Heat removal analysis in the AP1000 reactor’s refuelling process

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Abstract. A study has been conducted to find out how much heat removal in the refueling process in AP1000 reactor. The background of this research is, operation at a Nuclear Power Plant is strongly tied to periodical fuel replacement to guarantee future operations. The replacement of fuel assembly can be done by one-third of core or a half of core refueling. After the reactor shutdown, the spent fuel assemblies still generate decay heat that must be removed to prevent from fuel damage cause of heat buildup. The process of heat removal can be done by natural and forced convection modes. The purpose of this study was to obtain the cooling times after shutdown so that the fuel assembly can be transferred to the spent fuel storage pool (SFSP) with natural convection mode, and how minimum flow rate was required for cooling the 7 days spent fuel assembly by forced convection cooling in the SFSP. The requirement in the cooling was no boiling occurred in the channel. The calculation was carried out by assuming the inlet coolant temperature of 30°C and by using COOLOD-N2 code. The calculation results showed that for the one-third of core refueling and for cooling with natural convection, it took 21 days after shutdown, the new spent fuel assemblies could be transferred to SFSP, while for the forced convection cooling required minimum flow rate of 15 kg/s in the SFSP. The calculation result on the hot channel showed the outlet coolant temperature of 101.88°C, the maximum temperature of the outer cladding and the center fuel meat of 104.63°C and 105.21°C, respectively. As for a half of core refueling and for cooling with natural convection, it took 42 days after shutdown, the new spent fuel assemblies could be transferred to SFSP, while for the forced convection cooling required minimum flow rate of 25 kg/s in the SFSP. Calculations on the hottest channel showed the outlet coolant temperature of 90.54°C, the maximum temperature of outer cladding and the center fuel meat of 93.16°C and 93.75°C, respectively.

1. Introduction

Operation at a Nuclear Power Plant (NPP) is strongly tied to periodical fuel replacement to guarantee future operations. Technically, this process is known as Refueling Outage. Refueling outage, besides facilitating fuel replacement, allows for multiple maintenance and revision tasks to be carried out in equipment that might otherwise not permit inspection during normal operation. Each refueling process must be studied and planned in detail, such as to establish the duration, number of tasks to execute, and the contracting of technicians needed to carry out the process. During the tasks involved in refueling, a third of amount of the fuel assemblies housed in the reactor vessel are extracted by a mechanical arm on a platform, and transferred in pools to the transfer tube, to be sent to used-fuel storage pools. The process of loading new fuel follows the same path, yet inversely [1].
A study of heat removal capability in spent fuel storage pool (SFSP) was required as the background of this study. After the NPP reactor was operated at a certain time (about 12 to 18 months [2]), the reactor was shutdown for maintenance and fuel replacement. One third of amount of fuel assemblies in the core would be replaced with the new fuel assemblies. The spent fuel assemblies were transferred to the SFSP. Because the spent fuels still have decay heat, they need to be cooled, in order to prevent from fuel damage. The main problem of this study was to conduct cooling evaluation in AP1000’s SFSP from thermal-hydraulic aspect. For cooling spent fuel assemblies, there were two modes of cooling that could be used, namely natural convection and forced convection cooling modes. This study was performed in order to gain complete understanding of AP1000’s fuel thermal hydraulic aspect both in reactor operation and in spent fuel cooling. The previous research of thermal-hydraulic of AP1000 have been done for steady state reactor operation condition such as grid-spacer effect [3], the effect of radial-axial power fluctuation [4] and comparison analysis using fixed and temperature function of thermal conductivity [5].

The objective of this paper is studying to conduct evaluation in AP1000’s SFSP. The purpose of this study was to obtain the cooling time after shutdown so that the fuel assembly can be transferred to the SFSP with natural convection mode, and how minimum flow rate was required for cooling the 7 days spent fuel assembly by forced convection cooling in the SFSP. The requirement in cooling the spent fuel was no boiling occurred in the channel. The calculation was carried out by assuming the inlet coolant temperature of 30°C and by using COOLOD-N2 code. The calculation of cooling the spent fuel in SFSP would be carried out for 53 spent fuel (one third of core) [1] as first model and for 79 spent fuel (a half of core) as second model. That amount of fuel assemblies referred to amount of fuel assemblies in AP1000 core [2].

