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PERFORMANCE TESTS OF THE OAK RIDGE

NATIONAL LABORATORY FAST

SAFETY SYSTEM

J. R. Tallackson

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OAK RIDGE NATIONAL LABORATORY

operated by

UNION CARBIDE CORPORATION

for the

U.S. ATOMIC ENERGY COMMISSION

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Instrumentation and Controls Division

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FAST SAFETY SYSTEM

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J. R. Tallackson
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ABSTRACT

Tests were conducted to verify the performance of the Oak Ridge National Laboratory fast safety system. Safe operation of the Bulk Shielding Reactor II (BSR-II), a small pool-type reactor with an extremely short neutron lifetime (21×10^{-6} sec), requires a fast acting, reliable safety system, that is, a multichannel system in which all fixed-time delays are minimized and which features rapid insertion of the shim-safety rods.

The shutdown characteristics of the reactor, with and without the safety system, were determined by a comprehensive series of tests of the BSR-II, at the Special Power Excursion Reactor Test Facility, National Reactor Test Station, Idaho Falls, Idaho. The reactor was subjected to power excursions of ever increasing violence. Test severity was limited to peak powers less than that which produced incipient damage to the core. Self-shutdown of a transient with a 0.014-sec stable period resulted in a peak power of 226 Mw and was accompanied by very slight core damage.

The safety system employed both period ($\tau < 1$ sec) and flux level ($\phi > 130$ kw) signals to scram the reactor. The reactor was operated without a source so that the stable periods of the transients were completely established in the submilliwatt power region which was below the range of the safety system. Therefore, the tests resulted in a very conservative evaluation of the performance of the safety system. Under these conditions, the period safety channels held BSR-II power to safe values for transient excursions with stable periods down to 0.003 sec. The flux level channels were equally effective in limiting excursions with periods down to 0.005 sec. The tests demonstrated that, in this reactor, a fast safety system is effective in counteracting substantial and rapid additions of prompt excess reactivity.

This report describes, by diagrams, the safety system and its salient characteristics. Transient test procedure is reported in detail, test results are plotted and compared with calculated estimates, and general conclusions and recommendations are included. Data-reduction methods are outlined.

¹Now with the Tennessee Valley Authority.

INTRODUCTION

With very few exceptions, nuclear reactors are and have been equipped with safety systems designed to protect the reactor system and personnel in the immediate vicinity from the hazards associated with accidental, uncontrolled nuclear power excursions. In a typical reactor, the safety system is carefully designed, the system components are thoroughly tested, and the system performance is estimated by analysis and analog simulation.

A reactor design is thoroughly verified by performance tests after construction and installation are completed. In sharp contrast, it is seldom allowable and usually neither desirable nor practicable to generate the severe power transients required to test the safety system in an installed operating reactor. As a result, definitive safety system performance data have been virtually nonexistent.

The Bulk Shielding Reactor II (BSR-II), a newly designed swimming pool reactor with an extremely short neutron generation time and consequent rapid response to small amounts of excess prompt reactivity, required a fast safety system with experimentally verified performance. The ORNL Reactor Controls Department has established a continuing program to improve and to understand reactor safety; reactor safety system performance tests were included in this program. Self-shutdown experiments with the BSR-II were compatible with the experimental program at the SPERT-I facility.² Finally, the BSR-II core and safety system, small in size and simply constructed, were capable of easy and economical transport and installation at SPERT-I, where testing could be done without hazard.

This fortuitous combination of circumstances resulted in a comprehensive experimental program at the SPERT-I facility which was well equipped and staffed for the purpose.

These experiments, conducted during the winter of 1959-60, were a joint effort with personnel from both installations, ORNL and SPERT-I, participating. This report is concerned particularly with the tests which specifically demonstrate the ORNL fast safety system's ability to protect

²Special Power Excursion Reactor Test facility operated by the Phillips Petroleum Co. at the National Reactor Test Station, Idaho Falls, Idaho.

the BSR-II from the consequences of rapid additions of excess reactivity and, generally, in focusing attention on the merits of fast acting systems.

1. TEST OBJECTIVES

The fast safety system developed at ORNL was installed in the Bulk Shielding Reactor II (BSR-II) and tested at the SPERT-I facility at the National Reactor Testing Station, Idaho. Test objectives were as follows:

1. to provide experimental verification of analytically predicted reactor and safety system behavior for presentation to the Advisory Committee on Reactor Safeguards,
2. to obtain transient and nuclear data on the BSR-II, and
3. to test the performance of the ORNL safety system³ during short-period transients.

This report presents the performance of the ORNL safety system as installed in the BSR-II.

2. TEST PROGRAM

The tests conducted at SPERT-I were as follows:

- (1) Static measurements.⁴
 - (a) Critical loading and rod calibration.
 - (b) Neutron flux distribution.
 - (c) Neutron peaking in a water-filled hole.
 - (d) Temperature defect measurements.
 - (e) Void coefficient measurements.
- (2) Self-shutdown tests.
- (3) Level shutdown tests.
- (4) Period shutdown tests.
- (5) Reverse tests.

This report is concerned with items 2, 3, and 4 in the above tabulation.

³T. E. Cole, E. E. St. John, and S. H. Hanauer, The MTR Safety System and Its Components, ORNL-1139 (Apr. 4, 1952).

⁴F. Schroeder (ed.), Quarterly Technical Report SPERT Project, October, November, December, 1959, IDO-16616 (Aug. 24, 1960).

3. REACTOR AND SAFETY SYSTEM

3.1 The Bulk Shielding Reactor II

The BSR-II is a small (15-in.³) stainless-steel-clad, plate-type, swimming-pool reactor.⁵ It contains a maximum of 25 fuel elements. The fuel region in each element occupies a space 3 in.² and 15 in. high and contains 290 g of 93% U²³⁵ in a UO₂ stainless steel cermet distributed in 20 fuel bearing plates. The critical mass is approximately 5.8 kg. Its size and composition result in a short neutron lifetime (21 μ sec).

A detailed list of pertinent reactor data is in Appendix A.

