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AN INVESTIGATION INTO CRITICAL HEAT FLUX CORRELATIONS WITHIN CTF

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ABSTRACT

The safety margins of a nuclear reactor is something that is under constant scrutiny but is difficult to make progress towards due to the need for valid and accurate simulations since safety concerns severely limit the amount of actual tests that can be done. In order to work towards safety margin improvement, a validation of a model of a Westinghouse 17x17 Pressurized Water Reactor in CTF at a steady-state and 50% overpower condition and a comparison of CHF correlations including the W-3 correlation and the Groeneveld lookup tables was needed. The model was validated in CTF versus a provided model in COBRA-EN using factors such as temperature, density, void fraction, and pressure drop. The various CHF models were compared looking at mainly DNBR at various axial locations on the fuel rods. This model was determined to be valid and, in some areas, significantly more accurate than previous models. Both the CHF models proved accurate, and can be reasonably used to investigate alternative cladding materials in the future in order to potentially increase reactor safety margins.

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Chapter 1

Literature Review

One of the most important aspects of Nuclear Reactor Safety is Critical Heat Flux (CHF) and the Departure from Nucleate Boiling (DNB). Reactors draw their power from the heat produced by the power rods in the reactor's core, which heat water, which produces steam, which is used to generate energy. The structure of these rods is generally two part: a center portion that contains the nuclear material and a metallic coating called cladding that holds the material in place. This cladding has many functions, including moderating both the radioactive effects of the core and the heat transferred out of the core. These rods are arranged in various geometries and stored in water for a number of reasons. Water is one of the most favorable natural entities for the moderation of both temperature and a shield from radiation emission. The amount of heat transferred and change in temperature of the surface of the cladding is relatively linear until the cladding reaches the point of Critical Heat Flux [1]. This is the portion of the boiling curve I looked at in this study.

After reaching the point of CHF, further increases in heat transfer lead to very large and uncontrollable temperature changes. This is due to a vapor film of steam forming between the cladding and the water that is cooling it. This vapor film decreases the heat transfer efficiency and increases the temperature of the cladding and the rod as a whole [2]. This unstable boiling and temperature increase post CHF can in turn result in melting the cladding on the rod if it allowed to continue [3]. This boiling crisis can be classified as departure from nucleate boiling

(DNB) in a subcooled or low-quality region and dryout in the high quality region [3]. This research focused on the DNB case of post CHF reaction.

Due to the focus on the DNB case, the most common way to determine the safety margin for this type of accident is the Departure from Nucleate Boiling Ratio (DNBR). This is the ratio of the predicted correlation heat flux to the actual operating heat flux at any point on the power rod [4]. The standard safety margin for PWR reactors is that DNBR will be greater than or equal to 1.30 at maximum overpower conditions, and even at these conditions the number of rods reaching this DNBR is very small [3]. Overpower conditions for a PWR are generally considered to be at 112% of maximum design power [4]. If DNBR is less than one, there is a boiling crisis at that location on the rod. The simulation I ran had the reactor run at 150% maximum design power in order to induce CHF.

My research focuses on a simulation of a Pressurized Water Reactor (PWR), which is the most common form of a Light Water Reactor (LWR). LWR means that the form of water used to surround the power rods in normal water and PWR means that the water is pressurized in order to increase the boiling temperature of the water so that higher temperature can be used in the reactor [4]. This heated and pressurized water is used to heat unpressurized water and boil it to create steam which drives a turbine to create power. In this form of reactor, boiling is not desired and can be highly dangerous due to the boiling crisis described above. A boiling crisis in a PWR can lead to extreme temperatures that can spiral upwards uncontrollably and end up melting the cladding along with other disastrous consequences.

For the purposes of comparisons in this thesis, two different CHF correlations are considered: the W-3 correlation, and the Groeneveld Lookup Tables. It is important to look at the conditions at which these correlations are valid and legitimate in order to determine potential error and differences there will be for these correlations. For both correlations, there are unit differences in the general form of the correlations and the form used in the theory manual for the simulation used for this paper. The general correlations are in SI units whereas the correlation that is explicitly used in the CTF code is in Imperial units.

