

ADDRESSING SEVERE ACCIDENTS IN THE CANDU 9 DESIGN

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Abstract

CANDU 9 is a single-unit evolutionary heavy-water reactor based on the Bruce/Darlington plants. Severe accident issues are being systematically addressed in CANDU 9, which includes a number of unique features for prevention and mitigation of severe accidents. A comprehensive severe accident analysis program has been formulated with feedback from potential clients and the Canadian regulatory agency. Preliminary Probabilistic Safety Analyses have identified the sequences and frequency of system and human failures that may potentially lead to initial conditions indicating onset of severe core damage. Severe accident consequence analyses have used these sequences as a guide to assess passive heatsinks for the core, and containment performance. Estimates of the containment response to mass and energy injections typical of postulated severe accidents have been made and the results are presented. We find that inherent CANDU severe accident mitigation features, such as the presence of large water volumes near the fuel (moderator and shield tank), permit a relatively slow severe accident progression under most plant damage states, facilitate debris coolability and allow ample time for the operator to arrest the progression within, progressively, the fuel channels, calandria vessel or shield tank. The large-volume CANDU 9 containment design complements these features because of the long times to reach failure.

1. OVERVIEW

CANDU reactors possess two inherent supplies of water close to the fuel: the moderator which surrounds the fuel channels, and the shielding water which surrounds the calandria. 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 core coolant injection. This heat removal path is efficient enough to prevent UO₂ melting. The shield tank in a severe core damage accident can remove heat conducted through the calandria shell. 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.

For this reason, 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 and Loss of Emergency Core Coolant (LOCA + LOECC). 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.

For *severe accidents within the design basis*, typically a LOCA + LOECC, the fuel will heat up due to decay power and will heat up the pressure-tube through conduction, steam convection and radiation. At about 800°C, the pressure tube will start to plastically deform under the loads from the weight of the fuel and any residual coolant pressure, and strain or sag to contact the calandria tube. Since the calandria tube is cooled by the moderator, it will arrest the deformation of the pressure tube and provide a heat removal path to the bulk moderator. In this mode the pressure-tube acts as a fuse, deforming to allow efficient heat removal. The fuel bundles in such a sequence are severely damaged, with phenomena such as distortion of bundle geometry, oxidation of the clad, and, depending on the rate of oxidation, possible formation of a zirconium-uranium-dioxide eutectic at the clad/fuel interface. However the UO₂ itself does not melt.

As noted, severe core damage accidents beyond the design basis are residual risk events. A necessary requirement for severe core damage, defined for CANDUs as a widespread loss of channel integrity, to occur, is that the fuel channels not only be voided from within due to loss of HTS cooling and failure of ECC to inject, but that they additionally be voided from outside due to loss of moderator. In that case the fuel channels would gradually fail and collapse to the bottom of the calandria as the moderator boiled off. Blahnik (3) 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 onto the underlying channels. Eventually, the supporting channels (still submerged) collapse and the whole core, still almost completely solid, slumps to the bottom of the calandria. Rogers et al (1,4) have developed an empirically-based mechanistic model that shows that the end-state of core disassembly consists of a bed of dry, solid, coarse debris irrespective of the initiating event and the core disassembly process. Heat-up 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 from the calandria to the shield tank water are well below the critical heat flux at the existing conditions. The calandria can therefore prevent the debris from escaping. Should the shield tank water not be cooled, it will boil off, and the calandria 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.

Because of the two redundant, diverse, physically separate, fully capable, independent, testable, dedicated shutdown systems, a failure to shutdown when required is a very low probability event, typically less than 10^{-8} events per reactor year, as predicted by the Probabilistic Safety Analysis. Therefore, severe core damage accidents resulting from failure of the control system and both of the two shutdown systems to shut the reactor down when required, are not considered. Additionally, severe core damage sequences resulting in core-wide high pressure melt ejection are irrelevant to CANDU reactors; simply put, the pressure tube again acts as a fuse and a small number of pressure-tube failures will relieve the internal pressure before much melting has occurred. References 1,2,3 and 4 confirm that severe core damage can occur only at low pressures and channel damage resulting from loss of all heat sinks results in predominantly solid debris.

Severe accident mitigation capabilities are being systematically addressed early in the CANDU 9 design process, which includes more explicit mitigation of severe core damage accidents, as well as meeting the traditional requirements for design basis accidents. Drawing from the methodologies used for severe accident analysis for similar operating reactors and the extensive research and development activities in support of the CANDU reactors, the CANDU 9 program for severe accident analysis is composed of the following elements:

- Systematic plant review,
- Probabilistic Safety Assessment (PSA) level I,
- Severe Accident Consequence Analyses (PSA level II),
- Severe Accident Design Assessments,
- Severe Accident Management Program, and
- Severe Accident Research Programs.

A *Systematic Review* of the Plant Design has been performed to identify the initiating events. A preliminary Level I *Probabilistic Safety Assessment* (PSA) is then performed and identifies the potential accident sequences that dominate risk. For design basis events, which as noted previously, include some severe accidents, the design organization then compares the results (frequency, consequences) to acceptance criteria, and determines whether further accident mitigation (such as further redundancy in process or safety-related systems) is **required**. In addition the PSA identifies beyond-design basis severe accidents, including severe core damage events. Those which lie in a frequency band between 10^{-6} and 10^{-8} events per reactor year are then examined in more detail, to estimate the consequences (*Severe Accident Consequence Analyses* or PSA Level II), and to determine whether further mitigation is cost- and risk-effective (*Severe Accident Design Assessments*).

The scope of the Level I PSA for internal events, performed in the pre-project phase, concentrates on the following initiating event classes: LOCAs and HTS leaks, feedwater and main steam line breaks, support system failures, moderator system failures, and failures following reactor shutdown. Failures include potential hardware failures and post-accident human errors. The PSA results guide the designers in the provision of appropriate redundancy, to meet reliability targets. They also assist in refinement of operator response guidelines, control centre design and the environmental qualification process. Some external events such as loss of off-site power are likewise also evaluated at an early stage. Other external events are analyzed later once a site is selected.

The CANDU 9 Severe Accident Consequence Analyses draw from the results of earlier severe accident analyses for the reference plant and other CANDU reactors (e.g. references 1,2,3,4) and concentrate on features new to this implementation. Thus a preliminary design assessment of severe accident mitigation features in the CANDU 9 reactor was undertaken. The first step was to assess the containment design against the dominant severe core damage sequences.

2. CANDU 9 DESIGN FOR SEVERE ACCIDENT MITIGATION

The CANDU 9 is a single-unit evolutionary heavy-water reactor based on the Bruce/Darlington plants with an electric output of 925 MW. Its major reactor and process systems use designs proven in the reference plants and in the single-unit CANDU 6. It also incorporates safety improvements especially for severe accident prevention and mitigation, and to increase the time available to the operator to arrest the accident progression early.

CANDU 9 uses the standard CANDU core arrangement of horizontal fuel channels cooled by heavy-water primary coolant, placed in a square lattice within a low pressure and low temperature heavy water moderator, surrounded by a large tank of light water for shielding. The 480 fuel channels, each consisting of a zirconium-niobium pressure tube in turn surrounded by a zirconium alloy (Zircaloy) calandria tube, contain twelve fuel bundles each about 0.5m in length. The 37 fuel element fuel bundles contain natural uranium sheathed in Zircaloy. The reactor structure assembly shown in Figure 1 illustrates the two additional water volumes (calandria vessel with about 330 Mg. of heavy water and shield tank with about 530 Mg. of light water, each with their own independent cooling systems)