3.2 Safety System

Figure 1 is a highly simplified block diagram of the safety system tested,³ and Fig. 2 is a diagram of the reactor power ranges spanned by the safety and control instruments. The block diagram omits the positive current supply⁶ shown in Fig. 3. The positive current supply, suggested by S. H. Hanauer of the ORNL Reactor Controls Department, was used to ensure, when a transient was initiated, that the input to the log N amplifier was always a positive current of 5×10^{-11} amp and was identical for all transients. This value of diode current ensures that the log N amplifier is not operating in the "cutoff" condition and that no period scrams could be attributed to the false period signal produced as the amplifier comes out of "cutoff." This positive current bias produces a negligible degradation of period channel response referred to the response obtainable from a perfectly compensated chamber. The effect on safety system performance data was small.

Four shim-safety rods were located symmetrically about a vertical center line through the core (see Fig. 4). In most tests, three of these rods were used as safety rods, and the remaining rod was used as a shim to adjust the excess reactivity required to produce the transient and was not scrambled.

⁵E. G. Silver and J. Lewin, Safeguard Report for a Stainless Steel Research Reactor for the BSF (BSR-II), ORNL-2470 (July 16, 1958).

⁶S. H. Hanauer, A New Circuit for Preventing Certain Unnecessary Period Scrams, to be published.

Fig. 1. The BSR-II - SPERT-I Safety System.

Fig. 2. Reactor Power Ranges.

Fig. 3. Positive Current Supply.

Fig. 4. Plan View of BSR-II Core Used for Safety System Tests at SPERT-I.

Additional information describing the design and operation of the system is given in Appendix B.

The modular drive^{5,7,8} employs a spring to accelerate the shim-safety rod and a magnet designed to minimize release time.^{9,10} Magnet release times were 5 msec or less; this is a very substantial improvement when compared with rod magnets having release times from 20 to 50 msec. Figure 5 gives rod worth vs position for the three shim-safety rods as a group. Curve A, Fig. 6, displays the total reactivity vs time for these three

⁷Engineering Department Drawing E-30875 and Research Shops Drawing D-150-1.

⁸R. T. Santoro, T. F. Sliski, J. R. Tallackson, private communication, Mar. 6, 1961.

⁹J. R. Tallackson and R. T. Santoro, private communication, July 1, 1960.

¹⁰E. P. Epler et al., Reactor Safety System Response, ORNL-2318 (1958).

Fig. 5. Rod Worth vs Position of Shim-Safety Rods Nos. 2, 3, and 4 Grouped Together.

Fig. 6. Rod Worth Curves for BSR-II Shim-Safety Rods Nos. 2, 3, and 4 Acting as a Group and Having an Initial Withdrawal of 13.5 in. Curve A: rods accelerated by a spring producing 66 lb of force at 4.5 in. deflection. BSR-II - SPERT-I test conditions. Curve B: calculated for rods free-falling in water with an acceleration of 20 ft/sec².

spring-accelerated safety rods used in these experiments, and curve B affords a comparison with the same rods in the same core but without spring acceleration. Curve A was obtained by combining the rod-worth calibration obtained at SPERT-I with rod-drop test data⁸ gathered as the system was developed. Tests by Sorensen¹⁰ on the shim-safety rods used in the Bulk Shielding Reactor I were used to determine curve B. Sorensen's tests showed that these rods experienced an acceleration of 20 ft/sec² during free fall in water. This value for acceleration, combined with the same rod worth vs position data used to compute curve A, was used to calculate curve B.

It is entirely reasonable to assume that the free-fall dynamics of the BSR-II rods are not dissimilar from those tested by Sorensen (see ORNL-2470, Figs. 24, 27, and 28).

Inspection of Fig. 6 shows that the addition of springs produced a tenfold increase in the amount of negative reactivity inserted during the first 50 msec following release.

It is difficult to attach too much importance to the combination of short release time and accelerated rod insertion designed into this rod drive. These are the factors which share the credit for the performance data on the following pages.

4. CONTROL INSTRUMENTATION AND REACTOR CONTROL AND DATA EQUIPMENT

A low-level fission counter channel was provided for the critical runs preceding each transient and to provide assurance that the count rate immediately before initiating the transient was sufficiently low. The reason for the low pretransient count rate requirement is discussed in Sec 5.

The reactor control circuitry was very simple; a majority of the prohibitive and restrictive interlocks and control system niceties which are normally supplied to assist the reactor operator were omitted. The control requirements of the test program were well met by experienced operators exercising manual control.

All instrumentation used to measure reactor and safety system behavior during the transients was supplied by the SPERT-I facility.¹¹ The existing SPERT-I facility instrumentation had been specifically designed and developed for transient measurement and had established a record of successful application.

5. SPECIAL EQUIPMENT FOR PRODUCING TRANSIENTS

Transients were produced by the fast, spring-accelerated insertion of a two-bladed, "double-ended," rod, that is, a double-length rod containing both poison and fuel (see Fig. 7). The transient rod drive was identical to that used for the shim-safety rods and was centrally located in the core (see Fig. 4). The central fuel element and two adjacent fuel

¹¹F. L. Bentzen, SPERT-I Instrumentation, IDO-16316 (Mar. 5, 1957).

Fig. 7. Details of Burst Rod Used in Transient Tests of BSR-II at SPERT-I.

elements were reduced in size by removing fuel plates to accommodate the rod. It was also necessary to cut clearance holes in the grid plate so that the poisoned end of the transient rod could leave the core as the fueled end was being inserted. Curves of transient rod worth plotted against height and elapsed time after release are given in Figs. 8 and 9. These curves are derived from rod-worth calibration and rod-drop test data obtained at ORNL.

A conveniently available commercial power supply was used as the current source for the transient rod magnet. The transient was initiated by the SPERT-I programmed time switch which, by means of relays, interrupted the transient rod magnet current.

The transient data (see Appendix D) was recorded on Consolidated Electrodynamics Corporation oscillographs equipped with mirror galvanometers and useful to frequencies up to 5000 cps. The reader is referred to ref 11 for detailed information describing this instrumentation.

6. TEST PROCEDURE

The first tests with the safety system were conducted with a neutron source present; no other modus operandi had been contemplated since a source is a mandatory requirement for safe reactor operation. However,

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Fig. 8. Transient Rod Reactivity vs Rod Position.

with a source present the transient power, which was rising as the transient rod was moving into the core, initiated a scram in the period safety instruments before the transient rod was fully inserted. As a result the shim-safety rods were accelerated into the core before the transient rod had completed its stroke and it was not possible to attain the short, stable period characteristic of a step insertion of reactivity. The reactor powers attained during these excursions were too low to provide accurate measurements without making substantial revisions to the instrumentation. These revisions were not permissible in the time available.

