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THE EFFECTS OF COOLANT TEMPERATURE AND INITIAL
POWER LEVEL ON THE EXCURSION BEHAVIOR OF
A HIGHLY ENRICHED PLATE CORE IN SPERT III
--EXPERIMENT AND ANALYSIS

by
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ABSTRACT

Presented is the experimental data obtained from self-limiting power excursion tests performed on a highly enriched, stainless steel clad, plate core in the Spert III facility. The tests were initiated by step-insertions of reactivity with the reactor at power levels ranging from low power (10 W) to high power (28 MW) and with various combinations of other plant parameters: pressure (atmospheric to 2500 psig), temperature (ambient to 600°F), and flow (0 to 20,000 gpm). The reactivity insertions ranged from ≈ 0.4 to $\approx 1.2\%$, resulting in initial asymptotic reactor periods ranging from ≈ 1000 to ≈ 10 msec. The experimental data are correlated to show the effects of system temperature and initial power level on gross kinetic behavior. ART-04, a reactivity feedback model programmed for the computer to predict reactor transient behavior under operating conditions, is evaluated by comparing calculated and experimental transient data. The agreement between calculation and experiment is within the uncertainties associated with the data over the range of conditions considered.

SUMMARY

This report presents an analysis of experimental data obtained from power excursion tests performed with the Spert III, stainless steel clad, C-19/52 plate core to investigate the effects of initial system temperature and initial power level on excursion behavior. The experimental program consisted of self-limiting power excursion tests initiated under conditions that typify a PWR at hot and cold standby and at appreciable operating power. The reactivity additions ranged from ≈ 0.4 to $\approx 1.2\%$, resulting in initial reactor periods ranging from ≈ 1000 to ≈ 10 msec. The system parameters investigated were initial reactor power (10 W to 28 MW), system pressure (atmospheric to 2500 psig), system temperature (ambient to 600°F), and coolant-moderator flow rate (0 to 20,000 gpm). A major part of the analysis was devoted to an evaluation of the ART-04 computer code from the standpoint of reactor safety calculations. The experimental data were also subjected to a correlative analysis to provide information applicable to other similar reactor systems in at least an empirical manner.

The ART-04 computer code used the point kinetics model in conjunction with a temperature- and space-dependent adjunct feedback equation to predict the overall response of the reactor system in terms of hot-channel and nominal-channel behavior. The applicability of the code was assessed by comparison of its calculated results with similar experimental data. For the comparisons made, the code calculated overall reactor dynamic responses that were generally within the experimental uncertainties of the data. The results of this experimental-analytical program indicate that the ART-04 code is capable of reasonable predictions of the dynamic response of a reactor to reactivity perturbations for a large and important class of startup and operating condition accidents.

For the range of tests performed, correlation of the experimental data established several important points: (a) the system subcooling had no appreciable affect on the system response per se, the important parameter being the initial system temperature; (b) the overall severity of a power excursion initiated from low power was reduced as the initial system temperature was increased; and (c) the overall severity of a transient resulting from a given reactivity perturbation was increased as the initial reactor power was increased.

The results of the Spert III reactor safety studies program with the plate-type C-core confirm that, for this type of reactor, the kinetic response of the system is primarily moderator controlled. The comparison of analytical methods and experimental results also confirms that models based on a consistent physical picture of moderator effects provide a reasonable prediction of nondestructive excursion behavior of water-moderated and -cooled, plate-type reactors. Therefore, these models, with the experimental confirmation provided by the tests, can now be employed more usefully and reliably to assess the safety of other reactor systems.

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I. INTRODUCTION

This report presents an analysis of the data obtained from an experimental kinetics program carried out in the Spert III highly enriched, stainless steel clad, plate-core, pressurized-water reactor (PWR) to investigate the effects of system temperature and initial power level on reactor excursion behavior. The response of the system was studied for step, ramp, and loss-of-coolant-flow reactivity perturbations over a wide range of initial system and operating conditions. The major objectives of the experimental program and associated analysis were (a) to determine the gross effects on dynamic response of variations in initial system temperature up to 600°F and variations in initial reactor power level up to 28 MW, (b) to establish, in at least a qualitative manner, how the parameters of system temperature and initial power level affect the ability of the system to safely withstand sudden and sizeable changes in reactivity, and (c) to evaluate an existing analytical model by direct comparison of calculated and experimental transient data and, thus, determine the effectiveness of the model in calculating excursion behavior. Determination of the system condition effects upon the reactor kinetic response allows confirmation that, for a plate core, the kinetic response of the system is primarily controlled by moderator effects. Comparison of calculated results and experimental data enables confirmation that models based on a consistent physical picture of moderator effects will provide a reasonable prediction of the excursion behavior of water-moderated and -cooled, plate-type reactors. Experimental confirmation of the model provides a firmer basis for its acceptance and use in assessing the safety of other reactor systems.

The Spert reactor safety program has been concentrated primarily on identifying and understanding phenomena important to the excursion behavior of water-moderated reactor systems. The Spert III facility provides the capability for conducting kinetics experiments over a wide range of system temperature, pressure, and coolant flow rate conditions and with initial reactor power levels sufficient to produce power densities approaching those encountered in power-producing, pressurized-water reactors. Previously reported [1] tests carried out in Spert III investigated the sensitivity of excursion behavior to primary system coolant pressure (atmospheric to 2500 psig) and coolant flow rate (0 to 20,000 gpm). The presently reported work extends these investigations to include the effects of system temperature and initial power level to provide additional insight into the safety problems requiring consideration in the design, siting, and operation of reactor power plants.

A brief review of the results from previous Spert III excursion tests is given in Section II; included are pertinent fiducial test data and results from tests performed to investigate the effects of system pressure and coolant flow rate on excursion behavior. Section III presents a correlative analysis of the transient power and temperature data resulting from step insertions of reactivity from various initial system temperatures (ambient to 600°F) and initial power levels (10 W to 28 MW). The salient features of the data are discussed to illustrate the changes in dynamic response commensurate with changes in initial operating conditions. In Section IV, the ART-04 computer code [2,3], developed by Bettis Atomic Power Laboratories to predict the transient response of plate cores, is evaluated by comparing calculated transient variables (such as, power, energy, reactivity, and coolant temperatures) with measured values. Comparisons are made for tests initiated by step, ramp, and loss-of-coolant flow reactivity pertur-

bations over a wide range of initial system and reactor power conditions. Section V summarizes the conclusions derived from the analyses with respect to the program objectives.

Four appendixes provide detailed information to support and supplement the text. Appendix A presents a brief description of the Spert III plant, core, and test instrumentation in addition to a facility design data summary. Appendix B summarizes the experimentally measured static characteristics of the C-19/52 core. Appendixes C and D summarize the measured transient data and test conditions in both tabular and analog form.

II. PREVIOUS TEST RESULTS

1. FIDUCIAL AND PRESSURE TESTS

This section presents a brief resumé of the fiducial transient effects of system pressure and flow at ambient system temperature. A more detailed discussion of these effects is given in IDO-16918 [1].

The standard practice at Spert is to begin testing a new core with a series of fiducial tests performed at ambient temperature, atmospheric pressure, and with no forced coolant flow. In addition to experimentally examining the startup accident, these tests allow establishment of the inherent kinetic characteristics of the reactor and provide base-point data for comparison with later tests. Furthermore, these tests are used to set preliminary testing limits in order to avoid extensive damage to the core. Fiducial testing begins with very long-period transient tests and proceeds to shorter and shorter periods until the core begins to show or is predicted to sustain slight damage. The first such series of fiducial tests in Spert III was performed during 1960 and 1961 [4,5]; several similar tests were also performed during 1964 to obtain data for comparison with previous tests and to investigate the response characteristics of stainless steel clad thermocouples [6].

Tests conducted at elevated system pressure are discussed together with the fiducial tests since the effects of pressure are of small magnitude and only manifest themselves clearly when contrasted with atmospheric pressure tests. The data referred to as new data in the figures of this section were obtained during the recent series of tests (1964); whereas, the data referred to as previous data were obtained during the original series conducted in 1960 and 1961.

Figure 1 is a plot of peak power versus reciprocal period for ambient temperature tests at the two pressure extremes (atmospheric and 2500 psig) investigated in the Spert III C-19/52 core. These data illustrate the weak dependence of peak power on initial system pressure; the peak power attained in high-pressure tests is only slightly higher than that for corresponding atmospheric pressure tests. A more graphic illustration of the differences in these two types of tests is presented in Figure 2 wherein the energy release to time of peak power is plotted versus reciprocal (or inverse) period. For short-period tests, $\alpha_0 \geq 25 \text{ sec}^{-1}$, the data show that less energy release is required to quench the initial power burst for tests at atmospheric pressure than for tests at 2500 psig. The lower energy requirement of the short-period atmospheric pressure tests results from the onset of boiling steam formation as a shutdown mechanism; a more efficient mechanism in terms of reactivity compensated per unit energy release than the nonboiling mechanisms of fuel plate and moderator expansion. At 2500 psig, boiling is suppressed as a shutdown mechanism over the entire range of reactor periods investigated. The additional energy required to activate the nonboiling mechanisms for short-period, high-pressure tests is attained by (a) slightly higher peak powers, illustrated in Figure 1, and (b) slightly broader power burst shapes.

Shown in Figure 3 is peak power as a function of initial system pressure for several short period tests ($11.3 \text{ msec} > \tau > 10.6 \text{ msec}$). These data show that the

Fig. 1 Peak power versus inverse period.

major effect of pressure on peak power occurs in the range from 0 to about 100 psig; no additional pressurization is needed to suppress boiling prior to time of peak power for tests in this period range. For shorter period tests, this pressure threshold (≈ 100 psig) would be expected to increase.

Fig. 2 Energy release to time of peak power versus inverse period.

Fig. 3 Peak power versus system pressure.

2. FLOW TESTS

The tests referred to as flow tests were initiated by step reactivity insertions from low reactor power (≈ 10 W), ambient temperature, and with coolant flow as a parameter [7]. The coolant flow rate was varied from zero to about 18 ft/sec through the core at various pressures from about 200 to 2500 psig.

The principal effect of flow rate on the kinetic response of the C-19/52 core was that of altering the transfer of generated energy in the core. This effect manifested itself by reducing the reactivity compensation resulting from water and plate expansion. During long period tests (>150 msec), illustrated in Figure 4, core coolant flow of ≈ 2 ft/sec was sufficient to eliminate the initial power burst entirely. For shorter period tests (<25 msec), shown in Figure 5, the initial power burst was essentially unaffected by flow rate; however, the post-burst power level was increased in proportion to the flow rate. Since flow enhances the energy removal rate from the fuel plates, the observed fuel plate surface temperatures (also shown in above figures) were considerably lower than for corresponding no-flow tests.

Fig. 4 Power and plate surface temperature versus time as a function of flow rate at long periods.

Fig. 5 Power and plate surface temperature versus time as a function of flow rate at short periods.

III. SYSTEM TEMPERATURE AND INITIAL POWER LEVEL EFFECTS -- DATA CORRELATIONS

A primary objective of the Spert III reactor safety studies program is the determination of system condition effects upon the course and consequences of reactivity accidents in power reactor systems. Determination of these system condition effects upon the reactor kinetic response will provide confirmation that, for a plate core, the kinetic response is primarily controlled by moderator effects and can be adequately predicted by a physical model based on moderator effects. The study of system condition effects on reactor kinetic response has been accomplished by conducting power excursion tests over a wide range of operating conditions up to and exceeding those of current power reactor designs. The resultant test data and associated analyses provide a valuable supplement to nuclear safety technology and also a unique source of information to reactor designers and operators.

This section presents a correlative analysis of the experimental data obtained from tests designed to investigate the effects of initial system temperature and initial power level on kinetic behavior. The tests were conducted with initial system temperatures from ambient to 600°F (317°C), pressures from atmospheric to 2500 psig, coolant flow rates from 0 to 18 ft/sec velocity upward through the active core, and steady-state power levels from 10 W to 28 MW. Step reactivity insertions ranged from about 0.4 to 1.2\$, corresponding to reactor periods of ≈ 1000 to ≈ 10 msec.

The tests analyzed first are those initiated from low power, with system temperature as the variable parameter; both no-flow tests with system temperatures up to 400°F and flow tests with system temperatures up to 600°F are discussed. The correlative analysis of initial power level effects on kinetic behavior of the reactors for both constant and variable coolant flow rates is presented next. Slight variations in terminology are explained when necessary; details relating to the facility, core, design, static characteristics, and experimental data are given in the appendixes.

1. SYSTEM TEMPERATURE EFFECTS

1.1 No-Flow Tests

Correlated in this section are the data obtained from step reactivity insertion tests performed at initial system temperatures up to about 400°F (204°C), pressures up to 2500 psig, and no forced coolant flow. These system conditions provided a variation in subcooling from about 10 to 400°F and an overall change in moderator density of about 14 percent. Tests with three nominal reactor periods are considered in the analysis: (a) 20 msec (≈ 1.1 \$), for which moderator boiling occurred prior to the time of peak power at ambient temperature, atmospheric pressure; (b) 40 msec (≈ 1.05 \$), for which moderator boiling occurred at about the time of peak power at ambient temperature, atmospheric pressure; and (c) 150 msec (≈ 0.95 \$), for which no moderator boiling occurred at ambient temperature, atmospheric pressure.

In Figure 6, peak power is shown as a function of system temperature for several initial pressures. The system pressure is indicated only for the 20-msec tests since pressure had no measurable effect on the 40- and 150-msec tests. The 40- and 150-msec data clearly show that increasing the system temperature

Fig. 6 Peak power versus system temperature as functions of reactor period and system pressure.

causes a decrease in the measured peak power. Since little or no moderator boiling occurred prior to the time of peak power for these tests, the lower peak powers result from a change in the moderator reactivity coefficient, which increases as the system temperature is increased. Similar behavior occurs for the 20-msec tests, except that pressure effects are noticeable because boiling void formation provides an additional reactivity compensating mechanism for most of the 20-msec tests. For example, at a fixed initial system temperature, the measured peak powers increase as the system pressure is increased. This is a consequence of decreased boiling reactivity compensation.

Fuel plate surface temperature measurements in high-flux regions of the core are shown in Figures 7 and 8 for the 20-, 40-, and 150-msec tests. Figure 7 shows the temperature at time of peak power, $\theta(t_m)$, as a function of system temperature. Little or no boiling occurred prior to the time of peak power for the 40- and 150-msec tests; thus, $\theta(t_m)$ increases approximately linearly with increasing system temperature to provide the necessary reactivity compensation. For the 20-msec tests, boiling occurred prior to the time of maximum power for pressures up through 500 psig. For a given pressure in this range, $\theta(t_m)$ increases slightly above the moderator saturation temperature and is relatively independent of initial system temperature; this is a result of the relatively high heat transfer rates associated with nucleate boiling and the efficiency of boiling as a reactivity compensating mechanism. No boiling occurred prior to time of peak power for the 1000- and 2500-psig tests thus causing $\theta(t_m)$ to behave in a manner consistent with the 40- and 150-msec tests, ie, increase with increasing system temperature.

Fig. 7 Plate surface temperature at time of peak power versus system temperature as functions of reactor period and system pressure.