Table 1: Comparison of valid parameters for CHF correlations [4]

	W-3	Groeneveld Lookup Tables
Diameter (D)	0.015-0.018 m	3-25 mm
Length (L)	0.254-3.70 m	Length factor is included
Pressure (p)	5.5-16 MPa	0.1-21 MPa
Mass Flux (G)	1356-6800 kg/m^{2}s	0-8000 kg/m^{2}s
Quality (x)	-0.15-0.15	-0.50-0.90

The W-3 correlation is the most widely used correlation for evaluating DNB of a PWR and was developed by Tong [4]. It is Westinghouse’s standard CHF correlation. This correlation was originally developed for axially uniform heat flux, but has a correction factor for non-uniform heat flux that is built into the code used for this paper. This non-uniform heat flux correction factor was developed both by Tong and independently by Silvestri [4]. The general equation for uniform axial heat flux is as follows:

$$q''_{cr,u} = \{(2.022 - 0.06238p) + (0.1722 - 0.01427p)\exp[(18.177 - 0.5987p)x_e]\}[(0.1484 - 1.596x_e + 0.1729x_e|x_e|)2.326G + 3271][1.157 - 0.869x_e][0.2664 + 0.8357 * \exp(-124.1D_e)]\left[0.8258 + 0.0003413\left(h_f - h_{in}\right)\right]\ kW/m^2$$

In order to correct for non-uniform axial heat flux, a factor of F must be applied:

$$q''_{cr,n} = q''_{cr,u}/F$$

Where F is:

$$F = \frac{C\int_0^Z q''(Z')\exp[-C*(Z-Z')]\,dZ'}{q''(Z)[1-\exp(-C*Z)]}$$

Where Z is the axial location on the rod measure from the core inlet and C is:

$$C = 185.6\frac{[1-x_{eZ}]^{4.31}}{G^{0.478}}$$

The factor F in this form was proposed by Lin et al [4] and C is an experimental coefficient that describes the heat and mass transfer effectiveness at the bubble-layer/subcooled-liquid-core interface [4]. There are also corrections for cold wall effects and local spacer effects but those were not relevant for this study.

The specifically used formula is [5]:

$$\frac{q''_{w3}}{10^6} = \{(2.022 - 0.0004302p) + (0.1722 - 0.0000984p)\exp[(18.177 - 0.004129p)x_e]\}\Big[(0.1484 - 1.596x_e + 0.1729x_e|x_e|)\frac{G}{10^6} + 1.037\Big](1.157 - 0.869x_e)[0.2664 + 0.8357\exp(-3.151D_h)][0.8258 + 0.000794(h_f - h_{in})]$$

Where:

q''_{w3} = Critical Heat Flux (BTU/hr-ft^2)

p = Pressure (psia)

x_e = local quality

D_h = equivalent hydraulic diameter (in)

h_{in} = inlet enthalpy (BTU/lbm)

G = mass flux (lbm/hr-ft^2)

Using Imperial units in the equation means that we need the range of operating condition should be in Imperial units as well:

p = 800 to 2300 psia

$G/10^6$ = 1.0 to 5.0 lbm/hr-ft^2

D_h = 0.2 to 0.7 in

L = 10 to 144 in

The non-uniform correction factor remains the same except for the equation for C, which is now:

$$C = 0.15\frac{(1.0 - x_{DNB})^{4.31}}{\frac{G}{10^6}^{0.478}}$$

Where G is the local mass flux neat the rod surface in lbm/hr-ft^2. The units for C are in^{-1} which will be converted to ft^{-1} before being used in the equation for F.

The Groeneveld Lookup Tables are widely considered to be the "most useful and accurate current correlation" for CHF [4]. The look up tables are based on a normalized data bank for a vertical 8 mm in diameter water-cooled tube with uniform axial heat flux distribution. CHF (q''_{cr}) is based upon pressure, mass flux, and quality [4]. For a sub-channel or tube, the values are adjusted in this specific simulation to various diameters and flux distributions through correction facts using the following equation:

$$q''_{cr} = (q''_{cr})_{LUT}K_1K_4K_5$$

Where K_1 is the sub-channel or tube-diameter cross-section geometry factor:

For $3 < D_e < 25$ mm:

$$K_1 = (\frac{8}{D_e})^{1/2}$$

And for D_e > 25 mm:

$$K_1 = 0.57$$

K_4 is the heated length factor:

For L/D_e >5:

$$K_4 = \exp[\left(\frac{D_h}{L}\right)\exp(2\alpha_{HEM})]$$

Where

$$\alpha_{HEM} = \frac{x_e\rho_f}{x_e\rho_f + (1 - x_e)\rho_g}$$

Where α_{HEM} is the void fraction.

