The Thermalhydraulic Characteristics of the CANDU-6 Reactor Channel with a CANFLEX-RU Fuel Bundle

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ABSTRACT

This paper describes the thermalhydraulic characteristics of a CANDU-6 reactor's fuel channel loaded with CANFLEX-RU (CANDU Flexible Fuelling - Recycled Uranium) bundles. The distributions of the channel flow rate, channel exit quality, critical channel power(CCP) and critical power ratio(CPR) for the CANFLEX-RU fuel channels are evaluated by using the NUCIRC code. This code is incorporated with the recent models for pressure drop and critical heat flux(CHF) of the CANFLEX fuel bundle as well as a 37-element bundle. Especially, the effects of pressure tube creep and bearing pad height on the thermalhydraulic characteristics are also examined by comparing the various results on the uncrept, 3.3% and 5.1% crept channels loaded with CANFLEX bundles with the 1.4 mm height or 1.7 mm height bearing pads with those of the 37-element bundle. The distributions of channel flow rate and CCP for the CANFLEX-RU bundle show a typical trend for the CANDU-6 reactor channel, and the CPR is maintained above 1.455 at least in the uncrept channel. The CANFLEX-RU fuel bundle is considerably less sensitive to the CCP reduction due to pressure tube creep than the 37-element bundle. The CCP enhancement of the CANFLEX-RU fuel bundle due to the raised bearing pads is estimated to be about 2%□6%.

1. INTRODUCTION

KAERI(Korea Atomic Energy Research Institute) and AECL(Atomic Energy of Canada Limited) have been jointly developing the PHWR advanced fuel, CANFLEX®[1] (CANDU®[2] Flexible Fuelling) fuel bundle, since 1991 and completed the development in 2000. So, the DI(demonstration irradiation) of the 24 CANFLEX-NU(Natural Uranium) fuel bundles[1] was successfully completed at the PLGS(Point Lepreau Nuclear Generating Station) in Canada from September 1998 to August 2000, and another DI[2,3] has been on going at the Wolsung-1 plant in Korea from July 2002 till now.

On the other hand, KAERI, AECL and BNFL(British Nuclear Fuel Limited) launched a joint study for a CANFLEX-SEU(Slightly Enriched Uranium)/RU(Recycled Uranium) fuel development program[4] in 1996 year. The CANFLEX-RU fuel program is intended to use the recycled uranium in a PHWR, which was produced by reprocessing the spent fuel and is stored up to over the amount of 25,000 tons in France, United Kingdom and Japan. The recycled uranium can be directly used as a PHWR fuel without any enrichment and it will reduce the production rate of spent fuel by about 1/2 because it burns up 2 times more than the natural uranium. The price of the recycled uranium is also lower than that of natural uranium, so the use of recycled uranium will enhance the economy of a CANDU power plant. Thus, providing for the imbalance of uranium supply and demand in the future, the recycled uranium will increase the uranium utilization rate. The recycled uranium fuel is expected to enhance the reactor operating margin because it will be incorporated in the CANFLEX bundle[5,6,7].

In this paper, the thermalhydraulic characteristics of the CANDU-6 reactor's fuel channel loaded with CANFLEX-RU bundles are examined by using the NUCIRC code[8], which is incorporated with the recent models for pressure drop and critical heat flux(CHF) of the CANFLEX bundle. It is assumed that the CANFLEX-RU bundle is designed to have all the same geometrical configurations with a CANFLEX-NU bundle, but contain RU pellets of 0.92% U-235 enrichment instead of the NU pellets of 0.71% U-235 enrichment. In this case, the bundle configurations will not make any change to the pressure drop and

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1 CANFLEX® is a registered trademark of Atomic Energy Canada Limited (AECL) and Korea Atomic Energy Research Institute (KAERI).

2 CANDU®(Canadian Deuterium Uranium) is a registered trademark of Atomic Energy of Canada Limited(AECL)

CHF between the RU fuel and NU fuel bundle, but the axial and radial power distributions will cause a little change of the pressure drop and CHF between the RU fuel and NU fuel bundle. The sensitivity of the power distributions on the thermalhydraulic characteristics have been studied through a feasibility evaluation[9] of the RU fuel. The distributions of channel flow rate, channel exit quality, critical channel power(CCP) and critical power ratio(CPR) for the CANFLEX-RU fuel channels are compared with those for natural uranium fuel's channel such as CANFLEX-NU or the 37-element bundle. In addition, the effects of pressure tube creep and bearing pad height on the thermalhydraulic characteristics are examined by comparing the various results on the uncrept, 3.3% and 5.1% crept channels loaded with CANFLEX bundles with the 1.4 mm or 1.7 mm height bearing pads with those of the 37-element bundle.

2. ANALYSIS METHOD

2.1 Pressure Drop Model

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 at fuel dryout and melting. In the NUCIRC code, the single phase pressure drop is calculated by the K-Fuel model of equation (1), which is composed of skin frictional loss and form loss due to the appendages of a bundle.

