Proceedings of the 10o Brazilian Congress of Thermal Sciences and Engineering -- ENCIT 2004
Braz. Soc. of Mechanical Sciences and Engineering -- ABCM, Rio de Janeiro, Brazil, Nov. 29 -- Dec. 03, 2004

CIT04-0553

THE RELEVANCE OF THE POWER-TO-FLOW MAP OF A BOILING WATER NUCLEAR REACTOR AS REFERENCE FOR STABILITY ASSESSMENT

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Abstract. *The purpose of this work is to address the issue about the relevance of establishing exclusion regions of operation in the power-to-flow map as the only reference for BWR (Boiling Water Nuclear Reactor) stability. The DYNOBOSS computer code has been used for the analysis. Several results are presented showing that the power-to-flow map does not provide sufficient information about reactor stability and may tend to misleading conclusions if not complemented by additional results.*

Keywords. *boiling water reactor, two-phase flow, BWR instability, reactor safety.*

1. Introduction

Two-phase flow induced instabilities constitute an important operational issue for boiling water reactor power plants. As several BWR instability events indicate (Waaranpera, 1981; Sandoz, 1983; Gialdi et al., 1985; AEOD, 1988), the various coupled neutronic and thermal-hydraulic phenomena governing BWR dynamics are not fully understood. A short-term solution, which has been adopted by the industry, is directed towards expanding exclusion regions in the reactor power-to-flow operating map.

This work aims to show that this solution has several disadvantages. First, it may impose unnecessary restrictions on the allowed reactor operating conditions. Secondly, reactor instabilities usually occur in rather special circumstance and during transients rather than normal steady-state operation. Such conditions cannot be identified using standard power-to-flow maps, since they may involve several additional parameters (a multi-dimensional matrix), which do not affect the reactor power or coolant flow rate.

The computer code DYNOBOSS (DYnamics of NOnlinear BOiling SystemS), which is a time-domain computer code developed for studying two-phase flow instabilities in parallel multi-channel boiling systems as well as in an entire steam supply system of a boiling water nuclear reactor (BWR) was used in the analysis.

First, a brief description of the DYNOBOSS code is presented in order to show the variety of two-phase flow modeling and numerical method options implemented in the code as well as to demonstrate its capability to produce accurate time-domain calculations for a boiling water reator operating at natural circulation mode. Then, a numerical analysis showing that the power-to-flow control map should not be used as a stand-alone reference to indicate regions of system instability is performed.

2. Overview of the DYNOBOSS Code

DYNOBOSS (DYnamics of NOnliner BOiling SystemS) is a computer code for the analysis of transient and instabilities in boiling systems in general and in boiling water nuclear reactors (BWR), in particular. The code has two modeling options: parallel boiling channels and boiling loop systems (Rosa and Podowski, 1997).

The BWR model in DYNOBOSS accounts for all major components of the BWR nuclear steam supply system as shown in Figs. (1) and (2). These components are the reactor core, upper plenum, steam riser, steam separator and steam dome in the boiling (two-phase) region of the loop and the downcomers and lower plenum in the nonboiling (single-phase) part of the loop. The reactor thermal-hydraulics is based on a one-dimensional (1-D) modeling framework. In this approach, both kinematics (phasic slip) and thermodynamic (subcooled boilng) nonequilibrium are accounted for using a four-equation model of two-phase flow. The most important and complex component of the loop is the reactor core whose overall model includes a thermal-hydraulic model for the water coolant, a thermal model for the fuel elements and a neutron kinetics model. All these models are coupled to each other. In the hydraulic description of the core, several parallel channels are defined, as shown in Fig. (2), each associated with a group of fuel assemblies in a radial region of the core such that the channel has the characteristics of the average assembly of the group.

