NUCET NUCLEAR ENERGY AND TECHNOLOGY

Nuclear Energy and Technology 5(3): 225–229
DOI 10.3897/nucet.5.39319

Research Article

Experimental studies of temperature pulsations during the process of mixing non-isothermal coolant flows in nuclear reactor equipment components*

Sergey M. Dmitriev[1], Alexandr V. Mamaev[1], Renat R. Ryazapov[1], Aleksey Ye. Sobornov[1], Andrey V. Kotin[1], Dmitry Ye. Bescherov[2], Mikhail A. Bolshukhin[2]

1 *Nizhny Novgorod State Technical University n.a. R.Ye. Alekseev, 24, Minin str., Nizhny Novgorod, 603950, Russian Federation*
2 *JSC Afrikantov OKBM, 15 Burnakovsky proezd, Nizhny Novgorod, 603074, Russian Federation*

Corresponding author: *Alexandr V. Mamaev* (aleks_may@mail.ru)

Academic editor: *Boris Balakin* ♦ **Received** 11 June 2019 ♦ **Accepted** 20 August 2019 ♦ **Published** 25 September 2019

Citation: Dmitriev SM, Mamaev AV, Ryazapov RR, Sobornov AYe, Kotin AV, Bescherov DYe, Bolshukhin MA (2019) Experimental studies of temperature pulsations during the process of mixing non-isothermal coolant flows in nuclear reactor equipment components. Nuclear Energy and Technology 5(3): 225–229. https://doi.org/10.3897/nucet.5.39319

Abstract

One of the most important scientific and technical tasks of the nuclear power industry is to assure the reactor equipment life and reliability under random temperature pulsations. High-intensity temperature pulsations appear during the process of mixing non-isothermal coolant flows. Coolant thermal pulsations cause corresponding, sometimes very significant, fluctuations in the temperature stresses of the heat-exchange surface metal, which, added to static loads, can lead to fatigue failure of equipment components.

The purpose of this work was to conduct an experimental study of the temperature and stress-strain states of a pipe sample under the influence of local stochastic thermal pulsations caused by the mixed single-phase heat coolant flows.

To solve the set problems, an experimental section was created, which made it possible to simulate the process of mixing non-isothermal coolant flows accompanied by significant temperature pulsations. The design of the experimental section allowed us to study the thermohydraulic and life characteristics of pipe samples made of austenite steel (60×5 mm). Some tools were developed for measuring the pipe sample stress-strain state and the coolant flow temperature field in the zone of mixed single-phase media with different temperatures. The measuring tools were equipped with microthermocouples and strain sensors.

As a result, we obtained experimental data on temperature pulsations, time-averaged temperature profiles of the coolant flow in the mixing zone as well as statistical and spectral-correlation characteristics of thermal pulsations. Based on the results of measuring the relative strains, the values of fatigue stresses in the mixing zone were calculated.

In addition, some devices and methods were elaborated to measure the temperature and stress-strain states of the pipe sample under the influence of local stochastic thermal pulsations. The developed experimental section provided thermal-stress loading of the metal surface at a high level of alternating stress amplitudes causing rapid damage accumulation rates. The results were included in the database to verify the method for assessing the fatigue life of structural materials for nuclear power plants as applied to austenite steel 12Cr18Ni10Ti under the influence of random thermal cyclic loads.

Keywords

Equipment life, temperature pulsations, coolant, temperature field, stress-strain state

* Russian text published: Izvestiya vuzov. Yadernaya Energetika (ISSN 0204-3327), 2019, n. 2, pp. 114–127.

Introduction

The equipment of stationary and marine nuclear reactor plants exposed to thermal stresses is subject to various deterioration mechanisms affecting the strength properties of structural materials and leading to its early life expiration (Abib et al. 2013, Chapuliot et al. 2005, Faidy 2002). The danger is represented by alternating thermal stresses caused by temperature pulsations. Temperature pulsations are steady-state random processes, in which the equipment life expiration occurs by the mechanism of high-cycle thermal fatigue (Sudakov and Trofimov 1980, 1989). A high level of temperature pulsations is characteristic of the processes of mixing nonisothermal coolant flows, which are widespread in nuclear reactor equipment. An urgent problem is temperature pulsations in transportable nuclear reactors, which are characterized by high thermal-stress density due to a significant decrease in their physical data (Budov and Dmitriev 1989).

