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FLOW-INDUCED VIBRATION ANALYSIS OF STEAM GENERATORS AND FUEL ASSEMBLIES WITH THE VIBIC COMPUTER CODE

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Abstract

Degradation of components in nuclear power reactors due to flow-induced vibration and consequent fretting wear remains an important issue for reactor safety and life. This paper reviews AECL's existing technology for flow-induced vibration assessment of steam generator tubes and reactor fuel assemblies, and presents studies to assess the potential for vibration-related damage of steam-generator tubes at the Qinshan Nuclear Generating Station and fuel rods in a Generation-IV Super-Critical Water-cooled Reactor.

Introduction

The susceptibility of components to flow-induced vibration (FIV) and consequent fretting wear (FW) has always been an important consideration in the design and operation of nuclear power reactors. Historically, steam-generator tubes and fuel assemblies have suffered the most failures. In Canada, early steam-generator tube failures occurred at the Nuclear Power Demonstration (NPD) reactor, at Douglas Point Nuclear Generating Station (NGS), and at Pickering A NGS [1]. The NPD reactor was Canada's first nuclear power reactor and began operation in 1962, several months before a similar demonstration reactor in the U.S. In NPD's first nine years of operation, approximately 630 tubes had failures related to flow-induced FW. This failure mechanism continues to be a serious concern. Recently, unusually severe FW damage caused steam-generator tubes to leak at San Onofre NGS Units 2 and 3. On investigation, the steam generators were found to show premature wear at approximately 15000 locations on over 3000 tubes, less than three years after installation in both units [2]. On 2013 June 7, Southern California Edison announced plans to permanently retire these two units. FIV and FW damage has been identified as a key cause of fuel failure as well, mostly in pressurized water reactors (PWRs) [3].

When FW was identified as a serious degradation mechanism for nuclear reactor components, research organizations in Canada and other countries were quick to respond. Atomic Energy of Canada Limited (AECL) was one of the first research organizations in the nuclear sector to study and solve FIV and FW problems. As part of this work, computer codes such as VIBIC [4], [5] and PIPO were developed in the 1970s. PIPO was upgraded to its current version, PIPO-FE [6], in 2000.

FIV and FW problems in nuclear power plants are complex, but the likelihood of potential vibration problems can usually be predicted. At the design stage, comprehensive FIV and FW analyses using widely accepted codes such as VIBIC or PIPO-FE can be conducted to demonstrate that vibration levels and FW rates will be acceptable under design conditions. Once the plant is operational, these codes can also be used to investigate the causes of an unanticipated tube or fuel failure. If a vibration problem exists, the codes can be used to assess any proposed design modifications. In use since the

1970s, the codes have been continually updated with the latest analytical techniques and experimental findings.

Recently, the VIBIC code was used to assess the expected FIV and FW performance of a Generation-IV (Gen-IV) fuel assembly to be used in a fuel qualification test. The assessment showed that VIBIC is capable of providing predictions of wear damage due to FIV in Gen-IV fuel at the design stage. The predictions will be refined once the fluid forcing function has been verified and material wear properties have been measured at expected Gen-IV operating conditions.

This paper reviews AECL's existing FIV and FW assessment technology and its development. FIV and FW analyses of the Qinshan NGS steam generator tubes and Gen-IV fuel assemblies are included to demonstrate that the VIBIC code can be used to assess potential FIV and FW damage of steam generators and reactor fuel.

1. FIV and FW assessment technology and its development

AECL has studied FIV and FW in nuclear components for over 40 years. Because of the complicated nature of vibration in fuel assemblies and steam generators, experimental measurements have played a key role in characterizing the underlying mechanisms. Based upon experimental results obtained by AECL and others, formulations have been developed to describe vibration excitation phenomena and the associated component response. These formulations, in turn, have been used in computer codes developed by AECL and others in the nuclear and process industries. Formulations used in AECL codes have been described in many internal reports and external publications (for example, [6]).

