Performance Uncertainties of LEU Mo-99 Targets for HANARO

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ABSTRACT

The use of low enriched uranium (LEU) fuel target was examined on the feasibility of ^{99}Mo production in a research reactor, HANARO. Uncertainty analysis was done with respect to ^{99}Mo yield ratio, ^{239}Pu yield ratio, annual production rate, and decontamination requirement. An equilibrium core model for MCNP fixed-source problem was found by reactor design methodology, WIMS/VENTURE. Target design option with LEU was proposed. Variables related to target fabrication process and reactor physics condition was considered as uncertainty-inducing parameters. The most important factor affecting the overall uncertainty in LEU option was engineering tolerance in fabrication process of fuel film. It is acceptable to use LEU as a target fuel in view of radioactive purity of alpha emitter because the uncertainty in impurity level of ^{239}Pu is expected to be relatively small, only 6% in 95% confidence level.

I. INTRODUCTION

Molybdenum-99 is a parent isotope of technetium-99m which has been widely used in diagnostic medical imaging procedures. Commercial production of ^{99}Mo, however, has been almost exclusively dependent upon Canadian company Nordion. Therefore, a need for a reliable regional supply facility has been recognized worldwide.[1,2] Korea Atomic Energy Research Institute (KAERI), therefore, has developed molybdenum-99 production technology with 30MWt reactor, HANARO as a national R&D program since 1997.[3,4]
Current ^{99}Mo suppliers have used high enriched uranium (HEU) fuel target because of high ^{99}Mo yield ratio (Ci ^{99}Mo/gU) and low impurity level of α-emitters. RERTR program, however, recommends the use of LEU as a fuel or target in research or test reactors.[5, 6] Besides penalty in engineering economics, technical issues on the use of LEU as an alternative to HEU has not been resolved yet. Requirement of Pu impurity decontamination should meet the general international standards and U.S. Pharmacopoeia(USP) standard. Recently, experimental demonstrations of irradiation and chemical processing of LEU metal foil target were performed by Argonne National Laboratory. It was said that purity of the ^{99}Mo product does not come into question when HEU was converted to LEU. In this work, purity control of the ^{99}Mo against

gamma emitters, such as I-131, Ru-103, Te-132 and Np-239, etc, could be ensured within the international purity specification. The decontamination of alpha emitter, ^{239}Pu, however, was not proved in detail.[7,8] It is expected that LEU option have larger level of uncertainty in the amount of Pu impurity than HEU, because the LEU contains the 70 to 80 times as much ^{238}U as HEU and produces 30 to 50 times as much Pu for the same production goal of ^{99}Mo. The uncertainty reduction was a momentous issue because of the following reasons. If we don't know exactly the uncertainty in yield ratios of ^{99}Mo and ^{239}Pu from LEU, it is questionable to use a conventional Cintichem chemical process for HEU. Although purity specification might be satisfied, large uncertainty can cause economic defect for reliable quality control in post processes. Uncertainty evaluation should be performed for reactor safety analysis and environmental safety assessment prior to license application on ^{99}Mo production.
In this paper, a feasibility of the LEU as a target fuel for ^{99}Mo production was checked by evaluating the uncertainties of key response parameters, such as ^{99}Mo yield ratio in unit of Ci^{99}Mo/gU, ^{239}Pu yield ratio in unit of Ci^{239}Pu/gU, annual production rate in unit of Ci^{99}Mo/yr, and decontamination requirement in unit of minimum required decontamination factor satisfying the USP standards. In section II, physics model for isotopic inventory calculation is discussed. Section III describes the methodology to set up a reference core model at burned state for target design. Section IV is allocated to the results and discussion of uncertainty analysis.

Ⅱ. EVALUATION OF ANALYSIS TOOL

HANARO is not a dedicated reactor for ^{99}Mo production but a multi-purpose research reactor for experiment and miscellaneous isotope production. Hence, HANARO has very heterogeneous configuration with 25 vertical irradiation thimbles and 7 horizontal beam tubes, as shown Fig. 1.