In Figure 8, fuel plate surface temperature rise at time of peak power, $\theta_r(tm)$, is shown as a function of initial system temperature. The temperature rise for the 40- and 150-msec tests, which had little or no boiling prior to time of peak power, decreases as the system temperature increases. This decrease in $\theta_r(tm)$ is (as was previously observed in the discussion of Figure 6) a result of the increasing moderator reactivity coefficient which increases with increasing system temperature. Similar behavior is noted in the 20-msec test data; however, the addition of boiling as a reactivity compensating mechanism causes $\theta_r(tm)$ to decrease at a more rapid rate with increasing system temperature for a fixed pressure in the range 0 to 500 psig. The nonboiling 1000- and 2500-psig $\theta_r(tm)$ data behave similar to the 40- and 150-msec $\theta_r(tm)$ data.

The foregoing analysis of power and temperature data shows that increasing the initial system temperature produces no unexpected effects on gross kinetic

Fig. 8 Plate surface temperature rise to time of peak power versus system temperature as functions of reactor period and system pressure.

behavior over the range of conditions investigated. The lower peak powers observed with increasing system temperature are obtained at the expense of higher fuel plate temperatures although the measured temperatures are well below any significant damage threshold temperature such as that associated with fuel plate melting ($\approx 1450^{\circ}\text{C}$). The nucleate boiling observed in the shorter period tests increased the heat transfer rate from the fuel plates to the moderator and provided additional reactivity compensation through void formation. While not observed in these tests, the onset of localized film boiling would be expected to occur as the heat flux is increased at shorter periods; the resulting reduction in heat transfer rate would cause rapidly increasing fuel plate temperatures and possible fuel plate melting.

1.2 Flow Tests

This section presents an analysis of tests performed with forced coolant flow; special emphasis is placed on those tests performed at high system temperature. These tests were performed to gain insight into the behavior of a pressurized-water reactor operating in the hot standby mode. Furthermore, the thermal properties of the coolant-moderator undergo rapid changes above about 400°F compared with the behavior up to 400°F ; thus, any gross changes in kinetic behavior due to thermal effects will tend to be maximized in the temperature region from 400 to 600°F .

Figure 9 shows peak power as a function of inverse period with system temperature as the variable parameter. The high-temperature test data with

Fig. 9 Peak power versus inverse period as a function of initial system temperature.

forced coolant flow are shown in comparison with data from lower temperature tests conducted both with and without flow. The data presented are for several pressures in the range from 0 to 2500 psig, representing subcoolings from about 20 to 600°F. Since little or no boiling occurred prior to the initial power peak, changes in subcooling had no measureable effect on the data shown. Significant decreases in peak power as a function of increasing system temperature result from a continued increase in the magnitude of the moderator reactivity coefficient. Forced coolant flow is observed to have little effect on peak power data for these short-period tests since the coolant velocity through the core (≈ 18 ft/sec) was insufficient to remove a significant amount of the excursion energy released prior to the time of the initial power peak.

Peak power data for 20-msec tests as a function of initial system temperature are shown in Figure 10. Also shown are relative plots of the reciprocal expansion coefficient of water (equivalent to the fractional change in water volume per unit change in temperature) and the reciprocal isothermal temperature coefficient of reactivity; both are normalized to the power data at about 350°F. Comparison of these curves shows the close relationship of peak power to the expansion properties of the water moderator as functions of increasing system temperature. The moderator coefficient of reactivity as a function of system temperature is directly related to the expansion coefficient of water and is closely approximated in Spert III by the isothermal temperature coefficient.

In Figure 11, peak power is shown as a function of inverse period for tests initiated at 600°F with both high flow (≈ 18 ft/sec) and low flow (≈ 9 ft/sec). Previous tests at ambient temperature [1, 7] have shown that over the period range investigated (≈ 50 to 20 msec), the coolant flow rates considered do not remove sufficient energy from the core region to affect the magnitude of the initial power peaks; and since the reactivity coefficients associated with the plates and moderator are of comparable magnitude, measured peak powers are essentially independent of coolant flow rate. The data at 600°F in Figure 11, however, show increases in the peak power as the flow rate is decreased at a constant period. This behavior results from a combined temperature-flow effect. Increasing the system temperature from ambient to 600°F increases the moderator reactivity coefficient by a factor of about seven while the reactivity coefficient associated with fuel plate expansion remains about constant; furthermore, for a given flow rate, the convective heat transfer coefficient between the plates and coolant-moderator increases by a factor of about two, from ambient to 600°F. Thus, increasing the flow rate at 600°F causes an increased amount of energy to

Fig. 10 Peak power versus initial system temperature.

Fig. 11 Peak power versus inverse period; high and low flow tests.

be transferred to the coolant-moderator where it provides greater reactivity compensation than in the fuel plates and results in lower peak power values.

Figure 12 compares reactor power and fuel plate surface temperature rise time-histories for two nominal 22-msec-period tests, initiated with constant subcooling (39°C) but from different initial system temperatures. As might be expected, the test initiated from 320°C ($\approx 600^\circ\text{F}$) shows significantly lower power and plate surface temperature rise than the 242°C test, and further illustrates the effect of the increased efficiency of moderator expansion as a reactivity compensating mechanism at high temperatures.

Figure 13 shows power and temperature histories for two 42-msec

Fig. 12 Power and fuel plate surface temperature versus time for two 22-msec-period tests.

Fig. 13 Power and fuel plate surface temperature versus time for two 42-msec-period tests.

tests initiated at constant initial system temperature, but with variable subcooling (11 and 78°C). Little change is observed in the initial power burst; however, the post-burst power for the 11°C subcooled test is slightly lower because boiling is providing some contribution to the feedback reactivity, as evidenced by the plate surface temperature. These data, together with that in Figure 12, indicate that the effects of changes in subcooling have significantly less effect on gross kinetic behavior than do changes in system temperature.

In summary, no unexpected effects on kinetic behavior were observed for tests conducted with coolant-moderator flow and initial system temperatures up to 600°F. Moderator expansion becomes the increasingly dominant reactivity feedback mechanism at high system temperatures, with relatively small amounts of energy release producing sizeable reactivity compensation.

2. INITIAL POWER LEVEL EFFECTS

2.1 Constant Flow Rate Tests

The experimental data obtained from step reactivity insertion tests performed at constant initial system conditions and variable initial power levels are correlated in this section to show the effects of initial power level on gross kinetic behavior. The tests were initiated by reactivity insertions of approximately 1\$ with the system at 500°F (260°C) core inlet coolant temperature, 2500-psig pressure, and 18-ft/sec coolant flow rate; initial power levels were varied from about 10⁻⁵ to 28 MW.

Figure 14 shows measured power as a function of time after the initiation of the excursion for a series of tests initiated from steady-state power levels of 0.01, 0.1, 1.0, 9.5, and 28 MW. The reactivity insertions ranged from about 0.96 to 1.01\$ with an estimated uncertainty of about $\pm 0.02\%$. At low initial power, the transient power rises monotonically during the tests. As the initial power is increased, the magnitude of the transient power increases, and shape changes occur in the form of initial peaking followed by a power undershoot and a subsequent rise to a quasi-equilibrium level. No moderator boiling occurred during these tests; thus, moderator and fuel plate expansion provided the feedback reactivity required to self-limit the excursions. Increasing the reactivity insertion at constant initial power would be expected to result in a more pronounced peaking effect in the power at lower initial powers. The effect of changes in flow rate on the power is a combination of the effect of energy removal from the core and heat transfer from the fuel to the moderator. The net effect may either increase or decrease the power peaking, depending upon the specific conditions. As the initial power level was increased, the energy required to self-limit the power rise was released in less time, causing the initial power peak to occur earlier in time with respect to excursion initiation. Because moderator expansion is a delayed feedback mechanism, more energy was actually released prior to the time of peak power for the high initial power tests than was required to self-limit the power rise. The excess energy release caused a brief overcompensation of reactivity, resulting in the power undershoot observed following the initial peak. The combination of energy removal from the core region by the forced coolant flow and stabilization of the delayed neutron population caused the power to rise slightly and approach a quasi-equilibrium level in the run-out region.

Fig. 14 Five power transients started from different initial power levels.

The energy released from the beginning of the reactivity insertion to the time of peak power is shown in Figure 15 as a function of initial power level for the five tests previously shown in Figure 14. For comparative purposes, the peaks for the two lowest initial power tests were considered to be the points at which the slope of the power was changing at its maximum rate. The data show that for initial power levels above about 0.5 MW, the energy release to peak power is an increasing function of initial power level; continued increase at the approximate rate shown would be expected until sufficient energy release occurred to cause moderator boiling prior to the time of peak power. As noted previously, the excess energy release in this initial power region provides over-compensation of reactivity, resulting in noticeable changes in power behavior following the initial peak. The energy release data below 0.5 MW initial power indicate that about 1 MW-sec is sufficient to self-limit the initial power rise provided that sufficient time is available to transfer this energy to the moderator. The continual removal of energy from the core region by forced coolant flow eliminates any definite peaking behavior in this period region; thus, the magnitude of the energy release indicated is only approximate.

Fig. 15 Energy releases to peak power versus initial power level.

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Figure 16 shows peak power and run-out power (the power 1.2 sec after peak) versus initial power level for the tests shown in Figure 14; data are also shown for a 10^{-5} MW initial power level test. For low power levels (≤ 0.5 MW),

Fig. 16 Peak and run-out power versus initial power level.

the peak and run-out powers are insensitive to increases in the initial power level. Above 0.5 MW, the peak and run-out powers increase, with the peak power rising at a faster rate than the run-out power.

A different aspect of peak and run-out power behavior is shown in Figure 17. In this figure, the ratios of the peak-to-initial power levels and the run-out-to-

Fig. 17 Peak-to-initial and run-out-to-initial power ratios versus initial power level.

initial power levels are plotted as a function of their respective initial power levels. The peak ratio decreases, with a nearly constant slope with higher initial powers, until a starting power level of about 0.5 MW is reached. Above initial levels of ≈ 0.5 MW, the absolute value of the negative slope decreases and appears to approach a constant value. This suggests that 1\$ insertions made from initial levels higher than those tested would produce peak powers that would approximate some constant multiple of the initial level as long as boiling did not contribute significantly to the self-limitation of the initial power burst. When boiling shutdown becomes significant, the peak ratio would be expected to again show a decrease with increasing initial power. The run-out ratio behaves similarly to the peak ratio, but shows a slower change of slope at the higher initial power levels.

The reactivity insertions used to initiate the transients discussed in this section have been referred to as step insertions; whereas, in actuality, they are very fast ramp functions. The insertion rate used was approximately 15\$ per second; the 1-dollar insertions required about 70 milliseconds to complete. This time interval, required for the reactivity insertions, was negligible for transients started at low power levels since the insertions were completed before any appreciable energy, which could appear in the form of reactivity feedback effects, had been released. However, many of the transient tests were started from high initial power levels, resulting in significant energy releases from the start of

the reactivity insertions; therefore, significant reactivity compensation occurred before the reactivity insertions were completed. For example, as will be seen in the data figures, the time interval between the start of the reactivity insertion and the occurrence of the power peak for several tests was of the order of 80 milliseconds. This means that reactivity was being both inserted and compensated, simultaneously, from the start of the transient to within 10 milliseconds of the power peak. Reactivity inserted in this ramp manner causes a slight decrease in the magnitude of the measured peak power as compared with the peak value that would have been obtained had a true step insertion been made [8]. The effect would not, however, cause any appreciable change in the correlative analysis of the data from Figures 14 through 17.

Another illustration of the effect of initial power level on excursion behavior is shown in Figure 18 in which the power histories of two excursions where the power traces have been time-adjusted to coincide at time of peak power are compared. The excursions were initiated with approximately the same system conditions of temperature, pressure, and flow, but had different initial power levels of 10 W (Run 59) and 10 MW (Run 104). They were also initiated with

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Fig. 18 Comparison of transients initiated at different power levels.

slightly different reactivity insertions. The total reactivity of Run 104 was 1.04\$, comprising a 0.97\$ step insertion and 0.07\$ required to raise the power from the zero power critical level to 10 MW. The total reactivity insertion of Run 59 consisted of a 1.12\$ step insertion.

These particular transients were chosen for comparison because of their similarly shaped initial power bursts; after the initial bursts, the transients begin to show pronounced differences in behavior. First, the transient started from the lower initial level (Run 59) had a greater power undershoot than was obtained from Run 104. Second, following the stronger undershoot, as the power started to rise again, Run 59 displayed greater oscillatory behavior than Run 104. Third, a distinct difference in the magnitude of the run-out power levels existed after about 0.5 seconds. This difference was approximately equal to the difference between the initial power levels at which the two transients were started.

The different power undershoot and oscillatory behavior was caused by the relative difference in the delayed neutron population of the two tests just prior to the reactivity insertions, an effect previously predicted analytically by Canosa [9]. The third difference in behavior, the difference in the run-out power levels of the two tests, has an important implication from the standpoint of operational safety. By way of illustration, assume that a hypothetical DNB (departure from nucleate boiling) power level limitation had been placed on this core which stated that a steady-state power level of 45 MW should not be exceeded. It can be seen from the figure that for Run 104, if the core had not been damaged by the initial power burst that momentarily exceeded the DNB limit, damage could occur quickly from the high run-out power level which exceeded 45 MW immediately after the power undershoot. For Run 59, however, if the core had not been damaged by the initial power burst, more than a second of time would be available in which protective action could be taken before the DNB limit was reached. For this hypothetical case, the reactor hazard is appreciably greater for the transient started from the higher power level, even though the reactivity inserted for that transient was smaller than the reactivity inserted at the lower initial power level for Run 59. This illustrates the importance of initial power level on the ultimate course and consequences of reactivity accidents in operating power reactors.

In addition to reactor power measurements, fuel plate surface temperatures were also recorded during the tests performed to study initial power level effects. Several axial locations in the subassemblies were monitored by means of Chromel-Alumel thermocouples encased in stainless steel sheaths that were insulated with magnesium oxide. The ends of the sheaths containing the thermocouple junctions were flattened to a 0.007-inch thickness and spot-welded to the surface of the fuel plates. Earlier experiments had shown that there were significant cooling effects at the surface of the thermocouple sheaths because of direct contact with the coolant. These effects were minimized for this test series by shielding the flattened tips of the thermocouples from the direct flow with small, semicircular, sheet-metal hoods. The hoods were welded to the plate surfaces downstream of the thermocouple junctions and were open to the coolant on the upstream side.

Figure 19 shows fuel plate surface temperature rise data obtained from a hot spot thermocouple located in a central subassembly (S-11 lattice position,

Fig. 19 Maximum fuelplate surface temperature rise versus time for various initial power level tests.

see Figure B-1), 6-1/2 inches below the axial center line of the core. These data correspond to the power histories previously shown (Figure 14) with the traces time-adjusted to coincide at time of peak power. The response time of the stainless steel thermocouples was too slow to accurately record the fuel plate temperature in the vicinity of peak power; in the run-out region, however, the response was sufficient to provide a good representation of the fuel plate surface temperature history.

