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SPERT PROJECT
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SUMMARY

A series of tests was performed with the Spert III plate-type core to investigate the effects of the initial reactor power level on the dynamic response of the core. One of the primary purposes for these tests was to provide a basis for evaluating one of the more widely used computer programs for calculating reactor kinetic behavior, the ART-04 program. Comparisons of calculated and experimental results have been made for the following types of transient tests: (a) power excursions initiated by one-dollar stepwise reactivity insertions from initial power levels ranging from 0.01 to 28 MW, (b) power excursions initiated by ramp reactivity insertions of 0.05 dollars/sec from power levels of 1 and 25 MW, and (c) partial and total losses-of-flow from power levels of 10 and 28 MW. The initial conditions for most of the transients were 500°F core inlet coolant temperature, 2500-psig system pressure, and 20 ft/sec core coolant flow. From the results of these comparisons, it has been concluded that the ART-04 program is a useful analytical tool for calculating the transient behavior of cores similar to the Spert III plate-type core.

One of the problems in the experimental program with the Spert III plate-type core has been that of obtaining accurate fuel plate surface temperature data during short-period tests performed under initial conditions involving high temperatures, pressures, and flow rates. The fast-response thermocouples normally used at Spert are too fragile to withstand such environmental conditions. Therefore, for such tests, fuel plate surface temperature measurements have been made with MgO-insulated, stainless-steel-sheathed thermocouples. Tests, however, have shown that the use of these thermocouples can result in substantial errors in the measurements. Using a transfer function approach, a technique has been developed whereby the originally recorded temperature data from these thermocouples can be corrected to agree within approximately 10 percent at the time of peak temperature with that which would be given by the faster-response thermocouples.

During cooldown of the primary system following the last series of tests with the Spert III plate-type core, a failure of one of the liner-motor primary coolant pumps occurred. Subsequent disassembly and inspection of the pump showed that the rotor can, stator, and stator liner were damaged, requiring that the motor section of the pump be rebuilt. It was postulated that the damage to the pump was caused by either a foreign particle or a piece of the nitrided surface of the pump thrust-bearing disk becoming lodged between the rotor can and the stator liner. With continued use of the pump, several holes were worn in the stator liner which allowed water to enter the stator and resulted in a short circuit in the windings. The pump is currently being repaired and the remaining three primary pumps are being disassembled and inspected for similar damage.

A portion of the subassembly test program with the Capsule Driver Core (CDC) in Spert IV includes the planned destruction of test fuel samples. In preparation for such testing, a series of out-of-pile capsule and instrumentation tests is being conducted. The information from these tests is being used (a) to design a capsule for the testing of CDC fuel, (b) to obtain a fundamental understanding of the response of various materials to explosive loading conditions, and (c) to develop instrumentation techniques to be used in capsule tests. In the tests, chemical explosives have been used to mock up rapid

releases of nuclear energy within capsules or cylinders made of 6061-T6 aluminum, 304 stainless steel, and A106 grade B carbon steel.

From the results which have been obtained from these out-of-pile tests, a design for the capsule to be used in in-pile tests of the CDC fuel has been established. Basically, the capsule will consist of a 5-1/2-foot-long piece of two-inch, Schedule 80, 304 stainless steel pipe with a welded plug at the bottom and a bolted flange at the top. Results pertinent to obtaining a fundamental understanding of the response of the various materials have shown that the explosive pressures to which the materials were subjected had sufficiently short rise times ($\approx 10 \mu\text{sec}$) and durations (≈ 100 to $300 \mu\text{sec}$) to invalidate the use of static mechanical properties to calculate the dynamic response for stainless and carbon steel cylinders, but not necessarily for aluminum cylinders. The results also showed that stainless steel and aluminum cylinders failed in the shear mode but that carbon steel cylinders failed in the tensile mode when rapidly overpressured.

During this quarter fuel testing activities continued on the Power Burst Facility (PBF) fuel concept. Details of this activity will be summarized in later quarterly and topical reports. Title I design was essentially completed by the architect-engineer. A complete draft of the Preliminary Safety Analysis Report for the PBF was prepared and submitted to the AEC for review and comment.

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QUARTERLY TECHNICAL REPORT
SPERT PROJECT
JANUARY, FEBRUARY, MARCH, 1965

I. SPERT III

1. INTRODUCTION

During this quarter, preparations for an experimental program with a low-enrichment UO_2 rod-type core in Spert III and analysis of previous tests performed with the C-19/52 plate-type core have continued. This analysis includes (a) comparisons of experimental and calculated reactor behavior during transients initiated at high power levels, and (b) the development of a technique to obtain accurate fuel plate surface temperature data from measurements made during short-period power excursion tests with thermocouples lacking the required frequency-response characteristics. Descriptions of the techniques used and the results of these analyses are presented in the following sections. Also presented is a documentation of a failure of one of the Spert III primary coolant pumps.

