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Stochastic Safety Analysis of Natural Circulation Decay Heat Removal in Liquid Metal Reactor

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Abstract –*Safety analyses for nuclear reactors involve inherently various uncertainties. The uncertainties come from the input parameters, physical modeling, and numerical methods employed. The output from the stochastic analysis is used to obtain the tolerance limit and the confidence level of the computational results, and to identify the sensitive input parameters to the output. In the liquid metal reactor project, it is necessary to develop a safety analysis methodology based on the best estimate thermal-hydraulic code and the most probable assumptions and input parameters with uncertainty consideration. In the present study, a stochastic safety analysis of the natural circulation decay heat removal of the LMR is performed. It has been found that the maximum core coolant temperature follows normal distribution function and no outlying behavior is observed in the transient course of the response. It is due to the fact that the thermal-hydraulic features of the LMR are single phase flow. Therefore, no difficulty is foreseen in using the stochastic method in the safety design and regulation of the LMR. The importance of the individual input parameters from the viewpoint of uncertainty has been estimated using the variance of the conditional expectation and Pearson's correlation ratio. It is noted that the input parameters are sometimes dependent one another. A methodology has been developed for evaluating the dependencies among input parameters. The proposed methodology is effective for performing the stochastic safety analysis of the LMR and for understanding the characteristics of the uncertainty.*

I. INTRODUCTION

In the past, the natural circulation DHR was studied extensively in the world-wide LMR society. One of the authors has analyzed the natural circulation test performed at Japanese experimental fast reactor Joyo [1], [2]. He used a plant dynamics computer code and good agreement was obtained between the computation and the experimental measurement. Furthermore, the important thermal-hydraulic phenomena were identified through the intensive investigation.

The plant model and the input data used in the natural circulation test analysis are determined as the most probable values based on the individual separate tests and our knowledge and experience in the LMR whole plant analysis. On the other hand, analyses for licensing are generally based on conservative assumptions and data so that the system responses never exceed the safety limitation with appropriate safety margins.

Here, it should be emphasized that so-called the best estimate thermal-hydraulics codes are widely used for the safety analyses of the light water reactors. Modro et al. [3] mentioned that there existed significant potential for an increase in plant efficiency and economy if the best estimate safety analyses could be conducted with confidence and with quantified uncertainty of results. Furthermore, he added that conservative analyses might result in misleading sequences, unrealistic time scales, and some phenomena may be missed.

The United States Nuclear Regulatory Commission (USNRC) has revised the emergency core cooling system licensing rules to allow the use of the best estimate computer codes [4]. It requires explicit quantitative assessment of the uncertainties of the thermal-hydraulic calculations in the licensing and regulatory processes. The Code Scaling, Applicability and Uncertainty (CSAU) methodology was developed that provided a systematic approach for the investigation of uncertainty of safety related parameters [5].

In the feasibility study of the commercialization of the fast reactor cycle system in Japan, a Liquid Metal Reactor (LMR) has been selected as the most promising reactor design. [6] It is designated as the Japanese Sodium-cooled Fast Reactor (JSFR). The JSFR is a large-scale LMR having 1,500MWe power output with MOX fuel and the passive reactor shutdown system and the natural circulation decay heat removal (DHR) is a key feature of its safety design characteristics. A probabilistic safety analysis of the JSFR was performed and the results are published by Kurisaka.[7] According to the results, a loss-of-heat-sink accident (LOHS) was found to be one of the most risk-dominant sequences and the natural circulation DHR was a significant risk reduction measure. From this viewpoint, it is obvious that the natural circulation DHR following the reactor protection actuation is of importance in the safety design and analysis of the JSFR.

The present study is motivated by the necessity of the best estimate analysis with uncertainty consideration by an appropriate methodology in the safety analysis of the natural circulation DHR in the LMR. The authors applied the stochastic safety analysis method proposed by Mackay [8] to the LMR natural circulation DHR analysis. Latin Hypercube Method [9] is used for input parameter sampling. The importance of uncertainty of input parameters is evaluated by Pearson's correlation ratio. Further, a computational procedure is proposed to quantify the propagation of input parameter uncertainty when correlations among input parameters exist.

