TOPICAL REPORT EVALUATION

Report Title: Virginia Electric & Power Company Reactor Core Thermal-Hydraulic Analysis Using the COBRA IIIC/MIT Computer Code

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DSI Branch Involved: Core Performance Branch

Introduction

This report describes the Virginia Electric & Power Company (VEPCO) thermal-hydraulic model and its application to VEPCO's pressurized water reactor cores (i.e., North Anna, Surry). The accuracy of the VEPCO thermal-hydraulic model is demonstrated through comparisons with analyses which were used in the design and licensing of the Surry Nuclear Power Station. VEPCO has also submitted a number of check cases (Ref. 2), some of which have been compared to staff audit calculations, to further demonstrate the accuracy of their model.

VEPCO Thermal-Hydraulic Model

VEPCO's thermal-hydraulic model is an adaptation of the COBRA IIIC/MIT (Ref. 1). The major modifications to COBRA IIIC/MIT include:

1. The capability to do the thermal hydraulic analysis using single stage method which incorporates the geometries and methodologies used in traditional multistage analyses.

2. The two phase density in the subcooled void region is based upon the saturated vapor and subcooled liquid densities.

3. The DNBRs are printed out by channel numbers instead of rod numbers.

4. The W-3 L-grid and R-grid spacer factor correlations are options available in the code.

5. The fraction of the heat generated in the fuel and cladding are accounted for in the CHF calculations.

6. The calculation iterates at least twice before convergence but includes a variable damping factor input for more rapid convergence.

7. An option is added so that different crossflow resistance and mixing coefficients could be input and applied to the rod gaps.

8. Up to six different axial heat flux shapes could be input and applied to different fuel rods.

9. All water properties (enthalpy, specific volume, viscosity, conductivity and specific heat) are calculated using the HOH routines which are obtained from the PDQ7V2 computer code.

10. Saturated liquid properties are used to correct the calculations of the true (non-equilibrium) quality within the Levy subcooled model.

Description

The COBRA-IIIC/MIT computer code calculates the flow and enthalpy within interconnected flow channels by solving finite difference equations of continuity, energy, and momentum. The mathematical model is applicable to both steady state and transient conditions and the model considers both turbulent mixing and diversion crossflow. In formulating the model one-dimensional, two-phase, separated, slip-flow is assumed to exist during boiling. The two-phase flow structure is assumed to be fine enough to specify the void fraction as a function of enthalpy, flow rate, heat flux, pressure, position, and time. Sonic velocity propagation effects are not included. Within a channel, the diversion crossflow velocity is assumed to be small compared to the axial velocity to allow the use of a simplified equation for the conservation of transverse momentum.

The same finite difference equations are used for both steady state and transient computations. Initial conditions are obtained by performing a steady state calculation and then transient calculation is performed. Time dependent forcing functions consisting of inlet temperature, inlet flow, system pressure, and core average heat flux are used to establish boundary conditions at succeeding times. The calculation iterates over the first time step until the flow solution converges. The converged solution is then used as the initial condition for the new time, and the procedure continues for all the subsequent time steps. The correlations used in calculating turbulent mixing are of major importance. Once the flow solution is obtained, additional correlations are used in calculating the DNBR distribution. The COBRA-IIIC/MIT computer code allows user specification of the appropriate correlations.

Models and Correlations

The void fractions are predicted using the Smith correlation (Ref. 3) in conjunction with the Levy subcooled model (Ref. 4). In computing single and two-phase pressure drops, an isothermal friction factor correlation is used in conjunction with a wall viscosity correlation and a correlation for predicting two-phase friction multipliers.

In predicting the non-uniform critical heat flux, the W-3 correlation is used in conjunction with the F-factor correlation (Ref. 5). In determining the lower bound of the F-factor integral, the Jens and Lottes correlation (Ref. 6) is used to predict the axial position where nucleate boiling begins. When appropriate, the coldwall factor and the L-grid or R-grid spacer factor (Ref. 7) are used with the W-3 correlation to predict the critical heat flux.

Turbulent Mixing

The degree of turbulent mixing between adjacent channels is calculated using the following relationship,

$$w' = \beta sG$$

where w' is the turbulent transverse fluctuating flow rate per axial length, G is the average mass velocity of the adjacent channels, s is the common gap, and β is the mixing coefficient. The above relationship is used to predict both single and two-phase mixing.