2. COMPARISON OF EXPERIMENTAL AND CALCULATIONAL RESULTS
FROM SPERT III TRANSIENT TESTS INITIATED AT
HIGH POWER LEVELS

2.1 General

The last series of tests with the Spert III plate-type, C-19/52 core was performed to investigate the effect of the initial power level on the dynamic response of the core. In this series, tests were performed to determine the reactor response to (a) stepwise reactivity insertions of 1 dollar from initial power levels ranging from 0.01 to 28 MW, (b) ramp reactivity insertions of 0.05 dollar/sec from power levels of 1 and 25 MW, and (c) partial and total losses of flow from power levels of 10 and 28 MW. The majority of the tests were conducted from initial system conditions of 500°F core inlet coolant temperature, 2500-psig coolant pressure, and about 20-ft/sec average core coolant flow rate. The initial system conditions for the tests were chosen to approximate those used by current power-producing pressurized water reactors. The reactivity insertion of 1 dollar used in the step-induced transients was selected (a) to assure core integrity during the test series (larger reactivity insertions could possibly result in a departure from nucleate boiling condition), (b) as the stated upper limit of applicability for the analytical model with which the test results would be compared, and (c) as a reasonable reactivity value that could result from certain operational hazards such as the cold water accident (1 dollar is approximately equivalent to the reactivity perturbation that would result from a step decrease of 22°F in the core coolant-moderator temperature). The ramp reactivity insertion rate used, ≈ 0.05 dollar/sec, was also selected as typical of the insertion rates used in many power-producing reactors. A general description of the data obtained from these tests has been previously reported [1].

Prior to the performance of the test series, calculations were made using the ART-04 [2, 3] computer code to simulate the reactor response to the proposed perturbations. The results of these preliminary calculations were used as guides in the performance of the experimental investigations. In most instances, however, the experimental initial conditions were not the same as those used in the calculations. Therefore, similar calculations have been made during this quarter using the best-known values for the experimental initial system conditions as input to the computer program. The following discussions present the general results of the more-recent calculations.

2.2 General Description of Application of ART-04 Model

The ART-04 computer program is an analytical model designed by the Bettis Atomic Power Laboratory to calculate reactor transient behavior for plate-type pressurized water reactors. The program permits variation of the input parameters of reactivity, temperature, flow, and pressure. For the Spert III C-19/52 core, the program was set up as a single-pass model using one hot channel and three nominal channels to represent four radial regions of the core. The only difference in the method of treatment between the nominal and hot channels is in the manner in which the pressure drop across each channel is determined [2]. A plate-type fuel element geometry was assumed, with each channel divided into 15 axial regions. The treatment of fluid flow and convective heat transfer in each channel was based on a one-dimensional transient model in which variations in fluid velocity and fluid properties in directions transverse to the flow path were neglected. Core power as a function of time was determined by reactor kinetics calculations in which the feedback reactivity was determined using localized density coefficients and calculated bulk water density changes. Where applicable, experimentally obtained core properties were used in the analytical model.

Any assessment of the effectiveness of the ART-04 code in calculating transient behavior - as derived from the following comparisons of calculated and experimental results - should be tempered with the understanding that this model was primarily designed to calculate reactor behavior for subprompt-critical transients while the calculations performed for the step transients of this series used reactivity insertions of $\approx$ one dollar, the upper limit for reactivity insertions recommended by the designers of the model. In view of these facts, the good agreement found between calculated and experimental results in the transient region where conditions are changing most rapidly, ie, the initial peak power region, may be somewhat fortuitous. A more valid assessment of the effectiveness of the model can be expected from comparisons taken from the quasi-equilibrium region of the transients following the initial peak power.

2.3 Results from Stepwise Reactivity Insertions

For the tests initiated by 1-dollar stepwise insertions of reactivity, generally good agreement was obtained between the experimental and calculated reactor power behavior. Overall, the calculated peak power values were within 15 percent of the experimentally measured values and the run-out powers (defined for these tests as the power level 1.6 sec after the start of the reactivity insertion) agreed to within 35 percent.

A comparison between the calculated and measured reactor response to a 1-dollar stepwise reactivity insertion from an initial reactor power level of 28 MW can be seen in Figure 1. In the figure the calculated and measured reactor power response is shown as a function of time after the start of the reactivity insertion. As can be seen in the figure, the calculated value for the initial peak power is in good agreement with that which was measured for this test. The calculated run-out power was approximately 11 percent below that measured. This particular test yielded the best agreement between the experimental and calculated reactor behavior. Calculated and measured peak and run-out power levels for other step-initiated tests are presented in Figure 2 as a function of the reactor power level from which the respective 1-dollar reactivity perturbations were made. The calculated peak power values varied both above and below those measured over the range of initial powers for which tests were conducted; the calculated run-out powers, however, were consistently lower than the experimental measurements. The nature of the difference between the calculated and experimental run-out power levels suggests the existence of

Fig. 1 Experimental and calculated reactor power and nominal-channel coolant outlet temperature for Spert III C-19/52 core 1\$ stepwise reactivity insertion; initial system conditions: 28 MW, 500°F, 2500 psig, and 20 ft/sec coolant flow.

