classifications, design and fabrication, quality assurance, and seismic qualification, and (2) a series of Safety Design Guides that apparently provide design guidelines for such subjects as seismic analyses, code classification, and pipe rupture protection. SAR Subsection 3.2.7.2 states that where there is an individual reference in the CAN/CSA Standards to ASME Code Section III, Division 1 or 2, it is the intent of the CANDU 3U design certification process to incorporate all Articles in ASME Subsections NB, NC, ND, NF, CB, CC, and Divisions 1 and 2 Appendices in their entirety. However, for some SSCs classified as CSA Class 2, 3, 1C, 2C, or 3C, SAR Table 3.2-1 identifies the principal construction codes and standards as both ASME Section III and one of the CAN/CSA Standards. Because the certified design rule will be a part of the NRC regulations, all references should be made to U.S. codes and standards. The SAR should clearly discuss the acceptability of the design in those areas where the design deviates from the criteria in the ASME Code or U.S. industry standards.

b. SAR Table 3.2-1 lists the quality assurance for all SSCs as one of the Z299 series Quality Assurance Standards. SAR Section 3.2.4.3 contains only a brief description of these standards. With respect to the information in the SAR, it is not clear to the staff which of these standards, if any, contain a commitment to 10 CFR 50, Appendix B. The SAR should either state that Z299 meets all the requirements of Appendix B, or replace Z299 with a commitment to Appendix B for all SSCs classified as DBE (Seismic Category I).

210.2 Basis for the Alternative Safety Assessment

Atomic Energy of Canada Limited Technologies (AECLT) indicates that a Leak-Before-Break (LBB) approach may be used to demonstrate that catastrophic pipe failure has a very low probability of occurrence. If the LBB approach is to be used in the CANDU 3U design, the details of LBB methodology and acceptance criteria should be submitted in the SAR for staff review.

210.3 Computer Programs

It is not acceptable to provide only a list of computer programs. Additional information for each program in accordance with the guidelines of SRP 3.9.1.II.2 must be provided in the SAR to demonstrate that the program has been verified for its applicability and validity.

210.4 Experimental Stress Analysis

SAR Section 3.9.1.3 states that no experiments are required to qualify any mechanical systems or components; however, shake tests may be required on some fuelling machine components which will be specified in accordance with Section 3.10. It should be noted that experimental stress analysis is not an equipment qualification test. Therefore, AECLT should clearly state if experimental stress analysis is not used in the CANDU 3U design. If it is to be used, sufficient information must be presented in the SAR committing that the requirements of Appendix II to ASME Code, Section III, Division 1 will be met.

210.5 Seismic process Qualification Testing of Safety-Related Mechanical Systems

Per the guidelines of SRP 3.9.2, SAR Subsection 3.9.2.2 should include information on the seismic analysis methodology and approach for all Category I systems, components, equipment and their supports.

210.6 Dynamic Response Analysis of Reactor Structure and Fuel Channel Assemblies Under Operational Flow Transient And Steady-State Conditions

a. AECLT indicates that the dynamic response analysis is not planned at this time because the operational flow transient and steady-state conditions do not produce large loads to justify such analysis. Because the CANDU 3U design is the first of a design reviewed by the staff, documentation or references containing the results of required analyses and tests (either from a prototype of CANDU 3U or any other similar CANDU plants) must be provided for staff review.

b. In Sections 3.9.2 and 3.9.3 of the draft SAR, the following additional information as described in SRP Section 3.9.2 should be provided for verifying the design adequacy of the reactor vessel and internals:

   Detailed information on the reactor vessel and internal component design, including their configurations, major functions and design parameters, material requirements, definitions of operational and faulted condition design loads such as LOCA and SSE, design acceptance criteria such as stress and deflection limits, and methods and procedures for conducting tests and analyses to ensure their structural integrity and functional operability.

   A detailed description of the preoperational vibration assessment program for verifying the design adequacy against flow-induced vibrations, including designation of the prototype reactor, results of pre-testing vibration prediction analysis and acceptance criteria, preoperational flow testing program and planned instrumentation for vibration monitoring, and the post-testing visual inspection program.

210.7 Correlations of Reactor Structure Assembly Vibration Tests with the Analytical Results

AECLT indicates that the CANDU 3U design is based on proven technology and that the key features of the CANDU 3U design are essentially identical to those of operating plants. Since the CANDU 3U design is the first of a design reviewed by the staff, a discussion must be provided which describes the methods used (either for the CANDU 3U design or for any previous similar operating plants) to correlate the test results with those from dynamic analyses.

210.8 Component Supports

When snubbers are utilized as supports for safety-related systems and components, SAR Subsection 3.9.3.4 should incorporate relevant provisions as specified in the SRP 3.9.3 for establishing acceptable snubber operability assurance.

210.9 Control Rod Drive System (CRDS)

a. AECLT indicates that the SRP acceptance criteria are not applicable to the CANDU 3U design and that the Canadian Standard CSA-N285.0 is used in the design of CRDS. At this time, the staff has not confirmed that the SRP acceptance criteria are not applicable to the CANDU 3U design nor does the staff have a clear guidance position on how to review the acceptability of applying Canadian Codes and Standards to the CANDU 3U design. However, a simple reference to Canadian standards is not acceptable. The SAR should discuss why the SRP criteria are not considered applicable and clearly identify all criteria in these standards that are applicable to the CANDU 3U design.

b. In Section 3.9.4 of the draft SAR, provide a detailed description of the testing and analysis conducted for ensuring control rod insertion and safe shutdown of the reactor under faulted plant conditions such as seismic and LOCA events.

210.10 Calandria Vessel Internals (CVI) and Fuelling Machine (FM)

AECLT indicates that the SRP criteria are not applicable to the CANDU 3U design and that design alternatives including the use of Canadian Safety Design Guides and Standards are applied to the design of CVI. As discussed in 3.9.4, the SAR should discuss why the SRP criteria are not considered applicable and clearly describe the design alternatives and identify all criteria in those standards and design guides that are applicable to the CANDU 3U design.

210.11 In-Service Testing of Pumps and Valves

The information presented in this section should be expanded to address implementation of staff positions specified in SECY-90-016 and NRC Generic Letter 89-10. Although a detailed IST program will be developed by the COL applicant, an IST plan of sufficient information must be submitted to demonstrate that all safety-related pumps and valves including safety/relief valves of the CANDU 3U design can be adequately tested at the required frequency. Justifications for testing at cold shutdown or refuelling outage must be provided for those pumps and valves that cannot be tested quarterly.

210.12 Evaluation of Safety Issues

The following issues should be addressed:

Issue A-1	"Waterhammer"
Issue 70	"PORV and Block Valve Reliability"
Issue 79	"Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown"
Issue 87	"Failure of High-Pressure Coolant Injection Steamline Without Isolation"
Issue II.D.1	"Performance Testing of PWR Safety and Relief Valves"

210.13 Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment

a. The title of SAR Section 3.10 and its contents should be expanded to include not only seismic but also dynamic qualification of both electrical equipment and mechanical equipment.

b. Information and commitments must be provided in the SAR demonstrating that relevant qualification criteria specified in SRP 3.10, Rev. 2 have been met.

