Comparison of THALES and VIPRE-01 Subchannel Codes for Loss of Flow and Single Reactor Coolant Pump Rotor Seizure Accidents using Lumped Channel APR1400 Geometry

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1. Introduction

Subchannel analysis plays important role to evaluate safety critical parameters like minimum departure from nucleate boiling ratio (MDNBR), peak clad temperature and fuel centerline temperature.

In this study, two different subchannel codes, VIPRE-01 (Versatile Internals and Component Program for Reactors: EPRI) and THALES (Thermal Hydraulic AnaLyzer for Enhanced Simulation of core) are examined. Two transients, loss of feedwater and rocked rotor transients are compared for given core wide planar power distribution for sample departure from nucleate boiling analysis in Shin Kori 3&4 final safety analysis report.

2. VIPRE-01 and THALES Code

For many decades, there has been considerable interest in the area of technology known as subchannel analysis. This work has been motivated largely by the desire of nuclear reactor designers to predict accurately boiling transition and void distribution in fuel rod bundles [1].

Two different approaches can be used to define subchannels which are coolant-centered and rod centered division. Traditionally, coolant-centered approach is used more often in analysis. However, in two-phase flow, particularly in the annular regime, the liquid flow around the rod is difficult to accommodate by this approach [2].

VIPRE-01 predicts the three-dimensional velocity, pressure, and thermal energy fields and fuel rod temperatures for single and two-phase flow in PWR and BWR cores [3]. It solves the finite-difference equations for mass, energy and momentum conservation for an interconnected array of channels assuming incompressible thermally expandable homogeneous flow [3].

Modelling structure in VIPRE-01 is based on subchannel analysis. Full core or symmetric section of it is divided into flow channels which are described by flow area and wetted parameter. Connection between channels defined by channel width and distance between channel centroids. Through these connection crossflow exist between adjacent channels which is assumed to lose its sense of direction after entering the receiving channel.

VIPRE-01 is designed to evaluate safety critical parameters for nuclear power plants in steady-state and abnormal events or accidents. Information available in VIPRE-01 includes fluid velocity and state, pressure drop, critical heat flux, critical power ratio, and departure from nucleate boiling ratio (DNBR) at each computational location [4].

THALES has been developing by KNF to analyze core thermo hydraulics for OPR1000 and APR1400. It uses ASME steam tables to evaluate the coolant properties. As code is still in developing stage it doesn't take into account conduction model. So the outputs regarding fuel rods are missing from this code.

2.1 Two Phase Flow Correlations

The flow constitutive relations provide closure for mathematical solutions. Constitutive flow models can be classified in 3 groups:

1. Two-phase friction multiplier
2. Subcooled boiling models
3. Bulk void/quality relations

2.1.1 Two-Phase Friction Multiplier

The effect of two-phase flow on the friction loss can be expressed in terms of the single-phase friction pressure drop by the ratio [4]:

$$\phi_{fo}^{2} = \frac{\left.\frac{dP}{dX}\right|_{TP}}{\left.\frac{dP}{dX}\right|_{fo}} \qquad (1)$$

where;

ϕ_{fo}^{2} = two-phase friction multiplier

$\left.\frac{dP}{dX}\right|_{TP}$ = friction pressure drop in two-phase flow

$\left.\frac{dP}{dX}\right|_{fo}$ = friction pressure drop with the total two-phase flow considered to be all liquid.

Below table shows the availability of two-phase friction correlations in VIPRE-01 and THALES.

Table 1 Availability of two-phase friction multipliers correlations

Correlation	VIPRE-01	THALES
The Columbia/EPRI	+	-
Homogeneous	+	+
Armand	+	+
Beattie	+	-
Sher-Green and Martinelli-Nelson	-	+

2.1.2 Subcooled Boiling Models

Corelations used for subcooled boiling calculates flowing quality of a fluid where the bulk temperature of the fluid may still be subcooled. Calculated quality then be used in subccoled bulk void correlation in order to predict the subcooled void.

Below table shows the availability of the subcooled boiling correlations in VIPRE-01 and THALES.

