# CALCULATION OF FUEL AND MODERATOR TEMPERATURE COEFFICIENTS IN APR1400 NUCLEAR REACTOR BY MVP CODE

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***ABSTRACT:*** In this project, these fuel and moderator temperature coefficients were calculated in APR1400 nuclear reactor by MVP code. APR1400 is an advanced water pressurized reactor, that was researched and developed by Korea Experts, it's electric power is 1400 MW. The neutronics calculations of full core is very important to analysis and assess a reactor. Results of these calculation is input data for thermal-hydraulics calculations, such as fuel and moderator temperature coefficients. These factors describe the self-safety characteristics of nuclear reactor. After obtaining these reactivity parameters, they were used to re-run the thermal hydraulics calculations in LOCA and RIA accidents. These thermal-hydraulics results were used to analysis effects of reactor physics parameters to thermal hydraulics situation in nuclear reactors.

## I. INTRODUCTION

The Advanced Power Reactor 1400 (APR1400) [1,2,4] is of the pressurized water type using two reactor coolant loops. This reactor used uranium dioxide fuel, and many characteristics that were improved from OPR1000 reactor. The simulation and calculation of this new reactor is an important work that helps to obtain experiences and skills in analysis and assessment NPP technology.

Fuel temperature coefficient (Doppler effects) and Moderator Temperature Coefficients (MTC) are two important parameters that have to consider in design and operating of NPP. These parameters have to be negative in Pressurized Water Reactor (PWR), and also in APR1400. We carried out these factors to investigate change of reactivity that depend on temperatures of fuel and moderator. Then these results were used to assess effects of reactor physics factors to TH state in RIA and LOCA accidents.

## II. FUEL AND NUCLEAR DESIGN OF APR1400 REACTOR

APR1400 reactor is the newest water reactor that was result of Korea government project from 1992. This reactor has lager power, 1400 MWe. And there are many enhancements in fuel and nuclear design. Figure 1 and table 1 describe the full core of APR1400.

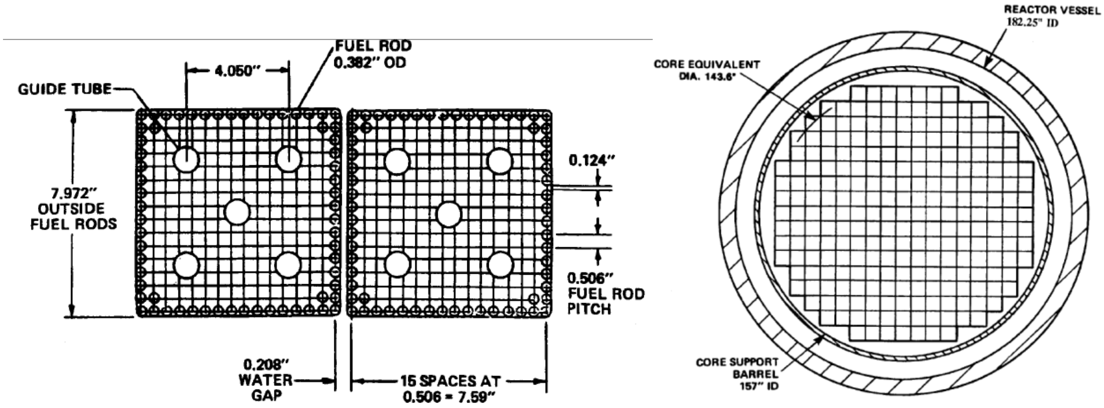


**Figure 1:** Reactor Core Cross Section 241 Fuel Assemblies [5].

**Table 1:** Mechanical Design Parameters [5].

| | |
|---|---|
| Number of fuel assemblies in core, total | 241 |
| Number of CEAs | 93 |
| Number of fuel rod locations | 56,876 |
| Spacing between fuel assemblies, fuel rod surface to surface inches (cm) | 0.208 (0.528) |
| Spacing, outer fuel rod surface to core shroud, inches (cm) | 0.214 (0.544) |
| Hydraulic diameter, nominal channel, feet (cm) | 0.0393 (1.198) |
| Total flow area (excluding guide tubes), $ft^2$ ($m^2$) | 60.8 (5.649) |
| Total core area, $ft^2$ ($m^2$) | 112.3 (10.433) |
| Core equivalent diameter, inches (cm) | 143.6 (3.647) |
| Core circumscribed diameter, inches (cm) | 152.46 (3.872) |
| Total fuel loading, lb U (kg U) (assuming all rod locations are fuel rods) | $228 \times 10^3$ ($103.42 \times 10^3$) |
| Total fuel weight, lb $UO_2$ (kg $UO_2$) (assuming all rod locations are fuel rods) | $258.6 \times 10^3$ ($117.3 \times 10^3$) |
| Total weight of Zircaloy, lb (kg) | 74,950 (33,996.7) |
| Fuel volume (including dishes), $ft^3$ ($m^3$) | 409.6 (11.6) |

This data is adequate for thermal-hydraulics and neutronics calculations in full core of APR1400 reactor.