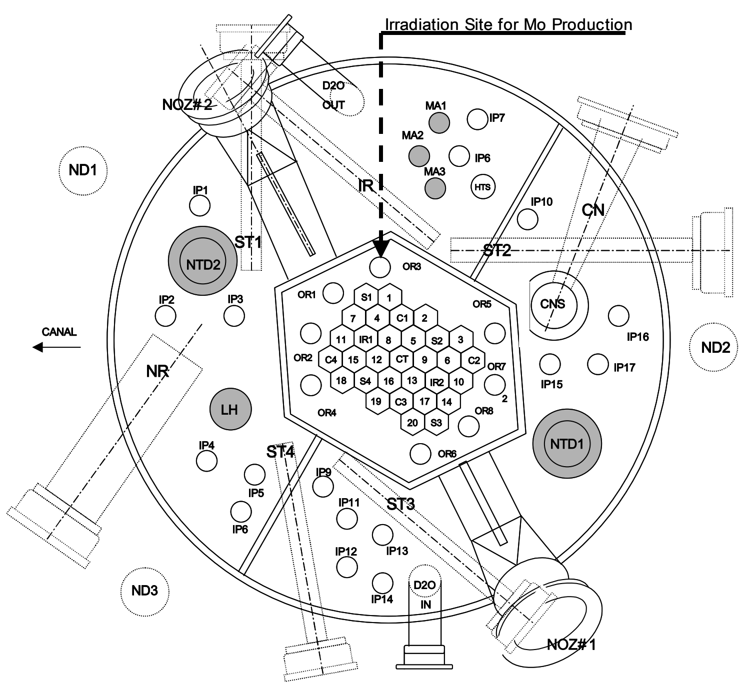

Fig. 1 HANARO Core and Target Irradiation Site

Fission Mo production target should be loaded at outer core hole, OR3 far from the center of the core and therefore within the heavy water reflector tank. Because outer core is supplied with neutrons from inner core, neutron spectrum in this region is softened to thermal range compared with that in the core. Heterogeneity and rapidly varying spectrum make the deterministic diffusion-theory code system with few-group cross section library unreliable for the evaluation of the fission reaction rate in the target. In this study, full core 3-D calculation of MCNP code was done for the evaluation of performance of core and target. And, ORIGEN2 code was used for the evaluation of isotope production and radioactive decay transient throughout the whole processes including post-irradiation chemical treatments.
Production-destruction schemes of Mo-99 and Pu-239 were searched and characteristics of these schemes were analyzed to evaluate exact isotopic amount by MCNP/ORIGEN2 code system. From this analysis, the amounts of ^{99}Mo and ^{239}Pu evaluated with modified cross section library of σ_f^{235}, σ_r^{238} agree well with the results with library including all modified cross section associated with production-destruction chain of ^{99}Mo and ^{239}Pu within 0.5% and 0.3% differences, respectively. Therefore, only these two cross sections generated by MCNP were modified, whenever the amount of isotope was evaluated.

III. REFERENCE TARGET DESIGN FOR BURNED CORE

III.A. CORE MODELING METHODOLOGY

An equilibrium core was modeled as a reference condition for target design optimization and uncertainty analysis. To perform an analysis with the MCNP code, number density of all isotopes and neutron source distribution should be defined. A condition of core at burned state was defined with WIMS/VENTURE design methodologies. Fig. 2 shows schematic diagram for equilibrium core modeling procedure proposed in this work for MCNP calculation.

