Recent Trends in Methodologies for CANDU® Finite-Core Analysis at AECL

by

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Paper presented at the Canadian Nuclear Society's Fifth International Conference on Simulation Methods in Nuclear Engineering, Montréal, Québec, 1996 September 8-11.

ABSTRACT

The last decade has seen the implementation several significant advances in calculational methodologies for CANDU finite-core analysis. Concurrent with these advances, reactor models have become less stylized, more complex and more realistic. The discipline is still witnessing advances in techniques and models. There are on-going incentives to make the simulations ever more realistic and to validate the computer codes and numerical methods. This review paper presents *some* of the recent developments in which AECL has been involved.

1. Introduction

The last decade has seen the implementation of several significant advances in calculational methodologies for CANDU finite-core analysis. Concurrent with these advances, reactor models have become less stylized, more complex and more realistic. These developments have sprung from the analyst's ever-present motivation to improve calculation methods and models, supplemented by the need to convincingly demonstrate to outside parties (e.g., the regulator) that simulations are accurate and true-to-life.

This review presents those recent developments in CANDU finite-core analysis with which AECL, in particular, has had involvement at least at some point. Thus it must be pointed out at the very start that many developments, which have been championed solely by other organizations, will be missing from the discussion. Also, advances in lattice codes *per* se are not covered, although the trend in the use of specific new codes is discussed.

2. Tracking of Core History with Flux Mapping and "Instantaneous" Fundamental Mode

Tracking a CANDU core history involves following the evolution of the flux, power, and irradiation distributions with time. Local changes in the flux distribution are induced by channel refuellings and reactivity-device movements (most often, changes in zone-control-compartment water fills).

The three-dimensional core flux distribution can be calculated with a diffusion-theory code such as Reactor Fuelling Simulation Program RFSP (see Section 3). However, for reactors (such as the CANDU 6) equipped with in-core (vanadium) flux-mapping detectors — see Figure 1 — the flux distribution can also be reconstructed using a linear expansion in a set of "flux modes", with the mode amplitudes determined so as to produce a best fit to the detector readings. Figure 2 shows schematically some of the flux modes (harmonics) used.

An on-line program, resident in the reactor's control computers, performs the flux mapping every 2 minutes, with the objective of providing a zonal flux distribution for the purpose of calibrating the in-core spatial-control detectors. This on-line program makes use of a fixed set of "time-average" flux modes and does not provide a sufficiently accurate calculation of the three-dimensional flux shape for application in long-term detailed core-follow simulations.

However, the accuracy of the flux mapping can be significantly enhanced by the adoption of a more appropriate fundamental mode. On any given day of the core history, this fundamental mode is the diffusion-theory-computed snapshot flux distribution corresponding to that day's actual core configuration (e.g., accounting for channels recently refuelled and current zone-control-compartment water fills).

An updated instantaneous fundamental mode was first adopted [1] at Hydro-Québec's Gentilly-2 reactor, using the HQSIMEX code [2]. Although the use of a fundamental mode calculated with an irradiation distribution based on the *mapped* fluxes first gave rise to growing tilts in the flux distribution, this problem disappeared when the irradiation tracking was based on *diffusion* fluxes. This method is displaying an excellent performance, regularly achieving a standard deviation in the differences between computed and measured detector fluxes in the range of 1.3 to 1.5%.

The instantaneous-fundamental methodology was later incorporated in AECL's RFSP [3] and is used in conjunction with a fundamental mode computed with history-based lattice properties (see Section 3) at New Brunswick Power's Point Lepreau reactor, where it is known as "power mapping". There, too, the method has resulted in excellent long-term agreement between predicted and measured flux shapes [4].

The level of agreement provided is somewhat better than can be achieved by diffusion-theory calculations alone, which is not altogether surprising perhaps, since the flux-mapping process is by its nature designed to optimize the fit to detector readings. On account of its excellent performance, therefore, this methodology has been adopted as the procedure of reference for core tracking and as the official source of the CPPF (Channel Power Peaking Factor) value, used to calibrate protection-system in-core detectors.

3. History-Based Methodology for Lattice Properties

Core modelling for diffusion-theory flux calculations requires the specification of lattice properties at every point (fuel bundle) in the core. In core-tracking snapshots, the variation of instantaneous fuel irradiation with position in the core is the most crucial factor to take into account.

As a result, the standard method of evaluating lattice properties has, for many years, consisted of interpolating in irradiation within "fuel tables" computed by a cell code, such as

POWDERPUFS-V [5]. Because irradiation is the single most important parameter, the fuel tables have generally been evaluated assuming core-average values of many other parameters, such as fuel temperature, coolant density, absolute flux level, moderator temperature, etc. I refer to this single-fuel-table approach as the "uniform-parameter" method.

However, in reality, lattice properties do depend on the local values of the parameters named above and also on the history of quantities such as the moderator poison concentration. The uniform-parameter method, by its nature, is "blind" to these effects.

It is possible to remedy this situation partly by providing a number of fuel tables, each evaluated with a chosen set of values (within a defined grid) for the most important parameters, e.g., flux level and coolant density. Lattice properties at any given core position can then be obtained not only for the local irradiation but also (by multi-dimensional interpolation in the grid of fuel tables) corresponding to local values of the other grid parameters. This is a "grid-based local-parameter" method. Grid-based methods suffer, however, from weaknesses such as:

- computational constraints, e.g. the impracticality of defining a large number of fuel tables (more than, say, 30 to 35), and
- the failure to address varying-history effects (e.g., when the moderator-poison concentration changes at some point in time, it is *in principle* not correct to provide lattice properties evaluated with the fuel irradiated throughout its life at the new poison concentration).

