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QUALIFICATION OF THE TH-3DNK COUPLED CODE RELAP5/PARCSV2.7 AGAINST A BWR UNSTABLE POINT AT PEACH BOTTOM NUCLEAR POWER PLANT

T. Barrachina, R. Miró*, A. Soler, G. Verdú
Chemical and Nuclear Engineering Department
Polytechnic University of Valencia
Camí de Vera s/n, 46021 Valencia, Spain
tebarcel@upvnet.upv.es, rmiro@iqn.upv.es; amsomar@etsii.upv.es, gverdu@iqn.upv.es;

1. INTRODUCTION

The possibility of an instability event in the core of a boiling water reactor induced by thermal-hydraulic and void reactivity feedback has been the subject of many analytical and experimental investigations. However, to improve the safety systems of these reactors, it is necessary to be able to detect in a reliable way these oscillations from the neutronic signals processing.

To characterize the unstable behavior of the Peach Bottom Unit 2 BWR, a number of perturbation analyses were performed: arrangements with Philadelphia Electric Company (PECo) were made for conducting different series of Low Flow-Stability Tests at Peach Bottom 2, during the first quarter of 1977.

The Low Flow Stability Tests intended to measure the reactor core stability margins at the limiting conditions used in design and safety analysis, providing a one-to-one comparison to design calculations.

The selection of this reactor was based on the fact that it is a large BWR/4 which reaches the end of its Number 2 reload fuel cycle early in 1977, with an accumulated average core exposure of 12.7 GWd/t.

Stability tests were conducted along the low-flow end of the rated power-flow line, and along the power-flow line corresponding to minimum recirculation pump speed. The actual reactor operating conditions at which the low-flow core stability testing was conducted are listed in table I and showed in Fig. 1 [1].

*Corresponding author

Table I. Peach Bottom-2 End-of-Cycle 2. Low-Flow Stability Test Conditions.

Test Number	Reactor Power		Core Flow Rate		Core Pressure[a]
	(MW_t)	(%Rated)	(kg/s)	(% Rated)	(MPa)
PT1	1995	60.6	6753.6	51.3	6.89
PT2	1702	51.7	5657.4	42.0	6.84
PT3	1948	59.2	5216.4	38.0	6.93
PT4	1434	43.5	5203.8	38.0	6.89

[(a)]Based on process computer edit (P1), corrected for steam separator pressure drop

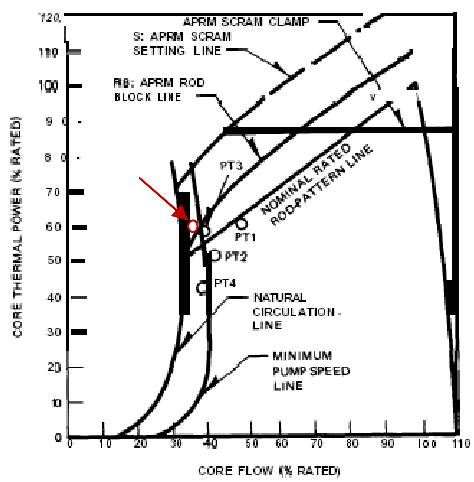

Figure 1. Peach Bottom 2 Low Flow Stability Tests Conditions.

In previous works [7], the results showed that point PT3 is a nearly stable point and it is at the end of cycle, while the obtained average axial power distribution shows a non bottom-peaked profile (stable).

In this work, three dimensional time domain BWR stability analysis were performed on a new analysis point (PT_UPV), which is inside the exclusion region with a core mass flow of 4660.1 kg/s (34% of the core rated mass flow) and total reactor power of 1997.8 MW (60.7 of the core rated reactor power), using the coupled code RELAP5-MOD3.3/PARCSv2.7. This point is achieved departing from test point 3 by the control rod movement as it is usual performed in Nuclear Power Plants. The transient starts with the control rod movement shown in Fig. 2[1]. The control rods move in 6 seconds; at the end of the movement the majority of the banks are completely withdrawn and only the bank 7 is almost completely inserted.

