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BENCHMARKING AND QUALIFICATION OF THE NUFREQ-NPW CODE FOR BEST ESTIMATE PREDICTION OF MULTI-CHANNEL CORE STABILITY MARGINS

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ABSTRACT

The NUFREQ-NPW[1,11] code was modified and set up at Westinghouse, USA for mixed fuel type multi-channel core-wide stability analysis. The resulting code, NUFREQ-NPW, allows for variable axial power profiles between channel groups and can handle mixed fuel types.

Various models incorporated into NURFREQ-NPW were systematically compared against the Westinghouse channel stability analysis code MAZDA-NF,[8] for which the mathematical model was developed, in an entirely different manner. Excellent agreement was obtained which verified the thermal-hydraulic modeling and coding aspects. Detailed comparisons were also performed against nuclear-coupled reactor core stability data. All thirteen Peach Bottom-2 EOC-2/3 low flow stability tests were simulated. A key aspect for code qualification involved the development of a physically based empirical algorithm to correct for the effect of core inlet flow development on subcooled boiling. Various other modeling assumptions were tested and sensitivity studies performed. Good agreement was obtained between NUFREQ-NPW predictions and data. Moreover, predictions were generally on the conservative side.

The results of detailed direct comparisons with experimental data using the NUFREQ-NPW code, have demonstrated that BWR core stability margins are conservatively predicted, and all data trends are captured with good accuracy. The methodology is thus suitable for BWR design and licensing purposes.

1.0 INTRODUCTION

Boiling flow instabilities must be considered in the design and analyses of many devices used in chemical processes and energy production, such as Boiling Water Nuclear Reactors (BWRs). As the power density and two-phase pressure drop of BWRs have increased, the possibility of thermal-hydraulic instabilities has been given increased attention both for design and licensing purposes. Such instabilities may cause divergent oscillations, boiling crisis, disturb the control system, and also cause mechanical damage to components from flow induced vibrations. The most important thermal-hydraulic instability in BWRs is nuclear-coupled density-wave oscillations.

An attempt to develop a detailed, mechanistic model for the analysis of nuclear-coupled density-wave oscillations in BWRs was made at Rensselaer Polytechnic Institute (RPI). This resulted in a large computer code called NUFREQ-N.[1]

One of the fundamental limitations of NUFREQ-N and other similar codes[2] is that these models assume constant system pressure. Consequently, such codes cannot be directly used to evaluate the system transfer function of a BWR excited by external system pressure perturbations. Since such techniques are particularly useful as experimental methods for investigating the stability of BWRs, as demonstrated by the stability tests performed on the Peach Bottom[3,4] reactor, it was deemed necessary that system pressure be used as an independent forcing variable in evaluating the transfer functions of BWR systems.

The NUFREQ-NP[5] code constitutes an extension of NUFREQ-N. It was developed by introducing system pressure as an independent variable in the equations. Modifications (at Westinghouse) have been introduced to include and model variable fuel assembly design dependent parameters (viz. fuel dynamics, spacer, and geometrical variations). This specific addition allows for the effective stability analysis of BWR cores with a variety of fuel designs, as would exist in a core with mixed fuel type. This version, which includes an empirical correction that accounts for the effect of inlet flow development on subcooled boiling, comprises the NUFREQ-NPW code, described in this paper.

The purpose of this paper is to present the benchmarking and qualification of the NUFREQ-NPW code for accurate prediction of multi-channel core stability margins in BWRs. A brief overview of the modeling process is also given for completeness. Thereafter details of the benchmarking and verification process, along with development of an algorithm, are presented.

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2.0 OVERVIEW OF MATHEMATICAL MODELING IN NUFREQ-NPW

2.1 System Modeling

The BWR system modeled for core stability analysis is shown schematically in Figure 1. As seen, the core is described as a collection of several parallel channels, each associated with one or more fuel assemblies. The flow field is divided into three regions: the single-phase region, the subcooled boiling region, and the bulk boiling region. In the subcooled boiling region a mechanistic thermal non-equilibrium model is used, whereas a thermal equilibrium model is applied in the bulk boiling region. A one-dimensional drift-flux model is used to represent two-phase flow dynamics. A single bypass channel is used to model the interstitial region between fuel assemblies.

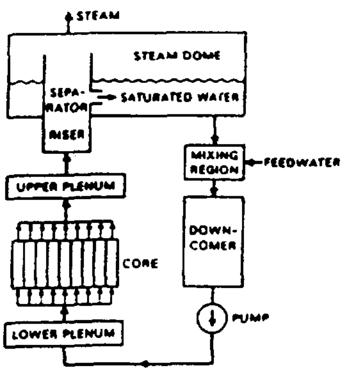

Figure 1. BWR loop model in NUFREQ-NP

In order to account for arbitrary non-uniform axial power, each channel is divided into a number of axial nodes with a uniform power profile in each node. Coolant thermal-hydraulics is coupled with fuel heat transfer, where a one-dimensional (radial direction) transient heat conduction equation is used.