Fig. 2 Experimental and calculated initial peak power, run-out power, and nominal-channel coolant outlet run-out temperature rise versus initial power level for Spert III C-19/52 core 1\$ stepwise reactivity insertions; initial system conditions: 500°F, 2500 psig, and 20 ft/sec coolant flow.

some consistent error or combination of errors either in the experimental results, the experimental data used as input data to the model, or in the model itself. The experimental power is known to an accuracy of ± 10 percent, and two likely sources of error in the model input data are the void coefficient of reactivity (known within ± 12 percent) and the core coolant flow-rate (known to ± 10 percent). Considering these facts it is apparent that the degree of precision with which the inherent inaccuracies of the model's analytical approximations can be evaluated is limited primarily by the uncertainties in basic plant parameters such as flow, power, and void reactivity.

The nominal channel coolant outlet temperature can be considered as representative of the gross average core coolant temperature behavior. There-

fore, the coolant outlet temperature was measured by use of a thermocouple located approximately one inch above the plate section in a nominal fuel assembly. A time-dependent comparison of the calculated nominal channel outlet temperature with that measured by the thermocouple is shown in Figure 1 for the test initiated by a 1-dollar insertion at a power level of 28 MW. Figure 2 presents a comparison of the experimental and calculated nominal channel coolant outlet temperatures as a function of the initial reactor power level for a time 1.6 sec after the start of the reactivity perturbation. For all of the tests, the calculated coolant outlet temperature increases were between 15 and 30 percent below those measured. It should be noted that the ART program calculates temperature values for local average conditions, while the measured temperatures discussed here are made in a pointwise manner. Therefore, the measurements and calculations are not directly comparable and agreement between the experimental and calculated temperature values should not necessarily be expected.

2.4 Results from Ramp Reactivity Insertions

Tests initiated by ramp reactivity insertions of approximately 0.05 dollar/sec were conducted from initial steady-state power levels of 1 and 25 MW. These moderate power transients were conducted to provide data for use in establishing the applicability of the ART program in calculating the reactor response to a particular type of reactor operator error, ie, a short term continuous rod withdrawal with the reactor initially operating at power. The operator error was simulated by withdrawing control rods continuously for 5 sec (equivalent to about a 0.3-dollar total reactivity insertion) then leaving them fixed and allowing the power to rise toward an equilibrium level. Also, as part of the series of ramp-initiated tests, a test was performed to determine the ability of the ART program to calculate the reactor response to a series of control rod manipulations. In this test, with the reactor initially at a power level of 1 MW, the control rods were withdrawn for 5 sec, left motionless for 3 sec, then reinserted for 3 sec. The reactor power was then allowed to seek an equilibrium level.

Because of the relatively low reactivity insertion rates used during these tests, the reactor power did not display initial peaks but monotonically increased or decreased during the periods of control rod withdrawal or insertion. The experimental and calculated results for these tests were in about the same generally good agreement that was obtained for the run-out powers and temperatures during the step-induced tests. The calculated powers and temperature increases were within 25 percent of the measured values and were generally lower than the measured values.

2.5 Loss-of-Flow Results

Comparisons have been made between the calculated and experimental reactor behavior for partial and total loss-of-flow tests initiated from steady-state reactor power levels of 10 and 28 MW with no automatic low-flow-scrum protective devices in operation. For each test the initial coolant flow rate

through the core was about 20 ft/sec ($\approx 22,000$ gpm). The calculated reactor response to the loss-of-flow perturbations was obtained from the ART-04 computer program using experimental flow coastdown characteristics of the Spert III system as part of the input data.

The comparative calculated and measured behavior of reactor power level and nominal channel coolant outlet temperature for the most extreme loss-of-flow test, the total loss of flow from 28 MW, are shown in Figure 3. The variables are plotted as a function of time after the electrical power to the coolant pumps was cut off. For this test the calculated reactor power level was consistently lower than that which was measured; 11 percent lower at 1 sec after the loss of flow and about 26 percent lower after 5 sec. The calculated nominal channel coolant outlet temperature, however, was in good agreement with the measured temperature.

Fig. 3 Experimental and calculated fractional reactor power and nominal-channel coolant outlet temperature during total loss-of-flow test; initial system conditions: 28 MW, 500°F, 2500 psig, and 20 ft/sec coolant flow.

The apparent time-dependence of the differences between the experimental and calculated values for power and temperature is believed to have its origin in errors existing in the experimental flow-coastdown data supplied as input to the calculational model. This data was derived from plant coolant loop differential pressure instrumentation which was not designed for use in transient measurements. However, the generally good agreement obtained between the calculations and the experiment indicates that such plant instrumentation may be adequate for measurements of this type.

2.6 Conclusions

For the tests performed, comparisons of calculated with experimental reactor response indicate that the ART-04 computer program is a useful analytical tool for predicting reactor transient behavior of plate-type, pressurized-water reactors similar to the Spert III C-19/52 core. The range of deviation of calculated values of power and temperature from those measured is relatively small and is generally within the combined uncertainties existing in the experimental values of the void coefficients, coolant flow rate, and reactivity insertions used as input data to the computer program. Therefore, on the basis of these results, it is concluded that the ART-04 program can be used to calculate similar transient behavior (eg, for use in hazards analyses) for cores similar in design to the C-19/52.

