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Calculational Benchmark Problems for VVER-1000 Mixed Oxide Fuel Cycle

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ABSTRACT

Specifications were provided jointly by the Russian and American participants in the Fissile Materials Disposition Program (FMDP) for a calculational benchmark problem set for fissile materials disposition with a VVER-type reactor. Both fresh and spent fuel storage were required; and the study used the following fuels: mixed oxide (MOX) with weapons-grade plutonium, MOX consisting of civil plutonium fuel (reactor-grade) and the traditional uranium dioxide (UOX) low-enriched fuel. Task I was a study of criticality safety in fresh fuel storage for the three types of fuel and is not discussed in this paper. Task II was a three-part task studying the shielding and radioactive characteristics when the fissile assembly is transported. Task IIa was a study of the radioactive characteristics of a fissile assembly of fresh fuel without a container. Task IIb was a study of a fissile assembly of fresh fuel within a cask. Task IIc was a study of a burned fissile assembly within a spent fuel cask. The cask model is typical of those used to transport fissile assemblies of spent fuel. Modules from the SCALE[1] code system were used for all calculations.

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Task IIa was a study of the radioactive characteristics of fresh fuel in a fissile assembly without a container (cask). The geometry model had the same lattice structure described above; however, the assembly of fresh fuel was ‘dry’. The temperature of the fissile assembly was 300 K. Calculations of dose rates at the surface of the assembly and at 0.5, 1 and 2 meters from this surface were done for all three fuel types using the one-dimensional SCALE module SAS1. The neutron- and gamma-source strengths were calculated using the SAS2 code sequence.

Task IIb was a shielding and heat generation study of fresh fuel within a cask model typical of those used to transport fissile assemblies of fresh fuel. The cross-section libraries used were the ENDF/B-IV 27 group burnup library and the 27 neutron, 18 gamma group library. Dose rates at the surface of the cask and at 0.5, 1 and 2 meters from the surface were calculated using the SAS2 module of SCALE. All three types of fuel were evaluated. Although the total sources for MOX-W and MOX-R are about an order of magnitude different, the gamma dose rates for MOX-R are higher by approximately a factor of 3, and the neutron dose rates are higher by approximately a factor of 9. In addition to calculating dose rates, a heat generation study for the fresh fuel cask was done using the ORIGEN module from SCALE.

Task IIc was a study of a cask model typical of those used to transport fissile assemblies of spent fuel. The cask contained 12 fissile assemblies. A pin irradiation with a burnup of 60 GWd/MTHM at average power of 166 W/cm was done. The fuel temperature was T=1027 K, and the temperature of the clad and the borated-light-water coolant was T=579 K. Each of the three types of fuel was analyzed. Dose rates at the surface of the cask and at 0.5, 1 and 2 meters from the surface were calculated using the SAS2 module of SCALE. Initially the gamma dose rates were thought to be relatively independent of the fuel type. Thus, the higher gamma dose rates for the MOX fuels were surprising. Investigation of these differences revealed that the relatively higher neutron leakage with the MOX fuels as compared with UOX fuel produced more captured gammas; and since nearly 90% of the gammas are captured gammas, the resulting dose rates for MOX are higher than for low-enriched uranium. In addition to calculating dose rates, a heat generation study for the spent fuel cask at various times of disposition was done using the ORIGEN module from SCALE.

ORNL results were compared to the results that were obtained by the Russians using several computer code sequences. The Russian cross-section libraries had different gamma group structures than the libraries used at ORNL; and, therefore, direct comparisons of gamma source results were inconclusive. Additional analysis using the same group structures would likely resolve this issue. Dose results for fresh fuel compared favorably with the Russian results except for UOX cases in which the Russians took into account the daughter products of ^{238}U while ORNL made no equilibrium assumptions. The ORNL dose predictions for spent fuel compared reasonably well with those of the Russians. Complete ORNL results are published in ref. 2.

REFERENCES

1. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, NUREG/CR-0200, Rev. 5 (ORNL/NUREG/CSD-2/R5), Vols. I, II, and III, March 1997. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-545.

2. M. B. Emmett, *Calculational Benchmark Problems for VVER-1000 Mixed Oxide Fuel Cycle*, ORNL/TM-1999/207, Lockheed Martin Energy Research Corp., Oak Ridge National Laboratory, in preparation.