

Reactor excursion behavior *

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Information accumulated in recent years on reactor excursion behavior indicates that significant sectors of the reactivity accident problem are well understood. Of particular practical importance is the high degree of safety that can be achieved by designing self-protective properties into both the inherent neutronic characteristics and the external controls of reactors systems. In many instances, the reactivity accident in power reactor systems can, through proper design, be reduced to secondary importance in the consideration of the maximum credible accident. The data which form the basis of this atmosphere of confidence will be reviewed, and an example of the interaction of design considerations with possible reactivity accidents will be given.

EXCURSION DATA

Available power burst data have covered a wide range of conditions and provide considerable information for attacking many excursion safety problems. The data will be grouped according to the three aspects of the reactivity accident on which the studies have focused: understanding the kinetics of excursions by identifying (or verifying) the various inherent reactivity effects and by developing appropriate analytical descriptions; investigating the effects of power plant type variables upon the fundamental kinetics; and investigating the nature and consequences of destructive reactivity accidents representative of the largest-scale accidents of this type. From the standpoint of practical safety applications, the data also naturally group into three categories distinguished by the extent of physical damage incurred in the tests. These groups can be characterized by broad reactivity ranges: low (no physical damage), medium (moderate damage), and high (destruction). In these ranges, different shut-down mechanisms dominate.

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Studies of reactor kinetics

This grouping includes the work carried out with the systems listed in Table 1, which describes briefly the reactors and some of the principal factors in the experimental and analytical studies. Excursion behavior for these different types of systems has been remarkably similar. For example, the curves of peak power attained in self-limiting bursts as a function of reciprocal period, (α), are qualitatively alike, and for thermal reactors, fall into a narrow range [14]. Many features of excursion behavior, such as peak power, energy release, burst shape, etc., can be described by simple analytical models using space-independent kinetics and an energy-dependent reactivity coefficient [15]. For reactors with prompt, essentially single-mode shut-down mechanisms, such as TREAT, TRIGA and the SPERT oxide core, these reactivity coefficients may be calculated directly and compare well with the experimentally determined coefficients.

For TRIGA, detailed analyses [16, 17] also showed that the fundamental spatial mode dominates during

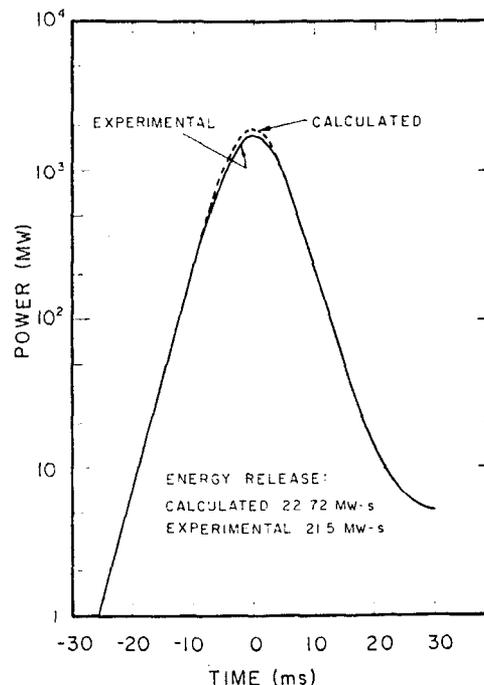


Figure 1. Comparison of calculated and experimental values for TRIGA reactor

Table 1. Reactor excursion studies

Reactor	Type	$\alpha_r(s^{-1})$	Range of reactivity studies		Principal shut-down mechanisms
			$\beta(\text{prompt})$	$\alpha(s^{-1})$	
TRIGA [1]	Heterogeneous, H ₂ O-ZrH-moderated, U-ZrH fuelled	100	0.74-3	0.57-345	Prompt leakage increase due to excitation of bound hydrogen in ZrH
TREAT [2]	Homogeneous, graphite-moderated, UC fuelled	8	≤ 4.15	≤ 25	Prompt leakage increase with moderator heating
KEWB [3]	Homogeneous, H ₂ O-moderated, uranyl sulphate fuel	127	≤ 5.8	≤ 900	Prompt moderator expansion, void formation
Fast core [4,5] (Godiva, FBR)	Bare, enriched-uranium metal, fast	$1 \times 10^6 - 5.5 \times 10^6$	≤ 1.11	$\leq 8.6 \times 10^4$	Prompt fuel expansion
SPERT	Plate-type, H ₂ O-moderated [6, 7, 8], fully-enriched, U-Al fuel	94-140	0.35-3.55	≤ 320	Prompt fuel expansion, delayed moderator expansion and steam void formation
	Plate-type, D ₂ O-moderated [9], fully-enriched, U-Al fuel	10	0.40-3.00	≤ 20	Prompt fuel expansion, delayed moderator expansion and steam void formation
	Plate-type, H ₂ O-moderated [10, 11, 12], fully-enriched, UO ₂ -SS fuel	310-435	0.40-2.00	≤ 300	Prompt fuel expansion, delayed moderator expansion and steam void formation
	Rod-type [13], H ₂ O-moderated, low-enrichment UO ₂ fuel, SS clad	280	0.50-2.60	≤ 450	Prompt Doppler effect, delayed moderator expansion and steam void formation

a pulse, the admixture of higher modes is small (and calculable), and the temperature coefficient and life-time may be obtained from modified perturbation theoretical formulae, with an appropriate weighting associated with local temperature. Figure 1 shows the agreement of calculated results with experiment.