Existing normative methods for assessing the reactor equipment life characteristics neither take into account the real laws of its thermal-stress loading, nor provide sufficient accuracy for damage accumulation calculations; therefore, this approach is rightly considered excessively conservative (NP-054-04 2004, Jhung 2013). Life tests of full-scale structures in normal conditions are not suitable due to the extreme complexity, high cost and long duration. In this regard, it seems appropriate to use a computational-experimental approach to assessing the longevity (useful life) of nuclear reactor equipment under random thermal cyclic loads, including the use of computational fluid dynamics (CFD) and stress-strain state modeling programs (Mahaffy et al. 2007, Smith 2010, Smith et al. 2010, 2015) as well as mathematical models of the fatigue damage accumulation process in the structural material.

This most quickly occurs at high values of the amplitude and number of loading cycles (Wakamatsu et al. 1995, Beaufils 2011, Courtin 2013). However, in the domestic and foreign literature there are no comprehensive experimental data on a single-phase coolant mixing at temperature drops of more than 100 °C (Miyoshi et al. 2016, Braillard and Edelin 2009, Chen et al. 2014, Kamide et al. 2009). Therefore, it was an important task to develop an experimental section for simulating the process of mixing coolant flows accompanied by significant temperature pulsations.

Currently, JSC Afrikantov OKBM is developing a methodology containing numerical simulation technologies using supercomputers to assess the effect of random thermal cyclic loads on the life characteristics of materials of the RITM-200 transportable nuclear reactor. As part of this work, the scientific team of the “Steam Generating Systems” Research Laboratory (“SGS” RL) at the NNSTU Nuclear and Thermal Stations Department conducts life tests of experimental models, the preparatory stage of which was the study of the temperature field and the stress-strain state (SSS) in the thermal-stress loading zone.

The article presents the results of the preparatory tests focused on the characteristics of temperature pulsations in the coolant mixing zone as well as the stress-strain state of the experimental sample material.

Description of the experimental facility

Experimental studies were carried out on the modernized FT-80 thermophysical facility (Dmitriev et al. 2006): its technical characteristics provide operating conditions corresponding to modern nuclear reactors. The experimental facility includes the following main components:

- coolant circuit;
- cooling circuit of the main coolant circuit equipment;
- reactor coolant makeup system;
- power supply system; and
- instrumentation system.

The equipment and pipelines of the facility are made of austenite steel 12Cr18Ni10Ti. High purity water was used as a coolant. The coolant circuit of the facility has five parallel channels for installing experimental models.

Description of the experimental section

The experimental section (Fig. 1) is a tee-branch consisting of the main 60×5×286 mm and peripheral 9×1,5 mm pipes and detachable joints. The pipes are made of steel 12Cr18Ni10Ti, which is used in reactor engineering. A diffuser is made in the guide flange. The slope angle of the peripheral pipe relative to the main one is 18°. The distance from the cut of the peripheral pipe to the inner wall of the main one is 7 mm. The main pipe is an experimental model (sample) subjected to temperature loading. Local loading of the inner sample wall is carried out by supplying a cold coolant flow from the peripheral pipe to the hot coolant flow.

Due to the detachable joints, both experimental and measuring models can be installed on the experimental section. The following measurement models were used in this work:

- model for measuring the temperature field (MMTP) and
- model for measuring the stress-strain state (MMSSS).

