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SPERT PROJECT

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NATIONAL REACTOR TESTING STATION
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Previous Quarterly Reports
Spert Project

<u>1957</u>	
<u>Quarter</u>	<u>IDO No.</u>
3	IDO-16416
4	IDO-16437
 <u>1958</u>	
1	IDO-16452
2	IDO-16489
3	IDO-16512
4	IDO-16537
 <u>1959</u>	
1	IDO-16539
2	IDO-16584
3	IDO-16606
4	IDO-16616
 <u>1960</u>	
1	IDO-16617
2	IDO-16640
3	IDO-16677
4	IDO-16687
 <u>1961</u>	
1	IDO-16693
2	IDO-16716
3	IDO-16726
4	IDO-16750

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SUMMARY

SPERT I

Integral-core destructive tests are planned for Spert I during the summer and fall of 1962. The highly enriched, aluminum-clad plate-type core to be used in these tests has been loaded and a number of static nuclear parameters have been measured. The core has a total available excess reactivity of about \$8.2 and a shutdown reactivity margin of about \$4.4. The average uniform void coefficient of reactivity is about $-36\phi/\%$ decrease in moderator density and the maximum measured coefficient for voids in the center of the core is approximately $-100\phi/\%$. The reduced prompt neutron lifetime measured by analysis of the statistical behavior of the neutron population in the sub-critical core is about 8.3 msec.

SPERT III

Previously reported calculations had been performed to determine if the predicted reactivity compensation resulting from fuel plate expansion and conduction heating of the moderator were adequate to account for the reactivity compensation of the time of peak power inferred from the power data, for non-boiling Spert III power excursions. The discrepancy which had been observed has led to efforts to further refine the method of calculation. The more accurate method of obtaining the local void coefficient used in the new calculations yields an increase of approximately 20% in the calculated values for the reactivity compensation, but the calculated value is still lower than the experimentally derived value by about 20% at short periods.

ANALYSIS AND DATA PROCESSING

A method for evaluating the frequency response of a system from its transient response has been programmed in FORTRAN for the IBM 650 computer. This program will be useful in analysis of excursion data leading toward identification of the physical mechanisms responsible for reactivity compensation.

A new analog-to-analog data processing method has been developed to shorten the time required to obtain quick replots of oscillographic data for operational and test-planning purposes.

ENGINEERING

A complete plant inspection has been made to determine the condition of the Spert III coolant system components, after approximately 1600 hours of plant operating time. Only minor defects were noted in the inspection.

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I. SPERT I

1. INTRODUCTION

Integral-core destructive testing has long been a part of the experimental program objectives for the Spert I facility. The present Spert I schedule includes the performance of such destructive tests during the summer and fall months of 1962. Of the many systems which were considered for such a program, that for which the largest volume of data and analysis is available concerning excursion behavior is the highly enriched, aluminum-clad plate-type core in an open reactor tank. The first integral-core destructive tests in the Spert program will be conducted with a simple system of this type. The initial test series, which will commence in 1962, will be devoted principally to a study of the reactor response, the transient pressure source, and the damage effects occurring to the reactor for an accident situation which results in violent core disassembly and/or meltdown. This test series will be performed in Spert I with a core composed of the Spert type "D" fuel assembly in which all fuel plates are removable, permitting replacement of instrumented and damaged plates as necessary to the program. Following the completion of tests on the oxide core in September 1961 [1], the Spert I facility was modified to accommodate the plate-type core which will be used for the destructive test series. This core was loaded to criticality early in March 1962 and an operational core loading was subsequently established which provided the excess reactivity necessary for the destructive testing. During the remainder of the first quarter of 1962, measurements were made of various static nuclear parameters for this Spert I "D" core. A discussion of these measurements and the results obtained is presented in the following sections.

2. PHYSICAL DESCRIPTION OF THE SPERT I "D" CORE

The Spert I "D" core, which will be used for the destructive testing program, is comprised of a 5 x 5 array of 3- x 3- x 24-in. long assemblies containing 12 fuel plates each. Each plate is composed of a highly enriched uranium-aluminum alloy clad with 0.020-in.-thick type-6061 aluminum cladding and contains 14 g of U-235. The active volume in each plate is 24 x 2.45 x 0.020 in. thick. The water channel between each plate in the assembly is a nominal 0.179 in. A plan view of the operational core loading, shown in Figure 1, indicates the relative positions of the fuel, transient rod, and control rods. There are four double-blade type control rods which are operated as a gang. The transient rod, also a double-blade type, is located in the center of the core and is operated independently of the control rods. Each assembly

Fig. 1 Plan view of the Spert I "D" core showing relative locations of the start-up instrumentation.

which contains a control or transient rod, has six fuel plates in contrast to the normal twelve plates. The poison sections of the control and transient rod blades are composed of "Binal"*, a boron-aluminum alloy containing 7 wt% boron. Each control rod has an aluminum follower-piece. The position of the poison section in the transient rod is in the lower section of the rod, the upper section being composed of aluminum. The principal mechanical features of the core are summarized in Table I.