Core modelling within RFSP has instead recently seen a major improvement in the form of the "history-based" methodology [6], for use especially in core tracking. In this method, *fuel tables are not employed.* Rather, at each core-follow snapshot, an *individual* POWDERPUFS-V calculation is performed within RFSP for *each* fuel bundle to update its properties over the incremental burnup step, using *locally appropriate* values of parameters (flux level, fuel temperature, coolant density, whatever parameters the user specifies) for that instant in the core history. Changes in lattice parameters are, therefore, captured when they actually occur, and the evolution of the nuclear properties of each individual bundle is more properly tracked.

Once the distribution of lattice properties for the current snapshot is established with the history-based methodology, the diffusion equation is solved as usual to obtain the flux distribution. It is to be noted that the method is not limited to RFSP but can of course be incorporated within any diffusion-theory core code.

The history-based diffusion method with RFSP and POWDERPUFS-V has been extensively validated against Point Lepreau in-core-detector measurements (e.g. References 7 and 8); see for example Figure 3. In particular, a paper by Carruthers and Donnelly [8], to be presented at this Conference shows clearly that, over a full-power-year of Point Lepreau core tracking, the history-based diffusion method offers greater accuracy than the uniform-parameter method in the comparison of calculated and measured detector fluxes.

The history-based method allows virtually any desired parameter to be modelled with a core variation. For example, the effect of non-uniformity in the moderator temperature, which would have been difficult to investigate previously, can be studied if information on this variation is available.

4. The Saturating-Fission-Product Driver

Because CANDU reactors are refuelled on-power, the core always has a certain number of fresh or near-fresh bundles. In fresh fuel, the concentration of saturating fission products, such as ^{135}Xe, ^{149}Sm and ^{151}Sm, ^{105}Rh, is at first zero but then builds up over a time interval of hours and days. In this interval, the absorption caused by saturating fission products is less than the steady-state value that is eventually attained, and the reactivity and power are correspondingly higher. The enhancement in the calculated power of refuelled channels is of the order of several percent for an 8-bundle-shift refuelling scheme and is, therefore, non-negligible when monitoring the power distribution.

An important advantage of the history-based method is that changes in the concentration of saturating fission products, caused by transient conditions such as refuelling or device movements, can easily be modelled at the individual-bundle level. A fission-product kinetics calculation (e.g., for $^{135}I/^{135}Xe$ and other appropriate parent/daughter pairs), can be performed (with a "saturating-fission-product driver") for each bundle to follow the concentration of each saturating fission product in time. These concentrations can then be fed to the POWDERPUFS-V history-based calculation, thereby entering the computation of local lattice properties in the same way as any other perturbation.

The saturating-fission-product driver installed within RFSP has been used to derive typical adjustment factors for the power of fresh bundles and channels and their neighbours. These adjustment factors are being used to obtain power distributions corrected for the fresh-fuel effect. Alternatively, one can now contemplate doing, on a routine and on-going basis, diffusion-theory core-tracking simulations that include *explicitly* the effects of the absence of saturating fission products in fresh fuel.

The history-based method, with the fission-product driver, has been validated on non-steady-state conditions with a comprehensive simulation [9] of the Point Lepreau restart after the 1992 outage. Measurements were made at various times during the return to full power, including both the nominal core and configurations with reactivity devices moved from their nominal position, e.g., adjuster banks withdrawn from core or shutoff-rods inserted in core. Flux measurements were also made in-core with a Travelling Flux Detector (TFD). The method attained excellent agreement with detector measurements (Figure 4 shows one example).

5. Alternative Lattice Codes

Since the start-up of Pickering, the first commercial nuclear generating station in Canada, POWDERPUFS-V has been the cell code used for CANDU core tracking. POWDERPUFS-V is, in fact, a semi-empirical code, developed at AECL specifically to capture the results of experiments conducted on heavy-water-moderated, natural-uranium-fuelled lattices in the research reactors at the Chalk River Laboratories. The good results that are consistently obtained with POWDERPUFS-V have, quite naturally, ensured its attractiveness to the fuelling engineer, and its continued use.

There are, nonetheless, good reasons to investigate the use of a cell code that has firmer foundations in transport theory, one which can be applied over a wider range of problems. One such computer program is WIMS-AECL [10], the AECL-elaborated version of the Winfrith code WIMS [11]. The proposed utilisation of advanced fuels (e.g., SEU, Recovered Uranium) and/or

different fuel-bundle designs (e.g., CANFLEX) in CANDU reactors is a strong impetus to use WIMS-AECL, recognizing that POWDERPUFS-V was not originally designed for these applications. But the adoption of WIMS-AECL as the cell code of reference in core calculations presupposes its validation. The previously referenced paper by Carruthers and Donnelly [8] shows such a validation of the WIMS-AECL/RFSP suite against one full-power year of Point Lepreau core history.

A practical consideration, when using WIMS-AECL, is the much longer computer time required to perform a cell calculation, relative to POWDERPUFS-V. Yet the superior performance of the history-based method compared with that of the uniform-parameter method creates an incentive to apply the methodology with WIMS-AECL as well. Work is, therefore, on-going to try to find ways of achieving this. The paper by Laughton and Jenkins at this Conference [12] investigates the idea of applying several computers in parallel to reduce the real-time requirement for the completion of the thousands of cell calculations inherent in a history-based simulation. Other possibilities being investigated are simplified models, approximations, or parametrizations (e.g., using sensitivity coefficients), which are relatively quick-running yet reproduce closely the results of full-fledged calculations.