1.1 Steam generator analysis

The three publications listed as References [7], [8] and [9] document AECL's recommended design guidelines for FIV and FW in steam generators and summarize vibration analysis procedures. Five main steps are outlined for conducting a vibration analysis of steam-generator tubes. Each of these steps can be conducted with different degrees of complexity depending on the purpose of a specific analysis. Two approaches, linear and non-linear, have been used to conduct FIV and FW analyses of steam generators at AECL. PIPO-FE and VIBIC, both of which use beam-type finite elements, have been developed to perform analyses using these two different approaches.

PIPO-FE uses a linear-analysis approach in which supports are simulated as pinned and the tube sheet is simulated as a clamped support. Neglecting the support clearance greatly simplifies calculations of the tube vibration response. A simulation run with PIPO-FE can typically be completed within several minutes for a typical CANDU®[1] -type steam generator tube analysis [10].

A non-linear approach is used in VIBIC to analyze vibrations of a steam generator tube with clearance supports. Tube and support intermittent contact makes the problem non-linear. A VIBIC simulation requires a much longer time due to calculations of the dynamic interaction between a tube and its supports. The VIBIC code uses a superposition of vibration modes to obtain the uncoupled modal equations of motion and transient response.

[1] CANDU® (Canada Deuterium Uranium) is a registered trademark of Atomic Energy of Canada Limited (AECL).

Vibration due to the three vibration mechanisms known to be important in steam generators, namely fluidelastic instability (FEI), random-turbulence excitation (RTE) and vortex shedding can be assessed by such codes. Using Archard's wear correlation, the fretting wear damage caused by random-turbulence excitation is estimated in PIPO-FE based on the energy approach proposed in Reference [11], and in VIBIC by integrating the calculated energy dissipated at contacts [12]. Because bounding values of the various empirical parameters are typically used in the calculations of damping and the three vibration mechanisms, the codes generally yield conservative predictions.

Archard's wear correlation states that the wear rate is linearly related to the power dissipated by friction through a constant wear coefficient [12]. Therefore, besides the work-rate calculated in the codes, the reliability of the wear damage prediction of a steam generator is also largely dependent on wear coefficients derived from experimental studies. Many of the nuclear industry's most relevant tests of material wear properties were performed at AECL over the past three decades. Two types of test machines, room-temperature and high-temperature, have been used to conduct fretting-wear experiments. Early test programs at AECL involved exploratory tests at or near room temperature, to determine ranges of interest and trends of all the important wear parameters [12]. Since the late 1980s, more complicated high-temperature machines have been used to test steam-generator materials at reactor secondary-side environmental conditions. Examples of experimental wear studies can be found in References [12] to [14] and in the references cited in these publications.

In addition to wear studies, various experimental studies on excitation mechanisms, damping and the corresponding vibration responses of steam generator tubes have been carried out at AECL. References [15] to [17] are example papers on the vibration behaviour and damping of tube bundles subjected to single- and two-phase air/water and Freon.

PIPO-FE and its predecessor, PIPO, were designed to conduct simplified analyses to provide a quick or initial vibration assessment. Both codes can be easily used by a non-specialist [10]. PIPO-FE checks the results of analyses to determine if certain basic criteria have been met for the three excitation mechanisms considered [7], [8]. Appropriate messages appear in the output as warnings if any of the appropriate acceptance criteria have not been met. VIBIC can be used to study issues that require a detailed simulation, such as the effects of tube-to-support gaps and preloads.

1.2 Reactor fuel analysis

VIBIC was originally developed to analyze FIV and FW of steam-generator tubes. VIBIC's main feature is the detailed modelling of loose supports, which includes the effect of stick/slip friction and squeeze film dynamics. Another attractive feature is its fast calculation speed and ease of use compared to general-purpose codes developed in the same era. Largely because of these features and the extensive validation (e.g., [18]), the code has been used to study vibration problems with fuel rods in PWRs, and fuel channels and fuel elements in CANDU reactors. Commercial studies have been conducted for Canadian and international nuclear organizations and manufacturers.

