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SEVERE ACCIDENT PHENOMENA AND RESEARCH FOR CANDU REACTORS

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ABSTRACT

Recent activities in the Canadian research program on severe accidents are described. The unique heat sinks in CANDU® are the moderator water and the additional water in the shield tank which can, respectively, cool the fuel in the event of a total loss of coolant and emergency coolant and delay or retain a melt in the calandria vessel should the moderator be lost. Experimental programs addressing fuel cladding relocation during loss of coolant and emergency coolant, the mechanistics of core collapse and the reasons for the Canadian participation in the Rasplav program sponsored by the OECD and Kurchatov Institutes are described.

I. CANDU CHARACTERISTICS RELEVANT TO SEVERE ACCIDENTS

CANDU reactors possess two inherent supplies of water surrounding the core—the moderator that surrounds the fuel channels and the shielding water that surrounds the calandria—that can function in emergencies to prevent or contain severe core damage, Fig. 1.

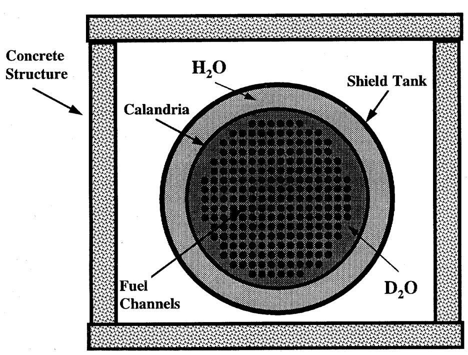

Figure 1. CANDU Core Showing Moderator Water (D_2O) and Shield Water (H_2O)

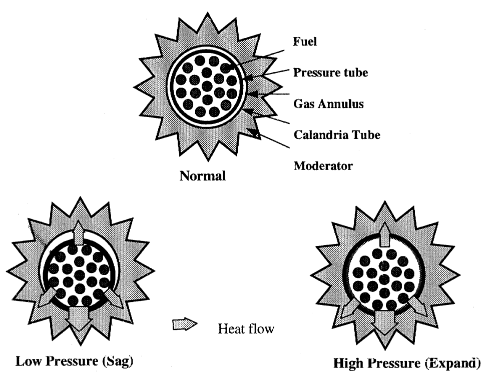

Figure 2. Mechanisms for Heat Transfer to Moderator by Deformation of Pressure Tube During LOCA-LOECC

CANDU is a horizontal pressure-tube reactor, with the fuel bundles located inside several hundred pressure tubes. An insulating gas gap separates each pressure tube from a concentric, surrounding calandria tube, which forms the outer boundary between the gas and the cool D_2O moderator. The moderator is contained within a cylindrical calandria vessel. The moderator system cools and circulates the moderator in the calandria, and can remove about 5% of reactor thermal power, equivalent to decay power shortly after shutdown. The short distance between the moderator and the fuel (1.5 cm), and the ability of the moderator to remove decay heat, allows the moderator to act as an emergency heat sink following a loss-of-coolant with failure of emergency coolant injection, Fig. 2. This heat removal path is efficient enough to prevent UO_2 melting.[1,2] The moderator specific volume is typically 8 litres/kW(th) at 1% decay power, or enough to absorb (through heat-up and boil-off) over 5 hours of decay heat from the fuel, assuming no heat removal from the moderator

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fluid; that is, even with no heat removal in a loss-of-coolant accident with simultaneous loss of emergency cooling and moderator cooling, it will slow down the rate of channel collapse and fuel melting to a timescale of hours.

The calandria vessel is in turn contained within a shield tank, Fig. 1, filled with ordinary water. In a severe core damage accident it can remove heat conducted through the calandria shell. Its cooling system can remove about 0.4 % of the thermal output of the core, the level reached a few days after reactor shutdown. The shield tank specific volume is typically 16 litres/kW(th) at 1% decay power, or enough to absorb (through heat-up and boil-off) more than ten hours of decay heat from the fuel, assuming no heat removal from the shield tank water. The shield tank cannot prevent fuel melting if all other heat removal systems, including the moderator, fail, but it can delay melt-through for hours and has the potential to indefinitely contain the melt within the calandria.

