# METHODOLOGY FOR THERMAL HYDRAULIC CONCEPTUAL DESIGN AND PERFORMANCE ANALYSIS OF KALIMER CORE

YOUNG-GYUN KIM, WON-SEOK KIM, YOUNG-JIN KIM, CHANG-KUE PARK
Korea Atomic Energy Research Institute,
Taejon, Republic of Korea

## Abstract

This paper summarizes the methodology for thermal hydraulic conceptual design and performance analysis which is used for KALIMER core, especially the preliminary methodology for flow grouping and peak pin temperature calculation in detail. And the major technical results of the conceptual design for the KALIMER 98.03 core was shown and compared with those of KALIMER 97.07 design core. The KALIMER 98.03 design core is proved to be more optimized compared to the 97.07 design core. The number of flow groups are reduced from 16 to 11, and the equalized peak cladding midwall temperature from 654℃ to 628℃. It was achieved from the nuclear and thermal hydraulic design optimization study, i.e. core power flattening and increase of radial blanket power fraction. Coolant flow distribution to the assemblies and core coolant/component temperatures should be determined in core thermal hydraulic analysis. Sodium flow is distributed to core assemblies with the overall goal of equalizing the peak cladding midwall temperatures for the peak temperature pin of each bundle, thus pin cladding damage accumulation and pin reliability. The flow grouping and the peak pin temperature calculation for the preliminary conceptual design is performed with the modules ORFCE-F60 and ORFCE-T60 respectively. The basic subchannel analysis will be performed with the SLTHEN code, and the detailed subchannel analysis will be done with the MATRA-LMR code which is under development for the K-Core system. This methodology was proved practical to KALIMER core thermal hydraulic design from the related benchmark calculation studies, and it is used to KALIMER core thermal hydraulic conceptual design.

## 1. INTRODUCTION

KALIMER (Korea Advanced Liquid Metal Reactor), a pool-type sodium cooled prototype reactor with thermal output of 392 MW (electric power of 150 MW), is currently under conceptual design study at KAERI. The objective of the KALIMER program is to develop an inherently and ultimately safe, environmentally friendly, proliferation-resistant and economically viable fast reactor concept. The KALIMER core is initially designed with 20% enriched U-10%Zr binary alloy metallic fuels, which generates a net negative reactivity with an inherent safety characteristics[1].

In this context, K-Core system, an integrated system of the KALIMER core design and analysis modules, is being developed in the KALIMER core design and technology development team, to provide major data links among the principal core design modules.

Core steady state thermal hydraulic performance analysis includes coolant flow distribution to the assemblies and core coolant/fuel temperature calculations. LMR core has generally configured duct assemblies with a triangular channel arrangement of fuel rods within, which made a closed circuit by themselves without any flow path between them. Thus the purpose of core thermal hydraulic design is to efficiently extract the thermal power of each assembly by distributing the appropriate sodium coolant flow.

### 1.1. Coolant Flow Distribution to the Assemblies

Sodium coolant flow is distributed to the assemblies with the overall goal of equalizing pin cladding damage accumulation and pin reliability. Sodium flow distributed in each assembly has to ensure the integrity of fuels and structures, and no sodium coolant boiling is allowed in both steady state and transient conditions[2,3].

In practice, initial flow grouping analysis attempts to equalize peak pin cladding midwall temperatures in all assemblies. Assuming peak fuel burnup to be equal in all assemblies, this flow grouping analysis can equalize fission gas pressure, cladding stress and damage accumulation.

Sodium flow is distributed according to the peak linear power of each assembly. The peak temperatures of the coolant assembly outlet, cladding midwall and fuel centerline are calculated in each assembly with previously distributed flows. Once the flow is distributed so as to equalize peak cladding temperatures in all assemblies, the remaining flow and thermal criteria should be checked to verify all criteria are met.

## 1.2. Coolant and Fuel Temperature Calculation

The key core temperature parameters are: (a) peak subchannel coolant temperature, (b) peak cladding midwall temperature, (c) peak thermal striping potential, (d) peak assembly outlet temperature, (e) peak fuel surface temperature and (f) peak fuel centerline temperature. The peak subchannel coolant temperature indicates the margin to coolant boiling. The saturation temperature of sodium at the depth of the core is greater than 950℃ with the pump pressure off and greater than 1,060℃ with the pumps on. So 950℃ is used as the conservative limit for this parameter[4].

Temperatures are computed by adding relevant temperature rises to the assembly inlet temperature. $2\sigma$ hot channel factors based on CRBR analyses are used in temperature predictions to account for core design, analysis, fabrication and operational uncertainties and variations[5].

## 1.3. Pressure Drop Calculation

Once the flow grouping and the peak pin temperature calculation was performed, the core pressure drop based on the fuel assembly with maximum flow, has to be calculated to see if it does not exceed the limit value. The primary pump of 0.8 MPa pressure head is used in KALIMER design. Thus the assembly bundle pressure drop has to be within 0.32 MPa with 20 % uncertainty, noting that pressure loss through the fuel element bundle and loss through the core contribute 40 and 80 % of the total reactor pressure loss, respectively[6].

Three models developed for the wire wrap rod spacing assemblies are used in bundle pressure loss calculations: (a) Novendstern model, (b) Chiu-Rohsenow-Todreas model, (c) Cheng-Todreas model[7].

## 1.4. Steady State Subchannel Analysis

Detailed core wide coolant temperature profiles are efficiently calculated using the simplified energy equation mixing model and the subchannel analysis method. SLTHEN code is a steady state thermal hydraulic analysis code based on the ENERGY model[8]. The efficiency of the ENERGY model in both computer storage and run time is due to the simplicity of its computational model where only the energy equations are solved, and the momentum and continuity equations are not directly included. The momentum coupling between coolant channels is indirectly taken into account using enhanced eddy diffusivity and the swirl velocity ratio with the experimental modellings developed for the wire-wrap spacing rod assemblies.

SLTHEN has a capability to calculate the multi-assembly whole core calculations. This code provide temperature maps for all pins in all assemblies and thus facilitate core wide failure probability studies.

