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# The Criticality Analysis of Spent Fuel Pool with Consolidated Fuel in KNU 9 & 10

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# 조밀화 집합체로 중간저장하는 경우 원자력 발전소 9, 10호기의 사용 후 핵연료 저장조의 임계분석

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## Abstract

Since the lack of the spent fuel storage capcity has been expected for all Korean nuclear power plants in the mid-1990s, the maximum density rack (MDR) with consolidated fuels can be proposed to overcome the shortage of the storage capacity in KNU 9 & 10 which have most limited capacities. To ensure the safety when the alternatives are applied in the KNU 9 & 10, the multiplication factor are calculated with varying the rack pitch and the thickness of consolidated storage box by the AMPX-KENO IV codes.

The computing system is verified by the benchmark calculation with criticality experiments for arrays of consolidated fuel modules, which was reported by B & W in 1981. Also an abnormal condition, i.e. malposition accident, is simulated. The results indicate that the KNU 9 & 10 storage pools with consolidated fuel are safe in the view of the criticality. Thus the storage capacity can be expanded from 9/3 cores into 27/3 cores even with considering equipments and cooling spaces.

## 요 약

1990년 중반에는 우리나라 모든 원자력 발전소의 사용 후 핵연료 저장조의 용량부족이 예견된다. 따라서 조밀화 집합체로 저장하는 MDR 방법을 가장 저장용량이 적은 9, 10호기 원전의 저장용량을 확장시키는데 적용하고자 하였다. 이러한 방법을 채택할 때 9,10원전의 사용후 핵연료 저장조의 안전성을 확인하기위해 격자 간격과 저장통 두께를 변화시키면서 중성자 증배계수를 AMPX-KENO IV 코드로 계산하였다. 그리고 이 전산체제를 검증하기 위해 1981년 B & W에서 실시한 임계실험에 대하여 검증계산을 수행하였다. 또한 가상사고로

써 malposition 사고도 모사하였다. 그 결과, 원전 9, 10호기의 핵연료 조밀화 저장법은 안전하며, 설비 및 냉각공간을 고려하여 9/3 노심분을 27/3 노심분의 저장 용량으로 확장할 수 있을 것이다.

## 1. Introduction

In the mid-1990s, the lack of the spent fuel storage capacity is expected for all Korean nuclear power plants. Thus alternatives to increase on-site storage capacity have been sugested and analyzed in a point of the safety view. Among them, the maximum density rack (MDR) with consolidated fuels[1] can be only sufficient to overcome the shortage of the storage capacity in KNU 9 & 10 nuclear power plants which have most limited capacities. A consolidation concept permits the coordinated disassembly of spent fuel assemblies and subsequent repacking of the rods into dense closely packed arrays.[2]

The purpose of this study is to calculate the criticality of the KNU 9 & 10 storage pools with the consolidated fuels to ensure its safety. For these calculations, a set of computing system has been established with the AMPX-KENO IV code. The neutron cross section libraries are collapsed by the AMPX modular code system from 218 group library[3] which is the CSRL-IV master library. These are used as inputs for KENO-IV which solves the neutron transport equation by Monte Carlo method.

The benchmark claculation for those AMPX-KENO IV computing system with 19 neutron energy group is performed for criticality experiments for arrays of consolidated fuel modules, which was reported by B & W in 1981. Finally this validated computing system is employed to perform the criticality anaysis in the normal condition and the abnormal condition for the fuel consolidation. For the abonormal condition, the malposition accident is selected where a consolidated fuel assembly is assumed to be laid on other assemblies due to crane handling mistakes.

## II. Calculational Method

To evaluate the criticality of consolidated spent fuel pool, the AMPX modules will supply the collapsed cross section for KENO-IV.[4] This cross section generation is performed in 2 steps; the first step is the group collapsing of the cross section using the energy spectrum as a weighting function and the second is the collapsing with the space and group dependent neutron flux.