We distinguish a **severe accident** in a CANDU, defined as one in which heat is not removed through the primary cooling system, from **severe core damage**, in which the pressure-tube geometry is lost. Severe accidents in which the moderator is available do not lead to severe core damage or fuel melting. Canadian safety practice has been to include the dominant-frequency severe accidents within the design basis (e.g., Loss of Coolant (LOCA) and Loss of Emergency Core Coolant (LOECC)) and require that they meet frequency and dose limits set by the regulatory agency. As a result, the frequency of severe core damage accidents has been reduced to the point at which they are residual risk events, typically less than 10^{-6} per year on an individual event basis. The redundant heat removal paths, combined with the provision of *two* independent, diverse shutdown systems, give a summed severe core damage frequency for current CANDUs for internal events of the order of 5×10^{-6} events per year.

II. SEVERE ACCIDENTS WITHIN DESIGN BASIS

The extensive body of R&D that supports those severe accidents in the design basis has been described at the last Pacific Basin Nuclear Conference.[1] It included a description of the extensive work done to date to confirm the effectiveness of the moderator as a heat sink. Work on fuel damage, fission product chemistry and hydrogen combustion research relevant to severe accident scenarios was also described. Because much of this work dealt with the physical properties of the core materials, it provides a good basis for extending our research to study the mechanisms of core melt progression for beyond-design-basis severe accidents, i.e., severe core damage.

In a LOCA and LOECC, the fuel will heat up due to decay power and oxidation of the sheaths, and will heat up the pressure-tube through conduction, steam convection and radiation.[3] At about 650°C, the pressure tube will start to plastically deform under the loads from the weight of the fuel and any residual coolant pressure. If the coolant pressure is high, typically above 1 MPa, the pressure tube will strain radially outward (balloon) until it contacts the cool calandria tube, Fig. 2. If the pressure is below 1 MPa, the pressure tube will preferentially sag, until again it contacts the cool calandria tube. As long as the calandria tube is cooled by the moderator so that dryout on its outer surface is prevented, it remains strong enough to arrest the deformation of the pressure tube. Heat can then be removed from the fuel, by conduction and radiation to the pressure tube and calandria tube, and then by convection to the bulk moderator. From there it is removed by the moderator cooling system. The fuel bundles in such a sequence are severely damaged—distortion of bundle geometry, oxidation of the clad, and, depending on the rate of oxidation, possible formation of a zircaloy-uranium-dioxide eutectic at the clad/fuel interface. However, the UO_2 itself does not melt.

By appropriate adjustment of moderator temperature and surface condition of the calandria and pressure tube (to optimize contact conductance), nucleate boiling on the calandria tube surface can be assured and good heat transfer to the moderator will occur, as described above. It simplifies the analysis, but is not a safety requirement, if the pressure tube does not fail during this sequence, so that much of the effort has been devoted to delineating the conditions required to assure pressure-tube integrity during the deformation. The ongoing work in this area is focussed on the effect of aging of the fuel channel on the deformation behaviour. Initial experiments on small specimens cut from irradiated pressure tubes indicate that there is a small reduction in the creep rates during temperature ramping. Ballooning experiments on full size sections of pressure tubing removed from power reactors are now underway at the Whiteshell Laboratories.

III. SEVERE CORE DAMAGE ACCIDENTS BEYOND DESIGN BASIS

For residual risk sequences in which the moderator is also assumed unavailable, the fuel channels would gradually fail as the moderator boiled off, and collapse to the bottom of the calandria. Blahnik,[4] using the MAAP_CANDU code, has characterized the degradation of a CANDU core with no cooling and gradual boiling-off of the moderator. The uncovered channels heat up and slump under their own weight until they are held up by the underlying channels. Eventually, as successive layers of channels pile up, the supporting channels (still submerged) collapse and the whole core, still almost

completely solid, slumps to the bottom of the calandria vessel. While there may be some melting of the fuel cladding, the UO_2 does not melt.