The measured temperatures indicate that the moderator remained subcooled throughout the duration of the tests ($\approx 93^\circ\text{C}$ rise required to exceed the saturation temperature). As expected, the observed temperatures increase as a function of increasing initial power level above about 0.5 MW. Over the range of

conditions investigated, however, the increases are insufficient in magnitude to be of concern with regard to possible fuel plate burnout.

In summary, the data correlated in this section show that varying the initial power level produced appreciable changes in reactor excursion behavior for step reactivity insertions of a given magnitude. The changes, as a function of increasing initial power, were in the form of higher initial power peaking, higher run-out power levels, and shorter time intervals between test initiation and achievement of high quasi-steady-state, run-out power levels. Therefore, in this regard, the consequences of an accidental insertion of reactivity from operating power are potentially more severe than for a comparable reactivity insertion at lower initial power.

2.2 Variable Flow-Rate Tests

Several tests were performed to investigate the effects of core coolant flow rate on kinetic behavior. The excursions were initiated by nominal 1\$ step insertions of reactivity from an initial power level of about 1 MW and with coolant flow rates of approximately 18, 15.2, 9, and 4.5 ft/sec. These data supplement those previously reported [1] on coolant flow rate effects for tests initiated from zero power in Spert III.

Figure 20 shows the measured power histories for four tests conducted at various flow rates, the traces having been arranged to coincide at the time of the initial power peaks. Changes in flow rate had little effect on the initial self-limiting behavior; however, marked differences appeared in the post-peak region. Increasing the coolant flow rate caused less power undershoot and a progressive increase in the run-out power level. These effects are proportional to the rate at which energy is removed from the core region by the forced coolant flow.

Fig. 20 Reactor power versus time for several coolant flow rates.

Fig. 21 Reactor power and fuel plate surface temperature rise data versus flow rate.

Figure 21 illustrates the dependence of power and temperature data on flow rate for the tests of Figure 20. Shown, in addition to peak power, are values of run-out power, fuel plate surface temperature rise, and bulk water temperature rise obtained 1.2 seconds after peak power. The fuel plate surface temperature data were recorded by a thermocouple positioned in a central subassembly (lattice position S-11), 6-1/2 inches below the axial core center line; the bulk water temperature was measured at the exit of a subassembly (lattice position N-33) located in an average radial flux region of the core.

The peak power data are observed to be relatively insensitive to flow rate because the maximum available flow rate was insufficient to remove any significant amount of energy from the core prior to the time of the initial peak. The run-out power data, however, do reflect the energy removal by flow and increase linearly in magnitude with increasing flow rate. As expected, fuel plate surface temperature in the core hot spot also increased linearly with flow although at a slower rate than the run-out power. The bulk water outlet temperature remained essentially constant with increasing flow rate thus reflecting the linearly proportional rise in run-out power level.

Briefly, the flow effects observed in these tests are similar in nature to those previously reported for Spert III excursions initiated at zero power. Thus, these data indicate that up to 1-MW initial power, variations in coolant flow rate produce no unexpected changes in dynamic response to step insertions of reactivity.

IV. EVALUATION OF THE ART-04 TRANSIENT MODEL

This section describes the comparisons made between the reactor response to reactivity perturbations as calculated by the ART-04 computer code and the experimentally measured response. These comparisons provide a basis for evaluating the effectiveness with which ART-04 could be expected to calculate reactor excursion behavior in plate-core water systems similar in design to Spert III.

Most of the analysis is devoted to tests initiated by step insertions of reactivity in the range from 0.68 to 1.16\$ and with initial system temperatures up to 600°F, flow rates up to 20,000 gpm, and pressures of about 2500 psig; initial steady-state power levels were varied from about 10 W to 28 MW. In addition, calculated and experimental comparisons are also made for the reactor response to short-term continuous rod withdrawals (ramp tests) and loss-of-coolant flow tests. Reactivity perturbations of this type are representative of accidents that might occur in operating power reactors.

The ART-04 code [2, 3] was developed by the Bettis Atomic Power Laboratory for the IBM-704 computer to predict the transient behavior of water-cooled and -moderated reactor systems with plate-type fuel assemblies. The code is capable of calculating transients of "intermediate speed", ie, ones in which reactivity is inserted faster than during a normal operational change and slower than in a superprompt-critical power excursion. Several of the Spert III tests considered had reactivity insertions ($\leq 1.16\$$) slightly greater than the stated upper limit ($\approx 1.00\$$) of the code; however, exceeding this limit had no detectable affect on the calculational results.

The ART-04 code approximates the reactor core by means of four radial coolant channels of which the properties were selected to be representative of four radial regions of the Spert III C-19/52 core. Each channel was divided into fifteen axial sections for the purpose of describing local flux and void coefficient distributions. The thermal calculations provide for the determination of the temperature in each axial section of each channel at three points: in the metal, at the clad-water interfaces (fuel plate surface temperature), and in the water channel (bulk water temperature). The treatment of fluid flow and convective heat transfer in each of the radial channels is based on a one-dimensional transient model in which variations in fluid velocity and fluid properties in directions transverse to the flow path are neglected.

Core power as a function of time is determined by point reactor kinetics calculations, with the reactivity feedback obtained solely from moderator-coolant density changes and localized water density coefficients. The localized moderator density coefficients were obtained by the use of an experimentally measured average core void coefficient and measured local neutron flux values.

The program assumes the reactor to be initially operating at a steady-state condition. It can then be subjected to variations in one or more of the parameters: (a) flow rate through the reactor, (b) coolant inlet temperature, and (c) reactivity change due to control rod motion. The calculations in this section were performed with options (a) and (c) to compute reactivity insertion and loss-of-flow transients.

The output of the code includes bulk water, fuel plate surface, and internal fuel plate temperatures, heat fluxes, core power, system reactivity, and the coolant flow rate in the separate channels as a function of time. In addition, the heat flux required to reach a DNB (departure from nucleate boiling) condition may be calculated at each time step.

1. LOW INITIAL POWER TESTS

Considered in this section are comparisons of the calculated and experimental behavior for transient tests initiated from ≈ 10 watts initial power and with $\approx 600^\circ\text{F}$ (316°C) system temperature, ≈ 2500 psig pressure, and coolant flow rates from ≈ 9 to 18 ft/sec through the core. Step-wise reactivity insertions, ranging from about 0.90 to 1.16% , resulted in initial reactor periods of ≈ 350 to 15 msec. The calculational results from the ART-04 code are compared with experimentally measured values of reactor power, hot channel bulk water exit temperature, and compensated reactivity. Fuel plate surface temperatures are not compared because the temperatures calculated are average temperatures; whereas, the measured temperatures were point measurements, and the stainless steel sheathed thermocouples (described in Appendix A, Section 3) used to obtain experimental fuel plate surface temperature data lacked the time-response characteristics required for accurate measurements. Additional details relating to the tests discussed are given in Appendix C.

1.1 Power Comparisons

Calculated and experimental power histories are shown in Figures 22 through 25 for tests having initial periods in the range from about 83 to 22 msec

Fig. 22 Calculated and experimental power versus time for an 83-msec-period test (Run 62).

Fig. 23 Calculated and experimental power versus time for a 64-msec-period test (Run 71).

Fig. 24 Calculated and experimental power versus time for a 29-msec-period test (Run 63).

Fig. 25 Calculated and experimental power versus time for a 22-msec-period test (Run 64).

and corresponding reactivity insertions from about 1.00 to 1.10\$. All of the comparisons display common behavior in the run-out region, ie, the calculated power is somewhat lower than the experimental power. In the vicinity of the initial power peak, however, the agreement between calculation and experiment is observed to be period dependent, with the calculated power being higher than the measured power for the two short-period tests (Figures 24 and 25). The calculated power shape compares relatively well with that measured except in the region following the initial power burst; the experimental power displays a secondary peaking behavior in this region which is not reflected in the calculated power. The secondary peaks observed experimentally are believed to be functions of the system thermo-hydraulics (heat transfer, pressure drop, energy balance in the water, etc). The absence of these peaks in the calculated power indicates that insufficient coupling exists between the time-step thermal-hydraulic and power calculations in the ART code.

The magnitude differences observed in the previous comparisons for the initial peak and run-out power regions are perhaps better illustrated in Figures 26 and 27. These figures show both calculated and experimental peak and run-out power levels as a function of inverse period for several tests in the period range from about 80 to 20 msec with the system at 600°F temperature, 2500 psig pressure, and 18 ft/sec flow rate.

The peak power data shown in Figure 26 indicate the ART code results to be consistently higher by about 20 percent than the experimental results; whereas, the run-out power comparison shown in Figure 27 indicates that the ART code underestimates the power in this region by about 35 percent. The run-out power

Fig. 26 Calculated and experimental peak power versus inverse period for several 600°F tests at high (≈ 18 ft/sec) flow rate.

data were obtained one second after the initiation of the tests because by this time most of the transient effects disappear and leave essentially a steady-state situation. Thus, based on these comparisons, the ART code overestimates the peak power by about 20 percent and underestimates the run-out power by about 35 percent.

The specific reasons for the differences between calculation and experiment have not been determined; however, several reasonable possibilities exist. With regard to peak power, the ART code has no provision for calculating feedback reactivity generated by expansion of the core components (fuel, cladding, etc); order of magnitude calculations indicate that this mechanism contributes as much as 5 to 10 percent to the compensated reactivity and, thus, if included in the ART

code, would reduce the calculated peak power. In addition, the experimental peak power may be as much as 5 percent low because of the method used to calibrate the ion chambers. The chambers were calibrated calorimetrically at steady state; and since a small portion of the fission energy release is delayed in time, the indicated peak power during a fast transient is an underestimate of the actual power peak. Furthermore, the random uncertainty associated with the power calibration is judged to be about ± 10 percent, which may account for some differences. In the run-out region, plate expansion becomes less important as a reactivity compensating mechanism; also, the experimental power data should be more precise. The run-out power is, however, strongly dependent on the void coefficient of reactivity and the void distribution. Uncertainties in these input data to the ART code could appreciably affect the calculated power.

1.2 Temperature Comparisons

Assessment of the ART code accuracy in calculating experimental temperatures is important for operational and safety analysis purposes. A comparison of calculated and experimental hot-channel exit, bulk-water temperature rise in the run-out region of several tests conducted at 600°F, 2500 psig, and 18 ft/sec flow is shown in Figure 28 as a function of inverse period. As expected, these comparisons are similar to the power comparisons in that the calculated run-out bulk-water temperatures are lower than those measured experimentally. Figure 29 shows a time-history comparison of the bulk-water temperature rise measured at the top of a hot channel and ART calculations of the temperature rise at a similar core position. Reasonably good agreement in magnitude is observed; however, the curve shapes do not agree very well in the vicinity of the initial temperature peak. This shape disagreement is attributed primarily to the relatively poor response characteristics of the thermocouple used to measure the experimental temperature.

Fig. 27 Run-out power (after 1 second) versus inverse period; calculated and experimental for several 600°F tests.

1.3 Reactivity Comparisons

Of all the self-shutdown mechanisms that have been postulated for water reactors, perhaps the two most widely investigated are boiling void formation and core density decrease due to thermal expansion of the moderator and core components. For the range of reactivity insertions investigated (0.90 to 1.16\$), the shutdown in Spert III during high-temperature, high-flow tests is primarily caused by the nonboiling density decrease of the moderator.

Figure 30 is a comparison of the reactivity compensated to the time of peak power as a function of inverse period for transients calculated by ART and experimental transients. The experimental data were derived from the reactor power traces by use of the computer program SPORT [10]. Also shown on the figure is a plot of the prompt reactivity ($\alpha_0 \Lambda / \beta$) which indicates that the amount

Fig. 28 Calculated and experimental hot channel temperature rise after 1 second versus inverse period for several 600°F tests.

of delayed reactivity compensated is approximately a constant 2% for the range of reactivity insertions investigated. Good agreement between calculation and experiment is observed; in addition, fairly good agreement was found at times prior to peak power. These data thus substantiate the premise that nonboiling density decrease of the moderator (the only shutdown mechanism considered by ART) is the primary reactivity feedback mechanism for high-system-temperature tests in Spert III. If the system temperature were reduced, fuel plate expansion would become more important as a shutdown mechanism, and calculations with the ART code would become less descriptive of the transient response.

The foregoing comparisons between the calculated and experimental response of tests initiated from low initial power indicate that the ART-04 code is capable of calculating initial power peaks to within about 20 percent, run-out power levels

Fig. 29 Calculated and experimental hot channel temperature rise versus time for a 22-msec-period test (Run 64).

to within about 35 percent, and bulk-water temperatures to within about the same degree of accuracy as the power. This evaluation was based on direct comparison of calculated and experimental data for tests with high temperature and pressure and with forced coolant flow. Possible reasons for the observed differences were postulated, eg, the neglect of fuel plate expansion as a reactivity compensating mechanism in the calculations, experimental power calibration uncertainties, and the sensitivity of the model to void coefficient magnitude and distribution. In view of these factors, the ART-04 code is considered to be capable of providing reasonably good estimates of the dynamic response of plate cores similar to the Spert III C-19/52 core over the range of conditions investigated in this evaluation.

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Fig. 30 Calculated and experimental reactivity compensated at peak power versus inverse period for several 600°F tests.

2. HIGH INITIAL POWER TESTS

The ART-04 code is evaluated in this section for transient tests initiated from initial power levels in the range from 0.01 to 28 MW by comparing calculated results with experimental data obtained with the system at $\approx 500^\circ\text{F}$ core inlet coolant temperature, with ≈ 2500 -psig pressure, and with coolant flow rates in the range from ≈ 4.5 to 18 ft/sec. The tests were initiated by nominal 1\$ step insertions of reactivity; in reality, the reactivity insertions were fast ramp insertions of approximately 15\$/sec and were so considered in the calculations. Discussed first are calculated and experimental comparisons of power and temperature data for several tests with constant flow rate (≈ 18 ft/sec) and various initial power levels, followed by similar comparisons for several tests with constant initial power (≈ 1 MW) and various coolant flow rates. Calculated results are compared to experimental measurements of reactor power and bulk-water temperature; additional details relating to the experimental data are given in Appendix D.

2.1 Variable Initial Power

Discussed in this subsection are the calculated and experimental results from five tests having initial power levels of nominally 28, 10, 1, 0.1, and 0.01 MW. Figures 31 through 35 show time histories of the calculated and experimental reactor power and nominal channel outlet temperature rise for these tests. The experimental bulk-water temperatures were obtained from a thermocouple located at the fuel channel exit of a near-nominal core subassembly, and are a good approximation to the average core coolant exit temperature.

The reactor power calculated by the ART code for these tests shows about the same relationship to the experimental power as was observed in the low initial power comparisons discussed in the previous section. For those tests having well-defined initial power peaks, the calculated power is slightly higher than the experimental power in one case (Figure 32) and slightly lower in the other two cases (Figures 31 and 33). On the average, the calculated peak powers were about 5 percent lower than the measured peak powers; the variations observed are believed to be within the uncertainties associated with the calculated and experimental data. In the run-out region, the calculated power is lower than the measured power by about 22 percent on the average. The calculated outlet temperatures behave in a manner consistent with the calculated powers, remaining approximately 15 to 20 percent below the measured temperature data in the run-out region.

Fig. 31 Comparison of measured and calculated response for Run 113.

Fig. 32 Comparison of measured and calculated response for Run 104.