CSRL-IV master library, which is the raw data in this study, has fine group structure of 218 groups. However it is too expensive to calculate the spatial flux distribution with the library which will be used as a weighting function for a given system. So the simultaneous energy and spatial weighting is usually done for relatively broad group structure. Four modules in the AMPX package[5] (i.e., AJAX, MALOCS, NITAWL and XSDRNPM) are employed to generate the final 19 group cross section library for KENO-IV. The flow chart for cross section generation is presented in Fig. 1. In this study, 13 nuclides including the component of $UO_2$, Zricaloy-4, and SS-304 for KNU 9 & 10 are selected by AJAX module. However, 14 nuclides inlcuding the component of $UO_2$, Al6061-T6, $H_3 BO_3$ are selected for benchmark calculation. In MALOCS module, the master 218 fine group library is collapsed to 51 broad group library, where the neutron flux will be spatially independent with assuming the infinite homogeneous medium. In NITAWL calculation, the Dancoff factor which takes into account the rod shadowing effect[6] is taken to be suitable for a closely packed fuel rods by interpolating values which is referred from Carlvik's Table.[7] The temperature of the system is assumed to be the room temperature. For the last step of the cross

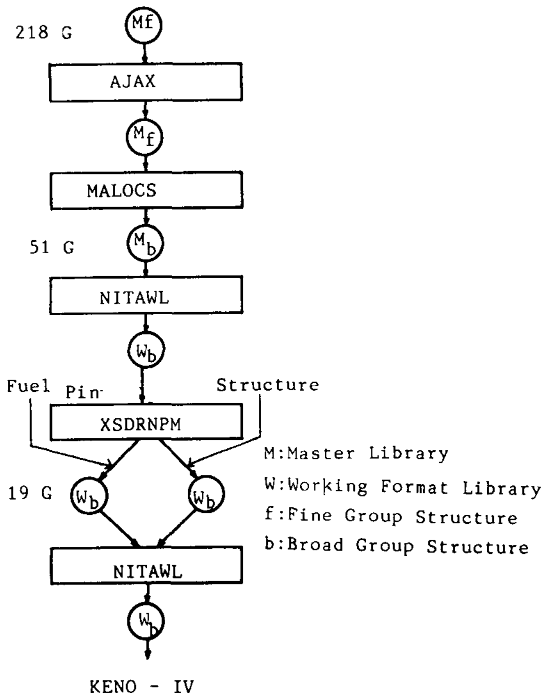


**Fig. 1. Flow Chart for Cross Section Generation.**

section generation, XSDRNPM module is employed. In this study, this module is used for group collapsing, especially to obtain the library representing the homogenized consolidated fuel. XSDRNPM is set up for the fuel pin geometry as well as for the structural geometry of the rack.

Using cell and zone weightings, the XSDRNPM calculation are implemented and the group collapsed cross section sets are obtained. These are then combined with NITAWL to provide the input for KENO-IV. Finally the KENO-IV geometry is modeled and $k_{eff}$ has been obtained by the KENO-IV code.

## III. Benchmarrks

Critical experiments[8] were carried out at the Babcock and Wilcox CX-10 facility in May 1979, to obtain basic data for validating analytical models of consolidated fuel storage racks. The fuel rods, which consisted of 2.46% enriched $UO_2$ pellets and clad of aluminum, were clustered in modules. And each assembly consisted of a 5×5 array of fuel modules. The variables measured in the experiment were the intermodular spacing, the spacing between fuel rods within the modules, the critical boron concentration, and to slight extent the critical water height. In the experiment, five basic core configurations were examined; Core I, II, and III employed a trangular pitch between fuel rods with different intermodular water gaps. Core IV and V empoyed a square pitch. Core I, II, and III are of current interest since the triangular pitch is highly dense and thus adopted in this study. Thus 3 benchmark calculations for those core configureations are performed to check the validity of the cross section libraries genrated from AMPX and the reliablity of KENO-IV code which used these libraries. Table 1 summarizes selected data for the five core types.

