Large-LOCA Calculations in CANDU

by
B. Rouben

In this Session we will look at typical CANDU calculations for large-loss-of-coolant accidents.

1. Introduction

A large loss of coolant (LLOCA) is one caused by the rupture of a large pipe such as a Reactor Inlet Header (RIH), Reactor Outlet Header (ROH), or Pump-Suction pipe (see Figure 1.

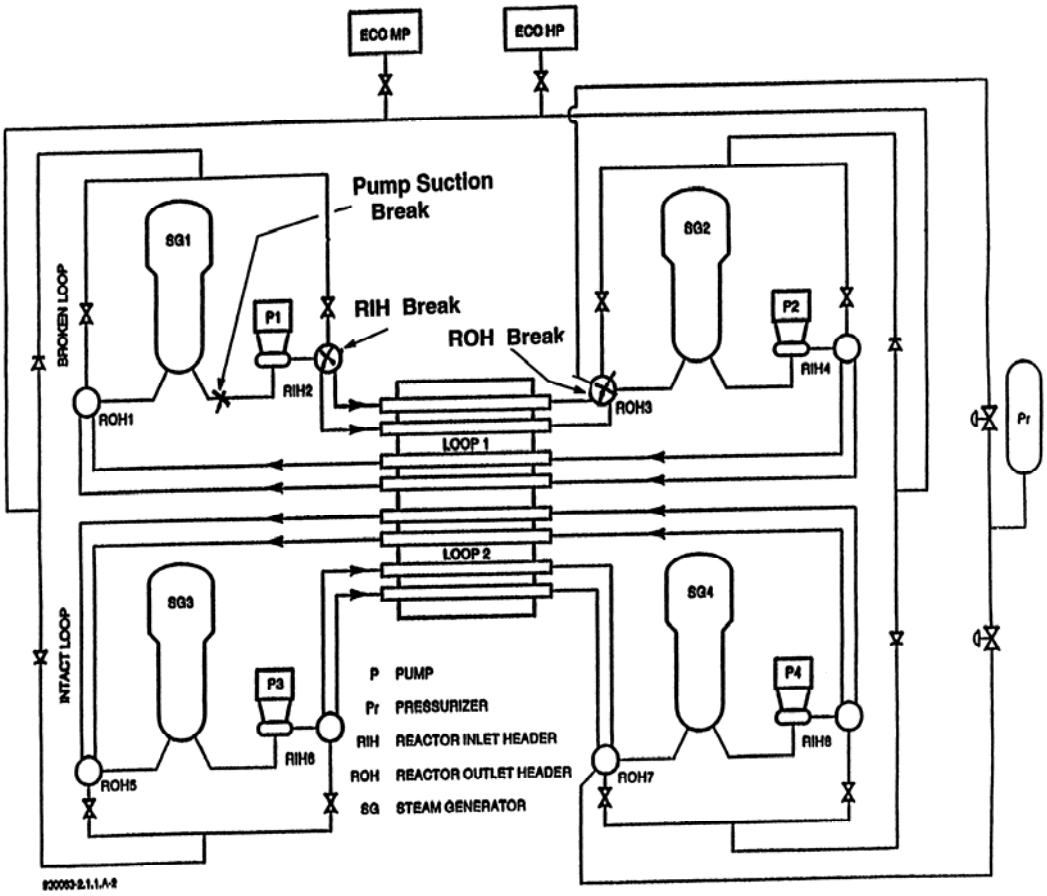


Figure 1 – Examples of Break Locations Giving Rise to Large LOCA

We have seen that the coolant-void reactivity is positive in CANDU. Therefore, as the coolant is lost, a positive reactivity insertion will materialize. The complete removal of the coolant may correspond to a void reactivity in the range of +15-20 milli-k. However, practical considerations (and designed subdivision of the heat-transport-system in some reactors) limit the amount of coolant loss that is possible in the first few seconds after the break. Therefore, even a large-pipe break will insert only of the order of 5-6 milli-k of positive reactivity in the first half s to 1 s.

This reactivity insertion is still sufficiently large to be beyond the capability of the Reactor Regulating System to control, and will therefore lead to a power pulse. One or other of the shutdown systems must be actuated very promptly (within about 1 second) to quickly terminate the transient. In fact, the LLOCA is the accident which presents the greatest challenge to CANDU shutdown systems in terms of the rate of positive reactivity insertion.

Neutron kinetics is the study of the evolution in time of the neutron flux and of the reactor power in transients. The term is often used in the narrow sense of **fast** kinetics, where large changes in power occur over intervals of seconds. This applies to fast neutronic transients of the type initiated by a LLOCA. Here, delayed-neutron effects play an extremely important role, and must be taken into account properly.

(Note: Neutron kinetics also embraces a slower type of transient, associated with the effects of the spatial and temporal redistribution of fission products such as I-135 and Xe-135 or similar pairs. These transients play out over time scales of the order of minutes, hours, or longer, and are not usually analysed with the same codes used for fast kinetics.)

2. Kinetics Methods

A simplistic treatment, such as assuming the power will evolve in time exponentially with reactivity, on a time scale obtained from some average of the prompt-neutron lifetime and delayed-neutron-precursor half-lives, is not adequate. A rigorous approach demands a proper mathematical treatment of delayed-neutron effects, i.e., of the rates of production of prompt neutrons and of production and decay of delayed-neutron precursors. Several approaches exist.

2.1 Point Kinetics

Point kinetics has already been covered in another session. In this approach, the reactor core is treated as a single point. The premise is that, predominantly, only the spatially uniform component of the power change need be examined and that spatial variations of the response can be ignored. Point kinetics determines the time variation of the global (average) values of power and of the concentrations of the various delayed-neutron precursors. This variation is then superimposed on the pre-event power shape.

