TIMING OF CORE DAMAGE STATES FOLLOWING SEVERE ACCIDENTS FOR THE CANDU® REACTOR DESIGN

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ABSTRACT

This paper documents the analytical methodology used to evaluate severe accident sequences. The relevant thermal-mechanical phenomena and the mathematical approach used in calculating the timing of the accident progression are described. An example of a specific accident scenario is provided in order to illustrate the application of the severe accident progression methodology. The postulated severe accidents analyzed mainly differ in the timing to reach and progress through each defined "core damage state".

1. INTRODUCTION

The determination of postulated severe accident progression, and modelling of physical processes which can lead to challenges to containment, form an integral part of the risk assessment of CANDU reactors. This paper presents the methodology and results of the timing and steam discharge calculations of a variety of accident sequences classified as having the potential to lead to core disassembly. 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 main factors affecting radioactive releases. The fault tree portion of the overall analysis determines the frequency of any accident scenario which, when coupled with the consequences of a specified event, defines the risk associated with that event.

The first challenge in determining the timing of an accident scenario is to define the system processes and their availabilities which 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.

The initiating events of these sequences include small and large break loss-of-coolant accidents (LOCAs), total loss of heat sinks, heat transport pump gland seal failures, single and multiple steam generator tube breaks, emergency coolant injection (ECI) blowback, large break LOCAs with loss of shutdown, and slow and medium loss of reactivity regulation (LORR). Each initiating event possesses a distinctive accident progression in the early stage prior to the onset of fuel damage. This initial phase of an accident usually involves blowdown and two-phase discharge which eventually depletes the coolant in the heat transport system (HTS).

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. When this happens, the bulk of the fuel decay heat in the hot channels is then transferred to the still subcooled moderator. However, since moderator cooling is assumed 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 vaporization process can occur for many cycles if the steam produced re-condenses in the containment and drains back to the FMD floor.

The accident progression described above is typical of postulated CANDU reactor severe accidents for Bruce and Darlington NGS. For analysis purposes it is convenient to describe the progression of a severe accident in terms of four core damage states (CDSs):

CDS1: hot fuel within the fuel channels
CDS2: fuel and debris in the calandria vessel
CDS3: fuel and debris in the containment
CDS4: boil-off of residual water by fuel and debris in the fuelling machine duct

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 (note: blowdown is defined here to mean from the start of the accident until the start of the first core damage state). Once the blowdown period is calculated, the methodology for determining subsequent CDSs is the same for almost all analyzed events (reactivity initiated events being the only exception). In other words, once the HTS inventory is depleted, the nature of the initiating event is of no further consequence. Therefore, the timing calculation of any accident scenario is taken as a two-step process. The first step examines the details of the particular event with attention given to the important phenomena associated with the event such as the availability of process and safety systems. In this first step, determination of the blowdown time and tracking of water mass inventory are key considerations. Once these parameters are determined, the next step involves using a computer program written to determine the timing and steam discharges into the containment for all CDSs.

The main task of the analysis is to evaluate the timing and the duration of each CDS and the associated thermal-mechanical phenomena for each accident sequence. The computational method used to assess CDSs is the same for all sequences. It is therefore economic to automate these repetitive computations. In the following, the relevant thermal-mechanical phenomena which occur in each CDS and the equations used to evaluate these phenomena will be described.

In parallel to the analytical methodology described in this document, the MAAP-CANDU code [1] 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.

2. ANALYTICAL MODELS

The timing calculation is essentially a transient solution of the conservation of mass and energy equations. It considers the availability of major heat sources (such as the decay and stored heat), and heat sinks (such as available water inventories and radiation heat sinks). Of note is that 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. These factors will be discussed in the following sections.

2.1 Heat Balance Rate Equation

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. 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:

$$\frac{d}{dt}\left(m_i c_{p,i} T_i\right) = Q_i - \sum h_{ij} A_{ij} \left(T_i - T_j\right), \tag{1}$$

where t is the time variable, m_i 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 1 is discretized for small time steps Δt_k with the incremental changes in temperature ΔT_{ik} .

$$m_{i,k} c_{p,i,k} \frac{\Delta T_{ik}}{\Delta t_k} = Q_{i,k} - \sum h_{ij,k} A_{ij} \left(T_{i,k} - T_{j,k}\right), \tag{2}$$

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 temperature at time

$$t_{k-1} = t_k + \Delta t_k , \tag{3}$$

is then

$$T_{i,k-1} = T_{i,k} + \Delta T_{ik} . \tag{4}$$

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.

The governing equations for each core damage state are discussed in the following sections.

2.2 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, i.e.,:

$$t_{cds1} = t_{blowdown} . \tag{5}$$

The blowdown time, $t_{blowdown}$, used here is the time period starting from the initiation of the accident until the HTS is empty of coolant. 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.

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.

2.2.1 Stage 1

In stage 1, most of the decay power goes to heat 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.

Equation 2 is applied to the fuel (i=1, j=2, the index numbers 1, 2 etc. refer to the materials specified before Equation 2), single phase liquid in the moderator (i=2, j=1 and 6), the CV wall (i=6, j=2 and 7), the single phase liquid in the ST (i=7, j=6 and 9) and the ST wall (i=9, j=7 and 10), respectively. When Equation 2 is applied to the fuel (i=1) or to the moderator (i=2), the temperature, T_{ik}, or T_{jk}, respectively, in the second term of the right hand side of the equation, is set equal to the initial temperature of the surface of the calandria tubes to account for the heat transferred to the moderator. The power source in Equation 2 for each of these materials is given, respectively, as:

$$Q_1 = Q(t) \, , \qquad Q_i = 0 \quad for \ i \neq 1 \, , \tag{6}$$

where Q(t) is the decay power as a function of time. It is noted that the calculations of heat transfer coefficients need to be carried out for solid to single phase liquid only. The time required to reheat UO_2 from 300°C to 1200°C is the duration of stage 1, DUR_{reheat}, and no steam is discharged out of the HTS during this period.

2.2.2 Stage 2

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. Equation 2 is applied to the single phase liquid in the moderator (i=2, j=6), CV wall (i=6, j=2 and 7), single phase liquid in the ST (i=7, j=6 and 9) and ST wall (i=9, j=7 and 10), respectively. The power source in Equation 2 is given, respectively, as:

$$Q_2 = Q(t) \, , \qquad Q_i = 0 \quad for \ i \neq 2 \, . \tag{7}$$

Again, the calculations of heat transfer coefficients are carried out for a solid surface in contact with single phase liquid only. The duration of stage 2, DUR_{sat}, 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.

For CDS1 the calculated duration and amount of steaming are then given by:

$$DUR_{cds1} = DUR_{reheat} + DUR_{sat} \, , \qquad STEAM_{cds1} = 0 \, . \tag{8}$$

2.3 Core Damage State #2

This core damage state involves fuel and channel debris (corium) present in the calandria vessel and it starts at t_{cds2} as given by:

$$t_{cds2} = t_{cds1} + DUR_{cds1} \, . \tag{9}$$

For computation purposes CDS2 is divided into three stages. At the beginning of CDS2, when the calandria discs rupture, some moderator inventory is being flashed into containment (stage 3). 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). Once the moderator inventory is boiled off, the remaining intact channels will collapse shortly afterwards (stage 5).

2.3.1 Stage 3

At the start 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 as follows.

