CONCLUSIONS

If it is not possible to separate the systematic and random effects which must be allowed for, prudent use of Criterion 1 is acceptable.

If separation of systematic and random contributions is possible, use of Criterion 2 is preferable.

Criterion 3 should be used if a more precise statistical treatment is required, because Student's t distribution is used to calculate confidence limits for k_{eff}. The formula needed for number of degrees-of-freedom has been derived.

Criterion 3 could be used in analyses of total risk.

Data for use with the criterion must be appropriate to the particular situation being studied.

Additional statistical refinements may not lead to real improvements in the quality of the final information.

1. K. C. RUSHTON, "The Monte Carlo Code MONK–A *Guide to Its Use for Criticality Calculations*," SRD R88 (1978).

2. V. S. W. SHERRIFFS, "MONK–A General Purpose Monte Carlo Neutronics Program," SRD R86 (1977).

3. G. WALKER, "The Monte Carlo Code MONK–Validity for Use in Criticality Calculations," SRD R87 (to be published).

4. D. W. ANDERSON, D. J. WESTERN, and W. MARSHALL, "A Criticality Safety Criterion for CEGB AGR Fuel," GDCD/NP 1109 (1977).

5. K. C. KENDALL, "The Use of the Code MONK on Some Critical Assemblies to Support the Validation of MONK for Critical Conditions," TCWP/P121/8 (1974).

6. W. MARSHALL, "A Discussion of the Statistical Treatment Used in the CEGB Criticality Safety Criterion for AGR Fuel," NHS/N40/79 (1979).

7. B. L. WELCH, *Biometrika*, **34**, 28 (1947).

2. Sheba: A Solution Critical Assembly, *R. E. Malenfant, H. M. Forehand, J. J. Koelling (LASL)*

Sheba, a clean geometry critical assembly employing fuel of 4.8% enriched uranium as the fluoride, was constructed during 1980. The primary applications of the machine are to evaluate accidental criticality alarm detectors for enrichment plants, to provide radiation spectra and intensity measurements to benchmark calculations on a low-enrichment solution system, and to provide radiation fields to calibrate personnel dosimetry. Although not in the immediate experimental plan, it is intended to work toward a solution burst machine. Even the name, Sheba, is the acronym for Solution High Energy Burst Assembly. To that end, experiments following the radiation measurements are aimed at evaluating prompt critical quench mechanisms in a bare, cylindrical, low-enriched solution. Design, construction, and applications of Sheba are discussed.

BACKGROUND

Goodyear Atomic Corporation had requested the assistance of the Oak Ridge National Laboratory in the evaluation of the spacing for nuclear criticality alarm detectors for enrichment plants. Due to uncertainties in the calculation and the high cost of safety through conservatism, Goodyear and ORNL requested that LASL run a solution critical to evaluate neutron and gamma leakage from a solution of relatively low-enrichment uranyl fluoride. An extension of the experiments to evaluate personnel dosimetry was obvious. It is interesting to note that all accidents since 1958