Similar to vibration studies conducted on steam generators, parameters used in VIBIC fuel simulations are determined or confirmed through experiments. Examples can be found in References [18] to [22]. To study fuel FW due to FIV, relative motion and contact forces between a vibrating CANDU fuel element and its support were measured using a work-rate measuring station [18]. Experimental investigation was conducted to verify the numerical simulations [19] and the results showed that, for three levels of excitation, the predicted work-rates at various gaps and preloads are in good agreement with the measured values. Reference [20] reported tests of various fuel elements conducted with a sliding-wear machine at room temperature to study the effect of CANDU fuel element motion on pressure tube wear. Reference [21] reported fretting-wear tests of Zircaloy bearing pads in contact with Zirconium alloy pressure tube samples at temperatures ranging from 25 to 315°C. In the early 2000s, an experimental study was conducted to investigate the effects of PWR fuel-rod motion and loading conditions on the wear of fuel-rod cladding made of Zircaloy-4 [22].

1.3 Current R&D programs

The operating experience of steam generators in the nuclear power industry demonstrates that a better understanding of vibration issues is still required, in particular fretting wear in the upper U-bend region of vertical recirculating steam generators in PWRs and CANDU reactors. An R&D program to test multispan U-bend (MSUB) tubes under various process conditions is currently underway at AECL. The test program is designed to improve understanding of vibration issues of in- and out-of-plane FEI and RTE in U-bend tubes with anti-vibration bar (AVB) supports. The goal of the program is to formulate more robust design methodologies and guidelines in order to mitigate steam generator structural vibration and prevent vibration damage. Reference [23] describes the basic test rig setup and early work conducted for preparing test instrumentation. This test program also provides an excellent opportunity to improve the VIBIC code, such as to further validate simulation capabilities and to add new calculations based on test findings.

AECL is also conducting studies of FIV in advanced reactors and smaller modular reactors. Due to significant changes in mechanical design and thermalhydraulic conditions, FIV and FW topics continue to be important for advanced nuclear reactors.

2. Qinshan steam generator analysis

In the early 2000s, AECL assessed the possibility of fretting-wear damage of Qinshan NGS steam generators with the VIBIC code. This section describes the VIBIC simulation conducted for the largest U-bend tube.

2.1 VIBIC model

Figure 1a, shows the VIBIC model of Tube MK#01, which is the largest U-bend tube in the steam generators. It has three flat-bar restraints and a collector-bar restraint on each side (cold and hot). The model consists of the full U-bend section plus straight leg portions on either side spanning three lattice grids. The cold leg is on the left-hand side of the tube. The origin of the coordinate system is located at the centre of the U-bend. The clearance supports and corresponding nodes are shown in red and blue, respectively. External steam/water density and flow velocity distributions predicted using AECL's THIRST code are shown in Figure 1b. The predicted external steam/water cross-flow mass flux distribution is shown in Figure 1c.

The collector-bar and flat-bar restraints and the top two lattice grids were modeled as clearance supports. To simulate the attachment of the U-bend section to the remainder of the tubes, the two lowest lattice-grid supports were fixed for the three directional translations and the x-directional rotation (torsion). In addition, a bending stiffness was applied at each end in the other two rotational directions to represent the restraint imposed by the cut-off straight leg sections. All of the clearance supports, with the exception of the end supports, were modeled by pairs of flat bars. Two pairs of flat bars located at adjacent nodes and oriented at 60° to each other were used to model the four lattice grids in the straight leg regions. The collector bars and the U-bend flat-bar restraints were modeled by pairs of flat bars oriented in the x-z plane.

Based on past experience, an effective contact stiffness of 10 N/µm was used for all of the clearance supports. A representative value of 0.5 was used for the dynamic coefficient of friction at all of the supports. To ensure accurate simulations of tube response during impact, a total of 115 modes were used in VIBIC. These sets of modes included modal frequencies up to 1500 Hz. Random turbulence excitation forces, modal damping and "negative damping" (when including the FEI effect) were calculated within VIBIC. The input data needed for the calculation include steam/water cross-flow velocities, densities and void-fraction distributions for every span.