Two-phase flashing ends when 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). With the internal heat generation in the moderator pool, the average void fraction α in the CV can be related to the steam generation rate ψ by:

$$\alpha = \frac{\psi}{1 + C_o \psi}. \tag{10}$$

The value of the constant, C_o, is taken to be 1.35, which is representative of a large number of water experiments. At the end of two-phase flashing, ψ is unity, and the final void fraction, α_f, of 0.4255 is obtained using Equation 10. This means that the remaining moderator inventory, after flashing, occupies the bottom 57.5% of the CV volume while the saturated steam occupies the top 42.5% of the CV volume. Afterwards, the steam flow rate, which is equal to the moderator vaporization rate, becomes very small (relative to the initial flashing rate) and the pressure in the CV, P_m, is slightly higher than in containment. The quality, χ, is related to the void fraction, α, by the following expressions:

$$\left(\frac{1-\alpha}{\alpha}\right) = \left(\frac{1-\chi}{\chi}\right)\frac{\rho_g}{\rho_l}, \qquad \chi = \frac{1}{1+\left(\frac{1-\alpha}{\alpha}\right)\frac{\rho_l}{\rho_g}}. \tag{11}$$

where ρ_l and ρ_g are the liquid and vapour saturated densities, respectively, at moderator pressure. Therefore, just before the end of the flashing stage, the quality χ is about 0.0004.

The remaining masses of moderator and steam in the CV at the end of the flashing period are given by:

$$M_{mw} = V_{cv}\rho_l(1-\alpha), \qquad M_{ms} = V_{cv}\rho_g\alpha, \tag{12}$$

where, V_{cv}, is the CV volume and the total amount of moderator discharged into the containment during the flashing, M_d, is given by:

$$M_d = M_{mod0} - M_{mw} - M_{ms}, \tag{13}$$

where M_{mod0} is the initial mass in the moderator. Of this total amount, the portion of liquid M_{wflash} and steam M_{sflash} discharged into the containment during the flashing period can be obtained using the following expressions:

$$\chi_c = \frac{(H_{flash} - H_f)}{H_{fg}}, \qquad M_{wflash} = (1-\chi_c)M_d, \qquad M_{sflash} = \chi_c M_d, \tag{14}$$

where χ_c is the quality, H_{flash} is the two-phase flow enthalpy corresponding to the saturated liquid enthalpy at the

CV disc rupture pressure, P_{m0}, and H_f and H_{fg} are the saturated liquid enthalpy and heat of vaporization of water at atmospheric pressure, respectively.

The initial flashing flow rate from the CV, W_{mf0}, is given by:

$$W_{mfo} = A_{mf} \sqrt{2P_{mo}\rho_{mlo}(1-\eta_{mo})}\ , \tag{15}$$

where P_{m0}, ρ_{ml0} are the pressure inside the CV and the saturated liquid density, respectively, at the start of the flashing and the critical pressure ratio, η_{m0}, is about 0.83 [2]. The available flow area, A_{mf}, is the flow area of the CV ruptured discs. This area depends on the number of open CV rupture discs.

To estimate the duration of flashing DUR_{flash}, it is assumed that the discharged mass flow rate varies linearly with time, then

$$DUR_{flash} = \frac{2M_d}{W_{mfo}}\ . \tag{16}$$

It should be noted that the duration of flashing 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.

2.3.2 Stage 4

At the end of moderator flashing, the pressure inside the CV has dropped to a value just slightly higher than atmospheric, and the boiling off of moderator begins. In this stage the moderator inventory in the CV consists of two separate phases: saturated water of mass, M_{mw}, occupying the volume, V_{mw}, and covering the area, A_{mw}, of the lower part of the CV and saturated steam of mass, M_{ms}, occupying the volume, V_{ms}, and covering the area, A_{ms}, of the upper part of the CV. The amount of water vaporized, ΔM_v, during the time interval, Δt_k, is given by:

$$\Delta M_v = \left[Q_4 - \sum h_{46k} A_{46}(T_{4k} - T_{6k})\right]\frac{\Delta t_k}{H_{fg}}\ . \tag{17}$$

Equation 2 is applied for the CV wall (i=6, j=4 and 7), the single phase liquid in the ST (i=7, j=6 and 9) and the ST wall (i=9, j=7 and 10). The power source in Equation 2 is given, respectively, as:

$$\begin{aligned} Q_4 &= Q(t)\,, \\ Q_i &= 0 \quad for\ i\neq 4\,. \end{aligned} \tag{18}$$

For every time step, the volumes, V_{mw} and V_{ms}, as well as the surface areas, A_{mw} and A_{ms}, are re-calculated. The heat transfer coefficients, h_{mw} and h_{ms}, are calculated for solid to single phase liquid and single phase steam, respectively. The aggregate equivalent heat transfer coefficient, h_{46k}, during the time interval, Δt_k, is given by:

$$h_{46k} = \frac{(h_{mwk}\ A_{mwk} + h_{msk}\ A_{msk})}{A_{46}}\ . \tag{19}$$

The duration of stage 4, $DUR_{boiloff}$, is the time required to boil off the residual moderator inventory. The amount of steam produced during stage 4 is thus

$$M_{sboiloff} = M_{mw} . \tag{20}$$

If some of the moderator inventory is assumed to be discharged into the containment through the annulus gas bellows due to the static head of the moderator inventory, during the moderator boiling off period, the mass flow rate of drained moderator, w_{drain}, is calculated as:

$$w_{drain} = A_{drain}\sqrt{2\ g\ \rho_l\ \Delta z_{drain}} , \tag{21}$$

where, A_{drain} is the available flow area for moderator draining, Δz_{drain} is the elevation difference between the moderator level and the draining level as a function of time, g is the acceleration due to gravity and ρ_l is the liquid density.

The corresponding amount of steam produced during stage 4 is then

$$M_{sboiloff} = M_{mw} - M_{wdrain} , \tag{22}$$

where the mass of drained moderator M_{wdrain} is given by:

$$M_{wdrain} = \int_{DUR_{boiloff}} w_{drain} dt . \tag{23}$$

2.3.3 Stage 5

Stage 5 begins following boiling off of the moderator inventory. 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. Thus,

$$DUR_{collapse} = 0.25 hr . \tag{24}$$

No steam is produced during this stage. Equation 2 is applied to the corium (i=1, j=6), the CV wall (i=6, j=1 and 7), the single phase liquid in ST wall (i=7, j=6 and 9) and the ST wall (i=9, j=7 and 10), respectively, for the duration $DUR_{collapse}$. The power source in Equation 2 is given, respectively, as:

$$\begin{aligned} Q_1 &= Q(t) , \\ Q_i &= 0 \quad for\ i \neq 1 . \end{aligned} \tag{25}$$

The heat transfer coefficient h_{16k} is the equivalent heat transfer coefficient due to natural convection and radiation between hot corium and the CV wall.

The duration of CDS2 as well as the duration and amount of steaming in this CDS are thus,

$$\begin{aligned} DUR_{cds2} &= DUR_{flash} + DUR_{boiloff} + DUR_{collapse} , \\ DUR_{steamcds2} &= DUR_{flash} + DUR_{boiloff} , \\ STEAM_{cds2} &= M_{sflash} + M_{sboiloff} . \end{aligned} \tag{26}$$

2.4 Core Damage State #3

CDS3 starts at time t_{cds3}:

$$t_{cds3} = t_{cds2} + DUR_{cds2} . \tag{27}$$

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 developed here 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 (cases a and b) are discussed.

2.4.1 Shield Tank Failure Prior to Calandria Vessel Failure

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.

2.4.1.1 Stage 6a

In this stage, 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 governing equations are the same as those governing stage 5, but starting at t_{cds3} , and the calculations end when the ST fails. The ST is assumed, conservatively, to burst along the seam at the bottom of the tank. The duration of stage 6a is called DUR_{heatup}.