Hydraulic Model Description

Eighth core symmetry is assumed, and thus 1/8 core segment is modeled. The location of hot assembly is assumed at the center of the core. The hot assembly is modeled as an array of subchannels, while the remaining assemblies are modeled as an array of lumped channels. For steady state analysis, a fine mesh geometry is used in which each lumped assembly and each hot assembly subchannel is modeled as an individual flow channel. For transient analysis, a coarse mesh geometry is used in which assemblies and subchannels are combined to form large channels. Because the coarser geometry contains fewer channels, less computational time per iteration will be required for the transient analysis.

A 53 channel model has been developed for steady state analysis and a 19 channel model has been developed for transient analysis of the Surry units. This 53 channel model consisted of 25 lumped assembly channels and 28 subchannels. The 19 channel model consisted of 4 lumped channels and 15 subchannels.

Thermal Model Description

The thermal model consists of an inlet flow distribution, radial and axial power distributions, and appropriate reactor operating conditions. For transient analysis, time dependent forcing functions of system pressure, inlet flow, inlet temperature, and core average heat flux are also specified.

Thermal hydraulic design parameters form the basis of the model. The radial power distribution is based upon the design value of $F^N_{\Delta H}$, and the axial power distribution is based on the reference axial shape. The thermal design flow rate is used in determining the core average mass velocity, and the thermal hydraulic design values for inlet temperature, system pressure, and power level are used as operating conditions.

The thermal model is then imposed upon the hydraulic model in order to obtain the complete thermal-hydraulic representation of the core. Since this representation is dependent upon thermal-hydraulic design parameters, revised representations must be considered in the event of any subsequent design changes. In general, the hydraulic model remains relatively fixed since it is affected only by changes in the mechanical design of the fuel. However, the thermal model can be significantly affected by changing any one of the design parameters.

Inlet Flow Distribution

The inlet flow distribution used in the 53 channel model for steady state analyses and 19 channel model for transient analyses, assumes a 5 percent flow reduction to the hot assembly while the peripheral assemblies have a flow fraction slightly greater than 1.0. Therefore, the average of all the fractions is approximately 1.0.

Power Distribution

The hot assembly as well as the adjacent assemblies are given relative powers of 1.475, while lower relative powers are assigned to the remaining assemblies. The average of all the assembly relative powers is 1.0. The assembly power distribution for the 19 channel model is derived from the 53 channel model by averaging all but the three central assembly powers. Within the subchannel array, a second power gradient exists, having a peak around the hot channel which is a thimble cell. The three fuel rods surrounding the hot thimble cell have relative powers of 1.55, the remaining fuel rods are at lower relative powers. The average of all the fuel rod relative powers is equal to the hot assembly relative power, 1.475.

The subchannel power distribution for the 19 channel model is also derived from the 53 channel model by averaging the relative power of the fuel rods located within the lumped subchannel. The same axial power distribution is used in both the 19 and 53 channel models.

Forcing Functions for Transient Analysis

For each reactor parameter that is changing with time, a forcing function is input as a table set with each entry consisting of the ratio of the transient condition to the initial condition and a corresponding time. The COBRA-IIIC/MIT computer code has the capability of handling four different forcing functions, e.g., core average heat flux vs time, inlet flow vs time, inlet temperature vs time, and system pressure vs time.

Engineering Uncertainties

After formulating the overall thermal-hydraulic representation of the core, engineering uncertainties are then applied to account for manufacturing tolerances used in the fabrication of the fuel. These fabrication tolerances are assumed to occur in the hot channel, and are called hot channel factors.

These factors consist of a pitch reduction, an engineering factor on the enthalpy rise ($F^E_{\Delta H}$), and an engineering factor on the heat flux (F^E_Q). The pitch reduction takes into account fuel rod spacing variations which may occur within the as-built fuel assembly. This reduced pitch is accounted for by modeling the hot channel with reduced gap spacing and a reduced flow area. Since a reduced flow area causes a greater pressure loss across the spacer grids, this effect is taken into account by using increased grid loss coefficients. Table 1 lists the hydraulic data which was used in modeling the hot channel.

The engineering factor of the enthalpy rise ($F^E_{\Delta H}$) takes into account the effect of enrichment and density variations which may occur in as-built fuel rods. This factor is accounted for by increasing the relative power of the hot fuel rod. For all the DNB analyses described within this report, the relative power of the hot fuel rod was multiplied by a factor of 1.02.

