Thermal-hydraulic conceptual design of the new multipurpose research reactor succeeding to JRR-3

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ABSTRACT

The core thermal-hydraulic conceptual design of the new multipurpose reactor succeeding to JRR-3 was performed for the forced convection cooling modes. The key criteria are first to avoid the nucleate boiling anywhere in the core and second to have enough safety margin to departure from nucleate boiling for normal operation conditions.
The results of the thermal-hydraulic conceptual design and analysis show that the optimum coolant velocity in the standard fuel element is about 6.0 m/s with the core inlet pressure of 1.49 kg/cm^2 abs at 30 MW of thermal power. At the coolant velocity of 6.0, the temperature margin against ONB temperature is 7.2 °C and the minimum DNBR is 2.3. The results obtained in this work establish the preliminary technical specifications for the core thermal-hydraulic design of the new multipurpose research reactor.

1. Introduction

Japan Research Reactor No.3 (JRR-3), which is one of the largest multipurpose research reactors in Japan, is a light water cooled and moderated pool type reactor with thermal power of 20MW at Japan Atomic Energy Agency (JAEA). JRR-3 is utilized from the research to industrial use as neutron beam experiments; irradiation tests of the reactor material, manufacturing radio isotopes for medicine use, and the silicon semiconductor by neutron transmutation doping (NTD).
However, the operation of JRR-3 was begun from 1990 and the aging problems are becoming apparent. Moreover, taking the neutron utilization after the stop of JRR-3 into account, the basic design of the new multipurpose research reactor should be investigated.
Our working group has started to look into basic concepts of the new research reactor that will be accepted as a successor of JRR-3. Our vision is to maintain national neutron source for neutron beam experiment, material irradiation, RI production and training et al., and our purpose is to develop a new research reactor to stably supply the specified neutron flux and neutron spectra that satisfies the needs of stakeholders.
As the first step in design study of the new multipurpose reactor, the thermal power, the basic

shape of the reactor core, the fuel element design and the reflector component were proposed to gain high neutron flux and satisfy the safety levels [1]. The new research reactor is desirable to be the pool type reactor because irradiation samples can be handled easily and the reactor facility system is simplified. Generally speaking, in the case of the thermal power density is raised for high neutron flux, the reactor type needs to be the tank type for pressurizing the reactor pool to prevent water around the core from boiling locally. However the new reactor is assumed to be able to increase thermal power by flattening power density and improving cooling performance. Thus the temporary power of it is set to 30MW and the reactor type is defined as the pool type. The reactor core will be surrounding by D_2O reflectors and aluminum irradiation holder. The neutron flux of the new reactor became much higher than JRR-3 similarly; the thermal neutron flux was over 1.5 times at the region of D2O reflectors, the fast neutron flux was over 2.9 times at the region of aluminum irradiation holder.

As the second step, this report presents the preliminary core thermal-hydraulic characteristics and safety margins for the beginning of cycle core configuration for three different core inlet pressures and various core flow rates.

2. Description of the reactor

The new rector core conceptual design will be utilized 20% enriched uranium fuel in the form of U-7Mo which is being investigated in each country. Figure 1 shows the core layout of the new research reactor. The core will comprise of 22 standard fuel elements and 4 control fuel elements on a 7×5 grid. Since the reactor is assumed to be operated under high thermal power density, the plate type fuel will utilized. Each standard and control fuel element will contain 17 fuel plates. Fuel meat dimensions for the proposed fuel are 63.5 mm × 0.51 mm × 800.0 mm. The cooling channel gap between fuel plates will have a thickness of 3.3 mm.

The reactor will be operated at a thermal power of 30 MW and will use demineralized light water as coolant. The heat generated in the core will be removed by a primary cooling system. For the core thermal-hydraulic conceptual design of new reactor, a forced convection cooling mode for high power operation is specified. As flow direction in the core for forced-convection cooling mode, downward flow is adopted in the view pint of attenuation of ^{16}N.

