The Abdus Salam
International Centre for Theoretical Physics

United Nations
Educational, Scientific and
Cultural Organization

IAEA
International Atomic Energy Agency

SMR/1848-T12 - T13 b

Course on Natural Circulation Phenomena and Modelling in Water-Cooled Nuclear Reactors

25 - 29 June 2007

T12 & T13 - The Boiling Water Reactor Stability (part 2)

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THE LA SALLE EVENT
(Wulff et al.,NUREG/CR 5816, 1992)
THE ACCIDENT PROGRESSION

The recirculation pump trip reduced the core flow rapidly to the flow rate of natural circulation, from 76% to 29% of normal full flow. The core flow reduction in turn decreased rapidly the fission power, from 84 to about 45% of full power, because of the increase in vapor void generation and void reactivity feedback. Because the fission power decreased, the dome pressure also decreased by 1.6 bar (24 psi), and the steam flow to the turbines, and therefore the extraction steam flow from the turbines to the feedwater heaters, decreased also, thereby providing increasingly insufficient feedwater preheating. The feedwater heater control system responded then to the loss of feedwater preheating and isolated automatically some of the steam extraction lines leading to the preheaters. The isolation is needed to prevent extraction steam condensate to back up into the main turbine.

THE LA SALLE EVENT

(Wulff et al.,NUREG/CR 5816, 1992)

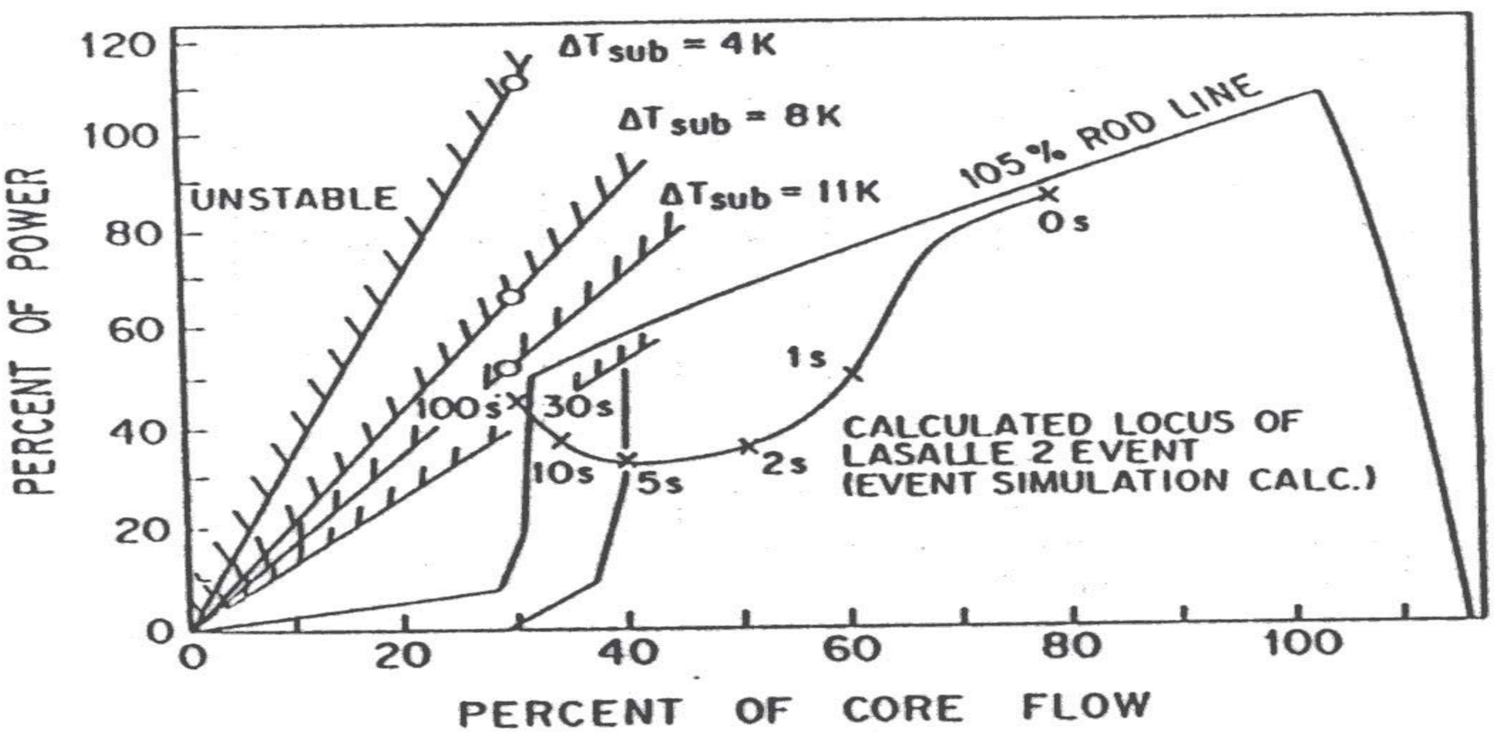
THE ACCIDENT PROGRESSION

The reduction in extraction steam flow rate and the partial isolation of steam extraction lines combined to reduce the feedwater preheating. Thus, cooler feedwater was supplied to the reactor pressure vessel and the core. The feedwater temperature dropped about 29 °C (53 °F) in five minutes. This decrease in feedwater temperature caused an increase in reactor power, as it caused both vapor void and moderator (coolant) temperature reductions, which combined with the negative void and moderator temperature coefficients of reactivity feedback to raise the fission power.

THE LA SALLE EVENT

(Wulff et al.,NUREG/CR 5816, 1992)

THE ACCIDENT PROGRESSION



Feedwater Temperature Transients used in Transient Calculations with LaSalle 2 Scinario.

THE LA SALLE EVENT

(Wulff et al., NUREG/CR 5816, 1992)

THE ACCIDENT PROGRESSION

Attempts to restart one of the two recirculation pumps, in order to increase the core flow, were unsuccessful. Then it was decided to scram the reactor manually. But just before the manual shut-down was completed, the reactor scrammed automatically on high neutron flux, at the APRM scram set point of 118% of full power. This occurred about seven minutes after the event-initiating recirculation pump trip.

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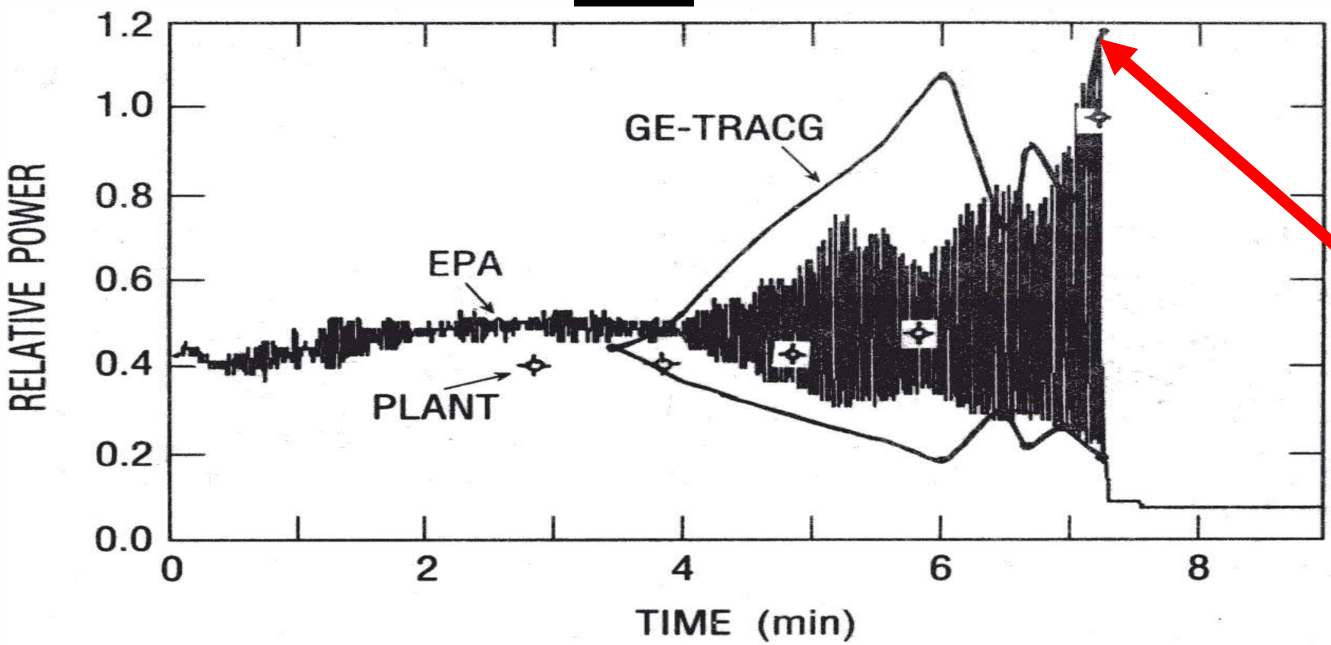
THE LA SALLE EVENT

(Wulff et al., NUREG/CR 5816, 1992)

THE ACCIDENT PROGRESSION



SCRAM

ure 2.5 Comparison of EPA-predicted Fission Power (thin line) with TRACG (GE) Prediction, carried out with imposed steam and feedwater flows (bold envelope) and Discrete Plant Data (circles), for LaSalle-2 Instability Event.

