AECL
TECHNOLOGIES INC.

May 10, 2004

Our File: 108US-01321-021-001
Your File: Project No. 722

U.S. Nuclear Regulatory Commission,
Document Control Desk,
Washington, D.C. 20555

Attention: Ms. B. Sosa
Project Manager, ACR

References:

1. E-mail R. Ion to B. Sosa, "ADVANCED copy: Regulatory treatment of LCDAs", May 07, 2004.

Re: **Regulatory Treatment of Limited Core Damage Accidents for the ACR-700**

In the April 5, 2004, meeting between AECL and the NRC, AECL was asked to prepare a position paper explaining the need for a separate classification of accidents for the ACR-700 described as Limited Core Damage Accidents (LCDAs). This paper (attachment 1 to this letter) provides that position and addresses the implications of that position in the context of safety analysis and protection of the public health and safety. An advanced copy of this paper was sent to you via electronic mail on May 07, 2004 (Reference 1).

If you have any questions on this letter and/or the enclosed material please contact the undersigned at (905) 823-9060 extension 6543.

Yours sincerely,

V. J. Langman for

Vince J. Langman
ACR Licensing Manager

/Attachment:

1. Regulatory Treatment of Limited Core Damage Accidents for the ACR-700

D070

Regulatory Treatment of
Limited Core Damage Accidents
For the ACR-700

1.0 Purpose

In the April 5, 2004 meeting between Atomic Energy of Canada, Limited (AECL) and the Nuclear Regulatory Commission (NRC), AECL was asked to prepare a position paper explaining the need for a separate classification of accidents for the ACR-700 described as Limited Core Damage Accidents (LCDAs). This paper provides that position and addresses the implications of that position in the context of safety analysis and protection of the public health and safety.

LCDAs are a class of accidents that are unique to the CANDU reactors due to their use of multiple, separated fuel channels, surrounded by a cool, low pressure heavy water moderator, contained within a calandria vessel rather than a LWR core contained within a reactor vessel. LCDAs are low probability, single channel events that involve consequential failure of a single pressure tube due to severe fuel overheating in the tube, or are accidents affecting the entire core that result in fission product release due to fuel overheating but do not result in consequential pressure tube failures.

LCDAs represent a class of accidents that, in terms of their consequences lie between design basis accidents (DBAs) and severe core damage accidents (SCDAs). These accidents generally should not be classified as DBAs because LCDAs have a lower probability of occurrence, some with very low probabilities in the severe accident range, and they should not be classified as SCDAs because LCDAs have a lower magnitude of fission product release from the fuel when compared with severe accidents in US light water reactors (LWRs).

2.0 Background

The genesis for a separate category of accidents for the ACR-700 is from the fundamental differences in the primary system between pressure vessel reactors and pressure tube reactors. In the ACR-700, each pressure tube is part of a fuel channel assembly. The pressure tubes separate the fuel bundles in one tube from the bundles in neighboring tubes. The materials separating the fuel bundles in any two channels are the two pressure tubes themselves, two gas-filled gaps, two calandria tubes, and the low-pressure, low-temperature, heavy-water moderator. Thus, a major design difference between the ACR-700 reactor and US LWRs is the use of two metal pressure boundaries and a substantial heat sink of water (moderator) that separate each of the primary system flow paths that cool the fuel bundles.

These design features contribute to the enhanced safety of the ACR-700. These physical barriers, geometries, and distances promote short term and long term cooling of the core and help prevent damage in a single channel from spreading across the core.

A number of other safety features also accrue from the ACR-700 design:

- The feeder tubes that connect to and deliver reactor coolant to the pressure tubes are small diameter pipes (i.e., the largest inlet feeders are 3 inch Schedule 80 pipe and the largest outlet feeders are 3.5 inch Schedule 80 pipe). No large diameter pipes are located below or near core elevation which could potentially empty the entire core quickly; a break in a feeder tube at the core elevation would result in degraded flow in only one pressure tube ~1/300 of the core. A guillotine break of a large diameter pipe, e.g., in an inlet or outlet header, would be above the core region and would continue to draw coolant through the fuel region during the initial stages of the accident.

