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THE TEST FUEL ASSEMBLY FOR IN-PILE TESTS TO DETERMINE DESIGN SAFE OPERATION MARGINS

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ABSTRACT

A design of the test fuel assembly of MR type for in-pile tests to determine design safe operation margins is presented in the paper. The tests will be performed in the pulsed uranium-graphite reactor of IGR type in Semipalatinsk (Kazakh Republic) under emergency models of a loss-of-coolant accident and a reactivity initiated accident.

Nowadays, the problems related to responsibility growth and requirements for scientific approach to substantiation of nuclear safety in research reactor design and operation are of current interest. The urgency of the work is induced by the siting of such reactors either in the boundaries of large cities or near to them as well as by license regulations.

The safety problem is a multipurpose one, its solution should be based on a deep understanding of steady and transient thermal, hydraulic, physical, radiation and other phenomena and processes which proceed in a reactor core.

For substantiation of reactor safety it is necessary to have experimental data on the behaviour of fuel elements and fuel assemblies in all postulated accidents and in a number of hypothetic ones for the purpose to use this information in calculation codes for analysis of accident progression and consequences as well as for development of safety standards and requirements to control and safety systems. The above approach is considered to be generally accepted presently [1-6].

Aluminium is a basic structural material for research reactor fuel elements and assemblies which are under development in the context of nuclear fuel enrichment reduction.

Low melting point of the material necessitates studies into accident conditions related to superheating of fuel elements.

The most important accidents are the following ones: a loss-of-coolant flowrate (LOCFR), that is, coolant flow termination resulted from a loss of power supply in circulation pumps or from pipe rupture; a reactivity initiated accident (RIA), that is, a sharp reactor power increase due to unauthorized introduction of positive reactivity.

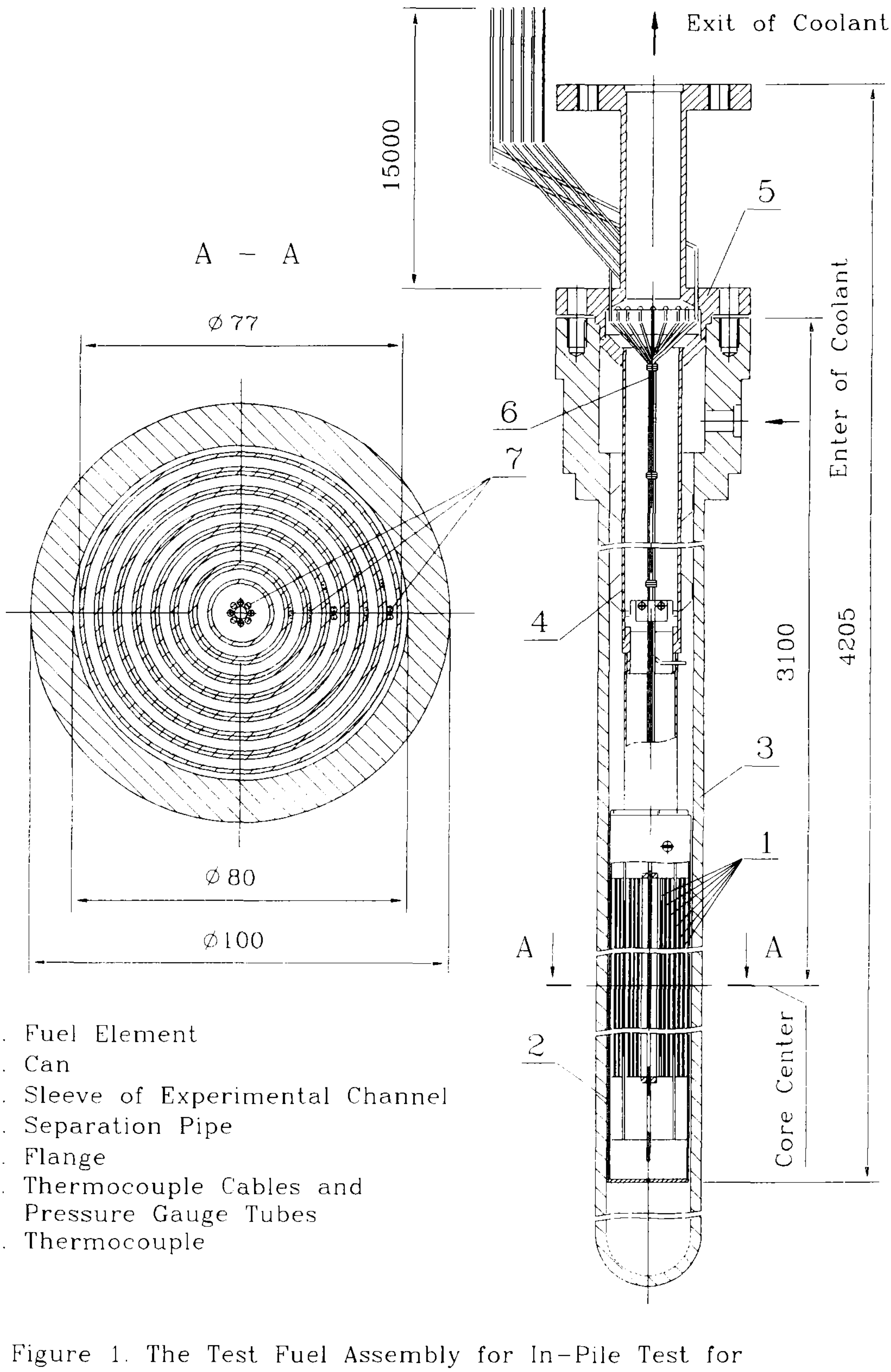
In both cases, emergency protection system actuation is bound to occur. However, it takes some time to form a signal, to actuate automatics, and to introduce emergency shutdown rods. Fuel element temperature should not exceed permissible level in this interval of time.