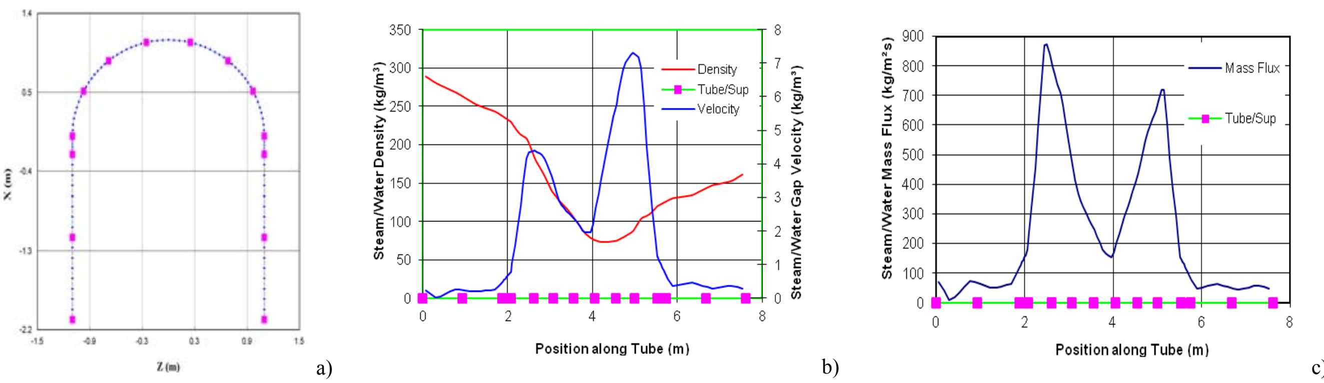


Figure 1 a) VIBIC model; b) Steam/water density and flow velocity distribution; c) Steam/water mass flux distribution.

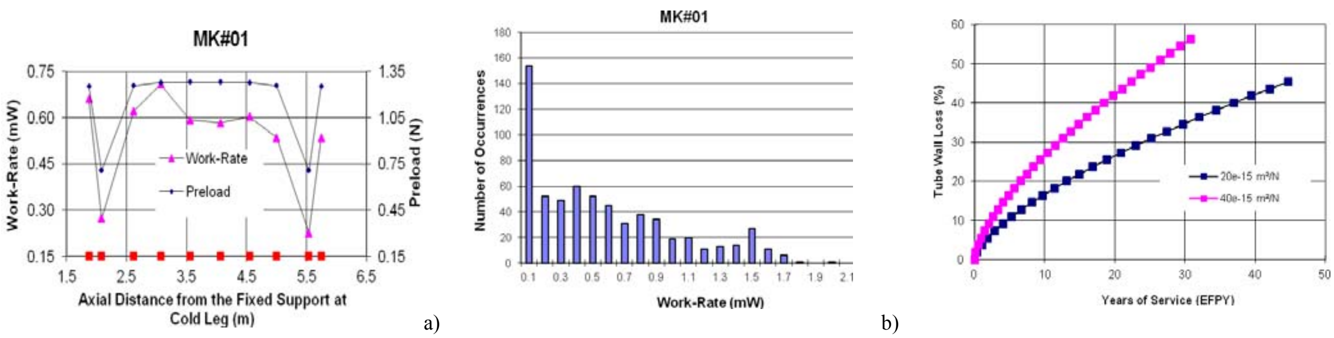
2.2 Predicted work-rates

The above model was run with different tube eccentricity within a clearance support, ranging from zero (centred or concentric) to 100% (touching or zero gap). When the tube is touching any of its supports, the exerted preload on the support can vary. Based on previous studies of tube preloads, eight possible preloads were considered: zero, 0.25, 0.5, 0.75, 1, 2, 3 or 5 N. The number of possible combinations of eccentricity and preload is very large for this tube with multiple supports. Therefore, 40 cases, each with randomly-selected sets of boundary conditions (i.e., clearance / preloads), were simulated systematically.

The average work-rates and preloads at each support are shown versus support position in Figure 2a. Average work-rates at flat bars in the U-bend region are generally higher than those at lattice grids in the straight leg portion and are always higher than those at collector bars. Predicted work-rates in the U-bend region range from zero to a maximum of 2.2 mW. The overall average work-rate (excluding the values at the lower lattice grids) is 0.53 mW. The greater average work-rate occurs on the cold leg side. This tendency is consistent with our previous fretting-wear studies. Distributions of work-rate at the collector and flat bars for each tube are shown in Figure 2b. Approximately 90% of the predicted work-rate values are less than 1.0 mW. Only 3% of the work-rate values are greater than 1.5 mW.