2.4.1.2 Stage 7a

During the flashing of the ST water inventory, it is conservatively assumed that the net energy transferred to the ST is sufficient to maintain its inventory as saturated water at the ST failure pressure, P_{st}. Thus, the flow rate when flashing is constant and given by:

$$w_{stf} = A_{stf} \sqrt{2 \, P_{st} \, \rho_{stl} \, (1-\eta_{st})} \; , \tag{28}$$

where ρ_{stl} is the saturated ST liquid density, η_{st} is the critical pressure ratio and A_{stf} is the available flow area (along the seam). The flashing duration, DUR_{flash}, is conservatively calculated assuming that the entire water inventory of ST is discharged:

$$DUR_{flashl} = \frac{M_{st}}{w_{stf}} . \tag{29}$$

To calculate the amount of liquid, $M_{wflashl}$, and steam, $M_{sflashl}$, discharged into containment at atmospheric pressure during flashing of the ST water inventory, a two-phase flashing enthalpy, H_{flashl}, corresponding to the saturated liquid enthalpy at P_{st} is considered. The quality, χ_{cl}, is given by:

$$\begin{aligned} \chi_{cl} &= \frac{(H_{flashl} - H_f)}{H_{fg}} , \\ M_{wflashl} &= (1 - \chi_{cl}) M_{st} , \\ M_{sflashl} &= \chi_{cl} M_{st} . \end{aligned} \tag{30}$$

During flashing, Equation 2 is applied to the corium (i=1, j=6) and the CV wall (i=6, j=1 and 7). The power source is given by Equation 25.

The heat transfer coefficient, h_{16k}, is the equivalent heat transfer coefficient due to natural convection and radiation between the corium and the CV wall. The ST water temperature is maintained constant at the saturation temperature corresponding to the ST failure pressure.

2.4.1.3 Stage 8a

After the ST water inventory is depleted, the CV wall heats up faster. 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. Equation 2 is applied to the corium (i=1, j=6) and the CV wall (i=6, j=1). The power source is given by Equation 25.

The duration of this stage, $DUR_{cvfailure}$, 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.

2.4.1.4 Stage 9a

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 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, DUR_{quench}, is conservatively assumed to be zero.

It is also assumed that the corium at temperature, T_1, calculated at the end of the previous stage, is quenched instantaneously and all of its stored heat is used to produce steam of mass $M_{squench}$, where,

$$M_{squench} = \frac{M_{cor}\ \Delta H_{cor}}{h_{fg}}, \tag{31}$$

here, M_{cor}, is the corium mass and ΔH_{cor} is the difference of corium enthalpy at temperature, T_1 and 100°C. For case a, the duration of CDS3 as well as the duration and amount of steaming in this CDS are given by:

$$\begin{gathered} DUR_{cds3} = DUR_{heatup} + DUR_{flashl} + DUR_{cvfailure}, \\ DUR_{steamcds3} = DUR_{flashl}, \\ STEAM_{cds3} = M_{sflashl} + M_{squench}. \end{gathered} \tag{32}$$

2.4.2 Calandria Vessel Failure Prior to Shield Tank Failure

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.

2.4.2.1 Stage 6b

In this stage, 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. The governing equations are the same as those governing stage 5, starting at t_{cds3} and the calculations end when the CV fails, with the corium melting through the CV wall. The duration of stage 6b is called DUR_{heatup}.

2.4.2.2 Stage 7b

When the CV fails, the corium is assumed to be displaced into the ST and quenched by water instantaneously. Therefore, the duration of stage 7b, DUR_{quench} is zero and the ST water temperature is increased instantaneously by ΔT_7, given by:

$$\Delta T_7 = \frac{M_{cor}\,\Delta H_{cor}}{M_{st}C_{p7}}. \tag{33}$$

where ΔH_{cor} is the difference of corium enthalpy at T_1 and T_7, at the time of the CV failure. Since this corium heat is assumed to exclusively be used in raising the ST water temperature, no steam is produced.

2.4.2.3 Stage 8b

In this stage the decay power is heating up the ST water. Equation 2 is applied to the ST water (i=7, j=9) and ST wall (i=9, j=7 and 10). The input power is given by:

$$\begin{aligned} Q_7 &= Q(t), \\ Q_i &= 0 \quad for\ i \neq 7. \end{aligned} \tag{34}$$

The duration of this stage, $DUR_{stfailure}$, 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.

2.4.2.4 Stage 9b

At the beginning of this stage, the ST is assumed to fail and its water inventory is flashed into containment. Equations 28 to 30 are used to compute the characteristics of the flashing period.

The duration of CDS3 as well as the duration and amount of steaming in this CDS are given by:

$$\begin{aligned} DUR_{cds3} &= DUR_{heatup} + DUR_{stfailure} + DUR_{flash1}, \\ DUR_{steamcds3} &= DUR_{flash1}, \\ STEAM_{cds3} &= M_{sflash1}. \end{aligned} \tag{35}$$

2.5 Core Damage State #4

At the end of CDS3, the CV and 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 a pour of a pool of molten corium, but a displacement of solid materials followed by gradual liquification of residual solids in the CV and ST and a gradual displacement of the slurry out of them.

The CDS4 starts at time t_{cds4} given by:

$$t_{cds4} = t_{cds3} + DUR_{cds3}. \tag{36}$$

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, $M_{wblowdown}$, water discharged from the moderator during the flashing following the rupture of the CV discs, M_{wflash}, moderator discharged through the annulus gas bellows (if applied), M_{wdrain}, water discharged from the ST after its failure, $M_{wflash1}$, and ECI inventory, M_{weci}, if it was injected. The total mass of water, M_w, is given by:

$$M_w = M_{wblowdown} + M_{wflash} + M_{wdrain} + M_{wflashl} + M_{weci} \,. \tag{37}$$

Of this total amount of water, two quantities of water inventory should be subtracted. The first amount, M_{sump}, 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, $M_{squench}$. 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 steaming rate is given by:

$$w_s = \frac{Q(t)}{H_{fg}} \,, \tag{38}$$

where Q(t) is the decay power and H_{fg} the vaporization heat of water at atmospheric pressure. The duration of CDS4, DUR_{cds4}, is thus equal to the time required to boil off the residual water on the floor of the FMD:

$$M_w - M_{sump} - M_{squench} = \int_{DUR_{steam}} w_s \, dt \,. \tag{39}$$

The duration of CDS4 as well as the duration and amount of steaming in this state are:

$$\begin{aligned} DUR_{cds4} &= DUR_{steam} \,, \\ DUR_{steamcds4} &= DUR_{steam} \,, \\ STEAM_{cds4} &= M_w - M_{sump} - M_{squench} \,. \end{aligned} \tag{40}$$

The calculation for duration of steaming presented in Equation 40 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 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.

3. RESULTS AND DISCUSSIONS

A severe accident is a combination of an initiating event and one or more process and system failure(s). In general, the following accident events are explicitly analyzed: large break LOCAs, small break LOCAs, reactivity initiated accidents (RIAs), such as large break LOCA with loss of shutdown systems (LSDS) and medium and slow loss of reactivity regulation (LORR), containment bypass events (includes pump seal breaks, feedwater line breaks, steam generator tube ruptures and ECI blowback) and a station blackout event. Note that for any accident scenario to be classified as "severe" implies that the 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 generally not credited even though there may be clear indications and sufficient time for intervention in order to halt the accident progression.

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 1. 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 shown as a function of time in Figure 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 heats 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.

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).

An example of the results of the large break LOCA accident scenario (for the Bruce NGS B design) is presented in the following paragraphs in order to illustrate predictions of the timing of the accident progression and tracking of the water inventory, both key parameters in the determination of the transient source term. Note that the large LOCA event must be coincident with failures of the ECI and/or ECR and moderator systems with no operator intervention to be able to lead to core disassembly.