Table 2 Availability of subcooled boiling correlations

Correlation	VIPRE-01	THALES
EPRI	+	-
Levy	+	+
Homogeneous	+	+
Drift Flux	+	-

2.1.3 Bulk Void/Quality Relations

In order to associate void fraction to flowing quality, void/quality correlations are introduced. Importance of those correlations is to take into account the effect of phase slip in two-phase flow. As phase slip is altered by many different factors such as flow rate, pressure, wall surface temperature and flow regime, many different correlations introduced for void quality relation.

Below table shows the availability of the bulk void correlations in VIPRE-01 and THALES.

Table 3 Availability of bulk void/quality correlations

Correlation	VIPRE-01	THALES
Modified Martinelli Nelson	-	+
Homogeneous	+	+
Modified Armand	+	+
Maier & Coddington	-	+
Chexal-Lellouche	-	+
Slip Model	+	+
Smith	+	-
EPRI	+	-
Zuber-Findlay	+	-

2.2 Heat Transfer Correlations

Heat transfer correlations are used to represent heat transfer phenomena on the different segments of boiling curve which consists of 4 major segments that are single-phase forced convection, subcooled and saturated nucleate boiling, transition boiling and film boiling.

Important step for the user is to select the best suited correlation for his problem. Below table shows the availability of the heat transfer correlation in VIPRE-01 and THALES.

Table 4 Availability of heat transfer correlations

Correlation	VIPRE-01	THALES
Subcooled Boiling		
Thom	+	+
Jens-Lottes	+	+
Saturated Boiling		
Thom	-	+
Chen	+	+
Wright	+	-
Schrock-Grossman	+	-

2.2.1 Single Phase Forced Convection

Single phase forced convection correlations are used where the wall temperature is below the onset of nucleate boiling and bulk temperature of the fluid is subcooled or saturated.

Equations for single phase forced convection can be categorized according to flow type. For turbulent flow;

$$H_T = \left(a_1 \operatorname{Re}^{a_2} \operatorname{Pr}^{a_3} + a_4\right)\left(\frac{k}{D_e}\right) \quad (2)$$

For laminar flow equation is in the same form but with different constants;

$$H_T = \left(b_1 \operatorname{Re}^{b_2} \operatorname{Pr}^{b_3} + b_4\right)\left(\frac{k}{D_e}\right) \quad (3)$$

The heat transfer coefficient is specified as the maximum of the values calculated with correlations for laminar and turbulent flow [4].

$$H_{SPFC} = \max\left(H_{turbulent}, H_{laminar}\right) \quad (4)$$

2.2.2 Subcooled Nucleate Boiling

The subcooled boiling region begins with the onset of nucleate boiling while the mean temperature is below the saturated temperature [5]. Bubbles are being generated near the hot wall. Those bubbles grow and collapse but they do not detach from the wall as the bulk temperature is still under saturation. This introduce small amount of void fraction which can be neglected. As the bulk of the coolant heats up, the bubbles can grow larger, and the possibility that they will detach from the wall surface into the flow stream increases [5]. This will further increase the void fraction.

Thom correlation is one of the available correlations for subcooled nucleate boiling which states heat transfer coefficient as;

$$H_{THOM} = \left[e^{P/1260} \left(T_w - T_f \right) / 0.072 \right]^2 / \left(T_w - T_b \right) \quad (5)$$

Another correlation for subcooled nucleate boiling is Jens-Lottes correlation which states heat transfer coefficient as;

$$H_{JL} = \left[e^{P/900} \left(T_W - T_f \right) 1.9 \right]^4 / \left(T_W - T_b \right) \quad (6)$$

Chen correlation which is originally developed for saturated boiling can also be prolonged to subcooled region. Chen correlation states heat transfer coefficient as;

$$H_{Chen} = \left(q^{"}_{NB} + q^{"}_{FC} \right) / \left(T_w - T_b \right) \quad (7)$$

where the nucleate boiling component defined as;

$$q^{"}_{NB} = H_{NB} \left(T_w - T_f \right) \quad (8)$$

and forced convection component defined as;

$$q^{"}_{FC} = 0.023 \left(\frac{k_l}{D_e} \right) \mathrm{Re}^{0.8} \, \mathrm{Pr}_l^{0.4} \left(T_w - T_b \right) \quad (9)$$

2.2.3 Saturated Nucleate Boiling

Nucleate boiling is occurring at the wall and where the flow pattern is typically bubbly, slug, or low-vapor velocity annular flow [6]. In this flow regime bulk temperature has reached saturation temperature and high heat transfer coefficients can be achieved.

