Decay Heat and Dryout Behavior of Spent Fuel Storage Pool with ORIGEN-ARP and MARS codes for the Station Blackout Accident

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1. Introduction

Spent nuclear fuels are stored in spent fuel storage pool (SFP) in nuclear power plants. SFP should be designed and operated to prevent the spent fuels from being critical and have a shielding capability against radiation. Borated water is usually used to prevent the fuel from being critical and provide radiation shielding. Borated water is also used for removal of the decay heat from the spent fuels. Since the fuels may be expected to be fail without cooling, the SFP should be maintained its temperature lower than safety limit. Electric power is always required for the SFP cooling system during all modes of operation to maintain cooling capability of SFP water. In this paper, we performed analysis of decay heat and dryout behavior of spent fuel pool for the station blackout accident (complete loss of AC power). The accident can be regarded as a most challenging one to the SFP and its support system. As a reference, SFP of Ulchin Unit 3 and its state of maximum storage is considered.

2. Decay Heat Calculation

Decay heat of spent fuels is calculated by using ORIGEN-ARP code [1] which is based on ORIGEN-S code for calculations of fuel depletion, actinide transmutation, fission product buildup and decay, and associated radiation source terms. Decay heat can be calculated from initial isotope inventory (just after reactor shutdown) and decay chain calculation during cooling time. To obtain initial isotope inventory lattice code calculation can be used which might take long computation time. In this paper, we used pre-calculated library for CE16x16 fuels in ORIGEN-ARP code with functions in fuel enrichment, burnup (power history), and moderator density. Because some nuclear fuels are loaded into reactor core twice and some are three times usually, burnup calculations of two fuel types are needed. Among 64 new nuclear fuel assemblies, we assumed 49 fuel assemblies are loaded into core three times and 15 fuel assemblies are loaded twice. Power density of nuclear fuel depends on its position in core and burnup. For simple analysis, we used same power density and fuels with same depletion time during same core cycle. Power density used is assumed as shown in Table 1. The depletion time for each core cycle and cooling time between core cycles are assumed 455 days and 30 days, respectively. Fig. 1 shows calculated decay heat of nuclear fuel.

Table 1. Power history used in burnup calculation

	Power density(W/g)	Burnup (MWD/MTU)
First cycle	44.58	20282
Second cycle	41.26	39056
Third Cycle	20.63	48443

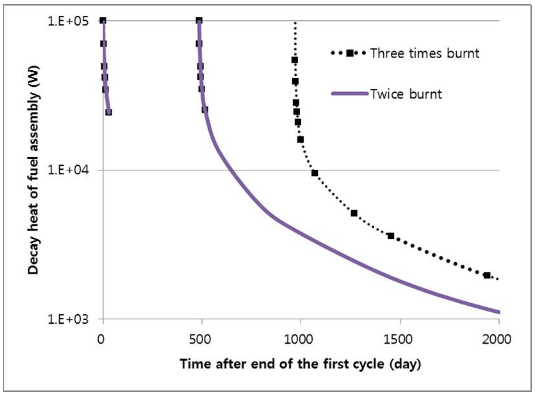

Fig.1 Decay heat of one fuel assembly

Table 2. Number of fuel assemblies with same burnup and cooling time in the SFP

cooling time (days)	20282 MWD/MTU	39056 MWD/MTU	48443 MWD/MTU
7	64	64	49
485		15	49
970		15	49
Every 485 days for 1455~7275 days		15 (x13)	49 (x13)
7760		15	49
8250		82	200

Decay heat of SFP depends on the number of fuel assemblies stored and their cooling time. SFP with 1498 fuel storage capacity is considered in this calculation. In this paper, we assumed SFP is full of 1498 fuel assemblies including 177 fuel assemblies (64 for once burnt, 64 for twice burnt, and 49 for three times burnt) transferred from reactor core with seven days cooling time. For every core cycle, 64 fuel assemblies (49 for three times burnt and 15 for twice burnt) are assumed to be discharged from reactor core. Table 2 shows the number of fuel assemblies with same cooling time and burnup. As shown in Fig.1, decay heat decrease as cooling time increase and decay heat with large enough cooling time might be very small. Thus we do not consider cooling time after 8250 days and assumed 282 fuel assemblies with longer cooling time have 8250 cooling time. The calculated decay heat at this point is 8.83MWth and decreases as shown in Fig. 2.

