Transactions of the Korean Nuclear Society Autumn Meeting
Goyang, Korea, October 24-25, 2019

Activation Evaluation on Reactor Pressure Vessel, Internal and Bio-Shield Applying Monte Carlo Methodology

Hee Dong Sohn [a*], Kwang Soo Park [a], Gyu Ho Jang [a], Chang Gyu Kim [a], Chang Je Park [b]
[a]*Doosan Heavy Industries and Construction, 22 Doosan Volvo-Ro, Seongsan-Gu, Changwon-Si, Gyeongsangnam-Do, 647-792, Republic of Korea*
[b]*Sejong University, Nuclear Engineering Department, 209 Neungdong-ro, Gwangjin-Gu, Seoul 143-747*
[*]*Corresponding author: heedong.sohn@doosan.com*

1. Introduction

In nuclear power plant, most activated components and structures are reactor pressure vessel, reactor vessel internal and bio-shield. Caused by neutron absorption reaction - (n, γ), (n, p), (n, 2n), and (n, α), these components irradiated by high intensity of neutrons for the period of nuclear power plant operation are activated.

Generally, the material for reactor vessel is carbon steel with stainless steel cladding inside wall and insulation outside wall of it, for reactor vessel internal is stainless steel and Inconel, for bio-shield is concrete.

When nuclear power plant is decommissioned, before actual dismantling activities are begun at site, activation evaluation should be done to prepare cutting plan, equipment design, radioactive waste quantity calculation, and to distinguish the waste level.

Monte Carlo methodology based on MCNP [1] and ORIGEN codes are used to calculate neutron flux on each position of components and specific activity after some years of cooling time. And finally, activation evaluation data is produced.

The main nuclides to impact activation of material – carbon steel, stainless steel and concrete – are Co^{60}, Ni^{59}, Ni^{63}, Fe^{55}, Mn^{54}, Nb^{94} for carbon and stainless steel [2], H^3, C^{14}, Co^{60}, Mn^{54} for concrete.

2. Methods and Results

2.1 Methodology

The exact method to analyze activation characteristics on each position of materials, measurement for the specimen collected from the original body is the best way. However, it does not guarantee the representativeness for the components and structures. So, computer program, especially, Monte Carlo methodology based MCNP and ORIGEN-S are widely used for neutron source term, neutron flux and specific activity calculations.

2.2 Geometry Description

2.2.1 Fuel Assembly

Westinghouse designed nuclear power plant which has full core loaded with 121 fuel assemblies of 14 x 14 type. It is composed of 179 fuel rods including UO_2 pellets with Zircaloy-4 clad, 16 guide thimble tubes and 1 instrumentation tube with Zirconium material. Fuel data on each cycle for U^{235} initial enrichment, power density, burn-up, etc., are referenced with NDR report. It is assumed that nuclear power plant has been operated for 40 years with 32 cycles.

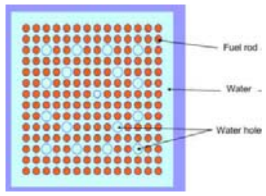

Fig. 1. Westinghouse Type 14 x 14 Fuel Assembly

2.2.2 Reactor Pressure Vessel, Internal and Bio-shield

Reactor vessel is the component enclosing internal structures and fuel assemblies. It is composed of shell, top/bottom heads, 2 inlet / 2 outlet nozzles and 33 ICI nozzles. Material is forged carbon steel except ICI nozzles with Inconel. There is stainless steel cladding inside wall of it and insulation outside of it.

Reactor vessel internal is the nearest component to fuel assemblies with supporting them. It is composed of stainless steel material with some Inconel parts and divided into upper and lower section. In case of Westinghouse type PWR nuclear power plant, the representative lower one is baffle, former, core barrel and thermal shield. Upper internal is composed of guide tube assembly, lots of support columns and plates, etc.

Bio-shield is the concrete structure surrounding all reactor pressure vessel and internal components. The shape is rectangular parallelepiped and it is located below reactor cavity structure. There are 8 ex-core detectors inside bio-shield with arraying circumferential direction around reactor vessel which is empty space with detectors.

2.2.3 Chemical composite composition for materials

Chemical composite composition for all materials – stainless steel, carbon steel, Inconel and concrete – applies the data described in NUREG/CR-3474 [3] which is the standard one and includes information of impurities as exactly as possible.