220.1 Sections 3.3.1.2 and 3.3.2.3 of the CANDU 3U Safety Analysis Report (SAR) state that the method for determining pressures generated on structures due to design wind is specified in accordance with the requirements of the National Building Code of Canada. Although Table 3.3-3 of the SAR states that the load determination criteria of SRP 3.3.1 and 3.3.2 (i.e., ANSI A58.2 and ASCE Paper 3269) are complied with, the extent and manner of compliance is not clear to the staff. AECLT should clearly demonstrate how compliance with the U.S. codes for wind load determination has been achieved and document the results in the SAR. The SAR should clearly discuss the acceptability of the design in those areas where the design deviates from the criteria in the ASME Code or U.S. industry standards.

220.2 Section 3.5.3 discusses the concrete missile barrier design criteria but does not address the design of steel missile-resistant barriers. AECLT should address compliance with SRP 3.5.3 guidelines for the design of steel missile-resistant barriers.

220.3 The CANDU 3U standard design defines two levels of earthquake: design basis earthquake (DBE) and site design earthquake (SDE). The design load combinations specified in Section 3.8 imply that the DBE is equivalent to the safe shutdown earthquake (SSE) and SDE is equivalent to the operating basis earthquake (OBE). The damping ratios specified in Table 3.7.1-2 comply with RG 1.61 for the DBE but the SDE damping ratios do not comply with the OBE damping ratios of RG 1.61. AECLT should provide justification for using damping ratios that are not in compliance with RG 1.61.

220.4 Table 3.7.1-1 of the CANDU 3U SAR provides the design basis earthquake ground response spectra in the horizontal direction for different damping values. In this table, the specific frequency values of the lower, intermediate and upper frequency ranges listed are not defined. AECLT should clearly define these frequency ranges to enable the staff to determine the compliance of CANDU 3U design response spectra with the RG 1.60 response spectra.

220.5 Table 3.7.5.1-1 of the SAR states that the CANDU 3U design ground response spectra envelope is the RG 1.60 spectra in the horizontal direction but the vertical design response spectra is less than the RG 1.60 vertical response spectra. AECLT should provide justification for using vertical design response spectra that are not in compliance with RG 1.60.

220.6 Table 3.7.5.1-2 of the SAR states that the seismic system analysis does not comply with SRP 3.7.2 guideline for considering 5% additional eccentricity to account for accidental torsion. AECLT should justify how accidental torsion effects will be considered in the CANDU 3U standard design in order to comply with the guidelines of SRP 3.7.2.

220.7 Section 3.7.2 of the CANDU 3U SAR does not provide the details of the seismic soil-structure interaction (SSI) analysis for compliance with the guidelines of SRP 3.7.2. This information should be provided.

220.8 Section 3.7.3 of the CANDU 3U SAR does not provide the criteria for the analysis and design of cable trays, conduits, HVAC and their supports. This information should be provided.

220.9 CANDU 3U standard design employs modular construction for various components of the containment internal structures. Although seismic analysis and design procedures for various systems and subsystems are presented in Sections 3.7 and 3.8, these generally address conventional safety related structures which may not be totally applicable to structures comprised of the modular units used in the CANDU 3U design. Sections 3.7 and 3.8 should discuss seismic behavior and design analysis methods for the CANDU 3U modular construction.

230.1 In Chapter 2, each section and subsection needs a COL applicant action item statement so the COL applicant will know what has to be done to assure that the site fits within the design assumption envelop. All references should be to U.S. codes and standards, NRC regulations, NRC regulatory guides and the NRC Standard Review Plan. The SAR should clearly discuss the acceptability of the design in those areas where the design deviates from the criteria in the ASME Code or U.S. industry standards. In particular, the SAR should discuss how an equivalent design margin is maintained.

230.2 The SAR should provide one summary table that lists all the assumed site parameters for which the CANDU 3U plant is designed rather than having it spread over various parts of the SAR.

230.3 The SAR should provide instructions for the COL applicant (or early site permit) on the type of information needed and types of studies to be performed to show that a site meets the geological, seismological and geotechnical engineering requirements.

230.4 The staff view is that sites with the potential for tectonic faulting at or near the ground surface are not acceptable for nuclear power plants. The SAR should state this.

230.5 Provide a complete description of the seismic instrumentation characteristics (i.e., solid state components, digital recording, bandwidth, dynamic range, ability to promptly determine responses spectra and cumulative absolute velocity, etc.)

230.6 Provide a discussion of the need for the COL applicant to have a program plan to perform pre-earthquake planning and post earthquake actions and an outline of such a program.

240.1 State the ground water level and the external flood level for which the plant is designed.

252.1 Turbine Missiles

This section of the SAR should include a figure showing the +25% degree low-trajectory turbine missile ejection zone. Further, the SAR should commit to meeting RG 1.115, "Protection Against Low-Trajectory Turbine Missile," Revision 1 which specifies that the probability of unacceptable damage from turbine missiles be less than 10^{-7} per reactor-year. Consistent with the staff's position taken for recently licensed plants, the probability of turbine missile generation should be no greater than 10^{-5} per reactor-year for unfavorably oriented turbines and 10^{-4} for favorably oriented turbines.

a. Paragraph 4.5.2.3 of the SAR indicates that Zirconium-Niobium pressure tubes meeting the requirements of Canadian Standard CAN/CSA-N285.6.1 will be used in the CANDU 3U reactor. The SAR should identify all criteria in these standards and guides that are different or deviate from those in the ASME Code or Standards endorsed by NRC. The SAR should clearly discuss the acceptability of the design in those areas where the design deviates from the criteria in the ASME Code or U.S. industry standards.

b. Paragraph 4.5.2.3 of the SAR also states that Canadian Standards CAN/CSA-N285.6.8 and CAN/CSA-N285.2 will be used for the selection of modified Type 403 stainless steel material and the use of a rolled joint design between the pressure tubes and the Type 403 material. AECLT must identify all criteria in these standards and guides that are different or deviate from those in the ASME Code or Standards endorsed by NRC. The SAR should clearly discuss the acceptability of the design in those areas where the design deviates from the criteria in the ASME Code or U.S. industry standards.

c. Paragraph 4.5.4.1 of the SAR states that lattice tubes and castings shall be fully radiographed over the maximum feasible volume. Section III of the ASME Boiler and Pressure Vessel Code (ASME Code) requires that nuclear components requiring radiography must be examined over the entire volume. The SAR should explain why CANDU 3U components cannot be examined over the entire volume as required by the ASME Code and discuss the acceptability of such an approach.

252.2 Heat Transport System and Connected Systems

a. Figure 5.2-1 of the SAR shows the heat transport system pressure boundary flow diagram. Using this diagram, the SAR should identify the materials of construction for the major pressure retaining components, e.g., D_2O storage tank - Austenitic Stainless steel Type 304L; D_2O feed pumps - Austenitic Stainless type 304L casing, 304L impeller, type 410 shaft; shutdown/bleed cooler - shell carbon steel, channel carbon steel, tubes carbon steel; bleed valves - body carbon steel, disk carbon steel, stem type 410 stainless steel; piping carbon steel.

b. Figure 5.6-1 of the SAR shows the moderator system flow diagram. Using this diagram, the SAR should identify the materials of construction for the major pressure retaining components of the moderator system.

c. The SAR should identify the impeller material for the heat transport and shutdown cooling pumps.

d. The SAR should identify all structural materials that will be exposed to high neutron fluence.

