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TECHNICAL MATERIAL

Calculation of Pellet Radial Power Distributions with Monte Carlo and Deterministic Codes

Motomu SUZUKI*, Toru YAMAMOTO and Tetsuo NAKATA

Japan Nuclear Energy Safety Organization,
TOKYU REIT Toranomon Bldg., 3-17-1 Toranomon, Minato-ku, Tokyo, 105-0001, Japan

The Japan Nuclear Energy Safety Organization (JNES) has been working on an irradiation test program of high-burnup MOX fuel at Halden Boiling Water Reactor (HBWR). MOX and UO_2 fuel rods had been irradiated up to about 64 GWd/t (rod avg.) as a Japanese utilities research program (1st phase), and using those fuel rods, in-situ measurement of fuel pellet centerline temperature was done during the 2nd phase of irradiation as the JNES test program. As part of analysis of the temperature data, power distributions in a pellet radial direction were analyzed by using a Monte Carlo burnup code MVP-BURN. In addition, the calculated results of deterministic burnup codes SRAC and PLUTON for the same problem were compared with those of MVP-BURN to evaluate their accuracy.

Burnup calculations with an assembly model were performed by using MVP-BURN and those with a pin cell model by using SRAC and PLUTON. The cell pitch and, therefore, fuel to moderator ratio in the pin cell calculation was determined from the comparison of neutron energy spectra with those of MVP-BURN. The fuel pellet radial distributions of burnup and fission reaction rates at the end of the 1st phase irradiation were compared between the three codes. The MVP-BURN calculation results show a large peaking in the burnup and fission rates in the pellet outer region for the UO_2 and MOX pellets. The SRAC calculations give very close results to those of the MVP-BURN. On the other hand, the PLUTON calculations show larger burnup for the UO_2 and lower burnup for the MOX pellets in the pellet outer region than those of MVP-BURN, which lead to larger fission rates for the UO_2 and lower fission rates for the MOX pellets, respectively.

KEYWORDS: pellet radial distribution, UO_2 pellet, MOX pellet, Halden boiling water reactor, burnup calculation, MVP-BURN, SRAC, PLUTON

I. Introduction

Recycling Plutonium from spent LWR (Light Water Reactor) UO_2 fuel has been started as reload MOX (Mixed Oxide) fuel since 2009 in Japan. Current licensed burnup of MOX fuel is lower than that of UO_2 fuel. In the future, MOX fuel is expected to irradiate to higher burnup. It is important to research performance of MOX fuel in high burnup. The Japan Nuclear Energy Safety Organization (JNES) has been working on an irradiation test program of high-burnup MOX fuel at Halden Boiling Water Reactor (HBWR).[1-3] MOX and UO_2 fuel rods had been irradiated up to about 64 GWd/t (rod avg.) as a Japanese utilities research program (1st phase). Using those fuel rods, in-situ measurement of fuel pellet centerline temperature was done during the 2nd phase of irradiation as the JNES test program. Evaluation of the fuel pellet temperature should be performed by adequately considering radial distribution of thermal power, fission gas release rate, fuel thermal conductivity and gap conductance etc.

The purpose of this analysis is to provide the calculation results of pellet radial fission rate distributions, which are proportional to the thermal power distributions, to be used in reproducing the measured values of the pellet temperature. As part of analysis of the measured temperature data, power distributions in a pellet radial direction were analyzed by a detailed calculation using an assembly model with a Monte Carlo code in this study. In addition, the calculated results of deterministic burnup codes SRAC[4] and PLUTON[5,6] for the same problem were compared with those of MVP-BURN[7] to evaluate their accuracy. The SRAC code is one of deterministic codes commonly used to calculate the neutronic characteristics of fuel and moderator cells in various nuclear design and analysis. The PLUTON code is a deterministic code mainly used included in a fuel mechanical-behavior calculation code FEMAXI.[8] The comparison of the results of MVP-BURN with those of the deterministic codes is aimed at assessing accuracy in calculation of the pellet radial distributions.

In the following, Section II describes the burnup calculation conditions, and Section III presents the calculation results and discussion. Section IV summarizes this paper.