3. MEASUREMENTS OF THE CONTROL ROD WORTH

The differential reactivity worth of the ganged control rods was measured by the period method. Incremental additions of boric acid to the water moderator were used as a reactivity shim to permit measurement of the rod worth over the total range of rod positions. Worth measurements of rod positions between those obtained by boric acid addition were made by using the transient rod as a reactivity shim.

The resulting rod-worth measurements are shown in Figure 2. There is a considerable amount of scatter in the data which has been attributed to uncertainty in rod position. Some of the scatter may also have been caused by incomplete mixing of the boric acid.

Integration of the rod-worth curve indicates a total available excess reactivity of about \$8.2 at 20°C. This value of excess reactivity indicates that the shortest possible period for excursions with the Spert I "D" core is about 1 msec. No specific measurements of the integrated transient rod worth were made; however, indications are that it is worth about \$7.1 in reactivity, corresponding to a minimum available period of about 1.3 msec. A conservative value for the "minimum" shutdown reactivity margin of \$4.4 is obtained by assuming that the differential control rod worth below the cold clean critical position (9.2 in.) is a linear function of position and is zero at zero in. withdrawal. The actual shutdown margin should be greater than \$4.4.

4. MEASUREMENT OF VOID COEFFICIENT

Measurements have been made of the void coefficient of reactivity since this quantity is needed for calculations of reactivity compensation during excursions and to permit making correlations with other cores. Two types of measurements were made using simulated voids, one for uniform void distribution throughout the core and the other for a 4-in.-high voided region at various axial positions in the central region of the core. Magnesium strips were used to simulate the voids.

The results of the uniform void coefficient measurements are shown in Figure 3. The average void coefficient corrected for absorption in the Mg is $-35 \phi/\%$ decrease in moderator density, or $-0.069 \phi/\text{cm}^3$ of water removed.

* Trademark for the Sintercast Corp. - aluminum-boron powder-metallurgy processed material.

TABLE I
MECHANICAL PROPERTIES OF SPERT I D-12/25 REACTOR

Reactor type	Open pool
Moderator-reflector	H ₂ O
Vessel size	10-ft ID x 16 ft high
Vessel material	Carbon steel, covered with corrosion-resistant paint.
Number of fuel assemblies	25
Standard fuel assemblies	20
Modified fuel assemblies	5
Approximate fuel assembly size	3 in. x 3 in. x 25 in.
Number of fuel plates per assembly	
Standard	12
Modified	6
Total number of fuel plates in core	270
Fuel plate thickness	0.060 in.
Meat thickness	0.020 in.
Clad thickness	0.020 in.
Active length of fuel plate	24 in.
Coolant channel thickness	0.179 in.
Meat	93% enriched U-Al alloy
U-235 per fuel plate	14 g
Total U-235 in core	3.8 kg
Total core volume	5.3×10^3 in. ³
Moderator volume	3.2×10^3 in. ³
Metal-to-water ratio	0.66
Heat transfer area	3.4×10^4 in. ²
Control rods	4 double-bladed, gang-operated
Transient rods	1 double-bladed
Control and transient rod poison material	Binal ^(a) (7 wt% boron-Al alloy) aluminum alloy)

(a) Binal - Trademark for the Sintercast Corporation; boron-aluminum powder-metallurgy processed material.

The uniform voids were simulated by inserting 30-in.-long x 0.610-in.-wide x 0.159-in.-thick Mg strips in alternate channels of each fuel assembly except those containing control rods. The voids were staggered to help reduce interaction, and only 24 in. of the length was considered as being void producing (ie, the active length of the fuel). The critical control rod position was then measured. Following the critical position measurement, some of the voids were removed in a uniform fashion and the critical position remeasured. This process was repeated until all the void strips were removed from the core. The reactivity loss for each void configuration was obtained by integrating the worth curve of Figure 3.

The measured uniform void coefficient was used to obtain a preliminary estimate of the isothermal temperature coefficient of reactivity. Using the thermal coefficient of expansion of water the temperature coefficient is found to be $-0.7\phi/^{\circ}\text{C}$.

Measurements of the axial variation in the void coefficient in the center of the core were performed by using 4-in.-long x 0.610-in.-wide x 0.139-in.-thick magnesium strips. The actual amount of void represented by the strips was obtained by measuring their water displacement. These voids were located in the core as shown in the plan view of Figure 4 and were attached to the transient rod mechanism to permit control of the vertical position of the voids. The axial profile of the void coefficient was obtained by measuring the control rod critical position as a function of void position. The results of this measurement are shown in Figure 5. The peak of the void worth curve occurs at about 8.5 in. from the bottom of the fuel.

Fig. 3 Spert I "D" core uniform void worth.

Fig. 2 Differential rod-worth for the Spert I "D" core.

Fig. 4 Plan view showing location of magnesium void strips in central portion of the Spert I "D" core.

The void coefficient at the position corresponding to this peak (not corrected for Mg absorption) is approximately \$1/% decrease in moderator density. As indicated in Figure 6, the reactivity loss at this position varies linearly (over the range of void measured) with the amount of moderator voided. An absorption correction will be applied to the central void worth measurement when flux distribution data are available.

Fig. 5 Reactivity loss as a function of axial void position in the central portion of the Spert I "D" core.

Fig. 6 Central core region void worth, Spert I "D" core.