The material and dimensions of both models and samples are identical. The MMTP is equipped with a set of eight micro-thermocouples (t1–t8) made of KTMS cable in a stainless capillary with an individual calibration characteristic (Fig. 2). The calibration error is ± 0.2 °C. To reduce the

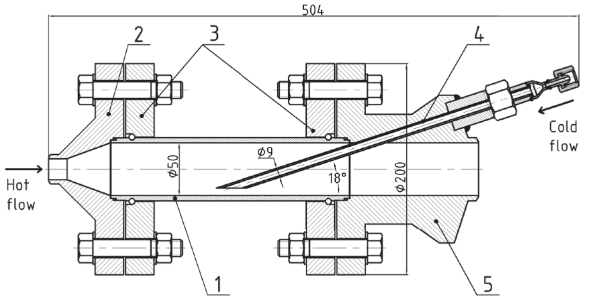

Figure 1. Experimental site 1. Experimental model; 2. Guide flange; 3. Cap flanges; 4. Peripheral tube; 5. Flange.

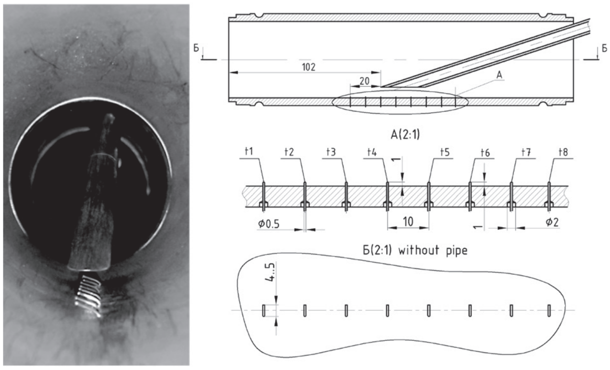

Figure 2. Location of the thermocouples on the MMTP.

thermal inertia of the thermocouples, the junction is made on the surface of the stainless capillary, and the sensitive part of the sensor is rolled to a diameter of no more than 0.5 mm (Kuschewski 2015). The thermocouples are installed in the central part of the pipe along the lower generatrix of the inner surface at a distance of 10 mm from each other. The sensors are oriented perpendicular to the main flow and spaced 1 mm away from the inner wall surface. Thermocouple t3 is located under the front edge of the peripheral pipe relative to the main flow direction. The MMSSS is equipped with eleven strain sensors and three thermocouples mounted on the outer surface of the model. The error in measuring the relative strain is $\pm 44 \cdot 10^{-6}$. The location zone of the sensors for the measuring models was selected so as to fully cover the proposed localization of the coolant mixing zone.

Experiment

For a reliable assessment of the durability of objects exposed to thermal-power impacts, it was necessary to study the complex processes of fatigue damage accumulation in structural materials of the equipment under simulated full-scale loading conditions, i.e.:

- pressure in the coolant circuit = 10 MPa;
- temperature range of the hot coolant flow (Th) = 283 °C–287 °C;
- temperature range of the cold coolant flow (Tc) = 35 °C–62 °C;

Table 1. Technical characteristics of instrumentation sensors.

Parameter	Sensor	Measuring range	Measuring error
Coolant temperature, °C	Thermocouple (KTHA 02.01-064-K1-I-S321-1-500/2000)	–40–700	±1
Coolant cold flow rate, kg/h	Flow rate converter (TPR 1-1-1)	10–36	0.5
Coolant pressure, MPa	Pressure sensor (S-10)	0–20	0.05
Coolant hot flow rate, kg/h	Differential pressure transducer (Yokogawa EJX110A)	300–800	15

- Reynolds number range (Reh) for the hot coolant flow 3.7·104–5.4·104;
- Reynolds number range (Rec) for the cold coolant flow 1·103–1.8·103;

The coolant flow rates were taken on the basis of the equality of the axial velocity components in order to create a quasi-stable vortex structure in the mixing region. T_h was selected on the basis of the maximum possible temperature gradients of coolants, characteristic of the reactor equipment exposed to thermal stresses. The resulting high-intensity temperature pulsations made it possible to obtain alternating stresses in the model material, providing fast rates of damage accumulation.

Coolant temperature pulsations in the MMTP were registered and recorded in the steady-state facility operation. The operating parameters were measured using transducers of temperature, pressure, differential pressure/temperature, and flow installed on the underwater sections of the experimental models. Technical characteristics of the sensors of the facility instrumentation complex are presented in Table 1.