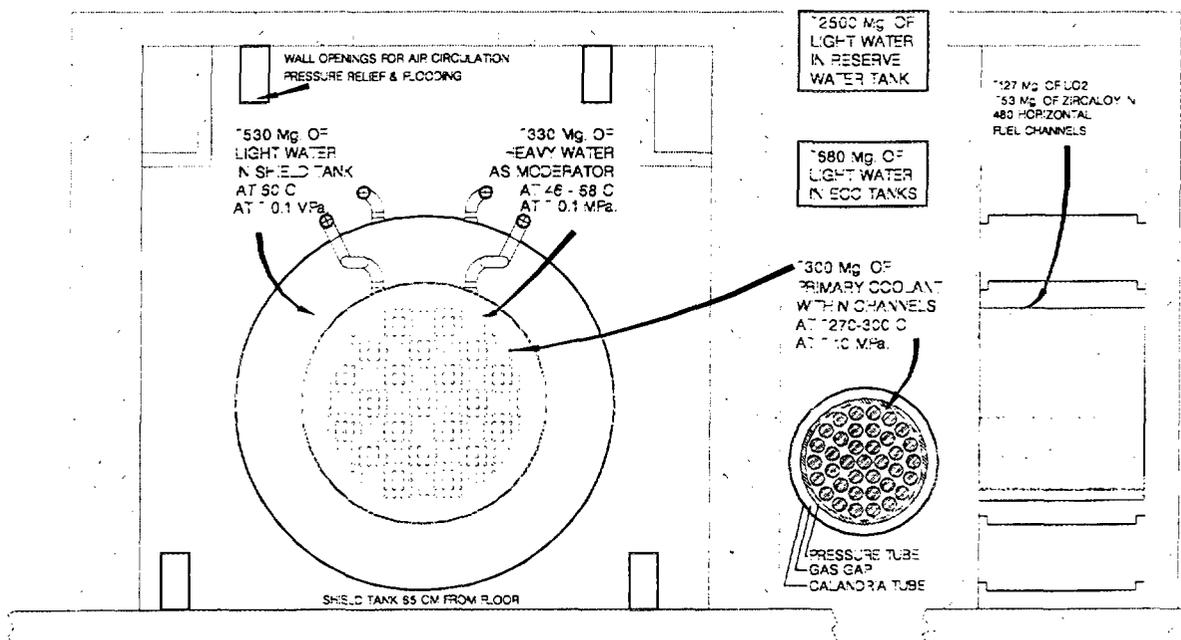


Figure 1: CANDU 9 Reactor Core and structures.

that, uniquely to CANDU, are instrumental in arresting the progression of severe accidents, such that potential debris is contained within the reactor structures (channels, calandria vessel or shield tank). CANDU 9 has an additional large tank of light water (about 2500 Mg.) located in the dome area of the containment. This Reserve Water Tank (RWT) supplies water automatically to the Emergency Core Cooling System pumps, and is also available to the operator to back up the normal heat removal systems and the front-line mitigation systems under accident conditions, specifically as emergency makeup to the steam generators, and to the heat transport system. This reduces the frequency of severe accidents. To reduce their consequences, the operator can use the RWT to keep the moderator and the shield tank filled with water, providing a means to cool the core even if the moderator and/or shield tank heat removal systems are unavailable. Water on the floor of the reactor building can be pumped back to the RWT to ensure the moderator and/or shield tank are always full, even if they leak.

The calandria-shield tank assembly, supported by two concrete reactor vault walls that span the two main concrete cross-walls of the reactor building internal structure, is located low, for structural stability against dynamic loads. This has the added benefit that the bottom of the shield tank is below the flood level.

2.1 CANDU 9 SEVERE ACCIDENT SCENARIOS

The dominant severe core damage sequences from the PSA Level I analyses are characterized as follows: channels not only lose cooling through the primary side heat transport loop, they also lose moderator as a potential heat sink. Some of the severe accident initiating events involve a loss of all heat sinks at high pressures. If unmitigated, this leads to an in-core failure of a high power channel at high heat transport system pressures. Such a failure depressurizes the heat transport system and no debris formation nor melt ejection at high pressure occurs. All subsequent core damage occurs at low pressures. Thus dominant sequences all involve channel collapse at low pressures. As reported in references 3,4, there are four severe accident end states, defined by the terminal location for debris, which will stay stable indefinitely if the specified heat sink is maintained. Three are severe core damage states; the other is a severe accident with the damaged fuel contained in the channel:

<i>Fuel / Debris location</i>	<i>Heat sink</i>	<i>Core Damage State</i>	<i>Illustration</i>
Fuel / debris in channels	Moderator water	CDS-1A	figure 2
Debris in the calandria vessel	Shield tank water	CDS-2A	figure 3
Debris in the shield tank	Base mat flood	CDS-3A	figure 4
Debris in the Reactor Vault	Base mat flood	CDS-4A	figure 5

Furthermore there are two possible variations of each of the above severe accident end states: Dry, hot fuel/debris ('A' state) or Debris covered by water (flooded - denoted as 'B' states, in later discussion). The latter implies that some recovery action has taken place to introduce water onto the debris. The "flooded debris" alternative is mainly of interest for evaluation of containment response, because it potentially involves a short period of rapid steaming (i.e. steam surge associated with the quenching of a large mass of hot debris) while the containment pressure is perhaps already elevated by earlier events. The steam surge also determines the required surge relief capacity of the various vessels or rooms (HTS, calandria vessel, shield tank, reactor vault). The surge can occur at the most inopportune time from the standpoint of other containment challenges and is so considered in the containment analyses.