5. MEASUREMENT OF REDUCED PROMPT NEUTRON LIFETIME

A static value of the reduced prompt neutron lifetime (ℓ/β_{eff}) has been determined for the Spert I operational core by analysis of the statistical behavior of the neutron population. A subcritical counting experiment was performed from which the variance-to-mean ratio of the counting rate was analyzed as a function of counting time and least-square fitted to the theory to give a value for the reduced lifetime of 8.3 ± 0.4 msec. The variance-to-mean statistical technique utilized in this experiment was similar to that previously described [2]. The data were fit to the theoretically derived equation which accounts only for those fluctuations in neutron population attributable to prompt neutrons. The neglect of delayed neutrons should have little effect on these results since the prompt root of the inhour equation is much greater than the largest delayed root for this system.

In order to solve the variance-to-mean relationship for the reduced lifetime, it was necessary to know the values of prompt reactivity at which counting data were taken. Prior to taking data, a double-blade stainless-steel regulating rod was inserted in the Spert I core in place of the transient rod. The total reactivity worth of this regulating rod was approximately \$2.5 with a maximum differential worth of about 18.5 ϕ /in. at about 8 in. withdrawn. Since relatively large movements of the regulating rod resulted in small reactivity changes the inherent uncertainty in rod position was negligible for the lifetime measurements.

Subcritical counting data were taken for three values of reactivity determined by use of the regulating rod worth curve. Control rod critical position was established with the regulating rod withdrawn about 12 in. The regulating rod was then inserted a predetermined distance and the reactivity of the

system was calculated from rod worth data. A summary of the reactor operating conditions for these measurements is given in Table II.

TABLE II
SUMMARY OF REACTOR OPERATING CONDITIONS

Run No.	Critical Regulating Rod Position (in.)	Sub-Critical Regulating Rod Position (in.)	Reactivity (ϕ)
1	12.00	11.1	-14.2
2	12.00	9.00	-51.7
3	12.00	10.00	-32.7

The statistical counting data collected during the experiment were recorded on punched paper tape, converted from tape storage to punch card storage, and processed to obtain the ratio of variance-to-mean by means of an IBM 650 computer program. Table III summarizes the counting conditions for the various measurements.

TABLE III
SUMMARY OF COUNTING CONDITIONS

Run No.	Reactivity (ϕ)	Number of Gate Times	Range of Gate Times (Sec)
1	-14.2	39	0.001 to 0.800
2	-51.7	46	0.001 to 3.000
3	-32.7	39	0.001 to 0.800

The variance-to-mean ratios calculated from the counting data have been fitted to theory for the purpose of evaluating the reduced prompt neutron lifetime (ℓ/β_{eff}). The fitting equation used was of the form originally derived by de Hoffmann [3] and may be written as,

$$\frac{\text{Var}}{\text{Mean}} (\tau) = 1 + K \left[1 - \frac{1 - e^{-\alpha\tau}}{\alpha\tau} \right] \quad (1)$$

where

$$\frac{\text{Var}}{\text{Mean}} (\tau) \equiv \text{variance-to-mean ratio} \equiv \frac{\overline{c^2} - \bar{c}^2}{\bar{c}},$$

τ = gate time

c = counts per gate time,

K = a combination of fixed variables (constants),

$$\alpha = \frac{1-k_p}{\ell}$$

k_p = prompt neutron reproduction factor,

ℓ = mean prompt neutron lifetime

A nonlinear least-squares fitting program for the IBM 650 computer was used to evaluate least-squares values of K and α for each set of data. The least-squares values of K and α determined were such that the sum of the squares of the deviations of the datapoints from the fitted curve was a minimum. The input to the program consisted of the measured values of $(\frac{\text{Var}}{\text{Mean}} - 1)_i$ and the gate times, τ_i , at which these measurements were made. The maximum gate time considered was 0.400 sec in order to minimize the effects of delayed neutrons. The least-squares values of K and α along with their standard deviations as calculated by the program are shown in Table IV. The fitted curves and the data points are plotted in Figures 7 through 9.

TABLE IV

LEAST-SQUARES VALUES OF K AND α

Run	α	K
1	135.6 ± 12.8	26.7 ± 0.5
2	171.7 ± 16.2	14.8 ± 0.3
3	163.9 ± 14.6	19.2 ± 0.3

Fig. 7 Variance-to-mean ratio of the counting rate minus one vs the gate time for a subcritical reactivity, $\rho = 14.2\%$.

Fig. 8 Variance-to-mean ratio of the counting rate minus one vs the gate time for a subcritical reactivity, $\rho = -51.7\%$.

Fig. 9 Variance-to-mean ratio of the counting rate minus one vs the gate time for a subcritical reactivity, $\rho = -32.7\%$.