Quantitatively, point kinetics is represented by a simple set of differential equations, which couple the neutron flux ϕ and the delayed-neutron-precursor concentrations C_i:

$$\frac{d\phi(t)}{dt} = \frac{\rho - \beta}{\Lambda}\phi(t) + \sum_{i=1}^{N} \lambda_i C_i(t)$$

(8)

$$\frac{dC_i}{dt} = \frac{\beta_i}{\Lambda}\phi(t) - \lambda_i C_i(t) \qquad i = 1,.., N$$

where

N is the number of delayed-neutron-precursor groups,

β_i and λ_i are the partial delayed-neutron fractions and decay constants for the various precursor groups, and

β is the total delayed-neutron fraction (the sum of the partial delayed fractions).

However, the point-kinetics approximation is inadequate when high accuracy is in demand, such as in modern safety analysis. Its weakness originates in the presence of spatially non-uniform effects. The following illustrates two very important sources of such non-uniformity, which cannot be addressed by point kinetics.

Voiding Transient

The coolant voiding in a large LOCA is certainly not uniform. For instance, in the CANDU 6, the heat transport system is subdivided into two side-by-side loops (see Figure 2), each servicing one half of the cylindrical reactor. The two loops are isolated from one another in a large LOCA. Thus, the break will induce a side-to-side asymmetry in the core coolant density, leading to a side-to-side asymmetry in the ensuing power pulse. Even in reactors without side-by-side heat-transport-system loops, there is often an asymmetry in the voiding transient between the inner core and the core periphery.

A **pre-accident** side-to-side or top-to-bottom asymmetry in the power distribution may also in some cases be postulated. If the initial power is assumed higher in the voiding half, the void reactivity will be enhanced by the higher flux, and the non-uniformity in the power pulse will be further accentuated.

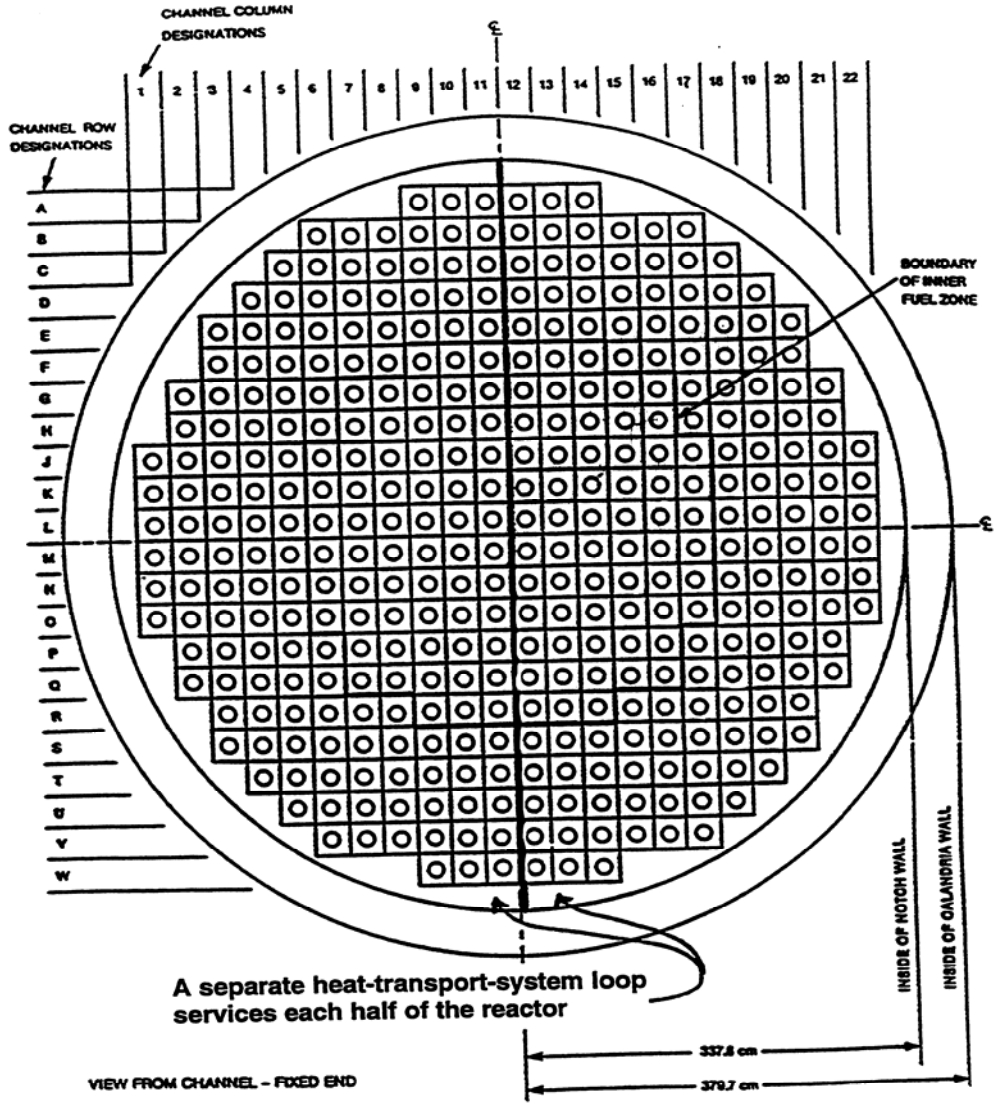


Figure 2 Side-to-Side Heat-Transport-System Loops in CANDU 6

Shutdown-System Coverage

Conservative assumptions are often used in traditional safety analysis. In particular, it may be assumed that not all parts of the shutdown system are operational, e.g., that two of the shutoff rods in SDS-1 (2 of 30 in a Bruce reactor) are not functioning. This results in non-uniformity in the shutdown-system spatial coverage.

2.2 Spatial Kinetics

The previous section illustrates the reasons why the capability to model spatial effects is important. This has led to the development of spatial-kinetics methods, which are used for the detailed analysis of fast transients over time scales of a few seconds. Point-kinetics methods are still useful as a means to continue the analysis over very long times, for instance to provide input to long-term thermalhydraulics simulations.

The development of spatial-kinetics codes for CANDU has yielded two different codes, CERBERUS (now integrated in RFSP-IST as its kinetics module) and SMOKIN.

