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Space-Time Neutronic Analysis of Postulated LOCA's in CANDU Reactors

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Abstract

Space-time neutronic behaviour of CANDU reactors is of importance in the analysis and design of reactor safety systems. A methodology has been developed for simulating CANDU space-time neutronics with application to the analysis of postulated LOCA's. The approach involves the efficient use of a set of computer codes which provide a capability to perform simulations ranging from detailed, accurate 3-dimensional space-time to low-cost survey calculations using point kinetics with some "effective" spatial content. A new, space-time kinetics code based upon a modal expansion approach is described. This code provides an inexpensive and relatively accurate scoping tool for detailed 3-dimensional space-time simulations.

Introduction

CANDU-PHW reactors employ Zircaloy pressure tubes to contain the primary D_2O core coolant. The pressure tubes, in turn, are contained within Zircaloy calandria tubes, with a gas-filled annulus between the two tubes providing thermal insulation between the high temperature coolant and the low temperature D_2O moderator. Each coolant channel is connected to inlet and outlet headers by a system of feeder tubes. In current CANDU designs the coolant pressure tubes are sectioned into two or three separate figure-of-eight coolant loops, with each loop having its own set of inlet and outlet headers, coolant pumps and steam generators.

The dimensions of a CANDU reactor are relatively large; for example, the cylindrical core of an 850 MWe reactor has a radius of approximately 4.0 m and a length of 6.0 m. These dimensions result in a neutronically de-coupled core which exhibits spatially dependent neutron behaviour. Furthermore, the CANDU lattice is over-moderated to preserve neutron economy which leads to a positive coolant voiding reactivity coefficient.

Present CANDU designs have two independent, equally capable safety shutdown systems. The first system, shutdown system #1 (SDS1), consists of a number of spring-assisted, gravity drop mechanical absorber rods. The second system (SDS2) consists of a number of horizontally oriented nozzles spanning the width of the core, through which a soluble poison (neutron absorber) is injected into the moderator. Both systems produce strong neutron flux distortions during the negative reactivity insertion transient.

The above-mentioned properties of the core place space-time neutron kinetics in a central role in the design and performance analysis of CANDU safety systems. In particular, the positive reactivity excursion and highly distorted flux shapes resulting from hypothetical sudden rupture of a coolant pipe in a loss-of-coolant accident (LOCA) dictate the use of detailed space-time kinetics tools to ensure that each shutdown system is capable of rapid reduction of fuel power over the whole reactor.

In subsequent sections the factors to be considered when analyzing LOCA's in CANDU reactors will be briefly described and a computational methodology for performing space-time neutronic analysis of these hypothetical accidents will be detailed.

CANDU LOCA Analysis

In order to demonstrate that a particular reactor design meets safety requirements a number of factors must be considered. These factors are:

a) Break Size

The whole spectrum of possible break sizes must be considered; ranging from an area equivalent to guillotine, or double-ended pipe break in a reactor header (the largest pipe in the coolant circuit), to a small break in a coolant feeder tube.

b) Break Position

With the coolant circuit divided into a number of separate loops, the position of the break is of importance since it will govern the resulting flux distortion during coolant voiding.

c) Safety Shutdown Systems

Licensing regulations require that analysis be performed for each of the two independent safety shutdown systems in turn. The assumption is made that the other system is not available for terminating the overpower transient resulting from a LOCA.

d) Reactivity Device Availability

Conservative assumptions regarding the number of reactivity devices that are out of service in a particular safety system at the time a LOCA occurs requires a search for the least favourable configuration.

Given the above considerations in a LOCA analysis, a large number of possible cases must be investigated in detailed studies. The strong dependence of the reactor space-time kinetics on the position of the break, the available safety shutdown system and available reactivity devices within the shutdown system precludes the direct extrapolation from one set of imposed assumptions to another. In order to handle the complexity of the space-time neutron kinetics analysis a methodology has been developed which allows the analyses to be performed within reasonable limits of time and cost. The approach is described below.