Findings

The experimental temperature realizations (Fig. 2) determined the statistical (Figs 3, 4) and spectral-correlation (Fig. 5) characteristics of the tewmperature field in the mixing zone. In Figs 4, 5, the coordinates of the MMTP thermocouples were marked along the x-axis. As a defined zero, a point spaced 10 mm away from Thermocouple t1 location was taken.

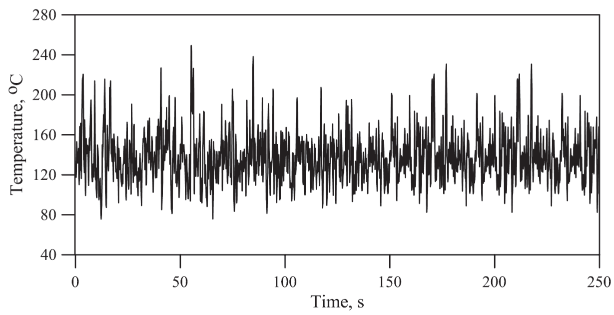

Figure 3. Experimental realization of the temperature-time dependence in the mixing zone.

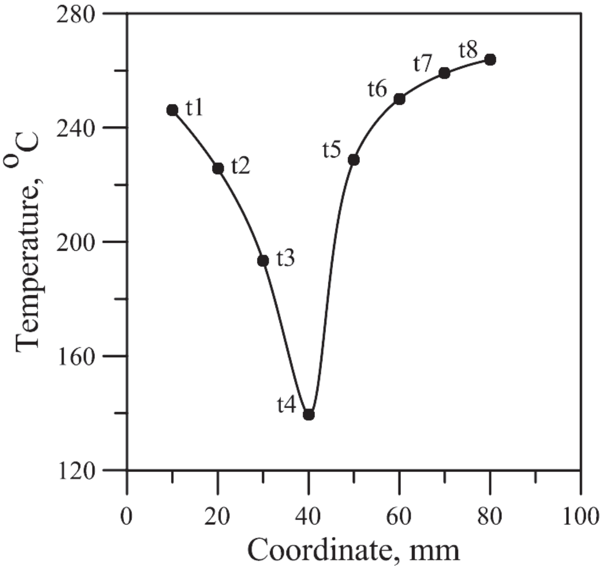

Figure 4. Averaged temperature profile in the coolant flow mixing zone.

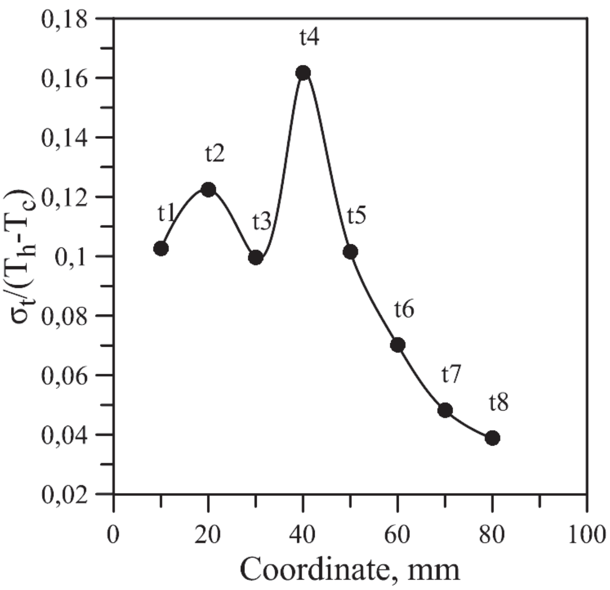

Figure 5. Relative intensity of temperature pulsations.

The maximum intensity of temperature pulsations (mean square deviation) σ_t (Chapuliot et al. 2005) was registered by Thermocouple t4. The largest range of thermal pulsations was 120 °C. The characteristic spectrum of temperature pulsations in the zone located under the peripheral pipe cut had a peak in the frequency range of 0.4–0.5 Hz.