## III. CALCULATION OF MULTIPLIER FACTOR (K-EFF) AND REACTIVITY COEFFICIENTS IN FULL CORE OF APR1400 REACTOR

Dopler effect is very important to design and operate a nuclear reactor [6,7,8]. When fuel temperature changes, cross section of $U^{238}$ and neutron reaction changes, and reactivity is changed. In this project, this effect was claculated when fuel temperature changes from 68 F (293 K)-room temperture to 2500 F (1644 K), that is reached when accident happens.

Fiure 2 is result of calculation for change of multiplier factor and reactivity when fuel average temperature varies.

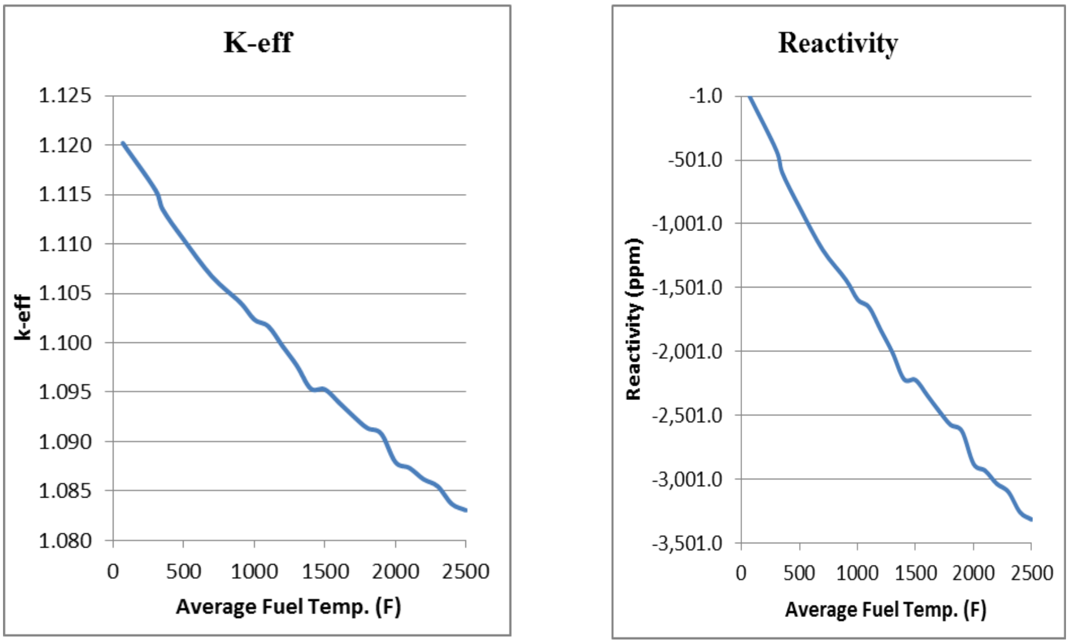


**Figure 2:** Dependent of multiplier factor and reactivity on fuel temperature by MVP code.

Results that were obtained by MVP code, run in personal computer and windows OS, has 0.036% errors. This is fit with results in Safety Analysis Report (SAR) [4].

Moderator Temperature Coefficient (MTC) strongly affects to reactor physics state, too. When moderator temperature changes, H-2, O-16 and B-10 nuclide densities change, that affects to neutron flux and reactivity in nuclear reactor. In this project, MTC was calculated in two states, the first state: hot zero power, 555$^{o}$F (290.6$^{o}$C) water temperature, no control element assemblies (CEA), clean, 1210 ppm acid boric; and the second state: Hot full power, 588$^{o}$F (308.9$^{o}$C), no CEAs, clean and 1088 ppm acid boric. The results are -2.19E-04 $\Delta\rho/^{o}$F and -2.63E-04 $\Delta\rho/^{o}$F , corresponding to the first and second states. These results have large different that comparing to results in SAR, MTC coefficients are -0.11E-04 $\Delta\rho/^{o}$F and -0.51E-04 $\Delta\rho/^{o}$F.