These very low values of peak shutdown powers were a conclusive demonstration that the safety system was intercepting the excursion as it was developing and, qualitatively, are ample proof of its ability to protect the reactor. However, quantitative data, adaptable to analysis and assessment by comparison to available analytical and experimental shutdown data, were required. The test procedure was changed so that the transient was established at a flux level well below the range of the period safety

Fig. 9. Reactivity Added by Transient Rod vs Elapsed Time After Release.

instrumentation. This was accomplished by removing the source and proceeding with extreme care during critical runs. The step-by-step procedure (see Fig. 10) was as follows:

1. All rods, including the transient rod, were fully inserted.
2. Safety rods (Nos. 2, 3, and 4) were withdrawn 13.50 in. above full insertion.
3. Shim rod No. 1 was withdrawn to make the reactor critical. The power was of the order of 5 mw, and, for most tests, the withdrawal of shim No. 1 was 3.25 in.

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Fig. 10. Sequence of Rod Position Used to Produce a Transient Test.

4. The double-ended transient rod was withdrawn thus inserting poison simultaneously with the removal of fuel. The reactor became subcritical.

5. The reactor neutron flux was permitted to decay until the maximum counting rate was 5 counts/min but preferably less. This usually required from 5 to 25 min.

6. Shim rod No. 1 was withdrawn a predetermined amount (see Fig. 11). This curve was plotted during the course of the experiments and, by extrapolation, served as a guide for estimating the amount of withdrawal required to obtain transient periods of ever-increasing severity.

7. The spring-loaded transient rod was released (see Sec 5), thereby initiating the transient. This operation was programmed on the sequence timer which also started the oscillograph recorders, set off alarms, etc.

Fig. 11. Reactor Period vs Incremental Withdrawal of Shim Rod No. 1.

It should be noted that the transient rod was accelerated and reached maximum speed while the reactor was subcritical, thus minimizing the time required to insert the necessary positive excess reactivity. For example, by referring to Figs. A-1 and 9, it can be seen that an excess reactivity of \$1.65 was required to obtain a 0.005-sec stable period and that this excess was inserted in approximately 0.080 sec, commencing 0.075 sec after transient rod release.

In order to maintain the reactor in a source-free condition for the duration of the test program, integrated reactor power was held to a minimum. This operating procedure ensured that decay-gamma-induced neutrons

would not constitute a distributed source. Hence, for all tests, the reactor was cold and clean. With the exception of a few calibration runs, the reactor was never operated for any length of time above the milliwatt region. The burst width during a transient was a few hundred milliseconds at most and did not represent a substantial contribution to the integrated power.

For these tests it was decided to establish 100% reactor power in the 100-kw region. This is a realistic number for pool reactors of this type. Ideally, then, the chambers would have been located so that the log N channels read six decades up-scale and the level safety channels indicated 100% power (see Appendix B) when the reactor was developing 100 kw of power as read in the SPERT-I calibrated chambers. The obvious and simple procedure was manual positioning of the chambers with the reactor running at rated power. This method was discarded since it would have violated the important requirement that integrated reactor power be held to a minimum. Furthermore, regulations at SPERT-I prohibited personnel in the reactor building while the reactor was critical. The method adopted was to estimate the required location as closely as possible, to conduct enough short full-power runs with the reactor to get equivalent readings in the log N and safety channels, and to establish the value of scram power.

7. TEST RESULTS

As applied to the BSR-II, the performance of the safety system was slightly better than the analog-computed estimates reported in ORNL-2470.¹² It was estimated (see Fig. 12 in ORNL-2470) that the ratio of peak to scram powers in the BSR-II would be 1000 if the reactor was scrammed while on a 0.0062-sec period and if the worth of the safety rods was the same, \$6.30, as that used in the tests at SPERT-I. The observed power increase ratio was 150. Figure 12 compares the analog-computed performance estimate¹² with the test results. The differences in test parameters which are tabulated on Fig. 12 are small, and the agreement obtained is excellent.

¹²E. G. Silver and J. Lewin, Safeguard Report for a Stainless Steel Research Reactor for the BSF (BSR-II), ORNL-2470 (July 16, 1958).

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THE CURVE PLOTS THE PERIOD, τ , OF THE TRANSIENT EXCURSION WHICH CAN BE REVERSED WITHIN THREE DECADES OF POWER RISE BY THE FAST SAFETY SYSTEM. THIS CURVE, OBTAINED WITH THE ANALOG COMPUTER, HAS BEEN EXCERPTED FROM FIG. 28 IN ORNL-2470 (REF 12). THE POINT, $\downarrow$, IS DERIVED BY EXTRAPOLATION OF DATA ON FIGURES 14 AND 15.

THE CONDITIONS GOVERNING THE ANALOG COMPUTATION AND THE TEST ARE COMPARED AS FOLLOWS:

	ANALOG CURVE	TEST
1. INITIAL SPRING FORCE	66 lb	66 lb
2. PROMPT NEUTRON GENERATION TIME	20 μ sec	24 μ sec
3. INITIAL SHIM-SAFETY ROD INSERTION	1.0 in.	1.5 in.
4. TOTAL ROD WORTH	VARIABLE	6.3 %

Fig. 12. Safety System Shutdown of the BSR-II; a Comparison of Analog Computations with Test Results.

Figure 13 compares peak powers reached during reactor self-shutdown with those obtained by safety system shutdown using three shim-safety rods. Fuel element damage in the form of slight fuel plate buckling appeared after the 0.014-sec period self-shutdown run which produced a peak power of 226 Mw. Reactor damage was prevented by limiting peak power in all subsequent tests to less than this value. The upper limits of test severity were:

Type of Test	Minimum Period (sec)	Excess Reactivity	Peak Power (Mw)
Self-shutdown	0.014	0.009 δ k	226
Safety system shutdown level scram	0.0046	0.012 δ k	92
Safety system shutdown period scram	0.0030	0.014 δ k	79

The self-shutdown tests are described by Bentzen in IDO-16722.¹³

As expected, with bursts starting at zero power, the period signal scrambled the reactor before the flux level signal became effective. If both period and level channels experienced equal delays, the separation

¹³F. L. Bentzen, The Merits of Inherent Shutdown vs Mechanical Shutdown of a Plate Type, Water-Moderated and Reflected Reactor in a Runaway Condition, IDO-16722 (Nov. 3, 1961).

