Core-Physics Aspects of Safety Analysis

by

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CANDU Emergency Shutdown Systems

- Two systems: SDS-1, SDS-2
- Physically, logically, functionally separate
- Each fully capable
- Each must meet requirement of 0.999 (99.9%) availability (demonstrated by testing)
- When both systems are called upon, probability of "no shutdown" = $10^{-3} * 10^{-3} = 10^{-6}$
  - incredible

Reactor physics provides an essential input into reactor safety analysis

Basic role of reactor physics is to:

- determine neutron balance (reactivity)
- determine rates of neutronic reactions in space and time
- provide evolution of fundamental quantities

Quantities of Interest

- core reactivity,
- neutron flux distribution,
- reactor bulk power and power distribution
- heat energy added to fuel (fuel enthalpy)

The physics analysis wants to:

- understand/model the action of the reactor protective systems in recognizing accident situations and actuating the shutdown system(s)
- demonstrate that the following are satisfied post-accident:

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Demonstrate:

- performance of either shutdown system is such that the core is rendered and remains subcritical,
- neutron flux and power are reduced to small values everywhere in reactor
- energy added to fuel does not result in unacceptable consequences

The physics analysis cannot be done in isolation from other components of safety analysis. It provides input to the thermalhydraulics, fuel, fuel-channel, and radiation-dose analyses. It also needs selected input, especially thermalhydraulics data, such as coolant density and temperature.

$$k_{eff} = \frac{\text{Rate of neutron production}}{\text{Rate of neutron loss (by absorption and leakage)}}$$

$$\rho = 1 - \frac{1}{k_{eff}}$$

Neutron Kinetics

Large-LOCA power pulse: fast transient

Power rises in fraction of a second, shutdown-system action is initiated within a second, and power is turned over and reduced to small values within a few seconds

Thus, detailed physics simulation usually extends to a few (~5) seconds after postulated time of break

1 milli-k = 0.001

A reactivity of 1 milli-k may seem small, but its effect may not be inconsequential, depending on other nuclear characteristics of the core.

The rate at which the neutron population (and, consequently, the power) will change will depend on the mean generation time T, the average time interval between successive neutron generations

Simplistic treatment of kinetics: power varies exponentially with reactivity and with time (in units of generation time T):

$$P = P_0 \exp\left(\frac{\rho t}{T}\right)$$

Simplistic treatment does not account for delayed neutrons. Neutrons produced in fission are either prompt or delayed. The prompt-neutron lifetime Λ (average time interval between birth of a neutron and its absorption in a subsequent fission reaction) in the CANDU lattice is approximately 0.9 millisecond.

If no delayed neutrons, mean generation time = prompt-neutron lifetime.

In that case, reactivity of 1 milli-k would lead to a power increase by a factor of 3 per second, a very fast rate of change!

(In LWRs, L is about 30 times shorter! The rate of change of power would then be 30 times as great for the same reactivity.)

Delayed neutrons, although only ~0.6 %, reduce rate of power change considerably.

Delayed neutrons are produced in beta decay of fission products (6 groups of precursors) with half-lives from 0.2 s to 50 s.

In CANDU, also photoneutrons; 9 groups of precursors, time constants = hundreds to tens of thousands of seconds.

“Effective” (weighted-average) mean generation time ~ 0.1 s, much (~ 100 times) longer than prompt-neutron lifetime.

Reactivity of 1 milli-k then leads to increase in power by only about 1.01 per second, compared to 3 per second without delayed neutrons.

Physics Analysis for Regional Overpower Protection (ROP)

Loss of regulation (LOR): Reactor Regulating System (RRS) loses control of global or local power

Regional Overpower Protection (ROP) system consists of in-core detectors which measure the local neutron flux

ROP system designed so that a reactor shutdown is initiated when detector readings reach appropriate "trip setpoints", judiciously selected to prevent fuel dryout in any fuel channel

ROP system is in fact a double system: one for each shutdown system

CANDU 6 ROP:

- 54 SDS-1 detectors (vertical)
- 34 SDS-2 detectors (horizontal)

Detectors "trip" on high power

Each ROP system is subdivided into three logic channels (D, E, F for SDS-1; G, H, J for SDS-2)

Triplicated logic

- If any detector in a logic channel reaches its trip setpoint, the channel is tripped
- When 2 of the 3 logic channels are tripped, the corresponding shutdown system is actuated

Role of the ROP analysis is to determine appropriate trip setpoints for detectors such that, for anticipated flux shapes under both normal and off-normal operation, fuel dryout will be avoided if there is an increase in local or global power

ROP analysis based on hundreds of flux shapes, spanning a variety of reactivity-device positions, power manoeuvres such as a power recovery following a short shutdown, xenon transients and oscillations, scenarios where zone-control compartments drain, etc.