The value of ℓ/β_{eff} was then calculated from the individual α values combined with the measured values of subcritical reactivity. The subcritical reactivity ρ may be defined as,

$$\rho(\beta) = \frac{1 - k_{\text{eff}}}{\beta_{\text{eff}} k_{\text{eff}}} \quad (2)$$

The least-squares value of α is defined as,

$$\alpha = \frac{1 - k_p}{\ell} \quad (3)$$

where the numerator of (3), $(1 - k_p)$, is defined as,

$$(1 - k_p) = (1 - k_{\text{eff}}) + \beta_{\text{eff}} k_{\text{eff}} \quad (4)$$

Dividing both sides of (4) by $\beta_{\text{eff}} k_{\text{eff}}$

$$(1 - k_p)/\beta_{\text{eff}} k_{\text{eff}} = (1 - k_{\text{eff}})/\beta_{\text{eff}} k_{\text{eff}} + 1 = \rho(\beta) + 1 \quad (5)$$

Then, taking (3), dividing the right side by $\beta_{\text{eff}} k_{\text{eff}}$, and substituting the result of (5) gives,

$$\alpha = \frac{(1 - k_p)/\beta_{\text{eff}} k_{\text{eff}}}{\ell/\beta_{\text{eff}} k_{\text{eff}}} = \frac{(\rho + 1) k_{\text{eff}}}{\ell/\beta_{\text{eff}}} \quad (6)$$

But k_{eff} is given by,

$$k_{\text{eff}} = \frac{1}{(1 + \rho \beta_{\text{eff}})} \approx (1 - \rho \beta_{\text{eff}}) \quad (7)$$

The value of β_{eff} in the small correction term $\rho \beta_{\text{eff}}$ may be estimated from the buckling and over the range of reactivities investigated results in less than 0.1% uncertainty. One may now rewrite (6) as,

$$\alpha \frac{\ell}{\beta_{\text{eff}}} = (\rho + 1) (1 - \rho \beta_{\text{eff}}) \quad (8)$$

Thus, ℓ/β_{eff} is obtained as the slope of the linear plot of $(\rho + 1) (1 - \rho \beta_{\text{eff}})$ as a function of α . The experimental plot obtained for the Spert I, D-12/25 core is shown in Figure 10. From a least-squares linear fit of the data,

$$\ell/\beta_{\text{eff}} = 8.3 \pm 0.4 \text{ msec} \quad (9)$$

The statistical data have been fitted to an analytical equation which neglects the effects of delayed neutrons. This approximation introduces a systematic

error which will tend to make the measured value of ℓ/β_{eff} larger than the true value. However, for systems with prompt neutron lifetimes less than 100 μsec , it is possible to collect counting data at gate times short enough so that delayed neutrons should have a negligible effect on the measured value of ℓ/β_{eff} .

The large amount of scatter in the variance-to-mean data plotted as a function of gate time suggests that extraneous noise may have been present in the counting system. A review of a portion of the raw counting data as well as visual observation of the magnetic tape playback recorded on a recording oscillograph has shown rather dense packing of pulses in certain areas as compared to the average count rate. No definite conclusions can be drawn from these observations since direct comparison with a known clean counting system is not now available. The measured value of ℓ/β_{eff} must, however, carry the qualifications that extraneous noise pickup may have created an uncertainty not accounted for in this analysis.

Fig. 10 Plot of $(\rho+1)(1-\rho\beta_{\text{eff}})$ vs α showing resultant ℓ/β_{eff} .

II. SPERT III

1. CALCULATIONS OF NON-BOILING REACTIVITY COMPENSATION IN SPERT III POWER EXCURSIONS

A previous quarterly report [4] described the calculations which were performed in an effort to determine if the predicted reactivity compensation resulting from fuel plate expansion and conduction heating of the moderator were adequate to account for the observed reactivity compensation at the time of peak power for non-boiling Spert III power excursions. The predicted values of reactivity compensation thus calculated were significantly less than the values inferred from the experimentally observed power data. As a first step in a program to further refine the calculational method to include effects not previously considered, the method of obtaining the local void coefficient has been improved.

For the original calculations, the Spert III core was considered to be divided into four radial regions and eighteen axial regions. The contribution to reactivity compensation was then calculated for each local region of the core and the total reactivity compensation obtained by summing over all regions of the core. In order to evaluate the reactivity compensation for each local region, the calculated void volume for each region was multiplied by a local void coefficient $C_{r,i}$, where r denotes a radial region and i denotes an axial region. Values for the local void coefficients were determined as follows:

An average void coefficient $\bar{C}_r$ was measured for each radial region, using aluminum strips to simulate voids. Values of the local void coefficient $C_{r,i}$ were also measured for the central four fuel assemblies (radial region 1). Since the axial neutron flux distribution is known throughout the core, it was assumed that the local void coefficient is proportional to the axial flux distribution and the measured average void coefficient $\bar{C}_r$ for each radial region. Thus a relationship of the form

$$C_{r,i} = \bar{C}_r \left(\frac{\phi_{r,i}}{\bar{\phi}_r} \right)_{r,i} K_{r,i} \quad (10)$$

was assumed. In Equation (10), $\phi_{r,i}/\bar{\phi}_r$ is the ratio of the flux in region (r,i) to the average flux for radial region (r) and $K_{r,i}$ is a proportionality constant. It was further assumed that the value of $K_{r,i}$ is the same for all radial regions of the core.