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Fig. 13. Safety System Performance in the BSR-II.

between the period and level shutdown power curves of Fig. 13 would be six decades (see Fig. 2). However, it is well recognized that the time delays in the period instrumentation¹⁴⁻¹⁶ are a variable function of period and signal strength. The observed separation, slightly more than three decades for a 0.005-sec period transient, is in good agreement with the value calculated by Ditto.¹⁴

Figure 14 displays normalized level scram test data and includes a calculated safety system response curve. This curve was calculated by the method of Appendix C using curve A in Fig. 6 to describe safety rod worth as a function of time. This curve may be represented by the equation

$$\delta k(t) = 0.0208 (1 - \cos 17t), \quad t \leq 0.050 \text{ sec.}$$

The computed shutdown curve A assumed a fixed delay, t_1 , of 0.005 sec. This figure was based on release time tests conducted in the laboratory and in situ at SPERT-I. The close agreement between calculated and observed power ratios is evident.

Curve B, Fig. 14, gives the computed ratio of reactor power at the time the clutch switch was actuated to the power at scram point. The computations assumed a magnet release time of 0.005 sec and a rod travel of 0.010 in. to actuate the switch. The spring-accelerated rod requires 0.003 sec to traverse this 0.010 in.; therefore, the total delay between scram point and switch actuation is 0.008 sec. Curve C is a similar curve for a total delay time of 0.005 sec (magnet release time only), and it can be seen that the clutch switch points obtained experimentally fall between these two curves. The clutch microswitches (see Fig. B-3, Appendix B) were adjusted in the field to obtain actuation by a very small incremental

¹⁴S. J. Ditto, Trans. Am. Nucl. Soc. 3(2) 456-57 (1960).

¹⁵J. D. Cummins, Diode Response and Time Derivative of Response to an Exponentially Increasing Input Current, AEEW-M10, Atomic Energy Establishment, Winfrith, Dorchester.

¹⁶B. B. Barrow, The Logarithmic-Diode Counting Rate Meter and Period Meter, AIEE No. 57-403, 2d Nuclear Engineering and Science Conference.

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Fig. 14. Normalized Level Scram Data.

motion following release. These switches have sensitivity and reproducibility within 0.005 in., and it is evident that the 0.005-sec release time was being attained in these tests.

As described in Sec 6 (see Fig. 10), the great majority of the flux level shutdown tests were run with three shim-safety rods for shutdown. A few level shutdown tests were run with only two safety rods to scram

the reactor and two tests with all four rods. These tests are compared in Fig. 15.

It is pertinent to note that in the four-rod test, the additional rod was, of necessity, the shim rod used to adjust the reactivity producing the transient. This rod, in the shorter period bursts, was about one-half withdrawn (see Fig. 11). Its incremental worth was maximum. Data so obtained are slightly biased in the direction of enhanced safety system evaluation.

Figure 16 gives curves of energy developed by the core to the peak of the excursion, during safety system shutdown (see Appendix D for the method of calculation). The complete power excursion curves (see Fig. D-1, Appendix D) were, by inspection, symmetrical. No serious error would result in

Fig. 15. Peak Power vs Reactor Period for Two, Three, and Four Rod Level Scrams.

Fig. 16. Energy to Peak of Burst vs Reactor Periods.

taking the total energy developed during such an excursion as twice the values on Fig. 16.

As pointed out in Sec 6, the reactor was cold and clean and operated without a source; the transient was not initiated until the neutron level, as measured by a fission counter, was indistinguishable from counting channel noise. The time intervals (see Fig. 17) between transient rod insertion and the peak of the burst were greatly in excess of the minimum intervals theoretically required for the reactor to reach peak power. For example; see Fig. 13, a safety system shutdown of a 0.010-sec period excursion resulted in a peak power of 10^6 w. Figure 17 shows a point which indicates that 1.0 sec was required to reach this peak after the transient rod was inserted. A simple calculation which assumes a continuous exponential increase in reactor power produces a starting power of 10^{-37} w (equivalent to a fission rate of 3×10^{-27} fission/sec).

This is obviously ridiculous. It is, however, apparent that the neutron population in the core at the beginning of these tests was extremely small. It is clear that, in this situation, a persistent chain reaction is not established simultaneously with the establishment of all of the conditions required to obtain criticality. The concept of criticality,¹⁷ in these circumstances, involves the statistical behavior of a relatively small number of particles, and the reactor's response to reactivity additions is not described by the usual kinetic equations. It is shown in Sec 8 that a reactor in this condition is potentially hazardous.

8. CONCLUSIONS AND RECOMMENDATIONS

This fast system will protect the BSR-II from damage caused by accidental insertions of excess reactivity up to \$2.00 regardless of the reactivity addition rate. Figure 13 provides a comparison of maximum powers vs inverse transient period of the three shutdown modes tested. It was pointed out in Sec 7 that incipient core damage was observed after a 0.014-sec period self-shutdown transient whereas safety system shutdowns of excursions, with periods as short as 0.003 sec left the core unharmed. Self-

¹⁷G. E. Hansen, Nucl. Sci. Eng. 8, 709-19 (1960).

Fig. 17. Delay Time from Transient Rod Seating to the Peak of the Burst vs Reactor Period.

shutdown of the BSR-II on a 0.003-sec period would appreciably damage, if not destroy, the core, and the possibility of hazard cannot be discarded.

The first series of tests provide a convincing demonstration of the anticipatory nature of the period safety channels. These first tests (see Sec 6) were conducted under normal reactor conditions by the use of a neutron source. Although nonproductive of numerical data, these tests

were entirely realistic and their importance cannot be overemphasized. The period safety instruments released the safety rods before the transient rod had completed its stroke. The net result was always a "rod race," and the transients were terminated at power levels so low as to be of no consequence. It was shown that the period channels initiated safety action as the accident was developing and not after the fact; it can be concluded that the period scrams are the most effective antidote for short-period (less than 1 sec) excursions regardless of power level. The data presented in Fig. 13 were collected by operating the reactor without a source. The total excess reactivity responsible for the transient was inserted before any safety system sensor was within its range of detection. This is not normally accepted reactor operating practice. It must be concluded that, using Fig. 13, an evaluation of the safety system's ability to protect the BSR-II will be extremely conservative and in the safe direction.