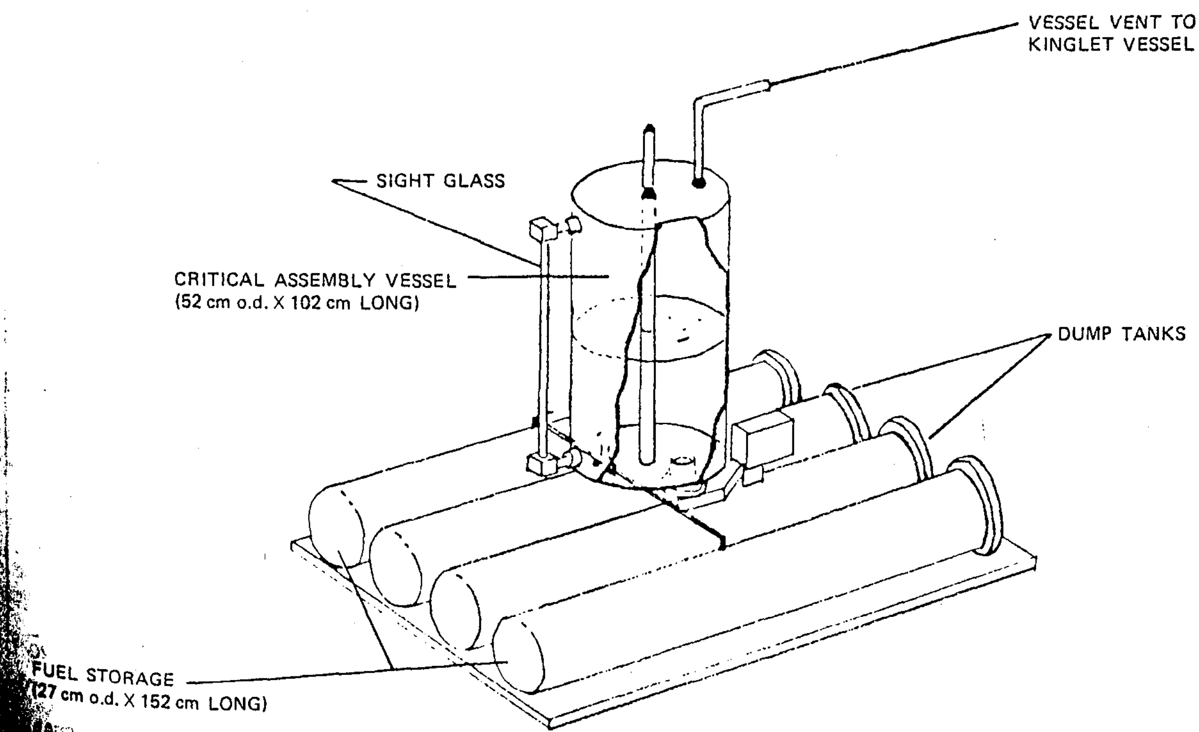

Fig. 1. Sheba: A solution critical assembly.

have involved solutions, although dosimetry is evaluated with metal assemblies such as the HPPR.

DESIGN

Sheba is of the simplest possible design in keeping with its application to benchmark calculations. The reactor vessel is a simple stainless-steel cylindrical tank with a 6.35-mm ($\frac{1}{4}$-in.)-thick wall. A single safety rod along the axis provides shutdown without sacrificing cylindrical symmetry. Control is affected by varying solution level through calibrated metering valves. Rapid shutdown is accomplished by solution dump through a three-inch pinch valve. The entire reactor assembly, including storage and dump tanks, is mounted on a pallet so the distance above the ground may be varied. To facilitate measurements out to 300 m, Sheba is housed in a thin metal shed outside of Kiva I at the Los Alamos Critical Assemblies Laboratory. Figure 1 illustrates Sheba's simplicity.

The concentrated uranyl fluoride [H/U $\approx$ 550, ρ(U) $\approx$ 1 kg/litre] is highly corrosive (pH $\approx$ 1). However, the nature of the facility is such that a leak could be tolerated. When operating at high power (~2 kW), radiolytic gas should form at the rate of 1 litre/min (indeed, a part of the experimental effort during late 1980 will be to evaluate the rate of formation of radiolytic gas). For purposes of the radiation experiments, the gas will be bled into a large holding tank to maintain a noncombustible atmosphere and to allow for the decay of fission product gases under confinement.

MEASUREMENTS

In addition to the evaluation of criticality alarm response to steady-state, ramp, and simulated burst operation, neutron and gamma spectra were evaluated and the responses of personnel dosimeters were determined. Measurements were made from near contact out to 300 m for several core elevations in free air, using shields of steel and concrete. Burst operation was simulated by establishing a point slightly below delayed critical and pulsing the system with a neutron generator.

EXTENSION

Two additional applications of Sheba are being pursued. We hope to continue to provide the facility for extensions of the measurements described above. In addition, we would like to pursue the development of a true solution burst capability with information generated from experiments with the present assembly. We propose to separately evaluate radiolytic gas formation and thermal expansion as quench mechanisms in solution burst machines. A high pressure (6 atm) head has been fabricated for Sheba which will allow the demonstration of depressurization of an operating solution assembly as a scram mechanism.

CONCLUSION

Sheba has demonstrated its utility to evaluate criticality alarm systems, personnel dosimetry, and benchmark calculations. In addition, the critical assembly machine is not only eminently useful for continued applications as designed but lends itself well to the analysis of prompt critical excursions in solution systems.

3. Application of Exponential Experiment to High Subcriticality Determination, *T. Suzaki, Y. Komuro, H. Tsuruta, I. Kobayashi (JAERI-Japan)*

From the viewpoint of criticality safety, there arises a need to develop better methods to evaluate high subcriticality of nuclear fuel systems with light water as neutron moderator. For this purpose, an exponential experiment technique was examined on water-moderated 2.6 wt% enriched UO_2 lattices at a critical assembly, TCA, of the Japan Atomic Energy Research Institute.