The paper by Arsenault *et al.* at this Conference [13] investigates the use of still another transport-theory code in the evaluation of reactivity-device incremental cross sections. This code is DRAGON [14], a latest-generation lattice program, developed at École Polytechnique. As a possible alternative to WIMS-AECL for CANDU applications, DRAGON needs to be validated against a database of heavy-water lattice experiments. This is covered in Marleau's paper at this Conference [15].

6. The Time-Average-Equivalent Reactor Model

The semi-continuous refuelling in CANDU reactors results in a global flux shape which does not vary to a great degree. One can think in terms of a time-average flux distribution about which a "refuelling ripple" is superimposed because of the burnup "journey" each channel traverses between its successive refuellings.

The picture has given rise to the CANDU time-average model, in which nuclear cross sections are defined at each bundle position in the core by averaging the lattice cross sections over the irradiation range $[\omega_{in}, \omega_{out}]$ "experienced" over time by fuel at that position. For example, the time-average thermal neutron absorption cross section at some core position r, $\Sigma_{a2}^{t.a.}(r)$, is written as

$$\Sigma_{a2}^{t.a.}(r) = \frac{1}{(\omega_{out} - \omega_{in})} \int_{\omega_{in}}^{\omega_{out}} \Sigma_{a2}(\omega)\, d\omega$$

where ω_{in} and ω_{out} are respectively the values of fuel irradiation when the fuel enters and exits that position in the core. The time-average calculation solves, self-consistently, for the core flux, the channel dwell times, the individual-bundle irradiation ranges $[\omega_{in}, \omega_{out}]$, and the lattice properties.

The time-average model is an important CANDU design tool. However, the complicated nature of the lattice properties (integrals over bundle-specific irradiation intervals) makes the time-average model difficult to apply in simulations of perturbations (for instance, in loss-of-

coolant accidents, where the coolant density is suddenly perturbed from, and to, different and location-specific values). In particular, the time-average model does not lend itself well to local-parameter or history-based applications.

To retain the usefulness of the time-average picture but allow the application of the history-based methodology, the *time-average-equivalent* model has been developed within RFSP. The idea is to find for each bundle a single *effective* value of irradiation, ω_{inst}, at which the instantaneous properties give an infinite multiplication constant $k_{\infty,inst}$ equal to the value implicit in the time-average properties at that location, $k_{\infty,t.av.}$. The *instantaneous* or *snapshot* model defined by these instantaneous irradiations is then the time-average-equivalent model, which can be used in place of the time-average model in history-based calculations; Reference 16 is an example of its application in a LOCA simulation.

7. True-Two-Group Diffusion Calculations

Diffusion codes based on POWDERPUFS-V lattice properties are often labelled "1.5-group" codes. The reason is that POWDERPUFS-V does not provide a fast-fission cross section; it calculates fast fission in ^{238}U only and lumps it into the thermal-fission cross section. Also, up-scattering is not included.

The use of WIMS-AECL has provided an impetus to generalize RFSP to a true-two-group code, to avoid the necessity of casting WIMS-AECL cross sections into the "1.5-group" format.

The first true-two-group version of RFSP has very recently been developed and tested [17]. It awaits extensive validation and application. Another opportunity for two-group or even multi-group core calculations is provided by the TRIVAC code [18], already incorporated in the Hydro-Québec program, HQSIMEX, and currently being added to RFSP.

8. Spatial Neutron Kinetics

In the last decade, in the aftermath of the Chernobyl accident, the simulation of loss-of-coolant accidents (LOCA) has come under increasing scrutiny. Spatial kinetics codes, whether CERBERUS [19] at AECL or SMOKIN [20] at Ontario Hydro, have been coupled to thermalhydraulics codes. To confirm the adequacy of the shutdown systems under different reactor configurations, many LOCA simulations have been performed (see, e.g., Reference 21) from initial states with various reactivity-device positions, tilted flux distributions, or high moderator-poison concentrations (for instance, following a long shutdown). In addition, esoteric effects such as fuel-string movement following inlet-header breaks have been modelled (see Reference 22).

To capture the spatial distribution of coolant voiding, the critical coolant pass (downstream of the break) is modelled as subdivided into several thermalhydraulic channel groups, each representing a number of channels with similar power and height in the core (see Figure 5). Each channel group has its own voiding characteristic.

The Improved Quasi-Static method [23] inherent in CERBERUS has been incorporated in the *CERBERUS module of RFSP. There it works with the history-based method, which allows any number of channel groups. In the limit, each channel can be modelled with its own voiding transient; the limitation is currently set by the thermalhydraulics codes. The *CERBERUS module also utilizes bundle-specific delayed-neutron fractions, computed consistent with the instantaneous or time-average-equivalent bundle irradiation. These features are an important improvement over the "axially homogeneous" models previously in use, since both the void reactivity and the total delayed fraction are dependent on the fuel irradiation, and these local effects can now be taken into account.

9. Modelling of the Reactor Regulating System

The modal method in the kinetics code SMOKIN makes it very quick running. This has been used to advantage by incorporating into it the capability to simulate the action of the Reactor Regulating System (RRS), a capability that has been used for many years at Ontario Hydro to study various types of operating transients.