Figure 2 shows the transient temperature and pressure response of the moderator and ST to the large LOCA after blowdown. In this case, the blowdown duration is about 0.1 hrs and results in 128.4 Mg and 105.8 Mg of discharge of water and steam, respectively, into containment. Initially, during CDS1 (duration of 0.6 hrs), most of the available decay power heats up the fuel to 1200 °C and this takes about 0.2 hrs. Once this temperature is reached, all available heat is transferred to the moderator as indicated in Figure 2 during the next 0.4 hrs. Heat transfer to the moderator continues until the saturation temperature corresponding to the rupture disc pressure is reached. At this point (approximately 0.6 hours following blowdown), CDS2 starts and some of the moderator inventory flashes and the moderator drops to near atmospheric pressure. The four CV rupture discs are conservatively assumed to open. The flashing process takes only one minute to complete. After flashing, the moderator inventory enters a long period of boiling off. Once this inventory is discharged, all channels begin collapsing to the bottom of the calandria vessel. CDS3 starts at the end of the channel collapsing period. For the large LOCA event, this occurs at about 4 hours after blowdown (clearly indicated by Figure 3). Starting from this time, the temperature of the ST inventory begins to increase significantly until it reaches the saturation temperature corresponding to the ST initial pressure, then the ST pressure begins to increase significantly following the saturation pressure of the ST temperature. Once the ST failure pressure limit is reached at about 13 hours following blowdown, the ST mechanically fails (see Figure 3) releasing all its inventory into containment (some flashing with the remaining liquid being dumped). Since the ST is assumed to fail at the bottom of the vessel, the pressure limit, which is calculated at the top of the vessel, is adjusted accordingly (i.e., the static head is accounted for). After the ST failure, the system drops to atmospheric conditions. Figure 3 tracks the liquid inventories of the moderator and ST. This figure shows the time of ST failure and the time when the moderator rupture disc perforates as well as illustrating the gradual boiling off of the remaining moderator inventory.

The temperature response of the corium (i.e., the bulk mass comprising the fuel, fuel channel and other in-core structures), CV and ST structures is shown in Figure 4. The initial heat up (to 1200 °C) of the corium is very short (about 0.2 hrs). It remains at a constant temperature during the period of moderator heat up and boil off. Once the moderator boil off ends (at approximately 4 hours following blowdown), the corium settles at the bottom of the calandria vessel and rapid heat up of the vessel wall begins. At this time, heat transfer from the CV to the ST water and ultimately the ST wall starts to become significant for about 9 hours. The corium and calandria vessel enters a period of cooldown until the ST is predicted to fail at approximately 13 hours after blowdown. Once this occurs, the large heat sink of the ST inventory is no longer available and both the corium and CV rapidly begin heating up again until the CV failure (melt-through of the corium) at about 15.5 hours following blowdown, at which point the corium drops onto the containment floor (assumed to be the fuelling machine duct floor) and immediately gets quenched by the ST inventory that was discharged due to the earlier ST failure. Since there is a large amount of water to keep the corium quenched, a long period of boiling off of the residual ST inventory begins (CDS4). Of note is that condensation of any generated steam is conservatively ignored in the timing calculation of severe accidents.

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. In this plot five different curves are presented to provide a direct comparison of various postulated accident scenarios. The times indicated on the plots are blowdown times and have the following correspondence: 0.1 hrs for the large LOCA, 0.91 hrs for small LOCA with 270 kg/s initial discharge, 5.34 hrs for multiple steam generator tube ruptures, 10 hrs for small LOCA with 40 kg/s initial discharge and 64.4 hrs for a single steam generator tube rupture. Again, note that these inventories are a function of time from blowdown such that the initial inventories are a result of the blowdown discharges. 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. The behaviour of the various accident scenarios are not noticeably different and Figures 5 and 6 help to illustrate that the timing predictions characterize the differences in various postulated accident scenarios. As the blowdown time increases, the periods of heating up and boiling off become longer. For all severe accidents analyzed for the Bruce NGS B design, shield tank failure was predicted to occur before calandria vessel failure.

The results of the timing and steam discharge calculations may be used in the source term estimates out of containment. Generally, earlier severe accident progressions and higher steam discharge rates lead 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 in the source term assessment to determine their effects, if applicable. The resultant source terms are used in determining the event consequence for a broad variety of severe accident scenarios. The product of the event consequence with the event frequency determines the risk associated with any particular event. By quantifying the risk of a number of events, a broad based categorization may be formed, leading to the identification of the dominant contributors to risk and indicating where risk can be reduced.

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-mechanical response of the reactor and other relevant systems. Details of the calculations involved in each stage of the severe accident progression are presented (relevant for a Bruce or Darlington type design). The results of a large break LOCA scenario for Bruce NGS B is also presented as an example of results obtained from application of this methodology. The presented results mainly focus on the pressure and temperature responses of various relevant reactor components. The postulated severe accidents analyzed mainly differ in the timing to reach and progress through each core damage state.

Ultimately, these calculations may be 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 results of the source term calculations leads to a broad based categorization of various postulated accident events and therefore a classification of the consequence associated with each particular categorization. The combination of probability and consequence of any particular event produces the risk associated with that event. Identification of the various risks associated to any accident sequence is the first step in attempting to reduce the risk of severe accidents in the future.

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TABLE 1: 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