The equations governing two-phase/single phase flows in the various components of the loop are:

Volumetric Flux Equation

$$\frac{\partial j}{\partial z} = \Gamma v_{fg} - v_g \alpha \frac{\partial \rho_g}{\partial t} \tag{1}$$

Void Propagation Equation

$$\frac{\partial \alpha}{\partial t} + \frac{\partial j_g}{\partial z} = v_g \left(\Gamma - \alpha \frac{\partial \rho_g}{\partial t}\right) \tag{2}$$

where Γ, the volumetric evaporation rate, is given by

$$\Gamma = \frac{q'}{A_{XS} h_{fg}} + \frac{[1 - \rho_f (1-\alpha) \frac{dh_f}{dp} - \rho_g \alpha \frac{dh_g}{dp}] \frac{dp}{dt}}{h_{fg}} \tag{3}$$

Mixture Energy Conservation Equation

$$\frac{\partial}{\partial t}[\rho_f h_l (1-\alpha) + \rho_g h_g \alpha] + \frac{\partial}{\partial z}[\rho_f j_l h_l + \rho_g j_g h_g] = \frac{q'}{A_{xs}} + \frac{\partial p}{\partial t}, \tag{4}$$

Mixture Momentum Conservation Equation

$$\frac{\partial G}{\partial t} + \frac{\partial}{\partial z}\left[\frac{\rho_f j_f^2}{1-\alpha} + \frac{\rho_g j_g^2}{\alpha}\right] + \frac{\partial p}{\partial z} + F_f + F_g = 0, \tag{5}$$

where G is the mass flux, q' is the linear heat rate and F_f and F_g are, respectively, the wall friction and gravity forces per unit area. All flow parameters are averaged over the flow cross sectional area, and the remaining notation in Eqs (1) – (5) is as in Lahey & Moody(1977).

The governing equations for single-phase liquid flow can be derived from the two-phase flow equations, Eqs. (1) - (5), by setting α, Γ and j_g to zero. In this case, Eqs. (1) and (2) reduce to the condition $j = j_l(z,t) = j_{in}(t)$. These equations can be used in the calculations for the single-phase region of the channels as well as for the single-phase part of the recirculation loop (lower and upper downcomers and lower plenum).

An important feature of the model is that it includes two options regarding the subcooled boiling phenomena: a profile-fit model (Levy,1966) and a mechanistic model (Lahey&Moody,1977)]. Another option is also available in which subcooled boiling is ignored. Similarly, the effects of phasic slip can be accounted for using different modeling assumptions, such as the EPRI correlation (Chexal and Lellouche,1985) or user specified drift-flux parameters, and either position-and-time-dependent or average parameters.

Table 1 summarizes the methods and approaches used in the modeling and respective numerical solution of the main components of a BWR loop implemented in the DYNOBOSS code.

The ability to predict the onset-of-instability conditions and the transient response of an unstable reactor can be used as a measure of accuracy and correctness of given mathematical and computational models. Therefore, in order to validate the code, the DYNOBOSS modeling calculations have been compared against the results of an exact analytical solution developed for NUFREQ-NP code (Peng at al.,1984). NUFREQ-NP is a BWR linear-stability-analysis code in the frequency domain. Because of that, the comparison was focused on the threshold of instability predicted by the codes. Figure (3) shows the marginal stability boundaries calculated with the DYNOBOSS and NUFREQ-NP codes for a typical boiling channel of a BWR core with several spacers along the channel. As can be seen in this figure, the marginal stability boundaries calculated by both codes compare remarkably well. Also, the DYNOBOSS code has been used successfully to reproduce a situation similar to the LaSalle instability event (Wulff et al.,1992). Whereas the comparison was mainly qualitative, the results clearly indicate that the onset of unstable oscillations, preceded by a quasi-stationary period, has been predicted accurately and, furthermore, that the reactor response properly reflects the superposition of self-sustained oscillations and the modulation due to feedwater flow rate fluctuations measured during the event.