The lattices were assembled in a tank of 1.83-m diam with fuel rods which were made of 94.9% TD UO_2 pellets of 12.5-mm diam and a 0.76-mm-thick Al cladding of 14.2-mm o.d. The fuel rods were arrayed in square lattices n X n, where n = 17, 15, . . . , 3, with lattice pitch of 19.56 mm. The central fuel rod was replaced by an Al tube of 18-mm o.d., in which a BF_3 counter was traversed to measure the vertical neutron flux distribution and a ^{252}Cf neutron source of 5.5 mCi was positioned at an elevation of 16.8 cm lower than the bottom of fuel active zone. A polyethylene rod, 18 cm long, was set between the source and the fuel active zone to shield the direct neutron beam. The active height of the lattices was fixed at 122.5 cm, which was the critical water level of the n = 17 lattice.

The measured distributions of thermal-neutron flux are shown in Fig. 1 for the lattices of n = 15 to 3. The decay constant ν obtained from the exponential distribution is related to an effective multiplication factor k_{eff} under the one-group diffusion model with continuously slowing down sources as

$$k_{eff} = \exp[-M^2(\nu^2 + B_V^2)] \ , \qquad (1)$$

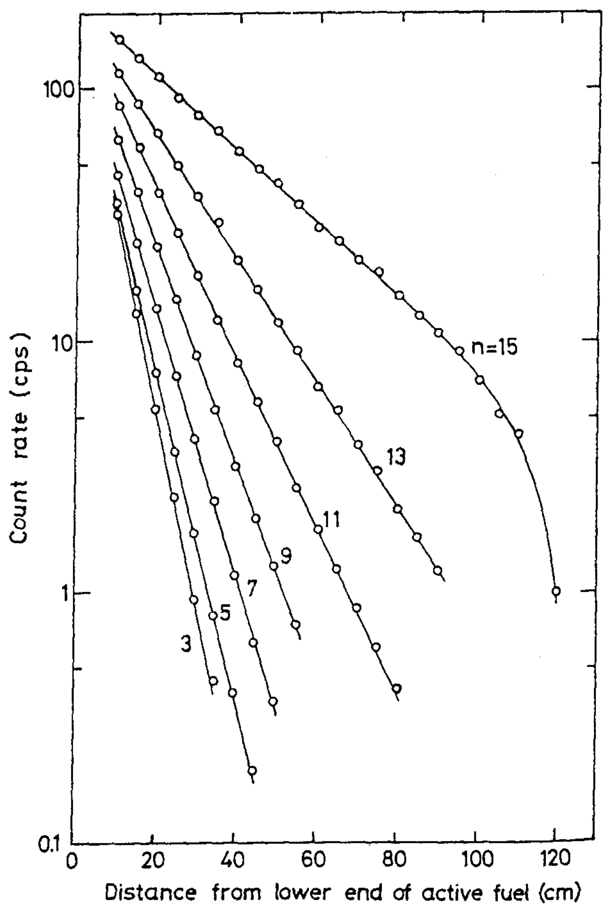

Fig. 1. Vertical flux distribution obtained by exponential experiment.

where M^2 is a migration area and B_V^2 is a geometrical buckling in the vertical direction. For these lattices, M^2 was obtained as $31.5 \pm 0.5\ cm^2$ by using the measured reactivity coefficient of water level at a nearly critical state and the calculated effective delayed-neutron fraction β_{eff} of 0.7478%.[1] The value of B_V^2 was determined from the water level and the vertical reflector saving, 12.2 ± 0.3 cm, which was obtained from a power distribution measurement. The resultant values of k_{eff} are shown in Fig. 2.

For comparison, k_{eff}'s were measured by a pulsed-neutron source technique on the same lattices as those of the exponential experiments. The prompt-neutron decay constant α is related to k_{eff} as

$$\alpha = [(1 - \beta_{eff})k_{eff} - 1]/\ell \text{ (Ref. 2)} . \qquad (2)$$

The prompt-neutron lifetime, ℓ, was obtained from the extrapolated value of α at $k_{eff} = 1$ and was found to be $(4.06 \pm 0.12) \times 10^{-5}$ s. Assuming ℓ as a constant, k_{eff} was evaluated by using Eq. (2). The results are shown in Fig. 2.

Criticality calculations were performed with the multigroup Monte Carlo code KENO IV.[3] Neutron cross sections of 137 energy groups were prepared by using the ENDF/B-IV library data. The calculated results of k_{eff} after the trace of 60,000 neutron histories are also shown in Fig. 2. Although the simple one-group diffusion model was used for the exponential experiment, the results agree well with those of KENO IV in the wide range of k_{eff} from 0.94 to 0.45. On the other hand, the results of the pulsed-neutron source experiments show a marked tendency to saturate in the subcritical state of $k_{eff} < 0.95$. Since the saturation is due mainly to the variation of ℓ, a proper estimation of ℓ is required for this technique.