TABLE 1

HOT CHANNEL HYDRAULIC DATA

<u>Hot Thimble Cell</u>

Pitch Reduction (inches)	0.0065
Reduced Flow Area (square inches)	0.1463
Reduced Fuel Rod to Fuel Rod Gap (inches)	0.1345
Reduced Fuel Rod to Thimble Tube Gap (inches)	0.0725

<u>Hot Unit Cell</u>

Pitch Reduction (inches)	0.0065
Reduced Flow Area (square inches)	0.1698
Reduced Fuel Rod to Fuel Rod Gap (inches)	0.1345

The engineering factor on the heat flux (F_Q^E) takes into account the effect of enrichment, density, diameter, and eccentricity variations which may occur in as-built fuel pellets. This factor is accounted for by applying a heat flux spike on the hot fuel rod at the position of MDNBR. Before the heat flux spike can be applied, however, a thermal analysis must first be performed in order to determine the axial position of MDNBR. Based upon these results, the axial heat flux shape for the hot fuel rod is then adjusted to include a heat flux spike at the determined position. The spike flux shape is included in a second thermal analysis from which the final results are obtained.

An engineering factor on the heat flux was applied to all the DNB analyses described in reference 1.

Thermal-Hydraulic Model Verification

VEPCO performed three steady state and six transient DNB analyses to verify the calculational accuracy of their thermal-hydraulic model. These analyses are listed in Table 2, and are representative of those contained in the original Surry FSAR and in subsequent licensing documents that update the FSAR. The minimum DNBRs obtained from the VEPCO model were compared to the values in the licensing documents. These values are provided in Table 3. This comparison results show that the VEPCO model to be consistant with previously submitted and approved licensing documents.

Staff Evaluation

Although VEPCO's original analyses showed fairly good agreement with the Surry FSAR calculations, the staff was still concerned that since the VEPCO code uses a single-pass method it should be benchmarked directly against a multi-pass code for a range of operating conditions. In response to concerns of the staff, VEPCO submitted (Ref. 2) the 7 check cases listed in Table 6. These cases are one for one comparisons

TABLE 2

LISTING OF VEPCO VERIFICATION ANALYSES

FSAR[8] Analyses

Steady State at 100% Power

Excessive Load Increase Transient

Uncontrolled Control Rod Assembly Withdrawal at Power Transient

Complete Loss of Reactor Coolant Flow Transient

Densification[9]/Positive Moderator Temperature Coefficient[10,11] Reanalyses

Steady State at 112% Power

Uncontrolled Control Rod Assembly Withdrawal At Power Transient

Complete Loss of Reactor Coolant Flow Transient

Low Flow Assumption[12] Reanalyses

Steady State at 102% Power

Complete Loss of Reactor Coolant Flow Transient

TABLE 3

	MDNBR		
	Surry FSAR	VEPCO	Channel Model
FSAR			
SS @ 100% Power	1.97	1.94 1.94	53 19
Excessive Load Increase Transient	1.55	1.53	19
Uncontrolled Control Rod Assembly Withdrawal	1.36	1.24	19
Complete Loss of RC Flow	1.46	1.48	-
Ref. 10	Ref. 10	VEPCo	
SS @ 112% Power	1.30	1.30 1.27	53 19
Uncontrolled	1.32	1.36	19
Complete loss of RC Flow	1.54	1.54	-
Low Flow Assumption Reanalysis Ref. 12			
SS @ 102% Power	1.50	1.49 1.49	53 19
Complete Loss of RC Flow Transient	1.33	1.35	-

TABLE 4

Range of Key Test Parameters

	Ranges
Pressure (psia)	1491-2433
Inlet Average Mass Velocity (Mlbm/hr-ft^2)	1.05-3.66
Inlet Temperature (°F)	433.0-617.0
Local Heat Flux (MBTU/hr-ft^2)	0.563-1.063

TABLE 5

W-3 Correlation Limits

Correlation	Ref. No.	Pressure Range (psia)	Mass Velocity (Mlb/h-f²)	Equiv. Diameter (in)	Local Quality	Axial Height (in)	Inlet Temp (°F)
W-3	1,2	1000-2400	1.0-5.0	0.2-0.7	≤0.15	10-144	>400
F-factor	1,2	1000-2400	1.0-3.0	0.2-0.7	≤0.15	10-144	
Coldwall Factor	1,2 3,4	1000-2400	1.0-5.0		≤0.15	>10	
Spacer Factor	3,4	1490-2440	1.5-3.7		≤0.15	96-168	404-624

