Journal of the Korean Nuclear Society
Volume 14, Number 2, June, 1982

《**Original**》

Criticality Safety Analysis of Spent Fuel Storage Facility for Ko-Ri Unit 1

Dong Ha Kim and Un Chul Lee

Seoul National University
(Received April 20, 1982)

핵연료 저장시설의 임계 안전성 분석

김 동 하·이 은 철

서 울 대 학 교
(1982. 4. 20 접수)

Abstract

In 1977, spent fuel storage capacity of Ko-Ri Unit 1 was raised to contain 4-2/3 core, by reducing the center-to-center spacing between fuel assemblies from 53.34cm to 36cm. In this paper the adequacy is discussed in detail by examining the previous design analysis report. According to the analytic method presented by Core Performance Branch, study on credible abnormal moderator density condition is performed by using KENO-IV for the redesigned spent fuel storage facility.

Result shows that 36cm for the center-to-center spacing between fuel assemblies is not enough to keep the storage safe at water density of 0.1143g/cm^3, which gives the maximum K_{eff} 0.9958$\pm$0.0048, which exceeds the CPB regulation limit 0.98.

From sensitivity study regarding to the center-to-center spacing, it should be maintained to space greater than 43cm in order to meet the CPB requirements.

요 약

1977년 집합체 cell 중심간의 간격을 53.34cm(162 연료 집합체 저장)에서 36cm(562 연료 집합체 저장)로 줄임으로 고리 1호기 부속 기사용 연료 저장 용량을 확장하였다. 확장된 저장 시설에 대하여 Monte Carlo방법을 이용한 KENO-IV코드로 Core Performance Branch에서 제시한 비정상적인 냉각수 밀도 조건에 따라 유효증배계수를 구하였다.

KENO-IV결과 밀도 0.1143g/cm에서 최대유효 증배계수 0.9958±0.0048을 얻었고, 이 값은 US NRC 기준과 CPB기준인 0.98을 초과하므로 새로운 집합체 cell중심간의 간격을 구하였다.

이 과정은 KENO-IV보다 보수적인 결과를 나타내는 확산 코드 KIDD를 이용하여 cell중심간의 간격에 따른 상관 연구로부터 새로운 cell중심간의 간격을 얻었다.

이로부터 현재의 집합체 cell중심간의 간격 36cm를 43cm 이상으로 늘려야 비정상적인 냉각수 밀도의 감소로 인한 사고의 경우에도 안전함을 알았다.

I. Introduction

The capacity of spent fuel storage of Ko-Ri Unit 1 was originally limited as 1-1/3 of core, which was equivalent to the maximum number of spent fuel assemblies unloaded from the first core during the refueling cycle plus the fuel assemblies contained in a full core load,[1] with center-to-center spacing between fuel assemblies of 53.34cm.[2]

However American nuclear policy had been changed not to allow reprocessing in other countries, which resulted in necessities of long term spent fuel storage. Thus expansion of original spent fuel storage facility was inevitable for at least 10 years. So in Ko-Ri plant, new facility was redesigned to store 4-2/3 of core, that was equivalent to 562 fuel assemblies, by reducing the center-to-center spacing between fuel assemblies from 53.34cm to 36cm to utilize the same space effecitively.

By the way, there are many points to be reviewed for the spent fuel storage design. The design criteria are quoted from USNRC Standard Review Plan[3] and Core Performance Branch.[4] They suggest that center-to-center spacing between fuel assemblies be sufficient to maintain the array in a subcritical condition,[5] and the maximum K_{eff} value shall not exceed 0.95 including total uncertainties such as transport correction calculational bias, and 95% confidence level. The design criteria also say that for postulated abnormal fuel or moderator distribution, the K_{eff} value shall not be greater than 0.98 even when all uncertainties are taken into account.

In this paper the adequacy of the spent fuel pool design is carefully examined for the variation of the abnormal moderator distribution. Core Performance Branch suggests this as the fifth assumption, which credible abnormal moderator density condition (e.g., due to loss of pool cooling) should be considered.