2.3 Tube wear rate prediction

The progression of fretting-wear damage in Qinshan SG tubes can be estimated from the work-rates summarized above if the wear coefficient and the geometrical relationships between wear volume and wear depth are known. Realistic wear coefficients for CANDU steam generators of the Qinshan Unit 1 / 2 design at operating conditions are in the range of 20 to 40 x 10^{-15} m^2/N. Following Reference [14] and assuming that the tube and support wear at the same volumetric rate and that the wear rate is constant, tube wall loss is calculated and shown versus years of service (EFPY) in Figure 2c. This estimate is based on the maximum work-rate predicted with nominal clearances without including the effect of fluidelastic instability (i.e., 2.2 mW). For the worst case, 2.2 mW for Support 7, tube wall losses for the most worn tubes reach 55% through wall depth after approximately 30 years of operation when a wear coefficient of 40 x 10^{-15} m^2/N is used.



a) b) c)

Figure 2 a) Average work-rate and preload at each support; b) Work-rate distribution at collector and flat bars; c) Tube wall loss in U-bend region versus time.

3. GEN-IV fuel rod analysis

A program has recently been started at the Rez Laboratories in the Czech Republic to use the LVR-15 research reactor to conduct a Fuel Qualification Test (FQT) at conditions of the Super-Critical Water-cooled Reactor (SCWR). This test facility will be the world's first nuclear facility operated with super-critical water.

AECL has developed a conceptual design of a SCWR and is helping to perform the FQT program. Following previous work to examine potential vibration issues that could affect a CANDU-type SCWR fuel assembly, AECL performed a vibration assessment of the FQT fuel assembly. The FQT features a cluster of four 0.6-m-long fuel rods confined by a square-shaped assembly box (referred to as a "flow tube" in this paper). Each fuel rod is wrapped with a wire helically wound in a counter clockwise direction. The flow tube is subject to both internal and external flow.

As the GEN-IV SCWR design is still in its conceptual phase, this analysis can only provide a preliminary assessment of vibration related issues for the proposed fuel assembly. The scope of work was to conduct an initial FIV and FW assessment. Future studies can build on this work to conduct more detailed simulations and to assess possible wear damage, once detailed fuel-assembly design information and flow data become available.

3.1 Fuel rod assembly

As shown by the reactor-core horizontal cross sectional view in Figure 3a, the test tube selected for the FQT is positioned at the F3 location. The test tube (cross sectional view in Figure 3a) includes an assembly of four fuel rods, the flow tube (assembly box) surrounding the four rods, inner and outer guide tubes and a pressure tube. Each fuel rod is supported (clamped) at both ends, with the bottom support designed to allow the rod to slide in the vertical (axial) direction.

The wire encircling each fuel rod is wrapped with an axial pitch of 200 mm. Figure 3b(i) shows the wire start positions at the top end of the fuel rods inside the flow tube. The four wires are positioned in-phase (i.e., the wire start and end positions on each of the rods are the same). Each wire is welded to the fuel rod at the top and bottom. The design gap between a fuel rod and the wire wrapped on a neighbouring fuel rod is 0.1 mm.

For this preliminary assessment, it was assumed that the same material, SS 321, will be used for the fuel flow tube, two guide tubes, helical wires and the pressure tube, and that SS 316 will be used for the cladding of the fuel rods. Uranium Dioxide will be the fuel used in the FQT. Coolant fluid densities corresponding to temperatures from flow simulations were obtained from the website of the US National Institute of Standards and Technology (NIST), assuming a constant pressure of 25 MPa.