Besides Thom correlation which is introduced in subcooled nucleate boiling section, there are 3 different correlations available for nucleate boiling regime.

Chen correlation applies to both the saturated nucleate boiling region and the two-phase forced convection region [4]. It uses fluid properties at saturation temperature and states heat transfer coefficient by assuming the superposition of forced convection correlation and pool boiling equation.

$$H_{Chen} = H_{SPL} + H_{NB} \quad (10)$$

where,

$$H_{SPL} = 0.023F \left(\frac{k_f}{D_e} \right) \mathrm{Re}^{0.8} \, \mathrm{Pr}^{0.4} \quad (11)$$

$$H_{NB} = 0.00122S \left[\frac{k_f^{0.79} Cp_f^{0.45} \rho_f^{0.49}}{\sigma^{0.5} \mu_f^{0.29} h_{fg}^{0.24} \rho_g^{0.24}} \right] \ldots$$

$$\left(T_w - T_f \right)^{0.24} \left(P_W - P \right)^{0.75} \quad (12)$$

Butterworth developed curve fits for both the Reynolds number factor, F, and the suppression factor, S, as follows

$$F \begin{cases} 1.0 & x_{tt}^{-1} \le 0.1 \\ 2.34 \left(x_{tt}^{-1} + 0.213 \right)^{0.736} & x_{tt}^{-1} > 0.1 \end{cases} \quad (13)$$

where,

x_{tt}^{-1} is inverse Martinelli factor defined as;

$$x_{tt}^{-1} = \left(\frac{x}{1-x} \right)^{0.9} \left(\frac{\rho_f}{\rho_g} \right)^{0.5} \left(\frac{\mu_g}{\mu_f} \right)^{0.1} \quad (14)$$

$$S \begin{cases} \left[1 + 0.12 \left(\mathrm{Re}'_{TP} \right)^{1.14} \right]^{-1} & \mathrm{Re}'_{TP} < 32.5 \\ \left[1 + 0.42 \left(\mathrm{Re}'_{TP} \right)^{0.78} \right]^{-1} & 32.5 \le \mathrm{Re}'_{TP} < 50.9 \\ 0.1 & \mathrm{Re}'_{TP} \ge 50.9 \end{cases} \quad (15)$$

Another correlation used in nucleate boiling regime is the Schrock and Grossman correlation. It estimates the heat transfer coefficient as;

$$\frac{H_{TP}}{H_{SPL}} = C_1 \left[\left(\frac{q^{"}}{G h_{fg}} \right) + C_2 \left(x_{tt}^{-1} \right)^{C_3} \right] \quad (16)$$

where,
H_{TP}=two-phase heat transfer coefficient
H_{SPL}=single-phase convection heat transfer coefficient based on liquid component flow

$$H_{SPL} = 0.023 \left(\frac{k_f}{D_e} \right) \left(\frac{G(1-x) D_e}{\mu_f} \right)^{0.8} \mathrm{Pr}_f^{0.4} \quad (17)$$

$$C_1 = 7.39x10^3$$

$$C_2 = 1.5x10^{-4}$$

$$C_3 = 0.667$$

Other option for nucleate boiling regime is the modified version of Schrock and Grossman correlation named as Wright correlation. The modifications are on empirical constants and index on the Prandtl number.

$$H_{SPL} = 0.023 \left(\frac{k_f}{D_e} \right) \left(\frac{G(1-x) D_e}{\mu_f} \right)^{0.8} \mathrm{Pr}_f^{1/3} \quad (18)$$

$$C_1 = 6.70x10^3$$

$$C_2 = 3.2x10^{-4}$$

2.3 Critical Heat Flux Correlations

The critical heat flux (CHF) phenomenon results from a relatively sudden reduction of the heat transfer capability of the two-phase coolant [5]. Effect of this phenomenon is considered and limited by MDNBR value at design stage for pressurized water reactors. Thus, accident analysis results shall be higher than this limit for a specific plant in order to indicate that plant is safe. CHF is also corresponds to upper limit of the nucleate boiling regime. For steady state analysis, CHF point defined as the location where the local heat flux exceeds the CHF computed with chosen CHF correlation. Whereas, for transient analysis it is defined as the location where the wall temperature exceeds CHF temperature as local heat flux is not known before the conduction equation is solved. The critical heat flux temperature is the wall temperature where the heat flux computed with the nucleate boiling correlation is equal to the critical heat flux [4].