## 1.5. Detailed Subchannel Analysis

The detailed subchannel analysis will be performed with MATRA-LMR code[9] which is being developed for KALIMER, and SABRE4 code[10] developed in the UK AEAT. MATRA-LMR is the detailed subchannel analysis code developed based on COBRA-IV-I[11] and MATRA[12]. MATRA-LMR solves the mass, axial and transverse momentum, and energy equations. MATRA-LMR is used for a single assembly analysis and will be extended for the multi-assembly whole core calculations. SABRE4 was initially developed for grid spacing assemblies and modified to wire wrap version. It is a three dimensional subchannel code which has good predictions especially in the blocked channel analysis.

It is expected they will form the licensing data basis and will be extensively used during preliminary and final design phases.

In this paper, the preliminary methodology for flow grouping and peak pin temperature calculation used for KALIMER core conceptual design is summarized. Figure 1 shows the overall conceptual design procedure for KALIMER core thermal hydraulics. The major technical results of the conceptual design for the KALIMER 98.03 core is shown and compared with those of KALIMER 97.07 design core. The steady state and detailed subchannel analysis methodology and the results are summarized in the other paper presented at this meeting[13].

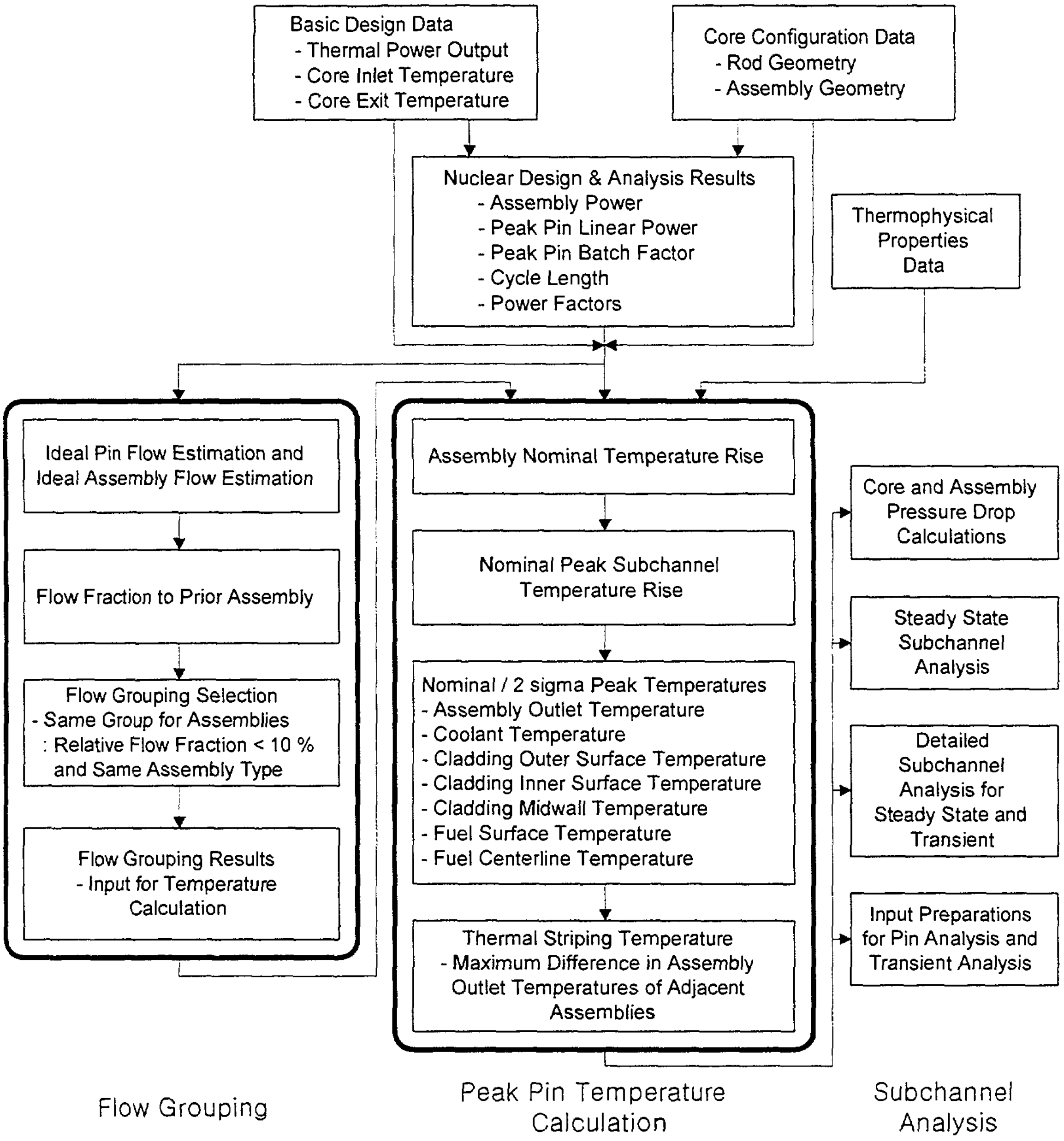


Figure 1. Conceptual Design Procedure for KALIMER Core Thermal Hydraulics

## 2. PRELIMINARY CONCEPTUAL DESIGN METHODOLOGY

### 2.1. Flow Grouping

Coolant flow controls, i.e. orificing flow restrictions, are located at the inlet orificing modules into which the assemblies are inserted at the lower nosepieces end. The flow groups remain in the same core locations for the plant lifetime. Thus it is associated with a core location and not with a particular assembly. Sodium coolant flow has to be supplied to the assemblies based on the peak linear heat generation rate for their whole lifetime to ensure the structural integrity of fuels, claddings and ducts. But small flow control is not expected with this method, so flow groups are limited in number.

During the conceptual design, reflector, shield and IVS were not considered for flow grouping, because their flow rates are small and the powers do not change so significantly throughout their lifetime to complicate their detailed orificing. The orificings for the reflector, shield and IVS were determined based on an average group assembly[4,5].

Total primary loop flow is set by the core power and the desired reactor temperature rise. A small portion of flow is assumed to bypass the core by leaking around the inlet module and assembly nosepieces seal rings, and to be used for structural cooling within the reactor. In general, 1.5% of the primary loop flow is assumed to bypass the core and is thus not included in the flow distribution to the assemblies, thus which is not included in the initial flow grouping.

Every assembly in the range of about 10% power difference was put in the same group. And separate flow groups were maintained for each assembly type, even where flows were virtually identical. This gives the conservative estimations in the number of flow groups, which will be improved at further design changes, considering the non-fuel assemblies. The flow groupings for these regions will be expanded later with better gamma heating calculations into those assemblies and thus increase the total number of flow groups.

