CONF-870102--2

THE ROLE OF INTEGRAL EXPERIMENTS AND NUCLEAR CROSS SECTION EVALUATIONS IN SPACE NUCLEAR REACTOR DESIGN

David L. Moses

Oak Ridge National Laboratory*

P.O. Box X

Oak Ridge, TN 37831

Commercial 615/574-6103
FTS 624-6103

Richard D. McKnight

Argonne National Laboratory

9700 S. Cass Avenue

Argonne, IL 60430

Commercial 312/972-6088
FTS 972-6088

CONF-870102--2

DE87 007004

MASTER

Manuscript prepared for:

Fourth Symposium
on Space Nuclear Power Systems
Albuquerque, New Mexico
12-16 January 1987

Author running head: D. L. Moses and R. D. McKnight

Title running head: INTEGRAL EXPERIMENTS AND NUCLEAR CROSS SECTIONS

Address correspondence to: D. L. Moses

*Operated by Martin Marietta Energy Systems under Contract No. DE-AC05-84OR21400 with the U.S. Department of Energy.

THE ROLE OF INTEGRAL EXPERIMENTS AND NUCLEAR CROSS SECTION EVALUATIONS IN SPACE NUCLEAR REACTOR DESIGN

David L. Moses
Oak Ridge National Laboratory
P.O. Box X
Oak Ridge, TN 37831
Commercial 615/574-6103
FTS 624-6103

Richard D. McKnight
Argonne National Laboratory
9700 S. Cass Avenue
Argonne, IL 60430
Commercial 312/972-6088
FTS 972-6088

At the previous symposia on Space Nuclear Power Systems, two papers [1,2] were presented which acknowledged the importance of the nuclear and neutronic properties of candidate space reactor materials to the design process, and two other papers[3,4] were presented which acknowledged the importance of using benchmark reactor physics experiments to verify and qualify analytical tools used in design, safety and performance evaluation. However, there are regulatory/legal precedents which make the case for using neutronic benchmark experiments somewhat stronger than merely that of good design practice.

With the single exception of "space-based reactors", the Department of Energy (DOE) implementing orders[5,6] stipulate that the program requirements for new ground-based reactors and presumably for pre-launch handling of space reactor units are to comply with the applicable federal regulations, specifically, the Code of Federal Regulations, Title 10, Part 50 (including subpart 50.34) and Part 100. Further, for all new reactors except the "space-based reactors," the safety analysis reports are required to follow the Nuclear Regulatory Commission's (NRC's) guidelines on the Standard Format and Content of Safety Analysis Reports.[7-9] Space-based reactors are to use "criteria consistent with space applications" (Ref. 5); however, the DOE policy statement[10] on these criteria merely directs the reactor system contractor to use as appropriate those documents referenced in the DOE implementing order[5] and to include a section on the nuclear design in the safety analysis report. As indicated above, the documents cited in the DOE order[5] are the current NRC guidelines for safety analysis documentation. The NRC guidelines (specifically, Section 4.3 of Refs. 7, 8 and 9) are explicit with regard to documenting the use of reactor physics experimental data, including critical experiments, to establish the accuracy and uncertainty in the prediction of power distributions, reactivity coefficients, shutdown margins, control worths, and subcriticality during fuel handling. These reactor physics parameters plus core lifetime are also expected to be issues in establishing "criteria consistent with space applications," and so the existing quality assurance guidance for demonstrating and documenting the safety and nuclear design performance of ground-based reactors is expected to be equally applicable to space-based reactors.

Since June 1966, the Cross Section Evaluation Working Group (CSEWG) has acted as an interagency forum for the assessment and evaluation of nuclear reaction data used in the nuclear design process.[11] Although CSEWG activities are not meant to support the specific safety analyses of particular reactor plants, CSEWG does direct a nuclear data testing program which quantifies the accuracy and uncertainty of nuclear data for those materials which are of most interest to the nuclear industry. Under project funding from the DOE, the CSEWG data testing has involved the specification[12] and calculation[13] of benchmark experiments for fast reactor criticality and reaction rate data, thermal reactor criticality and reaction rate data, shielding and dosimetry. The calculations of the benchmark experiments have been made by participants from industry and the national laboratories employing the data processing methods and core analysis tools which are used widely for commercial reactor design and safety analysis. Therefore, the results of such testing can be used where appropriate in making the safety case as required by the regulations and NRC guidelines. The specification of the fast reactor benchmark experiments preceded the issuance of the industry standards[14,15] for acceptance and use of "reference data", but the CSEWG benchmarks exceed the minimum acceptance criteria for such data.