Availability of critical heat flux correlations for VIPRE-01 and THALES are listed in below table.

Table 5 Availability of critical heat flux correlations

Correlation	VIPRE-01	THALES
KCE-1	-	+
CE-1	+	+
WRB-2	-	+
KNF-P	-	+
KNF-A	-	+
NGF	-	+
Machbeth	+	+
Biasi	-	+
Bowring	+	+
B&W-2	+	-
W-3	+	-
WSC-2	+	-
EPRI-1	+	-
Bezrukow	+	-
Smolin	+	-

For this study, CE-1 correlation is selected as the critical heat flux correlation to analyze APR 1400 core with PLUS 7 fuel assemblies.

3. Analysis &Results

In this section description about steady-state and transient analysis for APR 1400 with VIPRE-01 and THALES is discussed. Parameters and operating conditions are introduced for both steady-state and transient analysis.

The major concern for decreasing reactor coolant events is MDNBR. In order to investigate MDNBR, total loss of reactor coolant and single reactor coolant pump rotor seizure events are selected as transients for this study. Simulations has been performed for APR 1400 with PLUS 7 fuel using VIPRE-01 and THALES subchannel codes. Total simulation time for those events is 5 seconds as the MDNBR occurs during the first one to five seconds.

3.1 Steady-State Analysis

Steady-state analysis conducted with lumped channel geometry which is shown in figure 1 that is referenced from Shin Kori 3&4 final safety analysis report (FSAR).

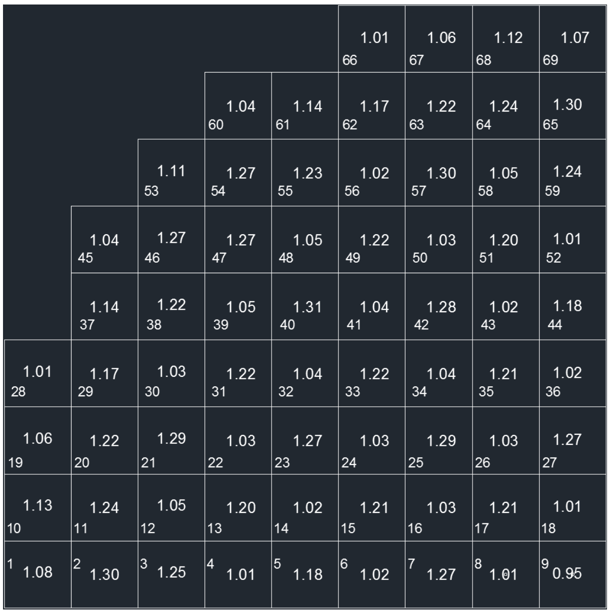

Figure 1 Core layout for analysis

In order to the compare the results from each code adequetly, same available models are selected for each code. Those mdoels are listed in below table.

Table 6 Selected models for simulations

Correlation	Name
Two-phase friction multiplier	Homogeneous
Bulk void/quality	Homogeneous
Subcooled nucleate boling	Thom
Saturated nucleate boiling	Chen
Critical heat flux	CE-1

Operating condition, core geometry and thermal-hydraulic parameters are indicated in table-7.

Results from VIPRE-01 and THALES shows small differences regarding MDNBR value. VIPRE-01 predicts MDNBR value for steady state case as 2.189 whereas THALES predicts as 2.173. Both codes agree with the location of the hot channel which is channel 65. Steady-state analysis results summarized in below table.