Space-Time Kinetics Analysis Methodology

The basis of the methodology is the efficient use of a series of computer codes which provide a hierarchy of analysis tools of varying complexity and spatial detail. The codes used are:

- CERBERUS, a 3-dimensional code based upon the Improved Quasi-Static (IQS) method (1,2).
- SMOKIN, a 3-dimensional modal expansion code (3,4).
- PARKIN, a point kinetics code which utilizes a self-checking collocation algorithm, (5).

The CERBERUS code provides the most detailed and accurate simulation tool. Typical reactor core models include 30,000 or more mesh points, 2 energy groups and between 6 and 15 delayed neutron groups depending upon the time scale of the transient. However, the code is expensive to run and its use is confined to those postulated accidents resulting in the most severe consequences -- often termed the "critical break" LOCA.

In order to augment the detailed space-time kinetics capability of the CERBERUS code an inexpensive but relatively accurate 3-D simulation code is required. The SMOKIN code was developed at Ontario Hydro to provide this function. This code is based upon modal expansion methods with refinements for modelling local effects not present in the mode expansion set. SMOKIN simulates detailed 3-D dimensional space-time neutronic behaviour with relatively good accuracy at low cost and is used to analyze the wide range of break sizes; perform sensitivity studies required to assess the performance of shut-down systems; and to evaluate the effectiveness of alternate designs. A summary of the main features of the code is presented subsequently.

The point kinetics code, PARKIN, is used primarily as a survey code in analysis and preliminary design activities. Some spatial information such as dynamic reactivities and spatial power peaking factors are utilized in the code.

However, such information is derived from the CERBERUS and SMOKIN space-time kinetics codes.

An important part of the simulation activity involves ongoing validation and consistency checking of the space-time solutions. Typically, CERBERUS, which has been validated against experimental data (6) is used as a benchmark code for validating SMOKIN results. However, validation of the SMOKIN code against experiments has also been performed for operational transients in power reactors (3). The PARKIN code provides an independent means of checking the consistency of the transient solutions from CERBERUS and SMOKIN based upon the dynamic reactivities generated by the two codes.

The Modal Kinetics Code, SMOKIN

The SMOKIN code is based upon one energy group modal expansion techniques (7), coupled with a local flux effects correction technique developed by one of the authors (3). The basis of the method is as follows.

The space-time distributions of neutron flux $\phi(r,t)$ and delayed neutron precursors $C_j(r,t)$, corresponding to a perturbation in the core physics properties from a reference core configuration, are expanded as a weighted series of fixed reactor λ-modes; ie

$$\phi(r,t) = \sum_{m}^{M} \psi_m(r) a_m(t)$$

$$C_j(r,t) = \sum_{m}^{M} \psi_m(r) b_{mj}(t)$$

where $\psi_m(r)$ are fixed spatial λ-mode functions, and $a_m(t)$, $b_{mj}(t)$ are time-dependent mode weighting amplitudes.

The mode functions $\psi_m(r)$ are the orthogonal eigenfunctions (λ-modes) of the static 2-group diffusion equations corresponding to a reference core configuration. These functions are generated by a separate static diffusion theory code (8).

Through substitution of the above modal expansions into the time-dependent neutron diffusion and delayed neutron source equations, and application of Galerkin weighted integration over the reactor core, a set of coupled differential equations are obtained for the mode amplitude weights. The space-time variations of core material cross-sections, with respect to the reference core cross-sections, explicitly define a set of modal reactivity terms which govern the temporal behaviour of the mode amplitude weights. The modal amplitude differential equations obtained from this procedure have the form of generalized point kinetics equations:

$$\frac{da_m}{dt} = \frac{1}{l_m^*} \sum_{k}^{M} \rho_{mk} a_k + \sum_{j}^{J} \bar{\lambda}_j b_{mj} + \frac{(\rho_{sm} - \beta) a_m}{l_m^*}$$

$$\frac{db_{mj}}{dt} = -\lambda_j b_{mj} + \frac{\beta_j}{l_m^*} a_m$$

where, ρ_{mk} is the modal reactivity coupling modes m and k

l^*_m is the prompt neutron generation time in mode m.

