THERMALHYDRAULIC EVALUATIONS FOR A CANFLEX BUNDLE WITH NATURAL OR RECYCLED URANIUM FUEL IN THE UNCREPT AND CREPT CHANNELS OF A CANDU-6 REACTOR

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The thermalhydraulic performance of a CANDU-6 reactor loaded with various CANFLEX fuel bundles is evaluated by the NUCIRC code, which is incorporated with recent models of pressure drop and critical heat flux (CHF) predictions based on high-pressure steam-water tests for the CANFLEX bundle as well as a 37-element bundle. The distributions of channel flow rate, channel exit quality, critical channel power (CCP), and critical power ratio (CPR) for the CANFLEX bundles (with natural or recycled uranium fuel) in the CANDU-6 reactor fuel channel are calculated by the code. The effects of axial and radial heat flux on CCP are evaluated by assuming that the recycled uranium fuel (CANFLEX-RU) has the same geometric data as the natural uranium fuel bundle (CANFLEX-NU), but a different power distribution due to different fuel composition and refueling scheme. In addition, the effects of pressure tube creep and bearing-pad height are examined by comparing various results of uncrept, and 3.3% and 5.1% crept channels loaded with CANFLEX bundles with 1.4 mm or 1.7 mm high bearing-pads with those of the 37-element bundle. The distributions of the channel flow rate and CCP for the CANFLEX-NU or -RU bundle show a typical trend for a CANDU-6 reactor channel, and the CPRs are maintained above at least 1.444 (NU) or 1.455 (RU) in the uncrept channel. The enhanced CHF of the CANFLEX bundle (particularly with 1.7 mm height bearing-pads) produces a higher thermal margin and considerably less sensitivity to CCP reduction due to the pressure tube creep than the 37-element bundle. The CCP enhancement due to the raised bearing-pads is estimated to be about 3% ~ 5% for the CANFLEX-NU and 2% ~ 6% for the CANFLEX-RU bundle, respectively.

KEYWORDS : CANFLEX-NU or -RU Bundle, CCP, Pressure Tube Creep, Bearing-Pad Height

1. INTRODUCTION

The CANFLEX (CANDU Flexible) fuel bundle has been jointly developed since 1991 by Korea Atomic Energy Research Institute (KAERI) and Atomic Energy of Canada Limited (AECL) as an advanced nuclear fuel for a pressurized heavy water reactor. It is a dual-sized 43-element bundle (8 rods with large diameter and 35 rods with small diameter), as shown in Figure 1, and is designed to improve the thermal hydraulic performance of the current 37-element bundle by using patented heat-transfer enhancing buttons attached to the surface of the elements. The development of the CANFLEX bundle with a 1.4 mm bearing pad (BP) (called a low BP or CANFLEX Mk-IV) was completed and a demonstration irradiation of 24 bundles with natural uranium (NU) fuel was performed from September 1998 to August 2000 at the Point Lepreau Generating Station (PLGS) in Canada [1]. A similar demonstration irradiation program was also carried out from July 2002 to February 2004 at the Wolsong-1 reactor in Korea [2]. A minor bundle-design modification (called a high BP or CANFLEX Mk-V) has been proposed to improve heat transfer by increasing the bearing-pad height from 1.4 mm to 1.7 mm. It is designed to increase the gap flow at the bottom of the bundle, where dryout preferentially occurs, and to decrease enthalpy imbalance between the top and bottom portions of the bundle by reducing the bundle eccentricity in the horizontal channel. Full-scale CHF tests, jointly funded by AECL and KAERI, have been performed at Stern Laboratories (SL) for the CANFLEX Mk-IV and CANFLEX Mk-V bundles [3,4]. In addition, a CANFLEX-RU (Recycled Uranium) fuel development program [5] was proposed to use recycled uranium in a PHWR. The fuel was produced by reprocessing spent fuel and is being stored in excess of 25,000 tons in France, the United Kingdom, and Japan. The recycled uranium can be directly used as PHWR fuel without any enrichment and it will reduce the production rate of spent fuel by about ½, because it burns up 2 times more than natural uranium. It is assumed that the CANFLEX-RU bundle is designed to have the same geometrical configurations as a CANFLEX-NU bundle, but contains RU pellets of 0.92% U-235 enrichment instead of NU pellets of 0.71% U-235 enrichment.

This paper evaluates the thermalhydraulic performance of the CANDU-6 reactor loaded with various CANFLEX fuel bundles by the application of recent models based on the pressure drop and CHF data for the CANFLEX bundle [3,4,6]. NUCIRC [7] is a steady state thermalhydraulic code designed to analyze the heat transport system of a CANDU reactor. It incorporates recent models of pressure drop and critical heat flux (CHF) based on high-pressure steam-water tests for a CANFLEX bundle as well as a 37-element bundle [8]. In this paper, the distribution of the channel flow rate, channel exit quality, critical channel power (CCP), and critical power ratio (CPR) for the CANFLEX bundles (with natural or recycled uranium fuel) in the CANDU-6 reactor fuel channel are calculated by the code. The effects of the axial and radial heat flux on the CCP are evaluated under the assumption that the CANFLEX-RU has the same geometric data as the CANFLEX-NU, but a different power distribution due to different fuel composition and different refueling schemes. In addition, the effects of pressure tube creep and bearing-pad height are examined by comparing the results of uncrept, and 3.3% and 5.1% crept channels loaded with CANFLEX bundles with 1.4 mm or 1.7 mm high bearing-pads with those of the 37-element bundle.

2. ANALYSIS METHOD

2.1 Pressure Drop Model

The CANDU-6 reactor has 380 horizontal fuel channels connected to both reactor inlet and outlet headers by individual feeder pipes and end fittings. The end fittings are external fuel channel assemblies connected to the core. NUCIRC is a steady state thermalhydraulic code designed to analyze the primary heat transport system of a CANDU nuclear reactor and to predict the critical channel power. The single phase pressure drop for a CANDU-type fuel string in the code is calculated by equation (1), which is composed of a skin frictional loss and a form loss due to the appendages of a bundle.

$$\Delta P_{1-\phi} = (f_{cor} f_{CW} \alpha_f \frac{L}{D_{hy}} + \alpha_k K_{form}) \frac{Q^2}{2\rho A^2} \tag{1}$$

In the above equation, $\Delta P_{1-\phi}$, Q, D_{hy}, L, ρ and A denote the single phase pressure drop (kPa), channel flow (kg/s), equivalent hydraulic diameter (m), fuel channel length (m), coolant density (kg/m^3), and flow area (m^2), respectively. K_{form} is the total form loss coefficient for flow obstructions such as fuel-string entrance and exit, spacers, bearing pads, and junctions. f_{CW} and f_{cor} are the Colebrook-White's skin friction coefficient and its correction factor, respectively. α_f and α_k are the correction factors to consider the pressure tube creep effects on the skin frictional loss and form loss, respectively. When equation (1) is applied to the CANFLEX bundle and the 37-element bundle, the same coefficients and correction factors can be used for the two bundles except for the form loss coefficients [7].