- The role of reactor physics is to calculate flux distributions and provide to the ROP code, for each flux shape, 3-d power distribution (channel powers) and corresponding detector fluxes
- ROP code selects detector positions and/or setpoints which protect the reactor against LORs but allow normal operation to proceed

Equation for ROP-Detector Setpoint

Let us use the following labels:
j for in-core detectors [j = 1, ...]
m for fuel channels [m = 1,...,380 for CANDU 6]
k for design-basis perturbations to the flux shape [k = 1, ...(hundreds)]

The basic ROP protection requirement is to protect against any perturbation k. That is, each logic (safety) channel must trip before the power in any fuel channel reaches the critical channel power for that channel. [While the minimum requirement is that two out of three logic channels must trip, we normally increase the requirement to **all** logic channels, for conservatism.]

Now, if

$\phi_0(j)$ = flux at detector j in reference (nominal) flux shape at 100% full power (FP)

$\phi_k(j)$ = flux at detector j in flux shape k at FP

D_o = the detector's nominal calibration at FP

$$r_{CPR}(k) = \underset{m}{Min}\left[\frac{CCP(m,k)}{CP(m,k)}\right]$$

= minimum critical power ratio for flux shape k
= minimum value of ratio of critical channel power (CCP) to actual channel power (CP) over all fuel channels

then, for detector j to protect against a loss of regulation in perturbed flux shape k, the trip setpoint TSP_j would have to be

$$TSP_j = D_o(j)\frac{\phi_j(k)}{\phi_o}r_{CPR}(k)$$

However, since there are many detectors in the core, not every detector need protect every flux shape.

All that is necessary is to define detector locations, trip setpoints (TSP) and channelizations such that for every design basis case there is at least one detector j in each safety channel which satisfies

$$TSP_j \leq D_o(j)\frac{\phi_j(k)}{\phi_o}r_{CPR}(k)$$

Physics Analysis for a Large LOCA

Coolant voiding in CANDU introduces positive reactivity and promotes an increase in power

[CANDU Positive Void Coefficient]

Positive Void Coefficient

Unlike situation in LWR, CANDU coolant is separate from moderator

Coolant provides a very small fraction of the moderation (much smaller volume)

Therefore loss of coolant does not act as a loss of moderator

The positive void coefficient is caused by changes in neutron spectrum and flux distribution in lattice cell. Mainly:

- Decrease in resonance absorption (+)
- Increase in fast-fission rate (+)
- In irradiated fuel, spectrum effects in Pu and fission products (-)

- Large loss of coolant (LOCA) presents the greatest challenge to CANDU shutdown systems in terms of rate of positive reactivity insertion
- Capability for a power surge (power pulse) beyond capability of Reactor Regulating System to control