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

In the above, $\Delta P_{1-\phi}$, Q, D_h, L, ρ and A denote the single phase pressure drop, channel flow, equivalent hydraulic diameter, fuel channel length, coolant density and flow area, respectively. K_{form} is the form loss coefficient. 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 the K-Fuel model of equation (1) is applied for the CANFLEX bundle and the 37-element bundle, the same coefficients and correction factors can be used for the two bundles except the form loss coefficients (K_{form}).

The CANDU-6 reactor is characterized by the occurrence of the coolant voids in many fuel channels at the 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 coolant and fuel channel and also by the friction between vapor and liquid, 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

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

In addition to the two phase friction multiplier(ϕ^2), the mass quality(X_a) and the void fraction(α) correlations, the OSV(Onset of Significant Void) correlation is needed to find the location of the initiation of boiling for the application of the above equation. Friedel's two phase friction multiplier, Saha-Zuber's mass quality and Armand-Massena's void fraction correlations were identically applied for the CANFLEX and 37-element bundles, but the OSV correlation is different for the two bundles. For the CANDU fuel bundle, the initiation of boiling is defined at the OSV point instead of saturation due to the enthalpy imbalance in the subchannels. The OSV points were obtained at the transition location from the single-phase to the two-phase flow region with Stern Labortories pressure drop measurements[6]. The OSV quality correlation for a CANFLEX bundle is expressed in equation (3). This correlation is applied for calculating the BLA(boiling-length-averaged) CHF as well as for predicting the two phase pressure drop.

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

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

$$E = \frac{D_{P/T} - D_{bundle}}{D_{P/T} - D_{inner}} \tag{4}$$

$D_{P/T}$ is

the diameter of the pressure tube, D_{bundle} is 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[5] were completed in the horizontal channel of Stern Laboratories by KAERI/AECL joint investment. The test series in the 3.3% and 5.1% peak crept channels as well as in the uncrept channel were performed in order to consider the pressure tube creep deformations in the plant's operating life time. The test fuel bundle string had the axially and radially non-uniform heat flux. In addition to the existing 1.4mm height bearing pads (low BP), 1.7mm and 1.8mm height bearing pads (high BP) were attached to the test bundle[7] designed to increase the gap flow at the bottom portions of the bundle 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, the CHF correlation for 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 OSV point, respectively.

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

Generally, the BLA concept was introduced to account for the AFD (axial flux distribution) effect on CHF. The BLA CHF values showed more consistency and less data scattering with the critical quality than the local CHF values in the crept channels[7]. The correlation was finally composed of the dimensionless parameters as shown in equation (6) and so can be directly applied to the heavy water fuel channel.

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

where

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

The above equation has the $b_1 \sim b_6$ coefficients, which are the function of bundle eccentricity to account for the local geometric effects on the BLA CHF of the CANFLEX bundle. Thus, the correlation is applicable for predicting the CHF values in the various crept channels.

2.3 Calculation Conditions

The thermalhydraulic analysis of the CANDU-6 reactor's fuel channel loaded with CANFLEX-RU bundles was done under the inlet header temperature of 265 °C, the outlet header pressure of 9.99 MPa and the header-to-header pressure drop of 1342 kPa boundary conditions of the NUCIRC code. This code is incorporated with the recent models for pressure drop and CHF of the CANFLEX fuel bundle as well as the 37-element bundle as described in the previous sections.

For the CANFLEX or 37-element fuel bundle, the NUCIRC code input parameters only relate to the fuel channel in the heat transport system of the CANDU-6. The form loss factors in the pressure drop model and CHF options have the different values between the two bundles, and mid-plane spacer loss coefficient of CANFLEX bundle attached with CHF enhancement buttons is much greater than that of 37-element bundle. The CHF of RU fuel bundle was calculated by applying the CHF correction factor 0.95 to the correlation of a CANFLEX bundle as described in the previous sections in order to account for the radial flux distribution effect. The BLA CHF correlation can account for the axial flux distribution effect by itself, so there is no need to apply the CHF correction factor for the axial flux distribution effect of the RU fuel bundle. So, it was assumed that the radial flux distribution of the 0.92% enrichment RU fuel bundle would bring about a 5% CHF reduction compared to the natural uranium fuel bundle.

The pressure tube of the CANDU-6 reactor suffers from the dimensional changes of diametral creep, axial elongation and creep sag resulting from the effects of neutron flux, stress and reactor operating temperatures over the plant's life time. In those variations, the diametral creep of the pressure tube affects the subchannel flow directly and it is known to have an exit skewed cosine-shaped profile along the fuel channel. Thus, the calculations were done for the 3.3% and 5.1% peak crept tubes as well as the uncrept tube as shown in Figure 1, where the maximum diametral creep was located at a 4.3 m (the 3.3% crept) and at a 4.8 m (the 5.1% crept) axial distance, and these profiles representatively simulate the actual fuel channels with plant ageing and were used for the water CHF tests.