II. Calculation Condition

1. Overview of Irradiation Experiment

Halden Boiling Water Reactor (HBWR) is an irradiation test reactor in Norway, in which heavy water is used as coolant and moderator.[9] Many irradiation experiments have been performed in the Halden Reactor Project. HBWR core consists of 110–120 assemblies including Instrumented Fuel

*Corresponding author, E-mail: suzuki-motomu@jnes.go.jp

Assembly (IFA) in hexagonal lattice. **Table 1** shows the core and assembly specifications. Fuel mechanical characteristics mainly vary with fuel burnup. Therefore the measurement data taken for the irradiated fuel at the HBWR can be applied to evaluate the fuel mechanical characteristics of the fuel irradiated at the normal LWR.

The irradiation program relevant to this study was implemented in the two phases. In the 1st phase, UO_2 and MOX fuel rods installed in a irradiation rig were irradiated up to burnup of about 64 GWd/t (rod avg.) in the Japanese utilities research program. In the program, the assembly is called IFA-609/626. The IFA-609/626 consists of UO_2 fuel rods, SBR (Short Binderless Route) MOX fuel rods, MIMAS (Micronized Masterblend) MOX fuel rods, which amount to 12 fuel rods. In the 2nd phase, the irradiation rig (IFA-702) was assembled by selected three fuel rods in IFA-609/626. The irradiation rig was irradiated up to burnup of about 70 GWd/t (rod avg.). In-situ measurement of fuel pellet centerline temperature was done during the 2nd phase of irradiation. **Table 2** shows the fuel specifications of IFA-702. Basic specifications of the fuel in Table 2 are those before irradiation (fresh fuel).

Table 1 Core specifications of HBWR[9)]

Core	
Power Level [MWt]	up to 20
Configuration	Open Hexagonal Lattice
Lattice Pitch [mm]	130
Assembly Shroud	
Material	Zircaloy-2
Inner diameter [mm]	71
Wall thickness [mm]	1
Coolant and Moderator	
Material	Heavy Water
Reactor pressure [MPa]	3.33
Heavy water saturation temperature [K]	513

Table 2 Fuel rod specifications of IFA-702[1-3)]

	702-1	702-2	702-3
Fuel pellet (before irradiation of IFA-609/626)			
Pellet type	SBR-MOX	UO_2	MIMAS-MOX
U-235 enrichment [wt%]	depleted	8	depleted
Put content [wt%]	8.4	-	8.4
Puf content [wt%]	6.1	-	6.1
Density [%TD]	95	95	95
Diameter [mm]	8.19	8.19	8.05
Cladding (before irradiation of IFA-609/626)			
Material	Zircaloy-4		
Outer diameter [mm]	9.5	9.5	9.5
Inner diameter [mm]	8.36	8.36	8.22
Fuel rod			
Average burnup before irradiation [GWd/t]	65*	65*	65*
Average burnup after irradiation [GWd/t]	70	69	69

*Rod average burnups were slightly larger than those of the 1st phase since lower burnup pellets were partly removed for the 2nd phase irradiation.

2. Monte Carlo Burnup Calculation

Burnup calculation of IFA-609/626 (1st phase) accounting for a large fraction of total burnup period was performed by using a Monte Carlo burnup code MVP-BURN, which has been developed by Japan Atomic Energy Agency (JAEA), and consists of a continuous energy Monte Carlo code MVP[10)] and a burnup calculation module. A two-dimensional calculation model was applied to the calculation. **Figure 1**[11)] shows the assembly model, which consisted of six fuel rods (four MOX fuel rods and two UO_2 fuel rods), a cylindrical shroud of Zircaloy-2 surrounding the fuel rods and a heavy water moderator. Each fuel pellet has eleven radial regions with an equal volume except the center of the fuel pellet. Since the center part of the fuel pellet was removed to insert a thermocouple in the irradiation of IFA-702, the radius of the center part was set to be a boring radius. An axial position was selected considering the temperature measurement in IFA-702. The calculation model employs perfect reflection condition. The burnup calculations reproduced the real operation period, and the linear heat density was adjusted to achieve the burnup. A void fraction in the shroud is also determined by the linear heat density. The fuel, cladding and moderator temperatures were set to be constant through the burnup steps. Burnup for each step was selected to be 0.5 GWd/t up to 12 GWd/t, and 1 GWd/t over 12 GWd/t. The JENDL-3.3 nuclear data library[12)] was used in the calculation. The number of neutron histories per batch was 40,000, and the number of batches was 120 at each burnup step, where the initial 20 batches were skipped in tallying. The Predictor-Corrector (PC) method was used for each burnup step in the burnup calculation.