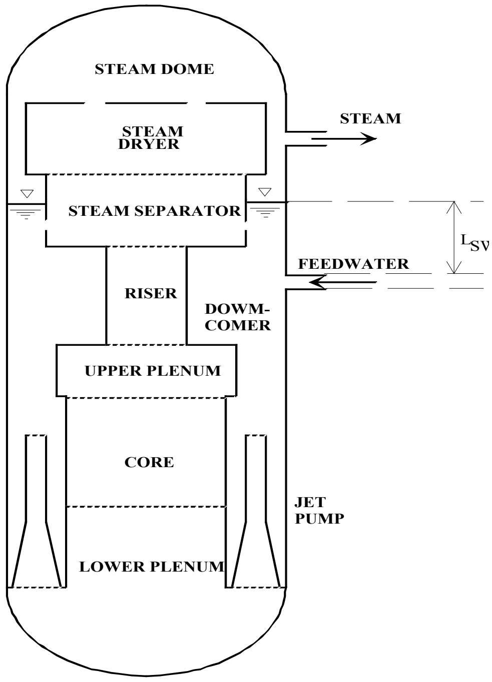

Figura 1. Schematic of a BWR Nuclear Steam Supply System.

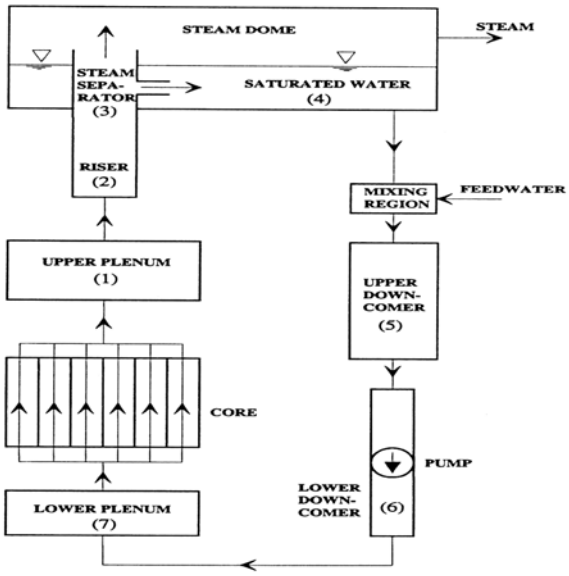

Figura 2. Reactor pressure vessel components and coolant flow diagram modeled in the DYNOBOSS code.

Table 1. Methods and Approaches

METHODS AND APPROACHES	COMPONENTS OF THE LOOP
Finite-difference method with two interpolation parameters	Reactor core, Steam riser and upper and lower downcomers
Lumped-parameter approach with implicit/semi-implicit methods for ODEs	Upper and lower plena, steam separator and upper downcomer
Method of characteristics (MOC)	Upper and lower downcomers

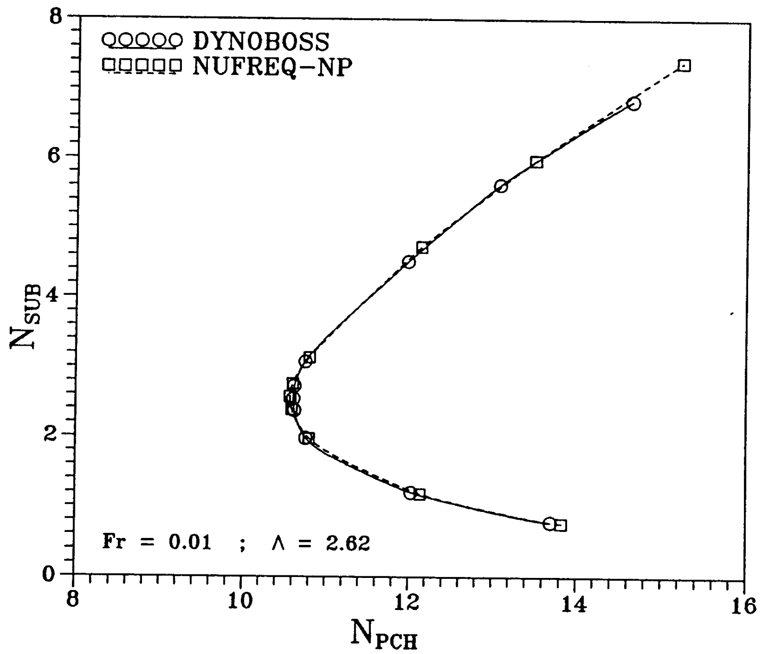

Figura 3. Comparison of the marginally stable boundaries calculated by the linear code NUFREQ-NP and the nonlinear DYNOBOSS, for a typical BWR boiling channel.