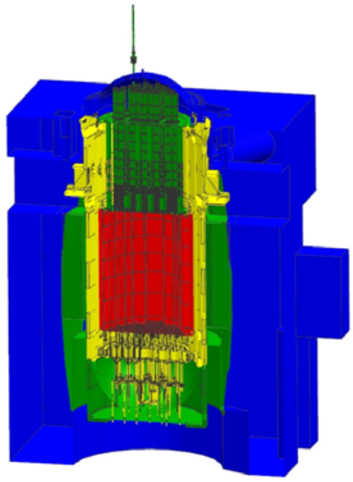

Fig. 2. 3D Model on Reactor Pressure Vessel, Internal and Bio-shield

2.3 Results and Discussion

2.3.1 Neutron Flux Calculation

Time dependent neutron flux is calculated on each position using MCNP program. First, neutron source term should be calculated. Full core 121 fuel assemblies are divided into two regions, the one is burned fuels, the other is newly loaded fuels. In this study, averaged initial enrichments and burn-up conditions are applied for all burned fuels, and specific characteristics are applied for newly loaded fuels. So, from the 1st cycle to 9th cycle, the outer region is newly loaded fuels and the inner regions are burned fuels, and from 10th cycle to 32nd cycle, vice versa.

As the result of neutron flux calculation on each components of reactor pressure vessel, internal and bio-shield, high neutron flux can be obtained for the components close to the reactor core to the range of ~ 10^{13} neutrons/cm^2.sec, and it is decreased as the distance is far from the core as depicted in Fig. 3.

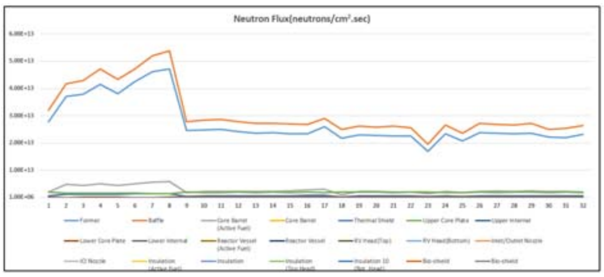

Fig. 3 Neutron Flux Calculation on Each Component

2.3.2 Specific Activity calculation

Based on neutron flux, specific activity should be calculated to determine the level of activity on each component [4]. Using ORIGEN-S program with flux irradiation option, carbon steel, stainless steel, Inconel, and concrete are irradiated by neutron flux for 32 cycles, and 10 years of cooling time. The relation between production and destruction for specific activated nuclides is expressed like Bateman equation below.

$$\frac{dN_i}{dt} = \text{Formation Rate} - \text{Destruction Rate} - \text{Decay Rate}$$

The representative activated nuclides on each material are H^3, C^{14}, Co^{60}, Ni^{59}, Ni^{63}, Sr^{90}, Nb^{94}, Tc^{99}, I^{129}, Cs^{137}, Eu^{154}, Mn^{54}, Mo^{93}, Fe^{55}. Likewise neutron flux, on the components close to the fuel assemblies, maximum specific activity reaches to the level of ~10^9 Bq/g. Table I shows the specific activity for each nuclide on each component.