252.3 Engineered Safety Features

This section must include a subsection 6.2.7, Fracture Prevention of Containment Pressure Boundary and commit that the containment liner, containment penetrations, equipment and personnel hatches will meet the ASME Code, Section III fracture toughness requirements.

252.4 Steam and Power Conversion System

Subsection 10.3.6 should be expanded to discuss those measures that have been taken in the CANDU 3U design to address the concerns of erosion/corrosion caused by single-phase or two-phase erosion/corrosion phenomenon as documented in Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning."

262.1 The detailed test program Individual Test Descriptions were not included in the CANDU 3U SAR; however, the applicant acknowledged the requirement to provide that information to the staff.

263.1 The AECLT reliability assurance program (RAP) was developed using the staff's interim position for a RAP, as stated in SECY-93-087. AECLT will need to revise the D-RAP to reflect the NRC's final position on RAP to include a description of the essential elements of D-RAP, details on how the applicable regulation for D-RAP is satisfied, and the ITAAC to verify implementation of D-RAP prior to fuel load.

290.1 The main control area (MCA) HVAC system is a non-safety system which does not conform to the single failure criterion for providing safety-related filtration and cooling functions. Therefore, the MCA does not meet the requirements of GDC 19 of 10 CFR Part 50, Appendix A.

420.1 Section 1.8 CONFORMANCE TO NRC REGULATORY GUIDES

The recommendations of Regulatory Guide 1.75 with regard to separation and independence of electric circuits are discussed only in SAR Section 8. AECLT should also discuss how the design meets the intent of Regulatory Guide 1.75 for the I&C systems discussed in SAR Section 7. The design should show that low-energy signal cables are routed separately from power cables, and that safety-related redundant I&C circuits and components are separated and isolated from non-safety-related circuits.

In AECLT's identification of the instrumentation and control systems important to safety and the acceptance criteria for these systems, many Canadian standards are referenced. However, there is no comparison provided between the criteria in the Canadian standards to that in comparable U.S. standards.

420.3 AECLT stated in Section 7.1.2.5.2 that a Failure Mode and Effects Analysis (FMEA) is not provided at this time since the detailed schematic and loop diagrams are not available. RG 1.70 calls for an FMEA for protection systems and components. The staff considers an FMEA for the CANDU 3U Special Safety Systems to be essential for its review.

420.4 Section 7.2 SPECIAL SAFETY SYSTEMS

AECLT stated in Section 7.2.1.2.8 that the technology proposed for the trip computers has been used in the Darlington Nuclear Generating Station. However, there is no documentation describing the hardware and software design, the verification and validation processes, configuration management and other aspects of the digital system design. The inspections, tests, analyses, and acceptance criteria (ITAAC) for these systems have not been included. The staff intends to address this lack of design detail in a manner similar to that for the GE ABWR and ABB-CE System 80+ designs by certification of a design development process as described in SECY-92-053.

420.5 In Section 7.7 (Control Systems not Required for Safety), there is no discussion of the instrumentation and controls for the on-power fueling machine and its interface with safety-related systems. This is a unique feature in the CANDU plant design which may pose potential system interaction concerns that should be addressed by AECLT.

435.1 Section 8.1.4.3.4.1 of the SAR only states that the design of the CANDU 3 is in compliance with the requirements of BTP-ICSB 4 (Rev. 2, 1981 July). The staff requires details of how the CANDU 3 design complies with this BTP in order to conduct its review.

435.2 Section 8.1.4.3.4.4 of the SAR only states that the design of the CANDU 3 is in compliance with the requirements of BTP-ICSB 18 (Rev. 2, 1981 July). The staff requires details of how the CANDU 3 design complies with this BTP in order to conduct its review.

435.3 Section 8.1.4.3.4.5 of the SAR only states that the design of the CANDU 3 is in compliance with the requirements of BTP-ICSB 21 (Rev. 2, 1981 July). The staff requires details of how the CANDU 3 design complies with this BTP in order to conduct its review.

435.4 Section 8.1.4.3.4.6 of the SAR states that the design of the CANDU 3 electric protection system complies with the requirements of BTP-PSB 1, and the details of voltage setpoint and time delay of the tripping shall be analyzed in the detailed design stage. The staff requires additional information on how the CANDU 3 design complies with this BTP. AECLT should provide details of the design and how it complies with each position of the BTP.

435.5 With regard to TMI item II.E.3.1, Emergency Power Supply for Pressurizer Heaters, Section 8.1.4.3.5.1 states that there is alternate compliance for this issue as the necessity for pressurizer heaters for continuing natural circulation can be challenged and dismissed by appropriate analysis. The staff requires the details of this analysis in order to properly perform its review of this issue.

435.6 There is no detailed discussion of how operational experience was incorporated into the CANDU 3 design. Specifically, in the electrical power area, a discussion of how the design incorporated the experience identified in Generic Letter 88-15, Electrical Power Systems - Inadequate Control Over Design Process, is required.

435.7 The staff requires additional information on the CANDU 3 lighting design described in SAR Section 9.5.3 in order to conduct its review. The following areas of information should be addressed:

a) Illumination ranges for the normal, standby, and essential/emergency lighting systems are required.

b) Additional design details are required on the different lighting systems.

440.1 Computer Codes used for Transient and Accident Analysis:

AECLT needs to provide detailed descriptions of all the computer codes used in the core design discussed in SAR Chapter 4 and the transient and accident analyses documented in Chapter 15 of the SAR. The documentation of each computer code should include a discussion of the purpose of the code, description of the calculational models, and verification of the computer code against test data or applicable operating data. The code documentation should demonstrate its acceptance to the staff by showing its validity of the governing mass, energy, and momentum equations. AECLT is also be requested to show the correct use of empirical correlations and steam-heavy water interfacial relationships, accuracy of the numerical solution scheme including modeling techniques, and adequacy of benchmark comparisons with existing data.

For the computer models used in the loss-of-coolant-accident (LOCA) analysis, the applicant must use, (as required for LWRs in Item (a)(1)(i) of 10 CFR 50.46), an evaluation model that includes sufficient supporting justification to show that the analytical techniques realistically describe the behavior of the reactor system during a LOCA. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and input must be identified and assessed so that uncertainty in the calculated results can be estimated. Alternatively, an evaluation model used for the LOCA analysis may be developed using an approach consistent with the requirements for LWRs described in Appendix K to 10 CFR 50. The discussion should include the references of test reports used to validate the models simulating key phenomena that could occur during transients and accidents.

471.1 Operational considerations - Section 12.1 should include a section on how operating experience from current generation plants has been incorporated into the CANDU 3U design. This section should provide examples of design changes and operational improvements based on lessons learned from past plant designs.

471.2 Contained radiation sources - Section 12.2 should include a description of contained radioactive sources (such as waste tanks, heat exchangers, filters, holdup tanks, resin tanks, etc.) in the plant. Information provided for each source should include source location within the plant, component material and geometry, and radionuclide contents with associated source strengths.

471.3 Plant layout drawings - Section 12.3 should contain detailed plant layout drawings which show: plant radiation zones, shielding wall thicknesses, personnel and equipment decontamination areas, health physics facilities, controlled access areas, contamination control areas, chemical and analysis labs, post-accident sampling stations, and the counting room. These plant layout drawings should be provided for each plant elevation and should also include the location of plant equipment and components as well as normal and post-accident personnel traffic patterns.