II. NATURAL CIRCULATION DECAY HEAT REMOVAL IN LMR

II.A. LMR System

In a LMR system, liquid sodium is used as coolant. The boiling point of the sodium is over 1000K at the ambient pressure. Therefore, the LMR is a single phase of liquid sodium heat transport system. It is a different situation from the light water reactor cases that the single-phase thermal-hydraulics is the major concern in the LMR safety design.

In the natural circulation phenomena, code output physical quantities are temperatures of the reactor core and the primary coolant boundary. The temperatures are determined based on the balance of the decay heat in the reactor core and the primary coolant mass flow.

Initiated by a transient, the reactor control rod insertion and reactor cooling pump trip follows. Although the reactor power decreases instantly by the cessation of the nuclear fission, the coolant flow decreases gradually because of the inertia of the centrifugal pump. Therefore, the coolant temperature in the reactor core decreases first by the initial over-cooling. The temperature increases subsequently due to the coolant mass flow reaches the minimum value. Afterwards, the temperature rise results in the development of a buoyancy force and the natural circulation flow develops. At this stage, the core coolant temperature descends and the plant arrives at an almost steady-state thermal-hydraulic condition.

Figure 1 shows the schematic of the LMR. The reactor system consists of the reactor vessel, the primary heat transport system, the secondary heat transport system and the DHR system (air cooler). The primary and the secondary heat transport systems are connected by the intermediate heat exchanger (IHX). In the secondary system, the steam generator and the air cooler are placed in parallel and the coolant flow path is switched from the steam generator to the air cooler after the reactor protection system is actuated. During the natural circulation DHR, the decay heat is transferred from the reactor core to the air cooler via the primary system, the IHX and the secondary system.

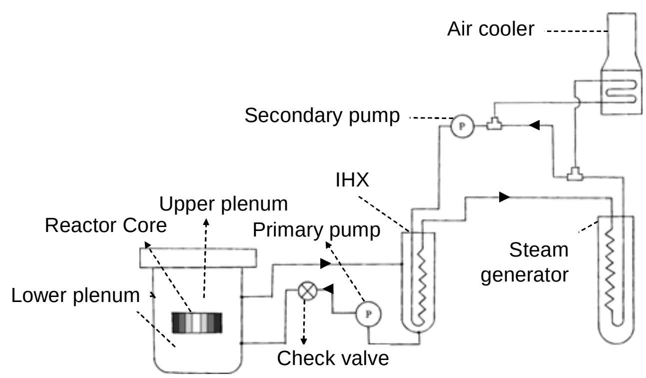

Fig. 1 Schematic of the LMR system

II.B. Preliminary Sensitivity Analysis

Here, a plant dynamics code developed by Kimura et al. [11] for an LMR is used and the sensitivity of input parameters is preliminarily investigated. Table 1 shows the results of sensitivity analyses. It is easily understood that a large number of parameters are related to the safety analysis. On the other hands, there is little information regarding the coefficient of variance (COV) of the parameters. The COV is defined as the standard deviation divided by the mean value for each parameter. They are determined as follows based on the engineering judgment in the present study.

First of all, the COV of the pressure loss coefficient is assumed to be 30%. In general, the pressure loss of the sodium components is evaluated according to water experiments with a scaled or mock-up model. The accuracy is dependent on the measurement error of the differential pressure and the flow velocity. In the natural circulation low-flow conditions, the error in the flow velocity measurement is relatively large comparing with the rated flow condition. Hence 30% of COV is assumed for the pressure losses. The COV of the flow coast down time constant is assumed to be 20 %. It is based on the author's experience in the natural circulation test analysis. The COV of the reactor decay heat is assumed to be 10 %. The error of the decay heat is well-investigated and the appropriate value is recommended (AESJ, 1989). The COV of the decay heat is thus determined based on the safety margins used in the safety analysis. For other parameters, 10% of COV is tentatively assumed.

The computational results are compared with the design limitation values to obtain the tolerance limit and the confidence level of the computational output. From the safety viewpoint, two quantities are used as the design limitation values. One is the cladding temperature in the reactor core fuel pin to prevent the fuel pin failure in the natural circulation DHR condition. The coolant temperature in the reactor core is used for this criterion. The other is the structural temperature of the primary coolant boundary so that the strength of the primary coolant boundary is not deteriorated. In the primary heat transport system except the reactor core, the coolant temperature is highest at the reactor upper plenum. Thus the upper plenum coolant temperature is to be used for the criterion regarding the coolant boundary integrity.