The coolant recirculation loop is divided into several components shown in Figure 1. The NUFREQ-NPW code can accommodate various neutronic models, such as point kinetics, 1-D, 2-D, and 3-D kinetics. The overall BWR model, given in terms of non-linear differential equations, is perturbed and then Laplace transformed for frequency domain analysis. Thereafter, results in the frequency domain can be applied to satisfy current licensing requirements, via curve fitting of the predicted transfer function profile, in order to derive the so-called "decay ratio."

2.2 Transfer Function Evaluation

Various transfer functions that are necessary for stability evaluation are available in NUFREQ-NPW. Details of the derivations along with mathematical transformations are given elsewhere[5] and hence will not be repeated herein. Only the salient features relating to transfer functions used in this paper for benchmarking and verification purposes are presented for the sake of completeness.

Channel (hydraulic) stability characteristics are represented by (for a channel i, nodalized into K nodes, k = 1 ... K)

$$\delta\bar{\Delta p}_i = P_{1,i}\,\delta\bar{j}_{in,i} + \sum_{k=1}^{K} P_{2,i}^{k}\,\delta\bar{q}'''_{k,i} + P_{3,i}\,\delta\bar{w}_{core} + P_{4,i}\,\delta\bar{p} \qquad (1)$$

where, Δp is the channel pressure drop, j_{in} is the channel inlet velocity, q''' is the volumetric internal heat generation rate, w_{core} is the total core flow, and P is the complex argument transfer function for which the form and derivations are given elsewhere.[5]

The coupling of equations for void-fraction perturbation with the neutron kinetics equations results in the system-pressure-to-power density (or neutron flux) transfer function (Qp)

$$\delta\bar{q}''' = Qp\ \delta\bar{p}\ . \qquad (2)$$

Equation (2) was used to compare with the pressure perturbation low flow stability data taken at the Peach Bottom-2 BWR/4 reactor.[6,7]

3.0 CODE BENCHMARKING

3.1 General

This section documents code verification and benchmarking against code calculations and experimental stability data. First, a systematic comparison against MAZDA-NF code predictions was made to test the various models used (e.g., slip flow, heater dynamics local losses in non-boiling and boiling regions, etc.). The MAZDA-NF nodal code[8,10] was developed at RPI, and benchmarked against experiments for multi-dimensional hydraulic stability analysis, in axially cross-connected channels undergoing density-wave oscillations. Such a code allows for examination of cross-flows on system hydraulic stability characteristics, without reactivity feedback. NUFREQ-NP on the other hand was developed for assessing nuclear-coupled core stability behavior. Due to the nature of the problem analyzed (i.e., cross-flows), the mathematical model in MAZDA-NF consists of discretized conservation equation sets, for each node, which were cast in matrix form. Thereafter, the characteristic equation was obtained by using a matrix reduction scheme for analyzing system dynamics. This is in sharp contrast to the process followed in NUFREQ-NP which consists of exact analytical solutions.[5] As such, the MAZDA-NF model was developed in an entirely different manner. The one dimensional option of MAZDA-NF thus constitutes a valuable verification tool for the modelling and coding aspects in NUFREQ-NPW. Detailed comparisons were also made against nuclear-coupled reactor core stability data, in which an algorithm to account for inlet flow development on subcooled boiling was developed for NUFREQ-NPW. These various comparisons along with sensitivity studies are described sequentially.

3.2 [illegible]

[illegible] various hydraulic models incorporated in NUFREQ-NPW in the single- and two-phase flow regions, respectively. Code assessment was done using operating parameters similar to a BWR/4 operating close to natural circulation. Channel geometry was that of a GE-8 × 8 fuel assembly. NUFREQ-NPW hydraulic transfer function values given by Eq. (1) with $\delta p = \delta q''' = 0$, were systematically compared directly against corresponding MAZDA-NF code predictions.

Typical transfer function calculations are shown in Figures 2 and 3 respectively. Figure 2 shows the base case (curve 1) as predicted for a boiling channel without local losses, slip flow, and heater dynamics under nominal conditions of low pressure and inlet subcooling. Results of varying power, flow and pressure are also shown therein (curves 2,3 and 4). The effects of slip flow conditions and skewed axial power shape are shown (curves 2 and 3) in Figure 3. Similar comparisons were also made to note variations with single-phase and two-phase orificing, inlet subcooling, and also the effect of heater dynamics. Excellent agreement was obtained between the predictions of MAZDA-NF and NUFREQ-NP codes. This tends to verify the modelling [illegible] in NUFREQ-NP. [illegible] results [illegible] are the same.