For investigation of these accident conditions for fuel assemblies of MR type with the fuel enriched in uranium-235 by 19.7 % researches are scheduled at the pulsed uranium-graphite reactor of IGR type in Semipalatinsk (Kazakh Republic).

Similar researches of the fuel assemblies of IVV-2M type with the fuel of 90 % enrichment were conducted at this reactor in 1990 [7,8]. Operating testing loop was designed and experimental technique was worked. The same facility and the same technique are planned to be used for testing fuel assemblies of MR type.

Since the MR-reactor is of channel type with cooling by pressurized water according to Field's scheme, its fuel assembly is well arranged in experimental channel of the IGR reactor. At the same time, the possibility exists for accurate simulation of coolant parameters (pressure at the fuel assembly inlet, flow rate, velocity, temperature).

The IGR-type reactor also allows one to provide needed thermal power of MR-type fuel assembly and similar power density distribution among fuel elements. Transient character of processes and a wide range of operating parameter variation are specific features of the tests. Automated quick-response system of wide-range instrumentation was designed and manufactured to monitor neutronic parameters. The assurance of synchronous space and time monitoring of thermal neutron flux within the range of 10^4 - 10^{17} $cm^{-2}s^{-1}$ in the central experimental channel, computer data acquisition and processing, the determination of fuel assembly power shape and amplitude are among prime objects in the design of the system. Polling freaquency of all detectors may be as much as 100 Hz when recording information from short reactor pulses up to 0.1 s [9].



1. Fuel Element
2. Can
3. Sleeve of Experimental Channel
4. Separation Pipe
5. Flange
6. Thermocouple Cables and Pressure Gauge Tubes
7. Thermocouple

Figure 1. The Test Fuel Assembly for In-Pile Test for Determining Design Safe Operation Margins