In Table 1, the increase of temperatures is shown when the input parameter is changed by the amount corresponding to one standard deviation. It is seen from this table that the pressure losses in the reactor core, primary coolant pump, and the check valve are sensitive to the core coolant temperature. Also, the reactor decay heat and the primary flow coast down time constant are influential input parameters as well. Concerning the upper plenum temperature, the pressure losses of the primary components and the reactor decay heat are sensitive in the same way as the core coolant temperature. In addition, the parameters for the air coolers in the DHR system are found to be sensitive. It is easily understood that the upper plenum temperature is governed by the long-term system behavior and the DHR system is related to the performance. On the other hand, the reactor core temperature is based on the short-term characteristics and the secondary system is not concerned.

II.C. Latin Hypercube Sampling

In the following discussion, the authors consider only the core coolant temperature for simplicity. Since the upper plenum temperature is to be important in case that the heat sink is lost and the system temperature monotonically increases. For the stochastic analysis of the natural circulation DHR, five parameters sensitive to the core coolant temperature are selected as the stochastic variables. Those are listed in Table 2. It is not due to the drawback of the proposed methodology such as the

TABLE I Sensitivity Analysis of natural circulation DHR

Input parameter	COV (%)	Core temperature change (K)	Upper plenum temperature change (K)
Reactor core pressure loss	30	1.7	0.3
Primary pump pressure loss	30	4.7	1.9
Secondary pump pressure loss	30	0.0	0.0
IHX pressure loss	30	0.7	0.1
Check valve pressure loss	30	8.1	3.3
Core decay heat	10	13.9	20.0
Gap conductance of fuel	10	0.0	0.0
ACS air flow rate	10	0.0	5.6
ACS air inlet temperature	10	0.0	1.3
ACS heat transfer coefficient	10	0.0	4.0
IHX heat transfer coefficient (primary)	10	0.0	0.1
IHX heat transfer coefficient (secondary)	10	0.0	0.1
Primary flow coast down time constant	20	2.8	0.0
Secondary flow coast down time constant	20	0.4	0.0

limitation of the number of random variables, but the authors want to make the discussion clearly understandable. The probability density function (pdf) of the input parameters is to be determined based on the statistical characteristics. Here the normal distribution is assumed for all the parameters.

TABLE II Statistical Properties of Input Parameters

Input Parameter	Pdf	COV
Pressure loss coefficient in the reactor core	Normal	30 (%)
Pressure loss coefficient in the primary pump	Normal	30 (%)
Pressure loss coefficient in the check valve	Normal	30 (%)
Decay heat in the reactor core	Normal	10 (%)
Time constant of the primary pump flow coast down	Normal	20 (%)

Random samples for five parameters selected as in TABLE II are generated according to the Latin Hypercube Sampling method [10]. Fifty samples are used for the design matrix of the stochastic numerical experiment. Therefore, the design matrix size becomes 50 times 5. Then, the combination of random samples for individual parameters is exchanged randomly in the design matrix. By this process, replicas of the design matrix are generated. In the present computations, ten replicas are used.

III. STOCHASTIC ANALYSIS AND IMPORTANCE INDEX OF INPUT PARAMETERS

III.A. Variance of Conditional Expectation

If input parameters are random variables, the uncertainties propagate through the code calculation and the code output is also a random quantity. Let the input variables vector $\mathbf{x}$ and the code output y, the computer code model is described as follows:

$$y = M(\mathbf{x}) \qquad (1)$$

The variance of the code output can be divided into two portions as:

$$V[y] = V\left[E\left(y \mid \mathbf{x}_s\right)\right] + E\left(V\left[y \mid \mathbf{x}_s\right]\right) \qquad (2)$$

where $V[\cdot]$ and $E(\cdot)$ are variance and expected value, respectively. $E\left(y \mid \mathbf{x}_s\right)$ and $V\left[y \mid \mathbf{x}_s\right]$ are expectation and variance on condition that a subset $\mathbf{x}_s$ of the input vector is fixed. Here the input variable vector is expresses as the union of $\mathbf{x}_s$ and the complementary subset $\mathbf{x}_{\bar{s}}$ as:

$$\mathbf{x} = \mathbf{x}_s \cup \mathbf{x}_{\bar{s}} \qquad (3)$$

The first term in the right hand side of Eq. (2) is the Variance of Conditional Expectation (VCE). The VCE expresses the magnitude of the correlation of the input parameter $\mathbf{x}_s$ and the code output y.