3. ANALYSIS RESULTS AND DISCUSSIONS

3.1 Channel Flow Rate and Channel Exit Quality Distributions

Figure 2 shows the distributions of the channel flow rate and channel exit quality of the CANDU-6 reactor channels loaded with CANFLEX-RU fuel bundles with respect to the core radius. The flow rates of channels, which are located in the inner-core, maintain a high flow more than 25 kg/s, but the flow rates in the outer-core are linearly decreasing as the core radius increases. The CANDU-6 reactor designed to have almost the same enthalpy rise in all the fuel channels causes the trend of the channel flow rate distribution. That is, the high power channels shall have high flow rates and the low power channels shall have low flow rates. Thus, the distribution of channel flow for CANFLEX-RU fuel bundles is very similar to the typical flow distribution of CANDU-6 reactor channels loaded with the natural uranium fuel bundles. Figure 2 also shows that the distribution of channel exit quality is opposite to that of the flow rate, where the exit qualities in the outer-core channels are linearly increasing as the core radius increases.

The channel flow rate increases and the exit quality decreases as the creep rate of the pressure tube increases, and this trend is more sensitive in high power channels than in low power channels as shown in Figure 3, which indicates that the increase of flow rate due to the crept 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 a high power/flow region is much greater than in a low power/flow region, the channel flow rate in the inner-core is relatively more affected by pressure tube creep than in the outer-core region.

In this paper, the separate CHF correlation options in the NUCIRC code were applied for the CANFLEX bundles with high bearing pads and with low bearing pads but the same form loss coefficients were used for the two bundles, so there were no changes of flow rates for the two bundles. Figure 4 shows the relative flow rates of CANFLEX-RU fuel bundles to the flow rates of the 37-element fuel bundles in the uncrept, 3.3% and 5.1% crept channels. In the case of the uncrept channel, the flow rates of the CANFLEX-RU fuel bundles increase by about 0%□2% in most channels compared with those of the 37-element fuel bundles, and in the case of the 3.3% and 5.1% crept channels, the changes of flow rate are negligible in the inner-core but are scattered (-2%□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 distributions as well as the input parameters for the fuel channel pressure drop such as the flow area and form loss coefficients.

3.2 Critical Channel Power and Critical Power Ratio Distributions

Figure 5 shows the distributions of CCP and CPR for the CANFLEX-RU fuel bundles attached with the low and the high bearing pads, respectively, in the uncrept pressure tube. It is found that the CCP of CANFLEX-RU fuel bundle attached with the low bearing pads in the uncrept pressure tube is maintained just above 9 MW in the inner-core, but is linearly decreasing in the outer-core, so the trend is very similar to the distribution of the channel flow rate. The CPR of CANFLEX-RU fuel bundle attached with the low bearing pads in the uncrept pressure tube is maintained above 1.455 at least in the inner-core, but is rather gradually increasing in the outer-core, although the CCP is decreasing, due to the relatively low channel powers. In Figure 5, many channels in the inner-core have CPR values close to the minimum CPR, and so the minimum CPR channel location is sensitive to the channel power and the axial power distributions. These trends of CCP and CPR can also be found for all the other cases of crept pressure tubes and natural uranium fuel bundles.

In Figure 6 and Figure 7, the CCP of the CANFLEX-RU fuel bundle with the low bearing pads is compared with the CCPs of the CANFLEX-NU bundle with the low bearing pads and the 37-element bundle, respectively, and in Figure 8 and Figure 9, the CCP of the CANFLEX-RU fuel bundle with high bearing pads is also compared with the CCPs of the CANFLEX-NU bundle with the high bearing pads and the 37-element bundle, respectively. Figure 6 and Figure 8 indicate that the CCP of CANFLEX-RU fuel bundle is very close to that of the CANFLEX-NU bundle in the case of the uncrept pressure tube and is increased by about 2%□4% in the case of 3.3% crept and 5.1% crept tubes. This means that the CCP of the CANFLEX-RU fuel would not be reduced but rather increased by the axial flux distribution of the RU fuel bundle, compared to the CANFLEX-NU bundle although the CHF decrease of 5% was assumed by the radial flux distribution of the RU fuel. The axial flux distributions of the RU fuel bundle[9] are flat or slightly concave in the channel center region and the peak heat flux locations are moved upstream in the fuel channel, compared to those of the natural uranium fuel(cosine-shaped profiles). So, the relatively low local heat flux of the RU fuel around the downstream in the fuel channel would make the CCP increase. Figure 7 and Figure 9 indicate that the CCPs of the low bearing pads CANFLEX-RU fuel bundle are increased by about 2%, 6% and 8% in the uncrept, 3.3% crept and 5.1% crept tubes, respectively, and the CCPs of the high bearing pads CANFLEX-RU fuel bundle are also increased by about 4%, 8% and 14% in the uncrept, 3.3% crept and 5.1% crept tubes, respectively, compared with those of the 37-element bundle in the inner-core.