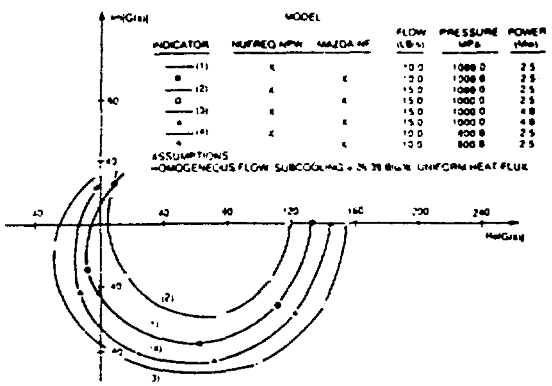

Figure 2. NUFREQ-NP verification runs (effect of flow, power and pressure)

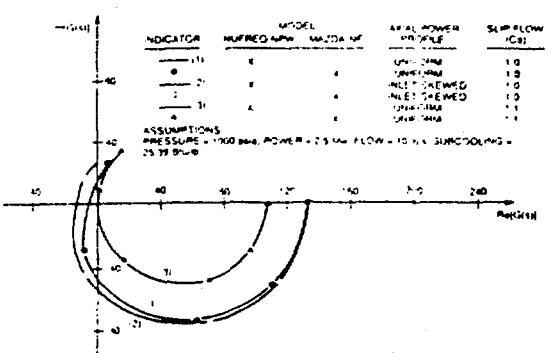

Figure 3. NUFREQ-NP verification runs (effect of slip and power shape)

3.3 Comparison with Peach Bottom [illegible]

3.3.1 General. Low flow stability tests were performed at the Peach Bottom-2 BWR/4 nuclear power plant prior to shutdown at the end of cycle-2 (3), and cycle-3 (4). Four tests were conducted at the end of cycle-2, whereas nine tests were performed at the end of cycle-3. The purpose of low flow stability testing was to determine the stability margin of the BWR core at an operating condition above the rated power-flow control line. These tests were also intended to demonstrate the practicality of using pseudo-random pressure perturbation tests to determine the stability margin of a large BWR core. These tests showed that small pressure perturbation testing offers a useful and practical method of measuring core stability margins, as was determined from the measured closed-loop pressure-to-average neutron flux transfer function data.

Detailed NUFREQ-NPW predictions were made for the conditions of all thirteen stability tests mentioned above, along with a sensitivity study with key parameters. The results of comparisons are shown first for tests at the end of cycle-2, and thereafter for those at the end of cycle-3.

3.3.2 Evaluation of Void-Reactivity Coefficient and Gap Conductance for Each Test. Best estimate values for void reactivity coefficients and rod-to-clad gap conductances for each test were obtained using Westinghouse design codes. Fuel resident in the core was segmented into various fuel groups, and the axially averaged gap conductance for each fuel type used was determined as a function of rod power. The gap conductance values used by NUFREQ-NPW were obtained by a weighted arithmetic mean for the GE-7 × 7 and GE-8 × 8 fuel type gap conductances, respectively. These values were used for the multi-channel simulations for stability analyses. For single (i.e., core average) channel stability analyses, the arithmetic mean of values evaluated for the GE-7 × 7 and GE-8 × 8 fuel types was taken.

3.3.3 Comparison Against Peach Bottom EOC-2 Stability Tests. As mentioned previously four tests were conducted at the end of cycle-2. These were referred to as PT1, PT2, PT3 and PT4, respectively. Design and operating data used for modeling the system were taken from references 6 and 7. For core-averaged analyses, a GE-7 × 7 fuel type was chosen because more than 75% of the resident assemblies were of this type.

Comparison of NUFREQ-NPW runs against the measured transfer function data ($\delta\gamma/\delta p$ where γ represents normalized core power density i.e., $\delta\gamma = \delta q'''/q'''$) in terms of magnitude and phase angle vs. frequency are shown in Figures 4 to 7, respectively (Cases 1 and 2). As noted therein, the results with inclusion of the subcooled boiling model (Case 1) vastly overestimate the data, especially the peak magnitude in $\delta\gamma/\delta p$. A parametric study was also conducted to adjust the location of net boiling incipience by 50% from that given given by the Saha-Zuber