Since the local void coefficients had been measured for radial region 1 ($r = 1$), in the original calculations the value of $K_{r,i}$ was determined as the ratio of the relative axial void coefficient to the relative axial flux for radial region 1. This ratio is written as $\left(\frac{R.V.E.}{R.F.} \right)_{1,i}$. Using this ratio for $K_{r,i}$ Equation (10) becomes:

$$C_{r,i} = \bar{C}_r \left(\frac{\phi_{r,i}}{\bar{\phi}_r} \right)_{r,i} \left(\frac{R.V.E.}{R.F.} \right)_{1,i} \quad (11)$$

In recent efforts to improve the calculational procedure, Equation (11) has been modified to give a more accurate determination of the local void coefficient. The modified expression for $C_{r,i}$ is determined in the following manner:

It is assumed that the local void coefficient is proportional to the flux ratio $[(\phi_{r,i})^2 / \overline{(\phi_{r,i})^2}]$ rather than to the ratio $(\phi_{r,i} / \bar{\phi}_r)$ used in the previous approach. Thus, a relationship of the form

$$C_{r,i} = \bar{C}_r \left[\frac{(\phi_{r,i})^2}{\overline{(\phi_{r,i})^2}} \right]_{r,i} K_{r,i} \quad (12)$$

is postulated. The proportionality factor $K_{r,i}$, is assumed to be given by the ratio of the relative axial void effect in radial region 1, $(C_{1,i} / \bar{C}_1)$ to the ratio $[(\phi_{1,i})^2 / \overline{(\phi_{1,i})^2}]$, and is considered to be the same for all radial regions of the core. Then

$$C_{r,i} = \bar{C}_r \left(\frac{C_{1,i}}{\bar{C}_1} \right) \left[\frac{(\phi_{1,i})^2}{\overline{(\phi_{1,i})^2}} \right] \left[\frac{(\phi_{r,i})^2}{\overline{(\phi_{r,i})^2}} \right] \quad (13)$$

As a check on the relative accuracy of Equation (13) as compared with Equation (9), the relative void effect $C_{1,i} / \bar{C}_1$ for radial region 1 was compared with the expressions $(\phi_{1,i} / \bar{\phi}_1) (\frac{R.V.E.}{R.F.})_{1,i}$ and $[(\phi_{1,i})^2 / \overline{(\phi_{1,i})^2}]$, as shown in Figure 11.

From this figure it is seen that the ratio $[(\phi_{1,i})^2 / \overline{(\phi_{1,i})^2}]$ is a much better approximation to the relative void effect than the term $(\phi_{1,i} / \bar{\phi}_1) (\frac{R.V.E.}{R.F.})_{1,i}$. Also, comparisons of the ratio $[(\phi_{1,i})^2 / \overline{(\phi_{1,i})^2}]$

with analogous ratios for radial regions 2, 3, and 4 have been made, as shown in Figures 12, 13, and 14, respectively. From these figures, it is seen that the term $[(\phi_{1,i})^2 / \overline{(\phi_{1,i})^2}] [(\phi_{r,i})^2 / \overline{(\phi_{r,i})^2}]$ is very close to unity for all r.

From these results, it was concluded that Equation (13) could be taken as a valid expression for determining the local void coefficient $C_{r,i}$ from the void and flux measurements made in Spert III.

Fig. 11 Comparison of relative void worth with various flux ratio functions for the Spert III core.

Fig. 12 Comparison of the flux ratio functions for core regions 1 and 2.

Fig. 13 Comparison of the flux ratio functions for core regions 1 and 3.

Fig. 14 Comparison of the flux ratio functions for core regions 1 and 4.

Use of Equation (13) instead of Equation (11) to calculate $C_{r,i}$ gave an increase of approximately 20% in the calculated values for the reactivity compensated at the time of the power peak as a result of fuel plate expansion and conduction heating of the moderator. Even with this sizeable correction, however, the results as shown in Figure 15, still indicate that the reactivity compensated at the time of the power peak by these two mechanisms is insufficient to account for the observed self-shutdown behavior during non-boiling power excursions in the Spert III reactor. The remaining discrepancy varies from about 20% for tests with periods of 10 msec to about 100% for periods of about 160 msec.

Further refinements of the calculational model will include consideration of the importance of possible variations in the neutron flux distribution during the course of power excursions.

Fig. 15 Comparison of the non-boiling calculated and experimental compensated reactivity at the time of peak power vs reciprocal period for the Spert III core.

III. ANALYSIS AND DATA PROCESSING

1. AN IBM 650 FORTRAN PROGRAM FOR FREQUENCY RESPONSE EVALUATION

The "effective" reactivity-feedback transfer function for a reactor can be determined by analysis of the measured time response of the reactor power to a step-wise input of reactivity. The observed variation in this deduced transfer function as a function of the size of the applied reactivity step can be useful in obtaining valuable insights to aid in the identification of the physical mechanisms responsible for reactivity compensation. For relatively modest reactivity steps, reactivity-feedback processes may be considered as essentially linear. Within this range, therefore, the only important effect of an increase in the reactivity step in the present connection is to increase the average power during the power burst. This leads to a proportionate increase in the steady-state value of the deduced reactivity-feedback transfer function, without any variation in the deduced time constants. For larger reactivity steps, the nonlinearities in the reactivity-feedback processes play a significant role in the reactor response. Under these conditions, both the steady-state value and the time "constants" in the deduced reactivity feedback function would be expected to vary.