Here the KENO-IV[6] is adopted to calculate the effective multiplication factor for the facility. KENO-IV employs Monte Carlo method[7,8] to solve the neutron transport equation and has a tool of weighting to get the statististical accuracy. Also the diffusion code KIDD[9] is employed for sensitivity study of assembly spacing.

II. Model and Codes Description

II.1. Model Description

Rack array of expanded spent fuel storage pool is shown in Fig. 1. For conservatism, boron is excluded in pool cooling water and concrete and stainless steel liner form the boundaries. 562 fuel assemblies are submerged in the pool. Full geometry is modeled to calculate the effective multiplication factor for KENO-IV, which is drawn in Fig. 2.

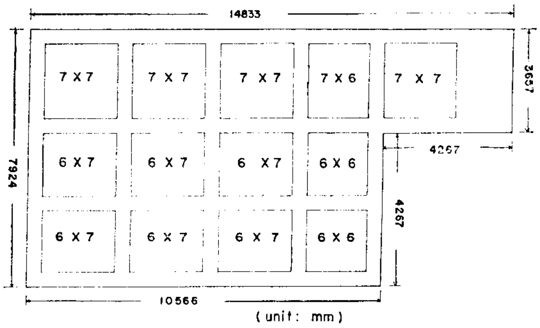

Fig. 1. Storage Rack Array of Ko-Ri Unit 1

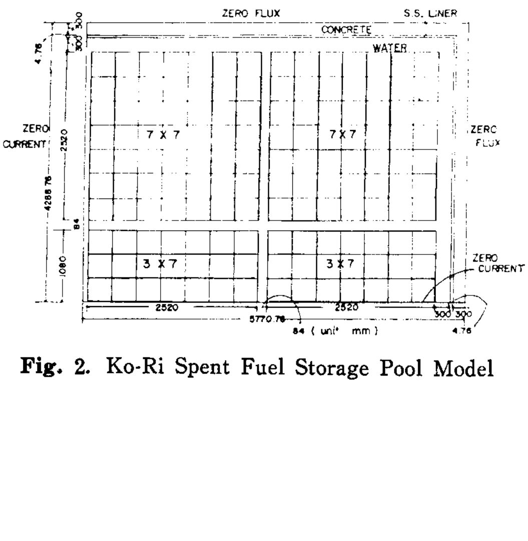

Fig. 2. Ko-Ri Spent Fuel Storage Pool Model

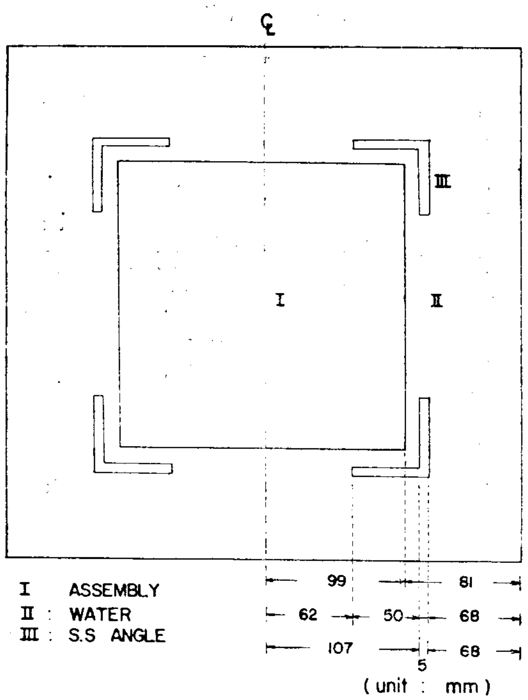

Fig. 3. Cell Description of Storage Pool

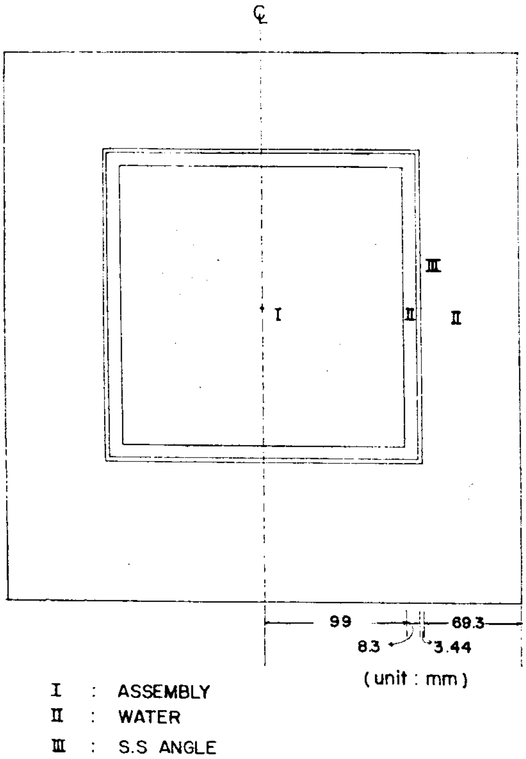

Fig. 4. Cell Model for KENO-IV

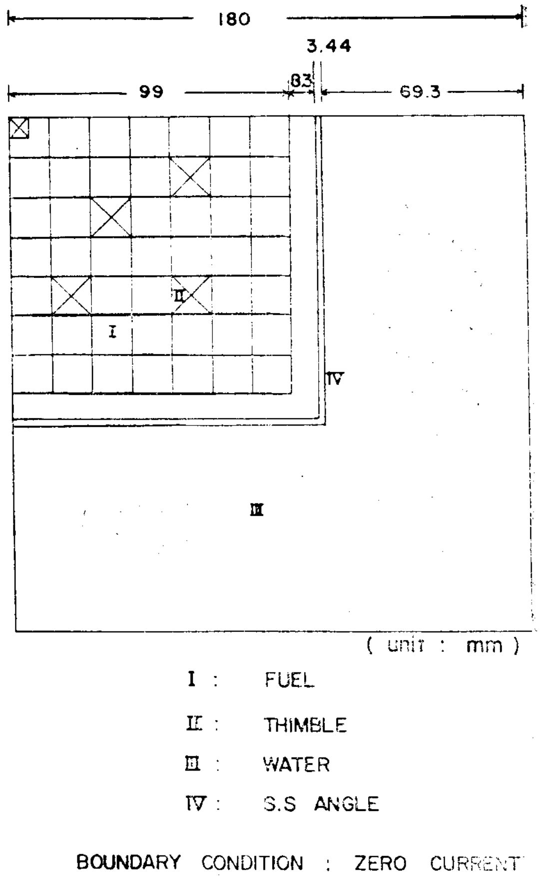

Fig. 5. 1/4 Cell Model for KIDD

Simplified cell description and modified cell model for KENO-IV are illustrated in Fig. 3 and Fig. 4, respectively. The fuel assembly is confined by stainless steel corner angle and the assembly pitch is assumed to be 36cm nominally. The rack is assumed to contain assemblies loaded with 3.2% enriched fuel without control rods which are filled with water instead. 1/4 cell model for sensitivity study with KIDD is shown in Fig. 5. Zero boundary condition is used to get the infinite multiplication factor, but this assumption meets the method which CPB suggested.

II. 2 Error Range and Comparison

KENO-IV is a three dimensional Monte Carlo criticality program. The starting point in developing any Monte Carlo simulation is the construction of mathematical models which describe the stochastic behavior of the variables in the process under study. So many neutron histories are needed to satisfy the probability theory and variance reduction techniques[10][11] such as splitting and Russian Roulette to obtain the statistical accuracy.

First, consider the uncertainties of KENO-IV. As described earlier, total uncertainties consist of transport correction, calculational bias, and 95% confidence level. Benchmarking test[12] with KENO-IV in case of critical homogeneous $U(1.4)F_4$ and paraffin with 16-group Hansen-Roach cross section[13] and history of 30,000 reduces the K_{eff} value of 0.99156 ± 0.00417. So calculational bias is $0.00844 \Delta K$ and 95% confidence level has the value of $0.00834 \Delta K$ from standard deviation of $0.00417 \Delta K$. Therefore the total uncertainties that KENO-IV could imply are up to $0.01678 \Delta K$.