Consider, for the purposes of discussing the phenomenology, a beyond-design basis severe accident that assumes loss of *all* heat transport system and emergency heat sinks at decay power levels (shutdown cooling system, main feedwater, auxiliary feedwater, Group II emergency feedwater, Emergency Core Cooling System, and moderator heat removal). In addition, assume an inability to depressurize the heat transport system (for which there are two independent signals). The analysis shows that only a very small number of channels is expected to fail at the high heat transport system pressures[4] (~6 to 10 MPa). Such a failure would first occur due to local straining in a high decay power channel located at a high elevation in the reactor core. The fuel in the channel would fall directly to the bottom of the calandria vessel, where it will be adequately cooled by the surrounding moderator. The failure of one or two channels would induce rapid depressurization of the partially voided heat transport system and allow the pressure tubes to strain into contact with the calandria tubes without failure. If the same accident took place at intermediate pressures (~1 to 6 MPa) the pressure tubes will balloon into full circumferential contact with the calandria tube. The channel will remain intact as long as the outside surface of the calandria tube does not undergo a prolonged period of film boiling or is surrounded by a void.[3] With loss of moderator heat removal, the moderator would gradually boil off. A voiding of the channel outside surface occurs when the moderator level in the calandria vessel falls below the channel. Again, one or a few high power channels located at a high elevation and uncovered early could fail. The number of channels that could fail by this mechanism is also expected to be very small. Therefore, as the moderator level falls further, the majority of channels at decay power levels are expected to fail under low system pressures (<1 MPa). The mechanisms of channel failure under those conditions are expected to be through excessive sag and/or local overheating. To determine the mechanisms, a research program was initiated. This program is at its initial stage; however some details of this program are included in this paper under the topic of channel collapse experiments.

In channels of high decay power the fuel sheath may exceed its melting point because of locally driven exothermic Zircaloy/steam reaction. However, the fuel temperatures will be considerably less than the melting temperature of UO_2 (~2800°C). Therefore, almost all of the fuel is expected to be in the solid state. The oxidized Zircaloy is also expected to be in solid state; the melting point of ZrO_2 is ~2700°C. A small fraction of unoxidized Zircaloy might be molten or solid, depending on its location and environment. The behaviour of molten Zircaloy in a fuel channel under such accident conditions, as well as in a number of other accident scenarios, was identified as one of the phenomena to be understood. Therefore, extensive research has been carried out on the relocation of molten material in a CANDU-specific subchannel geometry.[5]

Unlike in the vertically oriented Light Water Reactor (LWR) fuel rods, where the movement of the molten material is caused by gravitational forces (candling), and can take place over long vertical distances (metres), the movement of molten material in a CANDU subchannel geometry takes place over vertical distances of a few centimetres (the diameter of the pressure tube) and is influenced by additional capillary forces. This was observed when horizontally oriented CANDU bundles were exposed to simulated accident conditions reaching temperatures in excess of the melting point of Zircaloy.[6]

Post test examination of the bundles showed that molten Zircaloy had moved into the subchannels between the fuel elements. The peak temperatures attained in those tests were less than predicted. It was suggested that the lower-than-expected maximum temperatures may in part be attributed to the mitigating influence of the relocation of Zircaloy.[7] Analysis of the test results suggested that a reduction in the initial surface area of the sheath exposed to steam could have a significant effect on the reduction of heat generated by the exothermic Zircaloy/steam reaction. On the basis of these developments, experiments were undertaken at AECL Whiteshell Laboratories to study the relocation of molten material in a CANDU subchannel geometry and to evaluate its influence on the heat generation rate.

An experimental facility was built to conduct relocation tests in small samples of CANDU-specific subchannel geometries. The facility was capable of attaining hot-zone temperatures of about 1850°C. The exothermic Zircaloy/steam reaction was expected to raise the sample temperatures even higher. The heart of the facility is a vertically-oriented furnace heated by a graphite resistance element enclosed in an inert-gas-filled water-cooled steel chamber. The test sample was loaded into the bottom of the furnace tube, the system closed, evacuated and back-filled with ultra-high purity argon. The sample was then inserted part way into the furnace hot zone by a motorized ram, to be preheated to either ~1000°C or ~1400°C. Superheated steam was allowed to enter the bottom of the furnace tube while the sample was being inserted into the centre of the hot zone. The timing of the two actions was synchronized so that the sheath had oxidized prior to its arrival at the centre of the hot zone. By controlling the insertion rate and preheat temperature of the sample a wide range of heat-up rates could be achieved.