More recently, Hydro-Québec and New Brunswick Power have commissioned AECL to add to RFSP the ability to model the action of the CANDU 6 RRS. This capability has been incorporated by programming the actual algorithms used by the CANDU 6 RRS in a new RFSP module, *CERBRRS, which also performs the coupling to the neutronics calculation (see Figure 6).

Although its running time is much longer than that of SMOKIN, the *CERBRRS module now allows the simulation, based on finite-difference diffusion-theory methods, of operational transients, where reactivity devices are manipulated by the RRS to manoeuvre the reactor power or adjust the core power distribution. Reference 24 illustrates the use of *CERBRRS to simulate, for instance, RRS action following an operator-requested power reduction (see Figure 7) and a small in-core loss of coolant respectively (see Figure 8).

Validation of *CERBRRS against actual plant data is the next step in these studies.

10. Summary

CANDU finite-reactor analysis is still witnessing advances in techniques and models. There are on-going incentives to make the simulations ever more realistic and to validate the computer codes and numerical methods.

Future directions in the Canadian nuclear industry certainly include the increased use of WIMS-AECL, especially for advanced-fuel-cycle work and, perhaps, eventually for core tracking of existing reactors once new fuel types are introduced (for example, CANFLEX with either natural or enriched uranium), subject to the validation of the code and to the demonstration of practical advantages over POWDERPUFS-V. DRAGON may see use in the calculation of incremental cross sections and eventually, as a future alternative to WIMS-AECL. I believe that the push to evaluate lattice properties using a history-based method, or some equivalent approach, will be strong. Monte-Carlo applications will serve as possible "benchmarks".

Nodal methods to calculate the flux distribution may be adopted eventually; they would provide useful fine-structure information, especially regarding the flux at in-core detectors.

There will also undoubtedly be major incentives to quantify calculational uncertainties for as many computer-code applications as possible, for example, xenon transients, critical poison levels after long shutdowns, etc. There is still much work to be done.

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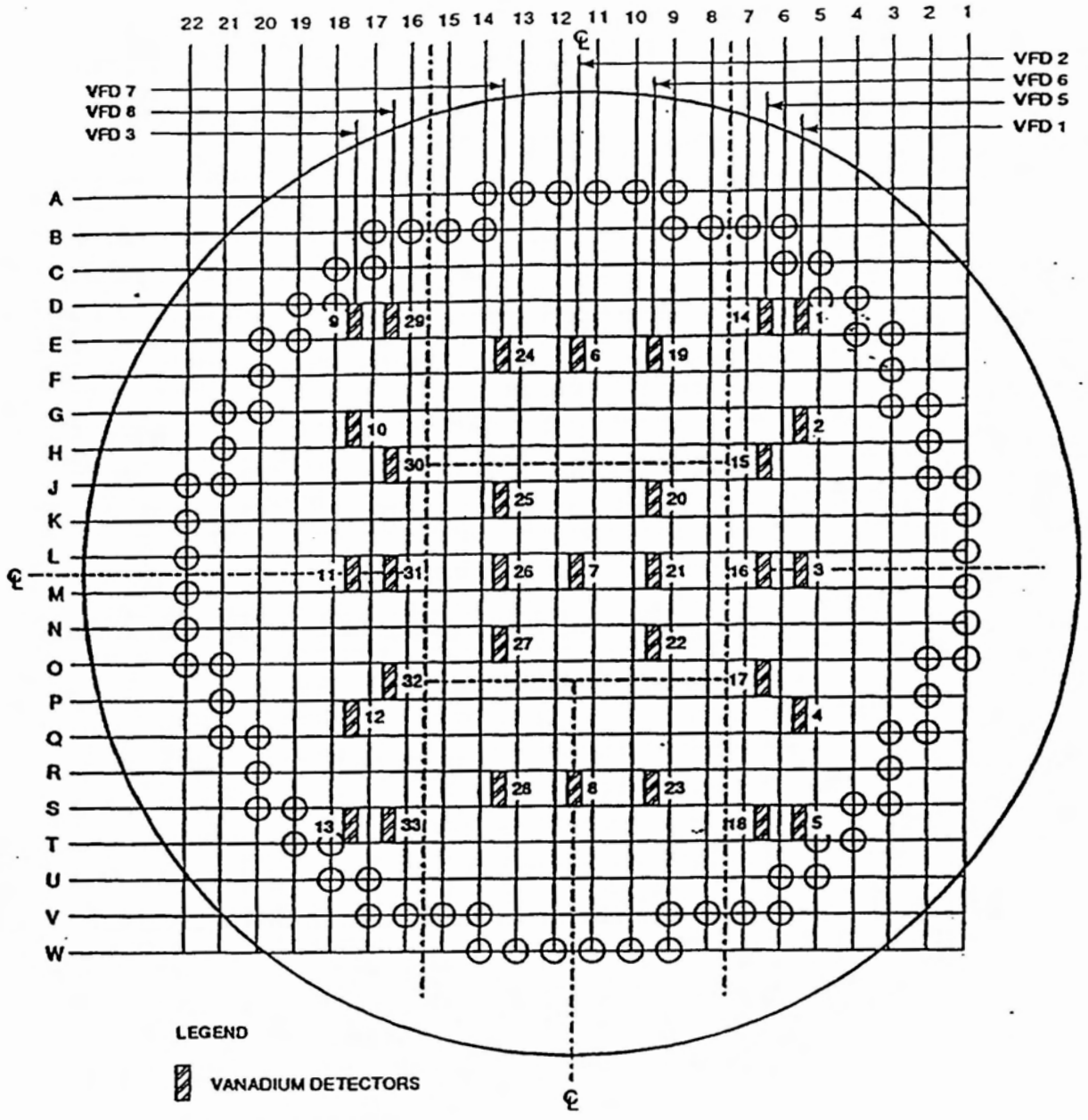