3.3 Pressure Tube Creep Effects

The effects of pressure tube diametral creep on the operating conditions of the reactor heat transport system are known to increase in a core flow with the rise of inlet header temperature and the reduction of header-to-header pressure drop. In addition, the low power/flow channels will suffer a reduction in flow while the high power/flow channels have a significant increase in flow because the outer-core channels are relatively less crept than the inner-core channels. But, it is difficult to predict the actual operating conditions because the pressure tube creep is slowly occurring for a long term over the plant's life time, accompanied with the heat transport system ageing such as crud depositions and flow-accelerated corrosion on the steam generator and feeder pipes, which increase the loop hydraulic resistance and are highly dependent on the plant operating histories. Thus, in this paper, the fixed operating conditions of the heat transport system were used in the case of the various crept channels in order to examine only the effects of 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 due to the subchannel flow re-distribution. Therefore, it is known that pressure tube creep will cause a reduction in CCP due to an increase in the bypassing flow, which is not effective for heat transfer through the inner subchannels of the bundle, although the channel flow increases. Figure 3 indicates that the flows in the 3.3% and 5.1% crept channels are about 10% and 16% greater than the flows in the uncrept channel in the inner-core region, but the relative increase in channel flows decrease rapidly to about 2% and 3% in the outer-core region. This means that the channel flow rate in a high power/flow region is relatively more affected by pressure tube creep than in a low power/flow region, because the fuel channel pressure drop portion of the header-to-header pressure drop in a high power/flow region is much greater than in a low power/flow region.

In Figure 10 and Figure 11, the CCPs in the crept channels are compared with those in the uncrept channel loaded with the CANFLEX-RU fuel bundle with the low bearing pads and the high bearing pads, respectively, as with the results for a 37-element bundle. Figure 10 shows that the CCPs of the CANFLEX-RU fuel bundle with low bearing pads in the 3.3% crept and 5.1% crept channels in the inner-core are about 2% and 4% less than those in the uncrept channel, while the CCP decreases for the 37-element bundle are about 6% and 12%, respectively. This means that the CANFLEX-RU fuel bundle is considerably less sensitive to the CCP reduction due to pressure tube creep than the 37-element bundle. Especially, Figure 11 indicates that the CCPs of the CANFLEX-RU fuel bundle with the high bearing pads in the 3.3% crept and 5.1% crept channels in the inner-core are about 3% less than those in the uncrept channel. 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

The bearing pads are 1.45 mm height 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[5], it was found that most CHF points were located on the bottom rods of the bundle and the flow imbalance was getting severe in the crept pressure tubes. So, a slight increase(about 0.3mm□0.4 mm) of bearing pad height was designed to improve the dryout power of the CANFLEX bundle, and the validation of the thermal performance was completed by the additional water CHF tests[7] of the CANFLEX fuel bundle attached with the high bearing pads.

The raised bearing pads on the outer elements keep the fuel bundle string closer to the central position of the pressure tube and increase the axial gap flow in the bottom positions of the bundle, thus increase the dryout power, compared with the CANFLEX fuel bundle with the low bearing pads. As an example of enthalpy imbalance reduction due to the raised bearing pads, it could be observed that the temperature difference between the bottom elements and top elements was rapidly decreased from 42 °C for the 1.4 mm height bearing pads to 35 °C for the 1.7 mm height bearing pads, based on the same single-phase flow condition data in the uncrept channel. As mentioned previously in Figure 7 and Figure 9, the CCP gains of the RU fuel bundle are about 2%, 6% and 8% in the uncrept, 3.3% crept and 5.1% crept tubes, respectively, for the low bearing pads, and the gains are about 4%, 8% and 14% for the high bearing pads, compared with those of the 37-element bundle. Based on these relative CCP gains for the two bundles, the CCP enhancement of the CANFLEX-RU fuel bundle due to the raised bearing pads is estimated to be about 2%□6%.

4. CONCLUSION

- The distributions of channel flow rate, channel exit quality, CCP and CPR of the CANFLEX-RU fuel bundle show the typical trends of CANDU-6 reactor channels loaded with the natural uranium fuel, and the CPRs of the CANFLEX-RU fuel bundle in the uncrept channel is maintained above 1.455 at least in the inner-core region.

- The amount of CCP reduction due to pressure tube creep for the CANFLEX-RU fuel bundle is much less than that for the 37-element bundle, and furthermore, the raised bearing pads can decrease such a CCP reduction significantly.

- The CHF enhancement and axial flux distribution of the CANFLEX-RU fuel bundle cause CCP increases of about 2%, 6% and 8% for the low bearing pads, and also cause CCP increases of about 4%, 8% and 14% for the high bearing pads in the uncrept, 3.3% crept and 5.1% crept channels, respectively, compared with the CCPs of the 37-element bundle. So, the CCP enhancement of the CANFLEX-RU fuel bundle due to the raised bearing pads is estimated to be about 2%□6%.

ACKNOWLEGEMENT

This work has been carried out under the Mid- and Long-Term Nuclear R & D Programs pushed by the Korea Ministry of Science and Technology

