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SCALING AND UNCERTAINTY IN BWR INSTABILITY PROBLEMS

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ABSTRACT

This paper deals with a critical review of activities, performed at the DCMN of Pisa University, in relation to the thermal-hydraulic oscillations in two-phase systems.

Stability analyses, including model development and achievement of experimental data, are generally performed for BWRs in order to achieve the following objectives:

- to reach a common understanding in relation to the predictive capabilities of system codes and to the influence of various parameters on the instability;
- to establish a data base for the qualification of the analytical tools already or becoming available;
- to set-up qualified tools (code/models + nodalization + user assumption) suitable for predicting the unstable behaviour of the nuclear plants of interest (current BWR, SBWR, ABWR and RBMK).

These considerations have been the basis for the following researches:

1) proposal of the Boiling Instability Program (BIP) [1]
2) evaluation of stability tests in PIPER-ONE apparatus [2]
3) coupled thermal-hydraulic and neutronic instabilities in the LaSalle-2 BWR plant [3]
4) participation to the "NEA-OECD BWR Benchmark" [4]

The RELAP5/MOD2 and RELAP5/MOD3 codes have been used.

1. INTRODUCTION

Thermal-hydraulic oscillations in two-phase systems have been well known since the beginning of the use of boiling water as a reactor coolant at the end of 1950s. Neutronic kinetics strongly interacts with thermal-hydraulics in boiling channels as the water is simultaneously both moderator and coolant (at least in western type BWRs). Notwithstanding the large amount of resources invested in this area, the problem of a full characterisation of instability including the coupling with neutronics has not been solved.

Following the observation of unstable situation in operating plants (e.g. [3]), the interest in this problem by the scientific community recently renewed. The following activities have recently been completed at DCMN of Pisa University:

1) a degree thesis performed in the framework of the Boiling Instability Program [1];
2) design of two-phase flow instability tests in a down scaled BWR simulator [2] [5];
3) the study of the coupled thermal-hydraulic and neutronic instabilities in the LaSalle-2 BWR plant [3];
4) the participation in the "NEA-OECD BWR Benchmark" [4].

In the first work the RELAP5/MOD3 code has been used to design the modifications in the PIPER-ONE facility[1] hardware which were needed to make it suitable for the BIP objectives and to define the scenarios of the BIP experiments. This work aimed at demonstrating the "extrapolability" [6] of experimental data obtained in PIPER-ONE, thermal-hydraulic oscillation in two-phase systems, to boiling water reactors.

In the second work the experimental program planned for the study of instabilities in boiling channels by the PIPER-ONE simulator is described. Analytical tools have been developed to predict instability conditions both in the time and in the frequency domain [2].

In the third work analyses made by the RELAP5/MOD2 code are presented highlighting the effect of different parameters on the predicted reactor stability behaviour. The basic thermal-hydraulic instability mechanisms (single channel density waves, parallel channel instabilities, loop instabilities) and reactor kinetic dynamics and feedback effects are considered in this work. The purpose is to give a contribution to both the evaluation of the phenomena occurred during the LaSalle reactor accident[2] and to the assessment of the capabilities of the RELAP5/MOD2 code in simulating the reactor behaviour.

In the frame of the last research the RELAP5/MOD2 code has been used for predicting stability performance of Ringhals-1 reactor. The activity has been carried out with the main purpose of extending the validation area of the adopted code and of the used methodology attempting to characterise the link between neutronics and thermal-hydraulics.

2. BIP ACTIVITY

The work carried out by using a qualified RELAP5/MOD3 code nodalization, can be subdivided into four main lines:

a) demonstration that unstable situations are predicted by the code in PIPER-ONE when power and core flowrate are in a range where instability occurs in BWR; typical values for these range are:
   - core power 40% - 60% of the nominal value;
   - core flowrate 20% - 35% of the nominal value.
b) demonstration that the prediction of unstable situations at item a) can be extrapolated by changing the geometrical dimensions and the number of parallel channels in the noding scheme [6];
c) demonstration that unstable conditions predicted at item a) are similar to those predicted at core power values allowed in PIPER-ONE apparatus (<25% core power);
d) planning of the experiments [7].

The range of parameters in the natural circulation flow maps where instabilities have been calculated can be seen in Fig. 1.

The matrix of the calculations [6] includes different nodalizations of multichannel ideal facilities that were set-up starting from the PIPER-ONE nodalization. The dimensions of the models and the number of parallel channels are increased to fully use the computer (IBM RISC 6000 320H) memory; the largest dimensions are those characterising the reference BWR and the maximum number of parallel channels is 50.