The SPERT oxide core transient experiments demonstrated the effectiveness of the Doppler mode of self-shut-down and provide a basis for analysis of accidents in similar power reactor systems. In the analysis of these excursions, a simple calculational model was developed for the temperature-dependent Doppler effect in a thermal oxide core with a non-uniform temperature distribution [18]. An analytical solution of the prompt-approximation, space-independent neutron kinetic equation was obtained using Doppler feedback as the only shut-down mechanism. Corrections were applied for the small contributions from moderator heating. The form of the Doppler temperature-dependence has been predicted to vary as the square root, the logarithm, or the inverse square root of the temperature. The use of a square-root dependence with the simple model produced a systematic agreement between calculated and experimental effects over the entire range of adiabatic fuel temperature rises in the short-period SPERT tests, whereas the other dependences gave results which differed significantly from experimental results. This strongly implies the validity of the square-root temperature

dependence for the Doppler effect in a thermal oxide core.

The state-of-the-art is not as well advanced in describing the behavior of a system which has a complex, perhaps delayed, shut-down process involving heat transport and phase changes such as in the SPERT plate-type cores [19]. For example, the error in calculating the reactivity effects for these cores is several times as great as that for the oxide core. A question could exist as to whether the source of the inability to better match theory and experiment lies in the understanding of the physical shut-down effects or in the simple way of handling the neutron kinetics. From the success of calculations for systems wherein the shut-down processes are dominated by a single and relatively simple mechanism, the problem would appear to lie in the treatment of the mechanisms of shut-down rather than in the way in which the neutronic behavior is handled.

Since, in practical considerations many dollars of excess reactivity may be potentially available for excursion initiation, and the damage resulting from excursions is influenced by the localized energy-density which can be achieved, it is of interest to compare the excursion data shown in Figures 2 and 3. Figure 2 shows the average energy density, on a volume basis, which is achieved by various experimental reactors as the initial excess reactivity is increased. Potential damage, of course, must be evaluated on the

basis of the properties of the material in which the fuel is contained. Figure 3 shows the derived energy-density shut-down coefficient. The figures point out three salient features: the high shut-down efficiency of reactors with large, prompt shut-down mechanisms, such as the prompt temperature coefficient that TREAT and KEWB possess; the small differences in shut-down efficiency for reactors not possessing this feature; and that many thermal reactors can be designed to self-control the addition of large amounts of excess reactivity without serious damage.

From the broad point of view of the safety evaluation of power reactors, the over-all significance of this work goes beyond the demonstrated ability to understand and predict detailed behavior of individual systems. The principal result of this group of experiments, by their very scope and depth, is a less tangible but more important effect. The determination of the existence and verification of the effectiveness of a great variety of quenching mechanisms, and the demonstration that the magnitude of these inherent shut-down mechanisms can be great enough to provide protection against even large reactivity perturbations, lead to confidence with regard to the state of understanding of the excursion problem and confidence that self-protection can be designed into these systems.

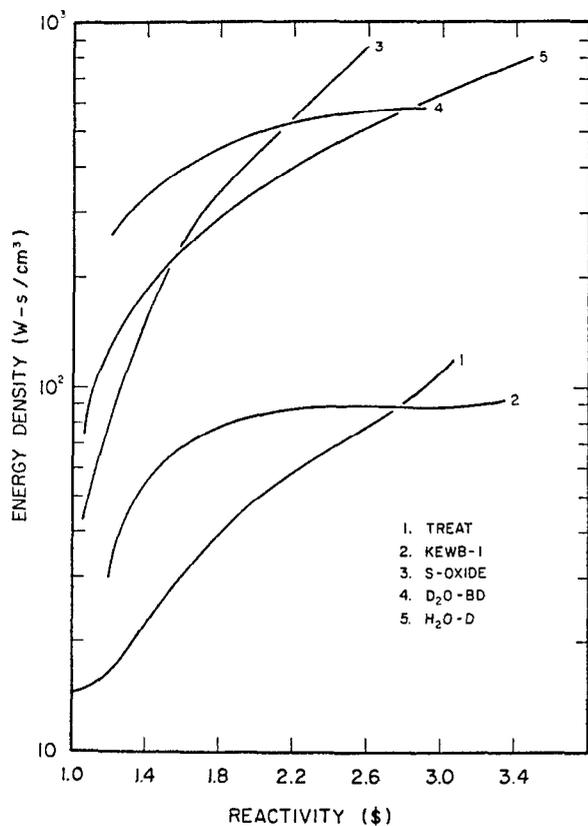


Figure 2. Energy density at time of peak power vs. initial reactivity insertion

Studies of operational variables

In operating power plants, the conditions under which excursions might occur, and the consequences that might follow, are influenced by the variables of plant operation, e.g., (in a boiling or pressurized water reactor) the steady-state pressure, coolant flow rate, temperature, power, and even the control-rod pattern (which may determine the effective core size). To date, only the SPERT plate-type reactors have been used in investigations of these factors, and the results are not in all cases applicable to systems with long heat-transfer time constants. The effect on the kinetic response of a reactor to changes in these system parameters is a complex situation that is not amenable to a concise, simple and unambiguous presentation, but the general trends can be discussed in a qualitative fashion.