Table 7 Results for steady-state analysis

THALES		
Value	Channel	Axial Level
2.173	65	29.02 in
VIPRE		
Value	Channel	Axial Level
2.189	65	25.79 in

Table 8 Core geometry, operating conditions and thermal – hydraulic parameters for simulations

Core Geometry	
Number of Fuel Assemblies	241
Number of Fuel Rods per Assembly	236
Number of Control Rod Thimbles per Assembly	20
Fuel Rod Diameter (nominal)	0.374 in
Fuel Pellet Diameter	0.3225 in
Clad thickness	0.0225 in
Control Rod Thimble Diameter (nominal)	0.816 in
Fuel Pm Pitch (nominal)	0.506 in
Assembly Pitch (nominal)	8.180 in
Active Fuel Length	150 in
Fuel Rod Pitch	0.506 in
Operating Conditions	
Power (thermal)	3983 MWt
Inlet Mass Flux	2.58 Mlbm/h-ft^2
Inlet Temperature	555 ºF
Pressure	2250 psi
Thermal-Hydraulic Parameters	
Hydraulic diameter (nominal)	0.498 in
Spacing between fuel assemblies, fuel rod surface to surface	0.216 in
Spacing, outer fuel rod surface to core shroud,	0.218 in
Total flow area (excluding guide tubes)	62.7 ft^2
Total core area	112.3 ft^2
Total heat transfer area	69470 ft^2

3.2 Loss of Flow transient

A complete loss of forced reactor coolant flow will result from the simultaneous loss of electrical power to all reactor coolant pumps (RCPs) [7]. Loss of electrical power to all RCP pumps can happen in the case of complete loss of offsite power.

For the loss of offsite power event, the MDNBR occurs during the first few seconds of the transient and the reactor is tripped by the Core Protection Calculators (CPCs) on the approach to the DNBR limit [7]. Thus, any single failure which lower the DNBR shall effect during first few seconds of the transient. Whereas, none of the single failures mention in safety analysis report have any effect on the minimum DNBR during this period of transient. So no single failure is considered for this analysis.

Initial conditions for the total loss of reactor coolant flow is referenced from Shin Kori 3&4 FSAR [8] as shown in below table.

Table 9 Initial values for loss of flow accident

Parameter	Assumed Value
Core Power Level, MWt	4062.66
Core Inlet Coolant Temperature ºF	550
Pressurizer Pressure, psia	2325
Core Mass Flow, Mlbm/hr	187.46

According to simulation results, both code agree with the radial location of the hot channel whereas MDNBR values shows small differences. VIPRE-01 predicts this value as 2.145 however THALES prediction is 2.107. Below table and figure summarizes the simulation results for loss of flow transient.

Table 10 Results for loss of flow accident

THALES		
Value	Hot Channel	Axial Level
2.107	65	25.79 in
VIPRE		
Value	Hot Channel	Axial Level
2.145	65	25.79 in

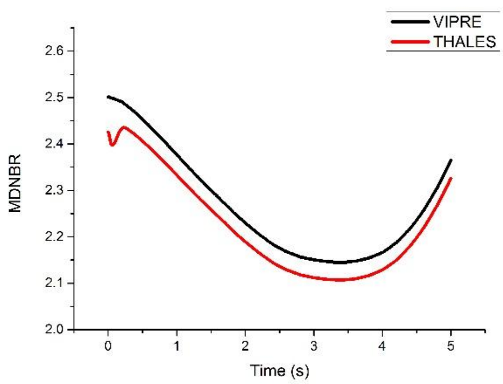

Figure 2 MDNR for loss of flow accident

3.3 Single Reactor Coolant Pump Rotor Seizure Transient

A single reactor coolant pump rotor seizure can be caused by seizure of the upper or lower thrust-journal bearings [7]. Likewise the loss of flow accident, MDNBR happens within few seconds of the accident. For this event, AC power is assumed to be lost which makes pressurizer pressure and level control systems, the reactor regulating system, the feedwater control system and steam bypass control system inoperable. But those system don't play any significant role on MDNBR during the transient so no single failure criteria cause more severe result regarding MDNBR. Initial conditions for single reactor coolant pump rotor seizure are as in below table.

Table 11 Initial conditions for single reactor coolant pump rotor seizure accident

Parameter	Assumed Value
Core Power Level, MWt	4062.66
Core Inlet Coolant Temperature ^{o}F	550
Pressurizer Pressure, psia	2337.2
Core Mass Flow, Mlbm/hr	153.52

Simulation results show that channel 65 is the hot channel. THALES predicts MDNBR as 1.704 whereas VIPRE-01 prediction is 1.737.Below table and figure summarizes the simulation results.

