

From Reference 2

**ROLE OF PHYSICS ANALYSIS
IN SAFETY AND LICENSING**

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1. Introduction

One of the licensing requirements for a CANDU plant is the submission of the Safety Report, in which it is demonstrated that under all credible postulated accident scenarios, the consequences are within acceptable limits specified by the Atomic Energy Control Board. Many diverse and potentially severe accidents are postulated and analyzed, and the special safety systems performance are predicted using sophisticated computer models. Such analysis involves multi-disciplinary studies of the event sequence and phenomena, starting with an initial plant state and a postulated initiating event, to transient system behaviours and safe reactor shutdown, and predicted activity releases. These studies call upon physics analysis of the reactor neutronic transient characteristics, thermalhydraulics analysis of the PHT and secondary side pressure / temperature transient responses, fuel and fuel channel analysis, containment analysis for activity releases, followed by atmospheric dispersion analysis for environmental contamination and dosage to the public.

At the front end of the accident analysis, physics calculations generally provide the definition of the initial core state. Then, according to the nature of the event, they model the changing core configuration to follow the neutronic characteristics as affected by fuel and coolant temperature, coolant density, reactivity device movements, power level changes and fission product evolution. These calculations predict the variations of flux and power as a function of time, which allows comparisons of the fluxes to regulation and protection system instrumentation set-points for reactor trip, power setback or stepback. The shutdown-systems actuation and control-systems device movements are modelled as they are actually controlled by the station computers. In these transient physics calculations, the changing thermalhydraulic conditions and fuel temperature in the PHT system are properly taken into account. On the other hand, the changing power level and power shape affect the thermalhydraulic behaviour. Therefore there is a need for physics and thermalhydraulics calculations to be coupled throughout the transient.

The special safety systems that are examined particularly from a physics viewpoint are the shutdowns system 1 and 2. The effectiveness of the system performance is measured in terms of its speed to mitigate and terminate any power excursion, so that the overpower transient and energy deposition in fuel do not lead to fuel and channel conditions exceeding acceptable limits. The most demanding accident event in terms of shutdown speed requirement is large LOCA, with a power pulse driven by the sudden de-pressurization and coolant void. Large LOCA analysis methodology will be discussed in detail in Section 3.

The shutdown system depth requirement is also set by considering a "most reactive" core state. The accident scenario postulates that certain unfavourable conditions simultaneously occur at the same time when the core configuration is in its most vulnerable state. This scenario involves a highly poisoned moderator which is then diluted by discharging coolant from an in-core break. The shutdown system must then maintain the reactor in a sub-critical state with sufficient margin at all times until operator intervention can be credited. This type of analysis is described in detail in Section 4.

With respect to licensing support analysis, two specific applications of physics input are discussed – guaranteed shutdown poison requirement and compliance to licensing power limits.

When the reactor is shut down, it is guaranteed to be sub-critical under all postulated credit accident scenarios. The most demanding accident scenario again involves an in-core LOCA and

moderator poison displacement. From an economics perspective, the time required to surrender guaranteed shutdown and remove poison to achieve criticality is to be minimized. Therefore it is important to establish the minimum poison requirement for guaranteed shutdown with adequate margin such that safety concerns are not compromised. The current method to establish this minimum poison requirement and safety margin is discussed in Section 5.

CANDU plants are licensed to operate at the rated power output under certain conditions, which include compliance to preset channel and bundle power limits. Often the initial powers assumed in safety analysis correspond to these licensing limits. Operations with power exceeding these limits for extended periods of time place the core in an unanalyzed regime, and constitute a license condition violation. Monitoring of powers at steady state operation and during routine operational manoeuvres is done by physics simulations of the reactor operations. Refuelling and burn-up history are tracked and the core state is simulated at frequent time intervals to give the power distribution. Uncertainties in the calculations must be properly addressed in order to determine the confidence level of compliance to the limits. The current compliance analysis method at Point Lepreau is discussed in Section 6.

Sample analysis results are given at the end of the discussion of each of the physics analysis applications. These results mostly pertain to CANDU 6 plants. The actual numerical values are not to be regarded as definitive, and may in fact be preliminary and evolving. They are quoted as typical results for the purpose to illustrate the analysis process and goals.

Details of the physics codes and methods have been presented in previous lectures, along with the important CANDU lattice physics characteristics. In order to facilitate understanding of the discussions in subsequent sections, a summary of the most pertinent reactivity effects due to changes in core state or in certain core parameters is presented in Section 2.

2. Reactivity Effects due to Changes in Core Parameters

The subject of CANDU neutronic characteristics has been addressed in some detail in previous lectures. Here we shall identify and describe qualitatively the "reactivity coefficients" that play an essential role in the accident transients. The term "reactivity coefficient" means the reactivity effect introduced by a change in certain physical core parameter, such as fuel temperature, or reactor power level, or moderator poison level. An understanding of the various reactivity components brought into play by the changing core state in an accident transient is essential to comprehend the trend of net reactivity and reactor power variations. Detailed reactivity effect assessments for CANDU 6 reactors are given in References 1 and 2.

Fuel temperature is affected by power level. The reactivity feedback is negative due to the "Doppler Broadening" of the fuel resonance absorption cross section. Therefore fuel temperature increase has a damping effect on a power excursion. On the other hand, there is a positive reactivity feedback upon a power reduction. This coefficient is dependent on fuel burn-up, the magnitude is $-0.01 \text{ mk/}^{\circ}\text{C}$ for fresh fuel and drops to $-0.006 \text{ mk/}^{\circ}\text{C}$ for mid-burnup fuel. If there is a power excursion and fuel temperature increase, the fuel temperature reactivity feedback provides about 0.6 mk per 100°C increase in an equilibrium core. If there is a rapid power reduction from full power and the fuel is cooled to room temperature in a fresh core, the reactivity feedback is about $+9 \text{ mk}$, which is outside of the range that can be compensated by the zone controller system.

The coolant density coefficient is negative, which means the coolant void coefficient is positive. When the reactor power increases, the coolant void increases and the lattice is more reactive and it feeds to the power excursion. This is compensated to some extent by the fuel temperature feedback. At equilibrium core nominal lattice conditions, complete voiding in all channels gives rise to about $+10 \text{ mk}$. This is further enhanced by pressure-tube creep, degraded coolant isotopic purity and presence of moderator poison. Coolant temperature coefficient on its own has a small reactivity feedback.

Poison in the moderator is often used to hold down excess reactivity in transient operation manoeuvres. In a reactor re-start, fission products (most notably xenon) are absent. The excess reactivity is held down by moderator poison. This represents a temporary situation when an in-core break will discharge coolant into the calandria and dilute the poison, leading to a positive reactivity insertion. The boron reactivity coefficient is about 8 mk/ppm .

The moderator isotopic purity is usually maintained at as high a level as possible since it strongly influences the economics of fuel consumption. The reactivity coefficient is about $+31 \text{ mk}$ per percent increase in moderator isotopic purity. The operating moderator purity is kept at around $99.9 \text{ atom percent}$. The discharging coolant with a lower isotopic purity, when mixed with the moderator will downgrade the moderator purity. It introduces negative reactivity.

Coolant isotopic purity has a much smaller coefficient on the system reactivity ($+0.57 \text{ mk/atom \%}$). However, it impacts on coolant void reactivity – downgraded coolant purity will increase coolant void reactivity. The coefficient is about $+0.56 \text{ mk}$ of void reactivity per percent degradation in coolant isotopic purity. The operating coolant isotopic purity is around 98.5 to $99.0 \text{ atom percent}$. The Operating Principles and Procedures stipulates that the operating coolant purity is not to be lower than $97.15 \text{ atom percent}$.

Moderator temperature reactivity feedback is positive. In the case of moderator temperature

raised by the hot coolant, positive reactivity is added. However, the magnitude of this coefficient is small -- the reactivity coefficient is $+ 0.07 \text{ mk/}^\circ\text{C}$. The heat capacity of the moderator inventory is relatively large, and hence the rate of moderator temperature change is usually slow. Note however, that the corresponding density reduction will lead to a reduction in poison concentration and can enhance the positive reactivity insertion.

3. Large Loss-of-Coolant Accident Analysis

3.1 General System Behavior and Analysis Approach

In a postulated large LOCA, steam and water would rapidly discharge into the reactor building. The PHT quickly de-pressurizes in the broken pass. The decreasing coolant density in the fuel channels downstream of the break would introduce positive reactivity at a rate which could not be compensated by the regulating system. This would lead to a rise in reactor power. The increase in heat generation and the degraded heat transfer would lead to fuel and sheath temperature rise. The highest temperature of fuel and of the pressure tubes are expected to occur for breaks where flow is reduced to near zero while the stored heat in the fuel is still high. In the intact loop where the heat transport pumps maintain forced circulation, fuel would remain well cooled. These are the general conditions in the first few seconds after the break.

Reactor trip signals would occur within about one second. Normally the neutronic set-points are the first reached, i.e. high neutron power and high rate of change of neutron power trips. The shutdown systems will actuate and turn over the power excursion and effectively terminate the fission process and shut down the reactor within 2 to 3 seconds. The primary safety concerns are the pulse energy deposition on the fuel, leading to fuel melting and break up of fuel pellets. The highest fuel temperature occurs at the pellet centre axis, thus centre-line melting is also often used as a criterion to indicate fuel failure. Other concerns are excessive heat transfer to the pressure tube, leading to a circumferential temperature gradient and breaching of the pressure tube integrity.

The role of physics analysis in large LOCA is to determine the power pulse due to the reactivity transient and the energy deposition in fuel. To maximize the effect of the potential power pulse consequences, certain assumptions are made of the initial core state and in the analysis methodology. These assumptions place the shutdown system performance under the most severe tests using a combination of worst but credible conditions. These conditions are sometimes known as the Minimum Allowable Performance Standards (MAPS). Detailed description of an extensive set of power pulse calculations for CANDU 6 reactor is given in Reference 3.

3.2 Pre-Event Reactor Conditions

The initial core state assumed in the analysis is qualitatively described as follows.

The accident is assumed to occur at a time when the moderator is heavily poisoned, i.e. at the time of a restart after a prolonged outage and the adjuster banks are all withdrawn. The absence of the saturating fission products and the adjuster bank being withdrawn both require reactivity compensation by moderator poison.

The pressure tubes are creeping diametrically and length-wise over their life time. The enlarged pressure tubes lead to higher coolant volume and higher void reactivity hold-up. Conservative estimates of the current pressure tube diameter increase due to creep are used in the lattice cell model and hence in the coolant void calculations.

The reactor is assumed to be operated with coolant isotopic purity at its lower limit.

A tilted flux shape existing at the time of the accident can aggravate the void effect. The PHT configuration in CANDU 6 is such that a break in one pass will initially affect a quarter of the

channels located to one side of the core. If a side-to-side flux tilt already exists and the high flux side coincides with the voiding side, then the void reactivity effect would be aggravated. Also, if a bottom-to-top flux tilt already exists at the time of accident, the effectiveness of the shutoff rods can be reduced since they take a longer time to reach the high-flux bottom of the core. Various initial tilted flux shapes are therefore assumed in LOCA analysis. Note also that the initial reactor power is reduced from full power to avoid early reactor trip on high power signals from the in-core detectors.

3.3 Other Analysis Assumptions

Other assumptions that maximize the power pulse and its consequences are:

- a. Trip set-points are to include uncertainty allowance.
- b. Trip time is to be based on the backup trip rather than the first trip, and on the third logic channel.
- c. The two most effective shutoff rods (or one most effective LIZZ nozzle) are assumed non-operational. The two most effective rods are selected with respect to the break type and the location being analyzed.
- d. Coolant void reactivity is deliberately augmented to allow for calculation uncertainty.
- e. Reactor Regulating System actions are ignored.

3.4 Analysis Tools and Methods

For full space-time kinetics calculations, two major computational tools are used:

- a. A thermalhydraulics code (e.g. FIREBIRD⁽⁴⁾, CATHENA⁽⁵⁾, SOPHT⁽⁶⁾) is used to calculate the time dependent coolant density distribution in the core, among other thermalhydraulic parameters of interest. The transient power distribution is required as input.
- b. A neutron kinetics code (CERBERUS⁽⁷⁾) to calculate the change in neutron flux and power with time. The transient coolant density and temperature, and fuel temperature are required as input.

The goal here is to calculate the power transient arising from a particular break size and location. The two codes can be executed in a de-couple mode. In this case, the thermalhydraulics code is first executed to compute the coolant density variation over the entire time of interest, using an estimated power pulse from a previous study. The predicted coolant density transient is then used in CERBERUS, which predicts the power transient for the given thermalhydraulic transient input. The process can be iterated to achieve consistency.