The results of the calculations performed by the developed nodalizations are compared with the reference calculation R5M3 (P7QC in ref. [7]). Table I shows the initial and boundary conditions of PIPER-ONE and ideal nodalizations.

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[1] PIPER-ONE is an integral test facility designed for reproducing the behaviour of BWRs in thermalhydraulic transients dominated by gravity forces.

[2] During a routine surveillance test, an instrument technician inadvertently caused the automatic shut-down of both recirculation pumps. As a consequence, the core flow rate was rapidly reduced to 29% of rated value, corresponding to natural circulation conditions, and this in turn decreased the fission power from 84% to about 45% of full power. The rapid power decrease, in turn, lead to the isolation of some of the steam extraction lines leading to the preheaters. The result of this action was a colder feedwater supply to the core, and oscillation in neutron flux that were stopped by SCRAM when maximum average power was 120% of nominal power.

Four types of instability have been distinguished, refs. [1], [6]:

a) loop instability 1, i.e. an oscillation mode involving all the circuit, which occurs before saturation conditions are reached in the lower plenum;
b) loop instability 2, i.e. an oscillation mode involving all the circuit, which occurs after saturation conditions are reached in the lower plenum;
c) local instability, characterised by oscillations in void fraction and flow rate occurring in the central core region and having small or no influence on core inlet and outlet parameters;
d) channel to channel instability, i.e. out-of-phase oscillations of flow rate and void fraction in separated channels.

Fig. 2 shows the "instability curve". This is obtained by calculating two quantities: the subcooling number (N_{sub}) and the phase change number (N_{pch}). N_{sub} and N_{pch} are defined as:

$$N_{SUB} = \frac{\Delta h_{SUB}}{h_{fg}} = \frac{h_L - h_{SAT,L}}{h_{SAT,V} - h_{SAT,L}}; N_{PCH} = \frac{P}{G_C * h_{fg}} = \frac{P}{G_C * (h_{SAT,V} - h_{SAT,L})}$$

This figure shows the instability map in the plane N_{sub} vs. N_{pch}, obtained by reporting the values of the above quantities during unstable periods.

2.1 DATA EXTRAPOLATION

In this report same concepts from the UMAE methodology [8] are applied to evaluate the "extrapolability" of the PIPER-ONE scenario to the reference plant. In the absence of experimental data, the calculation P7QC, [7], is used as reference.

Single valued parameters which identify the transient scenario have been used. The results are reported in ref. [6]: on the vertical axis, the ratio Y_r/Y_c[3] (reference calculation R5M3 over generic calculated value) is represented; on the horizontal axis the volume scaling factor is reported (K_V). Y is the value of any relevant quantity in a given transient. Examples of results are given in Figs. 3 and 4.

3. STABILITY TESTS IN PIPER-ONE APPARATUS

In the frame of the test program performed by the PIPER-ONE apparatus two experiments, for the investigation of instability phenomena, were run: PO-SD-5A and PO-SD-5B [9]. These tests were studied making use of techniques for data analysis in the frequency domain. In particular, by crosscorrelating the measured pressure drop signals, characteristic oscillation frequencies were identified (Fig. 5). Furthermore, the application of the Fourier analysis to the calculated and the measured trends, pointed out a qualitative agreement (Fig. 6).

4. INSTABILITIES IN THE LASALLE-2 BWR PLANT

The LaSalle accident has raised new concern about BWR stability, demonstrating the possibility that an instability event occurring during reactor operation could lead to scram because of high neutron flux. Other relevant instability events and incidents have been recorded all over the world in operating plants and constitute an experimental basis on which codes are currently validated. The basic thermal-hydraulic instability mechanisms (single channel density waves, parallel channel instabilities, loop instabilities) and reactor kinetic dynamics and feedback effects must be considered all together.

A detailed nodalization reproducing each geometrical zone of the reactor has been developed by RELAP5/MOD2 [10].

[3]In this case the ratio Yr/Yc was used. It is different with respect to the UMAE methodology, where the ratio Ye/Yc (experimental over calculated value) was considered.

Thermal-hydraulic analysis. Two different series of calculations have been performed. In the first series, the core power time trend has been imposed according to the information recorded during the event (STARTREC system). In the second series, the point neutron kinetics model available in the code has been used.