K_5 is the axial flux distribution:

For $x_e \leq 0$:

$$K_5 = 1.0$$

And for $x_e > 0$:

$$K_5 = \frac{q''}{q''_{BLA}}$$

Where q''_{BLA} is the boiling length averaged heat flux.

There are other factors that can be applied such as a bundle-geometry factor, a mid-plane spacer factor for a 37-element bundle (CANDU), a radial or circumferential flux distribution factor, a horizontal flow-orientation factor, and a vertical low-flow factor. All these factors are applied similarly to K_1, K_4, and K_5 but the program utilized in this study does not utilize these factors, so the specific equations are not relevant.

CTF includes two other CHF correlations but they were not relevant to my study since the Bowring correlation encountered multiple errors while running, and the Biasi correlation yielded results that were so far away from the W-3 correlation and the Groeneveld lookup tables that I deemed it not relevant to this study.

Chapter 2

Validation

The simulation code used is from Oak Ridge National Labs and North Carolina State University and is called CTF. CTF is an improvement upon earlier forms of COBRA-TF which is “a computational tool for assessing nuclear power plant behavior” [5]. COBRA-TF was originally developed in the 80’s by Pacific Northwest Laboratory and was sponsored by the Nuclear Regulatory Commission. CTF improves upon older forms of COBRA-TF by enhancing user-friendliness, improving quality assurances and modeling capabilities, as well as enhancing computational efficiency and user modeling documentation [5]. Initial validation took place by ensuring that the input linear heat rate equaled the output linear heat rate. Further validation took place comparing the model made with an existing model.

The simulation run is a reactor model of an eighth of a Westinghouse 17x17 pressurized water reactor (PWR). This model was run at a 50% overpower condition to induce CHF, and was run at steady-state. This was previously modeled in COBRA-EN which is a thermal-hydraulics code which was designed to be a section in other programs used to simulate light water power reactors and the analyze core dynamics [6]. COBRA-EN used the EPRI CHF correlation model, which is an older correlation model and is not offered in CTF. COBRA-EN was used as a comparison due to offering a model of the desired reactor as a sample case.

Geometrical data and information was available from diagrams and was used to build the model. This was especially helpful and necessary in the determination of the location of water rods as well as rod spacing and gap width, which are necessary parameters when simulating a reactor core. A diagram of the reactor core is shown in figure 2.