COOLOD-N2 code was applicable not only for research reactor in which plate type fuel adopted, but also for research reactor in which rod-type fuel adopted [6]. Other researches have been conducted by using COOLOD-N2 code such as analysis of the coolant velocity distribution in plate type fuel assembly [7], determining coolant flow rate distribution in the fuel modified TRIGA Plate Reactor [8], steady state thermal-hydraulics analysis of TRIGA research reactor [9], thermal hydraulic analyses of JRR-3 [10] and neutronic and thermal hydraulic analysis of TRIGA Mark II [11].

2. Theory

Calculation of residual heat

Evaluation of heat generated in a reactor after shutdown for determining cooling requirements under normal conditions and transient conditions is important to be done. The heat generation after reactor shutdown is the sum of the heat produced from the following: (1) fission from delayed neutron or photoneutron emissions; and (2) the decay of fission product, fertile materials, and other activation products from neutron capture. Both of these heat sources contribute equally amounts to the shutdown heat generation [12]. However, within a few minutes from shutdown, the fissions from delayed neutron emission are greatly reduced, so the amount is negligible.

2.1. Fission heat after shutdown.

The fission heat generation by delayed neutron is obtained by solving the neutron kinetic equation after a large negative reactivity insertion. The reactivity insertion will affect to the amount of neutron flux and neutron flux is directly proportional to reactor power [12]. The heat generated from fissions (the fission heat) after reactor shutdown was given in Equation 1 [12]:

\[
P_f = P_o \left(0.0625e^{-0.0124t} + 0.9375e^{-960t}\right) \quad \ldots \ldots (1)
\]

Where:
- \(P_f\) : heat generated from fissions after shutdown (MW),
- \(P_o\) : constant reactor operating power level (MW),
- \(t_s\) : times after shutdown (s).
In less than 0.01 second, the second term in Eq. 1 becomes negligible. Consequently, the reactor power will decrease exponentially over a period of approximately 80 seconds [12].

2.2. Heat from Fission Product Decay

The fission product decay is the major source of shutdown heat generation. The empirical formula for the rate of energy release due to $\beta$ and $\gamma$ emissions from decaying fission products are given by [12]

$$\beta \text{ energy release rate} = 1.40 \; t^-1.2 \; \text{MeV/fission . s} \quad (2)$$

$$\gamma \text{ energy release rate} = 1.26 \; t^-1.2 \; \text{MeV/fission . s} \quad (3)$$

Where $t^\prime = \text{time after the occurrence of fission (s)}$

The heat from fission product decay due to $\beta$ and $\gamma$ emission could be expressed as fraction of operating reactor power ($P_o$), given by Equation [9]:

$$P_\beta = P_o \left[ 0.035 \left( t^-0.2 - (t_s + \tau_s)^-0.2 \right) \right] \quad (4)$$

$$P_\gamma = P_o \left[ 0.031 \left( t^-0.2 - (t_s + \tau_s)^-0.2 \right) \right] \quad (5)$$

Thus, the total fission power decay heat rate ($P_2$) is given by Equation:

$$P_2 = P_\beta + P_\gamma$$

$$= P_o \left[ 0.066 \left( t^-0.2 - (t_s + \tau_s)^-0.2 \right) \right] \quad (6)$$

Where:

- $P_2$: total fission decay heat (MW)
- $P_\beta$: decay heat due to $\beta$ emission (MW)
- $P_\gamma$: decay heat due to $\gamma$ emission (MW)
- $P_o$: constant reactor operating power level (MW)
- $t_s$: times after reactor shutdown (s)
- $\tau_s$: reactor operation times (s)

3. Methodology

The methods in analyzing the decay heat removal from the fuel assembly to the spent fuel storage pool were done by 3 steps, i.e.:

3.1. Decay heat calculation

The decay heat from fission could be calculated by using Eq. 1, whereas the decay heat due to $\beta$ and $\gamma$ emission could be calculated by using Eq. 6. This paper would show the comparison between the fission heat after shutdown (Eq. 1), the total fission decay heat due to $\beta$ and $\gamma$ emission (Eq. 6) and the total heat (sum of Eq.1 and Eq. 6) generated after reactor shutdown at times interval from 1 to 150 seconds.