Three important conclusions were obtained:

1) power has an dominant effect on stability; high power levels favour the onset of core flow rate oscillations;
2) the phenomena are very sensitive to the variation of thermal-hydraulic parameters and in particular of the localised pressure drop coefficients;
3) sustained oscillations are calculated also without the neutronic feedback.

Neutronic and thermal-hydraulic analysis (reference case). Concerning the neutronic feedback, the bypass and the channel representing the central (roughly 1/3) part of the core have been considered. The comparison between experimental data and calculated results is shown in Figs. 7 and 8.

Neutronic and thermal-hydraulic analysis (sensitivity to parameters). Sensitivity analyses were carried out, basing on the reference case and changing some parameters (Figs. 9 and 10). The selected parameters are:

1) weight associated to the bypass channel in the evaluation of reactivity feedback effects;
2) number of core channels considered for the neutronic feedback;
3) void reactivity coefficient;
4) feedwater mass flow rate.

Since the bypass channel is generally full of liquid, it significantly contributes to lowering the average void fraction to be considered for reactivity feedback and significantly affects the system behaviour (Fig. 9). The increase of parameters at items 2) and 3) also leads to more unstable scenarios; while feedwater flowrate has a relatively smaller effect on the overall stability behaviour.

5. INSTABILITIES IN THE RINGHALS-1 BWR PLANT

The participation in the NEA-OECD BWR benchmark had the main purpose of extending the validation area of the adopted code and of the used methodology, attempting to characterise the link between neutronics and thermal-hydraulics.

The RELAP5/MOD2 including a 0-D kinetic model, has been used. An independently developed 1-D kinetic model has been coupled with the same code and applied to the benchmark problem.

Five main phases of the activity could be distinguished:

1) nodalization development;
2) nodalization qualification, trough the application to the calculation of the known Decay Ratio (cycles 14 and 15);
3) nodalization use - sensitivity analyses definitive of a methodology for code use;
4) nodalization use - prediction of situations relevant to cycles 16 and 17 ("blind" cases) [4];
5) use of 1-D neutronics.

Following step 2), after selecting a reference test, void coefficient was varied in the input deck up to matching the assigned DR (Decay Ratio) value (Fig. 11). This value of the void coefficient was used in all the subsequent analysis: the consideration of other tests led to confirm the validity of the choices (step 3)). In this same frame, thermalhydraulic parameters (N_{sub} and N_{pch}) were calculated for all the available tests. The "availability" of tests was judged considering these parameter values and the available axial power profile.

Blind predictions were carried out with reference to similar tests leading to the results in Fig. 12.

6. CONCLUSIONS

The applicability of RELAP5 code to the study of instability event was derived from analysing the comparison between calculated data and experimental results from PIPER-ONE tests (activity 2)).

The first work aimed at demonstrating the "extrapolability" of experimental data measured in PIPER-ONE. The main principal results are:

- identification and characterisation of four types of instability that are potentially present in the reference plant;
- demonstration of the global similitude of the scenario in the PIPER-ONE apparatus and in the facilities with increasing dimensions until the same dimensions of the reference plant had been reached;
- demonstration that PIPER-ONE apparatus is suitable in simulating only two of the four instability situations, i.e. a) and c) in sect. 2).

This analysis also showed that instability may be a system phenomenon and the applicability of simplified models (in reference to a boiling monochannel) is questionable; furthermore, the parameters N_{sub} (subcooling number) and N_{pch} (phase change number) appeared to be not sufficient to fully characterise instability situations.

From the LaSalle work (activity 3)) it can be stated that small variations of the input parameters, in the typical uncertainty ranges of plant data bring about great changes in calculation results. This does not apply to the simulation of other transients types, particularly the LOCA, and must be considered as a combined effect of both the characteristics of the instability mechanisms and of the code. On the other hand, it was seen that by "tuning" relevant parameters, it is possible to obtain a phenomenology in agreement with reality. The weight assigned to the bypass in the input data for neutron kinetics and the criteria adopted for core nodalization (e.g., number of parallel channels) have a strong effect on the prediction of system stability.

The participation to the NEA-OECD BWR Benchmark (activity 4)) confirmed independently the above findings making possible also use of "blind" test points. The trials to derive Decay Ratio values from noise analysis need additional developments; the strong influence of axial power profile on the stability behaviour was also confirmed.

