EVALUATION OF THALES SUBCHANNEL CODE BEHAVIOR FOR LOSS OF FLOW AND RCP ROTOR SEIZURE SCENARIOS

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ABSTRACT

KEPCO Nuclear Fuel (KNF) recently developed a new subchannel analysis code called THALES (Thermal Hydraulic AnaLyzer for Enhanced Simulation of core). To benchmark the code independently, its performance and characteristics are examined by simulating loss of flow and RCP rotor seizure scenarios of Shinkori 3&4. The core inlet and outlet conditions are taken from Shinkori 3&4 design with PLUS7 fuel. VIPRE-01 code is used for comparison. VIPRE-01 code was developed by EPRI and has been used widely. The above transients were simulated with the same correlation options.

For the comparison between the codes, we chose CE-1 CHF correlation, which are available in both codes. CE-1 correlation was developed for CE type fuel. The comparison shows that the code predictions are close. The key parameter is minimum departure from nucleate boiling ratio (MDNBR). The maximum difference in MDNBR is less than 3%. The axial location of MDNBR is very close though the channel exhibiting MDNBR is different. However, the channel is located symmetrically and the radial power values of the channel are the same.

Then, THALES code is used to examine PLUS7 fuel performance with KCE-1 CHF correlation which is developed for PLUS7 fuel. The effect of using KCE-1 correlation improves MDNBR showing the effect of improved fuel performance. Lastly, the effect of axial power shape is examined. Several different axial power shapes are examined to identify the most limiting shape, which is one of top-skewed profile.

1. INTRODUCTION

Subchannel means the flow channels inside fuel assembly in the reactor core. Subchannel analysis plays an important role in evaluating safety critical parameters in the core, such as departure from nucleate boiling, fuel and clad temperatures. Departure from Nucleate Boiling Ratio (DNBR) is the one of the key parameters to evaluate thermal margin of Pressurized Water Reactor (PWR).

THALES has been developed by KEPCO Nuclear Fuel (KNF). Since it is a newly developed code, it is worthwhile to examine the code performance independently, outside of KNF. From this perspective, we decide to compare its performance against a well-established subchannel code for benchmarking. There are several subchannel codes like THALES [1, 2]. Examples are VIPRE-01 [3], THINC-IV[4], Thermal-hydraulic Of Reactor Core (TORC) [5], and The Multichannel Analyzer for Transient and steady-state in

Rod Array (MATRA) [6]. These codes are developed by Electric Power Research Institute (EPRI), Westinghouse Electric Company (WEC), Combustion Engineering, Inc. (CE), and Korea Atomic Energy Research Institute (KAERI), respectively. Among the codes, we chose VIPRE-01 code since it has been widely used by Pressurized Water Reactor (PWR) nuclear utilities and well established in simulating thermal-hydraulic behavior and calculating DNBR in the nuclear fuel assembly.

The main purpose in this paper is safety analysis for Shinkori 3&4 with PLUS7 fuel for the benchmark of THALES subchannel code. The loss of flow and RCP rotor seizure accident scenarios of Shinkori 3&4 design are chosen.

Loss of flow and RCP rotor seizure are the limiting scenarios from the DNBR point of view. The core inlet and outlet boundary conditions are taken from Shinkori 3&4 design [7]. First, we simulate both codes with the identical constitutive relations to observe any differences. Then, the effects of KCE-1 CHF correlation and axial power shape are examined using THALES code.

2. SUBCHANNEL ANALYSIS CODE, THALES

THALES predicts thermal-hydraulic properties of local coolant by solving governing equations between the flow channels composed of rods as a minimum unit of core flow distribution analysis. This approach is similar to other subchannel codes. PLUS7 fuel is loaded in Shinkori 3&4. For PLUS7 fuel, KNF developed KCE-1 correlation based on PLUS7 CHF test data. The algebraic form of CE-1 correlation has been maintained in KCE-1 correlation.

To analyze fluid flow problem, Navier-Stokes Equations are converted to algebraic equation through the discretization such as Finite Difference Method (FDM), Finite Element Method (FEM), Finite Volume Method (FVM). THALES uses FDM. THALES provides two options for matrix solver; Iterative matrix solver and Gaussian Elimination Method. The iterative matrix solver, Preconditioned Bi-Conjugate Gradient Method (PBCGM) is generally used and the direct solver, Gaussian Elimination Method (GEM) is used for solving single channel problems.