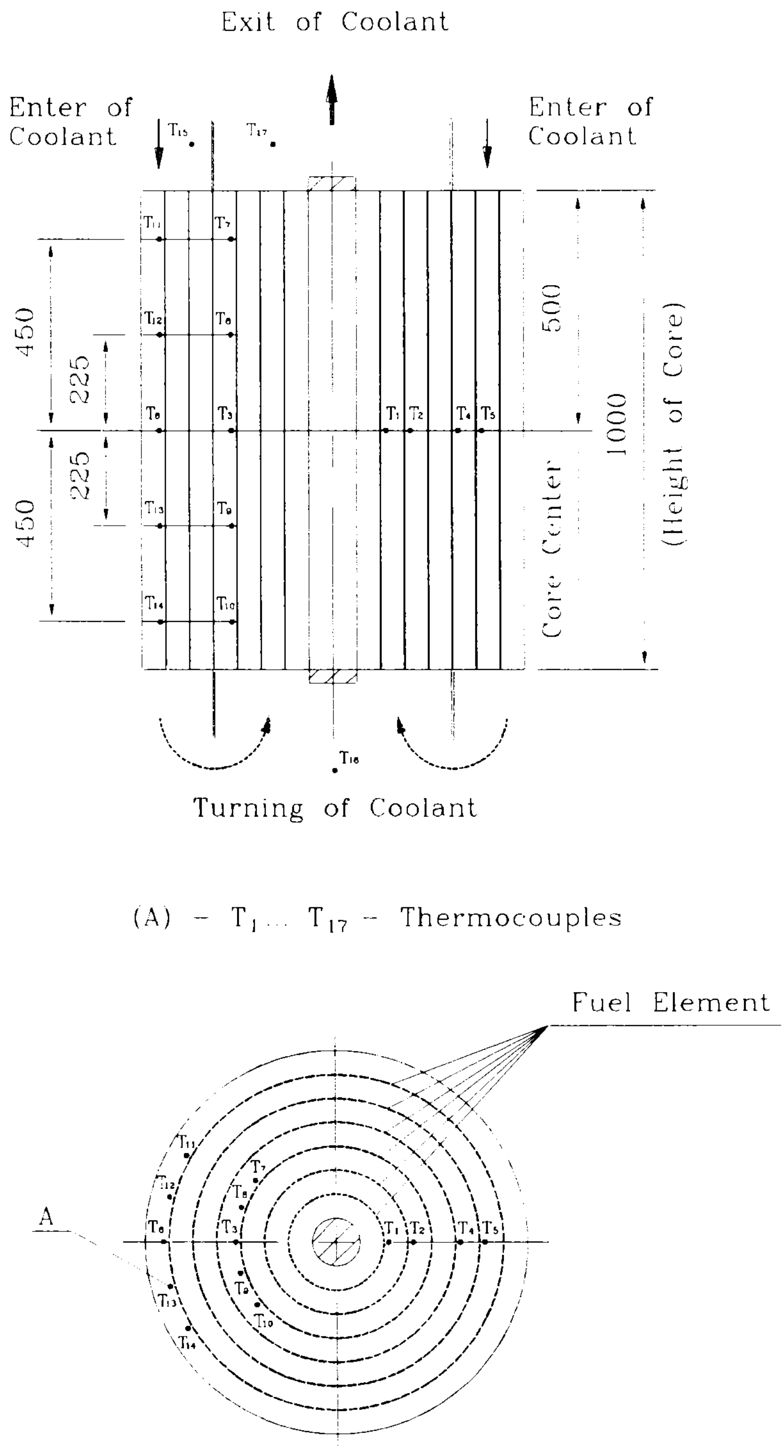


Figure 2. Position of Thermocouples in the Fuel Elements.