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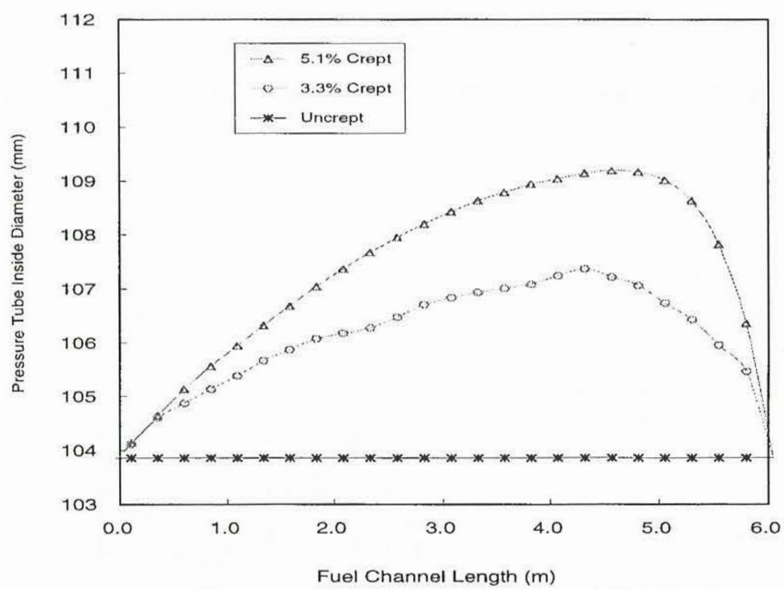
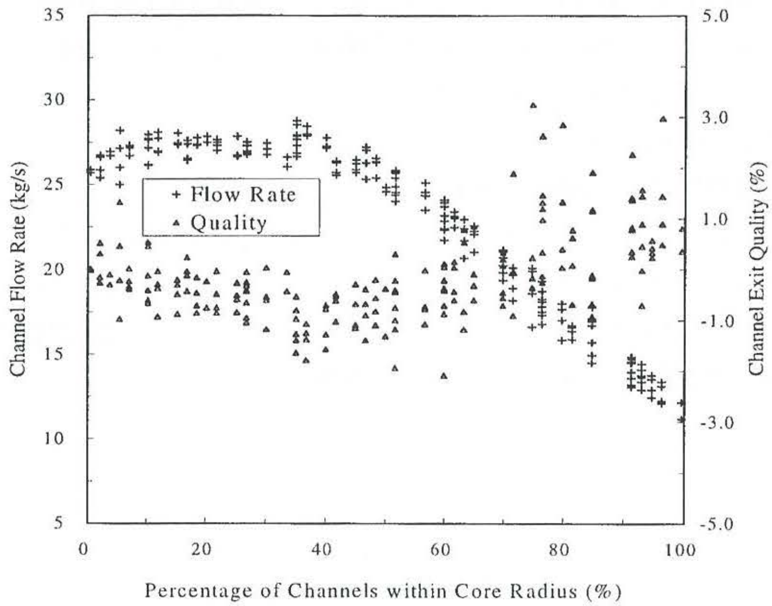
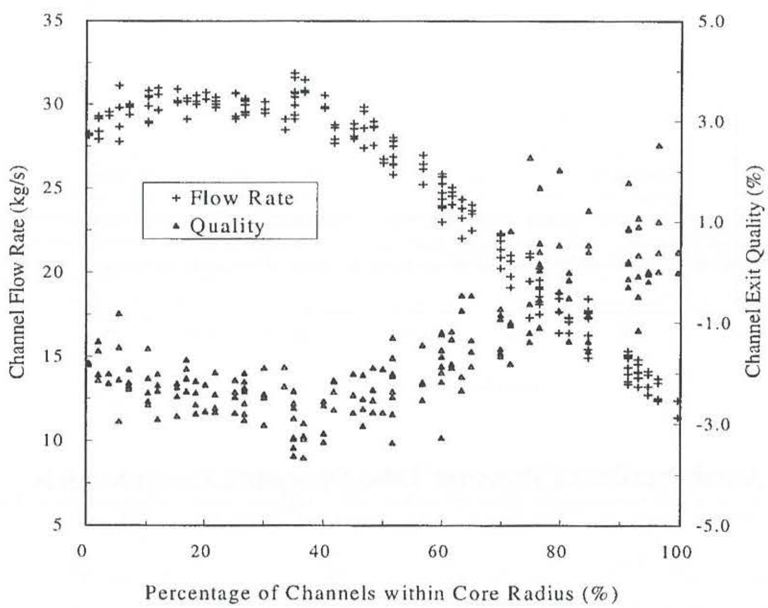


Figure 1. Axial Profiles of Pressure Tube Diametral Creep Models



(a) In the Uncrept Channel with the CANFLEX-RU Bundle Attached with the Low Bearing Pads



(b) In the 3.3% Crept Channel with the CANFLEX-RU Bundle Attached with the Low Bearing Pads

Figure 2. Channel Flow Rate and Channel Exit Quality Distributions

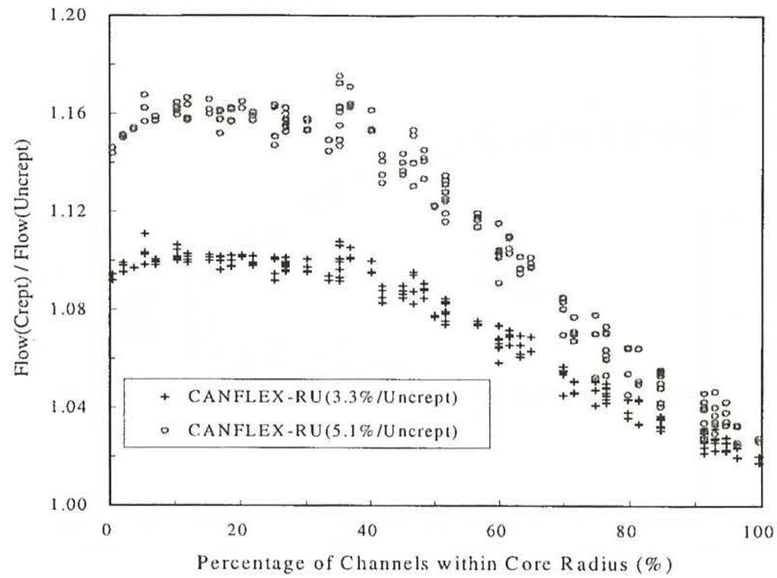


Figure 3. Comparison of Flows in the Crept Channels with Flows in the Uncrept Channel