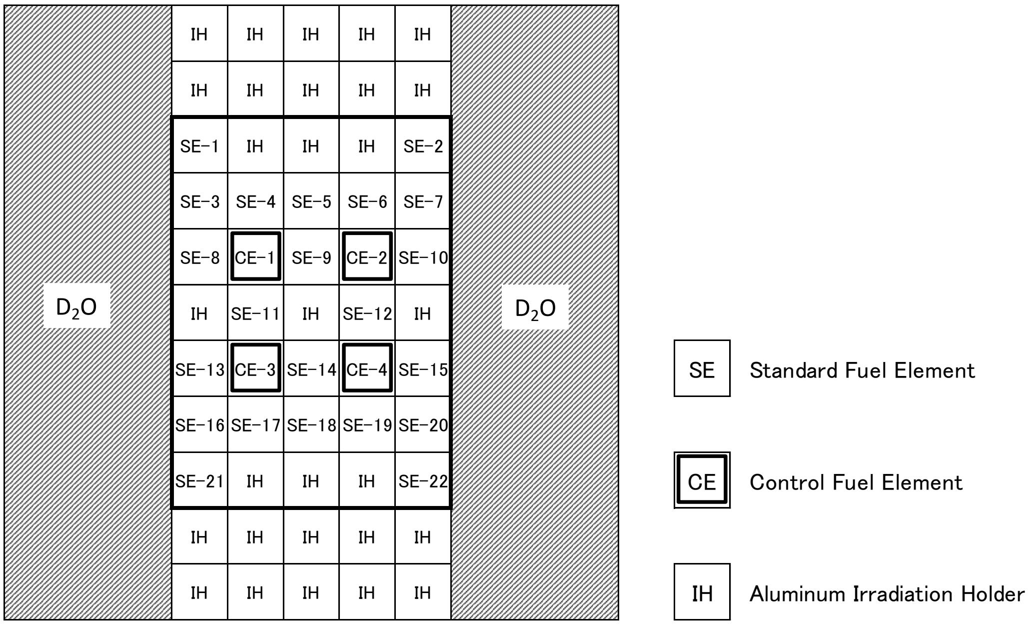

Figure 1: Core layout of the new multipurpose research reactor

3. Core thermal-hydraulic conceptual design

For the core thermal-hydraulic conceptual design of new research reactor, the design bases and the procedure applied for the JRR-3 were also adopted.

Two major design criteria were set up for the core thermal-hydraulic design of new reactor so that fuel plates may have enough safety margins for the conditions of normal operation. One is to avoid nucleate boiling of coolant anywhere in the core in order to give enough allowance against the burnout of the fuel plate even at the hottest spot in the core. Moreover it is also to avoid any flow instability by partial nucleate boiling in order to obtain stable neutron fluxes for experiments. For this criterion, the allowance in surface temperature of the fuel plate for the onset of nuclear boiling (ONB) temperature was evaluated at the hottest spot in the core, using hot channel model.

The other is to give enough margins against the burnout itself of the fuel plate under the normal operation condition so that there may be enough margins also for operational transients. The departure from nucleate boiling ratio (DNBR) was decided to be not less than 1.5 to meet the latter criterion. DNBR is the ratio of the heat flux at the departure from nucleate boiling (DNB) to the maximum heat flux, and the criterion is derived by DNB heat flux correlations which were developed by JAEA [2].

The coolant velocity in the standard fuel element which has the hottest spot is determined to give the maximum temperature margin of the fuel plates against the ONB. At the same time the DNBR is calculated under the condition of coolant velocity thus obtained and it is confirmed that the DNBR is larger than 1.5. A core inlet pressure is selected as design parameters.