ρ_{sm} is the subcriticality (eigenvalue seperation) of mode m $= \left(\lambda_1^{-1} - \lambda_m^{-1}\right)$.

$\beta_j, \bar{\lambda}_j$ are the delayed neutron parameters for group j.

A novel refinement of the modal expansion method involves the use of local flux effect functions to account for local flux changes induced by core material property changes. These local effect functions are generated from static diffusion theory solutions for source/sink type perturbations. They are incorporated directly into the modal reactivities as an incremental flux term in the perturbation-type equations defining the reactivities. In addition they extend beyond the site of the perturbation to surrounding regions of the core within a few lattice pitches of the perturbation.

The local effect spatial functions modify the modal flux expansion to yield a net perturbed flux distribution $\phi'(r,t)$, of the form:

$$\phi'(r,t) = \zeta(r)\ \phi(r,t)$$

where $\zeta(r)$ is the local effect spatial flux multiplier function defined by:

$$\zeta(r) = 1 + \alpha\left[\psi_1(r_o) / \psi_1(r)\right] \exp\left[\,|r - r_o|/M\,\right]$$

$$\alpha = \left[(\Delta\nu\Sigma_f - \Delta\Sigma_a) / (\Sigma_a + \Delta\Sigma_a)\right]_{r=r_o}$$

where $\left.\begin{array}{l}\Delta\Sigma_a \\ \Delta\nu\Sigma_f\end{array}\right\}$ are the incremental material cross-sections associated with a perturbation at r_o.

$\psi_1(r)$, $\psi_1(r_o)$ are the reference fundamental fluxes at positions r and r_o, (ref. 3).

M is an effective neutron migration length.

The most important feature of this approach is that it provides a direct means of separating the global "tilt" components of a perturbation from the strictly localized effect of the perturbation. Furthermore, a single set of λ-modes is required (the modes for the nominal core configuration), irrespective of the magnitude of change in reactor configuration from the nominal. This circumvents the problem of selecting an appropriate basis set of mode functions for a particular transient, thereby reducing the cost of generating the mode set to a fixed overhead cost to be shared between all subsequent transient simulations.

LOCA Simulation Results and Discussion

Space-time kinetics of hypothetical LOCA transients in a large CANDU-PHW reactor have been simulated with both CERBERUS and SMOKIN in the course of analysis to assess shutdown system performance. The break initiating the LOCA is postulated as equivalent to 40% of the maximum break area in an inlet header of one loop of a two loop coolant circuit. This LOCA leads to voiding in one half of the reactor core with an attendant side-to-side distorted overpower transient. The LOCA has been simulated assuming, in turn, availability of only one of the two safety shutdown systems. In each case the most effective reactivity devices in the shutdown systems are unavailable at the time of the accident. The broken loop was assumed to be the most conservative for the available shutdown reactivity devices.

The neutronic power pulses and dynamic reactivities in the first 2.5 seconds of the LOCA, obtained from the two space-time kinetics codes, are given in figures 1-4. The agreement between the two independent solutions is close throughout the entire transient. The fuel power integrated over the first 2.5 seconds and the peak local power at the end of the transient are important parameters since they govern the short term integrity of the fuel and end of blow-down fuel sheath temperatures. The differences in these quantities, as obtained from the two solutions, are less than 1.5% and 3% respectively. Space-time dependent results obtained from the two codes are in close agreement, both for global quantities such as dynamic reactivity, and for localized quantities such as power peaking.

Finally, the relative power distributions along radial and axial axes passing through the peak power locations at the end of the transients are given in figures 5-8. Comparison of these distributions indicates that SMOKIN can simulate the 3-dimensional power transients with relatively good accuracy. The agreement is particularly good in the case of the LOCA assisted by the second shutdown system (poison injection into the moderator). More importantly, the locations and values of the power peaks predicted by the two code are in close agreement in both cases.