3. SPERT III THERMOCOUPLE RESPONSE CORRECTIONS

Fuel plate surface temperature measurements are of importance in reactor kinetics testing since the more important reactivity-compensating mechanisms (moderator expansion, void formation, etc) in highly enriched, water-moderated systems depend on energy transfer from the fuel plate surfaces to the moderator. From the temperature data such information as the transient energy distribution, heat transfer rates to the moderator, and the onset and nature (nucleate, film, etc) of moderator boiling can be deduced. In addition, extrapolations of temperature measurements from the core "hot-spots" provide information as to the maximum reactivity insertion which a core can withstand without encountering clad melting.

One of the problems in the experimental program with the Spert III C-19/52 core has been that of obtaining sufficiently accurate fuel plate surface temperature data during short-period power excursion tests performed with high coolant flow rates and elevated initial temperature conditions. This problem arises because thermocouples which have the mechanical strength to withstand the environmental conditions do not have adequate response characteristics to follow the rapid fuel plate surface temperature increases which occur during such tests. In order to obtain measurements of fuel plate surface temperatures during power excursion tests performed with high coolant flow rates and elevated initial temperatures, stainless-steel-sheathed, MgO-insulated, 3-mil, chromel-alumel wire thermocouples have been used in the Spert III reactor. The response of these thermocouples, however, has been shown to lag considerably behind that of the fast-response, 0.5-mil, chromel-alumel thermocouples normally used at Spert in less severe environments [4]. Consequently, a study was undertaken to develop a technique for obtaining improved temperature data from the sheathed thermocouples.

In the study, stainless-steel-sheathed thermocouples were paired with 0.5-mil thermocouples at equivalent-flux positions in the Spert III C-19/52 core, and comparative fuel plate surface temperature data were obtained during power excursion tests with initial asymptotic periods ranging from approximately 88 to 19 msec. The tests were initiated at ambient temperature, with no forced coolant circulation through the core, and with the system pressure at either atmospheric or 2500 psig.

The temperature data from the thermocouples in these tests were used in a frequency response analysis in order to obtain a time-varying formula for correcting the stainless-steel-sheathed temperature data. In the analysis it was assumed that the 0.5-mil thermocouples indicated the correct fuel plate surface temperatures as a function of time. The FREQUENCY RESPONSE program [5] was used to obtain transfer functions for the error in the stainless-steel-sheathed thermocouples. These transfer functions were of the form

$$H(s) = \frac{T_{0.5}(s) - T_{ss}(s)}{T_{ss}(s)} \quad (1)$$

where $T_{0.5}(s)$ and $T_{ss}(s)$ are the Laplace transforms of the smoothed traces of the 0.5-mil and stainless-steel-sheathed thermocouples, respectively. Twenty-one transfer functions of the form given by Equation (1) were obtained from the thermocouples which had been installed in the core for the tests.

A previous analysis [6] of this type had provided good results in correcting temperature data obtained with 5-mil, chromel-alumel thermocouples. Unlike the transfer functions of the 5-mil thermocouples, however, the transfer functions of the stainless-steel-sheathed thermocouples exhibited an erratic behavior. For example, the Nyquist plots for different thermocouple pairs during the same transient and for the same pair during two successive transients showed little similarity to each other. Figure 4 shows this behavior in the Nyquist plots for a given pair of thermocouples on two successive transients with periods of 19.6 and 20.7 msec performed under the same initial system conditions. In a later series of transients, 2-mil brass shim-stock shields were fitted over the spaded tips of the sheathed thermocouples in an attempt to improve their response during coolant flow tests. However, the shields did not appear to help the response characteristics. The erratic behavior of the transfer functions is still seen in the Nyquist plots as shown in Figure 5.

Because of the erratic behavior of the stainless-steel-sheathed thermocouple transfer functions, no single approximation of $H(s)$ could be found which produced a good fit for all cases. A rational function approximation to $H(s)$ was obtained, however, that yielded a correction formula which generally agreed to within 10 percent of the 0.5-mil thermocouple temperatures at peak temperature. This approximation is of the form

$$Q(s) = \frac{s(s + 7.0)(s + 123.4)}{(s + 3.0)(s + 60.0)^2}. \quad (2)$$

Fig. 4 Nyquist plots for the error in Spert III stainless-steel-sheathed thermocouple for successive 20.7- and 19.6-msec-period tests.

Fig. 5 Nyquist plots for the error in Spert III brass-shielded, stainless-steel-sheathed thermocouple for successive 19.8- and 19.0-msec-period tests.