4. Calculations and analysis

4.1 Calculation code

The thermal-hydraulic calculations and analysis were carried out using the computer code COOLOD-N2 [3]. The COOLOD-N2 calculates local bulk temperature of coolant and the fuel plate surface heat fluxes, local coolant velocities and an outlet coolant temperature of the sub-channel in which the hottest spot exists. It also calculates the ONB temperature, the DNB flux, saturation temperature and pressure of coolant at every distance from the inlet of sub-channel, to evaluate the temperature margin against the ONB and heat flux margin against the DNB along the fuel plate which includes the hottest spot. The COOLOD-N2 code has been developed by JAEA for the steady state thermal-hydraulic analysis of research reactors in which plate-type fuel is employed. Thermal-hydraulic analysis of JRR-3 has been performed using the code.

4.2 Temperature calculation model

Fuel plates temperature are calculated considering one dimensional heat conduction in radial direction of fuel plates and assuming that the heat generation in fuel meat is constant along the radial direction. In the temperature calculation model for forced convection cooling mode, a coolant flow rate is given by input data. Fuel plates and coolant channel are divided into 20 segments in axial direction. The coolant is heated from both sides, and the flow direction is downward.

4.3 Pressure drop calculation model

COOLOD-N2 can deal with the calculation of fuel elements having a complicated shape inlet-outlet flow path. A pressure drop calculation model of the fuel element for a forced convection cooling mode was made taking the pressure loss due to friction and the pressure loss due to the change in the shape of the flow channel. Pressures of 1.49, 1.69 and 1.88 kg/cm^2 abs and the temperatures of 35 °C at the core inlet were used in the calculation.

4.4 Radial and axial peaking factors

The radial peaking factors were obtained from neutronic calculation using the MVP-BURN code [4] and the maximum radial peaking factor is at the position of standard fuel element SE-3, SE-7, SE-16 and SE-20. These positions were selected as the hottest channel. Figure 2 shows radial peaking factor calculated for beginning of the reactor cycle (BOC) core configuration.

Axial peaking factors in fuel elements were also determined by the neuronic calculation. The axial peaking factors were calculated based on the power distribution at the fueled region SE-3 of the BOC core configuration. The Figure 3 shows the axial peaking factors distribution used in the thermal-hydraulic calculations.

