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Review Article

BWR Stability Issues in Japan

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The present paper reviews activates relevant to the boiling water reactor (BWR) stability phenomenon, which has a coupled neutronic and thermal-hydraulic nature, from the viewpoint of model and code developments and their applications to the BWR stability solution methodology in Japan.

1. INTRODUCTION

The core power oscillation phenomenon inherently exists in BWR cores [1], as generally called by the BWR stability or instability. The BWR instability is possible even at the normal plant operation conditions, and significant core power oscillations may threaten core fuel integrity due to the fuel cladding dryout occurrence and/or due to the strong PCMI (pellet-cladding mechanical interaction). Therefore, an accurate prediction for the onset of BWR instability is indispensable for the safety of BWR core design and operation. Hence, numerous efforts have been paid to understand the complicated BWR instability mechanism and to develop the advanced analysis models.

The stability problem has become an important concern on safety of BWR operations, in particular, after the instability incident at LaSalle-2. It should be emphasized that the applied analysis code predicted a stable core condition while instability actually occurred. Therefore, GE and US BWROG (BWR Owners' Group) have improved the stability analysis models which can be adequately applicable to the actual core design and operation, and have developed the long-term stability solution methodologies with several modifications in the plant installation.

Also in Japan, similar activities have been proceeded by the BWR plant/fuel vendors and utilities to exclude any instability concern. Main goals in Japanese activities are as follows: (1) to analytically investigate the complicated BWR instability mechanism, the power oscillation onset/growth, and formation of the limit cycle oscillation, by using the three-dimensional time-domain code; (2) to empirically define the stability performance of the employed fuel design, and to assess the accuracy of calculation results by stability analysis codes using the experimental data, and (3) to establish the stability solution methodology, in which the selected control rod insertion (SRI) system is installed to automatically exclude the operated core from possibly unstable core condition.

The present paper describes the BWR stability issues in Japan. Researches related to the phenomena identification, models, and codes applicable to the design analysis and stability solution methodologies are described. Authors suppose that understanding the basis of the BWR stability issues can be useful for future improvements in the BWR stability solution methodology based on the advanced analysis models and codes. In the last section of the present paper, an outline of the on-going research on the advanced BWR stability solution methodology is to be introduced, which employs

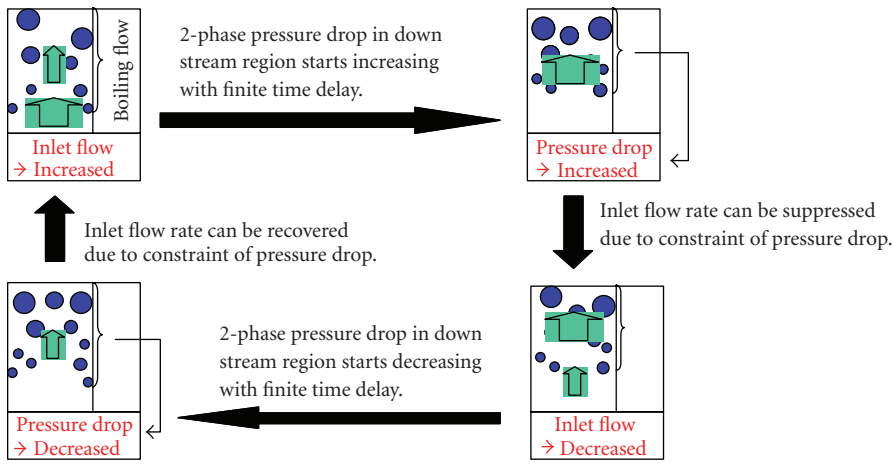


Figure 1: Schematic description for channel instability mechanism.

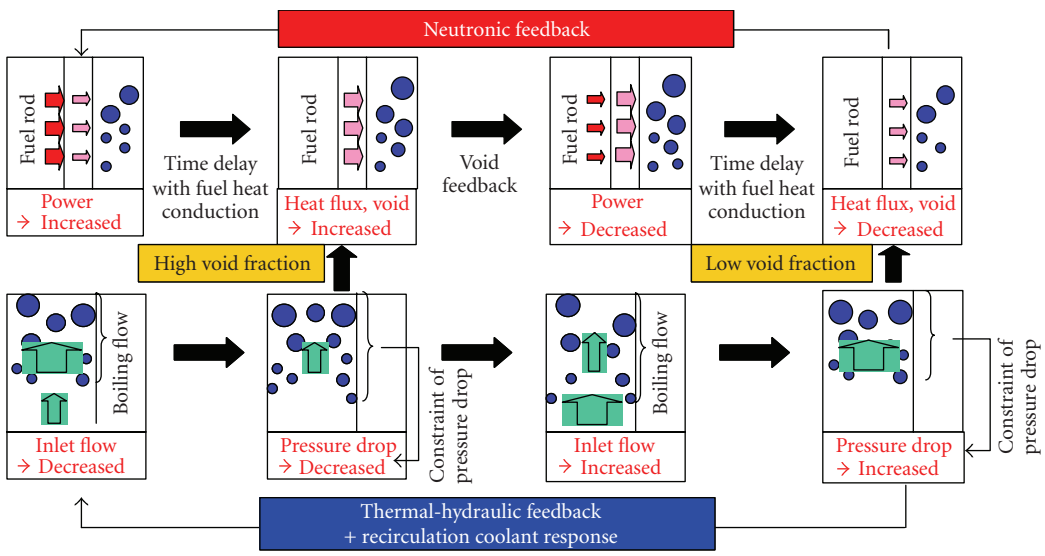


Figure 2: Schematic description for core instability mechanism.

Table 1: Features in frequency-domain and time-domain stability analysis codes.