CERBERUS

CERBERUS solves the finite-difference form of the 3-d, 2-energy-group time-dependent neutron-diffusion equation. The methodology includes the space-and-time-dependent delayed-neutron precursors (in 6 groups). All fission neutrons are assumed born in the fast group (g = 1)

The formulation of the problem is the familiar 2-group time-independent neutron-diffusion equation:

$$\frac{1}{v_1}\frac{\partial \phi_1}{\partial t} = \vec{\nabla} \cdot D_1(\vec{r},t)\,\vec{\nabla}\phi_1(\vec{r},t) - (\Sigma_{a1}(\vec{r},t) + \Sigma_{1\to 2}(\vec{r},t))\,\phi_1(\vec{r},t) + \beta\frac{\nu\Sigma_{f1}(\vec{r},t)\phi_1(\vec{r},t) + \nu\Sigma_{f2}(\vec{r},t)\phi_{f2}(\vec{r},t)}{k_{eff}(0)} + \Sigma_{2\to 1}(\mathbf{\vec{r},t})\phi_2(\mathbf{\vec{r},t}) + \sum_{j=1}^{6}\lambda_j C_j(\vec{r},t)$$

$$\frac{1}{v_2}\frac{\partial \phi_2}{\partial t} = \vec{\nabla} \cdot D_2(\vec{r},t)\,\vec{\nabla}\phi_2(\vec{r},t) - (\Sigma_{a2}(\mathbf{\vec{r},t}) + \Sigma_{2\to 1}(\mathbf{\vec{r},t}))\,\phi_2(\vec{r},t) + \Sigma_{1\to 2}(\mathbf{\vec{r},t})\phi_1(\mathbf{\vec{r},t})$$

where

v_1 and v_2 are the neutron velocities in group 1 and 2,

$\nu\Sigma_{f1}$ *and* $\nu\Sigma_{f2}$ are the fast-group and thermal-group neutron-yield cross sections,

Σ_{a1} *and* Σ_{a2} are the fast-group and thermal-group absorption cross sections,

$\Sigma_{1\to 2}$ is the slowing-down (moderation) cross section,

$\Sigma_{2\to 1}$ is the up-scattering cross section,

D_1 *and* D_2 are the fast-group and thermal-group diffusion coefficients,

C_j (j =1, …, 6) is the concentration of the j-th delayed-neutron precursor,

λ_j is the time constant of the j-th delayed-neutron precursor,

β is the total delayed-neutron fraction, and

$k_{eff}(0)$ is the initial steady-state (pre-transient) reactor multiplication constant, <u>not</u> a variable.

The space-time-dependent delayed-neutron-precursor concentrations satisfy the equations

$$\frac{\partial}{\partial t} C_j(\vec{r},t) = \beta_j(\vec{r})\left(\nu\Sigma_{f1}(\vec{r},t)\phi_1(\vec{r},t) + \nu\Sigma_{f2}(\vec{r},t)\phi_2(\vec{r},t)\right) - \lambda_g C_g(\vec{r},t) \qquad j=1,2$$

where the β_j is the partial delayed-neutron fraction for precursor group j.

Although this system of equations can be solved "directly" by advancing over time in small steps, and at each step solving for the flux shape in the familiar iterative way, this is very onerous when there are tens of thousands of mesh points. Thus, a different solution method is adopted in CERBERUS; this is the Improved Quasi-Static method, which is based on a factorization of the neutron flux into a space-independent amplitude and a space-and-time-dependent shape function:

$$\phi(\vec{r},t) = A(t)\psi(\vec{r},t)$$

with **the major time dependence cast into the amplitude A** by constraining a core integral of the shape function ψ to be constant in time. The diffusion equation can then recast into separate equations for the amplitude, the flux shape, and the precursor concentrations.

The amplitude is coupled to integrals of the delayed-neutron precursors by a set of point-kinetics-like equations, the difference being that parameters in these equations are core-integrated quantities. The equation for the flux-shape function is similar to the time-independent diffusion equation, except that it has additional terms in the amplitude and the precursor concentrations.

The Improved Quasi-Static method allows the problem to be solved in a two-tiered scheme of time intervals: short intervals for the (easy-to-solve) points-kinetics-like equations for the amplitude, and much larger time steps (~50-100 ms) for the flux-shape equation, which requires much greater computational effort (see Figure 3).

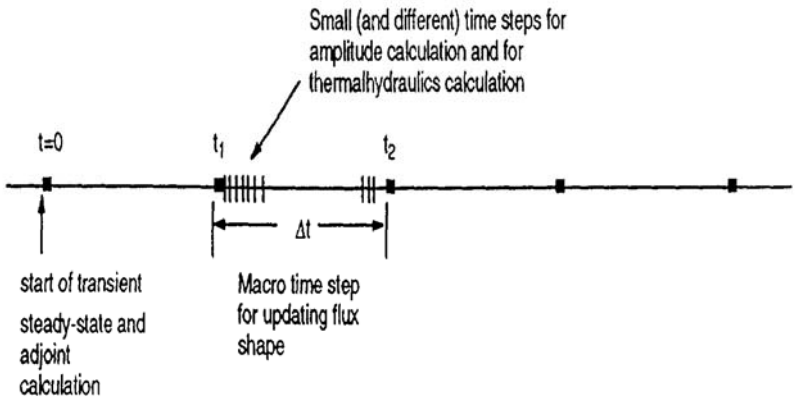


Figure 3 – Tow-Tiered Time-Interval Scheme in CERBERUS

SMOKIN

SMOKIN has been used mostly at Ontario Power Generation and Bruce Power. It is based on a **modal** method, in which the one-energy-group (thermal) neutron flux is expanded in a finite series of time-independent flux modes μ, pre-calculated only once:

$$\phi(\vec{r},t) = \sum_j a_j(t)\mu_j(\vec{r})$$

The modes μ are normally the flux "harmonics" of the time-independent neutron diffusion equation. These are actual "higher-frequency" solutions of the time-independent diffusion equations, identified with higher eigenvalues (smaller, i.e., subcritical k_{eff}) than the steady-state "fundamental" flux solution. They are essentially mathematical functions, featuring regions of negative flux, and are therefore not physically viable flux shapes on their own. Physically, they represent three-dimensional "global" flux components which are expected to be most "excited" (or promoted) by core perturbations. Added on to the "fundamental" flux, they can represent a real flux shape. The flux harmonics are calculated by a code such as RFSP-IST. Some 10 to 20 harmonics can practically be computed. Examples of the harmonic flux shapes obtained are sketched in Figure 4. They represent azimuthal (e.g., side-to-side), radial, and axial flux perturbations of various orders. Note that these modes reflect global shape changes and cannot represent fine details or very localized perturbations.