As a general conclusion, meaningful qualitative information was provided by the calculations, clearly showing the influence of different boundary conditions and modelling assumptions on the obtained results. Nevertheless, at present the great sensitivity of phenomena to parameter changes, together with scarcity of information on relevant plant details and intrinsic code limitations, make difficult to obtain reliable code predictions of BWR stability behaviour with the adopted codes.

NOMENCLATURE

ABWR	Advanced Boiling Water Reactor
BIP	Boiling Instability Program
DCMN	Dipartimento di Costruzioni Meccaniche e Nucleari
LOCA	Loss Of Coolant Accident
SBWR	Simplified Boiling Water Reactor
UMAE	Uncertainty Methodology based on Accuracy Extrapolation

SYMBOLS

Δh_{sub}	subcooling	$h_{SAT.L}$	liquid saturation enthalpy
h_{fg}	latent heat of vaporization	$h_{SAT.V}$	steam saturation enthalpy
N_{pch}	phase change number	P	power
N_{sub}	sub cooling number	G_c	core flowrate
h_L	liquid enthalpy		

PARAMETERS	unit	R5M3	PIX4	2CAN	P3QC	P5QC	P10Q	P20Q	20D2	P50Q	50X6	20BW	BWR
Core Power	MW	0.631	2.524	5.048	7.572	12.62	25.24	50.48	50.48	126.2	757.2	1514.	1514.
Channel flowrate	kg/s	1.29	5.16	5.16	5.16	5.16	5.16	5.16	5.16	5.16	5.16	5.16	5.16
Core inlet flowrate	kg/s	1.29	5.16	10.32	15.48	25.8	51.6	103.2	103.2	258.	1548.	3096.	3096.
Core inlet temperature	C	273.	273.	273.	273.	273.	273.	273.	273.	273.	273.	273.	273.
Pressure	MPa	7.2	7.2	7.2	7.2	7.2	7.2	7.2	7.2	7.2	7.2	7.2	7.2
Feedwater flowrate	kg/s	0.348	1.392	2.784	4.176	6.96	13.92	27.84	27.84	69.6	417.6	835.2	835.2
By-pass flowrate	kg/s	0.14	0.56	1.12	1.68	2.8	5.6	11.2	11.2	28.	168.	336.	336.
Steam-line flowrate	kg/s	0.348	1.392	2.784	4.176	6.96	13.92	27.84	27.84	69.6	417.6	835.2	835.2
Channels number	-	1	1	2	3	5	10	20	20	50	50	20	50
Kv	-	4.E-4	0.0017	3.3E-3	0.005	8.3E-3	0.017	0.0333	0.0333	0.083	0.5	1.	1.

Tab. I - Initial and boundary conditions of PIPER-ONE and multichannel calculations

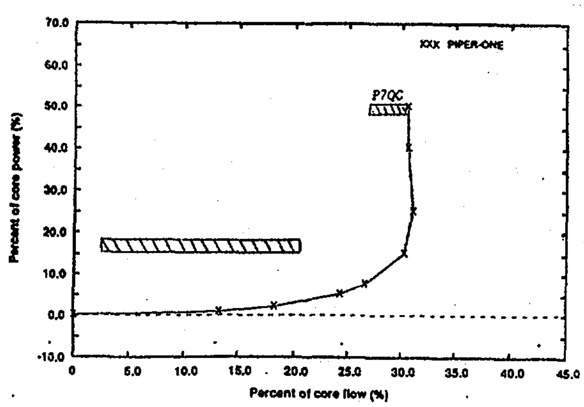

Fig. 1 - Natural circulation operating map (considered unstable areas).