The two codes can also be executed in a coupled mode. In this case, the transient of interest is simulated by the repeated execution of the two codes in sequence as shown schematically in Figure 3-1. The power distribution from CERBERUS at a flux-shape time step is fed to the thermalhydraulics code, which then evaluates the coolant densities and other parameters for the next time step, which are fed back to CERBERUS in the next flux-shape calculation. In

addition to changing PHT conditions, the changing device positions, such as shutoff rod insertion, are modelled in each flux-shape calculation.

In the thermalhydraulics model, channels are grouped according to their power and transient thermalhydraulics properties. Each channel type (see the example given in Figure 3-2 which shows 8 fuel types) is explicitly modelled in the PHT nodalization circuit. The transient properties for each channel type are fed to neutronics calculations. The power distribution generated from CERBERUS calculations is also collapsed to the groups of channels matching the thermalhydraulics channel types, and fed back to the thermalhydraulics calculations.

3.5 Break Types

Two break types are usually of the most interest – a large break (100% Pump Suction Break, or 100% Reactor Outlet Header Break) that leads to the highest energy pulse and energy deposition in fuel, and a critical break (about 20–30% Reactor Inlet Header Break) that leads to flow stagnation and most severe pressure tube temperature transient. The standard definition of break size is twice the pipe cross-sectional area for a 100% break.

3.6 Reactor Trip Time

The electronic circuitry for the neutronic trips are modelled in order to determine, as closely as possible, the actuation times of the shutdown systems. The calculated fluxes at detector and ion-chamber locations are fitted to a parabolic curve and fed to the TRIPDPG circuitry model which calculates detector response. By comparing to trip set-points, the high-power trip time of each in-core detector and the rate-log-power trip time of each ion-chamber are determined. Trip of all three logic channels is demanded, i.e. at least one detector in each logic channel has tripped.

Since the backup trip is to be credited, the shutdown system actuation time is then the later of the high-power and rate-log trip times. In case of SDS1, this actuation time is the time at which the current to the shutoff rod clutches is cut off.

Note also that in the calculation of rate-log trip time, the ion-chambers are assumed to be those located at the opposite side of the broken loop.

Typical high-power trip setpoint is around 122–124% for both SDS1 and SDS2, and typical rate-log trip setpoint for SDS1 is 10%/s and for SDS2 is 15%/s. An instrument-loop uncertainty is also normally assigned to the rate-log trip set-points. For example, the SDS1 rate-log trip is assumed to be at 11.5%/s in the analysis.

3.7 Shutdown-System Effectiveness

The primary measure of shutdown-system effectiveness is the margin to fuel-breakup. The energy stored in the fuel is the sum of the initial stored energy (i.e. steady state energy content) plus the energy added by the power pulse. The latter is the time integrated difference between power generated in the fuel and power removal from fuel by the coolant. In the adiabatic approximation, the power to coolant is ignored up to the time about when the shutoff rods are fully inserted.

The highest allowable bundle power in CANDU 6 is 935 kW. To assess fuel integrity, the energy stored in the hottest fuel element of a 935 kW bundle is evaluated. This fuel element is assumed to be subject to the power pulse of an actual bundle with the largest time-integrated power up to 5 seconds. The total energy stored is then compared to a conservative lower limit required for fuel breakup, typically taken as 840 J/g of fuel.

3.7 Sample results

For illustration purposes, the results from a sample case of 30% RHH break in a CANDU 6 plant from Reference 3 are described below.

The initial power is 100%FP. At time zero, a 30% break occurred in the pass represented by Channel Groups 1-5 in Figure 3-2. Channel Group 6 represented the other pass in the broken loop. Channel Groups 7 and 8 represented the intact loop. High neutron power trip was reached at 0.412 s, and high log-rate trip was reached at 0.495 s. The latter actuated the SDS1. The SOR drop characteristic curve was the same as measured at site plus uncertainty allowance. The results are summarized in the following Figures and Table:

Figure 3-3 Flux-square weighted coolant density transient in the thermalhydraulic channels

Figure 3-4 Void fraction in broken and unbroken loop

Figure 3-5 Flux-square weighted fuel and coolant temperature transient

Figure 3-6 System reactivity transient

Figure 3-7 Relative power transient in bundle with highest integrated energy deposition

Table 3-1 Detailed results for total power and loop power transient

When the power pulse for the bundle with highest integrated energy (to to 5 s) was applied to the hot pin of a 935 kW bundle, it added 212.8 J/g to the initially stored energy of 380.8 J/g. The total energy content of 593 J/g is significantly below the fuel break-up limit of 840 J/g.

4.0 In-Core Loss-of-Coolant Accident Analysis

4.1 General System Behaviour

The postulated spontaneous rupture of a pressure tube while the reactor is operating at power, is one of the events assessed in order to evaluate the effectiveness of the special safety systems. The calandria tube surrounding the ruptured pressure tube is assumed to have also failed. Primary circuit coolant discharges into the calandria. The discharge rate and force are dependent on the cause of the channel break, feeder flow resistance, and pressure build up in the broken channel. There can be a rapid pressurization of the calandria, relieved by subsequent bursting of the rupture discs in the pressure relief ducts. The discharging hot coolant, and possibly with ejected fuel bundles, can cause structural damage, disabling some shutoff rod guide-tubes and MCA guide-tubes.

The overall response of the primary circuit is similar to that of a small out-of-core break (0.5–1% Reactor Inlet Header break). For large in-core break, the pressure and inventory control system cannot make up for the discharge and the primary circuit would depressurize; voiding would occur in all channels. The discharging coolant will also mix with the moderator and dilute the poison concentration, and downgrade the moderator isotopic purity. The rate of voiding would be slow so that the reactor regulating system could compensate for the void reactivity. It is expected that low heat transport pressure trip and pressurizer low level trip set-points will be reached in 2–3 minutes. Moderator temperature increase would be relatively slow because of the high thermal capacity of the moderator. For a more detailed description of the system behaviour, see Reference 8.

4.2 Physics Considerations

The coolant void reactivity insertion rate from the rupture of a channel is much smaller than that as in the case of a large LOCA. It has been often assumed that in a small break, up to the time of reactor trip, the regulating system will compensate for the void reactivity insertion, and maintain the reactor bulk power at the demanded level. The maximum reactivity change rate of the zone controller system is 0.14 mk/s. This may or may not compensate for the positive reactivity insertion depending on the positive reactivity insertion rate which is a function of the coolant discharge rate and other factors such as moderator poison dilution rate. The RRS may also drive the mechanical control absorbers in the core. In such cases the power distribution will be more distorted.

Physics calculations in in-core LOCA accident analysis provide an evaluation of the shutdown system effectiveness, particularly in terms of sufficient depth of SDS1 when the system is partially impaired. Furthermore, the transient reactor regulating system response before reactor trip and hence the power distribution distortion and variations with time can also be modelled in physics kinetics calculations.

4.3 SDS1 Depth

After reactor trip, the shutoff rods are inserted and the reactor is sub-critical. The available number of shutoff rods may not be the full complement of the system – some rods are assumed not able to insert due to damaged guide tubes, and one or two of the remaining one are assumed to have failed. Note however the discharging coolant maintains the positive reactivity insertion after reactor shutdown. The shutdown system must be able to keep the reactor in a sub-critical

state up to a time when operator intervention can be credited, which is accepted at after 15 minutes of an unambiguous alarm indication of the accident event. Therefore it is required that the reactivity depth of the SDS1 shutdown system as designed is sufficient to maintain the reactor sub-critical 15 minutes into an in-core LOCA event.

4.3.1 SDS1 Depth Analysis Method

The current analysis methodology is to simulate the reactor core at 15 minutes after the initiation of the accident, modelling the core configuration as closely as possible:

- a. The moderator poison concentration as predicted by the most credible coolant discharge calculations and mixing model.
- b. The moderator temperature as predicted by moderator pressure and temperature transient calculations.
- c. The coolant density distribution in the four passes as predicted by thermalhydraulics transient calculations.
- d. The insertion of available shutoff rods which are not damaged by discharging fuel and coolant as predicted by the most credible damage assessment.
- e. The degradation of moderator isotopic purity due to mixing with the discharging coolant as predicted by the most credible mixing model.

Mitigating actions from the reactor regulating system and other safety systems are often not credited in the analysis: emergency coolant injection and boiler crash cool-down are not credited, RRS action is ignored and not modelled.

As in many other safety analyses, the initial core state is postulated to be one that would lead to the worst possible consequences. With respect to the requirement on shutdown depth, a highly poisoned moderator obviously leads to a more severe reactivity transient due to poison dilution. Also a highly poison moderator enhances the coolant void reactivity. Therefore the accident is often postulated to occur at a time when the poison level is the highest, such as at restart after a prolonged shutdown when the absence of fission product reactivity load is compensated by moderator poison, and the adjusters are out of core which requires compensation of moderator poison.

Damage to the shutdown system is assessed according to the cause of the in-core break. Three types of breaks are often considered: Pressure Tube Rupture, Flow Blockage and Feeder Stagnation. The pressure and temperature characteristics of leading to the channel break, and the discharge rates and hydrodynamics forces and subsequent physical damages are different for these break types. The selection of the broken channel and the location of the break are chosen to maximize the damage in terms of the number of shutoff rods disabled and the relative effectiveness of these disabled rods. Generally, Flow Blockage events have a higher temperature and pressure build-up and the size of the "sphere of influence" is larger. Note also that the break types also influence the assumption on the number of further shutoff rods assumed to have failed. For example, in flow blockage event, aside from the disabled rods, two further rods are assumed unavailable – one being tested and another randomly failed. Note that flow blockage events are postulated to be very unlikely to occur shortly after a restart when the flow verification test has just been conducted. In events that occur shortly after a restart and the

shutoff rods have just been tested, only one further rod is assumed unavailable.

The degradation of moderator isotopic purity allows a credit on negative reactivity insertion. It also stipulates a minimum difference between the moderator and coolant purities, thus places an upper limit on the operating coolant purity. If the difference is small, it will not be an operational constraint. It may be assumed that the coolant has the same purity as the moderator and no credit for moderator degradation is taken. If the analysis results demonstrate a sufficient shutdown margin, then there will be no upper limit to the operating coolant purity.

On the other hand, coolant void reactivity is aggravated by low coolant isotopic purity. Therefore the lower limit on operating purity is assumed in the simulation to maximize the void effect.

4.3.2 Mixing Models

The dilution of poison in the moderator is calculated according to certain mixing models. There are three mixing models that have been used in one form and another (Reference 9). With the "Piston Mixing" model, the discharging coolant is assumed to act as a "piston", displacing unmixed poisoned moderator which is expelled through the rupture discs. Note, however, the reactivity effect is often computed based on an average poison concentration in the calandria, i.e. assuming the poison is distributed uniformly. In the "Uniform Mixing" model, the discharging coolant is assumed to mix uniformly and instantly with the poisoned moderator so that the poison concentration of the expelled moderator is the same as the average poison concentration throughout the moderator. Recently a variant of the uniform mixing model, the "Delayed Mixing" model, is favoured. The basic assumption in this model is that the poison concentration of the fluid discharged through the rupture discs is equal to the average poison concentration at an earlier time T , which is the characteristic time over which the mixing takes place.

Mathematically,

if $P(t)$ = Average poison concentration at time t ,
 M = Mass of moderator,
 $m(t)$ = Coolant mass discharged at time t ,

then the three mixing models are represented by:

Piston Model: $P(t) = P(0) [1 - m(t) / M]$

Uniform Mixing Model: $P(t) = P(0) \exp [- m(t) / M]$

Delayed Mixing Model: $P(t) = P(0) \exp [- m(t) / M / (1 - m(T) / M)]$

The dilution factor at time t is simply $P(0) / P(t)$.

4.3.3 Calculation Uncertainty and Safety Margin

The simulation of the core state at 15 minutes after the accident gives a k_{eff} value, based on which we would attempt to conclude if the shutdown depth is adequate. However, to facilitate the calculation of the safety margin and judgement of its adequacy, the core simulation is often done with the moderator poison level floated to determine the "critical" poison level. This "critical" poison level is then compared to the pre-event "initial" poison level diluted to a level

as predicted by the most credible mixing model. The step to determine the safety margin is therefore:

1. An "initial" poison concentration $[Gd]_0$ at the start of the accident is calculated. This corresponds to the poison needed to compensate for the excess reactivity of the zero-power, hot restart core state after a long shutdown, with all adjusters out.
2. A dilution factor DF_{15} corresponding to coolant discharge up to 15 minutes is applied. The nominal "diluted" poison concentration at the 15th minute is $[Gd]_d = [Gd]_0 / DF_{15}$.
3. The "critical" poison concentration $[Gd]_{15}$ for the core state at the 15th minute with PHT partially voided, moderator poison diluted, moderator temperature increased and partial set of SOR inserted, is calculated. The margin to criticality is therefore given by

$$M = [Gd]_0 / DF_{15} - [Gd]_{15}$$

The adequacy of the safety margin cannot be judged in isolation. The calculation uncertainty inherent in M must be taken into consideration. Therefore, an assessment must be made to determine any bias error and random uncertainty in M . Any bias error so determined should be applied to adjust the margin. The margin to criticality is then measured in units of sigma, which is one standard deviation of the random uncertainty. This would give a probability level that the reactor will remain sub-critical and the adequacy of the safety margin is judged accordingly.