	Frequency-domain code (reduced-order model)	Time-domain code (3D kinetics model)
Computation speed	Fast	Slower than Freq. model
Numerical diffusion	No	Dependent on numerical scheme
Decay ratio	Determinable by the unique way from the Nyquist curve for the system transfer function	Sensitive to time-step size, disturbance condition to activate transient state, in numerical simulation
Model limitation	1st-order linear perturbation to the nonlinear physical systems	Basically No
Spatial behavior	No	Yes
Nonlinear behavior	No	Yes

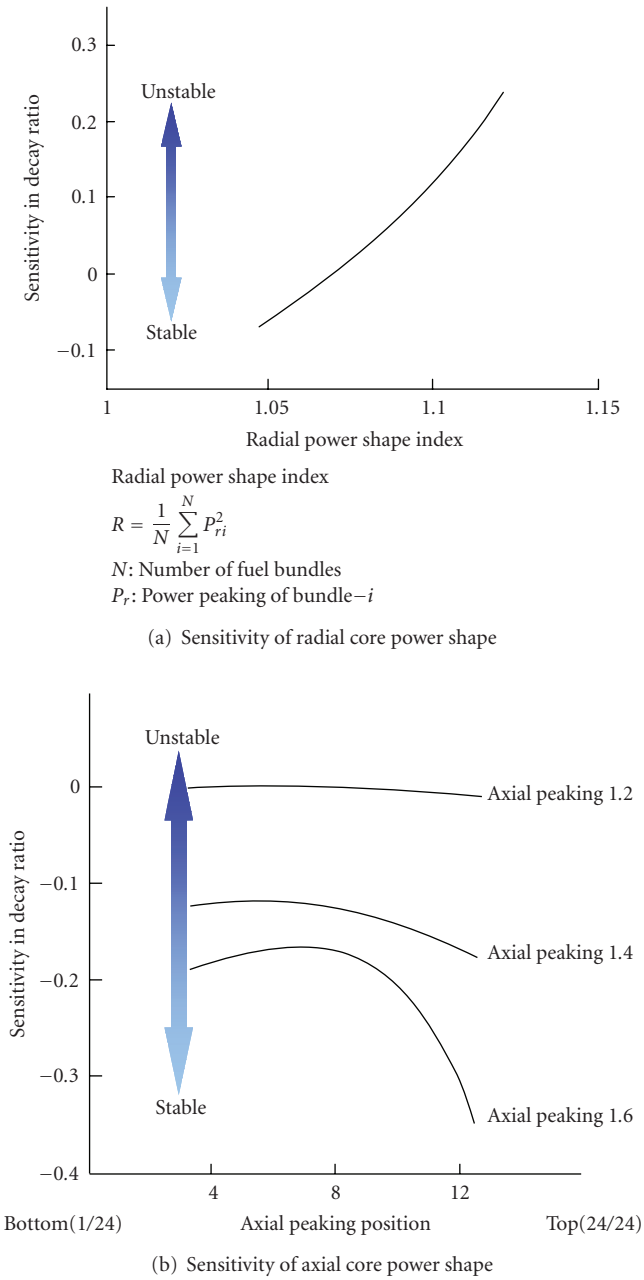


Figure 3: Sensitivity of core power shape to core stability decay ratio.

the best-estimate analysis code and the statistical approach in the safety evaluation methodology.

2. BWR INSTABILITIES

The BWR instability can be subcategorized into the three phenomena: (1) channel instability (density wave oscillation); (2) core instability (global core power oscillation); and (3) regional instability (powers in two halves of a core oscillate with an out-of-phase mode).

2.1. Channel instability

The channel instability is equivalent to the coolant density wave oscillation in a boiling channel, where the channel pressure drop is kept constant by any constraint [2, 3]. As shown in Figure 1, the coolant void sweeps in the boiling region, which significantly affects the 2-phase pressure drop, consequently leads to the coolant mass flow oscillation at the channel inlet. Hence, the channel instability can be invoked in a channel, where the 2-phase pressure drop is relatively larger than the single-phase pressure drop, for such conditions as (1) higher channel power and lower flow rate, (2) lower inlet coolant subcooling, (3) down-skewed axial power shape, (4) numbers of fuel rods and of fuel spacers which tend to generate the larger pressure drop in the 2-phase boiling region. In general, however, excitation of the channel instability can be suppressed by many other stable channels via the neutronic coupling effect among fuel bundles in an actual core.

2.2. Core instability

The coupled neutronic and thermal-hydraulic power oscillation can be categorized into the global instability and into the regional instability. In the first mode, the global core power oscillates in-phase, while in the regional oscillating mode, the power in a half core oscillates in an out-of-phase mode with respect to the other half. The core power oscillation is mainly driven by the negative coolant void feedback with the finite time del ay due to the fuel heat conduction [2]. This power oscillation can be actually excited by synchronizing with the mentioned density wave oscillation, as schematically described in Figure 2. a range from 0.3 to 0.6 Hz [4, 5], which are correlated with the wave propagation velocity through the core fuel channel.

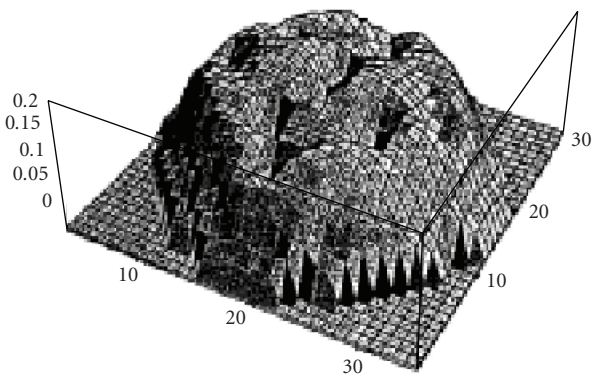
The core power oscillation becomes unstable under the lower flow and higher power core operation condition, corresponding to the density wave oscillation behavior. Large negative void feedback and faster fuel heat conduction make the core state unstable. In addition, the past investigation using frequency-domain stability analysis codes revealed interesting sensitivity with respect to the core power distribution, as shown in Figure 3 [6]. As for the radial power shape, fuel bundles with high power peaking factors tend to reduce the channel stability in the entire core, resulting in the core instability. The sensitivity regarding the axial power shape has more complicated nature as described below. The down-skewed shape leads to the longer boiling length, which makes the frequency of the density wave oscillation greater than the time constant in the fuel heat conduction. This mismatch tends to result in the stable core power oscillation. On the other hand, the flat and/or the middle-skewed shapes make the greater influence of neutronics in the high void region of the core, inducing the core instability due to increase in the negative void feedback.