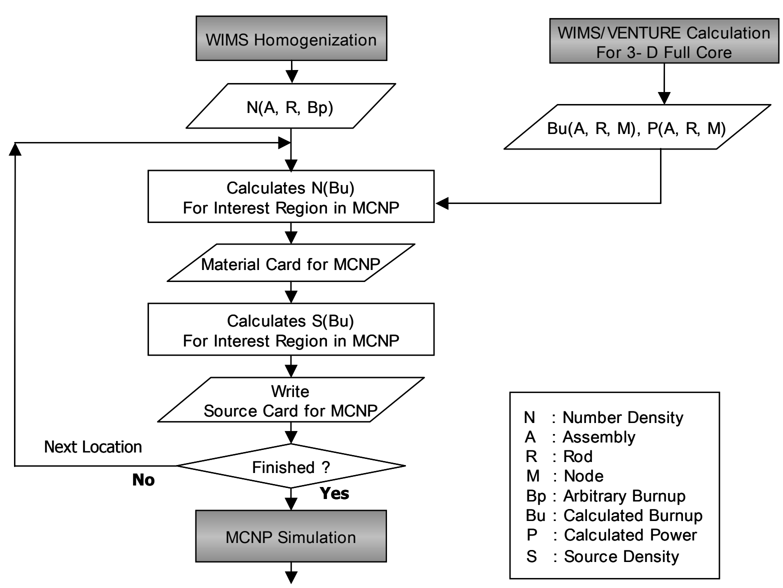

Fig. 2 Core Modeling Flow Chart by WIMS/VENTURE and MCNP

Burnup-dependent number density was calculated to each rod for each assembly by WIMS. Burnup and power of each node at any fuel region were calculated by VENTURE for

equilibrium core of HANARO. Based on burnup-dependent number density library calculated by WIMS, number density of each region in MCNP model was obtained by interpolation fitting to the calculated burnup rate from VENTURE. The same procedure was applied to obtain the source density for each region in MCNP model. Eight actinides and twenty fission products with relatively high macroscopic absorption cross section were treated in burnup model.
Full scope 3-D detailed description of HANARO for reactor analysis needs too much computational burden in MCNP model. Therefore, fixed source problem with homogenized region geometry is used for target design. Importance of each fuel assembly in core to an irradiation site was analyzed by MCNP code. Assuming that the neutron source would be generated from single assembly, the contribution of each assembly to irradiation site was calculated quantitatively like Eq. (1). W_i represents the ratio of thermal power of assembly i to average power and T_i represents the neutron flux at irradiation site originated from assembly i.

$$Contribution\ of\ assembly\ i = W_i T_i \qquad (1)$$

An importance of each assembly to irradiation site was evaluated by Eq. (2). The N represents the number of total fuel assemblies.

$$Importance\ of\ assembly\ i = \frac{W_i T_i}{\sum_{j=1}^{N} W_j T_j} \qquad (2)$$

The importance distribution of all fuel assemblies is shown in Fig. 3. The assembly regions that have importance of more than 1%, then should be treated heterogeneously were only nine assemblies, about 30% out of all. An accumulated contribution of nine assemblies to irradiation site was more than 93% in total. It was shown that if nine assemblies are modeled explicitly, neutron flux distribution at irradiation site is not varied by different level of material homogenization in other assemblies, as shown in Fig. 3. Therefore, nine assemblies were modeled with axially heterogeneous fuel rod. Except for these assemblies, fuel rod was homogenized axially.

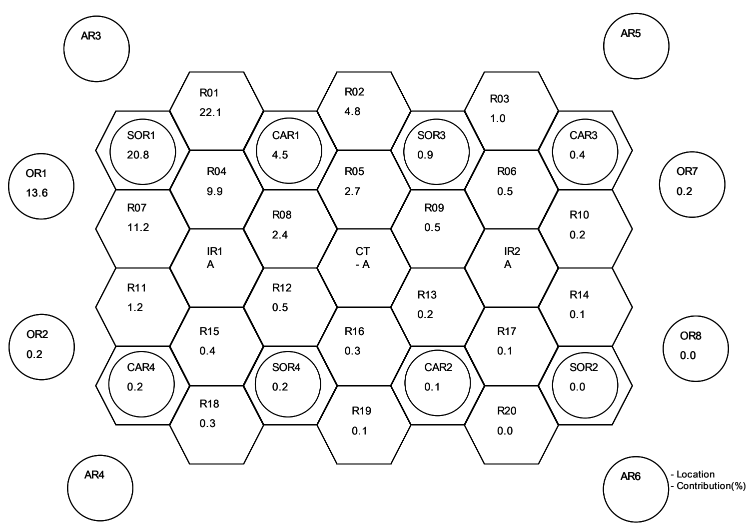

Fig. 3 Contribution of Each Fuel Assembly to OR3 Hole

To verify this geometry model, relative power distribution calculated by MCNP was compared with the results of WIMS/VENTURE calculation. The root mean square error was small as 2.5% and large errors occurred at the control rod locations where importance to target were small, as shown Fig. 4. Therefore, it is concluded that isotopic number density and source distribution for burned core was properly modeled for MCNP.

