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

October 26, 1993

Project No. 679

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THRU: Edward D. Throm, Acting Director
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FROM: Dino C. Scaletti, Sr. Project Manager
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Associate Directorate for Advanced Reactors
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Office of Nuclear Reactor Regulation

SUBJECT: DAILY HIGHLIGHT - FORTHCOMING MEETING WITH AECL TECHNOLOGIES (AECLT)

DATE & ME: October 28, 1993
2 p.m. - 5 p.m.

LOCATION: U.S. NRC Headquarters
11555 Rockville Pike, Room 1/F/19
Rockville, Maryland

PURPOSE: Discuss the request for additional information related to staff contract NRC-03-93-032, "CANDU 3 Containment Performance and Consequences" (see enclosure).

Contact:
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504-1104

PROJECT 679
(File Packet)

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Original signed by:

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Enclosure:
Items of Concern

cc w/enclosure:
See next page

*Meetings between NRC technical staff and applicants or licensees are open for interested members of the public, petitioners, intervenors, or other parties to attend as observers pursuant to "Open Meeting Statement of NRC Staff Policy," 43 Federal Register 28058, 06/28/78.

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CANDU
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1.0 INTRODUCTION

The contract statement-of-work tasks SEA with reviewing CANDU 3 documents submitted to NRC by AECL Technologies with regard to the major areas of the containment system design including containment system and structural response to both design basis and severe accidents, source terms and release rates, and containment performance analysis methods and acceptance criteria. SEA is to identify the preapplicant's criteria in each area and evaluate the technical and experimental bases supporting this criteria with particular regard to areas where design, materials, or acceptance criteria differ from accepted U.S. practice. The NRC criteria are found in NUREG-0800, "Standard Review Plan, Sections 3.8.1, 6.1.2, 6.2.1, 6.2.2, and 6.2.4, through 6.2.7" (SRP). Particular attention is to be paid to issues unique to the CANDU 3 design.

In our initial review, SEA has identified in several areas where additional information is needed to complete our task and has prepared a request for this information, as directed by the statement-of-work. The following request consists of a discussion of the documentation initially transmitted to SEA and a summary of the types of additional information needed, as well as lists of reports referenced within the initial documents which will at least partially fulfill those information needs.

2.0 STATUS OF INITIAL DOCUMENTATION

This section is a brief discussion of the completeness and applicability of the initially received documents relative to the "areas of review" specified in the SRP.

<u>2.1 Prevailing Design</u> The Technical Description and Conceptual Safety Report are both 1989 documents but the design has apparently continued to evolve since then. For example, the 1989 Technical Description shows 2 main reactor coolant circulation pumps, whereas, the 1992 Technical Outline shows 4. Our review based on the documents currently available for our review will not necessarily represent the current design.

<u>2.2 Concrete Containment</u> The current documents lack much of the detail needed to complete the review as outlined in the SRP. The containment building description consists of: a few sketches (one cross-section, two elevations, and an internal structural module), the basic dimensions (height, width, and wall thickness), the design pressure, and qualitative discussions regarding materials, access, construction, etc. The SRP areas of review include such items as main reinforcement and prestressing tendons, the anchorage of the liner, loads, and loading combinations which cannot be reviewed with the current level of technical description. Information, such as containment ultimate failure pressure, subcompartment failure pressure differentials, and containment structural design details is missing.

<u>2.3 Containment Source Terms and Containment Response</u> The safety analysis presented in the Conceptual Safety Report (CSR) is described as design-assist safety analysis functioning primarily to provide assistance and input to the design and is incomplete for license application, as well as not reflecting the latest design modifications. The CSR presents only minimal results for selected calculations such as a large LOCA, presented as a bounding LOCA. The SRP "areas of review" include the containment pressure and temperature due to a spectrum of LOCAs (break size and location) for both the primary and secondary systems. The verification that the LOCA source terms

to the containment are bounding as stated in the CSR cannot be determined from the 1989 CSR. Other important analytical details such as the containment temperature are not presented. Sensitivity and uncertainty analyses are not presented.

The CSR lacks detailed analytical results needed to evaluate the validity of the source terms to the containment and containment response, such as the containment analysis nodalization and basis for that nodalization, the impact on containment response of possible combustion (with or without igniters), containment response with or without air coolers, and the possible decomposition of protective coatings.

