# ASSESSMENT OF SUBCHANNEL CODE ASSERT-PV FOR SUPERCRITICAL APPLICATIONS

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**ABSTRACT**

The Canadian subchannel code ASSERT-PV has been modified and used in the development of a fuel assembly concept for the Canadian Supercritical Water-cooled Reactor (SCWR). This paper describes the modification to ASSERT-PV for the SCWR applications, and presents an assessment of the code against available experimental data. Full-bundle tests for the SCWR fuel assembly are not yet available; therefore, partial-bundle tests with a reduced number of rods and a shortened channel length were considered in the assessment. Three heat transfer experiments at supercritical pressures were selected for the assessment: (i) a Japanese 7-rod bundle water experiment; (ii) a Chinese 2×2 rod bare-bundle water experiment; and (iii) a Russian 7-rod bare-bundle Freon experiment. The rod surface temperature was taken as the key parameter in the assessment since the maximum fuel cladding temperature is a key criterion used in developing the SCWR fuel assembly concept. The ASSERT-PV predictions of rod surface temperature were compared against experimental data of measured wall temperatures. Six widely-known heat transfer correlations were assessed for their suitability in predicting the wall temperatures in rod bundles under conditions relevant to the Canadian SCWR. Overall, the Jackson correlation was found to be the most suitable for predicting the wall temperatures in the range of conditions covered by the selected experiments.

**KEYWORDS**
Subchannel code, Supercitical Flow, Thermalhydraulics, Heat Transfer, ASSERT

## 1. INTRODUCTION

Canada is participating in collaborative research and development for the next generation nuclear energy systems through the Generation IV (Gen-IV) International Forum. Canadian Nuclear Laboratories (CNL, formerly AECL) has been leading a national Gen-IV program in developing a pressure-tube based Supercritical Water-cooled Reactor (SCWR). Recently a fuel assembly concept has been established for the Canadian SCWR concept. It is a flask-like structure consisting of 64 fuel rods distributed in two rings around a central flow tube of 94-mm in outer diameter [1, 2]. The Canadian subchannel code ASSERT-PV, specifically modified for SCWR applications (V3R1m2), has been used in the development and thermalhydraulic performance optimization of the SCWR fuel assembly concept [2].

The ASSERT code has been employed by the Canadian nuclear industry in subchannel thermalhydraulic analysis of existing fuels and new fuel concepts in the past three decades. For single- and two-phase flows at subcritical pressures, ASSERT-PV [3] has been developed and qualified by CNL for prediction of flow and enthalpy distribution [4], critical heat flux (CHF) [5], and post-dryout (PDO) cladding temperatures [6] in fuel-bundle subchannels of a CANDU fuel channel. For flows at supercritical pressures, on the other hand, the recent Canadian SCWR program has resulted in the development of an interim version of the code, ASSERT-PV V3R1m2, for the SCWR applications.

ASSERT-PV V3R1m2 is essentially the same as the latest version ASSERT-PV 3.2 [3] as far as single-phase (or homogeneous two-phase) flows are concerned; the latter has incremental improvements in two-phase flow model of flow distribution, CHF, and PDO heat transfer [3]. ASSERT-PV V3R1m2 was enhanced to model supercritical flows under the SCWR conditions while retaining the flexibility in modelling a variety of subchannel geometries, fluids and flow orientations for CANDU PHWR, PWR and BWR applications. The primary modifications include an enhanced water property package HLWP 1.0 [7] covering supercritical flow conditions and a number of widely known empirical correlations for heat transfer in supercritical flows. These heat transfer correlations are described in subsequent sections.

The HLWP (heavy and light water properties) package includes a complete library of FORTRAN 77 subroutines capable of providing thermodynamic and transport properties of heavy and light water. The new version HLWP 1.0 [7], as compared to the previous version HLWP 0.1, includes a set of light water properties that extends into the supercritical pressure region. All the other aspects of this property package remain unchanged. In addition, the new version HLWP 1.0 is based on IAPWS-95 scientific formulation of water properties, which is more accurate than the subsequent IAPWS-97 formualtion. The 1997 formulation is a re-implementation of the 1995 formulation in which a slight loss of accuracy is traded off in favour of the computational efficiency of the formulation. Since accuracy, smoothness, and consistency of the source are important in the fitting procedure used to create the HLWP 1.0 routines, the 1995 formulation was chosen for use [7].

It should be noted that ASSERT-PV V3R1m2 can be used to model either sub- or supercritical flows, but not the transition between the two. This was judged to be sufficient, since modeling the transition is not required in a subchannel analysis at the stage of the SCWR concept development.

Owing to the fact that experimental data are unavailable for the Canadian SCWR fuel assembly concept, ASSERT-PVsubchannel code was assessed against the experimental data from partial-bundle tests conducted at operating conditions relevant to the Canadian SCWR. The objective of this paper is to assess the Canadian subchannel code ASSERT-PV for predicting flow and heat transfer under supercritical-flow conditions relevant to the Canadian SCWR, using partial-bundle experimental data that have become available. The experiments include (i) JAEA, Japan 7-rod bundle water experiment, an international benchmark [8, 9]; (ii) XJTU, China 2×2 bare-bundle water experiment [10]; and (iii) IPPE7, Russian 7-rod bare-bundle Freon-12 experiment [11, 12].