2.3. Regional instability

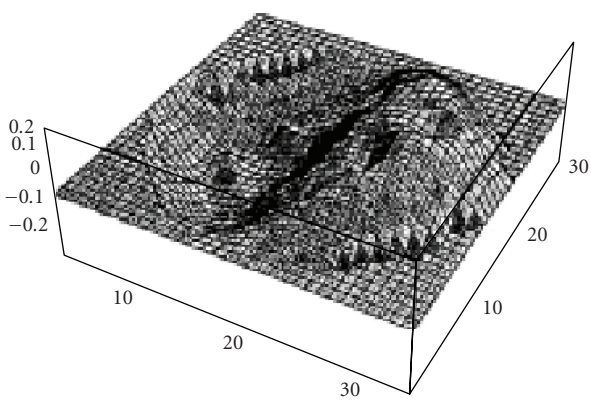
The basic phenomenon dominating the regional instability is similar to that for the core instability, and the coupled

Table 2: Three-dimensional stability analysis code in Japan.

Code	Development	Code qualifications
TOSDYN	Toshiba	Peach Bottom-2, Vermont Yankee, LaSalle-2, Caorso, Leibstadt, KRB-B/C, BWRS Startup Data, parallel loop channel stability test
STANDY	Hitachi Toshiba BWR utilities in Japan	Peach Bottom-2, Vermont Yankee, LaSalle-2, Caorso, KRB-B/C, etc.
TRACG	General electric GNF-A/GNF-J Toshiba Hitachi	LaSalle-2, Cofrentes, Leibstadt, Forsmark-1, etc.
DYNAS-2	Nuclear fuel industries	PB-2, WNP-2, KRB-B/C, KKK, Ringhals-1, parallel loop channel stability test, etc.
TRAC-BF1/ENTRÉE	TEPCO systems	Ringhals-1



(a) Fundamental mode



(b) Higher harmonics (1st Azimuthal) mode

Figure 4: Sample of spatial neutron harmonics modes.

neutronic and thermal-hydraulic oscillation can be individually excited in two halves of a core with an out-of-phase mode. Previous researchers proposed that the regional instability is equivalent to the oscillation of the higher harmonics (1st azimuthal mode) of the neutron flux distribution, while the core instability is to the oscillation of the fundamental mode (see Figure 4) [7]. Hashimoto derived the so-called 'modal point neutron kinetics equations in order to analytically represent the phenomenon, in stead of the ordinary point kinetics equations [8]:

$$\frac{dN_m(t)}{dt} = \frac{\rho_m^s - \beta}{\Lambda_m} N_m(t) + \frac{\rho_{m0}(t)}{\Lambda_m} N_0 + \sum_{n=0}^{\infty} \frac{\rho_{mn}(t)}{\Lambda_m} N_n(t) + \lambda c_m(t), \tag{1}$$

$$\frac{dc_m(t)}{dt} = \frac{\beta}{\Lambda_m} N_m(t) - \lambda c_m(t), \tag{2}$$

where

$$\rho_m^s = 1 - 1/k_m, \tag{3}$$

$$\rho_{mn} = \langle \phi_m^*, (\delta M - \delta L)\phi_n \rangle / \langle \phi_m^*, M_0 \phi_m \rangle. \tag{4}$$

m is the order of the higher harmonic mode ($m = 1, 2, \ldots$); N, c, and β are the core-averaged neutron flux, del ayed neutron precursor, and delayed neutron fraction, respectively. The other variables and notations are defined in the original paper [8]. Physically, ρ_m^s represents the subcriticality of the mth harmonic mode, which is mathematically corresponding to the eigenvalue separation, and is a negative value in the above definition. Hashimoto [8] and Takeuchi et al. [9] pointed out that a smaller absolute value of the subcriticality makes the feedback gain of the regional oscillation larger, which is correlated to the first term of the right-hand side of (1), inducing the regional instability.

As mentioned above, powers in two halves of a core oscillate with an out-of-phase mode, therefore, significant oscillations cannot be observed in the core-averaged power and inlet coolant flow responses. This results in that the hydraulic flow response via the recirculation loop is less sensitive to the regional stability.

3. BWR STABILITY ANALYSIS CODES, VERIFICATIONS, AND APPLICATIONS

Several stability analysis codes have been developed so as to investigate the BWR instability phenomena in detail, and to apply on the BWR core design in Japan. The analysis codes can be mainly classified into the two categories, the frequency-domain code and the time-domain code. Features