Table I. Specific Activity on Each Nuclide

Nuclide	Specific Activity (Bq/g, Cooling Time : 10 Years)												
	Baffle	Core Barrel (Active Fuel)	Thermal Shield	Upper Core Plate	Upper Internal	Lower Core Plate	Lower Internal	Reactor Vessel (Active Fuel)	Reactor Vessel Head (Top)	Reactor Vessel Head (Bottom)	Insulation (Active Fuel)	Bio-shield (Very Low)	Bio-shield (Clearance)
H-3	2.16E+04	5.89E+03	1.59E+03	4.56E+03	2.20E-03	2.31E+02	4.63E-01	2.44E+01	1.03E-06	2.20E-03	3.04E-01	9.16E+01	2.60E+00
C-14	2.49E+04	2.26E+03	5.66E+02	1.52E+03	8.34E-04	8.27E+01	1.13E-01	8.98E-01	1.86E-08	8.34E-04	1.42E-01	4.57E+00	2.02E-01
Co-60	4.30E+08	7.54E+06	1.65E+06	5.94E+06	2.89E+00	2.95E+05	8.80E+02	8.23E+02	6.03E-05	2.89E+00	3.32E+02	2.74E+00	3.50E-02
Ni-59	5.67E+06	1.20E+06	3.25E+05	8.38E+05	4.92E-01	4.86E+04	6.67E+01	1.88E+02	3.88E-06	4.92E-01	8.34E+01	6.63E-04	2.96E-05
Ni-63	1.11E+09	1.28E+08	3.25E+07	8.83E+07	4.91E+01	4.87E+06	7.08E+03	1.85E+04	4.24E-04	4.91E+01	8.11E+03	6.96E-02	2.97E-03
Sr-90	1.77E-03	1.45E-05	9.11E-07	6.96E-06	2.07E-18	2.06E-08	8.46E-14	7.61E-11	0.00E+00	2.07E-18	6.24E-14	3.75E-11	3.62E-13
Nb-94	3.63E+02	5.02E+01	1.30E+01	3.45E+01	1.95E-05	1.93E+00	2.64E-03	2.37E-02	4.66E-10	1.95E-05	3.29E-03	3.33E-04	1.48E-05
Tc-99	1.06E+02	1.91E+01	5.10E+00	1.33E+01	7.86E-06	7.77E-01	1.06E-03	9.77E-06	1.91E-13	7.86E-06	1.33E-03	1.11E-05	4.92E-07
I-129	3.47E-09	1.77E-12	1.06E-13	8.00E-13	2.29E-25	2.26E-15	4.10E-21	4.12E-18	0.00E+00	2.29E-25	6.39E-21	5.38E-20	9.66E-23
Cs-137	1.28E+00	2.14E-03	4.00E-05	5.84E-04	1.07E-22	1.06E-07	2.24E-15	1.45E-11	0.00E+00	1.07E-22	1.26E-15	1.41E-14	6.19E-18
Eu-154	9.93E+00	1.39E+02	4.87E+01	1.32E+02	7.18E-05	7.43E+00	1.73E-02	5.49E-01	2.84E-08	7.18E-05	9.21E-03	5.53E-02	1.23E-03
Mn-54	6.45E+05	5.40E+04	1.40E+04	4.93E+04	2.61E-02	2.58E+03	1.79E+01	2.03E+02	3.12E-05	2.61E-02	3.57E+00	2.37E-01	1.44E-03
Mo-93	1.49E+03	1.47E+02	3.72E+01	1.00E+02	5.52E-05	5.48E+00	7.47E-03	6.86E-05	1.35E-12	5.52E-05	9.33E-03	7.75E-05	3.44E-06
Fe-55	1.14E+09	9.93E+07	2.51E+07	8.73E+07	4.63E+01	4.68E+06	1.90E+04	3.15E+05	3.64E-02	4.63E+01	4.97E+03	5.66E+02	3.00E+00

3. Conclusions

Activation calculation using Monte Carlo Methodology is necessary to evaluate the status of activity for components around reactor core. Assuming 40 years operation with 121 fuel assemblies, ~ 10^{13} neutrons/cm^2.sec of neutron flux and ~10^9 Bq/g of specific activity can be obtained as maximum value. And based on the result of specific activity for each nuclide, we can distinguish the waste level on target components, and finally decide the disposal plan when it is decommissioned.

REFERENCES

[1] INEL-95/0419, Use of MCNP for Characterization of Reactor Vessel Internals Waste from Decommissioned Nuclear Reactors, E.F.Love, K.A.Pauley, B.D.Reid, Idaho National Engineering Laboratory, U.S. DOE, 1995.

[2] Andreas Loeb, Dieter Stanke, Lutz Demp, Decommissioning of the reactor pressure vessel and its peripheral facilities of the Nuclear Power Plant in Stade, Germany, WM2011 Conference, 2011.

[3] J.C.Evans, E.L.Lepel, R.W.Sanders, C.L.Wilkerson, W.Silker, C.W.Thomas, K.H.Abel, D.R.Robertson, Long-Lived Activation Products in Reactor Materials, NUREG/CR-3474, 1984.

[4] HakSoo Kim, SungHwan Chung, SungJun Maeng, Comparative Evaluation of Cutting Methods of Activated Concrete from Nuclear Power Plant Decommissioning, WM2013 Conference, 2013.