[1] Each number represents a position of the control rod, e.g. it correspond to a number of 3-inch notches withdrawn

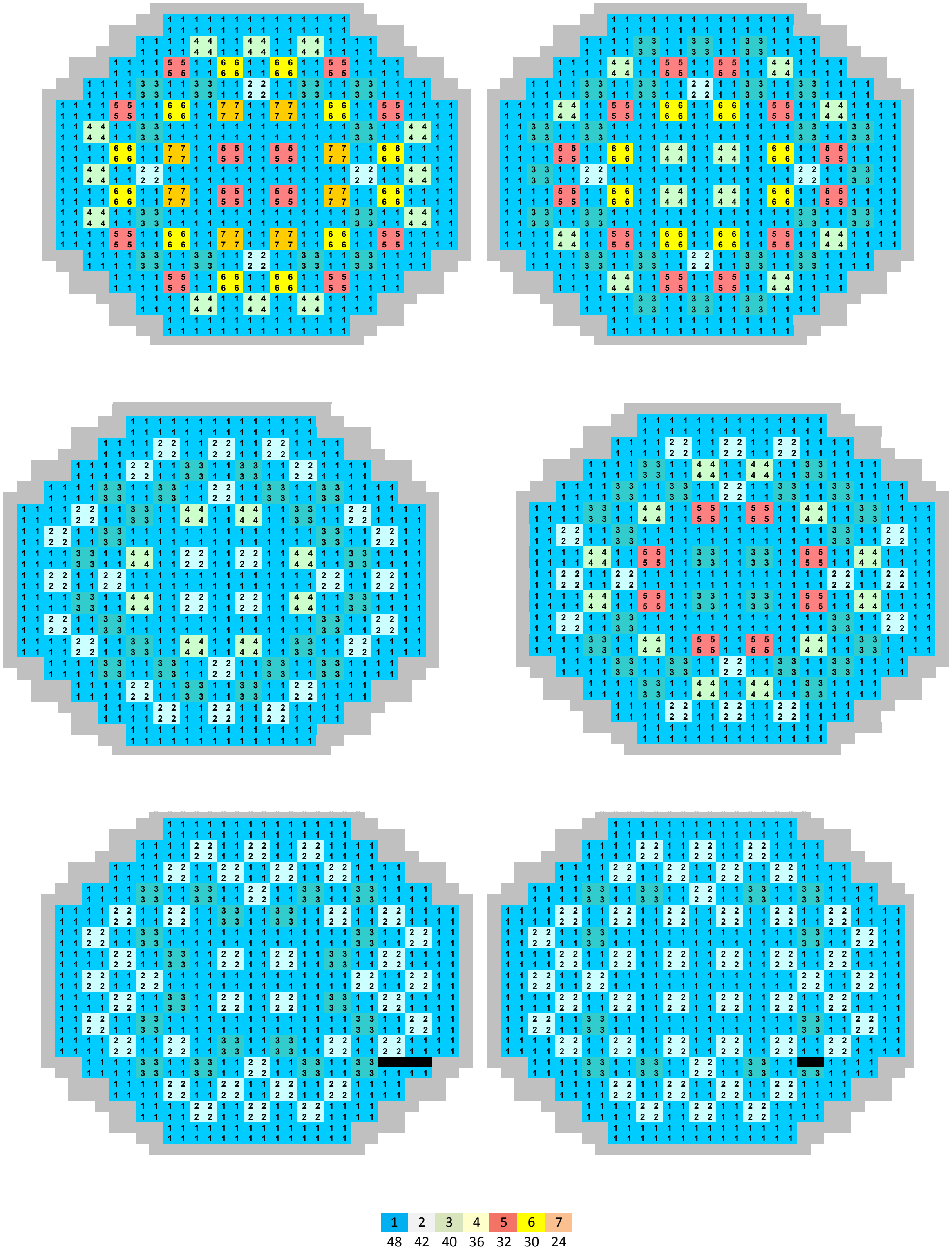

Figure 2. Control Rods Movement.