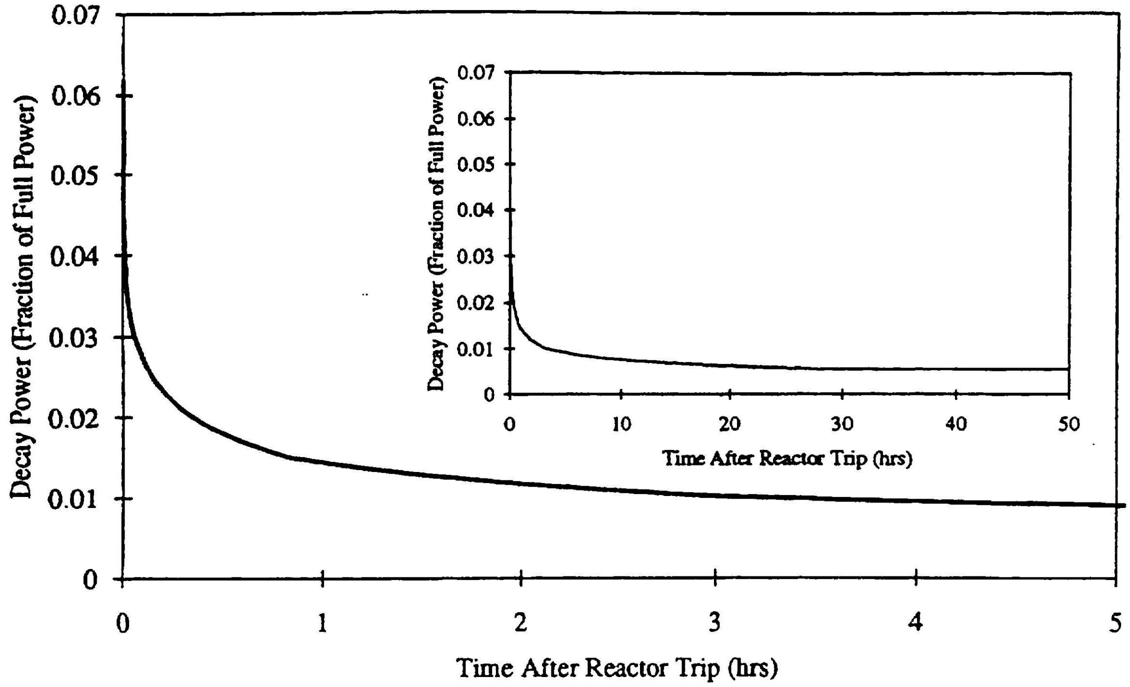

Figure 1
Decay Power Transient

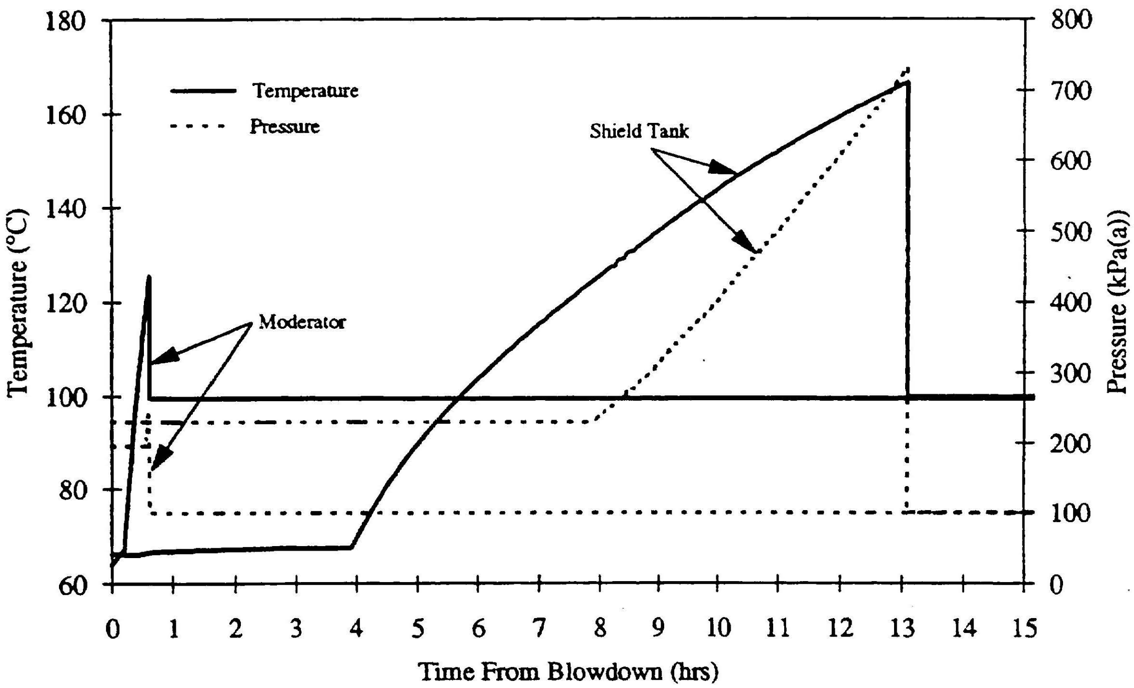

Figure 2
Moderator and Shield Tank Bulk System Response Following a Large Break LOCA

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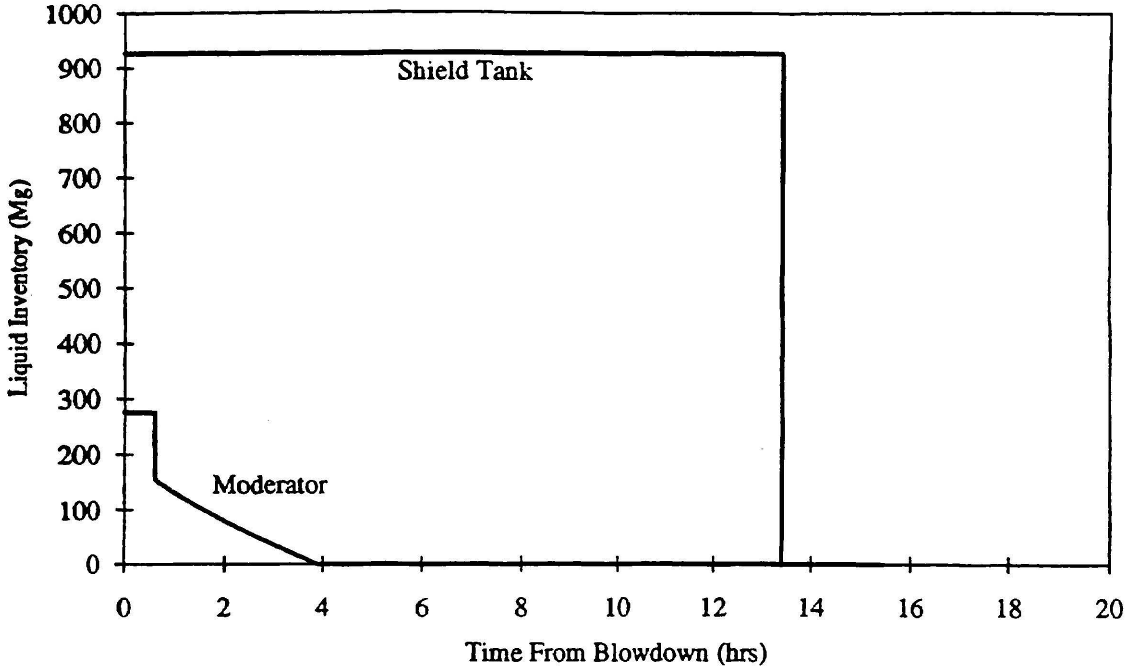

Figure 3
Liquid Inventory of Moderator and Shield Tank

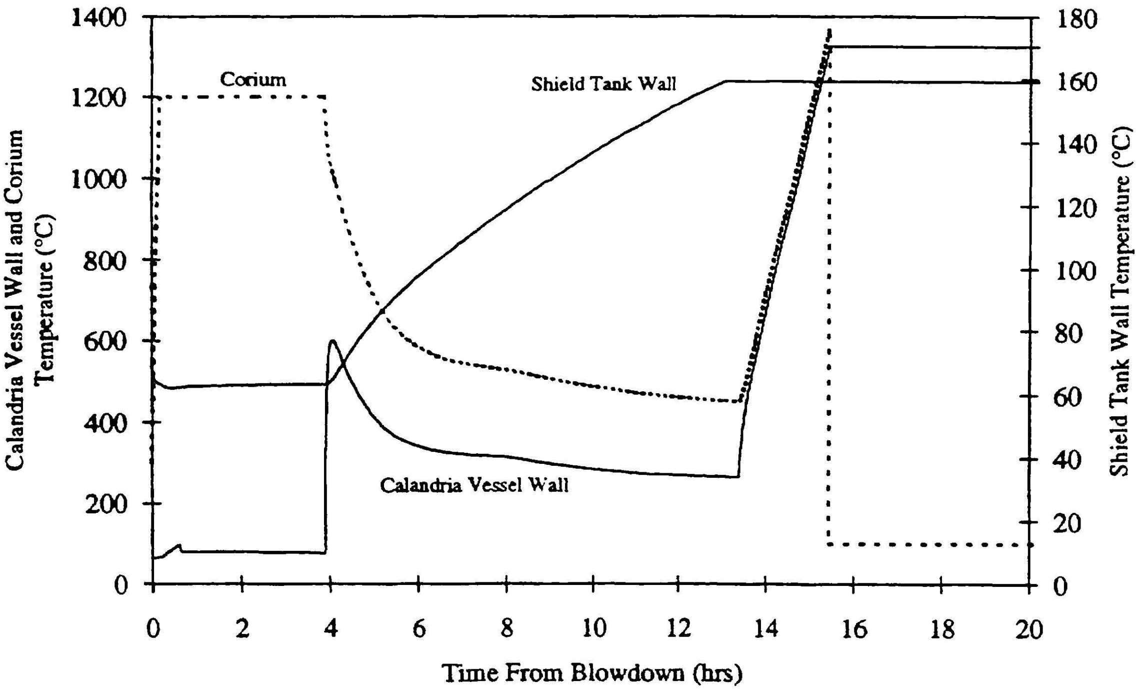

Figure 4
Calandria Vessel, Shield Tank, and Corium Temperature Response Following a Large Break LOCA