Figure 1
Location of Some Flux-Mapping Vanadium Detectors (Point Lepreau)

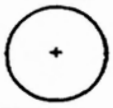
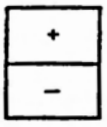
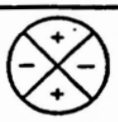
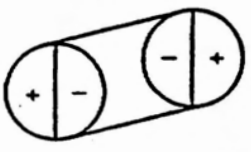
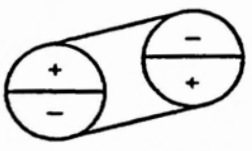
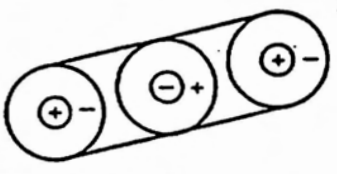
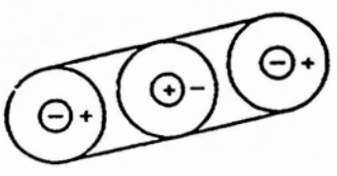
MODE NUMBER	DESIGNATION	SUBCRITICALITY mk	MODE SCHEMATIC (IDEALIZED)
0	FUNDAMENTAL	0	
1	FIRST AZIMUTHAL-A	16.2	
2	FIRST AZIMUTHAL-B	16.9	
3	FIRST AXIAL	27.1	
4	SECOND AZIMUTHAL-A	44.0	
5	SECOND AZIMUTHAL-B	47.0	
6	FIRST AZIMUTHAL-A X FIRST AXIAL	46.9	
7	FIRST AZIMUTHAL-B X FIRST AXIAL	47.7	
8	FIRST RADIAL X SECOND AXIAL-A	66.3	
9	FIRST RADIAL X SECOND AXIAL-B	80.6	