The purpose of this study is to qualify this coupled code against this kind of 3D complex accidents that take place inside the core.

The calculated results show that point PT_UPV is an unstable point and the obtained relative axial power distribution shows a bottom-peaked profile, which is characteristic of unstable cores. In Fig. 3 the change in the reactor axial power shape after the control rod movement is shown.

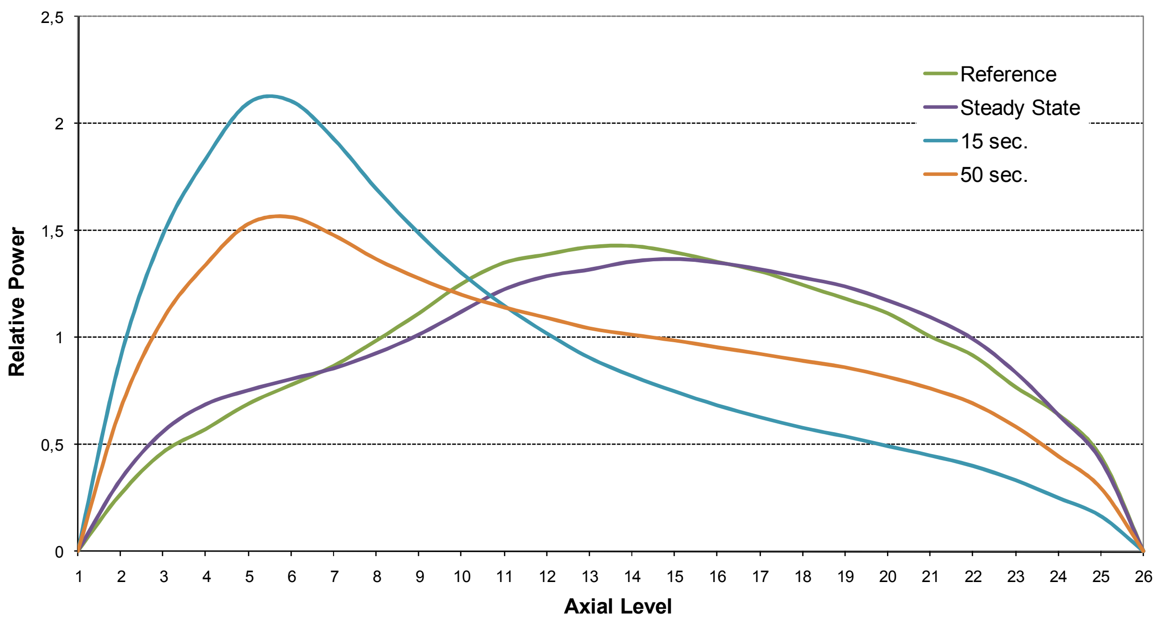

Figure 3. RELAP5/PARCSv2.7 Relative Axial Power.

The variation of the axial power profile, that assumes a bottom peaked shape, produces in the system an unstable behavior, just after the end of the control rod movement.

From the Fig. 4 it is possible to see that, after control rod movement, a limit cycle in-phase oscillation on the total reactor power evolution is obtained.