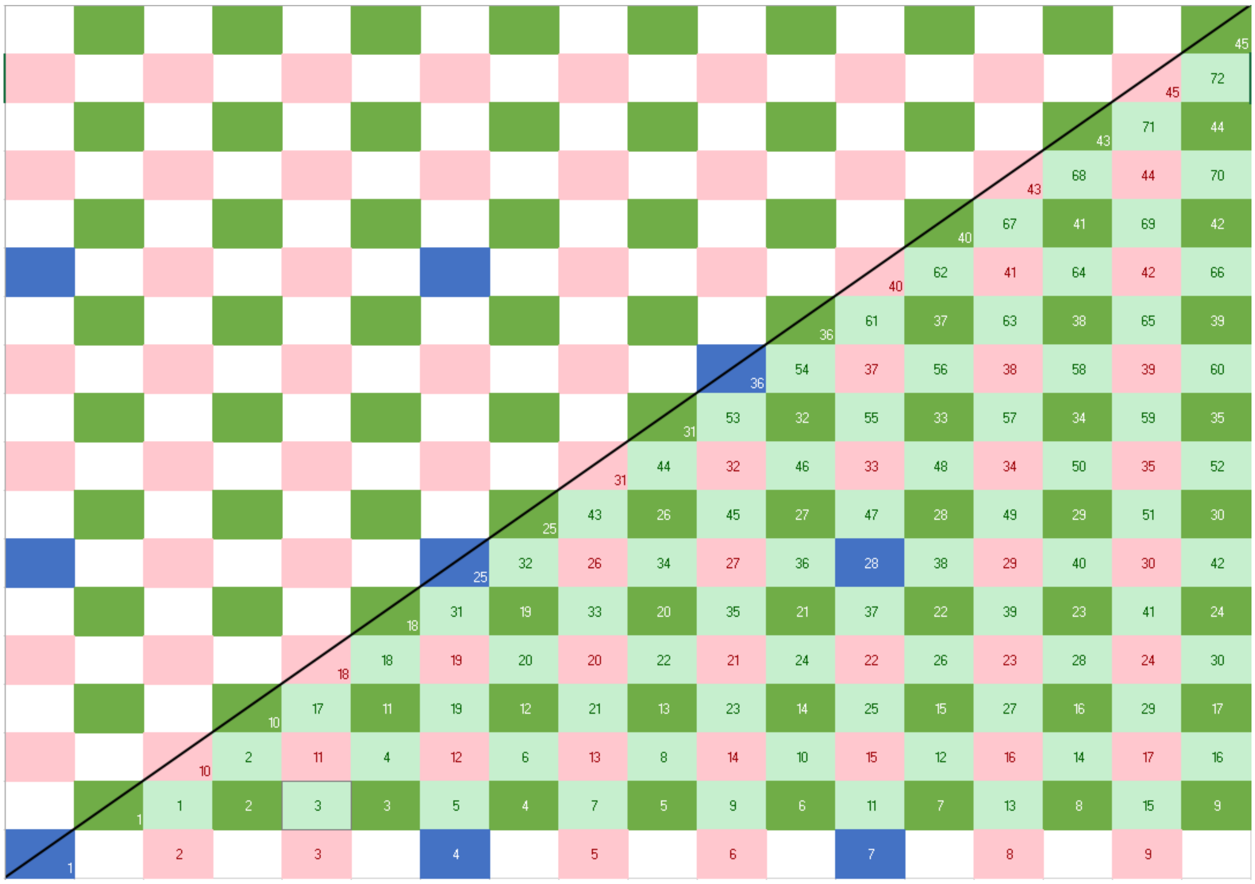

Figure 1: Numbered Excel diagram of a quarter of a Westinghouse 17x17 PWR

In Figure 2, the blue squares are water rods, the red squares are heated rods, the dark green squares are channels and the light green squares are gaps. Each of these different colored squares are numbered in a different sequence, with the exception of the two types of rods, which are in the same numerical sequence. This was done to assist in visualizing the numbering of everything since each rod, channel, and gap needs to be assigned a number in order to be accurately portrayed in CTF.

The output comparisons of various key values show that for the most part there is good agreement between the two models.

Table 2: Validation comparison between CTF and COBRA-EN

	Homogeneous Void Model	EPRI Model	CTF (W-3)	Percent Difference
Mean Coolant Void Fraction	0.12	0.12	0.15	25.58
Maximum Void Fraction	0.67	0.62	0.68	0.58
Mean Coolant Temperature [K]	597.8	597.8	598.38	0.097
Mean Coolant Temperature at Core Exit [K]	614.08	614.03	613.38	0.11
Mean Coolant Density [kg/m^3]	593.6	593.55	572.87	3.49
Mean Coolant Density at Core Exit [kg/m^3]	421.92	435.73	391.54	7.20
Mean Pressure Drop from Core Inlet to Outlet [kPa]	90.41	90.56	98.56	9.01
Mean Hydrostatic Pressure Head [kPa]	24.84	24.84	24.24	2.42

Looking at Table 2, there is very good agreement in the maximum void fraction, and both temperature values. There is some agreement in the mean coolant density and the mean hydrostatic pressure head. There is a little agreement for the mean coolant density at the core exit and the mean pressure drop. The by far largest percent difference comes with the mean coolant void fraction.