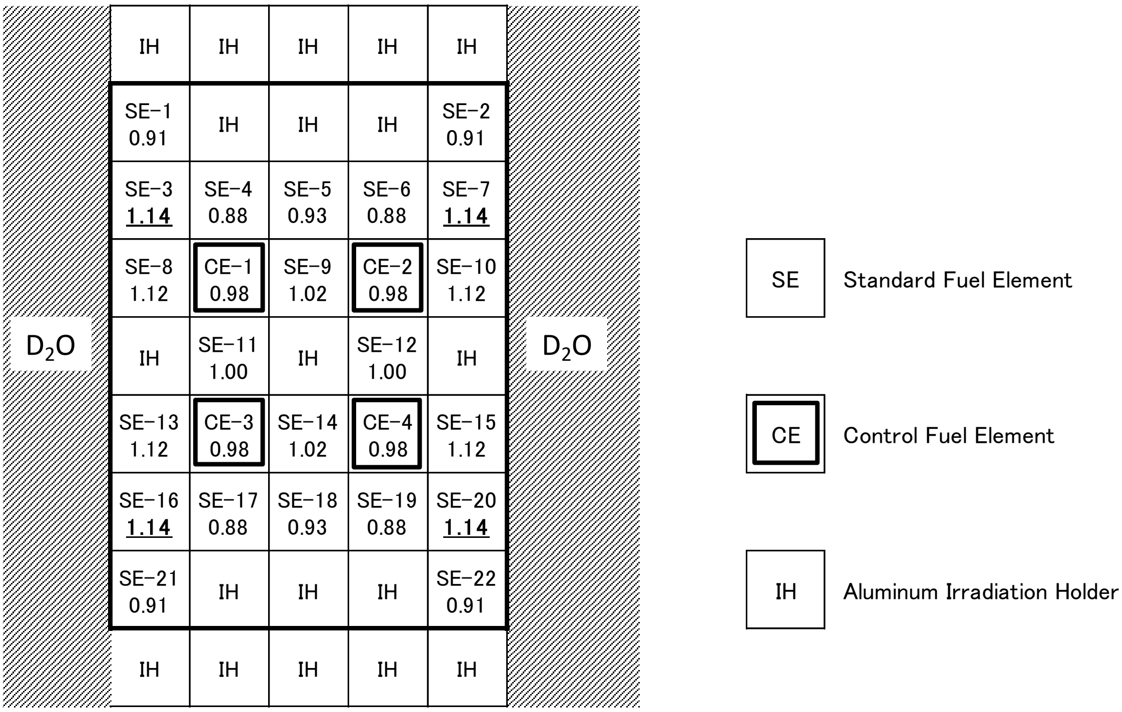

Figure 2: Radial peaking factors distribution at BOC core configuration

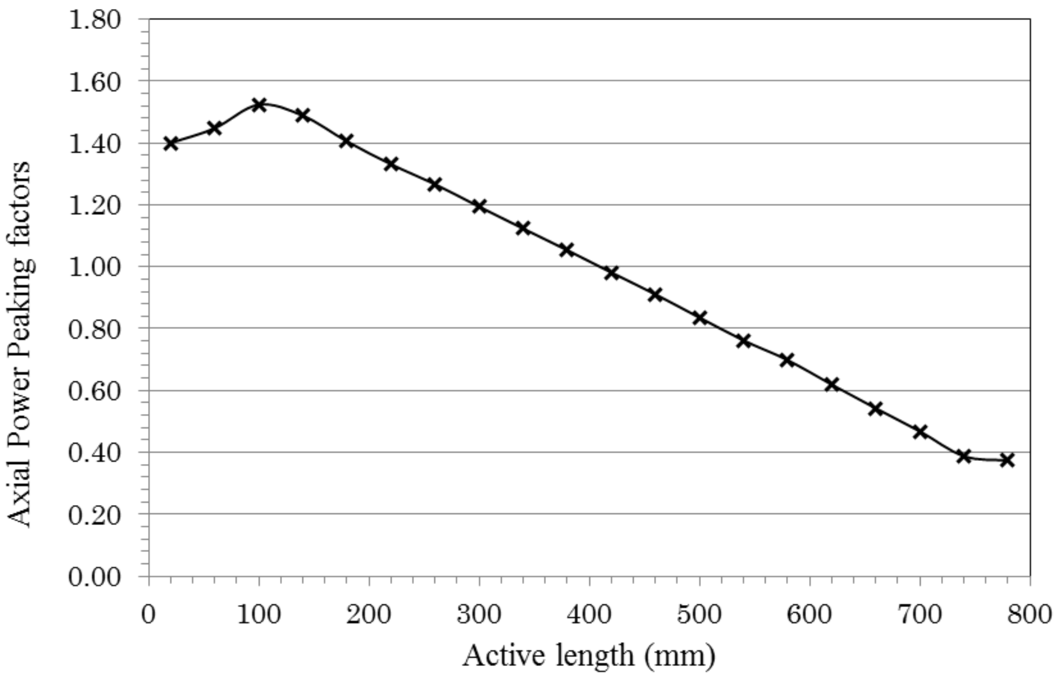

Figure 3: Axial peaking factors distribution used in the thermal-hydraulic calculation

4.5 Flow rate through fuel element

The performances of the proposed core under 30 MW operating condition, were evaluated under various flow rates through core ranging from 400 kg/s to 1100 kg/s which correspond to the core inlet temperatures of 35 °C, to determine optimum effective flow rate through the standard fuel elements.

For the thermal-hydraulic calculations, it was assumed that the flow rate through standard and control fuel elements to be the same. It was also assumed that the effective flow rate through the core to be produced only by the flow rates through all standard and control fuel elements in the core.

4.6 Thermal power and hot channel factors

The thermal power was fixed to 30 MW by the neutronic calculations. For these thermal-hydraulic calculations, it is assumed that all the thermal power is produced only by the fuel elements. The average power generated per fuel element was calculated as the ratio of the net heat generated in the core to number of effective fuel elements in the core.