## Coolant Temperature Along Pin

Maximum coolant temperature increases along the pin can be calculated from the bundle average temperature rise as follows.

Maximum subchannel coolant temperature rise, $\Delta T_{\max}$, can be expressed as

$$\Delta T_{\max} = f \, \Delta T_B \tag{1}$$

where

f is the peak pin $\Delta T$ peaking factor,
$\Delta T_B$ is the bundle average temperature rise,
$_B$ is for bundle.

And the bundle average temperature rise is calculated as

$$\Delta T_B = \frac{Q_B}{\dot{m}_B \, c_P} \tag{2}$$

where

$Q_B$ is the total heat generation in bundle,
$\dot{m}_B$ is the total mass flow rate in bundle,
$c_P$ is the coolant specific heat.

In the equation (2), total heat generation in bundle $Q_B$ is, from the number of pins in bundle ($n$) and the average heat generation from the fuel pin ($q_{avg}$), $Q_B = n\, q_{avg}$, and the bundle mass flow rate is

$$\dot{m}_B = \rho \, u_B \, A_B \tag{3}$$

where

$\rho$ is the coolant density,
$u_B$ is the bundle average velocity,
$A_B$ is the flow area in bundle.

Substituting the above relation and equation (3) in equation (2), the bundle average temperature rise becomes

$$\Delta T_B = \frac{n \, q_{avg}}{\rho \, u_B \, A_B \, c_P} \tag{4}$$

With equation (4) and (1), the maximum subchannel coolant temperature rise becomes

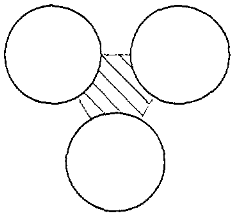
$$\Delta T_{\max} = f\left[\frac{n\, q_{avg}}{\rho\, u_B\, A_B\, c_P}\right] \tag{5}$$

Otherwise, maximum subchannel coolant temperatures that rise along the pin can be also calculated in the hottest channel as

$$\Delta T_{\max} = \frac{q_C}{\dot{m}_C\, c_P} \tag{6}$$

where

$q_C$ is the heat added to the subchannel by adjacent pins,
$\dot{m}_C$ is the subchannel mass flow rate,
$C$ is for subchannel,



and where $q_C$ is from 3 pins as shown in the figure below.

Assuming they are the hottest pins, $q_C$ becomes

$$q_C = 3 \times \frac{q_{\max}}{6} = \frac{q_{\max}}{2}$$

and the bundle mass flow rate $\dot{m}_C$ is

$$\dot{m}_C = \rho\, u_C\, A_C$$

So the equation (6) becomes

$$\Delta T_{\max} = \frac{q_{\max}/2}{\rho\, u_C\, A_C\, c_P} \tag{7}$$

Equalizing equations (5) and (7) with the assumption of uniform density and specific heat of the coolant, the following relation is obtained

$$\Delta T_{\max} = f\left[\frac{n\, q_{avg}}{\rho\, u_B\, A_B\, c_P}\right] = \frac{q_{\max}/2}{\rho\, u_C\, A_C\, c_P} \tag{8}$$

Peak pin $\Delta T$ peaking factor, *f*, can be expressed, from the equation (8), as

$$f = \left(\frac{q_{\max}}{q_{avg}}\right)\left(\frac{u_B}{u_C}\right)\left(\frac{A_B}{2\, n\, A_C}\right) \tag{9}$$

Each term in the right hand side of the equation (9) has the meaning as follows

$f_P = \dfrac{q_{\max}}{q_{avg}}$ is the bundle radial power peaking factor, (10)

$f_v = \dfrac{u_B}{u_C}$ is the bundle radial split factor, (11)

$f_g = \dfrac{A_B}{2\, n\, A_C}$ is the geometry factor. (12)

Using these factors and the bundle average temperature rise, $\Delta T_B$, the maximum coolant temperature rise, $\Delta T_{\max}$, can be calculated. Because temperature is directly related to power, these factors have to be considered in the flow grouping procedure, in modifying the peak pin linear power of each assembly.

## 2.2. Peak Pin Temperature Calculation

Temperatures are computed by adding relevant temperature rises to the assembly inlet temperature. Each temperature rise is computed from energy input, heat capacity, thermal conductivity and heat transfer coefficients. The temperature rises to be included are: (a) the coolant subchannel temperature rise from the inlet to the elevation modeled, (b) the film temperature rise between the subchannel bulk coolant and the cladding surface, (c) the cladding temperature rise, either to the midwall radius or to the inner surface, (d) the gap temperature rise between the cladding and the fuel surface and (e) the fuel temperature rise from the surface to the center[4,5].

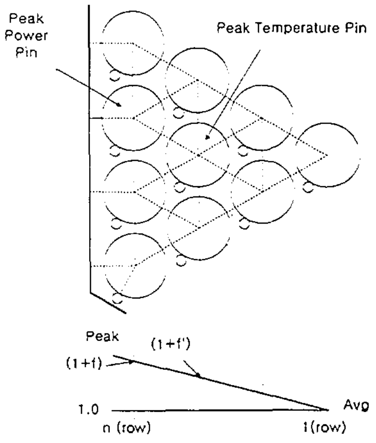
Hot channel factors are introduced in the temperature predictions to account for core design, analysis, fabrication and operational uncertainties and variations. $2\sigma$ uncertainty factors based on CRBR analyses were employed in the KALIMER design.

The temperatures which influence design conditions have to be checked with the limit values because they are important in the following senses. The limit to the nominal peak assembly outlet temperature is set to limit thermal aging effects and the thermal striping potential temperature is set to limit thermal fatigue effects in the UIS. The peak thermal striping potential temperature is a possible maximum difference in the assembly coolant discharge temperatures of adjacent assemblies. The maximum differences possible by combining $+2\sigma$ outlet temperatures adjacent to $-2\sigma$ are used. A limit of 190℃ is applied to this parameter to control high cycle fatigue of the permanent upper internal structures. Mixing between the core and UIS, about 1 meter above the core assembly coolant discharge reduces this temperature difference to an acceptable level for permanent structural components. These become important during transient analysis of duty cycle events and primarily serve to set maximum rates of power and temperature change during plant maneuvering. A considerable damage accumulation margin is reserved in assembly steady state structural analyses to accommodate these transients. The low powers in the control and USS assemblies require the minimum flow. These assemblies have low outlet temperatures. As a result, the control assemblies and their immediate neighbors set the maximum thermal striping potential. The equalized cladding midwall temperature is limited to about 650℃ for fuel and blanket assemblies to ensure the cladding material's structural integrity. The peak fuel surface temperature is limited to 700℃ to avoid liquefaction of a low melting temperature alloy formed by inter diffusion of cladding iron and fuel uranium and plutonium. Liquefaction greatly accelerates cladding internal wastage rates and shortens pin lifetime. The peak fuel centerline temperature indicates the margin to fuel melting.