- The reactivity control devices in the ACR-700 are in the low-pressure, low-temperature, moderator. Since the control rods are not in the high-pressure primary system, there is no potential for the equivalent of a control rod drive nozzle failure and subsequent control rod ejection from the core. In the ACR-700, an inadvertent reactivity insertion by withdrawing a control rod at the fastest rate possible is an event that does not lead to an accident with radiological releases.

- The ACR-700 reactor core has a low amount of excess reactivity due to the use of on-power refueling. Since a small amount of fuel is added on approximately a daily basis for an equilibrium core, there is no need for the more enriched fuels used in US LWRs, which are batch loaded once every one to two years. With a batch load, the reactor design must have strong neutron poisons to control the excess reactivity. In contrast, with on-power refueling, there is much less excess reactivity to be controlled. In the ACR-700, no boron is needed in the reactor coolant to hold down reactivity. Only relatively low concentrations (i.e., up to 10 PPM boron) are used in the moderator for initial start-up of a core to suppress some of the excess reactivity present before a new core attains appreciable burn-up. At all times, a reactivity insertion event is fundamentally limited by the lack of significant excess reactivity for the ACR-700 design.

- An inherent safety feature of the ACR-700 is a result of the significant design difference noted earlier. With the fuel bundles in separate pressure tubes, rather than in a single vessel, there could be accidents affecting the fuel in only a single channel. It is a design requirement that such single channel accidents will not propagate to the fuel in other channels and therefore the potential consequences are limited to ~1/300 of the core.

- For low frequency accidents affecting the cooling of the fuel in the entire core, e.g., loss of coolant accidents with coincident failure of the ECCS, the low-pressure and low-temperature moderator provides an alternate heat sink and limits the degree of fuel damage (i.e., fuel melting is precluded). This large moderator mass provides a heat sink that demonstrates application of defense-in-depth and ensures the goal of maintaining core coolability (i.e., limits the probability of core damage given accident initiation).

The following represent the spectrum of LCDAs in the ACR-700:

- Single Channel Events

  - *Severe flow blockage in a fuel channel*

    In a severe flow blockage event, the fuel bundles in a pressure tube can heat up with the reactor still operating, but with a reduction in flow in the affected tube. While the rate and extent of heat-up would depend on the degree of the blockage, in a severe flow blockage event the coolant flow area is dramatically reduced (more than 90% of original flow area is assumed blocked) resulting in superheated steam being formed in the pressure tube. This causes the fuel bundles and pressure tube to heat up rapidly, leading to pressure tube and calandria tube failure. After pressure tube failure, the contents of the tube, consisting of superheated coolant, fission products, and possibly some of the overheated fuel, are rapidly discharged into the moderator. Propagation to neighboring channels does not occur.

  - *Stagnation feeder break*

    In a stagnation feeder break event, a specific break size in the inlet feeder is postulated so that the flow from the break creates a pressure-flow balance that results in significant reduction of flow through a single pressure tube. This causes the fuel bundles and pressure tube to heat up rapidly, leading to pressure tube and calandria tube failure (i.e., similar to the behavior of the affected channel in a severe blockage). While the likelihood of achieving the precise break flow area required to produce a stagnation feeder break is very low, the physical possibility is acknowledged as a LCDA.

- Complete core events

  - *Loss of coolant accident (LOCA) plus loss of emergency core cooling (LOCA+LOECC)*

Such an accident would involve multiple failures. For an US LWR, such classes of accidents would be categorized as a beyond-DBA or a severe accident. However, given the enhanced safety features associated with the ACR-700's channel design and the available cooling from the moderator volume, the ACR-700 is able to accommodate this event with more benign consequences. Given the reduced consequences of such accidents in the ACR-700, these accidents have also been categorized as LCDAs.

In a LOCA+ LOECC event sequence, a large diameter pipe in the primary system (one of the two inlet headers) is postulated to break and discharge coolant into containment. The reactor trips, and although a signal is generated for injection of emergency core cooling flows, multiple failures are postulated that prevent adding water to the unbroken inlet header. Without effective addition of cooling water to the primary system, inventory continues to decrease and the heat transport pumps become less and less effective until they trip automatically.