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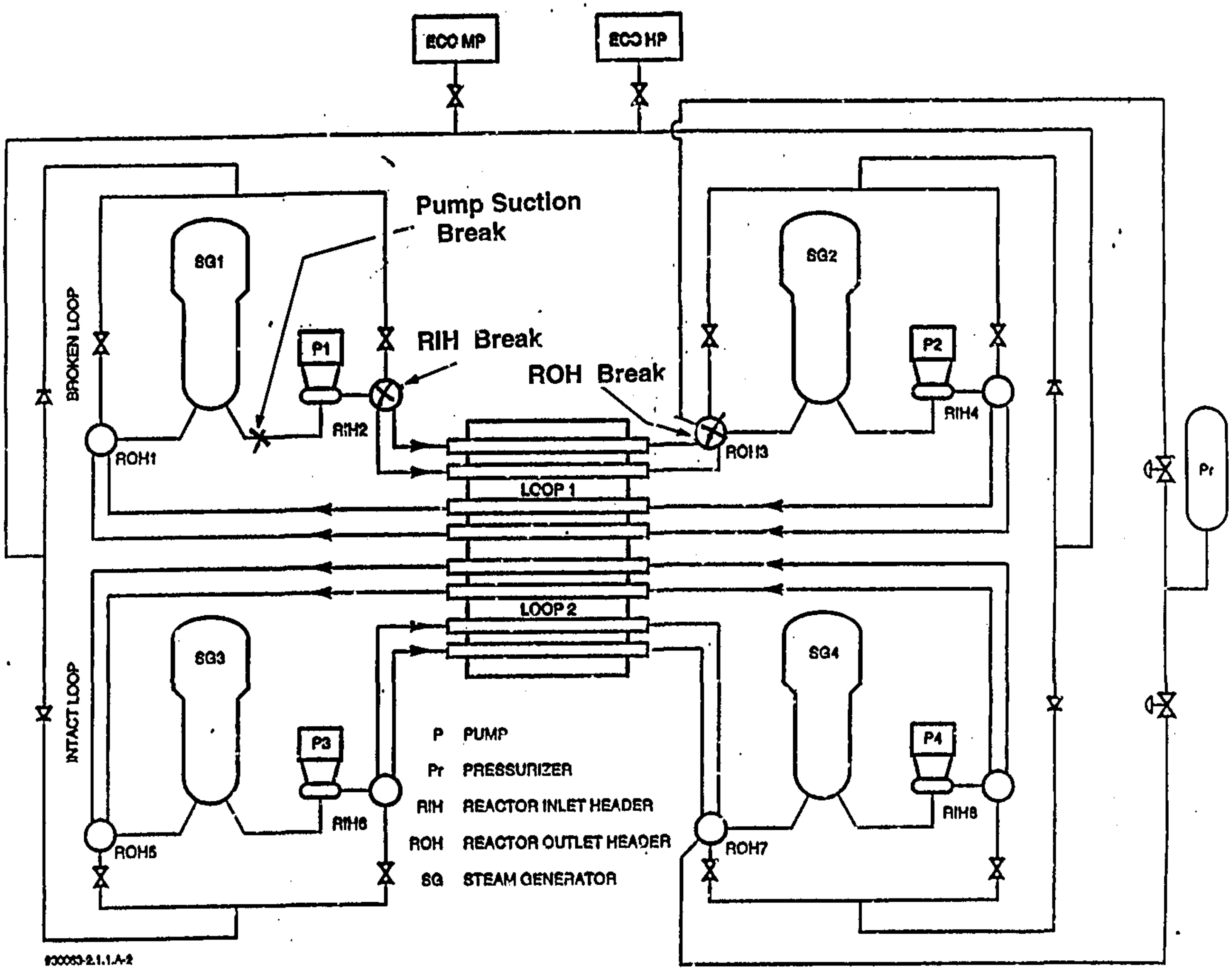

Figure 3
Examples of Break Locations Giving Rise to a Large LOCA

Main Steps in Physics LOCA Analysis

- simulation of pre-accident reactor configuration
- modelling of postulated perturbation (the LOCA, modelled in conjunction with a thermalhydraulics calculation)
- simulation of early part of the accident, prior to shutdown-system actuation

cont'd

Main Steps in Physics LOCA Analysis (Cont'd)

- calculation of the shutdown-system actuation time
- simulation of shutdown-system action
- simulation of the combined effects of the LOCA and the mitigating shutdown-system response

Physics LOCA Analysis

The quantitative results of the analysis:

- reactivity
- bulk power
- channel and bundle powers
- integrated powers and peak fuel enthalpy

are then available to assess the consequences of the accident.

Since the requirements on the CANDU shutdown systems are that they be separate, independent, and each fully capable of terminating any credible accident, including a large LOCA, the action of each shutdown system must be examined separately

Large LOCA Reactivity

Full-core void reactivity ~ 10-15 milli-k

But heat-transport subdivision and finite voiding time limit LOCA reactivity insertion to ~ 4-5 mk in first 0.5-1 s

SDS actuation typically within 0.5-1 s, and negative reactivity insertion ~ 50-100 mk within 2 s

Kinetics Methods

Proper treatment of delayed-neutron effects:

Proper mathematical treatment of rates of production of prompt neutrons and of production and decay of delayed-neutron precursors. Leads to a set of differential equations coupling the neutron flux and the delayed-neutron-precursor concentrations:

Point-Kinetics Equations

In point kinetics, reactor core is treated as a single point. Premise is that only the spatially uniform component of power change need be examined, and spatial variations of the response can be ignored. Method determines time variation of global (average) values of power and precursor concentrations. Variation then superimposed on pre-event power shape.