The Technical Description lacks detailed specification of the location of piping, valves, etc. within the containment, which piping penetrates the containment walls, and safety valve discharge location and capacity. During our initial review, for instance, we were unable as ascertain whether the main steam safety valves were located inside or outside the containment building or whether or not main steam lines had isolation valves.

The trip specification of a LOCA signal was not found. For instance, can a steam generator tube rupture signal a LOCA, thereby initiating secondary crash cooling to the atmosphere (assuming safety valves are located outside containment) of an non-isolated steam line (assuming these lines indeed do not have isolation valves), thereby, bypassing the containment?

2.4 Radionuclide Source Terms and Release Rates The source term analysis document, TTR-384, contains analysis specific to CANDU 6. While this analysis will contribute to overall understanding of CANDU source term methodology, the results from the analysis in TTR-384 are not directly applicable to CANDU 3. There are major differences between the CANDU 3 and CANDU 6 containment designs. Perhaps the most significant difference influencing the radionuclide source term to the environment is that the CANDU 6 containment employs sprays for pressure suppression and radionuclide control, whereas, the CANDU 3 design does not include sprays. The TTR-384 analysis involved sprays.

2.5 Design Bases and Severe Accidents The analyses in the documents provided to SEA are all based on design basis accidents. TTR-429 (June, 1992) states that information regarding containment response to severe core damage accidents will be provided to the NRC when this response is assessed.

2.6 Analysis Methods The SRP states that the applicant should use calculational methods that have been previously reviewed by the staff and found acceptable. Most of the computer codes used by the Canadians are CANDU specific and probably have not been evaluated by the NRC. Much of the technical basis for such review areas as the LOCA mass and energy source terms to the containment and the containment response to these sources is contained within the models of these Canadian codes and within the input and results of those codes. While a complete review of all of these codes and their associated experimental verification and validation to ensure conservative results is beyond the scope of this task, some preliminary understanding of these codes is necessary in order to make an assessment of the containment source terms. Descriptions of the models employed by these codes are almost completely missing in the initial documents. Further, the Canadians apparently use several separate codes in performing an accident analysis, as opposed to a fully integrated systems code, and their methods of integrating the code results are not discussed.

• M. A. Cormier, "Containment Node Link Model for the CANDU 3 Design", AR-74-68400-002.

• M. S. Quraishi, et al., "Assessment of Higher Leakage from Containment Following Postulated Accidents", TTR-168, Volume 2, 1986 September.

• J. M. Hopwood, R. S. Porter, S. Pang, B. A. Shalaby, S. D. Grant, E. Kohn, A. Lai, V. K. Molindra, "Large LOCA Power Transient Assessment for CANDU 3", AR-74-03500-016, 1989 January.

• S. D. Grant, V. I. Nath, "A Study of Pressure Tube Heat-Up Following Postulated Large Breaks in an Inlet Header", AR-74-03500-022, 1989 January.

• M. A. Wright and M. S. Quraishi, "Analysis of the Consequences of an End Fitting Failure", AECL Report TTR-153, 1985 May.

3.4 Radionuclide Source Terms and Release Rates We request CANDU 3 specific radionuclide source term and release analysis reports, if available and the following reports.

• G. I. Hadaller, G. H. Archinoff, and E. Kohn, "CANDU Fuel Bundle Behavior During Degraded Cooling Conditions", 4th Annual Conference of the Canadian Nuclear Society, Montreal, 1983 June.

• E. Kohn, G. I. Hadaller, R. M. Sawala, G. I. Archinoff and S. L. Wadsworth, "CANDU Fuel Deformation During Degraded Cooling - Experimental Results", Canadian Nuclear Society Conference, 1985 June.

3.5 Design Basis and Severe Accidents We request reports addressing CANDU 3 severe accident analysis, if available, including probabilistic analysis of beyond design basis events.

3.6 Analysis Methods We request reports discussing Canadian accident analysis codes and their technical and experimental basis in addition to the following referenced reports.

• M. R. Lin, S. Prawirosoehardjo, "FIREBIRD-III Mod 1 Program Description", TDAI-373, 1984 November.

• A. Lai, "FIREBIRD Model for CANDU 3 232-Channel, 4-Header Primary Heat Transport System Design, AR-74-03500-032, 1988 August.