3. SPECIFICATION FOR NUCLEAR FUEL AND REACTOR CORE

There are two types of reactors in Korea; one Westinghouse (WH) type, the other CE type. Shinkori 3&4 are based on CE type design. Nuclear fuel varies according to the reactor type. ACE7 fuel belongs to Westinghouse (WH) type. PLUS7 fuel belongs to KSNP type. PLUS7 fuel was developed by KNF and has been supplied to Shinkori 3&4.

3.1. PLUS7 Fuel

PLUS7 geometry data is used for THALES input [7, 8]. The channel/subchannel informations including cross section flow area, wetted perimeter, heated perimeter, and interface gap of channel/subchannel were used for THALES analysis.

3.2. Shinkori 3&4 Reactor Core

Shinkori 3&4 is an evolutionary advanced light water reactor based on the CE type plant, which is operation in Korea and incorporates the recent Nuclear Steam Supply System (NSSS) design [7]. The fuel loading operation of Shinkori 3&4 is expected in the near future with PLUS7 fuel in Korea.

For Shinkori 3&4, the total number of fuel assembly is 241. Code simulation is performed with 1/4. 1/4 core is adopted since the nuclear data of 1/4 core is symmetrical with respect to the horizontal and vertical axis each other. For this study, the typical thermal-hydraulic parameters of Shinkori 3&4 are listed in the Table I.

Table I. The typical thermal-hydraulic parameters of Shinkori 3&4

Parameter	Value	Unit
Total core heat output	3,983	MWt
Primary system pressure	158	kg/cm^2A
Reactor inlet coolant temperature	291	℃
Design minimum primary coolant flow rate	1,689,000	l/m

4. SUBCHANNEL ANALYSIS FOR SHINKORI 3&4 REACTOR CORE

Some of the subchannel analysis codes use multi-stage approach. For example, wide and coarse mesh simulations are used at the beginning. Then, fine mesh calculation is performed for the hot assembly. THALES code solves the single-stage model that considers the hot assembly/subchannel and the adjacent channels/nodes at the same time (Figures 1 and 2).

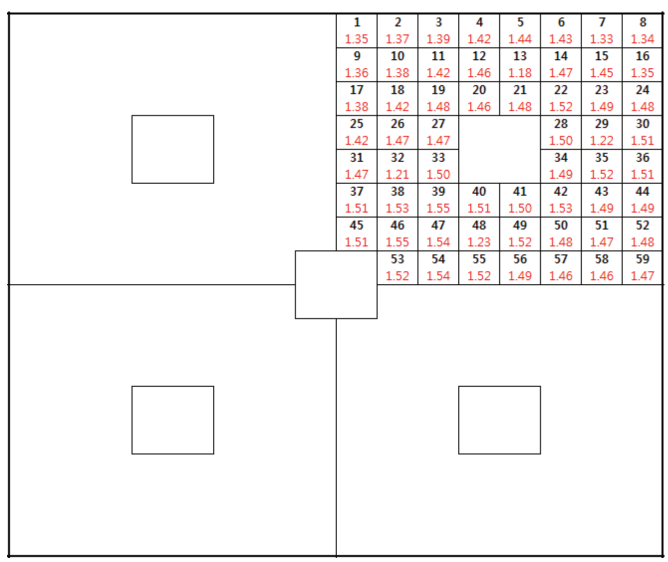

Figure 1. Typical Rod radial power factors in hot assembly quadrant for DNB analysis in Shinkori 3&4 and its application using THALES single-stage model.

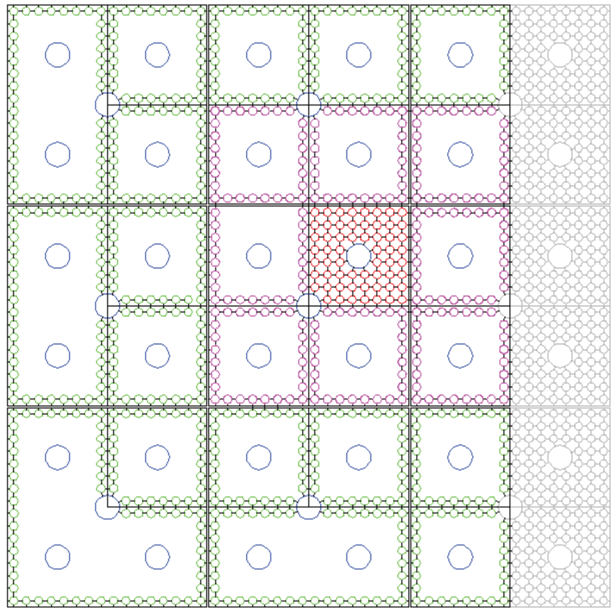

Figure 2. Configuration of subchannels and nodes using THALES single-stage model.