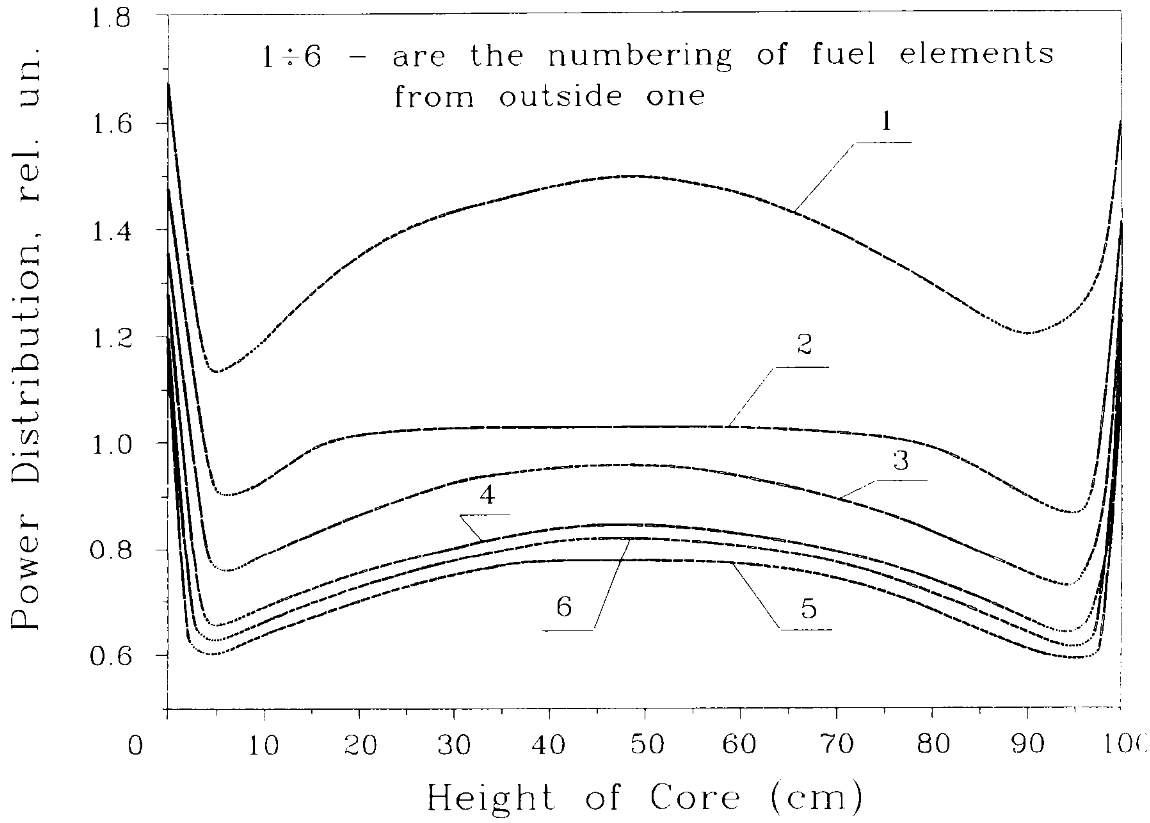


Figure 3. Results of Neutronic Calculation

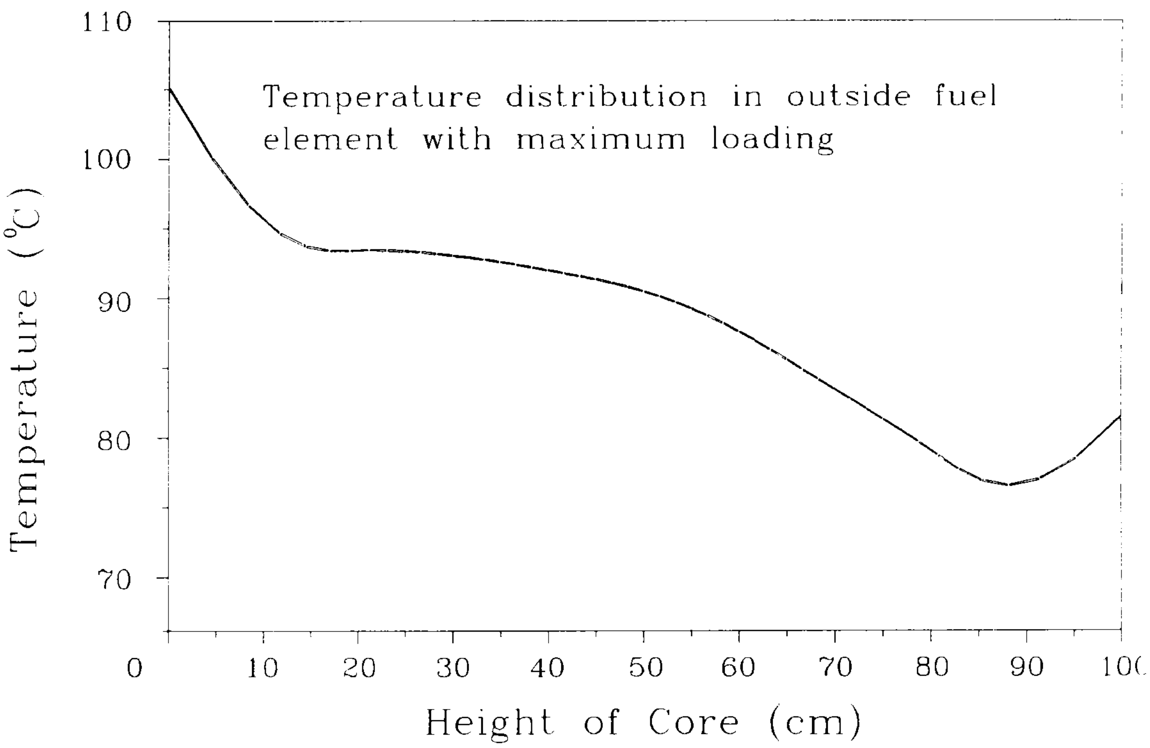


Figure 4. Results of Thermohydraulic Calculation

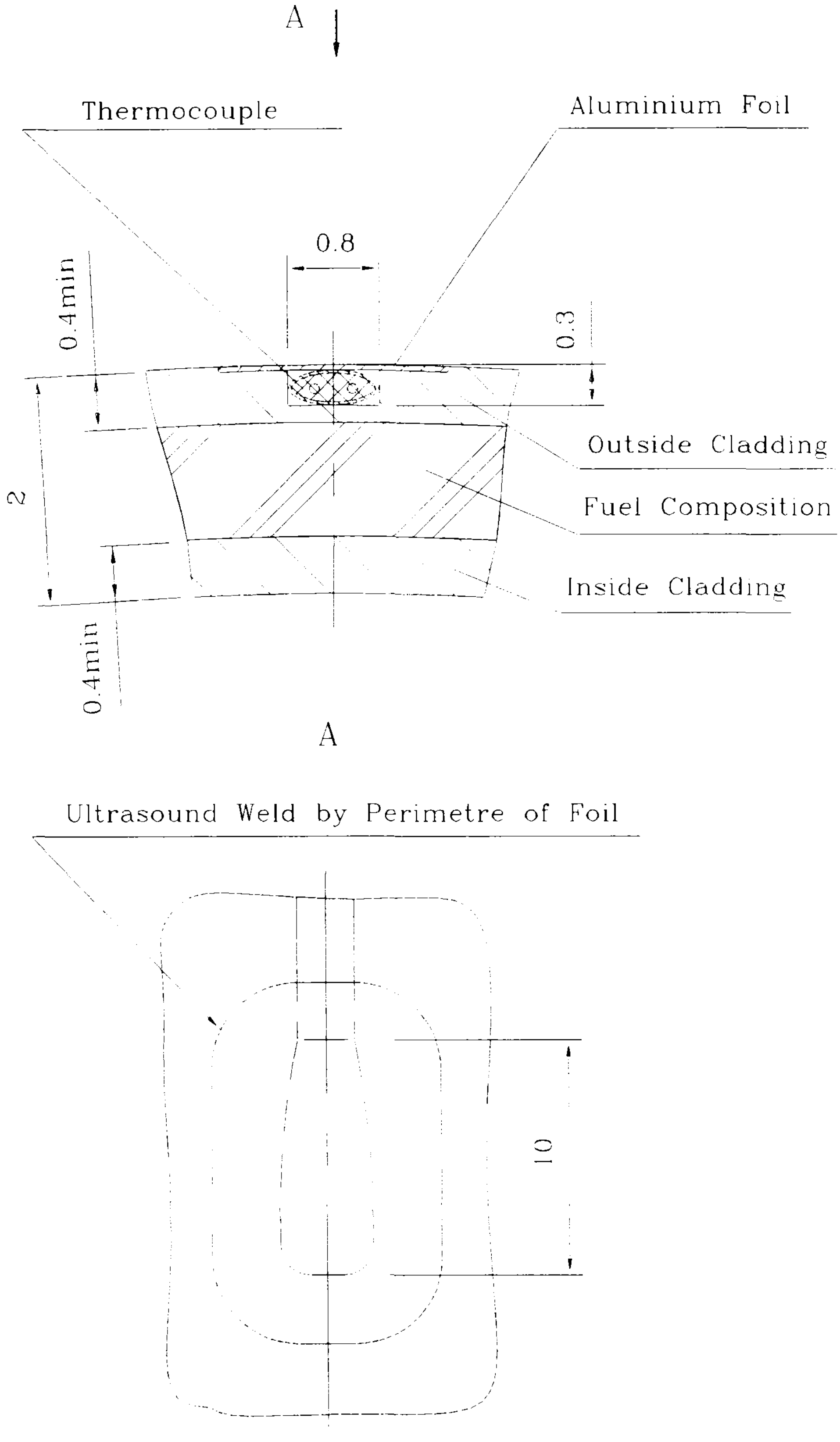


Figure 5. Position of Thermocouple in Fuel Element

Experimental fuel assembly is similar in design to standard MR-type fuel assembly that operates in MARIA reactor in Poland. Some changes refer to end section and top fitting but they have no effect on hydraulic characteristics of fuel assembly.

Experimental MR-type fuel assembly (see Fig.1) consists of six concentrically arranged fuel elements of round section, a top fitting, an end section and a displacer. The third from the outside fuel element where the separation of coolant flows occurs is a supporting one, ending parts of fuel assembly are fixed to it. Fuel elements are spaced in top fitting and in end section using combs and are free to move at a distance of expansion gap.

The coolant first passes from top to bottom through the slits on the outside of supporting fuel element, is mixed and then passes in the opposite direction through the inner slits.