Fig. 33 Comparison of measured and calculated response for Run 106.

Fig. 34 Comparison of measured and calculated response for Run 95.

Fig. 35 Comparison of measured and calculated response for Run 93.

The comparisons considered in this section indicate that increasing the initial power level has no detectable effect on the ability of the ART code to calculate reactor power and water temperature response to reactivity insertions of about 1\$. The differences between calculated and experimental response are similar to those observed in the previously discussed low initial power comparisons and are believed to result from the same reasons, viz, neglect of fuel plate expansion as a reactivity compensating mechanism in the calculations, uncertainties in the experimental power calibrations, and the sensitivity of the model to void coefficient input data.

2.2 Constant Initial Power

This subsection compares the calculated and experimental results for tests initiated by step insertions of approximately 1\$ reactivity from steady-state power levels of about 1 MW with average core coolant flow rates of approximately 15.2, 9, and 4.5 ft/sec. Figures 36, 37, and 38 show calculated and experimental comparisons of reactor power and bulk-water temperature rise, both as functions of time, with flow rate as the variable parameter.

These comparisons show features similar to those previously observed for low and high initial power tests. The calculated run-out powers are lower than those measured; however, no correlative trends

Fig. 36 Comparison of measured and calculated response for Run 105.

Fig. 37 Comparison of measured and calculated response for Run 109.

Fig. 38 Comparison of measured and calculated response for Run 114.

are apparent in the region about peak power. The calculated run-out temperatures are somewhat lower than measured, again showing the influence of reactor power. Reasons for the observed differences between calculation and experiment are believed to be the same as those discussed in previous sections, ie, possible errors in the experimental power data, possible errors in experimental void coefficients and void distribution, and the omission of a plate expansion reactivity feedback mode in the ART calculations.

3. RAMP TESTS

Data from two scoping tests performed by brief continuous control rod withdrawals, such as might occur from reactor operator errors, were compared to the reactor behavior calculated by the ART-04 code. These tests were conducted with the system at $\approx 500^\circ\text{F}$ core inlet coolant temperature, ≈ 2500 -psig pressure, and ≈ 18 ft/sec coolant flow rate. The reactivity insertion rate for both tests was approximately $5.3 \text{ } \beta/\text{sec}$, with a total reactivity insertion of about 27β .

Figure 39 presents time histories of calculated and experimental reactor power and bulk-water temperature for Run 101. This test was initiated from a steady-state power level of about 25 MW; the control rods were continuously withdrawn for about 5 sec, then held fixed for about 6 sec prior to scram in order to simulate an inadvertent operator error. The calculated power is observed to agree reasonably well with experiment until completion of the reactivity insertion. Beyond this point, the calculated power falls well below the measured power; and 10 sec after initiation of the control rod insertion, the calculated power is about 24 percent lower. A comparably low value is observed for the calculated exit bulk-water temperature rise.

Fig. 39 Comparison of measured and calculated response for ramp insertion Run 101.

In Figure 40, the same variables considered in the previous figure are again compared for Run 102. This test, however, was initiated from a steady-state power level of about 0.75 MW, and a slightly different method of control rod movement was used; the control rods were withdrawn for 5 sec, held fixed for 3 sec, then reinserted for 3 sec at the previous withdrawal rate. After the 3-sec insertion, the rods were held fixed for about 3 sec prior to scram. This simulates an accident whereby the operator realizes a mistake has been made and tries to compensate by inserting the rods. The agreement between the calculated and experimental power is similar to that previously observed. Eight

Fig. 40 Comparison of measured and calculated response for ramp insertion Run 102.

seconds after transient initiation, the calculated power was approximately 28 percent below that measured. Following the control rod reinsertion, the calculated power had fallen to about 33 percent below the measured power. The bulk-water temperature rise shown was too low in magnitude to be accurately compared to the calculation.

The comparative behavior considered in this section for the ramp tests is in general agreement with the previously discussed results obtained from step reactivity insertion comparisons.

4. LOSS-OF-FLOW TESTS

Several tests initiated at appreciable power levels were carried out to investigate the reactor response to a sudden loss of coolant flow. Both partial and total loss-of-coolant-flow tests were conducted, and their results were compared to calculational results obtained from the ART-04 computer code.

The tests were performed by stabilizing the reactor at the temperature, pressure, and flow conditions established as standard for this test series (500°F core coolant inlet temperature, 2500-psig coolant pressure, and 18 ft/sec coolant flow) and at the selected initial power level. The electrical power to the primary coolant pumps was then switched off, and the reactor power, temperature, pressure, and flow rate data were recorded during the flow coastdown. The reactor was scrammed within 5 to 10 seconds after the electrical power to the pumps was terminated. For the partial loss-of-flow tests, one of the two operating pumps in each of the two coolant loops was switched off, leaving the remaining pump in each loop running. The total loss-of-flow tests were initiated by simultaneously switching off all four primary system coolant pumps.

Partial loss-of-flow tests were performed from initial steady-state power levels of 10 and 27 MW. The experimental behavior of the reactor power, average core exit bulk-water temperature rise, and coolant flow rate are shown in Figures 41 and 42 as a function of time after the pump power was turned off. For comparative purposes, the reactor power, $P(t)$, and coolant flow, $G(t)$, are shown in the figures as fractional values of the initial power level, P_i , and the initial coolant flow rate, G_i .

Fig. 41 Reactor response to partial loss of flow, Run 107.

The removal of electrical power from two pumps in the partial loss-of-flow tests resulted in a reduction in flow to approximately 70 percent of the initial flow within 1 second; after which time, the flow rate remained stable for the duration of the experiment. The reactor power behavior during the partial loss-of-flow test that was initiated from 10 MW (Figure 41) decreased fairly rapidly during the first second of the flow transient, then after a few seconds stabilized at about 95 percent of the initial power level. The average bulk-water core outlet temperature change was small, less than 1°C. The same flow reduction from an initial power level of 27 MW (Figure 42) produced a similar behavior. The power level dropped to approximately 92 percent of the initial level within the first second after the flow reduction; and following minor fluctuations, the power level stabilized at that level. The average bulk-water core outlet temperature rise was slightly more pronounced in this transient, rising slightly over 1°C at 1 second after the flow reduction, then stabilizing at about 1/2°C above the initial value.

Fig. 42 Reactor response to partial loss of flow, Run 108.

Only slight changes in behavior resulted from the partial loss-of-flow transients; the data indicated that the reactor could have been operated at the resulting power and flow conditions indefinitely without any undesirable effect on the core.

The total loss-of-flow tests were performed from initial power levels of 10 and 28 MW. The reactor power, average bulk-water core outlet temperature, and flow rate data from these tests are shown in Figures 43 and 44 as a function of time after stopping the coolant pumps. Within 1 sec after the electrical power to all four coolant pumps was switched off, the flow rate had dropped to approximately 26 percent of the initial value. As shown in the figures, the flow rate decreased very rapidly following the pump

Fig. 43 Measured and calculated response to total loss of flow, Run 111.

Fig. 44 Measured and calculated response to total loss of flow, Run 117.

stoppage and was essentially zero within approximately 8 to 10 sec. The reactor power for the flow transient initiated from 10 MW (Figure 43) did not decay as rapidly as the flow. The data show that while the flow dropped essentially exponentially, the power decreased almost linearly with time through the duration of the test. The bulk-water temperature at the exit of the nominal subassembly showed an increase that was also approximately linear.

The total loss-of-flow transient that was started from a reactor power level of 28 MW (Figure 44) showed a more rapid power decay during the early part of the flow transient and decreased to approximately 80 percent of the initial level within the first second. The rate of power decrease slackened as the flow decrease became less rapid and became almost linear in behavior for the remainder of the test and reached 40 percent of the initial power when the flow had decreased to about 3 percent of initial flow rate. The bulk-water temperature at the core outlet of the nominal subassembly rose as a nearly linear function.

Comparison of the rates of decrease of power as a function of time between the two total loss-of-flow tests shows that the tests initiated from a reactor power of 28 MW had a proportionally greater rate of power decay than the transient started from 10 MW. This behavior is attributed to the stronger reactivity feedback originating in the greater energy release per unit of time that was available to the flow transient started from the higher power level. After 5 seconds, the power level of the 28-MW transient had decayed to 40 percent of its initial value while that of the 10-MW test had dropped to 60 percent of the initial power level. These data indicate that the core

would be capable of sustaining the effects of total loss-of-flow accidents for considerable periods of time before damage would be incurred.

The power and temperature response calculated by the ART-04 code for the total loss-of-flow transients are also included in Figures 43 and 44. As in the previous discussions of comparisons of calculated and experimental power behavior, the power levels calculated by the code were lower in magnitude than those measured experimentally. At 5 seconds after the start of the flow transient, the calculated power for the 10-MW test (Figure 43) was approximately 25 percent below that measured; and for the 28-MW test (Figure 44), the calculated power was about 23 percent low. This comparative agreement between experimental and calculated power behavior is essentially the same as encountered in the previous analysis of ART-04 computational results. However, a considerable change from the findings obtained in the earlier comparisons appeared when comparing the calculated and experimental temperature behavior of the loss-of-flow transients. As shown in Figures 43 and 44, the code calculated values that were considerably higher than those measured for the bulk-water temperatures rise at the core outlet of a nominal subassembly. Similarly, the calculated temperatures at other points in the core were found in all cases to be higher than the values given by the comparable thermocouple measurements. This is the reverse of the results that were found with the step transient comparisons. This difference is believed to result from the flow coastdown data supplied as input to the code, which was obtained from differential pressure transducers positioned in the main coolant loops. It is considered likely that the flow coastdown behavior measured in the loop may differ from that which occurs through the coolant channels of the core. In addition, the transducers used were steady-state devices and were not designed for transient measurements. Subsequent comparisons of experimental and calculated comparisons of the bulk-water temperature data indicate that a higher coolant flow existed in the region of the thermocouple than was used in the code calculations.

V. CONCLUSIONS

This report has presented the experimental and analytical results from tests designed to investigate the effects of system temperature and initial power level on the excursion behavior of a water-moderated and -cooled, highly enriched plate core in the Spert III PWR. The operating conditions of temperature, pressure, flow rate, and power level were chosen to approximate those encountered in the operation of pressurized-water power reactors. The correlative and calculational efforts described in the text allow formulation of the following conclusions regarding the kinetic behavior and inherent safety of this and similar systems over the range of conditions considered.

Correlation of the data obtained from step-reactivity-insertion, low-initial-power tests shows that for a given reactivity addition, increases in the initial system temperature over the range from ambient to 600°F reduce the resulting transient power, energy release, and fuel plate surface temperature rise. These reductions result from an increase in the moderator reactivity coefficient as the system temperature increases, an effect shown to be closely related to the thermal expansion properties of water as a function of temperature. Although the fuel plate temperature rise decreased, the absolute temperature of the fuel plates increased with increasing system temperature, but remained well below the melting temperature.

The data from tests performed with various initial power levels up to 28 MW show that increases in initial power above about 0.5 MW cause higher initial power peaking, higher run-out powers, and shorter time intervals between test initiation and achievement of quasi-steady-state run-out power levels. These changes in behavior demonstrate that the consequences of an accidental insertion of reactivity from high power are potentially more severe than at low power.

The data acquired during the experimental program were also used for comparative purposes in order to evaluate the effectiveness of the ART-04 computer code in calculating the kinetic behavior of the Spert III core. Experimental and calculational comparisons of reactor power and bulk-water temperature for tests performed with various initial conditions and reactivity perturbation modes formed the basis for the evaluation.

The comparative results show that the ART-04 code is capable of calculating initial power peaks to within about 20 percent and run-out power levels to within about 35 percent of those experimentally measured. The calculated peak powers showed little systematic behavior, being higher in some cases and lower in other cases than the experimental data. The calculated run-out power levels, however, were always lower in magnitude than the measured values. The comparative features of the bulk-water temperature data were similar to those of the reactor power, except for the loss-of-flow tests, in which case the calculated temperatures were somewhat higher than the measured temperatures.

Several reasons were postulated which, in at least a qualitative sense, are sufficient to explain the observed differences between the ART-04 calculational results and the experimental data. These include: (a) the neglect of fuel plate expansion as a reactivity compensating mechanism in the ART-04 code calculations, (b) uncertainties in the experimental reactor power calibration, (c) sensitivity of the ART-04 calculations to void coefficient magnitude and distribution,

and (d) uncertainties in the experimental flow coastdown data supplied as input to the code for loss-of-flow test calculations. When considered in terms of these uncertainties, which are associated with both the calculated and experimental results, the comparative agreement obtained is as good as could be expected. Thus, the evaluation indicates that the ART-04 code is capable of providing reasonable predictions of the nondestructive excursion behavior of water-moderated and -cooled, plate reactors, similar in design to Spert III, over a wide range of initial power levels for several modes of reactivity perturbation not greatly exceeding 1\$ in magnitude and for system temperatures such that moderator expansion is the dominant reactivity feedback mechanism.

In general, then, the results of the Spert III reactor safety studies program with the plate-type C-core confirm that, for this type of reactor, the kinetic response of the system is primarily moderator controlled. The comparison of analytic methods and experimental results also confirms that models based on a consistent physical picture of moderator effects provide a reasonable prediction of nondestructive excursion behavior of water-moderated and -cooled, plate-type reactors. Therefore, these models can now, with the experimental confirmation provided by these tests, be more usefully and reliably employed to assess the safety of other reactor systems.

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APPENDIX A
SPERT III FACILITY DESCRIPTION

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APPENDIX A

SPERT III FACILITY DESCRIPTION

This section presents a brief description of the Spert III facility and a summary of the pertinent design characteristics. A detailed description of each component of the system is presented in IDO-16721 [11].

The Spert III facility is designed for nuclear transient testing under environmental conditions that typify pressurized-water power reactors. To this end, the facility is provided with components designed for pressures to 2500 psig and temperatures to 650°F, with capabilities for flow up to 20,000 gpm (≈ 18 ft/sec core coolant flow rate). The primary heat removal system is capable of sustaining a reactor power level of 60 MW for approximately 30 minutes. Since the Spert III reactor is essentially unshielded, the complete plant is remotely operable from a control center located approximately one-half mile distant.

1. PLANT

The Spert III primary system, shown schematically in Figure A-1, includes a reactor pressure vessel, pressurizer vessel, two coolant loops and associated heat exchangers, and a secondary coolant system. The reactor pressure vessel is of laminated-welded construction, built around a Type 304L stainless steel liner. Coolant enters the reactor vessel by means of a bottom tee, flows upward through the active core, and is then forced down between thermal shields surrounding the core region to near the bottom of the vessel where it exits through six radially spaced exit nozzles. The control rod drives, transient rod drive, and access ports are located on the fully removable top head which is bolted to the cylindrical vessel shell. The pressure vessel is approximately 23-1/2 ft high with an inside diameter of 4 ft.

The original pressurizer vessel [12] was a carbon steel shell, clad internally with Type 304L stainless steel; however, the present vessel [13] is clad with nickel. The system is pressurized by heating water with sixteen 12-kW immersion-type heaters located in the pressurizer vessel. Pressure is transmitted to the primary coolant system by means of a 4-in. static line connected to the reactor vessel coolant outlet piping.