[redacted]

[redacted]

THE INSTABILITY EVENTS
(un-planned – status 1997, OECD/CSNI SOAR)

Power Plant	Status of Plant Before Event	Initial Condition Power/Flow (%)	Root Cause of Event	Type of Oscillations
Caorso	Startup	54/38	Operation within unstable area	out-of-phase
Caorso	Power Operation	-/-	Trip of one recirc. pump and loss of one feedw. preheater	out-of-phase
TVO I	Startup	62/40	Loss of one feedw. preheater train	in-phase
LaSalle 2	Power Operation	84/76	Trip of two recirc. pumps, loss of one feedw. preheater train	in-phase
Forsmark 1	Startup	63/42	Operation within unstable area	in-phase
Ringhals 1	Startup	73/50	Operation within unstable area	out-of-phase
Oskarshamn II	Power Operation	69/52	Operation within unstable area	in-phase
Cofrentes	Startup	41/31	Low feedwater temperature	out-of-phase
Isar 1	Power Operation	79/62	Trip of four recirculation pumps	in-phase
WNP-2	Startup	36/30	Skewed radial and axial power distribution	in-phase
Laguna Verde	Startup	35/38		in-phase

[redacted]

[redacted]

THE INSTABILITY EVENTS
(example of planned)

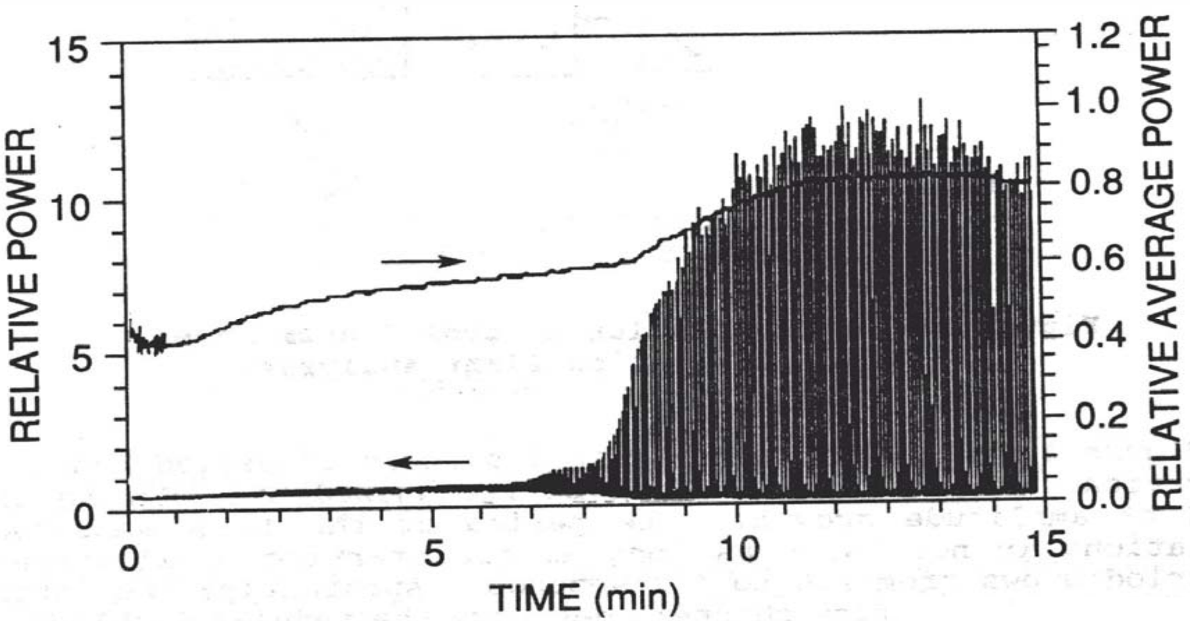
[redacted]

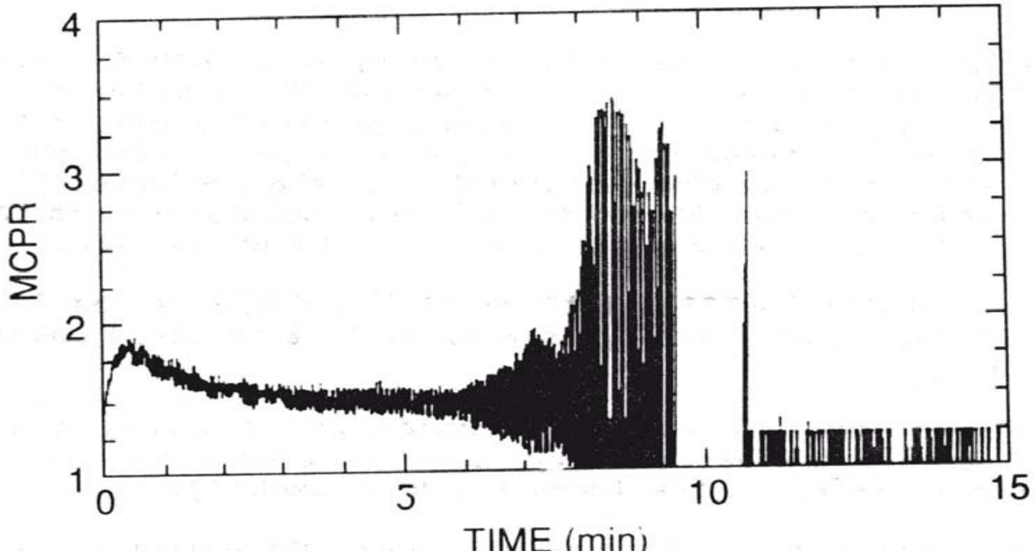
Case	Power	Core Flow	Global		Regional	
	%	kg/s	DR	f (Hz)	DR	f (Hz)
a	64.9	4116	0.3	0.43	---	---
b	64.9	3902	0.6	0.43	0.6	0.42
c	65.2	3662	0.69	0.43	0.57	0.43
d	70.2	3644	0.79	0.55	0.75	0.52
e	70	3876	0.67	0.51	0.6	0.5
f	70	4128	0.64	0.52	0.59	0.5
g	72.8	3672	0.8	0.56	0.99	0.54
h	75.2	3877	0.78	0.52	0.79	0.5
i	77.7	4104	0.71	0.5	0.63	0.49

Tab. 6-6 - Ringhals 1 BOC cycle 14 stability test results (900903-04), Ref. [6.52].