Conclusions

The methodology developed for analyzing space-time neutron kinetics of hypothetical LOCA's in Ontario Hydro CANDU-PHW reactors has proved to be very effective. The efficient use of the CERBERUS and SMOKIN codes has allowed analysis to be performed at a detailed, 3-dimensional level for a large number of postulated accident configurations. Such analysis using other methods has, in the past, been either prohibitively costly or has relied on extremely conservative assumptions to compensate for the lack of accurate models. The ability to perform numerous detailed space-time analyses has proven to be a distinct advantage when designing safety systems and evaluating their performance.

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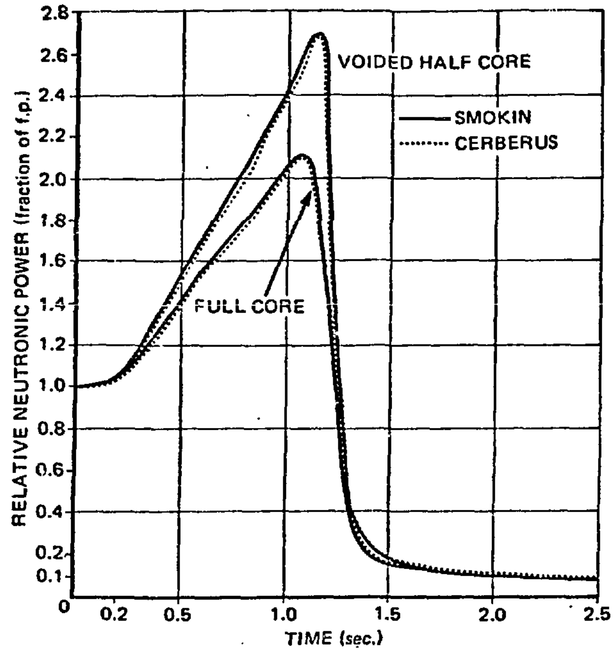

Figure 1
Relative Neutronic Power As A Function Of Time
(Half Core Loca With SDS2)

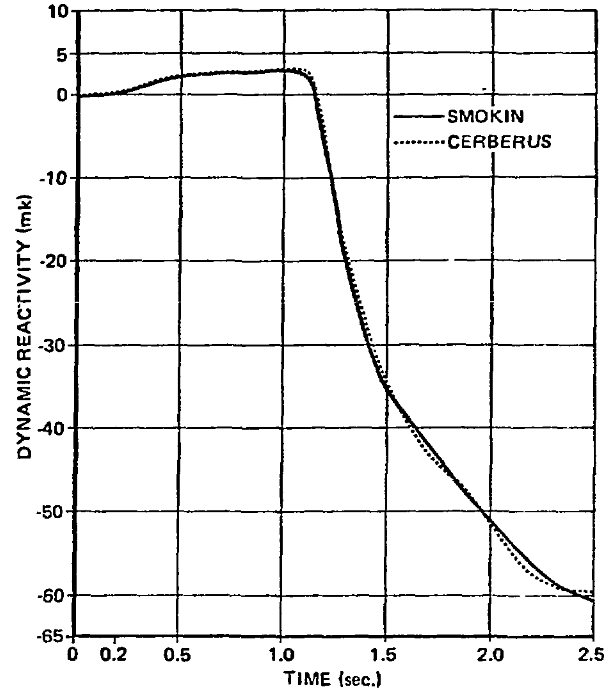

Figure 2
Dynamic Reactivity As A Function Of Time
(Half Core Loca With SDS2)