Use of this approximation in Equation (1) yields, in the time domain (3)

$$T_{0.5}(t) = 2.0T_{ss}(t) + 7.429 \int_0^t [0.599e^{-3\tau} + (476.33\tau + 1.06)e^{-7\tau}] T_{ss}(t-\tau) d\tau.$$

A program written for the IBM-7040 to evaluate Equation (3) was used to "correct" the sheathed thermocouple data. Results using this correction are shown in Figures 6 and 7. In Figure 6 it can be seen that the corrected trace is in good agreement with the temperature measured with the 0.5-mil thermocouple. In Figure 7, however, the correction is about 15 percent higher than the 0.5-mil thermocouple temperature at the time of the peak temperature (the original sheathed thermocouple trace is approximately 25 percent too low).

In 19 out of the 21 cases, this correction technique resulted in temperatures which were within 10 percent of the 0.5-mil thermocouple temperatures for temperatures near the maximum. For lower temperatures, during the initial rise, no specific conclusion can be made concerning the accuracy of the correction; however, in all cases the corrected temperatures were much closer to the 0.5-mil thermocouple temperatures than were the uncorrected sheathed-thermocouple temperatures. For the two cases in which the corrected temperatures were not within 10 percent of the 0.5-mil thermocouple temperatures (see Figure 7 for one case), the fuel plate surface temperature exhibited a sharp initial peak.

Fig. 6 Temperature rise versus time for Spert III 20.7-msec-period test showing 0.5-mil, stainless-steel-sheathed, and corrected stainless-steel-sheathed thermocouple traces.

Fig. 7 Temperature rise versus time for Spert III 20.8-msec-period test showing 0.5-mil, stainless-steel-sheathed, and corrected stainless-steel-sheathed thermocouple traces.

From the frequency response plot of the three-pole, three-zero approximation of $T_{ss}(s)/T_{0.5}(s)$, shown in Figure 8, the cutoff frequency of the stainless-steel-sheathed thermocouples is approximately 17 radians/sec. This compares with the cutoff frequency of approximately 140 radians/sec previously determined [6] for the 5-mil, chromel-alumel thermocouples.

From the results of this analysis it is concluded that the use of the stainless-steel-sheathed thermocouples has not completely solved the problem of the acquisition of accurate fuel plate surface temperature data during short-period transients performed under typical pressurized water reactor environmental conditions. The analysis has shown, however, that reasonably accurate values of the maximum temperatures attained can be obtained by use of the correction technique which has been developed.

Fig. 8 Frequency response plot of stainless-steel-sheathed thermocouples.

4. SPERT III PRIMARY COOLANT PUMP FAILURE

The Spert III facility was designed to permit the study of the kinetic behavior of reactor cores under forced coolant circulation conditions at rates up to 20,000 gpm through the core. To provide this flow rate, four, 500-hp, 5000-gpm, liner-motor primary coolant pumps were installed when the facility was constructed in 1958 [7]. These pumps have provided trouble-free service for seven years of intermittent operation, during which time they have been subjected to frequent starting, system pressures up to 2550 psig, and system temperatures up to 650°F.

During cooldown of the primary system following completion of the last series of tests with the C-19/52 core, however, the Number 2 primary pump stopped and could not be restarted. On inspection, primary system water was found leaking through the seals of the stator-winding terminal posts of the pump which indicated that there was leakage through the stator liner. Subsequently, the pump motor assembly was removed and sent to the manufacturer for disassembly, inspection, and assessment of damage. The rotor can, stator liner, and stator were found to be damaged. The extent of this damage will necessitate the complete rebuilding of the motor section which will require approximately six to eight months for completion.

The damage to the rotor can, shown in Figure 9, consisted of a band approximately two inches wide where the can was completely nonexistent. Heavy deposits of corrosion products were found on the rotor and some swelling of the can was observed. Figure 10 shows the stator liner following its removal from the stator assembly. The damaged area is opposite that of the

Fig. 9 Photograph showing damage to rotor can of Spert III primary pump.

Fig. 10 Photograph showing damage to stator liner of Spert III primary pump.

rotor can and consisted of several small holes worn through the liner plus one large ragged hole. Also, during the inspection, several cavities, one-half inch and smaller, were found where the nitrided surface of the thrust-bearing disk had flaked off.

Although no positive cause for failure was determined, it is the opinion of the manufacturer that the pump had operated for some time with a damaged rotor can. The manufacturer further postulates that initial damage to the can could have been caused by either a foreign particle which entered the motor from the primary system, or by a piece of the nitrided surface from the thrust disk. The largest particle which could enter the region between the stator liner and the rotor can from the primary system would be approximately 10 mils in diameter. However, a particle from the primary system could reach this region of the pump only by a tortuous path. Therefore, it was concluded that one of the pieces from the thrust-disk surface probably initiated the damage. With continued use, the rotor can wore several small holes in the stator liner which allowed water to enter the stator, resulting in a short circuit in the windings. The ragged hole in the liner, shown in Figure 10, is a result of this short circuit.

The other three pumps are currently being disassembled and inspected for worn or damaged parts. No damage to either the rotor cans or stator liners has been found in the two pumps which have been inspected thus far. However, the thrust-bearing disks of both pumps are in a condition similar to that of the damaged Number 2 pump. Therefore, new thrust-bearing disks will be installed in all of the pumps during reassembly.