A corollary may be developed from the preceding paragraph. Reactor and safety system designers are in the habit of discussing and specifying safety system requirements in terms of the system's response to excursions characterized by well-established stable periods. This, in all likelihood, is an outgrowth of the analytically tractable assumption that negligible error results in attributing credible short period excursions to step-wise additions of excess reactivity. The step insertion concept, when applied to a safety system which is capable of cancelling accidental excess reactivity as it is being added at the maximum credible rate, yields a most conservative estimate of the protection offered. A more accurate assessment results if the analytical model recognizes that both the accident and the safety system require finite times in which to become fully effective.

The long time delays (see Fig. 17) observed between transient rod insertion and peak transient power conclusively demonstrate the precept that source free operation is potentially disastrous. A long interval between a supercritical condition and the beginning of supercritical multiplication is an open invitation for the introduction of virtually unlimited excess reactivity at relatively low rates of addition. The effect is identical to a step-wise insertion of δk ; safety systems, however fast, cannot be designed for operational procedures which place no limits on the amount

and rate of reactivity addition. A properly located source must therefore be considered as an integral component of the safety system.

It is worthwhile to compare this fast safety system with somewhat similar systems which place no stress on speed of response nor on accelerating the insertion of shutdown reactivity. The reactivity insertion rates of the two systems are those shown on curves A and B in Fig. 6. These negative reactivity insertion curves are well described by the following equations:

$$\text{curve A, } \delta k(t) = -0.208 (1 - \cos 17t);$$

$$\text{curve B, } \delta k(t) = -0.712 t^{2.33};$$

$$0 < -\delta k(t) \leq \beta = 0.007.$$

Peak reactor powers were calculated as a function of the inverse stable period of the excursion by the use of these equations and varying the fixed delays preceding rod release. The method of calculation is described in Appendix C. The adverse effect of increasing the fixed delays preceding spring-accelerated rod release is displayed in Fig. 18. The merits of adding springs to accelerate the rods in an otherwise fast system are manifest from Fig. 19. The summary below (Table 1) is based on these and similar calculations which assumed the BSR-II was on 0.005- and 0.010-sec stable periods as it reached a normalized scram power of unity.

The response of a safety system may be defined in terms of sensor delays, safety rod release times, rod acceleration, etc. However, the overall response of the reactor with its associated safety system is the most important consideration. Therefore, the criteria of a fast safety system must be related to the reactor in which it is used; that is, the response of a safety system should be measured relative to the reactor's response to reactivity additions and to the credible rates of reactivity addition. Regardless of the numerical values of the response times involved, reactors equipped with fast safety systems have the following characteristics in common:

1. All credible accidents involving substantial excess reactivity above prompt critical appear as positive reactivity ramps to the safety system.

Fig. 18. Calculated Performance of the BSR-II Safety System with Spring-Accelerated Shim-Safety Rods and with Fixed Delay as a Parameter.

2. The safety system detects the addition of excess reactivity as it enters the core.

3. The safety system initiates reactivity removal with negligible delay.

4. The rate of reactivity removal is of the same order of magnitude as the maximum credible insertion rate.

Regardless of the speed or type of safety system employed, excess reactivity accidents are of two types: those which fall within the competence of the safety system and those which must be met with other measures, usually containment. This second category concedes partial or complete loss of the system. Fast safety systems have achieved limited use in reactor design, yet a fast, reliable safety system has the capacity to reduce the number of credible accidents falling into this second category. If so, the reactor and system designers are able to concentrate their efforts on eliminating or downgrading a reduced number of potential credible

Fig. 19. Calculated Comparison of Safety System Shutdown of the BSR-II Using Free-Falling and Spring-Accelerated Rods.

Table 1. Peak Excursion Powers in the BSR-II as Influenced by Safety System Response

Type of System ^a	Sum of Fixed Delays ^b (sec)	Stable Period of Excursion (sec)	Ratio of Maximum Power to Scram Power
A	0	0.010	6.0×10^0
A	0	0.005	1.6×10^2
A	0.005	0.010	9.5×10^0
A	0.005	0.005	4.4×10^2
A	0.030	0.010	1.2×10^2
A	0.030	0.005	2.4×10^3
B	0	0.010	3.2×10^2
B	0	0.005	5.3×10^6
B	0.005	0.010	5.2×10^2
B	0.005	0.005	1.5×10^7
B	0.030	0.010	6.3×10^3
B	0.030	0.005	2.2×10^9

^aSystem "A" employs spring-accelerated rods; system "B" employs free-falling rods in water; see Fig. 6 for rod-worth curves.

^bElapsed time between reactor scram point (power or period) and rod release.

accidents which require containment. Since it may cost but a little extra to build improved response into a safety system, the economic incentive can become substantial.

The penalty paid for fast response is a higher theoretical probability of false scrams produced by spurious noise signals. Operating experience at ORNL has shown that this source of false scrams is of little consequence in a properly maintained and operated system.

System time delays are a consequence of sensor time constants; of electronic, electrical, and electromechanical delays in the signal modifiers; of delays in the rod release mechanisms (magnets, latches, etc.); and of frictional and inertial effects which impede rod motion. Recommendations and comments are:

Sensors. - Neutron sensitive ion chambers are inherently very fast and present no problem from this standpoint. This assumes that the ion chambers in the period channels are properly compensated. Temperature, pressure, and flow signals would be equally useful as fast safety system inputs if suitable transducers existed. The development of reliable, high-speed transducers capable of enduring reactor environments and capable of in situ tests for integrity during reactor operation is recommended.

Signal Modifiers. - There are inherent delays in the log N period channel at low signal levels¹⁸⁻²⁰ but, as demonstrated by these tests, these delays do not prevent the period circuitry from providing the best response to fast transients at any flux level.

Rod Release Mechanisms. - The techniques used to design high-performance magnets are^{21,22} well understood. Reduction to practice has suffered

¹⁸S. J. Ditto, Trans. Am. Nucl. Soc. 3(2) 456-57 (1960).

¹⁹J. D. Cummins, Diode Response and Time Derivative of Response to an Exponentially Increasing Input Current, AEEW-M10, Atomic Energy Establishment, Winfrith, Dorchester.

