CONSEQUENCE ANALYSIS OF CORE DAMAGE STATES FOLLOWING SEVERE ACCIDENTS FOR THE CANDU REACTOR DESIGN

N.N. Wahba, Y.T. Kim[†], S.G. Lie and K.S. Dinnie

Ontario Hydro, Reactor Safety and Operational Analysis Department
700 University Avenue, Toronto, Ontario, M5G 1X6

[†]Atomic Energy of Canada Limited,
2251 Speakman Drive, Mississauga, Ontario, L5K 1B2

ABSTRACT

The analytical methodology used to evaluate severe accident sequences is described. The relevant thermal-mechanical phenomena and the mathematical approach used in calculating the timing of the accident progression and source term estimate are summarized. The postulated severe accidents analyzed, in general, mainly differ in the timing to reach and progress through each defined "core damage state". This paper presents the methodology and results of the timing and steam discharge calculations as well as source term estimate out of containment for accident sequences classified as potentially leading to core disassembly following a small break loss-of-coolant accident (LOCA) scenario as a specific example.

1. INTRODUCTION

The determination of postulated severe accident progression and modelling of physical processes which could result in challenges to containment, forms an integral part of the risk assessment of CANDU reactors. The timing and discharge calculations of a particular accident progression are required as key parameters defining the consequence portion of the overall risk assessment analysis. Ultimately, a source term estimate out of containment can be obtained for a broad range of accident scenarios using the timing of the event together with major driving forces such as steam generation as the basis of radioactive releases. These source term estimates, expressed in terms of both magnitude and timing of release to the environment, are used as the basis for grouping specific scenarios with similar release characteristics into general categories. Detailed frequency and consequence estimates for these categories provide the framework from which the overall risk estimates for the station can be determined.

The first challenge in attempting to determine the timing of an accident scenario is to define the system processes and availabilities that can lead an accident event into a severe accident scenario. Typically, each sequence of events is the result of an initial malfunction (or initiating event), followed by failures of other functions or systems designed to mitigate its effects. Table 1 shows the process and system failures that can potentially contribute to core disassembly. The initiating sequence is a combination of an accident initiating event with one or more process and system failure(s). In general, the following initiating events are considered: large break LOCA (LLOCA), small break LOCA (SLOCA), reactivity initiated accidents (RIA) such as LLOCA with loss of shutdown systems (LSDS) and slow loss of reactivity regulation (SLOR), containment bypass events (includes pump seal breaks, feedwater line breaks, steam generator tube ruptures and emergency coolant injection (ECI) blowback) and a station blackout event. Note that for any accident scenario to be classified as a "severe" accident implies that accident progression leads to core disassembly. Large LOCA, for example, is considered an inconsequential event in terms of severe accidents, if ECI and emergency coolant recovery (ECR) systems are available. In the analysis of severe accidents, an implicit assumption is that operator action is not credited after fuel damage occurs, even though there may be sufficient time for intervention in order to halt further accident progression.

The accident progression described in this paper is typical of postulated CANDU reactor severe accidents for Bruce and Darlington NGS. Once the fuel in the channels experiences severely degraded cooling, it heats up quickly. At some point the pressure tubes come into contact with calandria tubes, either due to pressure tube sagging or ballooning. Consequently, the bulk of the fuel decay heat in the hot channels is then transferred to the subcooled moderator. However, since moderator cooling is postulated unavailable in severe accident sequences, the moderator inventory will heat up rapidly and eventually reaches saturation conditions. When the pressure within the calandria vessel (CV) exceeds the setpoint of the CV rupture discs, the rupture discs perforate and the moderator inventory flashes with a portion discharging into containment via the calandria discharge pipes. The top rows of channels become uncovered and are only cooled by the steam produced through the evaporation of the remaining moderator inventory. This is not sufficient to prevent the channels from undergoing failure and disassembly. The moderator inventory is eventually boiled off, and debris composed of fuel, channel and other core materials (collectively called corium) starts to accumulate at the bottom of the CV. At this point the CV is still surrounded by the water contained in the shield tank (ST). However, the ST is not designed to withstand this magnitude of heat influx. When its water inventory becomes saturated at some pressure, the ST is assumed to burst along the seam at the bottom of the tank, causing the water inventory in the ST to drain rapidly to the fuelling machine duct (FMD). The corium at the bottom of the CV eventually melts through the CV and ST and falls into the FMD, where it is quenched. The corium will reheat and vaporize the remaining water on the floor of the FMD. This vapourization process can go on for a long period since the steam produced re-condenses in the containment and drains back to the FMD floor.

Once the accident scenario is specified, the accident progression needs to be defined. The accident progression is divided into four core damage states (CDSs). The defined states allow for analysis of a broad range of events and provide a basis for comparison in terms of timing and steam generation. Each state in the severe accident progression is defined based on conservative criteria. Precise descriptions of the timing of each state are not crucial since variations in timing become less significant by the broad representation of many accident sequences with a single, stylized source term. The following describes the condition of each CDS:

- CDS1 fuel heats up within the fuel channels due to loss of primary cooling
- CDS2 hot fuel channels disassemble and release their contents into the calandria vessel
- CDS3 bottom of calandria vessel and shield tank fail due to the load of hot core debris and the debris is released onto the fuelling machine duct floor
- CDS4 debris is quenched and cooled by evaporation of the accumulated water on the fuelling machine duct floor

After the core damage states are defined, it is found that the accident progression of all accident scenarios differ only in the "blowdown" phase of the transient. Once the blowdown period is calculated, the methodology for determining subsequent CDSs is the same. Once the timing and tracking of water mass inventory of the blowdown are determined, the timing and steam discharges for all CDSs are determined by using a computer program. The details of these calculations are discussed in Reference 1.

The estimation of the source term from various postulated accident scenarios represents the event consequence in the overall risk assessment of severe accidents. With the timing calculation providing the timeframe for any particular accident progression, the source term estimation considers the major factors that could potentially lead to releases to the environment. The categorization of the severity of any modelled accident can then be accomplished by considering the amount and timing of radioactive releases. Potential release of radioactive substances into the environment is expressed as a fraction of the total core inventory divided into four fission product groups; namely, noble gases, volatile, semi-volatile and non-volatile aerosols. The grouping represents substances with similar volatility and chemical affinity.

The estimation of radioactive releases into the environment is determined by a combination of basic fluid dynamics and expert opinion (mainly in the form of estimation of fission product releases from the core for each CDS), as well as cross-comparison with the MAAP-CANDU code. Analytical calculations are used to obtain transient fission product releases by considering relevant driving forces (for example steam generation, pressure gradients, global hydrogen gas burns and core-concrete interactions) in combination with the status of containment systems such as containment impairment, emergency filtered air discharge system, hydrogen ignitor systems and other impairments, which are taken together to determine the amount of fission products released out to the environment. The core inventory is tracked from the reactor core to containment and out to the environment, as applicable.

In parallel to the analytical methodology described in this document, the MAAP-CANDU code (Reference 2) may be used to assess the progression of severe accidents. MAAP has the capability of simulating the accident progression in a much more detailed and comprehensive manner. One of the main reasons to use the analytical approach is to gain a better general understanding of, and insights into these complex severe accident phenomena. To achieve these goals, it is necessary to substitute some of the complex calculations by simpler, more conservative ones, to keep the analytical approach easy to understand and perform.

This paper presents the methodology and results of the timing and steam discharge calculations as well as source term estimate out of containment for accident sequences classified as potentially leading to core disassembly following small break LOCA.

2. ANALYTICAL MODELS

The timing calculations of any accident scenario are performed as a two-step process. The first step examines the details of the particular event with emphasis on the important phenomena associated with the event such as the availability of process and safety systems. In this first step, determination of the blowdown timing and tracking of water mass inventory are key considerations. Once these parameters are determined, the next step is achieved by using a computer program written to determine the timing and steam discharges for all CDSs [1].

2.1 Blowdown State

This initial phase of an accident usually involves blowdown and two-phase discharge that eventually depletes the coolant in the heat transport system (HTS). The blowdown time used here is the time period starting from the initiation of the accident until the HTS is empty and the first core damage state starts. The coolant blowdown time depends on the accident initiating event (such as small or large break LOCA, etc.), and the failure of process and safety systems (such as emergency coolant injection system, emergency coolant recovery systems or steam generator cooldown). Therefore, the blowdown time is determined for each sequence separately.