In a CANDU bundle the majority of subchannels have a trefoil geometry. Therefore, in most of the tests a sample consisting of two and one-half sheathed, hollow UO_2 pellets arranged horizontally in a trefoil geometry was used. The

temperature of the sheath during a test was monitored with a thermocouple spot-welded to the sheath and with a pyrometer focused onto the sheath. The rate of hydrogen evolution was monitored throughout the test using a volume displacement technique. The samples were sectioned to evaluate the extent of molten material relocation and the surface area of relocated material exposed to steam. The composition of various phases was also determined using a scanning electron microscope.

A photograph of a trefoil specimen after being sectioned perpendicularly to the pellet axis is shown in Fig. 3. A white oxide layer (A, ZrO_2) completely surrounds the three pellets and the molten material is contained inside that "crucible". The darker material beneath this layer (at B, for example)'is oxidized previously-molten material containing $U((Zr,U)O_{2-x})$. Some of the molten material relocated only a short distance or remained essentially in place (at B, for example). Other molten material relocated into the triangular region between the three pellets (C). In some specimens, the material in the triangular region was only partially oxidized; material that remained unoxidized contained Zr, U and the Zircaloy-4 alloying elements (D). The formation of the crucible reduces the original surface area of the sheath exposed to steam and restricts steam access to the inside of the triangular interstice, where noxidized molten material was observed. Analysis of the temperature transients confirmed that the reduced hydrogen production rate resulting from the relocation process could be satisfactorily explained if the area changes were considered in the calculations. The above tests demonstrated that unlike in a vertically oriented LWR geometry, where the molten material moves under the candling effect, the horizontal CANDU bundle restricts the molten material movement into the subchannels, where the melt is contained inside the zirconium oxide crucible surrounding the melt and continues to oxidize. These processes reduce the peak temperatures attained in an accident and reduce the hydrogen source term.

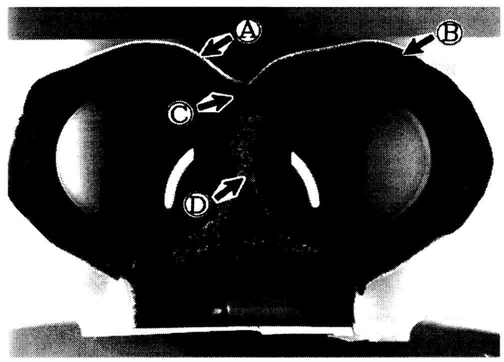

Figure 3. Trefoil Specimen Showing Relocated and Oxidized Cladding

A. Channel Collapse Experiments

As discussed above, currently a research program is underway to study the behaviour of CANDU fuel channels under severe accident conditions. In this program small-scale tests will be conducted to accomplish the objective.

Consider the loss-of-all-heat sinks model scenario previously described. As channels are uncovered during moderator boil-off, their temperatures rise and they begin to sag under gravity loads. The axial creep rate of the pressure tube material Zr-2.5 Nb of the channel increases rapidly with temperature and excessive sagging of the channel is expected to occur above ~1200°C. In Blahnik's[4] model, a sagging channel comes into contact with the next lower row of channels. The lower row of channels may or may not be cooled adequately by the moderator, depending on whether it is submerged in the moderator or not. As the moderator level continues to decrease the lower row of channels is uncovered and sags under its own weight as well as that of the supported channels. This process continues as more channels are uncovered. As sagging increases, channel segments separate near the bundle junctions by sag-induced local strain. A suspended debris bed is thus formed which moves downward with the falling moderator level. Since a submerged channel can support only a finite number of channels, the ends of those channels are expected to fail by shear. This process will increase the loading on the channels below leading to progressive failure of the channels resulting ultimately in the collapse of the reactor core into the liquid pool in the bottom of the calandria vessel.