471.4 Radiation protection design features - In accordance with Chapter 12 of the SRP, Section 12.3 of the CANDU 3U SAR contains a good description of facility design features incorporated to ensure that personnel exposures are maintained ALARA. However, the SRP also states that Section 12.3 of the SAR should describe how the plant design considers such major exposure accumulating functions as maintenance, refueling, in-service inspections, decommissioning, etc. Section 12.3 of the SAR should include this information, as well as 1) a discussion of source term control, 2) a description of how robotics have been incorporated into the plant design to minimize personnel doses, and 3) a description of any accessible plant areas having a potential for dose rates greater than one Sv/hr.

471.5 Dose assessment - Section 12.4 should contain a dose assessment performed in accordance with the guidelines of Regulatory Guide 8.19, "Occupation Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates".

471.6 DBA dose consequence computer codes - In Sections 15.1.3, et al., codes used by AECLT in the CANDU consequence analysis are referenced. These codes need to be described, and the basis for their validity needs to be presented.

471.7 Calculational methodologies and assumptions - The dose calculation methodologies and input assumptions used to analyze the radiological consequences of postulated accidents need to be provided. The information provided in this area needs to be adequate to allow NRC to perform independent calculations of the dose consequences of DBAs.

471.8 Accident source term - The source term used in the radiological consequence analyses should be specified. NRC has done extensive work on the accident source term for light water reactors (LWRs); NRC has little experience with heavy water reactors. Therefore, the basis for the CANDU accident source term needs to be described.

620.1 The staff's review of Human Factors Engineering (HFE) topics can be performed in three levels: programmatic, implementation plan, and completed (see attached information). AECLT should identify the level for each topic area. At present it appears that the HFE program plan and the Operating Experience Review may be completed. Most of the other topics can be treated at an implementation plan level of detail. V&V appears to be at a programmatic level of detail. Please discuss.

620.2 While no specific document reference is provided, the SAR discussion seems to indicate that a Human Factors Engineering Program Plan (HFEPP) exists which may provide some of the additional detail required for the review. AECLT should provide the HFEPP.

620.3 Great emphasis is placed on the use of the predecessor plant's analyses and operating experience as a basis for the CANDU 3 design. AECLT should provide a discussion of the relevant HFE analyses performed for that design which provide starting points for the new design

620.4 The specific documentation which is already available for review to support the HFE discussions and conceptual design in the SAR and that will be developed as part of the detailed design should be clearly identified.

620.5 The SAR contains missing figures, as well as cross-references, which when checked, provide little additional information (e.g., Figure 18.3.4-3 missing; Section 18.1.6.1.2, "complete list of documents," is missing; and there are numerous references to Section 18.1 which do not really expand upon the topic under discussion). These aspects of the SAR should be modified.

620.6 Several review topics were either minimally addressed or not addressed at all. AECLT should provide detailed information on these topics. Most notably, these include:

- Task analysis (minimally addressed)
- HRA-HFE integration (appears to be omitted)
- Minimum Inventory (appears to be omitted)
- CDD/ITAAC/DAC (appears to be omitted)

GENERAL GUIDANCE

HFE Review Levels:

The HFE PRM (NUREG-0711) can be used to review applicant submittals at three levels: *Programmatic Review*, *Implementation Plan Review*, and *Complete Element Review*. At a *Programmatic Review* level, the SAR does not include detailed methodology and, therefore, detailed evaluations using the HFE PRM acceptance criteria are beyond the scope of the staff review for design certification. At a programmatic level review, the HFE PRM criteria are used to determine whether the program provides a top-level identification of the substance of each criterion which, after design certification, will be developed by the applicant into a detailed implementation plan. The value of the programmatic level review is that it provides assurance that the implementation plan will address all HFE PRM criteria. Applicant commitment to the development of such a detailed implementation plan should be described in ITAAC/DAC. The staff will review this plan during post certification review activities. ITAAC/DAC are also needed for completing the implementation plan and providing the results to the staff for review.

To perform an *Implementation Plan Review*, the applicant's submittals should describe the applicant's proposed methodology in sufficient detail for the staff to determine whether the methodology will lead to products that meet the HFE PRM acceptance criteria for the element. An implementation plan review provides the applicant the opportunity to obtain staff review and concurrence on the applicant's full approach prior to design certification. The actual completion of the plan will then likely take place after design certification. Such a review is desirable from the staff's perspective since it provides the opportunity to resolve methodological issues and provide input early in the analysis or design process when staff concerns can more easily be addressed than when the effort is completed. While some implementation plans can be reviewed on their own merits, the staff may request a sample analysis which demonstrates the application of the methodology and its results. ITAAC/DAC are needed for completing the implementation plan and providing the results to the staff for review.

A *Complete Element Review* can only be performed when the finished products (e.g., main control room (MCR) design) are available for the staff to evaluate. This means that the applicant has submitted an analysis results report(s) and design team review report(s). An analysis results report provides the results of the applicant's efforts on an HFE PRM element with respect to the review criteria. A reviewer will utilize the report as the main source of information for assessing compliance with the review criteria. An applicant's design team review report provides the independent evaluation of the activities addressed for the element by the design team. On resolution of staff concerns regarding the analysis or its results, the review topic can then be closed prior to design certification.

Piping Design:

In accordance with 10 CFR 52.47(a)(2), an application for a standard design should contain sufficient design information to enable the NRC to make final safety determinations. For piping design, acceptable approaches consist of either (1) having documented and available for audit a complete design of all safety-related piping systems or (2) providing comprehensive descriptions of design acceptance criteria (DAC) for piping in the SAR and sample representative piping analyses. The piping DAC should contain information in the following areas:

- applicable codes and standards
- methods to be used for completing the piping design analysis
- modeling techniques
- pipe stress analyses criteria
- pipe support design criteria
- criteria for postulating high-energy line breaks
- Leak-before-break (LBB) approach applicable to CANDU-3 (analyses are required for all candidate LBB piping)

The SAR should clarify the approach and upgrade Sections 3.7.2, 3.9.2 and 3.9.3 of the draft SAR as necessary to address the above areas of concern.

United States Nuclear Regulatory Commission

Atomic Energy of Canada, Limited Technologies

Receipt of Application for Design Certification

Notice is hereby given that the Nuclear Regulatory Commission (the Commission) has received an application from Atomic Energy of Canada, Limited Technologies dated September 30, 1994, filed pursuant to Section 103 of the Atomic Energy Act and 10 CFR Part 52, for the standard design certification of the CANDU 3U Pressurized Heavy Water Reactor Plant. A notice relating to the rulemaking pursuant to 10 CFR 52.51 for design certification, including provisions for participation of the public and other parties, will be published in the future.

The CANDU 3U is a 450 MWe pressurized heavy water reactor design. A unique feature that distinguishes the CANDU 3U from the current generation of light water reactors designed in the United States is the use of natural uranium fuel contained in a pressurized heavy water coolant system and a separate heavy water moderator system. The CANDU 3U application includes the entire power generation complex, except those elements and features considered site specific.

The staff has determined that the application does not contain all information required by 10 CFR 52.47. A docket number is being assigned to the application to facilitate public access to correspondence and review information. Although no formal review schedule will be established until an updated Safety Analysis Report and schedules for all information required has been received, the NRC staff will continue limited review of the application. This is consistent with the letter of September 30, 1994, "...that no major activity will be initiated by the NRC beyond the acceptance review...."