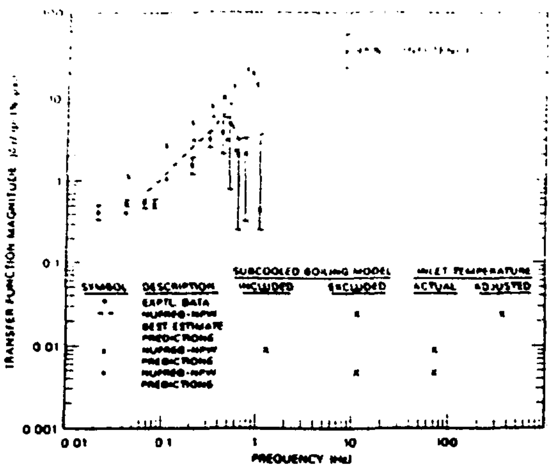

Figure 4a. NUFREQ-NPW predictions for Peach Bottom test PT1 (gain)

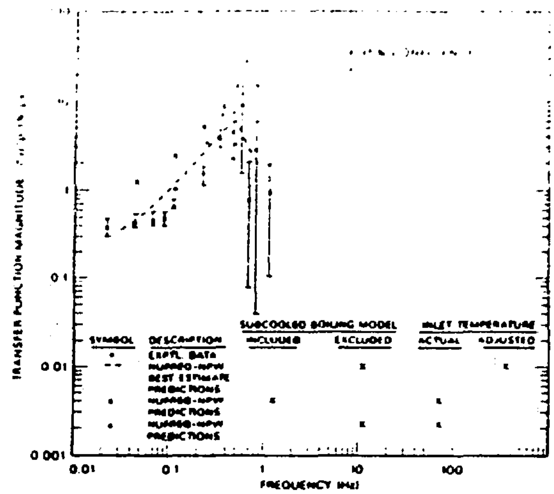

Figure 5a. NUFREQ-NPW predictions for Peach Bottom test PT2 (gain)

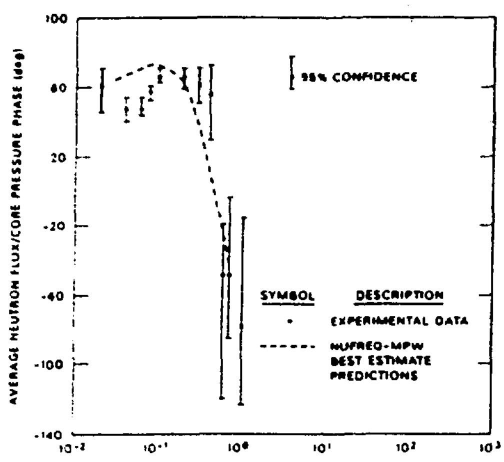

Figure 4b. NUFREQ-NPW predictions for Peach Bottom test PT1 (phase)

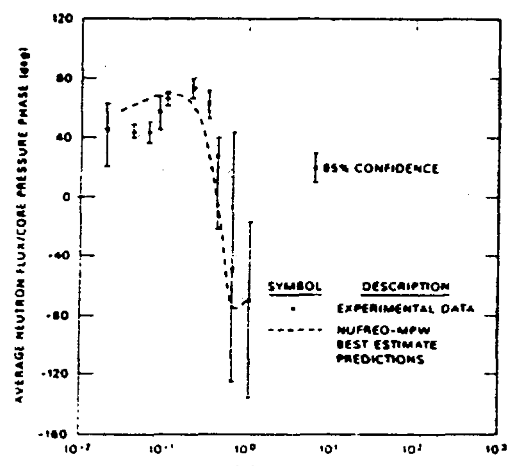

Figure 5b. NUFREQ-NPW predictions for Peach Bottom test PT2 (phase)

correlation. The results obtained with subcooled boiling still deviated significantly from the data. The results obtained by excluding the subcooled boiling model (Case 2) are significantly better in terms of matching the peak magnitude.

However, the resonant frequency was underestimated for PT1, PT2 and PT4 test cases where the inlet subcooling was considerably larger than for test case PT3. Evidently, neglecting the effect of subcooled boiling leads to a longer non-boiling length, and shifts the predicted resonant frequency to a lower value. Clearly, neither the use of the subcooled boiling model nor the neglect of this model was ideal. This is likely due to the fact that models and correlations (e.g., Saha-Zuber) which are valid only for fully developed flows were used in these evaluations. However, due to significant orificing, the flow at the inlet of a BWR fuel assembly experiences intense cross-flows and turbulent mixing. It was apparent that accounting for a shorter non-boiling length due to the phenomenon of subcooled boiling was essential for accurately predicting nuclear-coupled core stability behavior. To a good first approximation this can be physically visualized as the coolant having a reduced inlet subcooling, which causes it to boil before the bulk coolant becomes saturated. Indeed, such an occurrence is predicted by the well known Saha-Zuber correlation represented as,

$$h_f - h_d = f(q'',G) \qquad (3)$$

where,

q'' is the heat flux, G is the mass flux, h_f is the saturation enthalpy, h_d is the departure enthalpy, and the difference $(h_f - h_d)$ can be