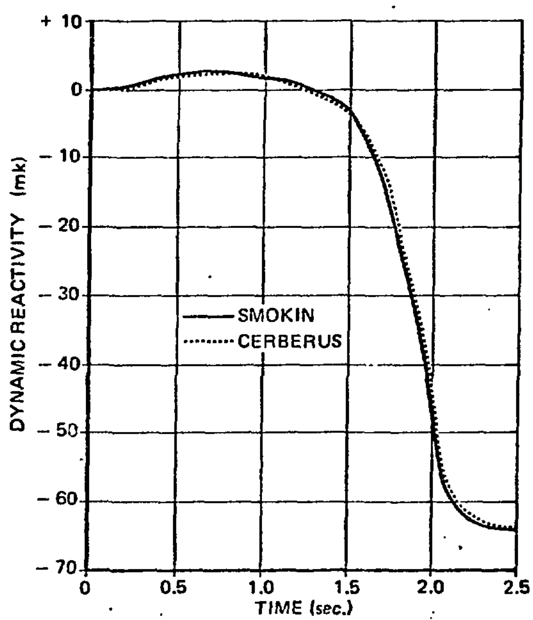

Figure 3
Dynamic System Reactivity
As A Function Of Time
(Half Core Loca With SDS1)

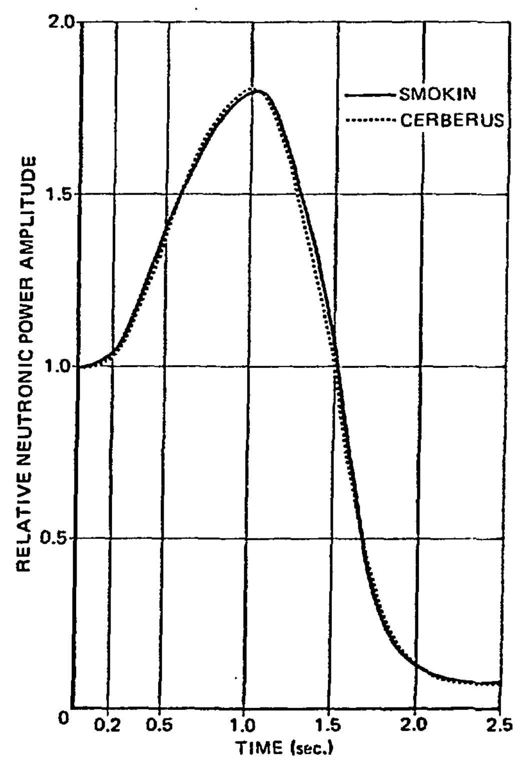

Figure 4
Relative Neutronic Power Amplitude
As A Function Of Time
(Half Core Loca With SDS1)

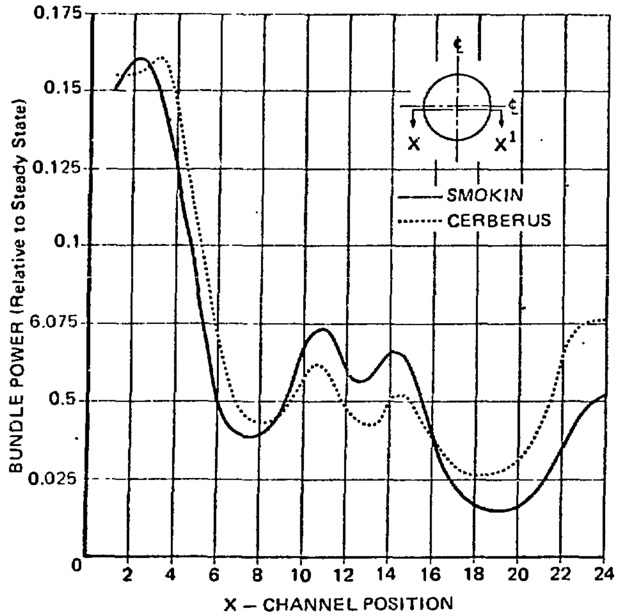

Figure 5
Radial Power Distribution
Along X – X¹ At Time = 2.5 Seconds
(Half Core Loca With SDS1)