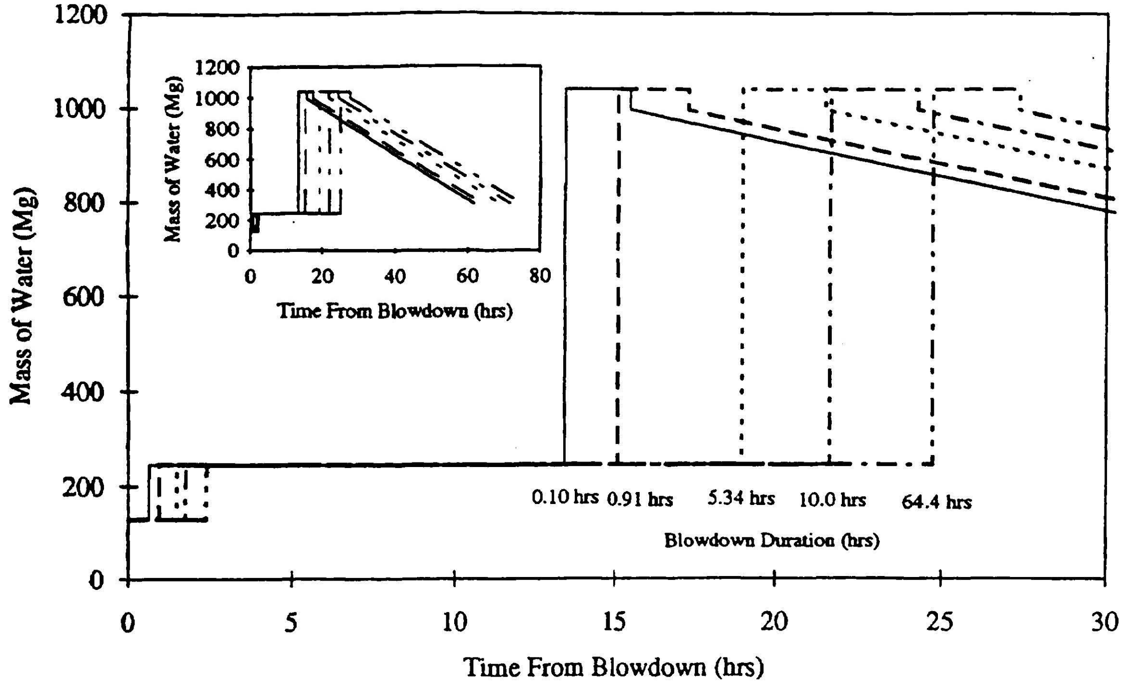

Figure 5

Liquid Inventory in Containment

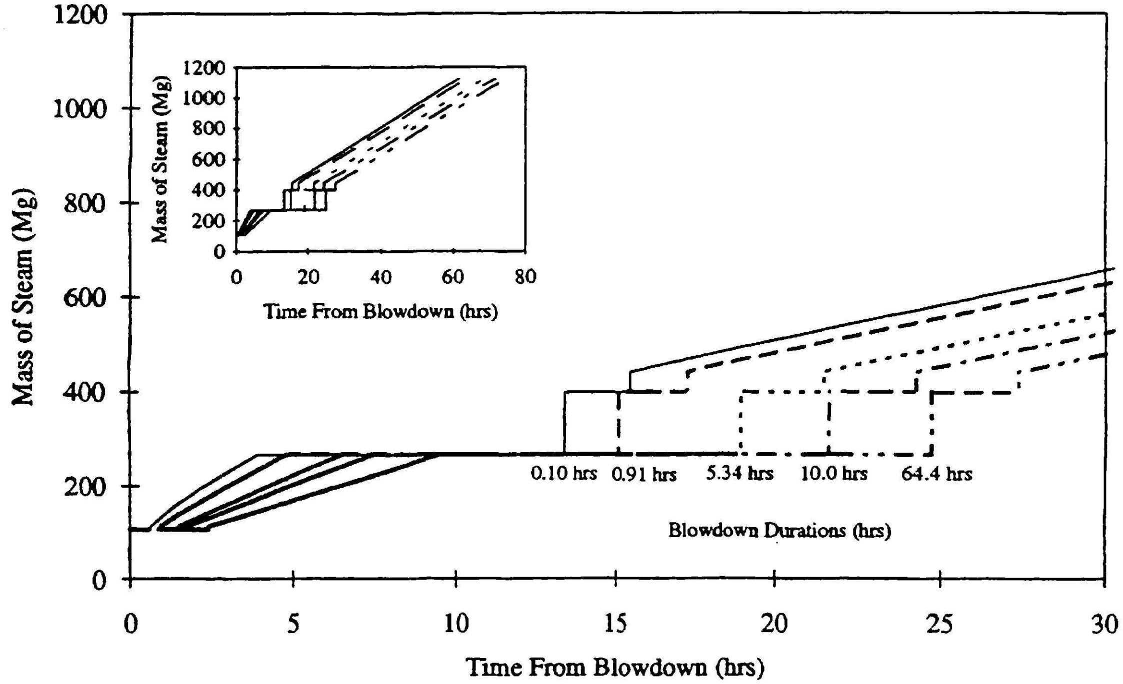

Figure 6

Steam Inventory in Containment

FAILURE MODE AND EFFECT ANALYSIS EXPERIMENTAL RELIABILITY DETERMINATION FOR THE CANDU REACTOR EQUIPMENTS.

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Abstract

This paper describes the experimental tests performed in order to prove the reliability parameters for certain equipments manufactured in INR Pitesti, for NPP Cernavoda. The tests were provided by Technical Specifications and test procedures. A comparison, referring to the reliability parameters, between Canadian equipments and INR manufactured equipments ones is also given. The results of tests and the conclusions are shown..

1. INTRODUCTION

The information regarding the reliability of products are obtained, in principal, by following the behavior during the real operation or during the laboratory tests.

Each of these 2 ways presents advantages and limitations, and in case of real operation, all the phenomena appeared during product operating are recorded.

At the moment of the conclusion formulation these may present just a historical significance, the purpose of the reliability tests being used to improve a performance level of the current manufacturing. To these limitations of the methods of the real operation, are added the difficulties connected to accurate acquisition of data or deficiencies of the informational system.

Without excluding this method which present a lot of disadvantages it is necessary to use the method of the laboratory tests where the samples are operating in certain conditions close to the real ones, in the NPP, being necessary the existence of special testing devices and qualified personnel. During the tests a systematic record is necessary which will stay at the final decision of the test, such as:

- the time (beginning of the test, the occurrence of failures, etc.).
- details on the stress and environment conditions.

The main reliability parameter considered was the MTBF and the assumed theoretical reliability law was exponential.

A special attention has been granted to the accelerated tests, where stress level applied to the components is above the level established by reference condition (stated by design).

2. FAILURE MODE AND EFFECT ANALYZE (FMEA)

During laboratory tests (type and production) the failures which occurred have been noted.

For functional tests we got no failures. Environment tests were carried out in two stages:

- burn-in test (to cover the initial failure period of "bath-tube" curve) - see fig. no. 1;
- accelerated thermal aging tests (to demonstrate that the life period for equipments which are intended to be used in NPP at SDS#1 must be 30 years).

The analyzed reactor instrumentations were :

- ***Dynamic Signal Compensation Module (DSCM);***
- ***Trip Test/Alarm Control/Buffer Amplifier (TT/AC/BA);***

During the laboratory tests there were analyzed, through the FMEA (Failure Mode and Effect Analyze) method the nuclear instrumentations included in SDS#1 (Shutdown system no.1), manufactured in INR Pitesti, for NPP CANDU Cernavoda (see fig.2) :

Burn-in test was performed for these instrumentations for a period of 240 hours, divided into 10 cycles of 24 hours each, and the following failures were recorded :

for DSCM

- *2 due to the fuse (short);*
- *2 due to the input cable (for the input signal) - short;*

for TT / AC / BA

- *1 due to the resistor incorporated in the supply part of the module;*
- *1 due to the fuse;*

As a consequence, the decision was to replace the fuse type, cable type and the resistor.