The nuclear factors, radial and local were determined by the neutronic calculations. The engineering hot channel factors which are determined with manufacturing tolerance of the fuel element, error of heat transfer correlation adopted and uncertainties of parameter affecting the bulk coolant temperature rise of coolant and the film temperature rise were assumed equal to engineering hot channel factors of the JRR-3 reactor. Table 1 shows the nuclear factors and engineering hot channel factors used in the calculations.

Table 1: Hot channel factors

Nuclear factors	
Radial (F_R)	1.14
Local (F_L)	1.30
Engineering hot channel factors	
Bulk coolant temperature rise (F_b)	1.32
Film temperature rise (F_f)	1.36
Heat flux (F_q)	1.16

5. Results and discussion

5.1 Calculation results

Thermal-hydraulic calculations were carried out for the forced convection cooling mode at the reactor thermal power of 30 MW with three different core inlet pressures of 1.49, 1.69 and 1.88 kg/cm^2 abs and core inlet temperature of 35 °C under various flow rates through the core ranging from 400 kg/s to 1100 kg/s. Calculation results of the ONB temperature (T_{ONB}) and fuel plate surface temperature (T_W) at the hottest spot where the difference between T_{ONB} and T_W is a minimum in the hot channel are given in Figure 4 as a function of coolant velocity in the channel. These temperatures are shown as a function of the coolant velocity because the coolant velocity is only the dominant variable to fuel plate surface temperature, once the pressure and temperature at the core inlet are fixed. And it can be observed that T_{ONB} decrease with an increase of the coolant velocity because an increase in the coolant velocity gives lower local pressure according to the increase of pressure loss.

The temperature margin against the ONB as a function of the coolant velocity is shown in Figure 5 for each of the three pressures at the core inlet. Figure 5 shows that an increase of the coolant velocity gives a higher temperature margin against the ONB temperature. In the range of coolant velocities shown in Table 2, no boiling occurs in the hot channel. If the coolant velocities are greater than approximately 8.3 to 9.3 m/s, however, the pressure at fuel plate exit becomes negative by the increase of pressure loss. On the other hand, two-phase flow occurs with nucleate boiling at coolant velocities less than approximately 5.3 m/s, 5.0 m/s and 4.8 m/s for the core inlet pressure of 1.49 kg/cm^2, 1.69 kg/cm^2 and 1.88 kg/cm^2, respectively.

Table 2: Range of coolant velocities in which $T_{ONB} > T_W$

Core inlet pressure (kg/cm^2 abs)	Range of coolant velocities (m/s)
1.49	5.3 to 8.3
1.69	5.0 to 8.8
1.88	4.8 to 9.3

Figure 6 shows the departure from nucleate boiling ratio (DNBR) and the onset of flow instability ratio (OFIR) as a function of coolant velocity. For the reactor design purpose, acceptable data on burnout heat flux are needed since DNB is potentially a limiting design constraint.

On the other hand, OFIR is the ratio of the heat flux at the onset of flow instability (OFI) to the maximum heat flux. Flow Instability referees to flow oscillations of constant or variable amplitude that are analogous in mechanical system. Flow oscillations are undesirable for several reasons: First sustained flow oscillations may cause undesirable forced mechanical vibration of components, second, flow oscillations may cause system control problems, which are of particular importance in water cooled reactors where the coolant also acts as moderator; third, flow oscillations affect the local heat transfer characteristics and the boiling crisis. The burnout heat flux under unstable flow conditions may be well below the burnout heat flux under stable flow conditions. Thus for plate type fuel design, the critical heat flux that leads to the OFI may be more limiting than of stable burnout.

In figure 6, it can be observed that an increase of the coolant velocity gives higher safety margins of DNBR and OFIR.