It is assumed that reactor trip occurs shortly after the break. It is also assumed that the HTS coolant occupies a fixed volume, V, during the blowdown duration, i.e.:

$$V = \frac{M}{\rho}, \qquad 1$$

where M and ρ are the instantaneous mass and density of HTS water inventory. The mass discharge rate, W, to the atmosphere is proportional to the square root of the density and pressure difference, ΔP, between HTS and atmosphere.

$$W \alpha \sqrt{\rho \Delta P}, \qquad 2$$

The duration of each stage of the blowdown is estimated assuming that the discharged mass flow rate varies linearly with time. Once the heat transport system (HTS) pressure is dropped to a specific value, pumped ECI is assumed to be initiated. This will maintain core cooling until the ECI tank is depleted. In severe accident sequences, it is assumed that the moderator cooling is unavailable, and ECR fails consequently.

Following the reactor trip, the two-phase discharge continues and HTS pressure drops rapidly. If the steam generator (SG) cooldown is assumed available, the HTS pressure drops to the steam generator pressure prior to the initiation of the controlled cooldown. During steam generator cooldown, the HTS pressure is assumed to follow the SG pressure transient and the discharge from the break is estimated as an average two-phase discharge between the starting and final pressures of the cooldown phase. The duration of SG cooldown is determined by its temperature reduction rate. Without steam generator cooldown, the HTS pressure is conservatively assumed to drop to the steam generator pressure and maintained constant.

The two-phase discharge from the break continues until the phase separation occurs in the HTS. Subsequently, only steam will be discharged from the break. The HTS pumps are assumed running. If SG controlled cooldown is credited, the SG acts as an effective heat sink, which is very capable of removing the decay heat generated in the core. In this case, the steam discharge rate at the end of phase separation is governed by the degraded pump head in a steam environment. If the SG cooldown is assumed unavailable, the steam discharge at the end of phase separation is governed by one of two mechanisms: either (a) the degraded pump head in a steam environment; or (b) the steaming rate required to reject the decay heat. Both mechanisms are considered, and the discharge rate is conservatively taken as the greater of the two rates.

2.2 Core Damage States

The timing calculation is essentially a transient solution of the conservation of mass and energy equations. The progression of the CDSs is governed by the heat balance between the heat sources and the heat dissipation to the available heat sinks. The fission product decay in the fuel after reactor shutdown represents the main heat source. The heat sinks include the residual moderator inventory, the water in the ST, the corium, the CV and the ST walls and the heat radiation from their surfaces to the surrounding environment. This analysis considers many of the key system components in terms of a "lumped-parameter" model. For example, the core is modelled as a lumped mass of UO_2, Zircaloy and steel (representing all other materials). Although many conservatisms are taken into account in the estimation of CDS timing, some minor factors are not accounted for in this analysis. The heat produced by the exothermic Zircaloy-steam reaction is relatively small (a few percent of the decay heat) and is neglected in the current methodology. This is more than compensated by not modelling all the known heat sinks, in order to focus the analytical computation on the main contributors. In particular, the transient thermal behaviour of the shutoff and control devices and the steel balls in the end-shield and the ST are ignored.

The balance of heat at any instant of time in the CDSs can be described by the following rate equation:

$$m_i c_{p_i} \frac{d}{dt}(T_i) = Q_i - \sum h_{ij} A_{ij}(T_i - T_j), \qquad 3$$

where t is time, m_i is the mass of the i[th] material, c_{pi} is the specific heat of the i[th] material which is a function of temperature T_i, and Q_i is the power source. The h_{ij} and A_{ij} are the heat transfer coefficient and the interface area between the materials i and j, respectively. The materials considered are:

1. fuel (early in the accident) or corium (later as debris);
2. moderator as a single phase liquid;
3. moderator as a homogeneous two phase mixture;
4. moderator as a separated two phase fluid (liquid in the bottom and steam in the top of the CV);
5. moderator as a single phase steam;
6. calandria vessel wall;
7. shield tank inventory as single phase liquid;
8. shield tank inventory as single phase gas;
9. shield tank wall; and
10. gas in containment.

For numerical computation purposes, Equation 3 is discretized for small time steps Δt_k with the incremental changes in temperature ΔT_{ik}.

$$m_{i,k}\, c_{p_{i,k}} \frac{\Delta T_{ik}}{\Delta t_k} = Q_{i,k} - \sum h_{ij,k}\, A_{ij}\left(T_{i,k} - T_{j,k}\right), \qquad 4$$

where the variables $m_{i,k}$, $c_{pi,k}$, $Q_{i,k}$, $h_{ij,k}$, $A_{ij,k}$, $T_{i,k}$, and $T_{j,k}$ are values computed at time t_k. The time, t_{k+1} is expressed as:

$$t_{k+1} = t_k + \Delta t_k, \qquad 5$$

the temperature, $T_{i,k+1}$, at t_{k+1}, is then

$$T_{i,k+1} = T_{i,k} + \Delta T_{ik}. \qquad 6$$

The relevant heat transfer interface areas are:

A_{12}	total surface area of all calandria tubes;
A_{16}	radiation surface area of corium in the calandria vessel;
A_{26}	inner surface area of calandria vessel;
A_{36}	inner surface area of calandria vessel;
A_{46}	inner surface area of calandria vessel;
A_{56}	inner surface area of calandria vessel;
A_{67}	outer surface area of calandria vessel;
A_{68}	outer surface area of calandria vessel;
A_{79}	inner surface area of shield tank;
A_{89}	inner surface area of shield tank; and
$A_{9\ 10}$	outer surface area of shield tank.

The indices refer to the materials listed previously.

2.2.1 Core Damage State #1

This core damage state involves hot fuel within the intact fuel channels. This state starts at the end of the blowdown stage. During CDS1, the fuel is heating up and the channels are hot but remain intact. The end point of CDS1 is defined as when the calandria discs rupture. Shortly thereafter the channels start to disassemble. The criteria for channel disassembly are based on two conditions which must be satisfied: a) the channel is hot (defined by channel temperatures exceeding 1200°C); and b) the moderator water level is at least 1 meter below the top channel row, thus creating enough lateral space to allow for significant channel sagging and mechanical stresses leading to channel failure. The second condition is always met following the rupture of calandria discs. Condition (b) can be conveniently expressed in terms of moderator liquid temperature. It is satisfied if the moderator is saturated or boiling at the rupture disc setpoint pressure. The time required to reach condition (b) represents the duration of CDS1. For computation purposes CDS1 is divided into two stages.

In stage 1, most of the decay power heats up the total fuel mass (only UO_2 is considered and the piping is conservatively neglected) from about 300°C (the assumed fuel temperature at the end of the blowdown) to 1200°C, while some heat is transferred to the moderator from the surface of the calandria tubes. At about 1200°C the pressure tubes are assumed to sag or to balloon, providing good contact with the calandria tubes. Hence, in stage 2, most of the decay power is conducted to the moderator liquid until it reaches the saturation temperature corresponding to the pressure at which the first rupture disc is designed to open. The duration of stage 2 is the period from the end of stage 1 until the moderator liquid reaches the saturation conditions and the opening of the first rupture disc occurs. No steam is discharged out into containment during this period.

2.2.2 Core Damage State #2

This core damage state involves fuel and channel debris (corium) present in the calandria vessel. For computation purposes CDS2 is divided into three stages (stages 3 to 5).

At the beginning of CDS2, the CV rupture discs are perforated which leads to flashing of the moderator into containment. The amount of two-phase moderator flashed into containment and the duration of flashing are estimated assuming that the discharged mass flow rate varies linearly with time. It should be noted that the duration of flashing (stage 3) is relatively short and changes in the temperatures of various materials are expected to be insignificant during this period; therefore, calculations of temperature changes are not required.

At the end of moderator flashing, phase separation occurs in the moderator and steam begins discharging out of the CV through the moderator discharge pipes. During this steam discharge period, the pressure in the CV is slightly higher than within containment which is at atmospheric pressure (enough to discharge the amount of steam produced due to decay power). The top rows of channels are uncovered and channels start to disassemble while the moderator inventory is being boiled off by the heat from the debris falling into the pool as well as the remaining hot channels still submerged (stage 4). The duration of stage 4 is the time required to boil off the residual moderator inventory.

Once the moderator inventory is boiled off, the remaining intact channels will collapse shortly afterwards (stage 5). During this stage, all remaining intact channels are assumed to collapse within 15 minutes (i.e., core disassembly is complete in 15 minutes) since the channels are already hot on the inside.

2.2.3 Core Damage State #3

At the beginning of CDS3, the corium piles up at the bottom of the CV and heats up the CV wall which is surrounded by the ST water. This process continues until either, a) the CV wall temperature reaches its melting point and fails, or, b) the ST temperature reaches the saturation temperature corresponding to its failure pressure and fails. The analytical methodology does not make *a priori* assumptions, it simply proceeds to calculate both the critical variables for a given accident sequence. Once one of the failure criteria is reached, the subsequent calculations proceed accordingly. The accident progression following vessel failure is different depending on which (the CV or ST) fails first. Both scenarios are discussed in the following subsections.