II. SPERT IV

1. INTRODUCTION

A portion of the subassembly test program which will be conducted with the Capsule Driver Core (CDC) in Spert IV includes the planned destruction of reactor-fuel test samples. In order to obtain information on containment considerations for such tests, a detailed out-of-pile investigation of capsule materials and designs has been initiated. The objectives of this investigation are twofold: (a) to design and proof-test a capsule to be used during in-pile subassembly tests of the CDC fuel. These subassembly tests, designated "lead-rod" experiments [1], will include determinations of the response characteristics and operational limits of the CDC, and the destructive thresholds of the CDC fuel; and (b) to obtain information on the response of various capsule materials to explosive-loading conditions for use in the design of capsules for future portions of the Spert subassembly testing program.

The first of these objectives has been attained. A lead-rod capsule has been proof tested and will be ready for service in the CDC in the near future. Three categories of tests were performed in the development of the lead-rod capsule: (a) tests were conducted with cylinders to evaluate the explosive response of various capsule materials; (b) tests were performed with sealed capsules to determine the effects of containing explosions within closed systems; and (c) tests were performed to investigate the water-hammer effect which resulted from the explosions within the closed capsules and caused the capsules to be raised from their initial test positions. An investigation of this effect was made in order to obtain data necessary for the design of a capsule-constraining (hold-down) mechanism.

Information pertinent to the second objective is currently being obtained. Since the responses of most materials to rapidly applied loads is markedly different from corresponding responses to slowly applied loads, this change in material responses must be included in the development of capsule design criteria.

The success of the detailed investigation of capsule materials is dependent upon obtaining reliable data regarding the strain responses of the capsules to the explosive pressures to which they are subjected. Therefore, in conjunction with the investigations of capsule materials and designs, a program also is being conducted to develop instrumentation techniques to be used in the extreme environmental conditions generated in explosive tests.

2. OUT-OF-PILE CAPSULE TESTS

2.1 General Considerations For Tests

Among the factors needed to be considered in the design of the CDC lead-rod capsule are: (a) physical size, (b) containment of pressure or mechanical

energy generation, (c) the figure-of-merit ^[a], and (d) provisions for instrumentation. A discussion of the relationship and importance of these factors as applied to capsule design was presented in the last Spert quarterly report [1]. With regard to containment, it was pointed out that due to the unknown magnitude and transitory nature of pressure pulses which will be generated in reactor-fuel experiments conducted in the CDC, specification of only the static pressure capacity of the capsule is not a sufficient criterion for ensuring the adequacy of the capsule under test conditions. It was, therefore, decided that the basic capsule requirement for containment would be the ability to withstand up to 10 percent of the energy generated within the capsule during a 2-msec-period test in the form of mechanical energy produced by a high explosive. Using a reasonable value for the figure of merit of the system (≈ 2.1), the total energy release within the capsule during a 2-msec-period lead-rod test would be about 0.9 MW-sec and the peak energy density would be 1640 W-sec/g UO₂.

The main problems in the design of a capsule using a criterion based on the energy released within the capsule during short-period tests are: (a) the unknown conversion efficiency from nuclear energy generated in the test fuel to strain energy in the capsule wall, and (b) the unknown response of the capsule to rapidly applied loading conditions.

Regarding the first problem, data from Spert [8] and SNAPTRAN [9] experiments indicate that a conversion efficiency from nuclear to mechanical energy of 10 percent represents a reasonable upper limit for the range of testing of CDC fuel which is proposed in the lead-rod experiments. However, no similar data are available for the geometry-dependent conversion efficiency from mechanical to strain energy in the capsule. Secondly, the response of the capsule to the applied loading conditions is a function of the strain rate to which the capsule is subjected. For high-speed impact, this strain-rate dependency can become a critical factor in the deformation response of some materials [10, 11]. At the present time, knowledge of this subject is insufficient to permit a meaningful theoretical solution to the problem. It was, therefore, decided to initiate an explosives-testing program in which selected types of cylinders would be subjected to more severe loading conditions than would be expected from comparable energy releases of nuclear origin. The results of this program will be correlated using a calculational model based on the static mechanical properties of the test materials, and attempts will be made to include the strain-rate effect on the material response in the correlations.

In the experiments for which the CDC lead-rod capsule is being designed, a 30-inch section of CDC fuel will be subjected to transient power bursts. Previous experience indicates that some modes of fuel rod failure could result in full-length longitudinal rips in the fuel rod cladding [8]. Such a mode of failure, resulting in expulsion of hot UO₂ into the water annulus surrounding the rod, could very likely have the appearance of a linear explosive source. Therefore, this type of explosion was mocked up using linear charges of chemical

[a] Figure-of-merit is defined as the ratio of the average radial power density at the axial power peak in the test fuel to the maximum power density in the core fuel.

explosives in the testing program. The explosive being used in this testing program is Primacord [a] which has a burning rate of about 21,000 ft/sec, a heat of explosion of about 1385 cal/g, and which releases 100 percent gaseous products. In work completed elsewhere [12] the lowest suggested value for the conversion efficiency from explosive heat to mechanical energy for PETN was 38 percent. For conservatism, however, a nominal conversion efficiency of 30 percent was assumed in this investigation.

