*Transactions of the Korean Nuclear Society Spring Meeting*
*Jeju, Korea, May 22, 2009*

# Cladding Heatup Prediction between Spacer Grids for the Downstream Effect Evaluation

J. Y. Park* and M. W. Kim
*Korea Institute of Nuclear Safety, Thermal-Hydraulic Research Dept., 19 Guseong-dong Yuseong-gu, Daejeon 305-338, KOREA*
**Corresponding author:k385pjy@kins.re.kr*

## 1. Introduction

Since a recirculation sump clogging issue by debris generated from high energy pipe line break had been invoked as GSI-191 in the US, many researches on this issue have been undertaken. Previous researches on this topic are well summarized in Bang et al. [1].

Due to comprehensive nature of the issue, it includes many area of research and one of them is the area of downstream effect evaluation. The downstream effect is involved with adverse effects of debris passing the sump screen on the downstream systems, components and piping including core and it can be further divided into an ex-vessel downstream effect and an in-vessel downstream effect. In the ex-vessel downstream effect, focus is laid on plugging of spray nozzle, wearing and abrasion of moving parts of pump and valve and etc. Otherwise, a debris effect on reactor core is focused in the in-vessel downstream effect. Since debris can be ingested in the core or the systems of downstream of sump screen during recirculation, basically the downstream effect influences long-term core cooling phase.

With respect to the in-vessel downstream effect, an up-to-date evaluation methodology is well summarized in a topical report submitted to the US nuclear regulatory commission by the pressurized water reactor owners group (PWROG) [2]. The report evaluates various aspects of debris ingestion in the core such as blockage at the core inlet, collection of debris on fuel grids, plating-out of fuel, chemical precipitants, protective coatings effect and etc. Most of them are evaluated qualitative manner based on previous research results and geometrical consideration on fuel rod bundles but some of them are also backed up by quantitative calculations to corroborate the qualitative decisions. One of them is a cladding heatup calculation between spacer grids. This is done to demonstrate that the cladding temperature of a fuel rod between grids with debris deposited on the clad surface in a post-LOCA recirculation environment is below acceptance limit of long-term core cooling phase (i.e. 800°F) at extreme case of involved variables.

Since this evaluation can be done analytic manner and it is directly related to a fundamental question of how high the clad temperature could be under debris deposition, we perform the cladding heatup calculation for OPR1000 reactor with PLUS7 fuel just as the same manner as done in the topical report.

## 2. Evaluation model and governing equations

To evaluate cladding heatup, first of all, a simplified fuel rod model should be devised. Figure 1 shows devised model in the present study. As shown in the figure, the fuel rod with active length of $L$ is assumed to be covered by uniform thickness of oxide, crud and precipitate layers. Typical radii of cladding, oxide, crud and precipitate layers are represented by $r_1$, $r_2$, $r_3$, $r_4$ and $r_5$, respectively and $Q$ represents heat flow from fuel pellet. No spacer grid is assumed either. In heat transfer terms, the present evaluation model neglects axial heat transfer and assume uniform linear power density along the fuel rod, negligible contact thermal resistances among various layers and steady state condition.

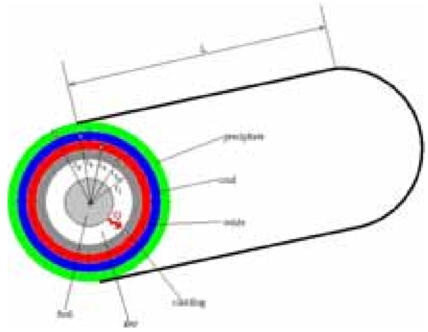

Fig. 1. A fuel rod model for cladding heatup calculation.

For the schematic of the fuel rod model seen above, thermal resistances of each layer are given by [2]

$$R_{clad}=\frac{1}{2\pi L}\frac{\ln(r_2/r_1)}{K_{clad}},\ R_{oxide}=\frac{1}{2\pi L}\frac{\ln(r_3/r_2)}{K_{oxide}} \quad (1)$$
$$R_{crud}=\frac{1}{2\pi L}\frac{\ln(r_4/r_3)}{K_{crud}},\ R_{pre.}=\frac{1}{2\pi L}\frac{\ln(r_5/r_4)}{K_{pre.}}$$

where $R$ and $K$ are thermal resistance and thermal conductivity of each layer. From the definition of thermal resistance, heat flows for each layer are given as

$$Q_{clad}=\frac{T_1-T_2}{R_{clad}},\ Q_{oxide}=\frac{T_2-T_3}{R_{oxide}} \quad (2)$$
$$Q_{crud}=\frac{T_3-T_4}{R_{crud}},\ Q_{pre.}=\frac{T_4-T_5}{R_{pre.}}$$

where $T_i$ represents temperature at $r_i$. Since

$$Q_{clad}=Q_{oxide}=Q_{crud}=Q_{pre.}=Q \quad (3)$$

in a steady state condition, following equation holds.

$$Q\equiv\frac{T_1-T_5}{R} \quad (4)$$

where $R$, a total thermal resistance from cladding layer to precipitate layer is given by

$$R=\frac{1}{2\pi L}\left(\frac{\ln(r_2/r_1)}{K_{clad}}+\frac{\ln(r_3/r_2)}{K_{oxide}}+\frac{\ln(r_4/r_3)}{K_{crud}}+\frac{\ln(r_5/r_4)}{K_{pre.}}\right). \quad (5)$$