$$V\left[E\left(y \mid \mathbf{x}_s\right)\right] = \int \left(E\left(y \mid \mathbf{x}_s\right) - E(y)\right)^2 f_{\mathbf{x}_s}\left(\mathbf{x}_s\right) d\mathbf{x}_s \qquad (4)$$

where $f(\cdot)$ denotes the pdf of the variable. The second term in the right hand side of Eq. (2) is the within-group variance or the residual:

$$E\left(V\left[y \mid \mathbf{x}_s\right]\right) = \iint \left(y - E\left(y \mid \mathbf{x}_s\right)\right)^2 f_{y|\mathbf{x}_s}(y) f_{\mathbf{x}_s}\left(\mathbf{x}_s\right) dy d\mathbf{x}_s \qquad (5)$$

It reflects the variation of the individual data within the group from the group-mean value and has nothing to do with $\mathbf{x}_s$. Therefore, one considers Eq. (5) is the residual with respect to the uncertainty of $\mathbf{x}_s$.

The relative importance of the uncertainty of the input parameter with respect to the output uncertainty is defined by the Pearson's correlation ratio as follows:

$$\eta_{\mathbf{x}_s}{}^2 = \frac{V\left[E\left(y|\mathbf{x}_s\right)E\right]}{V[y]} \qquad (6)$$

III.B. Natural Circulation DHR Analysis

Natural circulation DHR analysis is performed with stochastic input parameters given in Table 2. The temperature transient is shown in Figure 2. As mentioned above, the core coolant temperature firstly decreases by 150K in a few seconds. Subsequently, it increases over the initial temperature. The peak temperature is observed at 40-50 second when the coolant mass flow rate becomes the minimum value. From Fig 2, the maximum temperature ranges from 550°C to 600°C.

The cumulative probability distribution function of the maximum temperature is evaluated and the result is shown in Fig. 3. The normal cumulative function is also shown for comparison in the figure. It is seen that the function of the maximum temperature is in accordance with the normal distribution. The natural circulation of the LMR is a single-phase thermal-hydraulic phenomenon and the maximum core coolant temperature follows the normal distribution. In other words, no discontinuity or outlying behavior of the system response is observed within the present variation range of input parameters. Hence it can be said that the stochastic method is useful for the DHR issue of the LMR safety analysis.

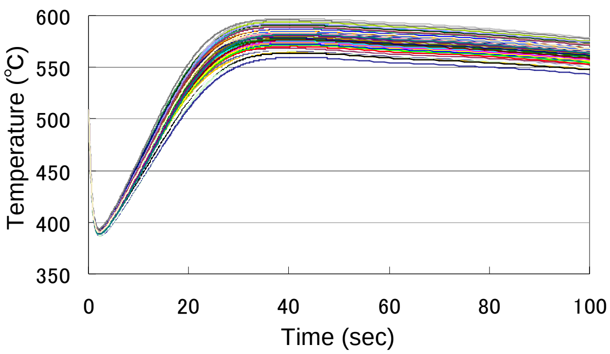

Fig. 2. Transient courses of the core coolant temperature during the natural circulation DHR.

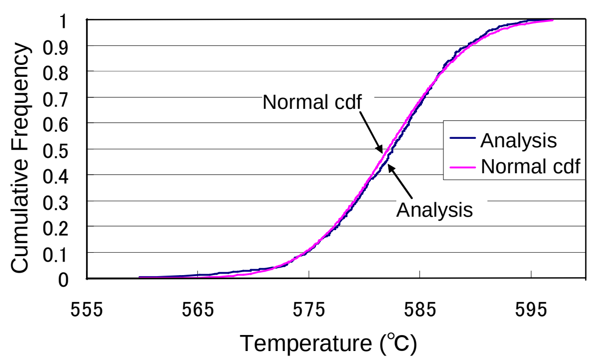

Fig. 3. Cumulative distribution function of the maximum coolant temperature. Cumulative normal distribution function is shown for comparison.