Increasing system temperature (ambient to 400°F) [20, 21]

In general, increasing the system temperature at constant pressure results in a reduction of peak power, Φ_m ; energy to peak power, E_m ; fuel plate surface temperature rise at peak power, $\Delta\theta_m$; and maximum fuel plate surface temperature rise, $\Delta\theta_{(max)}$ for all initial periods. The reduction is largest for the shorter-period tests and for temperatures that are nearest to

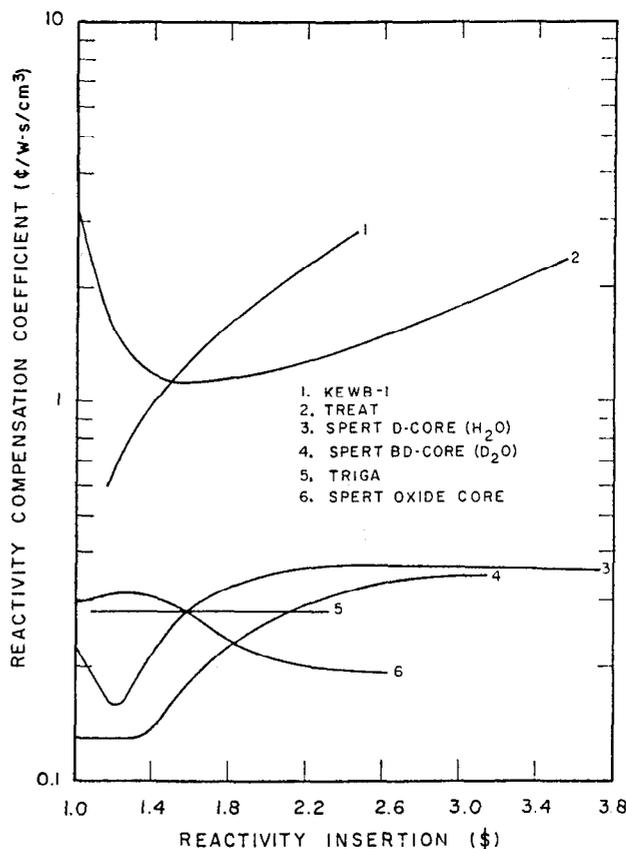


Figure 3. Energy density reactivity compensation coefficient vs. initial reactivity insertion

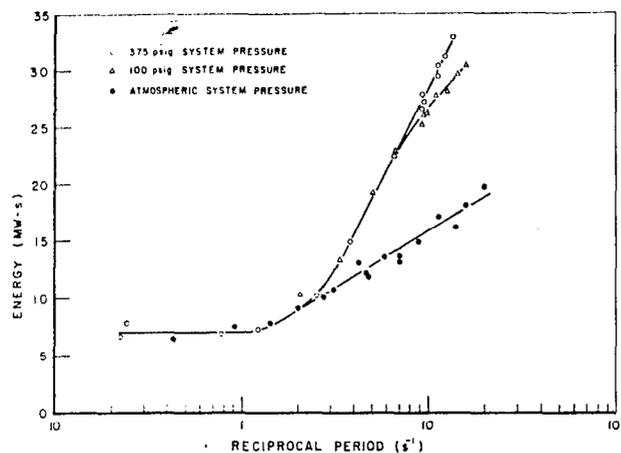


Figure 4. Energy release vs. reciprocal period for various system subcoolings — SPERT 2

the saturation temperature. The actual fuel plate surface temperature (initial temperature plus rise) for the peak power and maximum conditions is increased at the longer periods and decreased at the shorter periods.

Decreasing system (moderator) density (approximately 14%) [22, 23]

These tests were performed by increasing the system temperature and pressure and maintaining constant subcooling. The results were generally of the same type as observed for the increase in system temperature and were consistent with those expected from the increase in the expansion coefficient of water as the density decreases.

Decreasing system subcooling (600°F to 0°) (decreasing system pressure 2 500–0 psig) [23, 24]

A decrease in subcooling has no observed effect on longer-period tests, where boiling does not contribute to self-shut-down. For shorter-period tests, where boiling does normally contribute to shut-down, a decrease in subcooling resulted in a decrease in Φ_m , E_m , $\Delta\theta_m$, θ_m , $\Delta\theta_{(max)}$, and $\theta_{(max)}$. The effect is larger as the period of the test is decreased. The major effect of pressure change occurs in the 100–0 psig range, which is consistent with the change in saturation temperature with pressure. Figures 4 and 5 show this effect for SPERT 2, where boiling is the principal shut-down mode for α greater than ~ 3 , and for SPERT 3, where boiling is of secondary importance in shut-down.