The results of AMPX-KENO IV calculations are presented in Table 2. The 95% confidence level camouflages the deviation and the AMPX-KENO

**Table 1. Selected Data of Benchmarks on 5 Types Cores.**

| Core | No. of Fuel Rods | Moderator Boron Conc. (ppm) | Critical Water Level(cm) | Intermodular Spacing(cm) |
|---|---|---|---|---|
| I | 6,075 | 435 | 143.56 | 1.778×1.945* |
| II | 6,075 | 361 | 142.54 | 2.539×2.709* |
| III | 6,075 | 121 | 145.64 | 3.807×3.976* |
| IV | 5,525 | 886 | 145.00 | 1.778 |
| V | 4,125 | 1,156 | 144.85 | 1.792 |

*The first number is the separation between the staggered row edges of adjacent modules and the second is the distance between even row edges of those.

**Table 2. Calculated multiplication factors of Core I, II and III for Benchmarks.**

| Core | Intermodular Spacing(cm) | Calculate $k_{eff}$ |
|---|---|---|
| I | 1.778×1.945 | 1.00766±0.00563 |
| II | 2.539×2.709 | 1.00044±0.00457 |
| IV | 3.807×3.976 | 0.99991±0.00456 |
| Total Number of Results | | 3 |
| Mean Value ($\mu$) | | 1.00267 |
| Standard Deviation($\sigma$) | | 0.00433 |
| 95% Confidence Level Deviation($2\sigma$) | | 0.00866 |
| Bias ($\mu$-1.0) | | 0.00267 |
| Uncertainty + Bias | | −0.00267 ±0.00866 |

IV computing system can be used successfully in the criticality analysis of consolidated fuel with the bias of 0.00267 for the 3 benchmark calculations.

## IV. Criticality Calculation in Spent Fuel Pool of KNU 9 & 10

### 4.1. Normal Conditions

Spent fuel pool of KNU 9 & 10 accommodates 472 fuel asemblies (9 cores) blocked into 2 of 8×6, 3 of 7×6 and 7 of 6×6 assemblies, and the center-to-center distance between racks is 37 cm. Devices for water circulation are installed and boric acid is normally diluted in water with 2,000 ppm. In this study, CE dimensions[9] (CE MAX-CAP) employed in consolidated fuel storage box thickness and rack pitch. For the calculation of the criticality, some assumptions (referred from US NRC-SRP[10] and ANSI N210-1976[11]) are made as follows:

1) Stored fuels are fresh and their enrichment is 3.5% which is the limit value written in KNU 9 & 10 FSAR.

2) No neutron poisons are in water at 20°C and 1 atm.

3) Fuel assemblies are arrayed infinitely in X, Y directions and the pool water as a radiation shield is flooded 200 cm over the top of the consolidated fuel assembly and 30 cm below the bottom.

4) The 95% theoretical density of $UO_2$ pellet is 10.406 g/cm$^3$

The consolidated fuel assembly includes two spent fuel assemblies into a tightly packed triangular configuration and is surrounded with a tight fitting membrane (SS-304) which is called consolidation fuel storage box as shown in Figure 2. An eligible simplification of this model is to use a homogenized representation of the peripheral fuel pins as well as the interior fuel pins. Thus, two distinct peripheral homogenized regions are defined as illustrated in Figure 3.

The number of fuel rods in each region is 494.5 and 11.5 and 22, respectively. In this model, the spectral calculation is carried out for a unit cell in region 1 with XSDRNPM. The resultant set of microscopic corss sections is used for all three homgenized regions; only nuclide number densities are all different in 3 rigions. The nuclide number densities are calculated according to volume fractions of pellet, gap, clading and water. The Dancoff Factor for NITAWL is to be 0.885. The geometrical model for the KENO-IV code is the rack cell model, whose top view is shown in Fi-