The model in COBRA-EN does not have mixing or cross-flow turned on, while CTF does. Essentially COBRA-EN treats each channel as an independent entity, instead of the entire thing as a system. This is what likely caused the edge channels to display zero for the void fraction and bring down the void fraction average enough to give a 25% difference. COBRA-EN also uses a different slip ratio than CTF. This means that the two models assume different ratios of the velocities of the liquid and gaseous portions of the coolant [4]. COBRA-EN assumes a slip ratio of 1 whereas the CTF outputs yields slip ratio values between 1.0 and 1.2 [5]. CTF also has a special focus on improving simulations regarding heat equations and void fraction. This leads me to believe that the errors and issues lie within the COBRA-EN code and model. A more detailed and specific explanation and comparison is unavailable without having access to the full COBRA-EN code, which was not available for this study.

Chapter 3

CHF Comparisons and Analysis

Small differences exist in the values used for comparisons with COBRA-EN for different CHF correlations, most likely due to slightly different predictions of the departure from nucleate boiling. For actual CHF comparison purposes, the W-3 correlation are compared to the Groeneveld lookup tables since the tables are widely deemed the most accurate source of CHF and DNBR information.

Table 3: CTF W-3 compared to COBRA-EN

	COBRA-EN Homogeneous Model	CTF W-3 Model	Percent Error
Mean Coolant Void Fraction	0.12	0.15	25.58
Maximum Void Fraction	0.67	0.68	0.58
Mean Coolant Temperature [K]	597.8	598.38	0.097
Mean Coolant Temperature at Core Exit [K]	614.08	613.38	0.11
Mean Coolant Density [kg/m^3]	593.6	572.87	3.49
Mean Coolant Density at Core Exit [kg/m^3]	421.92	391.54	7.20
Mean Pressure Drop from Core Inlet to Outlet [kPa]	90.41	98.56	9.01
Mean Hydrostatic Pressure Head [kPa]	24.84	24.24	2.42

According to the comparison in Table 3, the average percent error for this comparison with the W-3 correlation is 6.06% and if the apparent outlier of the Mean Coolant Void Fraction is excluded, it is 3.27%.

Table 4: CTF with Groeneveld Lookup Tables compared to COBRA-EN

	COBRA-EN Homogeneous Model	CTF Groeneveld Lookup Tables	Percent Error
Mean Coolant Void Fraction	0.12	0.15	26.01
Maximum Void Fraction	0.67	0.68	1.21
Mean Coolant Temperature [K]	597.8	598.38	0.097
Mean Coolant Temperature at Core Exit [K]	614.08	613.35	0.12
Mean Coolant Density [kg/m^3]	593.6	572.67	3.53
Mean Coolant Density at Core Exit [kg/m^3]	421.92	391.43	7.23
Mean Pressure Drop from Core Inlet to Outlet [kPa]	90.41	98.64	9.10

The average percent error for this comparison in Table 4 with the Groeneveld lookup tables is 6.22%. If the outlier of the Mean Coolant Void Fraction difference is excluded, the average percent error is 3.39%.

In terms of actual CHF correlations, it is best to compare the Departure from Nucleate Boiling Ratio at various vertical locations along one of the power rods. For the comparison I chose rod number two (see Figure 2) since it is closest to the center of the core of the simulated reactor, and thus theoretically going to experience the most heat flux and therefore the most CHF and is most likely to achieve departure from nucleate boiling (DNB). Rod number 2 proved to experience the highest average CHF values and lowest DNBR of all the rods in the simulation