The CANDU-6 reactor is characterized by the occurrence of coolant voids in many fuel channels at normal operating power conditions. The two phase pressure drop of the horizontal fuel channel is composed of the two phase frictional loss ($\Delta P_{2-\phi,fric}$) and the acceleration pressure drop (ΔP_{acc}), as shown in equation (2). The two phase frictional loss, which is caused by the friction between the two-phase mixture and solid walls, is far greater than the acceleration pressure drop, which is caused by the momentum change due to the coolant density gradient.

$$\Delta P_{2-\phi} = \Delta P_{2-\phi,fric} + \Delta P_{acc} \tag{2}$$

where

$$\Delta P_{2-\phi,fric} = \phi^2 \Delta P_{1-\phi}, \quad \Delta P_{acc} = G^2 \Delta \left[\frac{X_a^2}{\alpha \rho_g} + \frac{(1-X_a)^2}{(1-\alpha)\rho_f} \right]$$

In addition to correlations determining the two phase friction multiplier (ϕ^2), mass quality (X_a), and void fraction (α), correlations for OSV (Onset of Significant Void) are needed in order to obtain the location of the initiation of boiling for the application of the above equation. The correlations of Friedel for the two phase friction multiplier [9], Saha-Zuber for mass quality [10], and Armand-Massena for void fraction [11] were applied to both the CANFLEX and the 37-element bundles. Different correlations of OSV were used for the two bundles. For the CANDU fuel bundle, the initiation of boiling is defined at the OSV point instead of at the saturation point due to the enthalpy imbalance among the subchannels in the bundle. The OSV points were obtained at the transition location from the single-phase to two-phase flow region with Stern Laboratories pressure drop measurements [6]. The OSV quality correlation for a CANFLEX bundle is expressed in equation (3). This correlation is applied to determine the transition point between single-phase and two-phase flow in the calculation of the BLA (boiling-length-average) CHF and the two-phase pressure drop.

$$x_{OSV} = C_1 \left(\frac{q''_{local}}{(1-E)^{C_2}\, G\, H_{fg}} \right)^{C_3} \left(\frac{(1-E)^{C_4}\, G\, D_{hy}}{\mu_f} \right)^{C_5} \left(\frac{\rho_g}{\rho_f} \right)^{C_6} \quad (3)$$

where ρ_g and ρ_f are the saturation densities of vapor and liquid phase, respectively. q''_{local} is the cross-section averaged local heat flux, G is the local mass flux, H_{fg} the latent heat of vaporization, D_{hy} the equivalent hydraulic diameter, μ_f the dynamic viscosity of the fluid at saturating conditions, $C_1 \sim C_6$ are constants, and E is the bundle eccentricity defined as

$$E = \frac{D_{P/T} - D_{bundle}}{D_{P/T} - D_{inner}} \quad (4)$$

$D_{P/T}$ is the diameter of the pressure tube, D_{bundle} the diameter of the bundle, and D_{inner} is the inner diameter of the bundle based on the equivalent-annuli approach [6]. The bundle eccentricity is introduced to account for the pressure tube creep profiles.

2.2 Critical Heat Flux Model

A series of water CHF tests for the 6 meter full-scale CANFLEX bundle simulator [3] were carried out in the horizontal channel of Stern Laboratories by a KAERI/AECL joint project. Test series in 3.3% and 5.1% peak crept channels as well as in an uncrept channel were performed so as to quantify the impact of the pressure tube creep deformation during the plant operating life time. The test fuel bundle string had an axially and radially non-uniform heat flux profile. In addition to the existing 1.4mm bearing pads (low BP), 1.7mm and 1.8mm bearing pads (high BP) were attached to the test bundle [4]. The modification of bearing-pad height is designed to increase the gap flow at the bottom portion of the bundle and to improve the heat transfer. The test flow conditions covered a wide range of pressure, from 6 to 11 MPa, flow rate, from 7 to 29 kg/s, and inlet temperature, from 200 to 290°C.

Based on the analysis of the test data, a CHF correlation for the CANFLEX bundle was derived and incorporated into the NUCIRC code. The correlation was based on the BLA CHF defined as the following equation (5), where Z_{DO} and Z_{OSV} are the distance of the dryout location and the OSV point, respectively.

$$q''_{BLA} = \frac{1}{Z_{DO} - Z_{OSV}} \int_{Z_{OSV}}^{Z_{DO}} q''_{local}\, dz \quad (5)$$

Generally, the BLA concept is introduced to account for the AFD (axial flux distribution) effect on CHF. The BLA CHF values show greater consistency and less data scattering with the critical quality than the local CHF values in uncrept and crept channels [4]. The correlation is finally composed of dimensionless parameters, as shown in equation (6), and hence can be directly applied to the heavy water cooled fuel channel.

$$Bo_{BLA} = \frac{1}{10000} \left(b_1 \left(\frac{\rho_f}{\rho_g} \right)^{b_2} We^{b_3} - b_4 \left(\frac{\rho_f}{\rho_g} \right)^{b_5} We^{b_6}\, x_{DO} \right) \quad (6)$$

where

$$We = \frac{G\, D_{hy}^{0.5}}{\rho_f^{0.5}\, \sigma^{0.5}}, \quad Bo_{BLA} = \frac{q''_{BLA}}{G\, H_{fg}}$$

The above equation has the coefficients $b_1 \sim b_6$, which are a function of the bundle eccentricity to account for the local geometric effects on the BLA CHF of the CANFLEX bundle. Thus, the correlation is applicable for predicting CHF in uncrept and crept channels. The correlations of OSV and CHF for the 37-element bundle are of the same type as those for the CANFLEX bundle, but with different values of coefficients $b_1 \sim b_6$ and $C_1 \sim C_6$ in equations (3) and (6).

2.3 Calculation Conditions

The thermalhydraulic analysis of the CANDU-6 reactor fuel channel loaded with CANFLEX-NU or -RU bundles was performed with the NUCIRC code with an inlet header temperature of 265°C, an outlet header pressure of 9.99 MPa, and a header-to-header pressure drop of 1342 kPa. This code incorporates the recent models of pressure drop and CHF for the CANFLEX fuel bundle as well as the 37-element bundle, as described in the previous sections. For simulating the heat transport system of the CANDU-6 reactor with the CANFLEX or the 37-element fuel bundle, the same NUCIRC input parameters are employed, except for those related to the fuel channels, e.g. the form loss factor in the pressure drop model and the selection of CHF correlation are dependent on the bundle type in the fuel channel. The pressure loss coefficient of the mid-plane spacer of the CANFLEX bundle attached with the CHF enhancement buttons is much greater than that of the 37-element bundle.