By examining the magnitudes of the "effective" steady-state feedback reactivity and of the "effective" time constants, as well as their variation with the size of the reactivity step applied, it is possible to develop useful tests of the adequacy of various proposed models for the reactivity feedback processes as well as to obtain suggestions for the physical basis for such model.

A method for evaluating the frequency response of a system from its transient response has been programmed in FORTRAN for the IBM 650 computer. The frequency response,

$$G(j\omega) = \frac{\int_0^{\infty} A(t)e^{-j\omega t} dt}{\int_0^{\infty} B(t)e^{-j\omega t} dt} \quad (13)$$

is evaluated from digital data representing the time-dependent system output $A(t)$ resulting from an arbitrary time-dependent input $B(t)$. The functions $A(t)$ and $B(t)$ must be continuous and finite and must approach constant final values. However, an option is also provided in which $B(t)$ is specified as a unit impulse.

Within computer running time limitation, any number of points equally spaced in t may be used to represent $A(t)$ and $B(t)$. The frequency response function may be evaluated at up to twenty values of ω . The output of the program includes the real and imaginary parts as well as the magnitude and phase of $G(j\omega)$. In addition, $20 \log_{10} |G(j\omega)|$ is computed.

The computer running time for this program is approximately 1.2 sec per time point per value of frequency. Because of the rather excessive running time, it is planned to reprogram the method directly in IBM 650 machine language.

The program will also be useful in evaluation of the performance of data instrumentation channels, permitting the detailed frequency response to be determined from simple transient-response measurements.

2. NEW ANALOG-TO-ANALOG DATA PROCESSING EQUIPMENT

Because of the rapidly expanding kinetics testing program at Spert, there exists an increasing need for prompt reduction of power, temperature, period, and energy data shortly after the completion of each test. Since this information is required principally for operational and test-planning purposes, high precision is not demanded.

With the normal method of data processing in use at Spert, this demand could not easily be met because of the time required to digitize the analog-type oscillographic traces and then replot them using punched-card input to a digital plotter. A new auxiliary data processing method has been developed in which the oscillographic data are replotted directly on an x-y plotter without an intermediate digitizing step. With this method, test data in a reduced analog form are made available within two to three hours after the completion of the test.

This analog-to-analog data reduction method uses an oscillograph curve follower, designed and constructed at Spert. One operator using this equipment can produce approximately forty reduced plots per day. By comparison, two operators, using the digital conversion equipment, can digitize only about 36 traces of data per day which then must be plotted by means of the relatively slow digital plotter.

The unit consists of an oscillograph curve follower connected through electro-mechanical servo-mechanisms, including log converter and integrator circuits, to an analog x-y plotter. With this equipment, the operator maintains the reading head index on the oscillograph trace as the head is translated over the record by a foot-pedal-controlled motor drive. The log converter circuit and integrator circuit permit the reduced plot to be a linear or logarithmic representation of the variable or its integral. This reduction operation can be carried out by a trained technician with an accuracy of better than $\pm 2.5\%$ of the full scale deflection of the oscillograph trace. Gain and zero offset controls on the amplifiers permit the reduced plot to be made in real units with compressed scales.

IV. ENGINEERING

1. SPERT III PLANT INSPECTION

Since a number of relatively untried construction features had been included in the high pressure, high temperature, Spert III primary coolant system, periodic inspections of the complete system have been included in the operating schedule. The first of these inspections had been planned for early in 1963. Because of the failure of the Spert III pressurizer vessel in October 1961 [5] and the resultant shutdown of the facility pending procurement and installation of the replacement vessel, this inspection was performed during the first quarter of 1962.

Because of the cyclic nature of operation of the Spert III primary coolant system, particular interest was centered on the condition of the spun-cast stainless steel piping, the heat exchanger outlet mixing nozzles, and the multi-layer reactor vessel. The spun-cast piping includes all pipe between 8 in. and 16 in. diameter, which constitutes the majority of the primary system piping.

The primary system piping welds were inspected by the Zyglo Fluorescent penetration process developed by the Magnaflux Corporation. Of the primary spun-cast piping, the following component welds were inspected:

West Loop:

Reducer - 16 in. to 10 in. - inlet to heat exchanger

Heat exchanger head inlet - 10 in.

Check valve section - downstream of No. 1 primary pump

Tee - 16 in. x 16 in. x 8 in. - downstream of main flow-control valve

East Loop:

Inlet nozzle to No. 4 primary pump

Mixing nozzle section - outlet of heat exchanger

Removable spool section - between heat exchanger and mixing nozzle

Reactor Vicinity:

Removable spool section - in the torus in the reactor outlet piping

Tee - 12 in. x 12 in. x 8 in. - reactor outlet piping to torus

Tee - 12 in. x 12 in. x 16 in. - reactor return piping to pumps from torus

The two spool sections, one in the reactor outlet piping and the other in the east loop, were removed from the primary piping. The sections were

dye-checked and dimensionally inspected both inside and outside. Outside dimensional inspections were also made at various locations on other sections of the spun-cast primary piping. No noticeable weld defects or evidence of material yielding were found during the inspection of the spun-cast piping and the heat exchanger mixing nozzles.