4.1. Common Input Parameters

The thermal-hydraulic models (correlations) applied in this study are distinguished for comparison with VIPER-01 and for THALES specific analysis as shown in Table II. For the benchmarking purpose, we chose the correlations that are available for both VIPRE-01 and THALES code. The correlations sets for Shinkori 3&4 analysis are also shown in the table for comparison.

The options used are default PBCGM iterative solver as a matrix solver and NIST water & steam table.

Table II. Thermal-hydraulic models selected in THALES

Model	For comparison with VIPRE-01	For THALES specific analysis
Node division modelling	Channel only	Subchannel/Channel combination
Turbulent mixing correlation, WP	AAT*(Re)**BBT*Gavg*HDav	AAT*(Re)**BBT*Gavg*HDav *CGW*NSCCB/(GWCB/12.)
Two phase friction multiplier correlation	Armand	Sher-Green and Martinelli-Nelson
Void model	Armand	Armand
Flowing quality	Equilibrium	Equilibrium
Subcooled nucleate boiling correlation	Jens-Lottes	Jens-Lottes
Inverse Peclet number, *1/Pe* (Thermal Diffusion Coefficient, TDC)	-	Design Value
Critical Heat Flux, CHF correlation	CE-1	KCE-1
Boundary condition for core inlet	Inlet flow distribution	Inlet flow distribution
Boundary condition for core outlet	Uniform outlet pressure distribution	Outlet pressure distribution

THALES uses the single-stage approach. Core power distribution and radial power factors are based on Shinkori 3&4 design. Typical core power distribution is shown in Figure 3. Typical rod radial power factor in hot assembly are shown in Figure 1. The consideration of the adjacent channels and nodes are shown in

Figure 2. The detailed radial power factors and those node divisions are for THALES specific analysis described in Section 5.

VIPRE-01 code uses the single-stage approach. But VIPRE-01 cannot apply in the same way with THALES as shown in Figures 1 and 2. For the comparison with VIPRE-01 code, we consider each assembly in Figure 3 as a lumped volume and the maximum values are used for radial power factors.

No. of Assembly					1	2	3	4
Max					1.01	1.06	1.12	1.08
Avg.					0.61	0.76	0.83	0.81
			5	6	7	8	9	10
			1.04	1.14	1.17	1.22	1.24	1.30
			0.65	0.88	0.98	1.09	1.12	1.19
		11	12	13	14	15	16	17
		1.11	1.27	1.23	1.02	1.30	1.05	1.25
		0.74	1.08	1.10	0.95	1.20	0.97	1.15
	18	19	20	21	22	23	24	25
	1.04	1.27	1.24	1.05	1.22	1.03	1.20	1.01
	0.65	1.08	1.15	0.97	1.14	0.96	1.12	0.94
	26	27	28	29	30	31	32	33
	1.14	1.22	1.05	1.31	1.04	1.28	1.01	1.18
	0.88	1.10	0.97	1.22	0.97	1.19	0.95	1.11
34	35	36	37	38	39	40	41	42
1.01	1.17	1.02	1.22	1.04	1.22	1.03	1.21	1.02
0.61	0.98	0.95	1.14	0.97	1.15	0.97	1.13	0.95
43	44	45	46	47	48	49	50	51
1.06	1.22	1.29	1.03	1.27	1.03	1.29	1.03	1.27
0.76	1.09	1.20	0.96	1.19	0.97	1.21	0.96	1.19
52	53	54	55	56	57	58	59	60
1.13	1.24	1.05	1.20	1.02	1.21	1.03	1.21	1.01
0.83	1.12	0.97	1.12	0.95	1.13	0.96	1.13	0.93
61	62	63	64	65	66	67	68	69
1.08	1.30	1.25	1.01	1.18	1.02	1.27	1.01	0.95
0.81	1.19	1.15	0.94	1.11	0.95	1.19	0.93	0.90

Figure 3. Typical core power distribution for sample DNB analysis in Shinkori 3&4.