The average axial velocity of the coolant along the fuel rod section was estimated from the super-critical water density data obtained from the test facility with the following assumptions:

1. The coolant axial mass flow rate is constant, i.e., there is no “shock wave” effect near the critical condition (due to sudden density changes) causing fluctuating mass flow rates;
2. The cross-sectional flow area inside the flow tube was calculated by assuming that the wire is wrapped on the rod tightly, leaving no gap between the wire and rod, that the wires are in contact with the flow tube at the start and end of each curved corner and that the curved corners have an opening angle of 90°;
3. In between each fuel rod and the flow tube, the coolant was assumed to flow in the wire helical-angle direction for about one-fourth of the fuel rod perimeter;
4. The coolant surrounding the rest of the fuel rod surface flows in two directions – mainly axially away from the wire, and in the helical angle direction near the wire;
5. The coolant temperature was assumed to increase linearly with axial position along the fuel rod, and the temperature difference around the circumference of the fuel rod can be neglected.

3.2 VIBIC model

The VIBIC model of an individual fuel rod is depicted in Figure 4, for the #3 fuel rod shown in Figure 3b(ii). The red line in Figure 4 represents the rod. The “flat bar” support option built into VIBIC was used to model potential contacts with either another fuel rod or the flow tube through the wrapped wires. In the model, each such contact condition or “support” is characterized by a pair of flat bars, one on either side of the rod. There are 16 contact/support locations along the fuel rod (in the x (axial) direction in the model) represented by 19 pairs of flat bar supports (shown as squared shapes in Figure 4) and two end supports (shown as diamond shapes). These supports are numbered from the fuel rod top (left end) to bottom (right end). Two pairs of flat bars are used at each of five locations and one pair of flat bars (marked inside the brackets) was used to represent each possible contact of the #3 fuel rod with the two neighbouring rods (#1 and #4 fuel rods) through the wrapped wire.

Viewed from the top, there are five possible contact locations around the circumference of the #3 fuel rod, as shown by the five rectangular areas outlined in green in Figure 3b(ii). Note that only one of each pair of flat bars was positioned close enough to potentially touch the #3 fuel rod; the other flat bar of each pair (not shown in this figure) was positioned further away from the modelled fuel rod. The tabulated data in Figure 4 show the positions of the 19 flat bars along the fuel rod and the flat bar angles. The 19 clearance supports can be characterized as having one of five flat-bar configurations, as listed in Figure 5, and simulate either a 0.1-mm gap or a just-in-touch contact condition.

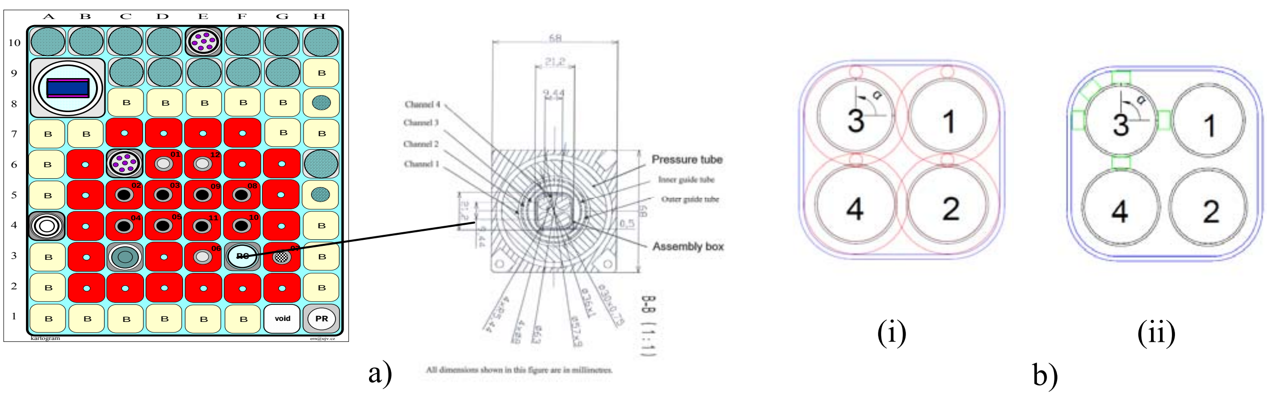
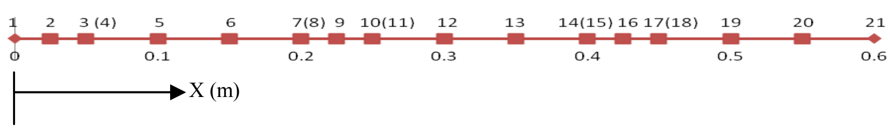


Figure 3 a) Core configuration and fuel bundle; b) Top view of fuel rods with wires (i) and wire model (ii).