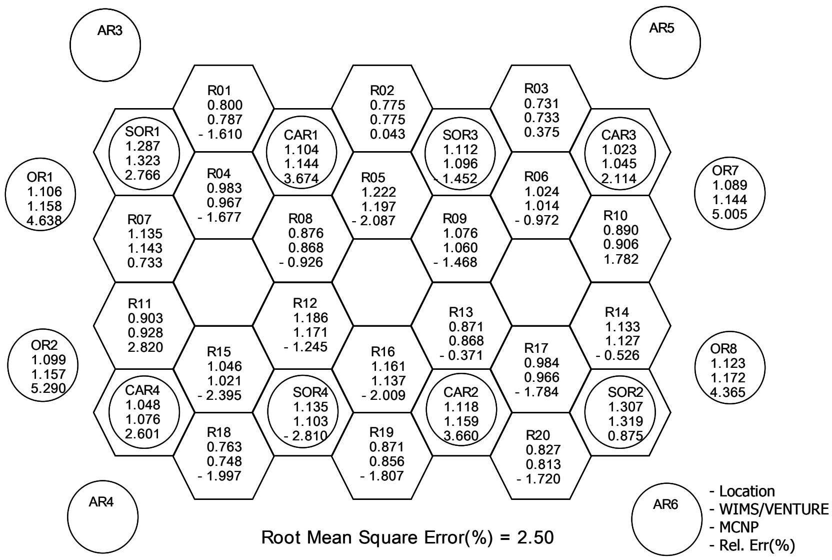

Fig. 11 Relative Power Distribution by WIMS/VENTURE and MCNP at MOC

Ⅲ.B. DESIGNED LEU TARGET

The design optimization was performed for the operational condition of HANARO. The optimization goals were high ^{99}Mo production yield ratio, low temperature in target cladding and fuel, high engineering feasibility in target fabrication, high ^{99}Mo recovery ratio in post-chemical processes, and low amount of radioactive waste. Sensitivities of design parameters such as axial irradiation location, fuel thickness, radius, axial length, the number of target, choice of ^{99}Mo recoil loss barrier, and choice of fuel material were evaluated in detail to the design performance parameters such as ^{99}Mo yield ratio in unit of Ci^{99}Mo/gU, cladding surface temperature, annual production rate in unit of Ci^{99}Mo/yr and waste production rate.[9,10] Based on these characteristics, LEU target design option was proposed. The dimensions and evaluated parameters of designed LEU target are described in Table 1 and Table 2, respectively. This target uses 19.75w/o enriched U metal fuel embedded in both Al-cladding. The Ni recoil-barrier is bonded to inner surface and outer surface of fuel film. This design reflects the concept proposed by Argonne National Laboratory.[11,12] Further design optimization will be done for a real application with this target, however, required modification would not be large.

Table 1. Design parameters for LEU Target

Axial Length (cm)	Target Cladding		Thickness(μm)			U Loading (g/target)
	Outer Tube O.D.(cm)	Inner Tube I.D.(cm)	Cladding	Recoil Barrier	UO_2 Fuel	
10	4.40	4.088	1,500	10	100	24.40

Table 2. Response Parameters for LEU Target

Reactivity Worth ($\%\Delta\rho \pm 1\sigma$)	Yield Ratio ($Ci^{99}Mo/gU$)	Production ($Ci^{99}Mo/yr$)	Max. Clad. Temp.($^{\circ}C$)	ΔT($^{\circ}C$) (Outlet-Inlet)
0.0761±0.0158	6.967	5,440	88.8	1.05