A copy of the application is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, N.W., Washington, D.C. Previous correspondence on this application is filed under Project number 679. The new docket established for this application is STN-52-005.

Dated at Rockville, Maryland this 15th day of December1994.

Dennis M. Crutchfield

Dennis M. Crutchfield, Associate Director
for Advanced Reactors and License Renewal
Office of Nuclear Reactor Regulation

CANDU
Project No. 679
Docket No. 52-005

cc: Louis N. Rib, Licensing Consultant
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Steve Goldberg, Budget Examiner
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Director, CANDU 3U Safety & Licensing
AECL Technologies, Inc.
9210 Corporate Boulevard, Suite 410
Rockville, Maryland 20850

A. D. Hink, President
AECL Technologies
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Rockville, Maryland 20850

Attachment 2

NRC/AECLT MEETING

COST AND SCHEDULE

January 30, 1995

NRC/AECLT Meeting
January 30, 1995

AGENDA:

1. Introduction

   - AECLT: Purpose of Meeting/Background

   - NRC: Current Status of NRC Cost/Schedule Review

2. AECLT: Comparison of FDA Review Costs

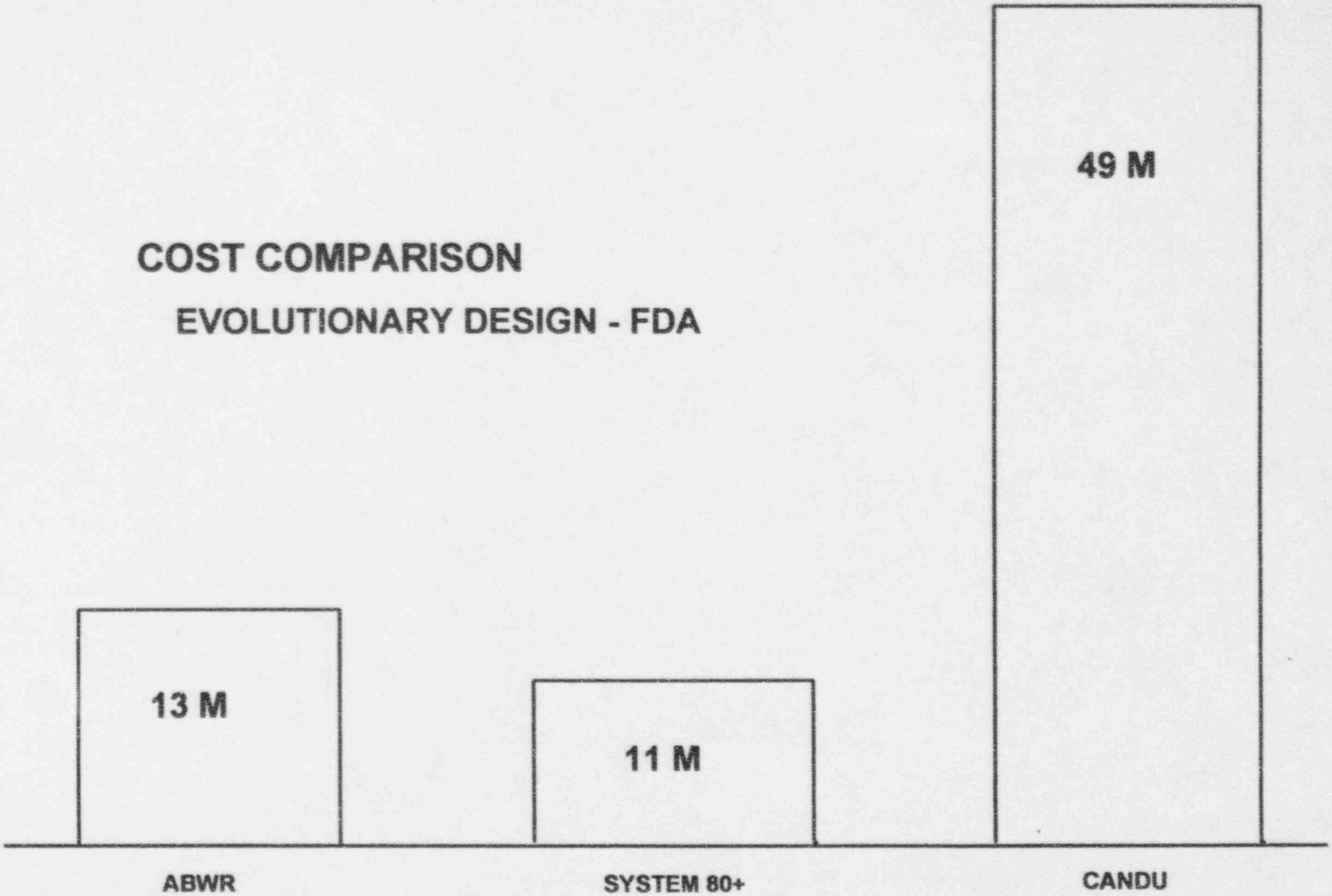
3. AECLT: Evaluation of FDA Review Costs

4. AECLT: Proposed FDA Review Resources and Schedule

5. AECLT: The Next Step

COST COMPARISON

EVOLUTIONARY DESIGN - FDA



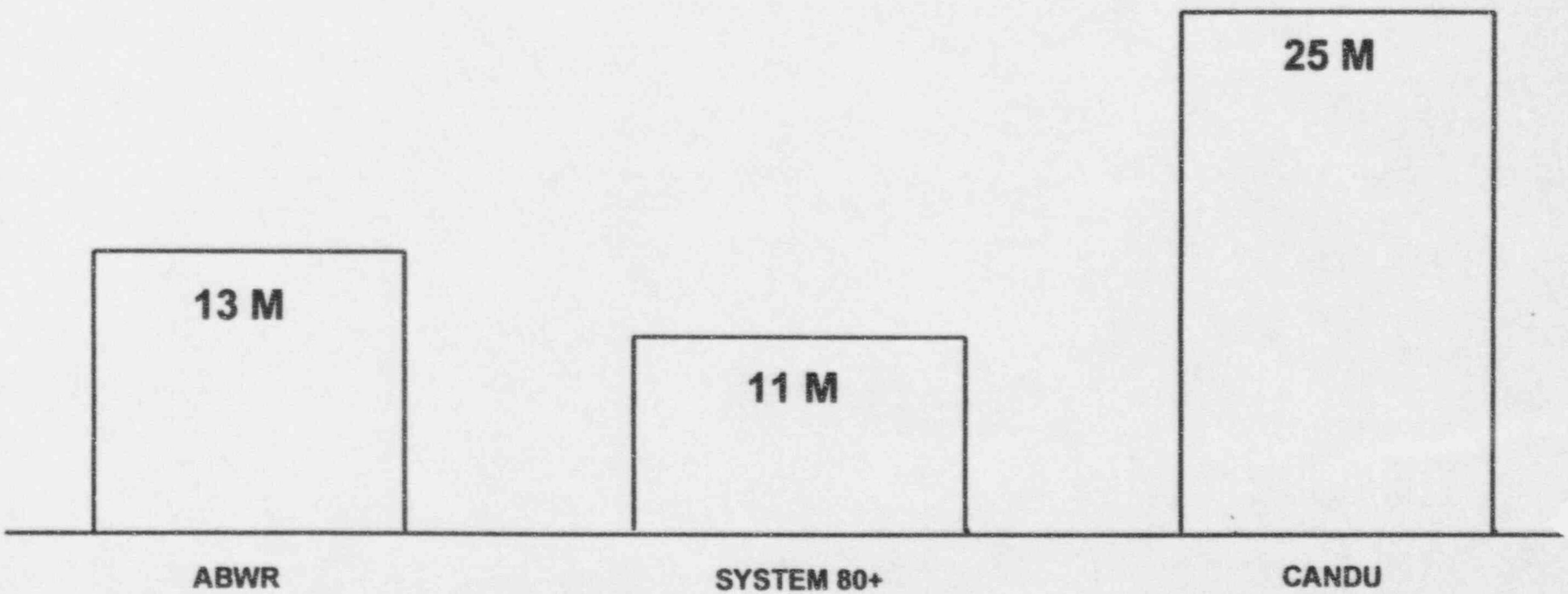
AECLT POSITION ON RESEARCH:

- Standard Review Plan Review of Safety Margin and R&D Support of CANDU Design is Adequate and Sufficient

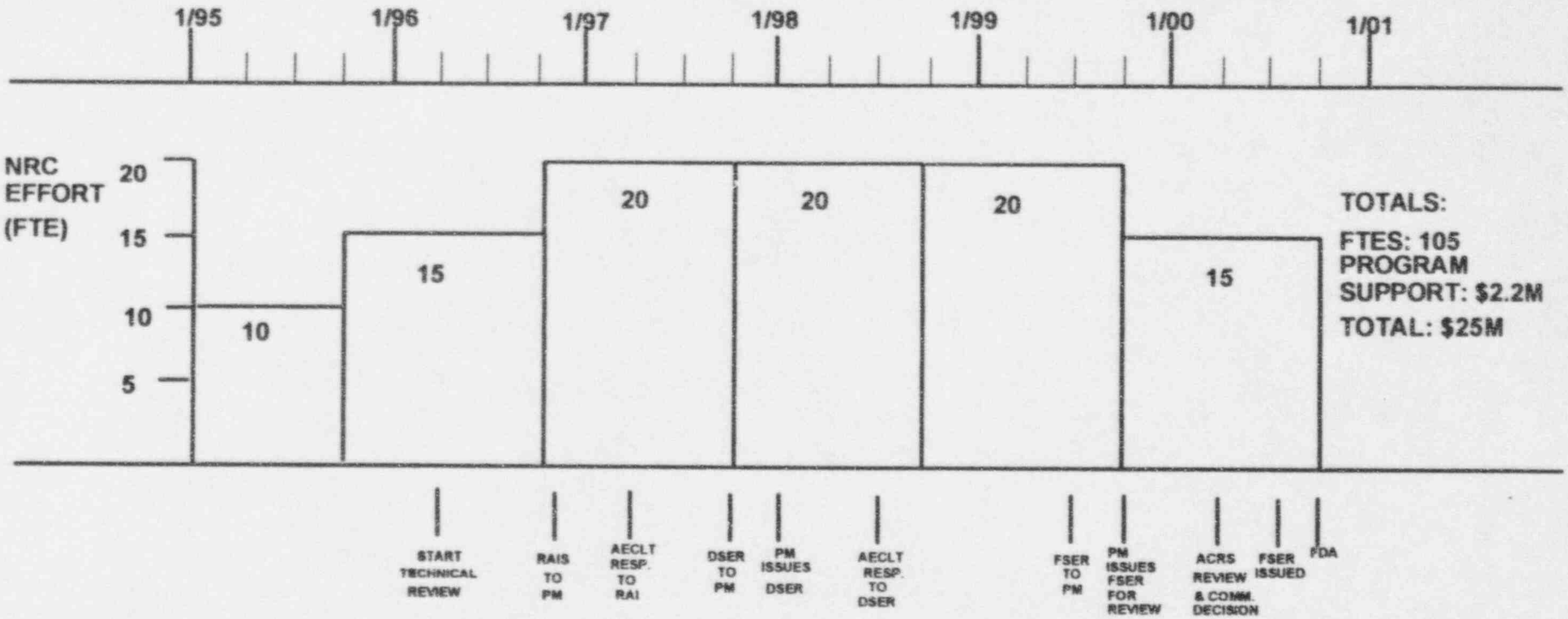
- No Additional Research is Required, CANDU Technology is Mature and Based on Operating Experience and AECL Research programs

COSTS WITHOUT RESEARCH

EVOLUTIONARY DESIGN - FDA



NRC INITIAL ESTIMATE SECY-94-079 (NRR ONLY)