[redacted]

THE ATWS - La Salle event without scram





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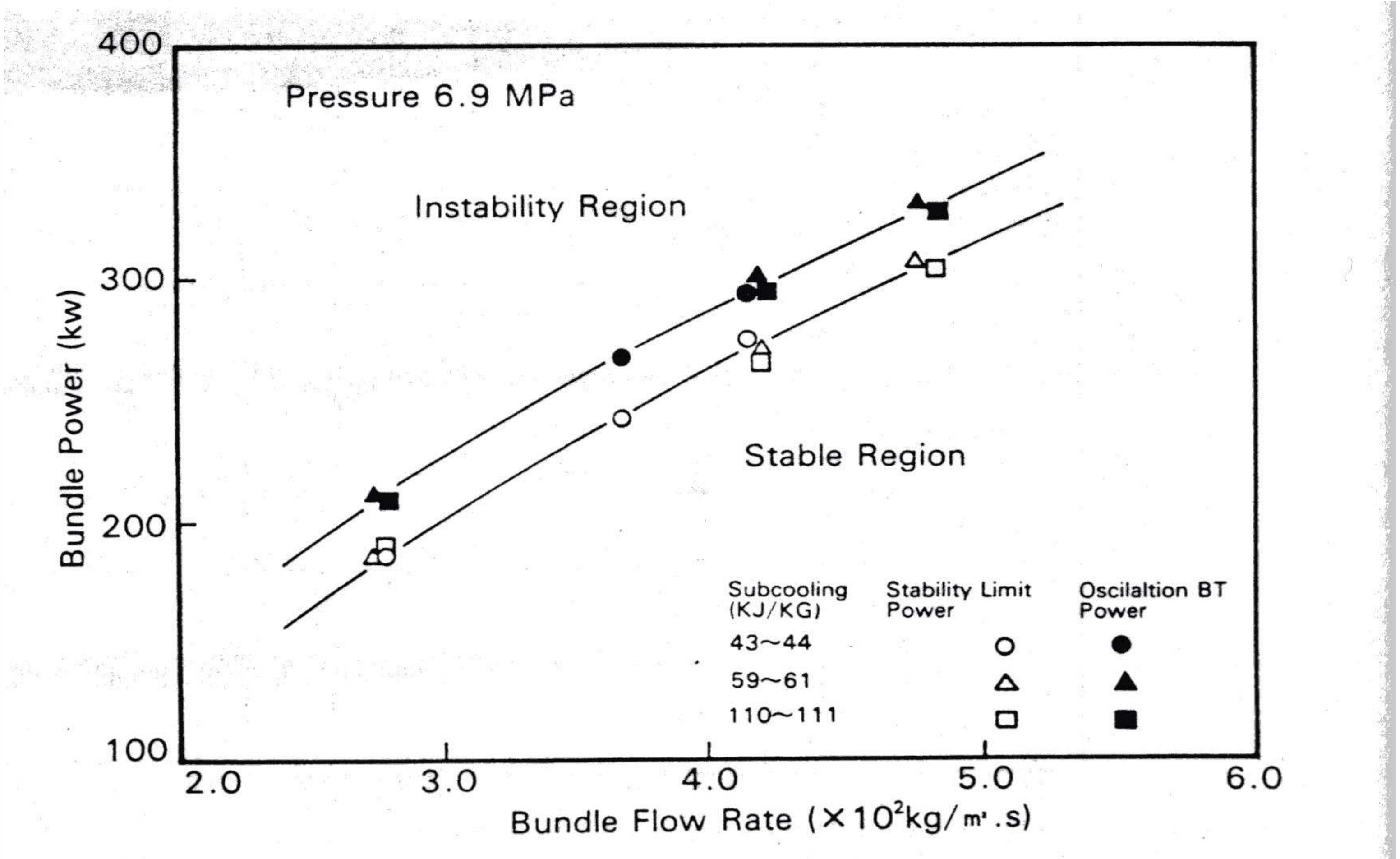
SIGNIFICANT RESULTS (Brookhaven OECD Proc.)



. 16. Relationship of critical stability power of bundle power, and bundle flow rate.

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SIGNIFICANT RESULTS (Brookhaven OECD Proc.)

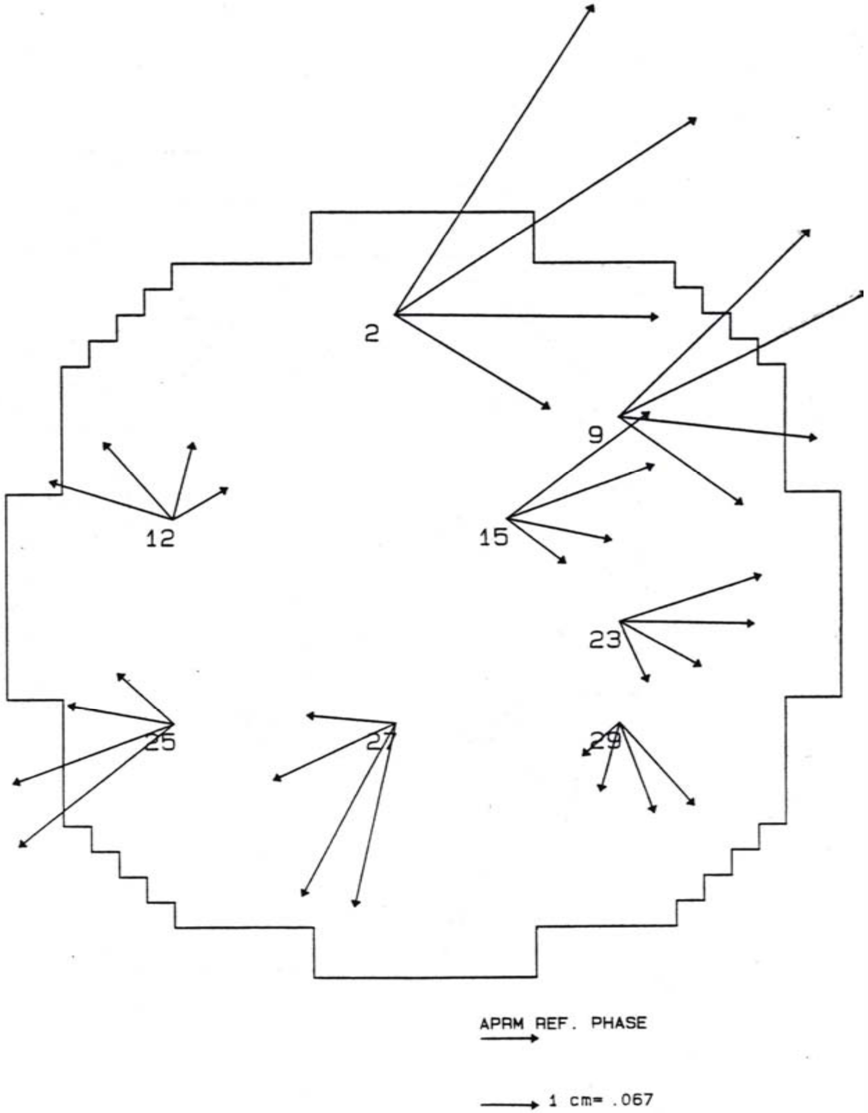
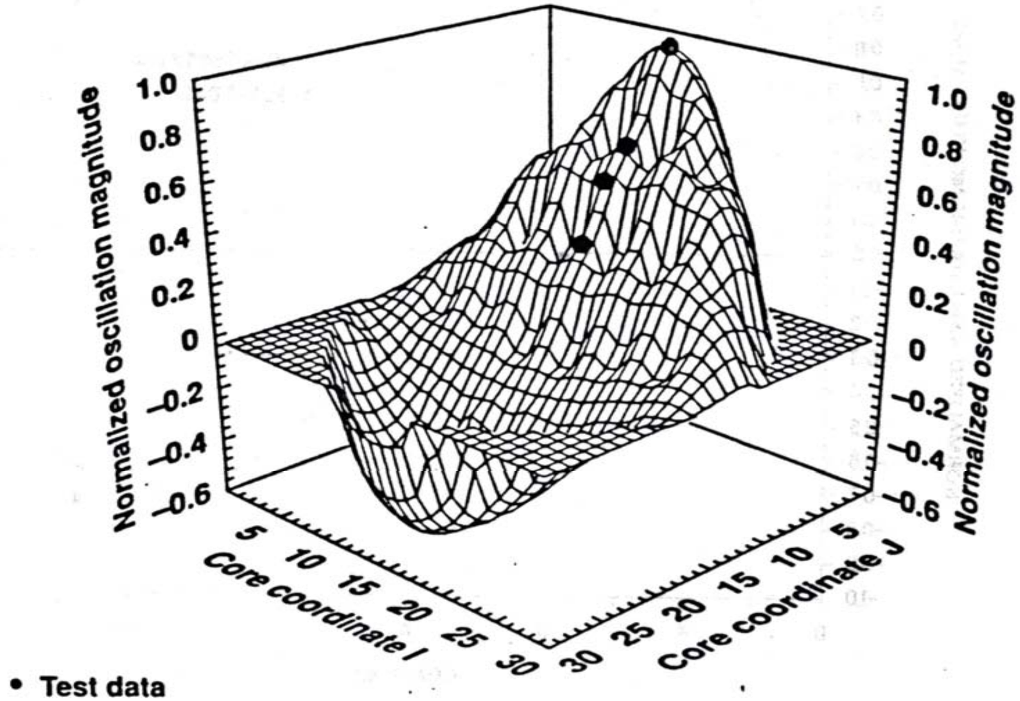


Figure 2. Measured LPRM signal relative oscillation amplitudes, 25.92 - 33.28 s.