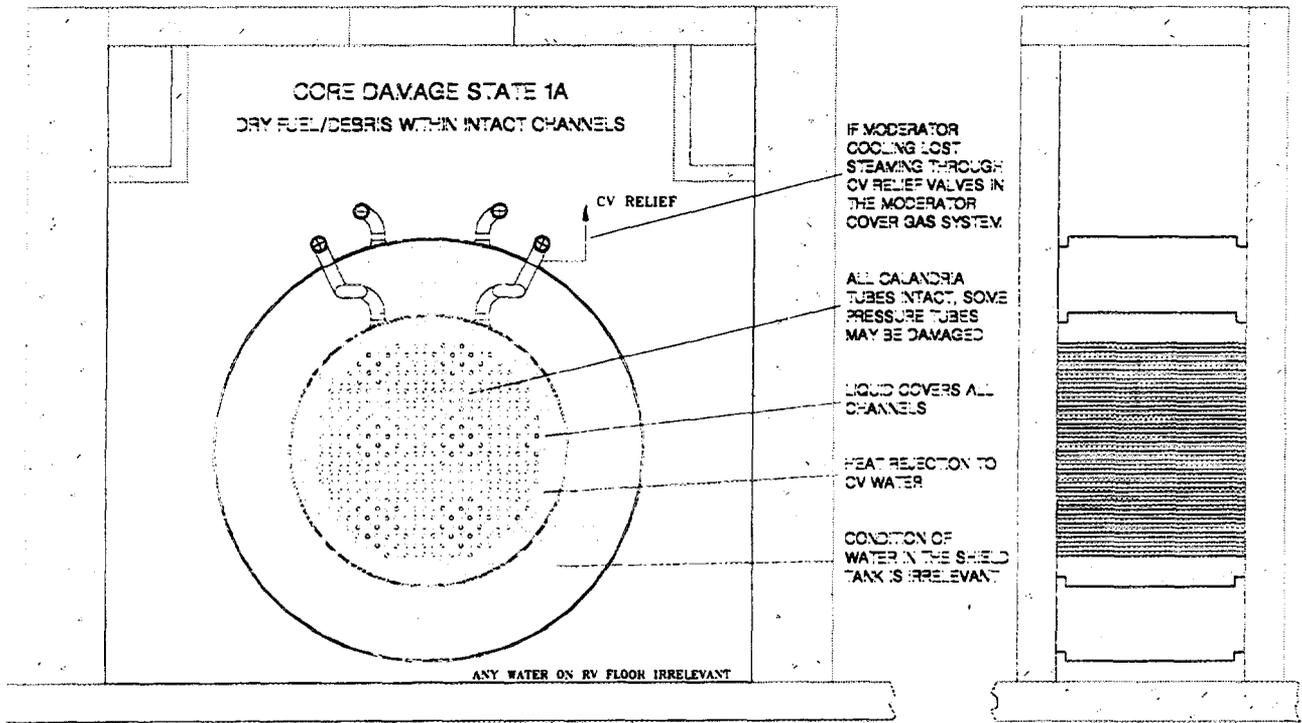


Figure 2: Hot Fuel / debris in intact channels - CDS 1A

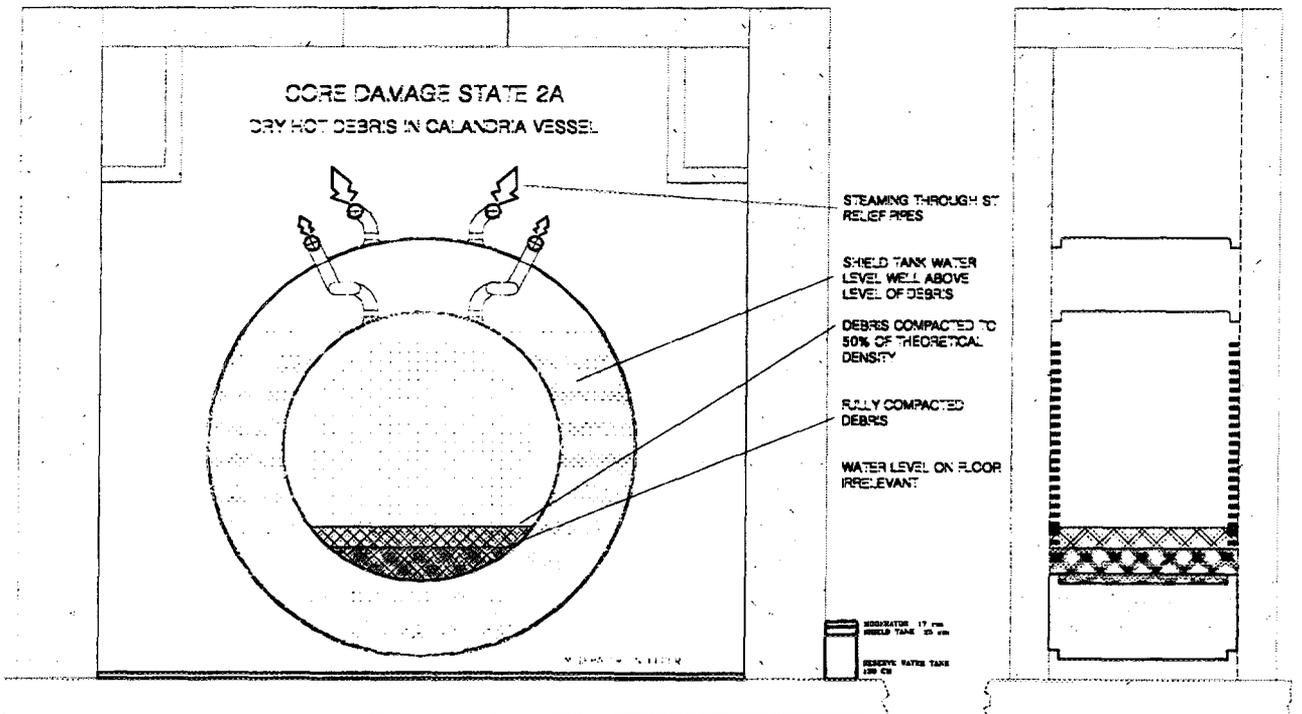


Figure 3: Hot debris in calandria vessel - CDS-2A

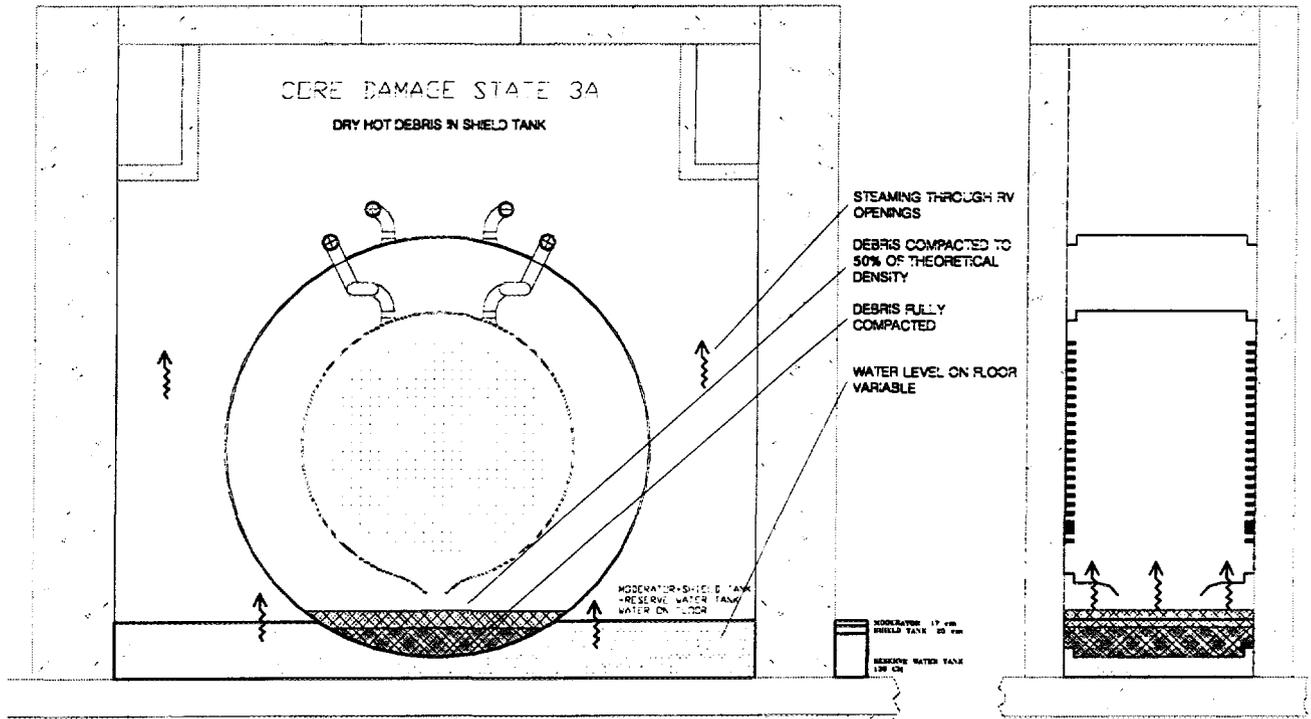


Figure 4: Hot debris in shield tank - CDS 3A [Potential contributors to the flood level on the floor are listed in Table 1]

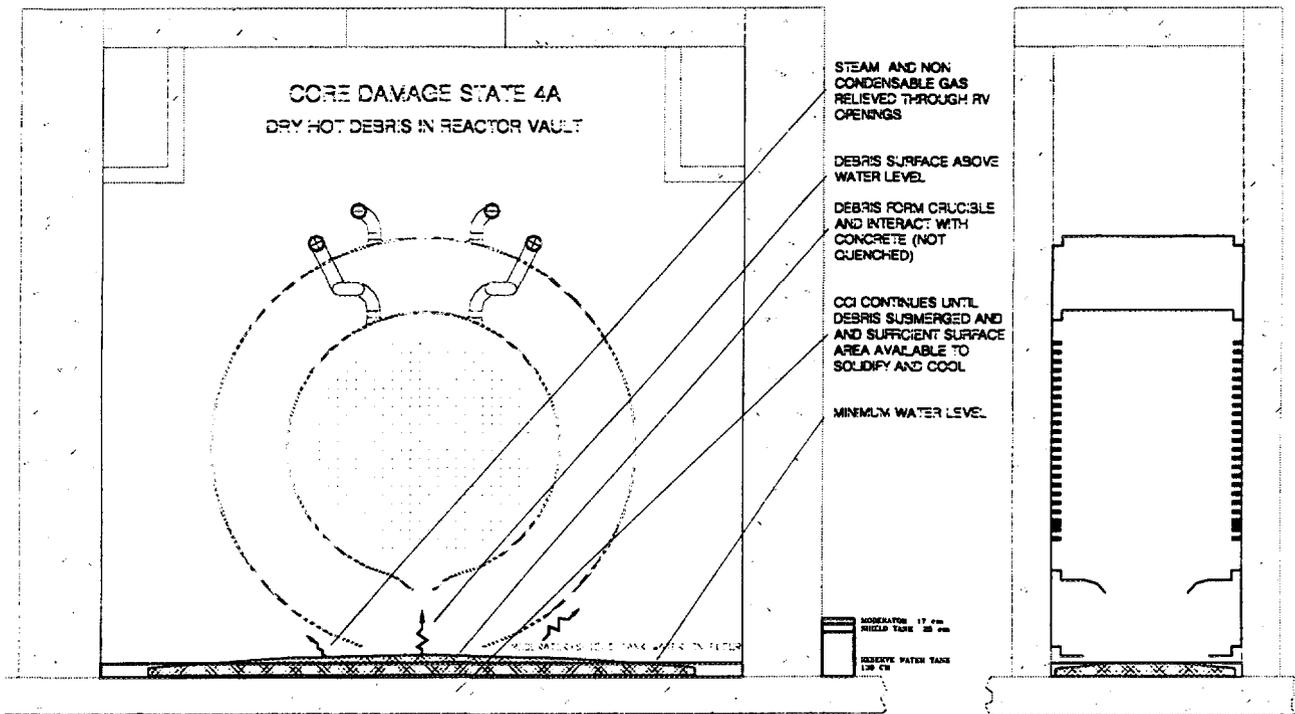


Figure 5: Debris in Reactor vault - CDS 4A

2.2 CONTAINMENT PHENOMENA

In order to assess the containment ability to mitigate severe accident events, the following potential phenomena are identified:

SOURCE TERM	ACCIDENT PHENOMENA / NOTES
Steady release of steam/water mixtures.	Due to fuel/debris surrounded by water; for each of the core damage states in section 2.1
Surge release of steam/water mixtures	Quenching of fuel/debris by water (operator actions or progression from one CDS to another).
Release and accumulation of non-condensable gases	Short term H ₂ from steam reactions with zirconium, long term by radiolysis, corrosion, interactions with concrete (also CO).
Hydrogen combustion	Slow accumulation of H ₂ ; igniters and recombiners limit the hydrogen concentration below the deflagration limit and permit only local burning.
Debris-water interaction within calandria and shield tank vessels	Energetic interactions precluded as debris not molten. H ₂ estimates are low (<< 10% of molten zirconium may react - Reference 3)
Debris-water interaction on Reactor Vault floor	Floor always covered with water if debris on floor
Fission product (FP) interactions following release from debris	FP carry significant (up to 40%) decay heat. Steady containment heating by fission products can be simulated analytically; most of the fission products will be in the water pool on the floor.
Boundary (e.g. seal) failure due to prolonged high temperature/radiation exposure	Loss of containment integrity.
Mechanical impingement by jets, flying debris, insulation.	These are phenomena common to design-basis severe accidents, and are addressed in the layout, by provision of barriers, and in the design of the ECC sumps.
Hot gases, fires	To be evaluated
Vacuum	Coolers, sprays can induce vacuum for certain scenarios, induce structural loads.

In the preliminary stages of the evaluations of the CANDU 9 severe accident mitigation capabilities, only the first two source terms are explicitly considered. The effects of some others are covered by the successful actions of the mitigating systems or dealt with in later analyses.

2.3 ACCEPTANCE CRITERIA

The following acceptance criteria are used in the preliminary assessments :

- 1) The maximum containment pressure is lower than the containment failure pressure for up to 24 hours after the onset of a severe accident. (This analysis uses $P_{max} \leq 450$ kPa (g), the pressure below which the steel liner stays intact; "true" containment failure pressure calculations pending.)
- 2) The hydrogen concentration remains below the limits for deflagration (a conservative value of 9.0% by volume is used in this analysis) in any given volume of the containment.
- 3) The maximum pressure/temperature/radiation field at containment seals, penetrations and doors are below the failure limits for the seals and the containment, whichever is lower.
- 4) The long-term heat removal capacity within the containment must exceed all heat sources such that conditions 1 and 3 are met.
- 5) The debris has adequate area to spread in the reactor vault (a lower limit of 0.02 m²/MWT debris spread area is targeted for some reactors -Reference 5) and any debris in the reactor vault are covered with water.

2.4 SEVERE ACCIDENT MITIGATING SYSTEMS

2.4.1 MODERATOR AS A HEAT SINK

In certain severe accident scenarios, the fuel channels are intact, hot and voided. This can occur, for example following a loss of primary coolant and a failure to initiate emergency core cooling. The moderator surrounds all the channels and removes the decay heat and the metal-water reaction heat from the hot channels (Figure 2). If moderator cooling is available or if any moderator inventory loss can be replenished in a timely manner, this core damage state can be maintained indefinitely. Various analyses (References 2, 3) have shown that the channels maintain their integrity as long as they remain submerged in the moderator. While the channel integrity is maintained, the fuel sheaths will fail and the bundles will slump in most locations, depending on the timing of the accident sequence. Even for the worst case (steam flow chosen to maximize the metal-water reaction in each channel), fuel remains below its melting temperature and only partial sheath melting is predicted.

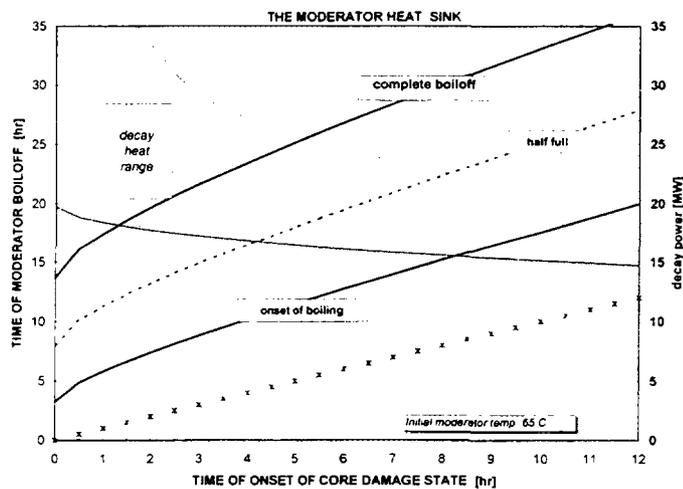


Figure 6: Moderator boiloff by submerged channels [Example: If the moderator heatup starts at 7 hours after trip, it begins to boiloff at about 13.5 hours and is completely boiled off by 28 hours after reactor trip; during this period the decay heat is between 23 and 16 MW]

If the moderator cooling is lost, the moderator begins to heat to saturation and then to boil. Pressure inside the calandria is relieved by rupture discs and/or by relief valves. After about three or four rows of channels are uncovered, the channels begin to fail by thermo-mechanical loads. The debris falls progressively into the moderator which eventually boils away. Conservative estimates of moderator boil-off time due to heat from the submerged channels is shown in Figure 6; the operator typically has many hours to replenish the moderator. This is a straightforward operation, consisting of opening the valves from the elevated Reserve Water Tank to the moderator, and refilling it by gravity. If the operator does not do this, estimates of moderator boiloff by debris collapse are shown in Figure 7; the debris collapse into the moderator can boil off a significant portion of the remaining moderator and induce a high steam surge load.

The calandria vessel over-pressure protection is provided by relief valves in the cover gas system and rupture disks at the end of four large pipes on top of the calandria vessel. The over-pressure protection system is designed to assure structural integrity of the calandria vessel (CV) against increase in pressure caused by in-core rupture of a channel, or loss of moderator cooling at full power. The rupture disk burst pressure is of course higher than the relief valve opening pressure. These relief systems also mitigate over-pressure in the calandria in severe core damage sequences, for example steam boil-off in a LOCA/LOECC/loss of moderator cooling triple failure.

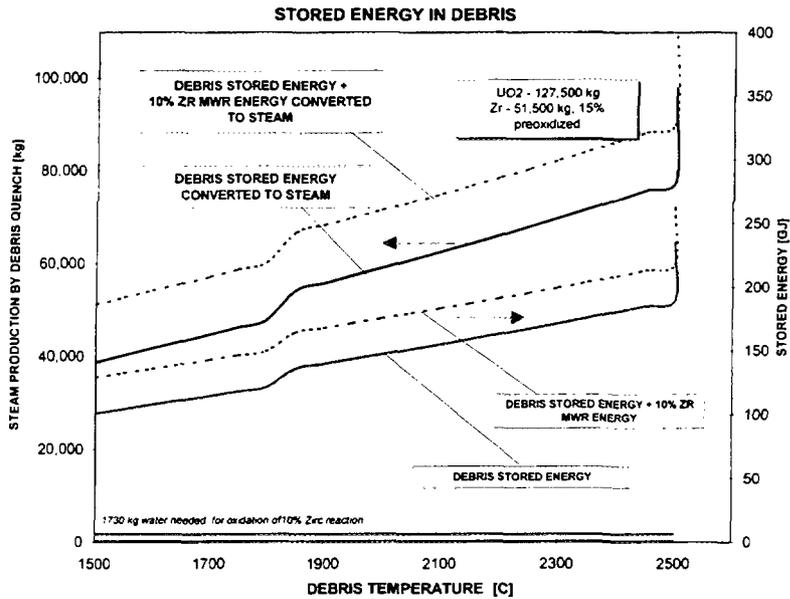


Figure 7: Moderator boiloff by debris [Example: Debris at an average temperature of 1900° C have a stored energy of about 140 GJ, enough to boiloff about 56 Mg of water by quenching]