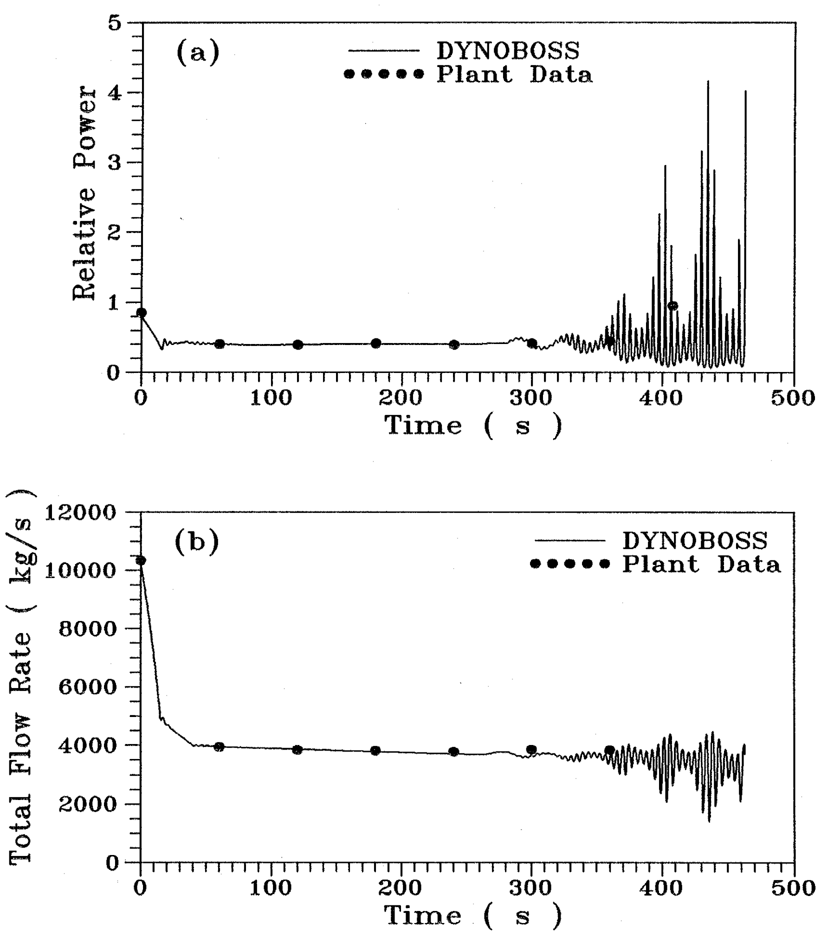

Figura 4. BWR response to inadvertent shutdown of both circulation pumps (LaSalle instability event).

3. Results of Analysis

As an example to show that the power-to-flow control map should not be used as a stand-alone reference to indicate regions of system instability the following calculations to quantify the effect of the feedwater parameters on the stability of the system operating at natural circulation have been performed. The feedwater temperature and steam flow rate depend on the power plant operating conditions including the turbine and feedwater line. Specifically, for a given thermal power and feedwater temperature versus steam flow rate curve shown in Fig. (5.a), the appropriate steady-state parameters were calculated. Now assuming that the feedwater temperature can deviate 20 ºC above and below the reference values, the steady-state parameters were calculated for the same reactor powers calculated for the reference feedwater curve. Fig. (5.b) shows the effect of feedwater temperature on the decay ratio for different power levels. The decay ratio is defined here as ratio between two consecutives oscillation amplitudes in time, therefore, if the decay ratio is smaller than unity the oscillation amplitudes are decreasing in time and the system is said to be stable, conversely, if the decay ratio is grater than unity the oscillation amplitudes are increasing in time and the system is unstable. As can be seen, lowering the feedwater temperature by 20 ºC increases the decay ratio by approximately 10%; conversely, increasing the feedwater temperature by 20 ºC the decay ratio decreases by approximately 7%. Since the power-to-flow control map is generally used to indicate possible regions of instability, these results were plotted in that map and it is shown in Fig. (6.a). It is interesting to notice that for a given power, different feedwater temperature has little effect on the location of the operating points along the natural circulation line. This is clearly observed in Fig. (6.b), which is a blow up of the appropriate section of the map in Fig. (6.a).