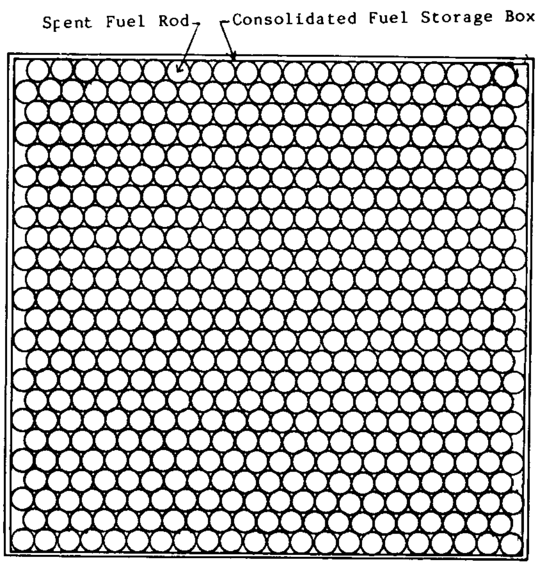


**Fig. 2. Consolidated Fuel Storage Box Model in Spent Fuel Pool of KNU 9 & 10(22×24)**

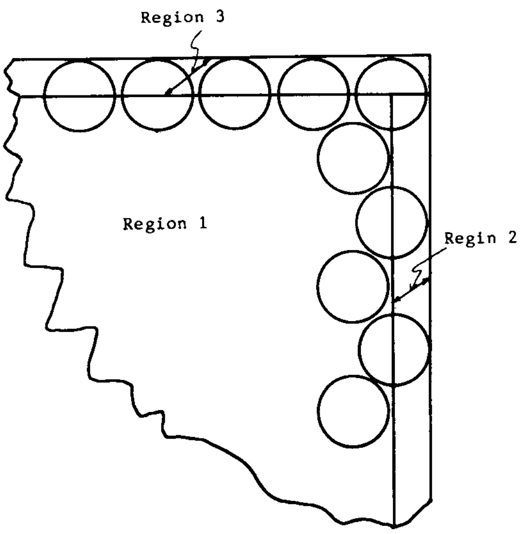


**Fig. 3. Homogenized Fu el Module Representation.**
1,2,3: Homogenlzed Fuel Assembly
4: SS-304 Tight Fitting Membrane
5: Pure Water(20°C)
6: S.S.-304 Can
7: Pure Water (20°C)

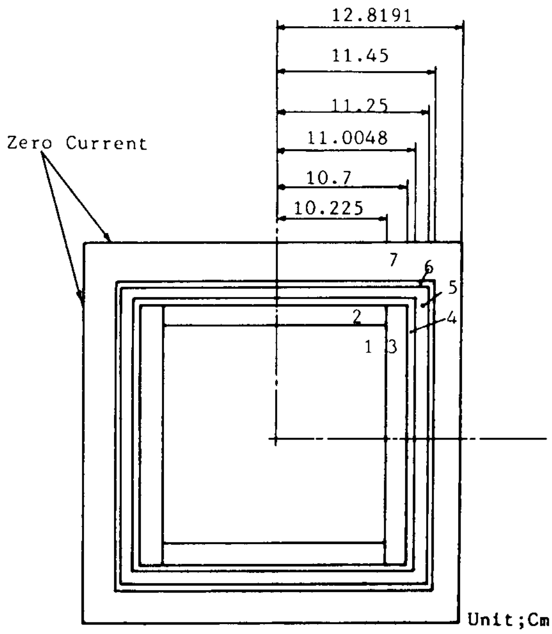


**Fig. 4. Normal Conditin KENO Model.**

gure 4. In KENO-IV code, various biasing methods, which are used in other Monte Carlo codes, are utilized to reduce the standard deviation and computing time. In this study, neutron weights for Russian Roulette and splitting are given as 1/3 and 3 times of the average weight, respectively. 300 neutron histories per batch for 53 batches in which the first 3 are excluded in stochastic calculation are used and this number of histories make the standard deviation to be below 0.01, which is a criterion vlaue for using the result of Monte Carlo claculation to system design. The fundamental generator of random number for sampling and tracking the neutron history is the internal function RANF in CYBER 170-875 computer at KAERI. This function utilizes the multiplicative congruential method with module $2^{48}$, i.e.

$$S(N+1)=a\times X(N)(\text{mode } 2^{48})$$

whhere a: the multiplier
$X_0$: the starting value or 'seed'
N: the seuqence number

Reflective boundary conditions are given at cell boundaries for the describing the inifnite array like Figure 4. Moreover, for the conserfivative calculation, the eccentricity and the cell thickness tolerance case are claculated.