This paper also would show the amount of the fission decay heat due to $\beta$ and $\gamma$ emission at several days after reactor shutdown (based on Eq.6) for 157 fuel assemblies (full core), 53 fuel assemblies (one third of core) and 79 fuel assemblies (a half of core) of AP1000 reactor.

3.2. Calculation model

In modeling the calculation of cooling the decay heat from the fuel assemblies in the spent fuel storage pool, it was made 2 models refueling. The first model is one third of core refueling, and the second model is a half of core refueling. In one third of core refueling means 57 spent fuel assemblies (15,048 fuel rods) were taken out from the core and sent to the spent fuel storage pool for cooling, and changed with the new amount one. Whereas, in a half of core refueling means 79 spent fuel assemblies (20,856
fuel rods) were taken out from the core and sent to the spent fuel storage pool for cooling and changed with the new amount one. There are 2 kinds of coolant circulation for cooling the each model in the fuel storage pool, i.e., by using natural convection and forced convection. For cooling by using natural convection, it is necessary to determine how many days after shutdown, so that the fuel assemblies can be cooled without boiling. Whereas, for cooling by using forced convection, it is necessary to determine how the minimum flow rate, so that the 7th days fuel assemblies after reactor shutdown can be cooled without boiling. The modeling and calculation were conducted by using COOLOD-N2 code.

3.3. Analysis the calculation results.

The last step was comparing the calculation results between of one third of core refueling and a half of core refueling. The analysis was conducted for cooling by natural convection and forced convection for each model by using COOLOD-N2 code, by assuming the inlet coolant temperature of 30.0°C. The calculation of heat removal by natural convection from spent fuel to the storage pool of AP1000 reactor varied for 7, 14, 21, 28, 35 and 42 days after reactor shutdown were performed for one third and a half of core models. Whereas, the calculation of heat removal by forced convection from spent fuel to the storage pool of AP1000 reactor varied of flow rate of 15, 20, 25, 30 and 35 kg/s at 7 days after reactor shutdown.

4. Results and discussion

4.1. Decay heat calculation result

Figure 1. The graph of fission heat, fission decay heat due to $\beta$ and $\gamma$ emission and the total heat generated after reactor shutdown fission at times interval from 1 to 150 seconds

Figure 1 showed the comparison graphs between the fission heat generated after shutdown (Eq.1), the fission decay heat due to $\beta$ and $\gamma$ emission (Eq.6) and the total heat generated after reactor shutdown (sum of Eq.1 and Eq. 6) at times interval from 1 to 150 seconds, by using the assumption of a 3400 MW of constant thermal power of AP1000 reactor that operated for 1 year. At 1 second after reactor shutdown, the fission heat is amount of 209.88 MW (6.2% of reactor power), the fission decay heat due to $\beta$ and $\gamma$ emission is amount of 217.30 MW (6.4% of reactor power), and resulting the total heat generated after shutdown is amount 427.18 MW (12.6% of total power). The fission heat is decreased slowly but continuously. It was shown, at 10 minutes after shutdown, the fission heat is remaining of 0.125 MW (less than 0.01% reactor power). So that, in the order of days after reactor shutdown, the amount of fission heat can be negligible. While the fission decay heat rate decreases exponentially (steep at the beginning but ramps at the end). It was shown, at 10 minutes after shutdown the fission
decay heat is still 55.33 MW (1.6% reactor power). The fission decay heat due to β and γ emission at several days after reactor shutdown was shown in Table 1.

