CONF-840614--90

BENCHMARKING CRITICALITY SAFETY CALCULATIONS WITH SUBCRITICAL EXPERIMENTS

John T. Mihalczo

Oak Ridge National Laboratory*
Oak Ridge, Tennessee 37831

CONF-840614--90

DE84 014264

Summary of paper to be presented at
the 1984 Annual Meeting
American Nuclear Society
June 3-8, 1984
New Orleans, Louisiana

---

*Operated by Union Carbide Corporation under contract W-7405-eng-26 with the U.S. Department of Energy.

BENCHMARKING CRITICALITY SAFETY CALCULATIONS WITH SUBCRITICAL EXPERIMENTS*

John T. Mihalczo

Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831

Calculation of the neutron multiplication factor at delayed criticality may be necessary for benchmarking calculations but it may not be sufficient. The use of subcritical experiments to benchmark criticality safety calculations could result in substantial savings in fuel material costs for experiments. In some cases subcritical configurations could be used to benchmark calculations where sufficient fuel to achieve delayed criticality is not available. By performing a variety of measurements with subcritical configurations, much detailed information can be obtained which can be compared directly with calculations. If this variety of detailed information is confirmed by calculation, then the result should be an adequate benchmark of the calculational method. In many plant applications the neutron multiplication factor of a subcritical configuration in a process operation, as it exists in the plant, should be of interest to safety specialists, as well as what it takes to make such configurations achieve delayed criticality. Criticality safety calculations are used to assess the safety of plant designs and to size components without knowledge of the accuracy of the calculations for the actual application. The assumption is usually made that if calculations are valid for and benchmarked with critical experiments they are valid for subcritical configurations. Properties such as neutron energy spectra and fission rate distributions may depend on subcriticality. To compensate for this inadequacy conservatism is usually used. This paper discusses several measurements that can be performed with subcritical assemblies and presents examples that include comparisons between calculation and experiment where possible. Where not, examples from critical experiments have been used but the measurement methods could also be used for subcritical experiments.

---

*Research sponsored by the U.S. Department of Energy under contract W-7405-eng-26 with the Union Carbide Corporation.

In a subcritical experiment a neutron source is inserted into the assembly and the fission chain multiplication process provides a steady state flux distribution throughout the assembly. Neutron detectors can provide steady state count rates for a variety of source-detector configurations. This type of data is usually obtained in inverse multiplication experiments.[1] Fixed source calculations provide neutron fluxes which can be combined with nuclear cross sections of the detectors and detector geometry to calculate count rates for the experimental source-detector geometry.

The energy distribution of neutrons both internal to the subcritical configuration and leaking from the assembly can be measured by a variety of techniques. Threshold reactions and the use of bare- and cadmium-covered foils can provide integral data which can be calculated by the appropriate integrals of the neutron flux over the reaction cross section for the foil geometry. The energy distribution of neutrons using proton recoil techniques can be measured from 1 Kev to 10 Mev. Methane-filled ionization chambers have been used successfully in critical experiments for internal spectrum measurements above 1 Kev analagous to the method of Bennett.[2] For neutron spectrum measurements >0.5 Mev proton recoil methods using liquid scintillators[3,4] would have greater efficiency. The energy distribution of neutrons can be obtained directly from the calculated fluxes as a function of energy and compared with the experimental data.[2]

The time dependence of the neutron flux can be measured for subcritical assemblies using pulsed neutron techniques.[5,6,7] Rossi-α type measurements[8] with steady state neutron sources can also be used to obtain time dependent information. Combining the pulsed neutron technique with neutron spectrometry the energy distribution of neutrons as a function of time after the neutron pulse could be measured. The data from pulsed neutron type experiments can be calculated by several methods[9,10] and in some cases by diffusion theory.[11] For experiments where the prompt neutron decay can be described by a single exponential decay, multigroup methods can be used to determine the decay constant.

By varying neutron source position in a subcritical assembly, the spatial distribution of the neutron importance can be measured. Activation of fission foils distributed appropriately in the assembly can provide the spatial distribution of the fission rate. Both the neutron importance and the fission density are quantities that can be calculated and compared with measurements.[12]

Recent measurements using the ^{252}Cf-source-driven neutron noise analysis method to obtain the subcriticality [13-16] also provide data which can be used to verify calculational methods. The ^{252}Cf-source-driven neutron noise method obtains the subcriticality from ratios of measured power spectral densities ($G^*_{12} \cdot G_{13} / G_{11} \cdot G_{23}$), where subscript 1 refers to an ionization chamber (the source) containing ^{252}Cf, that is placed in or adjacent to the subcritical assembly and that provides source neutrons to initiate the fission chain multiplicative process. The subscripts 2 and 3 refer to a pair of detectors that are also located in or near the assembly and that detect neutrons from the fission chains. The main advantages of this method are: (1) the subcriticality is obtained from measurements only at the state of interest, and thus a reference measurement near critical is not required; and (2) the interpretation does not depend on relative or absolute value of the detection efficiency or the intensity of the inherent source. The cross power spectral densities (CPSD) between the detectors and the ^{252}Cf source and between the neutron detectors can be calculated using the JPR kinetics code.[17] The dependence of the CPSD on frequency is related to the neutron energy spectrum in the subcritical assembly. The ratio of spectral densities can be interpreted to obtain the neutron multiplication factor which can then also be used to verify calculational methods.