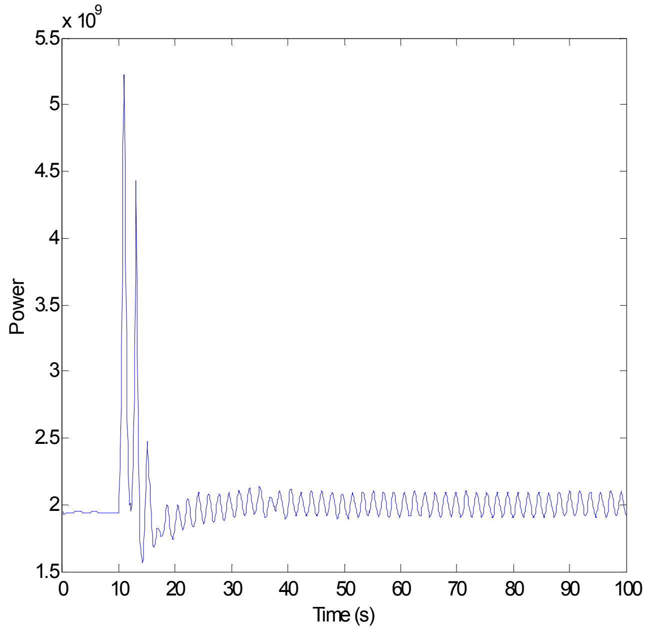

Figure 4. Power evolution during transient.

With the aim to make a more careful analysis of the instability recognized in the RELAP5/PARCS simulation, using the nuclear cross-sections provided by the transient calculation performed with the coupled codes, a number of analyses with the nodal modal code VALKIN has also been carried out.

The core neutronic data used in all the calculations are specified in [2].

For all the calculations, it has been developed the same detailed thermal-hydraulic nodalization reproducing each geometrical zone of the plant [3], [4].

For the core, 33 thermalhydraulic channels have been modeled to represent the active part of the core and one channel for all by-passes. For the rest of the plant a coarse nodalization has been adopted for limiting the needed computer resources.

For the neutronic code, a nodalization with a 3D core mesh composed with 764 axial nodes has been modeled. A large set of cross section data including 435 compositions has been adopted in neutronic input deck [2].

Moreover, the above presented results for the power modal analyses have been complemented by considering the information provided by the simulation of the LPRM signals by RELAP5/PARCS. A modal decomposition of the neutronic power was performed from the local power distribution in the reactor core (made available by the coupled codes) and then, this decomposition was compared with the one achieved from the LPRM outputs also simulated by the same coupled codes.

Finally, in order to observe the difference between the results obtained using different numbers of modes or different updating times, different transient calculations have been carried out.

REFERENCES

1. L. A. Carmichael, R. O. Niemi, "Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2", EPRI Report NP-564, 1978.
2. K. Ivanov, et al., "Boiling Water Reactor Turbine Trip (TT) Benchmark", Volume 1: Final Specifications, NEA/NSC/DOC (2001).
3. F. Maggini, R. Miró, F. D'auria, G. Verdú, D. Ginestar, "Peach Bottom Cycle 2 Stability Analysis using RELAP5/PARCS", *International Conference Nuclear Energy for New Europe*, Ljubljana, Slovenia (2003).
4. A. M. Sánchez, R. Miró, G. Verdú, D. Ginestar, "Test de Estabilidad en el Reactor Peach Bottom Unit 2 con el Código Acoplado TRAC-BF1/VALKIN", *29 Reunión Anual de la Sociedad Nuclear Española*, Zaragoza, Spain (2003).
5. R. Miró, D. Ginestar, G. Verdú, D. Hennig, "A Nodal Modal Method for the Neutron Diffusion Equation. Application to BWR Instabilities Analysis", *Annals of Nuclear Energy*, **29**, pp.1171-1194 (2002).
6. G. Verdú, R. Miró, D. Ginestar, V. Vidal, "Transients Modal Analysis using TRAC-BF1/VALKIN", *PHYSOR 2002*, Seoul, Korea (2002).
7. R. Miró, A. M. Sánchez, G. Verdú, F. Maggini, F. D'Auria, D. Ginestar, "Peach Bottom-2 Low-Flow Stability Test using TRAC-BF1/VALKIN and RELAP5/PARCS Codes", *PHYSOR 2004*, Chicago, USA (2004).