Experimental fuel assembly is arranged in a can with inner diameter equal to that of fuel channel of MR-type reactor. Experimental fuel assembly is suspended in a sleeve of experimental channel with the help of separating pipe which serves as a continuation of separating fuel element.

The separating pipe is welded to a flange which serves for suspension of the all experimental unit in a sleeve of experimental channel. Thermocouple cables and pressure gauge tubes are brought out through holes in the flange. With the help of upper flange the experimental section is coupled with cooling system.

To measure fuel cladding temperature the experimental fuel assembly is equipped with thermocouple in the form of chromel-alumel thermoelectric converters 1.5 mm in diameter with an insulated thermocouple junction. The diameter of thermocouple at the ends decreases down to 0.5 mm, and thermal junction is flattened down to the thickness of 0.3 mm. In all, 17 thermocouples are mounted in fuel assembly (see Fig.2). Thermocouples are positioned in fuel elements in accordance with neutronic and thermohydraulic calculation results presented in Figs.3
and 4.

Junctions of thermocouples are ultrasound welded into fuel claddings using milled grooves in them (see Fig.5). In such a case, fuel element tightness is preserved.

Thermocouples on fuel elements are arranged as follows:

- one sensor for every fuel element in a center of active part;

- for two fuel elements with the most high temperatures expected, thermocouples are distributed along the height of active part.

To measure coolant pressure the tubes of 2 mm inner diameter are used. The tubes are located at the inlet and at the outlet of fuel assembly as well as in area of coolant turn.

Presently, two such experimental fuel assemblies are under manufacture. The tests are scheduled in early 1996.

References

1. CANDU fuel behaviour in severe fuel damage conditions /Lau J.H.K., Blahnik C., Akalin O.// Nuclear Power Performance and Safety: Proc. Int. Conf., Vienna, 28 September - 20 October. Vienna, 1988. V.4. P.257-273.
2. Andreev V.I. and others. Methodical aspects of study into fuel element behaviour under transients// Atomnaya tekhnika za rubezhom. 1985. No 3. P.3-7 (in Russian).
3. Burukin V.P., Klinov A.V., Toporov Yu.G. Foreign programs of reactor researches in emergency and transient operating conditions of nuclear power installation// Atomnaya tekhnika za rubezhom. 1989. No 5. P.3-7 (in Russian).
4. Arkhangel'skij N.V., Dikarev V.S., Egorenkov P.M., Ryazantsev S.P. Enhancement of research reactor safety// Atomnaya energiya. 1988. Vol.64. Iss.5. P.331-338 (in Russian).
5. Belen'kij I.Ya., Gotovskij M.A., Kirsanov G.A., Fokin B.S. Experimental study of thermophysical state in PIK reactor core at sharp pressure reduction// Voprosy atomnoj nauki i tekhniki. Seriya: Fizika i tekhnika yadernykh reaktorov. 1990. Iss.1. P.34-38 (in Russian).
6. Burukin V.P., Klinov A.V., Toporov Yu.G. Reactor installations to test fuel elements and fuel assemblies under emergency and transient operating conditions// Atomnaya tekhnika za rubezhom. 1988. No 6. P.7-15 (in Russian).

7. Kartashev E.F., V.F.Lisovoj, A.V.Sokolov and others. Tests of the IVV-2M fuel assemblies under pulsed loading and coolant supply failure conditions// Voprosy atomnoj nauki i tekhniki. Seriya: Yadernaya tekhnika i tekhnologiya. 1992. Iss.4. P.56-67 (in Russian).
8. V.G.Aden, E.F. Kartashov, V.A. Lukichev and others. In-pile tests substantiating fuel enrichment reduction in research reactors// Proceedings of the 16th International Meeting on REDUCED Enrichment for Research and Test Reactors. October 4-7, 1993, Oarai, Japan// March 1994. Editorial Working Group. Department of JMTR Project. Japan Atomic Energy Research Institute. JAERI-M, 94-042, p.183-190.
9. V.V.Gusev, A.I.Efanov, S.I.Kryukov, V.M.Malinkin. Automatic on-line physical parameters monitoring system during dynamic tests of fuel elements and fuel assemblies at the IGR reactor// Proceedings of the Topical Meeting on Computational and Experimental Validation of Nuclear Power Safety and Fuel Cycle Investigations. Moscow, s/c "Volga", 5-9 September, 1993. Vol.1. P.189-191.