### Second Row Peak Pin Modification

The peak power pin appears generally in the last row facing the core center in each assembly. But the channel area surrounding that pin is larger than the interior channels, thus it is cooled more than other pins. So the real peak temperature channel is in one row inside. This peak temperature channel factor can be considered by the linear interpolation method, as shown in the figure below.

$$f' = f\left(\frac{n-2}{n-1}\right) \tag{13}$$



linear interpolation

| pins | rows | $\left[1-\frac{1}{n-1}\right]$ |
|---|---|---|
| 271 | 10 | 0.8889 |
| 217 | 9 | 0.8750 |
| 169 | 8 | 0.8571 |
| 127 | 7 | 0.8333 |
| 91 | 6 | 0.8000 |
| 61 | 5 | 0.7500 |
| 37 | 4 | 0.6667 |
| 19 | 3 | 0.5000 |
| 7 | 2 | 0.0000 |

And from the equation (13), the power factor in one row inside can be obtained

$$1+f' = 1+f\left(\frac{n-2}{n-1}\right) = 1+\left(f_p-1\right)\left[1-\frac{1}{n-1}\right] \tag{14}$$

where

$1+f = f_p = q_{\max}/q_{avg}$ is the bundle radial power peaking factor.

The bracket term in the right side of the equation (14) has only the geometry dependency as shown in the table above.

Thus the linear pin power can be modified with these factors as follows

$$q'\big|_{row2} = \left(\frac{1+f'}{1+f}\right) q'\big|_{row1} = \frac{\left[1+(f_p-1)\left\{1-\dfrac{1}{n-1}\right\}\right]}{f_p} q'\big|_{row1} \tag{15}$$

Subchannel $\Delta T_{\max}$ can be modified in the same way. The peak subchannel temperature rise $\Delta T_{\max}$ is expressed in the equation (7). But the real peak temperature channel is in one row inside, so the equation (7) becomes

$$\Delta T_{\max,row2} = \frac{q_{\max}/2}{\rho\, u_C\, A_C\, c_P}\left(\frac{1+f'}{1+f}\right) \tag{16}$$

The above equation (16) can, using the equations (4), become

$$\begin{aligned}\Delta T_{\max,row2} &= \left[\Delta T_B \cdot \frac{\rho\, u_B\, A_B\, c_P}{n\, q_{avg}}\right] \cdot \frac{q_{\max}/2}{\rho\, u_C\, A_C\, c_P}\left(\frac{1+f'}{1+f}\right) \\ &= \Delta T_B \left(\frac{q_{\max}}{q_{avg}}\right)\left(\frac{u_B}{u_C}\right)\left(\frac{A_B}{2\, n\, A_C}\right)\left(\frac{1+f'}{1+f}\right)\end{aligned} \tag{17}$$

The meaning of each term in the right hand side is in the equations (10), (11), (12) and (14).

Thus, the real maximum subchannel coolant temperature rise can be obtained from the equation (17), (14) and the relation $q_{\max}/q_{avg} = f_p = 1+f$,

$$\begin{aligned}\Delta T_{\max,row2} &= \Delta T_B \cdot \left(f_v \cdot f_g\right) \cdot \left(1+f'\right) \\ &= \Delta T_B \cdot FF \cdot \left[1+(f_p-1)\left\{1-\frac{1}{n-1}\right\}\right]\end{aligned} \tag{18}$$

where

$FF = f_v \cdot f_g$ is the bundle radial flow peaking factor.

Each temperature rise is recalculated in order to account for the uncertainties, i.e., design, analysis, fabrication and operational uncertainties and variations. $2\,\sigma$ hot channel factors based on CRBR analyses are used in KALIMER analysis.

The $2\,\sigma$ temperature rises are obtained from the nominal temperature rise, with the corresponding hot channel factors

$$\Delta T_{i,hcf} = \Delta T_i \cdot HCF_i \tag{19}$$

where

$HCF_i$ is the corresponding hot channel factors for i component

## 3. CONCEPTUAL DESIGN CHARACTERISTICS OF KALIMER CORE

KALIMER core thermal hydraulic conceptual design was performed with the above mentioned methodology. In 97.07 design core, there were 16 flow groups and the equalized cladding midwall temperature was 654℃. Some nuclear design optimizations were done to improve the performances[14].

Table 1 and figure 2 show core configuration and the basic design data of KALIMER 98.03 design respectively. The core flow grouping and the peak pin temperature calculation results are shown in table 2 and

Table 1. Basic Design Data of KALIMER 98.03 Design Core

| | |
|---|---|
| Core Thermal Output (MWth) | 392.2 |
| Core Electric Power (MWe) | 150.0 |
| Net Plant Thermal Efficiency (%) | 38.2 |
| Core Inlet/Out Temperature (℃) | 386.2/530.0 |
| Total Flow Rate (kg/s) | 2143 |
| Active Core Height (cm) | 100.0 |
| Core Diameter (cm) | 344.73 |
| Core Configuration | Radial Homogeneous |
| Number of Core Enrichment Zones | 2 |
| Feed Fuel Enrichments (w/o%) (IC/OC) | 14.41/20.00 |
| Fuel Form | U-10%Zr Binary Alloy |
| Refueling Interval (months) | 12 |
| Refueling Batches (Driver/Radial Blanket) | 3/6 |
| Duct Inside Flat to Flat Distance (mm) | 149.8 |
| Pins per Fuel Assembly (Driver/Radial Blanket) | 271/217 |
| Pin Outer Diameter (Driver/Radial Blanket) (mm) | 7.67/12.0 |
| Pin P/D Ratio (Driver/Radial Blanket) | 1.167/1.083 |
| Average/Peak Fuel Burnup (MWD/kg) | 25.35/42.67 |
| Avg/Peak Linear Power for Driver (BOEC) (W/cm) | 160.2/208.5 |
| Peak Fast Neutron Fluence (E>0.1 MeV) ($\times 10^{23}$n/cm$^2$) | 1.399 |
| Cladding Material | HT9 |