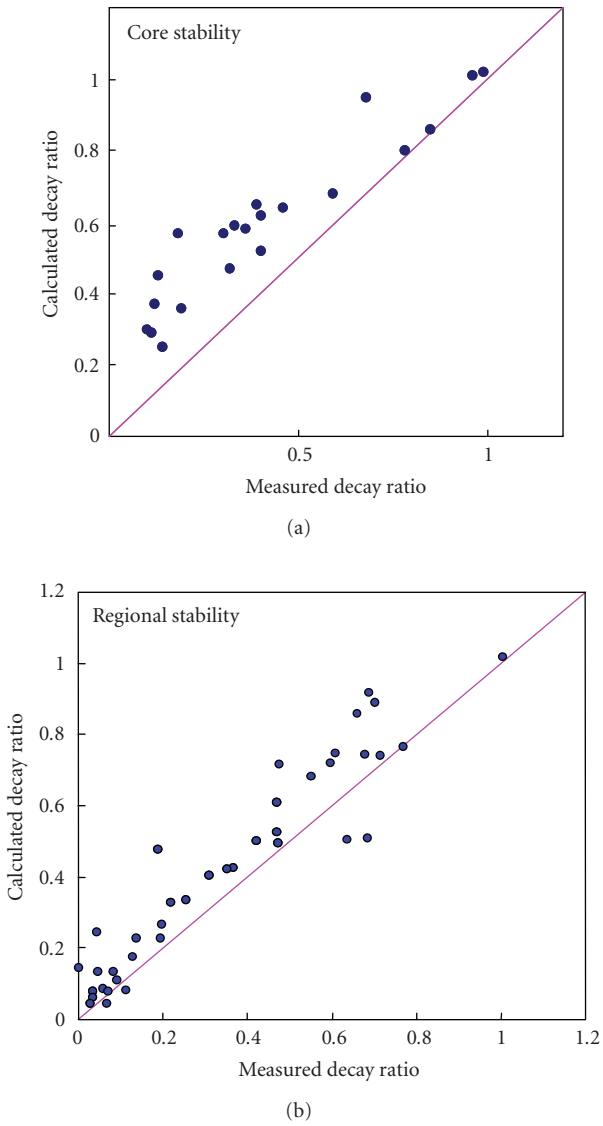


Figure 5: Sample of verification for the frequency-domain code using stability test data.

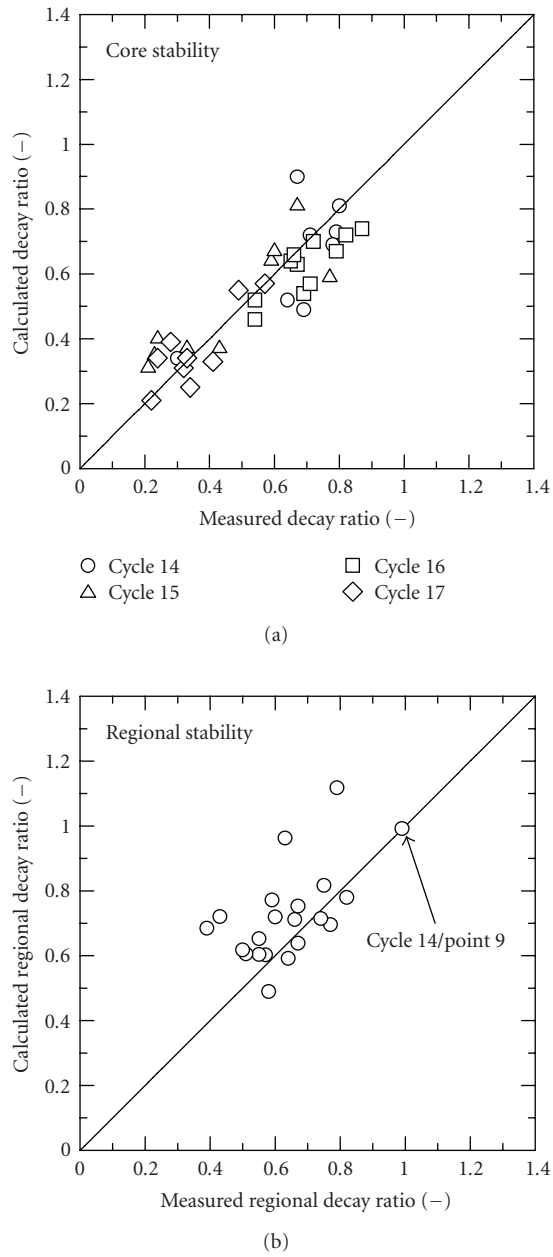


Figure 6: Sample of verification for the time-domain code using Ringhals-1 stability test data.

of the frequency-domain stability analysis code and of the time-domain code are summarized in Table 1, respectively.

3.1. Reduced-order frequency-domain codes

In general, the frequency-domain code employs the reduced-order model like the point neutron kinetics, to mathematically simplify the phenomenological representation, and to attain the faster computation time. In addition, the decay ratio, representing the stability degree of an oscillation, is determinable by the unique methodology based on the system transfer functions. These features are favorable in the design analysis. All the equations representing the physical phenomena are linearized for small perturbations to yield the system transfer functions via the Laplace transformation, which characterize the channel, core, and regional stabilities.

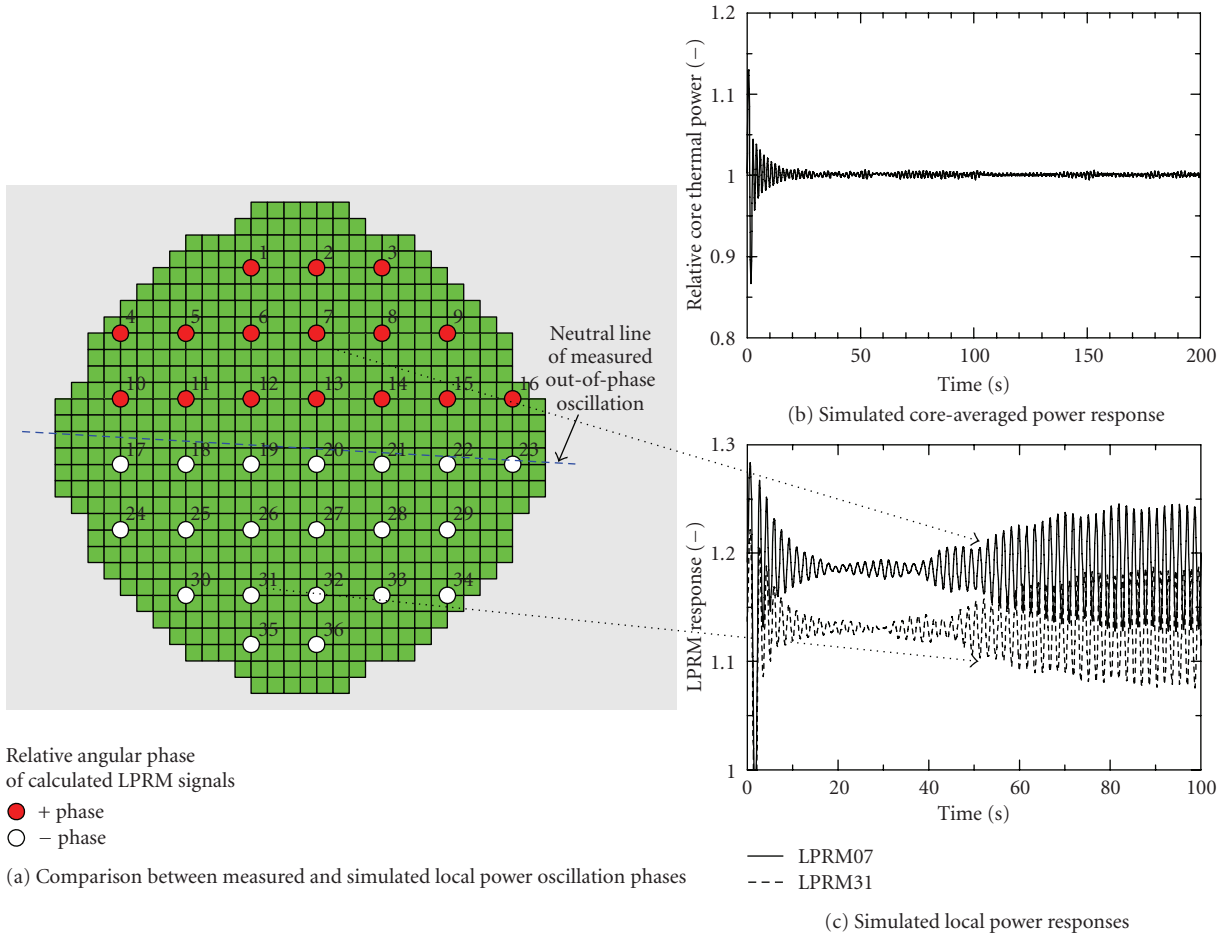
The primary physical equations employed in a representative frequency-domain code are the followings:

(1) mass, energy, and momentum equations for 2-phase mixture boiling coolant flow;

(2) radial one-dimensional fuel heat conduction equations; and

(3) point neutron kinetics equations.

The thermal-hydraulic behavior in a core is modeled with the parallel channel geometry, and the fuel heat conduction is accounted in each hydraulic calculation node. As for the regional stability analysis, the point neutron kinetics equation

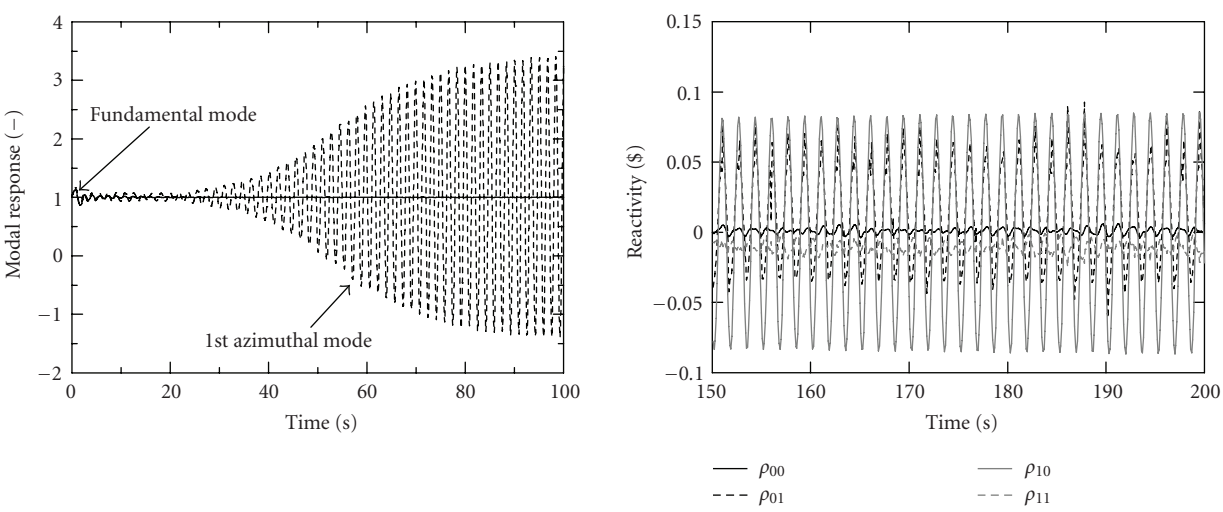


(a) Comparison between measured and simulated local power oscillation phases

(b) Simulated core-averaged power response

(c) Simulated local power responses

Figure 7: Simulated regional instability at Ringhals-1 C14/PT9 stability test.



(a) Simulated modal responses

(b) Simulated modal reactivity responses

Figure 8: Modal parameter responses under simulated Ringhals-1 C14/PT9 regional instability.

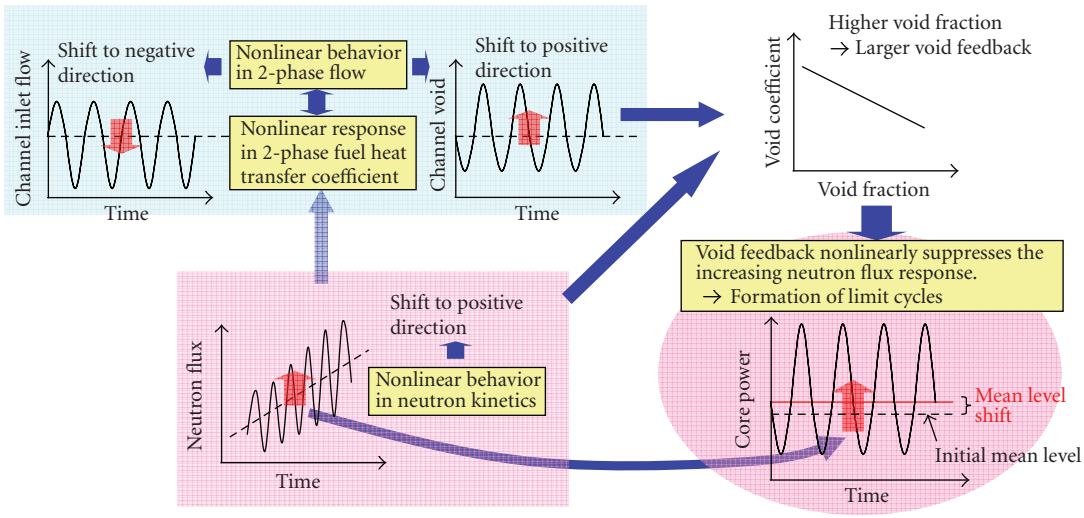


Figure 9: Mechanism in formation of limit cycle oscillations.