The current set of CSEWG fast reactor benchmarks were developed to support the breeder reactor program. As such, the major constituent materials of the critical experiments are not the same as those anticipated for space reactor core application. There are minor constituents of molybdenum and carbon in a few of the critical experiments, and boron-10 has been extensively examined as a control poison. However, a number of the candidate structural materials for space reactors have been subjected to central worth measurements. The state of the art for predicting central worth for these materials is reflected in Table 1. The data presented in Table 1 represent a single set of calculations by Los Alamos National Laboratory using ENDF/B-V nuclear data.[16] Problems experienced in accurately calculating the central worths have been addressed by studies at Argonne National Laboratory.[17,18] These problems center on the inability of the analytical methods to simulate the precise configuration in which the experiment is performed. Some experiments, such as BIG TEN, have been found to employ techniques that minimize the calculation-to-experiment error observed in the worth of small, heavy metal samples; however, as shown in Table 1, this observation does not apply to lighter materials for which neutron scattering is the important effect.

Thus, although the current CSEWG benchmarks provide a place to start in assuring the accuracy and uncertainty of nuclear data important to space reactor applications, there need to be developed better benchmarks to quantify the accuracy and uncertainty of the differential and integral data. CSEWG is aware of the numerous critical experiments performed in support of space reactor programs during the 1960s and early 1970s, and the authors have compiled an extensive library of these data sources. Further effort will be required to develop from these a set of benchmarks consistent with the basis for the CSEWG benchmark specifications[12] and industry standards for reference data.[14,15] The development of such benchmarks and the subsequent data testing will require interagency support and participation in order to be useful and effective. New experiments are most certainly required to meet the regulatory guidelines for the safety analysis of specific reactor concepts. CSEWG's role is to support industry in making the most effective use of existing and new data sources so as to minimize the cost and time involved in the design effort and licensing-equivalent certification.

Acknowledgments

This work has been performed under the auspices of the U.S. Department of Energy.

References

1. Condliff, A., and R. J. Marsh (1987) "Selection of Structural Materials for Space Nuclear Power Generators," in Space Nuclear Power Systems 1986, M.S. El-Genk and M.D. Hoover eds., Orbit Book Co., Malabar, FL.

2. Little, W., and J. Barner (1987) "Neutronics Characteristics of Space Power Reactors," in Space Nuclear Power Systems 1986, M.S. El-Genk and M.D. Hoover eds., Orbit Book Co., Malabar, FL.

3. Bartholomew, R. J. (1986) "Monte Carlo Simulations for Reactor Design and Criticality Safety," in Space Nuclear Power Systems 1985, M.S. El-Genk and M.D. Hoover eds., Orbit Book Co., Malabar, FL.

4. McFarlane, H. F., et. al. (1987) "Recent BeO-Reflector- Controlled Reactor Experiments in ZPPR", in Space Nuclear Power Systems 1986, M.S. El-Genk and M.D. Hoover eds., Orbit Book Co., Malabar, FL.

5. U.S. DOE (1981) Environmental Protection, Safety, and Health Protection Program for DOE Operations, DOE 5480.1A, U.S. Department of Energy, Washington, D.C., August 1981.

6. U.S. DOE (1981) Safety Analysis and Review System, DOE 5481.1A, U.S. Department of Energy, Washington, D.C., August 1981.

7. U.S. NRC (1978) Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, Regulatory Guide 1.70, Revision 3, U.S. Nuclear Regulatory Commission, Washington, D.C., November 1978.

8. U.S. AEC (1973) HTGR Edition of the Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, WASH-1266, U.S. Atomic Energy Commission, Washington, D.C., July 1973.

9. U.S. AEC (1974) Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LMFBR Edition, U.S. Atomic Energy Commission, Washington, D.C., February 1974.