Support #	Clearance Support #	Location	Support Type	Angle1*	Angle2**	Rod 3***	Rod 1	Rod 4
		(m)		(degree)	(degree)			
1		0	fixed					
2	1	0.025	flat bar	135		x		
3,4	2,3	0.05	flat bar	180	0	x	x	
5	4	0.1	flat bar	270		x		
6	5	0.15	flat bar	0		x		
7, 8	6,7	0.2	flat bar	90	270	x		x
9	8	0.225	flat bar	135		x		
10, 11	9,10	0.25	flat bar	180	0	x	x	
12	11	0.3	flat bar	270		x		
13	12	0.35	flat bar	0		x		
14, 15	13,14	0.4	flat bar	90	270	x		x
16	15	0.425	flat bar	135		x		
17, 18	16,17	0.45	flat bar	180	0	x	x	
19	18	0.5	flat bar	270		x		
20		0.55	flat bar	180		x		
21		0.6	fixed					

* Angle of the flat bar representing wire wrapped on Rod 3.

** Angle of the flat bar representing the wires wrapped on two neighbor rods.

*** x is used to mark the flat bar representing the wire wrapped on each rod.

Figure 4 VIBIC Model for Fuel Rod 3 Shown in Figure 3b

Twenty modes were used to simulate the response of the fuel rod. This set of modes includes modal frequencies up to 2500 Hz. To ensure accurate simulation of contact forces for the assumed contact stiffness of 5 N/µm, time steps of 30 µs were used. For this initial modelling, a damping ratio of 3% was used for all the modes specified. A friction coefficient of 0.5 was assumed. A total of 10 s of simulation time was used to ensure transient effects were minimised. The final work-rate predictions were based on cumulative work rates over the total simulation time.

19 Clearance Supports

Clearance Support #	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19
Clearance Setup #	2	3	5	4	5	1	4	2	3	5	4	5	1	4	2	3	5	4	5

Five Clearance Setups

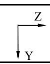
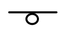
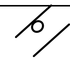
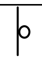
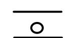
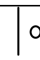
Clearance Setup #	1	2	3	4	5
					
Angle (deg)	90	-45	0	90	0
Gap between Fuel Rod and nearby Flat Bar (µm)	0	0	0	100	100

Figure 5 Boundary condition at clearance supports.

With the absence of experimental excitation data for wire wrapped fuel rods, the circumferential component of the helical flow was treated as tangential flow. The VIBIC model was run separately with excitation due to tangential flow only, and both tangential and axial flows. The random data used in the calculation of excitation forces due to the tangential flow turbulence were generated using an inverse FFT filter. The turbulence excitation amplitudes for the tangential flow component were calculated within VIBIC based on the empirical correlation between force and cross flow velocity for steam generator tubes since there is no forcing function available for super-critical water flow. Six correlation indices were used to set the forces acting on the rod within each 100-mm increment of axial length in either the y or z direction to be correlated. Forces that were not within the 100-mm range or not in the same direction were uncorrelated.

Simulation runs were conducted using this model with excitation forces applied along the fuel rod at the nodes between the two end supports. Preload values were applied to Node 21 (250 mm below the top end of the fuel rod). In one case, a preload at Node 21 was used to simulate the possible contact force due to differential thermal expansion of the fuel rod caused by a temperature variation around the circumference of the fuel rod. Pinned-pinned end conditions were used for two of the simulations in order to estimate the largest possible vibration levels.

3.3 Analysis results

Results of the simulations show that 1) the predicted rod vibrations vary little between different simulation cycles (one-second periods) and, therefore, it is sufficient to run the simulations for 10 s; and 2) predicted vibration response is dominated by the fundamental mode.