The research program is aimed at understanding how the majority of the channels of a CANDU core would collapse by conducting small-scale experiments. From those experiments the key parameters controlling the channel collapse will be identified and a failure criterion will be developed. Since the tests will be conducted using a small-scale geometry, a scaling analysis was done. One of the key parameters for scaling calculations is the axial stress in the channel because the creep behaviour of the channel depends significantly on the stress. A full size CANDU channel was scaled down to about the size of a CANDU fuel sheath by maintaining the same stress level. Since most of the load is carried by the pressure tube at low system pressures and the creep behaviour depends strongly on the material selected, the simulated channel is made of Zr-2.5 Nb. Other scaling parameters include the ratio of the channel length to the length of a CANDU bundle and the ratio of the channel length to pitch height between the neighbouring channels of a CANDU core. The axial temperature distribution of the full size channel will be maintained in the scaled down channel to simulate the accident scenario. To aid in the design of the experiments the commercially available code ABAQUS is used.

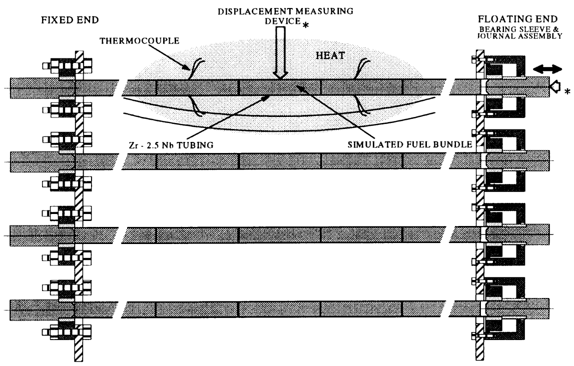

Figure 4. Conceptual Core Disassembly Facility

Figure 4 shows a schematic of the test facility designed to study the disassembly behaviour. Since one end of the channel is fixed in a CANDU reactor and the other end allows free axial movement the channels of the small-scale tests will simulate those boundary conditions. The bundles are simulated by scaled down ceramic pellets. The load exerted by the channel at the fixed end, the axial temperature distribution, the axial and longitudinal displacement including any localized straining of the channel, will be monitored. Since neighbouring channels influence the disassembly behaviour, several levels of channels are modelled in the small-scale tests. Data obtained from the experiments will provide a better understanding of the disassembly of a CANDU core. The results of these tests will reduce the uncertainty of channel disassembly behaviour under those accident conditions.

IV. CORE MELT RETENTION

Rogers, et al,[8] have developed an empirically-based mechanistic model of the collapse process, that shows that the end-state consists of a bed of dry, solid, coarse debris irrespective of the initiating event and the core collapse process. Heat-up of the debris bed is relatively slow, because of the low power density of the mixed debris and the spatial dispersion provided by the calandria shell, with melting beginning in the interior of the bed about two hours after the start of bed heat-up. The upper and lower surfaces of the debris remain well below the melting point and heat fluxes to the shield tank water are well below the critical heat flux at the existing conditions. The calandria vessel is protected by a solid crust of material on the inside, and by water on the outside, so it can prevent the debris from escaping. Should the shield tank water not be cooled, it will boil off, and the calandria vessel will eventually fail by melt-through, but this will not occur in less than a day, giving ample time for operator action such as flooding the shield tank from emergency supplies. The analysis shows considerable margin against dryout of the calandria vessel (CV) as long as the shield water is present. However, an opportunity has arisen in the LWR community by which we can obtain data that would give additional support to our analysis.

Thus, Canada has become a participant in the OECD/NEA/Russia-sponsored Rasplav program[9,10] being carried out in Russia's Kurchatov Institute. The objective of this program is to investigate corium melt-vessel interactions and local heat flux distributions resulting from the relocation of core material downward to the lower head of the reactor pressure vessel (RPV) during the late phase of a severe accident, as occurred in the TMI-2 accident. Although primarily driven by a need to understand and formulate accident management strategies for retention of the molten core inside the RPV for current and future designs of LWRs via ex-vessel cooling, the program is directly relevant to the CANDU reactor in which inherent cooling of the relocated core in the calandria vessel (CV) is provided by the large volume of shield tank water. Note that the calandria vessel is a low-pressure vessel, containing moderator at essentially atmospheric pressure and with walls typically 2.5 cm thick, so that heat transfer is facilitated relative to the 15-cm walls of a pressure vessel.