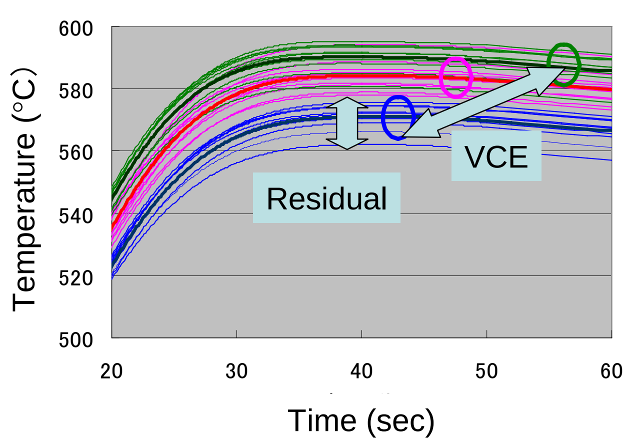

Fig. 4. Close-up of the transient courses of the core coolant temperature. Concepts of the VCE and the residual are illustrated.

Figure 4 shows the close-up view of Fig. 2 from 20 second to 60 second. It conceptually explains the VCE and the residual defined in Eq. (4) and Eq. (5), respectively. In the figure, green lines are the computational results when the reactor decay heat takes the largest value. On the other hand, the blue lines correspond to the results with the minimum values for the decay heat. The red lines assume intermediate values for the decay heat. Other input parameters are sampled randomly from its own pdf. In each group, the mean curve within the group is shown by the thick line. The VCE express the variance among the thick lines. In other words, the VCE means the scattering of the expected curves (the thick lines in Fig. 4) of the coolant temperature on condition that the decay heat is fixed, i.e. $V\left[E\left(y \mid \mathbf{x}_S\right)\right]$. In this case, y is the core coolant temperature and $\mathbf{x}_S$ is the reactor decay heat. On the other hand, the scattering of the lines of the same color means the variance of the code output when the decay heat level is fixed at a level, $V\left[y \mid \mathbf{x}_S\right]$. Therefore, the expected value of the variance with respect to the decay heat level is $E\left(V\left[y \mid \mathbf{x}_S\right]\right)$ that is the residual defined in Eq. (4). It expresses the variance of the computational results that is not related to the selected input parameters $\mathbf{x}_S$.

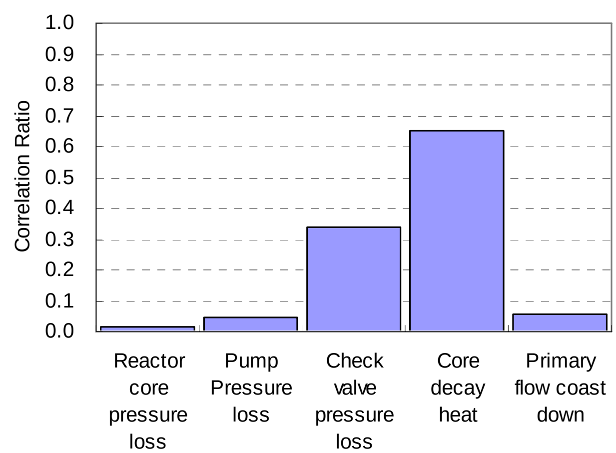

Fig. 5. Correlation ratio for the natural circulation DHR.

The correlation ratio is shown in Fig. 5. If the correlation ratio is unity, the variance of the code output $V[y]$ is equal to the VCE $V\left[E\left(y|\mathbf{x}_S\right)\right]$ as seen from Eq. (6). It means $\mathbf{x}_S$ governs the uncertainty of the code output. On the other hand, uncertainty of $\mathbf{x}_S$ has no relation to the uncertainty of the code output if the correlation ratio is zero. As seen from Fig. 5, the sensitive parameters from the viewpoint of output uncertainty are the core decay heat and the check valve pressure loss.

The natural circulation DHR capability is determined by the balance of the reactor decay heat and the natural circulation flow. The primary flow coast down time constant is a few second while the peak temperature appears at 40-50 second. Therefore, the result that the time constant has small correlation ratio is consistent with our engineering intuition. However, it may be strange that the correlation ratio of the pressure losses of the reactor core and the primary pump are not significant.