The assessment of calculation uncertainty in M is therefore an essential component in the analysis. Evidently the uncertainty in M is dependent on the uncertainties in $[Gd]_0$, $[Gd]_{15}$ and in DF_{15} . Generally speaking, the uncertainty in these calculated quantities can be estimated through comparisons to corresponding measurement data. For example, the measured poison concentration at the time of restart after a prolonged outage is often compared to RFSP calculations. This would give an estimate of uncertainty in $[Gd]_0$. However, there are situations where measurements are not possible or not available, such as the calculation of $[Gd]_{15}$ which involves accident core conditions. The uncertainty estimate for $[Gd]_{15}$ is then based on the reactivity components introduced by the perturbations as the abnormal core conditions, and accuracy of RFSP capturing the reactivity effects of these perturbations. While the detailed assessment method is outside the scope of this lecture, an illustrative example of the results are given in the next section.

4.3.4 Sample Results of SDS1 Depth Analysis

For illustration purposes, the results from a recent study for CANDU 6 plants for the case of a pressure-tube and calandria-tube rupture event (Reference 10) are described below.

The initial core at time zero corresponded to a restarted core after a prolonged shutdown at zero power hot conditions. All adjusters were withdrawn. To further increase the excess reactivity, fuelling ahead of 5 mk while the reactor was shut down was assumed. The critical poison level $[Gd]_0$ was calculated to be 6.86 ppm boron.

Channel E11 was assumed to rupture at time zero. The PHT blow-down was computed by SOPHT, and the thermalhydraulics conditions at 907 second were modelled in RFSP "all-effects-included" simulation. A total of six SOR's were assumed non-operational: five disabled rods and one additional unavailable rod. The moderator D_2O isotopic purity was 99.94 atom percent, and was not degraded by the coolant discharged. However, in the coolant void

effect calculation, the coolant isotopic purity was degraded to 95.08 atom percent to enhance the void reactivity. The just critical poison level $[Gd]_{15}$ at 907 s was determined to be 3.16 ppm boron.

The delayed mixing model was used to compute the dilution factor with a characteristic time of 15 s. The dilution factor DF_{15} was determined to be 1.38. Therefore the diluted poison level $[Gd]_d$ at 907 s would be $6.86 / 1.38 = 4.98$ ppm boron. Comparing this to the just critical poison level of 3.16 ppm boron, there is a safety margin of 1.82 ppm boron, which is equivalent to about 15 mk.

The adequacy of such a margin was judged in the context of calculation uncertainty. The 1- σ uncertainty in $[Gd]_d$ was estimated to be $\pm 14\%$, which was the combined uncertainties in $[Gd]_0$ and in DF , which was respectively $\pm 5\%$ and $\pm 13\%$. The uncertainty in $[Gd]_{15}$ was more complicated since it involved abnormal core conditions. The details can be found in Reference 10. The 1- σ uncertainty was estimated to be also $\pm 14\%$. The 1- σ uncertainty in the safety margin was then given by $[(4.98 \times 0.14)^2 + (3.16 \times 0.14)^2]^{1/2} = \pm 0.83$ ppm boron. The safety margin is more than two-sigma and hence there is greater 98% probability that the reactor remains sub-critical.

4.4 RRS Response Modelling in In-Core LOCA without Reactor Trip

In the event of an in-core break, reactivity perturbation is introduced and the reactor regulating system will respond to compensate the excess reactivity. If the power error is large and positive, the mechanical control absorbers are inserted and the initial power distribution will be distorted. It can be postulated that the guide tubes for the MCA can be damaged as well and the partially impaired MCA system may not totally compensate for the reactivity insertion. Power excursion may occur and the reactor will trip on high neutron power or power stepback will be initiated on neutronic set-points being reached. The reactor will also trip on some process parameters such as low coolant flow. However, if all these trips and power stepback and setback are not credited, the response to the RRS will continue to counteract the reactivity transient and control and power to the setpoint level. For such analysis, neutron kinetics calculation using such codes as CERBERUS is necessary, together with the capability to model the RRS actions.

The capability of modelling RRS in CERBERUS calculations has been recently implemented⁽¹¹⁾. The RRS control algorithms used in the station computers in CANDU 6 (specifically those in G2) have been closely mimicked in the *CERBRRS module in RFSP. This allows coupled neutronics and RRS response transient simulations for accident analysis where the control actions lead to significant feedback to the reactivity and power calculations.

As a typical application, an in-core LOCA scenario presented in G2 Safety Report⁽⁸⁾ was re-analyzed using *CERBRRS. The initial reactor power was at 75% FP, with a substantial amount of moderator boron. An in-core break at Channel E11 was postulated and the subsequent coolant discharge, coolant density transient in the four passes, poison dilution and moderator isotopic purity degradation were predicted using the SOPHT-G2 and COMTES-G2⁽¹²⁾ codes. The reactivity insertion due to poison dilution as a function of time is shown in Figure 4.4-1. Reactor trip, power stepback and setback were discredited. Two of the MCA's were assumed knocked out of service.

The subsequent reactivity transient, zone fill changes and the movements of the remaining two MCA rods were predicted using the *CERBRRS module. The power transient is shown in Figure 4.4-2, and the device positions as a function of time are shown in Figure 4.4-3. At the initiation of the LOCA, the average zone fill immediately went up to compensate for the diluting poison. The MCA rods were soon called into action when the excess reactivity became too large. The power excursion was eventually turned over when the two MCA rods were inserted into the core. The response of the devices throughout the transient was strictly according to the set of rules for each device as specified in the RRS design specifications. This example clearly illustrates the capability of physics kinetics calculations coupled with RRS modelling for potential applications in accident analysis.

5.0 Guaranteed Shutdown State Poison Requirement

Closely related to in-core break physics analysis is the determination of the guaranteed shutdown state (GSS) poison requirement. When the reactor is shut shown, from safety viewpoint, the moderator poison level should be such that the reactor is guaranteed to be sub-critical under all conditions. From an operations viewpoint, the moderator poison level is to be just sufficient without compromising safety concerns, but to restart the reactor, the time needed to reach GSS removal is at a minimum.

The safety concerns are addressed by postulating a most reactive core state with a combination of abnormal accident conditions – an in-core break leading to poison dilution, moderator temperature increase, and complete PHT voiding. There is no credit for operator action, and hence the available PHT inventory will be assumed to be emptied and mixed with the moderator. With the diluted poison, the reactor must still remain sub-critical, accounting for calculation uncertainties in the simulations of the "most reactive" core state and in the calculation of the dilution factor. Also, it must be demonstrated that there is a high level of confidence that the safety margin (to criticality) is adequate.

The SDS1 depth analysis and the GSS poison requirement analysis share a lot of common elements – an in-core LOCA diluting the poison and coolant voiding as the most limiting scenario. However, there are some essential differences: the SOR's are not inserted in GSS, and there is no 15-minute time frame. Thus the PHT is assumed to be completely voided, and the dilution is with all available PHT inventory.

5.1 Methodology

The current methodology to establish the GSS poison requirement is described as follows.

Define:

$[Gd]_{gss}$	GSS poison concentration.
DF	Moderator poison dilution factor.
$[Gd]_d = [Gd]_{gss} / DF$	Diluted poison concentration.
$[Gd]_c$	Moderator poison concentration at which the core will be just critical for the most reactive core state.
M	Margin to criticality after poison dilution, in units of ppm poison concentration.

Using these definitions, we have

$$M = [Gd]_{gss} / DF - [Gd]_c \quad \text{Eq. 1}$$

The calculated quantities $[Gd]_c$ and DF have uncertainties. We further define

σ_c (in %)	One-Sigma uncertainty in $[Gd]_c$. (The bias error in $[Gd]_c$ is also to be assessed and accounted for).
σ_{df} (in %)	One sigma uncertainty in DF. Presumably there is no bias error in DF.

The margin M deduced from Equation 1 will have inherent uncertainty, denoted by σ_m . The definition of "sufficient" margin M is to be specified, and should be expressed in terms of σ_m . Here we make the assumption that if $M \geq 2 \sigma_m$, then the margin is sufficient.

From Equation 1, if M has been specified, we can simply deduce

$$[Gd]_{gss} = ([Gd]_c + M) * DF \quad \text{Eq. 2}$$

or alternatively,

$$[Gd]_{gss} = ([Gd]_c + 2 \sigma_m) * DF \quad \text{Eq. 2a}$$

Therefore, procedurally we can establish $[Gd]_{gss}$ by following these steps:

- a. Establish $[Gd]_c$ (and σ_c),
- b. Establish DF (and σ_{df}),
- c. Estimate σ_m and deduce $[Gd]_{gss}$ using Eq. 2a.

However, there are really two unknown quantities, $[Gd]_{gss}$ and M . We cannot establish σ_m because $[Gd]_{gss}$ is not known and hence $[Gd]_d$ (given by $[Gd]_{gss} / DF$) is not known, and we cannot establish $[Gd]_{gss}$ from Equation 2 because M (or equivalently σ_m) is not known.

To circumvent the cyclic situation described in Section 2.3, one can revise Equation 2 to:

$$[Gd]_{gss} = [Gd]_c \times DF \times UF \quad \text{Eq. 3}$$

where UF is an Uncertainty Factor to be related to σ_c and σ_{df} .

We can rearrange Equation 4 as:

$$[Gd]_{gss} / DF - [Gd]_c = [Gd]_c \times (UF - 1) = M \quad \text{Eq. 4}$$

Evidently the margin M is given by $[Gd]_c \times (UF - 1)$.

One functional form of UF relating to σ_c and σ_{df} is:

$$UF = 1 + \left[(n \sigma_c)^2 + (n \sigma_{df})^2 \right]^{1/2} \quad \text{Eq. 5}$$

with n being the number of sigmas one wishes to cover. Note that the uncertainty factor is applied to the product of $[Gd]_c$ and DF , hence it is justifiable to combine σ_c and σ_{df} the way it is stated.

A further uncertainty allowance was made to cover off residual uncertainties arising from "engineering assumptions":

$$UF = 1 + \left[(n \sigma_c)^2 + (n \sigma_{df})^2 \right]^{1/2} + C \quad \text{Eq. 6}$$

The value assigned to C has been somewhat arbitrary and is based on judgement.

5.2 Critical Poison Level Calculation

The current method used to calculate $[Gd]_c$ is an all-effects-included RFSP simulation of the most reactive core state, which corresponds to a hot, pressurized, zero-power state after restarting from a prolonged outage, all adjusters out, zones drained and PHT voided, and with

fuelling ahead while it was shut down. All temperature reactivity feedback is implicitly included in the simulation. The moderator poison concentration is floated in the calculation to determine $[Gd]_c$. The coolant void reactivity is a function of moderator poison level, and therefore corresponds to $[Gd]_c$ ppm of poison in moderator. Furthermore, the moderator temperature is raised by the discharging hot coolant, and the moderator temperature reactivity effect and the poison dilution reactivity effect due to reduced moderator density are all included in the simulation.

Previously, for convenience and due to lack of modelling capability, $[Gd]_c$ was established through a summation of the reactivity components from the various core conditions and the critical poison concentration was deduced using a poison reactivity coefficient. This procedure gave a rough estimate which ignored any cross-component compounding effects, such as the void reactivity being a function of the moderator poison level.

The assumptions made in the RFSP all-effects simulation have implications on how the uncertainty in $[Gd]_c$ is to be assessed. For the purpose of GSS methodology characterization, it is appropriate that "best-estimate" assumptions are made in the RFSP calculation. However, if for certain components, a bias in the calculation method is generally acknowledged either implicitly or as a prudent measure that has been consistently applied (e.g. void reactivity uncertainty allowance), then the bias allowance is included in the RFSP simulation. Note further that in the assessment of σ_c , if site-specific operation limits impose bounds on the uncertainties, they are to be taken into consideration.