In SMOKIN, because the flux modes are pre-calculated, the thousands of flux unknowns in the time-dependent neutron diffusion equation are replaced by the few unknown mode amplitudes a_j by re-casting the diffusion equation in the guise of a small number of linear differential equations in the a_j. Because of the very small number of unknowns, the numerical solution of the equations can be computed extremely quickly.

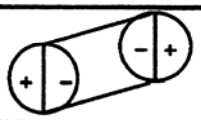
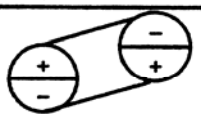
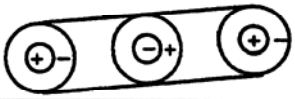
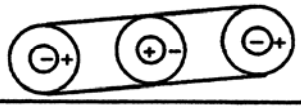
MODE NUMBER	DESIGNATION	SUBCRITICALITY δk	MODE SCHEMATIC (IDEALIZED)
0	Fundamental	0	
1	First Azimuthal-A	16.2	
2	First Azimuthal-B	16.9	
3	First Axial	27.1	
4	Second Azimuthal-A	44.0	
5	Second Azimuthal-B	47.0	
6	First Azimuthal-A x First Axial	46.9	
7	First Azimuthal-B x First Axial	47.7	
8	First Radial x Second Axial-A	66.3	
9	First Radial x Second Axial-B	80.6	

Figure 4 – Schematic of Harmonic Flux Shapes

Pros and Cons

The diffusion-theory methodology in CERBERUS has a more rigorous basis and contains fewer approximations than the modal method, and thus that CERBERUS is the more accurate code. On the other hand, the advantage of SMOKIN is the relatively small numerical effort required to analyze even long transients; safety analysis and scoping studies can therefore be performed in a shorter time frame.

3. Physics Analysis for a Large LOCA

[Example of physics analysis for transient.]

The main quantitative results of the physics analysis of a large loss-of-coolant accident are:

- The reactivity transientThe bulk (whole-reactor) power, channel & bundle powers vs. time
- Time-integrated powers and values of fuel enthalpy.

Following its actuation, either of the special shutdown systems (SDS-1 or SDS-2) can insert a large amount of negative reactivity to turn the power over and terminate the power pulse. Even with 2 shutoff rods missing, SDS-1 introduces at least –80 milli-k of reactivity within about 1.5 s. SDS-2 can also introduce about this amount in the first second or two, but the negative reactivity insertion becomes even greater, and can reach several hundred negative milli-k, as the poison spreads within the volume of the moderator.

LLOCA analysis requires assembling complex models and inputs, making decisions about how to treat assumptions on the values of different parameters, and running the computer programs. These various considerations, decisions and inputs are described in the following sections.

3.2 Neutronics and Thermalhydraulics Models for LLOCA Analysis

With the evolution of computer capacity and performance, and the analyst's desire to capture increasingly closely the physical phenomena at play, the models used for reactor analysis have increased substantially in size and complexity over the last few years.

The evolution of physics models for LLOCA analysis has gone hand in hand with that of thermalhydraulics models. Twenty years ago, a LLOCA calculation would use a single coolant density transient over the entire broken PHTS loop. Present-day calculations are carried out with much more detailed and realistic input:

- different density transients are modelled in different passes of the same primary-heat-transport-system loop (the "critical", or downstream-of-the-break, pass features much more severe voiding than the upstream pass)
- the voiding transients are not uniform along a channel, but are instead functions of axial position along the channel

- the thermalhydraulics model now groups channels according to local conditions which may affect the voiding transient, e.g. high-power vs. low-power, inner-core high-power location vs. peripheral location, elevation in core (correlating with feeder length), etc.

A typical LLOCA calculation now models the critical pass with 5-20 different thermalhydraulics channel groups (see example in Figure 5) representing channels with different conditions, instead of a single "average" group. The groups are chosen according to the criteria listed immediately above. Non-critical passes, where the voiding is much slower, can be modelled by one channel group.