Table 12 Results for single reactor coolant pump rotor seizure accident

THALES		
Value	Hot Channel	Axial Level
1.704	65	25.79 in
VIPRE		
Value	Hot Channel	Axial Level
1.737	65	25.79 in

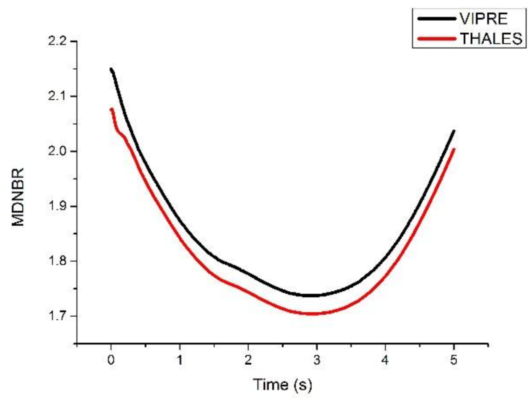

Figure 3 MDNR for single reactor coolant pump rotor seizure accident

3.4 Effect of fuel Conduction model of VIPRE-01

Another set of simulations conducted to investigate effect of conduction model that is available in VIPRE-01. For this model built-in material properties are used for clad and fuel materials which are available in VIPRE-01. Conduction model handles heat transfer to coolant with solving the conduction equations throughout the fuel region whereas without conduction model heat transfer to coolant occurs via heat flux boundary conditions.

Simulation with conduction model gives lower MDNBR and time of MDNBR occurrence is delayed for the both transients. The results are summarized in below tables and figures.

Table 13 Results for loss of flow accident with conduction model in VIPRE-01

THALES		
Value	Hot Channel	Axial Level
2.107s	65	25.79 in
VIPRE with Conduction Model		
Value	Hot Channel	Axial Level
2.026	65	25.79 in

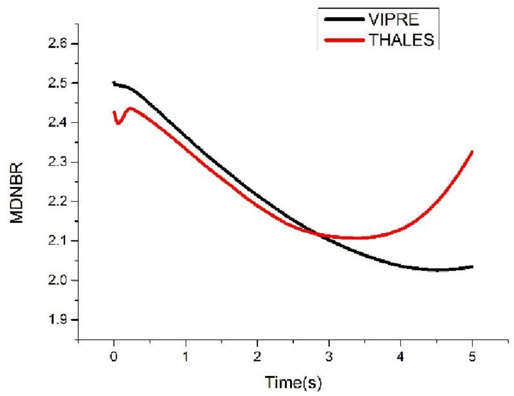

Figure 4 MDNBR for loss of flow accident with conduction model in VIPRE-01

Table 14 Results for single reactor coolant pump rotor seizure accident with conduction model in VIPRE-01

THALES		
Value	Hot Channel	Axial Level
1.704	65	25.79 in
VIPRE with Conduction Model		
Value	Hot Channel	Axial Level
1.652	65	25.79 in

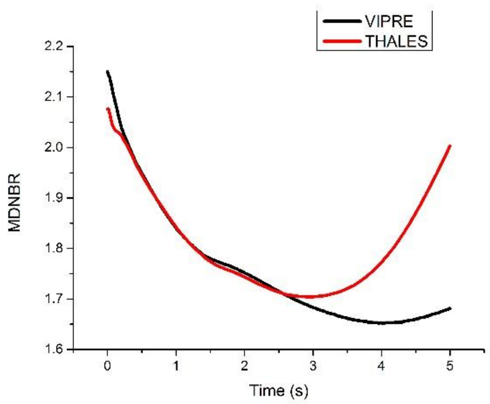

Figure 5 MDNBR for single reactor coolant pump rotor seizure accident with conduction model in VIPRE-01

4. Conclusion

In this study, two different transient cases for which MDNBR result play important role are selected to conduct analysis with THALES and VIPRE-01 subchannel codes. In order to get comparable results same core geometry, fuel parameters, correlations and models are selected for each code.

MDNBR results from simulations by both code are agree with each other with negligible difference. Whereas, simulations conducted by enabling conduction model in VIPRE-01 shows significant difference from the results of THALES. Regarding this issue more detailed analysis are plan to be conduct for future. By conducting analysis with finer geometry and subchannel layout for hot assembly may reveal more precise and reliable results for the effect of conduction model.

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