The code predictions were assessed against wall temperature measurements. Sensitivity studies were performed for six wall-to-coolant heat transfer correlations, including the well-known Dittus-Boelter [13] correlation and five newly implemented correlations that were developed to cover supercritical conditions. Note that experimental uncertainties (total uncertainties) in measured wall temperatures were not available for the experiments. Prediction statistics from comparison with the measured temperatures using each of six correlations in the subchannel analyses, such as bias and standard deviation, are not discussed in this paper due to length limitation; they will be presented in an extended paper.

## 2. SUPERCRITICAL ROD BUNDLE EXPERIMENTS USED FOR CODE ASSESSMENT

A brief description is provided in this section for each of the three experiments used in the assessment; detailed information is available in the referenced papers. Experimental conditions are summarized in Table I for the tests selected for the present assessment. The test numbers for the JAEA and IPPE experiments are the same as originally assigned by the investigators [8, 11], whereas those for the XJTU experiment are used herein only for the convenience of reference.

## 2.1. JAEA 7-Rod Bundle Water Experiment – an International Benchmark

The 7- rod vertical bundle with honeycomb spacers was the subject of a recent blind benchmark exercise to assess the capabilities of CFD and subchannel codes in predicting supercritical flows. Detailed synthesis and additional information pertaining to the benchmark can be obtained from [9]. The measurements, which were not released to the participants until the end of the exercise, consisted of wall temperatures at different axial, radial and azimuthal positions on the rods. By being part of a blind benchmark exercise, participants had no opportunity to fine-tune the models to improve the outcome. The experiment was conducted in a supercritical water test facility at JAEA, which consists of a rod bundle of seven hexagonally arranged heating rods, the details of which can be found in [8].

Figure 1 presents the schematic of a cross-section of the flow channel used for the assessment (subchannel and rod numbers will be referred to later). The flow channel wall was designed such that it represents more closely a subsection (a partial bundle) of a bundle of a larger triangular rod array. The rod diameter is 8 mm. All gaps are 1 mm including the rod-to-rod and rod-to-channel gaps. The length of the uniformly heated section is 1.5 m.

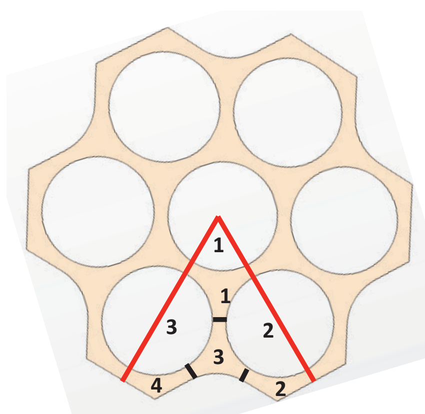


**Figure 1 Cross-sectional View of ASSERT PV Geometry Model for 7-Rod Bundle Water Tests.**

## 2.2. XJTU 2×2 Bare-Bundle Water Experiment

The 2x2 bundle tests provided measurements of circumferential wall-temperature distribution around the heated rods. The data reported in [10] corresponded to the measurements obtained at an axial location 10 cm upstream of the downstream end of the heated section.

The schematic of the cross-section of the flow channel used for simulations is presented in Figure 2 . The flow channel wall is rounded at its corners for a span of 90 degrees. The rod diameter is 8 mm. All gaps are 1.44 mm including the rod-to-rod and rod-to-channel gaps. The length of the uniformly heated section is 0.6 m.

## 2.3. IPPE 7-Rod Bare-Bundle Freon-12 Experiment

The experiment was conducted on a vertical 7-rod bare bundle cooled with supercritical Freon-12 at IPPE in Russia [11, 12]. The tests were performed at system pressures of ~4.65 MPa for several different combinations of wall and bulk-fluid (or simply bulk) temperatures that are below, at, or above the pseudo-critical temperature.  The measurements by IPPE included data on variation of wall temperature along the length of the test section.