Point-Kinetics Equations

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

$$\frac{dc_i}{dt} = \frac{\beta_i}{\Lambda}\phi(t) - \lambda_i c_i(t)$$

Inadequacies of Point Kinetics

Important sources of non-uniformity:

- Voiding transient: side-by-side loops, variations from channel to channel, and axial variations
- SDS coverage: not uniform, analysi assumes missing SDS components

Therefore, need spatial kinetics

Spatial Kinetics

Two categories of methods to solve time-dependent neutron diffusion equation:

- Nodal - finite difference: CERBERUS
- Modal - flux synthesis: SMOKIN

CERBERUS

- Three spatial dimensions
- Two energy groups
- Finite-difference model with tens of thousands of mesh points
- Factorization into amplitude and shape:

$$\Phi(\vec{r}, t) = A(t)\Psi(\vec{r}, t)$$

SHAPE EQUATION

$$\left(-M + F_p\right)\psi(\vec{r},t) + \frac{1}{A(t)}\sum_{g=1}^{G}\lambda_g C_g(\vec{r},t)\begin{pmatrix}1\\0\end{pmatrix} = \left(\frac{1}{v}\right)\frac{\dot{A}(t)}{A(t)}\psi(\vec{r},t) + \frac{\partial\psi}{\partial t}$$

M is the leakage, absorption, and scattering operator

Fp is the prompt-production operator

$\beta(\vec{r},t)$ is the total delayed fraction at position $(\vec{r},t)$

PRECURSOR SPATIAL EQUATION

$$\frac{\partial}{\partial t}C_g(\vec{r},t) = \beta_g(\vec{r})\frac{\nu\Sigma_f(\vec{r},t)}{k_0}A(t)\psi_2(\vec{r},t) - \lambda_g C_g(\vec{r},t)$$

$C_g(\vec{r},t)$ = space-time concentration of group-g delayed-neutron precursor

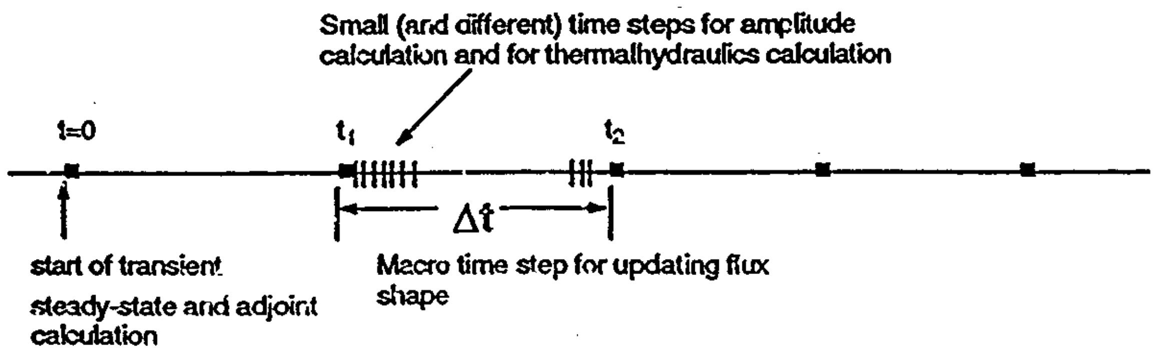

Figure 6.1 Scheme of Macro and Micro Time Intervals in CERBERUS

SMOKIN

- Expanded in finite series of pre-calculated flux modes ("harmonics")
- One energy group (thermal)
- Mode amplitudes are the unknown variables in the equations:
- small number of unknowns (10-20)
- very fast solution

Kinetics Effects on Flux Shape and Reactivity

Delayed-neutron precursors have a marked influence on the reactivity worth of shutdown systems.

Delayed neutrons retard flux-shape changes, i.e. there is a neutron source in region of action of shutdown system.

This enhances the reactivity worth of the shutdown system.

Example

CANDU 6, 26 of 28 rods operational

- Static reactivity worth (*time-independent*, non-physical calculation) = -55 mk.
- Dynamic reactivity worth (time-dependent, kinetics calculation) = -80 mk.

Analysis Methods, Models, and Assumptions

- More detailed, realistic thermalhydraulics and neutronics models
- Coupled thermalhydraulics-neutronics
- Void reactivity + uncertainty allowance
- Poison in moderator (after long shutdown)
- Flux tilt
- Pressure-tube radial creep

Analysis Methods, Models, and Assumptions (cont'd)

- Protection-system modelling, including detector delayed response, logic, electronics
- Partial shutdown-system configuration, rod-drop characteristic
- Fuel-string relocation (fuelling against flow)
- Decay-heat included
- Fuel enthalpy calculation

Analysis Methods, Models, and Assumptions (cont'd)

Conservative analysis:

back-up trip, 3/3 logic instead of 2/3, allowances on setpoints, worst power pulse at hottest bundle, least effective shutdown-system configuration

Newly emerging methodology: "best estimate + uncertainty analysis" - much more computationally intensive

In-Core LOCA

Small break, slow LOCA.