²⁰B. B. Barrow, The Logarithmic-Diode Counting Rate Meter and Period Meter, AIEE No. 57-403, 2d Nuclear Engineering and Science Conference.

²¹E. P. Epler et al., Reactor Safety System Response, ORNL-2318 (1958).

²²E. P. Epler and C. Michelson, private communication, Sept. 6, 1957.

as a consequence of insufficient development engineering aimed toward producing a consistent, long-lived, and economical product. Electromechanical latches are attractive in that they permit greater forces to accelerate the rods and also permit the use of smaller magnets.

Ways and means to produce accelerated insertion of negative reactivity should be studied. The obvious method, spring acceleration of a safety rod, as in the BSR-II, is definitely limited. These limitations are imposed by definite upper limits (magnetic saturation) on magnet force and by power available to cock the spring. A program to study and to develop other methods of rapidly inserting negative reactivity should be initiated. Consideration should be given to the effective use of compressed gas as an accelerating means; to the development of simple, reliable safety rod latches which will enable small, low-power magnets to release large forces; and to the storage of kinetic energy, as by the continuous rotation of a flywheel, to provide the energy to accelerate a safety rod.

A reactor operator, possessing certain knowledge that his reactor would soon experience a dangerous addition of reactivity, as by a cold slug, would push the scram button. He would not wait for the reactor flux to rise. Fast safety systems have emphasized the use of flux information to the virtual exclusion of other danger signals. It is suggested that safety system design place additional emphasis on ways and means to anticipate accidental additions of δk in addition to flux sensing. Flux changes are after the fact. Admittedly flux sensing is, in a majority of situations, the best or only source of reactivity information; however, the effectiveness of a safety system is a very strong function of its ability to anticipate, and design should be guided accordingly. For example, a temperature sensor with response time equal to that of an ion chamber would provide an effective way to deal with the cold slug accident.

Reactor Testing. - The tests herein discussed were not entirely realistic. A better idea of the ultimate capability of a safety system would develop from tests with the following conditions:

1. The rate of excess reactivity addition to be a variable parameter.

2. The upper limit of test severity should be extended until the test program is naturally terminated by severe damage to or by destruction of the core.

3. Initial position of the shim safety rods to be a variable parameter.

It would be highly desirable, during severe tests, to measure the forces acting on the safety rods.

It is inevitable that the shutdown data in Fig. 13 will suffer extrapolation. In the absence of data, any extrapolation is open to question. Tests conducted under the conditions outlined above would provide information as to whether reactor shutdown at very short periods would result from the combined effects of safety system and self-shutdown mechanisms. There is reason to believe that a fast safety system which is acting immediately to downgrade the effects of the initial insertion of large positive δk would provide time for self-shutdown to become effective and augment the safety system.

Test results show that simple hand calculations based on rod-worth dynamics and reactor characteristics produce good predictions of safety system performance when the excess reactivity is above prompt critical. Predictions so obtained yield conservative evaluations of safety system performance since they assume a step insertion of excess reactivity.

It cannot be concluded that a fast, high-performance safety system is the universal panacea for hazards brought about by excess reactivity accidents. Safety system design augments rather than supplants careful reactor and reactor system design and operation. The role played by reactor safety systems will be discussed more generally and thoroughly in a report now being prepared by the Reactor Controls Department at ORNL.

ACKNOWLEDGMENTS

The work herein reported represents the participation and individual effort of a large number of persons. It is a pleasure and a privilege to acknowledge the contributions of R. K. Adams of the Instrumentation and Controls Division who completed the computer codes which were invaluable in the reduction of data, M. M. Bowelle of the Instrumentation and Controls Division who assumed the responsibility for control circuit layout and detailed design, T. F. Sliski of the Instrumentation and Controls Division who assisted with a wide variety of mechanical problems, C. B. Stokes of the Instrumentation and Controls Division who fabricated the control system and displayed patience and initiative with the more onerous aspects of data reduction, K. M. Henry and E. G. Silver of the Neutron Physics Division who helped with pretransient rod drive tests and provided substantial assistance during the test program, and C. W. Angel of the Engineering and Mechanical Division and members of his staff who designed the special fixtures required by the tests.

It is particularly appropriate to emphasize that these experiments were a joint effort by ORNL and the SPERT facility operated by the Phillips Petroleum Corporation. The cooperation extended by W. E. Nyer, S. J. Forbes, F. L. Bentzen, T. R. Wilson, R. Heffner, J. C. Haire, A. Nelson, and members of their staff at SPERT was most generous. No request for assistance was considered too large or too small, and all requests were handled expeditiously.

Appendix A

NUCLEAR DATA FOR THE BSR-II

Nuclear and related data for the Bulk Shielding Reactor II based on tests at Oak Ridge National Laboratory²³ and SPERT-I²⁴ are as follows:

1. Critical mass 5.778 kg
(This mass of 93% U²³⁵ produced \$0.16 excess reactivity during initial critical tests.)
2. $\ell^*/\beta_{\text{eff}}$ $(3.0 \pm 0.2) \times 10^{-3}$ sec
3. ℓ^* 21×10^{-6} sec
4. β_{eff} $0.007 \pm 10\%$
5. Shim-safety rod worth 3 rods \$6.29
6. Temperature coefficient of reactivity (Average from 58°F to 184°F) 5×10^{-5} k/°F
7. Void coefficients²⁴
(Void coefficients were measured by simulating voids evenly distributed throughout the core with both magnesium strips and with Styrofoam. Values for both sets of measurements are as follows.)
 - a. Magnesium 1.10×10^{-3} per cm³ (\$0.46 per % moderator void)
 - b. Styrofoam (equivalent to air) 1.40×10^{-3} per cm³ (\$0.60 per % moderator void)
8. Core dimensions (active fuel bearing region) 15 × 15 × 15 in.
9. Fuel plate dimensions
 - a. Stainless steel cladding thickness 0.008 in.
 - b. Fuel region thickness 0.014 in.

Figure A-1 is the curve of excess reactivity vs asymptotic period. A detailed description of the BSR-II is contained in ref 23.

²³E. G. Silver and J. Lewin, Safeguard Report for a Stainless Steel Research Reactor for the BSF (BSR-II), ORNL-2470 (July 16, 1958).

²⁴F. Schroeder (ed.), Quarterly Technical Report SPERT Project, October, November, December, 1959, IDO-16616 (Aug. 24, 1960).