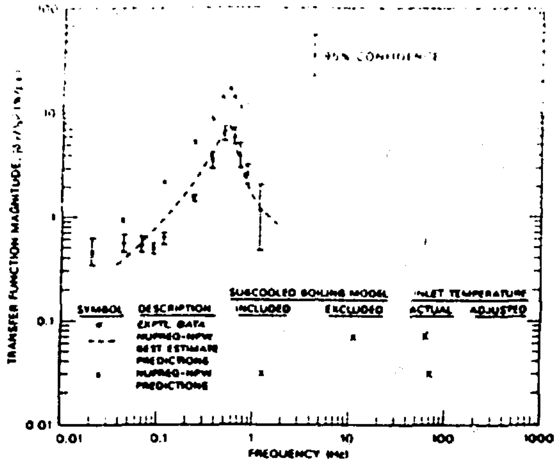

Figure 6a. NUFREQ-NPW predictions for Peach Bottom test PT3 (gain)

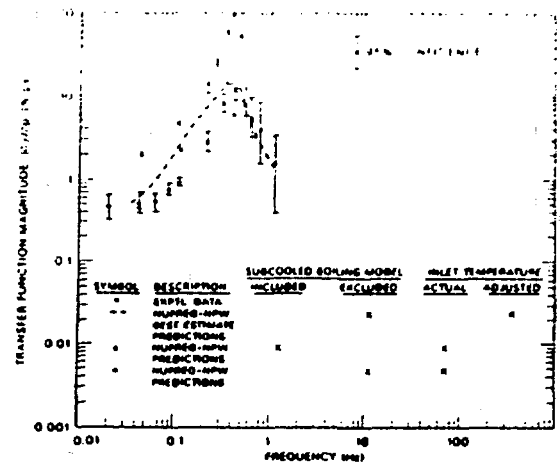

Figure 7a. NUFREQ-NPW predictions for Peach Bottom test PT4 (gain)

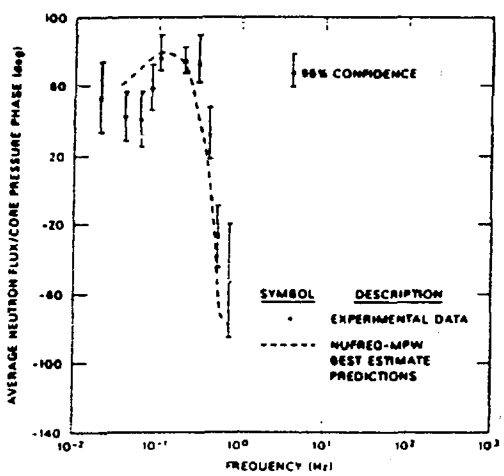

Figure 6b. NUFREQ-NPW predictions for Peach Bottom test PT3 (phase)

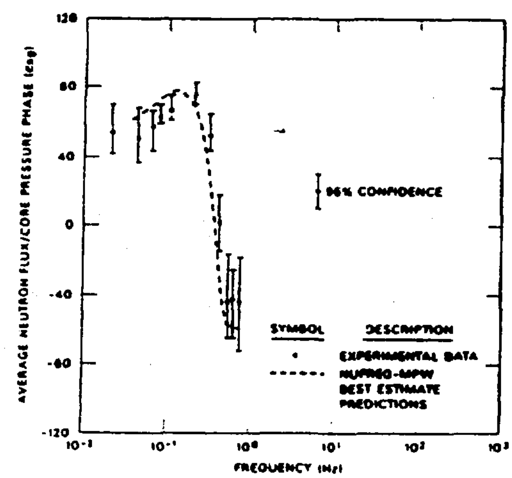

Figure 7b. NUFREQ-NPW predictions for Peach Bottom test PT4 (phase)

viewed as being a virtual reduction in the inlet subcooling which causes (subcooled) void formation at a bulk coolant enthalpy of h_d instead of h_f.

A physically based algorithm was thus developed to adjust the inlet subcooling by a specified amount, dependent on the operating conditions. This algorithm is based on the postulate that due to flow development subcooled boiling will not be initiated for the first six inches from the core entrance. For the fuel assemblies in the Peach Bottom-2 reactor, this amounts to about 10 L/D for sufficient entrance effect decay before allowing for the inception of significant subcooled voids.