Assumptions which worsen consequences:

- Pre-event poison present, dilution
- Poison maximized by long shutdown, plutonium peak
- pressure-tube radial creep
- damaged number of shutoff-rod guide tubes

In-Core LOCA (cont'd)

Physics analysis must make accurate assessment of evolution of reactivity with time, i.e. eventual SDS-1 reactivity "depth"

External input:

- Guide-tube damage area
- Poison dilution model (piston model, instantaneous uniform mixing, or delayed mixing)

Summary

Physics analysis essential to quantitative understanding of core behaviour following hypothetical accident

Provides important quantitative information used in rest of safety analysis

Typically, many conservative assumptions

Neutronics methods and models have greatly evolved over last twenty years

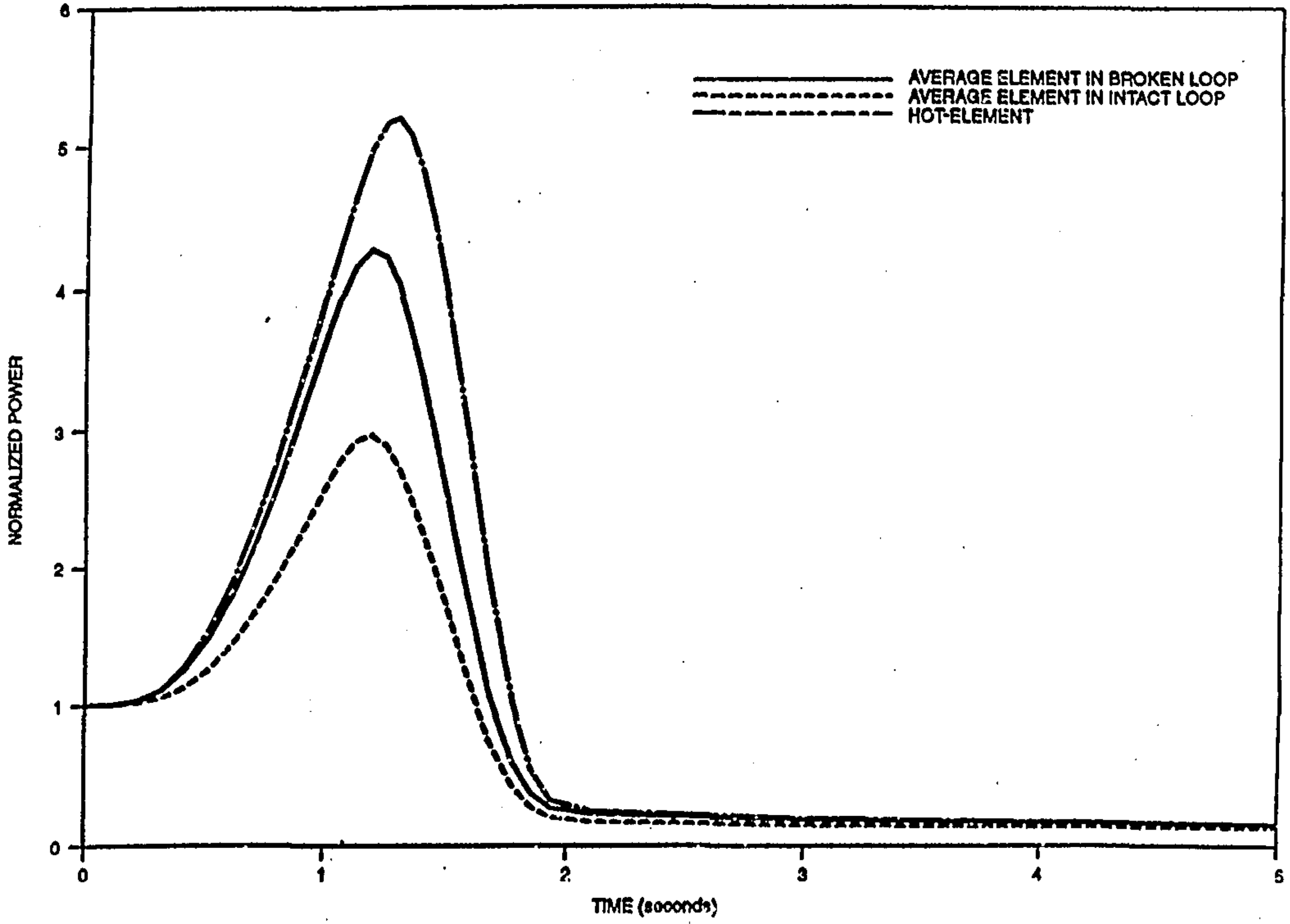

Figure 14
Typical Power Pulses for an Individual Bundle and for Core Halves