Also, the DYNOBOSS code has been used to calculate the natural circulation line in the power-to-flow map, as indicated in Fig. (**7**). Two models of two-phase flow in the core and riser have been used. One is the homogeneous equilibrium model (HEM) and the other is the slip equilibrium model (SEM). Figure (8) shows the stability boundaries in the $N_{SUB}\, x\, N_{PCH}$ plane of a boiling channel system calculated by the DYNOBOSS code, which simulates the two-phase flow experiment by Saha(1974), for different two-phase flow modeling assumptions. To the left to this boundary the system is stable and to the right the system is unstable. Figure (9) shows the time responses of the channel inlet mass flux to a small and short perturbation in the channel wall heat flux for three different operating conditions for the homogeneous equilibrium model (HEM) and the slip equilibrium model (SEM). As can be seen in Figures (8) and (9) the reactor stability characteristics obtained from these models are substantially different from each other, although the natural circulation line of the control map generated with these models are very close to each other as shown in Fig. (**7**).

From these results, it is obvious that the power-to-flow map does not provide sufficient information about reactor stability and may tend to misleading conclusions if not complemented by additional results.

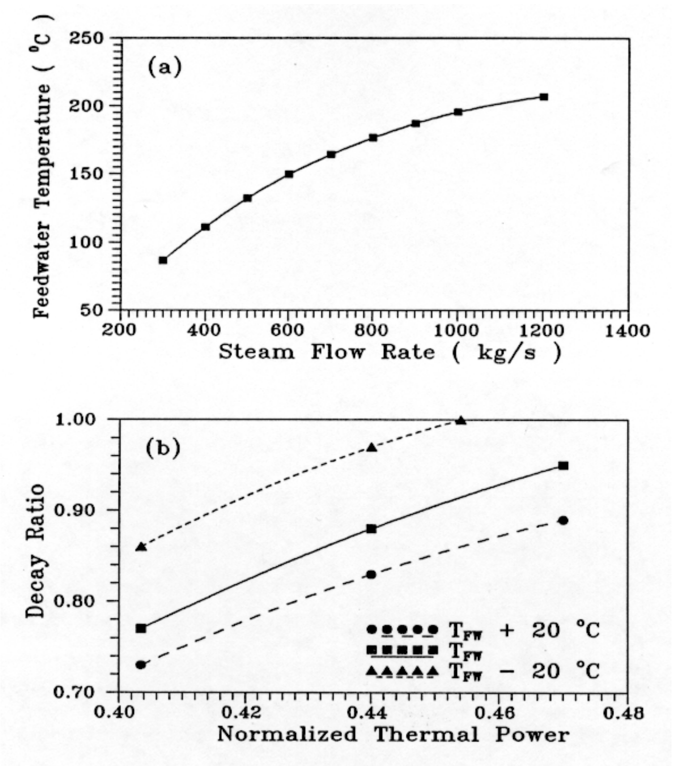

Figura 5. The effect of feedwater temperature on BWR stability: (a) the feedwater model (Gitnick eat al., 1992), (b) the sensitivity of decay ratio to changes in the feedwater temperature.