2.2.3.1 Shield Tank Failure Prior to Calandria Vessel Failure (Case a)

First, consider case a, where the shield tank fails while the calandria vessel is still intact. For computational purposes, CDS3 is divided into four stages, stage 6a through 9a. In stage 6a, the corium heats up the CV wall which transfers the heat to the surrounding water in the ST. This process continues until the ST water reaches the saturation temperature corresponding to its failure pressure. The ST is assumed, conservatively, to burst along the seam at the bottom of the tank.

During the flashing of the ST water inventory (stage 7a), it is conservatively assumed that the net energy transferred to the ST is sufficient to maintain its inventory at saturation temperature, at the ST failure pressure, P_{st}. Thus, the flow rate during the flashing is constant. The flashing duration is conservatively calculated assuming that the entire water inventory of ST is discharged. To calculate the amount of liquid and steam discharged into containment at atmospheric pressure during flashing of the ST water inventory, a two-phase flashing enthalpy corresponding to the saturated liquid enthalpy at P_{st} is considered.

After the ST water inventory is depleted, the CV wall heats up faster (stage 8a). During this stage, it is conservatively assumed that there is no further heat transferred from the CV wall to the remaining steam in the ST. The duration of this stage is the time required to heat up the CV to its melting temperature, at which point the CV is assumed to fail. No steam is produced during this stage.

When the CV fails as the corium melts through it, it is assumed that the corium is displaced into the ST and instantaneously melts through the ST and falls into the fuelling machine duct. This is conservative, since melting through the ST, with the steel balls at its bottom, does require some time. The corium is quenched by the water (originated from the moderator and the ST) lying on the floor of FMD. The duration of stage 9a is conservatively assumed to be zero.

2.2.3.2 Calandria Vessel Failure Prior to Shield Tank Failure (Case b)

Now consider the alternate scenario where the CV fails while the ST remains intact (case b). Again, for computational purposes, CDS3 is divided into four stages, stages 6b through 9b. In stage 6b, the corium heats up the CV wall which transfers the heat to the surrounding water in the ST. This process continues until that the CV wall temperature reaches its melting point and the CV fails.

When the CV fails, the corium is assumed to be displaced into the ST and quenched by water instantaneously (stage 7b). Therefore, the duration of this stage is zero and the ST water temperature is increased instantaneously. Since the corium heat is assumed to exclusively be used in raising the ST water temperature, no steam is produced. In stage 8b, the decay power is heating up the ST water. The duration of this stage is the time required to raise the ST water temperature to the saturation temperature corresponding to its failure pressure. No steam is produced during this stage. At the beginning of stage 9b, the ST is assumed to fail and its water inventory is flashed into containment.

2.2.4 Core Damage State #4

At the end of CDS3, both of the CV and the ST have failed and the corium is assumed to be displaced into the FMD directly beneath the reactor vault. The process of corium displacement from the CV and ST is not necessarily rapid. It is not like pouring of a pool of molten corium, but a displacement of solid materials followed by gradual liquefaction of residual solids in the CV and ST and a gradual displacement of the slurry out of them.

At the beginning of CDS4, the FMD floor, which is at the lowest elevation inside containment, is covered by water. This is water discharged from the heat transport system during the blowdown phase, water discharged from the moderator during flashing following the rupture of the CV discs, moderator discharged through the annulus gas bellows (if applied), water discharged from the ST after its failure, and ECI inventory if it was injected. Of this total amount of water, two quantities of water inventory should be subtracted. The first amount represents the quantity of water likely to end up in the ECI recovery sump and hence not available on the FMD floor. The second quantity accounts for the water transformed into steam when the corium falls into the water pool and is initially quenched. After quenching there is still a significant quantity of water remaining on the floor. The corium reheats and begins to evaporate the residual water. The duration of CDS4 is thus equal to the time required to boil off the residual water on the floor of the FMD.

The calculation for duration of steaming assumes implicitly that once the water on the floor is vapourized, this amount of water is unavailable for further cooling. However, the steam produced will be condensed inside containment, either by engineered (e.g., air cooling units) or natural means or both. Since the FMD floor is the lowest point in containment, the condensed steam would drain back onto the floor replenishing the water, such that the vapourization process may go on for a very long time. To a lesser extent, this also applies to the case where there is an opening in the containment envelope, since the discharge of the steam outside containment through an opening is a slow process.

2.3 Source Term Calculation

The source term estimate is governed by the timing of the initiating event coupled with the "state of containment" at the time of the accident and subsequent progression. This release is expressed as a fraction of the total core inventory of four fission product groups representing radioactive isotopes/compounds of similar volatility and chemical affinity. The four groups are classified as noble gases, volatile, semi-volatile and non-volatile fission products.

The source terms are evaluated for three source term periods (STP); defined as short term (0 to 6 hours), intermediate term (6 to 24 hours) and long term (greater than 1 day). The choice of utilizing STPs is to divide the accident progression in terms of short and long term responses as well as simplification of the estimation calculation. In general, the later the release, the smaller the consequences to the public because radiologically significant short lived isotopes have time to decay. The relationship between STPs and CDSs is dependent on the timing of the accident progression. The STPs are fixed periods whereas the CDS is dependent on a particular modelled severe accident.

The major factors in determining the potential release of radioactive fission products to the environment are:

- the quantity and mix of radionuclides released into containment (engineering judgments);
- the timing of release with respect to the start of the accident (from the timing calculation);
- the available dilution volume (DV) inside containment (station specific);
- the existence of an internal source of pressurization (i.e., a driving force to expel airborne aerosols); and
- the status of the containment envelope and subsystems prior to and during the accident progression (from the fault tree analysis of modelled systems).

The dilution volume is the containment volume throughout which the airborne fission products are dispersed. For calculational purposes, any radioactivity inside containment is assumed to be homogeneously dispersed within the dilution volume.

For any significant release of fission products from containment to the environment, two conditions must be satisfied. The first is that a release pathway must exist and the second is that a driving force must exist. The driving force is a positive pressure differential between containment and the outside environment. This differential is created by the balance between addition of gases to the containment atmosphere (for example, steam generation, instrument air and non-condensable gases produced in core-concrete interactions) and removal of gases from the containment atmosphere (for example by steam condensation or the filtered air discharge system (FADS)). These types of gradient tend to be sustained throughout the STP, therefore they are termed "sustained discharges". A short-lived gradient (or "temporary discharge") may also be created by either an uncontrolled hydrogen gas burn or a rapid pressurization of the moderator.

The effectiveness of containment response in preventing or limiting off-site releases depends on the state of containment at the time of the accident. The state of containment is defined in terms of the integrity of the envelope and availability of the containment subsystems.

The major containment subsystems which affect the off-site releases include:

- containment envelope integrity;
- steam pressure control;
- non-condensable gas pressure control; and
- filtered discharge.

The containment envelope is the final physical barrier against airborne releases and its state has a major impact on the magnitude of releases. If the containment envelope is intact, release of radioactive aerosols is negligible regardless of the conditions inside containment. However, an impaired containment envelope, whether due to a pre-existing opening or a consequential failure, is assumed to provide a direct pathway out to the environment with the releases driven out when the containment pressure exceeds atmospheric pressure.

Steam is a primary driving force in expulsion of airborne fission products and as such, the containment subsystems designed to mitigate steaming become important once a severe accident is initiated. The two major systems that can mitigate large quantities of steam are the vacuum building (and dousing system) early in the accident progression - i.e., during blowdown, and the air cooling units (ACUs) in the long-term.

The other primary driving force is pressure buildup due to production of hydrogen gas and other non-condensable gases during the accident progression. In the case of hydrogen, there is added concern whether concentrations are conducive for an uncontrolled burn which leads to a consequential failure of the containment envelope. Another source of non-condensable gases is instrument air in-leakage. There are two containment subsystems designed to mitigate non-condensables: the hydrogen ignitor and FADS. If the FADS is available and containment remains intact, only noble gases are released.

The source term estimate is an analytical calculation primarily based on a simple rate equation. The aerosol deposition characteristic is modelled as an exponential decay with the settling half-life based on simulations from the MAAP code. The half-life determination is based on the environment during the accident progression. A steam environment with ACUs available would lead to a significantly shorter half-life than a dry environment without forced convection. Currently, due to a lack of transient information about aerosol behaviour during the severe accident progression, an attempt is made to be conservative in terms of aerosol deposition half-lives (i.e. generally, a longer half-life increases the time period of suspension of aerosols which leads to higher release estimates).

Once the timing of the accident progression is determined, the analytical calculation of the source term can accommodate a variety of containment states for quick analysis. This is the main reason for using an analytical method of calculating source terms since the MAAP code requires a very detailed model of the accident scenario and containment state, it is not currently used when a significant number of simulations is required. Five selected unique containment states were analyzed. These states were chosen to give a broad based estimate given that only certain key states significantly affect the releases. These key states include the status of the containment envelope (intact, small pre-existing opening (size 0.1 m^2) and large pre-existing opening (size 1.6 m^2)), the availability of the ACUs and the effect of an uncontrolled hydrogen burn.

