Safety Analyses for MOX Fuel in a VVER-1000 Reactor

J. J. Carbajo, G. L. Yoder
Oak Ridge National Laboratory
Oak Ridge, Tennessee, 37831

and

V. K. Ivanov
RRC- Kurchatov Institute
Moscow, Russia

INTRODUCTION

This summary presents safety thermal-hydraulic (T-H) calculations performed for a reactor core with mixed uranium-plutonium oxide (MOX) fuel at Balakovo nuclear power plant, a site with four VVER-1000 (V320) units located near Saratov, Russia. Excess plutonium from weapons will be converted into MOX fuel and burned at this power plant. It is planned to start loading 3 MOX fuel assemblies in the core and gradually increase the number of MOX assemblies until approximately 41% of the core is utilizing MOX fuel.

The Balakovo reactors are similar to the pressurized water reactors (PWRs) built in the United States (US), with the main difference (regarding T-H modeling) being horizontal steam generators (SG) versus vertical SGs in US PWRs. The reactor has four loops, four main primary coolant pumps, and four steam generators. The core contains a total of 163 hexagonal assemblies.

These thermal-hydraulic calculations discusses herein are required by the Russian Nuclear Regulatory Authority (GosAtomNadzor or GAN) as part of the licensing submittal before MOX is loaded in the reactor. A total of about 40 different transients need to be considered and a selection of six of them is currently being analyzed. The six selected transients are: trip of the 4 primary loop pumps, locked rotor of one primary loop pump, uncontrolled withdrawal of a regulating control rod group, control rod ejection, loss of coolant accident (LOCA), and main steam line break. Calculations are being performed in Russia by the Russian Research Center-Kurchatov Institute (RRC-KI) and OKB-Gidropress (OKB-GP), and by Oak Ridge National Laboratory (ORNL) in the US.

Different codes are being employed for these calculations. The Russian codes DINAMICA-97 and TETCH-M-97 [Ref. 1] are being employed by OKB-GP. RELAP5 MOD3.2 [Ref. 2], RELAP5 MOD3.3 [Ref. 3], and RELAP5-3D [Ref. 4] are being used by ORNL. RRC-KI is using RELAP5 MOD3.2 and BIPR8KN [Ref. 5].

The RELAP5-3D [Ref. 4] code is a 3-Dimensional (3-D) version of the RELAP5 MOD 3.2 code [Ref. 2] with the 3-D neutron kinetics model based on the NESTLE code [Ref. 6]. BIPR8KN is also a 3-D kinetic code that has been coupled to RELAP5 MOD3.2 to perform 3-D kinetic calculations by KI.

DESCRIPTION OF THE WORK

The RELAP5 decks used in these studies were prepared by RRC-KI and ORNL. Both, point and 3-D kinetics calculations have been completed with some results presented in the past, but these previous calculations employed either UO_2 fuel or only 3 MOX assemblies. Pump transients were modeled in Ref. 7, and LOCAs were modeled in Ref. 8. Calculations with point and 3-D kinetic models with 3 MOX assemblies were presented in Ref. 9 for a control rod ejection accident and in Ref. 10 for main steam line break accidents. Ref. 11 shows results for a multidimensional thermal-hydraulic model.

Calculations for 1/3 of the core MOX have been completed and compared to results for a core with only 3 MOX fuel assemblies and to a core with only UO_2. The current equilibrium core loading is 54 MOX assemblies (1/3 of the core) and 109 UO_2 assemblies with 12 month refueling cycles. UO_2 assemblies have up to 4.2% enrichment, and MOX assemblies up to 3.62% Pu. The fuel design includes 6 Uranium-Gadolinium (U-Gd) poison rods in the UO_2 assemblies and 16 U-Gd poison rods in the MOX assemblies. The final design of the MOX core is still being determined, and the number of MOX assemblies may be increased up to 41% of the total core and the refueling cycle length may be increased to 18 months.

RESULTS

The results for most transients for 1/3 core MOX are very similar to previous results for

3 MOX assemblies or for UO_2 cores, with some differences for the control rod ejection accident. Point kinetic results with the code RELAP5 MOD3.2 for a *control rod ejection accident* are presented for 1/3 core MOX (54 assemblies), for a core with 3 MOX assemblies, and for a core with only UO_2 . The reactor is operating at 104% nominal power when a control rod is ejected at 50 s into the transient. The worth of the ejected control rod is conservatively assumed to be 0.25%. Reactor power increases after the control rod is ejected and the calculated peak power is higher in the core with 1/3 MOX than in the other cores because of the higher reactivity of the MOX fuel (Fig. 1). The results for 3 MOX assemblies and for the UO_2 core are virtually the same. The shutdown control rods are inserted at 51 s. All of the calculated parameters are within allowed safety limits. Additional calculations with 3-D kinetics are underway because this transient together with main-steam-line-breaks [Ref. 10] are space dependent transients that are better analyzed with 3-D codes. Also, sensitivity calculations are being performed varying fuel burnup and thermal properties (to account for uncertainties), and for LOCA transients, LOCA sizes and locations are varied.