TABLE 6

SUMMARY OF COMPARISONS WITH THINC-I

Case No.	Pressure (psia)	Power (%)	Flow (%)	Tin (°F)	FΩE	Axial Offset	Minimum DNBR COBRA	THINC- I
1	2200.	112.	100.	554.	1.24	zero	1.27	1.30
2	2400.	118.	100.	563.	1.03	zero	1.30	1.33
3	2400.	101.	100.	563.	1.03	large positive	1.32	1.39
4	2400.	81.7	100.	618.4	1.03	large negative	1.30	1.33
5	1855.	112.	90.	515.	1.03	zero	1.47	1.47
6	2220.	100.5	76.5	547.	1.03	zero	1.32	1.32
7	2400.	101.	100.	563.	1.03	large positive	1.42	1.50

between the COBRA 19 channel single-pass and the THINC-I multi-pass 1/8 core models. These cases were chosen to span the general licensed operating range of the VEPCO plants. A consistent set of input was used for both computer codes.

Case 1 is a recalculation of the VEP-FRD-33 Section 6.3.2 statepoint. It represents a point on the 2200 psi core thermal limit line at 112 percent power and conditions based on the densification/positive moderator temperature coefficient reanalysis.

Case 2 computes a point which would exist on a COBRA generated 2400 psia core thermal limit line at 118 percent power. Thermal design flow and the currently applicable heat flux spike were used.

Case 3 is a recomputation of Case 2, using a large positive axial offset power distribution instead of the normal cosine shaped axial power distribution. The power was adjusted to yield a COBRA minimum DNBR approximately equal to 1.30 (the actual value is 1.32). This set of operating conditions was then applied to the THINC-I calculation.

Case 4 shows the effect of reduced power and a large negative axial offset power distribution. As in Case 4, the power was adjusted to yield a COBRA minimum DNBR approximately equal to 1.30.

Case 5 shows the effect of reduced pressure and flow. This statepoint corresponds to conditions on the existing Surry 1855 psia core thermal limit line at 112 percent power. Current fuel stack height reduction and heat flux spike factors were used in this calculation, as opposed to the densification reanalysis values used in the limit line generation.

Case 6 shows the effect of reduced flow. The loss-of-flow transient reported in VEP-FRD-33 Section 6.4.3 was rerun with COBRA using corrected values for reduced fuel height and heat flux spike to yield minimum DNBR results identical to those from THINC-III.

Case 7 is the same as Case 3 except a different axial power distribution was used.

In all these cases the COBRA 19 channel single-pass model yielded either identical or conservative minimum DNBR's when compared with the THINC-I multi-pass model results.

As part of our review, the staff chose Cases 1 and 7 to perform audit calculations of VEPCO methodology using COBRA-IV. Our analyses were performed in two steps. First, an eighth core symmetric core-wide calculation was performed to determine the flow to the hot assembly. This crossflow was stored on tape and used as a boundary condition in the subchannel calculations. For the subchannel case, an octant of the limiting assembly was modeled on a rod-by-rod basis. The cases selected were those that had the greatest deviation between COBRA-IIIC/MIT and THINC-I. Table 7 contains the operating conditions for the cases analyzed while Table 8 presents a comparison of the results.

From Table 8, it can be seen that there is good agreement between the staff's detailed audit calculations and VEPCO's COBRA-IIIC/MIT results. In both cases, VEPCO is conservative when compared to our results.

TABLE 7

OPERATING CONDITIONS

Power Shape	Peak Value	Pressure (Psia)	T_{IN} (^oF)	Average 9" ($MBTU/hr-ft^2$)	Mass Velocity ($MLBM/hr-ft^2$)
Cosine	1.550	2200.0	554.0	0.22231	2.273
Upskew	1.748	2400.0	563.0	0.20000	2.251

TABLE 8

CALCULATIONAL RESULTS

Power Shape	VEPCo COBRA-IIIC/MIT	THINC-I	NRC COBRA-IV
Cosine	1.27	1.30	1.33
Upskew	1.42	1.50	1.53