Figure 5 - Example of Thermalhydraulic Channel Grouping for LLOCA Calculation

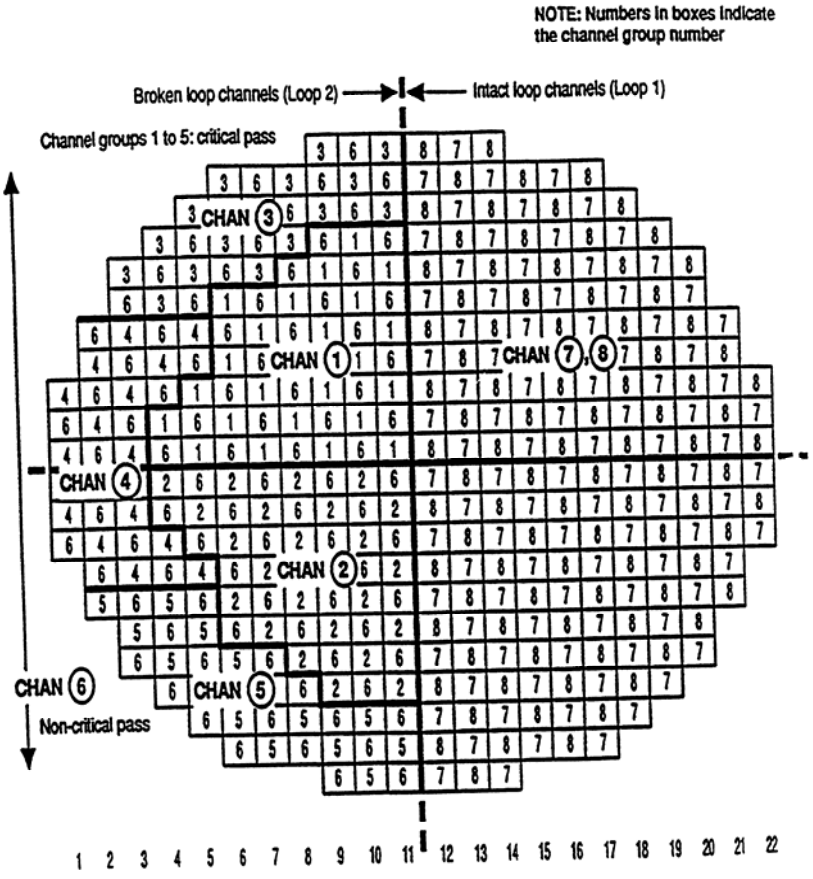


Figure 6 gives examples of results obtained for the coolant density in the various channel groups, in specific LLOCA calculations.

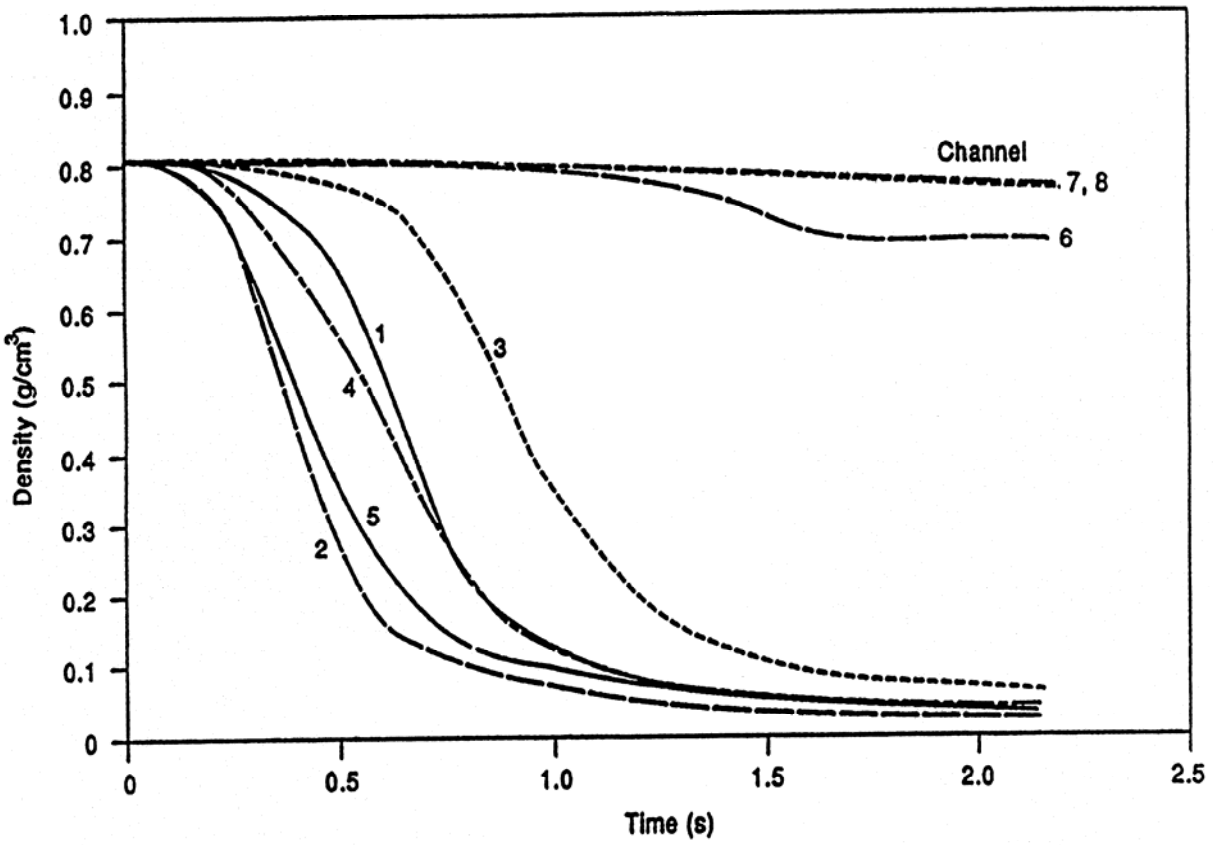


Figure 6 Coolant-Void Fractions in Various Channel Groups

LLOCA analysis is now performed with coupled neutronics and thermalhydraulics codes, so that the greatest benefit can be garnered from the evolution of the models. The coupling can be done in either of two ways:

1. Cycle between the thermalhydraulics and neutronics calculations over the entire LOCA simulation interval, starting from a ”guessed” power transient, and repeating until convergence is achieved. This method has been used mostly with the SMOKIN/TUF combination, but occasionally with CERBERUS/TUF as well.
2. “Walk” through the transient only once, sequencing the thermalhydraulics and neutronics calculations at each time step used for the CERBERUS shape-function calculation. This method is being used with the CERBERUS/CATHENA combination.

Typical power pulses calculated for a CANDU-6 reactor for an individual fuel bundle and for the core halves containing the broken and intact loops are shown in Figure 7.