The axial and radial heat flux distribution for the NU fuel and the RU fuel bundles are shown in Figure 2 and

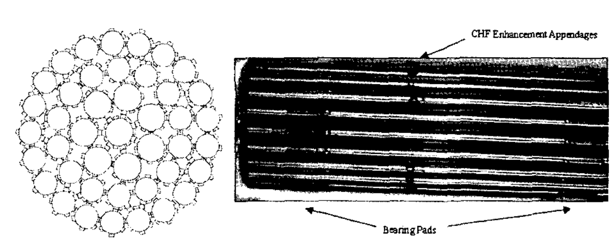


Fig. 1. Picture of the CANFLEX Fuel Bundle

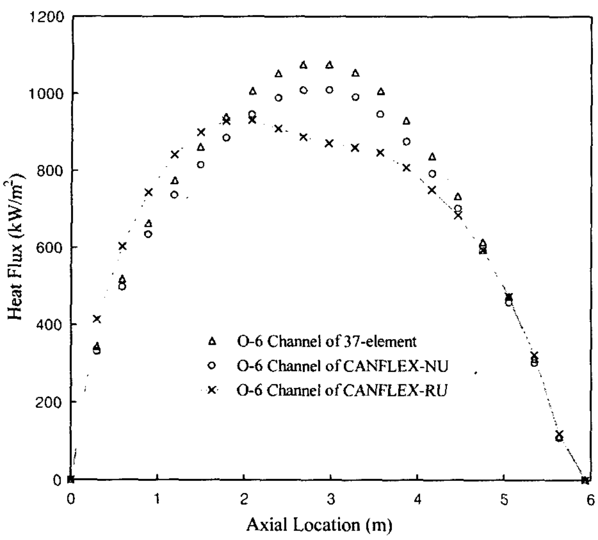


Fig. 2. Axial Heat Flux Distributions for the NU and RU Fuel Bundles

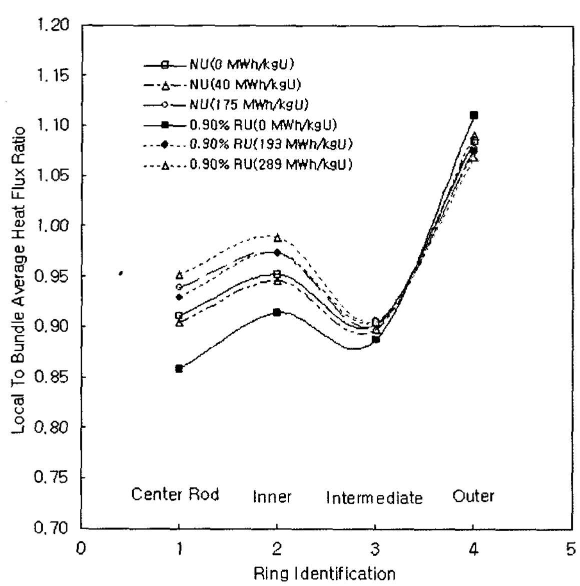


Fig. 3. Radial Heat Flux Distributions of the CANFLEX-NU and -RU Bundles

Figure 3, respectively. The power distributions are dependent on the fuel composition and the refueling scheme. The recycled uranium fuel bundle uses a 4-bundle refueling scheme to meet the current CANDU fuel performance criteria instead of the 8-bundle refueling scheme of the natural uranium fuel bundle. The axial peak location of the RU fuel bundle tends to move upstream, while the NU fuel bundle maintains a cosine shape. The axial heat flux distribution of the RU fuel bundle is caused by the use of the 4-bundle refueling scheme in place of the 8-bundle refueling scheme of the NU fuel bundle. The axial heat flux distribution of the RU fuel bundle is optimized to increase the CCP of the CANDU-6 reactor, because the local heat flux is lowered at downstream where dryout preferentially occurs [12]. The BLA CHF correlation alone can account for the effect of the axial heat flux distribution, and thus there is no need to apply additional correction factors for the effect of the axial heat flux distribution of the RU fuel bundle.

As shown in Figure 3, the change in the radial flux distribution of the RU fuel bundle is slightly greater than that of the NU fuel bundle with burnup. It is difficult to evaluate the effect of the radial heat flux distribution on CHF without various experimental data. AECL performed a sensitivity study on the effect of the radial flux distribution on CHF based on a small data base (with only 3 radial heat flux distributions). By applying this method to RU fuel, it was predicted that the CHF change in the RU fuel bundle with burnup would be less than 2% [13]. This estimation may have a large degree of uncertainty, because it is not based on experimental data of large parameter ranges. Hence, it was tentatively assumed that the radial heat flux distribution of the RU fuel bundle would approximately yield a 5% CHF reduction when compared to the NU fuel bundle [14].

The pressure tube of the CANDU-6 reactor suffers from the dimensional change of diametral creep, axial elongation, and creep sag resulting from the neutron irradiation, and mechanical and thermal stresses over the plant life time. The diametral creep of the pressure tube has a skewed cosine-shaped profile along the fuel channel and directly affects the subchannel flow. Thus, calculations