Figure 2
Flux Harmonics Used in CANDU 6 Flux Mapping

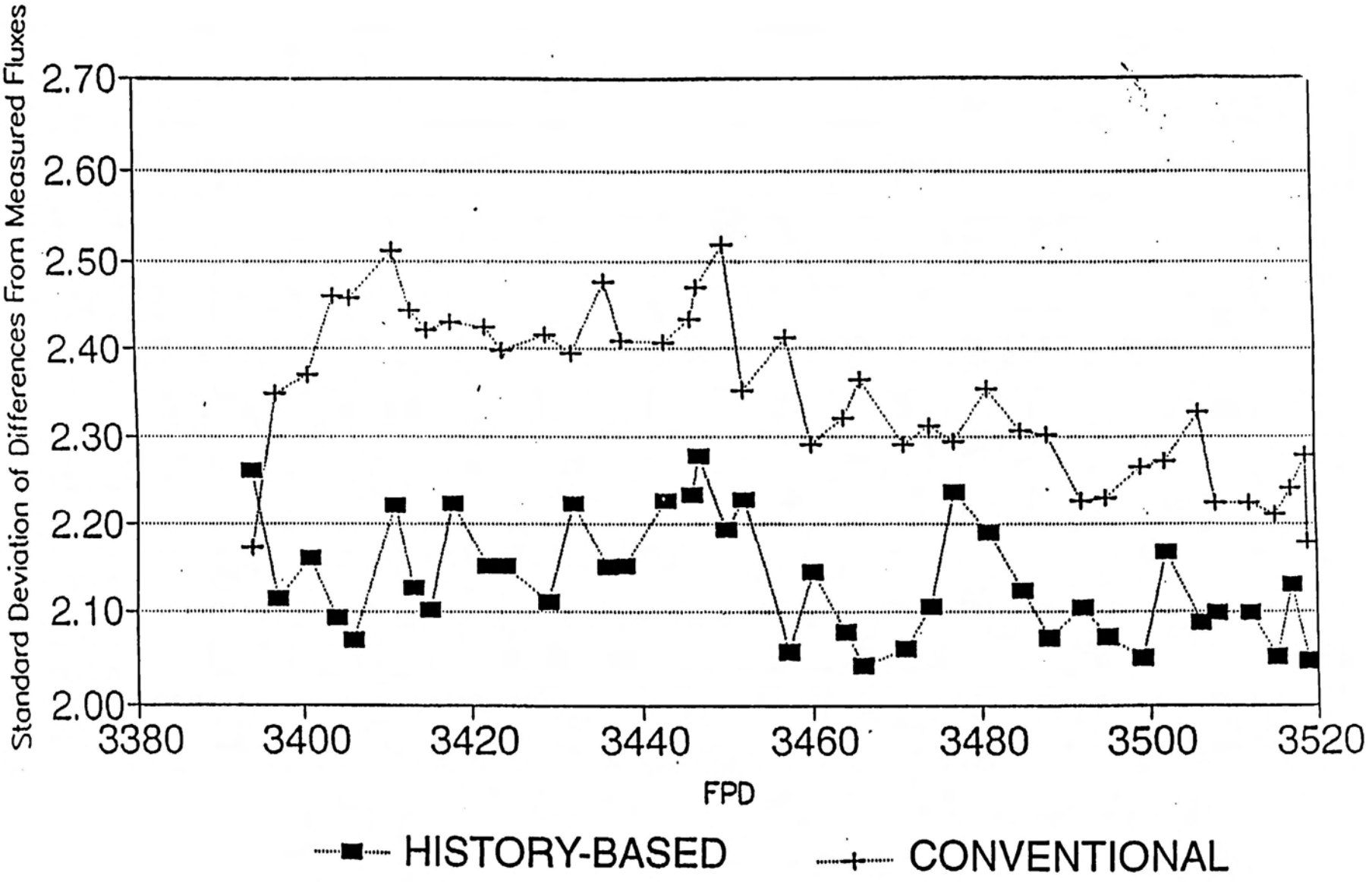


Figure 3
Comparison of History-Based and Conventional Core-Follow Methods

SIMULATION METHOD	PERTURBATION						AVERAGE
	MCA1/4 Half-In	SOR19 Half-In	Zone2 Drain	AA18 Out	1-Bank Shim	4-Bank Shim	
Standard Diffusion	1.2	1.4	2.8	2.6	2.5	1.5	2.0%
History-Based Diffusion	1.2	1.3	1.6	2.3	2.5	1.8	1.8%
Mapping	0.6	0.7	0.3	0.3	0.3	0.3	0.4%