Increasing system flow (0 to 18 ft/s) [21, 23, 25]

For long periods (> 100 ms), the addition of forced coolant circulation tends to eliminate the initial power-peaking characteristics of the typical power excursion. The power instead rises monotonically until equilibrium is established between heat production and removal. For intermediate periods (100 ms to 50 ms),

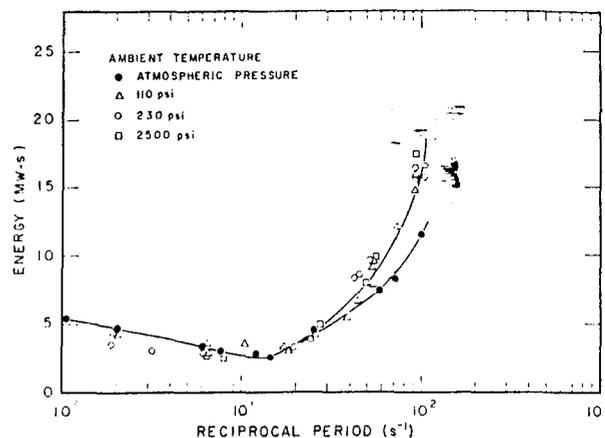


Figure 5. Energy release vs. reciprocal period for various system subcoolings — SPERT 3

the power rises to slightly higher initial peaks than for tests without flow, and the power burst shape tends to be slightly broadened. For short periods (< 50 ms), the addition of forced coolant circulation has no significant effect on the power burst as the coolant transit time through the core, (~ 170 ms), is long compared with the period of the burst. As would be expected from the power behavior, the tests with forced coolant circulation resulted in higher total energy releases than for the corresponding tests without flow. In the short-period region where there was little effect on the peak power or burst shape, there was little difference in the energy released during the burst.

Although the reactor powers that were attained with flow were, in all cases, equal to or greater than those without flow, the fuel-plate surface temperatures were observed to decrease with increasing flow.

Changes in initial power (0–100 kW) [26, 27]

Some testing has been performed with initial power as a variable. For equivalent reactivity insertions, an increase in initial power results in (a) a longer-period excursion, and (b) reduced peak power. Over the range of the tests, the resulting effects were very weakly dependent upon starting power. However, tests have yet to be conducted under power conditions typical of operating power reactors.

Transient pressures were not of significance from a safety point of view. Indeed, transient pressures sufficiently large to be a safety problem probably can be generated only as a consequence of fuel melt-down or vaporization.

In summary, the experiments indicate that although the effects on reactor kinetic behavior of the operating plant parameters can be complex, the effects are qualitatively understood. Most important, the capability of these plants to safely self-limit excursions induced by reactivity insertions considerably in excess of prompt critical was not significantly changed by operation under power-plant-type conditions.

Tests in the destructive range

The extension of excursion tests into the potentially destructive region has been desirable for a number of reasons. One very important one is the concern that some major destructive effects might not be extrapolable from nondestructive data, that is, that their onset might in effect be a threshold phenomenon. Thus, there is a need for determination of the conditions necessary to produce a destructive burst. Furthermore, uncertainties in many fundamental quantities under these extreme conditions makes preanalysis of the consequences difficult. Finally, there is the desire to demonstrate that nothing fundamental has been overlooked in the consideration of major postulated accidents.

Table 2 summarizes the main features of the tests carried out to date, with the available results of the SL-1 accident included for completeness. BORAX 1, SL-1, and the SPERT 1 D-core were highly-enriched uranium fuelled, aluminum-plate cores with somewhat differing physical characteristics; the SPERT 1

oxide core was made up of low-enrichment UO_2 -fuelled stainless-steel clad rods; and the SNAPTRAN 3 test was of a ZrH-U fuelled reactor designed for space applications. BORAX 1, SL-1 and SNAPTRAN 3 were essentially single destructive excursions, but the SPERT 1 D-core tests were a series which progressed through limited melting of the core to the final destructive excursion. The oxide core tests, although damage to the core was experienced, did not result in major destructive effects. The operability of the reactor facility was not impaired.

The tests of greatest interest are the SPERT D-core and the SPERT 1 oxide core. The D-core results indicate clearly the existence of a destructive threshold effect with increasing reactivity. This is shown by the fact that for two preceding tests, in which $\sim 0.3\%$ and $\sim 3.0\%$ of the core were melted, no pressure effect beyond that normally extrapolable from previous tests was observed. The final test, on the other hand (Figure 6), in which $\sim 35\%$ of the core melted, revealed a very sharp, large pressure pulse which occurred after the power burst had been terminated