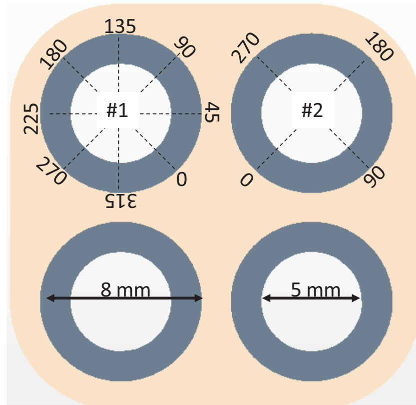


**Figure 2  Cross-sectional View of ASSERT-PV Geometry for 2x2 Rod Bundle Water Tests.**

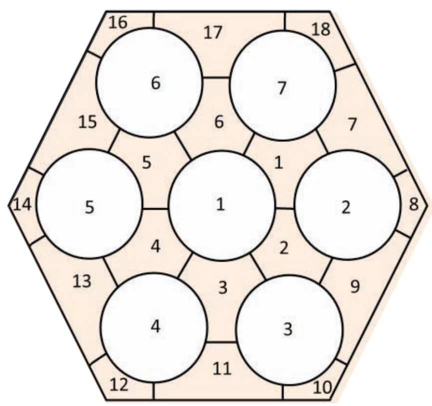


**Figure 3  Cross-sectional View of ASSERT PV Geometry Model for 7-Rod Bundle Freon Tests.**

Figure 3 shows the schematic of a cross-section of the flow channel.  The flow channel wall is hexagonal which is different from that in Figure 1.  The rod diameter is 9.5 mm.  The rod-to-rod gap is 1.8 mm and rod-to-channel gap is 1.3 mm.  The length of the uniformly heated section is 1.2 m.

## 3. ASSERT-PV MODELS OF SUPERCRITICAL HEAT TRANSFER

There are a plethora of empirical and semi-empirical correlations for wall-to-fluid heat transfer developed for supercritical flows, all based on flows inside circular tubes; thus their applicability to bundle flows, especially under the SCWR conditions, need to be assessed [1].  A total of five widely-known correlations were implemented into ASSERT-PV V3R1m2.  They are Kurganov-Ankudinov [14], Swenson et al. [15], Jackson [16], Bishop et al. [17], and Mokry et al. [18], which are referenced in tables and figures using the name of the first author for simplicity.  All correlations except Kurganov-Ankudinov are summarized in Table II, which also includes the Dittus-Boelter correlation that was used in comparison.  The correlations listed in Table II can be expressed in the following general form:

**Table I. Test Conditions for the ASSERT-PV Simulations**

| Experimental Data Set (Org., Bundle, Test#) | Pressure (MPa) | Inlet Temp. ($^{o}$C) | Mass Flux (kg/m$^2$.s) | Heat Flux (kW/m$^2$) | Fluid |
|---|---|---|---|---|---|
| **JAEA 7-rod, Test #B1** | 25 | 246.4 | 1448 | 597* | Water |
| **JAEA 7-rod, Test #B2** | 25 | 180.4 | 1433 | 905 | |
| **XJTU 2x2, Test #1** | 25 | 416.7 | 1000 | 400 | Water |
| **XJTU 2x2, Test #2** | 25 | 375.2 | 1000 | 400 | |
| **IPPE 7-rod, Test #3** | 4.65 (25) | 74.4 (311) | 509 (491) | 19.4 (344) | Freon-12 (water equivalent ǂ) |
| **IPPE 7-rod, Test #9** | 4.64 (25) | 119.3 (386) | 516 (498) | 43.5 (771) | |

* The centre rod has a lower heat flux of 522 kW/m$^2$.
ǂ Fluid-to-fluid modeling (see [12], e.g.).

$$h = \frac{k_x}{d_h} C Re_x^n \overline{Pr_x}^m F \tag{1}$$

Where: $h$, $k$ and $d_h$ are the heat transfer coefficient, thermal conductivity and hydraulic diameter, respectively; $C$, $F$, $n$, $m$ and $x$ are given in Table II; Re is the Reynolds number and $\overline{Pr_x}$ is the average Prandtl number defined by:

$$\overline{Pr_x} = \frac{\overline{C_p}\mu_x}{k_x} \tag{2}$$

The average specific heat, $\overline{C_p}$, is calculated from:

$$\overline{C_p} = \left(\frac{H_w - H_b}{T_w - T_b}\right) \tag{3}$$

In the above, $\mu$, $H$ and $T$ are the dynamic viscosity, enthalpy and temperature. The subscripts $w$ and $b$ denote values at the wall and bulk fluid. Note that unlike other correlations, all properties in the Swenson et al. correlation are based on wall temperature instead of bulk temperature.

The Kurganov-Ankudinov correlation is a semi-empirical one that was developed to account for heat transfer deterioration in supercritical-flow heat transfer. The formulation is provided in Appendix A.

The key thermalhydraulic performance parameter of the SCWR fuel assembly concept is the maximum cladding temperature, which is affected primarily by the heat transfer correlation [1]. Therefore, models other than the heat transfer correlations, such as the frictional pressure drop and inter-subchannel turbulent mixing, were not modified at this stage of conceptual design. All the "flow-distribution models", compared to heat transfer models, used in the present analysis are exactly the same as employed in the analysis of the Canadian SCWR [2] and in the submission to the international blind benchmark [9], all based on the recommended model set [3, 4] without any change. The fuel model in ASSERT-PV, required for accounting for the conjugated heat transfer (CHT), was not used since our "base-case" model set for the SCWR fuel assembly concept does not include a fuel model and since it is known that CHT plays only a secondary role in determining the cladding temperature distribution under the bundle and flow conditions relevant to the present assessment [8]. Geometry models specific to an individual experiment are described along with assessment results in the next section.