5.3 Sample results

The following values (from Reference 13) are typical of a CANDU 6 plant, and are used for illustrative purposes only. The critical moderator poison concentration was determined by RFSP simulation of the "most reactive" core to be 10.5 ppm boron. The dilution factor was calculated using the delayed mixing model. Typical values for a CANDU 6 plant, the mass of moderator is 232 Mg which did not include the amount that initially entered the relief duct and hence not available for mixing. The mass of available PHT inventory is 220 Mg which included all the mass that could be discharged such as the mass in the pressurizer and storage tank. Assuming a discharge rate of 0.18 Mg/s obtained from thermalhydraulics simulation and a characteristic time of 15 s, a dilution factor of 2.6 was obtained. The nominal GSS poison requirement is therefore $10.5 \times 2.6 = 27.3$ ppm boron.

Preliminary uncertainty assessments suggested the $1-\sigma$ uncertainty for $[Gd]_c$ was $\pm 7.5\%$ and for DF was $\pm 5.0\%$. Using a $2-\sigma$ uncertainty allowance and an additional 20% for unspecified contingencies in Eq. 6, we obtained a Uncertain Factor UF 1.38. The GSS poison requirement was therefore $27.3 \times 1.38 = 38$ ppm boron. The safety margin was about 4 ppm boron.

6.0 Compliance to Licensing Power Limits

The licensee of an operating station is required to demonstrate to the AECB that the licensing channel and bundle power limits are observed. The utilities therefore routinely carry out compliance analysis according to certain established procedures. The current practices in CANDU 6 plants are to demonstrate compliance through simulations of reactor operations which are carried out at frequent intervals using reactor physics codes and models.

It is recognized that these simulations have inherent errors and it is important that the magnitude of these errors is carefully determined and justified, and factored in the comparisons to the licensing limits. To allow for these uncertainties, certain administrative power limits are defined in the operating procedures to assist in ensuring compliance. The fuelling engineer makes every effort to ensure the fuelling schedule results in peak powers below these administrative limits. The operating history versus performance targets in terms of transgressions above these administrative limits are carefully tracked and analyzed. The frequency at which the compliance calculations are carried out is currently 2 or 3 times a week in a CANDU 6 plant. Certain actions are required to be taken if the administrative limit is exceeded, which could potentially be an immediate reduction in reactor power.

The compliance calculations are performed usually at a time with xenon at equilibrium with flux distribution, i.e. at a time when transient xenon effects in the refuelled channels have settled. Thus the calculated maximum channel power and bundle power used for compliance should have allowance not only for uncertainties in the calculation, but also for transient powers between surveillance times. This also points out a weakness of the current after-the-fact surveillance method. The future direction is moving towards developing on-line surveillance methods which support the current practices and provide continuous assurance of compliance.

As an example to provide more details on the compliance analysis and procedure, the current simulation method, error allowances, transient power variation estimates, and refinement in methodology being developed for Point Lepreau are discussed in the following subsections.

6.1 Simulation Method

The flux / power mapping option in RFSP is used for core tracking purposes at Point Lepreau. It is based on best fitting the 102 in-core vanadium detector readings by a linear combination of a set of pre-calculated basis functions which are eigen-functions of the two-group diffusion equation. The fundamental flux shape function corresponds to a solution obtained for the latest core configuration.

The calculations are done every Monday and Thursday morning. Typically refuelling of 7-10 channels starts after each calculation, and the xenon transient effects would have settled by the time of the next calculation. The sum of the mapped bundle powers is normalized to total reactor power.

6.2 Steady State Calculation Errors

There are inherent errors in the flux and power mapping process: detector measurement errors, detector position uncertainty, accuracy and completeness of the basis shape functions, uncertainties introduced by detector flux interpolation from the mesh fluxes, and conversion of

cell fluxes to bundle powers. Furthermore, there are limitations of the mapping calculation as well. The in-core detectors are located in the central core region and do not cover the peripheral region; the harmonic flux shapes in the flux synthesis have inherent errors due to core modelling approximations and the diffusion method. Normalization to the total reactor power also introduces errors since there are uncertainties in the measured total power.

The assessment of the mapped flux error is based on comparisons to special flux scan measurements using a travelling miniature fission chamber. The assessment of the mapped channel power error is based on comparisons to heat balance data derived from predicted coolant flow rate and measured temperature increase. Currently there is an on-going extensive channel power and bundle power uncertainty assessment program, with the comparisons covering data from both Point Lepreau and G2 over extended periods of operation. The measurement data itself has uncertainty and must be considered as well. All known sources of errors are identified and examined, and their contribution to the net error quantified. Furthermore, possible correlations between the various error terms to core physics parameters (such as fuel burnup) or to core model uncertainty (such as adjuster position) are investigated.

An interim channel power and bundle power calculation uncertainty of $\pm 2.7\%$ has been in use at Point Lepreau for compliance analysis purposes. It represents the channel and bundle calculation random one-sigma uncertainty. The administrative limits are set at one-sigma and two-sigma level below the licensing limits.

6.3 Transient Powers

Transient power distributions due to xenon-free effects are estimated by means of corrections to the steady powers. These corrections are applied to the refuelled channels and their immediate eight neighbors. The correction factors were derived from detailed simulation studies of power transients after refuelling and comparing the power just after refuelling to the equilibrium power. The magnitude is of the order of a few percent, and is dependent on the location of the refuelled channel in the core. Typical values in use are those recommended in Reference 14 and given here in Table 6-1.

After the applications of the xenon-free corrections simultaneously to all affected channels, a transient power map representing the highest possible powers for each channel in between the surveillance times is created. Compliance statistics for this transient power map are also compiled.

6.4 Compliance and Transgression Statistics

The *MARGINS module in RFSP⁽¹⁵⁾ has been designed to track compliance statistics. With the steady state power map and the transient power map, the statistics of channels and bundles falling in bins of half-sigma width are compiled. Transgression over the one-sigma and two-sigma administrative limits are immediately noted.

The *TRANSG module in RFSP⁽¹⁵⁾ has been designed to compile the transgression records over time, such as number of transient violations per channel, cumulative transient violation hours per channel, overpower excursion duration periods. Typical channel power transgressions over the one-sigma and two-sigma limits are shown in Table 6-2. The transient xenon effects were

included in this tabulation.

6.5 Refined Compliance Analysis Methodology – A Probabilistic Approach

A refined compliance analysis method based on ROP style probabilistic calculation has been under development. In this approach, given a snapshot power distribution, the probability that no channel (or bundle) exceeding its corresponding licensing limit is computed.

The current framework allows three error terms: a bias error common to all channels and bundles, for example, the bias error in the bulk reactor power used in the normalization; a channel independent random uncertainty, for example, the RFSP mapping calculation random uncertainty; and a random uncertainty common to all channels and bundles, for example, the random uncertainty in the bulk reactor power.

The treatment of xenon-free correction has also been refined. The average xenon-free corrections are taken as a bias error. The random variations about the average are considered as an additional component to the channel random uncertainty. Furthermore, since these corrections will gradually disappear as xenon builds up in the fresh bundle, a time dependence of these correction factors is included in the model and used to create instantaneous snapshot power distribution at selected time instants.

For a given instantaneous power distribution, the evaluation of the compliance probability consists of three steps:

- a. Modify the power distribution to include the bias errors.
- b. For each channel (or bundle), calculate the margin to limit.
Express the margin in units of sigma of the channel random uncertainty, and evaluate the probability P_i for channel i that the limit will not be exceeded. Ignoring the common random uncertainty, the probability that each and all channel is less than its respective limit is given by the product of P_i .
- c. The probability density evaluated in step b is then combined with the probability density for the common random uncertainty. The compliance probability that no channel exceed its limit is then deduced.

The major advantage of such a probabilistic approach is that all channels and bundles are taken into account in the evaluation of compliance calculation to give an numerical and tangible quantity to measure compliance, whereas the current procedures only give a sense of over or under the administrative limits.

Sample applications of this probabilistic approach are shown in Figure 6-1 and Figure 6-2.

7.0 Conclusion

The materials presented in the previous Sections clearly illustrate the extent to which physics analysis is involved in safety and licensing of the CANDU plants. In particular, physics analysis has an essential role in defining the shutdown system performance requirements in terms of both speed and depth, and to demonstrate that the SDS's as designed can effectively mitigate any reactivity excursion or reactivity increment in credible accident scenarios.

The material presented also illustrates the general approach in physics analysis – the philosophy of defining a worst possible core state leading to the most severe consequences and most stringent demands on system performance, and making the most unfavorable assumptions in the analysis process.

In transient accident analysis, it is the neutronic kinetics behavior that drives the power variations. The neutron kinetics is directly affected by the changing core conditions, reactivity feedbacks and device movements. Thus physics analysis is closely linked to changing thermalhydraulic conditions and regulating system responses. Such couplings between physics and other disciplines have also been illustrated.

Physics analysis also plays an indispensable role in meeting routine operation requirements. Physics simulations of the reactor core operation give essential performance data such as channel and bundle power distributions which are used to ensure compliance to licensing power limits.

As discussed in previous lectures and also evident from the materials presented here, physics calculation methods have been generally well-established. It has also been shown that the uncertainties in the calculation results are important elements in the assessment of safety margins and in providing a high level of confidence of the analysis conclusions.

Table 3-1

Reactivity and Selected Powers Versus Time (30IH100)

Case No.	Time from Break (s)	Reactivity (mk)	Total Relative Power * (As if Prompt)		Broken-Loop Relative Power* (As if Prompt)		Intact-Loop Relative Power* (As if Prompt)		Max. Channel Power (As if Prompt)		Max. Bundle Power* (As if Prompt)	
			MW	Relative	MW	Relative	MW	Relative	MW	Position	kW	Position
1	0.000	0.0000	2058.4	1.0000	1025.1	1.0000	1033.3	1.0000	6.851	P-12	775.5	S12-6
3	0.100	0.0955	2067.7	1.0045	1031.9	1.0067	1035.7	1.0024	6.879	P-12	780.0	S12-7
4	0.200	0.5206	2121.5	1.0307	1070.0	1.0438	1051.6	1.0177	7.094	J-11	802.4	E11-7
5	0.300	1.2577	2260.9	1.0984	1163.2	1.1347	1097.7	1.0624	7.737	H-8	892.1	H6-7
6	0.400	2.0497	2508.1	1.2185	1321.4	1.2890	1186.7	1.1485	8.357	H-8	1029.3	H6-7
7	0.500	2.8267	2863.4	1.3911	1544.3	1.5064	1319.2	1.2766	10.447	N-7	1211.4	H6-7
8	0.600	3.4484	3321.6	1.6137	1829.6	1.7848	1492.0	1.4439	12.480	N-7	1442.0	H6-7
9	0.700	3.7399	3851.2	1.8710	2152.4	2.0997	1698.9	1.6441	14.752	M-6	1702.9	H6-6
10	0.800	3.9787	4409.5	2.1422	2488.8	2.4279	1920.7	1.8588	17.066	M-6	1973.8	H6-6
11	0.883	4.1981	4887.3	2.3743	2776.3	2.7083	2111.0	2.0430	19.060	M-6	2207.3	H6-6
12	0.972	4.1803	5408.7	2.6276	3088.9	3.0132	2319.8	2.2450	21.260	M-6	2459.9	H6-6
13	1.037	3.9954	5746.9	2.7919	3292.7	3.2121	2454.2	2.3751	22.832	M-6	2628.6	J5-6
14	1.103	3.5684	5949.1	2.8901	3420.4	3.3366	2528.7	2.4472	24.129	O-6	2776.8	N5-6
15	1.169	2.9158	5928.5	2.8801	3422.1	3.3383	2506.4	2.4257	24.952	O-6	2875.5	P5-6
16	1.229	2.1414	5683.6	2.7612	3293.6	3.2130	2390.0	2.3130	24.980	O-6	2892.8	P5-6
17	1.280	1.2960	5303.2	2.5764	3084.4	3.0088	2218.9	2.1474	24.360	O-6	2837.1	P5-6
18	1.329	0.4691	4819.9	2.3416	2813.3	2.7444	2006.6	1.9419	23.210	P-7	2722.9	Q6-6
19	1.380	-0.3789	4257.7	2.0684	2492.6	2.4316	1765.0	1.7081	21.535	P-7	2543.6	Q6-6
20	1.429	-1.3737	3699.8	1.7974	2170.7	2.1176	1529.1	1.4798	19.725	Q-8	2331.1	Q6-6
21	1.476	-2.5971	3161.0	1.5357	1857.1	1.8116	1304.0	1.2619	17.748	Q-8	2097.2	Q6-6
22	1.521	-4.1264	2650.5	1.2876	1557.9	1.5198	1092.6	1.0574	15.646	Q-8	1866.8	S8-6
23	1.564	-6.0550	2178.0	1.0581	1280.0	1.2487	898.0	0.8691	13.668	S-8	1636.5	S8-6
24	1.635	-12.0003	1293.3	0.6283	758.9	0.7403	534.3	0.5171	9.171	S-10	1097.9	S8-6
25	1.750	-22.9944	680.7	0.3307	399.8	0.3900	280.8	0.2718	5.171	S-10	627.6	T9-6
26	1.827	-40.8616	383.0	0.1861	227.1	0.2215	156.0	0.1509	2.824	T-9	343.0	T9-7
27	1.910	-62.6625	234.8	0.1141	141.2	0.1377	93.6	0.0906	1.270	U-10	156.5	N5-10
28	2.088	-74.0410	177.8	0.0864	107.7	0.1051	70.1	0.0678	0.921	N-5	130.1	N5-10
29	3.000	-73.7604	136.6	0.0663	82.4	0.0804	54.2	0.0524	0.701	N-5	99.7	N5-10
30	4.000	-73.5523	112.7	0.0548	67.8	0.0661	44.9	0.0435	0.576	N-5	82.4	N5-10
31	5.000	-73.3163	96.7	0.0470	58.0	0.0566	38.7	0.0374	0.492	N-5	70.8	N5-10