Visualisation of regional, out-of-phase oscillations

[redacted]

SIGNIFICANT RESULTS (Brookhaven OECD Proc.)

Table 2-1: Summary classification of density wave instabilities

PHYSICAL MECHANISM	OSCILLATION MODE
Pure Thermalhydraulic	Single Channel Instability
	Parallel Channels Instability
Coupled Neutronic and Thermalhydraulic	Single Channel Oscillation
	Core Wide Oscillations (in-phase)
	Regional Oscillations (out-of-phase)

Instability Classification

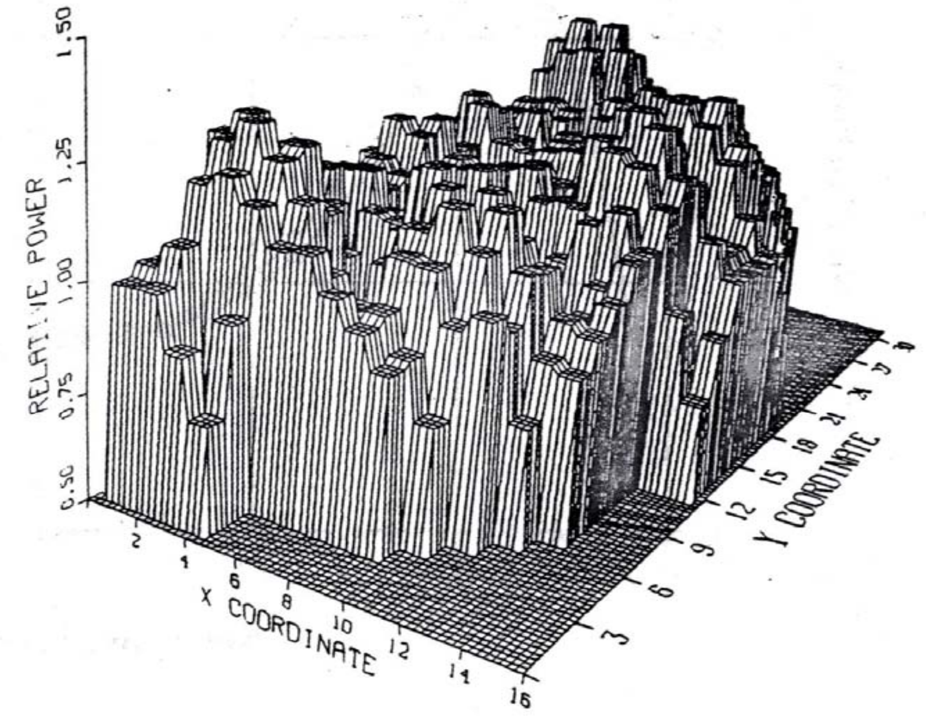


Figure 5. Initial radial power distribution, half-core model

[redacted]

[redacted]

RECENT FINDING

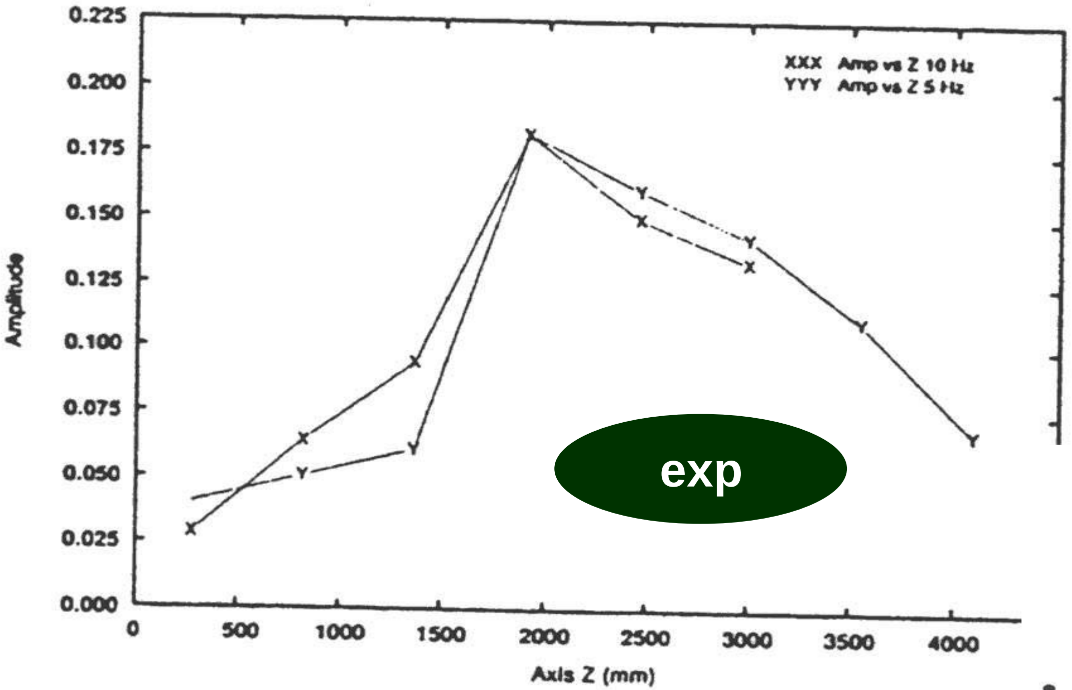


Fig. 11 - Local instability in experimental dat

Punctual Instability

"Standing DP wave" (peaked at FA middle)

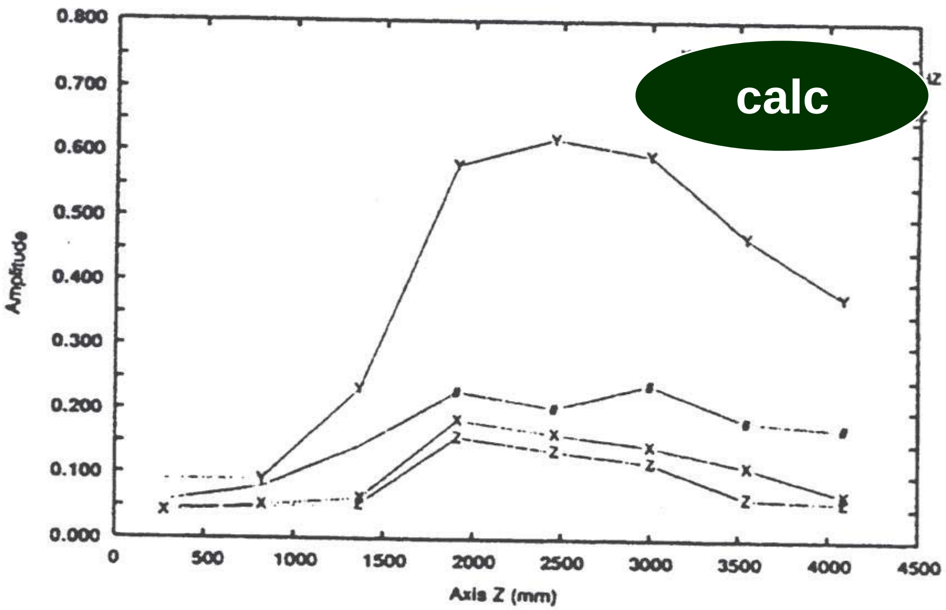


Fig. 16 - Local instability in calculated data

[redacted]