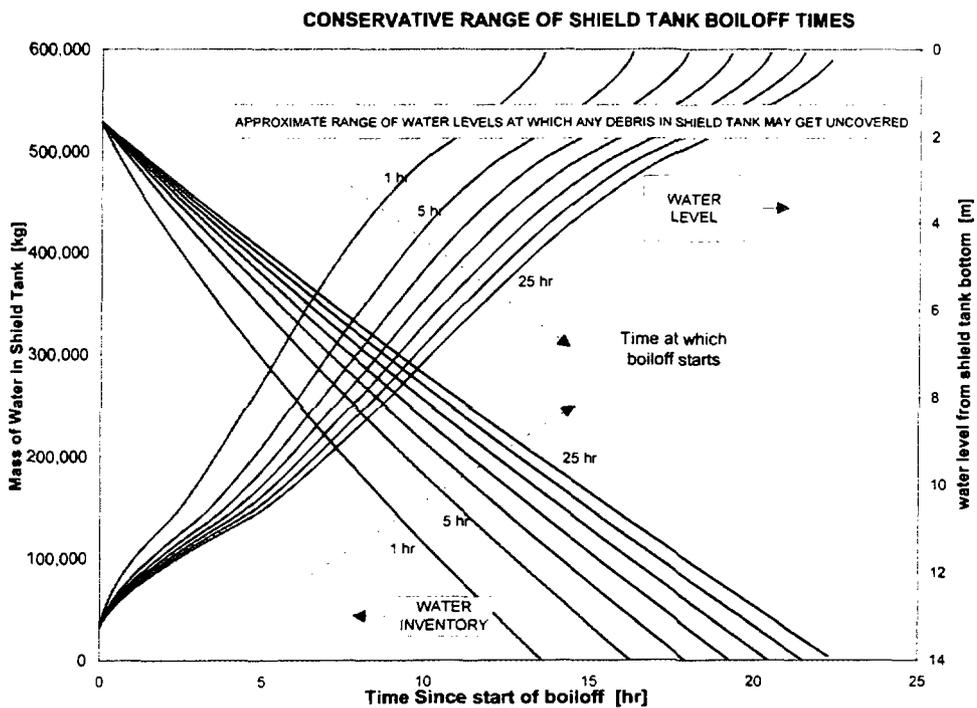


Figure 8: Shield tank boiloff times as a function of time at which boiloff starts (CDS 2A) [Example: If the Shield tank water begins to boil at 5 hours after reactor trip, its level drops to about 2m after another 13 hours and is depleted by just over 16 hours after onset of boiloff]

2.4.2 SHIELD TANK AS A HEAT SINK

In certain severe accident scenarios progressing to severe core damage, dry hot core debris may lie at the bottom of the calandria vessel (Figure 3) with decay heat removal by the shield tank water. This core damage state can be maintained as long as the shield tank water (~530 Mg.) surrounds the debris. Figure 8 shows the shield tank water boil-off estimates as a function of onset of the boil-off (which determines the decay power). Even without credit for the shield tank cooling system, which can remove about 0.3% of full power, the operator typically has more than 10 hours before the water level in the shield tank falls below the level of debris in the calandria vessel (~2m, see Figure 3). Replenishing the shield tank water is likewise a straightforward operation, consisting of opening the valves from the elevated Reserve Water Tank to the shield tank and refilling it by gravity. A shield tank over-pressure protection system prevents shield tank over-pressurization and allows decay heat to be released as steam to containment.

2.4.3 EXTERNAL FLOODING OF THE SHIELD TANK

Consider a more extreme case of Section 2.4.2 (Shield tank as a heat sink): In the unlikely scenario that the shield tank water is also lost (failure of operator to replenish the tank, or a break in the tank), debris may melt through the calandria vessel and end up in the shield tank. By this time the major liquid inventories (HTS, moderator and the shield tank water) are mostly on the reactor building floor (even in absence of coolers the majority of water is predicted to rain out) and along with potential contributions from ECCS and the Reserve Water Tank, flood the outside of the shield tank (see Figure 4) - there is no basement beneath the reactor as on operating CANDU plants. Thus, the CANDU 9 containment layout permits one more level of defence against vessel melt through by debris. The reactor centre line is at an elevation of 7.3 m from the basemat floor. With an external shield tank diameter of 13.3 m, the distance from the floor to the bottom of the shield tank is only 65 cm. The water level depends on the accident sequence, but with the ECC and RWT water inventories, can be as high as 2.5 m. The operator can also manually dump the RWT inventory on the floor to facilitate shield tank flooding.

2.4.4 RESERVE WATER TANK FOR SEVERE ACCIDENT MITIGATION

The Reserve Water System is a CANDU 9 innovation with significant accident mitigation capabilities. It is a passive, backup gravity-fed light water supply system that requires no pumps to deliver its inventory to critical locations. It consists of the Reserve Water Tank, located at a high elevation in the reactor building, and piping connections, with remotely actuated isolation valves, to the shield tank, HTS, calandria, Steam Generators and ECCS. The total capacity of the RWT is about 2500 m³ and it can be replenished from the reactor building sump by two 100% recovery pumps. Injection from the Reserve Water Tank is initiated by the operator.

2.5 CONTAINMENT DESIGN FEATURES FOR SEVERE ACCIDENT MITIGATION

2.5.1 CONTAINMENT LAYOUT

The general containment building schematics and equipment layout is shown in Figure 9. The CANDU 9 reactor building is a large dry containment, made of pre-stressed concrete with a full internal steel liner. It has a flat circular cylindrical base slab, it is 57m in internal diameter and it has a 42 m high circular perimeter wall, topped by a hemispherical dome for a total ceiling height of about 72m. The thickness of the exterior perimeter walls is 1.5m, with the dome wall thickness varying from 1.5 to 1.0 m. The internal structures are supported on the base slab and impose no load on the perimeter walls or the dome. The building is designed to maximize human access during operation to test, repair or replace components while minimizing radiation exposure. Areas containing potential heavy water leakage sources (reactor vault, fuelling machine vaults, steam generator enclosures, moderator pump and heat exchanger rooms and the shutdown bleed cooler area) are inaccessible while the reactor is on power and have a separate, controlled atmosphere. Blowout panels connect these sub-volumes in an accident to prevent local over-pressure and to ensure hydrogen mixing.