When reactor core material (composed mainly of a coarse mixture of UO_2, ZrO_2 and Zr) relocates as debris to the bottom of the CV in a CANDU severe accident, it is cooled by any water remaining in the CV. This water eventually boils off due to the continuous internal decay heat generation, and subsequent cooling is via heat rejection to the shield tank water by conduction along the bottom surface, and by primarily radiation/convection from the top surface. These processes occur over a long transient period during which any excess internal energy generated that is not rejected to the shield tank goes to heat up the debris. Some additional heat generation due to Zircaloy oxidation by the ambient steam may also occur. After a period of several hours, a steady-state condition will be reached with a hot and possibly molten pool of material surrounded by a crust, Fig. 5. The distribution of heat flux through the flat upper surface, and the curved lower surface, depends on the imposed boundary conditions and the natural convection process occurring within the molten pool. At issue is whether the heat flux delivered to the shield tank water is sufficient to induce dryout on the outer surface of the CV, thereby increasing the local steel temperature and threatening CV integrity.

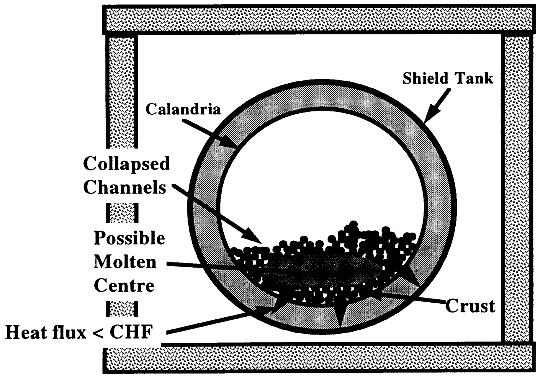

Figure 5. Collapsed Core Showing Melt Retention Through Shield Water Cooling

Natural convection in a heat generating pool is mainly characterized by two dimensionless numbers, namely, the Prandtl number and a Rayleigh number, in the mass and energy conservation equations. The heat flux from the pool to the various boundaries (top, bottom, side) is characterized by the Nusselt number, Nu. Experiments to date using fluids such as salt, water and freon for geometries approaching semi-scale appear to confirm the Nu=Constant Ra^n type relationship where Ra is the modified Rayleigh number. However, more prototype experiments are needed to confirm application of the existing correlations for natural convection to the reactor case.

The Rasplav program will be the first of its kind to provide such experimental data for real corium melts at reasonably large scale (equivalent to one-half reactor pool height, which governs the Ra number, for CANDU). The program will also address the effects of crust formation (and any feedback effect on heat transfer) and corium-vessel interactions. Sufficient external cooling is provided to prevent melting of the steel vessel. Approximately 2-3 corium experiments are anticipated using 200 kg of melt in a 2-D slice geometry of radius 0.4 metres and 20 cm-thick. At the time of writing, the first corium experiment is planned for July, 1996.

Extensive separate effects studies of scaling, heating method, melt retention materials and design, instrumentation development, and so on, as well as two 1:2.5 scale tests, have been performed by the Russians to ensure that prototypical conditions are obtained up to Ra ~ 10^{11}. For example, it became apparent that corium tests using internal heat generation by electrolytic conduction was not practical. However, it was shown by the Russian modellers that side-wall heating was equivalent to volumetric heating as long as the thickness of the 2-D slice was not so small that boundary layer effects dominated, and not so large that lateral temperature gradients became important.

Also included in the Rasplav program is a more extensive series of tests with similar geometry using various molten salts. The salt tests will provide reference data for comparison with the corium tests, and allow extrapolation of the natural convection phenomena up to Ra ~ 10^{15} while retaining the ability to form a crust. The higher Raleigh numbers (and hence higher Nu) are relevant to molten pool heights on the order of 2 metres for some LWRs. They are less relevant to CANDU since the maximum molten pool height is only about 1 metre, due to the larger diameter vessel (by approximately a factor of 2) and cylindrical versus hemi-spherical geometry in CANDU. Another important purpose of the salt tests is to compare heating methods (volumetric heat generation versus side-wall heating) for the same Ra numbers as in the corium tests to verify the modelling predictions referred to earlier. These tests are therefore important to demonstrate that the heat transfer correlations obtained from the side-wall heated corium tests are valid when applied to the reactor case in general, and to CANDU application[8] in particular.