Table 6-1 Recommended Xenon-Free Correction Factors

Parameter	Suggested Increase for Outer Core (%)	Suggested Increase for Inner Core (%)
Channel power of refuelled channel	6.0	3.5
Channel power of first (nearest) neighbours of refuelled channel	5.0	3.0
Channel power of diagonal neighbours of refuelled channel	5.0	2.5
Bundle power of bundle in refuelled channel	8.0	5.5
Bundle power of bundle in first (nearest) neighbours of refuelled channel	6.5	6.5
Bundle power of bundle in diagonal neighbours of refuelled channel	6.0	4.5

Table 6-2 Sample Channel Power Transgressions

FPD	3994	3997	4001	4003	4006	4011	4015	4017
Channels with CP within 2.7% of limit	N16	N05						
Number of channels within 2.7% of limit	1	1	0	0	0	0	0	0
Channels with CP within 2.7-5.4% of limit	N15 O16	N06 N07 M05 O05 L18 P16 T09	P07 P08 F15 P06	P11 P10 L05 M05 K16	N16 M15 N15 M09 N08	P17 P16 K08 J16 S13 K09 H16 L08 J09 J08		R16
Number of channels within 2.7-5.4% of limit	2	7	4	5	5	10	0	1

Note: There was a 1% derating in the period FPD 3997 to 4001.
There was a derating of up to 2% in the period FPD 4015 to 4017.
These deratings have been "ignored" here.