Summary

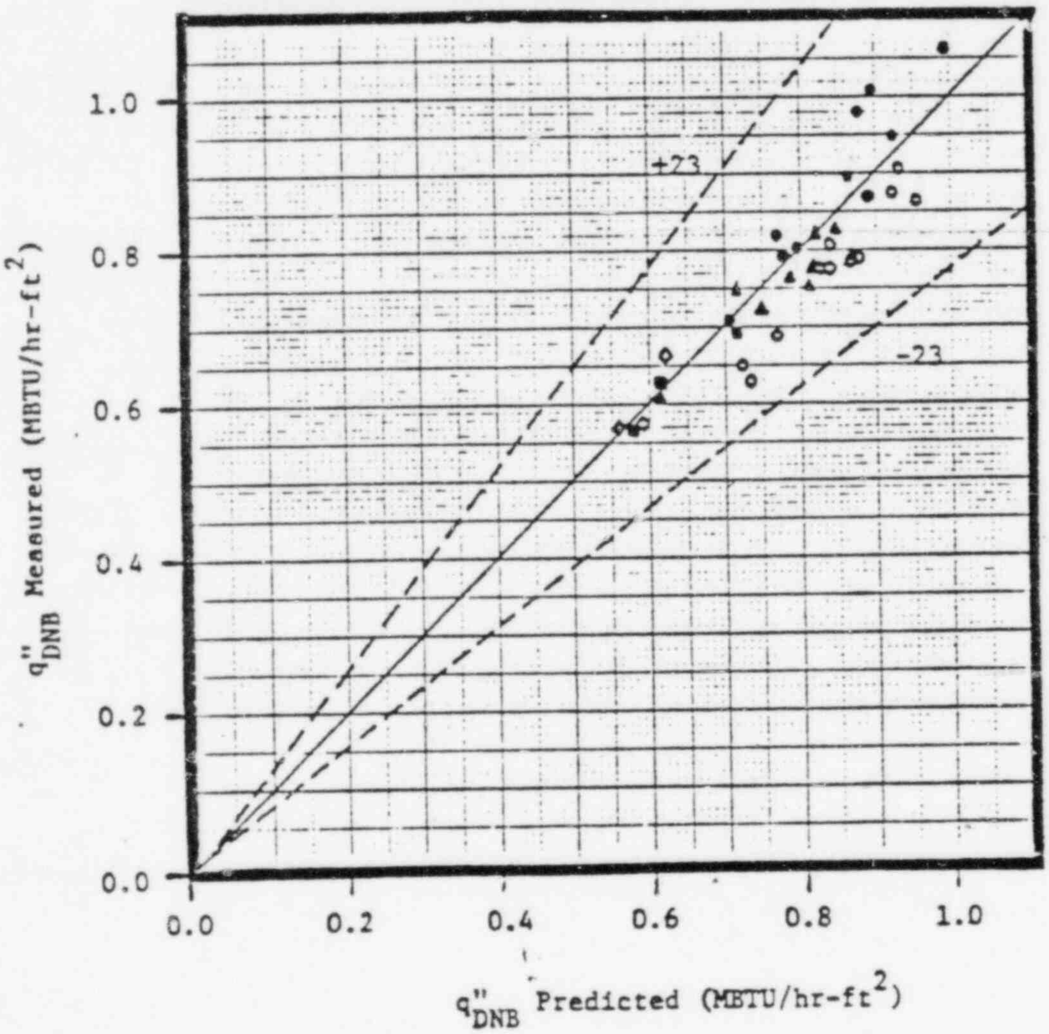
In order to verify the calculational accuracy of their thermal-hydraulic model, VEPCO initially performed steady state and transient DNB analyses duplicating their original Surry FSAR analyses. This comparison showed the VEPCO model to be consistent with previously submitted and approved licensing documents. In addition, VEPCO submitted 7 check cases comparing their model on a one for one basis with the THINC-I multi-pass 1/8 core model. In all these cases the VEPCO model yielded either identical or conservative minimum DNBRs. Finally, the staff performed audit calculations of the VEPCO analyses using COBRA-IV. These audit calculations showed VEPCO's analyses to be both consistent and conservative relative to the staff's results.

Based on the staff review of the methodology presented in reference 1 and on the verification analyses performed by VEPCO and the staff, the staff finds reference 1 to be acceptable for referencing by VEPCO in future reload licensing submittals for North Anna, Surry and future plants of the same design.

FIGURE 1

COMPARISON OF DNB DATA WITH COBRA PREDICTIONS

Heated Length	Axial Flux Distribution	Grid with Vane	Grid without Vane
8'	u sin u cosine u	● ○	▲ –
14'	u sin u cosine u	■ ◇	– –



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