The main experimental parameters to be measured in the corium and salt tests are the local melt temperatures, crust thicknesses, wall temperatures and heat flux distribution, and a system heat balance. These data will yield the Nusselt number distribution along the pool boundary. In the corium tests, post-test metallurgical examinations will determine whether any eutectic has formed between the melt and steel vessel. This is less likely to be a concern in CANDU application due to the low inner-wall temperature of the relatively thin steel CV. The tests will also address the issue of whether the less dense metallic Zr can stratify towards the upper surface of the molten pool and affect the upward heat transfer and/or side heat transfer where it may be in direct contact with the steel vessel and cause dryout. The uncertainty regarding the potential to cause local hot spots on the CV was alluded to in Ref. 8. However, it appears highly unlikely that such stratification will occur due to the highly turbulent, churning motion accompanying natural convection in the melt.

It should be noted that the Rasplav program is not intended to determine the critical heat flux on the ex-vessel side. Rather, the inner-wall heat flux data from the Rasplav tests is to be used to derive the heat transfer correlations such as employed in Ref. 8. In addition, any transient thermal effects associated with debris heat-up, melting relocation, resolidification and remelting during the long transient period before steady-state is reached, will also be evident in the Rasplav corium test data. This is directly relevant to CANDU, where it is shown[8] that the starting debris temperature is always well below melting temperature, i.e., no molten material is initially in contact with the vessel wall. The Rasplav corium tests should therefore be useful for developing a transient debris bed melting model.

An important aspect of the Rasplav program is the measurement of thermophysical, thermodynamic and thermochemical properties of the corium mixture as a function of temperature. Three different corium compositions are being investigated, as a minimum, namely C-22, C-75 and C-100, representing various ranges of Zr oxidation in a starting mixture which is approximately 80% UO_2 + 20% Zr by mass. Properties to be measured include solidus and liquidus temperatures, density, thermal conductivity, dynamic viscosity, and thermal expansivity. Since application to the reactor case requires a knowledge of these properties for specific corium mixture compositions, depending on the accident scenario, extensive use and possibly expansion of this materials properties data base is anticipated, regardless of reactor type. This aspect of the Rasplav program addresses one of the uncertainties identified in Ref. 8.

Another major component of the Rasplav program is the development of models for describing the natural convection behaviour of molten material with imposed boundary conditions which result in phase change. These models, which include the effects of using either side-wall or volumetric heating, turbulence models at high Ra numbers, and so on, have provided good insight into the physical processes involved in the design of the experiments. Application of the Rasplav data may take on one of two forms. They may be used either to derive simple heat transfer correlations in phenomenologically based calculations (such as the code MOLPOOL in Reference 8), or to validate more fundamental Computational Fluid Dynamics (CFD) codes which may be used for actual reactor calculations.

Finally, it should be noted that the Rasplav experiments are viewed as confirmatory experiments. We do not expect any major surprises, particularly for CANDU application. In view of the relatively low power density, long heat-up time, shallow molten pool height, and large surface area for heat rejection to the cylindrical shield tank, the maximum heat flux (~150 kW/m^2) is predicted in Ref. 8 to be well below the limiting heat flux for dryout (~300 kW/m^2) so long as the shield tank water is present.

V. SUMMARY

A review of the assumptions associated with the analysis of core melt progression has been completed and several areas for confirmatory experiments have been identified. One study is examining the relocation of molten cladding in the CANDU horizontal bundle as this determines the input to the analysis of pressure tube integrity and the ultimate temperatures in the fuel channel. These temperatures are determined by the Zirconium-steam reaction which in turn depends on the exposed surface area of zirconium.

A scaleable model of a CANDU core using 1/10 scale Zr-2.5% Nb tubes to simulate pressure tubes has been developed and experiments are planned to heat the assembly to the collapse condition and validate the models for core collapse.