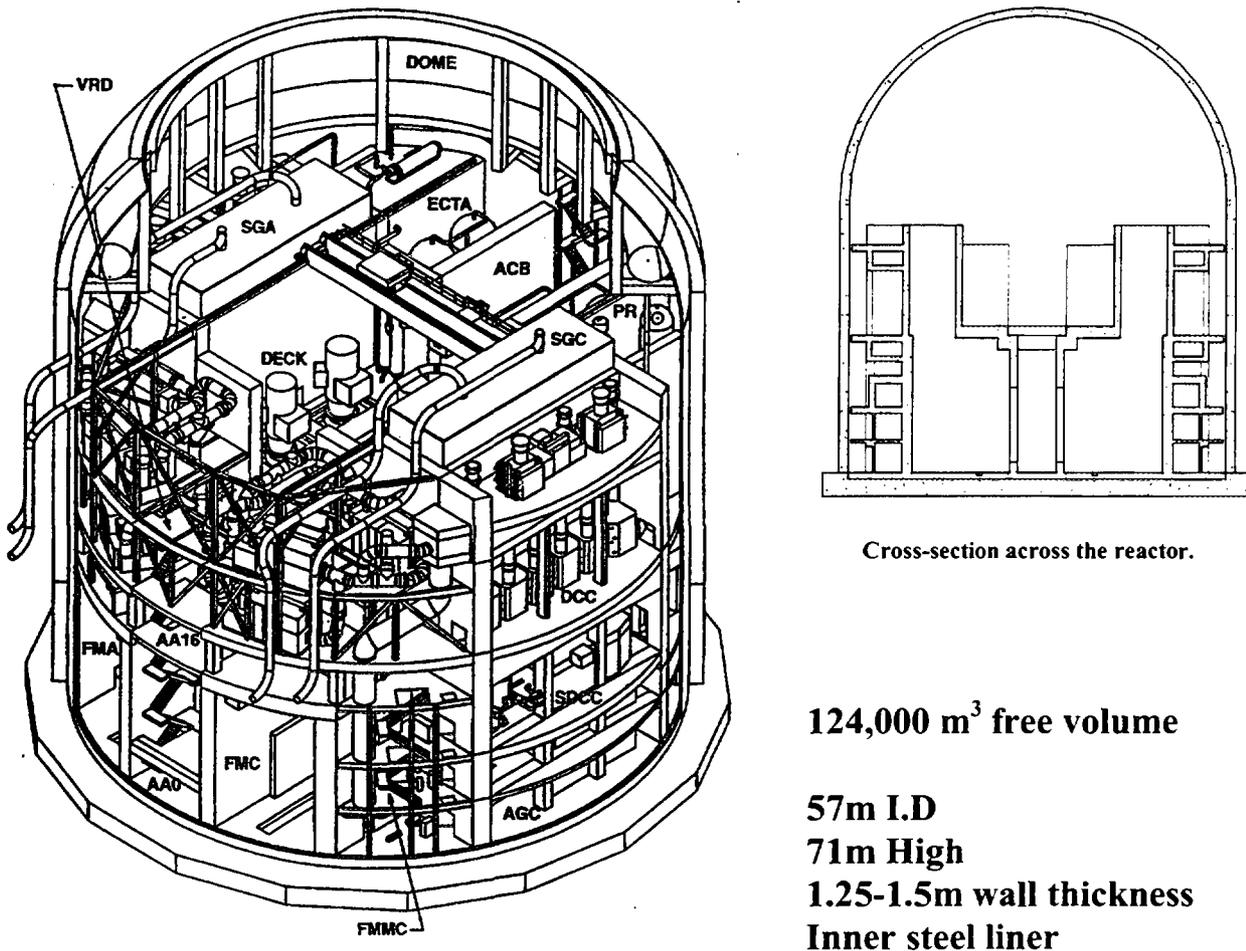


Figure 9: A schematics of the containment layout

The containment net free volume is one of the largest in the world, estimated to be about 124,000m³ with an estimated equipment volume of about 11000m³. The large net free volume limits the rate of rise in internal pressure and global hydrogen concentrations in accidents.

Ground-level openings between various structures and rooms allow for the unimpeded spreading of any water spilled in the reactor building. The openings between the reactor vault and the fuelling machine vaults extend to above the flood level. Special high level openings have been designed in the reactor vault walls near the feeder cabinets to relieve steam and gases released in the vault and to promote natural circulation in the absence of forced circulation by air coolers.

2.5.2 CONTAINMENT STRUCTURES AS HEAT SINKS

The large internal metal and concrete surfaces (over 50,000 m²) provide significant heat sinks following release of steam into the containment. These estimates do not include floor and ceiling surface areas. They also help remove fission product aerosols by condensation, an effect not yet credited.

2.5.3 REACTOR VAULT DESIGN FOR SEVERE ACCIDENT MITIGATION

As noted previously, the reactor vault layout allows progressive containment of debris from a severe core damage accident in CANDU 9: first the calandria vessel, then the shield tank. In the most unlikely event that the debris melts through the shield tank and pours onto the floor, it will always fall into a pool of water (Figure 5) . The reactor vault has ample wall openings ($> 30 \text{ m}^2$) to preclude pressurization of the vault. The large floor area of the CANDU 9 reactor vault ($\sim 116 \text{ m}^2$) is conducive to debris melt spreading: the floor area corresponds to about $0.041 \text{ m}^2/\text{MW}$ of initial core thermal power. Guidelines for advanced reactor designs suggest a design target debris spread area of $0.02 \text{ m}^2/\text{MW}$ rated thermal power (Reference 5) (higher values are better).

Severe accidents develop because the process and safety heat sinks become unavailable. In most cases that means that the water inventories that can potentially remove heat are either unavailable or are discharged into the containment by breaks or boiloff. Calculations for containment flood level estimates show that the basemat level surface area is about 1803 m^2 . Table 1 lists the water inventories of major reactor systems. Also shown are the individual contributions of the various sources of water on basemat flood levels. If all the water from the HTS, calandria, shield tank, ECC tanks and the Reserve Water tank should end up on the floor, the flood level may reach about 2.5 m, enough to cover the lower portion of the shield tank. This estimate does not include potential fluid loss from non-seismically qualified systems (estimated to contain about 300 m^3 of water) and any losses from the feedwater system (estimated to contain about 2000 m^3 of water). While these two sources can add another 1.23m to the flood level, it is noted that actual flood levels are scenario specific and a simple addition of contribution from all service water, process and safety systems cannot be made. Further scenario specific calculations are pending.

Table 1: Fluid inventories

SOURCE	MATERIAL	Volume [m^3]	LEVEL [m]
HTS	D ₂ O	363	0.22
SG SEC. SIDE	H ₂ O	327	<i>not credited</i>
MODERATOR	D ₂ O	307	0.17
END SHIELD	H ₂ O	20.6	0.01
SHIELD TANK	H ₂ O	529	0.29
ECCS	H ₂ O	680	0.38
RESERVE WATER TANK	H ₂ O	2500	1.39

2.5.4 CONCRETE COMPOSITION TO MINIMIZE CORE CONCRETE INTERACTIONS

Core-concrete interactions are sensitive to accident specific details such as corium composition and attack characteristics and concrete properties, etc. However they are precluded by the various barriers described above. Nevertheless, the composition of concrete in the reactor vault (floor and lower sections of walls) is being optimized to minimize non-condensable gas production by interaction with solid and molten corium.

2.5.5 COOLERS FOR LONG TERM PRESSURE SUPPRESSION

Containment coolers and the ventilation system provide air cooling, exchange and distribution and maintain the containment pressure slightly sub-atmospheric under normal operation. The heat removal from the inaccessible areas is by two banks of ducted containment air coolers, each equipped with 4 air coolers on each side of the reactor and designed to maintain a temperature lower than the maximum permissible for equipment and concrete. These ducted air coolers draw air from the top of the steam generator enclosures (Figure 10), isolated from the dome area by blow-out panels, and discharge cool air at the bottom of the fuelling machine vault and reactor vault at two different

elevations, thus ensuring good mixing. A common, separate environment is thereby maintained in the reactor vault, fuelling machine vaults, feeder cabinet areas and steam generator enclosures. The four unducted air coolers, under the dome, cool the air in the accessible areas.

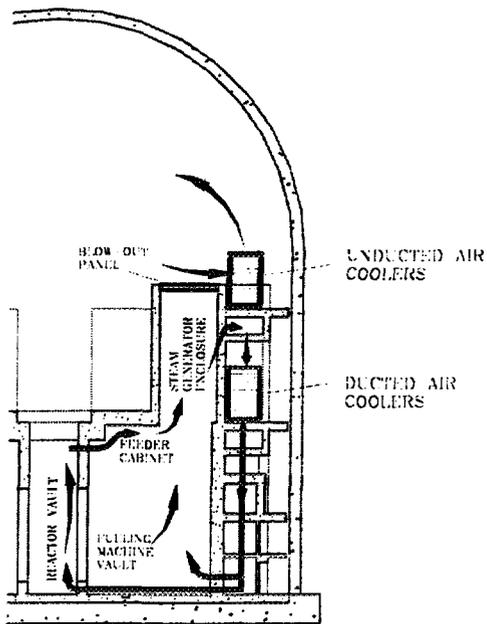


Figure 10: Air flow patterns in one half of the reactor under normal operating conditions