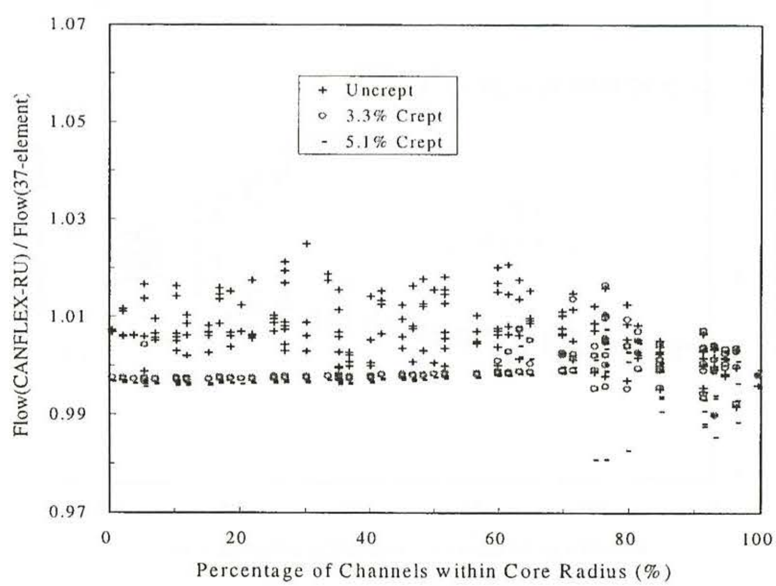
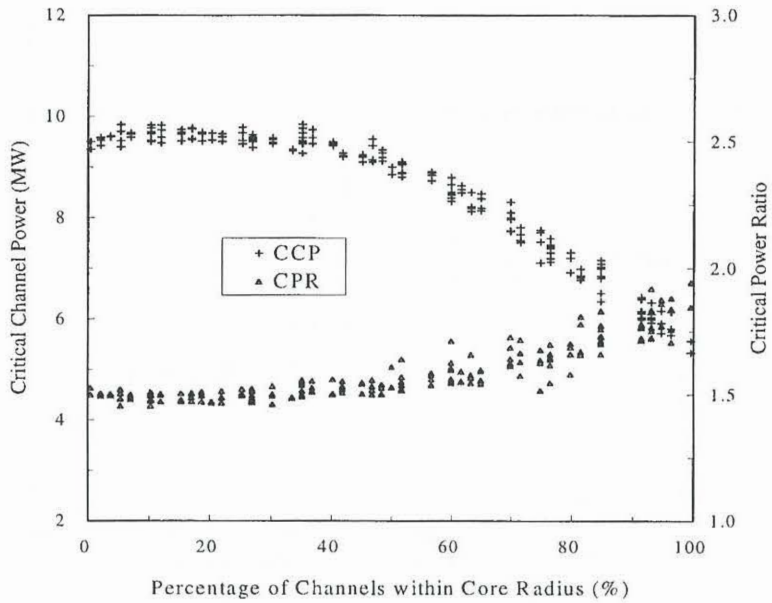
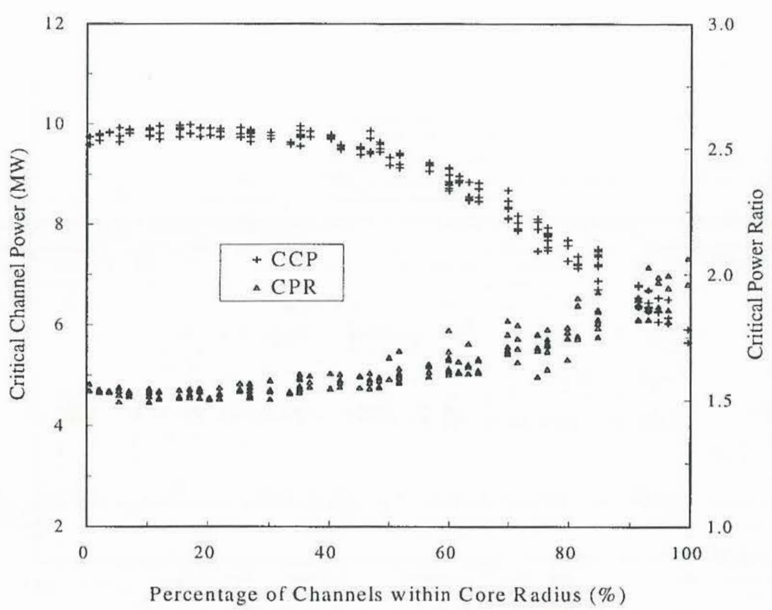