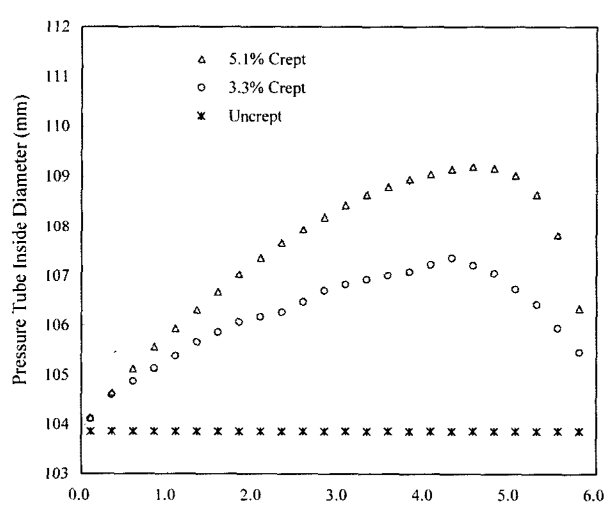


Fig. 4. Axial Profiles of the Pressure Tube Diametral Creep Models

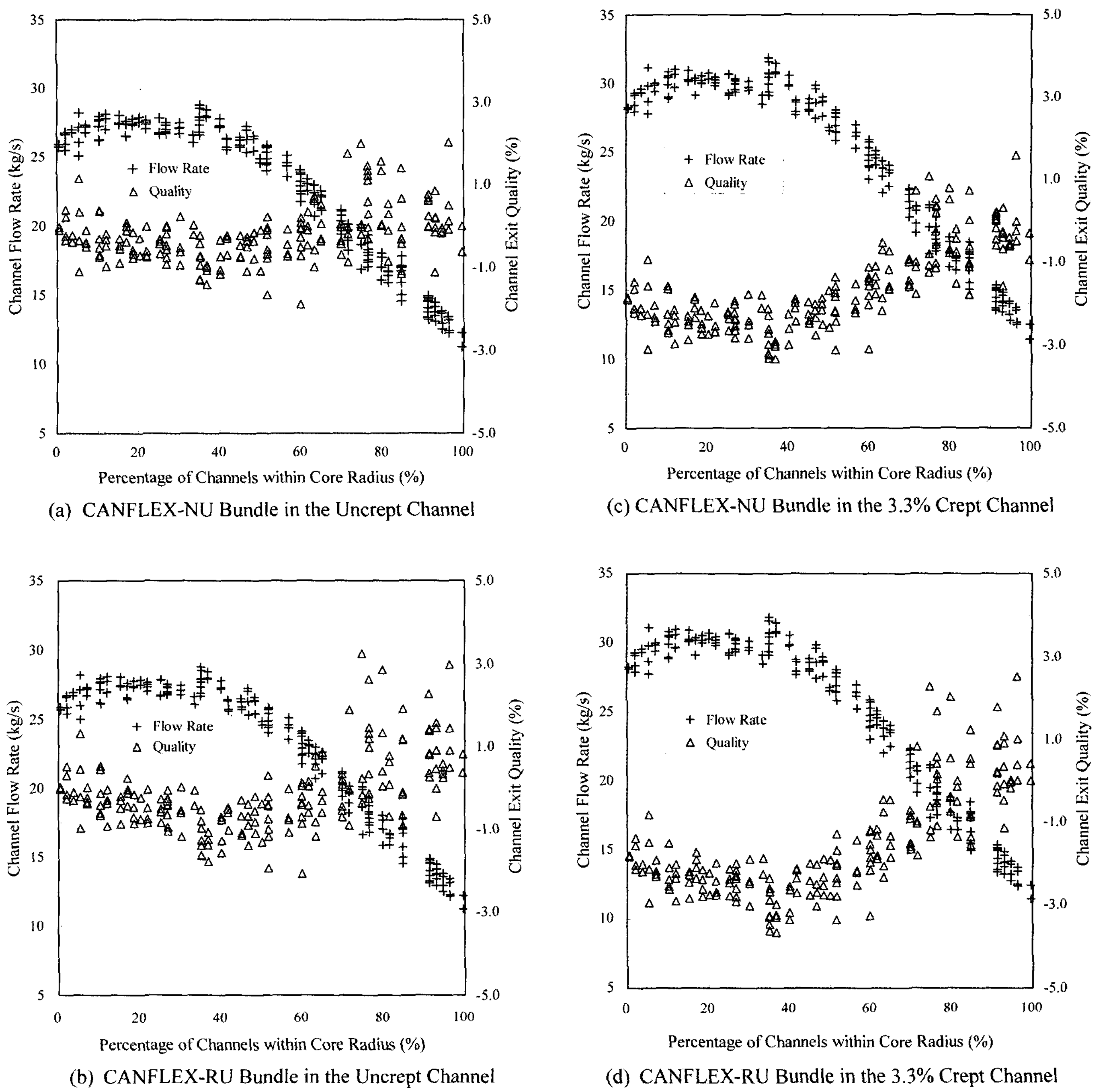


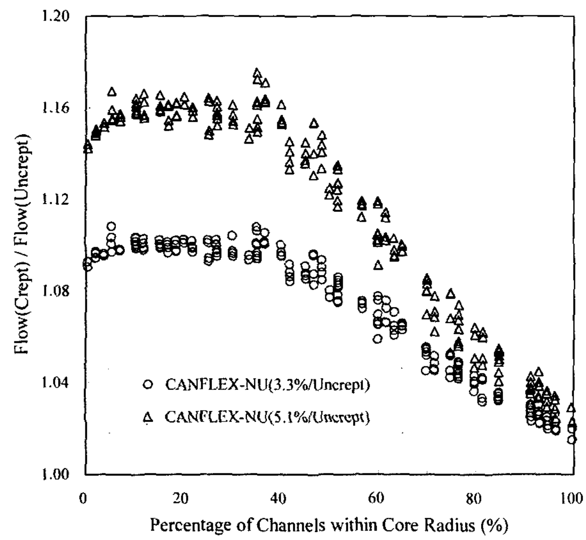
Fig. 5. Channel Flow Rate and Channel Exit Quality Distributions

were done for the 3.3% and the 5.1% peak crept tubes as well as the uncrept tube, as shown in Figure 4. The maximum diametral creep was located at an axial elevation of 4.3 m (the 3.3% crept) and at 4.8 m (the 5.1% crept) axial distance. These profiles representatively simulate the prototypical fuel channels with plant ageing and were used for the water CHF tests.

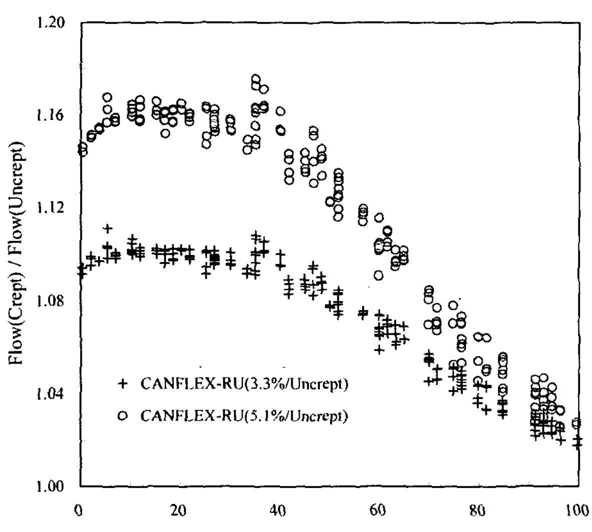
3. RESULTS ANALYSIS AND DISCUSSION

3.1 Channel Flow Rate and Channel Exit Quality

Figure 5 shows the distribution of the channel flow rate and channel exit quality of the CANDU-6 reactor channels loaded with CANFLEX-NU or the CANFLEX-RU fuel bundles with respect to the core radius. The flow rates in the channels, which are located in the inner-core, maintain high values of more than 25 kg/s whereas the flow rates in the outer-core linearly decrease as the distance to the core center increases. In the CANDU-6 reactor, the channel flow rate distribution is designed to have almost the same enthalpy rise in all the fuel channels. That is, the channels with high power have high flow rates and the low power channels have low flow rates. Figure 5 also shows that the distribution of the channel exit quality is opposite to that of the flow rate, where the exit qualities in the outer-core channels linearly increase as the distance to the core centre increases. The channels at the same core radius have different channel powers, because they are not symmetrically located in the core. This channel power distribution causes a scattering in the channel flow and in the quality at the same core radius. The distribution of the channel flow and exit quality for the CANFLEX-RU fuel bundles are very similar to those of the CANDU-6 reactor channels loaded with natural uranium fuel bundles. A slightly different distribution of channel flow and quality for the two bundles is shown in Figure 5. The difference is likely