During accelerated thermal aging test we did not observe any failures. This test has been carried out in the following conditions: **14 cycles of 96 hours each at 65°C and RH=30÷40%.** After every cycle, the temperature was decreased at 55°C, 1 hour, for reading the parameters. The test is based on Arrhenius's law, and the duration of the test was calculated taking into consideration the specific activation energies for every component belonging to DSCM and TT/AC/BA such as : DSCM was exposed 1354 hours and TT/AC/BA 2007 hours, respectively.

The implied failure rate is less than .45284 $F*10^{-6}*h^{-1}$ (allowed $.5*10^{-6}*h^{-1}$).

The estimation of the relative probabilities of the modes, as percentage, are presented in Table no.1.

Reliability actions taken as a consequence of these failures were:

- use of redundancy circuits for module supply system. As a result, the reliability increases as shown in Table no. 2.
- use of the specific screen to detect (eventually) the unreliable components when, particularly, failures modes or mechanism were known or suspected to be present

3. EXPERIMENTAL RELIABILITY DETERMINATION REACTOR INSTRUMENTATION FOR THE NUCLEAR, RELIABILITY TESTS

During carrying-out the laboratory tests for the tested equipments, no failures were recorded. In this situation, the experimental reliability parameters were calculated based on the parametric reliability model, following the behavior of certain parameters vs. time, for every equipment taken into consideration, in the testing intervals. The failures occurred during "burn-in" tests were not considered because were repaired, in time, and they were not repeated during reliability tests. The following parameters were measured: U0 (output voltage) for **DSCM, the report neutron flux rate vs. neutron flux for FFLS, volumic activities for HTSRM.** It was stated that the sum of the determined errors can be assimilated with a straight line, therefore the gaussian character of the repartition was assumed. To establish the defect fractions, the following steps were considered :

- the ranking of the calculated errors values, from minimum to maximum;
- the determination for every "i" values of the repartition function,Yi=(i-1/2)X100/N, where N is the no. of the repeating measurements and " I " is the ranking number;
- graphical representation of the pair values "mi, Yi ";
- graphical representation of the envelope curve errors, for every interval;
- the determination of the defect fraction in every interval;

The determined defect fractions were transferred in the Weibull probability paper, and using the graphic formula, β, η and γ factors were determined. To calculate the experimental reliability parameters the following relations were used :

Reliability, R(t), was calculated with the formula:

$$R(t) = \exp[-(\frac{t-\gamma}{\eta})^{\beta}] \qquad (1)$$

where β, η, γ are the above mentioned parameters graphical determined;

MTBF was calculated with the formula:

$$MTBF = \gamma + \eta \cdot \Gamma(\frac{1}{\beta}+1) \qquad (2)$$

where $\Gamma(\frac{1}{\beta}+1)$ is the Euler function, first type.

The hazard rate, z(t), was determined with formula :

$$z(t) = \frac{\beta}{\eta} \cdot (\frac{t-\gamma}{\eta})^{\beta-1} \qquad (3)$$

where: β, η, γ, are specific parameters which can be determined in a graphic mode.

Practical example :

For **HTSRM** the following parameters were determined, as follows :

β = 3.2 ; η = 1,900 ; γ = 0 ; Γ = 0.896

$$MTBF = \gamma + \eta . \Gamma(1+ \frac{1}{\beta}) = 0 + 1{,}900 * 0.896*(1+ 0.325) = 1702.4 * 1.03125 = 1756 \text{ h}$$

$\Sigma\lambda = 569.7$ F / 10^6 h

The activity carried out was focused on experimental reliability determination for the nuclear reactor instrumentation, manufactured in INR Pitesti, as follows:

- **Failed Fuel Location System (FFLS);**
- **Heat Transfer System Radioactivity Monitor (HTSRM);**
- **Dynamic Signal Compensation Module (DSCM);**

The reliability tests were performed in two steps:

- **qualification tests;**
- **accelerated tests;**

These tests were performed in accordance with the TS for every equipment.

3.1 Experimental Reliability Tests for FFLS

In a **CANDU** reactor the purpose of the **FFLS** is to locate and to find in that channel what particular bundle pair is failed. To do so, D2O samples from each channel are sequentially monitored to detect a comparatively high level of delayed neutron activity. Qualification tests last about 1,268 Hrs. and the intensive tests, last about 496 Hrs. in 12 cycles. Intensive tests were done for approx. 496 Hrs. within 12 individual cycles, each cycle consisting of 7 automatic scannings. During these scannings the following failures were noticed:

- *1 failure due to locking pin (failure to function);*
- *1 failure due to error in positioning of carriage (failure to remain in position).*

These events were not the result of an electronic component failure, and were eliminated by increasing the hysteresis of the discriminator that treats transducer head signals, and by providing a constant force on the locking pin coil. These were done by changing a resistor value in the feedback loop of the discriminator. The performance of the reliability test (as shown above) on the FFLS equipment was in accordance with the specific reliability procedure prior approved by Canadian part. The sum of these failure is less than **.1 $F*10^{-6}*h^{-1}$**, and not affects significantly the MTBF value, and also the operation of the tested equipment.

Total test time : 2,016 Hrs. **No. of failures: 2**

The experimental reliability parameters thus obtained (for non-parametric errors) are given in Table 3.

3.2 Reliability Tests for HTSRM

HTSRM is a complex dozimetric equipment which check the state of the fuel from CANDU reactor by monitoring the fission products. Is an equipment complementary (as function) with FFLS and manufactured to measure the activity of 4 radionuclides (**Kr-88, Xe-133, Xe-135, I-131)**, characteristic for PHWR CANDU. The reliability test was developed on the first sample of product by operation in laboratory conditions for 1,000 Hrs. During the test we noticed no failures, and because of that, processed the parametric defects values, calculating the relative errors on the measured volumic activities, by using Ba-133 source, consisting of 4 pairs of vessels in increasing order of decades. The displayed values for volumic activities, the activities in the currents and the detector-generated impulses constituted the data bank necessary for reliability performances calculation. Experimental reliability parameters obtained from the reliability tests allowed the pointing out of the reliability performances of HTSRM product.

The experimental reliability parameters for HTSRM are given in Table 4.

3.3 Experimental Reliability Tests for DSCM

The purpose of DSCM module is to eliminate the time constants of **30s** and **2,500 s** which appear within output signal of the Self - Powered Platinum Detector from SDS #1, specific for a PHWR CANDU Reactor Type. This module is associated with the TT/AC/BA(see also fig.2).

Reliability tests for DSCM were performed in two steps:

- **qualification of tests** for a period of 200 Hrs.
- **accelerated tests**, for a period of **1,400 Hr.**, 4 devices (samples) being exposed in the following conditions: temperature **55°C, RH =80 %.** During these tests, the recorded parameter was U0 (output voltage). No failures have been recorded.

The evolution in time of the parametric error corresponding to U0 parameter was followed.

The experimentally reliability parameters are shown in

Table 5.

As a conclusion, the experimental reliability parameters are superior to those calculated by provisional standard methods.

4. IMPROVING THE RELIABILITY OF COMPONENTS AND SYSTEMS MANUFACTURED IN ROMANIA FOR NPP CERNAVODA.

The following have been taken into consideration:

- improving the reliability of components and systems manufactured based on experimental test program.
- comparison of the generic data base being used for the PSA of NPP Cernavoda with the specific results of components reliability experimental results.
- the use of short test duration to estimate the performance characteristics of components with respect to actual mission time.