Figure 4. Comparison of Flow Rates for CANFLEX-RU Bundle with Flow Rates for the 37-element Bundle in the Uncrep, 3.3% and 5.1% Crept Channels



(a) In the Uncrept Channel with the CANFLEX-RU Bundle Attached with the Low Bearing Pads



(b) In the Uncrept Channel with the CANFLEX-RU Bundle Attached with the High Bearing Pads

Figure 5. Critical Channel Power and Critical Power Ratio Distributions

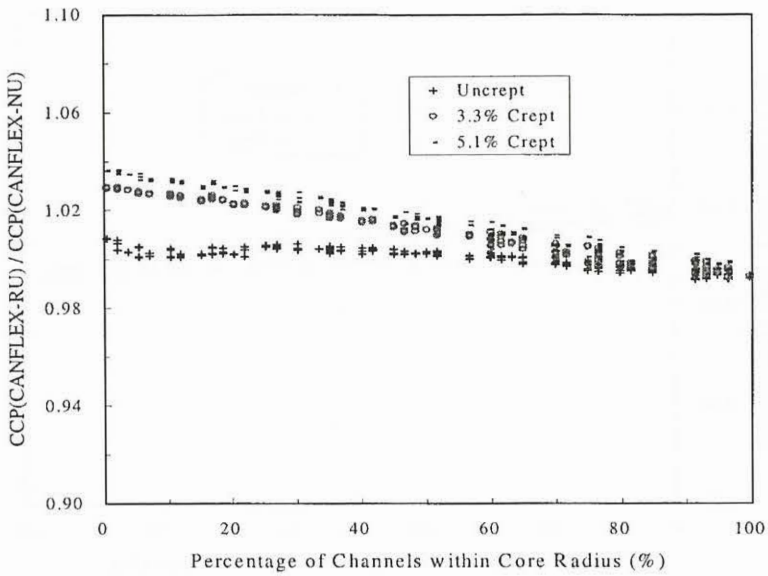


Figure 6. Comparison of the CCPs for the RU Bundle Attached with the Low Bearing Pads with Those for the NU Bundle with the Low Bearing Pads

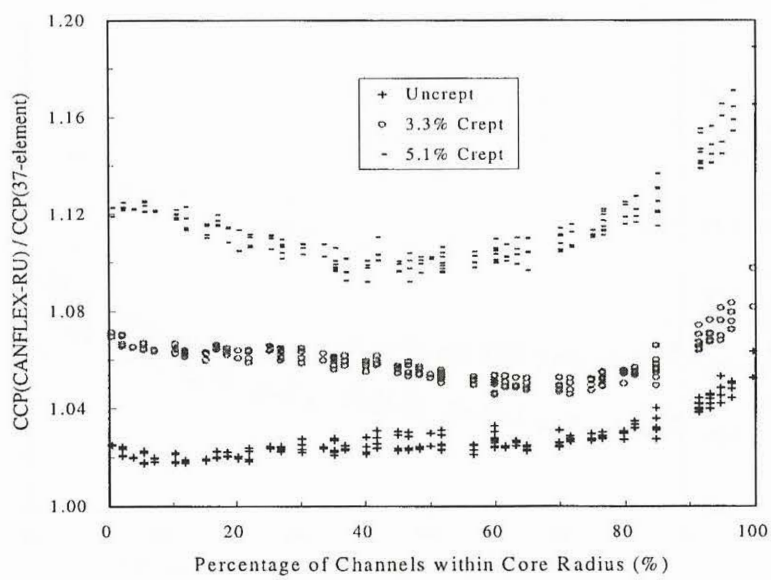


Figure 7. Comparison of the CCPs for the CANFLEX-RU Bundle Attached with the Low Bearing Pads with Those for the 37-element Bundle

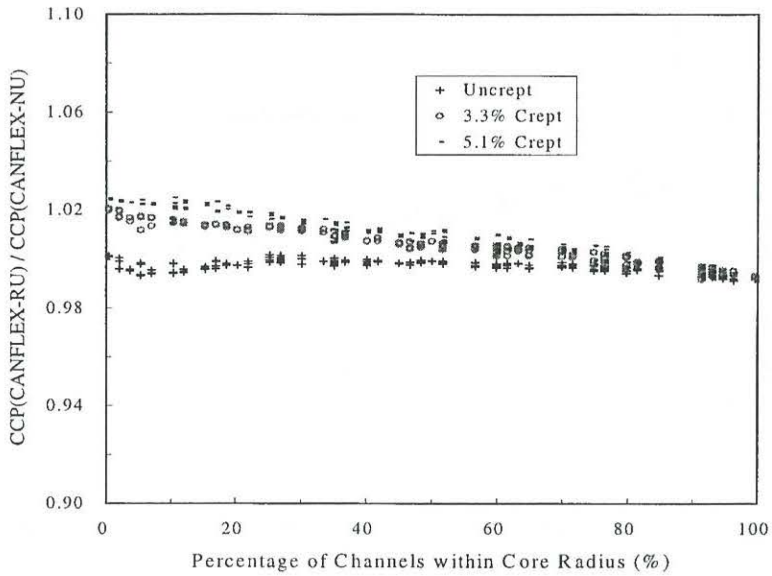


Figure 8. Comparison of the CCPs for the RU Bundle Attached with the High Bearing Pads with Those for the NU Bundle with the High Bearing Pads