As flow decreases, fuel and pressure tube temperatures increase. Fuel heatup, clad oxidation, clad failure and consequent fission product release occurs in most channels. Eventually, the pressure tubes may become hot enough to deform under the weight of the fuel in the channel. By this time, the primary system pressure is too low to drive pressure tube creep. Any subsequent heat up of the pressure tubes may result in sagging and contact with the calandria tubes. The portion of a pressure tube, which contacts a cold calandria tube, drops in temperature significantly. Heat is transferred from the calandria tube to the surrounding moderator. The moderator cooling system removes heat from the moderator. As a result, unlike a severe accident in an US LWR that results in core melt and consequential release of a large amount of fission products, for the ACR-700 there is no fuel melting, significantly less fission product release, no failure of the fuel channels and therefore, core coolability is maintained. As a result, this category of accidents for the ACR-700 is classified as a LCDA.

3.0 Proposal for Regulatory Treatment of Accidents in ACR-700

The design and inherent safety features of the ACR-700, which involve physical separation of fuel channels and the provision of an alternate heat sink, creates a corresponding need for three categories of events: design basis accidents, severe core damage accidents, and limited core damage accidents. These categories are explained further below along with the distinctions between categories for US LWRs and the ACR-700.

3.1 Treatment of DBAs in the ACR-700

For DBAs for US LWRs, the NRC has subdivided the DBAs into three event classifications as published in Regulatory Guide 1.70 (Reference 1):

> Incidents of moderate frequency - these are incidents, any one of which may occur during a calendar year for a particular plant.
>
> Infrequent incidents - these are incidents, any one of which may occur during the lifetime of a particular plant.
>
> Limiting faults - these are occurrences that are not expected to occur but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material.

For each initiating event, the subsequent DBA sequence considers the worst single failure of safety-related equipment and a loss of offsite ac power (LOOP). The LOOP is not considered a single failure. Acceptance criteria are provided in the Standard Review Plan sections for DBAs (Reference 2) and vary for different accident sequences.

For the ACR-700, the initiating event frequency would still be used to determine the event classification, corresponding to the three categories from Regulatory Guide 1.70. This results in a system of classification of DBAs for the ACR-700 that is generally consistent with US LWR practices, i.e., the initiating event frequency determines the classification. Dose consequences for DBAs for the ACR-700 will be evaluated against an increasing set of allowables, but in all cases the consequences are within 25 Rem TEDE (and are substantially less than 25 Rem for the higher frequency events).

3.2 Treatment of SCDAs in the ACR-700

SCDAs for the ACR-700 will be identified and defined in a manner similar to severe accidents for US LWRs. For an accident sequence to be classified as a severe accident, the event sequence starts with an initiating event, and considers increasing failures of safety-related equipment (i.e., multiple failures) and a LOOP. The initiating event is followed by a sequence of failures or successes (i.e., system, function, or operator performance) that leads to undesired consequences, with a specified end state (i.e., core damage) (References 3 and 4).

The severe accident sequence is analyzed using probabilistic risk methodology to determine the core damage frequency, where core damage is defined as follows: uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage (including fuel melting) is anticipated and involving enough of the core to cause a significant release. The core damage frequency is added for all severe accident sequences and evaluated against 1) the NRC guideline of less than 1.0 E-4/yr as documented in SECY-89-102 (Reference 5) and 2) an ACR-700 adopted objective of less than 1.0 E-5/yr.

3.3 Treatment of LCDAs in the ACR-700

LCDAs have no exact analog to accident sequences for US LWRs. In a LCDA, core damage in one pressure tube is stopped from propagating to another pressure tube by the use of design features (i.e., pressure boundaries and distances between fuel channels; and moderator cooling) that do not exist in US LWRs. As a result, it would not be appropriate to classify LCDAs as either a DBA or severe accident, as those terms are applied to an US LWR.

However, there are parallels within the US regulatory system that demonstrate that the LCDA concept is not completely removed from US experience. For example, References 6 and 7 establish acceptance criteria for the control rod ejection accident evaluated in Chapter 15 of the safety analyses of US PWRs. These acceptance criteria are established specifically for this group of PWR accidents in recognition of the special safety implications of the accident. As noted in Reference 6:

> The Regulatory staff has reviewed the available experimental information concerning fuel failure thresholds. In general, failure consequences for UO_2 have been insignificant below 300 cal/g for both irradiated and unirradiated fuel rods. Therefore, a calculated radial average energy density of 280 cal/g at any axial fuel location in any fuel rod as a result of a postulated rod ejection accident provides a conservative maximum limit to ensure that core damage will be minimal and that both short-term and long-term core cooling capability will not be impaired.