The improvements are as follows :

4.1 For FFLS Equipment the following reliability improvements have been done:

- modifications of software, in certain parts, to assure a more correct operation of the "watch-dog" and for a better communication between those two microcomputers belonging to the FFLS;
- improvement of the optical decodification system for carriage positioning (increased accuracy of positioning);

4.2 For HTSRM the following improvements were performed:

- modification of software in the acquisition system in certain subroutines ;
- replacement of the step by step motors through direct-current motors for a better accuracy of the collimator positioning and a better operation of the equipment.

4.3 For DSCM module the following reliability improvements measures were performed :

- use of redundant power supply circuits;
- use of the components with a reduced stress factor;
- replacement of the fuses with a new type, more reliable;

These reliability improvements measures, together with others taken during the manufacture of the above equipment (use of military components with a lower failure rate, use of 100% screening before mounting, use of "burn-in" for electronic components and for the equipment), have contributed to the reliability parameters increasing, as shown in the comparative Table 6.

5. COMPARISON OF THE GENERIC DATA BASE BEING USED FOR THE PSA OF NPP CERNAVODA WITH THE SPECIFIC RESULTS OF COMPONENTS RELIABILITY PARAMETERS EXPERIMENTAL RESULTS

A comparison of the generic data base used for our PSA analysis for NPP Cernavoda, with the specific experimental reliability results, are shown in Table 7.

These data are used in PSA Analysis for NPP Cernavoda for Shutdown System no.1 (SDS#1), together with other generic data.

6. USE OF SHORT TEST DURATION TO ESTIMATE THE PERFORMANCE CHARACTERISTICS OF COMPONENTS WITH RESPECT TO ACTUAL MISSION TIME.

The main reason to estimate the performance characteristics of the equipment with respect to actual mission time (30 years) by use of short test duration was to prove that such nuclear reactor equipment will be able to operate in safe conditions up to the last day of their lifetime period.

The accelerated thermal aging(for DSCM) and intensive tests (for FFLS and HTSRM) were performed for the above mentioned equipment, separately.

Procedures for intensive tests were prior approved by Canadian experts and the background for accelerated thermal aging has been based on Arrhenius law. The duration of test necessary to aging the electronic module for ca. 30 years at 23.5°C was calculated with respect to the specific activation energies of every electronic component belonging to the equipment.

7. CONCLUSIONS

Reliability data are very important in the NPP Safety, particularly in PSA. We have not yet at our disposal the plant specific reliability data but only a few from the operating of some Canadian NPP, as far as I know. In the absence of the reliability data from operational experience,) we made the above mentioned laboratory tests. These tests have the advantage that the failure mechanism can be more easily identified, and testing can be accelerated. The reliability data such obtained were included in INR Reliability Data Bank, to be used for our PSA analysis. The results obtained ascertain that the tested equipments are reliable, in accordance with TS.

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FIGURES AND TABLES

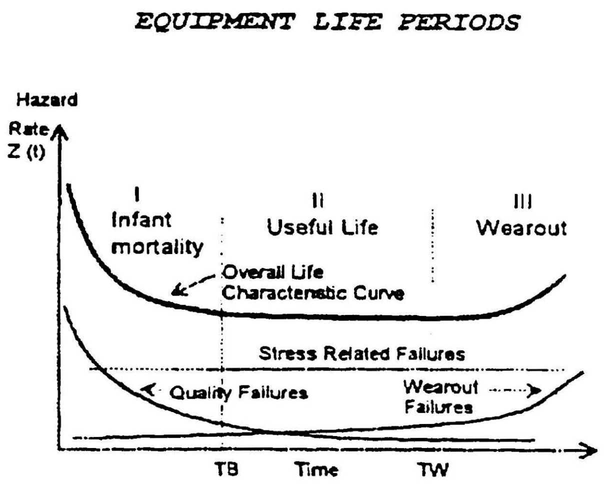

FIG. 1 Components of failure

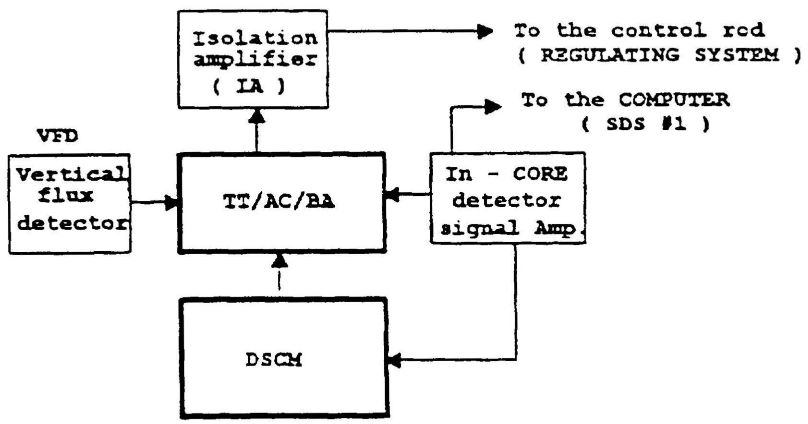

FIG. 2 Connection between DSCM and TT/AC/BA
and other components of SDS #1 (partial) NPP CANDU

Table 1

Mode	Est. (%) of total failures*	Remarks
I	57	Fuse
II	29	Input cable
III	14	Resistance

Table 2

	Reliability	
	no redundancy	with redundancy circuit
DSCM	.8843	.950784
TT/AC/BA	.8568	.98273257

Table 3

	Reliability	
	Provided in TS	Experimental
MTBF (Hours)	.936	1,008
R (10 Hr.)	.9836	.990
R (24 Hr.)	.974	.976
R (168 Hr.)	.835	.846

MTR = 2 h

Table 4

	Provided in TS	Experimental
MTBF (Hours)	>1,600	1,756
R(24 hr.)	.987	.99
R(600 hr.)	.721	.97
R(1,200 hr.)	.52	.79

MTR = 8 h

Table 5

	Provided in TS	Experimental
MTBF (Hours)	81,500	142,572
R(1,000 hr.)	.987	.993
R(10,000 hr.	.885	.931
A (availability)	99.96	99.97

MTR = 4 h

Table 6

Equipment	FFLS	HTSRM	DSCM
A. Failure rate (F/10^6*h)			
- estimated	1,068.37	625.0	12.171
- experimental	992.06	87.4	7.014
B. Reliability (R) for 10 Hr.:			
- estimated	.9893	*	*
- experimental	.990	*	*
for 24 Hr.:			
- estimated	.974	.987	*
- experimental	976	.99	*
for 168 Hr.:			
- estimated	.835	*	*
- experimental	.846	*	*
for 600 Hr.:			
- estimated	*	.721	*
experimental	*	.97	*
for 1,200Hr.			
- estimated	*	.52	*
- experimental	*	.79	*
for 1,000 Hr.			
-estimated	*	*	.987
-experimental	*	*	.993
for 10,000 Hr.			
- estimated	*	*	.885
- experimental	*	*	.931
MTBF (HR.)			
- estimated	936	>1,600	81,500
- experimental	1,008	1,756	142,572
Availability(A)			
- estimated	.9978	.995	.9996
- experimental	.998	.9953	.9997

* - Not calculated for that interval.

Table 7

Equipment	MTBF (Hr.) *			Reliability **		
	TS	generic	exper.	TS	generic.	exper.
DSCM	100,000	81,500	142,572	.990	.987	.993
FFLS	*	936	1,008	*	.974	.976
HTSRM	*	>1,600	1,756	*	.721	.970

* Not available from Canadian sources.
** See also Diagrams no 3 and 4