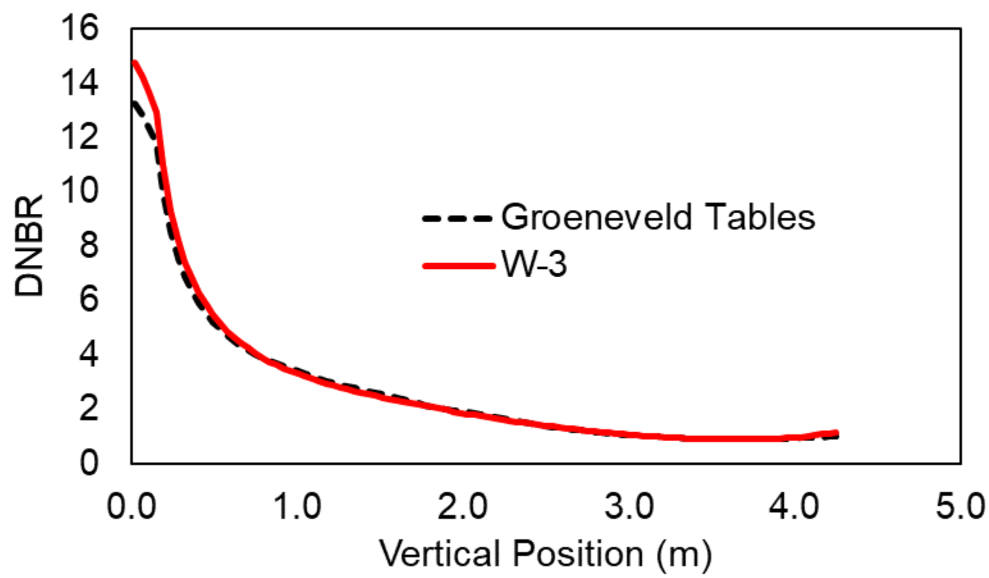

Figure 2: Graph of Vertical Position vs DNBR for both CHF correlations

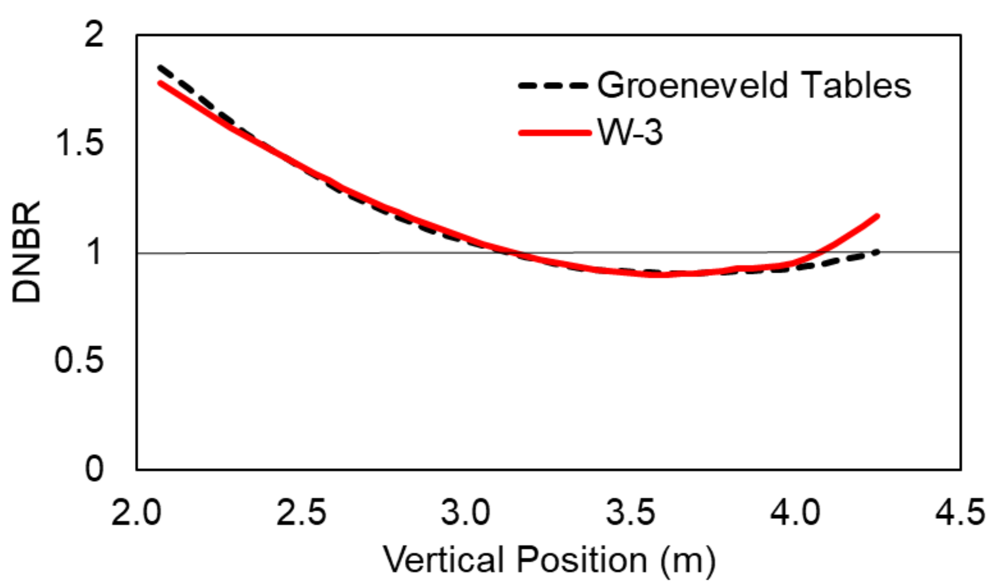

Figure 3: Graph of Position versus DNBR for the upper half of the rod

Figures 4 and 5 show position versus DNBR and it is readily apparent that the Groeneveld tables and W-3 correlation are fairly similar. DNBR less than one means that a boiling crisis is likely to occur, and is predicted to occur at a similar position of between 3.2 to 4.0 m up the rod by both the Groeneveld tables and the W-3 correlation.