Figure 4
Comparison of Perturbations in RFSP and Measured V-Detector Fluxes

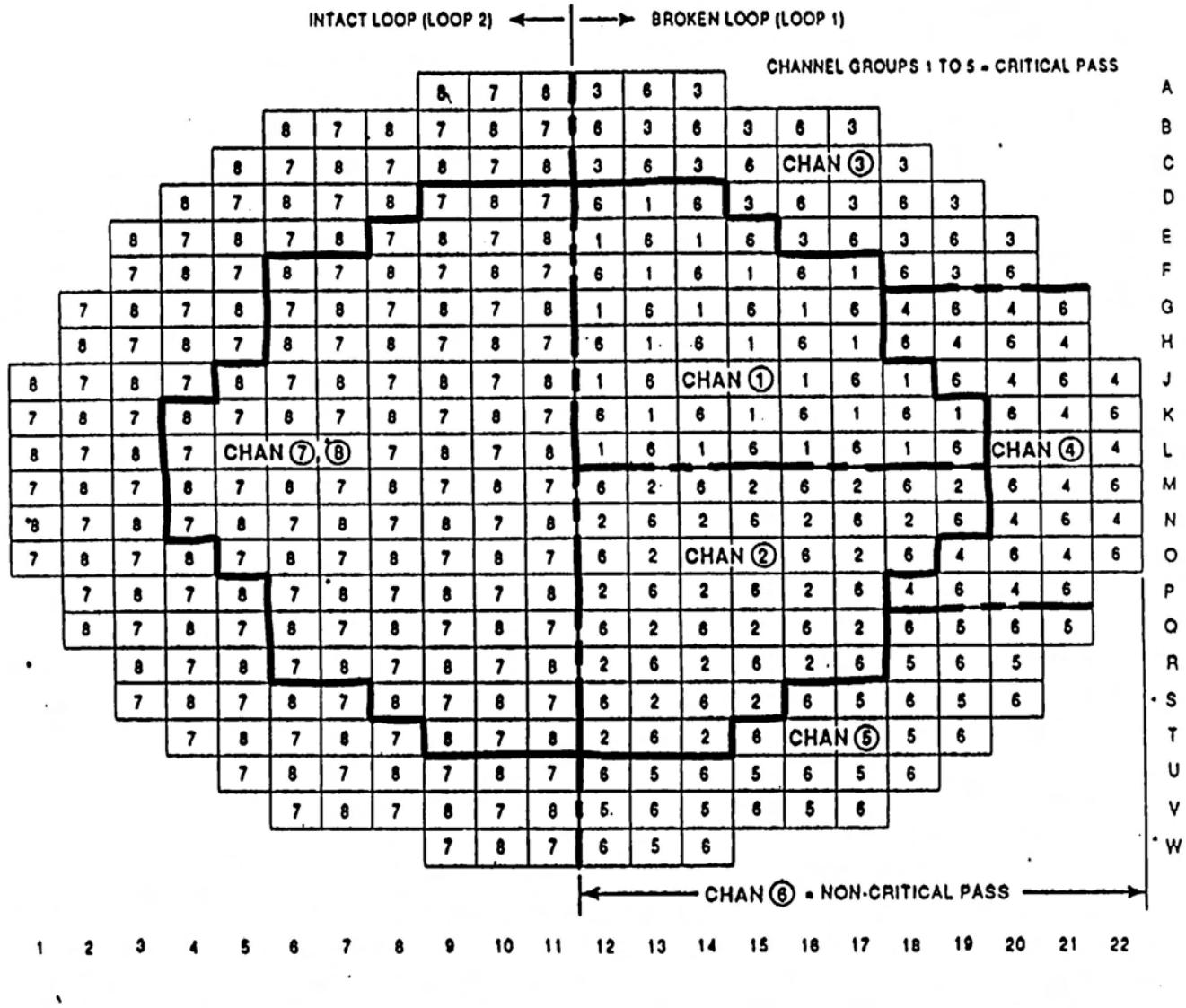


Figure 5
Subdivision of Channels in Thermalhydraulic Channel Groups

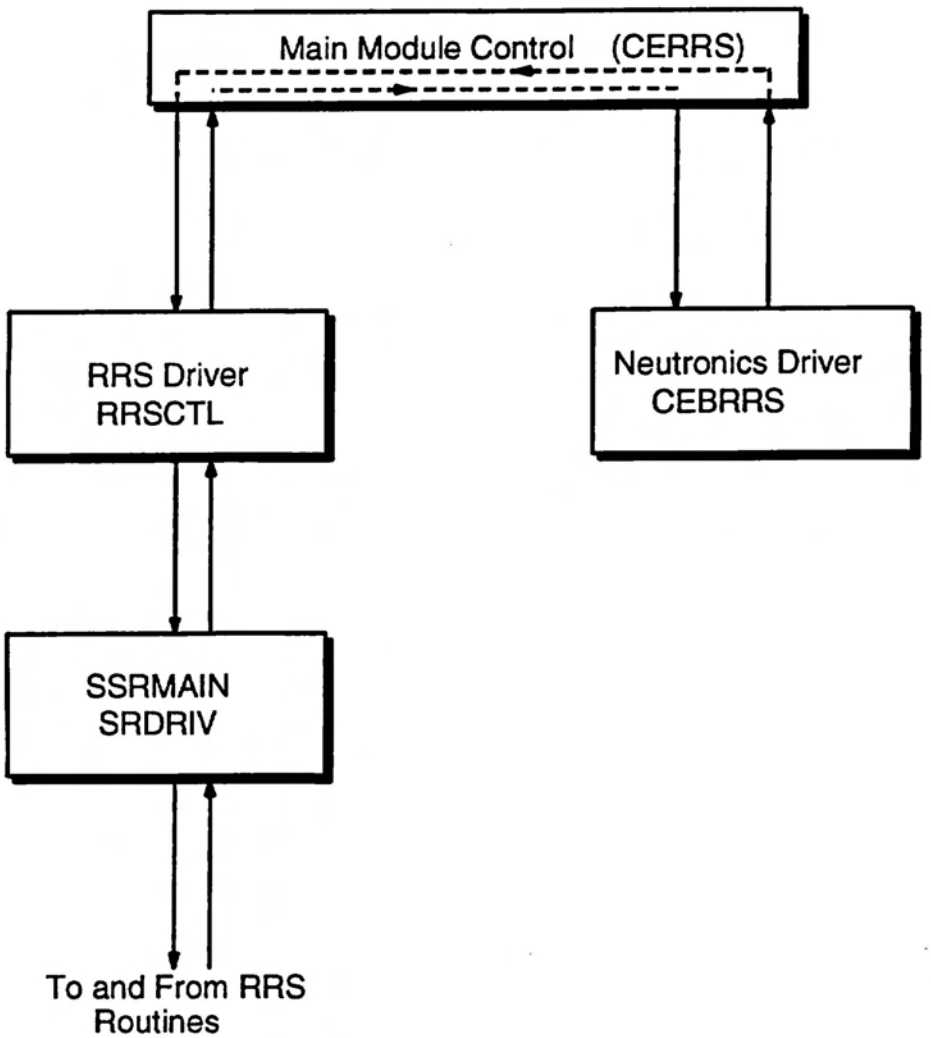


Figure 6