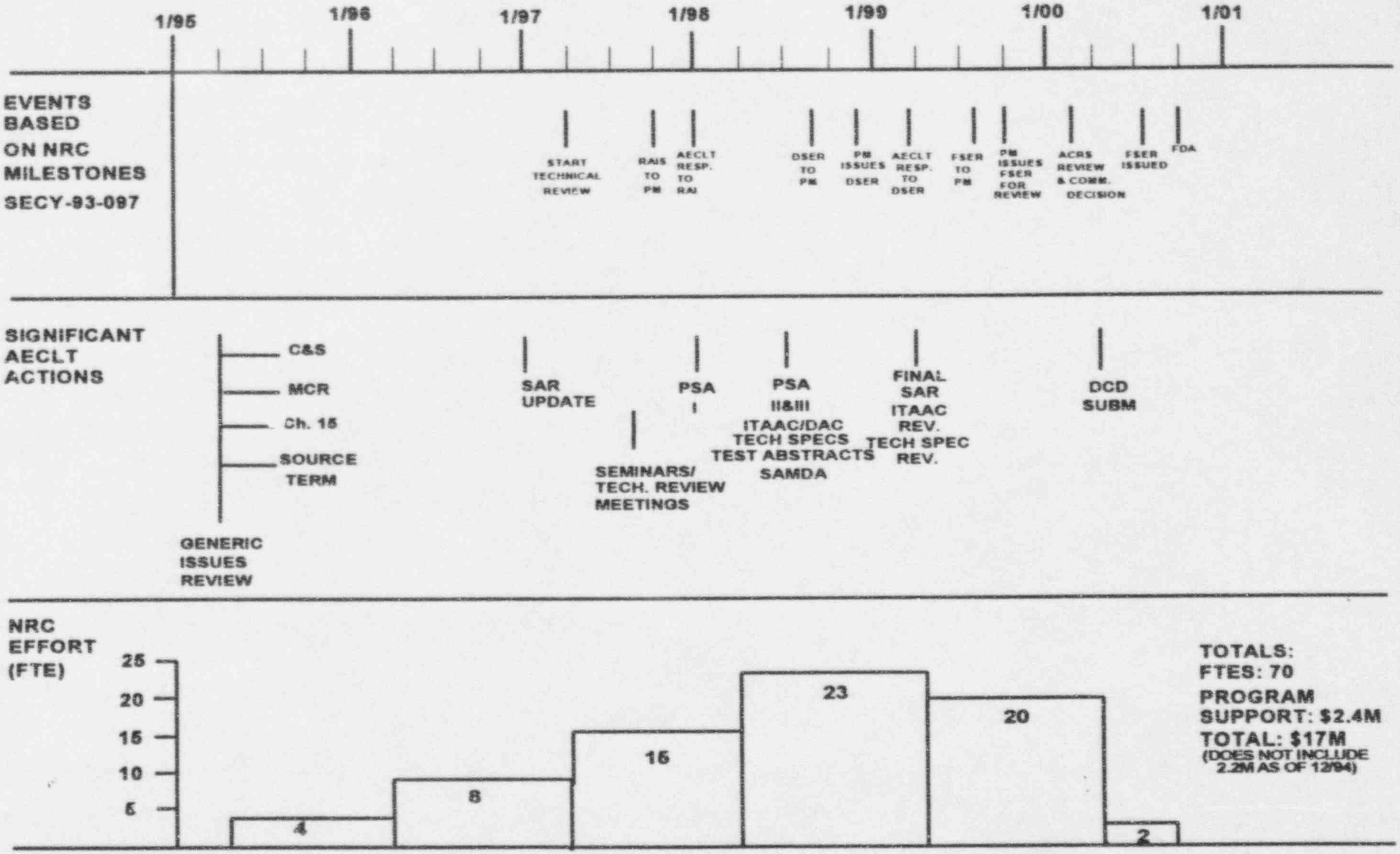


NRC/AECLT Meeting
January 30, 1995

COST ESTIMATE EVALUATION - PROPOSED EFFICIENCIES:

- Start with Program for Resolution of Generic/Policy Issues
- Conduct Technical Issues Seminars During Detailed Review (SNUPPS approach, tends to minimize number of RAIs)
- Apply "Load Follow" versus "Base Loaded" Resources
- AECLT Improve SRP Comparisons in SAR
- Reach Agreement to Pursue Regulatory Efficiency Generally and to Control Budget and Schedule Specifically to Achieve This Objective

AECLT PROPOSED PROGRAM



TOTALS:
FTES: 70
PROGRAM SUPPORT: $2.4M
TOTAL: $17M
(DOES NOT INCLUDE 2.2M AS OF 12/94)

THE NEXT STEP:

- Develop Agreement/MOU Regarding Control of Cost and Schedule
- Conduct Periodic (Weekly/Biweekly/Monthly) Project Status Meetings
  - Monitor Status
  - Anticipate Problems
  - Facilitate Responses to Questions

NRC/AECLT Meeting
January 30, 1995

CANDU GENERIC/POLICY ISSUES:

CODES AND STANDARDS

ACCIDENT ANALYSIS

SOURCE TERM

CONTAINMENT PERFORMANCE

POSITIVE VOID REACTIVITY

CONTROL ROOM - SECONDARY CONTROL AREA

SEISMIC DESIGN

QUALITY GROUP BOUNDARIES

FIRE PROTECTION

ON-POWER FUELLING

HUMAN FACTORS ENGINEERING

PRA

BACKUP SLIDE

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February 2, 1995

Docket No. STN-52-005
File No. 09000401
Control No. 950202001

Document Control Desk
U.S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, DC 20555

Subject: **Response to NRC Request for Information Regarding CANDU 3U Application for Final Design Approval and Design Certification**

Refs: 1. NRC letter to AECLT (D. M. Crutchfield to A. D. Hink) dated December 15, 1994: Results of the Acceptance Review for AECL Technologies' Application for Final Design Approval and Standard Design Certification for the CANDU 3U Design

2. NRC Meeting Summary, dated January 12, 1995: Summary of Meeting Held with AECL and AECLT in Ottawa, Ontario

Gentlemen:

This letter responds to the results of the NRC staff acceptance review of the subject application as documented in Reference 1.

AECL Technologies Inc. (AECLT) has developed a schedule for submittal of additional information as requested in Reference 1. The schedule for updating the Safety Analysis Report (SAR) and submitting other information (ITAAC, Technical Specifications, etc.) is provided in Enclosure 1. The schedule set forth in the enclosure is based on the review milestones developed by the NRC staff in SECY-93-097; however, the milestones have been relocated to coincide with an AECLT-proposed schedule for the review. This schedule was presented to members of the NRC staff at a January 30, 1995 meeting to discuss cost and schedule for the CANDU review. The schedule proposes a period of generic licensing issue review followed by the specific plant design review. This timing is based on the current AECL CANDU plans and schedules for developing design information for the full CANDU plant product line. AECLT is planning to meet with the NRC staff to identify

Attachment 3 D037

and address several technical and safety issues that are generic to the review of CANDU plants using NRC regulatory requirements and guidance. Therefore, the period prior to initiating the specific design review would be used productively to resolve generic issues such as those identified in SECY-93-092 and in Reference 2. These issues involve all major aspects of CANDU technology; therefore, their evaluation and resolution would result in a more efficient review of the application. Reference 1 contained specific questions in the form of a request for additional information. Many of these questions will be answered, and result in SAR updates, during the planned generic issue review effort. AECLT intends to provide responses to the remaining questions in an SAR update submitted no later than January 1997. AECLT proposes to initiate discussions with the NRC staff in the near future to identify and schedule reviews of generic CANDU issues. A list of candidate issues is provided in Enclosure 2.

At the January 30 meeting, AECLT again stated that (1) the cost for an FDA review should be more in line with that of other evolutionary plants (with due consideration of unique design differences) and (2) that AECLT could not proceed under the previous estimate of approximately $50 million if that amount were proposed to AECLT for NRC cost recovery. AECLT is looking forward to further discussions after the decision of the NRC Commission regarding confirmatory research costs and NRR's reassessment of resources for the CANDU review are available.

In the final paragraph of Reference 1, ongoing program support tasks are mentioned. It is requested that NRC staff provide AECLT with task descriptions and an estimate of the anticipated fees that will be associated with these tasks over the next year - to March 31, 1996.

Should you have any questions regarding this letter, please contact the undersigned.

Very truly yours,

M. H. Fletcher

M. H. Fletcher
Director, Safety and Licensing

Enclosures: 1. Proposed Program Milestones
2. CANDU Generic/Policy Issues

cc: D. Scaletti NRR

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Enclosure 1

Proposed Program Milestones

The following milestones are based on the program for CANDU review discussed with the NRC staff at a meeting on January 30, 1995. The proposed program involves an initial two-year review of generic/policy issues followed by a three-year detailed review of the CANDU design. The salient points affecting the program schedule are: (1) recognition of CANDU as a well-developed reactor plant design; (2) the necessity of pursuing innovative and efficient review methods to limit costs; (3) the need for AECLT to provide timely and complete information to the NRC; (4) maintaining a level of effort within AECLT budget during the generic review phase, i.e., $1M (US) the first year building up to $2M in the second year; and (5) elimination of costs for research that duplicates existing CANDU research. The milestones are also shown graphically on Figure 1.