Next, it is necessary to compare the differences between the two codes, KENO-IV and KIDD for analyzing the same test cases. So infinite multiplication factor is calculated to find the differences for a 14×14 Ko-Ri Unit 1 assembly with 3.2% enriched fuel.

At full water density the value of 1.312 ± 0.0053 and 1.3995 are obtained from KENO-IV and KIDD, respectively. The result of KIDD is 5.5% higher than that of KENO-IV with 95% confidence level. These differences might come from cross section library, fission spectrum, and the solving method. However, equal discrepancies are found in other situations, which would give the similar trends to both calculations.

III. Discussion and Conclusion

It is generally known that the KENO-IV with 16-group Hansen-Roach cross section is adequate, when calculating homogenizing K_{eff} of PWR

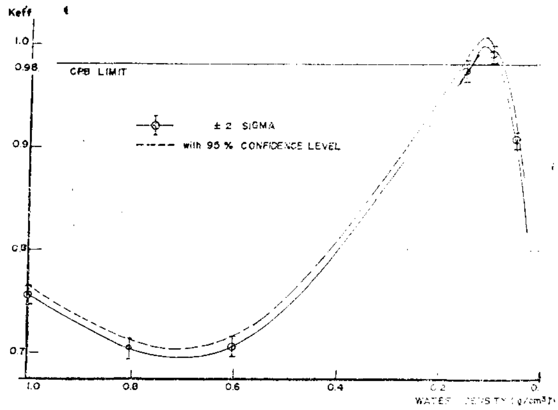

Fig. 6. K_{eff} vs Water Density in Spent fuel Storage Array

fuel assemblies containing uranium oxide fuels (consistently overpredicted by an average of 2.6%)[14].

Now look into the result derived from various conditions regarding to water densities with 15,000 neutron histories. Fig. 6 shows that the K_{eff} value varies from 0.756 ± 0.0041 at normal density, minimum at about 0.65 g/cm³, and gets to the maximum at density 0.11 g/cm³, and then falls sharply again. This trend shows how the moderator density reduction affects the multiplication factor.

Now examine the relations between moderation effects and absorption reduction. KENO-IV uses resonance corrected cross sections that are to compensate for the self-shielding of incident neutrons due to the nature of materials. For homogeneous system, cross section sets included in KENO-IV library are selected according to this resonance corrected cross sections. Moderator number density as well as fuel number density is considered to get resonance corrected cross sections. So moderator number density has an influence on the selection of cross section sets and that alters the values of thermal utilization factor and resonance escape probability, etc. At last it can be said that multiplication factor depends on the relative contribution of moderation and absorption effect which are sensitive to moderator number density.

As shown in Fig. 6, this tendancy is proven that at density above 0.65g/cm^3, absorption effect is greater than moderation effect, so K_{eff} becomes low, while below that density, moderation effect is dominant over absorption effect, then increasing the K_{eff} to the maximum. This phenomena is also proven in case of Commercial Nuclear Plant of B&W.[15]

To find the peak point in this curve, we get relation for the K_{eff} values and water densities of 0.5, 0.15, 0.1 and 0.05g/cm^3. From polynomial fitting, the empirical formula is derived;

$$Y=43.2068X^3-33.424X^2+5.94668X+0.68825$$

where X is water density, and Y is K_{eff}. The maximum value calculated from the above fitting is estimated as 0.9958 at water density of 0.1143g/cm^3. With considering total uncertainties of KENO-IV, the maximum value becomes 1.0126, which surely exceeds the regulation limit 0.98.

As the center-to-center spacing is one of the main parameters which affect the maximum value even at the same water density, KIDD is used to obtain the new center-to-center spacing. The sensitivity study, regarding the spacing from 22.144cm to 46.392cm is performed under the condition of 22 psia and 77°F which satisfies the CPB requirements.