4.2. Steady-state Condition

The operating condition for the steady-state is shown in Table I. Total core heat output is converted to core average heat flux and core mass flow rate is converted to core average mass velocity as follows:

- $Core\ average\ heat\ flux = 518{,}949\ kcal/hr - m^2$
- Active core length : 3.81 m
- $Core\ average\ mass\ velocity = 12{,}588{,}013\ \ kg/hr - m^2$

Axial power distribution is examined by ASI. For Shinkori 3&4, ASI are classified to several types and axially separated to fifty nodes. ASI-1 is provided as a default axial power distribution, and is used in the steady-state analysis. ASI-1 is shown in Figure 4 and the power fraction of top part is higher than that of bottom part as much as 7%. The additional analysis using other ASI types at the steady and transient conditions is described in Section 5.

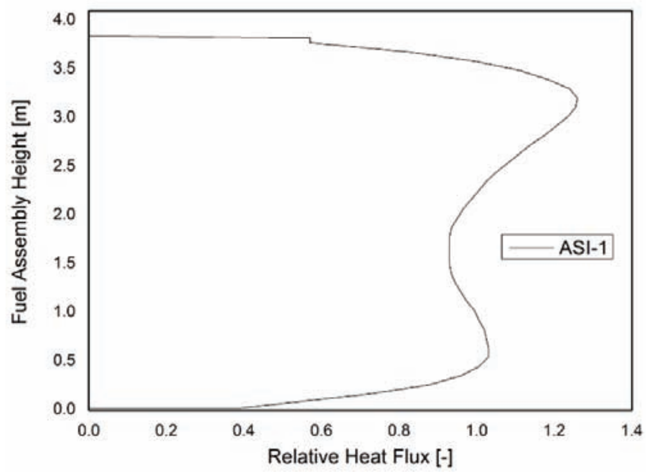

Figure 4. The representative ASI-1.

To compare MDNBR between THALES and VIPRE-01, the thermal-hydraulic models of THALES were selected to match those of VIPRE-01 as shown in Table II (left row). At steady state, MDNBR of THALES was 2.957 on the #10 of channel and MDNBR of VIPRE-01 was 2.954 on the #62 of channel as shown in Table III. The difference of MDNBR between THALES and VIPRE-01 was 0.1%. The channel of MDNBR was different from each other. However, the radial power values of them were symmetric and same (Figure 3). The axial location of MDNBR is close to each other. The result shows that the two codes predict the steady state reasonable close to each other.

Table III. Comparison of MDNBR of hot channel between THALES and VIPRE-01 at the steady-state condition

	THALES	VIPRE-01
MDNBR	2.957	2.954
Channel number	#10	#62
Axial location (channel)	3.357 m (#42)	3.276 m (#41)

4.3. Transient Accident

There are four accident analyses that are related to subchannel analysis in Shinkori 3&4 safety analysis. Two scenarios are more limiting and have the concern on MDNBR. These two accidents are total loss of reactor coolant flow and single reactor coolant pump rotor seizure with loss of offsite power.

4.3.1. Total Loss of Reactor Coolant Flow

4.3.1.1. Introduction and Causes of Accident

Total loss of reactor coolant flow accident happens when all Reactor Coolant Pumps (RCPs) lose their power simultaneously. The cause that all RCPs lose their power at the same time is the loss of offsite power. The loss of offsite power, in addition, causes turbine trip and make steam dump and bypass system not to operate. Therefore, Nuclear Power Plant (NPP) should be cooled through Main Steam Safety Valves (MSSVs) and Atmospheric Dump Valves (ADVs).

Total loss of reactor coolant flow accident causes more severe MDNBR than other loss of flow accidents. The factors which decrease DNBR locally are as follows:

- Increase of coolant temperature
- Decrease of coolant pressure
- Increase of local heat flux
- Decrease of coolant flow

4.3.1.2. Analysis and its Results

For Shinkori 3&4 safety analysis, the reaction of NSSS by total loss of reactor coolant flow accident was simulated using transient analysis code. The boundary conditions required for the subchannel analysis are core coolant inlet pressure, temperature, mass flow rate, and core average heat flux. The initial conditions for total loss of flow accident are shown in Table IV. The core coolant inlet pressure and temperature are kept constant for five seconds to give conservatism though the temperature and pressure of system are decreased. The core average heat flux and mass flow rate change through the transient and their behaviors are shown in Figure 5.