The application of all of these existing measurement techniques to a subcritical assembly should provide sufficient measured data to benchmark calculations. For many configurations of fissile materials with $k_{eff} \approx 0.8$ the fissile inventory is about half that required for a critical experiment. The material costs for these experiments could be

reduced since for experiments with some new fissile materials the materials costs are much more than half the cost of the experiment. With limited safety research funds available this may be a method for more cost effective utilization. Most of these experimental data can be calculated by existing calculational methods which have been in use for some time; i.e., diffusion,[18] transport theory,[19] and Monte Carlo methods.[20] For three dimensional diffusion and transport theory applications, some code development would be required for direct calculation of the CPSD from the ^{252}Cf-source-driven neutron noise analysis method.

REFERENCES

1. American National Standard for Safety in Conducting Subcritical Neutron-Multiplication Measurements In Situ, ANS-8.6, ANSI N16.3-1975, Revision of N16.3-1969.

2. T. J. Yule and E. F. Bennett, "Measured Neutron Spectra in a Number of Uranium and Plutonium-Fueled Reactor Assemblies," Nucl. Sci. Engr. 46, 236-243 (1971).

3. V. V. Verbenski, W. R. Burrus, Nucl. Inst. & Methods 67, 181 (1979).

4. J. H. Thorngate, D. W. Reuppel, R. V. Griffith, "Neutron and Gamma-Ray spectrum Measurements at the Los Alamos Critical Assembly Sheba, Trans. Amer. Nucl. Soc. 39, 557 (1981).

5. L. Passel, J. Bengston, D. C. Blair, UCRL 4808 (1957).

6. Proceedings of the Symposium of Pulsed Neutron Research IAEA, Karlsruhe, (May 1965).

7. J. D. Orndoff, Nucl. Sci. Engr. 2, 450-460 (1959).

8. J. T. Mihalczo, Nucl. Sci. Engr. 53, 393-414 (1974).

9. S. A. Dupree, H. A. Sandmeier, G. E. Hansen, W. W. Engle, and F. R. Mynatt, "Time-Dependent Neutron and Photon Transport Calculations Using the Method of Discrete Ordinates," LA 4557-ORNL-4462, (August 1970).

10. M. B. Emmett, "The Morse Monte Carlo Radiation Transport Code System," ORNL 4972, (February 1975).

11. K. S. Smith and A. F. Henry, "Analytic NODAL Method for Solving the Two Group, Multidimensional, Static, and Transient Neutron Diffusion Equations," Nucl. Engr. Thesis, Department of Nucl. Eng., MIT, Cambridge, MA, (February 1979).

12. J. T. Mihalczo, "Neutron Importance and Fission Density in Uranium-235-Enriched Uranium and Plutonium Metal Spheres," Nucl. Sci. Eng., 56, 271-290 (1975).

13. J. T. Mihalczo, V. K. Paré, G. L. Ragan, M. V. Mathis, and G. C. Tillett, "Determination of Reactivity from Power Spectral Density Measurements with ^{252}Cf," Nucl. Sci. Eng., 60, 29 (1978).

14. W. T. King and J. T. Mihalczo, "Power Spectral Density Measurements with ^{252}Cf for a Light-Water-Moderated Research Reactor," Trans. Am. Nucl. Soc., 33, 796 (1979).

15. J. T. Mihalczo and W. T. King, "Californium-^{252}-Source-Driven Neutron Noise Method for Measuring Subcriticality of Submerged HFIR Fuel Elements," Trans. Am. Nucl. Soc., 43, 408 (1982).

16. J. T. Mihalczo, R. C. Kryter, and W. T. King, "Preliminary Investigation of ^{252}Cf-Driven Neutron Noise Analysis for Subcritical Fuel Solution Systems," Trans. Am. Nucl. Soc., 38, 359 (1981).

17. J. A. Renier, "Multi-Group, Multi-Dimensional Investigations of the Power Spectral Densities of the Georgia Tech Research Reactor and the Fast-Thermal Argonaut Reactor," PhD Dissertation, Georgia Institute of Technology, Atlanta, GA (1976).

18. D. R. Vondy, T. B. Fowler, and G. W. Cunningham, "VENTURE: A Code Block for Solving Multigroup Neutronics Problems Applying the Finite-Diffusion-Theory Approximation to Neutron Transport," Version II, ORNL-5062, Oak Ridge National Laboratory, Oak Ridge, TN (October 1975).

19. W. A. Rhodes, D. B. Simpson, R. L. Childs, W. W. Engle, Jr., "The DOT-IV Two Dimensional Discrete Ordinates Transport Code with Space-Dependent Mesh and Quadrature," Oak Ridge National Laboratory, TM-6529 (1979).

20. L. M. Petrie and N. F. Cross, "Keno-IV: An Improved Monte Carlo Criticality Program," ORNL-4938, Oak Ridge National Laboratory, Oak Ridge, TN (November 1975).