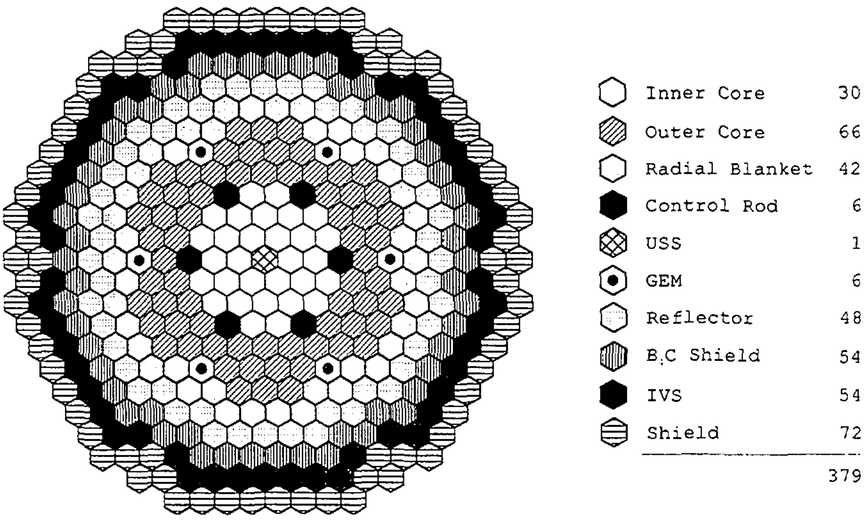


Figure 2. KALIMER 98.03 Design Core Configuration

figure 3 on the 1/6 core configuration map. Table 3 shows an example of the peak pin temperature calculation results for three peak flow assemblies of inner and outer enrichment zones, and radial blankets. Design data and core configuration are shown in table 5 and figure 4 respectively for KALIMER 97.07 design, and the flow grouping and the peak pin temperature calculation results in table 6 and figure 5.

Both cores have the same thermal power of 392 MW (electric power of 150 MW). The core exit temperature is the same for both cores (530℃), but the core inlet temperature is increased from 361.4℃ to

Table 2. Flow Grouping and 2 σ Peak Pin Temperature Calculation Results of KALIMER 98.03 Design Core

| Flow Group | Assy Type | Assy Count | Assy Flow (kg/s) | Zone Flow (%) | Assy Outlet (℃) | Thermal Striping (℃) | Cladding Midwall (℃) | Fuel Surface (℃) | Fuel Center (℃) |
|---|---|---|---|---|---|---|---|---|---|
| *limit* | | | | | *590* | *190* | *650* | *700* | *1000* |
| 1 | IC | 6 | 21.6 | | 575 | 160 | 628 | 635 | 680 |
| 2 | IC | 24 | 20.4 | | 578 | 158 | 628 | 635 | 674 |
| 3 | OC | 24 | 23.7 | | 571 | 148 | 628 | 636 | 691 |
| 4 | OC | 12 | 20.6 | | 561 | 31 | 628 | 635 | 675 |
| 5 | OC | 12 | 19.2 | | 556 | 78 | 628 | 634 | 667 |
| 6 | OC | 18 | 15.8 | 90 | 554 | 79 | 628 | 633 | 650 |
| 7 | RB | 12 | 4.6 | | 508 | 78 | 628 | 632 | 642 |
| 8 | RB | 18 | 3.3 | | 501 | 79 | 628 | 631 | 641 |
| 9 | RB | 12 | 2.7 | 7 | 498 | 71 | 628 | 631 | 641 |
| 10 | CTL | 6 | 3.1 | | 397 | 158 | - | - | - |
| 11 | USS | 1 | 3.1 | 1 | 393 | 160 | - | - | - |
| Total primary loop flow : 2143 kg/s<br>Total bypass flow : 2.0 % | | | | | | | | | |

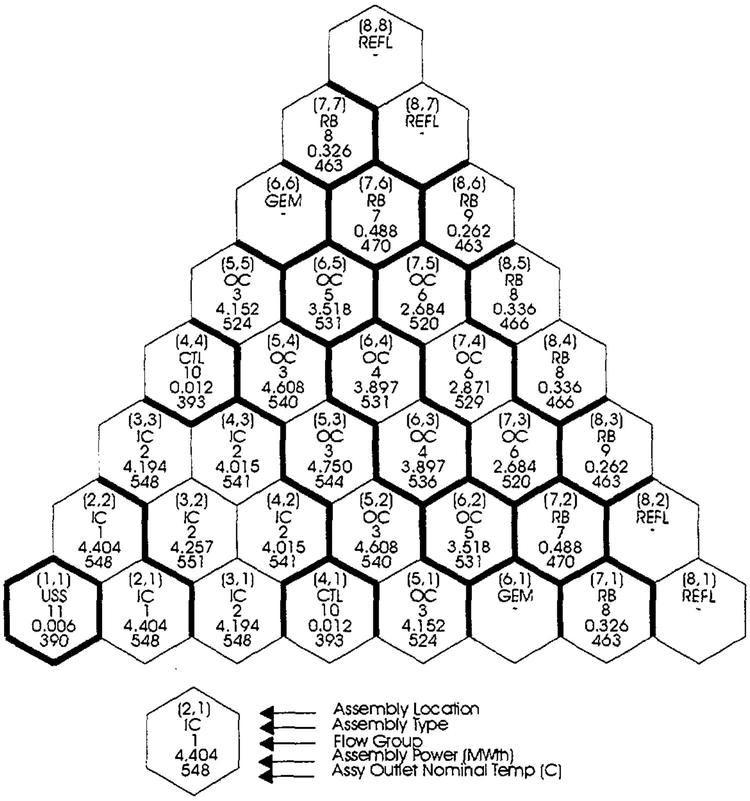


Figure 3. Flow Grouping and Temperature Calculations of KALIMER 98.03 Design Core (1/6 Core Map)

386.2℃ (the average temperature rise in the core is decreased from 168.6℃ to 143.8℃). The pin's outer radius is increased also from 7.4mm to 7.67mm and the fuel enrichment in the outer core is decreased from 15.0% to 14.4%, based on the nuclear and thermal hydraulic design optimization study[14].