Ⅳ. UNCERTAINTY ANALYSIS ON FISSION MO TARGET

Fission Mo production project includes target fabrication, irradiation in reactor, cooling for radioactive decay, chemical processing, and waste disposal. These processes have large uncertainties themselves. If we don't account for these uncertainties properly, incorrect decisions for ^{99}Mo purification step in a chemical process would be given rise to. We should remind that ^{99}Mo purification process should be established as few step as possible for better economics. It is not easy, however, to establish an appropriate chemical process with the least purification step because of uncertainties involved in the isotopic amount estimation, especially when it comes to a case for a technology still in the experimental phase like LEU target. It is, therefore, significant to take these uncertainties into consideration in the estimates of isotopic amount such as ^{99}Mo and ^{239}Pu. The sources of uncertainty for response parameters in ^{99}Mo production are as follows; At first, reactor physics parameters induce the uncertainty. These sources of uncertainty arise from inaccurate core model in MCNP and insufficient consideration of all reactor condition when response parameters were evaluated. Secondly, manufacturing tolerance of fabrication and composition induce the uncertainty. These sources of uncertainty arise from inaccurate target dimension and material composition in the engineering process. Thirdly, chemical processing tolerance induces the uncertainty. This source of uncertainty arises from the condition of chemical processing. The uncertainty in this study means that the scatter of response parameters, such as ^{99}Mo yield ratio, ^{239}Pu yield ratio, annual ^{99}Mo production rate, and minimum required decontamination factor (MRDF) to satisfy the USP standards. Hereafter, we refer to these parameters as response parameters. The uncertainty caused by nuclide number density calculated by WIMS/VENTURE code system, MCNP code bias and chemical processing is not included in this study because of the lack of experimental data.

Ⅳ.A. DETERMINATION OF INPUT PARAMETERS

Ⅳ.A.1. Reactor Physics Parameters

In MCNP code, the source position is preferentially sampled according to source density distribution converted from thermal power distribution. And then, the source energy is sampled based on fission neutron spectrum to generate the neutron source. In deterministic code system, the number of neutrons generated in fuel region *i*, and energy interval *g* can be expressed by

$$S_{ig} = \sum_j \chi_g^j \nu^j \left[\sum_{g'} \Sigma_{fg'}^j \Phi_{g'} V \right]_i$$

$$\cong \sum_j \chi_g^j \frac{\nu^j}{E_r^j} \left(\frac{[\Sigma_f]^j}{\sum_k [\Sigma_f]^k} \right)_i P_i = \sum_j \chi_g^j S_i^j \qquad (3)$$

where, i = fuel region index,
j,k= fissile nuclide index,
g = energy interval index,
χ= fission spectrum,
P_i = thermal power produced at region i,
ν = average number of neutron emitted per fission,
E_r = recoverable energy per fission
S_i^j = number of neutrons produced from fission of nuclide j.[13]

In Monte Carlo code system, the number of neutrons generated in fuel region i, and energy interval g can be expressed by

$$S_{ig} = C\sum_j w_i^j \chi_g^j S_i \text{ where, } w_i^j = \frac{S_i^j}{\sum_j S_i^j}. \qquad (4)$$

Each term in right location of Eq. (4) means probability density function normalized to 1.0. That is to say, S_i represents the probability that neutron is generated in the region i, normalized to whole core. And, w_i^j is the probability the neutron source spectrum of nuclide j is selected in region i. The C factor represents normalization constants in MCNP. In this section, we elaborate on the uncertainties associated with terms of the source formulation.

A) Source Spectrum

After completing position sampling, it is determined that which fission source spectrum would be applied to sample the neutron energy. It should be considered that fission spectra of ^{235}U is different from that of ^{239}Pu. It is definite that the content of Pu for each region varies from different burnup, although it exists in the same assembly. This means that fission reaction ratio of Pu to total fission is varies for each region. Therefore, the probability that χ^j is selected as a neutron source spectrum should be defined for each region i. For example, if the probability of ^{239}Pu fission is P, and the probability of ^{235}U fission is $1-p$ in certain region i, the chance that ^{239}Pu spectrum is selected should be defined with the probability of P to that region. It was difficult, however, to define the probability that χ^j is selected for each region, and subsequently it was not applied for each region but for each assembly just like Eq. (5). The l denotes the assembly index in Eq. (5).

$$S_{ig} = C\sum_j w_l^j \chi_g^j S_i \qquad (5)$$

Finally, the regions included in the same assembly have the same probability that ^{239}Pu spectrum is selected. Therefore, the uncertainties resulting from incorrectly defined fission source spectrum should be evaluated.