Complete dye-checks and dimensional inspections were made of the small diameter (4 in. and under), type 304L, seamless, stainless steel primary piping. The condition of this piping was of particular interest since this was the first comprehensive inspection of the piping since design allowables for the type 304L material had been included in the ASME code. On the basis of the new code this piping would be considered to be overstressed for operation at 650°F. However, this inspection showed no measurable yielding or any other evidence of distress. It was therefore concluded that on the basis of operating history and the complete inspection that the presently installed piping is adequate from a safety standpoint.

During the complete primary piping inspection, no evidence was observed of significant oxidation or corrosion damage. The small diameter seamless piping was coated with a thin oxidation film deposit less than 1/64 in. thick. The spun-cast piping had no oxidation deposits that showed a blue-black discoloration.

When the removable spool section in the torus was examined, a thermocouple, which had been installed in the spool section, was observed to be damaged. This thermocouple and two others had been installed during plant construction to permit measurement of the inside wall temperature of the piping. These thermocouples were completely sheathed with stainless steel using magnesium oxide for insulation. Upon discovery of the damaged thermocouple, the other two were checked and one was found to be damaged. The damage consisted of a split in the sheath (see Figure 16) which extended nearly the entire length of the thermocouple that was within the pipe. Since these thermocouples were not of great importance, they were removed from the system, and the pipe thermocouple bore was plugged. The split thermocouples are currently being examined by metallurgists to determine the probable cause of the failure.

After removal of the core barrel, thermal shields, and lower grid from the reactor vessel, dye-checks were made of all internal welds and suspicious areas, and a complete visual inspection and dimensional check was made of the reactor vessel interior. A number of small flaws in the cladding were detected by the dye-check but these were easily removed by grinding. The resulting pits were less than 1/16 in. deep. Since the design thickness of internal clad was selected only to provide for corrosion allowance, the small ground-out pits do not reduce the design strength of the vessel. The pits are not deep enough to appreciably reduce the corrosion allowance.

The visual inspection of the interior of the reactor vessel showed evidence of oxidation. Thin layers of oxidation covered the entire reactor inner surfaces but in certain areas, there were sharp transitions in the thickness. A very sharp transition in the thickness occurred along a vertical line near the south side of the vessel, while on the north side the oxidation film was fairly uniform in thickness. Since the transition line was present on two separate internal cladding plates, it was decided that the cladding was not the cause for the transition in thickness of the oxidation film. The reason for the transition cannot be attributed to any known operating environment.

Fig. 16 Photograph of a Spert III primary system thermocouple showing the rupture of the sheath material.

During the vessel inspection, the reactor inlet and outlet piping were also inspected. The inlet piping consists of a 20-in. tee, in which the two primary flow loops combine to form a single inlet at the bottom of the reactor. This bottom tee was not removed for inspection, but it was possible to descend into the tee for a close visual inspection. No defects were noted other than thin oxidation deposits on the walls. Inspection of the thermal shields in the 8-in. outlet nozzles revealed no defects other than a slight oxidation coating.

The core region thermal shields and core barrel which were removed for the vessel inspection were not disassembled; however, they were subjected to a careful visual inspection. It was noted that the oxidation deposits were darker in color and thicker on the core barrel than on the thermal shields. However, the film thickness on the core barrel was not significant. No material defects were noted for the complete assembly.

The east and west heat exchangers were inspected on both the shell and the tube sides. Considerable oxidation had occurred on the shell side of the heat exchanger and very minor oxidation occurred on the tube side. One tube in the west heat exchanger was blocked by welds. This was done during construction when preliminary testing revealed a leak in the tube. No material defects other than the oxidation film were noted during the heat exchanger inspection.

After approximately 1600 hours operating time on the Spert III plant, only minor defects have been noted during the plant facility inspection.

V. LIST OF SPERT PUBLICATIONS ISSUED DURING THE FIRST QUARTER OF 1962

1. INTRODUCTION

This section presents a listing and short summary of the reports issued by the Spert staff during the first quarter of 1962. Similar listings for 1961 appear in previous quarterlies [6, 7].

2. AEC REPORTS

(a) IDO-16743, Spert III Pressurizer Vessel Failure, R. E. Heffner, T. R. Wilson, D. P. Halls, W. W. Hickman, and K. R. Hoopingarner, (Issued January 29, 1962).

On October 26, 1961, at approximately 8:00 p.m. a failure occurred in the Spert III pressurizer vessel, an integral component of the primary coolant system of the Spert III reactor facility. At the time of failure, the reactor contained a subcritical array of fuel, and non-nuclear tests were being conducted to evaluate experimental procedures and the performance of plant equipment in preparation for a series of cold water accident studies scheduled to follow. Because of an error in the indication of the existing level-control instrumentation on the vessel, it was possible for one or more of the electric heaters to operate above the steam-water interface thus causing superheating of the steam and producing overtemperature in the upper half of the vessel. The failure mechanism was third-stage creep at elevated temperatures.

This report contains the results of the study of the cause of failure.

(b) IDO-16741, Spert III Thermowell Failure and Replacement, R. E. Heffner, S. W. Gleave, J. A. Norberg, (Issued February 2, 1962).