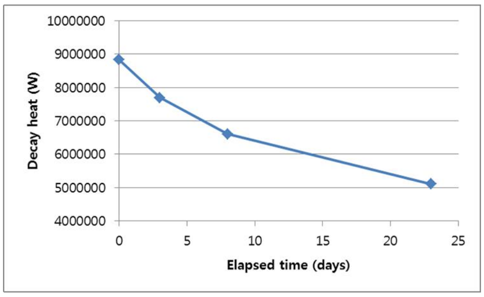

Fig.2. Decay heat of spent fuel storage pool

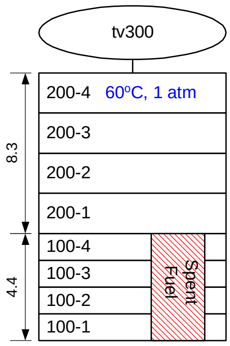

Fig. 3 MARS modeling of spent fuel storage

3. Thermal-Hydraulic Calculation

Dryout behavior of fuel cladding under borated water condition was calculated by using MARS code [2]. The spent fuel storage pool is modeled as shown in Fig.3. Rack region which contains fuels and support racks and pool region were modeled by 4 volumes, respectively. The initial water temperature was assumed 60°C which is the maximum temperature of SFP during the normal power and refueling operation. Atmosphere above the pool was assumed to be a saturated one condition. As described above, two cases of decay heat were used: (1) case of constant heat, 8.83 MWth and (2) case of actual decay heat (Fig. 2). Decay heats are interpolated in two time intervals (0~3 days and 3~8 days).

The calculated water level and fuel cladding temperature are shown in Figures 4 and 5, respectively. The uncovery of fuel was found at 45 to 50 hours after accident and the cladding heatup was at 50 to 55 hours. And the difference in timing of heatup between the case 1 and case 2 was less than 5 hours. The significant increase of cladding temperature after heatup was due to rapid oxidation of the cladding.

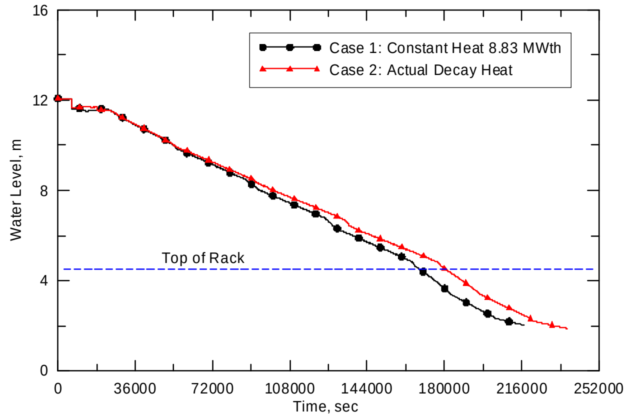

Fig.4 Comparison of water level

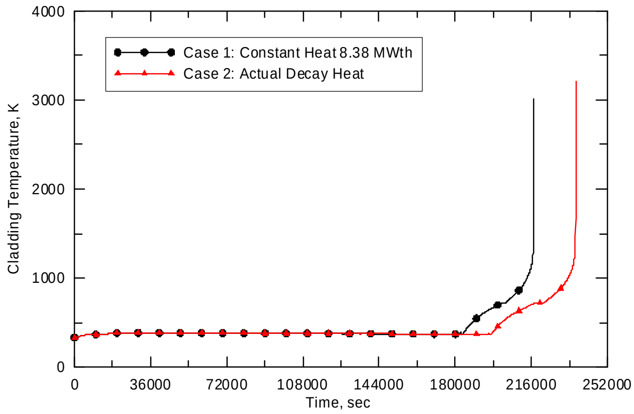

Fig.5 Comparison of cladding temperature

4. Conclusions

Decay heat and dryout behavior of spent fuel pool for the station blackout accident was analyzed with the ORIGEN-ARP and MARS codes. The spent fuel pool of Ulchin Unit 3 at its maximum status of storage was considered. The calculated decay heat with 1498 spent fuel assemblies including 113 fuel assemblies to be reloaded to the reactor core was 8.83 MWth and decreases to 6.61 MWth after 8 days. Uncovery of the fuel was started at 45 to 50 hours after accident and the cladding heatup was followed. Thus, the cooling capability should be recovered considering this analysis result.

REFERENCES

[1] SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, ORNL/TM-2005/39 Version 6, January 2009
[2] KAERI, MARS Code Manual, Volume I: Code Structure, System Models, and Solution Methods, KAERI/TR-2812/2004, December 2009.