B) Source Density Modeling in MCNP

For the proper sampling of starting position of neutron source, P_i in eq. (3) should be defined exactly for each region. The major uncertainty associate with P_i depends on the accuracy of the core physics calculation. The P_i that was calculated from WIMS/VENTURE code system has the uncertainty itself. This implies that source position sampled has the probability of biasing and response parameters evaluated have the uncertainties.

C) Fissile Consumption in Target
When the amount of ^{99}Mo and ^{239}Pu was calculated in MCNP/ORIGEN code system, it was assumed that the fission reaction rate in the target was constant during 5-days irradiation. It is obvious that fissile material in the target is consumed by nuclear fission, which makes the fission reaction rate decrease during the irradiation period. Inappropriate consideration of this phenomenon in MCNP/ORIGEN2 code system induces the uncertainty of response parameters.

D) Variation of Reactor Power Level
An absolute reactor power was assumed with 30MWt for the normalization in MCNP/ORIGEN2 system. It is difficult to maintain the reactor 30MWt constant, and subsequently the power level vibrates from the 30MWt normal power. This fact should be considered in the uncertainty analysis.

Ⅳ.A.2. Geometrical and Composition Tolerance of Target
To reduce the uncertainty, the target should be fabricate in accordance with the dimension that was optimized. It is difficult, however, to make the target dimension the same with designed value. Fabrication tolerances which should be considered are thickness and axial length of the fuel and cladding. It is also difficult to make the material composition the same with designed value. For example, it is expected that the enrichment of the LEU fuel should be made 19.75w/o. The material with biased enrichment makes the response parameters have undesired value. These parameters should be considered as an uncertainty-inducing factors.

Ⅳ.B. SCREENING BY SENSITIVITY ANALYSIS
It is difficult to perform the uncertainty analysis considering the all parameters previously mentioned. The screening of input parameters whose effect of uncertainty of response parameters was negligible was done as a first step of uncertainty analysis. The eliminated parameters were thickness, axial length and composition variation of Al cladding in LEU target. The uncertainty from these parameters was too small to be neglected.

Ⅳ.C. Quantification of Input Parameters
Each input parameter can be treated as random variable having a probability density function. However, it is difficult to determine appropriate statistical distribution when there are not exact historical data. Uniform distribution, triangular distribution, normal distribution were considered as an input parameter distribution. We can find that uniform distribution has the largest variance among these distributions. This fact means that uniform distribution make the uncertainty of response parameters large conservatively. Triangular distribution can reflect the expert opinion deduced from a few data. This fact implies that this distribution can be applied to decrease the conservatism of uncertainty. Prior to performing this work, most of experimental data related to input parameters were insufficient. Additionally, experts for fission Mo production were not enough. Therefore, input parameter distribution having insufficient experience was treated as uniform distribution for conservative assumption.
A statistical distribution of Pu fission ratio to total fission for each assembly should be defined to estimate the uncertainty resulting from improperly applied fission source spectrum χ. At first, Pu fission ratio to total fission for each region was calculated from WIMS/VENTURE result. Because 36-rod assembly was modeled with 504 nodes in WIMS/VENTURE calculation, we can

obtain 504 data point. And then, these data was manipulated statistically. This distribution doesn't represent certain formal distribution. Therefore, fission ratio of Pu for each assembly was assumed with uniform distribution, because this conservative distribution would be compensate for the uncertainty induced by inaccurate isotopic number density for equilibrium core. In the target design optimization step, Pu fission ratio, namely, probability that ^{239}Pu fission spectrum is selected, was set to most likely value. The distribution of P_i in Eq. (3) to obtain S_i in Eq. (4) was treated as normal distribution with 4.5%(1σ) uncertainty according to FSAR of HANARO. This means that relative assembly power calculated by WIMS/VENTURE can deviate from exact value with ±4.5% difference in 1σ level. This probability distribution function is expressed in Eq. (6) with mean of r_i and standard deviation of 0.045 r_i. The r_i means assembly power peaking factor.