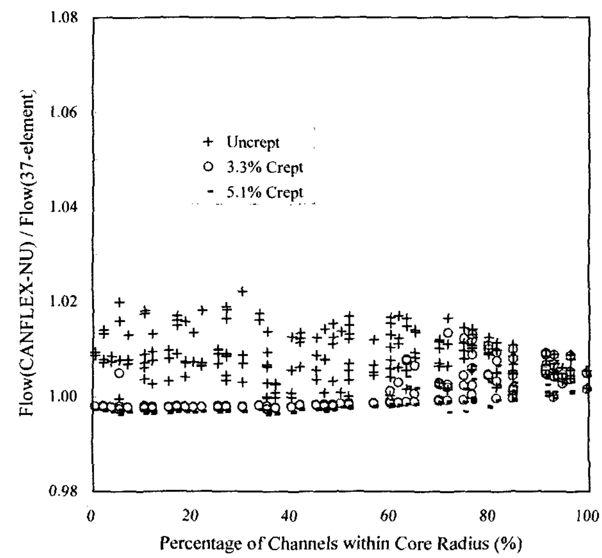


(a) CANFLEX-NU Bundle

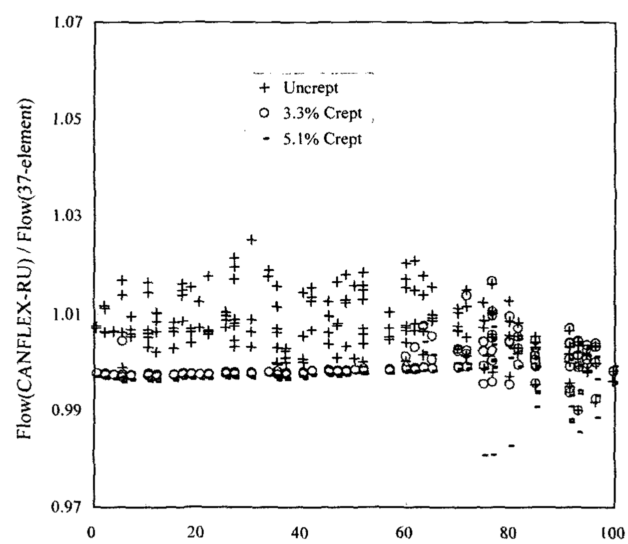


(b) CANFLEX-RU Bundle

Fig. 6. Comparison of the Flows in the Crept Channels with the Flows in the Uncrept Channel



(a) CANFLEX-NU Bundle

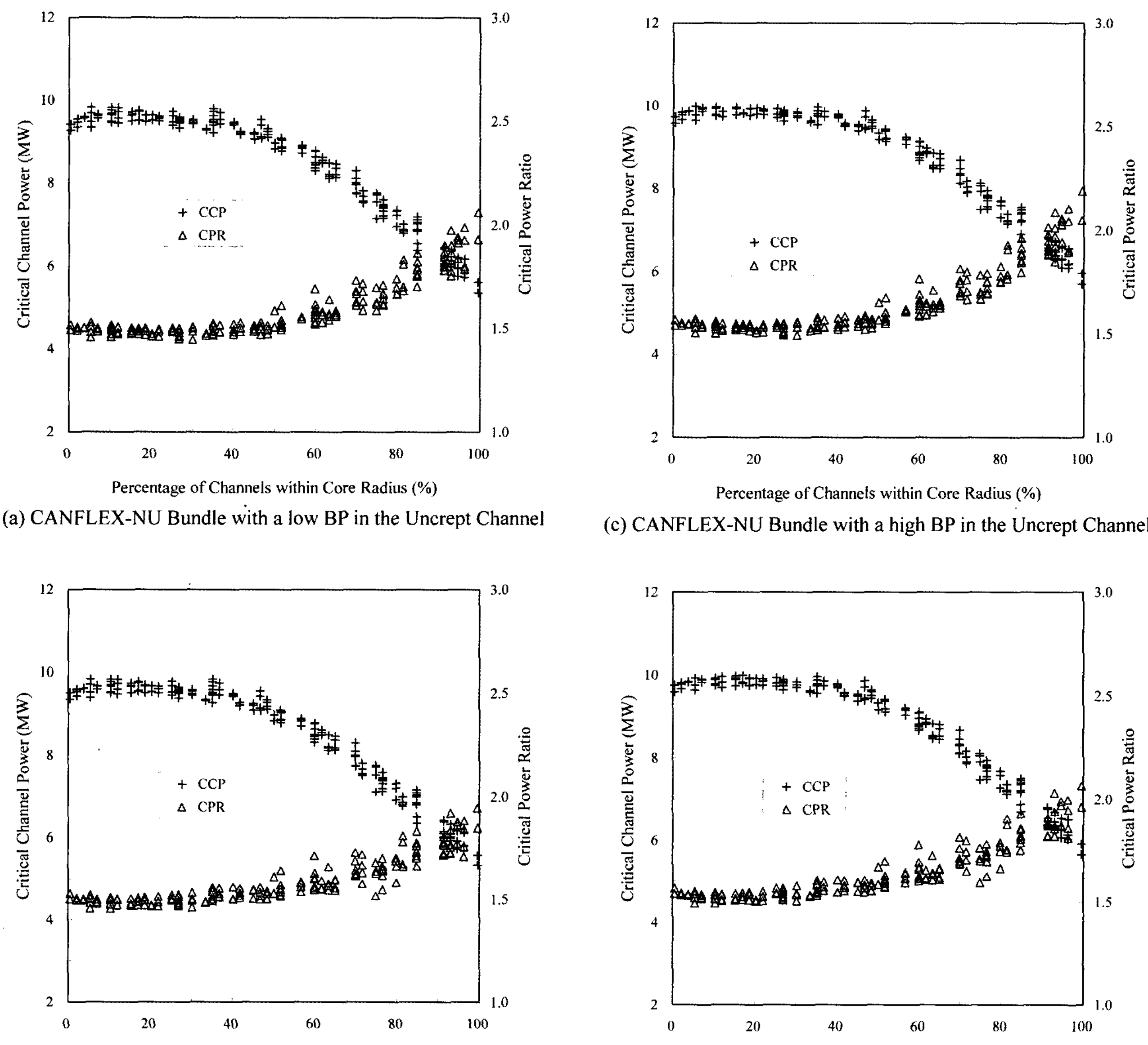


(b) CANFLEX-RU Bundle

Fig. 7. Comparison of the Flow Rates for the CANFLEX-NU or -RU Bundles with the Flow Rates for the 37-Element Bundle in the Uncrep, 3.3% and 5.1% Crept Channels

due to changes in the channel power and axial heat flux distributions between the NU and the RU fuel bundles. The channel flow rate increases and the exit quality decreases as the creep rate of the pressure tube increases. This trend is more sensitive in the high power channels than in the low power channels. As shown in Figure 6, the increase in the flow rate due to the creep of the pressure tube is much greater in the inner-core than in the outer-core channels. Because the fuel channel pressure drop portion of the header-to-header pressure drop in the high power/high flow region is much greater than that in the low power/low flow region, the channel flow rate in the inner-core is more affected by pressure tube creep than in the outer-core region.