Fig. A-1. Asymptotic Period vs Excess Reactivity.

Appendix B

ORNL SAFETY SYSTEM USED WITH THE BSR-II TESTS AT SPERT-I

General Description

This type of safety system, Fig. 1, was originally designed for the MTR and has been described by Mruk²⁵ and by Cole²⁶ et al. It has the following features.

The input sensors are neutron-sensitive ion chambers. These chamber currents, after suitable amplification and modification, are used to initiate safety action; that is, a rapid decrease of current to the electromagnets which hold up the shim-safety rods. Rod release so obtained takes place automatically if (1) reactor power level exceeds a preset safe maximum value or (2) if the positive reactor period is less than a preset safe value.

The system allows wide latitude in redundancy of individual channels. In Fig. 1, it may be noted that the four safety channels, two level and two period, terminate at a common junction, the sigma bus. The electronic

²⁵W. S. Mruk, ISA J. 4, 206 (1957).

²⁶T. E. Cole, E. E. St. John, and S. H. Hanauer, The MTR Safety System and Its Components, ORNL-1139 (Apr. 4, 1952).

design is such that the sigma bus voltage is that produced by the safety channel having the highest output. Any reasonable degree of redundancy may be used, depending on the particular reactor design and operation.

Figure B-1 diagrams the system used in these tests.

Period Channel²⁷

This channel is designed to span the six decades of reactor operation below 100% rated power. The primary sensor is a neutron chamber with adjustable compensation.^{28,29} Chamber current at 100% power is adjusted to be 50×10^{-6} amp. It is impossible, in a practical operating reactor, to attain ideal compensation under all conditions. It is well recognized that an over-compensated chamber may be an initiating source of false scrams during reactor startup. They cannot be related to the period on which the reactor power is rising. Test transients were established at extremely low neutron levels; therefore, the compensated chamber output was well below its lower threshold of 5×10^{-11} amp.

It was not possible before each test run to determine whether or not the chamber was properly compensated and data obtained under these conditions would be suspect. This difficulty was surmounted by using an auxiliary circuit, Fig. 2, suggested by Hanauer³⁰ of the ORNL Reactor Controls Department. This device, a well-filtered, high-impedance positive current supply, acts as a variable source of synthetic neutrons and does not substantially alter the response of the log $\underline{N}$ amplifier. As used, the potentiometer was adjusted before each transient so that the log $\underline{N}$ recorder was barely on scale, thus ensuring that the positive current flow into the log $\underline{N}$ amplifier was 5×10^{-11} amp.

²⁷W. H. Jordan, H. D. Frey, and George Kelly, An Instrument For Measuring the Logarithm of Neutron Level and the Period of the Pile, ORNL-110 (Nov. 23, 1948).

²⁸H. E. Banta and S. H. Hanauer, private communication, Feb. 8, 1954.

²⁹J. L. Kaufman, private communication, May 25, 1960.

³⁰S. H. Hanauer, A New Circuit for Preventing Certain Unnecessary Period Scrams, to be published.

Fig. B-1. Block Diagram of BSR-II - SPERT-I Safety Elementary.

The log N amplifier produces a voltage proportional to the logarithm of chamber current and spans six decades of reactor operation. The period amplifier³¹ differentiates the output of the log N amplifier and its voltage output is, therefore, the inverse of the reactor period. In these tests the period channels were set to scram for periods of one second or less.

Level Safety Channel

The level safety channel uses a neutron-sensitive, parallel, circular plate ion chamber³² as the input sensor. The system is designed so that chamber output at 100% reactor power is set at 30×10^{-6} amp. Compensation is not required because, in the region of operating power, the prompt fission gammas are overwhelmingly preponderant and augment the neutrons in contributing to chamber output. A simple, linear preamplifier (cathode follower) is used between the chamber and the sigma amplifier. The useful span of the level channel is slightly over the two decades above 1% rated reactor power. Safety action (scram) was initiated at 150% rated power. Time delays in the electronic circuitry up to and including the magnet amplifier output stage are less than 0.001 sec.

Level channel performance tests were made possible by removing the period channels from the system. This was accomplished by the relay contacts shown in Fig. 1. This prevented all shutdowns from being period channel scrams.

Sigma Bus and Magnet Amplifiers

Each individual channel terminates in a sigma amplifier which supplies a voltage to the sigma bus.³³⁻³⁵ The sigma bus voltage is determined by the channel having the highest output voltage. The addition,

³¹H. E. Banta and S. H. Hanauer, private communication, Feb. 8, 1954.

³²R. K. Abele and J. C. Gundlach, The Neutron Sensitive P.C.P. Ionization Chamber, ORNL-1080 (Sept. 4, 1951).

³³T. E. Cole, E. E. St. John, and S. H. Hanauer, The MTR Safety System and Its Components, ORNL-1139 (Apr. 4, 1952).

³⁴C. S. Walker, private communication, June 24, 1959.

³⁵J. B. Ruble and S. H. Hanauer, private communication, Mar. 7, 1955.

within limits, of extra safety channels produces a slight but unimportant loading effect. Figure B-2 displays curves of magnet current and sigma bus voltage for both a single and a pair of level channels in parallel.

The magnet amplifier³⁶ supplies the necessary current to the control rod magnets. Reactor scram is accomplished by the sharp reduction of magnet current as sigma bus voltage increases above the 100% reactor power value which is nominally 37 v.

Fig. B-2. Safety System Characteristics as Influenced by the Number of Input Channels Used.

Control Rod Drives

The control rods³⁷ were driven by a modular rod drive.³⁸ The rod drive is designed so that the drive motor, rod magnet, control rod, and the nonstandard fuel element in which the control rod operates form a self-contained unit. This module occupies the same space in the core as

³⁶H. E. Banta and S. H. Hanauer, private communication, Feb. 8, 1954.

³⁷E. G. Silver and J. Lewin, Safeguard Report for a Stainless Steel Research Reactor for the BSF (BSR-II), ORNL-2470 (July 16, 1958).

³⁸Engineering Department Drawing E-30875 and Research Shops Drawing D-150-1.

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Fig. B-3. Detail of BSR-I Shim-Safety Rod Drive Showing Accelerating Spring and Magnet.

a standard fuel element and may be inserted in any grid plate hole. The design will not allow the control rod to fall out of the core; it eliminates the possibility that during normal operation a control plate, wedged or otherwise stuck in a fuel element, will withdraw the entire fuel element from the core and subsequently release it after the servo has corrected for the withdrawal. Removal of the rod drive with its securely attached fuel element always decreases reactivity.