Table 2. Destructive power excursion summary

Reactor	Reactivity addition (β)	α (s^{-1})	Peak power (MW)	Energy release (MW s)	Maximum temperature ($^{\circ}C$)	Maximum energy density (watt seconds/cm ²)	Maximum pressure (psi)	Remarks
BORAX 1 [28]	3.1	384	$\leq 19\ 000$	135	$\leq 1\ 800$	$\leq 6\ 500$	6 000-10 000	Destroyed core, vessel, and some associated equipment. Small fission-product release. Steam explosion proposed as cause.
SL-1 [29]	3.0	280	$\sim 19\ 000$	133	$> 2\ 075$	$> 7\ 300$	10 000	Destroyed core, bulged vessel, local fission-product contamination. 10% fission-product release. Steam explosion—minor contribution from metal-H ₂ O reaction
SPERT 1 [30]	2.6	200	1 130	11	585	2 000	7	Melted $> 0.5\%$ of core
D-12/25	2.7	218	1 270	19	680	2 300	8	Melted $\sim 3\%$ of core
	3.55	313	2 250	31	1 360	4 600	$\leq 4\ 000$	Melted $\sim 35\%$ of core. Destroyed core and associated equipment, bulged tank. $\sim 4\%$ fission-product release. Probable steam explosion—Al ₂ O ₃ analysis indicates ~ 3.5 MW s energy release from metal-H ₂ O reaction
SPERT 1 [31] oxide core	2.6	455	17 400	155	1 800	2 200	70	Two fuel rods ruptured. Discoloration and/or deformation of 25% of fuel rods. Negligible fission-product release
	3.3	645	35 000	155	1 800	2 200	130	Two fuel rods ruptured. Discoloration and/or deformation of 25% of fuel rods. Negligible fission-product release.
SNAPTRAN 3 [32]	3.5	1 400	$\sim 20\ 000$	50	$> 2\ 500$	7 100	$\sim 4\ 000$	Burst pressure vessel. All fuel rods ruptured, $\sim 1/2$ of fuel reduced to powder form. Negligible fission-product escape.

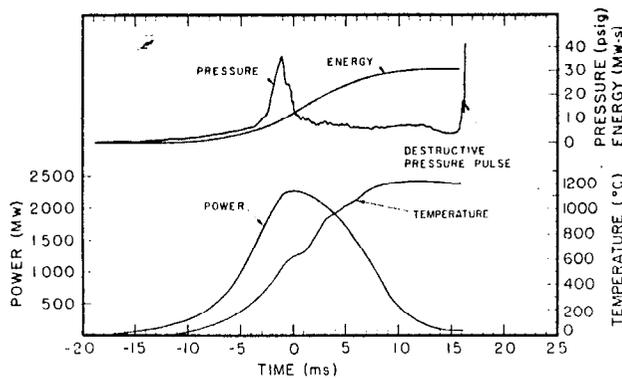


Figure 6. SPERT 1 destructive experiment—D-core

by the normal inherent shut-down mechanisms. Little recorded information is available about the nature of the explosion which destroyed the core after the fuel melted. Of the postulates which have been advanced, most of the available information supports the hypothesis of a steam explosion resulting from momentary superheating of the water in the moderator channels.

In considering the destructive effects that might result from the rupture of UO_2 fuel rods, the possibility existed that spilling or expulsion of the hot oxide into intimate contact with water could result in large pressures being generated in the core through a steam explosion mechanism. In the first destructive-region test of the oxide core, during which two fuel rods were ruptured, the expulsion of the oxide and subsequent heat transfer to the moderator acted as a threshold shut-down mechanism, as can be seen in Figure 7. Here the shut-down coefficient decreases as would be expected from a Doppler shut-down mechanism until about the time of peak power, at

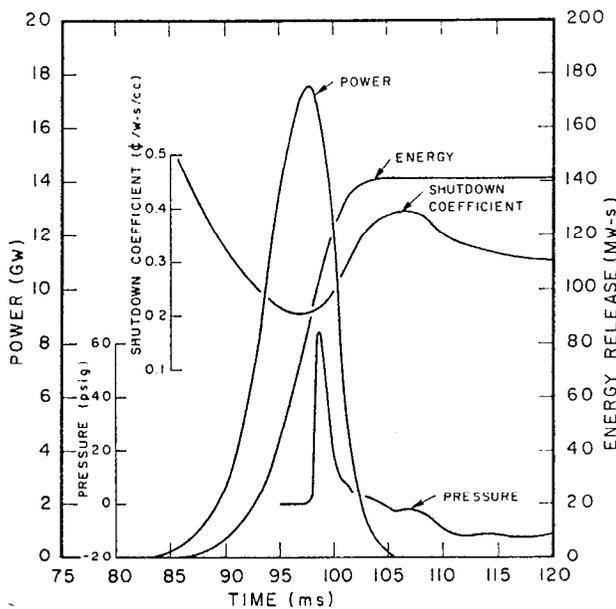


Figure 7. SPERT 1 destructive experiment—oxide core 1

which time it increases rapidly, and is accompanied by a sharp pressure increase. In this case the rapid heat transfer to the moderator acted as a prompt void-formation shut-down mechanism.

The second oxide core test, in which more reactivity was added than in the first, yielded essentially the same results. Even though the period was shorter and the peak power attained was higher, the total energy release and physical effects on the core were not greater. Figure 8 shows the power burst during the test and compares it with the predicted burst based on Doppler shut-down.

The following results were obtained from the tests: (a) the power excursion behavior was predictable until time of fuel rod rupture, both by calculations using Doppler reactivity feedback theory and extrapolation from previous test results; (b) the bursting of the fuel rods caused steam formation that aided in shut-down of the reactor and decreased the expected total energy release of the excursion; (c) less than 1% of the heat energy in the fuel of the ruptured fuel rods was converted into mechanical energy in the form of pressure generation; and, (d) the failure of a fuel rod and consequent dispersal of powdered fuel into the water during a severe power excursion did not result in pressures sufficiently large to initiate failure of additional fuel rods or seriously damage other reactor components.