In this paper, different CHF correlation options in the NUCIRC code were applied to CANFLEX bundles with high bearing pads and with low bearing pads. The same form loss coefficients were used for the two bundles, although the raised bearing-pad height was expected to increase the pressure drop due to greater concentric flow; thus there were no changes in the flow rates for the two



(a) CANFLEX-NU Bundle with a low BP in the Uncrept Channel

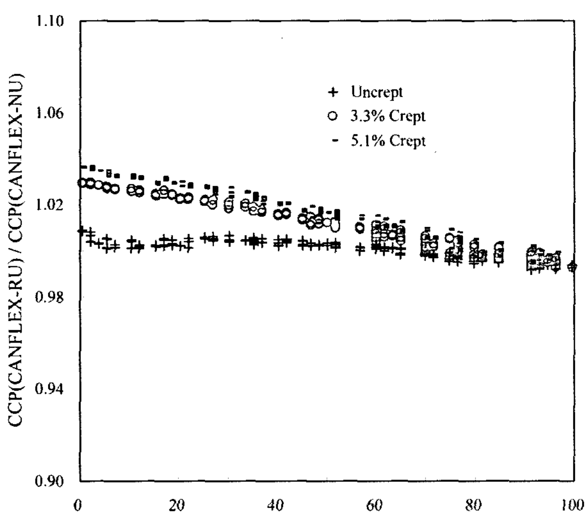
(c) CANFLEX-NU Bundle with a high BP in the Uncrept Channel

(b) CANFLEX-RU Bundle with a low BP in the Uncrept Channel

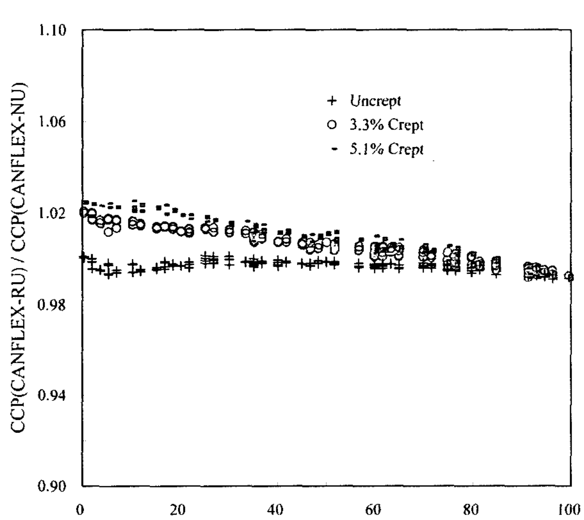
(d) CANFLEX-RU Bundle with a high BP in the Uncrept Channel

Fig. 8. Critical Channel Power and Critical Power Ratio Distributions

bundles. Figure 7 shows the relative flow rates of the CANFLEX fuel bundles to the flow rates of the 37-element fuel bundles in the uncrept, and 3.3% and 5.1% crept channels. In the case of the uncrept channel, the flow rates of the CANFLEX fuel bundles slightly increased by up to 2% in most channels relative to those of the 37-element fuel bundles. In the case of the 3.3% and 5.1% crept channels, the change of the flow rate was negligible in the inner-core but was scattered by ± 2% in the outer-core region. These variations of the flow rates for the two fuel bundles were affected by the channel power and axial heat flux distribution as well as other input parameters such as the flow area and form loss coefficients.



(a) low BP



(b) high BP

Fig. 9. Comparison of the CCPs for the CANFLEX-NU with CANFLEX-RU Bundle

3.2 Critical Channel Power and Critical Power Ratio

Figure 8 shows the distributions of CCP and CPR for the CANFLEX-NU and CANFLEX-RU fuel bundles attached with low and high bearing pads, respectively, in the uncrept pressure tube. It was found that the CCP of the CANFLEX fuel bundle attached with low bearing pads in the uncrept pressure tube was maintained just above 9 MW in the inner-core, but decreased in the outer-core. Thus, the trend is very similar to the distribution of the channel flow rate. The CPR of the CANFLEX fuel bundle attached with the low bearing pads in the uncrept pressure tube was maintained above 1.444(NU) or 1.455(RU), at least in the inner-core. However, it gradually increased in the outer-core, despite decreasing CCP, due to the relatively low channel powers. In Figure 8, many channels in the inner-core have CPR values close to the minimum CPR. Therefore, the location of the minimum CPR channel is sensitive to the channel power and the axial power distributions. These trends of CCP and CPR were also found for all the cases with crept pressure tubes and natural uranium fuel bundles.

In Figure 9, the CCP of the CANFLEX-RU bundle is compared with the CCP of the CANFLEX-NU bundle in the uncrept and crept channels. It is observed that the CCP of the CANFLEX-RU fuel bundle is very close to that of the CANFLEX-NU bundle in the uncrept channel and is increased by about 2% ~ 4% in the 3.3% crept and 5.1% crept channels. This indicates that the CCP of the CANFLEX-RU fuel is not reduced but rather is increased by the axial heat flux distribution of the RU fuel bundle when compared to that of the CANFLEX-NU bundle, although a CHF decrease of 5% was assumed from the radial heat flux distribution of the RU fuel. The axial heat flux distributions of the RU fuel bundle are flat or slightly concave in the channel center region and the peak heat flux locations are moved upstream in the fuel channel relative to those of the natural uranium fuel (cosine-shaped profiles). Hence, the relatively low local heat flux of the RU fuel around the downstream region in the fuel channel would lead to an increase in CCP [12].

In Figure 10, the CCP of the CANFLEX-NU and -RU bundles is compared with the CCP of the 37-element bundle in the uncrept and crept channels. Figures 10 (a) and (c) indicate that the CCP of the low BP CANFLEX-NU bundle is increased by about 2%, 4%, and 7% and the CCP of the high BP CANFLEX-NU bundle is increased by about 5%, 7%, and 12% in the uncrept, 3.3% crept, and 5.1% crept channels, respectively, relative to those of the 37-element bundle in the inner-core. Figures 10 (b) and (d) also show that the CCP of the low BP CANFLEX-RU bundle is increased by about 2%, 6%, and 8% and the CCP of the high BP CANFLEX-RU bundle is increased by about 4%, 8%, and 14% in the uncrept, 3.3% crept and 5.1% crept channels, respectively, relative to those of the 37-element bundle in the inner-core.