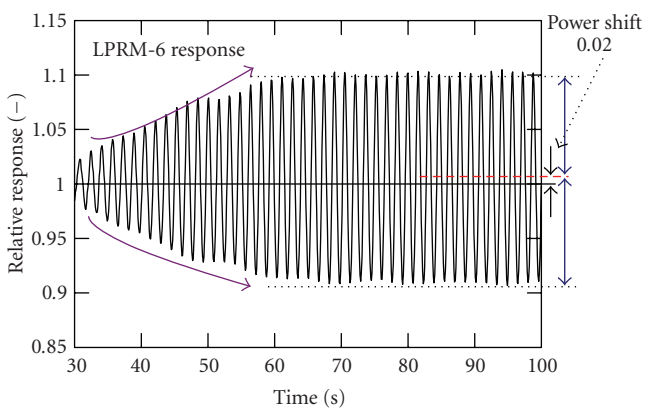


Figure 10: Average power shift in simulated Ringhals-1 C14/PT9 regional instability.

is replaced by the modal point kinetics equation as mentioned in the previous section.

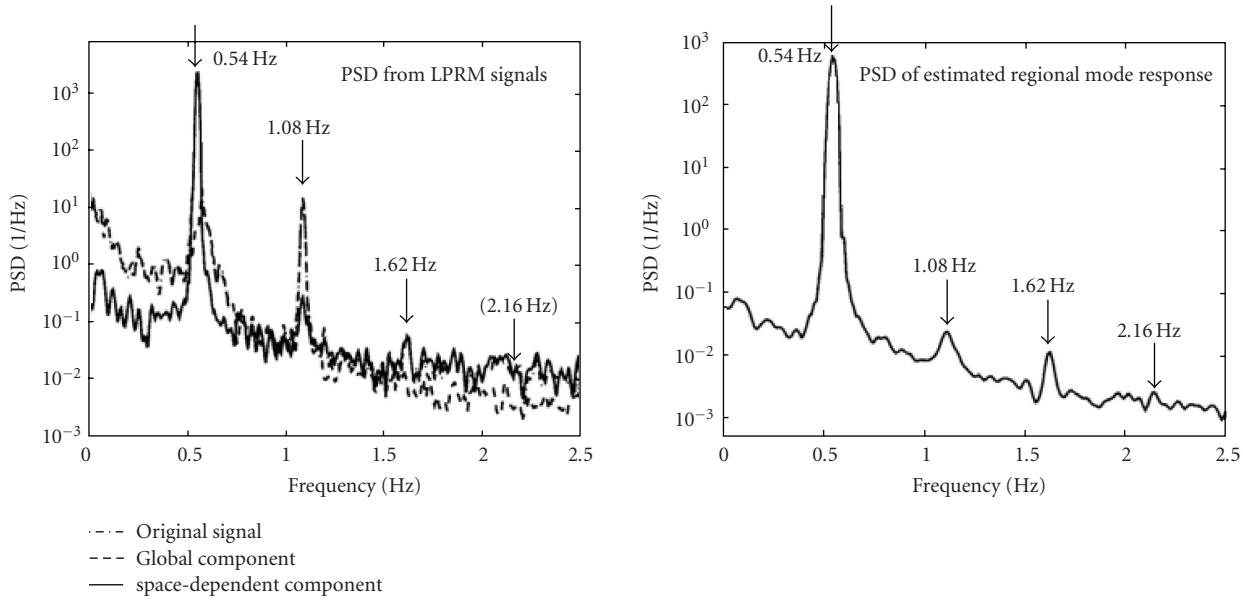
Figure 5 shows a sample of verification result for the frequency-domain code, which is currently applied on the BWR core design analysis. The code is able to derive good correlations over the wide stability range for the core stability analysis as well as for the regional stability analysis, while the code models are conservative a priori.

3.2. Three-dimensional time-domain codes

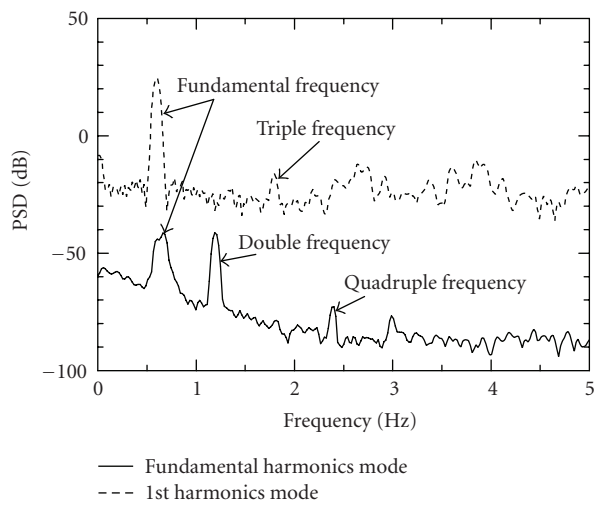
As described above, the frequency-domain codes generally employ the simpler fundamental equation set in order to avoid mathematical difficulties in derivation of the system transfer functions representing the coupled neutronic and thermal-hydraulic phenomena in a BWR. The time-domain code, on the other hand, adopts the more sophisticated physical models, like the spatial neutron kinetics model. In fact, their implementation on a code is simple and straightforward, while it consumes larger computational time than the mentioned reduced-order model. Since 1990s, however, the significant advance in computation technologies has facilitated development of time-domain codes that employ the complicated three-dimensional and multigroup neutron diffusion kinetics model [1, 10–14]. Several time-domain codes developed by Japanese organizations are listed in Table 2. The most significant advantage in these codes is that the detailed spatial kinetics behavior in a core can be explicitly simulated, namely, both the core stability and the regional stability can be evaluated using a single three-dimensional time-domain code without any modification. However, users have to pay attention to the applied numerical time step size, which is sensitive to the simulated oscillation and decay ratio [15, 16].