CONCLUSIONS

In conclusion, this paper reviews safety calculations required for licensing MOX fuel in VVER-1000 reactors. All of the calculations completed have shown that results for 1/3 core MOX are similar to results for UO_2 cores. Based upon these calculations, MOX fuel can be burned in VVER-1000 reactors without exceeding any safety limits.

REFERENCES

1. International Atomic Energy Agency, **"**Guidelines for Accident Analysis of WWER Nuclear Power Plants," IAEA-EBP-WWER-01 (1995).

2. Idaho National Engineering and Environmental Laboratory, RELAP5 Code Development Team, "RELAP5/MOD3.2 Code Manual," NUREG/CR-5535, INEL-95/0174 (1995).

3. Information Systems Laboratory, Inc., Nuclear Safety Analysis Division, "RELAP5/MOD3.3 Code Manual," NUREG/CR-5535, Rev. 1 (2002).

4. Idaho National Engineering and Environmental Laboratory, RELAP5-3D Code Development Team, "RELAP5-3D Code Manual," INEEL-EXT-98-00834, Rev. 2.0 (2002).

5. V. K. IVANOV et al, "Results of Coupled Dynamic Calculations of Neutron Physics and Thermalhydraulics of a VVER, with 3-Dimension Core Representation," *Proceedings of the 12th Symposium of AER*, 543, Sunny Beach, Bulgaria (2002).

6. R. M. AL-CHALABI, et al, "NESTLE, A Nodal Kinetics Code," *Trans. Am. Nucl. Soc.*, **68** (1993).

7. J. J. CARBAJO et al, "VVER-1000 Thermal-Hydraulic Transients Modeled with the RELAP5/MOD3.2 Code," *Trans. Am. Nucl. Soc.*, **80**, 185 (1999).

8. J. J. CARBAJO et al, "LOCA Simulation in a VVER-1000 with MOX Fuel," *Trans. Am. Nucl. Soc*., **81**, 200 (1999)

9. J. J. CARBAJO et al, "Simulation of a Control Rod Ejection in a VVER-1000 Reactor with the Program RELAP5-3D," *Proceedings of the ICONE-9 Meeting*, Nice, France (2001).

10. J. J. Carbajo et al, "Main-Steam-Line-Break Accident Analyses in a VVER-1000 Reactor," *Trans. Am. Nucl. Soc.,* **85**, 124 (2001).

11. E. POPOV et al, "VVER-1000 Three Dimensional Model Assessment," *Trans. Am Nucl. Soc..* **88,** 290 (2003)

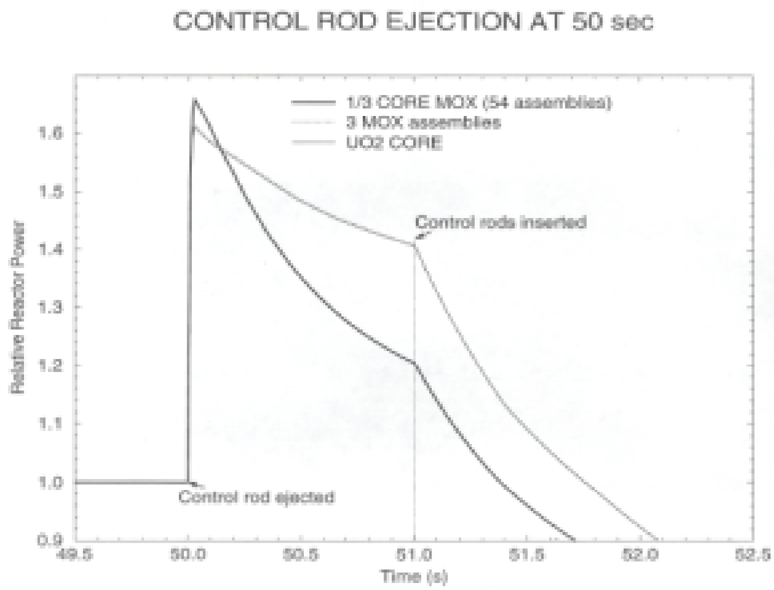

Fig. 1. Calculated relative reactor power after the control rod is ejected