**Table II. Heat Transfer Correlation Parameters implemented and used in ASSERT-PV**

| Correlation | x | C | n | m | F |
|---|---|---|---|---|---|
| Dittus-Boelter | b | 0.023 | 0.8 | 0.33 | 1, with $\overline{\mathrm{Pr_x}}$ replaced by $\mathrm{Pr_b} = \frac{\mathrm{C_{p,b}}\mu_b}{\mathrm{k_b}}$ |
| Bishop | b | 0.0069 | 0.9 | 0.66 | $\left(\frac{\rho_w}{\rho_b}\right)^{0.43}$, with entrance effects neglected |
| Mokry | b | 0.0061 | 0.904 | 0.684 | $\left(\frac{\rho_w}{\rho_b}\right)^{0.564}$ |
| Jackson | b | 0.0183 | 0.82 | 0.5 | $\left(\frac{\overline{C_p}}{C_{p,b}}\right)^{nj}\left(\frac{\rho_w}{\rho_b}\right)^{0.3}$, with $\overline{Pr_x}$ $replaced\ by\ Pr_b = \frac{C_{p,b}\mu_b}{k_b}$ |
| Swenson | w | 0.00459 | 0.92 | 0.613 | $\left(\frac{\rho_w}{\rho_b}\right)^{0.231}$ |

Jackson Exponent, *nj*

$nj = 0.4$ for $[(T_{rb} \le T_{rw})$ and $(T_{rw} \le 1)]$ or $[(T_{rb} \ge 1.2)$ and $(T_{rw} \ge T_{rb})]$
$nj = 0.4 + 0.2(T_{rw} - 1)$ for $[(T_{rb} \le 1)$ and $(T_{rw} \ge 1)]$
$nj = 0.4 + 0.2(T_{rw} - 1)[1 - 0.5(T_{rb} - 1)]$ for $[(T_{rb} \ge 1)$ and $(T_{rb} \le 1.2)$ and $[(T_b \le T_w$
$T_{rb} = T_b / T_{pc}$
$T_{rw} = T_w / T_{pc}$
$T_{pc}$ = pseudocritical temperature, *b* = bulk, *w* = wall. All temperatures used in the correlations are in degrees K.

## 4. ASSESSMENT OF ASSERT-PV WITH EXPERIMENTS

Assessment of ASSERT-PV with experimental data described in Section 2 is presented in this section. Comparisons of predicted and measured wall temperatures for the selected tests (Table I) are presented in Figure 5 through Figure 9. Comparison was made for the six heat transfer correlations listed in Table II. Note that the Jackson correlation was used in the optimization of the SCWR fuel concept [2], and the implication was discussed in [1].

### 4.1 JAEA 7-Rod Bundle Water Tests

Two tests were selected from the experiment, a low (B1) and a high (B2) heat-flux case as described in Table I. The two tests were the only ones used in the international blind benchmark [9] for wall temperature predictions. The measured wall temperatures as well as bulk temperatures for Test B1 are well below the psudo-critical point (~385 °C), whereas those for Test B2 are well above.

In modeling the benchmark geometry, a 1/6 symmetry was assumed resulting in a total of three rods or partial rods (R1: centre, R2 and R3: outer rods), and four subchannels (S1: inner, S2 to S4: outer subchannels) as presented in Figure 1. Since the two outer subchannels (S2 and S4) were identical by symmetry, there were only four wall temperatures as independent output at any axial location; i.e., R1-S1, R2-S1, R2-S3 and R2-S2. Since the wall temperatures provided by the benchmark are all at an outer-rod facing an inner subchannel (at different axial locations for different rods), only the axial distribution of wall temperatures at R2-S1 was required for comparison with the benchmark data.

An axial grid of 60 uniformly distributed nodes (control volumes) was employed. As with all ASSERT-PV simulations, grid convergence tests were performed to ensure that the grid system is sufficient for the accuracy required in the present analysis. The JAEA 7-rod bundle had five grid spacers, of 2.5 cm in length, installed respectively at 0.0, 0.3, 0.7, 0.9, 1.3 m from upstream end of the heated section. They were modeled using a k-factor of 0.7 provided by reference [8] for a similar spacer grid. All the models including the axial grid are the same as used in our blind-benchmark submission, in which the result using the Jackson correlation was included (see the benchmark synthesis paper [9]).

Figure 4 shows the result of comparison for the lower heat flux case (B1) where the wall and bulk temperatures are well below the pseudo-critical point. The Jackson and the Kuganov-Ankudinov correlations resulted in wall temperatures that are very close to the Dittus-Boelter prediction; the three are barely distinguishable from each other in the figure. By comparison, the prediction is somewhat higher by Bishop et al., and slightly higher still by Mokry et al. and Swenson et al. (the two appears to be overlapping in the figure). The Jackson correlation stood well in this comparison.