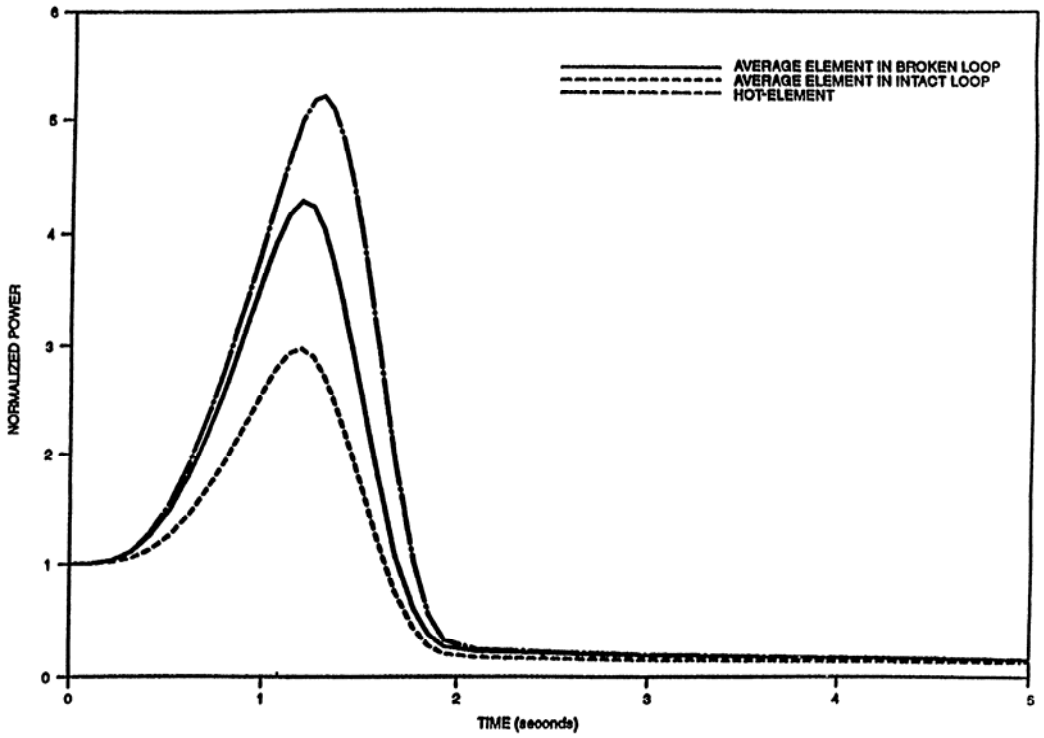


Figure 7 - Typical Power Pulses for Core Halves and for Maximum-Power Bundle

3.3 Uncertainty Allowance on Coolant-Void Reactivity

Void reactivity enters the calculation via the lattice parameters (nuclear cross sections) computed with the cell code, e.g., POWDERPUFS-V or WIMS-IST. As the loss of coolant proceeds, these lattice parameters will change in regions of voiding, inserting positive reactivity.

If we know or suspect that there is a difference between the value of void reactivity obtained from the cell code and the measured value, this should be taken into account by an “uncertainty allowance” made to the cell-code value. The allowance is positive (or negative) if the cell code underestimates (or overestimates) the void reactivity. The allowance may, for example, be made by changing a selected parameter so as to artificially change the void reactivity by the postulated amount. Or else the allowance may be made by artificially modifying a lattice property (e.g., neutron-absorption cross section) by a user-input multiplier.

3.4 Reactor Pre-Accident Configuration

For purposes of conservatism in the safety analysis, LLOCA calculations are often, performed assuming a number of extreme values of operating parameters or reactor configurations. Assuming that the coolant purity is at the lowest value permitted in the operating license, even if extremely unlikely, is an instance of this. This Section describes other examples.

Poison in Moderator

Moderator poison (boron or gadolinium) is used to maintain criticality in certain situations where excess reactivity must be suppressed. Such instances are:

- the "young" reactor core, from initial criticality to first refuelling,
- after a long reactor shutdown, when Xe-135 and other saturating fission products have decayed away,.
- in periods of intentional overfuelling in anticipation of planned fuelling-machine maintenance

However, the presence of moderator poison increases void reactivity. The reason is a redistribution of neutron flux in the lattice cell on coolant voiding: the flux increases in the fuel region and decreases in the moderator, causing a reduction in the rate of neutron absorption in the poison. As a consequence, the void reactivity increases with increasing concentration of moderator poison.

Configurations which induce larger void reactivity will thus be those in which the moderator-poison concentration is high, for example the plutonium-peak core (where the lattice reactivity is maximum), and after a long shutdown. A critical core following a long shutdown at the plutonium peak will feature the highest poison concentration. This has been the configuration assumed in many LOE analyses of LLOCA for the CANDU 6.

Pre-existing Flux Tilts

may increase the power pulse associated with a LLOCA. For instance, in the CANDU 6, a pre-existing side-to-side flux tilt will increase the neutronic importance of the void (and therefore the void reactivity) when the coolant loss is on the high-flux side. In fact, this pre-accident configuration has been found to induce the most severe power pulses in the CANDU 6. Again, this has been the initial configuration in recent LLOCA analyses for the CANDU 6.