2.2 Explosive Tests With Cylinders

A series of explosive tests using cylinders was conducted to investigate the response of various materials to explosive loadings and to select a suitable material for the lead-rod capsule. All tests were performed in water using three-foot-long cylinders. The charges of Primacord were positioned along the centerlines of the cylinders and extended through the cylinders. The materials of primary interest in these tests were 6061-T6 aluminum, 304 stainless steel, and A106 Grade B carbon steel. Carbon steel was not considered for use as a capsule material but was investigated for possible use in model studies of stainless steel vessels. For each material, a series of tests was performed, increasing the charge loading in the Primacord for each successive test, until failure of the cylinders was obtained.

Examples of the types of failures observed in carbon steel, aluminum, and stainless steel are shown in Figures 11 through 14. It is interesting to note that all of the carbon steel cylinders failed in the tensile mode with little attendant permanent deformation. All failures observed in aluminum and stainless steel cylinders, however, occurred in the shear mode at somewhat larger deformations. Because of this difference in the strain response and mode of failure between the carbon steel and stainless steel cylinders, it was concluded that carbon steel would not be appropriate for use in the model studies of stainless steel vessels.

Comparisons of the results from failure of static loading and Primacord loading are given for the three materials in Table I. The static loading values for the various materials were obtained from their respective manufacturers.

The results given in Table I indicate that, with the possible exception of aluminum, the explosive pressures to which the cylinders were subjected had sufficiently short rise times ($\approx 10 \mu\text{sec}$) and durations (100 to 300 μsec) to invalidate the use of static mechanical properties of the materials in the analysis of cylinder responses to the applied loads. The effect of the rates of strain to which the cylinders were subjected is a probable explanation of this type of material response.

The figure-of-merit of the system, and hence, the total nuclear energy release within the capsule, is a function of the capsule wall thickness and material. Calculations which were made [1] to determine the optimum capsule wall thickness as a function of energy release in the capsule showed that the

[a] Trade name for detonating cord manufactured by the Ensign Bickford Company, which contains an explosive core of pentaerythritol tetranitrate (PETN). When used in this report, the unit gr/ft means grains of PETN/ft of Primacord.

Fig. 11 Tensile failure of 2-inch, Schedule 80, A106 grade B carbon steel cylinder. Total charge: 100 gr/ft.

Fig. 12 Shear failure of 2-inch, Schedule 80, 304 stainless steel cylinder. Total charge: 450 gr/ft.

Fig. 13 Shear failure of 2-inch, Schedule 80, 6061-T6 aluminum cylinder. Total charge: 200 gr/ft.

Fig. 14 Shear failure of 2-inch-OD x 0.065-inch thick 304 stainless steel cylinder. Total charge: 400 gr/ft.

TABLE I

COMPARISON OF STATIC AND DYNAMIC DEFORMATION RESPONSE OF CYLINDERS

Material	A106 Grade B Carbon Steel	6061-T6 Aluminum	304 Stain- less Steel	304 Stain- less Steel
Cylinder size	2 in., Sch. 80 pipe	2 in., Sch. 80 pipe	2 in., Sch. 80 pipe	2-in. OD x 0.065-in. thick tubing
Elongation as a result of static failure (%)	30	10 to 12	67	≈ 60 to 65
Elongation as a result of failure using Primacord (%)	4 to 6	9	≈ 15	≈ 30
Mode of failure	Tensile	Shear	Shear	Shear
Charge loading of Primacord which resulted in fail- ure (gr/ft)	175	200	450	400

two-inch, Schedule 80, 304 stainless steel pipe would have a figure-of-merit of about 2.1. When the previously stated energy criterion for the capsule design (10 percent of the energy released during a 2-msec-period test) is applied to this material, and using the assumed 30 percent conversion efficiency from explosive heat to mechanical energy, the pipe must be capable of withstanding the energy release from a 30-inch length of 400 gr/ft Primacord. The two-inch, Schedule 80, 304 stainless steel pipe did withstand explosive charges of 400 gr/ft, thereby demonstrating the suitability of this material for use in the initial lead-rod capsule. Since the aluminum cylinder failed at a charge loading of 200 gr/ft, this material was dropped from consideration for use in the lead-rod capsule.