• W. M. Collins, "PRESCON2 Program Description", AECL Report TDAI-292, Volume 1, 1982 September.

• W. M. Collins,"MICROPRESCON 2 Program Description", TDAI-368, Vol. 1, 1985 February.

• . P. Muzundar, R. L. Sakaguchi, J. K. Presley, "HOTSPOT-II Fuel Bundle Thermal Response Code", Ontario Hydro Report NSSD 83058, 1983 April.

• H. Keil, "MODHT: A Computer Program to Predict Transient Moderator Temperatures -Model Description", TDAI-183, Volume 1, 1980 March.

• D. J. Richards, B. N. Hanna, N. Hobson, and K. H. Ardon, "CATHENA: A Two-Fluid Code for CANDU LOCA Analysis, (renamed from ATHENA)," Presented at the Third International Topical Meeting on Reactor Thermal Hydraulics, Newport, RI, 1985 October.

• D. J. Richards, "Validation of the CATHENA Two-Fluid Code", presented at the Third International Topical Meeting on Reactor Thermal-Hydraulics, Soeul, Korea, 1988 November 14-27.

• J. P. Mallory, M. A. Wright and H. Huynh, "Validation of CATHENA at High Temperature Conditions Using CHAN Thermal-Chemical Experiment Results," Presented at the 12th Annual CNS Conference, Saskatoon, SK, 1991 June.

• J. E. Kowolski, V. S. Krishnan, "Transient Stratified Flow Experiments in a Horizontal Channel Containing Rod Bundles and Numerical Simulations Using the ATHENA Two-Fluid Code", Presented at the International ANS-ENS Topical Meeting on Thermal Reactor Safety, San Diego, Ca., 1986 February 2-6.

<u>3.7 Performance of Unique Features</u> We request the following referenced reports.

• AECL, "Unique Aspects of the CANDU 3 Design", Atomic Energy of Canada, Limited, June 1989.

• S. D. Grant and J. M. Hopwood, "The Effect of Fuel Heat Transfer on Early Void Production Following a Large Pipe Break in CANDU Reactors", Canadian Nuclear Society Simulation Symposium, Winnipeg, Manitoba, 1988 April.

• P. G. Gulshani, "Prediction of Pressure Tube Integrity for Large Loss-of-Coolant Accident in CANDU", American Nuclear Society, 1987 Winter Meeting, Los Angles, Ca, 1987 November 15-19.

• V. I. Nath and Kohn, "High Temperature Oxidation of CANDU Fuel During a LOCA", Proceedings of the Fifth International Meeting on Thermal Nuclear Reactor Safety, Karlsruhe, 9-13 September, 1984, Kraftwerk Union Report KFK 388011, 1984 December.

<u>3.8 Regulatory and Standards Documents</u> We request the following reports.

• Hedges, K. R., M. Bonechi, and E. M. Hinchley, "CANDU 3 Meets ALWR Requirements", Modern Power Systems, 1990 December.

• AECB, "Requirements for the Safety Analysis of CANDU Nuclear Power Plants", Atomic Energy Control Board, Consultative Document C-6, Ottawa, Ontario, Canada, 1980 June.

• The following Canadian Standards. The Canadian Standards Association (CSA) Standards have apparently already been submitted to NRC.

CAN3-A23.3-M84,	Design of Concrete Structures for Buildings
CAN3-N287.1-M82	General Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants
CAN3-N287.2-M82	Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants

CAN3-N287.3-M82	Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants
CAN3-N287.4-M83	Construction, Fabrication, and Installation Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants
CAN3-N287.5-M81	Testing and Examination Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants
CAN3-N287.6-M80	Pre-Operational Proof and Leakage Rate Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants
CAN3-N287.7-M80	In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants
CAN3-N289.1-80	General Requirements for Seismic Qualification of CANDU Nuclear Power Plants
CAN3-N289.2-M81	Ground Motion Determination for Seismic Qualification of CANDU Nuclear Power Plants
CAN3-N289.3-M81	Design Procedures for Seismic Qualification for CANDU Nuclear Power Plants
CAN3-N289.4-M86	Testing Procedures for Seismic Qualification of CANDU Nuclear Power Plants