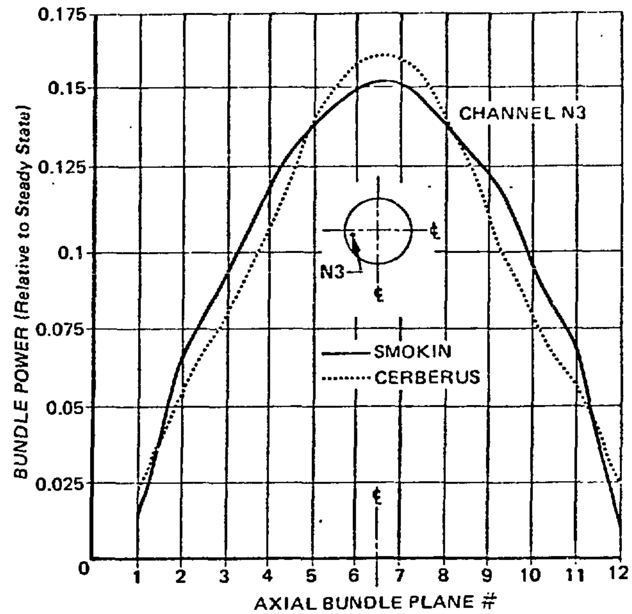

Figure 6
Axial Power Distribution
Along Channel N3 At Time = 2.5 Seconds
(Half Core Loca With SDS1)

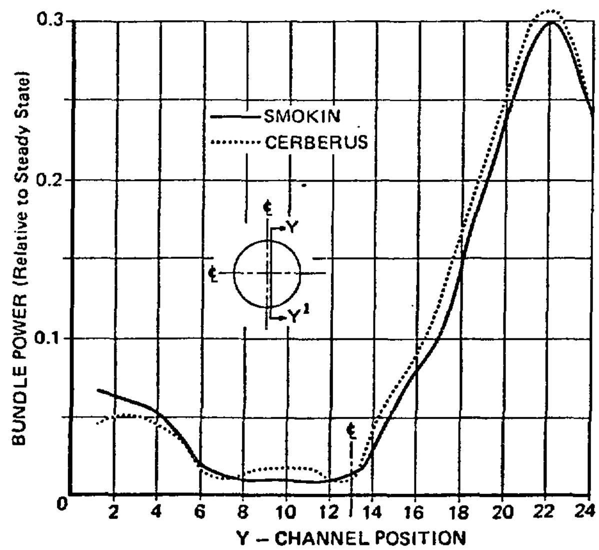

Figure 7
Radial Power Distribution
Along Y – Y¹ At Time = 2.5 Seconds
(Half Core Loca With SDS2)

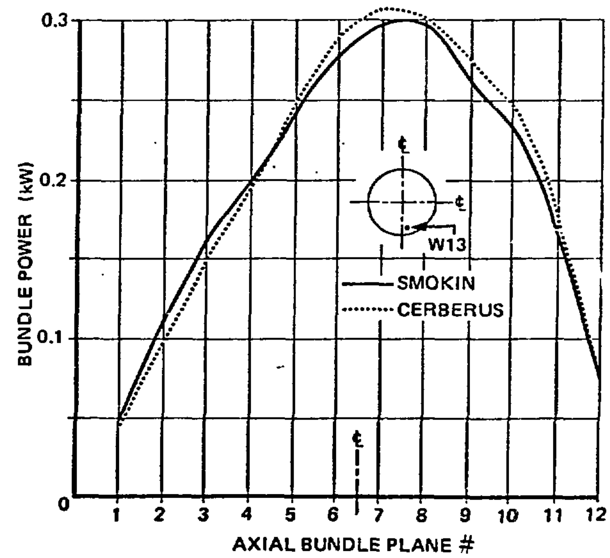

Figure 8
Axial Power Distribution
Along Channel W13 At Time = 2.5 Seconds
(Half Core Loca With SDS2)