2.3 Prototype Capsule Proof-Tests

From the results of the tests with cylinders, two-inch, Schedule 80 stainless steel pipe was selected as the basic material for the lead rod capsule. Therefore, a series of prototype proof-tests was performed with 5-1/2-foot-long sealed capsules made of this material. These capsules were sealed with welded plugs at the bottom and threaded caps at the top. Primacord charges 30 inches long were detonated in the bottom 30 inches of the capsules, with an air space of from 1 to 2-1/2 feet in the top of the capsules. All tests were conducted with the capsules immersed in water. Charges in the range from 40 to 400 gr/ft were detonated to investigate: (a) the response of the capsule material in the closed system, (b) the effect of the air space on the energy of motion imparted to the capsule, and (c) the maximum energy of motion imparted to the capsule at the design charge loading of 400 gr/ft. The energy of motion imparted to the capsule was determined by measuring the height to which an unconstrained

capsule of known weight was raised from its initial position by the Primacord explosions. The results of the capsule tests gave the following information:

- (1) Two-inch, Schedule 80, 304 stainless-steel capsules, 5-1/2 feet long, sealed with welded plugs at the bottom and threaded caps at the top, can contain the design energy charge of 400 gr/ft without failure. Therefore, the initial lead-rod capsule will be basically of this design, except that a bolted flange will be used at the top rather than the threaded cap. This change is being made to facilitate removal of instrumentation leads from the capsule.
- (2) The effect of the air space on the energy of motion was not significant for the air spaces considered.
- (3) The maximum energy imparted to the capsule in the tests did not exceed 300 ft-lb. This result defined the requirements of the hold-down mechanism needed to keep the capsule positioned in the test space, within the CDC. Such a mechanism is being designed to be added to the existing core and capsule support structure.

Currently, the lead-rod capsule is in the final design stage with consideration being given to nuclear-test instrumentation.

3. CAPSULE INSTRUMENTATION DEVELOPMENT

An essential part of the out-of-pile capsule testing program is the development of instrumentation and techniques which will provide measurements of capsule pressures (inside and outside of the capsule) and capsule strains in an explosive environment. Although it was found that pressure and strain instrumentation functioned reasonably well in the pressure field adjacent to low-yield Primacord explosions, serious problems were encountered when the charges exceeded 100 gr/ft. The following problem areas were defined by the results of the initial tests:

- (1) Strain gauges attached to the outer walls of the capsules failed when subjected to strains in excess of about 2,000 μ in./in.
- (2) Difficulties were encountered in trying to separate pressure signals from noise signals which were generated by the charge firing circuit and vibrations of the pressure transducers and lead-out cables attached to the capsule wall.
- (3) Damage to pressure transducers was incurred when charge loadings exceeded 100 gr/ft.

Because of these difficulties, a series of free-field tests in water was performed (a) to evaluate the response of various types of pressure transducers to the large-magnitude, fast-rise-time pressure pulses generated by Primacord explosions; and (b) to investigate the free-field characteristics of the resulting pressure waves, such as, peak-pressure dependence on charge loading, and peak-pressure attenuation with distance.

The initial series of free-field tests was performed using bonded-strain-gauge and tourmaline-crystal type pressure transducers. The data obtained from the bonded strain gauge pressure transducers contained high frequency oscillations, indicating that the rise time of the pressure source exceeded the response capabilities of the transducers. However, the response of the tourmaline crystal pressure transducers was well behaved, and this type of transducer was used in subsequent tests.

Figure 15 shows the response of a tourmaline crystal pressure transducer which was positioned six inches from a 28-inch length of 350 gr/ft Primacord. The resulting pressure had a peak magnitude of 16,000 psi and a duration of about 250 μ sec. It should be noted that the rise time of the recorded pressure (16 μ sec) is the limit of the signal conditioning and recording system.

To gain further insight into the variation of peak pressure with charge loading and distance from the pressure source, pressure transducers were located at distances of 1, 6, 12, and 18 inches from 28-inch-long explosive charges. These locations were on the perpendicular bisector of the charges. For tests which were conducted with charges exceeding 100 gr/ft, no measurements could be made at the one-inch position. The results of a series of tests using 28-inch charge lengths are summarized in Figure 16 which shows peak pressure as a function of charge loading and distance from the source.

The curves shown in the figure were fit to the data using the relation

$$p \propto w^x$$

Fig. 15 Pressure pulse recorded by a tourmaline-crystal pressure transducer positioned at a radial distance of 6 inches from a 28-inch-long charge of 350 gr/ft Primacord detonated in water.

Fig. 16 Free-field peak-pressure as a function of radial distance from 28-inch-long charges of Primacord in water.

where

p = peak pressure

w = charge loading

x = an empirically derived number.

The values of x varied from 0.50 at a distance of 18 inches to 0.64 at a distance of six inches from the pressure source. The uppermost curve, showing peak pressure as a function of charge at a distance of one inch, has been extrapolated using the data obtained from the six-inch position for charges exceeding 100 gr/ft.

It was found that in the region from 1 to 6 inches from the charges, peak pressures, p, were attenuated over distance, r, approximately according to the relation, $p \propto r^{1/2}$. For distances in excess of 12 inches, peak pressures were approximately proportional to the inverse of the distance from the pressure source.

Currently, a series of tests is being conducted in which flat plates of various thicknesses and materials are positioned between the explosive charges and the various strain and pressure transducers. The primary purpose for these tests is to investigate the effects of such variables as the charge strength, system geometry and materials, and the transducer type and position on the characteristics of the measured strains and pressures. The results of these tests will provide information regarding the relative importance of these factors and will assist in the instrumentation and analysis of tests in the Spert subassembly testing program.

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