3. RESULTS AND DISCUSSIONS

The results presented in this section focus on the Bruce NGS B design. The relevant parameters for the analysis of Bruce NGS B severe accident conditions are summarized in Table 2. This includes the core power, the initial values of material masses, pressures and temperatures, the heat transfer areas as well as the critical values for specific criteria which must be met to define core damage states. The decay power, as a fraction of the total core power, following the reactor shutdown is a function of time [1]. This decay heat is adjusted to account for the transient release of volatile fission products (starting at the time of core disassembly) since they constitute approximately 40% of the total decay heat. The specific heat for various materials are determined as a function of temperature. The heat transfer coefficients are determined as a function of pressure, temperature and material phase.

An example of the results of the small break LOCA accident scenario (for the Bruce NGS B design) is presented in the following sections in order to illustrate predictions of the timing of the accident progression and tracking of the water inventory. Both are key parameters in the determination of the transient source term. Note that the LOCA event must be coincident with failures of the ECI and/or ECR and moderator systems with no operator intervention to potentially lead to core disassembly. Two sizes of break are considered, namely, 40 and 270 kg/s. The calculations are performed for both by crediting and not crediting SG cooldown cases.

3.1 Blowdown State

The blowdown phase of the overall severe accident progression is the key difference between the various postulated accident scenarios analyzed. Since the HTS inventory is a fixed and known quantity, the timing to the end of the blowdown phase (when most of the HTS inventory is discharged) is the defining difference between cases. After blowdown, the severe accident progression is generally the same for all accident scenarios (the exception being RIA events). Figures 1 and 2 show the transient of liquid and steam inventory in containment, respectively, during the blowdown phase following a small break LOCA of 40 kg/s.

The blowdown progression may be divided into four stages. In the first stage, a two phase discharge continues at rate of 40 kg/s until the reactor trip occurs at 15 minutes from the accident initiation. The amounts of discharged water and steam into containment, during this period, are 22.7 and 13.3 Mg, respectively. Following the reactor trip, the two-phase discharge continues and HTS pressure drops rapidly. If the steam generator (SG) cooldown is assumed available, the HTS pressure drops to the steam generator pressure (about 4.65 MPa) prior to the initiation of the controlled cooldown. During steam generator cooldown, the HTS pressure is assumed to follow the SG pressure transient and the discharge from the break is estimated as an average two-phase discharge between the starting and final pressures of the cooldown phase. The duration of SG cooldown is determined by its temperature rate of reduction. This stage lasts for 0.56 hour and the amounts of discharged water and steam into containment, during this period, are 21 and 12.4 Mg, respectively. In the third stage of the blowdown, the two-phase discharge from the break continues until the phase separation occurs in the HTS. This stage lasts for 6.36 hours, and the amounts of discharged water and steam into containment, during this period, are 84.6 and 49.7 Mg , respectively. In the fourth stage of the blowdown, only steam will be discharged from the break. The HTS pumps are assumed running. Since the SG controlled cooldown is credited, the SG acts as an effective heat sink, which is very capable of removing the decay heat generated in the core. In this case, the steam discharge rate at the end of phase separation is governed by the degraded pump head in a steam environment. This stage lasts for 3.1 hours, and the amount of discharged steam into containment, during this period, is 30.5 Mg

If the SG cooldown is assumed unavailable, the HTS pressure is conservatively assumed to drop to the steam generator pressure (4.65 MPa) and is maintained constant until phase separation. Thus the second stage of the blowdown does not apply. The steam discharge at the end of phase separation is governed by one of two mechanisms: either (a) the degraded pump head in a steam environment; or (b) the steam rate required to reject the decay heat. Both mechanisms are considered, and the discharge rate is conservatively taken as the greater of the two rates. The results are represented as dotted lines in Figures 1 and 2. In the case of 40 kg/s break, the blowdown duration is about 10.25 hours if the SG cooldown is available. The blowdown duration becomes much shorter (about 3.7 hours) if the SG cooldown is not credited.

Similar results are shown in Figures 3 and 4 for a small break LOCA of 270 kg/s. In this case, automatic reactor trip is assumed to occur at 4 minutes after the break. At the time of the reactor trip, 60 Mg of water is discharged from HTS. Because the break is relatively large, the phase separation occurs during the SG cooldown (if it is assumed available), and the third stage of the blowdown does not apply. It can be concluded that for larger break size, the blowdown period is shorter and the effect of SG cooldown is less pronounced than smaller break size. The blowdown of the HTS results in 128.4 Mg and 105.8 Mg of discharge of water and steam, respectively, into containment.

3.2 Core Damage States

The severe accident progression is generally characterized by periods of heating up, flashing, boiling off and quenching within containment. Figures 5 and 6 track the liquid and steam inventories discharged into containment, respectively, for the duration of the accident scenario for the break of 40 kg/s. Similar results are shown in Figures 7 and 8 for the break of 270 kg/s. These inventories are a function of time from blowdown such that the initial inventories are a result of the blowdown discharges. In each plot, two different curves are presented to illustrate the effect of crediting the SG cooldown. The rapid increases in the liquid inventories are a result of flashing, with the first being attributed to rupture disc perforation and the second corresponding to the ST failure. The rapid decrease in liquid inventory is attributed to the quenching of hot corium after the melt-through. For the steam inventories, the first instance of flashing (following CV rupture disc opening) does not show up on the scale used in the plot but the second occurrence (following ST failure) is much more significant. The differences are due to the quality of water at these times. The gradual increase in the steam inventory after flashing is due to moderator boil off. The rapid increase is due to flashing after shield tank failure and the third rapid increase is attributed to quenching of the corium on the FMD floor. After quenching, the gradual increases in steam inventory is due to the long term boiloff of the residual inventory remaining on the FMD floor. This is the final state of the accident progression (CDS4) and its duration is dependent on the particular accident scenario. Due to the long time scale of the analysis, the behaviour of the various accident scenarios are not noticeably different. For smaller break size, the blowdown time increases, the periods of heating up and boiling off become longer. It can also be concluded that crediting the SG cooldown results in longer periods of blowdown, heating up and boiling off. This effect is more pronounced for smaller break size. For all severe accidents analyzed for the Bruce NGS B design, shield tank failure was predicted to occur before calandria vessel failure.

3.3 Source Term

The results of the timing and steam discharge calculations are used in generating the source term estimates out of containment. Generally, earlier severe accident progressions and higher steam discharge rates led to greater fission product releases and ultimately, higher public doses. The source term assessment deals with the status of containment and driving forces (for example steam discharges), in its estimation. Other key severe accident phenomena, such as core-concrete interaction, steam explosions and global hydrogen gas ignition, are explicitly considered as to whether they are applicable and their effects are quantified, if found to be applicable, in the source term assessment. The resultant source terms are used in determining the event consequences for a broad range of severe accident scenarios.

Given the timing calculations of the specific small LOCA example for the Bruce B design, the source term estimates for five different containment states are presented in Table 3. It must be stressed that if the containment envelope remains intact, as designed, then any releases to the environment will be limited to noble gases with possibly some insignificant amount of radioactive aerosols due to leakage. If the containment envelope is intact, there are no calculated amounts of radioactive aerosols released to the environment.

As an example in order to illustrate some source term results, comparison of the discharge characteristics for the 270 kg/s discharge with steam generator controlled cooldown unavailable is shown in Figures 9a to 9d. Since the containment envelope is assumed to remain intact for containment state 'A' (Table 3), the discharge rate to the environment is very small. This discharge rate is based on instrument air leakage rates (from seals, gaps and cracks in containment). With the exception of the containment state 'A', the differences in discharge characteristics between the various containment states is mainly due to the differences in their respective net steaming (i.e., steam generated minus the amount condensed either through the ACU system or natural condensation) rates. Since steaming is a primary driving force to expel airborne fission products, it has a significant impact on the magnitude of releases to the environment, especially given the fact that the analysis assumes a direct pathway if an opening is present in the containment envelope. Of note, is that for containment states B to E, there is a very high rate of steaming (off-scale on the presented plots) due to the failure of the shield tank vessel. In the long-term (over 40 hours after the accident starts for the examined case), with the FAD system assumed available, there are insignificant radioactive aerosol releases. Figure 10 shows the distribution of fission product releases to the environment for containment state 'D' for each of the three STPs (i.e., not cumulative releases), as an example. This case models failure of the ACUs coincident with a large pre-existing opening in containment. It is representative of the worst source terms that can be expected from all small break LOCA analysis cases. Since, in this case, most of the releases occur early (during the first STP), it results in very high potential consequences, with corresponding high-risk potential (however, the associated risk is also dependent on the probability of the event). Therefore, this particular case may be one of the cases identified as needing further study to reduce its overall risk value.