## IV. USING THE NEUTRONICS CALCULATION RESULTS FOR THE THERMAL HYDRAULICS CALCULATIONS

The thermal hydraulics code - RELAP5 code, was used to carry out the Reactivity Initial Accident (RIA) and Loss Off Coolant Accident (LOCA) calculation in WINDOW OS [3], on PC. The calculations used the neutronics calculation results, that showed above. Figure 3 indicated change of DNBR parameter in control valve - 777 in 6 s after RIA accident. Results obtained from the new and old input data, are similar. That indicates that MTC and Doppler effects affect to thermal hydraulics state of reactor very weakly.

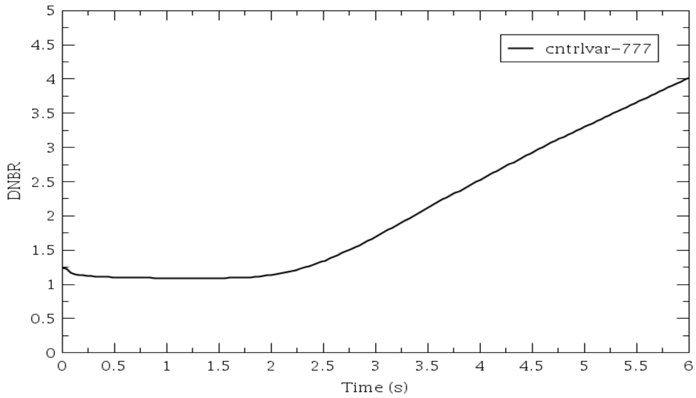


**Figure 3:** DNBR change in RIA accident.

## V. CONCLUSION

In this project, the calculated group completed all of proposed works, described detailed structure of fuel assembly and full core in APR1400, knew and calculated multiplier factor, reactivity coefficients, and using nuclear physics results for thermal-hyraulics calculation. Calculated results are used for nuclear reactor safety analysis, although this data is not high correct level but it was the first step in full core calculation for APR1400 technology, and understood effect of and neutronics results to thermal-hydraulics states.

## REFERENCES

[1] Design Features, Safety Assessment and Verification of Key Systems, and Economic Advancements for APR1400, Sung Jae Cho, Eui Jong Lee, Engineering Support Center, Nuclear Environment Technology Institute, Korea Hydro & Nuclear Power Co., LDT, 103-16, Munji-Dong, Yuseong-Gu, Daejeon 305-380, Korea.

[2] APR1400 Design Description, Center for Advanced Reactors Development, Nuclear Environment Technology Institute, 로고 Korea Hydro & Nuclear co., Ltd, 03/2002.

[3] Lê Văn Hồng và các cộng sự. Báo cáo tổng hợp kết quả nghiên cứu khoa học công nghệ nghị định thư “Hợp tác nghiên cứu phân tích, đánh giá an toàn vùng hoạt lò phản ứng năng lượng nước nhẹ trong các chuyển tiếp và sự cố, Viện Năng lượng Nguyên tử Việt Nam, Hà Nội, 2011.

[4] Sung-Quun Zee, Modure 2: Reactor Core and Components, Nuclear Power Reactor Technology, Core Dseign and Analysis Technology Dept., Korea Atomic Energy Research Institute, <http://www.kntc.re.kr/openlec/nuc/NPRT/module2/module2_2/module2_2_2/2_2_2.htm#3.3%20Burnable%20Poisons>

[5] Islamic Azad University. Computation of concentration changes of heavy metals in the fuel assemblies with 1.6% enrichment by ORIGEN code for VVER-1000, Mohammad Rahgoshay, Department of Nuclear Engineering, Faculty of Engineering, Science and Research Branch, , Tehran, Iran, 2006.

[6] K. Okumura, T. Kugo, K. Kaneko, K. Tsuchihashi. SRAC2006: Acomprehensive Neutronics Canculation Code System, Japan Atomic Enerrgy Angency, 2007.

[7] Y. Nagaya, T. Mori, K. Okumura, M. Nakagawa. MVP/GMVP: General Purpose Monte Carlo Codes for Neutron and Photon Transport Camculations based on Continuous Energy and Multigroup Methods Version 2, Japan Atomic Energy Research Institute, 2004.

[8] Lê Đại Diễn. Báo cáo tổng kết đề tài khoa học công nghệ cấp cơ sở “Sử dụng chương trình MVP tính toán cho mô hình bó nhiên liệu HEU và LEU của lò phản ứng hạt nhân Đà Lạt”, Hà Nội, 2007.