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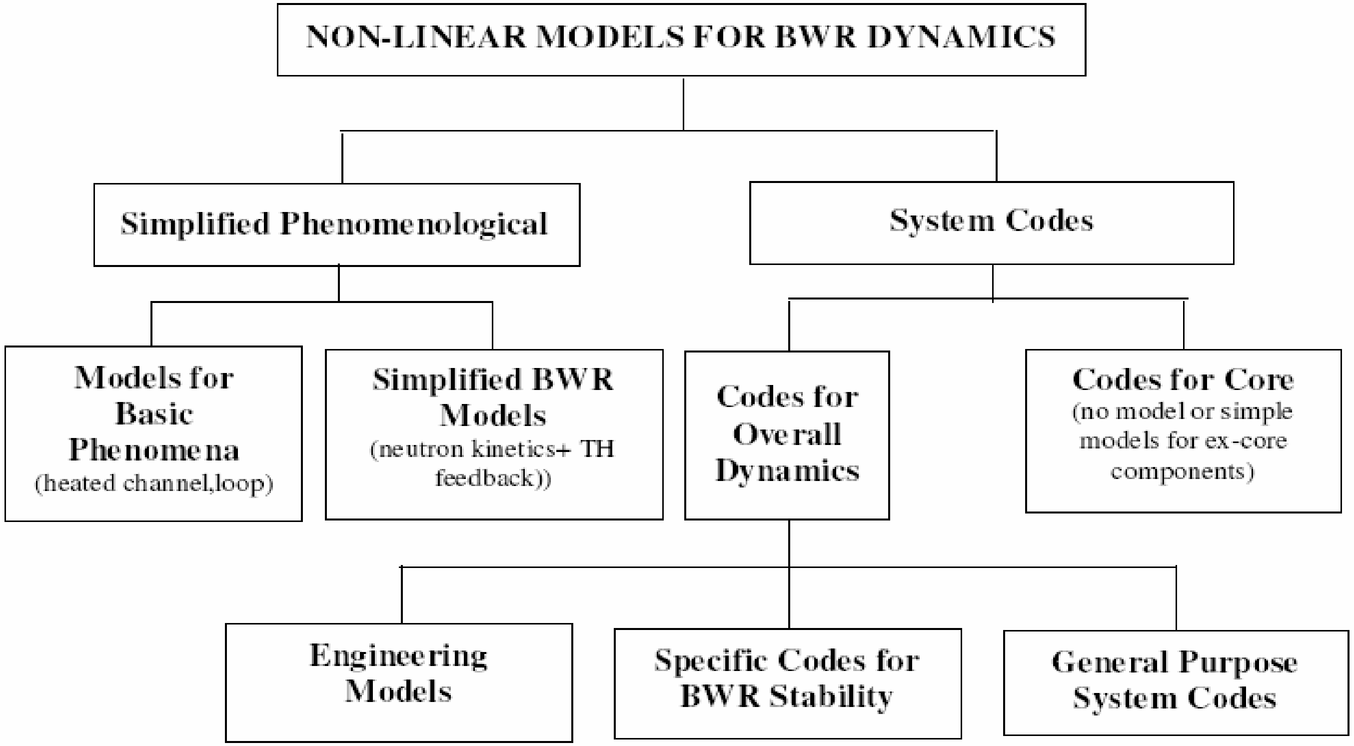


BWRS CODES (typically <u>0D-NK</u>) ARE CAPABLE OF REPRODUCINGDETECTED PHENOMENOLOGIES

PREDICTIVE CAPABILITIES
(outside qualification domain)
ARE QUESTIONABLE

[redacted]

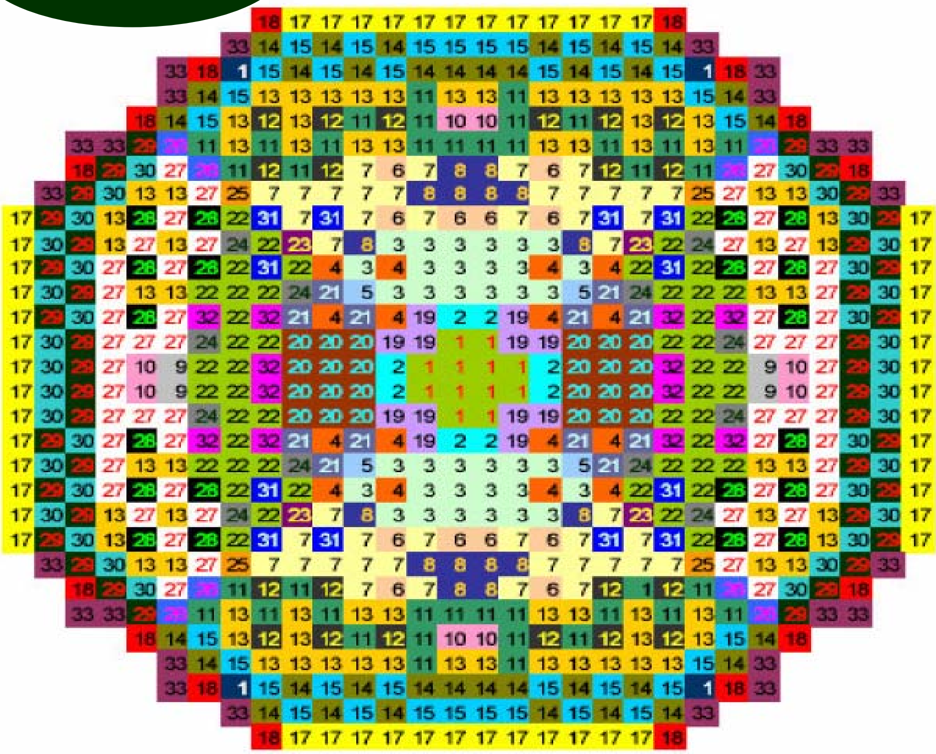
[redacted]

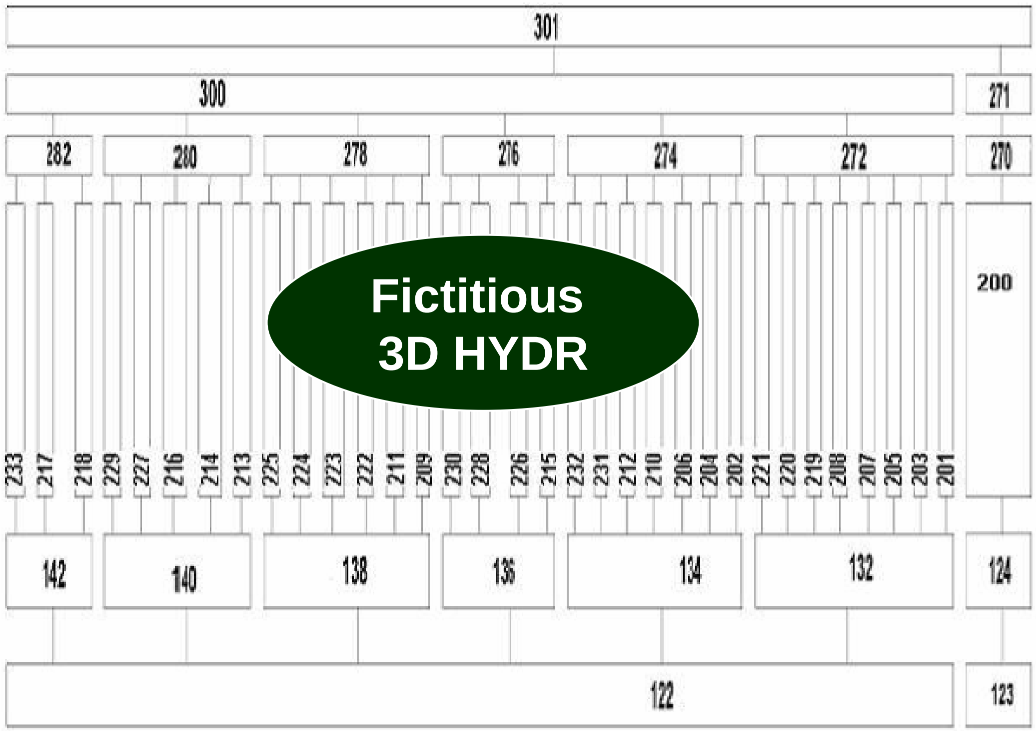
TYPICAL BWR CORE NODALISATIONS FOR <u>3D NK-TH</u> COUPLED CALCULATION ARE GIVEN BELOW

1-by-1 FA NK and several tens (over ~ 800) of parallel hyd-ch modelling possible. Fully 1-by-1 recently achieved.