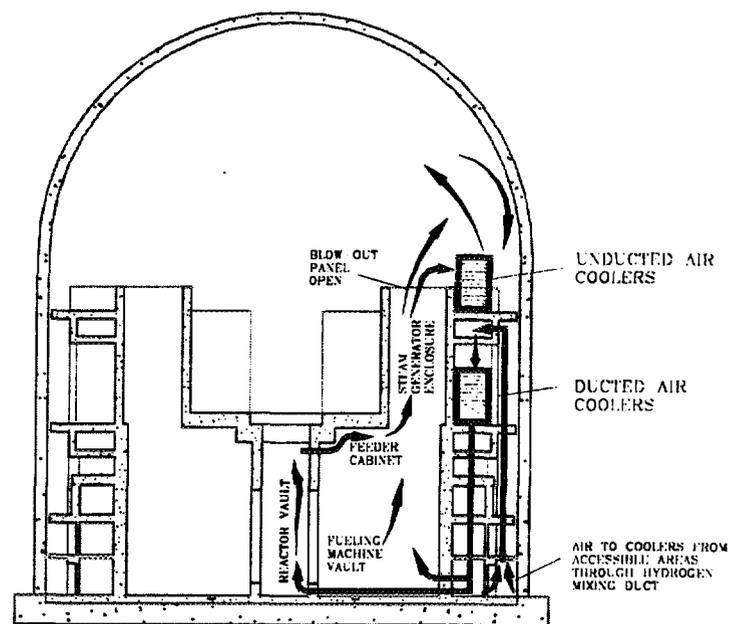


Figure 11: Containment air flow patterns after an accident [Flow patterns in the other half are identical]

In event of a loss of coolant accident, a high pressure or high activity signal on either of two fully independent containment isolation systems isolates the containment by the four valves in series in each of the ventilation duct penetrations in the reactor building wall. The blow-out panels and dampers between the steam generator enclosures and the accessible areas open and all coolers become available for containment cooling (Figure 11). In order to enhance mixing in the air environment, coolers begin to draw relatively cool and clean air from the accessible areas through the hydrogen mixing ducts located in opposite sides of the reactor building. The air coolers are important in evaluation of severe accident consequences. Enhanced heat removal capabilities of the air coolers under conditions of high humidity are well documented. Analytical models have been developed to compute heat removal capabilities under a wide range of steam concentrations and ambient temperatures. Analyses are underway to establish long term cooler survival and functionality under accident conditions.

2.6 HYDROGEN MITIGATION SYSTEMS

During the course of a severe accident, hydrogen may be produced, in the short-term, within the fuel channels and by debris in the calandria vessel, shield tank or the reactor vault. Longer term sources of hydrogen include radiolysis, corrosion and core-concrete interactions. A large scale release of energy associated with hydrogen (and carbon monoxide) deflagration can pose potential threats to containment and equipment integrity. Efforts are underway to identify all short and long term sources of hydrogen and to mitigate them.

During severe accidents in CANDU reactors, concentrations of hydrogen build up slowly and reach combustible values over many hours. Therefore, the potential for hydrogen deflagration (at 9-10% volumetric concentration of H₂ at low steam concentrations) can occur only if spatial concentrations are allowed to build up. The containment layout precludes pockets or regions where hydrogen can accumulate. Hydrogen distribution in containment, in the absence of forced circulation, is governed

by diffusion, condensation and natural circulation processes. The containment layout is conducive to natural circulation of gas mixtures in absence of forced circulation. Since these processes are not easily quantifiable, hydrogen mitigation systems that allow recombination or early ignition and burning are provided in various locations within the containment. Specifically the containment is equipped with both igniters and recombiners, to ensure that the hydrogen concentration remains below the critical value (9-10%) for deflagration. The design has been done before the confirmatory three-dimensional transient calculations, a technology which is only now available; the modelling of the hydrogen mitigation systems and estimates of hydrogen source terms is underway.

The igniters are made available from the onset of the accident to instigate local burns as soon as the local hydrogen concentration exceeds the ignition threshold (4-6 volumetric %, depending on the steam concentration). Igniters are not a panacea; their drawbacks include: a) their inability to operate in steam inerted environments, i.e., at steam volumetric concentrations > 55%, so that they may operate in some cases only after the steam has condensed and thus potentially initiate deflagration at high H₂ concentrations; and b) the potential to initiate deliberate ignition in a room with unknown, high hydrogen concentration.

The catalytic recombiners work over a wide range of hydrogen concentrations (from ~2%) and are unaffected by steam concentrations. They are a long-term hydrogen mitigation system and their H₂ removal capacity is of the order of tens of kg/hr per unit (typically 3-4 m³ in size). Required and available recombination rates are generally small, and it would take hours to days to effect a measurable change in the containment hydrogen concentration. The placement of recombiners will be reviewed, once a detailed hydrogen distribution analyses for a range of severe accident scenarios is performed.

2.7 SEVERE ACCIDENT SPECIFIC INSTRUMENTATION

While some of the normal plant operation instrumentation can help ascertain the accident progression and reactor state, additional dedicated instrumentation is provided (12 qualified temperature, pressure, humidity and radiation monitors) to help identify the reactor state under accident conditions. Generic requirements for special instrumentation, dedicated to the monitoring of severe accident progression, and capable of surviving the anticipated harsh environment and operating in the range of anticipated extreme conditions are being developed as a part of this assessment. Further evaluation of the adequacy of the current post-accident monitoring instrumentation is planned.

3. CONTAINMENT RESPONSE TO STYLIZED SEVERE ACCIDENT CHALLENGES

In lieu of detailed analyses of core disassembly process for specific sequence of events, the containment response to a series of stylized loads is examined. Some sample results are presented here. In an initial simple simulation, the containment is subjected to a constant steaming load, representing any of the following core damage states:

- Hot dry intact channels (CDS-1A) submerged in boiling moderator
- Hot dry debris in calandria vessel (CDS-2A) with boiling shield tank water
- Hot dry debris in shield tank (CDS-3A) with boiling outside the submerged shield tank
- Submerged debris (CDS-4A) on reactor vault floor

In all these cases, it is conservatively assumed that all decay heat goes into boiling and that the initial pressure in the containment is atmospheric. The effect of pressure spikes due to the initial break and later quenching of fuel or debris is considered separately. The containment dome pressure and temperature transients are shown in Figure 12 and Figure 13 for a constant steam injection rate of 12.6 kg/s, corresponding to a constant decay power of 1%, typical during the long time-scales for CANDU. With 6 ducted air coolers operating, the containment over-pressurization is limited to less

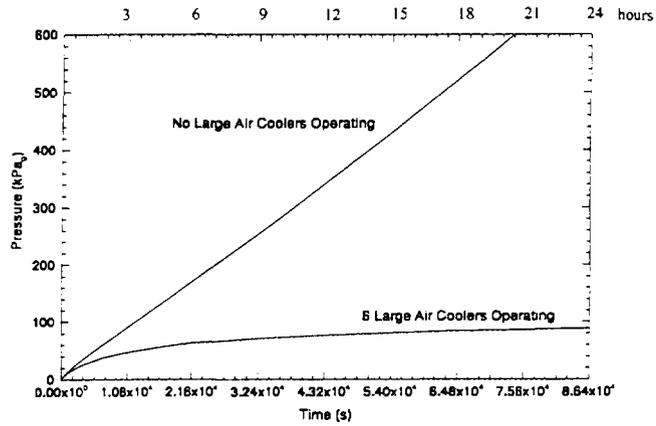


Figure 12: Containment dome pressure transient for constant steam load at 1% decay power.

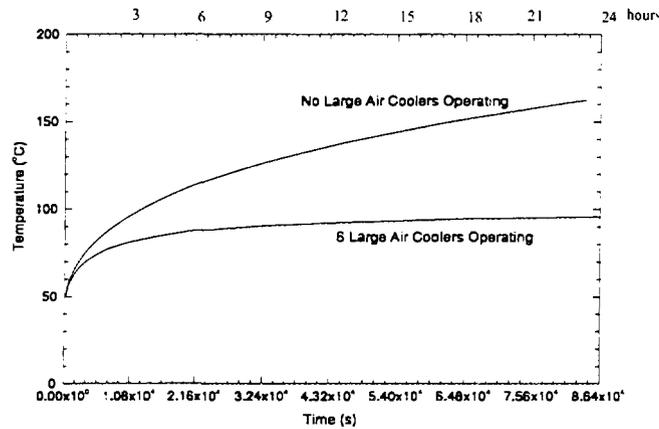


Figure 13: Containment dome temperature transient for constant steam load at 1% decay power.

than 100 kPa(g) and the temperature does not exceed 100° C. Without coolers, the containment pressurizes to the reference pressure of 450 kPa(g) after about 15 hours, at which time the gas temperature (Figure 13) reaches over 140° C. It is obvious that the coolers are important in the long term.