Each primary coolant loop contains two circulating pumps capable of delivering 5000 gpm each, a U-tube heat exchanger capable of dissipating 30 MW of heat, and the necessary flow and temperature control equipment. System temperature control is accomplished by passing a portion of the primary coolant stream in each loop around the heat exchangers.

The secondary coolant system includes the shell side of the heat exchangers, two secondary coolant pumps, an auxiliary line for filling the primary system, and the requisite controls. Water vapor generated on the shell side of the heat exchangers is vented to the atmosphere.

Fig. A-1 Spert III plant process diagram.

Auxiliary equipment for Spert III includes water treating facilities, water storage tank, auxiliary cooling system for the primary pump motors, various storage and hot waste disposal equipment, as well as compressed air for operation of the control rod and transient rod drive system. Emergency power equipment is also maintained.

For high-nuclear-power operation, demineralized water is supplied to the shell side of the primary heat exchangers in a quantity limited by the capabilities of the water treating equipment and the demineralized water storage tanks. Because of these restrictions, full-power operation (60 MW) of the reactor is limited to about 30 minutes duration.

2. REACTOR SYSTEM

2.1 Core

The Spert III C-19/52 core consists of 44 fuel assemblies, 8 fuel-bearing control rods yoked together in banks of two, a transient rod, and upper and lower grid structure. The fuel assembly, shown in Figure A-2, is made up of fuel plates, side plates, a center support strut, and end boxes. The fuel plates are composed of a 93 percent enriched UO_2 stainless steel cermet meat section, approximately 20 mils thick, surrounded by a 5-mil Type 304L stainless steel clad. The water

Fig. A-2 Spert III Type 1-S fuel assembly.

channel between plates is a nominal 0.128 inch thick. To provide space for the transient rod, the four assemblies in the center of the core are slightly reduced in size. These assemblies (designated Type 2S) contain 32 fuel plates and contain 442 grams of U-235 per assembly. The remainder of the fuel assemblies (Type 1S) consist of 38 fuel plates and contain 638 grams of U-235 per assembly. The Type 2S assemblies are 2.5 by 2.5 inches, and the Type 1S are 3.0 by 3.0 inches. Both types have an active fuel length of

36 inches. Those positions in the core lattice not occupied by fuel assemblies or control rods contain nonfueled filler pieces to ensure proper flow distribution in the core (see Figure A-3).

Fig. A-3 C-19/52 core cross section.

2.2 Control Rods

The absorber sections of the control rods are of Type 18-8 stainless steel containing 1.35 weight percent Boron-10. The control rods have a fuel follower section containing fuel plates in a Type 2S configuration and contain 414 grams of U-235 per follower section. A view of the control rods is presented in Figure A-4. The transient rod is cruciform shaped with an absorber section containing 1.35 weight percent Boron-10 in Type 18-8 stainless steel with a stainless steel follower section. The transient rod is used to initiate nuclear power excursions by rapid ejection of poison from the core.

Fig. A-4 Spert III control rod pair.

3. INSTRUMENTATION

Since the data presented in this report represents the response of certain detectors, a brief discussion of the detectors used is included. A detailed discussion of the reactor transient instrumentation is given elsewhere [14].

The reactor transient power was measured by four gamma-uncompensated ionization chambers which were physically positioned to measure power over the complete range encountered in these excursion tests. The operational power detectors consisted of two pulse counters for startup and two ionization chambers for higher power levels.

Temperature was measured by thermocouples positioned to measure fuel plate surface temperature and bulk-water temperature associated with the hot channels. For reactor system temperatures up to about 400°F, a bare-wire, Chromel-Alumel-type couple was used; for the higher temperature tests, a stainless steel sheathed, MgO-insulated, Chromel-Alumel couple was used. These stainless steel sheathed couples, because of their bulk, had a slower transient response time; therefore, an analytical procedure was developed to correct the measurements. Figure A-5 is a photograph of the two types of thermocouples; the details of these thermocouples as well as those of the corrective analysis are presented elsewhere [14].

Fig. A-5 Stainless steel sheathed, Chromel-Alumel thermocouples.

Pressure was measured at 3 positions adjacent to the core by unbonded-strain-gage-type pressure transducers.

4. DESIGN DATA SUMMARY

A summary of the characteristics of the reactor and major plant equipment is tabulated in this section. The nuclear characteristics of the reactor, as presented, represent those used for nuclear design of the core.

4.1 General Reactor Design Data:

Type -- pressurized-water experimental reactor suitable for boiling or
pressurized-water experiments
Function -- experimental, reactor transient behavior, and safety
Moderator -- H_2O
Coolant -- H_2O
Neutron energy -- thermal
Core type -- heterogeneous, plate type
Heat power (max) -- 60 MW (limited)
Power density (max) -- 217 kW/liter
Design pressure and temperature (max) -- 2500 psig at 650°F

4.2 Reactor-Component Design Data:

Core:

Configuration -- cylinder

Diameter -- ≈ 24 in.

Active height -- 36 in.

Volume -- $2.76 \times 10^5 \text{ cm}^3$

Composition:

Water -- 73.5 volume percent

Stainless steel -- 25.2 volume percent

U-235 -- 1.3 volume percent

Fuel load, U-235 -- 30.8 kg

Number of fuel assemblies, all types -- 52

Fuel Elements:

Type -- rectangular, plate-type, stainless steel, Type-304L

Number:

Type 1-S -- 40

Type 2-S -- 4

Control-rod fuel section -- 8

Overall Dimensions:

Type 1-S -- 2.95 by 2.95 by 47.125 in.

Type 2-S -- 2.476 by 2.476 by 47.125 in.

Control rod fuel section -- 2.476 by 2.476 by 39 in.

Number of fuel plates/element:

Type 1-S -- 38

Type 2-S -- 32

Control rod fuel section -- 32

Fuel content:

Type 1-S -- 638.4 g

Type 2-S -- 441.8 g

Control rod fuel section -- 414.4 g

Plate spacing (nom) -- 0.128 in.

Flow area (inside)/element:

Type 1-S -- 6.93 in.^2

Type 2-S -- 5.85 in.^2

Control rod fuel section -- 6.05 in.^2

Heat-transfer area:

Type 1-S -- 3160.9 in.^2

Type 2-S -- 2115.1 in.^2

Control rod fuel section -- 2115.1 in.^2

Fuel Plates:

Type -- rectangular, flat

Materials:

Clad -- stainless steel, Type 304L

Fuel core -- 93.5 percent enriched UO_2 - stainless steel

Overall Dimensions:

Type 1-S plates -- 37 by 1.445 by 0.030 in.

Type 2-S plates -- 37 by 1.208 by 0.030 in.

Control rod fuel plates -- 37 by 1.208 by 0.030 in.

Fuel-core dimensions:

Type 1-S plates -- 36 by 1.155 by 0.020 in.

Type 2-S plates -- 36 by 0.918 by 0.020 in.

Control rod fuel plates -- 36 by 0.918 by 0.020 in.

U-235/plate:

Type 1-S plates -- 16.8 g

Type 2-S plates -- 13.8 g

Control rod fuel plates -- 12.95 g

Fuel-core composition:

UO_2 -- 18.7 weight percent

Control Rods:

Type -- rectangular; upper section is absorber material, lower section is a fuel subassembly

Number -- 8

Composition:

Absorber section -- 1.35 weight percent Boron-10 in Type 18-8 stainless steel; 3/16-in.-thick hollow square box

Fuel section -- see fuel-assembly section

Overall Dimensions:

Absorber section (nom) -- 2.476 by 2.476 by 38 in.

Fuel section (nom) -- 2.476 by 2.476 by 39 in.

Travel -- 42 in. maximum

Average scram time -- 0.350 sec

Withdrawal rates:

minimum -- 6.66 in./min

maximum -- 20 in./min

Transient Rod:

Type -- cruciform, lower section absorber material, upper section Type 18-8 stainless steel

Number -- 1

Composition -- 1.35 weight percent Boron-10 in Type 18-8 stainless steel

Dimensions:

Thickness of blades -- 3/16 in.
Blade width -- 5.25 in.
Absorber-section length -- 38 in.
Travel -- 42 in. (max)
Average release time -- 0.2 sec.

Control Rod Drives:

Type -- ball nut and screw; air pressure maintains rod in contact with screw and scram rods
Number -- 5
Motor type -- 480 V, 3 ϕ , 3-speed, constant torque
Motor rating -- 1 hp

Reactor Vessel:

Construction:

Shell -- laminated
Top head -- forged, full-opening flanged, hemispherical
Bottom head -- laminated
Inside diameter -- 48 in.
Shell thickness (including clad) -- 3.25 in.
Head thickness -- 3.5 in.
Materials:

Outer shellplate -- VMS 1146 plates 1/4 in. thick
Inner shellplate -- ASTM 225 Gr B with Type 304L stainless steel clad
Top head -- VMS-1146
Clad -- 309 Cb stainless steel

Clad thickness -- 0.125 in.

Design pressure:

Working pressure -- 2500 psig
Instrument shock pressure -- 3500 psig

Design temperature -- 700°F

Overall length, includes head -- 19 ft 11-1/8 in.
Maximum diameter of head flange -- 5 ft 7 in.
Number of thermal shields -- 5
Total thickness of shield -- 5.75 in.
Average length of shields -- 45.5 in.
Inside diameter of inner shield -- 34-1/4 in.

Weight of vessel:

Shell	55,784 lb
Head	10,606 lb
Bottom tee	5,580 lb
Shield	<u>11,430 lb</u>
Total	83,400 lb

Insulation -- 4 in. of foamglass

4.3 Thermal Design Data:

Coolant -- deionized water
Heat power (max) -- 60 MW
Coolant flow (max) -- 20,000 gpm
Coolant passes through core -- one
Velocity in core (max flow) -- 17.5 ft/sec
Heat transfer area -- 1055 ft²
Average heat flux (core) -- 194,200 Btu/hr-ft² at 60 MW

4.4 Primary Coolant System Design Data:

Primary Pumps:

Type -- 10 by 10 by 11, single-stage, single-suction, double-volute,
canned rotor
Number -- 4
Capacity, ea -- 5000 gpm
Net differential head -- 328 ft of H₂O at 650°F, 2500 psig
Motor:
Rating -- 500 hp
Type -- 480 V, 3 ϕ , 60 cycle
rpm -- 3550
Weight, ea -- 8500 lb
Material -- stainless steel, Type 304L

Heat Exchangers:

Type -- U-tube, evaporative
Number -- 2
Heat transfer capacity, ea -- 30 MW
Tube side fluid -- primary coolant, deionized H₂O
Shell side fluid -- deionized water
Design pressure:
Tube side -- 2500 psig
Shell side -- 75 psig
Design temperature:
Tube side -- 668°F
Shell side -- 320°F
Heat-transfer surface -- 1940 ft²
Number of tubes -- 367 U's
Tube size, OD -- 5/8 in., 16 min BWG
Tube straight length -- 191 in.
Tube pitch -- triangular
Pressure drop -- 10 psi
Materials:
Tubes -- stainless steel, Type 304L
Tube sheet -- A-266-5S, CL 2, clad with stainless steel, Type 304L
Channel -- A-266-5S, CL 2, clad with stainless steel, Type 304L
Channel cover -- A-266-5S, CL 2, clad with stainless steel, Type 304L
Shell -- A-212, Gr A

Weight:

Full -- 43,200 lb
Empty -- 23,400 lb
Tube bundle -- 11,600 lb

Pressurizer Vessel:

Number -- 1
Design -- Section VIII ASME Boiler and Pressure Vessel Code
Design Pressure -- 2750 psig
Design Temperature -- 700°F
Dimensions:

Overall height -- 15 ft 2-11/16 in.
Overall diameter -- 3 ft 3.3 in.

Volume -- $\approx 83.0 \text{ ft}^3$
Shell thickness -- 3.15 in.
Head thickness -- 2.9 in.
Materials:

Backing -- A-212, Gr B, FBX
Clad -- Nickle 200 (ASTM-B-162), 0.10 in.

Heaters -- electric immersion, 480 V, 3 ϕ :

Number -- 16
Capacity, ea -- 12 kW
Total capacity -- 192 kW

Weight -- $\approx 26,000 \text{ lb}$

Primary System Piping:

Type -- centrifugally cast stainless steel as per ASTM-A-362-52T,
modified

Wall thickness:

10 in. to 16 in. -- Schedule 160
8 in. -- Schedule 140

Material -- stainless steel, Type 304L
Insulation -- 3 in. of 85 percent magnesia

Makcup Pump:

Number -- 1
Type -- Triplex, direct flow, vertical reciprocating pump
Piston diameter -- 7/8 in.
Stroke -- 2-1/2 in.
Capacity rating (max):

gpm -- 5
Head -- 3000 psi

Motor:

Type -- 480 V, 3 ϕ , 60 cycle, gear motor
hp -- 10
rpm -- 280

4.5 Secondary Coolant System Pumps:

Type -- single-stage, single-suction, vertical-split-case, centrifugal pump

Number -- 2

Capacity, ea -- 250 gpm

Total disc. head -- 200 ft of H₂O at 70°F

Motor:

Rating -- 20 hp

Type -- 480 V, 3 ϕ , 60 cycle

rpm -- 3500

Weight -- 800 lb

Materials -- stainless steel, Type 316

Piping:

All piping to the upstream side of the heat-exchanger level-control valves

Type -- seamless or welded

Wall thickness -- Schedule 40

Material -- stainless steel

All other piping:

Type -- seamless

Wall thickness -- Schedule 40

Material -- carbon steel

4.6 Auxiliary Equipment Water-Treatment System:

Softener:

Type -- sodium zeolite

Capacity/cycle -- 50,000 gal

Flow rate (max) -- 90 gpm

Deionizer:

Type -- mixed bed

Capacity/cycle -- 20,000 gal

Flow rate (max) -- 100 gpm

Effluent water resistivity -- 1.0 + megohm-cm

Deionized Water Storage Tank:

Construction -- carbon steel, rubber-lined, shallow-dished heads

Capacity -- 12,000 gal

Outside diameter -- 108 in.

Length straight section -- 300 in.

Raw-Water Cooling System:

Cooling Tower:

Type -- double-flow, forced-draft evaporative

Water flow rate -- 300 gpm

Fan motor -- 5 hp, 480 V, 3 ϕ

Overall Dimensions:

Length -- 10 ft 0 in.

Width -- 8 ft 6 in.

Pump:

Type -- close coupled, vertical turbine pump

Number -- 1

Capacity -- 300 gpm

Total disc. head -- 60 ft H₂O

Motor:

Type -- 480 V, 3 ϕ

Rating -- 7-1/2 hp

rpm -- 1770

APPENDIX B
STATIC CHARACTERISTICS

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APPENDIX B

STATIC CHARACTERISTICS

This section presents a brief summary of the C-19/52 core static characteristics which were determined in order to implement the analysis of the C-core transient test results. A detailed description of the original static tests is given in IDO-16586 [15]. Additional static experiments have been performed and reported in various Quarterly Technical Reports [16, 17, 18, 19, 20].