3.3 Pressure Tube Creep Effects

The effects of the pressure tube diametral creep on the operating conditions of the reactor heat transport system are known to increase in a core flow with a rise of inlet header temperature and a reduction of header-to-header pressure drop. In addition, low power/low flow channels will suffer a reduction in flow, while high power/high flow channels will experience a significant increase in flow, because the outer-core channels are relatively less crept than the inner-core channels. However, it is difficult to predict the actual operating conditions because the pressure tube creep slowly occurs over the plant life time. This is accompanied by heat transport system ageing such as crud deposition and flow-accelerated corrosion on the steam generator and feeder pipes, which increases the loop hydraulic resistance and is highly dependent on the plant operating histories. Thus, in this paper, fixed operating conditions

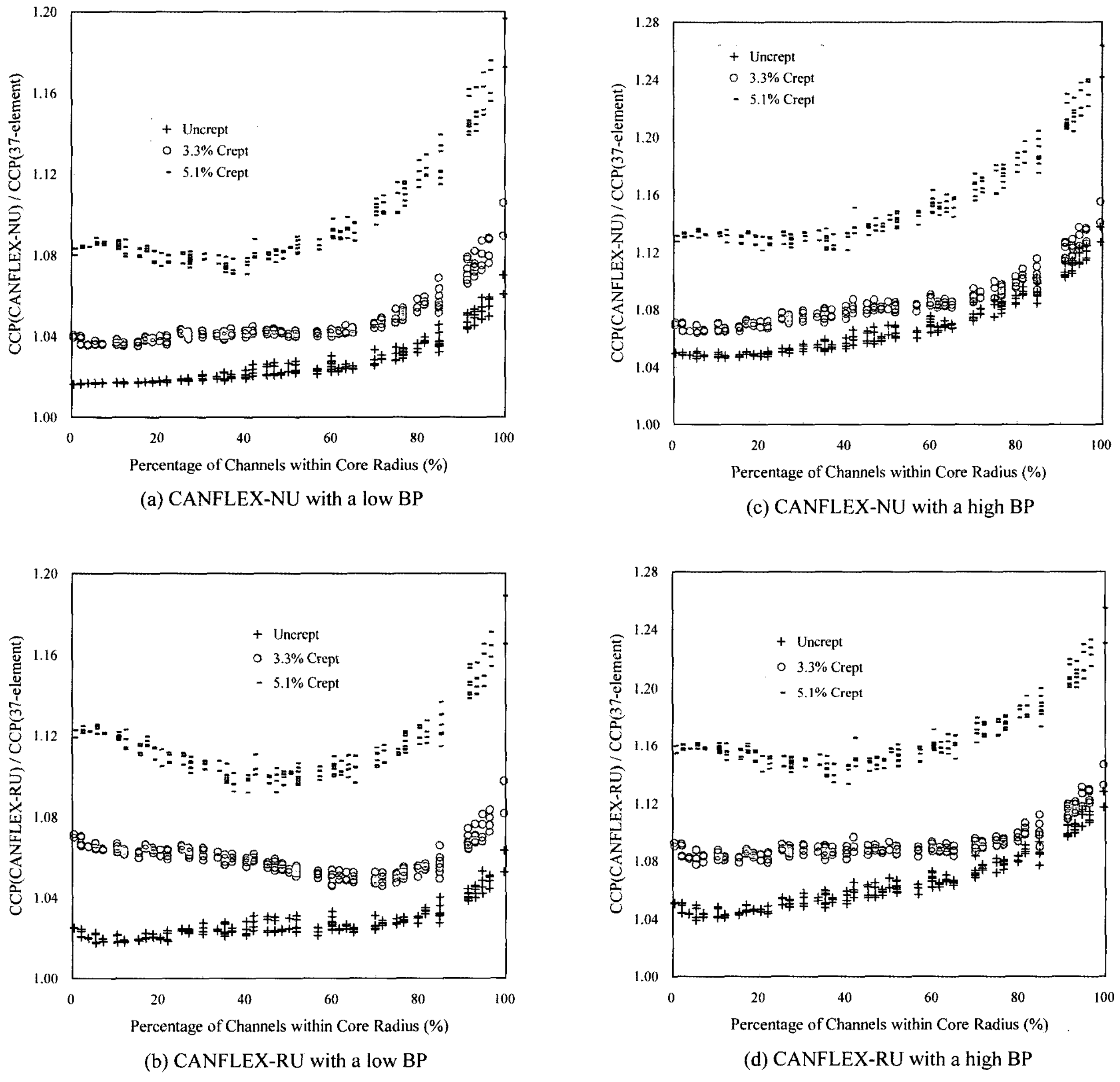
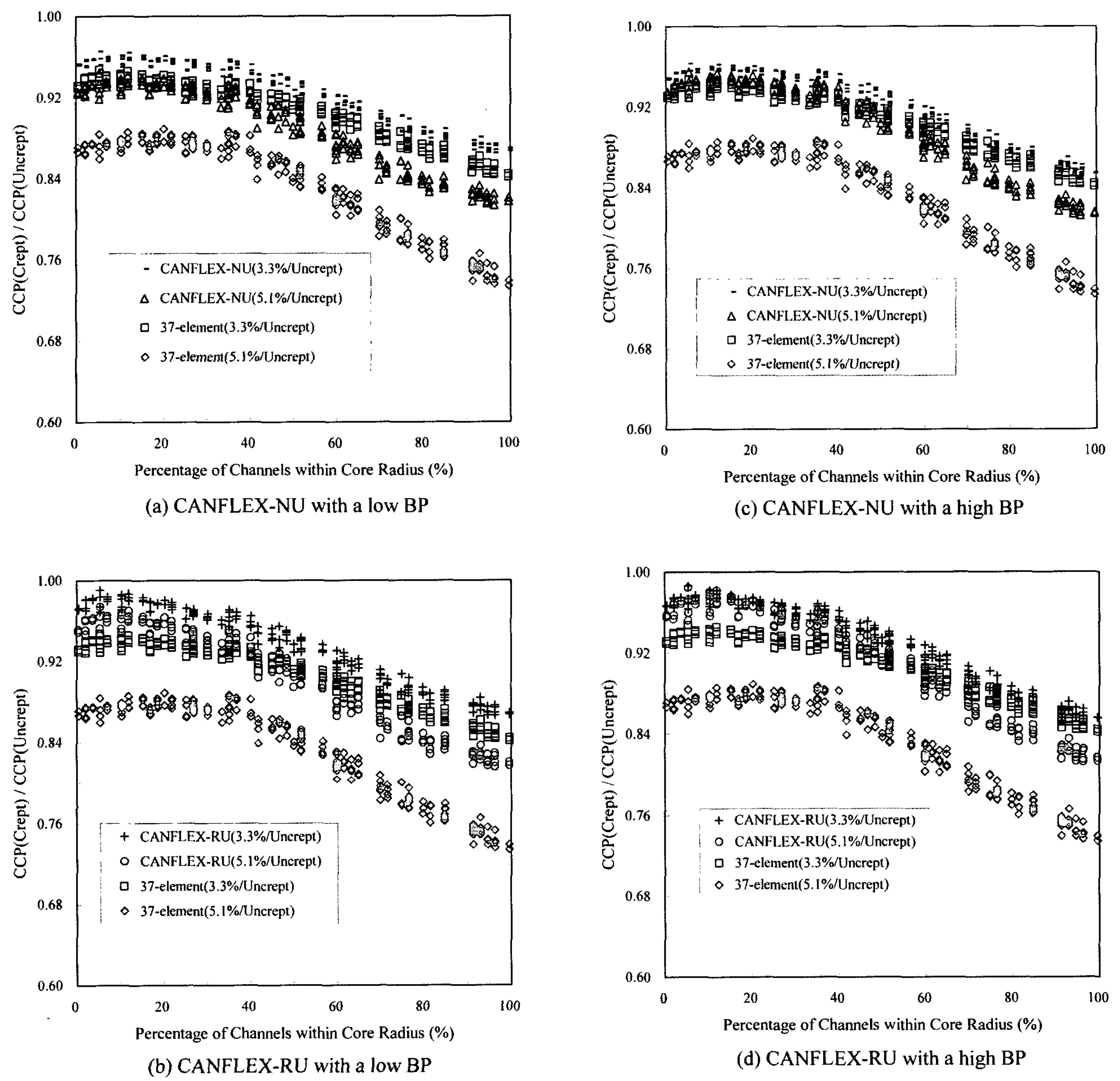