The eccentricity means that 4 consolidated fuel assemblies make one sided concentration into the center of 4 storage racks. And the KNU 9 & 10 storage racks, which were designed by KHIC, retain the intainsic clearance ($0.2\pm0.05$ cm). Due to SS-304 nuetron absorption, the claculation of cell thickness tolerance is calculated with 0.15 cm cell thickness.

#### 4.2. Abnormal Conditions

In this study, in order to ensure the safety of consolidated spent fuel pool, an unliked abnormal condition is simulated. The malposition accident is assumed for a consolidated fuel assembly which was laid on the 20 $\times$ 7 assemblies due to crane handling mistakes, as shown in Figure 5. In that case, the top nozzle part is neglected and the consolidated fuel assembly is assumed to be located in the center of the other assemblies for conservative calculations.

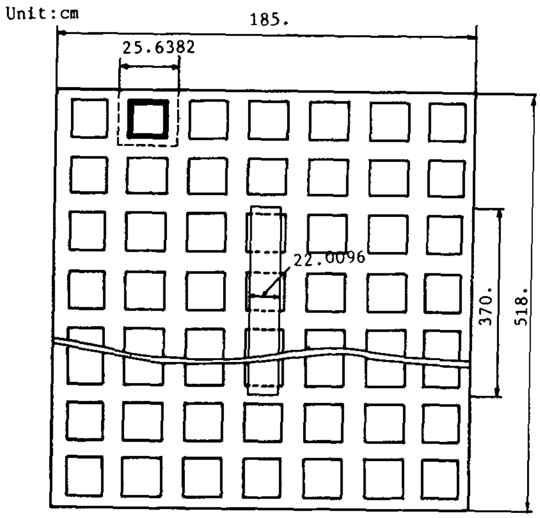


**Fig. 5. Maplposition KENO Model.**

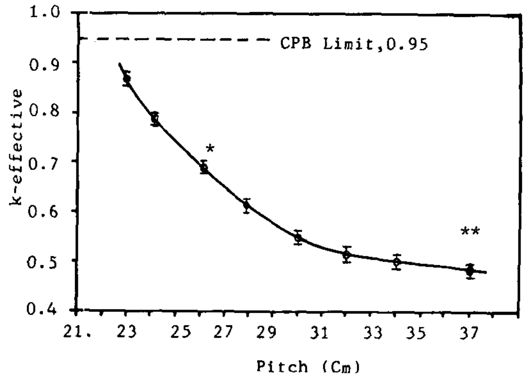


**Fig. 6. Multiplication Factors vs. Cell Pitch.**
*Consolidation Cell Pitch of KNU 9 & 10.
**No consolidation Cell Pitch of KNU 9 & 10.

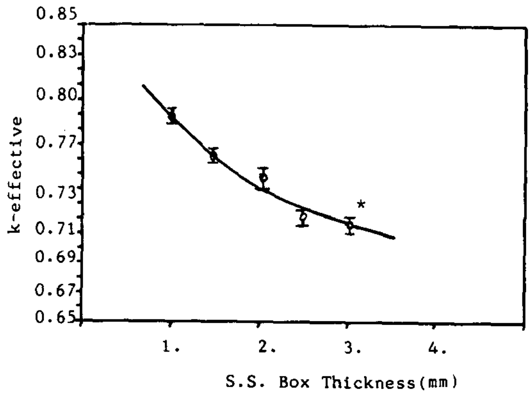


**Fig. 7. Multiplication Factors vs. Box Thickness.**
*CE Dimension (CE MAX-CAP)

## V. Results and Discussion

The effective multiplication factor, $k_{eff}$ for KNU 9 & 10 storage pool where the consolidated pitch is 25.6 cm is calculated to be $0.71786 \pm 0.00571$. Also the sensitivity of rack pitch size is studied. As shown in Figure 6, the multiplication factor increases with the decrease of pitch size due to the increase of moderator-to-fuel ratio in the overmoderated region. The sensitivity of consolidated fuel storage box is also carried out as shown in Figure 7. The multiplication factor decreases with increasing the storage box thickness due to the neutron absorption of SS-304, as expected. The nuclear criticality analysis should demonstrate that each LWR spent fuel storage facility system is subcritical ($k_{eff}$ should not exceed 0.95). Also methods used to calculate that subcriticality should be validated in accordance with Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety," which endorses ANSI NI6.9-1975. The evaluated multiplication factor of fuel in the spent fuel storage racks, $k_s$, should be equal to or less than an established maximum allowable multiplication factor, $k_a$.