Table IV. Assumed initial conditions for total loss of reactor coolant flow

Parameter	Assumed value	Unit
Total core heat output	4062.66	MWt
Primary system pressure	163.45	kg/cm^2A
Reactor inlet coolant temperature	287.78	℃
Core mass flow rate	$85.03x10^6$	kg/hr

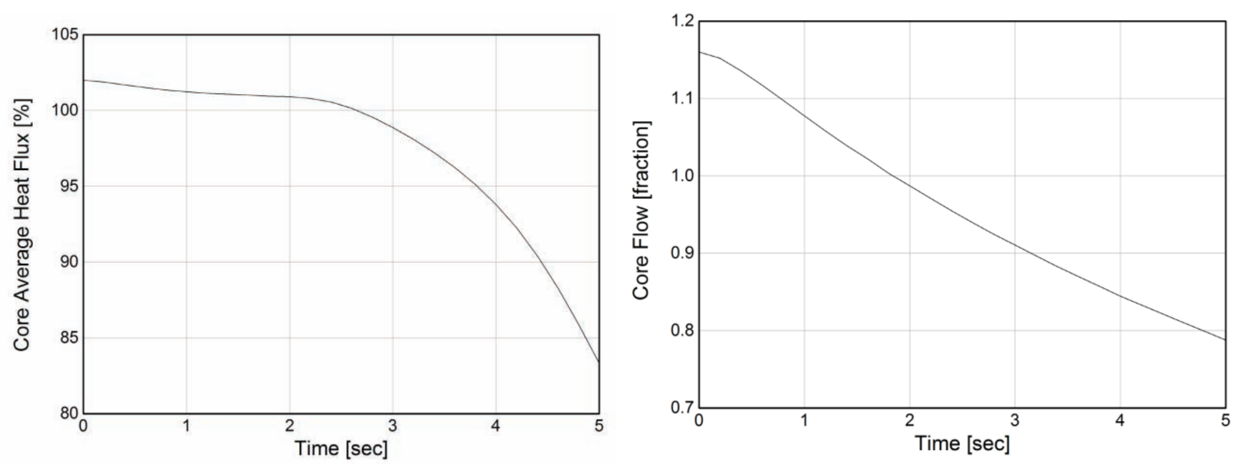

Figure 5. Thermal-hydraulic profiles of hot channel for total loss of reactor coolant flow.

Thermal-hydraulic models shown in Table II (left row) are used and ASI-1 is used as an axial power distribution for both codes. MDNBR of THALES was 2.953 on the #10 of channel while MDNBR of VIPRE-01 was 2.879 on the #29 of channel as shown in Table V. The maximum MDNBR difference between THALES and VIPRE-01 was 2.84%. The channel of MDNBR was different from each other. The axial location of MDNBR was close to each other. The variation of MDNBR with time was very close to each other with a small offset between them as shown in Figure 6. The results are reasonable considering the difficulty in matching all the conditions for the transient accident.

Table V. Comparison of MDNBR of hot channel between THALES and VIPRE-01 for total loss of reactor coolant flow

	THALES	VIPRE-01
MDNBR	2.953	2.879
Channel number	#10	#29
Axial channel (channel)	3.357 m (#42)	3.276 m (#41)

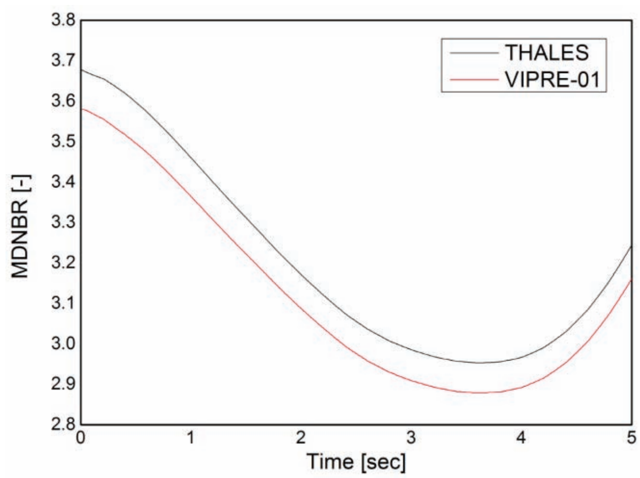

Figure 6. Comparison of MDNBR profiles of hot channel between THALES and VIPRE-01 for total loss of reactor coolant flow.