3. Deterministic Burnup Calculation

Burnup calculation by deterministic codes SRAC and PLUTON was performed in a pin cell model consisting of fuel rod and a surrounding moderator. SRAC is a general purpose neutronics code which has been developed by JAEA. Burnup calculations of 107 neutron energy groups were made using the collision probability module (Pij) of the SRAC code. Resonance cross sections were obtained by a hyper-fine energy group calculation module PEACO[13)] from 961.2 eV to a thermal cutoff energy 1.86 eV. The JENDL-3.3 nuclear data library was used.

PLUTON is a three-group neutronics code which has been also developed by JAEA. PLUTON has adopted a theoretical shape function of neutron attenuation in pellet.

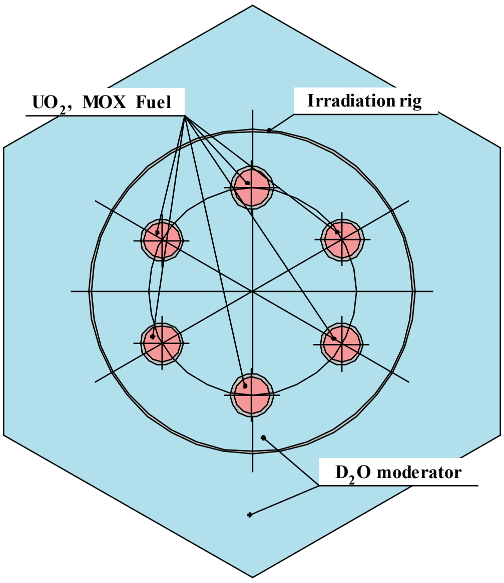

Fig. 1 Assembly model for Monte Carlo calculation in 1st phase (IFA-629)[11)]

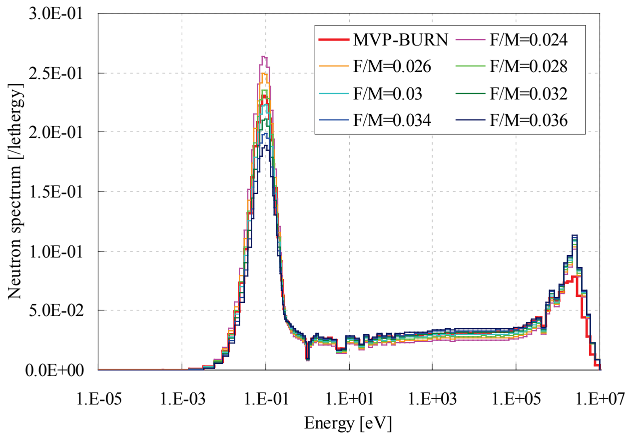

Fig. 2 Comparison of rod average spectrum for UO_2 between MVP-BURN and SRAC at the end of the 1st phase irradiation (EOC)

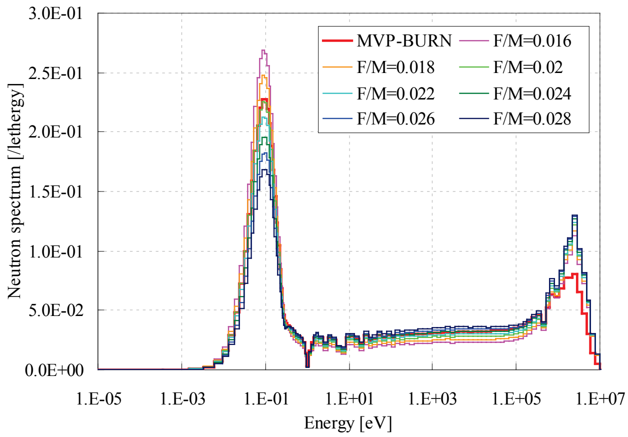

Fig. 3 Comparison of rod average spectrum in SBR-MOX between MVP-BURN and SRAC at the end of the 1st phase irradiation (EOC)

The code calculates power densities, burnup, concentrations of trans-uranium elements, plutonium build up, depletion of fissile elements, and fission products in fuel rod such as UO_2 and MOX fuels in water reactors. Calculated burnup, power densities and fission yield for Xenon and Krypton are transferred to a fuel behavior calculation code FEMAXI to evaluate a fuel and a cladding temperatures, stress and strain etc. PLUTON has been performed verification by comparison with the HBWR experiment.[5,6)]

The calculations were performed for three cases of UO_2, SBR-MOX, MIMAS-MOX fuel rods. The operation period and the linear heat density and burnup steps, the fuel, cladding and moderator temperatures were set to be same condition of the Monte Carlo calculation. However, the cell pitch was adjusted by reproducing a neutron spectrum in the assembly model calculation as mentioned in the next section. A void fraction was set to be 0%.