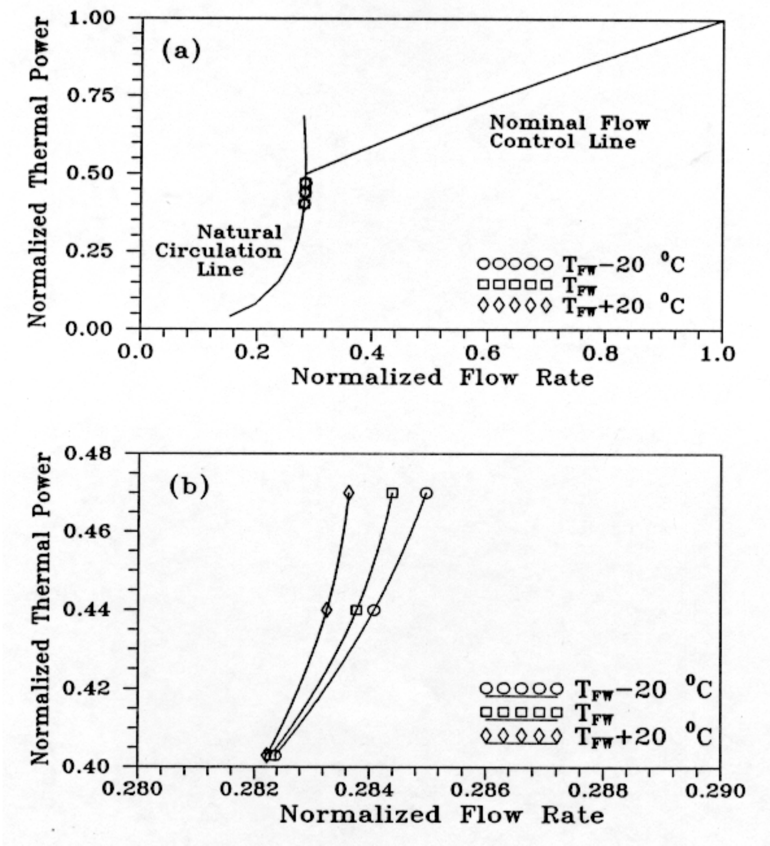

Figura 6. The BWR power-to-flow map; (a) the location of the points in Figure 5.b, (b) the blow up of a small region around the points in (a).

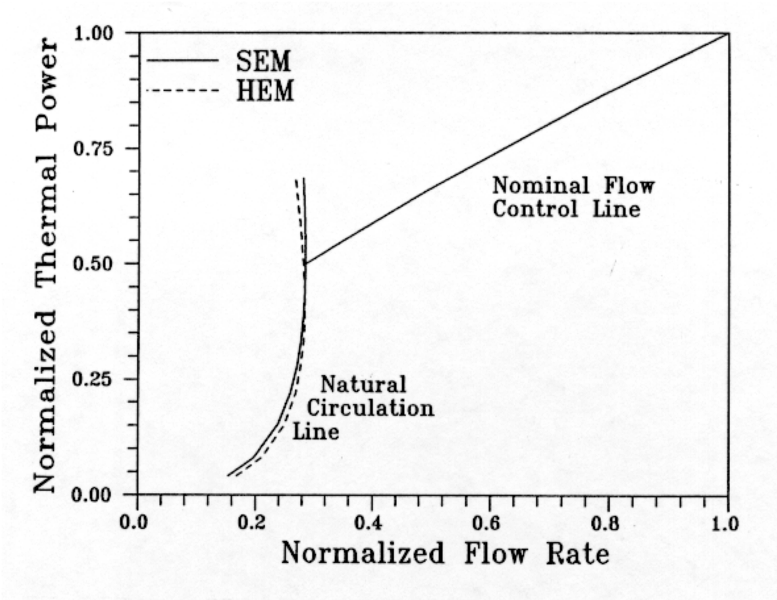

Figura **7**. Natural circulation lines for the homogeneous equilibrium model (HEM) and the slip equilibrium model (SEM), calculated by the DYNOBOSS code for a typical BWR-4.

