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.

One of the fundamental limitations of NUFREQ-N and other similar codes 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 stability tests performed on the Peach Bottom 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 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 a 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 in subcooled boiling, comprises the NUFREQ-NP code, described in this paper.

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

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The BWR system modelled 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 NUFRQ-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 NUFRQ-NP 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 NUFRQ-NP. Details of the derivations along with mathematical transformations are given elsewhere and hence will not be repeated here-in. Only the salient features relating to transfer functions used in this paper for bench-

\[ \Delta P = P_{1,1} \Delta \xi_{1,1} + \frac{1}{2} P_{2,1} \Delta \xi_{2,1} + \frac{1}{4} P_{3,1} \Delta \xi_{3,1} \]

where, \( \Delta P \) is the channel pressure drop, \( j_{in} \) is the channel inlet velocity, \( q^{\prime\prime\prime} \) is the volumetric internal heat generation rate, \( v_{core} \) is the total core flow, and \( P \) is the complex argument transfer function for which the form and derivations are given elsewhere.

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 (gp)

\[ \Delta \xi'''' = gp \Delta P \]

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

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, as developed at RPI, and benchmarked against experiments for multidimensional 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. NUFRQ-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 NUFRQ-NP which consists of exact analytical solutions. 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 NUFRQ-NP. 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 NUFRQ-NP. These various comparisons along with sensitivity studies are described sequentially.
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 NUWEQ-NP codes. This tends to verify the modelling.
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:

$$h_f - h_d = f(q'', C)$$

where,

$q''$ is the heat flux, $C$ 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
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, \( \Delta T_{\text{sub}} \), is reduced by an appropriate amount, which leads to an adjusted inlet subcooling, \( \Delta T_{\text{adj}} \) at which bulk boiling begins, given by,

\[
\Delta T_{\text{adj}} = \text{Max} (\Delta T_{\text{sub,m}}, T_{\text{in}} - \Delta T_r).
\]

where

\[
\Delta T_{\text{sub,m}} - \Delta h_{\text{sub,m}} = \frac{0.5 q''}{T_r}.
\]

\[
\Delta T_r - h_f h_d = 154 \text{ q''/C}
\]

where

\( q'' \) is the average heat flux, \( p_h \) is the heated
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 modeling 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 (CE-7, 7 and CE-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 as the EOC-3. Details are given in references 4 and 5, 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 ($p$) to normalized power density ($\dot{q}$) to normalized power density ($\dot{q}$) 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 10a. MUFREQ-NPU predictions for Peach Bottom test 3PT3 (gain)

Figure 10b. MUFREQ-NPU predictions for Peach Bottom test 3PT3 (phase)

Figure 11a. MUFREQ-NPU predictions for Peach Bottom test 4PT1 (gain)

Figure 11b. MUFREQ-NPU predictions for Peach Bottom test 4PT1 (phase)

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

3.3.5 Comparison of Predicted and Experimental Decay Ratios. Decay ratios predicted by MUFREQ-NPU 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 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 threshold).

3.3.6 Variation of Decay Ratio with Void Reactivity Coefficient (Cv) and Gap Conductance (Hg). Several computer runs were performed to note the sensitivity of the predicted decay ratio to Cv 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

<table>
<thead>
<tr>
<th>Test</th>
<th>Prediction</th>
<th>Experiment</th>
<th>Resonance Frequency (radians/sec)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1 zero, 2 pole</td>
<td>Optimized</td>
<td>1 zero, 1 pole</td>
</tr>
<tr>
<td>PT1</td>
<td>0.097</td>
<td>0.075</td>
<td>0.126</td>
</tr>
<tr>
<td>PT2</td>
<td>0.201</td>
<td>0.167</td>
<td>0.126</td>
</tr>
<tr>
<td>PT3</td>
<td>0.367</td>
<td>0.234</td>
<td>0.344</td>
</tr>
<tr>
<td>PT4</td>
<td>0.271</td>
<td>0.326</td>
<td>0.295</td>
</tr>
</tbody>
</table>
The results of comparisons with experimental data using the NUFREQ-NPW code, demonstrated 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