4.3.2. RCP rotor seizure with loss of offsite power

4.3.2.1. Introduction and Causes of Accident

Single Reactor Coolant Pump Rotor Seizure (SRCPRS) is caused by the seizure of thrust-journal bearing at the upper and lower parts. And this accident is assumed that the offsite power loses by turbine trip. Thus, it leads to loss of main feedwater flow, condenser inoperability since onsite loads lose power, and a coastdown by the RCPs remained at the same time.

The factors which are decreased to DNBR locally as follows:
- Increase of coolant temperature
- Decrease of coolant pressure
- Increase of local heat flux (included radial and axial power distribution effect)
- Decrease of coolant flow

4.3.2.2. Analysis and its Results

For Shinkori 3&4 safety analysis, the reaction of NSSS by total loss of reactor coolant flow accident was simulated using transient analysis code. The boundary conditions required for the subchannel analysis are core coolant inlet pressure, temperature, mass flow rate, and core average heat flux. The initial conditions for RCP rotor seizure with loss of offsite power accident are shown in Table VI. The core average heat flux and mass flow rate change through the transient and their behaviors are shown in Fig. 7.

Table VI. Assumed initial conditions for RCP rotor seizure with loss of offsite power

Parameter	Assumed value	Unit
Total core heat output	4062.66	MWt
Primary system pressure	163.34	kg/cm^{2}A
Reactor inlet coolant temperature	287.8	℃
Core mass flow rate	69.64x10^6	kg/hr

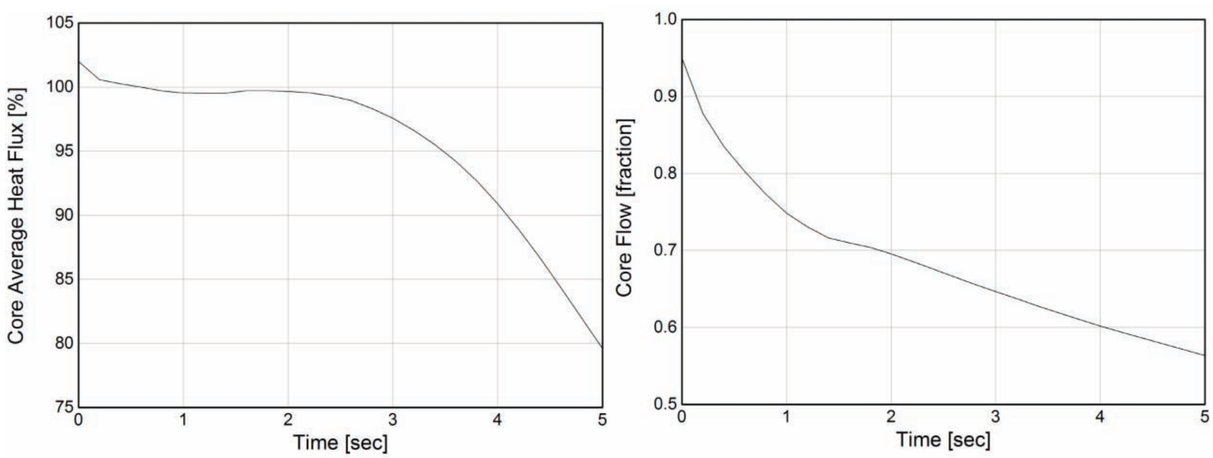

Figure 7. Thermal-hydraulic profiles of hot channel for RCP rotor seizure with loss of offsite power.

To compare MDNBR between THALES and VIPRE-01, thermal-hydraulic models chosen for steady-state condition are used and ASI-5 is used as an axial power distribution for both codes. MDNBR of THALES was 1.715 on the #10 of channel while MDNBR of VIPRE-01 was 1.671 on the #10 of channel as shown in Table VII. The maximum MDNBR difference between THALES and VIPRE-01 was 2.6%. The channel of MDNBR and the axial location are the same for this accident scenario. The variation of MDNBR with time was almost same as shown in Fig. 8. The comparison shows that the two codes predict reasonably close to each other considering the complexity in predicting the transient accident.