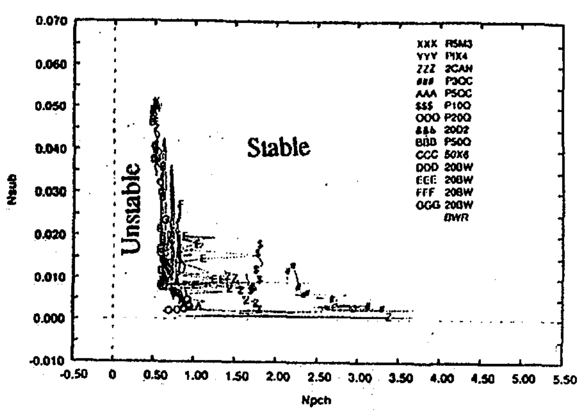

Fig. 2 - Instability map

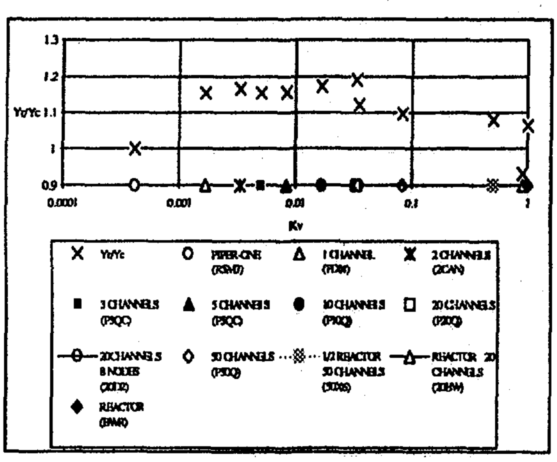

Fig. 3 - Loop instability 1: core inlet flowrate

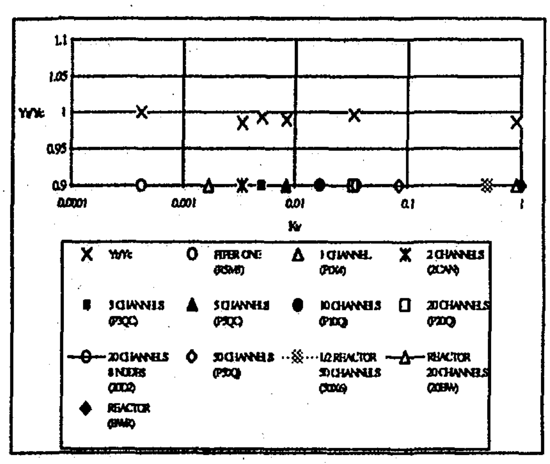

Fig. 4 - Local instability: core inlet temperature

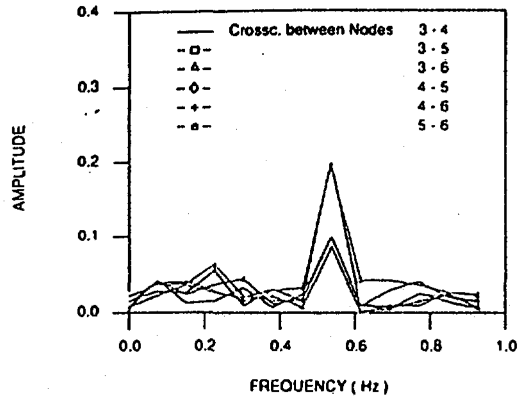

Fig. 5 - Test PO-SD-5: results of the application of the crosscorrelation to experimental differential pressure along the core (different curves relate to different positions) [2]

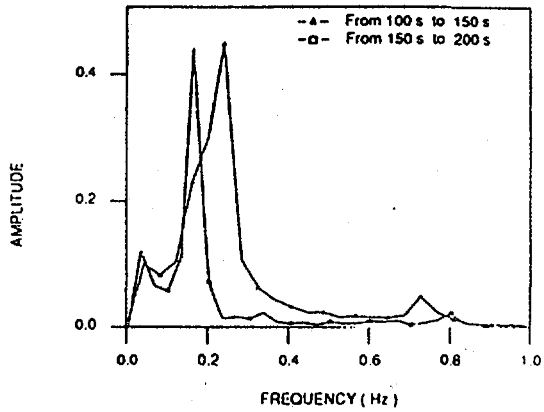

Fig. 6 - Test PO-SD-5: results of the application of the crosscorrelation to a single RELAP5 calculated differential pressure in the core, considering two different time spans