Figure 5 shows the result of comparison for the higher heat flux case (B2) where wall temperatures and bulk temperatures are well above the pseudo-critical point. The Dittus-Boelter, Mokry et al. and Jackson correlations predicted the experiments reasonably well, with Mokry et al. closer to the measurements only near the outlet of the heated section. The Kurganov-Ankudinov and the Bishop et al. correlations predicted similar values, whereas the Swenson et al. predicted a temperature rise and fall, similar to a deteriorated heat transfer not observed in the experiment; i.e., a rise in wall temperature at ~0.8 m and a drop at ~1.0 m.

It is worth noting that among the three experiments, this is the only one that provided data of channel pressure drop. There was a third case of the blind-test benchmark for the pressure drop prediction, Test A1, for which ASSERT-PV provided the most accurate prediction among all participants (about 10 in total), with a relative prediction-measurement difference of ~3% [9].

### 4.2 XJTU 2×2 Bare-Bundle Water Tests

Two tests were selected from the experiment as described in Table I. Test #1 is a case with high inlet temperature where the bulk temperature is well above the pseudo-critical point at the axial location of rod circumferential temperature measurement (0.5 m [10]). Test #2 is a case with an inlet temperature slightly below the pseudo-critical point where the bulk temperature is near the pseudo-critical temperature of ~385 °C at 0.5 m. The heated length is 0.6 m, but only the upstream 0.5 m was modeled in the present analysis. An axial grid of 100 uniformly distributed nodes was employed. All the models including the axial grid are the same as used in a previous paper [1], in which the result using the Jackson correlation was included.

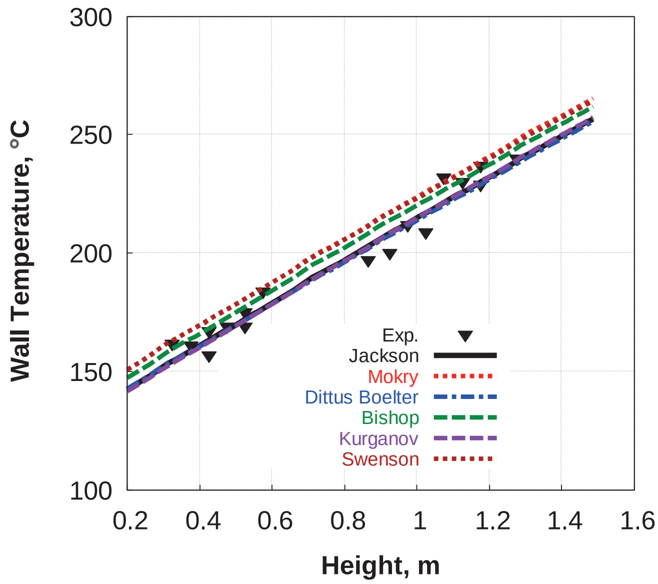


**Figure 4 Axial Variation of Wall Temperature of 7- Rod Bundle with Honeycomb Spacers: below Pseudo-Supercritical Point (Test B1).**

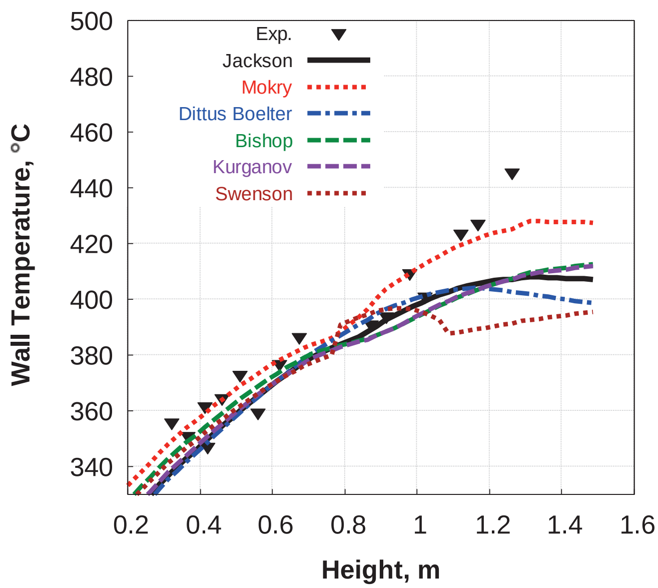


**Figure 5 Axial Variation of Wall Temperature of 7- Rod Bundle with Honeycomb Spacers: above Pseudo-Supercritical Point (Test B2).**

For both tests #1 and #2, as seen in Figure 6 and Figure 7 respectively, the six correlations predicted significantly varied circumferential wall temperature distributions (at Rod 1 shown in Figure 2). Overall for the two tests, the Jackson correlation and the Kurganov-Ankudinov correlation predicted the wall temperature distribution that agrees better with measurements than the other correlations.