i.e. $k_s \leq k_a$ (5.1)

The factor, $k_s$, should be evaluated by

$$k_s = k_{sn} + \Delta k_{sb} + \Delta k_u + \Delta k_{sc} \quad (5.2)$$

where $k_s$: the computed effective multiplication factor

$\Delta k_{sb}$: the bias in the calculation procedures which is obtaind from the comparisons with experiments

$\Delta k_u$; the uncertainty in the benchmark experiments

$\Delta k_{sc}$: the combined uncertainties.

The combined uncertainties, $\Delta k_{sc}$ include:

1) Statistical uncertainty in the calculated results if a Mcnte Carlo calculation is used.

2) Uncertainty resulting from comparison with calculational and experimental results

**Table 3. Calculated Multiplication Factors for Normal and Abnormal Condition of KNU 9 & 10.**

| | Case | $k_{eff}$ |
|---|---|---|
| Normal Concition | Standard | $0.71786 \pm 0.00571$ |
| | Eccentricity | $0.72448 \pm 0.00548$ |
| | Mechanical Tolerance | $0.72123 \pm 0.00512$ |
| | $k_s$(with all uncertainties) | 0.72998 |
| Abnormal Condition | Maplosition Accident | $0.72100 \pm 0.00449$ |

3) Eccentricity of fuel bundle location and dimensional tolerances.

Table 3 shows the $k_{eff}$ under normal condition and that including the eccentricity or dimensional tolerance. The evaluated multiplication factor, $k_s$ from Equation 5.2. is calculated to be 0.72998. In abnormal conditions, according to KENO-IV calculations for malposition model $0.72100 \pm 0.00449$ is obtained. The current spent fuel pool of KNU 9 & 10 accommodates 472 fuel assemblies (9 cores), which can not admit all discharged fuels during plant operation life. Thus the storage capacity should be expanded byconsolidating the fuel pins from 2 fuel assemblies into a cell. However, in this case, spent fuel pool must have the spaces to dismember assemblies and the cooling space to dissipate heat that does not exceed 6,600 W[2] for consolidated PWR fuel assemblies. So, cooling periods is evaluated to about 2 years. Also it should have spaces to accommodate the assemblies of a core in the case of accidents. The storage capacity of KNU 9 & 10 is calculated by the following equations in the unit of 1/3 cores.

$$Nc = \frac{N \times d^2 - Ac}{d_c^2} \times f \quad (5.3)$$

$$C = \frac{Nc - n_c - 2 \times n_b}{n_b} \times 2 + 5 \quad (5.4)$$

Where N: current number of racks in KNU 9 & 10, 472.

Ac: Fuel consolidation equipment space ($9.29\ m^2$)

f: Storage increase rate obtained by setting up additional racks between current rakcs and walls

d: rack pitch of no consolidated fuel

$d_c$: rack pitch of consolidated fuel

$n_c$; total assemblies in a core

$n_b$: the number of a replaced assemblies

$N_c$: the number of racks except consolidation equipment space

C: storage capcity in the unit of 1/3 cores

Now the storage capacity of KNU 9 & 10 by consolidation can be expanded from 9/3 cores into 27/3 cores, i.e., the number of stored fuel assemblies will be increased from 472 into 1423.

## VI. Conclusion and Recommendation

The calculation model based on a homogenized fuel module representation which is suitable for use in AMPX-KENO IV system is defined and verified by the benchmark calculation. The results of this analysis show that the neutron multiplication factor in the consolidated spent fuel pool of KNU 9 & 10 does not exceed the limit value of 0.95 under normal and abnormal conditions.

Therefore, it is concluded that the storage capcity is able to expanded from 9/3 cores into 27/3 cores in the standpoint of the criticality.

For further works, as shown in figure 7, the storage box thickness can be reduced into the smaller value to increase the safety. But the stress analysis of storage box should be performed for optimizing design of storage rack.

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