The direct applicability of these results to operating oxide-fueled reactors should take into consideration the following: (a) the tests were initiated from very low power levels; (b) there was no significant fission-product inventory, and thus no build-up of entrapped fission gases; and (c) the fuel used was originally intended for critical assembly work, and must be considered to be substandard compared to production fuel.

The SNAPTRAN 3 test, which resulted in total destruction of the reactor, illustrates the predictability of some threshold mechanisms; in this case, the release of hydrogen from the ZrH-U lattice when the dissociation temperature was reached. The results of the experiment were in good agreement with pre-analysis.

APPLICATION TO POWER REACTOR SYSTEMS

The excursion data can be used as handbook-type information to a limited extent, as was possible in the case of the NS *Savannah*, where the plant was very similar to one of the SPERT cores [33]. But the real value of the work is much broader and can be best indicated by outlining an approach to some of the safety considerations that enter into the design of power reactor systems. The first step in the approach is to establish the general excursion characteristics of the system. Then, limits on permissible reactivity

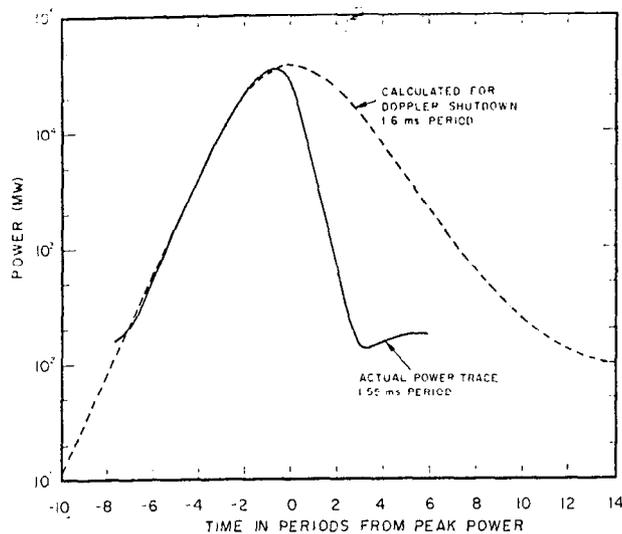


Figure 8. SPERT 1 destructive experiment — oxide core 2

increments in control or fuel units can be set. Finally, design details of mechanical and procedural controls may be determined. In this process, the economic objectives, design problems, safety requirements, and safety data interact.

Since the pressurized-water and boiling-water reactors in the USA account for a major portion of the total power reactor population, the discussion will be limited to these types. The present systems are mainly light-water cooled and moderated and use UO_2 fuel rods with either Zircaloy or stainless steel cladding. The trend toward larger cores and higher power ratings is continuing. In these reactors, three core characteristics affect the reactivity accident safety considerations in a major way: the large initial excess reactivity (\$20–\$30) for accommodating high burn-up and relatively large negative power coefficients; the large physical size, with attendant small neutron leakage; and, most important from the safety standpoint, the ^{238}U Doppler effect.

The first characteristic requires consideration of both the reactivity increment that can be invested in control devices or fuel bundles, and the rate at which this investment can be added to the system by moving these elements either singly or in groups. The actual numbers will vary considerably with the details of a particular design and with the detailed procedures for loading and control rod withdrawal. The fuel bundles and control blades which are used in these lattices have insertion rates in the range of several dollars per second for abnormal, accident situations involving mechanical failure of normal design safeguards that provide protection against such accidents.

These large cores, which actually contain several critical masses, have relatively weak spatial coupling of the neutron flux. Thus, two widely-spaced regions of the core may be in a supercritical or subcritical

state, both possessing nearly asymptotic periods, as was demonstrated in special experiments performed in the Dresden reactor, where it took several minutes for two such regions to reach a common asymptotic period [34]. It is also theoretically possible to achieve regions of greater buckling by withdrawing control rods preferentially. The highly peaked neutron importance function results in larger incremental reactivity worths of fuel bundles and/or control elements than is the case for the large core geometry. There exists, then, the possibility of localized excursions in addition to gross core excursions involving the entire core acting as a unit.

The negative reactivity contribution of the ^{238}U Doppler effect to the self-limitation or quenching of nuclear excursions depends on the amount of resonance capture in the lattice, the oxide-fuel time constant, and the reactivity insertion rate for the excursion. Generally, for relatively short-period excursions the Doppler coefficient is the major shut-down mechanism for these reactor types, since the low values of the thermal diffusivity for the oxide fuel limits the rate at which heat can be transferred out of the fuel and, in turn, results in increased fuel temperature and a corresponding negative reactivity effect. The oxide core tests provide reference points for evaluating the characteristics of the Doppler shut-down mechanism.

In the dynamic response of these cores to nuclear excursions, such parameters as neutron lifetime, delay fraction, reactivity coefficients, scram reactivity insertion rates, etc., are important. The values of these parameters vary considerably with the details of the design. However, from the available published information (for typical lattices) [35–38], the ranges of these parameters are the following: the ratio of the delayed neutron fraction to the prompt neutron lifetime, which is defined as α_R , 100 to 750 s^{-1} ; the power coefficient, -2 to $-5\text{c}/\%$ change in power; the moderator temperature coefficient, 0 to $-5 \times 10^{-4} \Delta k/^\circ\text{F}$; the moderator void coefficient, 0 to $-3 \times 10^{-3} \Delta k/\%$ void; the Doppler coefficient, -1 to $-5 \times 10^{-5} \Delta k/^\circ\text{F}$; the scram reactivity insertion rate, \$10 to \$20/s average rate including delay time, with peak rates two or three times greater.