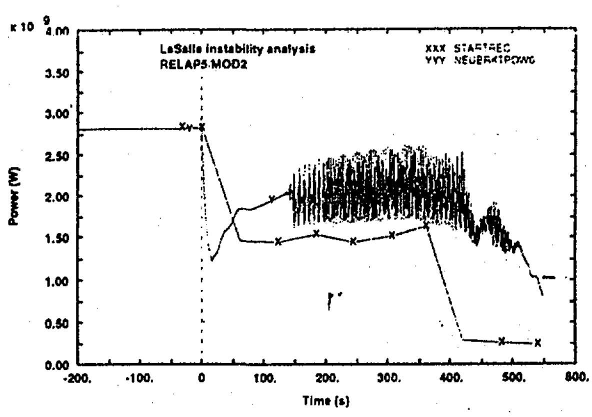

Fig. 7 - Reference case with neutronics: core power

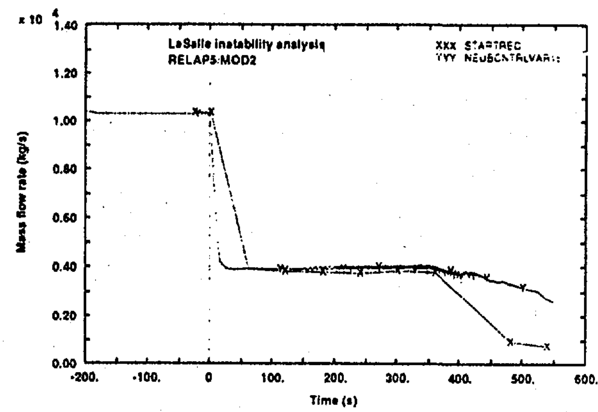

Fig. 8 - Reference case with neutronics: total core mass flowrate

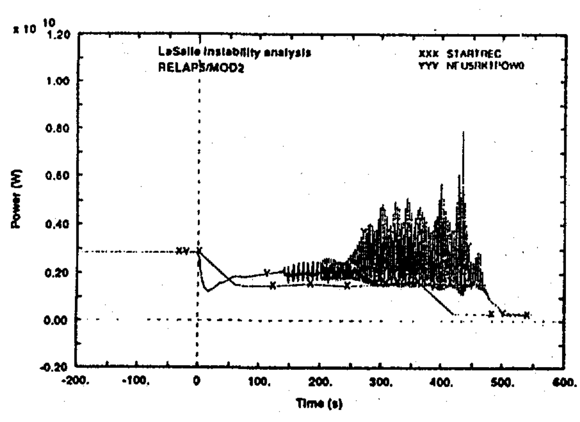

Fig. 9 - Sensitivity analysis with neutronics: core power with lower bypass weight (case 6)

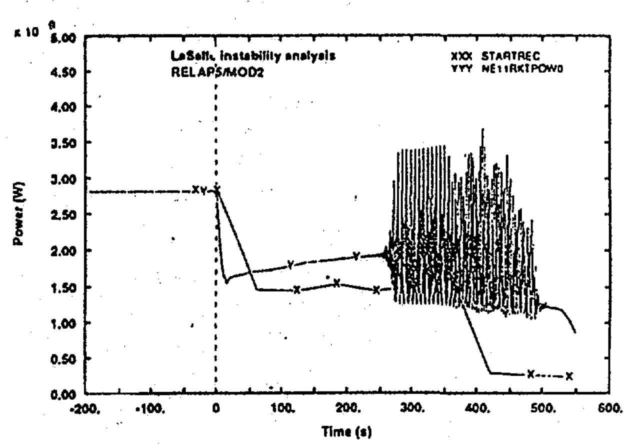

Fig. 10 - Sensitivity analysis with neutronics: core power considering three core channels for feedback (case 12)

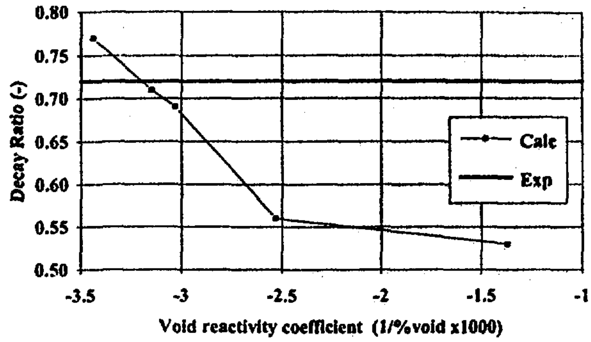

Fig. 11 - Void reactivity coefficient versus decay ratio

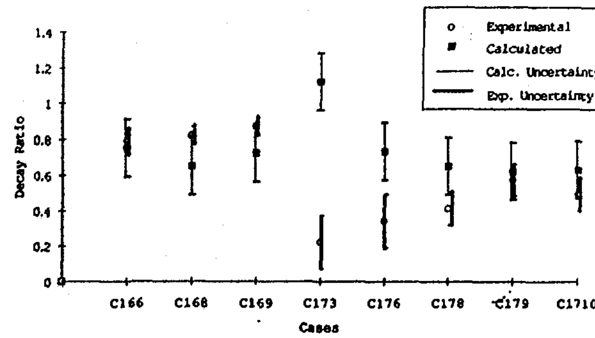

Fig. 12 - Decay Ratio in calculations and experimental cases

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