The following are some key observations about the results of the source term calculations. As seen in Table 3, the trends for the source terms is are summarized in the following:

- no significant releases to the environment if the containment envelope is intact (state A);
- pre-existing opening provides a direct pathway out to the environment which significantly increases the amount of fission products (the key release being volatile aerosols);
- the unavailability of the ACUs has a major impact in increasing the amount of releases (comparison of state C with state D);

- the larger of the two small LOCA events is always slightly worse in terms of releases when comparing the same containment state, as expected;
- the unavailability of steam generator controlled cooldown has a significant impact on the 40 kg/s small LOCA event because the timing of the accident progression is pushed into STP1 (i.e., 0 to 6 hours); and
- the availability of the steam generator controlled cooldown does not significantly affect the source term of the 270 kg/s SLOCA event because the timing is not significantly affected by the cooldown.

The calculated source terms are used in risk assessment as a basis to develop consequence categories. These categories provide the framework from which the final estimate of accident frequency and risk can be developed.

4. SUMMARY

Due to the broad range of accident events to be analyzed for the risk assessment of CANDU reactors, a simple and straight-forward,yet thorough methodology has been developed in order to calculate the timing and general containment conditions expected from severe accidents. This methodology is based upon a clearly defined severe accident progression, termed *core damage states*. The timing and steam generation calculations during these core damage states considers the thermal-mechanicalresponse of the reactor and other relevant systems. The methodology of the calculations involved in each stage of the severe accident progression are summarized (relevant for Bruce or Darlington type design). The results of a small break LOCA scenario for Bruce NGS B is also presented as an example typical of the results obtained from application of this methodology. The postulated severe accidents analyzed mainly differ in the timing to reach and progress through each core damage state. For smaller break sizes, the blowdown time increases, the periods of heatup and boiloff become longer. Crediting the SG cooldown results in longer periods of blowdown, heating up and boiling off. This effect is more pronounced for smaller break sizes. For all severe accidents analyzed for the Bruce NGS B design, shield tank failure was predicted to occur before calandria vessel failure.

Ultimately, these calculations are utilized in determining the composition and magnitude of radioactive releases from the station for each identified accident scenario (with specific details of each sequence determined by fault tree analysis). The source term estimate consider several significant factors which affect releases out to the environment. The most important factors in determining potential source terms is the status of the containment envelope and containment subsystems. Therefore, the timing of the accident progression coupled with the state of containment will determine the ultimate releases to the environment. Results are summarized for estimated source terms of a small break LOCA event. It must be noted that if containment remains intact and operates as designed, this analysis assumes that there are negligible radioactive releases to the public. If, on the other hand, there is an opening in containment coincident with failures of certain key containment subsystems (such as the ACUs), the estimated releases can be quite high. In general, a larger break size leads to higher source term due to the impact by the earlier timing of the severe accident progression. The results of the source term calculations lead to a broad based categorization of various postulated accident events used in probabilistic safety assessment as a framework to calculate risks.

The timing and steam generation calculations, relevant to several existing CANDU reactor designs, can be used in determining the event consequence for a broad range of severe accident scenarios. From these results, a broad based categorization is formed which constitutes the framework for developing the overall frequency and consequence estimates from the study, leading to the identification of the dominant contributors to risk and indicating where risk can be reduced.

5. REFERENCES

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2. M.H. CHOI, M.T. KWEE, R.K. LEUNG and S.G. LIE, "MAAP-CANDU Simulation of Severe Accidents in Darlington NGS", Paper presented at the 19th CNS Nuclear Simulation Symposium, Hamilton, Ontario, October 15-17, 1995.

6. ACKNOWLEDGEMENT

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Table 1
Definitions of Process and Safety Systems Failures

System	Failure Label	Failure Definition
Emergency Coolant Injection System (ECIS)	ECI	High pressure ECIS unavailable due to injection valves failing to open on signal or the accumulator and low pressure pumps unavailable when required. For ECI blow-back event, by definition, the ECI is not available.
Emergency Coolant Recovery System (ECRS)	ECR	ECRS unavailable due to injection valves failing to open (consequential failure due to ECIS injection valve failure) or ECI recovery system unavailable.
Moderator System	MOD	Moderator system failure mode represents absence of all heat sinks. In LSDS, moderator is unavailable as a consequential failure (overpower transient fails the calandria vessel and drains the inventory). For an in-core LOCA, moderator heat sink unavailable due to inventory being discharged into containment via annulus gas bellows or channel end-fitting failure, rate determined by the static head of the moderator inventory.
Steam Generator Cooldown (SGC)	SG	Failure of the steam generator to provide crash cooling. For pump gland seal and steam generator tube ruptures, instrumented SRVs do not open since the required signal is conditioned on low HTS pressure and either on a) high reactor building pressure, b) high moderator level, c) high reactor building temperature, or d) sustained low reactor building pressure. Also for steam generator tube ruptures only, the spring-loaded SRVs are assumed to open and be stuck open as the postulated pathway for containment bypass. For all other events, SGC on or off is assessed.

Table 2

Relevant Parameters Used in the Accident Progression for the Bruce NGS B Design

Parameter	Value
Core Power	2704 MW
Mass of UO_2	136.3 Mg
Mass of Zircaloy	54.6 Mg
Total Mass of Water in Heat Transport System	234.2 Mg
Mass of Water in Moderator	276 Mg
Mass of Water in Shield Tank	925.9 Mg
Mass of Water Filling ECI Recovery Sump in Pressure Relief Duct	310.9 Mg
Mass of Calandria Vessel Walls	46 Mg
Mass of Shield Tank Walls	335.7 Mg
Heat Transfer Area Between Calandria Tubes and Moderator	1181.3 m^2
Surface Area of Calandria Vessel	180.3 m^2
Surface Area of Shield Tank	704.7 m^2
Inner Diameter of Calandria Vessel	8.4582 m
Average Pressure of Heat Transport System	9.97 MPa(a)
Average Temperature of Heat Transport System	283 °C
Surface Temperature of the Calandria Tubes	73 °C
Average Pressure of Moderator	195 kPa(a)
Average Temperature of Moderator	64 °C
Average Pressure of Shield Tank	229.5 kPa(a)
Average Temperature of Shield Tank	66.3 °C
Effective Discharge Area Per Rupture Disc	0.1231 m^2
Rupture Disc Pressure Setpoint	239 kPa(a)
Shield Tank Failure Pressure	880 kPa(a)
Melting Temperature of Calandria Vessel Walls	1327 °C
Nominal Head of PHT Pump	213.4 m
Nominal Volumetric Flow of PHT Pump	3.303 m^3/s
Steam Generator Nominal Pressure	4.65 MPa
Maximum Steam Generator Cooldown Rate	2.8 °C/min
Steam Generator Temperature at the End of Cooldown	165 °C
Containment Dilution Volume	96167 m3
Maximum Filtered Air Discharge System Rate (given as % of dilution volume)	1.91 %/hr
Maximum Instrument Air Leakage Rate (given as % of dilution volume)	0.358 %/hr
Steam Condensation Rate of ACUs in Reactor Vault	2.45 Mg/hr

Table 3
Source Term Estimates for Small LOCA Example

Case I.D.	Cumulative Releases in STP1 (% total core inventory)				Cumulative Releases in STP2 (% total core inventory)				Cumulative Releases in STP3 (% total core inventory)				State
	NG	VA	SVA	NVA	NG	VA	SVA	NVA	NG	VA	SVA	NVA	
SLA1-1	0	0	0	0	2.6	0	0	0	91.9	0	0	0	A
SLA1-2	0	0	0	0	31.1	3.0	0.3	0.03	100	3.9	0.4	0.04	B
SLA1-3	0	0	0	0	44.0	6.1	0.6	0.06	100	7.3	0.7	0.07	C
SLA1-4	0	0	0	0	49.2	21.6	2.2	0.22	100	28.8	2.6	0.26	D
SLA1-5	0	0	0	0	49.2	21.7	2.2	0.22	100	28.9	2.6	0.26	E
SLA2-1	0	0	0	0	4.8	0	0	0	92.1	0	0	0	A
SLA2-2	7.5	0.9	0.1	0.01	68.3	8.9	0.9	0.09	100	8.9	0.9	0.09	B
SLA2-3	11.4	1.5	0.1	0.01	69.7	11.6	1.2	0.12	100	11.9	1.2	0.12	C
SLA2-4	13.9	3.1	0.3	0.03	71.6	35.4	3.5	0.35	100	39.2	3.6	0.36	D
SLA2-5	13.9	3.1	0.3	0.03	71.6	36.0	3.6	0.36	100	39.8	3.7	0.37	E
SLB1-1	0	0	0	0	5.5	0	0	0	92.2	0	0	0	A
SLB1-2	36.0	6.8	0.7	0.07	71.3	11.0	1.1	0.11	100	11.0	1.1	0.11	B
SLB1-3	43.2	9.4	0.9	0.09	80.2	13.5	1.3	0.13	100	13.8	1.3	0.13	C
SLB1-4	46.7	25.2	2.5	0.25	90.1	39.2	3.9	0.39	100	41.8	3.9	0.39	D
SLB1-5	46.7	25.2	2.5	0.25	90.1	39.8	4.0	0.4	100	42.4	4.0	0.4	E
SLB2-1	0	0	0	0	5.6	0	0	0	92.2	0	0	0	A
SLB2-2	36.9	7.3	0.7	0.07	73.1	9.3	0.9	0.09	100	9.3	0.9	0.09	B
SLB2-3	43.4	9.8	1.0	0.1	82.6	11.8	1.2	0.12	100	12.1	1.2	0.12	C
SLB2-4	46.7	25.5	2.6	0.26	92.2	37.7	3.8	0.38	100	40.3	3.8	0.38	D
SLB2-5	46.7	25.5	2.6	0.26	92.2	40.0	3.8	0.38	100	40.6	3.8	0.38	E