$$f(x) = r_i \frac{1}{\sigma\sqrt{2\pi}} e^{-\frac{(x-1)^2}{2\sigma^2}} , \quad -\infty \le x \le +\infty \tag{6}$$

The distribution of reactor power level was derived with the minimum value of 97.4% and the maximum value of 102.6% of 30MWt. This data is based on the sufficient operating experience. The variation of fissile nuclides for irradiation period is described in Table 3. In Tables 4 and 5, the fabrication tolerance and composition tolerance are described based on experience data, respectively. The distributions associated with these parameters are assumed with uniform distribution.[14]

Table 3. The Variation of fissile Nuclide for Irradiation Period

Amount of Fissile Nuclide in Target (#/barn-cm)		
Min. Value(a)	Max. Value(b)	Deviation(%)
9.1870E-03	9.613E-03	±2.2
Uniform Distribution : $f(x) = \frac{1}{b-a}, a \le x \le b$ $= 0$, *otherwise*		

Table 4. A Distribution of Input Parameters with Fabrication Tolerance

		Min.(a)	Max.(b)	Deviation(%)
Fuel	Thickness(μm)	80	120	±25.0
	Axial Length(cm)	9.5	10.5	±5.00
Uniform Distribution				

Table 5. A Distribution of Input Parameters for Fuel Composition

Concentration [w/o]					
U-235			U-238		
Min.(a)	Max.(b)	Deviation(%)	Min.(a)	Max.(b)	Deviation(%)
19.50	20.00	±1.27	80.00	80.50	±0.31
Uniform Distribution					

IV.D. UNCERTAINTY ANALYSIS METHODOLOGY

The Crude Monte Carlo sampling method was applied as a uncertainty analysis methodology.[15,16] Input variable set is composed by random sampling for each input parameter, and then this is applied to the complicated computer model. Repeating this procedure, we can obtain the exact distribution of response parameter. No correlation of input parameters was convinced by χ^2 test. This methodology has disadvantage that we can't found by what parameter the uncertainty was induced. Therefore, to clarify the uncertainty importance of parameters, it should be identified that what input parameter introduces the largest spread in response parameters. When we sample input variable set of power distribution P_i to make the source distribution, permission/rejection method was applied to compensate for correlation of each assembly. The only input variable set which make the mean of sampled P_i the average linear power rate $\overline{P}$ is applied for computer model. If the many response parameters is calculated, the confidence interval is quantified by Eq. (7) with the weighing of the statistical error.

$$\overline{C} - T_{95,95} S_c \leq X \leq \overline{C} + T_{95,95} \tag{7}$$

where, $$\overline{C} = \frac{\sum_{i=1}^{n} \frac{C_i}{S_{C_i}^2}}{\sum_{i=1}^{n} \frac{1}{S_{C_i}^2}}, \quad S_c = \frac{n}{n-1}\left(\frac{\sum_{i=1}^{n}\left(\frac{C_i}{S_{c_i}}\right)^2}{\sum_{i=1}^{n}\left(\frac{1}{S_{c_i}}\right)^2} - \overline{C}^2 \right),$$

and, $T_{95,95}$ represents two-sided tolerance limit factor.[17]