Table 1 showed the amount of the fission decay heat after the reactor shutdown (based on Eq.6) for 157 fuel assemblies (full core), 53 fuel assemblies (one third of core) and 79 fuel assemblies (a half of core) of AP1000 reactor.

| Times after shutdown (days) | Fission decay heat from 157 fuel assemblies MW | % | Fission decay heat from 53 fuel assemblies (one third of core) MW | % | Fission decay heat from 79 fuel assemblies (a half of core) MW | % |
|-----------------------------|---------------------------------------------|---|---------------------|---|---------------------|---|
| 1                           | 16.01                                       | 0.471 | 5.59                | 0.164 | 8.06                | 0.237 |
| 7                           | 8.58                                        | 0.252 | 3.08                | 0.091 | 4.32                | 0.127 |
| 14                          | 6.58                                        | 0.194 | 2.40                | 0.071 | 3.31                | 0.097 |
| 21                          | 5.55                                        | 0.163 | 2.05                | 0.060 | 2.79                | 0.082 |
| 28                          | 4.87                                        | 0.143 | 1.82                | 0.053 | 2.45                | 0.072 |
| 35                          | 4.38                                        | 0.129 | 1.65                | 0.048 | 2.20                | 0.065 |
| 42                          | 3.99                                        | 0.117 | 1.52                | 0.045 | 2.01                | 0.059 |
| 49                          | 3.69                                        | 0.108 | 1.41                | 0.041 | 1.85                | 0.055 |
| 56                          | 3.43                                        | 0.101 | 1.32                | 0.039 | 1.73                | 0.051 |

4.2. Cooling times analysis by using natural convection

The calculation results of heat removal by natural convection from spent fuel to the storage pool of AP1000 reactor varied for 7, 14, 21, 28, 35 and 42 days after reactor shutdown was shown in Figure 2. As requirements, it is necessary to determine the cooling time after shutdown, so that the spent fuel assemblies can be cooled without boiling.

Figure 2 (a) showed the graph of coolant temperature of the hottest fuel rod varied 7, 14, 21, 28, 35 and 42 days after reactor shutdown using natural circulation mode for the first model. Up to 14 days after reactor shutdown, coolant boiling was still occurred. The new spent fuel assemblies could be moved into the spent fuel storage pool at least 21 days after reactor shutdown. At 21 days after reactor shutdown, the calculation results using the COOLOD-N2 code indicated that the outlet coolant temperature of the average channel was 81.48°C, the maximum temperature of outer cladding and center fuel meat were 83.50°C and 83.77°C, respectively. However, for the hottest channel showed that the outlet coolant temperature was 101.76°C, the maximum temperature of outer cladding and center fuel meat were 104.21°C and 104.51°C, respectively. Although the outlet coolant temperature of the hottest channel was 101.76°C (more than 100°C) at the end of the channel, but since it was still lower than the saturation temperature of 104.92°C (at 1.618 atmospheric pressure), so the boiling was not be occurred.
Figure 2. Graph of coolant temperature on the hottest fuel rod varied of time after shutdown, cooling by using natural convection. (a) one third of core refueling, (b) a half of core refueling.

Note: $T_f =$ coolant temperature and $T_{sat} =$ saturated coolant temperature.

Whereas, figure 2 (b) showed the graph of coolant temperature of the hottest fuel rod for the second model. Up to 35 days after reactor shutdown, coolant boiling was still occurred. The new spent fuel assemblies could be moved into the spent fuel storage pool at least 42 days after reactor shutdown. At 42 days after reactor shutdown, the calculation results indicated that the outlet coolant temperature of the average channel was 80.48°C, the maximum temperature of outer cladding and center fuel meat were 82.27°C and 82.40°C, respectively. However, for the hottest channel showed that the outlet coolant temperature was 100.37°C, the maximum temperature of outer cladding and center fuel meat were 102.52°C and 102.72°C, respectively. Comparing the second refueling model to the first one, the second refueling model that has 26 fuel assemblies more than the first one, requires 21 days after reactor shutdown (twice times) more than the amount days for the first model.

4.3. Coolant flow analysis by using forced convection

In the cooling process by natural convection mode, obtained that the average coolant velocity was 0.0078 m/s. If each fuel assembly has a flow area of 0.02472 m$^2$, respectively, and the coolant density was 977.8 kg/m$^3$ at mean coolant temperature of 69.06°C, it could be calculated that the total flow rate was:

$$\text{Flowrate} = \rho v A = 977.8 \frac{kg}{m^3} \times 0.0078 \frac{m}{s} \times 53 \times 0.02472 m^2 = 9.99 \frac{kg}{s}$$

The natural convection cooling has total flow rate almost 10 kg/s. It took a long time of 21 days for the new spent fuel assemblies to be transferred to the spent fuel storage pool. Therefore, it is necessary to analyze how the new spent fuel cooling by using forced convocation mode, although with a small flow rate pump.