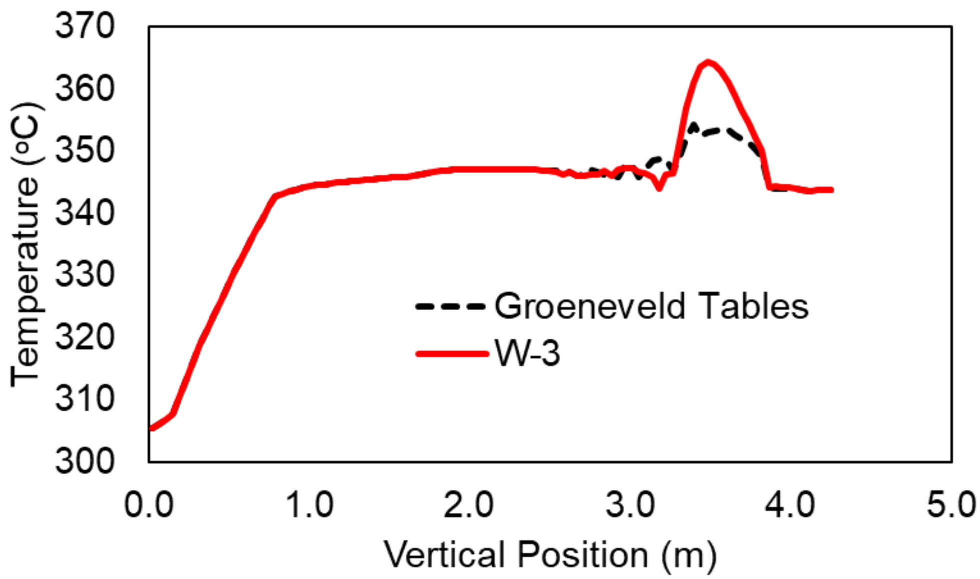

Figure 4: Graph of Vertical Position vs Rod Surface Temperature for both CHF correlations

The temperature comparison also shows the boiling crisis predicted by the Groeneveld tables and W-3 with the temperature spikes at the same vertical positions that DNBR is below one. W-3 predicts a larger peak temperature than Groeneveld by about 10 degrees Celsius. This temperature difference is despite having nearly identical DNBR values in this region. The temperature does trend back down as the DNBR gets back closer to one. Both of these temperature spikes are potentially dangerous to the cladding material.

Based Figures 4-6, the W-3 correlation and the Groeneveld tables agreed quite well. The overall percent error for DNBR between W-3 and Groeneveld is 2.18%, and when both correlations are predicting a boiling crisis (DNBR <1) the percent error is just 0.86%. The only disagreement between the two models occurs at the tail end of the graphs, which is the top and bottom of the rod in question. Since the W-3 correlation is only considered accurate up to 3.7 m on length of the rod, that is one potential source for this error. Another is the quality at these points is outside of the listed accurate range of W-3 of -0.15 to 0.15. The area outside of the quality range is above approximately 2.9 m on the rod and below 1.2 m on the rod. The percent error when W-3 is outside of the quality limit is 3.79% and drops to 2.25% while inside the acceptable quality range. The quality difference might also help to explain the temperature difference in Figure 6, since the temperature spike occurs at a vertical position above 2.944, which is when the quality of the channel and rod in question go outside the parameters given for W-3. However, W-3 is still a very accurate and viable CHF correlation for this model. The two other CHF correlations in CTF were not viable options since the Bowring correlation would not run with this model and the Biasi correlation gave results that had a percent error over 200% when compared to the Greoneveld lookup tables.

Chapter 4

Conclusion

This model of a Westinghouse 17x17 PWR in CTF is a viable and useful simulation for looking at this reactor core bundle, and is a drastic improvement upon what COBRA-EN can do, especially in terms of heat equations and void fraction. For the CHF correlations, the Groeneveld look up tables are consistently stated in literature as the most accurate model and are taken as such here. The W-3 correlation is a good approximation and compares favorably to the Groeneveld tables as long as the attributes of the rod and channel are within the given accuracy range of the W-3 correlation. Therefore, both can be used and trusted to be accurate for this model. The other CHF options in CTF, the Bowring correlation and the Biasi correlation, encountered enough problems as to be unsuitable for this model.

The validity of this model in CTF allows future work to be done on a number of topics. Primarily, this model can now be run in a transient state, as opposed to the steady-state it was run for this simulation. This will allow for studies on more advanced topics such as accident tolerant fuels (ATF). A promising form of ATF is FeCrAl. FeCrAl was identified as having characteristics that positively affect its CHF compared to the current cladding material, Zircaloy-4. In previous studies, FeCrAl has been shown to have more average surface roughness and a lower contact angle, both important factors in increasing CHF, than typical Zircaloy-4 [7]. There are even indications that proper FeCrAl alloys (specifically $Fe_{13}Cr_4Al$) can improve CHF by up to 60% [7]. Initial studies such as this paired with a viable model to run core bundle simulations are both foundations that can be built upon to properly investigate the viability of FeCrAl along with other accident tolerant fuels.

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