III. Results and Discussion

1. Equivalent Cell Pitch in Deterministic Burnup Calculation

The equivalent cell pitch was selected from comparison of neutron spectra of 107 energy groups in a lethargy unit between the MVP-BURN and SRAC calculations. The used spectra were averaged values in the fuel region at the end of the 1st phase irradiation (EOC). The burnup calculations by using a parameter as fuel to moderator ratio (F/M) with SRAC were performed to determine the equivalent cell pitch. **Figures 2** and **3** show the comparisons between MVP-BURN and SRAC for the UO_2 rod and the SBR-MOX rod as a representing MOX rod, respectively. The statistical errors (one σ) in each energy region of MVP-BURN were 0.7% for the UO_2, and 0.9% for the SBR-MOX. Root Mean Squares (RMS's) of differences in the 107 groups between MVP-BURN and SRAC were evaluated. In all the three fuel cases, the SRAC calculation shows larger values than that of MVP in the neutron energy over about 0.5 MeV, which would be due to the difference in the assembly and pin cell models. Thus determined cell pitches correspond to F/M=0.03 for the UO_2, F/M=0.022 for the SBR-MOX, and the MIMAS-MOX rods. In the following, the results of SRAC and PLUTON are those based on the determined cell pitches.

2. Radial Distributions for Burnup and Fission Reaction Rate

Figure 4 shows the calculated results of the pellet radial distribution of burnup for the UO_2, SBR-MOX, and MIMAS-MOX rods at the end of the 1st phase irradiation (EOC). It is observed that a large burnup peaking in the pellet outer region for the UO_2 and MOX pellets. The SRAC calculations show good agreement with those of MVP-BURN. The PLUTON calculations also show good agreement with those of MVP-BURN in the inside fuel pellet. In the outer pellet region, the calculated values of PLUTON are larger by about 5% than those of MBP-BURN

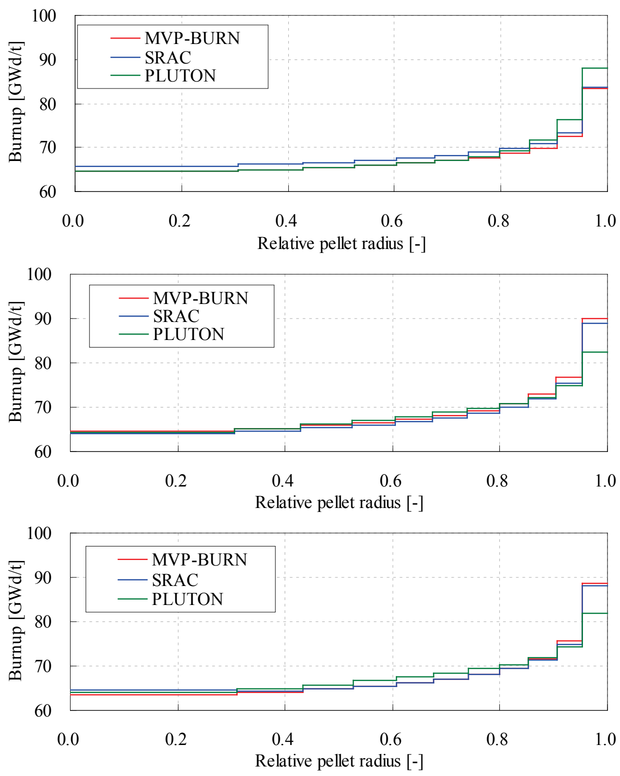

Fig. 4 Difference in radial burnup distribution between MVP-BURN, SRAC, and PLUTON at the end of the 1st phase irradiation (EOC) (Top: UO_2, Middle: SBR-MOX, Bottom: MIMAS-MOX)