Furthermore, a simulator has been implemented on the recent time-domain codes in order to accomplish the more realistic dynamic simulation reflecting the actual core state including the fuel history data thus being seamlessly consistent to the static core design [17, 18]. Figure 6 shows a sample of verification for the three-dimensional time-domain stability analysis code, SIMULATE-Kinetics, using the Ringhals-1 stability test data [19, 20]. It can be confirmed that the code is basically targeting on the best-estimate stability analysis on the contrary to conservative approach applied in the frequency-domain code. The Ringhals-1 cycle-14 PT9 stability test, where a regional instability was observed, was accurately simulated as shown in Figure 7 [19]. In addition, the results of numerical simulation demonstrated that the observed regional instability is equivalent to an oscillation of the higher harmonics mode (1st azimuthal, N_1 defined by (3), and that modal reactivities (ρ_{10} and ρ_{01} defined by (4) are dominant in the regional event as shown in Figure 8.

A feature of the three-dimensional time-domain code is that it is applicable to the analytical investigation of the limit cycle oscillation which is driven by the complicated nonlinear effects [21–23]. Figure 9 schematically describes the del iberated mechanism in the formation of limit cycle oscillation.



(a) Harmonics excitation in measured modal responses [24]



(b) Harmonics excitations in simulated modal responses

Figure 11: Harmonics excitations under regional limit cycle oscillations.

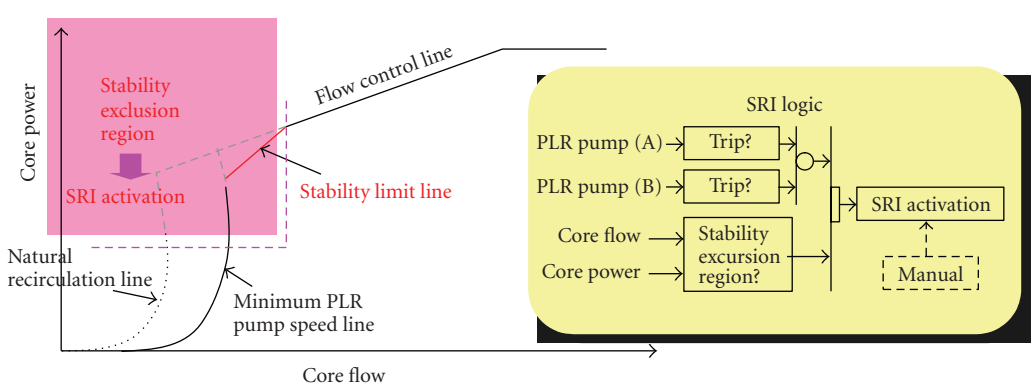


Figure 12: Outline of approved stability solution methodology in Japan.

The nonlinear behavior in the 2-phase boiling flow tend to increase the core-averaged void fraction and the negative void feedback, which suppresses the growing neutron flux oscillation due to nonlinearity in the neutron kinetics, resulting in the limit cycles. Any power shift observed in the measured core power responses and/or in the numerically simulated power responses (see Figure 10) is due to the above nonlinearities.

As for another scientific interest on the regional limit cycle oscillation, the bifurcation behavior observed via the spectrum analysis of the measured core power responses [24], Farawila theoretically proposed that the nonlinear interaction in the modal reactivities defined by (4) plays an important role in this phenomenon [25]. In addition, Ikeda et al. have numerically demonstrated that the nonlinearity excites the different higher harmonics of the core-averaged and regional power responses, respectively, as shown in Figure 11 [23], which was obtained by applying a spectrum analysis to the simulated fundamental and higher modal responses (refer to Figure 8).

4. CURRENT BWR STABILITY SOLUTION METHODOLOGY

Since the instability incident at LaSalle-2 [26], GE, and US-BWROG has developed several long-term stability solution methodologies [27, 28]. Also in Japan, a similar stability solution methodology was established, where the adequate stability margin must be ensured in the core design process, and the selected control rod insertion (SRI) system is equipped to exclude the BWR core from the unstable operation region (stability exclusion region) as shown in Figure 12 [29].The SRI system is activated to suppress the core power when the core coolant recirculation pumps are tripped and the core goes into the preliminary determined stability exclusion region. The stability exclusion is to be determined by using stability design codes certified via the regulatory assessments, with the conservative stability criteria (decay ratio is less than 0.8). Consequently, this methodology is targeting on that the BWR instabilities are not possible in the operated core in Japan.

5. RESEARCH ON ADVANCED BWR STABILITY SOLUTION METHODOLOGY

The current stability solution methodology is effectively contributing to safety of BWR plant operations in Japan. However, considering the recent occurrences of BWR instabilities [30, 31], authors suppose that any improvement may be indispensable for the future stability solution methodology, which is able to correspond to the recent modifications in the existent BWR plants as the extended core thermal power-uprate [32] with the advanced fuel designs [33–35]. An approach to resolve this concern is that sufficient stability margin is to be introduced, namely, the plant operable region is limited by the wider stability exclusion region, which can be determined by using the current conservative stability analysis code, as shown in Figure 13. This approach, however, possibly leads to the economical loss by