This acceptance criteria does not preclude the predicted occurrence of clad melt, nor fuel melting. As a result, certain aspects of the PWR control rod ejection accident evaluation do not address the entire set of acceptance criteria normally associated with a breach in the primary PWR pressure boundary (e.g., peak cladding temperature limits, avoidance of fuel melting), separate criteria have been identified to localize and control the core damage from the initiating event so that the health and safety of the public continues to be assured.

In a similar fashion, it is possible to establish separate acceptance criteria for the ACR-700 that will ensure the long-term safety of the plant following LCDAs. These criteria are proposed as follows:

- Demonstrate using design-centered assumptions and analyses that the accident sequence for each type of LCDA would not result in fuel melting or propagation beyond the initially affected channels, thereby preventing significant releases.

- Demonstrate using design-centered assumptions and analyses that the doses from LCDAs will not exceed the 25 rem TEDE dose limit in 10 CFR 50.34.

- Demonstrate that the frequency of the event sequence for each type of LCDA is low, i.e., in the range from 10E-4 to 10E-6/yr.

- Apply defense-in-depth to ensure a low frequency of occurrence of LCDAs, including conservative design provisions, qualification requirements, applicable construction codes, and QA. Defense-in-depth will also be applied to help mitigate LCDAs, including the moderator cooling system, calandria design, and containment building.

Given the nature of the LCDAs and these acceptance criteria, it would not be appropriate to apply individual, classical DBA and severe accident acceptance criteria to LCDAs. The DBA acceptance criteria were developed for vessel reactors for which the need exists to prevent common and widespread fuel assembly damage. Similarly, it would not be appropriate to measure LCDA frequencies against core damage frequency goals and large early release frequency goals, since the consequences from the LCDAs will be shown to be markedly smaller than severe accidents.

The proposed treatment of LCDAs is similar to the regulatory framework discussed in Issues 4 and 5 in SECY-03-047 (Reference 9). As discussed therein, the NRC staff has suggested greater emphasis on the use of risk information and the evaluation of events based upon their specific consequences and probabilities. In the application for certification of the ACR, AECL will provide the information needed for the risk evaluation of the LCDAs: the consequences that are specific to the LCDA scenarios described above and the estimates of the frequencies for these events (along with the basis for those frequencies).

4.0 Conclusions

Limited core damage accidents are accidents that exhibit low frequencies of occurrence and limited consequences. LCDAs are unique to the CANDU reactors and arise from the use of multiple fuel channels, separated and surrounded by a low pressure, low temperature heavy water moderator, contained within a calandria vessel. Since LCDAs represent a separate set of accidents from those evaluated for US LWRs, the current body of US regulations and guidance do not apply to these events. Rather, a separate accident category and associated acceptance criteria for these events, as proposed in this report, should be developed and incorporated as part of the ACR-700 certification process. This approach ensures that the real differences in risk between LCDAs and SCDAs for channel-type reactors are recognized for any further risk-informed decision making assessments that may arise.

References

1. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Revision 3, November 1978.

2. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Sections for Chapter 15, Accident Analysis, various revisions.

3. Analysis Basis Document: <u>Probabilistic Safety Assessment Methodology,</u> 108-03660-AB-001 Revision 1, July 23, 2003.

4. Analysis Report: <u>Preliminary Design Assist PSA Level 1- Selected Full Power Event Trees,</u> 10810-03660-AR-001 Revision 1, January 28, 2004.

5. Staff Requirements Memorandum to the EDO on SECY-89-102, "Implementation of the Safety Goals," June 15, 1990.

6. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," Revision 0, May 1974.

7. Standard Review Plan 15.4.8, "Spectrum of Rod Ejection Accidents," Revision 2, July 1981.

8. NRC Safety Evaluation Report, BWR Owners Group - Topical Report NEDC-32975(P), "Regulatory Relaxation for BWR Loose Parts Monitoring Systems" (TAC NO. MA9643) ML010310367, January 25, 2001.

9. SECY-03-047, "Policy Issues Related to Licensing Non-Light-Water Reactor Designs", March 2003.