Flow Chart of *CERBRRS Module

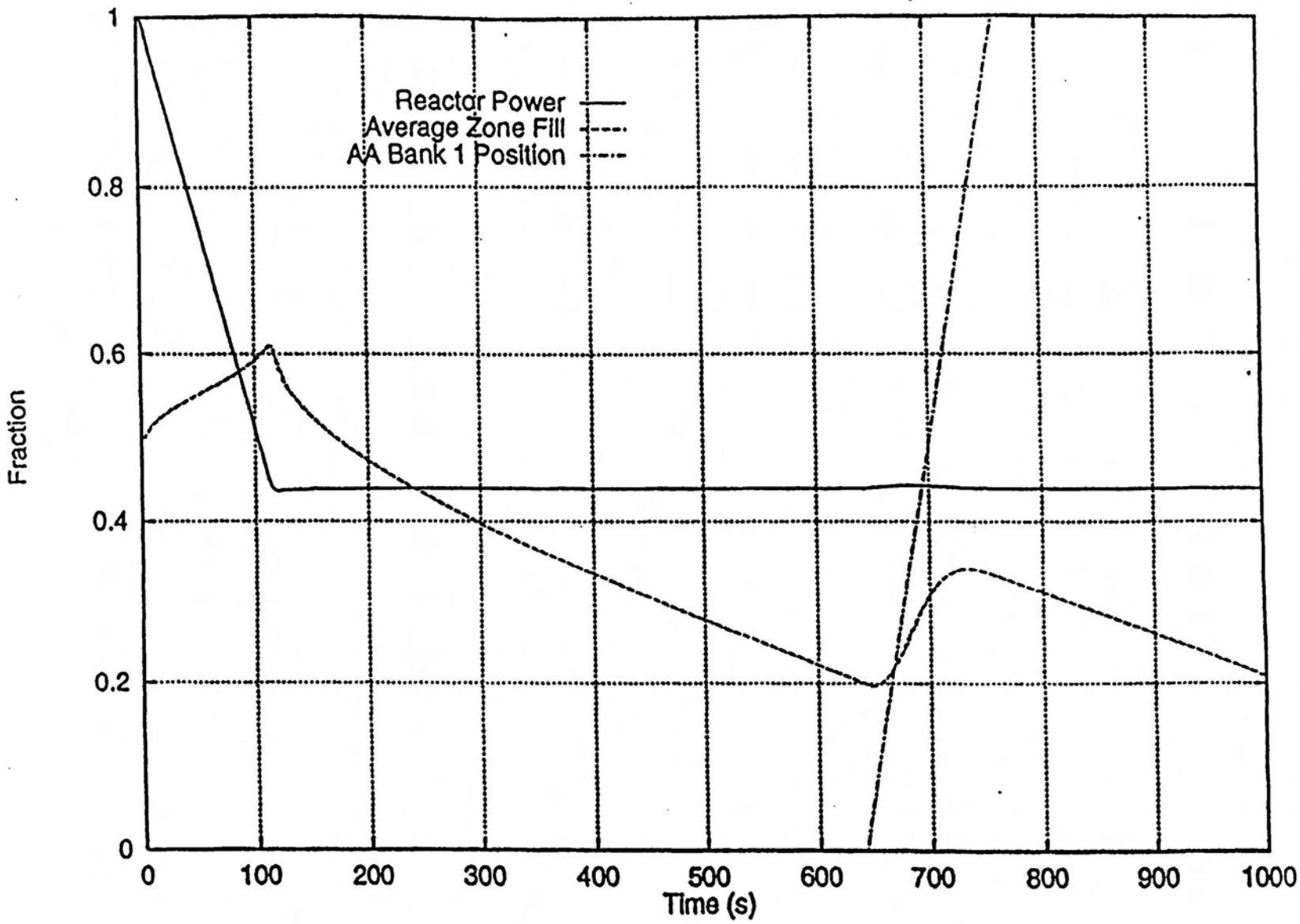


Figure 7
Power-Reduction Test - 100% to 44% FP (CERBRRS Results)

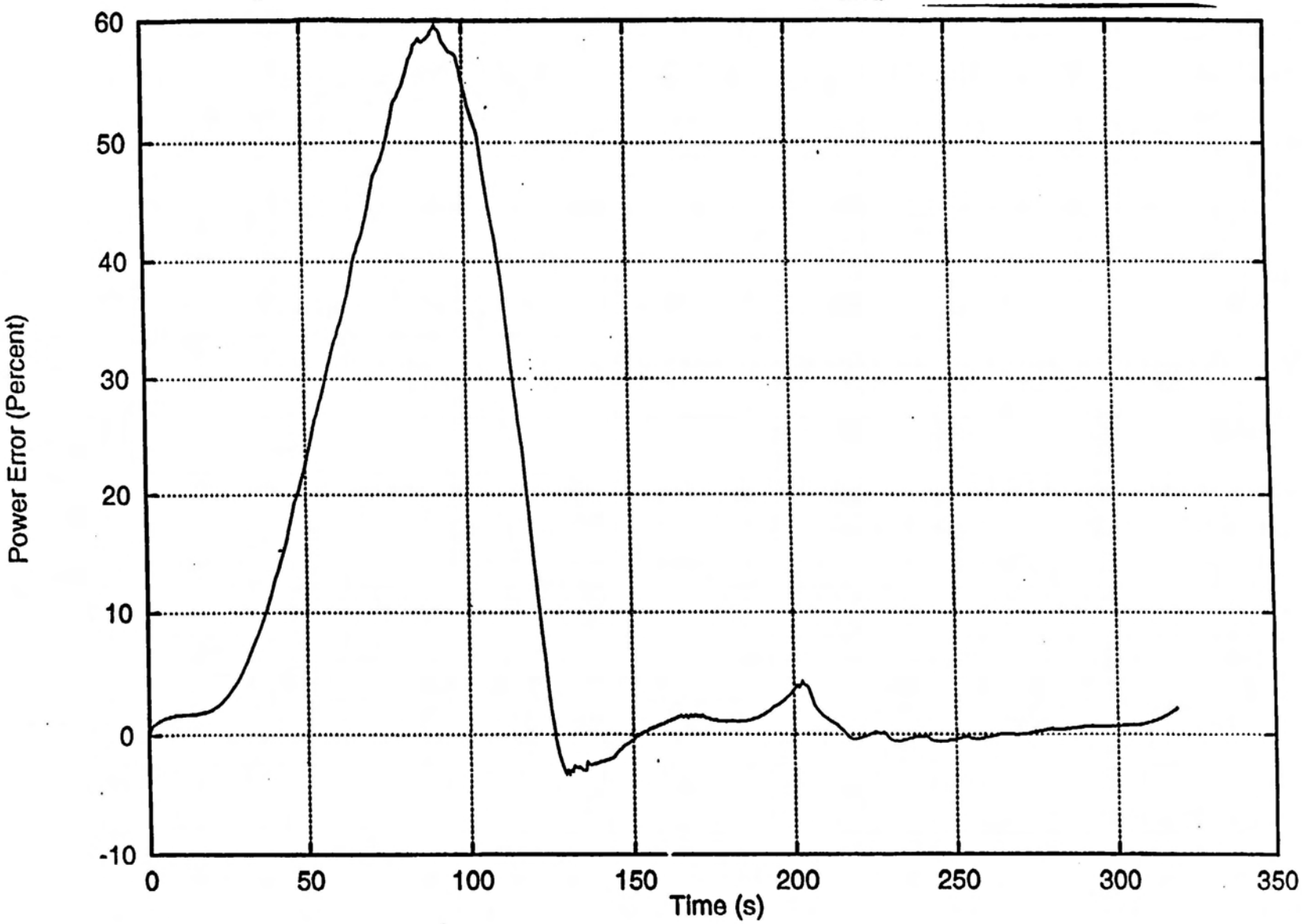


Figure 8
In-Core LOCA Test - Power Transient (CERBRRS Results)