Therefore, a heat removal calculation of spent fuel storage pool of AP1000 reactor were performed for one third and a half of core refueling at 7 days after reactor shutdown by using forced convection mode varied of flow rate of 15, 20, 25, 30 and 35 kg/s. At 7 days after reactor shutdown, the first model has power of 3.08 MW, and the second model has power of 4.32 MW. The calculation was conducted by using COOLOD-N2 code, by assuming the inlet coolant temperature of 30.0°C as shown in Figure 3. As requirements, it is necessary to determine how much flow rate needed, so that the new spent fuel assemblies can be cooled without boiling.
Figure 3. Graph of coolant temperature on the hottest fuel rod varied of time after shutdown, cooling by using forced convection. (a) one third of core refueling, (b) a half of core refueling. 

Note: $T_{sat}$ = saturated coolant temperature, $Nat-Cir$ = coolant temperature using natural circulation, and $P_{xx}$ = coolant temperature by using pump (forced convection).

Figure 3 (a) showed the graph of the coolant temperature in the hottest fuel rod at 7 days after reactor shutdown for one third of core refueling by using forced convection cooling varied for 10 (natural circulation), 15, 20, 25, 30 and 35 kg/s of flow rates. The graph showed that for storage pool inlet temperatures of 30.0°C, a cooling pump with minimum flow rate of 15 kg/s was required to cool the new spent fuel assemblies in order to make no boiling in the fuel channels. In cooling conditions with a minimum flow rate of 15 kg/s, the calculation results using the COOLOD-N2 code for the average channel indicates that the outlet coolant temperature was 81.57°C, the maximum outer cladding temperature and the center fuel meat were 83.90°C and 84.34°C, respectively, and maximum heat flux was 2.27 kW/m². And for the hottest channel was obtained that the outlet coolant temperature was 101.88°C, the maximum temperature of the outer cladding and the center fuel meat were 104.63°C and 105.21°C, respectively, and the maximum heat flux was 3.17 kW/m². Although the outlet coolant temperature of the hottest channel was 101.88°C (more than 100°C) at the end of the channel, but since it was still lower than the saturation temperature of 104.92°C (at 1.618 atmospheric pressure), so the boiling was not occurred.

Figure 3 (b) showed the graph of the coolant temperature in the hottest fuel rod at 7 days after reactor shutdown for a half of core refueling by using forced convection cooling varied for 10 (natural circulation), 15, 20, 25, 30 and 35 kg/s of flow rates. The graph showed that for storage pool inlet temperatures of 30.0°C, a cooling pump with minimum flow rate of 25 kg/s was required to cool the new spent fuel assemblies in order to make no boiling in the fuel channels. In cooling conditions with a minimum flow rate of 25 kg/s, the calculation results showed that for the average channel indicates that the outlet coolant temperature was 101.88°C, the maximum temperature of the outer cladding and the center fuel meat were 104.63°C and 105.21°C, respectively, and the maximum heat flux was 3.17 kW/m². Although the outlet coolant temperature of the hottest channel was 101.88°C (more than 100°C) at the end of the channel, but since it was still lower than the saturation temperature of 104.92°C (at 1.618 atmospheric pressure), so the boiling was not occurred.

5. Conclusion
The analysis of heat removal from the new spent fuel to the storage pool by using COOLOD-N2 has been carried out. The analysis was conducted for one third of core and a half of core refueling, that cooling by using natural convection and forced convection. On cooling using natural circulation, the new spent fuel could be moved to the fuel storage pool at 21 days after reactor shutdown for first model, and 42 days for second model. Whereas, on cooling the new spent fuel by using forced convection at 7 days after reactor shutdown, it was required 15 kg/s for first model and 25 kg/s for second model, so the boiling was not occurred.
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