Shielding and Criticality Safety Analysis of KSC-1 Cask for the High Burnup PWR Spent Fuels

Hyoung-Mun Kwon*, Jung-Nam Jang, Yong-Hwa Hwang, In-Chan Kwon, Duck-Kee Min, Yong-Bum Chun
Korea Atomic Energy Research Institute, 1045 Daedeok-daero, Yuseong-gu, Daejeon 305-353, South Korea
**Corresponding author: django@kaeri.re.kr*

1. Introduction

KSC-1 (KAERI Shipping Cask-1) was designed and manufactured with a pure domestic technology in 1985 in order to transport a PWR spent fuel assembly from nuclear power plant to PIEF (Post-Irradiation Examination Facility) of KAERI. Since the first transportation of the fuel assembly from Kori-1 NPP was carried out by the cask in 1987, 19 shipments for the PWR spent fuels have been done successfully by now.

Maximum discharge burnup of PWR in Korea has been extended from the late 1990s in order to reduce the cost of power generation. From this cause, allowable design values of the initial enrichment and the cooling time for the cask have been changed three times: year 2003, 2007 and 2010.

Radiation shielding and criticality of KSC-1 were analyzed for all the PWR fuel type irradiated in Korea NPP to renew the design approval.

2. Methods and Results

In this section methods and results of shielding and criticality safety analysis for KSC-1 are described.

2.1 KSC-1

KSC-1 is wet-type cask for shipping one PWR fuel assembly. It consists of cask body, basket, impact limiter and tie-down device. The cask body is divided into three parts by cylindrical stainless steel shells. The inner part includes spent fuel assembly, basket and cooling water. Lead shields gamma radiation emitted from a spent fuel in the middle part. The outside part contains a mixture of 50% ethylene glycol and 50% water, which provides freezing protection and the function of neutron shielding[1]. Figure 1 shows the KSC-1 schematic drawing.

The safety analysis of KSC-1 was performed for three categories of shipping contents as in table I.

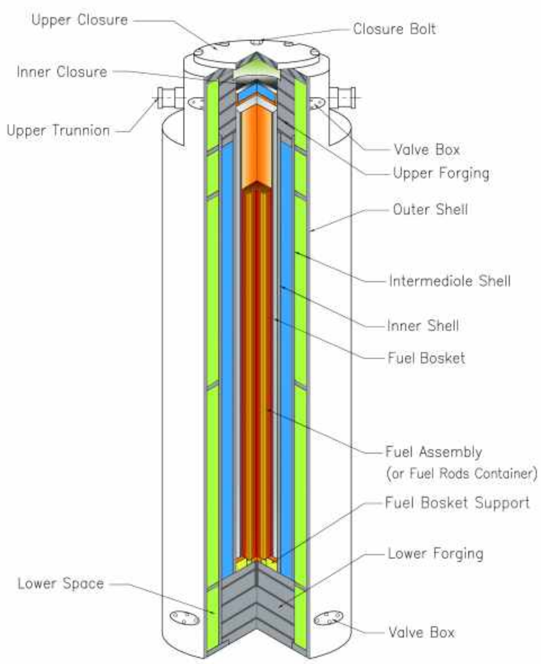

Fig. 1. KSC-1 cask schematic drawing

Table I: Design values of loading condition

Content	Design values of loading condition		
	One conventional fuel assembly	One high burnup fuel assembly	12 high burnup fuel rods
Design basis fuel	14x14 WH STD 16x16 WH STD 17x17 WH STD	14x14 KNF OFA 16x16 KNF SFA 17x17 KNF V5H	16ACE7, 17ACE7 17RFA, Guardian, PLUS7
Maximum discharge burnup, GWd/tU	45	52.5	70
Minimum cooling time, year	1.0	1.5	0.5
Max. decay heat, kW	6.02	4.78	0.67
Maximum enrichment, wt% U-235	3.5	4.2	5.0
Radioactivity, PBq	51,7	41.9	5.39

2.2 Radiation Shielding

Neutron and gamma radiation sources were evaluated by ORIGEN-S[2] code, and the shielding analysis was performed by MCNP[3] code after selecting fuel types with higher radiation source.