Based upon the above postulate for stability conditions, the actual inlet subcooling, ΔT_{sub}, is reduced by an appropriate amount, which leads to an adjusted inlet subcooling, ΔT_{adj} at which bulk boiling begins, given by,

$$\Delta T_{adj} = \mathrm{Max}\,(\Delta T_{sub,m},\ T_{in} - \Delta T_r)\ , \tag{4}$$

where

$$\Delta T_{sub,m} - \Delta h_{sub,m} = \frac{0.5\ q''\ P_H}{w_T}\ , \tag{5}$$

$$\Delta T_r - h_f = h_d = 154\ \bar{q}''/G \tag{6}$$

where

$\bar{q}''$ is the average heat flux, p_H is the heated perimeter, w_T is the total flow rate, h_f is the saturated enthalpy, h_d is the departure enthalpy [illegible] G is the mass flux. $\Delta T_{sub,n}$ and ΔT_i are inlet subcooling adjustments/reductions given by Eqns. (4) and (5) that account for void formation ahead of bulk fluid saturation.

While the use of a six inch "dead zone" at the inlet is somewhat arbitrary, it produces conservative predictions of stability margins and modifies the results in a physically realistic manner. That is, it approximates the developing nature of the flow at the inlet of heated channels and its effect on initiating subcooled boiling. It should be noted that while using the algorithm described by Eq. (4), the subcooled boiling modelling option in NUFREQ-NPW is turned off. Thus, the effect of subcooled boiling is approximated by Eq. (4). The results of NUFREQ-NPW calculations (best estimate) with adjusted inlet temperatures (case 3) are shown alongside the previous two results in Figures 4 to 7. It should be noted that based upon the algorithm of Eq. (4) no inlet temperature correction was necessary for test case PT3 (Figure 6). Hence the absence of triangles.

As noted in Figures 4 to 7, use of the algorithm [Eq. (4)] for case 3 captures both the peak magnitude, as well as the resonant frequency, with good accuracy. Excellent agreement is seen in the important region around resonance. It should be noted that tests PT1 to PT4 were conducted over a fairly wide range of operating conditions. Note also that a log-log scale tends to accentuate small differences at low frequencies. However, the results are generally conservative.

The relative effect of using a single averaged channel to represent the core versus modeling the core as consisting of two different fuel types (GE-7 × 7 and GE-8 × 8) was also studied. The two approaches gave results that were close and in good agreement with one another, indicating the suitability of the gap conductance averaging process.

3.3.4 Comparison against Peach Bottom EOC-3 Stability Tests. In addition to the four EOC-2 stability tests, nine additional tests were conducted on the Peach Bottom-2 reactor at the EOC-3. Details are given in references 4 and 7 respectively. The EOC-3 tests generally showed higher decay ratios, and, significantly, the inlet subcooling was about twice as large than for EOC-2 tests. Hence, the EOC-3 stability tests proved a valuable data set for assessing the subcooled boiling correction developed for core stability analysis during benchmarking of the EOC-2 tests in Sec. 3.3.3.

Typical results of NUFREQ-NPW best estimate predictions against transfer function data are shown in Figures 8 to 11 respectively. As noted therein, good agreement is observed between predictions and data. Once again, NUFREQ-NPW predicts fluctuations of the pressure (δp) to normalized power density ($\delta\gamma = \delta q'''/q'''$) (i.e., $\delta\gamma/\delta p$) well and generally in the conservative direction. The resonant frequency is also well predicted, although it is a bit lower than the experimental value.

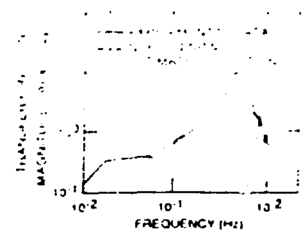

Figure 8a. NUFREQ-NPW predictions for Peach Bottom test 2PT3 (gain)

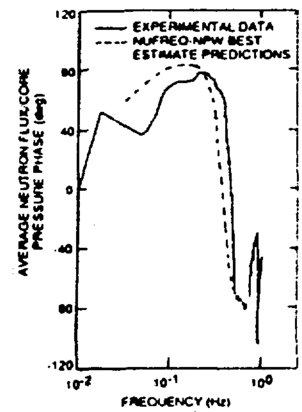

Figure 8b. NUFREQ-NPW predictions for Peach Bottom test 2PT3 (phase)

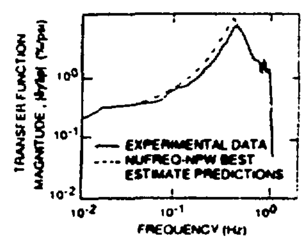

Figure 9a. NUFREQ-NPW predictions for Peach Bottom test 3PT2 (gain)