Figure B-3 details the section of the rod drive occupied by the magnet and accelerating spring. It can be seen that the spring is always fully compressed when the control rod armature is attached to the magnet; therefore, spring force does not depend on rod position. When fully compressed 4.5 in. the spring exerts 66 lb of force on the shim safety rod and this force plus gravitational attraction produces an initial rod acceleration of 6.5 g.

Low Level Counting

The fission chamber channel, Fig. B-4, is not standard in that a second linear pulse amplifier was used as a repeater at the control room end of the half-mile-long cable. No difficulties were experienced with this arrangement.

Fig. B-4. Low Level Counting Channel.

Installation of BSR-II and SPERT-I

Figure B-5 shows the building layout of SPERT-I. These locations are noted on Fig. B-1. Figure B-6 is a photograph of the grid plate and rod drives in the tank and was taken before loading. Figure B-7 is another

Fig. B-5. SPERT-I Plot Plan.

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Fig. B-6. Grid Plate and Rod Drives in the Tank Before Loading.

Fig. B-7. Another View of the Grid Plate and Rod Drives in the Tank.

Fig. B-8. Chamber Locations.

view. Figure B-8 shows the location of the various chambers with respect to the core.

Unlike most of the SPERT-I self-shutdown tests these tests were conducted with the core as fully submerged as the tank would permit; water level above core midplane was 9.8 ± 0.2 ft for all tests.

Appendix C

SIMPLE METHOD OF CALCULATING SAFETY SYSTEM PERFORMANCE

J. R. Tallackson

The ratio of peak power $\phi_{\max}$ to scram power ϕ_s is a logical figure of merit in the evaluation of a fast safety system.

It is well recognized that short-period reactor excursions, characterized by excess reactivity exceeding β , are adequately described by the kinetic equation which ignores the delayed neutrons. If we also ignore, in the one-velocity one-region equation, the product $(\delta k)\beta$, the reactor power ϕ is described by

$$\frac{d\phi}{dt} \cong \frac{\phi}{\ell} (\delta k - \beta) . \quad (1)$$

Expressing the control system's insertion of negative reactivity as an explicit function of time, δk in Eq. (1) becomes

$$\delta k = \delta k_0 - \delta k(t) \quad (2)$$

Consider the somewhat idealized situation in which the reactor, at $t \leq 0$, is on a transient of constant short period τ_0 , which is being produced by excess reactivity δk_0 , then

$$\tau_0 = \frac{\ell}{\delta k_0 - \beta} \quad (3)$$

at $t = 0$ (the signal level in the safety system is at the scram point ϕ_s). The safety system experiences a variety of delays; electrical, mechanical, magnetic, etc., and hence negative reactivity insertion does not commence until t_1 , at which time power is ϕ_1 . At t_1 reactivity begins to change in accordance with Eq. (2). Integrating Eq. (1), inserting the above initial conditions, and taking into consideration the simple exponential rise

in power during the interval t_1 , the total rise in power after the reactor has reached scram level is given by

$$\frac{\phi_{\max}}{\phi_s} = \exp \left[\frac{\delta k_o - \beta}{\ell} T - \frac{1}{\ell} \int_{t_1}^T \delta k(t) dt \right], \quad (4)$$

where T is the time of $\phi_{\max}$ and the time at which net excess reactivity equals β .

The foregoing is diagrammed in Fig. C-1 and it can be seen that the ratio

$$\frac{\phi_{\max}}{\phi_s} = \exp(\text{area } ABCDE) .$$

Inspection of the diagram shows the merits of minimizing fixed delays and benefits obtained by the rapid insertion of negative reactivity.

Fig. C-1. Graphic Solution for the Ratio $\phi_{\max}/\phi_s$.

Appendix D

DATA PREPARATION AND REDUCTION

All test data reported herein were extracted from oscillograph records. Figure D-1 is a section cut from a typical recording and displays the raw data recorded during the excursion.

The individual traces are interrupted momentarily and in sequence. The sequential location of the short gaps thus identifies the traces.

After the individual trace was identified, the procedure was as follows:

1. Scram time was located. Scram time, in this report, is the average time at which the three shim-safety rod clutch switches opened. Switch opening times are located by a jump in the oscillograph traces, which were used to identify switch condition.

2. The time at which the transient rod was fully inserted was located. The rod seat switch was used as the signal source in the same manner as the clutch-switches. See (1) above.

3. The power traces were read. In order to obtain sufficient range with the linear power instrumentation³⁹ three channels of decreasing sensitivity were usually required (see Fig. D-2). On Fig. D-1, record A, these are traces numbered 20, 21, and 22. The flat-topped portion of trace 20 is the region of amplifier saturation and hence useless. In the burst region the 10-msec timing marks were further subdivided into 2.5-msec intervals, and trace deflections were measured with a Gerber Oscillograph Automatic Tracer (GOAT).

These data, GOAT deflection readings, times, and static power trace calibration constants in units of power per millimeter of trace deflection, were reduced by the Oracle which was programmed for ORBIT⁴⁰ (Oracle Binary Internal Translator) Algebraic Programming Code. The Oracle performed the following operations:

1. converted GOAT readings to millimeters of deflection,

³⁹F. L. Bentzen, SPERT-I Instrumentation, IDO-16316 (Mar. 15, 1957).

⁴⁰

J. D. Cummins, Diode Response and Time Derivative of Response to an Exponentially Increasing Input Current, AEEW-MIO, Atomic Energy Establishment, Winfrith, Dorchester.

2. computed reactor power and tabulated same as a function of time,
3. normalized computed power and recorded normalized values on punched tape for subsequent linear and logarithmic curve plotting.

Figure D-3 is the data processing flowsheet.

An automatic curve plotter, using the punched tape as input information, produced linear and logarithmic curves of reactor power vs time. The linear plots were graphically integrated to provide energy to the peak of the burst.

Fig. D-1. Section Cut from a Typical Recording.

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Fig. D.2. Wide Range Recording of Burst Power.

Fig. D-3. Oracle Flowsheet for Orbit Code. Reduction of transient test data, BSR-II safety system tests at SPERT-I.

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