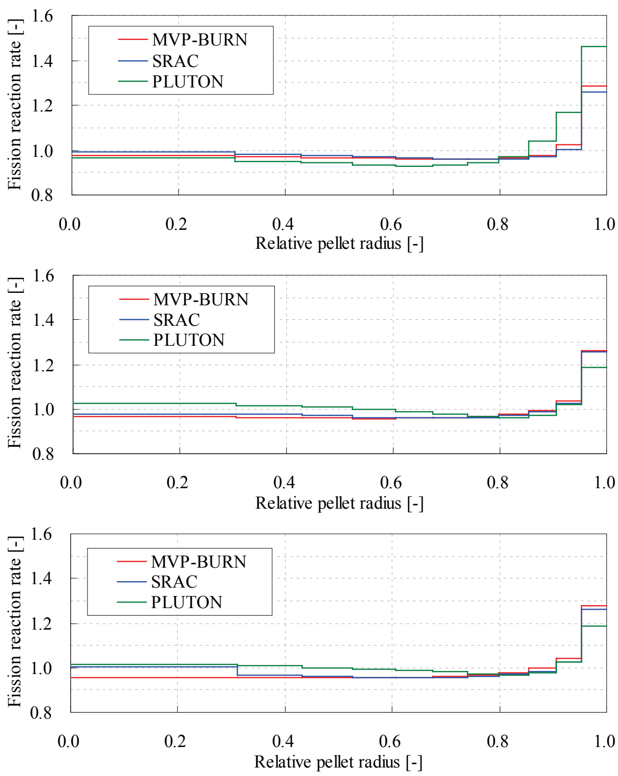

Fig. 5 Difference in radial distribution of fission reaction rate between MVP-BURN, SRAC, and PLUTON at the end of the 1st phase irradiation (EOC) (Top: UO_2, Middle: SBR-MOX, Bottom: MIMAS-MOX)

for the UO_2 fuel, and lower by about 8% for the SBR-MOX and MIMAS-MOX.

Pellet radial fission rate distributions represent pellet radial power distributions. The fission reaction rates for the radial regions were normalized so that average fission reaction rate equal to 1.0 in the fuel pellet. The statistical errors (one σ) of the fission reaction rate of each region calculated by MVP-BURN were lower than 0.2% at each burnup step. **Figure 5** shows the comparisons of the calculated fission reaction rate distributions between MVP-BURN, SRAC, and PLUTON at the end of the 1st phase irradiation (EOC) for the UO_2, SBR-MOX, and MIMAS-MOX rods**.** The radial fission rate distribution varies with pellet average burnup; however, the change in a radial burnup distribution with pellet average burnup is more moderate than that of the radial fission rate distribution. The radial fission rate is also influenced by the fissile density which inversely changes with burnup. It is also observed that a large peaking of the fission rates by the rim effect in the pellet outer region for the UO_2 and MOX pellets. The SRAC calculations show good agreement with those of MVP-BURN, however, those are larger by about 5% than those of MVP-BURN in the center of the pellet for the MIMAS-MOX rod. The calculated results by PLUTON are larger by about 15% than those of MVP-BURN in the pellet outer region for the UO_2 rod and lower by about 5% for the SBR-MOX and MIMAS-MOX rods.

The pellet radial fission rate distributions are correlated to those of radial burnup distributions. In the UO_2 rod, the PLUTON calculation gives a larger burnup and, therefore, a larger buildup of Pu in the pellet outer region than those of MVP-BURN so that the fission rates in the outer region are larger than those of MVP-BURN. In the MOX rods, the PLUTON calculation gives a lower burnup in the pellet outer region, which leads to a higher residual of Pu and also a lower buildup of Pu. It is assumed that the latter effect exceeds the former and the fission rates are smaller than those of MVP-BURN.

IV. Summary

JNES has been working on the irradiation test program of high-burnup MOX fuel at HBWR. As part of analysis of the measured temperature data of the irradiated pellets, fission rate distributions in the pellet radial direction was analyzed by using the assembly model with the Monte Carlo burnup code MVP-BURN. In addition, the calculated results of deterministic burnup codes SRAC and PLUTON for the same problem were compared with those of MVP-BURN to evaluate their accuracy. The calculation results show a large peaking in the burnup and fission rates at the pellet outer region for the UO_2 and MOX pellets. The SRAC calculations give very close results to those of the MVP-BURN. On the other hand, the PLUTON calculations show larger burnup for the UO_2 and lower burnup for the MOX pellets in

the pellet outer region than those of MVP-BURN, which lead larger fission rates for the UO_2 and lower fission rates for the MOX pellets.

The calculation methods adopted in this study will be validated by using measurement data of nuclide distributions in the pellet radial direction, which is scheduled to be obtained in the future.

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