Figure 6a shows an example of the vibration response due to combined axial and tangential flow for a fuel rod with pinned top and pinned (but free to slide) bottom supports. The maximum predicted root-mean-square (RMS) amplitude is 102 µm. When the "pinned" boundary conditions for the top and bottom supports are replaced with "clamped" conditions, the shape of the vibration response along the fuel rod remains similar, but the vibration amplitude is about 10 times lower. In both cases, the maximum response is close to the middle of the rod (see Figure 6b). The maximum normal work rate is 0.095 mW at Support 10 (Figure 6c at Location 0.25 m). The effect of preload was also checked. As the preload at Support 10 increases, the work-rate at Support 10 increases as shown in Figure 6d).

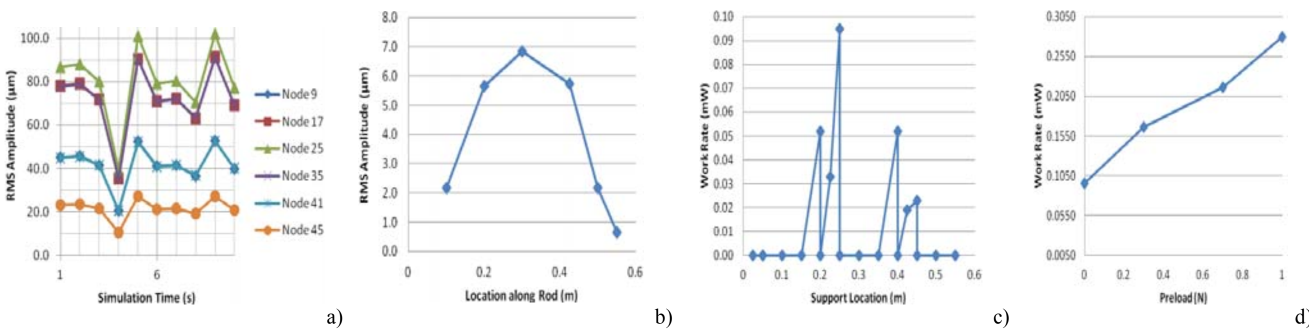


Figure 6 a) Rod response for pinned-pinned but free slide bottom end supports; b) Rod response of 3rd-simulation cycle for clamped end supports and zero preload; c) Cumulative work-rate for clamped end supports and zero preload; d) Work-rate vs. preload at Support 10.

The material loss of fuel rod and spacer wire or flow tube was estimated based on two assumptions: 1) volumetric wear of two contacting components occurs at the same rate and 2) the work rate and, hence, wear rate, is constant. An initial estimate shows that the most worn wire would reach 50% material removal in 35 years, and the fuel cladding has a maximum wear depth of 0.7 mm after 60 years of operation. For this study, a wear coefficient of 40×10^{15} m^2/N was used to estimate wear damage. Further investigation is required to more accurately assess potential wear damage to the fuel cladding and wire wraps once more detailed data for the fuel assembly, including also the supercritical-flow excitation function and material wear properties, become available.

4. Conclusions

FIV and FW assessment technology has been developed at AECL based on international and AECL studies. The VIBIC and PIPO-FE computer codes have been developed and maintained to resolve vibration issues in the nuclear power industry. Numerous code verification and validation studies have been conducted. Together with AECL's extensive past experience in vibration and wear simulation, the two case studies included in this paper have shown that the VIBIC code is capable of analysing flow-induced vibration of steam-generator tubes and fuel assemblies under realistic conditions.

VIBIC was used to assess the design of the Qinshan NGS steam generators. This analysis predicted that: 1) work-rates are small and tube failure is not an immediate concern (this has been confirmed now) and 2) tube wall losses for the most worn tubes reach 55% through wall depth after approximately 30 years of operation.

The FQT fuel assembly was analysed with a focus on determining ways to assess vibration in the SCWR fuel. The preliminary analysis showed that the fuel rod cladding was the component in the fuel assembly most prone to wear damage and the wire wraps would also undergo wear damage. To provide more reliable predictions on potential vibration-related damage for a complicated system, such as FQT fuel assemblies, a combined approach of experimental studies and numerical simulations should be conducted. In particular, studies are required to examine excitation and damping of wire wrapped cylinders in axial flow and wear under super-critical water flow conditions.

5. References

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