10. U.S. DOE (1982) Nuclear Safety Criteria and Specifications for Space Nuclear Reactors, OSNP-1 REV 0, U.S. Department of Energy, Washington, D.C., August 1982.

11. Howerton, R. J. (1985) "A Review of Nuclear Reaction Data Evaluation in the United States," in Transactions of the American Nuclear Society, 49:192-193.

12. BNL (1974) Cross Section Evaluation Working Group Benchmark Specifications, BNL-19302 (ENDF-202), Brookhaven National Laboratory, Upton, NY, November 1974.

13. BNL (1982) Benchmark Data Testing of ENDF/B-V, BNL-NCS-31531 (ENDF-311), Brookhaven National Laboratory, Upton, NY, August 1982.

14. ANS (1983) American National Standard for the Determination of Neutron Reaction Rate Distributions and Reactivity of Nuclear Reactors, ANSI/ANS-19.3-1983, American Nuclear Society, La Grange, IL, June 1983.

15. ANS (1978) American National Standard Requirements for Reference Reactor Physics Measurements, ANSI/ANS-19.5-1978, American Nuclear Society, La Grange, IL, November 1978.

16. Kidman, R. B. (1981) ENDF/B-V, LIB-V, and the CSEWG Benchmarks, LA-8950-MS, Los Alamos National Laboratory, Los Alamos, NM, August 1981.

17. Collins, P. J., R. D. McKnight and S. G. Carpenter (1982) "Critical Review: Fast Reactor Benchmarks," in Transactions of the American Nuclear Society, 41:573-574.

18. Schaefer, R. W., and K. S. Smith (1985) "Calculation of Small Sample Worths by Monte Carlo," in Transactions of the American Nuclear Society, 49:430-431.

Table 1. LANL ENDF/B-V Calculation-to-Experiment Ratios for Selected Central Worth Measurements in CSEWG Benchmarks

CSEWG Fast Reactor Benchmark Number/Name	H-1	Li	Li-6	Li-7	Be	B	B-10	C	O-16	Zr	Nb-93	Mo	Hf	Ta-181	V	U 235	U 238
1 JEZEBEL	0.0421				1.0133		0.9968	1.1619	1.1134	0.7835	1.1791[a]	1.0840		1.1080	1.0590	1.0712	0.9637
3 ZPR-3/48							1.0407	5.9877				1.3464		1.0373		1.1533	1.1643
4 ZEBRA-3	1.5749		0.9062			1.3755	0.9678							-1.0755		1.1464	1.0785
5 GODIVA	0.9833				0.9854		0.8345	0.5273			0.9860[a]					0.9798	1.0028
6 VERA-1B	1.0748															1.0076	1.5535
7 ZPR-3/6F	0.6388				1.0731		0.9659	0.7466		19.1536	1.1658	-1.4212	1.4778	1.1330		0.8800	2.9267
8 ZPR-							0.9308	1.6832	1.2867			1.2699		0.9754		1.1101	1.0094
9 ZPR-								0.2382			1.1075	1.1661		1.0178		0.9885	0.9337
10 ZEBRA-2	1.0495	0.6415	0.8330			0.6077	0.8730	0.5967					1.1778	1.2527		1.0924	1.0337
11 ZPPR-2							0.9648	1.3331			1.2815	1.1275		0.8715	1.6355		
12 ZPR-6/7							1.2175	1.3303				1.3167		1.4453		1.1127	1.0388
13 ZPR 3/56B							0.8957	1.8523						0.9738		1.0582	1.1487
15 ZPR-6/6A							0.9436	0.4114						1.7377		1.1191	1.1739
16 SNEAK-7A							0.9026							0.6633		1.0176	1.0898
17 SNEAK-7B							0.8962							0.7567		1.0185	1.0361
18 ZPR-9/31			1.1165				1.0287	2.3430								1.1413	1.0308
20 BIG TEN	1.0935		0.9186	1.3112	1.4409		0.9099	1.3482	1.4030							1.0120	1.0191
20 BIG TEN			0.8936[a]				0.8443[a]										

a Central Reaction Rate Ratios Normalized to U-235 Fission.