In January 1961, an immersion section of a primary coolant system thermowell was found in the bottom of the Spert III reactor vessel. Subsequent evaluation of the failure and the methods to verify the design adequacy of all primary system thermowells are presented in this report.

(c) IDO-16745, Spert IV Facility, R. E. Heffner, E. L. Morris, D. T. Jones, D. R. Seaman, and M. K. Shane, (Issued February 22, 1962).

The Spert IV reactor is a swimming-pool type nuclear research reactor which has been constructed to provide a facility for the conduct of reactor kinetic behavior and safety investigations especially in the field of reactor stability. The facility has been designed to incorporate a wide range of flexibility in flow rates and direction of flow, and has a heat removal capacity of 1 Mw, in addition to the heat sink capacity of the reactor pools. This report describes the engineering features of the reactor and supporting process equipment as constructed at the National Reactor Testing Station.

(d) IDO-16636, Review of Several Models for Power Behavior and Reactivity Feedback in Reactor Excursion Studies, R. W. Garner (Issued February 6, 1962).

One of the overall objectives of the Spert Program for reactor kinetics is to obtain, by experimental and theoretical studies, an understanding of the functional dependence of reactor kinetic behavior on various reactor parameters. A part of the program is concerned with the investigation of the inherent self-limiting response of the reactor system to large reactivity perturbations. A large portion of some analytical work performed at Spert which has been reported in various quarterly reports has been compiled. The work described in this report has been directed primarily at the development of analytical models which describe and correlate the experimental data obtained and permit some extrapolation of the data beyond present experience. Calculations of the reactivity compensated at the time of peak power have been extended to include values obtained for a Clipped Exponential burst shape model and some salient features of the Two-term burst shape model have been emphasized. As a result of investigating the effects of various forms of coupling between reactivity and power behavior, four correlative models are described and compared with experimental results. The qualitative accomplishments of the correlative approach to reactor dynamic behavior are pointed out as well as the shortcomings of each coupling equation.

(e) IDO-16751, Self-Limiting Power Excursion Tests of a Water-Moderated Low-Enrichment UO₂ Core in Spert I, A. H. Spano, J. E. Barry, L. A. Stephan, and J. C. Young, (Issued February 28, 1962).

This report presents the results of a Spert I experimental investigation of the dynamic behavior of a heterogeneous, light-water-moderated reactor, employing low-enrichment UO₂ rod-type fuel elements. Self-limiting power excursion tests, initiated by step insertions of reactivity, were conducted over a wide range of initial asymptotic reactor periods, from 30 sec to 3.2 msec. In the long-period region, reactor shutdown was provided both by the various heat-transfer-dependent mechanisms effecting loss of moderator from the core and by Doppler broadening of the U-238 absorption resonances. In the short-period region, where the reactor period was small compared to the thermal time constant of the UO₂ fuel rods, reactor shutdown was provided essentially by the Doppler reactivity effect alone, which permitted a direct measurement of the dynamic Doppler coefficient of the core. An additional reactivity effect, positive in sign and arising as a result of the systematic bowing of the fuel rods during the transient power burst, was observed to introduce a significant change in burst shape behavior. Separation of this fuel rod bowing effect from other reactivity feedback effects was obtained by performing two series of reactor transient experiments: one in which fuel rod bowing occurred and the other in which bowing was effectively suppressed by mechanically constraining the fuel rods. The shortest asymptotic reactor period test performed was a 3.2-msec test, wherein the maximum power reached was about 7500 Mw, the total nuclear energy released in the transient was about 107 Mw-sec, and the maximum steam pressure developed was about 8 psi. No damage occurred as a consequence of this test.

(f) IDO-16755, Contributions to Two Problems in Space-Independent Nuclear Reactor Dynamics, A. A. Wasserman, (Issued March 30, 1962).

The analytical and experimental studies on the describing function of a nuclear reactor and analytical studies on the effect of a reflector on reactor dynamics which have been performed at Spert in the last several years are described in this report.

Earlier analytical work on the reactor describing function at both low and high power levels has been substantially extended and clarified. Some of the earlier results on the high-power reactor describing function have been shown to be incomplete or incorrect. Several IBM 650 programs have been written to facilitate accurate calculation of the low- and high-power reactor describing function for a variety of reactor types, and illustrative calculations have been presented.

The analytical results for the low-power describing function have been confirmed by pile-oscillator, frequency-response measurements in a small, enriched, plate-type, H₂O-moderated and -reflected reactor (the Spert I P-18/19 reactor), for reactivity amplitudes up to \$0.145 and for frequencies from 0.0002 to 18.4 cps. A novel, reflector-located, rotary, reactivity-oscillator mechanism of adjustable reactivity amplitude was used for these tests.

Agreement between the experimental and theoretical describing-function results was shown to be better when a slightly modified version of Keepin and Wimett's U-235 fast-fission delayed-neutron data was used for the calculations than when Keepin and Wimett's thermal-fission data [8] were used, regardless of the reactivity amplitude or the method of data handling.

VI. REFERENCES

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8. G. R. Keepin and T. F. Wimett, "Reactor Kinetics Functions: A New Evaluation", Nucleonics 16, p 86 (October, 1958).