Fig. 10. Comparison of the CCPs for the CANFLEX-NU or -RU with the 37-Element Bundle

for the heat transport system were used in the case of various crept channels in order to examine the effect of the pressure tube diametral creep.

The pressure tube diametral creep causes a direct increase in the flow area between the top of the bundle and the pressure tube, and reduces the effective hydraulic resistance in the channel. The increase in the bypass flow is not effective for heat transfer through the inner subchannels of the bundle, although the total channel flow increases. Therefore, it is known that pressure tube creep will cause a reduction in CCP due to the decrease in CHF. Figure 6 shows that the flows in the 3.3% and 5.1% crept channels are respectively about 10% and 16% greater than the flows in the uncrept channels in the inner-core region, but the relative increase in channel flows decreases rapidly to about 2% and 3% in the outer-core region. This means that the channel flow rate in the high power/high flow region is relatively more affected by pressure tube creep than in the low power/low flow region, because the fuel channel pressure drop portion of the header-to-header pressure



(a) CANFLEX-NU with a low BP
(c) CANFLEX-NU with a high BP
(b) CANFLEX-RU with a low BP
(d) CANFLEX-RU with a high BP

Fig. 11. The Ratio of the CCPs in the Crept Channels to Those in the Uncrept Channel

drop in the high power/high flow region is much greater than that in the low power/low flow region.

In Figure 11, the CCP in the crept channels is compared with that in the uncrept channel loaded with the CANFLEX -NU, CANFLEX-RU bundles (with low BP or high BP), and the 37-element bundle. Figure 11 (a) shows that the CCP of the low BP CANFLEX-NU bundle in the 3.3% crept and 5.1% crept channels in the inner-core is about 4% and 8%, respectively, less than that in the uncrept channel. Figure 11 (c) shows that the CCP of the high BP CANFLEX-NU bundle in the 3.3% crept and 5.1% crept channels in the inner-core is about 4% and 6%, respectively, less than that in the uncrept channel. Figure 11 (b) shows that the CCP of the low BP CANFLEX-RU bundle in the 3.3% crept and 5.1% crept channels in the inner-core is about 2% and 4%, respectively, less than that in the uncrept channel. Figure 11 (d) shows that the CCP of the high BP CANFLEX-RU bundle in the 3.3% crept and 5.1% crept channels in the inner-core is about 3% less than that in the uncrept channel. However, it shows that the CCP decreases for the 37-element bundle are about 6% and 12%, respectively, greater than those of the CANFLEX bundles. This means that the CANFLEX bundle is considerably less sensitive to CCP reduction due to pressure tube creep than the 37-element bundle. It is also found that the amount of CCP reduction due to pressure tube creep increases in the outer-core region.

3.4 Bearing Pad Height Effects

As shown in Figure 1, the bearing pads are 1.45 mm high appendages attached to the outer elements at 3 planes of the bundle in order to maintain the gap between the horizontal pressure tube and the fuel bundle. Based on the water CHF test data of the CANFLEX bundle attached with a minimum height of 1.4 mm bearing pads [3], it was found that most CHF points are located on the bottom rods of the bundle and the flow imbalance becomes severe in the crept pressure tubes. Hence, a slight increase (about 0.3 mm ~ 0.4 mm) of the bearing pad height was employed in order to improve the dryout power of the CANFLEX bundle. The validation of the thermal performance was made by additional water CHF tests [4] of the CANFLEX fuel bundle with high bearing pads.

The raised bearing pads on the outer elements kept the fuel bundle string closer to the central position of the pressure tube and increased the axial gap flow in the bottom position of the bundle. They also increased the dryout power relative to the CANFLEX fuel bundle with the low bearing pads. As an example of the reduction in enthalpy imbalance due to the raised bearing pads [3,4], it was observed that the temperature difference between the bottom elements and top elements rapidly decreased from 42°C for the 1.4 mm high bearing pads to 35°C for the 1.7 mm bearing pads, based on the same single-phase flow condition data in the uncrept channel. Based on the relative CCP gain, as displayed in Figure 10, the CCP enhancement of the CANFLEX bundle due to the raised bearing pads is estimated to be about 3%, 3%, 5% (NU), and about 2%, 2%, 6% (RU) in the uncrept, 3.3% crept, and 5.1% crept channels, respectively.

4. CONCLUSIONS

The distributions of the channel flow rate, channel exit quality, CCP, and CPR of the CANFLEX bundle reflect typical trends of CANDU-6 reactor channels. The CPR of the low BP CANFLEX bundle in the uncrept channel is maintained above 1.444 (NU) or 1.455 (RU) in the inner -core. The CCP gains of the CANFLEX bundles are estimated to be about 2%, 4%, 7% (low BP, NU), 5%, 7%, 12% (high BP, NU), 2%, 6%, 8% (low BP, RU), and 4%, 8%, 14% (high BP, RU) in the uncrept, 3.3% crept, and 5.1% crept channels, respectively, relative to those of the 37-element bundle in the inner-core. From the relative CCP gains between the low BP and the high BP bundle, the CCP enhancement of the CANFLEX bundle due to the raised bearing pads is estimated to be about 3%, 3%, 5% (NU), and about 2%, 2%, 6% (RU) in the uncrept, 3.3% crept, and 5.1% crept channels, respectively. The shape of the axial heat flux profile of the RU fuel (the peak skewed to upstream) produces an increase in the CCP (about 2% ~ 4% in the 3.3% crept and 5.1% crept channels), although a CHF decrease of 5% was predicted based on a comparison of the radial heat flux distributions of both the RU fuel bundle and the NU fuel bundle. The reduction in CCP due to pressure tube creep in the CANFLEX bundle is much less than that in the 37-element bundle. Furthermore, the raised bearing pads can significantly decrease this CCP reduction.

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