Pressure-Tube Creep

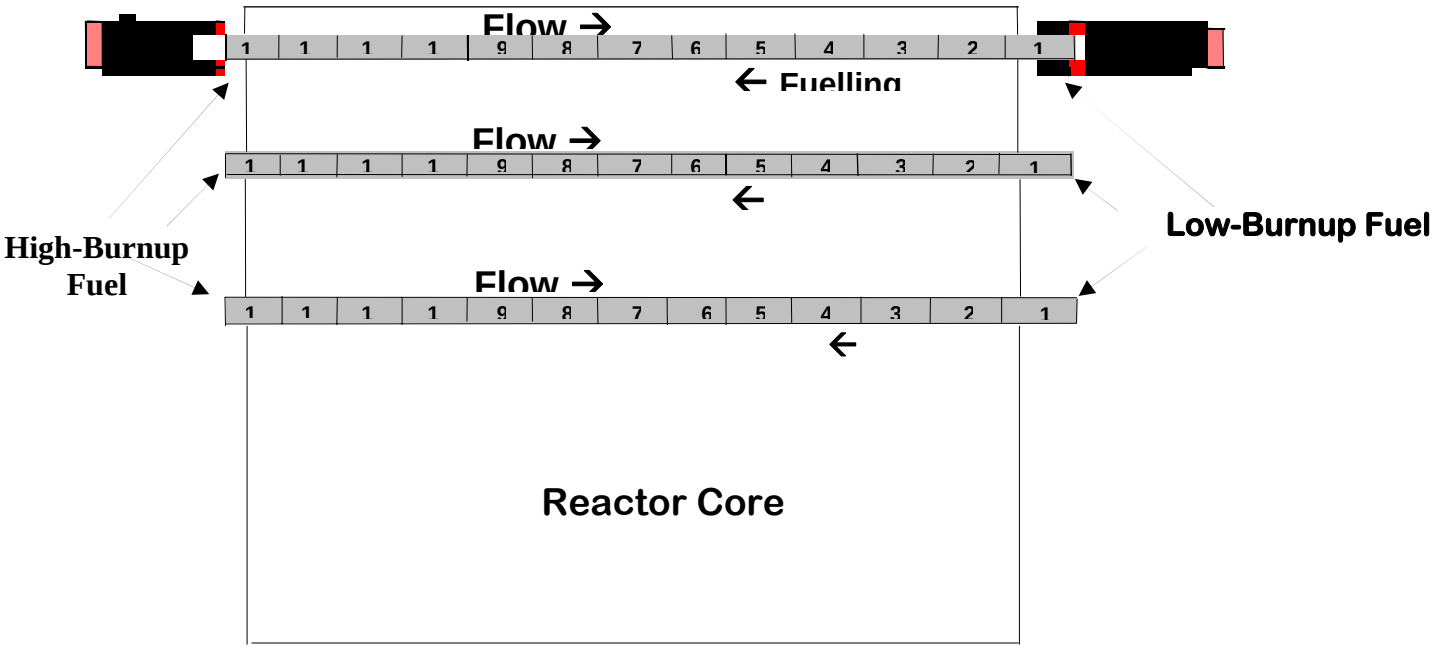
One of the effects of reactor aging is pressure-tube radial creep under prolonged neutron bombardment. The increased pressure-tube radius results in a greater volume of coolant in the core. To this is associated a larger void reactivity. The safety analysis for aging

reactors therefore uses values of radial creep consistent with the actual or anticipated age of the pressure tubes.

Fuel-String Relocation

Axial creep also needs to be taken into account in reactors where refuelling is done against the coolant flow. In normal operation, the fuel string in a channel is pushed by the force of the flow to the coolant-outlet end of the channel. Elongation of the pressure tube due to axial creep means therefore that the fuel string may be partly out of the core at the channel outlet. In an inlet-header (RIH) break, pressure differentials may push the fuel string back into the core. If the refuelling scheme in the reactor is against the direction of coolant flow (such as in the Bruce reactors), the irradiation distribution of bundles in the channels is such that the movement of the fuel string back into the core ("fuel-string relocation") would introduce positive reactivity **in addition to** the void reactivity. This must be taken into consideration in the kinetic analysis of the LLOCA. The magnitude of the effect depends on the length of the gap in the channels, which changes with the pressure-tube age (axial creep).

FUEL-STRING RELOCATION AND LLOCA



3.5 Protection-System and Detector Modelling

The physics simulation determines the actuation time of the shutdown system in response to the LLOCA. In order to calculate the actuation time, the code needs:

- the position of in-core ROP detectors and out-of-core ion chambers
- the logic channelization of these detectors - i.e., which in-core detectors and ion chambers are connected to which logic channel
- the delayed-response characteristics of the in-core detectors (various detector types have different prompt-response values and delayed-response terms and time constants)
- the electronic components (amplifiers, compensators, etc.) to which the detectors are connected
- the trip setpoints of the detectors: in-core detectors trip on high flux, while ion chambers trip on high rate-of-change of log of flux ("high log rate").

Many conservative assumptions are often made in safety analysis, which have an impact on the shutdown-system actuation time, for example:

- all three logic channels are required to trip before the trip is credited, while in reality two channels are sufficient to cause the trip
- the first trip signal is ignored; only the back-up (second) trip signal is credited
- error allowances are added to the trip setpoints
- the log-rate trip time is maximized by voiding the loop opposite the ion chambers corresponding to the shutdown system studied (CANDU 6).

3.6 Shutdown-System Configuration

Analysis often assumes, for conservatism, that only part of the shutdown system functions. For instance, for SDS-1, it is assumed that two of the shutoff rods do not drop into the core, with **the two missing rods selected in such a way that the remaining rod configuration is the least effective**. The missing rods are usually adjacent (see Figure 8), leaving an uncovered region in which the power pulse will be higher.