NG = noble gases, VA = volatile aerosols, SVA = semi-volatile aerosols and NVA = non-volatile aerosols

The Case I.D. corresponds as follows:

"SLA1" represents a small LOCA with 40 kg/s initial discharge and with steam generator cooldown available
"SLA2" represents a small LOCA with 40 kg/s initial discharge and with steam generator cooldown unavailable
"SLB1" represents a small LOCA with 270 kg/s initial discharge and with steam generator cooldown available
"SLB2" represents a small LOCA with 270 kg/s initial discharge and with steam generator cooldown unavailable

"State" denotes a containment state assumed for the calculation where:

A - all containment subsystems operational with the containment envelope remaining intact
B - all containment subsystems operational with a pre-existing small opening (0.1 m^2) in the containment envelope
C - all containment subsystems operational with a pre-existing large opening (1.6 m^2) in the containment envelope
D - same as C except the ACUs are assumed to be unavailable throughout the severe accident progression
E - same as D with additional release due to a global hydrogen burn inside containment (highest modelled estimates)

FIGURE 1: Liquid Inventory in Containment during Blowdown

Break Size 40 kg/s

FIGURE 2: Steam Inventory in Containment during Blowdown

Break Size 40 kg/s

FIGURE 3: Liquid Inventory in Containment during Blowdown
Break Size 270 kg/s

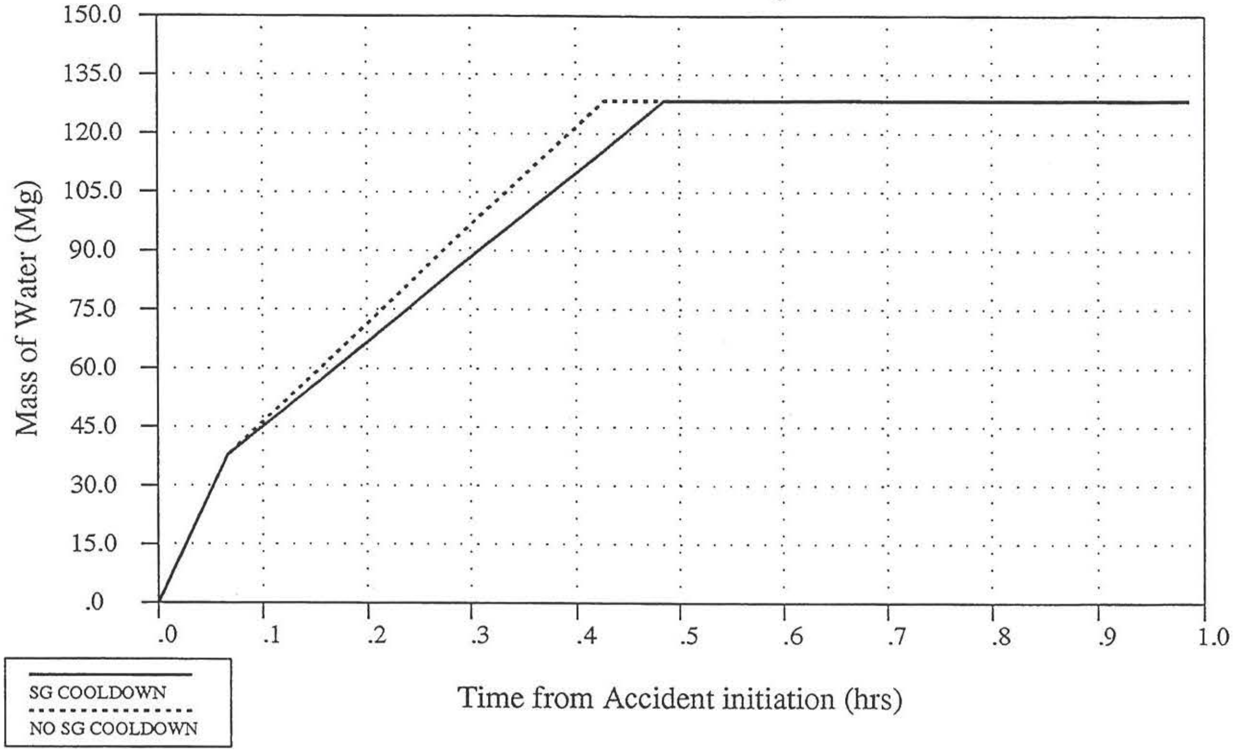

FIGURE 4: Steam Inventory in Containment during Blowdown
Break Size 270 kg/s

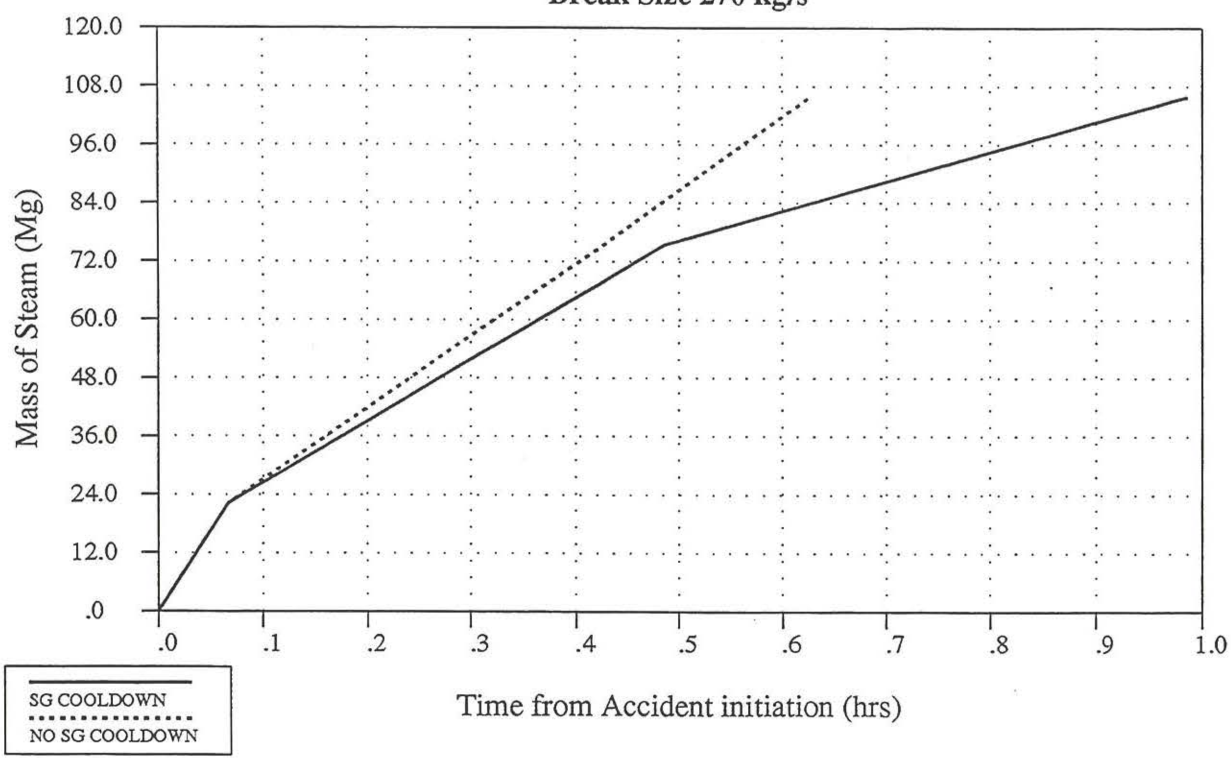

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FIGURE 5: Liquid Inventory in Containment during CDSs
Break Size 40 kg/s