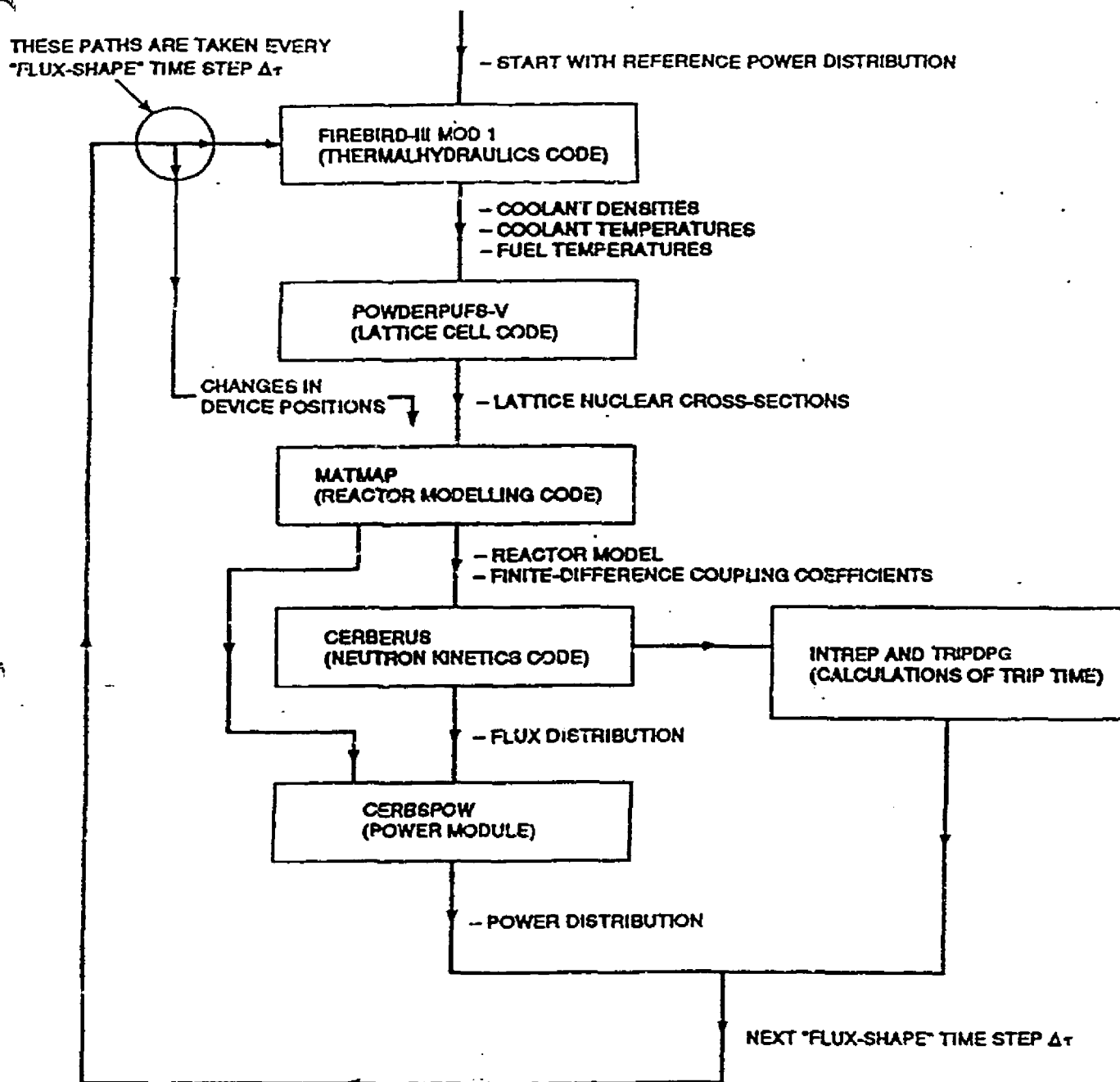


Figure 3-1 Sequence of Calculations in Simulation of Transient

NOTE: Numbers in boxes indicate the channel group number

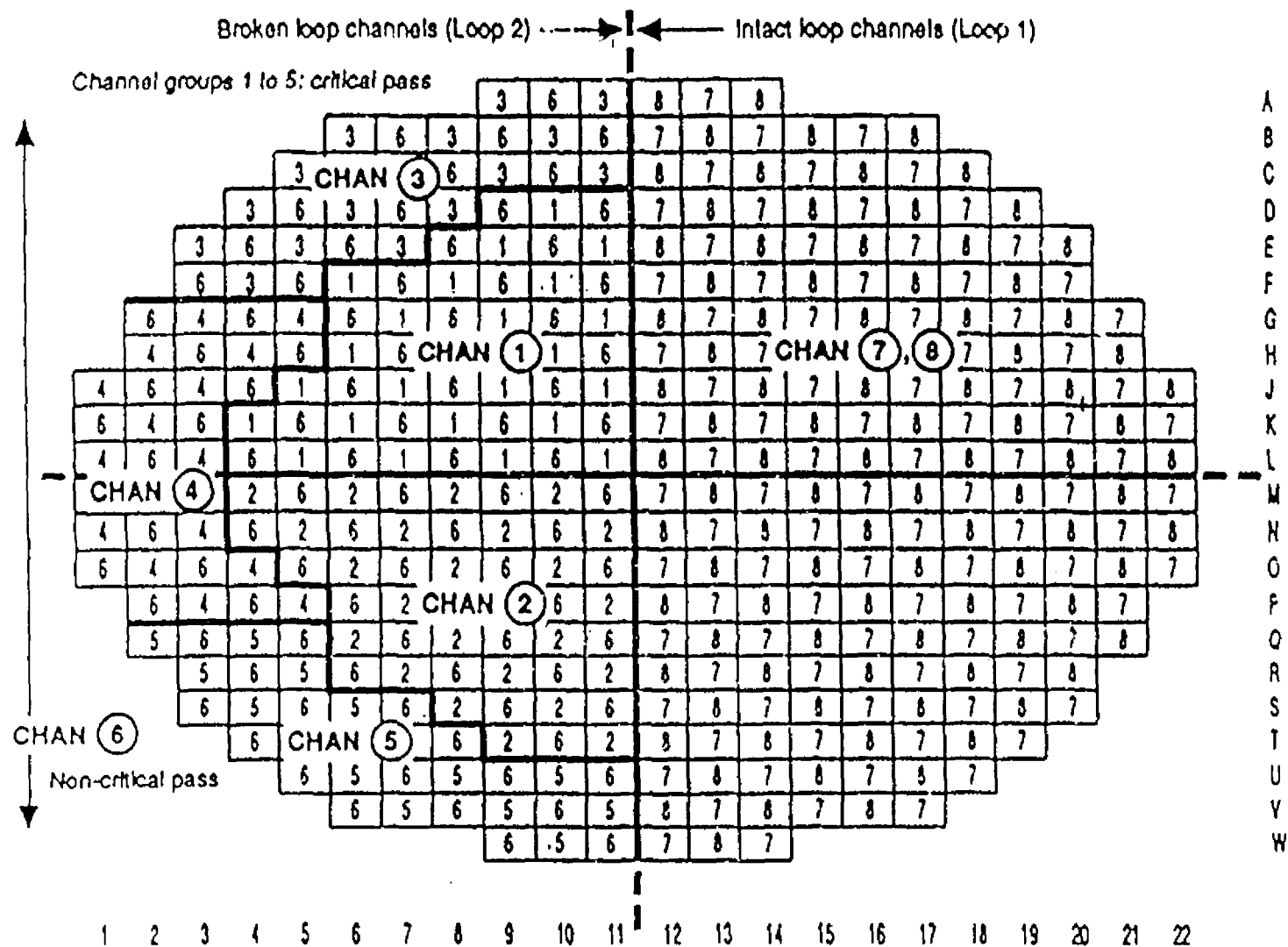


Figure 3-2 Thermohydraulic channel group representation

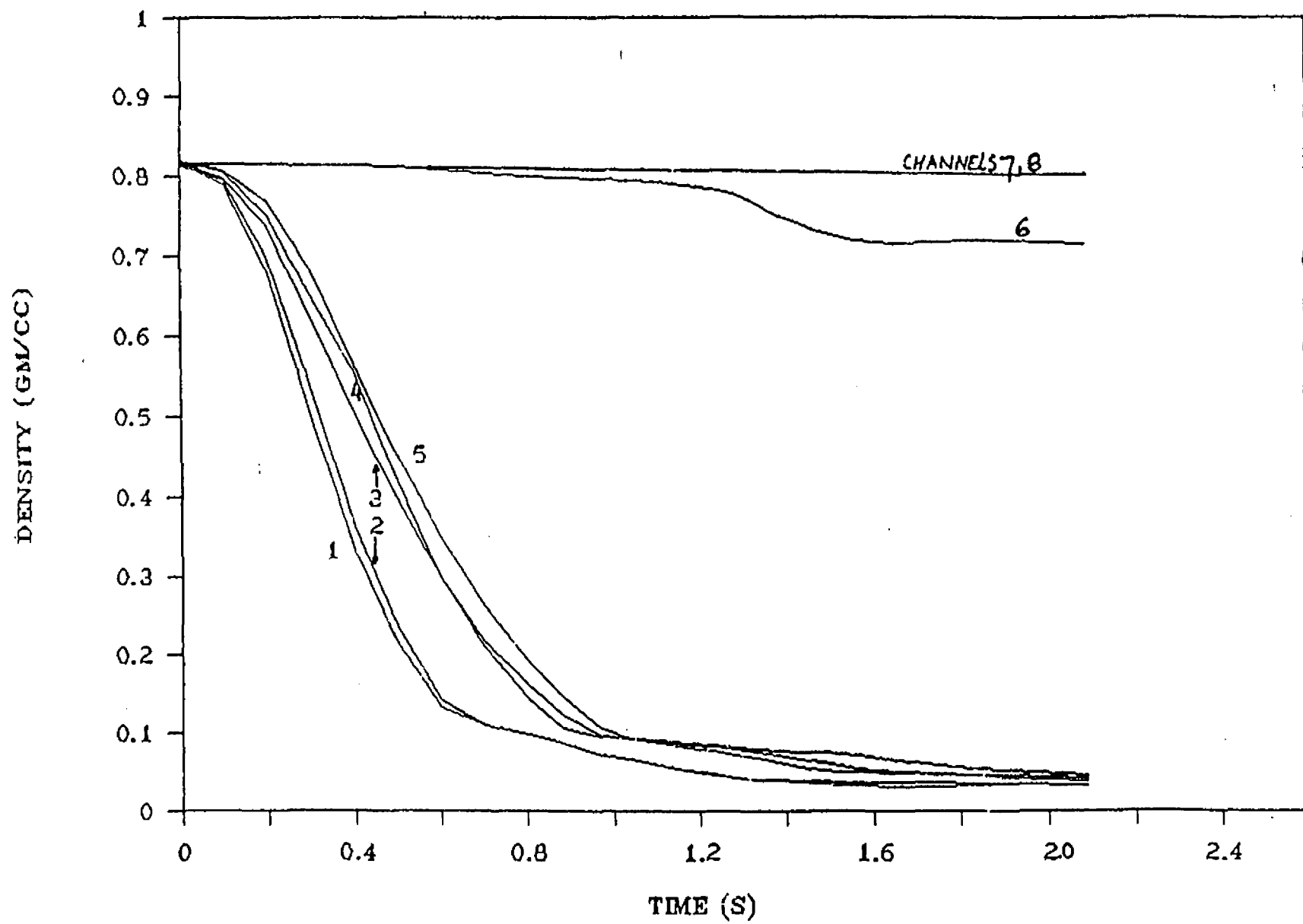


Figure 3-3 Flux Squared Weighted Coolant Density - 30% RHH Break

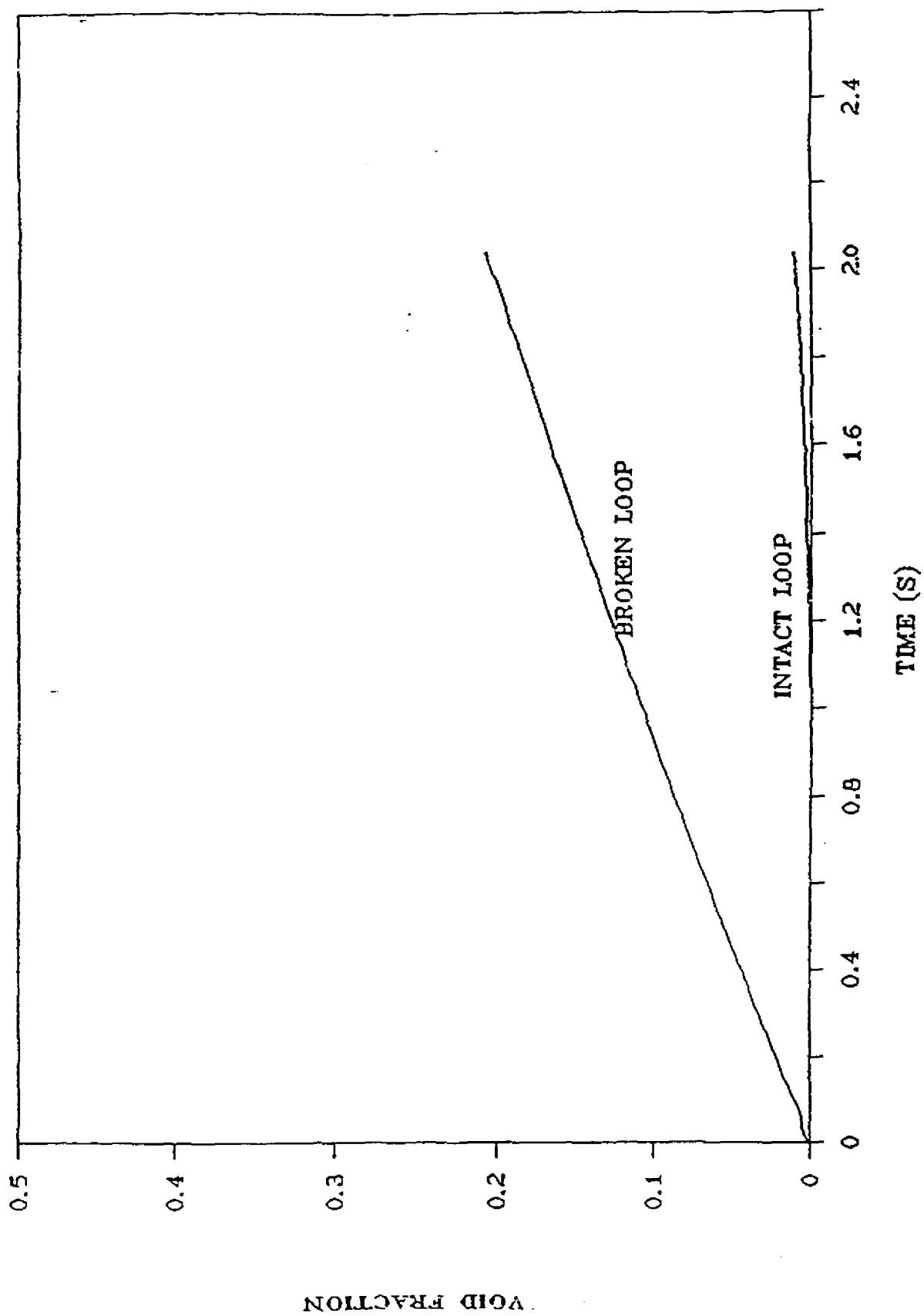


Figure 3-4 Total Loop Void Fraction - 30% R/H Break