III.C. Quantification of Dependency of Input Parameters

It may be common that we consider the pressure loss correlation equations of the reactor core, the primary pump and the check valve are dependent one another. In the natural circulation mode, the coolant flow mass is a few percent of the rated value. For example, the flow velocity in the reactor core and the piping system is approximately 5-10 cm/s. The LMRs are designed based on the concept that decay heat be removed by the forced circulation. So the natural circulation is not a design basis event. Therefore, the pressure loss coefficient of a component is mostly measured in a higher flow rate region, i.e. 10-20 % of the rated flow. In the natural circulation range, the uncertainty in the pressure loss correlation coefficients seems to be large. The pressure loss is measured using a mock-up test in the water. Accordingly, one may consider that the pressure loss data and experimental correlation equations of sodium components are dependent each other.

In this section, the quantification method of the input parameter dependencies is presented, which can be applied to the stochastic safety analysis. Let us consider a component K and its pressure loss coefficient x_K as the input parameter. It is assumed the pressure loss coefficient x_K follows the normal distribution with the mean μ_K and the standard deviation σ_K. Let us express the relation as:

$$x_K \sim N(\mu_K, \sigma_K) \tag{7}$$

It is considered that the uncertainty of the pressure loss coefficient consists of the independent part inherent to its own and common part to other components. Hence the pressure loss coefficient is expressed by the sum of the independent part x_K^I and the common part x_K^C as:

$$x_K = x_K^I + x_K^C \tag{8}$$

For generating random samples of x_K, we consider the probabilistic features of the dependency. Letting the variance of x_K^I be ξ_K^2 and the variance of x_K^C be η_K^2, the variance of x_K that follows the normal distribution is expressed as:

$$x_K \sim N(\mu_K, \sigma_K) = N\left(\mu_K, \sqrt{\xi_K^2 + \eta_K^2}\right) \tag{9}$$

$$\sigma_K^2 = \xi_K^2 + \eta_K^2 \tag{10}$$

If we define the fraction of the common part of the input parameter variance be ϕ,

$$\xi_K^2 = (1-\phi)\sigma_K^2, \;\; \eta_K^2 = \phi\sigma_K^2 \tag{11}$$

It is easy to generate samples of x_K following Eq. (9). Standard normal random sample r_K and $\hat{r}$ that are independent each other are generated first. Then, the sample of x_K is given by:

$$\tilde{x}_K = \mu_k + \xi_K r_K + \eta_K \hat{r} \tag{12}$$

Since the mean values of r_K and $\hat{r}$ are zero and the standard deviations are unity, the mean and the variance of Eq. (12) are μ_k and ${\xi_K}^2 + {\eta_K}^2 = {\sigma_K}^2$, respectively. It is consistent with the definition of Eq. (7).

For another component K' with the common characteristic to the component K, the independent and common parts of the variance are calculated using the fraction ϕ of the common part by:

$$\xi_{K'}^2 = (1-\phi)\sigma_{K'}^2, \, \eta_{K'}^2 = \phi\sigma_{K'}^2 \tag{13}$$

Hence, samples for components K' are given by:

$$\tilde{x}_{K'} = \mu_{k'} + \xi_{K'} r_{K'} + \eta_{K'} \hat{r} \tag{14}$$

It is noted that Eq. (12) and Eq. (14) have the common random variable $\hat{r}$ that expresses the dependency of the two components. The fraction to the total variance is ϕ. This procedure can be easily extended to three or more component cases and two or more dependencies cases.

We assume the dependency among the three input parameters for pressure loss coefficients of the reactor core, the primary pump and the check valve. Figure 6 shows the numerical results for independent case ($\phi = 0$), moderate dependency case ($\phi = 0.2$), and large dependency case ($\phi = 0.8$). In the moderate dependency case, the correlation ratio of the pressure loss coefficients of the reactor core and the primary pump exceed 0.1. It seems to be reasonable if the input parameters are dependent one another. When we assume $\phi = 0.8$, it is seen from Fig. 6 that the contributions from the three pressure loss coefficients are almost the same. It is matter of course that the importance of each input parameter is almost the same when they are very much correlated.