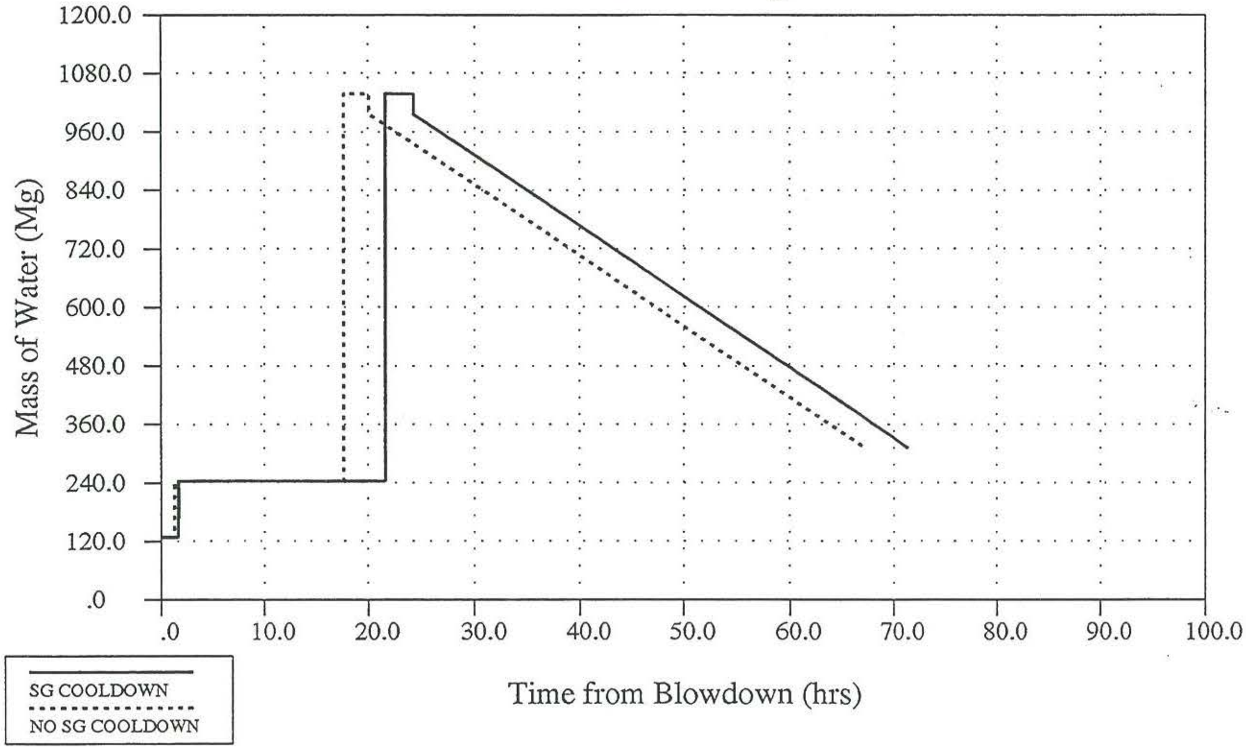

FIGURE 6: Steam Inventory in Containment during CDSs
Break Size 40 kg/s

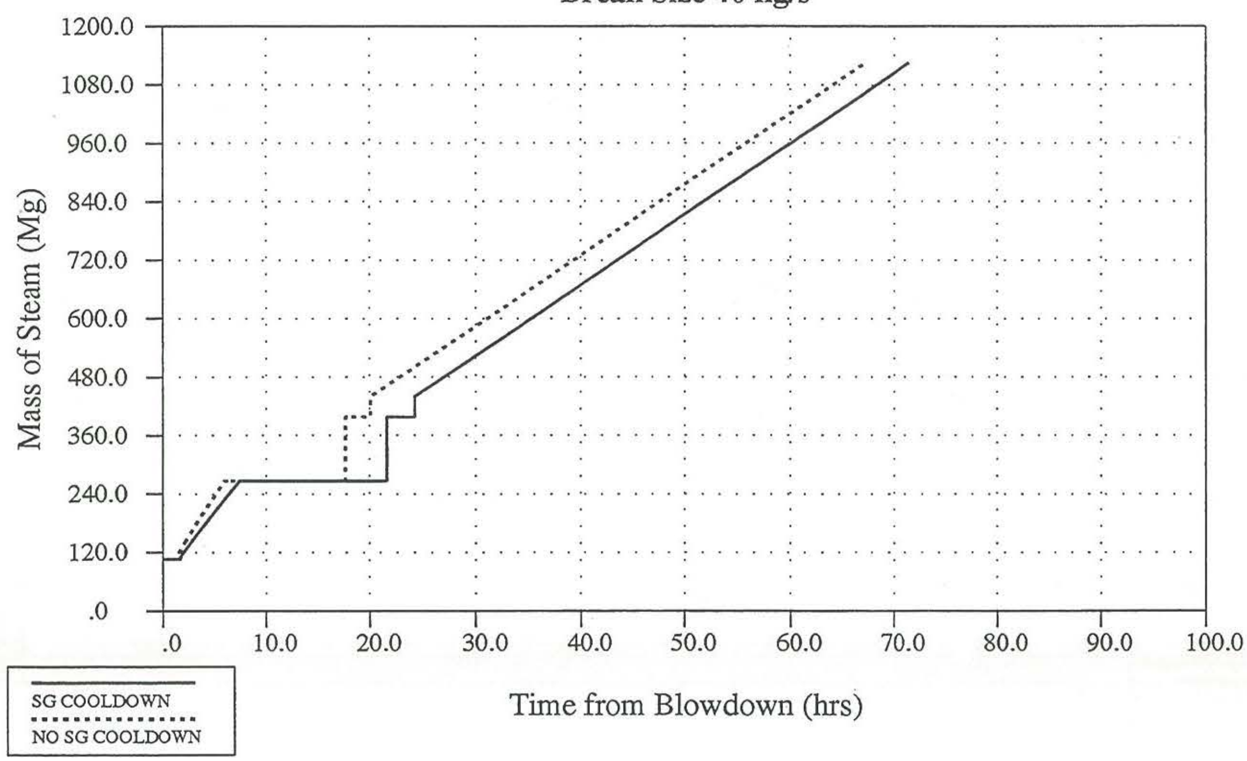

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FIGURE 7: Liquid Inventory in Containment during CDSs
Break Size 270 kg/s