[redacted]

3D NK





Fictitious 3D HYDR

[redacted]

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LIST OF APPLICABLE BWRS CODES

TD = Time Domain
D-F = Drift-Flux
NE = Thermal non-equilibrium
PNE = Partial Thermal Non-Equilibrium
RSF = Radial Space Function

FD = Frequency Domain
EQ = Thermal equilibrium
SB = Subcooled Boiling model
SM = Specific Models
Pk. = Point kinetics

Code Name	Type	Property/Developer	Thermal-hydraulics	Neutronics	Ex-core Systems
ATHLET	TD	GRS (Germany)	1D D-F NE	Pk.	1D T-H (vessel and BoP) + SM
DYNOBOSS	TD	IEA (Brazil), RPI (USA)	1D D-F EQ + SB (4 eqs.)	Pk.	Vessel modelling
DYNAS-2	TD	NFI Ltd (Japan)	1D D-F NE (5 eqs.)	3D	Vessel modelling
EPA	TD	BNL, NRC (USA)	1D D-F NE (4 eqs.)	Pk.	Vessel and BoP modelling
FABLE/BYPSS	FD	General Electric, USA	1D slip EQ + SB	Pk.	Gain and phase lags from recirculation
HIBLE	FD	Hitachi (Japan)	1D slip EQ (3 eqs.)	Pk.	Recirculation loop momentum balance
K2	FD	Toshiba, (Japan)	1D D-F EQ + SB (3 eqs.)	Pk.	Recirculation loop momentum balance
LAPUR-5	FD	ORNL, NRC (USA)	1D slip EQ + SB	Pk.	Recirculation loop momentum balance
NUFREQ-NP	FD	RPI (USA)	1D D-F EQ + SB (3-4 eqs.)	Pk. 1D, 3D	Vessel component modelling
ODYSY	FD	General Electric (USA)	1D D-F NE (5 eqs.)	1D	SM
PANTHER	TD	Nuclear Electric (UK)	1D D-F EQ + SB	3D	(inf. not available)
QUABOX/ CUBBOX-HYCA	TD	GRS (Germany)	1D D-F	3D	Coupling with ATHLET
RAMONA-3 RAMONA-4B	TD	NRC, BNL (USA) ScandPower N)	1D D-F NE (4 eqs.)	3D	1D T-H in vessel + BoP models
RELAP5/MOD2-3	TD	INEL, NRC (USA)	1D two-fluid (6 eqs.)	Pk.	1D T-H (vessel and BoP) + SM
RETRAN-3D	TD	EPRI (USA)	1D slip EQ (5 eqs.)	1D	1D T-H + SM
SABRE	TD	PP&L (USA)	1D slip EQ (3 eqs.)	Pk.	Vessel modelling
SIMULATE-3K	TD	Studsvik (S, USA)	1D EQ (3 eqs.)	3D	Coupling with external SM (TSI, J)
SPDA (EUREKA-RELAP5)	TD	JINS (Japan)	1D two-fluid PNE (5 eqs.)	3D	1D T-H (vessel and BoP) + SM
STAIF	FD	Siemens (Germany)	1D D-F NE (5 eqs.)	1D	Vessel modelling
STAIF-PK	FD	NFI Ltd (Japan)	1D D-F NE (5 eqs.)	Pk.	Vessel modelling
STANDY	TD	TEPCo, Toshiba, Hitachi	1D D-F + SB (3 eqs.)	3D	Vessel modelling
TOSDYN-2	TD	Toshiba (Japan)	1D D-F NE (5 eqs.)	3D	Vessel modelling
TRAB	TD	VTT (Finland)	1D D-F (4 eqs.)	1D + RSF	1D T-H (vessel and BoP)
TRACG	TD	General Electric (USA)	1D and 3D two-fluids (6 eqs. for 1D)	3D	1D T-H (vessel and BoP) + SM
TRAC-BF1	TD	INEL (USA)	1D and 3D two-fluids (6 eqs. for 1D)	1D	1D T-H (vessel and BoP) + SM

SOAR ON BWRS

[redacted]

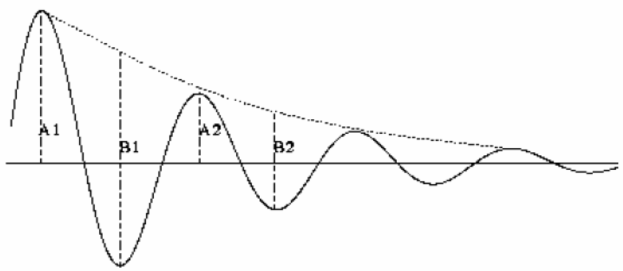
[redacted]

- **IT IS IMPORTANT TO DETERMINE WHETHER THE ACTUAL OPERATING CONDITIONS ARE IN THE STABLE REGION OF OPERATION FOR THE SELECTED NPP.**
- **THE DECAY RATIO (*the ratio of two consecutive maxima of the impulse response)* MUST BE DERIVED AT EACH TIME.**
- **SEVERAL METHODS AVAILABLE TO CALCULATE DR FROM LPRM & APRM SIGNALS**

[redacted]

If the system that models the signals is a second-order oscillator of the form

$$\ddot{x} + 2\alpha\dot{x} + \omega^2 x = 0 \qquad (4.17)$$



the general solution for the system is

$$x(t) = Ae^{-\alpha t}(\cos(\sqrt{\omega^2 - \alpha^2 t} + \varphi)) \qquad (4.18)$$

The DR parameter gives us a measurement of the damping of the system and it is defined as the ratio between two consecutive maxima of the signal. For the second-order system this parameter is a constant, and is given by

$$DR = e^{-\frac{\alpha 2\pi}{\sqrt{\omega^2 - \alpha^2}}} \qquad (4.19)$$

[redacted]