For Test #1, the bulk temperature is about 431 °C, well above the pseudo-critical point. As shown in Figure 6, the ASSERT code (with the Jackson option) captured the wall temperature distribution well, but slightly over-predicted the wall temperature at the narrow-gap region (i.e., 180°) and slightly under-predicted that at the centre subchannel region (i.e., 0°). The difference is largely attributed to the neglect of CHT via the heated wall, which was not included in this assessment as explained in Section 3. However, this should not affect the correlation-to-correlation comparison in this analysis as demonstrated here. All the other correlations except Kurganov-Ankudinov did not performed as well in predicting the experiment. The Mokry et al. correlation predicted the highest maximum temperature (and thus the greatest deviation from the experimental peak cladding temperature), followed by the Swenson et al.

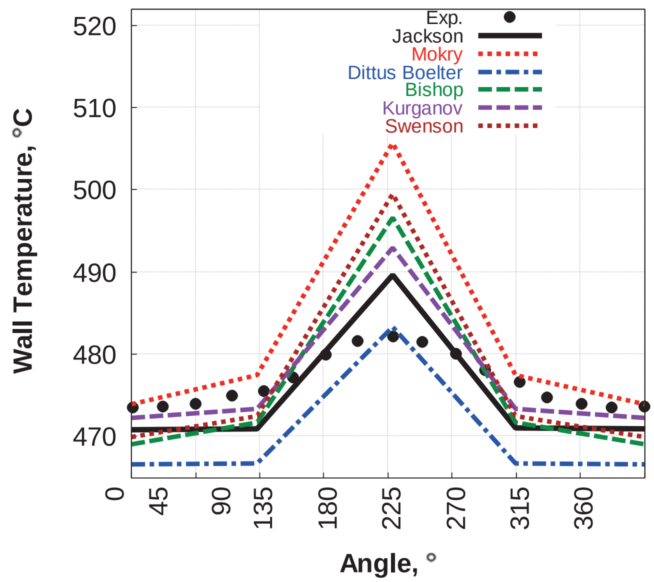


**Figure 6 Rod Circumferential Wall-Temperature Variation of 2×2 Bundle: Above Pseudo-Critical Point (Test#1).**

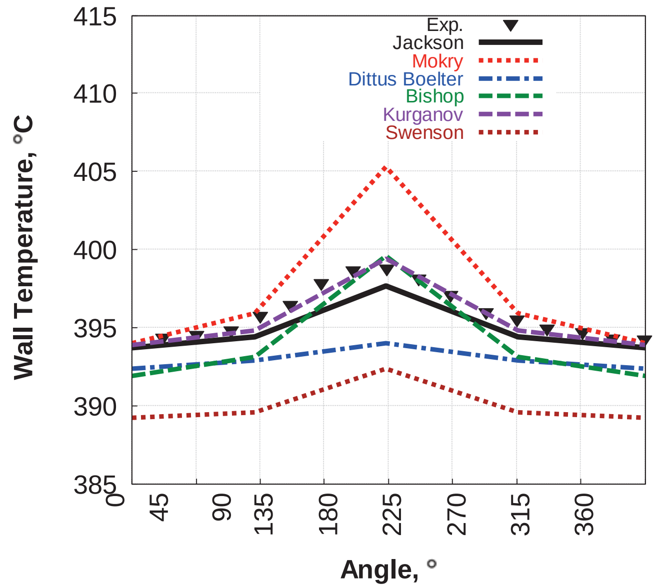


**Figure 7 Rod Circumferential Wall-Temperature Variation of 2×2 Bundle: Near Pseudo-Critical Point (Test #2).**

For Test #2, the bulk temperature is about 384 °C, well within the region where fluid properties undergoing sharp variations (at or near the pseudo-critical point). As seen in Figure 7, the ASSERT code (with the Jackson option) predicted the wall temperature distribution well, with a slight under-prediction of the peak (at the rod-to-flow-tube gap region). Again all the other correlations except Kurganov-Ankudinov fared less well in comparison. The Mokry et al. correlation predicted the highest maximum temperature whereas the Swenson et al. predicted the lowest.

It was pointed out in [1] that the most recent correlation, Mokry et al., was developed as an improvement over other existing correlations but it was found to over-predict cladding temperatures for subchannel sizes (equivalent diameters) significantly smaller than 10 mm. The results shown in Figure 6 and Figure 7, with the smallest subchannel at the corner (at around 180°) being ~3 mm, confirmed the observation.

### 4.3 IPPE 7-Rod Bare-Bundle Freon-12 Tests

Two tests were selected from the experiment, a low (#3) and a high (#9) heat-flux case as described in Table I. The two tests were the ones that had sufficient descriptions and phenomenon observations by the investigators [11]. The wall temperatures were measured along the heated length at three circumferential locations at the centre rod: TC1, TC2 and TC3, 180° apart from each other and each facing a surrounding subchannel. Due to the symmetry conditions, only one predicted wall temperature distribution, at R1-S1, was required for comparison against the measurements. An axial grid of 100 uniformly distributed nodes was employed in the ASSERT analysis for the heated length of 1.0 m.