- Present - 4/97 Review of CANDU generic licensing issues
- Present - 1/97 AECLT updates to SAR
- 4/97 NRC staff initiate specific design review
- 5/97 (approx.) AECLT initiate review seminars in selected design topics for NRC staff technical reviewers
- 1/98 AECLT submit response to NRC RAIs
- 1/98 AECLT submit Level I PSA
- 11/98 NRC issues Draft Safety Evaluation Report
- 7/98 AECLT submit Level II & III PSA, ITAAC & DAC, Technical Specifications, Initial Test Program Test Abstracts, and Severe Accident Mitigation Design Alternatives
- 3/99 AECLT respond to Draft Safety Evaluation Report and submit Final SAR, Final Technical Specifications, and Final ITAAC & DAC
- 10/99 NRC issues Final Safety Evaluation Report for Review
- 4/00 AECLT submit Design Control Document
- 9/00 FDA issued

AECLT PROPOSED PROGRAM

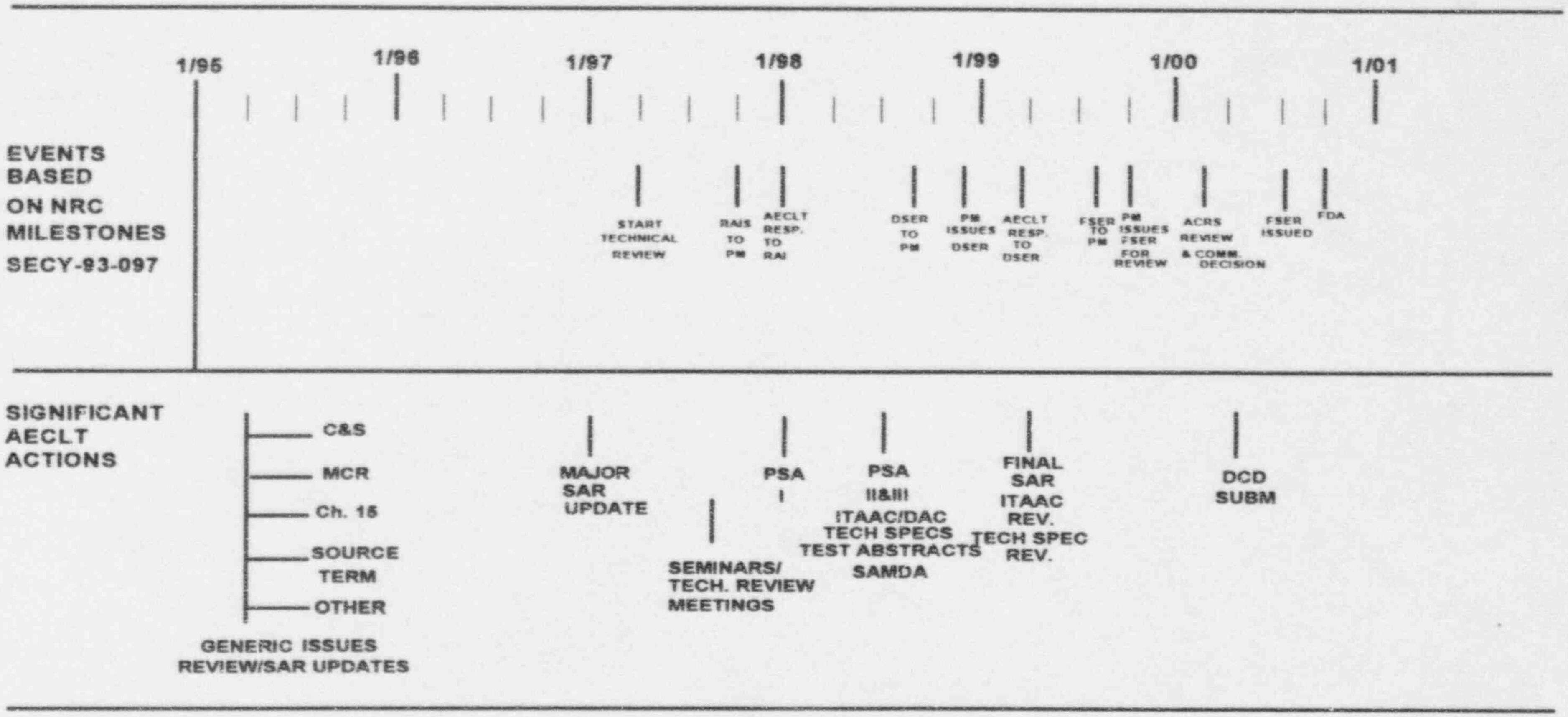

FIGURE 1

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Enclosure 2

CANDU GENERIC/POLICY ISSUES

1. Codes and Standards (structures, pressure-retaining systems, testing/inspection, electrical)

2. Accident Analysis* (event selection and evaluation, categorization of events, acceptance criteria, assumptions, analytical tools, severe accidents, external events)

3. Source Term* (mechanistic evaluation: fuel performance, transport, event-specific source terms)

4. Containment Performance* (offsite dose limits, ASME Level C limits or equivalent)

5. Positive Void Reactivity* (probability, consequences)

6. Control Room/Secondary Control Area* (CANDU philosophy, NRC GDC-19)

7. Seismic Design (assumptions for SDE, seismic margins)

8. Quality Group Boundaries (reactor coolant pressure boundary, containment penetrations, Class 6 systems, differences between U.S. and CANDU definitions)

9. Fire Protection (CANDU methodology and NRC Standard Review Plan)

10. On-Power Fuelling (safety issues and operating experience)

11. Human Factors Engineering (CANDU human factors engineering plan and criteria)

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* Issue identified in SECY-93-092

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Mr. H. D. Hink, President
AECL Technologies, Inc.
9210 Corporate Boulevard
Suite 410
Rockville, Maryland 20850

SUBJECT: RESPONSE TO YOUR LETTER OF FEBRUARY 2, 1995

Dear Mr. Hink:

Enclosed is a copy of SECY-95-XXX, "Change in Plans for Design Certification Review of the CANDU 3U Reactor." In the Commission paper, the staff details plans for the CANDU 3U review in response to the February 2, 1995, letter from Mr. Fletcher of your staff.

In our letter of December 15, 1994 we responded to your request for limited continuing program support. The staff plans to continue a low level of effort in accordance with your needs on key issues such as void reactivity and shutdown system reliability. We will track our resource expenditures closely, inform you when we are close to your budget limit, and stop work when the limit is reached, unless you indicate that further support is available. We request that you give us your budget limits for our fiscal year (October 1 through September 30) to assist us in our planning. We have also advised the Office of Nuclear Regulatory Research of your letter and stated that they should cease all design-specific research in support of the CANDU application until further notice.

The staff has three contracts nearing completion; these contracts address fuels, core physics, and reactivity analysis. The contractors are preparing the draft reports for these projects, which should be complete within the next few weeks. The February and March costs to finish the three draft reports should be less than $20,000. The staff issued one contract for FY95 for $128,562 to assess the CANDU shut down system reliability.

Certain areas of your proposed review schedule will be difficult for us to accommodate as discussed in the Commission paper. The AECLT schedule calls for the staff to issue the final CANDU safety evaluation report (SER) in October 1999, allowing only 30 months for the review. Past staff experience with evolutionary and passive design review schedules, and the limited resources available, will not support your request for a 30-month schedule. The staff cannot prepare a detailed review schedule until July 1998, when it has received all of the required information such as the completed probabilistic safety assessment, inspection, test, analysis, and acceptance criteria; technical specifications; tests programs; and severe accident mitigation design alternative.

Attachment 4

If you have any question regarding this letter please contact the NRC project manager, Dino C. Scaletti at 301-415-1104.

Sincerely,

Dennis M. Crutchfield, Associate Director
for Advanced Reactors and License Renewal
Office of Nuclear Reactor Regulation

Docket No. 52-005

cc w/enclosure:
See next page