IV.E. RESULTS

Table 6 lists the some statistical characteristics of uncertainty distributions for each response parameter through the simulation of 100 input variable sets for LEU target. It turned out that normal distribution fits the best the uncertainty distributions of the majority of each response parameter through Kolmogorov-Smirnov goodness of fit. It was shown that the 95% confidence intervals of ^{99}Mo yield ratio, ^{239}Pu yield, annual production rate, and MRDF were 6.23~ 7.70Ci^{99}Mo/gU, 4,047~ 6,382Ci^{99}Mo/yr, 1.381E-5~ 1.900E-5Ci^{239}Pu/gU and 4,834~ 5,596, respectively. These showed the uncertainties of about 9.5%, 20%, 14%, and 7% 2σ level. The major part of uncertainty to ^{99}Mo and ^{239}Pu yield ratio originates form target fuel film thickness. In case of MRDF, the major part of uncertainty arose from the uncertainty of power distribution calculated by WIMS/VENTURE code system. The demand of fission-produced ^{99}Mo is constant during a relatively short period of one year. This implies that stable and reliable supply of ^{99}Mo is crucial. A larger uncertainty in ^{99}Mo production rate induces overproduction to ensure the reliable supply. Let's suppose the two case of ^{99}Mo annual production rates with 4,000~ 4,200Ci and 4,000~ 5,000Ci in 95% confidence level. The latter case means that overproduction of 5,000Ci is possible to supply the 4,000Ci/yr of ^{99}Mo with the 95% confidence level. Therefore, technology to reduce the variation of thickness should be developed to increase the economics of LEU target. In view of Pu decontamination, if the decontamination factor (DF) of Pu for each step, namely, dissolution/precipitation, purification I, and purification II in cintichem process were 10, the total DF would be about 1,000. Therefore, more purification step for Pu decontamination would be required considering the uncertainty. But the uncertainty of MRDF in LEU revealed only 6% in 95% confidence level, though the uncertainties of ^{99}Mo and

^{239}Pu yield ratio are about 9.5% and 20%, respectively. This is due to the fact that correlation coefficient of ^{99}Mo and ^{239}Pu yield ratio is positive with the value of 0.91. The small uncertainty of MRDF in LEU means that the quality of ^{99}Mo/^{99m}Tc generator in view of alpha impurity would be definitely ensured, if one more purification step would be added to Cintichem process.

Table 11. Overall Uncertainty Evaluation Results

Target	Response Parameter	Mean	Standard Deviation	2.5% Quantity	97.5% Quantity	Rel. Err.(%) (2σ level)
LEU	$^{c}Mo-99(Ci/gU)$	31.591	1.497	28.247	34.934	9.48
	$^{c}Pu-239(Ci/gU)$	1.6402E-5	1.1620E-6	1.3807E-5	1.8998E-5	14.17
	$^{c}MRDF$ $(10^{-7}Ci-Pu)/(mCi^{99}Mo)$	5215.32	170.60	4834.19	5596.44	6.54
	$^{R}Mo-99(Ci/gU)$	6.9627	0.3299	6.2255	7.6996	9.48
	$^{R}Production\ Rate\,(Ci^{99}Mo/yr)$	5215.67	521.987	4049.55	6381.79	20.02

C: The value evaluated at chemical processing, R : The value evaluated at 6-day reference

Therefore, it is concluded that although the production amount of Pu in LEU is 50 times higher than that of HEU, applicability of LEU would be feasible because of small variation of alpha impurity in ^{99}Mo/^{99m}Tc generator.

Ⅴ. SUMMARY AND CONCLUSIONS

The applicability of LEU as a fission-produced ^{99}Mo target fuel was examined by quantifying the uncertainty of ^{99}Mo yield ratio, ^{239}Pu yield ratio, annual production rate, and MRDF satisfying the USP standards. The results of the uncertainty importance analysis resulting from each input parameters indicated that the inherent uncertainty of current fabrication process of LEU fuel film should be reduced. Therefore, technology to reduce the variation of thickness should be developed to increase the economics of LEU target. And, one more purification would be required considering the uncertainty because the 95% confidence interval of MRDF was 4,834~ 5,596. However, the uncertainty of LEU was only 6% in 95% confidence level, because correlation coefficient of ^{99}Mo and ^{239}Pu yield ratio is positive with the value of 0.91. This fact concludes that the quality of ^{99}Mo/^{99m}Tc generator would be ensured in clinical procedure. Therefore, if we overcome the economic disadvantage by massive commercial production using LEU, the use of LEU target looks to be feasible for ^{99}Mo production in HANARO. The uncertainties evaluated from this work will be applied for detailed design of chemical process.

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