According to the fuel assembly structure and the material, radiation source analyses were done for fuel/grid, plenum, top nozzle and bottom nozzle. The scale factor applied to the source analysis for hardware except UO_2 fuel as in table II.

The analysis aims to evaluate the shielding performance of KSC-1 in the normal transport and the hypothetical accident condition as compared with Korean regulatory guidelines. The accident condition was assumed as the loss of the neutron moderator.

As results of the analysis, maximum surface dose rates were 0.241 mSv/h and 0.304 mSv/h for V5H and WH-STD type in normal transport. Maximum dose rates at 1 m distance from the surface were 0.661 mSv/h and 0.702 mS/h for V5H and WH-STD in accident condition.

Table II: Scale factor

Region	ORNL[4]	PNL[5]
Top end fitting	0.011	0.1
Gas plenum	0.042	0.2
Bottom end fitting	0.011	0.2

2.3 Criticality

The criticality analysis was carried out by CSAS26 control module within SCALE 5.1. 238-group ENDF/B-VI cross section library(V6-238) was used for this analysis.

The concerns of conservative assumption for the modeling of the criticality analysis are as follows;

- Unirradiated fresh fuel
- 20 °C at all the material
- Burnable poison rod and axial blanket are not considered
- Ignorance of dishing and chamfering of the fuel pellets
- Water is substituted for hardware
- Usage of manufacturing tolerance to maximize the reactivity

USL (Upper Subcritical Limit) was determined as 0.93858 by using the criticality experiment data from NUREG/CR-6361[6] in accordance with ANSI/ANS-8.17[7]. As the results, effective k-values for 11 fuel types were evaluated below USL.

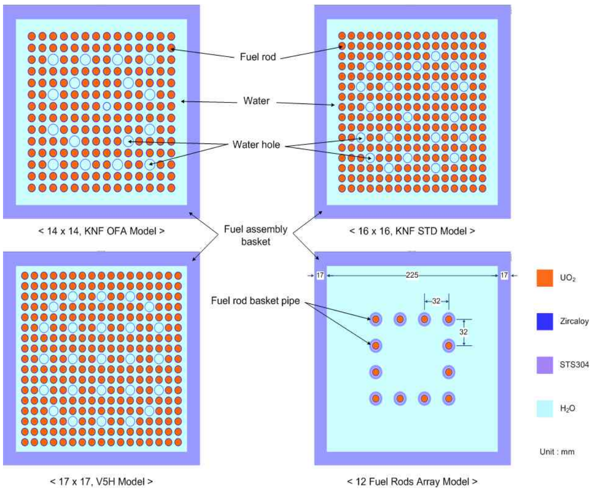

Fig. 2. Cross section view of geometrical model for fuel assembly type (criticality analysis)

3. Conclusions

In order to renew the design approval of KSC-1, shielding and criticality safety analysis were performed for high burnup PWR spent fuels. KSC-1 cask was evaluated to satisfy safety requirements of the radiation shielding and the criticality.

REFERENCES

[1] Eonho Gwak et al., Safety Analysis Report of KSC-1 Shipping Cask, KAERI/TR-77/85, 1985.

[2] I.C. Gauld, O.W. Hermann, and R.M. Westfall, "ORIGEN-S: Scale System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," ORNL/TM-2005/39, Ver. 6, Vol. II, Sect. F7, Oak Ridge National Laboratory, 2009.

[3] MCNP - A General Monte Carlo N-Particle Transport Code, Version 5, LA-UR-03-1987, Los Alamos National Laboratory, 2005.

[4] A.G. Croff, et al., "Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code," ORNL/TM-6051, Oak Ridge National Laboratory, 1978.

[5] S.M. Short, et al., "Spent Fuel Hardware: Characterization and 10CFR61 Classification for Waste Disposal: Volume 2, Calculated Activity Profiles of Spent Nuclear Fuel Assembly Hardware for Pressurized Water Reactors," PNL-6906-Vol.2, Pacific Northwest Laboratory, 1989.

[6] J. J. Lichtenwalter, S. M. Bowman, M. D. DeHart, and C. M. Hopper, "Criticality Benchmark Guide for LWR Fuel in Transportation and Storage Packages:, NUREG/CR-6361, ORNL/TM-13211, March 1997.

[7] American National Standard for Nuclear Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors, ANSI/ANS-8.17-1984.