Table 3. Temperature Calculation Summary Table of 98.03 Core (Example)

| ASSEMBLY & GROUP MODEL DATA | Assy 1 (IC) | | Assy 11 (OC) | | Assy 11 (RB) | |
|---|---|---|---|---|---|---|
| Type & Location | F | (2,1) | F | (5,3) | RB | (7,2) |
| Orifice Group & Flow Rate (kg/s) | 1 | 21.6 | 3 | 23.7 | 7 | 4.6 |
| Inlet Temp. (℃) & Avg. Cp (J/kg-℃) | 386.2 | 1267.8 | 386.2 | 1267.8 | 386.2 | 1267.8 |
| Film & Gap Coefficients (kW/$m^2$-℃) | 132.5 | 132.5 | 132.5 | 132.5 | 88.3 | 132.5 |
| Pin OD & ID (mm) | 7.67 | 6.61 | 7.67 | 6.61 | 12.0 | 10.9 |
| Clad Thickness & Pellet OD (mm) | 0.53 | 5.8 | 0.53 | 5.8 | 0.54 | 10.1 |
| Adjustment For 2nd Row Pin Power | 0.8889 | – | 0.8889 | | 0.8333 | |
| Power Factor: Peak Clad, Peak Fuel | 0.41 | 0.92 | 0.41 | 0.92 | 0.35 | 0.73 |
| Temperature Factor: Peak Clad, Peak Fuel | 1 | 0.71 | 1 | 0.71 | 1 | 0.85 |
| | BOL | EOL | BOL | EOL | BOL | EOL |
| Peak Pin W/cm, BOL & EOL | 189.5 | 186.7 | 207.8 | 201.7 | 69.6 | 80.3 |
| Assembly Total Power (MWth) | 4.404 | 4.377 | 4.750 | 4.654 | 0.488 | 0.581 |
| Bundle Radial Power Peaking Factor | 1.042 | 1.040 | 1.063 | 1.062 | 1.520 | 1.467 |
| Bundle Radial Flow Peaking Factor | 1.13 | 1.13 | 1.13 | 1.13 | 1.21 | 1.21 |
| Fuel Porosity Factor | 0.5 | 0.7 | 0.5 | 0.7 | 0.5 | 0.7 |
| ASSY OUTLET TEMPERATURE | | | | | | |
| Assy T Rise Based On Core Cp (℃) | 143.6 | 142.6 | 140.2 | 137.0 | 66.2 | 82.2 |
| Assy Avg T Based On Core Cp (℃) | 466.9 | 466.4 | 465.2 | 463.6 | 428.2 | 436.2 |
| Cp (J/kg-℃) | 1266.5 | 1266.6 | 1266.7 | 1267.0 | 1272.9 | 1271.5 |
| Assy Nominal T Rise (℃) | 143.7 | 142.7 | 140.3 | 137.1 | 65.9 | 81.9 |
| Nominal Outlet Temp (℃) | 547.7 | 546.7 | 544.3 | 541.1 | 469.8 | 485.9 |
| Nominal Peak Subchannel T (℃) | 574.5 | 574.0 | 574.8 | 570.9 | 531.2 | 553.7 |
| Nominal Peak Clad Midwall T (℃) | 581.0 | 579.5 | 580.9 | 576.8 | 532.5 | 555.2 |
| Nominal Peak Fuel Surface T (℃) | 587.0 | 585.3 | 587.4 | 583.2 | 533.6 | 556.5 |
| Nominal Peak Fuel Centerline T (℃) | 637.6 | 610.2 | 647.5 | 614.9 | 541.3 | 556.3 |
| +2$\sigma$ ASSEMBLY PEAK T (℃) | | | | | | |
| +2$\sigma$ Coolant Outlet | 574.8 | | 570.9 | | 507.6 | |
| +2$\sigma$ Thermal Striping | 160.0 | | 26.9 | | 78.0 | |
| +2$\sigma$ Cladding Midwall | 628.5 | | 628.6 | | 628.1 | |
| +2$\sigma$ Fuel Surface | 635.1 | | 635.8 | | 629.5 | |
| +2$\sigma$ Fuel Centerline | 680.4 | | 691.2 | | 631.6 | |
| +2$\sigma$ Peak Fuel Surface At Scram | 667.1 | | 667.9 | | 660.8 | |
| +2$\sigma$ Peak Fuel Center At Scram | 718.3 | | 730.5 | | 663.2 | |

Table 4. Comparisons of the Calculation Results of KALIMER 98.03 Core

| Temperature (℃) / Code | Peak Subchannel | | | Bundle Average | | |
|---|---|---|---|---|---|---|
| | IC | OC | RB | IC | OC | RB |
| ORFCE | 574.5 | 574.8 | 531.2 | 547.7 | 544.3 | 469.8 |
| SLTHEN | 548.2 | 544.6 | 481.9 | 530.0 | 527.0 | 458.9 |
| SABRE4 | 565.3 | 561.4 | 485.1 | 533.0 | 530.0 | 462.5 |
| MATRA-LMR | 556.2 | 552.5 | 482.7 | 533.0 | 530.0 | 462.5 |

Table 5. Basic Design Data of KALIMER 97.07 Design Core

| | |
|---|---|
| Core Thermal Output (MWth) | 392.0 |
| Core Electric Power (MWe) | 150.0 |
| Net Plant Thermal Efficiency (%) | 38.3 |
| Core Inlet/Out Temperature (℃) | 361.4/530.0 |
| Total Flow Rate (kg/s) | 1824 |
| Active Core Height (cm) | 100.0 |
| Core Diameter (cm) | 344.3 |
| Core Configuration | Radial Homogeneous |
| Number of Core Enrichment Zones | 2 |
| Feed Fuel Enrichments (w/o%) (IC/OC) | 15.0/20.0 |
| Fuel Form | U-10%Zr Binary Alloy |
| Refueling Interval (months) | 12 |
| Refueling Batches (Driver/Radial Blanket) | 3/3 |
| Duct Inside Flat to Flat Distance (mm) | 149.0 |
| Pins per Fuel Assembly (Driver/Radial Blanket) | 271/217 |
| Pin Outer Diameter (Driver/Radial Blanket) (mm) | 7.4/12.0 |
| Pin P/D Ratio (Driver/Radial Blanket) | 1.189/1.083 |
| Average/Peak Fuel Burnup (MWD/kg) | 28.0/47.3 |
| Avg/Peak Linear Power for Driver (BOEC) (W/cm) | 151.9/234.6 |
| Peak Fast Neutron Fluence (E>0.1 MeV) ($\times 10^{23}$n/cm$^2$) | 1.434 |
| Cladding Material | HT9 |