According to the results obtained on the MMSSS, the stress-strain state of the sample was calculated. The maximum stress value in the loading area was 152 MPa.

The results presented in the work were obtained at T_h = 285 °C, $Re_h = 5.15 \cdot 10^4$, T_c = 35 °C, $Re_c = 1.08 \cdot 10^3$.

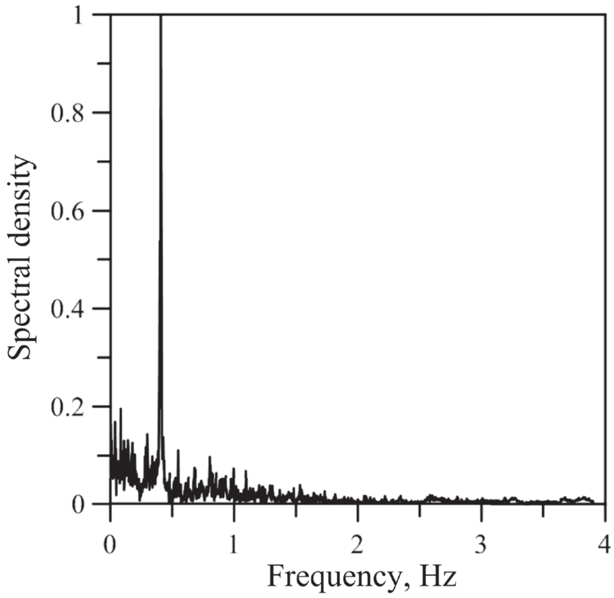

Figure 6. Normalized spectral density of temperature pulsations.

Conclusion

An experimental facility was designed and constructed to study the fatigue damage accumulation kinetics in the material of the experimental models as a result of loading by temperature pulsations due to mixed non-isothermal coolant flows. The loading of the experimental models with temperature pulsations was simulated with coolant parameters similar to the standard parameters of a nuclear reactor steam generator cassette. As a result of the experimental work,

- experimental data were obtained on the temperature field in the mixing zone; and
- the stress-strain state of the experimental sample was determined.

The results were included in the database to verify the method for assessing the fatigue life of structural materials for nuclear power plants as applied to austenite steel 12Cr18Ni10Ti under the influence of random thermal cyclic loads.

Acknowledgments

The study was carried out with a grant from the Russian Science Foundation (project No. 18-19-00473).