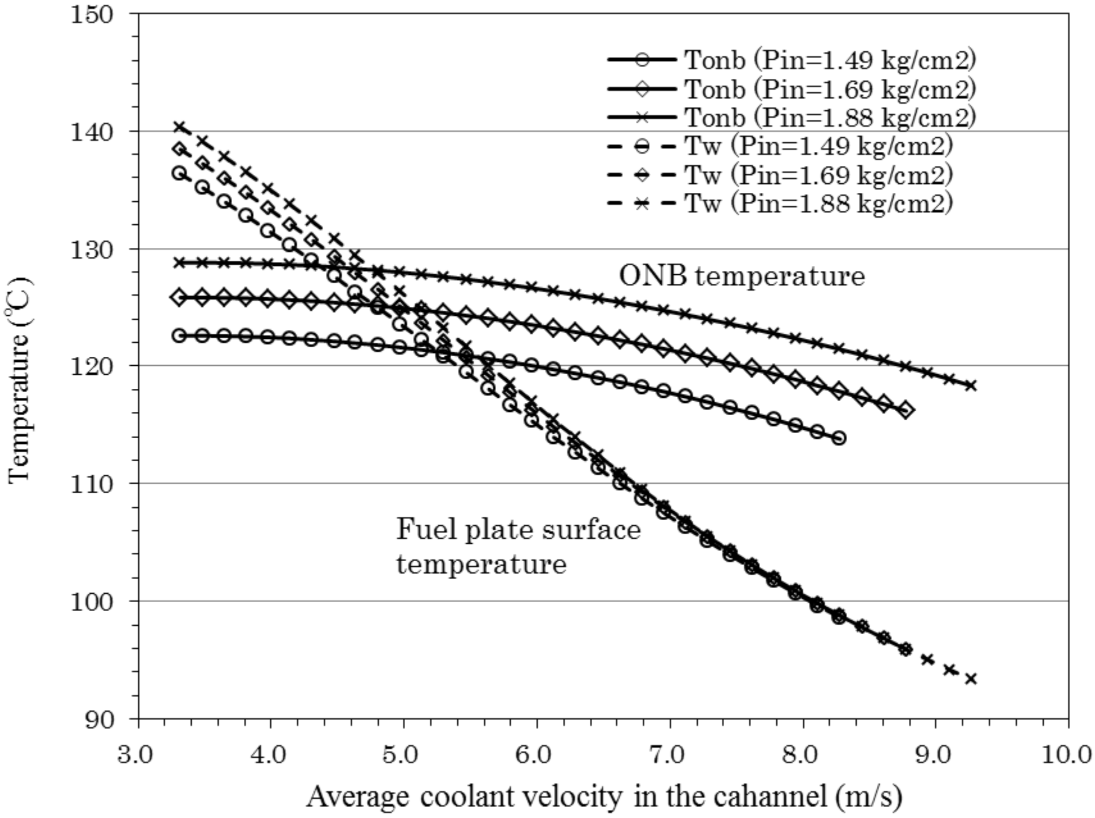

Figure 4: ONB temperature (T_{ONB}) and fuel plate surface temperature (T_W) vs coolant velocity for the standard fuel element at 30 MW of thermal power.