The maximum K_{eff} is expected to occur in the water density range of 0.1 and 0.2g/cm^3 [16] and this study confirms the KENO-IV results. So sensitivity study is carried out at water densities of 0.1 and 0.15g/cm^3. The result is shown in Fig. 7. Infinite multiplication factor of redesigned structure with center-to-center spacing of 36cm is lower than the regulation limit at normal density, but it exceeds the limit 0.98 at density 0.1g/cm^3. Multiplication factor should be limited to the value of 0.98 under normal and postulated accident conditions including uncertainties of 0.02ΔK which are originated from

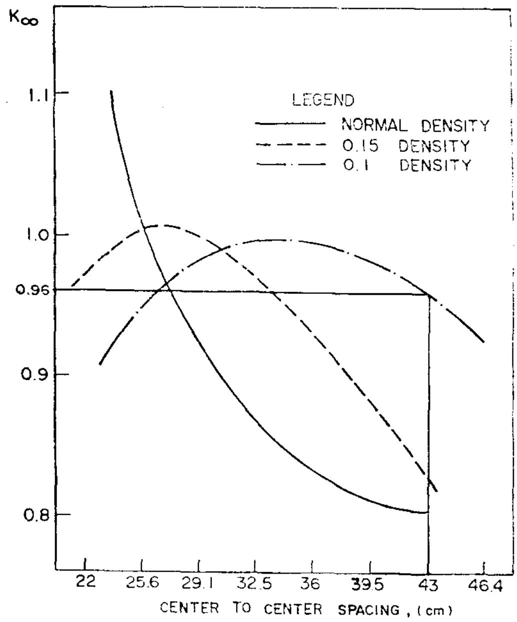

Fig. 7. K_∞ vs Center Spacing between Fuel Assembilies.

KENO-IV and new spacing can be extracted from the curve of water density 0.1g/cm^3.

As presented in Fig. 7, the minimum spacing between fuel assemblies should be greater than 43cm to make the K_{eff} lower than 0.98. So it should be maintained to space greater than 43cm in order to meet the CPB requirements.

References

1. US Nuclear Regulatory Commission Standard Review Plan, Section 9.1.2, Spent Fuel Storage, Ⅲ.1, NUREG-75/087 (1978).
2. "Spent Fuel Storage Capacity Expansion Services for Ko-Ri Unit No. 1", Korea Nuclear Engineering (1978).
3. NUREG-75/087, op. cit.
4. Branch Technical Position CPB 9.1-1, "Criticality in Fuel Storage Facilities".
5. 10 CFR 50, Appendix A, Criterion 62, USNRC (Jan., 1979).
6. L.M. Petrie and N.F. Cross, "KENO-IV-An Improved Monte Carlo Criticality Program", ORNL-4938, Union Carbide Corporation (1975).

7. J. Spanier and E.M. Gelbard, "Monte Carlo Principles and Neutron Transport Problems", Addison-Wesley Publishing Company, Inc. (1969).
8. Herman Kahn, "Application of Monte Carlo," AECU-3259, RAND Corporation (1954).
9. S.K. Lee, K.S. Moon, B.J. Jun, J.H. Chang, and J.B. Lee, "KIDD, A KAERI Improved Diffusion Depletion Program for Nuclear Reactor Analysis", Korea Advanced Energy Research Instituie, to be published.
10. L.L. Carter and E.D. Cashwell, "Particle Transport Simulation with the Monte Carlo Method," TID-26607 (1975).
11. E.J. McGrath and D.C. Irving, "Technique for Efficient Monte Carlo Simulation", Vol. III, Variance Reduction, ORNL-RSIC-38 (1975).
12. G.R. Handley and C.M. Hopper, "Validation of the KENO Code for Nuclear Criticality Safety Calculation of Moderated, Low Enriched Uranium System," Y-1948, Oak Ridge Y-12 Plant (1974).
13. G.E. Hansen and W.H. Roach, "Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies", LAMS-2543, Los Alamos Scientific Laboratory (1961).
14. William Bromley, "Safety Calculation and Benchmarking of PWR Fuel Storage Pool Studies", *Trans. Am. Nucl. Soc.*, **27**, 408 (1977).
15. F.M. Alcorn, "Criticality Evaluation of Low-Density Moderation in PWR Fuel Storage", *Trans. Am. Nucl. Soc.*, **27**, 416 (1977).
16. J.T. Brantley, et al., "Criticality Analysis of Dry Fuel in ANO-2 Fuel Storage Pool", *Trans. Am. Nucl. Soc.*, **27**, 421 (1977).