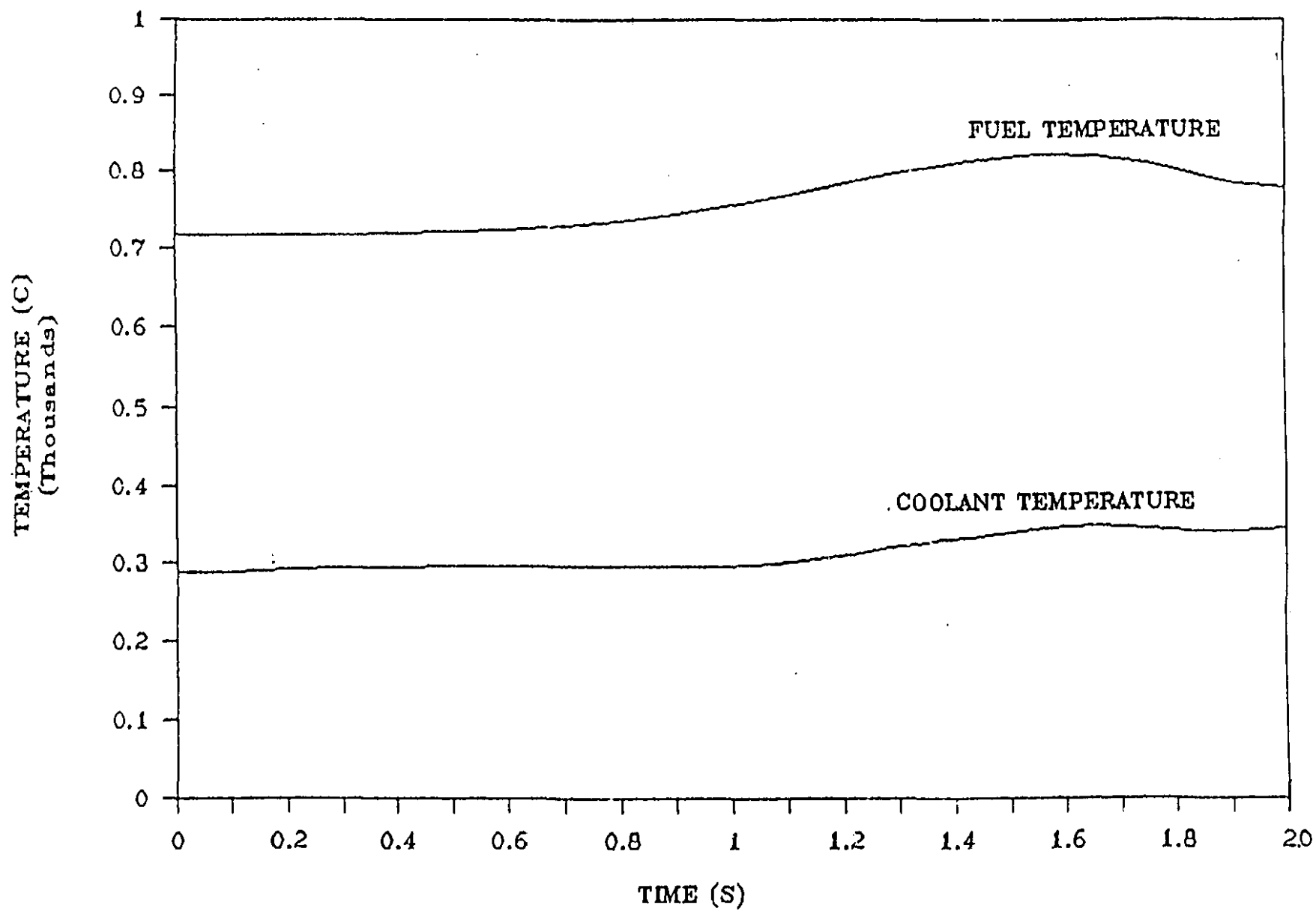


Figure 3-5 Flux Squared Weighted Fuel and Coolant Temperature - 30% RIH Break

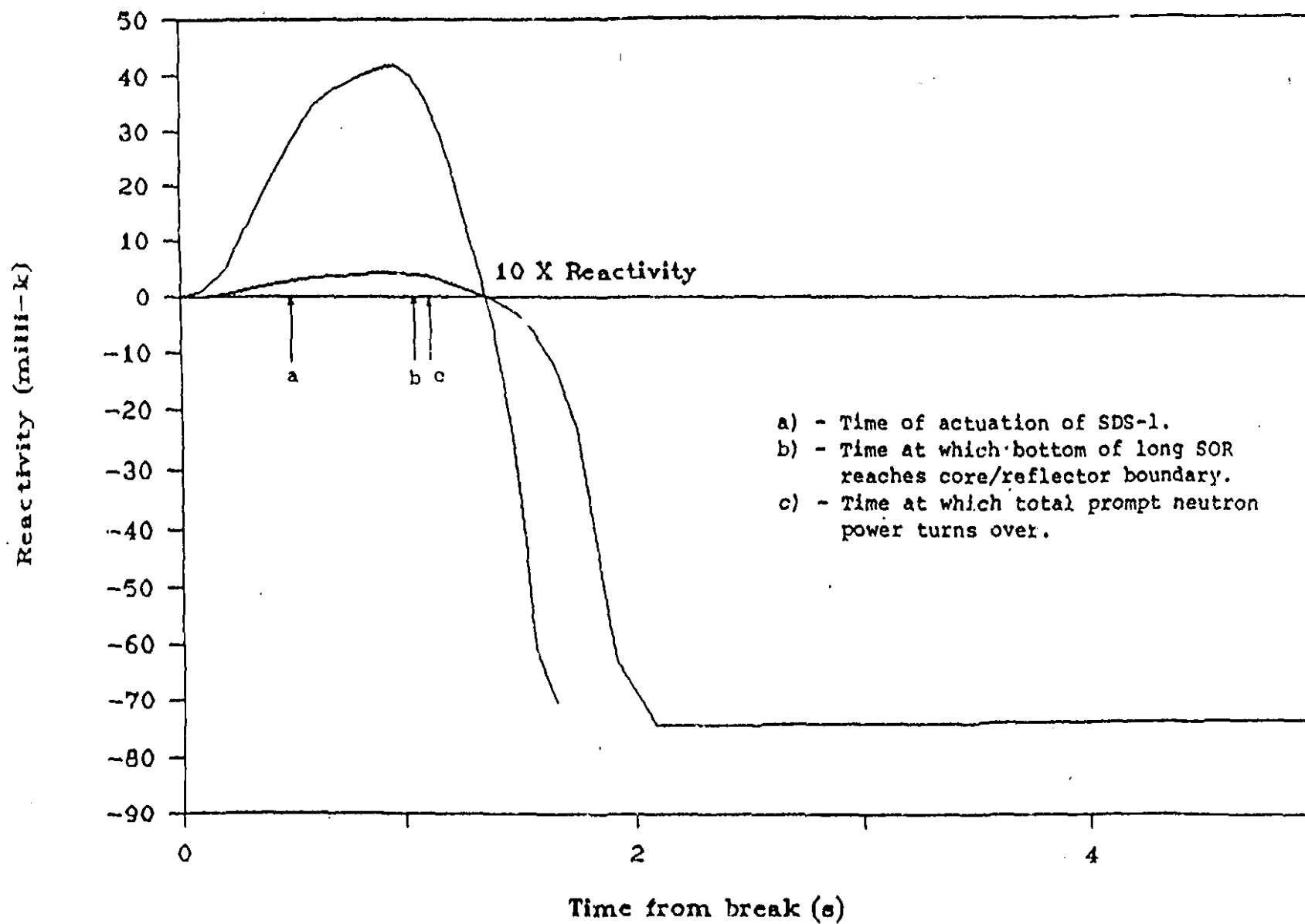


Figure 3-6 Reactivity vs. Time - 30% RIH Break

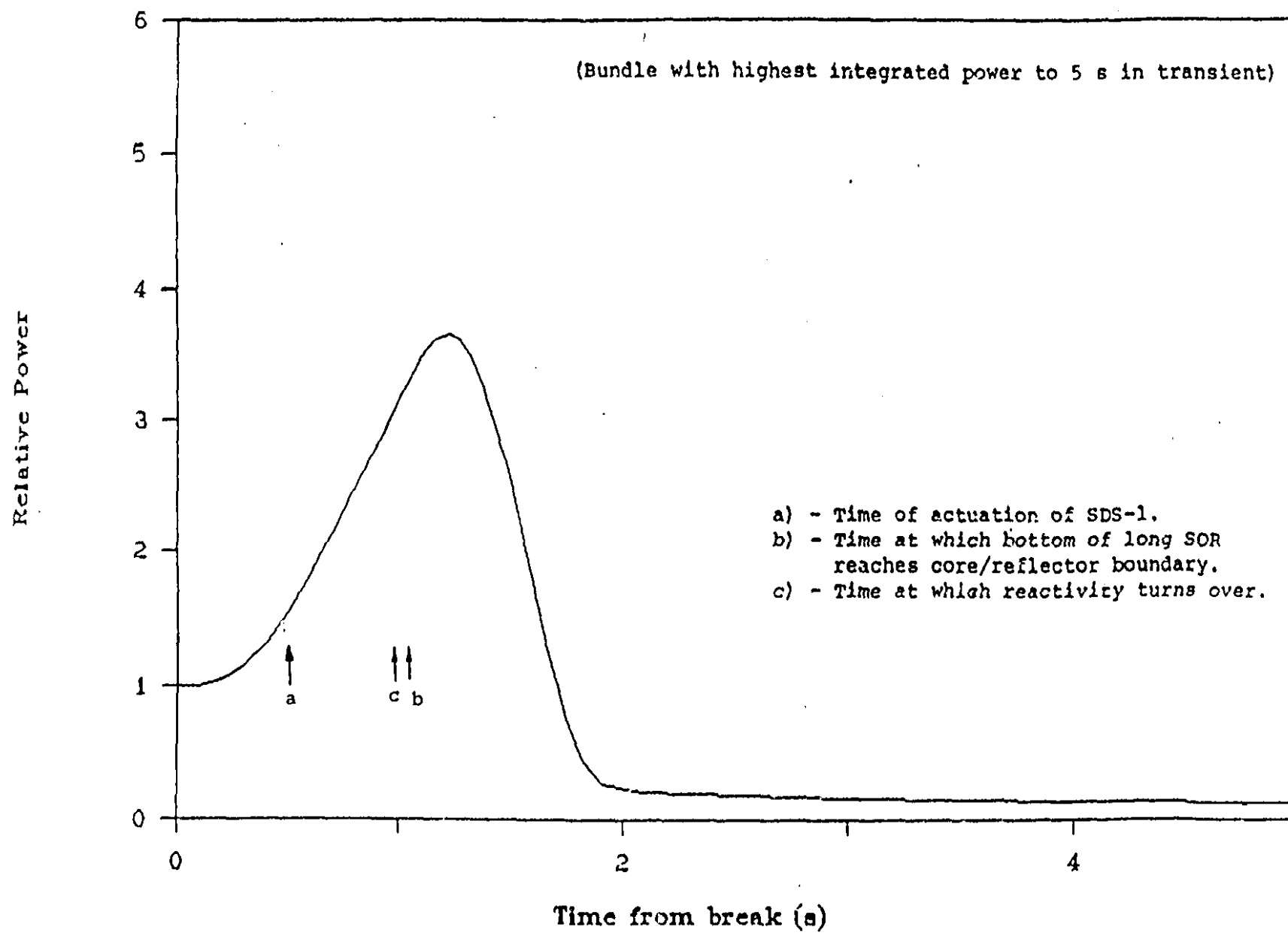


Figure 3-7 Power of Bundle Q6/7 (Initial Power 741.3 kW) - 30% RIH Break

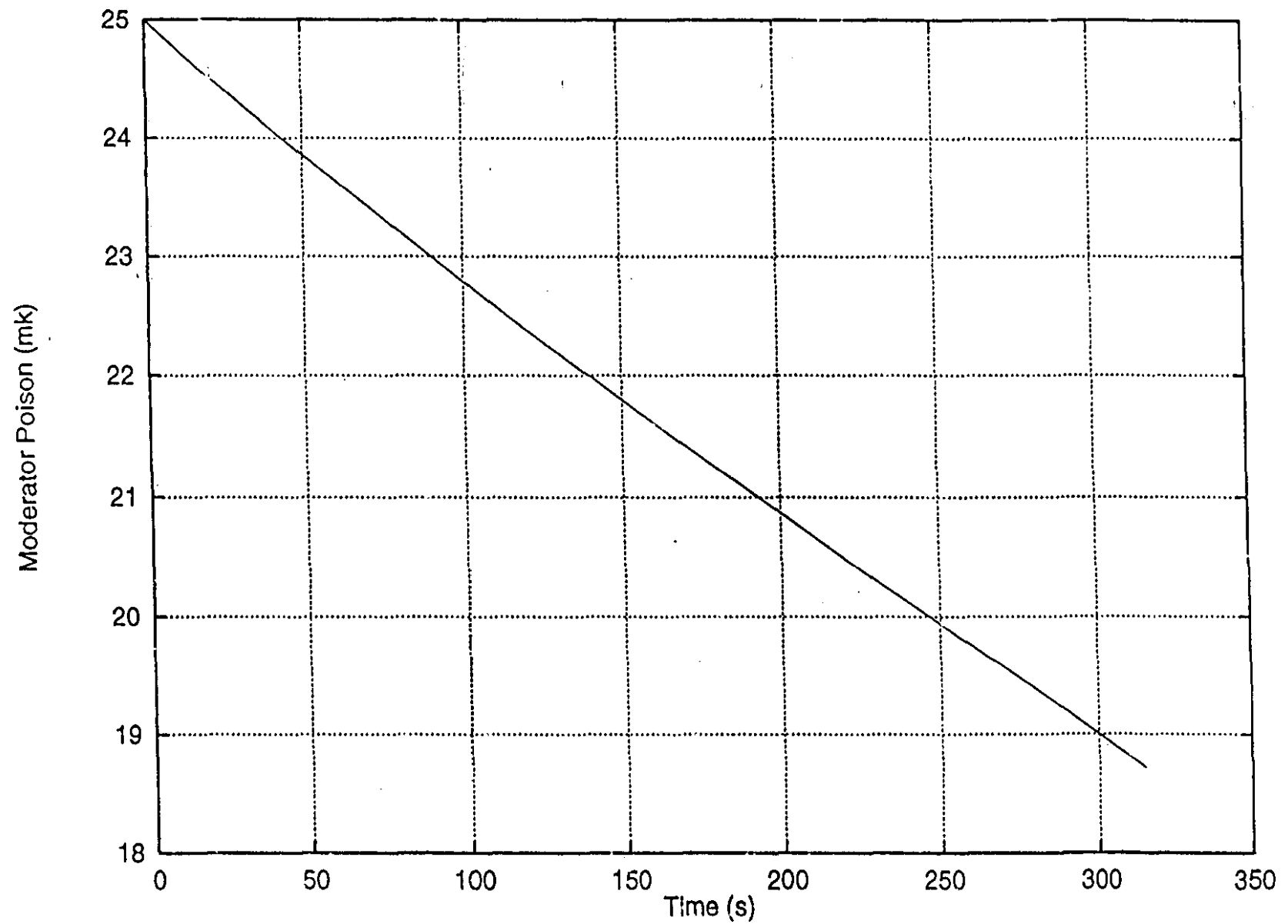


Figure 4.4-1 In-Core LOCA Test – Moderator Poison Load Transient (COMETES Results)