(TYPICAL) OUTPUT FROM MONITORING SYSTEM

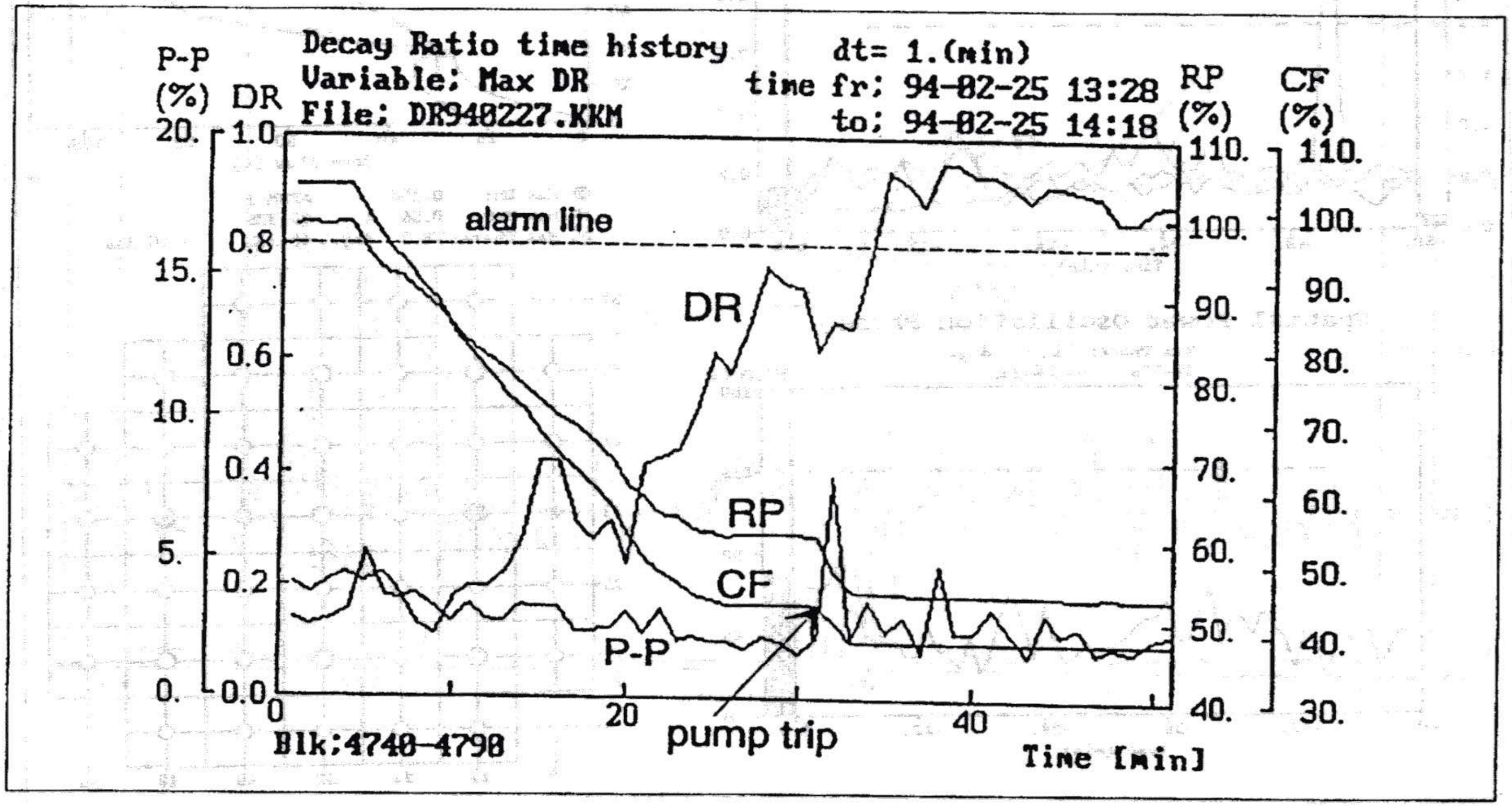
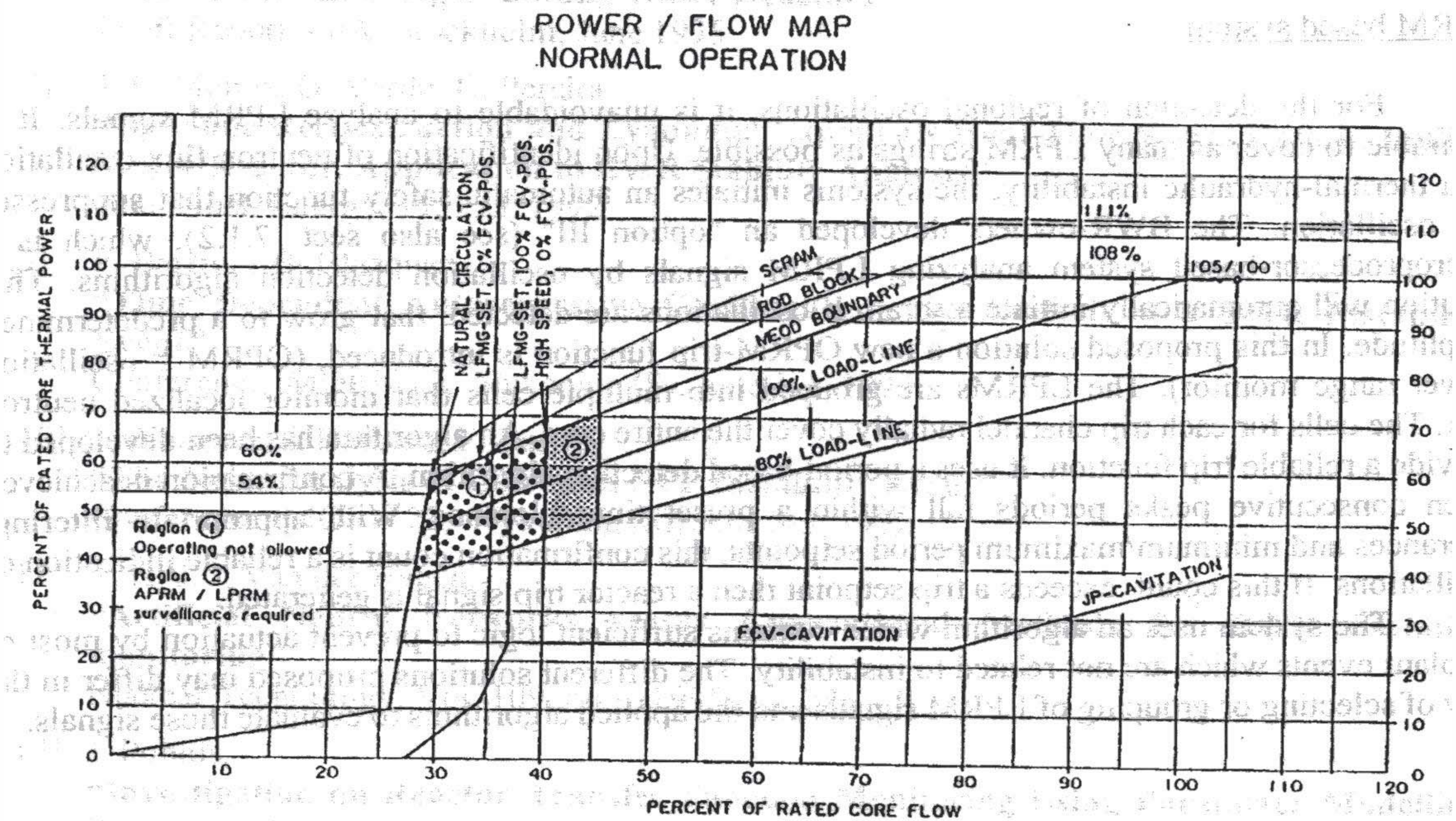


Fig. 5-4 - Time history of the stability parameters during the power decrease by reducing the recirculation flow and successive recirculation pump trip at Mühleberg BWR; P-P: peak-to-peak, DR: decay ratio, RP: reactor power, Ref. [5.1].

[redacted]

LICENSING STRATEGIES:
A) REGIONAL EXCLUSION (SEE BELOW)
B) 'QUADRANT APRM' (GE- BWR-2 design)
C) LPRM BASED (envisaged/desirable)



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RELEVANT QUESTIONS (US NRC & W. Wulff)

(1) What are the causes of large amplitude oscillations and under what conditions can they occur in a BWR?

(2) What are the inherent limits, if any, on the amplitude of power oscillations in the case of scram failure? If limit cycle oscillations occur, what then are the amplitude-limiting mechanisms?

(3) Can core-wide power and flow oscillations occur during any type of Anticipated Transient Without Scram (ATWS)? What effects can power and flow oscillations have during ATWS events, especially on Suppression Pool Temperature?

(4) What are the amplitudes of fuel pellet and cladding temperature oscillations associated with limit-cycle power oscillations?

(5) Can the safety limit of minimum critical power ratio (MCPR = 1.05) be violated during limit-cycle oscillations?

(6) For isolation events, how do the time rates of suppression pool temperature and of containment atmosphere temperature rise depend on the amplitude of limit-cycle power oscillations?

(7) Can suppression pool temperature and pressure exceed technical specification limits?

(8) What are the causes of, and conditions for, out-of-phase, i.e. region-wise, power and flow oscillations?

(9) What are the inherent amplitude limits, if any, on region-wise power oscillations?

(10) Which control rod patterns can cause region-wise power oscillations?

(11) Will a set of selected local power range monitors (LPRM) detect and signal all important region-wise power oscillations?

(12) Are available computer codes reliable for predicting BWR instability?

(13) Are stability analyses useful if they are performed with imposed neutron flux oscillations?

(14) When should frequency or time domain computer codes be used? When should point kinetics, and when should space-time kinetics codes be used?

[redacted]

RELEVANT ANSWERS (W. Wulff et al.)
Q1 (causes of oscillations)

Answer:

The instability at LaSalle-2 was a thermohydraulic instability and caused by the combination of three phenomena, namely by:

(1) core flow reduction due to the tripping of both Recirculation Pumps,

(2) radial power peaking and axial power shape with strong bottom peaking as a result of fuel burn-up, and

(3) feedwater temperature reduction due to inadvertent closure of some of the valves admitting extraction steam to the feedwater heaters.

[redacted]

RELEVANT ANSWERS (W. Wulff et al.)
Q2 (inherent limits & related mechanisms)

Answer:

(1) The EPA predicts for LaSalle-2 conditions, but with postulated scram failure, power peaks as high as 13 times the rated power (see Fig. 3.5). Under circumstances with lower feed water temperature (turbine trip, no extraction steam for feedwater preheating) and no feedwater flow reduction (100 % Bypass flow, no operator intervention), the power peaks could be higher (up to 16 times rated power; see Fig. 4.2, p. 71 and Sect. 4.6, p. 98).