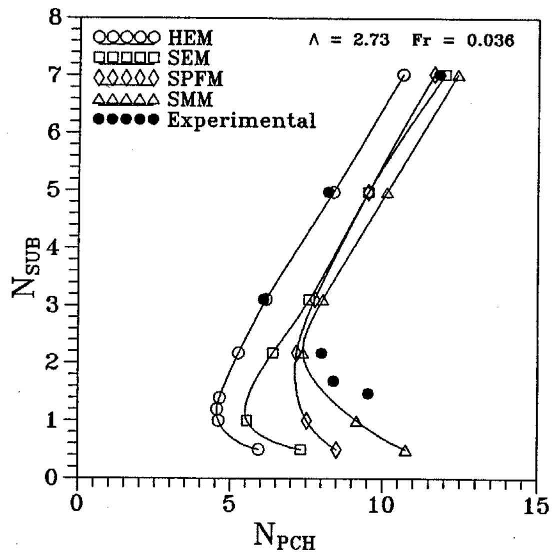

Figura 8. Comparison of the marginally stable boundaries calculated with the DYNOBOSS code using: the homogeneous equilibrium model (HEM); the phasic slip equilibrium model (SEM); the phasic slip profile-fit model (SPFM); the phasic slip mechanistic model (SMM). Also shown are the experimental results of Saha(1974).

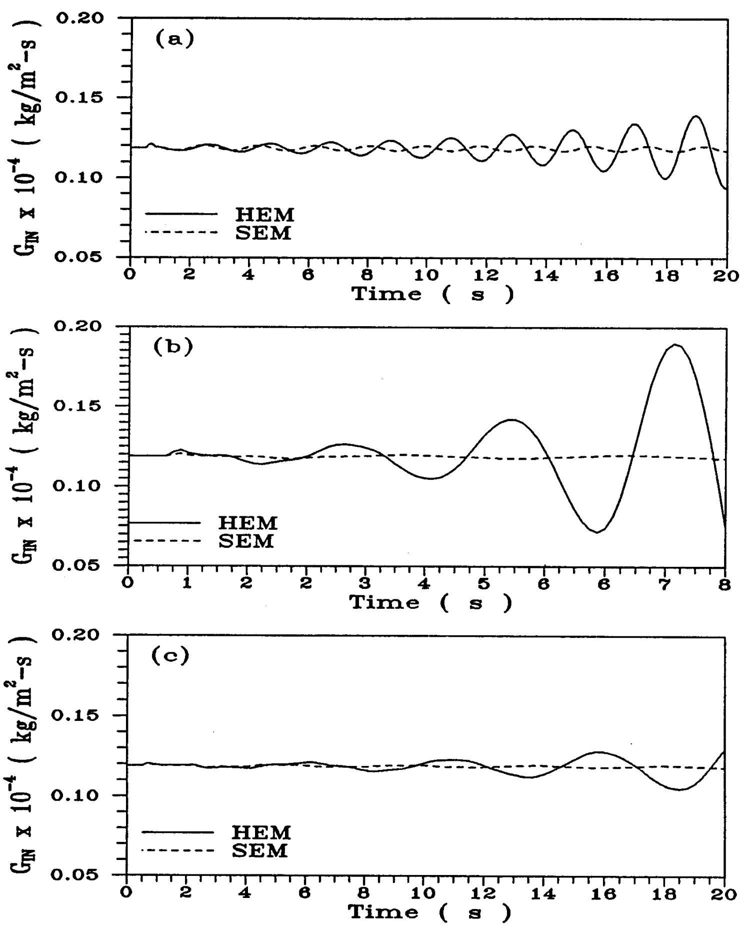

Figura 9. Illustration of the time responses of a boiling channel using the homogeneous (HEM) and the slip (SEM) equilibrium models for different operating conditions.

4. Conclusion

The analysis results clearly show that operating power-to-flow control map cannot be used as the only reference for establishing permissible operating conditions from the view point of BWR stability, since several important parameters, which affect the stability of the system, are not accounted for in this map. Furthermore, the approach of adopting exclusion regions in the power-to-flow control map has several disadvantages. First, it may impose unnecessary restrictions on the allowed reactor operating conditions. Secondly, reactor instabilities usually occur in rather special circumstance and during transients rather than normal steady-state operation. Such conditions cannot be identified using standard power-to-flow maps, since they may involve several additional parameters (a multi-dimensional matrix), which do not affect the reactor power or coolant flow rate.

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