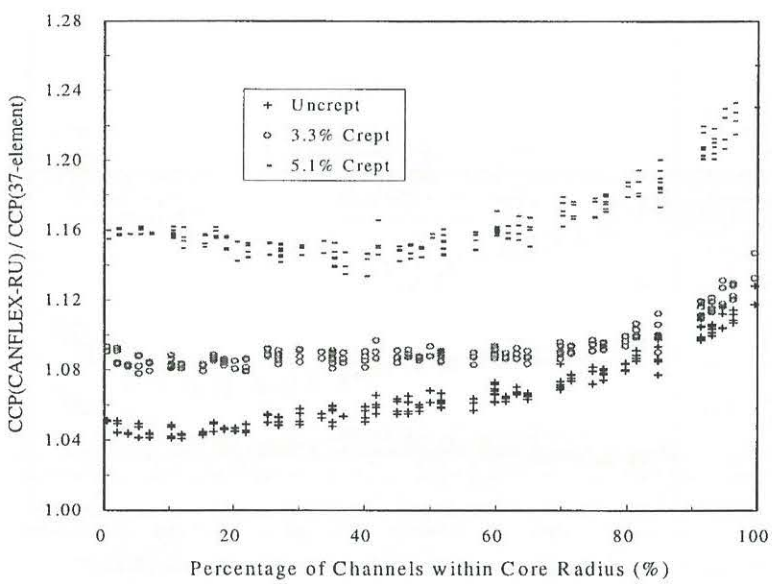


Figure 9. Comparison of the CCPs for the CANFLEX-RU Bundle Attached with the High Bearing Pads with Those for the 37-element Bundle

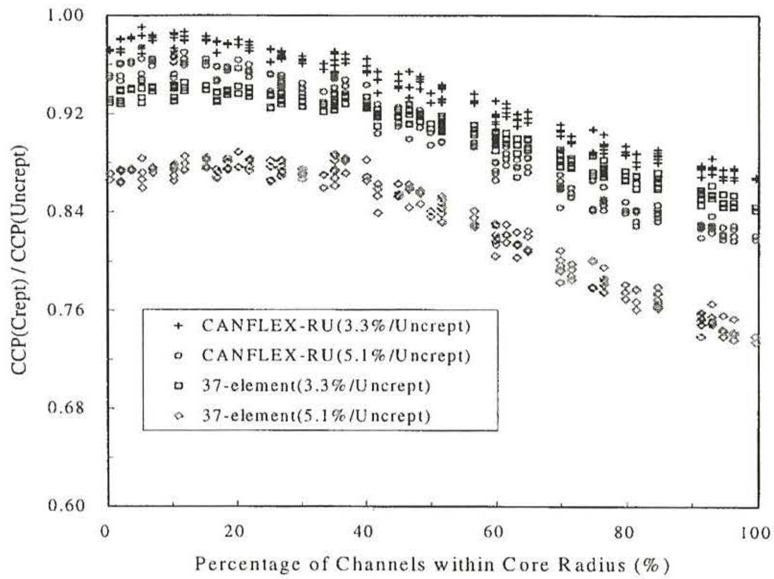


Figure 10. Comparison of the CCPs for the RU Bundle Attached with the Low Bearing Pads in the 3.3% and 5.1% Crept Channels with Those in the Uncrept Channel

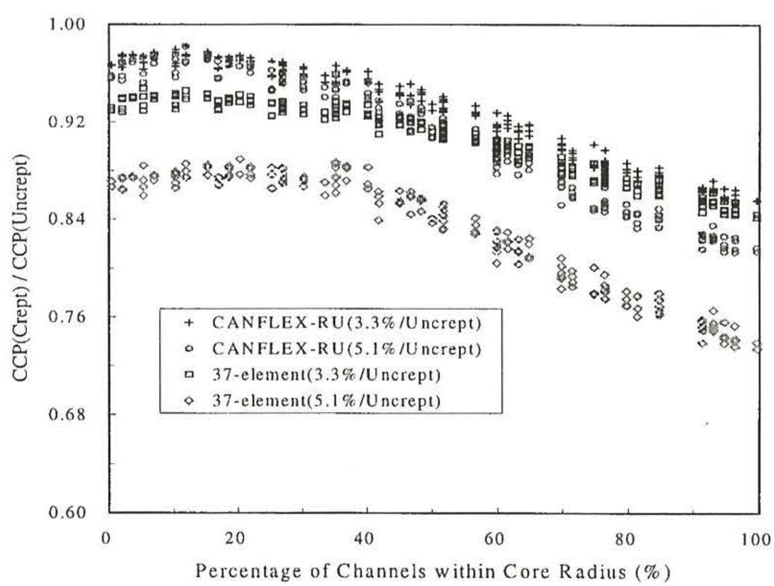


Figure 11. Comparison of the CCPs for the RU Bundle Attached with the High Bearing Pads in the 3.3% and 5.1% Crept Channels with Those in the Uncrept Channel