A relation between $T_5$ and $T_{bulk}$ surrounding fluid temperature near the fuel rod can be derived by Newton's law of cooling. From figure 2, it follows that

$$Q = hA(T_{wall} - T_{\infty}) = hA(T_5 - T_{bulk}) \quad (6)$$

where $h$ and $A$ are heat transfer coefficient and heat transfer area, respectively. Therefore, $T_5$ is given by

$$T_5 = Q/(hA) + T_{bulk} = QR_{external} + T_{bulk}. \quad (7)$$

Having been all required equations derived, the fuel cladding temperature, $T_2$ can be derived by combining equations such as $T_1 - T_2 = QR_{clad}$, $T_1 - T_5 = QR$ and $QR_{external} = T_5 - T_{bulk}$ as

$$T_2 = \frac{Q}{2\pi L}\left(\frac{\ln(r_3/r_2)}{K_{oxide}} + \frac{\ln(r_4/r_3)}{K_{crud}} + \frac{\ln(r_5/r_4)}{K_{pre.}} + \frac{1}{hr_5}\right) + T_{bulk}. \quad (8)$$

Here, $A = 2\pi r_5 L$ has been used for above derivation.

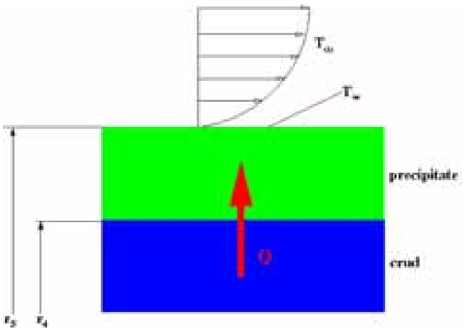


Fig. 2. Schematic of the fuel rod surface.

## 3. Calculation example for Uljin3/4 with PLUS7 fuel

Since fuel cladding temperature equation having been derived as Eq. (8), specific evaluation of cladding heatup is performed for PLUS7 nuclear fuel of Uljin3/4 nuclear power plant. First, a peak linear heat flux $Q/L$ is determined as follows. From figure 3 of normalized decay heat curve of Uljin3/4, averaged $Q$ at the beginning of recirculation cooling is given by

$$\overline{Q} = 2980MWt \times 0.01956 = 58.28MWt \quad (9)$$

where $2980MWt$ includes 102% of nominal core power and RCP power. $L$ is determined by

$$L = 41{,}772 \times 150inch = 150{,}379.2m \quad (10)$$

where 41,772 is total number of fuel rods and $150inch$ is active length per fuel rod. Therefore, averaged linear heat flux $\overline{Q}/L$ is given by

$$\overline{Q}/L = 366.25W/m. \quad (11)$$

Applying total heat flux factor value of 2.42 referred from Uljin3/4 FSAR table 4.4-1 to the averaged linear heat flux, then the peak linear heat flux $Q/L$ is given by

$$Q/L = 366.25W/m \times 2.42 = 886.325W/m. \quad (12)$$

As for $r_2$, a nominal diameter of PLUS7 fuel is used.

$$r_2 = 0.5 \times 0.374inch = 0.0047498m. \quad (13)$$

Based on nuclear industry experience, oxide and crud thicknesses are assumed to be $0.004inch$, respectively. Therefore, corresponding $r_3$ and $r_4$ are determined to be $0.0048514m$ and $0.004953m$.

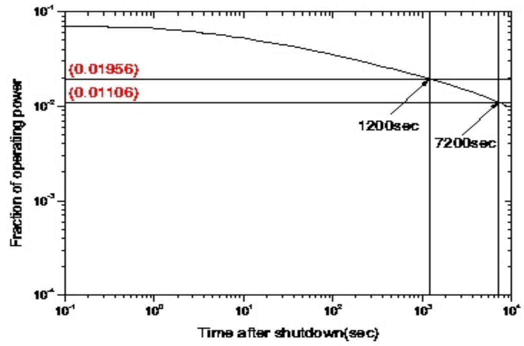


Fig. 3. Normalized decay heat curve of Uljin3/4.

For precipitate thickness, $0.05inch$ is considered to be conservative value and the resulting $r_5$ is given by $0.006223m$. For thermal conductivities, $2.2W/m-K$, $0.52W/m-K$ and $0.17W/m-K$ are used for $K_{oxide}$, $K_{crud}$ and $K_{pre.}$, respectively based on industry experience, EPRI suggestion and conservative consideration. $h$ and $T_{bulk}$ are also assumed to be $3{,}688.4W/m^2-K$ and $394.3K$, respectively which are considered as typical values of $h$ and $T_{bulk}$ at recirculation phase. Using above specified numerics, the fuel cladding temperature $T_2$ for PLUS7 is calculated as

$$T_2 = 560.26K = 287.1^oC = 548.8^oF. \quad (14)$$

Note that this value is similar to $T_2 = 558.7^oF$ for typical Westinghouse fuel case. [3]

## 4. Conclusion and Discussion

Considering debris deposition on the fuel surface, quantitative cladding heatup evaluation for PLUS7 fuel of Uljin3/4 was performed. The calculated peak cladding temperature of PLUS7 is almost the same as that of Westinghouse fuel and satisfies the acceptance criteria of the long-term core cooling.

However, much of uncertainties were involved in determining oxide/crud/precipitate thickness, thermal conductivities, heat transfer coefficient and bulk fluid temperature, care should be in mind to corroborate the present result.

## REFERENCES

[1] Y. S. Bang, I. G. Kim, D. Y. Oh, B. G. Huh, S. W. Woo and I. S. Lee, Development of Regulatory Technology on ECCS Water Sources for Long-term Recirculation Cooling, KINS/RR-475, February 2007.

[2] F. P. Incropera and D. P. De Witt, *Introduction to Heat Transfer*, John Wiley & Sons, 1985.

[3] Westinghouse Electric Company, Evaluation of Long-term Cooling Considering Particulate and Chemical Debris in the Re-circulating Fluids, WCAP-16793-NP, May 2007.