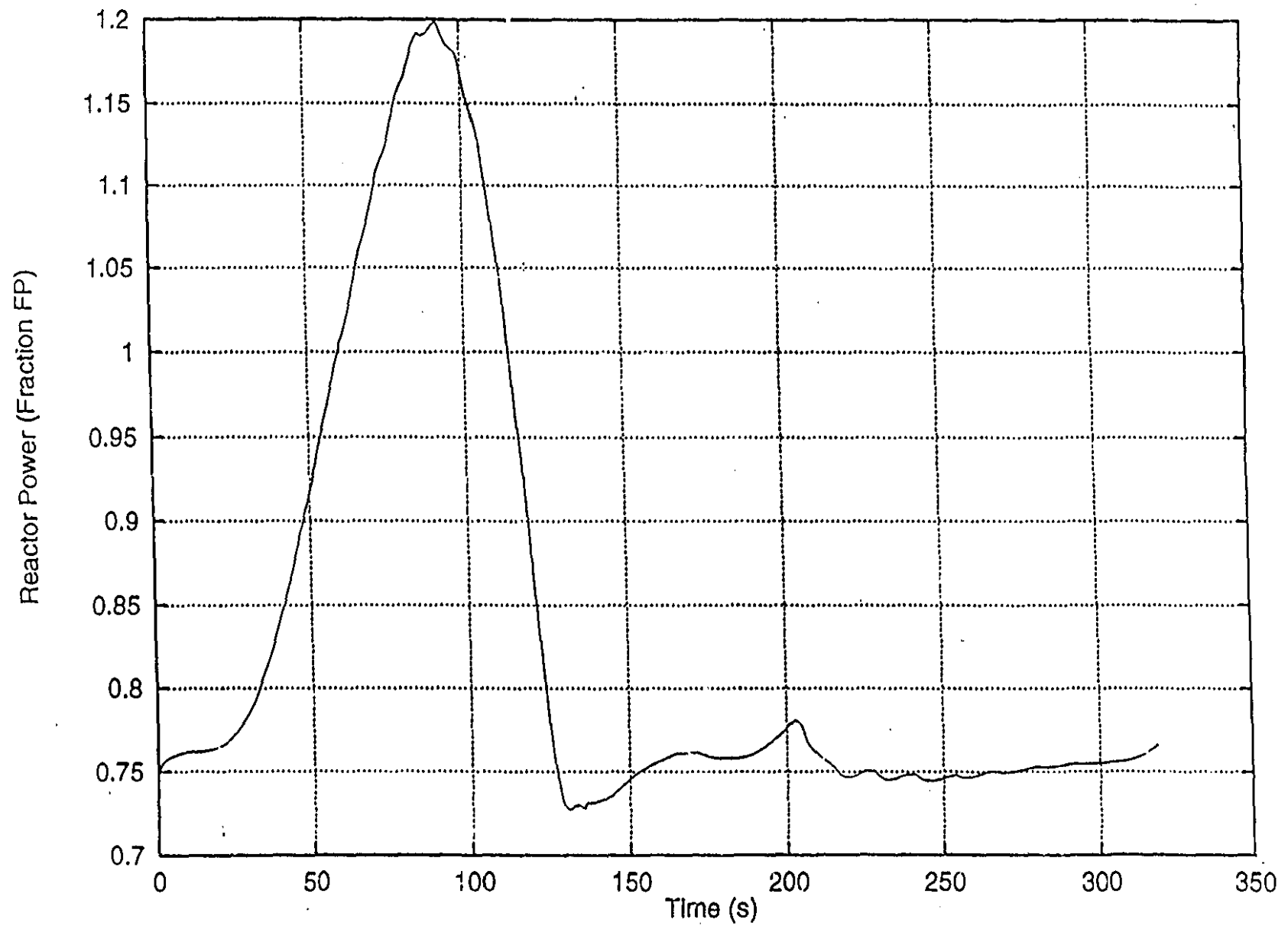


Figure 4.4-2 In-Core LOCA Test – Reactor Power Transient

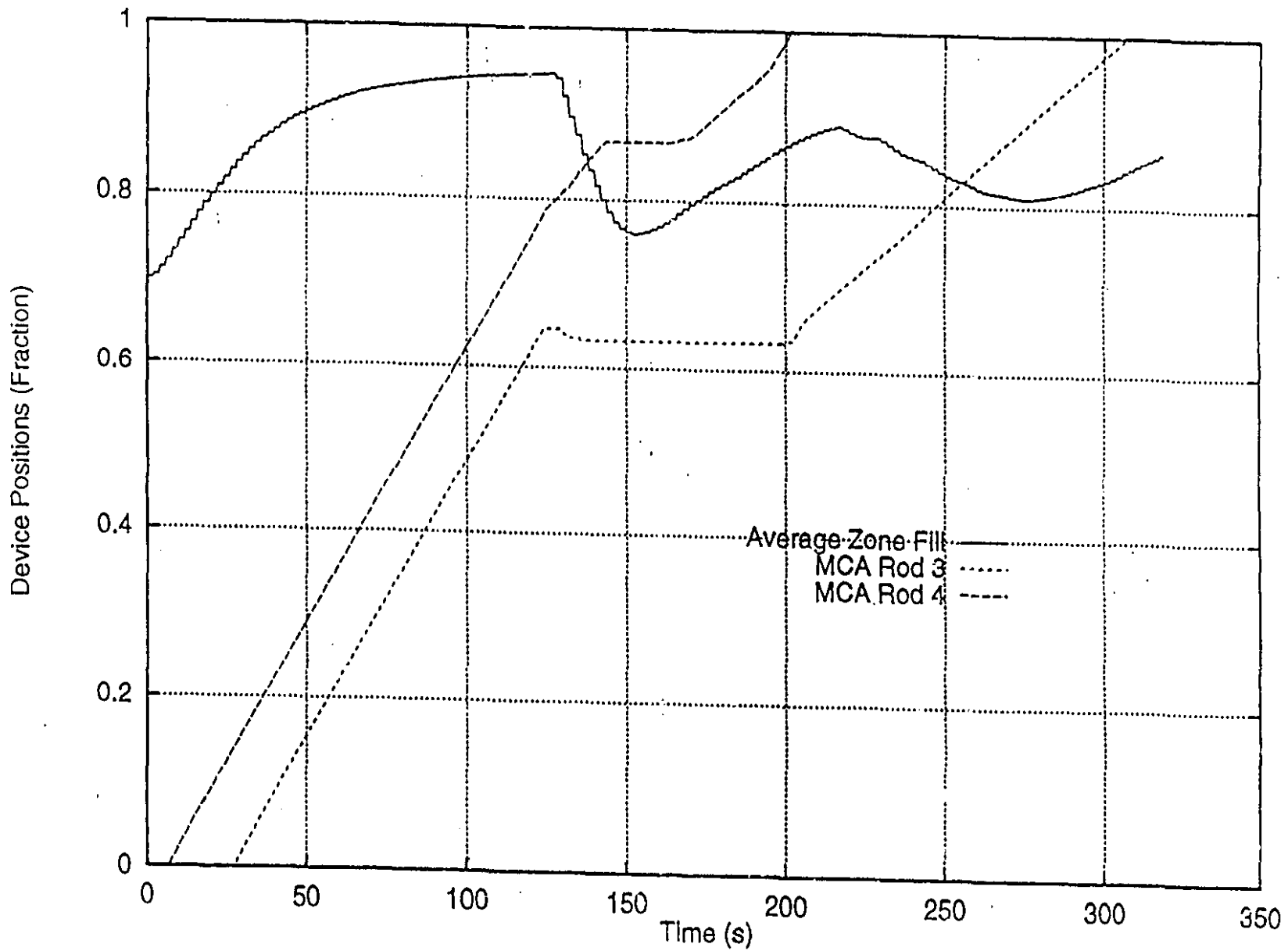


Figure 4.4-3 In-Core LOCA Test – Average Zone Fill and Device Positions

Figure 6-1 Sample results of Compliance Probability Calculations

Summary of Compliance - (All Probabilities in Percent)

Previous "Steady State"

3835.635 FPD CP Compliance Probability 98.00 BP Compliance Probability 98.00

Time (FPD)	Channel Refuelled	CP Compliance Probability Before Fuelling	CP Compliance Probability After Fuelling	BP Compliance Probability Before Fuelling	BP Compliance Probability After Fuelling
3835.948	Q09	98.00	97.30	98.00	99.84
3836.663	N05	97.52	97.63	99.83	99.84
3836.764	P14	97.61	97.68	99.84	99.84
3836.852	P15	97.66	89.52	99.84	99.77
3836.953	N02	91.78	92.10	99.79	99.78
3837.604	V11	96.55	96.61	99.83	97.81
3837.700	K11	96.82	96.94	98.59	98.63

Current "Steady State"

3840.467 FPD CP Compliance Probability 97.52 BP Compliance Probability 99.83

Interval Averages: CP Compliance Probability 96.79
BP Compliance Probability 99.33

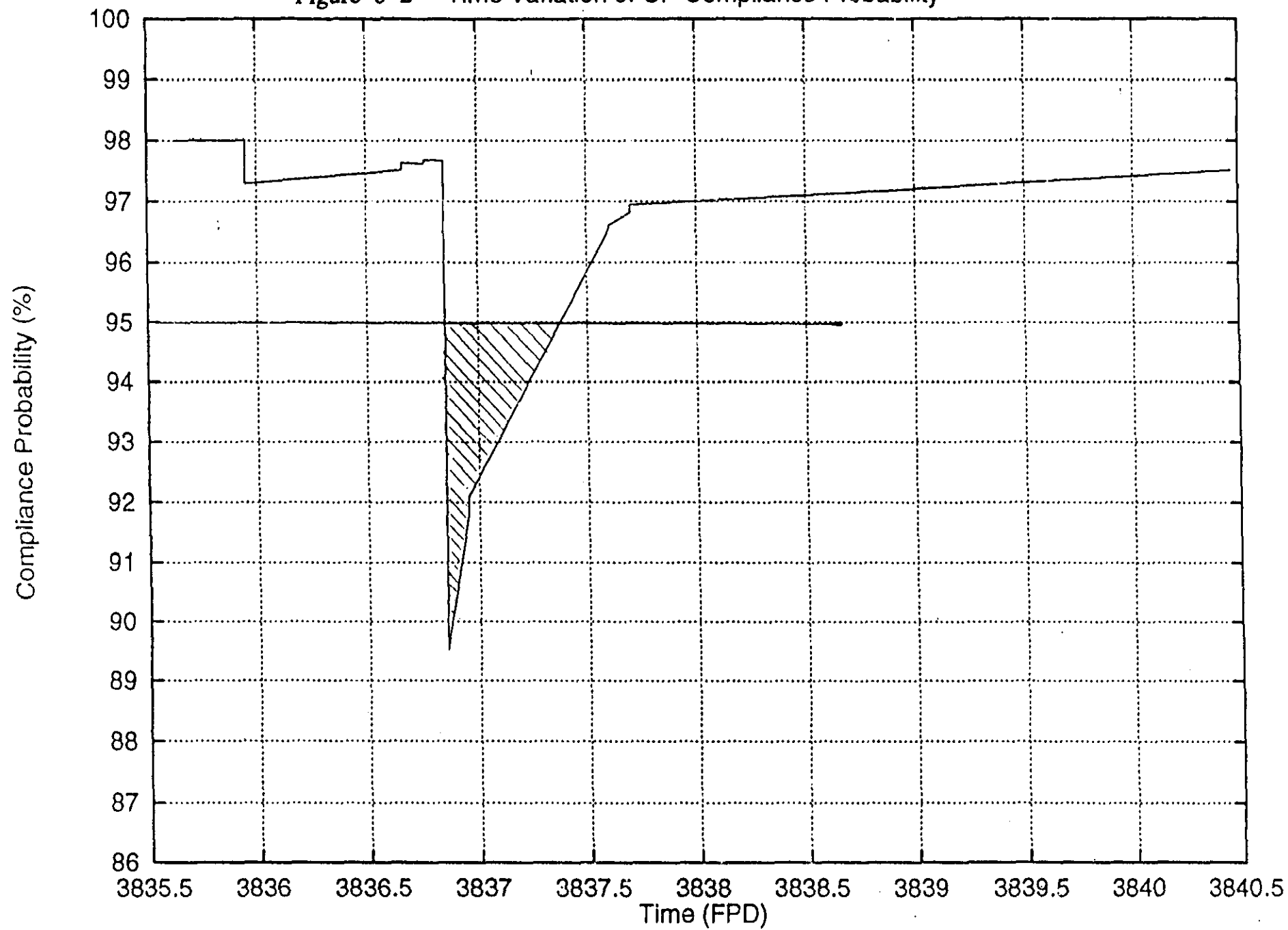
Lowest CP Compliance Probability within Interval: 89.52
Occured after fuelling of Channel P15

Lowest BP Compliance Probability within Interval: 99.77
Occured after fuelling of Channel P15

CP Non-Compliance Statistics for Assumed Instantaneous Limits

Target Limit(%)	From (FPD)	To (FPD)	Total Time(h)	Average Probability While Below Limit(%)	Integrated Area Under Target Limit (%-h)
94.00	3836.852	3836.953	2.42	90.65	14.44
	3836.953	3837.231	6.67	93.05	
			9.09	92.41	
95.00	3836.852	3836.953	2.42	90.65	25.29
	3836.953	3837.378	10.19	93.55	
			12.61	92.99	
96.00	3836.852	3836.953	2.42	90.65	39.68
	3836.953	3837.524	13.71	94.05	
			16.13	93.54	
97.00	3836.852	3836.953	2.42	90.65	58.04
	3836.953	3837.604	15.63	94.32	
	3837.604	3837.700	2.30	96.71	
	3837.700	3837.978	6.66	96.97	
			27.02	94.85	
98.00	3835.948	3836.663	17.17	97.41	141.04
	3836.663	3836.764	2.43	97.62	
	3836.764	3836.852	2.11	97.67	
	3836.852	3836.953	2.42	90.65	
	3836.953	3837.604	15.63	94.32	
	3837.604	3837.700	2.30	96.71	
	3837.700	3840.467	66.39	97.23	
			108.45	96.70	

Figure 6-2 Time Variation of CP Compliance Probability



Lecture 11 FUELING STRATEGIES AND CYCLES

Introduction by G. Brenciaglia

Lecture #6 explained how the initial fuel load is selected and Lecture #8 covered the target fueling for which the reactor was designed. The fueling strategy that the operators need to get there must answer the following questions:

- a) When does fueling start?
- b) What criterion is used to decide the fueling priority of individual channels?
- c) How much of the low burnup fuel can be recycled?

To answer these questions the fueling engineer needs information that usually comes from experience in similar plants:

- the maximum sustainable fueling rate (typically channel visits per day) that the fueling machines can maintain during their break-in period;
- threshold for sudden power increases that might lead to fuel defects, as a function of fuel burnup.

Chapters 5 and 6 of Reference 5 describe the experience obtained in commercial CANDU power reactors (Pickering and Bruce). That experience shows that even after equilibrium fueling has been reached many problems can move fueling away from the target pattern. An organization with simulation codes capable of providing guidance to the fueling engineer is needed to recover from these perturbations. The reference reports many comparisons of the powers calculated by the simulation tools with instrumentation readings. This type of monitoring is essential in judging the accuracy achieved in the simulations. The monitoring must confirm the results of the commissioning tests at the start of reactor operations.

Many perturbations move the reactor away from equilibrium fueling as shown in Reference 5. I want to briefly review strategic changes in fuel cycle that can be intentionally introduced after years of operation. CANDU reactors are particularly flexible for these changes because of their on-power fueling. The timing of the change and its rate can be adapted to the objectives of the new cycles. Just a listing of these potential changes gives a measure of the flexibility that may be required.

Table 1. Potential Fuel Cycles

Cycle	Objectives
Slightly Enriched Fuel (SEU)	Increased power, decreased used fuel out
Mixed Oxide Fuel (MOX)	Dispose of weapon plutonium
Actinide Burner (AB)	Convert non-fissile actinide to fissile
Thorium Fuel	Converts fertile thorium to fissile uranium

The heavy water reactors have also dynamic characteristics that are relatively insensitive to the fuel material, because of the long lifetime of their thermal neutrons - 0.14s versus 0.0002s for light water cores. Therefore, they can use different types of fuel without changes to their control mechanisms.

Differential loading of fissile material between different channels and between elements of individual bundle makes it possible to reduce maximum element rating and maximum channel power with advanced fuel cycles. Hence it becomes feasible to increase reactor output for current cores (up to 17% in recent work). It also makes it feasible to modify the void coefficient of reactivity to achieve a higher degree of passive safety. Figure 1 shows how slight enriched elements (SEU) can be mixed in a fuel bundles with elements containing depleted uranium (DU) and a burnable neutron absorber (Dy) to achieve the reduction in rating and in void coefficient. Figure 2 gives the characteristic burnup and power transients on loss of coolant for this fuel relative to natural uranium fuel in a CANDU reactor.

These major changes introduced in an operating reactors are a topic of many recent studies, but will not be further discussed here, unless time and interest warrant it. We can, however, conclude that there are a variety of opportunities for new cycles, and the reactor physicist must prepare his simulation tools to handle them.