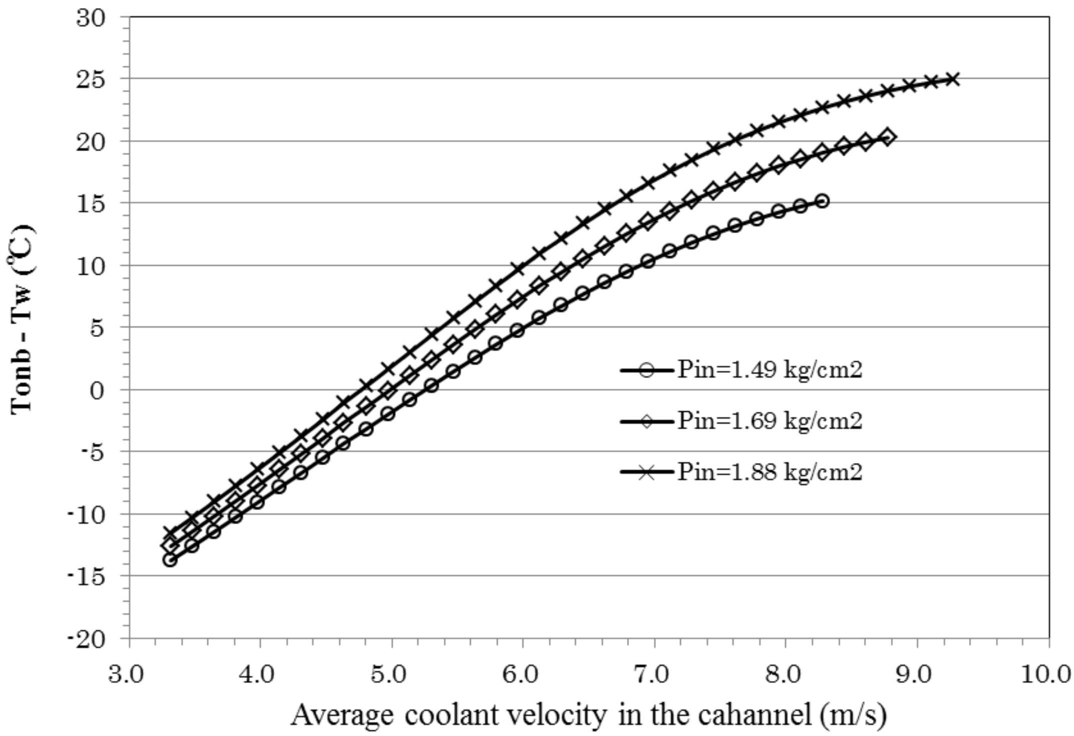

Figure 5: Temperature margin against the ONB temperature vs coolant velocity for the standard fuel element at 30 MW of thermal power.

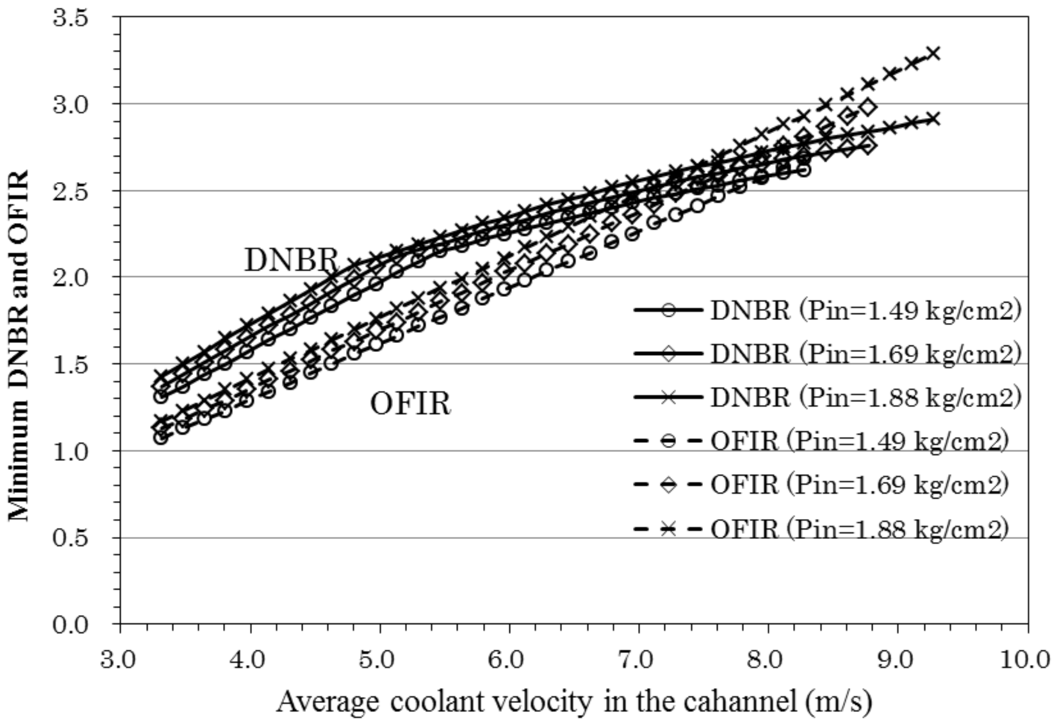

Figure 6: DNBR and OFIR vs coolant velocity for the standard fuel element at 30 MW of thermal power.