Table VII. Comparison of MDNBR of hot channel between THALES and VIPRE-01 for RCP rotor seizure with loss of offsite power

	THALES	VIPRE-01
MDNBR	1.715	1.671
Channel number	#10	#10
Axial location (channel)	0.737 m (#10)	0.737 m (#10)

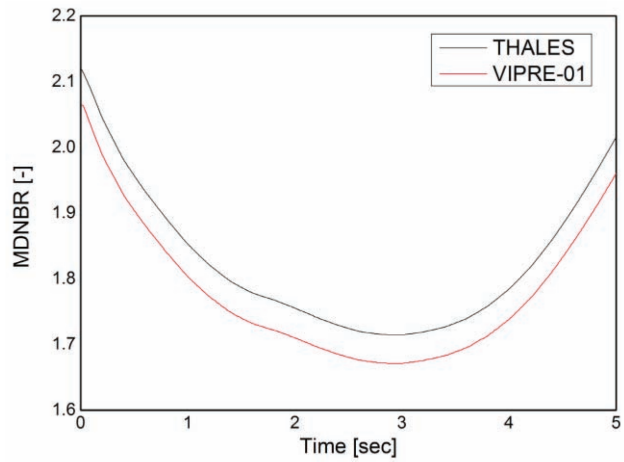

Figure 8. Comparison of MDNBR profiles of hot channel between THALES and VIPRE-01 for RCP rotor seizure with loss of offsite power.

5. THALES SPECIFIC ANALYSES

In this section, the effect of CHF correlation and axial power shape are examined to better understand the behavior of THALES. The thermal-hydraulic models for THALES specific analysis are applied as shown in Table II (right row).

5.1. CHF Correlation

For comparison between the two codes, THALES and VIPRE-01, CE-1 CHF correlation [9] was chosen in the section 4. CE-1 correlation was developed for CE type fuel. When PLUS7 fuel was developed, KNF performed CHF tests and developed KCE-1 CHF correlation [8]. KCE-1 correlation utilized the general form of CE-1 correlation. KCE-1 correlation is not available for VIPRE-01 code. Therefore, the effect of using CE-1 and KCE-1 correlations is examined with THALES. The MDNBR based on CE-1 correlation is substantially lower at the steady-state condition as expected. Though the location of MDNBR was similar, the exact location was not matched as shown in Table VIII.

Table VIII. MDNBR at the steady-state by condition CHF correlations

	KCE-1	**CE-1**
MDNBR	2.542	2.085
Channel number	#49	#48
Rod number	#40	#99
Axial location (channel)	3.194 m (#40)	3.439 m (#43)

Figure 9 shows the MDNBR profile for total loss of reactor coolant flow. PLUS7 fuel has a better thermal performance than CE type fuel and KCE-1 correlation predicts with high value of DNBR than CE-1 correlation reflecting those of geometric characteristics, respectively. Hence, it's expected that the results with CE-1 correlation have lower CHF values than those of KCE-1 despite the same PLUS7 geometry. The CHF variation with time is similar because the KCE-1 correlation used the same form as that of CE-1 correlation. The maximum difference of MDNBR between KCE-1 and CE-1 is about 19.3%.

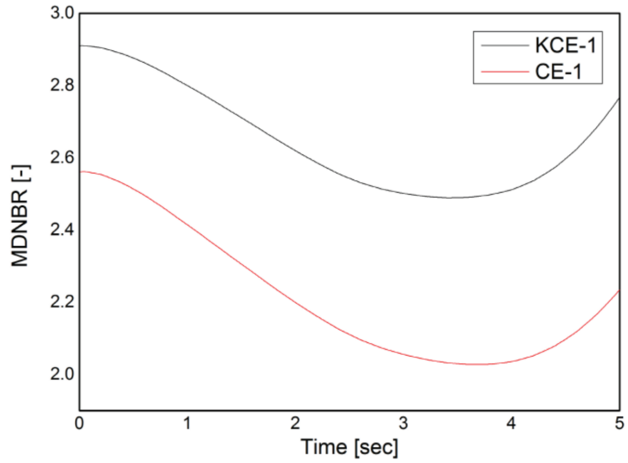

Figure 9. MDNBR profiles of hot channel for total loss of reactor coolant flow by CHF correlations.

5.2. Axial Power Distribution

In Section 4, ASI-1 is used for calculating DNB. In this section, the effect of various axial power distribution is examined. ASI is generated by normalizing each axial power factor. For PLUS7 fuel design, several types of ASI can be used in thermal-hydraulic design as shown in Figure 10.
MDNBRs were calculated using THALES code for types of ASI. The values and its locations of hot channel at the steady-state are listed in Table IX. The lowest MDNBR was obtained for ASI-2. MDNBR is 1.737 at the channel number 49 and rod number 40. The results show that MDNBR is dependent on their axial peak power. MDNBR values of top peak are lower than those of bottom peak in general. The location of MDNBR is higher than the location which has axial peak value. It seemed that these two facts are due to upflow effect.