In another stylized scenario (Figure 14), a source of steam, consistent with decay heat production, is introduced into the containment at 15 minutes after reactor trip to simulate steam production from: Hot dry intact channels (CDS-1A) submerged in boiling moderator; or Hot dry debris in calandria vessel (CDS-2A) with boiling shield tank water. A subsequent 1800° C debris quench at 23 hours simulates:

- Reflood of hot dry fuel in intact channels (CDS-1B) or core collapse into the moderator at 23 hours, or
- Reflood of hot dry debris in the calandria vessel (CDS-2B) or debris melt through into the moderator at 23 hours, or
- Reflood of hot dry debris in the shield tank (CDS-3B) or debris melt through onto the reactor vault floor at 23 hours, or
- Debris dropping onto the reactor vault floor after melt-through of the shield tank (onset of CDS-4A) at 23 hours

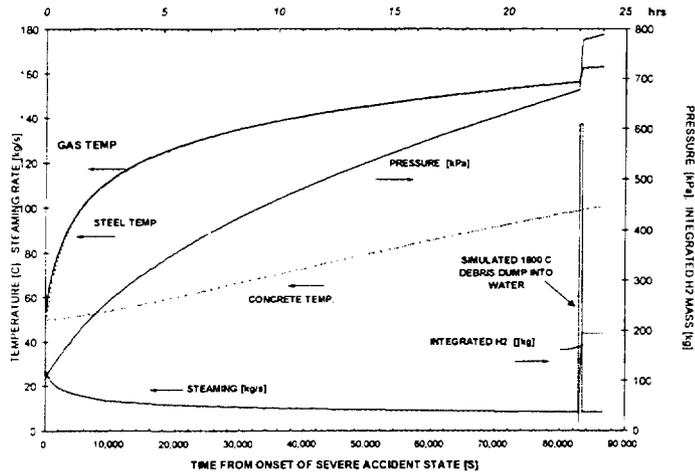


Figure 14: Containment transients for early steaming and late debris quench.

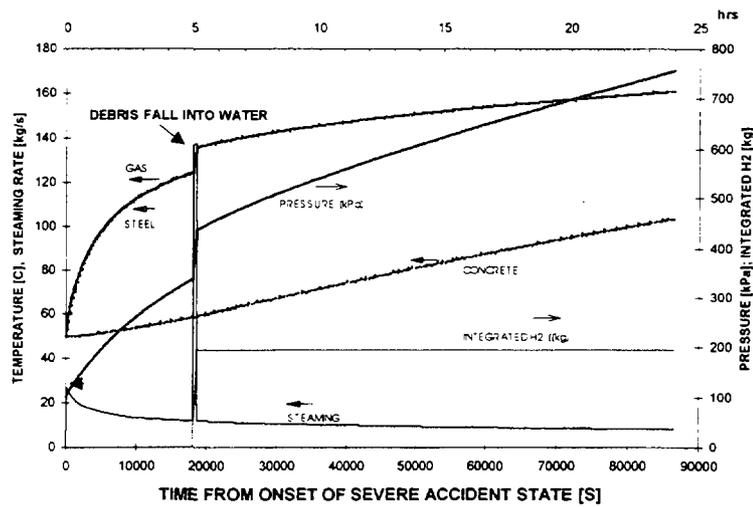


Figure 15: Containment transients for early steaming and early debris quench

A 10% oxidation of the un-oxidised Zircaloy is assumed to accompany the debris quenching process (to simulate the hydrogen source term). The containment coolers are assumed to be out of service. The containment response plotted in Figure 15 assumes debris quenching at 5 hours, instead of 23. Containment over-pressurization (pressure > 450 kPa (g)) occurs at about 10 hours instead of 15 due to early pressurization of containment by debris quenching at 5 hours. The debris quench contributes about 100 kPa to the containment pressure in both cases- i.e., the timing of debris quench has little effect on containment pressurization.

An operator action to reflood the debris at 5 hours, followed by restoration of cooling (termination of steaming from debris) is modelled. The containment response is plotted in Figure 16. The debris quench again contributes about 100 kPa to the containment pressure. The containment starts to depressurize as steam injection into it is terminated by operator action and the containment structures become the dominant long-term heat sinks. Containment response following a similar stylized severe accident following an early LOCA is presented in Figure 17. In this case, a steam surge by debris quench at 20 hours is simulated. In all cases the containment coolers are assumed to have failed at the onset of the accident. With coolers operating, containment over-pressurization is avoided with anticipated response similar to that in Figure 12.

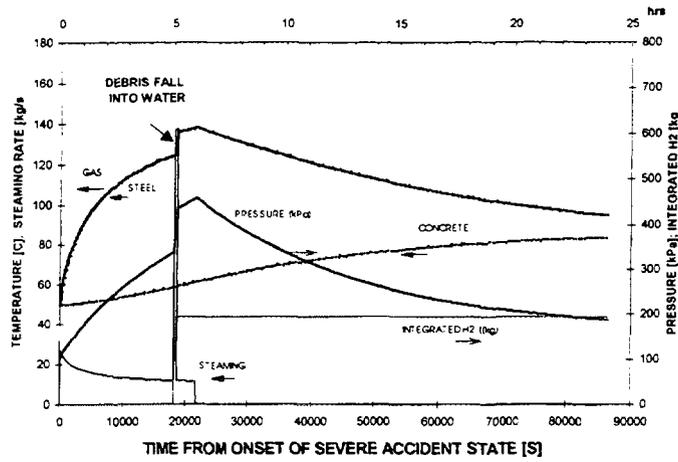


Figure 16: Containment transients for early steaming, debris quench and accident termination

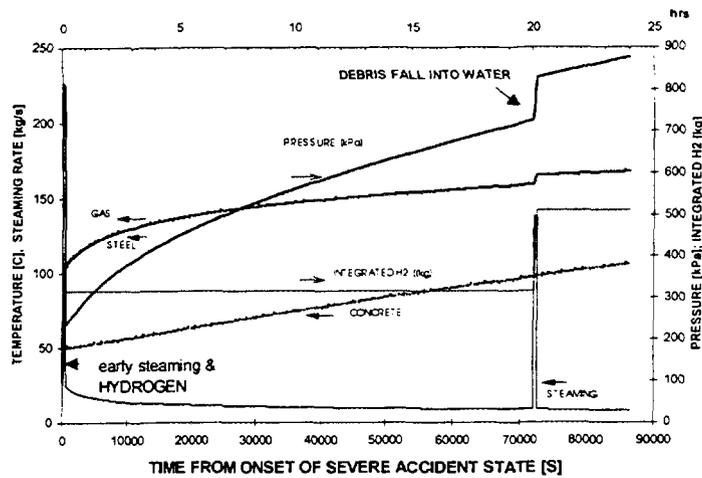


Figure 17: Initial LOCA followed by core disassembly.

The containment response to the stylized containment loads (typical of many severe accident sequences) presented above illustrates:

1. The coolers are effective in limiting containment pressurization,
2. If they fail, ample time is available to restore operation of the coolers, or initiate an alternative means of providing cooling or pressure suppression (now under review).
3. Overall containment pressurization is insensitive to the exact timing of events for a given class of severe accidents
4. Containment pressurization rate is relatively slow.
5. Structural heat sinks offer significant heat removal capability.

4. CONCLUSIONS

The CANDU moderator and shield tank water volumes provide unique severe accident mitigation capabilities. The reserve water tank in CANDU 9 affords additional time to arrest severe accident progression. Preliminary results confirm that containment air coolers are effective in avoiding containment failures for the whole range of accident progression pathways. Other features of the CANDU 9 containment include:

- The large CANDU-9 containment and the equipment layout results in large, open volumes with good potential for natural circulation and no apparent hydrogen traps.
- The pre-stressed concrete boundary with a steel liner results in high failure pressure.
- The large structural heat sinks significantly augment heat, humidity and fission product aerosol removal from the containment atmosphere by the air coolers.
- Reactor building flooding levels permit external cooling of debris in the shield tank and provide an extra boundary to arrest severe accident progression.
- Hydrogen mitigation systems allow systematic and timely dispersion and reduction of hydrogen.
- The reactor vault concrete floor composition and geometry minimize core-concrete interactions in the most unlikely event of debris arriving at the reactor building basement.
- Instrumentation is provided for measurements and control under severe accident conditions

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