Lastly let us discuss how the dependency among input parameters are defined and quantified. Examples are given in the following. If a bias in the measurement of the pressure loss at an extremely low flow condition have the same tendency for all the components, for example, we considers the experimental estimates of the pressure losses for different components are correlated to some extent. Another example is that conservative tendency in an engineering judgment may exist in developing the pressure loss correlation, which leads to the underestimate of the natural circulation mass flow for a safety reason.

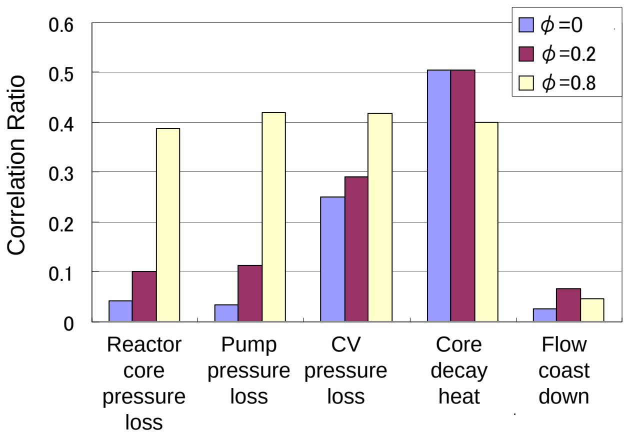

Fig. 6. Correlation ratio for the natural circulation DHR when the dependency among the input parameters are considered.

To evaluate the dependency quantitatively, one needs further study of the individual input data characteristics returning back to the origin. In this study, the influence of the degree of dependency is investigated. We assume representative values for the common part fraction. From the practical viewpoints, identification and quantification of the root cause of the dependencies among the input parameters are important. Since it is not simple and not an easy task, screening of the input parameters would be effective as performed in section II.

IV. CONCLUSIONS

The LMR is the most promising fast reactor design in Japan. Making provision for the construction of the demonstration LMR, a safety analysis methodology is desirable based on the best-estimate thermal-hydraulic code and the most probable assumptions and input parameters with uncertainty consideration. In this study, the stochastic safety analysis of natural circulation DHR of the LMR has been proposed. The thermal-hydraulic response involves inherently various uncertainties. The uncertainties come from the input parameters, physical modeling, and numerical methods employed. The natural circulation of the LMR is a single-phase thermal-hydraulic phenomenon and the maximum core coolant temperature follows the normal distribution. In other words, no outlying behavior of the system response is observed within the present variation range of input parameters. Hence it can be said that the stochastic method is useful for the DHR safety design and regulation of the LMR.

The importance of the individual input parameters from the viewpoint of uncertainty has been estimated using the variance of the conditional expectation (VCE) and Pearson's correlation ratio. It has been found that the decay heat and the primary pressure loss coefficient are dominating the output uncertainty. It is noted that the input parameters are sometimes dependent one another. A methodology has been developed for evaluating the dependencies among input parameters. The proposed methodology is effective for the stochastic safety analysis and understanding the results.

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NOMENCLATURE

Acronyms

ACS	Air Cooler System
COV	Coefficient Of Variance (Standard deviation divided by mean)
CSAU	Code Scaling, Applicability and Uncertainty
DHR	Decay Hear Removal
IHX	Intermediate Heat Exchanger
JSFR	Japanese Sodium-cooled Fast Reactor
LMR	Liquid Metal Reactor
LOHS	Loss-Of-Heat-Sink accident
pdf	probability density function
USNRC	United States Nuclear Regulatory Commission methodology
VCE	Variance of Conditional Expectation

Variables and Functions

$E(\cdot)$	Expected value
$f(\cdot)$	Probability density function
K	Identifier of a component
$M(\cdot)$	Simulation model
$N(\cdot,\cdot)$	Normal distribution
$r, \hat{r}$	Standard normal random sample
$V[\cdot]$	Variance
$\mathbf{x}$	Input parameter vector
x_S	Subset of $\mathbf{x}$
$x_{\bar{S}}$	Complementary subset of $\mathbf{x}$
y	Code output
η^2	Pearson's correlation ratio
μ	Mean value

σ, ξ, η Standard deviation

ϕ Fraction of common variance

Superscript
I Independent part
C Common part

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