Another application of the exponential technique has been examined on the lattices loaded with blade-type absorbers. The fuel rods were arrayed in a 15 X 20 lattice, which was separated by an absorber into two 15 X 10 regions. The subcriticality was varied with the absorbers which had seven B_4C concentrations. In this case, the k_{eff} from the exponential experiments showed good agreement with those from the pulsed-neutron source experiments in the range of $0.97 > k_{eff} > 0.90$.

In conclusion, the exponential experiment is an effective technique to determine k_{eff} of a highly subcritical system, when M^2 of the system is known by some method and there is a sufficient region of uniform composition in at least one direction.

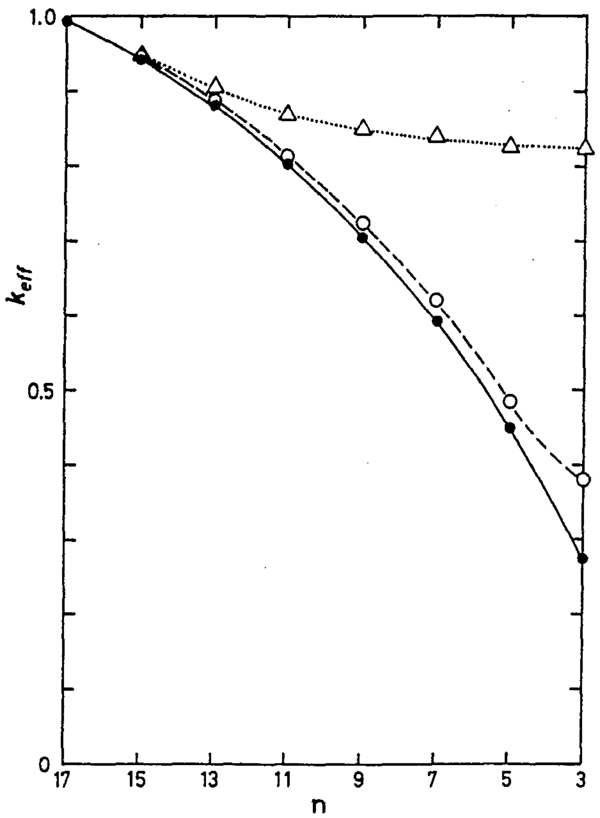

Fig. 2. Comparison among experimental and calculational effective multiplication factors k_{eff}. Closed circles, open circles, and triangles represent KENO IV, exponential, and pulsed-neutron source results, respectively. N is of n X n square lattices.

1. H. TSURUTA et al., "Critical Sizes of Light-Water Moderated UO_2 and PuO_2-UO_2 Lattices," JAERI 1254 (1978).

2. G. R. KEEPIN, *Physics of Nuclear Kinetics* (1965).

3. L. M. PETRIE and N. F. CROSS, "KENO IV—An Improved Monte Carlo Criticality Program," ORNL-4986 (1975).

4. SCALE System Cross-Section Validation for Criticality Safety Analysis, *A. M. Hathout (Al-Azhar Univ-Egypt), R. M. Westfall (UCC-ND), H. L. Dodds, Jr. (Univ of Tenn)*

The purpose of this study is to test selected data from three cross-section libraries for use in the criticality safety analysis of UO_2 fuel rod lattices. The libraries, which are distributed with the SCALE system,[1] are used to analyze potential criticality problems which could arise in the industrial fuel cycle for PWR and BWR reactors. Fuel lattice criticality problems could occur in pool storage, dry storage with accidental moderation, shearing and dissolution of irradiated elements, and in fuel transport and storage due to inadequate packing and shipping cask design. The data were tested by using the SCALE system to analyze 25 recently performed critical experiments.

The SCALE system consists of a driver module, control modules, functional modules, and a data base. One of the control modules is CSAS2, which stands for Criticality Safety Analytical Sequence No. 2. The CSAS2 control module reads a single unified set of input. After "messaging" the input and performing several auxiliary calculations (e.g., Dancoff factor determination), problem-dependent cross-section processing and a subsequent calculation of the system multiplication factor are performed. The execution path includes the functional modules BONAMI and NITAWL for problem-dependent cross-section processing, and the KENO-IV module for a Monte Carlo determination of the system multiplication factor.

The three cross-section libraries utilized in the analysis presented here are the 27-group CSRL library,[2] the 123-group GAM-THERMOS library,[3] and the 16-group Hansen-Roach library.[4] The 27-group library was collapsed from the 218-group CSRL library,[5] which was derived from ENDF/B-IV data.[6] Each of these libraries is available in the SCALE system data base.

Two series of experiments[7] involving 4.75 wt% enriched UO_2 rods were analyzed. The rods had a fuel diameter of