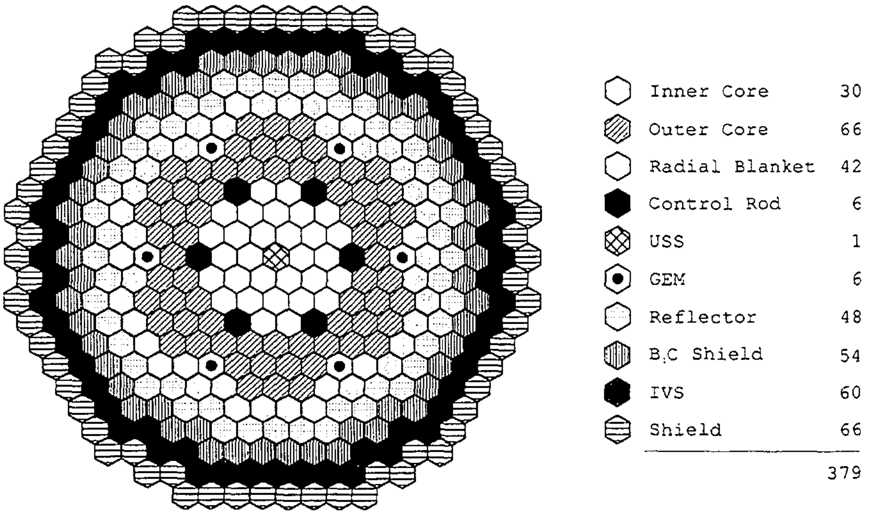


Figure 4. KALIMER 97.07 Design Core Configuration

Eleven flow groups were specified, as shown in table 2, for the present KALIMER 98.03 design core: six groups are used for fuel assemblies, the radial blanket assemblies utilize three flow groups and the six control assemblies and the USS assembly use a single flow group each. Both the control and USS assemblies have sliding rod bundles and their flow grouping is subject to a minimum flow that maintains turbulent flow cooling during a scram drop. These assemblies strongly affect thermal striping potential temperatures during conceptual design. The minimum flow was assumed to be 3.15 kg/s based on ALMR design analyses[3]. One flow group for GEMs are subject to the high pressure coolant plenum with no orificing restriction and serves as the sodium inlet and outlet connection which activates the GEM reactivity feedback upon loss of pumped flow. With the 1.5 % bypass flow, an additional 0.5% of primary flow was omitted in the conceptual design study and is reserved for the perimeter assemblies not explicitly being orificed. Thus 2.0% of primary flow is not distributed in the initial flow grouping analyses.

Table 6. Flow Grouping and 2 $\sigma$ Peak Pin Temperature Calculation Results of KALIMER 97.07 Design Core

| Flow Group | Assy Type | Assy Count | Assy Flow (kg/s) | Zone Flow (%) | Assy Outlet (℃) | Thermal Striping (℃) | Cladding Midwall (℃) | Fuel Surface (℃) | Fuel Center (℃) |
|---|---|---|---|---|---|---|---|---|---|
| *Limit* | | | | | *590* | *190* | *650* | *700* | *1000* |
| 1 | IC | 6 | 18.4 | | 593 | 190 | 654 | 663 | 728 |
| 2 | IC | 6 | 17.8 | | 592 | 33 | 654 | 663 | 718 |
| 3 | IC | 6 | 17.3 | | 591 | 172 | 654 | 662 | 713 |
| 4 | IC | 12 | 16.4 | | 587 | 169 | 654 | 661 | 704 |
| 5 | OC | 18 | 26.2 | | 538 | 116 | 654 | 663 | 734 |
| 6 | OC | 6 | 17.4 | | 584 | 166 | 654 | 663 | 733 |
| 7 | OC | 12 | 16.1 | | 579 | 62 | 654 | 661 | 711 |
| 8 | OC | 12 | 14.6 | | 577 | 98 | 654 | 661 | 701 |
| 9 | OC | 6 | 12.5 | | 561 | 101 | 654 | 660 | 676 |
| 10 | OC | 12 | 11.8 | 92 | 557 | 100 | 654 | 660 | 673 |
| 11 | RB | 12 | 2.0 | | 540 | 98 | 654 | 657 | 666 |
| 12 | RB | 6 | 2.4 | | 500 | 57 | 654 | 657 | 667 |
| 13 | RB | 12 | 2.3 | | 493 | 101 | 654 | 657 | 666 |
| 14 | RB | 12 | 2.1 | 5 | 487 | 100 | 654 | 657 | 665 |
| 15 | CTL | 6 | 3.1 | | 409 | 172 | - | - | - |
| 16 | USS | 1 | 3.1 | 1 | 385 | 190 | - | - | - |
| Total primary loop flow : 1824 kg/s<br>Total bypass flow : 2.0 % | | | | | | | | | |