For the lower heat flux test, Test #3, where both the wall and bulk temperatures are well below the pseudo-critical point, a good agreement was obtained between measured and predicted wall temperatures along the heated length, as shown in Figure 8, practically for all tested correlations. The slight under-prediction is not significant considering the data scatter among the three radial locations, where the measured temperatures at TC1, TC2 and TC3 would have been identical without experimental uncertainty.

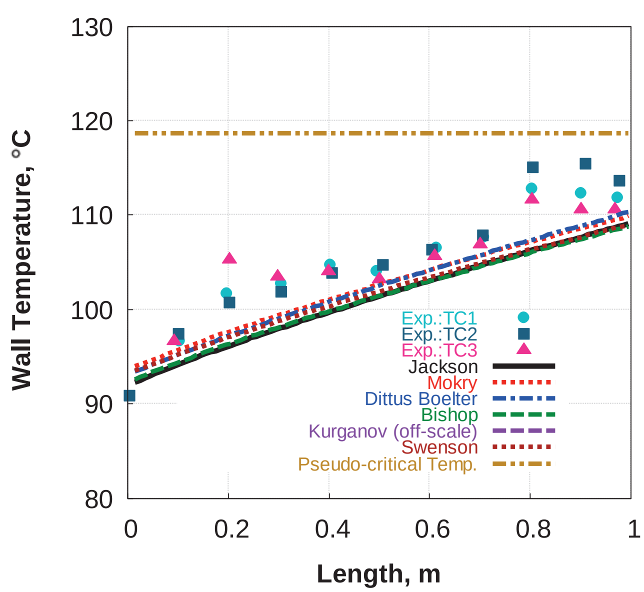


**Figure 8 Axial Wall Temperature Variation of 7-Rod bundle Freon Test: Below Pseudo-Critical Point (Test #. 3).**

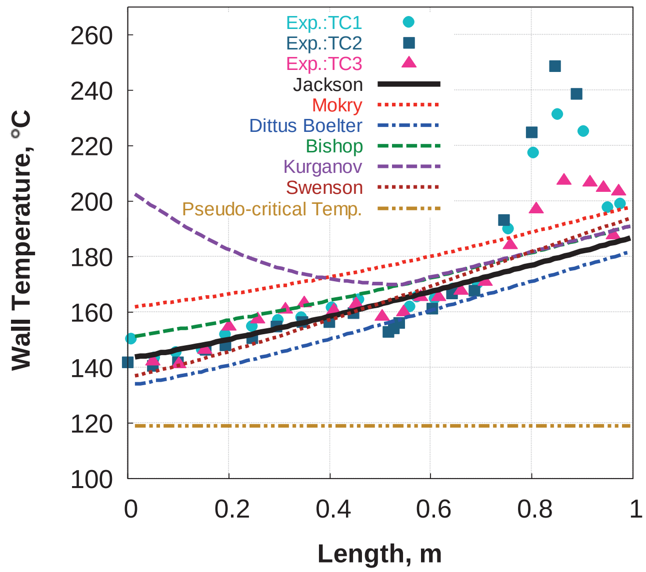


**Figure 9 Axial Wall Temperature Variation of 7-Rod bundle Freon Test: Above Pseudo-Critical Point (Test # 9).**

For the higher heat flux test, Test #9, where both the wall and bulk temperatures are well above the pseudo-critcal point, the results are shown in Figure 9. All correlations except Kurganov-Ankudinov appear to do a reasonable job in predicting wall temperatures from the inlet to about 0.8 m downstream. The Kurganov-Ankudinov correlation, on the other hand, over-predicted by a large margin the wall temperatures in the upstream half of the heated length. The likely cause is that these test conditions are significantly outside the range of those used to develop the Kurganov-Ankudinov correlation. Among the rest of the correlations, the Jackson and the Swenson et al. correlations appear to provide better predictions than the other three, with the Dittus-Boelter prediction being the lowest and the Mokry et al. the highest.

None of the correlations could reproduce the measured temperature peak at the last 20 cm of the heated section, which is not believed to be a result of the heat transfer deterioration phenomenon for two reasons. First, the scatter between the measured temperatures at the three "identical by symmetry" locations is too large (up to ~40 °C) for the data at the exit region to be judged reliable. Second, the flow condition near the downstream end was further away from the pseudo-critical point for a large heat transfer deterioration to occur. Kirillov and co-workers [11] stated that the temperature peak might be a result of material coming out of solution at the higher temperature and depositing on the wall, thereby increasing the wall-to-fluid heat transfer resistance.

## 5. SUMMARY AND CONCLUSIONS

The Canadian subchannel code ASSERT-PV was assessed against available experimental data of measured cladding temperatures in rod bundles cooled with water and Freon-12 at supercritical pressures under flow and heat transfer conditions relevant to the Canadian SCWR. Based on this assessment the following salient points can be inferred.