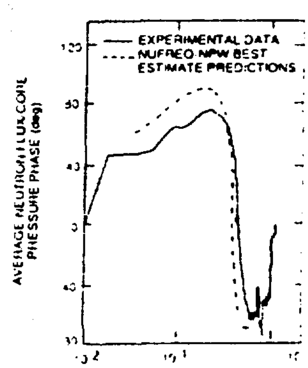

Figure 9b. NUFREQ-NPW predictions for Peach Bottom test 3PT2 (phase)

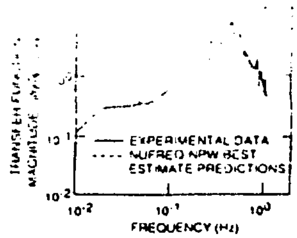

Figure 10a. NUFREQ-NPW predictions for Peach Bottom test 3PT3 (gain)

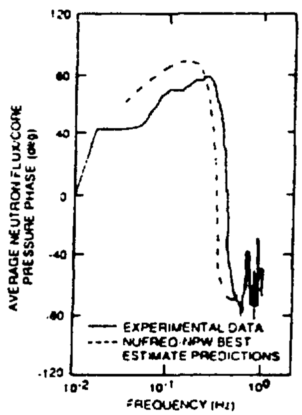

Figure 10b. NUFREQ-NPW predictions for Peach Bottom test 3PT3 (phase)

These detailed and "direct" comparisons with the Peach Bottom data strengthen and further verify the approach developed for evaluating core stability using NUFREQ-NPW.

3.3.5 Comparison of Predicted and Experimental Decay Ratios. Decay ratios predicted by NUFREQ-NPW using two approaches (i.e., "1 zero, 2 poles" fitting; "optimized" fitting) are displayed against experimental decay ratios in Table 1. As noted therein, the predicted decay ratios are in excellent agreement with measured

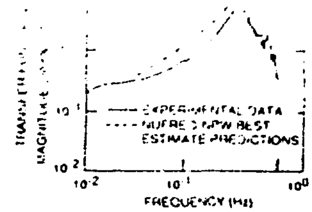

Figure 11a. NUFREQ-NPW predictions for Peach Bottom test 4PT1 (gain)

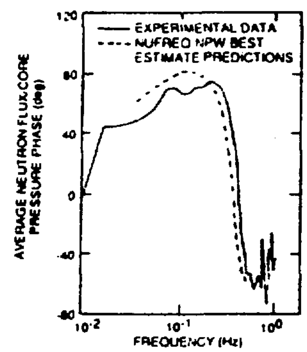

Figure 11b. NUFREQ-NPW predictions for Peach Bottom test 4PT1 (phase)

values. The spread between the "1 zero, 2 poles" and "optimized" fit decay ratio is to be expected, but should get close to zero as the predicted decay ratio approaches unity (i.e., near the instability threshhold).

3.3.6 Variation of Decay Ratio with Void Reactivity Coefficient (C_α) and Gap Conductance (Hg). Several computer runs were performed to note the sensitivity of the predicted decay ratio to C_α and Hg for each of the thirteen stability tests. Sample results for test case PT3 (EOC-2) are displayed in Figure 12. As noted, the predicted decay ratio is quite sensitive to these two parameters. From the studies conducted, it was found that a 10% change in

Table 1a. Comparison of best estimate decay ratio and natural frequency with experimental values for Peach Bottom EOC-2 stability tests

Test	Decay ratio: Prediction: 1 zero, 2 pole	Decay ratio: Prediction: Optimized	Decay ratio: Experiment: 1 zero, 2 pole	Decay ratio: Experiment: 1 zero, 3 pole	Decay ratio: Experiment: Coherence	Resonance frequency (radians/sec): Prediction	Resonance frequency (radians/sec): Experiment
PT1	0.097	0.075	0.1206	0.134 ± 0.046	0.2539	2.68	2.76
PT2	0.201	0.167	0.1206	0.168 ± 0.136	0.303	2.4	2.8
PT3	0.367	0.284	0.344	0.34 ± 0.108	0.331	2.67	2.74
PT4	0.271	0.326	0.295	0.263 ± 0.09	0.271	2.2	2.4

[illegible]

Test	Decay Ratio			Resonance Frequency (radians/sec)	
	Prediction		Experiment		
	1 zero, 2 pole	Optimized	Coherence Weighted	Prediction	Experiment
1PT1	0.264	0.322	0.236	2.24	2.76
1PT2	0.285	0.333	0.314	2.20	2.58
2PT2	0.483	0.497	0.435	2.30	2.73
2PT3	0.492	0.456	0.509	2.40	2.71
3PT3	0.45	0.494	0.391	2.16	2.57
4PT1	0.272	0.326	0.355	2.05	2.5
4PT2	0.22	0.296	0.213	2.0	2.4
4PT3	0.145	0.22	0.210	2.0	2.4