References

- Abib E, Bergholz S, Rudolph J (2013) German experiences in local fatigue monitoring. International Journal for Nuclear Power 58: 284–289.
- Beaufils R (2011) Analysis of the Father Experiment with an Engineering Method Devoted to High Cycle Thermal Fatigue. In: Proceedings of the ASME 2011 Pressure Vessels & Piping Conference (PVP 2011), July 17–21, Baltimore, Maryland. https://doi.org/10.1115/PVP2011-57630
- Braillard O, Edelin D (2009) Advanced experimental tools designed for the assessment of the thermal load applied to the mixing tee and nozzle geometries in the PWR plant. In: Advancements in Nuclear Instrumentation, Measurement Methods and their Applications, ANIMMA 2009, June 7–10, Marseille, France. https://doi.org/10.1109/ANIMMA.2009.5503718
- Budov VM, Dmitriev SM (1989) Forced heat exchangers of water cooled nuclear power unit. Energoatomizdat Publ., Moscow, 174 pp. [in Russian]
- Chapuliot S, Gourdin C, Payen T, Magnaud JP, Monavon A (2005) Hydro-thermal- mechanical analysis of thermal fatigue in a mixing tee. Nuclear Engineering and Design 235: 575–596. https://doi.org/10.1016/j.nucengdes.2004.09.011
- Chen MS, Hsieh HE, Ferng YM, Pei BS (2014) Experimental observations of thermal mixing characteristics in T-junction piping. Nuclear Engineering and Design 276: 107–114. https://doi.org/10.1016/j.nucengdes.2014.03.052
- Courtin S (2013) High Cycle Thermal Fatigue Damage Prediction in Mixing Zones of Nuclear Power Plants: Engineering Issues Illustrated on the FATHER Case. Procedia Engineering 66: 240–249. https://doi.org/10.1016/j.proeng.2013.12.079
- Dmitriev SM, Spiridonov DV, Vostrikov AA, Dmitrieva TS (2006) Nestacionarnoe temperaturnoe sostoyanie i ocenka dolgovechnosti teploobmennoj poverhnosti parogeneriruyushchego ehlementa s dvustoronnim obogrevom. Trudy RNKT-4. Moscow, 4: 88–91. [in Russian]
- Faidy C (2002) High Cycle Thermal Fatigue: Lessons Learned From Civaux Event. In: Materials Reliability Program: Second International Conference on Fatigue of Reactor Components (MRP-84), July 29–August 1, Snowbird, Utah.
- Jhung MJ (2013) Assessment of thermal fatigue in mixing tee by FSI analysis. Nuclear Engineering and Technology 45: 99–06. https://doi.org/10.5516/NET.09.2012.026
- Kamide H, Igarashi M, Kawashima S, Kimura N, Hayashi K (2009) Study on mixing behavior in a tee piping and numerical analyses for evaluation of thermal striping. Nuclear Engineering and Design 239: 58–67. https://doi.org/10.1016/j.nucengdes.2008.09.005
- Kuschewski M (2015) Development and application of flow measurement methods for the investigation of near-wall temperature fields. Doctoral dissertation, University of Stuttgart, No D93.
- Mahaffy J, Chung B, Dubois F, Ducros F, Graffard E, Heitsch M, Henriksson M, Komen E, Moretti F, Morii T, Mühlbauer P, Rohde U, Scheuerer M, Smith BL, Song C, Watanabe T, Zigh G (2007) Best practice guidelines for the use of CFD in nuclear reactor safety applications. NEA/CSNI/R(2007)5.
- Miyoshi K, Kamaya M, Utanohara Y, Nakamura A (2016) An investigation of thermal stress characteristics by wall temperature measurements at a mixing tee. Nuclear Engineering and Design 298: 109–120. https://doi.org/10.1016/j.nucengdes.2015.12.004
- NP-054-04 (2004) Norms for Calculating the Strength of Equipment Elements and Pipelines for Ship Nuclear Steam Generating Units with Water-Cooled Reactors. Rostekhnadzor Rossii Publ., Moscow, 57 pp. [in Russian]
- Smith BL (2010) Assessment of CFD codes used in nuclear reactor safety simulations. Nuclear Engineering and Technology 42: 339–364. https://doi.org/10.5516/NET.2010.42.4.339
- Smith BL, Andreani M, Bieder U, Ducros F, Graffard E, Heitsch M, Henrikkson M, Höhne T, Houkema M, Komen E, Mahaffy J, Menter F, Moretti F, Morii T, Mühlbauer P, Rohde U, Scheuerer M, Song CH, Watanabe T, Zigh G (2015) Assessment of CFD Codes for Nuclear Reactor Safety Problems – revision 2. OECD/NEA/CSNI/R(2014)12.
- Smith BL, Bestion D, Hassan Y (2010) Experiments and CFD Code Applications to Nuclear Reactor Safety (XCFD4NRS). Nuclear Engineering and Design (Special Issue) 240: 2075–2382. https://doi.org/10.1016/j.nucengdes.2010.06.037
- Sudakov AV, Trofimov AS (1980) Stresses at temperature pulsations. Atomizdat Publ., Moscow, 64 pp. [in Russian]
- Sudakov AV, Trofimov AS (1989) Pulsations of temperature and electrical equipment's component life. Energoatomizdat Publ., Leningrad, 179 pp. [in Russian]
- Wakamatsu M, Nei H, Hashiguchi K (1995) Attenuation of temperature fluctuations in thermal striping. Journal of Nuclear Science and Technology 32: 752–762. https://doi.org/10.1080/18811248.1995.9731770