Finally, we are working to improve our understanding of the mechanisms by which a melt can be retained in a vessel by external cooling. Our own efforts are supplemented by our participation in the Rasplav program which is addressing the similar problem of core melt retention by externally flooding a pressure vessel. The very large international effort in this project will advance the technology for the CANDU scenarios as well as for the LWR community.

The Canadian approach to severe accidents has been, as noted, to emphasize prevention, protection and mitigation over accommodation, and to include the frequency-dominant severe accidents within the design basis, so that their consequences are minor in terms of public health. The CANDU R&D programme on severe accidents has focussed, therefore, on those within the design basis, to provide guidance to operators in management of an accident, assurance to the regulator that the consequences are indeed no greater than predicted, and confirmation of the Probabilistic Safety Analyses used to guide the design and confirm that the risk to the public is low.

This left little incentive for extensive work on residual risk events, since the scope of severe core damage accident R&D has to be very carefully chosen to ensure a safety payback for the money spent. Nevertheless, customers and regulators are increasingly interested in the low-frequency residual risk events, and with assurance that there is no "cliff-edge" at very low frequencies. This requires prediction of consequences of core melt, albeit very approximate. The CANDU R&D programme is therefore expanding, beyond examining those severe accidents within the design basis, to those features of residual risk events that are unique to CANDU, specifically in the mode of collapse of the channels and the behaviour of the debris bed in the calandria shell.

VI. REFERENCES

1. L.A. Simpson, "Severe Accident Research in Canada," Proc. 9th Pacific Basin Nuclear Conference, Sydney, Australia, May 1-6, 1994, p.595-602, Australian Nuclear Association Inc. and The Institute of Engineers, Australia.

2. V.G. Snell, S. Alikhan, G. Frescura, J.Q. Howieson, F. King, J.T. Rogers, and H. Tamm, "CANDU Safety Under Severe Accidents: An Overview," IAEA/OECD International Symposium on Severe Accidents in Nuclear Power Plants, Sorrento, Italy, March 1988. Also Atomic Energy of Canada Ltd. Report, AECL-9802.
3. J.Q. Howieson, "CANDU Moderator Heat Sink in Severe Accidents," Proc. 2nd International Topical Meeting on Nuclear Power Plant Thermohydraulics and Operations, Tokyo, Japan, April, 1986, p.5-140-5-143.
4. C. Blahnik, et al., "Modular Accident Analysis Program for CANDU Reactors," Proc. 12th Annual Canadian Nuclear Society Conference, Saskatoon, Saskatchewan, Canada, June 9-12, 1991, p.235-242.
5. P.M. Mathew, et al., "Relocation of Molten Zircaloy in a CANDU Subchannel Geometry," 4th International Conference on CANDU Fuel, Pembroke, Ontario, Canada, Oct. 1-4, 1995, sponsored by Canadian Nuclear Society, CANDU Owners Group and IAEA.
6. E. Kohn, et al., "CANDU Fuel Deformation During Degraded Cooling (Experimental Results)," Proc. 6th Annual Canadian Nuclear Society Conference, Ottawa, Ontario, Canada, June 3-4, 1985, p.16.39-16.45.
7. O. Akalin, et al., "Fuel Temperature Escalation in Severe Accidents," Proc. 6th Annual Canadian Nuclear Society Conference, Ottawa, Ontario, Canada, June 9-12, 1985, p.16.26-16.32.
8. J.T. Rogers, et al., "Coolability of Severely Degraded CANDU Cores," ICHMT International Seminar on Heat and Mass Transfer in Severe Reactor Accidents, Cesme, Turkey, May 21-26, 1995. Also Atomic Energy of Canada Ltd. Report, AECL-11110.
9. T.P. Speis and A. Behbabani, "Rasplav: A Unique OECD/Russian Experimental/Analytical Program in Severe Accident Management/Mitigation," ENS Topsafe '95, Budapest, Hungary, September 24-27, 1995, p.180-195.
10. T.P. Speis and V. Asmolov, "Rasplav: Refining RPV Integrity Strategies for Accident Management," Nuclear Europe Worldscan (1995), v. 15(9-10), p.46-47.