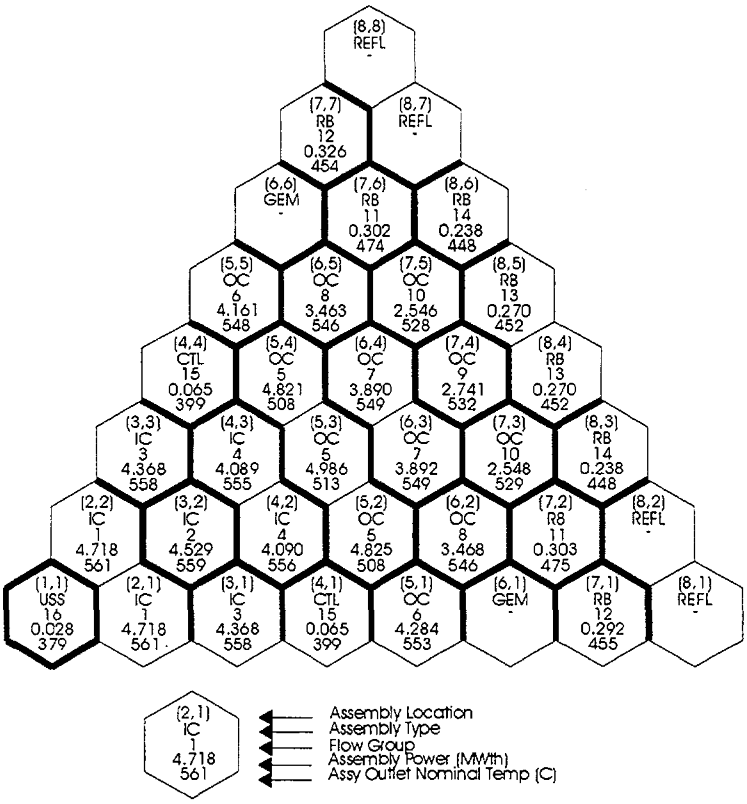


Figure 5. Flow Grouping and Temperature Calculations of KALIMER 97.07 Design Core (1/6 Core Map)

The key peak temperatures are calculated for the peak power pin of each flow group and compared to the appropriate limits in table 2 and 6. The criteria used for steady state thermal performance are the same as those for design basis events. As shown in table 2, all the criteria are met with a margin, not in the case of table 6. In the KALIMER 98.03 design core, given the core power distribution and total flow rate, the equalized cladding midwall temperature is about 628℃ for fuel and blanket assemblies which is below the limit temperature of about 650℃. The maximum +2$\sigma$ cladding temperature is limited to ensure adequate fuel lifetime. Table 3 lists some of the detailed peak pin temperature calculation results for the peak flow assemblies of inner and outer enrichment zones, and radial blankets. It shows an example of the calculation results for each assembly.

Table 4 shows the comparison for KALIMER 98.03 core design results of this preliminary calculation with SLTHEN, MATRA-LMR and SABRE4 code calculations. It is done for the nominal subchannel peak and bundle average temperatures in the peak flow assemblies of inner and outer enrichment zones, and radial blankets. Table 4 shows that ORFCE calculation can be used in the preliminary conceptual design with a sufficient margin.

In summary, the number of flow groups are reduced from 16 to 11, the equalized peak cladding midwall temperature from 654℃ to 628℃ for the KALIMER 98.03 design core. It was achieved through a core design optimization study, i.e., power flattening and radial blanket power fraction increase. The peak assembly flow rate is decreased from 26.2 kg/s to 23.7 kg/s, and thus the average flow velocity from 4.26 m/s to 4.05 m/s and the maximum bundle pressure drop from 0.21 MPa to 0.18 MPa. The KALIMER 98.03 design core is proved to be more optimized compared to the 97.07 design core.

## 4. CONCLUSION

A Methodology for thermal hydraulic conceptual design and performance analysis of KALIMER core is summarized, especially the preliminary methodology for flow grouping and peak pin temperature calculation in detail. For its preliminary steady state thermal hydraulic performance analysis, ORFCE-F60 and ORFCE-T60 modules are used for the flow grouping and the peak pin temperature calculation respectively. The major technical results of its preliminary conceptual design for KALIMER 98.03 design core are shown and compared with the results of KALIMER 97.07 design core. The number of flow groups are reduced from 16 to 11, as is the equalized peak cladding midwall temperature from 654℃ to 628℃ through a core design optimization study. The comparison for the KALIMER 98.03 core design results of this preliminary calculation with SLTHEN, MATRA-LMR and SABRE4 code calculations shows that ORFCE modules can be used in the preliminary conceptual design with a sufficient margin.

At later design stages, the basic subchannel analysis will be performed with the SLTHEN code, a steady state thermal hydraulic analysis code based on the ENERGY model, and the detailed subchannel analysis will be done with the MATRA-LMR code which is being developed based on COBRA-IV-I and MATRA. MATRA-LMR is now used for a single assembly analysis but will be extended for the multi-assembly whole core calculations. And SABRE4 code, a three dimensional subchannel code developed in the UK AEA, will be used as benchmark calculations. These three codes provide temperature maps for all pins in all assemblies and thus facilitate core-wide failure probability studies. It is expected they will form the licensing data basis and will be extensively used during preliminary and final design phases.

## REFERENCES

[1] PARK, C.K., "Development of Korea Advanced Liquid Metal Reactor", IWGFR (1998).
[2] YOKOO, T., et al., Design Study on Metal Fuel FBR Cores, T91013, CRIEPI (1991).
[3] HAHN, D.H., et al., KAERI Concept Study, GE (1995).
[4] PARK, C.K., et al., KALIMER Design Concept Report, KAERI/TR-888/97 (1997).
[5] KIM, Y.G., et al., "Thermal Hydraulic Analysis of KALIMER 98.03 Design Core", Proceedings of KNS Spring Meeting (1998).
[6] TANG, Y.S., et al., Thermal Analysis of Liquid-Metal Faster Breeder Reactors, ANS (1978).
[7] WALTER, A.E., REYNOLDS, A.B., Fast Breeder Reactors, Pergamon Press, New York (1981).
[8] YANG, W.S., "An LMR Core Thermal Hydraulics Code Based on the ENERGY Model", Journal of KNS, Vol.29, No.5, pp.406-416 (1997).

[9] KIM, W.S., et al., "Code Development for LMR Core Thermal Hydraulic Subchannel Analysis", Proceedings of KNS Spring Meeting (1998).

[10] MacDOUGALL, J.D., LILLINGTON, J.N., "The SABRE Code for Fuel Rod Cluster Thermohydraulics", Nucl. Eng. and Des., V. 82, pp. 171-190 (1984).

[11] WHEELER, C.L., et al., COBRA-IV-I : An Interim Version of COBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores, BNWL-1662 (1976).

[12] YOO, Y.J., HWANG, D.H., "Development of Subchannel Analysis Code MATRArm alpha -version", Proceedings of KNS Autumn Meeting (1997).

[13] KIM, W.S., KIM, Y.G., "Recent Development of Subchannel Analysis Code MATRA-LMR for KALIMER Subassembly Thermal-Hydraulics", IAEA TCM, 27-31 July 1998, Obninsk, Russia (1998).

[14] KIM, Y.I., et al., "KALIMER Core Design and Analysis", Proceedings of KNS Autumn Meeting (1997).