5.2 Discussion of Core thermal-hydraulic conceptual design parameters

Generally the research reactor with plate-type fuels are operated under the condition of no nucleate boiling of the coolant anywhere in the core in order to give enough allowance against the burnout of the fuel plate even at the hottest spot in the core, to avoid any flow instability induced by partial boiling in the core and to obtain stable neutron fluxes for experiments.

ONB is taken as a limit for single-phase cooling and is not a limiting criterion in the design of a fuel element. The heat transfer regime should be clearly identified for proper hydraulic and heat transfer considerations i.e. single-phase flow. The nucleate boiling occurs at a fuel plate surface temperature over ONB temperature.

Operational transients and accidents analysis should be carried out based on steady-state thermal-hydraulic calculat1ations results. During an operational transient, DNBR and OFIR will decrease due to a decrease of coolant flow rate and/or an increase of the reactor power. So, the steady-state condition of the research reactors should have enough safety margins.

From the calculation results for the hot channel described before, the coolant velocity of 6.0 m/s is proposed as conceptual design velocity for the proposed fuel elements. As seen in Figure 5, Temperature margin against ONB temperature (ΔT_{ONB}) increases linearly with increase of coolant velocity up to approximately 6.0 m/s for each of core inlet pressures and after that ΔT_{ONB} increase ratio is decreases because of increase of core pressure drop. The coolant velocity of 6.0 m/s is the maximum coolant velocity in the linear part of ΔT_{ONB} vs coolant velocity and that is why 6.0 m/s is selected as the conceptual design velocity for the standard fuel elements. The coolant velocity of 6.0 m/s corresponds to a flow rate of 720 kg/s (2600 m^3/h). At the coolant velocity of 6.0 m/s, ΔT_{ONB} are calculated to be 4.7 °C, 7.2 °C and 9.7 °C for the core inlet pressure of 1.49 kg/cm^2, 1.69 kg/cm^2 and 1.88 kg/cm^2, respectively. These ΔT_{ONB} are enough for the steady-state condition of new research reactor from safety point of view.

The minimum DNBRs obtained with the coolant velocity of 6.0 m/s are 2.25, 2.30 and 2.34 for the core inlet pressure of 1.49 kg/cm^2, 1.69 kg/cm^2 and 1.88 kg/cm^2, respectively. With the three different core inlet pressures, it was confirmed that these DNBRs were larger than 1.5 and enough safety margins against DNB were obtained for 30 MW of thermal power.

A core inlet pressure of 1.69 kg/cm^2 abs is proposed as a conceptual design core inlet pressure. With the core inlet pressure of 1.69 kg/cm^2 abs, the top of core will be located at 7 m below the surface of the water of the reactor pool and the reactor pool depth will be about 10 m. A core inlet pressure of 1.88 kg/cm^2 abs is also proposed. With this pressure, the top of core will be located at about 9 m below the surface of the water of the reactor pool and the reactor pool depth will be about 12 m. On the other hand, in case a core inlet pressure is 1.49 kg/cm^2 abs, the top of core will be located at 5 m below the surface of the water of the reactor pool and the new reactor may have a problem on the shielding design.

6. Conclusions

The core thermal-hydraulic conceptual design of the multipurpose research reactor succeeding to JRR-3 was performed for forced convection cooling mode. The coolant velocities in the standard fuel element which have the hottest spot were determined to give the maximum temperature margin of the fuel plates against the ONB temperature with three different core inlet pressures. At the same time, the DNBRs were calculated under the condition of coolant velocity thus obtained and it ware confirmed that the DNBRs were larger than 1.5.

The results obtained in this work established the preliminary technical specifications for the core thermal-hydraulic design of the new research reactor.

References

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