- Among the six heat transfer correlations evaluated, the Jackson correlation appears to be the best overall in predicting the measured cladding temperatures for the three experiments selected for the assessment.  Therefore, it is recommended that the Jackson correlation be continually used as the base model in further development of the SCWR fuel assembly.

- The partial-bundle experimental data used in the assessment are still limited, and the thermalhydraulic parameter ranges covered by the data are not sufficiently wide.  Therefore, it is recommended to supplement the Jackson prediction with a sensitivity study using other correlations such as the Mokry et al. and Swenson et al. until a better correlation for the Canadian SCWR fuel assembly becomes available.
- Reliable experimental data at a wider range of thermalhydraulic conditions, especially at the combination of higher heat flux and lower mass flux, or with longer heated section, are required for further assessment of the code in predicting cladding temperatures up to 800 °C.

## NOMENCLATURE

| | |
|---|---|
| AECL | Atomic Energy of Canada Limited |
| BWR | Boiling Water Reactor |
| C, m, n | Constant in Equation (1) |
| CANDU | CANada Deuterium Uranium |
| CNL | Canadian Nuclear Laboratories (formerly AECL) |
| d | Diameter |
| F | Parameter in Equation (1) |
| g | Gravitational Constant |
| G | Mass Flux |
| IPPE | Institute of Physics and Power Engineering |
| IAPWS | The International Association for the Properties of Water and Steam |
| JAEA | Japanese Atomic Energy Agency |
| L | Channel Heated Length |
| PWR | Pressurized Water Reactor |
| PHWR | Pressurized Heavy Water Reactor |
| Re | Reynolds Number |
| SC | Supercritical |
| SCWR | Canadian Supercritical Water-cooled Reactor |
| XJTU | Xi'an Jiaotong University |

*Greek Letters*

| | |
|---|---|
| $\beta$ | Thermal Expansion Coefficient |

*Subscripts*

| | |
|---|---|
| h | Hydraulic |
| n | Normal |
| x | Based on Bulk Fluid (b) or Wall Temperature (w) according to Table II |

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## APPENDIX A The Kurganov-Ankudinov Correlation [14]:

Based on a mechanistic approach, Kurganov and Ankudinov derived the following semi-empirical correlation for the heat transfer coefficient *h*:

$$\frac{h}{h_n} = \begin{bmatrix} 1, & for\ \widetilde{K} \le 1 \\ \widetilde{K}^{-m}, & for\ \widetilde{K} > 1 \end{bmatrix} \tag{A-1}$$

In the above, $h_n$ is the heat transfer coefficient for normal heat transfer without deterioration; calculated from:

$$h_n = \frac{d_h(f_{/8})Re_b\overline{Pr}_b}{k_b\left\{1 + \frac{900}{Re_b} + 12.7(f_{/8})^{0.5}\left(\overline{Pr}_b^{2/3} - 1\right)\right\}} \tag{A-2}$$

The parameter $\widetilde{K}$ accounts for the effects of buoyancy and acceleration resulting from fluid density variation near the wall:

$$\widetilde{K} = \left\{\frac{\varepsilon_u}{\gamma} + S\frac{\mathrm{Gr}}{\mathrm{Re}_b^2}\right\}\frac{1}{f\{1 - exp(-Re_b/30000)\}} \tag{A-3}$$

With:

$$\gamma = 1 - 0.8exp\left[-3\left\{\frac{\varepsilon_u}{8}\frac{4L/d_h}{ln\left(\rho_{in}/\rho_b\right)}\right\}^2\right] \tag{A-4}$$

The acceleration coefficient, $\varepsilon$, is given by:

$$\varepsilon = \frac{8\phi\beta_b}{GC_{p,b}} \tag{A-5}$$

And the Grashof number, Gr, is calculated from:

$$Gr = \frac{gd_h^3\rho_b^2}{\mu_b^2}\left(1 - \frac{\rho_w}{\rho_b}\right) \tag{A-6}$$

The friction factor correlation given by equation (10) is recommended by Cheng and Schulenberg [19] for the calculation of frictional pressure drop in supercritical conditions. The friction factor, *f* at supercritical conditions is calculated from:

$$f = \left\lfloor\frac{0.55}{log\ (Re_b/8)}\right\rfloor^2\left(\frac{\rho_w}{\rho_b}\right)^{0.4} \tag{A-7}$$

*S = +1* for upflow and *-1* for downflow, indicating that for downflow buoyancy forces, quantified by the ratio of Grashof number to the square of the Reynolds number, enhance, rather than degrade heat transfer. The exponent *m* depends on the heated length and is calculated from:

$$m = 0.55[1 - exp(0.02L/d_h)] \tag{A-8}$$

Values of $\widetilde{K}$ greater than 1 are indicative of a strong effect of buoyancy and acceleration, resulting in heat transfer deterioration. It is noted that since equation (A-2) is based on a minimum value of 1 for $\widetilde{K}$, the maximum value of the heat transfer coefficient is limited to its normal value with no net enhancement.