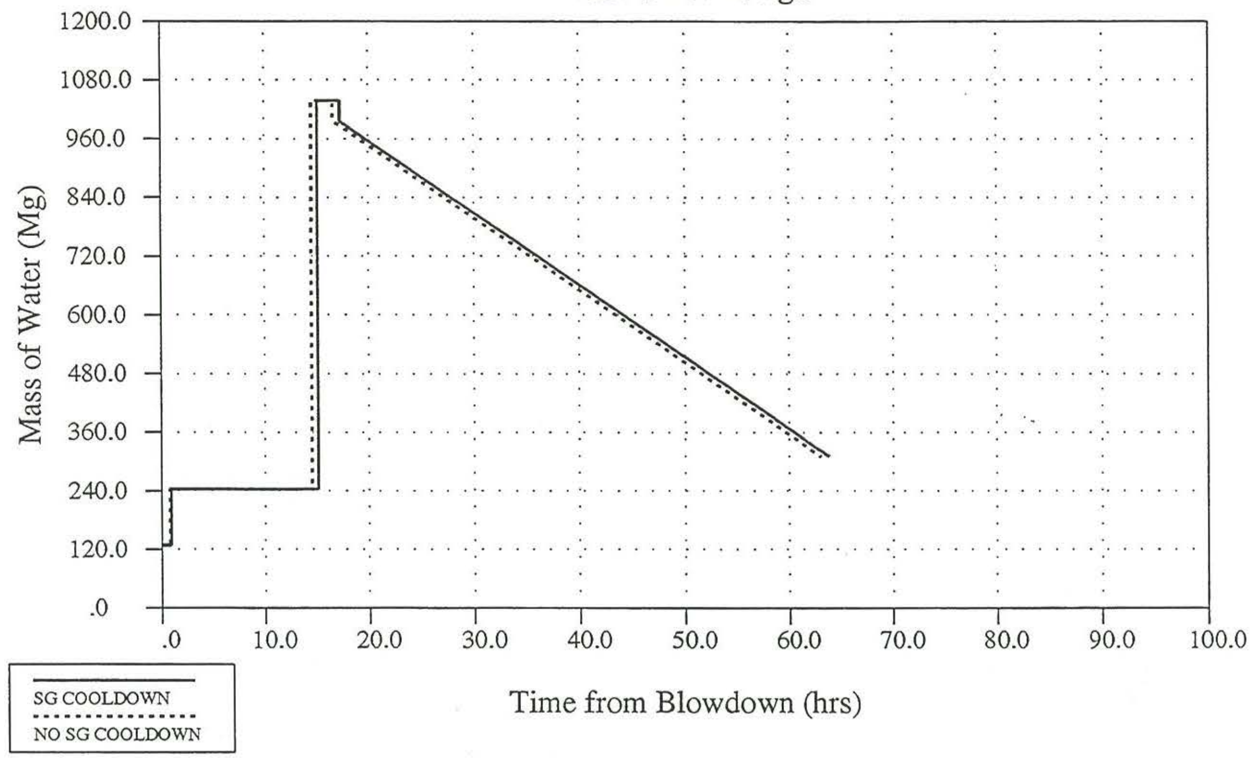

FIGURE 8: Steam Inventory in Containment during CDSs
Break Size 270 kg/s

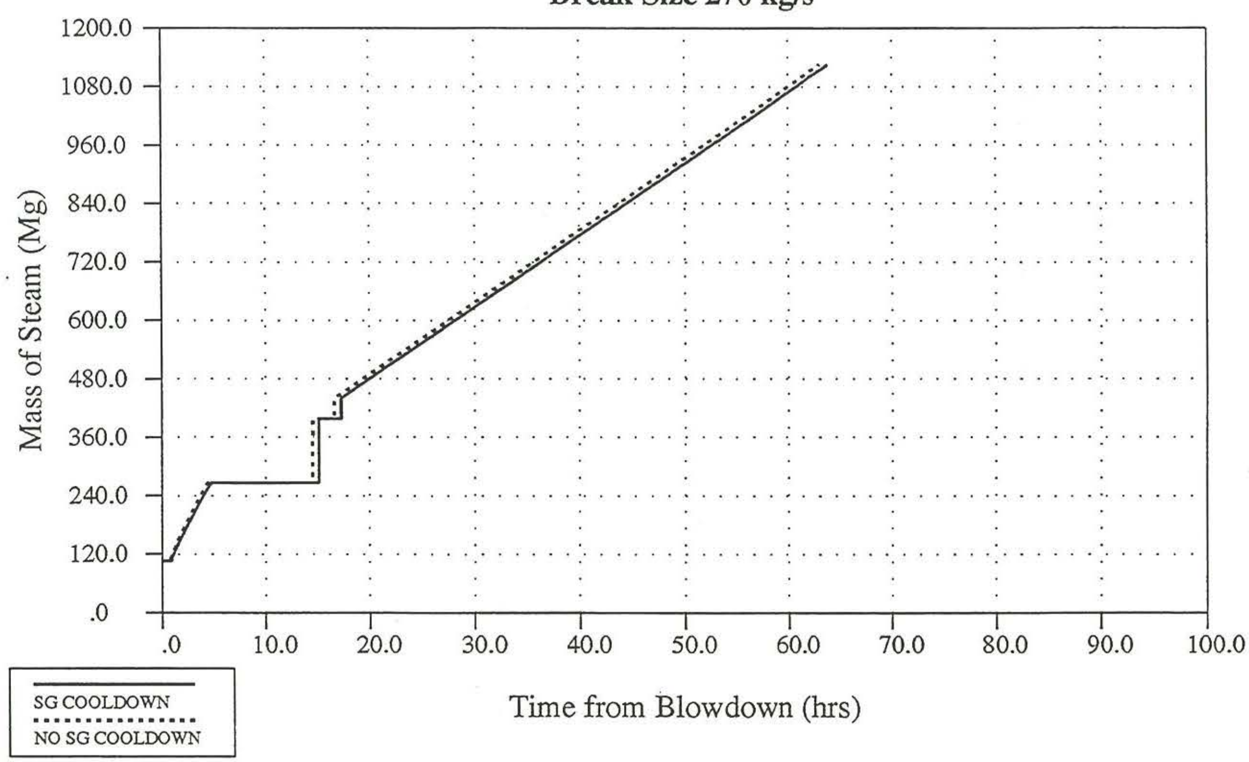

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Figure 9a: Discharge from Containment
Case SLB2, Containment State A

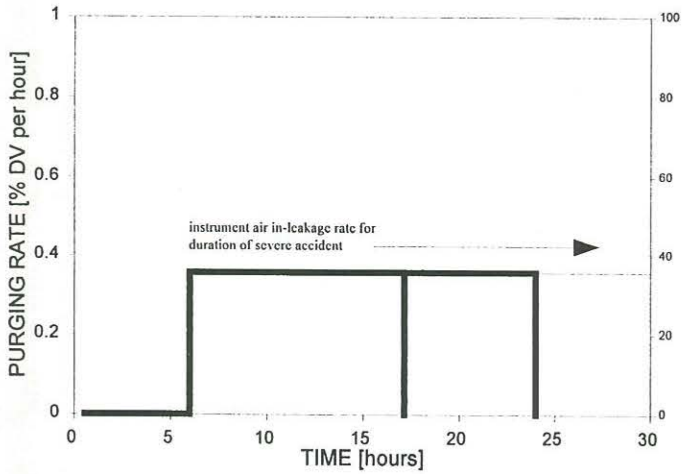

Figure 9b: Discharge from Containment
Case SLB2, Containment State B

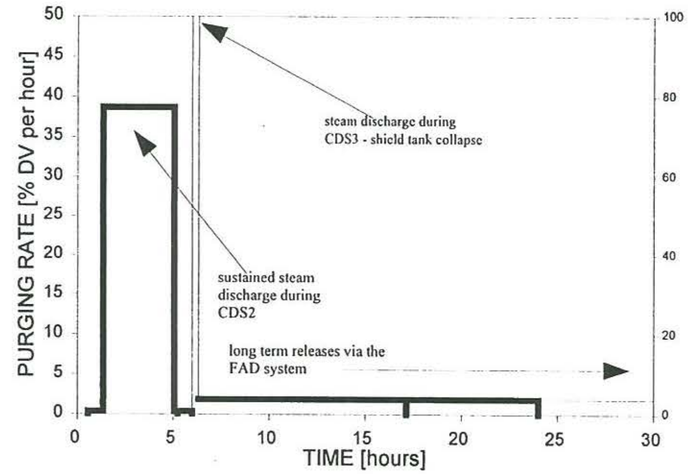

Figure 9c: Discharge From Containment
Case SLB2, Containment State C

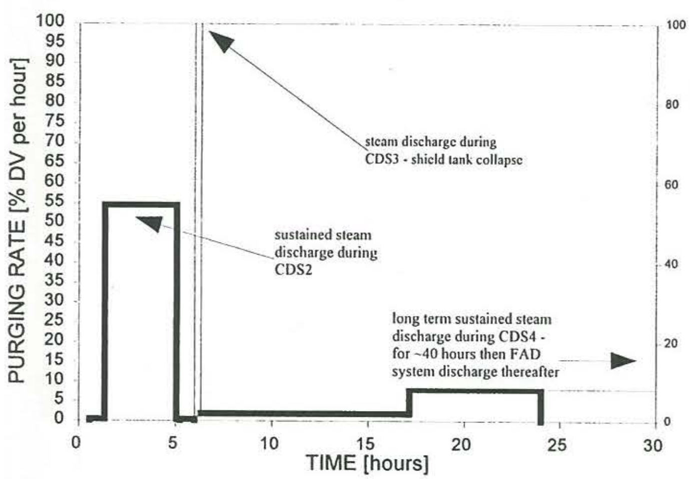

Figure 9d: Discharge From Containment
Case SLB2, Containment State D

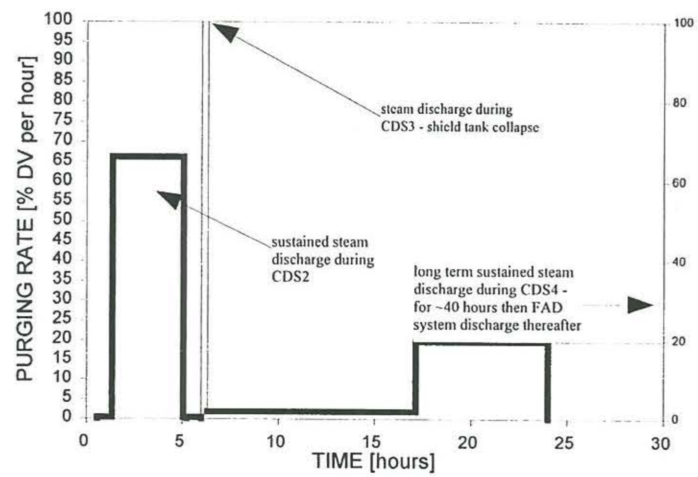

Figure 10: Source Term to Environment
Case SLB2, Containment State D