The measured reactivity coefficients of the Spert III C-19/52 core are summarized in Table B-1. The neutron flux distributions have been determined in the Spert III core as a function of temperature over the range from ambient to 600°F. The vertical neutron flux profiles for position S-11 are shown in Figure B-1. Figure B-2 is a cross section of the operational C-19/52 core illustrating the lattice configuration.

The power-measuring ion chambers were calibrated calorimetrically using the reactor as the heat source and also by flux-measuring techniques. The calorimetric calibrations were performed at temperatures of 200, 300, 400, and 500°F; however, the most extensive data were taken at 500°F. An indication of the temperature dependence of power-measuring chambers is shown in Figure B-3.

TABLE B-1

MEASURED STATIC CHARACTERISTICS OF C-19/52 CORE

Mass of U-235	
Control rod follower fuel assemblies (8)	3.3 kg
Fixed fuel assemblies (44)	27.3 kg
Total	30.6 kg
Control Rod Critical Positions (70°F)	
4 rods at	10.2 in.
1 rod at 0 in., 3 rods at	11.7 in.
2 rods at 0 in., 2 rods (diametrically opposed) at	15.9 in.
2 rods at 0 in., 2 rods (adjacent quadrants) at	13.8 in.
3 rods at 0 in., 1 rod at	21.8 in.
Pressure Coefficient of Reactivity (measured at ambient temperature)	+0.014¢/psi
Pressure Coefficient of Reactivity (estimated at 500°F)	+0.044¢/psi
Isothermal Temperature Coefficient of Reactivity (at 2500 psig)	
100°F	- 1.0¢/°F
400°F	- 3.2¢/°F
500°F	- 4.6¢/°F
600°F	- 8.2¢/°F
650°F	-16.0¢/°F
Water Density Coefficient of Reactivity	-44.0¢/ % decrease in moderator density
Temperature Defect (80°F to 650°F)	18\$
Clean Core Excess Reactivity	
80°F	20.5\$
650°F	2.5\$
Thermal-Hydraulic Core Characteristics for Conditions of 500°F Inlet Temperature, 2500-psig Coolant Pres- sure, 20,000 gpm Coolant Flow, 30-MW Power Level	
Neutron flux peaking factors:	
Axial peak-to-average ratio	2.04
Radial peak-to-average ratio	1.30
Heat fluxes:	
Maximum core heat flux	2.6×10^5 Btu/hr-ft ²
Average core heat flux	9.72×10^4 Btu/hr-ft ²
Mass flow rate:	
Total system flow	3.29×10^6 lb/hr-ft ²
Flow through core (95% total)	3.1×10^6 lb/hr-ft ²
Reduced Prompt Neutron Lifetime, ℓ/β_{eff}	2.5 msec

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Fig. B-1 Spartan III lattice diagram.

Fig. B-2 Relative neutron distribution versus distance at various temperatures for Lattice Position S-11.

Fig. B-3 Nuclear chamber constant versus system temperature.

APPENDIX C
DATA SUMMARY -- LOW INITIAL POWER TESTS

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APPENDIX C -- DATA SUMMARY -- LOW INITIAL POWER TESTS

Appendix C summarizes the experimental data from the low initial power excursion tests performed with the C-19/52 core in Spert III. Tabular data for each run are given in Table C-1. Time plots of the reactor power, energy release, reactivity, and fuel plate surface temperature are included in Figures C-1 through -40, for a representative portion of the tests. As may be seen from Table C-1 these plots represent only a small fraction of the total number of tests performed. The temperature data presented are those obtained from the highest reading thermocouple in each test. The time plots of temperature behavior show temperature rise rather than absolute temperature. Although some earlier ramp-initiated tests are not shown, Table C-1 is otherwise complete with the exception of data presented in Appendix D.

The nomenclature used in Table C-1 is as follows:

- E_T = the total system energy release (MW-sec)
- $E(t_m)$ = the energy release to the time of maximum power (MW-sec)
- G = the coolant flow rate (gal/min)
- P = the system pressure (psig)
- P_i = the initial system pressure (psig)
- $P(\max)$ = the maximum measured transient pressure (psig)
- $\rho_c(t)$ = the reactivity compensated at time t with respect to transient initiation (\$)
- $\rho_c(t_m)$ = the reactivity compensated at time of maximum power (also $K_c t_m$)
- ρ_o = the reactivity insertion (\$)
- α_o = the initial inverse reactor period (sec^{-1}) (also referred to as reciprocal period)
- α = the instantaneous inverse reactor period (sec^{-1})
- θ_i = the initial system temperature ($^{\circ}\text{C}$ or $^{\circ}\text{F}$)
- $\theta(\max)$ = the maximum recorded fuel surface temperature ($^{\circ}\text{C}$)
- θ_{Sat} = the saturation temperature ($^{\circ}\text{F}$)
- $\theta(t_m)$ = the maximum recorded fuel plate surface temperature at the time of maximum power ($^{\circ}\text{C}$)
- ℓ/β_{eff} = the reduced prompt neutron lifetime (sec)
- τ_o = the initial reactor period (sec or msec)

$\phi(t)$ = the total core power at time t (W or MW)

$\phi(t_m)$ = the maximum reactor power (initial burst) (MW)

The identification number is a run number for Transients 1 to 82, the remaining numbers are oscillograph identification numbers.

TABLE C-1

COMBINED SUMMARY OF DATA FROM TRANSIENT
TEST SERIES SPERT III C-19/52 CORE

Identification Number	P _i (psi)	θ _i (°C)	τ _o (msec)	p(0) (\$)	Flow (gpm)	θ _{tm} (MW)	E _(tm) (MW-sec)	p _c (tm) (\$)	θ _{tm} Rise (°C)	θ _{max} Rise (°C)	P _{tm} (psi)	P _{max} (psi)
4242	Atm	18	22	1.10	0	160	8.37	-0.12	134	154	---	---
2	Atm	18	174	0.96	0	7.78	3.50	-0.10	55	106	---	---
1	Atm	18	504	0.87	0	3.39	5.01	-0.18	38	89	---	---
9	Atm	19	20.8	1.14	0	215	6.97	-0.17	101	106[e]	21.0[c]	21.0[c]
10	Atm	19	39.1	1.06	0	60.4	4.38	-0.09	69	89[e]	---	---
11	Atm	19	104	0.99	0	12.7	2.81	-0.07	41	94[e]	---	---
3568	Atm	21	40	1.05	0	62	4.39	-0.08	61	98	---	---
3169	Atm	21	93.5	0.98	0	11.8	2.42	-0.06	45	90	---	---
3171	Atm	21	120	1.15	0	8.61	2.43	-0.07	No TC's connected for this run			
3570	Atm	21	160	0.96	0	6.72	2.64	-0.09	45	95	---	---
3548	Atm	22	11.2	1.20	0	770	12.9	-0.23	205	226	---	---
3546	Atm	22	18.6	1.11	0	277	8.39	-0.13	146	157	---	---
1859	Atm	22	22.7	1.08	0	185	7	-0.10	125	127	---	7
2233	Atm	23	10.9	1.22	0	650	10.2	-0.26	175	201	---	---
1863	Atm	23	11.2	1.22	0	802.6	10.8	-0.26	127	140	---	15
1604	Atm	24	13.9	1.14	0	392	8.6	-0.16	142	163	---	12
2124	Atm	24	17.4	1.14	0	258	6.45	-0.16	138	155	---	---
2221	Atm	24	18	1.13	0	252	6.59	-0.15	150	162	---	---
3167	Atm	24	21.5	1.09	0	163	5.80	-0.11	109	110	---	---
1593	Atm	24	39	1.04	0	56.6	4.05	-0.07	80	101	---	---
1422	Atm	24	130	0.97	0	9	2.94	-0.08	41	83	---	---
1292	Atm	24	940	0.77	0	1.97	5.19	-0.20	56	74	---	---
4226	Atm	25	10.4	1.22	0	734	11.7	-0.28	207	319	---	---
1958	Atm	25	16.8	1.14	0	258	6.66	-0.18	113	135	---	---
4145	Atm	25	20.4	1.11	0	272	8.32	-0.13	134	179	---	---
4136	Atm	25	38	1.05	0	77.7	5.03	-0.08	97	109	---	---
4083	Atm	25	150	0.96	0	7.9	2.99	-0.09	40	89	---	---
1307	Atm	25	490	0.87	0	3.23	4.40	-0.17	48	84	---	---
1425	Atm	25	9300	0.41	0	478[b]	61.6[a]	-0.40[b]	38	38	---	---
1676	Atm	26	10.1	1.24	0	810	11.3	-0.29	201	239	---	8
2127	Atm	26	10.9	1.21	0	638	9.88	-0.28	190	216	---	---
1595	Atm	26	16.9	1.13	0	284	7.25	-0.15	56	186	---	7
1309	Atm	26	162	0.94	0	8.8	3.48	-0.07	46	84	---	---
1470	Atm	27	68.3	1.01	0	19	2.60	-0.06	41	81	---	---
1446	Atm	27	81.9	0.99	0	15.1	2.79	-0.05	41	80	---	---
1967	Atm	28	10.6	1.20	0	620	9.76	-0.23	178	206	---	11
2246	Atm	28	10.7	1.23	0	621.3	9.42	-0.26	180	210	---	---
2157	Atm	28	18	1.13	0	220	6.02	-0.15	107	122	---	---
3163	Atm	28	19.2	1.11	0	238	7.11	-0.12	104	117	---	---
1311	Atm	28	4800	0.49	0	422[b]	5.08[a]	-0.21[b]	36	36	---	---
4745	Atm	29	21	1.10	0	180	5.97	-0.13	119	120	---	---
4695	Atm	29	38.3	1.04	0	53	3.94	-0.07	75	105	---	---
3620	Atm	30	9.9	1.25	0	810	12.2	-0.29	202	226	---	---
1955	Atm	34	11.4	1.21	0	560	9.09	-0.26	172	202	---	14
4305	Atm	60	24.3	1.09	0	119	4.11	-0.11	81	81	---	---
4302	Atm	60	47.1	1.03	0	32.4	2.89	-0.07	44	67	---	---
4299	Atm	60	190	0.95	0	3.94	2.18	-0.11	31	50	---	---
4323	Atm	82	21.2	1.11	0	114	3.46	-0.15	59	63	---	---
4320	Atm	82	38	1.05	0	42.6	2.51	-0.09	31	36	---	---
4317	Atm	82	143	0.97	0	5.4	1.79	-0.08	23	31	---	---
4399	Atm	90	20	1.12	0	105	2.94	-0.17	56	61	---	6
4387	Atm	90	45.2	1.04	0	28.4	2.29	-0.09	41	42	---	---
3165	20	29	18.5	1.10	0	258	8.49	-0.12	120	132	---	---
3551	50	22	11.3	1.21	0	860	14.6	-0.24	219	244	---	73
3550	50	23	18.7	1.12	0	290	9.25	-0.13	167	178	---	20
4314	50	60	23.8	1.10	0	154	5.88	-0.12	118	125	---	8
4311	50	60	45.4	1.03	0	36.3	3.13	-0.06	48	101	---	---
4308	50	60	192	0.94	0	4.50	2.4	-0.11	36	71	---	---
4414	50	90	20.7	1.12	0	139	4.01	-0.14	93	102	---	---
4409	50	90	38.9	1.05	0	39.3	2.96	-0.08	54	82	---	7
4404	50	90	159	0.96	0	4.81	2.04	-0.10	26	62	---	---
4548	50	127	19.8	1.12	0	110	3.22	-0.14	46	57	---	---
4543	50	127	38.4	1.04	0	30.8	1.88	-0.06	40	45	---	---
4538	50	128	165	0.96	0	4.39	1.87	-0.10	21	37	---	---
2137	100	27	148	0.96	0	8.2	3.26	-0.09	46	101	---	---
2132	100	27	480	0.87	0	2.49	3.75	-0.19	50	89	---	---
4375	100	82	20.3	1.09	0	140	5.24	-0.11	120	127	---	---
4370	100	82	39.6	1.05	0	41.8	3.10	-0.07	58	99	---	6
4365	100	82	157	0.96	0	4.96	2.17	-0.10	31	67	---	8
4382	100	90	167	0.96	0	4.65	2.03	-0.10	26	30	---	---
4490	100	127	19.8	1.13	0	131	3.84	-0.16	74	79	---	---
4485	100	127	42.3	1.03	0	30.9	2.57	-0.06	36	58	---	---
4480	100	127	154	0.96	0	4.7	1.02	-0.09	50	51	---	---
4624	100	154	20.4	1.10	0	88	2.53	-0.13	38	45	---	---
4615	100	154	152	0.95	0	3.94	1.67	-0.09	19	27	---	---
4620	100	155	43.5	1.03	0	24.1	1.67	-0.07	24	30	---	---
4623	110	22	18.1	1.12	0	275	8.91	-0.13	162	196	---	---
2235	110	23	10.8	1.22	0	882	13.2	-0.26	285	312	---	---
1868	110	24	505	0.87	0	3.33	4.69	-0.17	53	90	---	---
2141	110	26	54.1	1.02	0	25.3	2.73	-0.06	53	136	---	---

TABLE C-1 (Contd.)