Even higher power peaks are possible in the unlikely event that the feedwater control system failed in the maximum demand position after a turbine trip with 100% Bypass flow.

(2) Doppler and void reactivities limit the growth of the fission power amplitude (see Fig. 3.14 on p.53). During large-amplitude, limit-cycle power and flow oscillations, the reactor remains subcritical on the average over an oscillation period, with the mean total reactivity of approximately -4.0 $, while the instantaneous total reactivity swings between -9.3 $ and +1.04 $.

For very large power oscillations, both void and Doppler reactivity curb the fission power rise, but the Doppler reactivity feedback determines both peaking time and magnitude of the peak, because the Doppler reactivity drops off very sharply before the void reactivity peaks. (See Sect. 3.3.2, p. 54; Sect. 3.4, p. 62; Sect. 4.6, p. 97.)

[redacted]

[redacted]

RELEVANT ANSWERS (W. Wulff et al.)
Q3 (ATWS)

(1) An ATWS, caused by scram failure and the simultaneous tripping of turbines and both recirculation pumps would lead, without any further operator intervention, to:

(a) large core-wide power and flow oscillations with **power peaks of 1,600 % of rated power, larger than are predicted for the conditions of the LaSalle-2 instability** in 1988 (see Sect. 4.2, p. 68).

(b) the rise in the **pool temperature** to its **limit** of 353 K (80°C, 175°F) in only **7.2 minutes.**

(2) Large limit-cycle power and flow oscillations in a BWR give rise to an **increase in time-mean fission power** above that which is attained during stable natural circulation after a dual recirculation pump trip. The rise is 2.2% for a 100% increase in peak power (see Fig. 3.7), i.e. 29% for the 1,300% increase in power reported under Question No. 2 above.

(3) No steam is discharged into the Suppression Pool during large power oscillations, as long as the turbines are not tripped and the Main Steam Isolation Valves (MSIV) do not close (on high radioactivity in the steam), because the vessel pressure remains below the setpoint for Safety and Relief Valve opening. If steam were to be discharged into the suppression pool during an ATWS with large power and flow oscillations in the core, be it by operator action, automatic MSIV closure or Turbine Trip, then the **elevated mean fission power** would cause the suppression **pool temperature** to reach its **limit** of 353 K (80°C, 175°F) **faster** than it would during normal ATWS conditions with over-pressurization of the vessel. See Fig. ES-1 for the time spans.

[redacted]

[redacted]

RELEVANT ANSWERS (W. Wulff et al.)
Q4 (fuel pellet)

Answer:

(1) For the March 9, 1988 LaSalle-2 conditions, but with postulated scram failure, the EPA predicts the fission power to oscillate between 30 and 1,300% of rated power, and the fuel centerline temperature between 1,200 and 1,755 K (1,700 and 2,700°F), while the fuel cladding temperature is oscillating between 563 and 569 K (554 and 565°F). (See Sect. 3.4.)

(2) Under conditions with 100% Bypass flow and automatically controlled feedwater flow, the fission power is predicted to oscillate between 40 and 1,700% of rated power, the fuel centerline temperature between 1,033 and 2,088 K (1,400 and 3,300°F), the fuel mean temperature between 726 and 1,089 K (850 and 1,500°F), and the cladding temperature between 563 and 569 K (553 and 565°F). (See Sect. 3.4.)

(3) Should the feedwater regulator fail in the full demand position, at 100% Bypass flow, then one would have to expect even larger temperature oscillations. The EPA predictions are based on a rewet model (cf. Section 9.1.3) which could not be confirmed for periodic flow conditions, because no experimental data were available.

[redacted]

[redacted]

RELEVANT ANSWERS (W. Wulff et al.)
Q5 (MCPR)

Answer:

(1) The Minimum Critical Power Ratio does not fall below MCPR = 1.05 during power and flow oscillations, if the scram system shuts down the reactor before the power peaks exceed 118 % of full power. Then there is no fuel damage expected. (See Sect. 2.7, p.38)

(2) If the scram system fails to shut off the reactor, then the safety limit will be temporarily violated, as shown in Fig. 3.19. However, the widely accepted MCPR correlation in the EPA could not be confirmed for oscillatory flow conditions, because there were no data available (see Sects. 3.3.4, Fig. 3.19; Sect. 3.4 and also Sect. 9.1.3 regarding the rewet model limitations in the EPA, p. 162).

[redacted]

[redacted]

RELEVANT ANSWERS (W. Wulff et al.)
Q6-Q7 (PSP related – not strictly connected with the subject)
Q8-Q11 (answer below, not given by Wulff et al. because 3D NK-TH needed)

Q8 (causes for regional oscillations): **LOADING PATTERNS, DIFFERENT FUEL, ASYMMETRIC PERTURBATION, DP OF INDIVIDUAL FA**

Q9 (inherent amplitude limits for regional oscillations): **SAME AS FOR IN-PHASE OSCILLATIONS**

Q10 (control rod patterns causing regional oscillations): **VARIOUS, INDIVIDUAL NPP-CORE ANALYSIS (PLANT SPECIFIC) NEEDED**

Q11 (are LPRM useful for detection of regional oscillations): **BASICALLY YES, UPGRADINGS OF CURRENT DESIGN COULD IMPROVE THE CAPABILITY**

[redacted]

RELEVANT ANSWERS (W. Wulff et al.)

Q12 (Reliability of computer codes): general comment provided in the Conclusions

Q13 (are stability analyses useful if they are performed with Imposed neutron flux oscillations?): NO

Q14 (use of frequency or time domain codes): … part 1

[redacted]

Answer:

We answer this four part question without any calculations.

(1) Frequency domain computer codes are based on linearized equations. They require no time integration, as they are developed to obtain the stability boundary via the so-called Decay Ratio, or ratio of two successive amplitudes (which is a growth ratio, if greater than unity) from the leading eigenvalue of the characteristic systems equation.

Having no time discretization errors, frequency domain codes should predict more precisely the Decay Ratio than time domain codes. Requiring no numerical integration of

[redacted]

[redacted]

Q14 (use of frequency or time domain codes): … part 2

partial differential equations, frequency domain codes are less expensive than time domain codes. Being based on linearized equations, frequency domain codes are restricted to the determination of decay ratios and related parameters of linear perturbation analysis.

Thus, frequency domain codes are superior to time domain codes in their restricted capability of predicting decay ratios and stability boundaries.

(2) Time domain codes are indispensable for the analysis of all nonlinear effects, i.e. for the determination of amplitudes of power, flow or fuel temperatures during oscillations, for the determination of plant responses to operator actions, malfunctions and functions of control systems.

(3) Both time and frequency domain codes can be used effectively as scoping analysis tools, if they are designed for efficiency. This is known in the case of frequency domain codes and demonstrated in this report for the time domain code HIPA of the EPA.

(4) Computer simulations with point kinetics are suitable for analyzing core-wide, in-phase power and flow oscillations, provided the time-dependent radial and axial distortions of the fission power distributions can be modeled as shown in Chapter 2 of this report and confirmed through the use of plant data, such as the data from the LaSalle-2 instability of March 9, 1988.

(5) Computer simulations with three-dimensional neutron kinetics are indispensable for the analysis of region-wise, out-of-phase power and flow oscillations, as well as for the study of all transients with asymmetric power and flow distributions.

- BWR INSTABILITIES ARE WELL KNOWN TO THE TECHNOLOGY AND DEEPLY INVESTIGATED SINCE THE BEGINNING

- BWR NPP OPERATION HAS BEEN AFFECTED BY INSTABILITY EVENTS THAT (SO FAR) DID NOT POSE UNACCEPTABLE SAFETY RISK

- COMPUTATIONAL TOOLS ARE AVAILABLE THAT REPRODUCE KNOWN STABILITY CONDITIONS. CAPABILITY TO PREDICT INSTABILITIES OUTSIDE THE QUALIFICATION DOMAIN IS QUESTIONABLE

- MONITORING IS EFFECTIVE IN PREVENTING INSTABILITIES. THE LICENSING FRAMEWORK APPEARS ADEQUATE

- NEW FRONTIER CONSTITUTED BY THE COUPLED 3D NK-TH COMPUTATIONAL TECHNIQUES CAPABLE OF SIMULATING THE PERFORMANCE OF INDIVIDUAL FUEL ASSEMBLIES