On the basis of these characteristics and the present understanding of Doppler coefficients and excursion kinetics, the principal features of nuclear excursions for these cores can be estimated, as shown in Table 3. Note that it takes an energy density of approximately 1.1×10^4 watt seconds/cm³ to melt UO_2 , and approximately two to three times that to vaporize UO_2 , assuming that the UO_2 is initially at room temperature. In the low-ramp-rate range there is no fuel melting, the periods are relatively long, and significant heat transfer to the moderator occurs during the burst. The shut-down mechanisms are the Doppler effect and the negative moderator coefficient. The expected

Table 3. Excursion parameters for pressurized-water and boiling-water reactors

Range	Reactivity insertion rate ($\$/s$)	Equivalent step ($\$$)	Minimum period (ms)	Energy density (av) ($W/s/cm^2 \times 10^{-4}$)	Energy density (peak) ($W/s/cm^2 \times 10^{-4}$)	Principal shut-down mechanism
Low	< 2.5	< 1.8	> 4	< 0.14	< 0.44	Doppler effect, negative moderator coefficient
Medium	2.5–25	1.8–3.5	4–1.2	0.14–0.48	0.44–2.2	Doppler effect
High	> 25	> 3.5	< 1.2	> 0.48	> 2.2	Doppler effect, core disassembly

consequences of such an excursion are generally trivial, although in some cases the scram system must be relied upon to prevent clad damage [35, 36]. The medium range has faster ramp rates, shorter periods, and only a small amount of heat transfer to the moderator during the burst. Shut-down is predominantly from the Doppler effect. At the upper end of this range, there may be a small amount of fuel clad failure during the burst with fuel (UO_2) melting and, possibly, vaporization of UO_2 . After the termination of the burst, further clad failures may occur at a relatively slow rate while the energy stored in the fuel is transferred to the clad and the water. No significant mechanical effects or transient pressures are postulated for excursions in this range. In the high range (very fast ramp rates and very short periods), the predominant shut-down mechanism is the Doppler effect, possibly supplemented by fuel dispersion and moderator expulsion during the burst. Significant fuel melting and vaporization could occur with, possibly, substantial mechanical effects and transient pressures, if the stored energy in the fuel can be transferred rapidly to the water. No experimental data are available on these high-ramp-rate accidents. However, the most extreme tests shown in Table 2 extend well into the medium-ramp-rate range. The results of these tests generally confirm the qualitative differences between the low-ramp-rate and medium-ramp-rate ranges discussed above.

In the above discussion, the ranges for the energy release and consequences were established on the basis of experimental excursion studies, analysis of kinetic behavior, and the lattice characteristics. However, the credibility, or even possibility, of achieving the postulated excursions was not considered. The physical limits on attainable reactivity insertion rates [15], and the inherent safety characteristics of the particular system, must be considered in relation to the following principal types of accidents: (a) mechanical malfunction of the control system, causing rapid removal of the control rods; (b) inadvertent, rapid, manual removal of control rods; (c) procedural operator error in withdrawing control rods, or in changing control-rod

patterns during operation (e.g., the classical start-up accident); (d) fuel-handling or fuel-loading accident, including potential accidents associated with handling of temporary poisons in the form of sheets or strips [37]; (e) accidents arising from moderator effects, such as cold-water accidents in PWRs and BWRs, and positive-pressure ramps in BWRs; and, (f) accidents associated with malfunctions of the soluble-poison control system, resulting in accidental removal of the poison or insufficient shut-down margin in the liquid poison system and leading to a refuelling accident.

Both the potentiality for occurrence of such accidents and the possible magnitude of the consequences can be reduced by limiting the reactivity increment and insertion rate in fuel or control units. In the design of this type of power reactor these limits are significantly affected by safety considerations. Important factors which enter into the specification of the mechanical control system are reactor shut-down margin, maximum control-rod worth, and scram requirements. Once design criteria have been established for the first two these factors (which tend to be limiting), the excess reactivity which may be loaded into the reactor core for a given control-rod spacing and the cold-to-hot operating reactivity swing for the fuel may be determined. The key to interrelating the fuel cycle requirements and the movable control system is an approximate relationship between thermal control-rod strength, thermal diffusion length, and control blade (span/pitch) ratio [39], which is essentially a restatement of absorption area theory [40].

With proper safeguards design and operating procedural limitations for the plant, most of the potential excursion accidents in the above categories fall into the low range (no fuel damage). For example, the classical start-up accident leads to reactivity insertion rates of the order of cents per second to ten cents per second [35–38]. Accidents that are outside this range can be conceived by postulating compound violations of procedural controls with concurrent, unrelated mechanical or equipment failures.