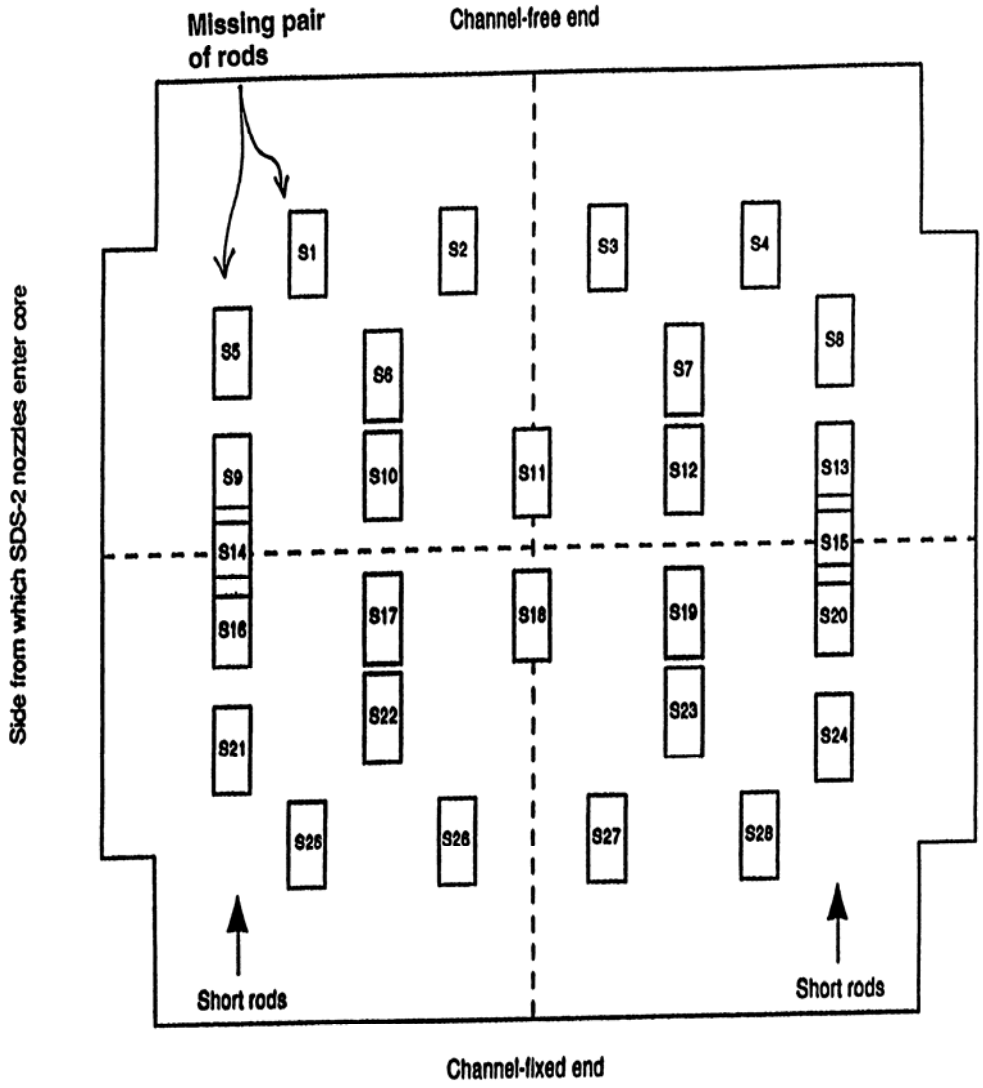


Figure 8 Assumed Missing Shutoff Rods

The analysis may also make a conservative assumption regarding the speed of insertion of the shutoff rods. The "insertion characteristic" used is usually based on actual field tests of shutoff-rod speed, further slowed down by an additional (2-σ) allowance.

In simulations of SDS-2 (poison injection in the moderator), conservative assumptions are made for the pressure in the injection tanks and the poison concentration in the tanks. One of the (6 or 7, depending on the reactor) poison tanks is assumed to be non-functional.

3.7 Decay Heat

In the calculation of the power distribution, it is necessary to remember that the thermal power produced in the reactor has two components:

- the “prompt”, or neutronic, component, which appears very quickly following fission, and
- the decay heat, which is produced in the decay of fission products and which appears delayed (seconds/minutes to weeks/months) following fission.

In steady-state operation, the decay heat is approximately 7% of the total thermal energy generated. In a transient situation, the decay power has a time variation which is very different from that of the prompt (neutronic) power. While the prompt power increases quickly and is reduced quickly (within seconds), the decay power decreases very slowly. This is illustrated schematically in Figure 9.

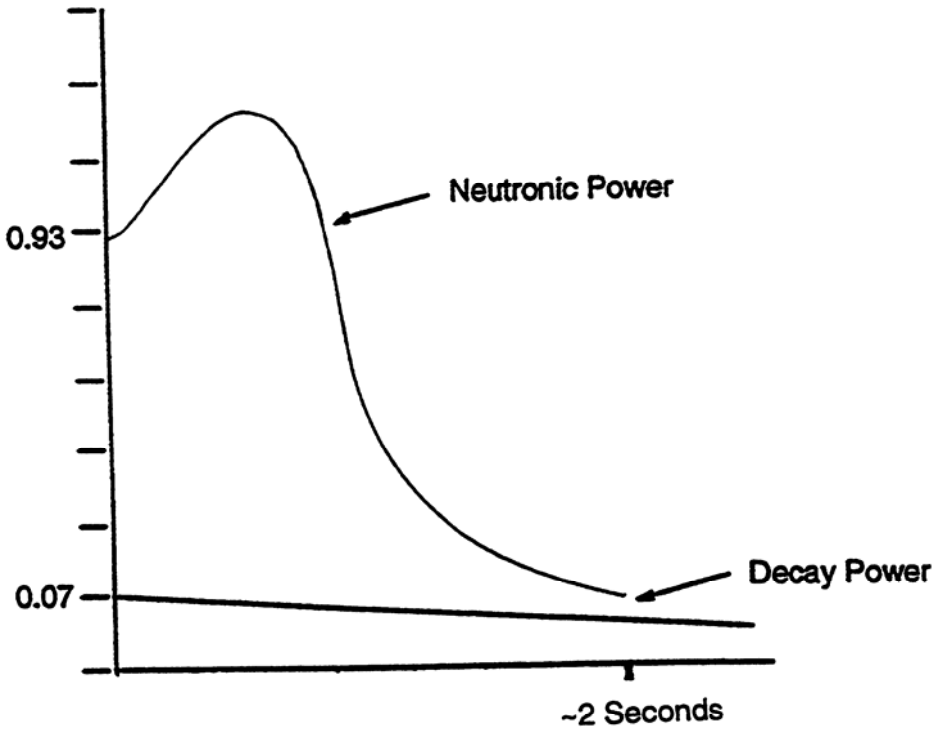


Figure 9 Schematic of Prompt and Decay Heat

In addition to the decay heat from fission products present before the accident, there is a decay-heat component originating in fission products newly created in the power pulse. This component should also be taken into account.

4. Summary

Modern LOCA simulations are most often coupled neutronics-thermalhydraulics calculations. The models are large and more and more detailed, and require a lot of care in the preparation. The assumptions and the parameter values used require careful selection. As a result of all these characteristics, modern LOCA calculations are time-consuming and quite onerous, but are an essential component of the reactor safety analysis.