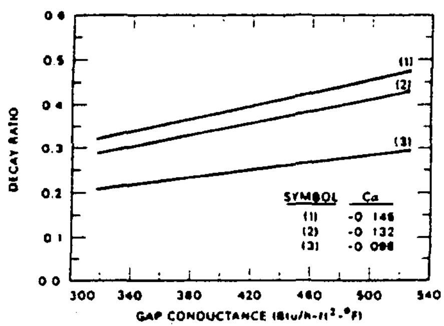

Figure 12. Decay ratio variation with Cα and Hg; Peach Bottom-2 EOC-2 Test PT3

C_α causes roughly the same change in predicted decay ratios as a 25% change in Hg.

4.0 SUMMARY AND CONCLUSIONS

This paper has focused on a methodology for core stability analysis of BWRs using the NUFREQ-NPW code and the benchmarking and verification of the code. All major BWR components for stability analysis are represented. Multi-channel simulation of the core is possible. One of the key distinguishing features of NUFREQ-NPW is its ability to model system pressure perturbation as an external forcing function, which allows for direct comparison against pressure perturbation test data instead of inferring equivalent information via curve fitting of other transfer functions.

The solutions of NUFREQ-NPW were systematically compared against MAZDA-NF code predictions in order to check the various models and coding. Excellent agreement was obtained in all cases.

The NUFREQ-NPW code was extensively assessed against the Peach Bottom-2, BWR/4 stability data at the end of cycles 2 and 3. All thirteen tests were simulated and compared against. A physically motivated empirical algorithm was developed to correct for the effect of flow development on subcooled boiling. [illegible] variations, with excellent [illegible]. Good agreement was also obtained between the experimental and predicted decay ratios. These predictions were generally conservative.

The results of comparisons with experimental data using the NUFREQ-NPW code, demonstrate that BWR core stability margins are predicted with good accuracy. Thus it appears that this methodology is suitable for design and licensing applications.

5.0 REFERENCES

1. G. C. PARK, M. Z. PODOWSKI, M. BECKER, and R. T. LAHEY, JR., "The Development of NUFREQ-N: An Analytical Model for the Stability Analysis of Nuclear Coupled Density-Wave Oscillations in Boiling Water Nuclear Reactors," *NUREG/CR-3375* (1983).

2. "Stability and Dynamic Performance of the General Electric Boiling Water Reactor," *NEDO-21506* (1977).

3. L. A. CARMICHAEL, and R. O. NEIMI, "Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2," *EPRI NP-564, Project 1020-1*, (1978).

4. F. B. WOFFINDEN, and R. O. NEIMI, "Low-Flow Stability Tests at Peach Bottom Atomic Power Station Unit 2 During Cycle 3," *EPRI NP-0972, Project 1020-2*, (1981).

5. S. J. PENG, M. Z. PODOWSKI, and R. T. LAHEY, JR., "NUFREQ-NP: A Digital Computer Code for the Linear Stability Analysis of Boiling Water Nuclear Reactors," *NUREG/CR-4116*, (1984).

6. N. H. LARSEN, "Core Design and Operating Data for Cycles 1 and 2 of Peach Bottom 2," *EPRI NP-563, Project 1020-1*, (1978).

7. G. L. HOLLOWAY, and L. M. SHIRAISHI, "Core Design and Operating Data for Cycle 3 of Peach Bottom 2," *EPRI NP-0971, Project 1020-2* (1981).

8. R. TALEYARKHAN, M. Z. PODOWSKI, and R. T. LAHEY, JR., "Ventilated Channel Instability Analysis," *Journal of Heat Transfer*, Vol. 107, 175-181, February 1985.

9. P. SAHA, and N. ZUBER, "Point of Net Vapor Generation and Vapor Void Fraction in Subcooled Boiling," *Proc. Fifth Int. Heat Transfer Conf.*, Vol. IV (1974).

10. R. TALEYARKHAN, M. Z. PODOWSKI, and R. T. LAHEY, Jr., "An Analysis of Density-Wave Oscillations in Ventilated Boiling Channels," *NUREG/CR-2972*, 1983.

11. S. J. PENG, M. Z. PODOWSKI, R. T. LAHEY, Jr., and M. BECKER, "NUFREQ-NP: A Computer Code for the Stability Analysis of Boiling Water Nuclear Reactors," *Nucl. Sci. Engr.*, 88, 404-411 (1988).