COMBINED SUMMARY OF DATA FROM TRANSIENT
TEST SERIES SPERT III C-19/52 CORE

Identification Number	P _i (psi)	θ _i (°C)	τ _o (msec)	p(0) (\$)	Flow (gpm)	θ _{tm} (MW)	E _(tm) (MW-sec)	p _c (tm) (\$)	θ _{tm} Rise (°C)	θ _{max} Rise (°C)	P _{tm} (psi)	P _{max} (psi)
2152	110	27	18.6	1.12	0	235	7.75	-0.13	179	201	---	---
1941	110	28	22.4	0.78	0	135	8.38	-0.19	138	178	---	---
2145	110	28	37.2	1.04	0	56.6	4.04	-0.07	82	148	---	---
1934	110	28	57.8	1.01	0	24.4	2.80	-0.06	52	131	---	---
1872	110	28	155	0.96	0	8.1	3.20	-0.09	40	111	---	---
1945	110	29	18.3	1.12	0	278	9.21	-0.14	193	209	---	---
2148	110	29	25.4	1.08	0	109	5.16	-0.11	112	164	---	---
1938	110	29	39.1	1.04	0	57.4	4.26	-0.08	81	148	---	---
1948	110	30	13.5	1.15	0	465	11.6	-0.20	224	238	---	---
1952	110	31	10.8	1.21	0	720	10.5	-0.27	272	303	---	---
1878	110	32	93.1	0.99	0	16	3.13	-0.06	42	130	---	---
3562	175	21	10.6	1.22	0	870	13.5	-0.25	246	279	---	80
3518	175	23	11.2	1.18	0	816	37.1	-0.19	260	260	---	---
2850	200	26	39.2	1.05	0	67.4	4.77	-0.08	66	126	---	---
3612	200	32	11.4	1.21	0	790	13.8	-0.23	273	286	---	---
4476	200	116	17.1	1.13	0	184	5.94	-0.15	110	125	---	---
5058	200	116	19.2	1.15	20,000	152	4.92	-0.16	50	65	---	---
25	200	116	20.4	1.14	10,000	139	4.88	-0.15	34	69	---	---
4471	200	116	47	1.03	0	24.8	2.30	-0.07	36	83	---	---
5053	200	116	65.8	1.01	20,000	61.2[b]	65.6[a]	-0.27[b]	53	55	---	104
23	200	116	66.5	1.02	10,000	16.1	67.5[a]	-0.07	16	78	9.7[c]	11.7[c]
24	200	116	78.3	1.01	0	11.7	1.894	-0.07	24	68	---	---
4465	200	116	190	0.947	0	3.47	1.84	-0.11	24	49	---	---
5048	200	116	195	0.94	20,000	20.5[b]	9.60[a]	-0.07[b]	16.9	19.7	---	24
26	200	116	224	0.94	10,000	31.1	167.7[a]	-0.26	45	70	---	---
22	200	116	347	0.90	10,000	31.9	562[a]	-0.23	41	88	---	42.5
4554	200	153	77.6	1.00	0	9.72	1.59	-0.06	20	54	---	---
4571	200	154	22.1	1.12	0	106	3.87	-0.15	62	70	---	---
4559	200	154	41.3	1.05	0	29.6	2.32	-0.08	34	54	---	---
4575	200	154	175	0.95	0	3.90	1.70	-0.09	22	45	---	---
4565	200	155	155	---	0	---	---	---	---	---	---	---
4770	200	185	21.7	1.11	0	66.8	1.91	-0.15	30	32	---	---
4765	200	185	31.3	1.06	0	35.9	1.63	-0.10	25	27	---	---
4775	200	185	175	0.95	0	3.11	1.35	-0.09	17	20	---	---
3093	220	28	310	0.89	5,000	24.7[b]	64.7[a]	-0.31[b]	113	114	---	---
3096	220	31	516	0.86	200	2.48	4.45	-0.214	48	98	---	---
3040	225	31	155	0.95	5,000	17.1[b]	17.4[a]	-0.18[b]	85	85	---	35
3101	230	27	485	0.85	1,000	4.88[b]	15.4[a]	-0.32[b]	126	120	---	---
2982	230	28	10.1	1.24	5,000	1067	15.5	-0.30	324	387	---	52
3050	230	28	300	0.91	0	3.06	3.32	-0.14	52	46	---	---
3091	230	28	313	0.89	1,000	7[b]	20.3[a]	-0.30[b]	123	124	---	90
3049	230	28	314	0.91	500	3.92	3.79	-0.15	57	112	---	---
3099	230	28	525	0.85	0	2.51	3.22	-0.17	50	86	---	---
2993	230	29	37.1	1.04	2,000	7.71	4.27	-0.07	70	113	---	13
3047	230	29	155	0.96	500	6.46	2.50	-0.09	44	113	---	---
3105	230	29	157	0.94	2,000	8.1[b]	11.4[a]	-0.24[b]	89	96	---	17
3098	230	29	510	0.85	500	3.5[b]	19[a]	-0.43[b]	111	113	---	---
2968	230	31	21.9	1.10	0	200	7.82	-0.11	163	232	---	35
3103	230	31	25.3	1.09	20,000	142	6.20	-0.11	79	106	---	---
2900	230	31	39.4	1.04	1,000	54.5	4.11	-0.08	56	105	---	9
2941	230	31	40.8	1.04	0	42	3.30	-0.07	66	140	---	9
2906	230	31	40.9	1.04	5,000	46	3.76	-0.08	56	81	---	---
3597	230	31	500	0.87	20,000	27[b]	25[a]	-0.07[b]	22	22	---	---
3608	230	32	10.9	1.20	20,000	730	15.2	-0.22	265	308	---	---
2972	230	32	21.3	1.11	5,000	202	7.45	-0.02	148	204	---	38
2980	230	32	22.9	1.09	0	192	7.85	-0.11	145	209	---	---
2914	230	32	39.3	1.05	0	51.8	3.98	-0.08	70	149	---	---
2908	230	32	40.4	1.05	10,000	45.9	3.69	-0.08	50	125	---	---
3616	230	32	511	0.83	20,000	152[b]	713[a]	-0.34[b]	136	136	---	66
2994	230	33	9.7	1.23	0	1030	17.5	-0.27	366	366	---	---
3614	230	33	21.5	1.10	5,000	164	4.50	-0.11	120	174	---	---
3045	230	33	157	0.95	0	6.9	2.99	-0.09	38	90	---	---
3044	230	33	164	0.95	2,000	8.2	13.1[a]	-0.24[b]	60	91	---	---
3037	230	33	186	0.90	10,000	43[b]	63.2[a]	-0.22[b]	105	105	---	---
2974	230	34	21.9	1.09	10,000	164	6.66	-0.11	112	150	---	---
3618	230	34	40.4	1.04	20,000	51	4.05	-0.07	44	154	---	---
2909	230	34	52.6	0.78	0	30.1	3.60	-0.17	54	127	---	21
3038	230	34	163	0.94	0	7	0.90	-0.07	37	103	---	---
2987	230	35	9.7	1.21	0	950	12.7	-0.23	294	368	---	---
3610	230	35	10.6	1.24	5,000	805	13.2	-0.26	263	282	---	60
2911	230	36	37	1.04	20,000	58	4.64	-0.07	68	150	---	---
28	230	182	260	0.93	10,000	21.8[b]	27.8[a]	-0.21	---	20	---	28.4
29	230	182	337	0.90	20,000	23.3[b]	19.6[a]	-0.10	---	17	---	20.3
27	230	183	294	0.92	0	2.44	2.06	-0.15	18	23	---	---
2985	235	33	9.7	1.25	10,000	1051	14.9	-0.30	293	392	---	49
3577	250	23	10.9	1.21	0	820	15.1	-0.24	273	285	---	---
3557	250	25	18.3	1.12	0	275	9.46	-0.14	182	222	---	21
3350	250	26	10.9	1.23	0	933	14.8	-0.31	267	273	---	65
3042	250	33	166	0.93	20,000	63	58.1[a]	-0.16[b]	82	82	---	70

TABLE C-1 (Contd.)

COMBINED SUMMARY OF DATA FROM TRANSIENT
TEST SERIES SPERT III C-19/52 CORE

Identification Number	P _i (psi)	θ _i (°C)	τ _o (msec)	p(0) (\$)	Flow (gpm)	Q _{tm} (MW)	E _(tm) (MW-sec)	p _c (tm) (\$)	θ _{tm} Rise (°C)	θ _{max} Rise (°C)	P _{tm} (psi)	P _{max} (psi)
2977	255	31	22	0.78	20,000	187	9.71	-0.19	108	140	---	60
2990	255	33	9.9	1.23	20,000	971	16.2	-0.25	236	369	---	---
34	255	183	99.1	0.99	20,000	40.2[b]	28.4[a]	-0.20	---	31	---	39.1
35	260	183	67	1.01	20,000	49.5[b]	52.5[a]	-0.26[b]	---	34	---	33.3
33	260	183	117	0.98	20,000	33.7[b]	21.2[a]	-0.15[b]	---	29	---	28.3
36	260	184	22.9	1.10	20,000	84.9	3.47	-0.12	22	37	5.8	22.8
2853	270	26	44.9	1.04	200	41.4	3.54	-0.07	52	137	---	9
2325	500	21	17.9	1.17	0	278	9.15	-0.13	167	248	---	---
1961	500	25	16.8	1.12	0	299	9.42	-0.13	213	272	---	---
1964	500	28	11.7	1.21	0	870	15.4	-0.24	328	356	---	---
3560	500	29	10.7	1.21	0	900	16.6	-0.24	283	297	---	60
4691	500	29	171	0.95	0	6.39	3	-0.11	27	62	---	---
4755	500	162	44.7	1.04	0	25.3	2.16	-0.07	31	70	---	---
4633	500	162	118	0.96	0	5.11	1.52	-0.06	19	45	---	---
4760	500	163	17.4	1.11	0	185	5.59	-0.15	90	105	---	---
4750	500	163	150	0.95	0	4.24	1.63	-0.07	16	40	---	---
37	500	183	31.1	1.07	20,000	47.8	2.81	-0.09	13[c]	52	---	31.2[c]
4823	500	203	62.6	1.02	0	10.1	1.25	-0.06	20	44	---	---
4952	500	204	170	0.96	0	2.81	1.38	-0.11	17	34	---	---
4827	500	205	22.9	1.10	0	79	2.37	-0.12	47	53	---	16
4947	500	205	37.4	1.05	0	26.2	1.97	-0.08	35	46	---	16
41	550	235	26.3	1.08	500	53.9	1.93	-0.13	15[c]	19	2.3	13.4
40	550	239	46.3	1.04	500	17.7	1.33	-0.08	12[c]	15[c]	---	16.5
43	570	231	14.7	1.16	20,000	161	4.39	-0.20	31	32	13.6	46.5
42	570	233	29.4	1.07	20,000	40.1	2.36	-0.10	10[c]	23	15.8	47.3
39	585	233	147	0.96	20,000	39.6[b]	48.9[a]	-0.24	---	24	7.5	30.3
38	585	233	394	0.89	20,000	588[h]	0.28[a]	-0.03[b]	[c]	[c]	13	30.3
2227	1000	23	18.5	1.10	0	269	9.06	-0.12	202	300	---	20
2064	1000	24	16.4	1.12	0	329	10.4	-0.13	216	301	---	---
2237	1000	25	11	1.21	0	910	14.5	-0.26	328	370	---	---
2067	1000	27	10.6	1.22	0	950	15.2	-0.25	348	394	---	---
4961	1000	204	19	1.11	0	120	4.48	-0.12	88	99	---	17
4956	1000	204	40.4	1.04	0	20.5	1.72	-0.07	28	60	---	8
4966	1000	204	148	0.96	0	2.99	1.24	-0.10	17	34	---	12
46	1000	246	22.1	1.10	20,000	64.9	2.67	-0.12	26	57	14.1	46.1
47	1000	246	36.3	1.05	500	34.3	2.18	-0.08	25	43	---	18.3
45	1000	246	42.4	1.04	20,000	21.4	1.71	-0.08	11[c]	38	12.3	49.8
44	1000	246	105	0.98	20,000	48.1[b]	85.2[a]	-0.32	---	39	8.2	45.2
51	1000	280	46.1	1.04	500	16.6	3.61	-0.07	10[c]	11[c]	---	9.2
50	1050	278	18.4	1.13	20,000	83.5	2.70	-0.15	10[c]	30	---	37.3
48	1050	278	211	0.94	20,000	22.1[b]	21.8[a]	-0.19[b]	---	17	4.3	28.7
49	1052	278	42.3	1.04	20,000	1.74	1.42	-0.08	[d]	16[c]	6.4	36.6
71	1775	316	64.3	1.01	10,000	6.32	0.7	-0.06	[d]	12[c]	---	15.7
68	1775	317	19.4	1.12	20,000	43.8	1.80	-0.14	9[c]	13[c]	---	28.4
69	1775	317	109	0.98	20,000	19[b]	21.1[a]	-0.24[b]	[c]	11[c]	---	18.2
70	1775	317	34.8	1.06	1,000	26.4	1.50	-0.10	9[c]	12[c]	---	11.2[c]
66	1775	322	43.3	1.04	20,000	11.7	1.06	-0.07	[d]	13[c]	8.6	33.5
67	1775	323	154	0.96	20,000	11.8[b]	10.1[a]	-0.16[b]	---	8[c]	10.8	21.7
56	1780	318	278	0.92	20,000	5[b]	2.77[a]	-0.07[b]	---	[d]	1.8	13
2117	1900	21	17.9	1.11	0	254	8.23	-0.13	202	320	---	---
2230	1900	23	17.4	1.14	0	299	9.28	-0.15	208	333	---	20
2121	1900	24	11.5	1.21	0	800	14.1	-0.23	373	439	---	---
2240	1900	25	11	1.23	0	972	16.5	-0.26	296	415	---	---
76	1900	297	35.2	1.06	10,000	19.3	1.31	-0.08	[d]	23	---	23.2
79	1960	297	75.7	1.00	20,000	28.8[b]	30.3[a]	-0.26[b]	---	25	---	24.5
75	1975	296	55.7	1.02	20,000	28.9[b]	32.4[a]	-0.52[b]	---	28	9.4	42.5
74	1975	297	19.3	1.12	20,000	79.7	2.63	-0.14	15[c]	32	9.4	37.8
73	1975	297	30	1.07	20,000	27.9	1.51	-0.09	9[c]	30	7.2	24.8
72	1975	297	207	0.94	20,000	15.8[b]	11.6[a]	-0.15	---	14[c]	11.2	26.1
77	1975	298	151	0.96	1,000	2.46	0.97	-0.10	[d]	10[c]	---	---
78	1975	300	50.4	1.03	1,000	14.7	1.35	-0.07	9[c]	26	---	14.8
3575	2000	23	10.2	1.21	0	941	15.3	-0.26	348	407	---	90
59	2050	264	19.1	1.12	20,000	73.6	2.67	-0.14	20	51	---	36.9
58	2050	264	41.7	1.05	20,000	19.2	1.76	-0.09	8[c]	62	29	65.3
57	2050	264	82.6	1.00	20,000	44.5[b]	57.4[a]	-0.30[b]	---	45	7	13.9
60	2050	267	55.3	1.02	500	13.2	1.37	-0.06	12[c]	40	---	34.3
65	2450	317	40	1.05	700	19.1	1.54	-0.08	12[c]	33	---	12.8[c]
16	2500	16	90.4	1.00	0	15.4	2.71	-0.07	44	125[e]	---	10.9[c]
13	2500	17	50	1.04	0	35.8	3.52	-0.08	56	146[e]	---	11[c]
12	2500	17	81.7	1.00	0	17.9	2.84	-0.06	38	155	---	11.3[c]
15	2500	17	83.5	1.00	0	17.2	2.94	-0.07	35	124	---	---
14	2500	17	124	0.98	0	10.3	2.85	-0.08	41	117[e]	---	---
30	2500	17	128	0.97	0	10.3	3.13	-0.05	59	113	---	19.9[c]
8	2500	18	20.7	1.14	0	231	8.42	-0.15	142	278[e]	---	20.1
18	2500	18	22.4	1.12	0	234	8.50	-0.14	148	264	---	19.6
17	2500	18	31.7	1.08	0	91.5	5.39	-0.11	87	177	---	---
6	2500	18	36.1	1.07	0	69	4.50	-0.10	78	165	---	18.6
4	2500	18	154	0.96	0	7.88	3.13	-0.10	47	113	---	14[c]
31	2500	19	19	1.12	0	252	9.59	-0.14	96	312	---	44.8
19	2500	19	20.3	1.14	0	242	8.60	-0.16	150	277[e]	---	19[c]

TABLE C-1 (Contd.)