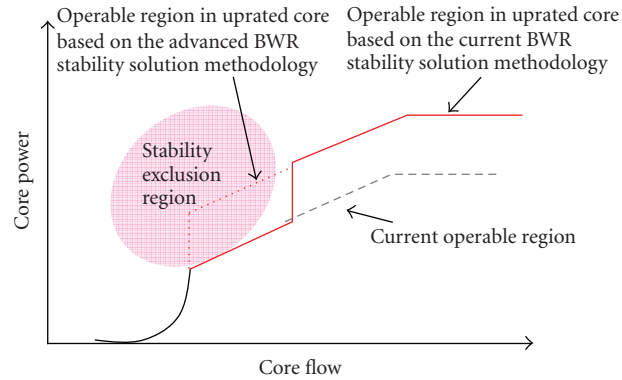
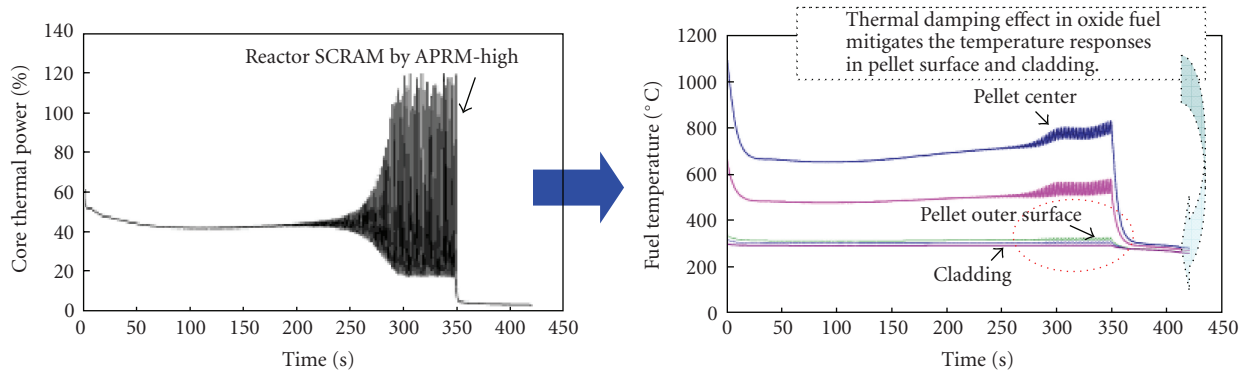


Figure 13: Plant operable region for the power-uprated core.

consuming longer time for the plant startup operation. Reactors with the larger stability exclusion region are generally allowed to adopt few continuous withdrawals of control rods under the lower power condition. This is because the continuous-withdrawal operation induces significant increase in core power at the fixed core flow condition, possibly removing the core into the prohibited stability exclusion region. Therefore, a lot of control rod operations, which must be conducted slowly and intermittently to maintain the fuel mechanical integrity, are required under the higher power condition to attain the target control rod pattern at the rated power operation. Consequently, the overall plant startup time tends to become longer in the BWR plant with the larger stability exclusion region.

In order to reasonably enhance the operable region even under the power-uprated core, a joint research group organized by several Japanese industrial and academic organizations has started a development of an advanced stability solution methodology based on the best-estimate code system [36]. Basis of the present research is to apply the original regulatory criterion with respect to the BWR instability [37], that is, "*exceeding specified acceptable fuel design limits (SAFDLs) are not possible*", not prevention from the instability occurrence. From the viewpoint of the applicable SAFDLs on the BWR instability, the PCMI and the material fatigue via the power oscillation possibly make no significant affect on the fuel integrity, because temperature responses of the fuel pellet and cladding are negligibly small as shown in Figure 14. Therefore, occurrence of the core coolant boiling transition (BT) can be a primary cause for the fuel failure under the BWR instabilities. So as to accurately and mechanistically predict the BT onset even under the BWR instabilities, the research group is applying an advanced code system based on the best-estimate plant simulator, TRAC-BF1/ENTRÉE [13], and the 2-fluid/3-field subchannel code, NASCA [38]. As schematically described in Figure 15, TRAC-BF1/ENTRÉE provides the pin-by-pin-based power responses in each fuel bundle; the subchannel thermal-hydraulic behavior and BT onset on the local rods are evaluated by NASCA with the boundary conditions supplied by the TRAC-BF1/ENTRÉE.



(a) Postulated BWR power oscillation

(b) Fuel temperature responses

Figure 14: Fuel temperature responses under the representative BWR instability.

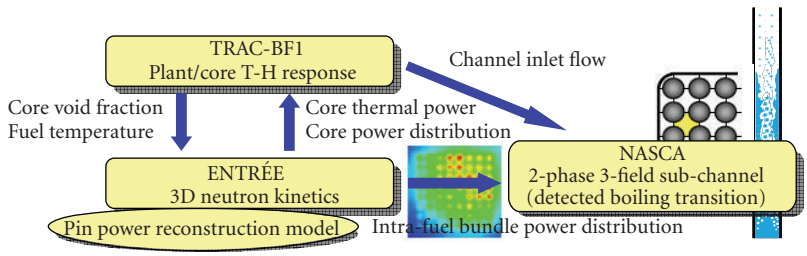


Figure 15: Outline of TRAC-BF1/ENTRÉE and NASCA code system.

The research group is also investigating the possibilities to introduce the statistical safety evaluation methodology [39] so as to establish the reasonable conservatism in the stability exclusion region determined by using the above best-estimate code system. The research, in particular, currently pays a lot of efforts to establish the phenomena identification ranking table (PIRT) applicable to BWR instabilities including the subchannel thermal hydraulics, based on the existent stability PIRTs [40–42]. This is the basis of the uncertainty evaluation for the best-estimate BWR stability analysis.

6. CONCLUSIONS

Many efforts have been paid to research on BWR stability issues in Japan, as introduced in the present paper. The industrial organizations have developed and improved the BWR stability analysis using computational tools specific for the reduced-order frequency-domain and three-dimensional time-domain codes. The first category is currently applied to the BWR stability design analysis, while the latter one has been exploited to understand the complicated phenomena related to BWR stability. The current stability solution methodology based on the SRI system with the stability exclusion region is successfully preventing the occurrence of BWR instabilities in Japan. However, authors suppose that the future application of the extended core power uprate requires further improvements to the current solution methodology in order to reasonably minimize the stability exclusion region. A Japanese research group is currently proposing to apply the best-estimate analysis code with the statistical safety evaluation methodology. This will allow better evaluation of the stability exclusion region, and will be consequently applied to the BWR plants with the extended core power uprate.

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