SUMMARY

The excursion studies have shown that for many types of reactor systems excursion behavior is well understood and that the self-limiting capabilities can be sufficient to provide protection against substantial reactivity disturbances. The studies have been principally devoted to systems with relatively strong quenching properties and under initial test conditions such that substantial energy releases would be required to cause damage. Thus, neither the effects of locally positive reactivity coefficients nor excursions initiated under high power conditions have been examined. The data from excursions that resulted in mechanical damage have provided information on the shut-down mechanisms that operate under such conditions, the magnitude of the transient pressure effects, and the radiological consequences.

In considering the possible sources of the maximum credible accident, attention must be given to the means of initiation and credibility of occurrence of reactivity accidents, including simple plant failures. However, the body of excursion data and the present state of understanding of reactor excursions are such that the reac-

tivity accident, for many power reactor systems, can, through appropriate design, be reduced to secondary importance in the consideration of the maximum credible accident. Of particular importance to many present day designs of pressurized-water and boiling-water reactors is the fact the light-water oxide-core power burst data can be reproduced analytically using models developed from first principles. This lends confidence to the model predictions of the excursion behavior of power reactor cores which are reasonably similar to the experimental configurations. Information on the consequences of very short period (high reactivity range) excursions in these cores is limited since this problem is not very amenable to analysis and only a small amount of experimental data is available. In particular, no data are available to determine if there exists a destructive threshold for these cores similar to that observed in other systems, although present information does not indicate that such a threshold occurs with the rupture of a few fuel rods. Information in this general area can be useful to the reactor designer in optimizing nuclear system design from the standpoint of safety and performance.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

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Etude des sautes de puissance d'un réacteur par W. E. Nyer et al.

Il n'est pas exclu que les sautes de puissance dans les réacteurs nucléaires puissent, dans des cas extrêmes, causer des dommages dans l'installation, et libérer des quantités importantes de produits de fission; leur étude est donc un aspect important de l'évaluation de la sûreté d'un réacteur de puissance. Lorsque l'on fait cette évaluation, il faut déterminer les mécanismes physiques intrinsèques qui ont pour effet d'arrêter ou d'amplifier la saute de puissance, déterminer aussi leur comportement pendant la saute de puissance et bien comprendre leur effet sur l'allure de la saute de puissance. Ces mécanismes intrinsèques, qui sont en partie des propriétés du type de réacteur, peuvent être sensiblement modifiés par des caractéristiques particulières. Il faut examiner soigneusement ces caractéristiques pour s'assurer que certains détails du projet n'ont pas d'effets indésirables, pour évaluer les moyens possibles de provoquer des sautes de puissance accidentelles, et enfin pour fixer certaines spécifications et mesures de contrôle qui devront s'appliquer au fonctionnement de l'installation aussi bien qu'à la conception elle-même.

On dispose maintenant de nombreuses indications touchant la sûreté des réacteurs de puissance du point de vue des sautes de puissance grâce aux études expérimentales faites dans des conditions diverses avec des réacteurs de différentes filières: TRIGA, qui est ralenti au ZrH; TREAT, qui est ralenti au graphite; KEWB, qui est homogène et aqueux; Godiva, qui est un assemblage entièrement métallique à neutrons rapides; SPERT 1, qui est ralenti à l'eau et dont le combustible est en aiguilles ou en plaques; SPERT 2, qui est ralenti à l'eau lourde et dont le combustible est en plaques; SPERT 3, qui est ralenti à l'eau à haute pression et dont le combustible est en plaques. On a

pu démontrer que les divers mécanismes d'arrêt étaient efficaces dans le cas de l'autolimitation des sautes de puissance provoquées délibérément, et mesurer leur effet quantitatif sur la réactivité. Ces mécanismes étaient la production de gaz par radiolyse, des modifications (dues à la chaleur) dans la géométrie de la configuration du combustible, des variations de la densité du fluide de refroidissement et de refroidissement, des variations des sections efficaces de diffusion, les effets Doppler et dans plusieurs cas, le désassemblage du réacteur. La plupart de ces mécanismes sont importants dans les types courants de réacteurs de puissance; toutefois, l'effet quantitatif sur la réactivité varie selon le type de réacteur et même selon les conditions de fonctionnement de l'installation (comme l'ont montré les expériences faites avec SPERT 2 et SPERT 3). Comme on s'y attendait pour ces réacteurs, une augmentation de la température et de la pression initiales au cours d'une série d'essais a eu pour effet un accroissement significatif, mais non extrême, de l'énergie libérée. Des débits élevés du fluide de refroidissement ont également eu pour effet un accroissement de l'énergie libérée, mais non de la température du combustible. Qui plus est: lorsque l'on a simulé les conditions qui règnent dans les réacteurs de puissance, on n'a pas fait décroître sensiblement la capacité qu'ont ces installations de limiter elles-mêmes les sautes de puissance provoquées par des additions de réactivité considérablement supérieures à celle que provoquent les neutrons instantanés. Les résultats obtenus avec tous les systèmes considérés indiquent une remarquable similitude dans la façon dont la forme de la saute de puissance, la puissance de crête et l'énergie varient en fonction de la grandeur de la réactivité ajoutée, et les descriptions mathématiques en sont remarquablement simples. On ne pense pas que ces caractéristiques soient qualitativement différentes dans les réacteurs de puissance. Ce qui est particulièrement important, c'est que dans les expé-