COMBINED SUMMARY OF DATA FROM TRANSIENT
TEST SERIES SPERT III C-19/52 CORE

Identification Number	P _i (psi)	θ _i (°C)	τ _o (msec)	p(0) (%)	Flow (gpm)	Q _{tm} (MW)	E _(tm) (MW-sec)	Q _c (tm) (%)	θ _{tm} Rise (°C)	θ _{max} Rise (°C)	P _{tm} (psi)	P _{max} (psi)
20	2500	19	29	1.09	0	119	6.08	-0.11	93	190	---	---
21	2500	19	43	1.05	0	49.6	4.01	-0.09	48	144	---	13.8[c]
5	2500	19	87.8	1.00	0	14.5	2.66	-0.07	45	124[e]	---	---
3	2500	19	467	0.87	0	3.46	4.45	-0.17	46	87	---	---
7	2500	20	19.6	1.12	0	241	8.54	-0.16	144	277[e]	10.0[a]	24.2
3566	2500	21	10.7	1.22	0	1130	21.4	-0.23	72	113	---	40
3564	2500	21	18.2	1.10	0	304	10	-0.11	208	341	---	17
32	2500	21	19.0	1.12	0	249	9.31	-0.14	96	304	---	42.5
3573	2500	23	36.3	1.04	0	68.8	4.76	-0.07	40	77	---	9
3572	2500	23	156	0.95	0	8.3	3.12	-0.08	46	112	---	---
2244	2500	27	11	1.22	0	949	17.6	-0.24	397	552	---	---
3112	2500	28	125	0.97	0	7.10	2.28	-0.08	74	212	---	---
3111	2500	31	134	0.96	2,000	8.77	11.9[a]	-0.27[b]	175	176	---	---
3107	2500	32	22.1	1.10	20,000	179	6.80	-0.11	131	174	---	60
3115	2500	33	9.6	1.23	20,000	1057	18	-0.25	437	550	---	90
3109	2500	33	19.8	1.10	0	222	8.09	-0.12	244	426	---	44
5040	2500	101	20.1	1.11	0	201	7.39	-0.13	133	225	---	---
5044	2500	101	25.9	1.09	16,000	90.3	4.01	-0.11	38	49	---	---
5036	2500	101	43.9	1.03	0	27.7	2.59	-0.06	36	84	---	---
5028	2500	205	19.2	1.12	0	137	4.74	-0.14	103	163	---	---
5032	2500	205	172	0.95	0	2.66	1.23	-0.09	21	37	---	---
5021	2500	206	34.8	1.05	0	35	2.48	-0.19	40	82	---	---
82	2500	261	44.7	---	20,000	16.3	---	---	9	42	9.2	31
80	2500	261	50.4	1.03	20,000	15.7	2.09	-0.08	12[c]	30	10.6	34.9
81	2500	261	175	---	20,000	27.8[b]	---	---	---	29	13	40
54	2500	278	42.2	1.04	20,000	16.8	1.53	-0.08	10[c]	47	5.3	56.8
53	2500	278	86.5	1.00	20,000	34.2[b]	49.7[a]	-0.28[b]	---	42	5.2	51.6
55	2500	279	26.5	1.08	20,000	32.9	1.81	-0.11	9[c]	42	11.6	59.8
52	2500	284	37.8	1.05	500	28.1	1.99	-0.08	25	46	---	19.4
61	2500	322	140	0.96	20,000	10.9[b]	6.53[a]	-0.14[b]	---	10[c]	---	24.5[c]
63	2500	323	29.3	1.06	20,000	19.8	1.22	-0.08	7[c]	20	[d]	24.1
62	2500	323	82.6	1.00	20,000	17.1[b]	16.1[a]	-0.21[b]	---	17	7.4	36.4
64	2500	324	22.2	1.10	20,000	40.2	1.64	-0.12	10[c]	26	[d]	41.3

[a] Indicates total burst energy.

[b] Indicates maximum measured value where no peak occurred.

[c] Indicates oscillograph deflection was between 5 and 10 mm (best obtainable accuracy).

[d] Indicates oscillograph deflection was less than 5 mm (questionable accuracy).

[e] Indicates temperatures were measured with 0.5 Al-Cr TC's for Run 1-82. All other temperatures for Run 1-82 were measured with SS sheathed TC's.

Fig. C-1 Run 1, 504 msec period; log power, energy, reactivity versus time.

Fig. C-2 Run 1, 504 msec period; linear power, temperature versus time.

Fig. C-3 Run 2, 174 msec period; log power, energy, reactivity versus time.

Fig. C-4 Run 2, 174 msec period; linear power, temperature versus time.

Fig. C-5 Run 3, 468 msec period; log power, energy, reactivity versus time.

Fig. C-6 Run 3, 468 msec period; linear power, temperature versus time.

Fig. C-7 Run 4, 154 msec period; log power, energy, reactivity versus time.

Fig. C-8 Run 4, 154 msec period; linear power, temperature versus time.

Fig. C-9 Run 5, 88 msec period; log power, energy, reactivity versus time.

Fig. C-10 Run 5, 88 msec period; linear power, temperature versus time.

Fig. C-11 Run 7, 19.6 msec period; log power, energy, reactivity versus time.

Fig. C-12 Run 7, 19.6 msec period; linear power, temperature versus time.

Fig. C-13 Run 9, 20.8 msec period; log power, energy, reactivity versus time.

Fig. C-14 Run 9, 20.8 msec period; linear power, temperature versus time.

Fig. C-15 Run 11, 104 msec period; log power, energy, reactivity versus time.

Fig. C-16 Run 11, 104 msec period; linear power, temperature versus time.

Fig. C-17 Run 33, 117 msec period; log power, energy, reactivity versus time.

Fig. C-18 Run 33, 117 msec period; linear power, temperature versus time.

Fig. C-19 Run 36, 22.9 msec period; log power, energy, reactivity versus time.

Fig. C-20 Run 36, 22.9 msec period; linear power, temperature versus time.

Fig. C-21 Run 37, 31.1 msec period; log power, energy, reactivity versus time.

Fig. C-22 Run 37, 31.1 msec period; linear power, temperature versus time.

Fig. C-25 Run 49, 42.3 msec period; log power, energy, reactivity versus time.

Fig. C-26 Run 49, 42.3 msec period; linear power, temperature versus time.

Fig. C-27 Run 50, 18.4 msec period; log power, energy, reactivity versus time.

Fig. C-28 Run 50, 18.4 msec period; linear power, temperature versus time.

Fig. C-29 Run 61, 140 msec period; log power, energy, reactivity versus time.

Fig. C-30 Run 61, 140 msec period; linear power, temperature versus time.

Fig. C-31 Run 63, 29.3 msec period; log power, energy, reactivity versus time.

Fig. C-32 Run 63, 29.3 msec period; linear power, temperature versus time.

Fig. C-33 Run 64, 22.2 msec period; log power, energy, reactivity versus time.

Fig. C-34 Run 64, 22.2 msec period; linear power, temperature versus time.

Fig. C-35 Run 66, 43.4 msec period; log power, energy, reactivity versus time.

Fig. C-36 Run 66, 43.4 msec period; linear power, temperature versus time.

Fig. C-39 Run 76, 35.2 msec period; log power, energy, reactivity versus time.

Fig. C-40 Run 76, 35.2 msec period; linear power, temperature versus time.

APPENDIX D
DATA SUMMARY -- HIGH INITIAL POWER TESTS

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APPENDIX D -- DATA SUMMARY -- HIGH INITIAL POWER TESTS

Presented in this appendix is a summary of the experimental data obtained during the high power transient test series. Included are tabular data (Table D-1) summarizing the test results of each test in terms of transient data and run-out data. The transient data are those data associated with the initial power peak, where such peak occurred, while run-out values are those obtained 1.2 seconds after the initiation of the transient unless otherwise specified.

Representative time histories of the power, temperature, and energy generated are included in Figures D-1 through D-23.

TABLE D-1

SUMMARY OF EXPERIMENTAL DATA (High Initial Power)

Step Transient Tests														
Initial Conditions						Transient Values								
Identification Numbers		Power Level	Reactivity Insertion	Core Inlet Temperature	Maximum Plate Temperature	Nominal Channel Outlet Temperature	Flow	Pressure	Peak Power	Energy	Power	Excess Reactivity	Maximum Plate Temperature Rise	Nominal Channel Outlet Temperature Rise
Run No.	Transient No.	(MW)	(\$)	(°C)	(°C)	(°C)	(gpm)	(psig)	(MW)	(MW-sec)	(MW)	(\$)	(°C)	(°C)
1. $\approx 0.7\%$ Insertion, Variable Initial Power														
92	3-06706	0.01	0.69	260	265	261	20,000	2,500	---	---	0.04 ^[b]	0.68 ^[b]	7 ^[c]	7 ^[c]
94	3-06720	0.01	0.79	260	260	260	20,000	2,500	---	---	0.9 ^[b]	0.63 ^[b]	6 ^[c]	6 ^[c]
112	3-06938	0.89	0.72	260	260	260	20,000	2,500	---	---	8.3 ^[b]	0.55	6	6
2. $\approx 1.0\%$ Insertion, Variable Initial Power														
57	3-06352	1×10^{-5}	1.00	260	263	262	20,000	2,050	---	0.7	11.9 ^[b]	0.78	13	7
93	3-06713	0.01	1.01	260	260	260	20,000	2,500	---	1.09	30.3 ^[b]	0.78	30	5
95	3-06723	0.09	0.98	260	260	260	20,000	2,500	---	1.06	32.3 ^[b]	0.77	34	5
106	3-06879	0.98	0.99	260	260	260	20,000	2,500	24.5	1.04	35.5	0.69	44	7
104	3-06863	9.5	0.97	260	273	262	20,000	2,500	76.38	2.98	68.0	0.61 ^[e]	70 ^[r]	7 ^[r]
59	3-06360	1×10^{-5}	1.12	260	262	263	20,000	2,050	74.4	2.76	51.1 ^[d]	0.75 ^[e]	55 ^[r]	21 ^[r]
113	3-06946	28.0	0.96	260	260	272	20,000	2,500	172.9	6.58	102.8	0.51	113	25
3. Coolant Flow Variable, $\approx 1.0\%$ from ≈ 1 MW														
106	3-06879	0.98	0.99	260	263	260	20,000	2,500	24.5	1.04	35.5	0.69	44	7
105	3-06871	0.97	0.97	260	273	264	16,850	2,500	20.1	0.90	27.8	0.64	50	12
109	3-06913	1.03	0.99	260	263	260	10,000	2,500	24.2	1.0	20.6	0.62	37	9
114	3-06954	0.65	0.97	260	276	268	5,000	2,500	20.1	0.98	11.9	0.56	45	15
4. Comparative Transients, $\approx 1.0\%$ from ≈ 1 MW														
96	3-06723	0.96	0.99	260	272	266	20,000	2,500	18.2	0.89	42.6	0.71 ^[h]	49	12
100	3-06831	0.98	0.95	260	262	260	20,000	2,500	17.7	0.90	5.46	0.9 ^[h]	---	---
110	3-06921	0.97	0.97	260	260	260	20,000	2,500	18.5	0.81	34.0	0.65	39	7
Loss of Flow Tests														
Initial Conditions						Transient Values ^[b]								
Identification Numbers		Power Level	Core Inlet Temperature	Maximum Plate Temperature	Nominal Channel Outlet Temperature	Flow	Pressure	Power	Maximum Plate Temperature Rise	Nominal Channel Outlet Temperature Rise	Flow			
Run No.	Transient No.	(MW)	(°C)	(°C)	(°C)	(gpm)	(psig)	(MW)	(°C)	(°C)	(gpm)			
1. Partial Loss of Flow														
107	3-06887	9.9	260	276	262	20,000	2,500	9.2	17	4	$\approx 14,400$			
108	3-06904	26.7	260	275	266	20,000	2,500	24.7	42	7	$\approx 14,400$			
2. Total Loss of Flow														
111	3-06929	10.0	260	275	263	20,000	2,500	6.8	32	8	$\approx 1,400$			
117	3-07020	27.5	260	298	269	20,000	2,900	16.1	72	21	$\approx 1,400$			

TABLE D-1 (Contd.)

SUMMARY OF EXPERIMENTAL DATA
(High Initial Power)

Ramp Reactivity Insertion Tests												
Initial Conditions								Transient Values ^[j]				
Identification Numbers		Power (MW)	Reactivity Insertion Rate (\$1 sec)	Total (\$)	Core Inlet Temperature (°C)	Maximum Plate Temperature (°C)	Nominal Channel Outlet Temperature (°C)	Flow (gpm)	Pressure (psia)	Power (MW)	Temperature Rise (°C)	Nominal Channel Outlet Temperature Rise (°C)
Run No.	Transient No.											
99	3-06821	0.91	0.053	0.27	260	266	263	20,000	2,500	1.7	None Detected	None Detected
102	3-06846	0.75	0.30	0.27	260	265	263	20,000	2,500	1.7	None Detected	None Detected
101	3-06836	24.50	0.30	0.28	260	293	263	20,000	2,500	39.2	51	8

[a] Data taken 1.2 seconds after the occurrence of peak power.

[b] No peak in power, deflection taken 1.2 sec after start of transient.

[c] No detectable change.

[d] Taken at 0.058 seconds before control rod seat.

[e] Extrapolated, power scrambled 1.09 sec after peak power.

[f] Taken just before scram.

[g] Trace oscillating, data taken at high point of oscillation.

[h] Scrammed 0.84 seconds after peak power.

[i] All transient values taken 4.0 seconds after pump trip.

[j] All transient values taken 7 seconds after start of transient.

Fig. D-1 Run 57, 1.0\$ step from 10 W; linear power, temperature, energy.

Fig. D-2 Run 59, 1.12\$ step from 10 W; linear power, temperature, energy.

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Fig. D-3 Run 92, 0.70\$ step from 10 kW; linear power and energy.

Fig. D-4 Run 93, 1.01\$ step from 10 kW; linear power, temperature, energy.

Fig. D-5 Run 94, 0.79\$ step from 100 kW; linear power, temperature, energy.

Fig. D-6 Run 95, 0.98\$ step from 90 kW; linear power, temperature, energy.

Fig. D-7 Run 96, 0.99\$ step from 1 MW; linear power, temperature, energy.

Fig. D-8 Run 99, 0.27\$ ramp from 0.91 MW; linear power, temperature, energy.

Fig. D-9 Run 100, 0.95\$ step from 1 MW; linear power, temperature, energy.

Fig. D-10 Run 101, 0.28\$ ramp from 25 MW; linear power, temperature, energy.

Fig. D-11 Run 102, 0.27\$ ramp from 0.75 MW; linear power, temperature, energy.

Fig. D-12 Run 104, 0.97\$ step from 9.5 MW; linear power, temperature, energy.

Fig. D-13 Run 105, 0.97% step from 1.0 MW; linear power, temperature, energy.

Fig. D-14 Run 106, 0.99% step from 1 MW; linear power, temperature, energy.

Fig. D-15 Run 107, partial loss of flow from 9.9 MW; linear power, temperature, energy.

Fig. D-16 Run 108, partial loss of flow from 26.7 MW; linear power, temperature, energy.

Fig. D-17 Run 109, 0.99\$ step from 1 MW; linear power, temperature, energy.

Fig. D-18 Run 110, 0.97\$ step from 1 MW; linear power, temperature, energy.

Fig. D-19 Run 111, total loss of flow from 10 MW; linear power, temperature, energy.

Fig. D-20 Run 112, 0.72\$ step from 1 MW; linear power, temperature, energy.

Fig. D-21 Run 113, 0.96\$ step from 28 MW; linear power, temperature, energy.

Fig. D-22 Run 114, 0.97\$ step from 0.65 MW; linear power, temperature, energy.

Fig. D-23 Run 117, total loss of flow from 27.5 MW; linear power, temperature, energy.

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