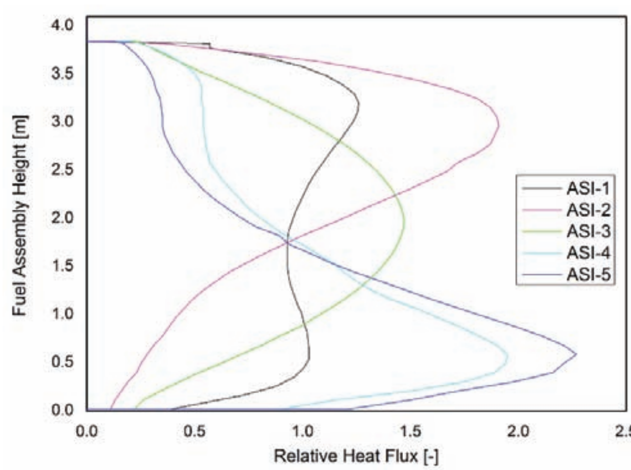

Figure 10. Several types of ASI used in the thermal-hydraulic design for subchannel analysis.

Table IX. MDNBR values and its locations of hot channel at the steady-state condition by ASI types

ASI types	MDNBR	Channel number	Rod number	Axial location (channel)	Note
ASI-1	2.542	#49	#40	3.194 m (#40)	
ASI-2	1.737	#49	#40	3.194 m (#40)	Lowest
ASI-3	2.464	#49	#40	2.375 m (#30)	
ASI-4	2.151	#48	#39	0.655 m (#9)	
ASI-5	1.819	#49	#40	0.737 m (#10)	

In case of total loss of reactor coolant flow accident, the MDNBR is also lowest with ASI-2. The value is 1.697. The MDNBR results by different ASI types are shown in Figure 11.

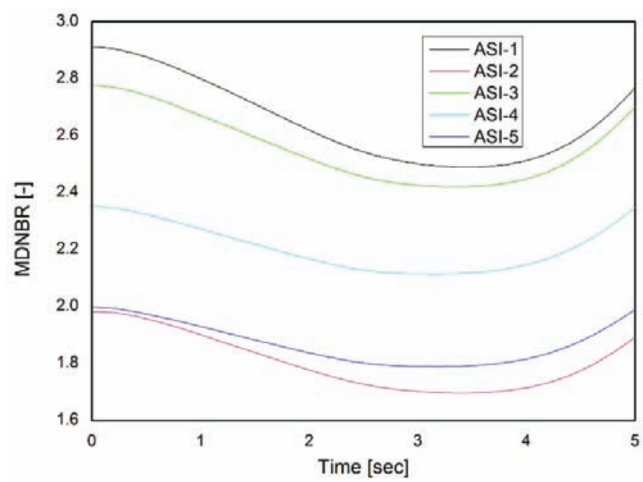

Figure 11. MDNBR profiles of hot channel for total loss of reactor coolant flow by ASI types.

6. CONCLUSIONS

In this study, THALES code was benchmarked. To compare MDNBR between THALES and VIPRE-01, the same thermal-hydraulic correlation sets of THALES were selected as that of VIPRE-01. For steady state, the two codes predict reasonably close to each other. The difference of MDNBR between THALES and VIPRE-01 was 0.1%. The location of MDNBR was different from each other. However, the locations were symmetric and radial power values of them were the same. The axial location is close to each other. For the two transients, the result shows that the MDNBR profiles are similar with a small offset. Considering the complexity in simulating the transients, the result indicates that THALES code is well benchmarked against VIPRE-01 code for the two accident scenarios.

The additional analyses were performed about the effect of CHF correlation, and ASI. KCE-1 and CE-1 CHF correlations were developed for PLUS7 and CE type fuels reflected their own characteristics. KCE-1 correlation is proprietary information and is not available for VIPRE-01 code. As expected, KCE-1 correlation yields higher MDNBR than CE-1 correlation with PLUS7 fuel geometry. ASI is generated by normalizing each axial power factor. For PLUS7 fuel design, several types of ASI can be used in thermal-hydraulic design. The results show that the MDNBR is dependent on their axial peak power and the MDNBR values of top peak are lower than those of bottom peak overall. In addition, the location of MDNBR is higher than the location which has axial peak value.

Two transient accidents were simulated by two subchannel codes, THALES and VIPRE-01 with lumped volume. And this paper is just for study, so there is no connection with the real design.

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