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A PROPOSAL FOR A
REPETITIVELY PULSED TEST FACILITY (RPTF)

R. G. Fluharty, F. B. Simpson,
G. J. Russell, and R. H. Morris

IDAHO NUCLEAR CORPORATION
NATIONAL REACTOR TESTING STATION
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by

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IDAHO NUCLEAR CORPORATION

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ESSENCE OF THE PROPOSAL

I. WHAT IN SUMMARY

(See Appendices A, B, & C for further details)

This is a proposal to build a Repetitively Pulsed Test Facility (RPTF) at the National Reactor Testing Station (NRTS) in Idaho to be completed in FY 1973. Currently a study program of such systems is being carried out by Idaho Nuclear Corporation and a definitive concept is expected by the end of FY 1968. Title I and Title II phases are scheduled for completion in mid FY 1969 and FY 1971, respectively, and project construction to start at the beginning of FY 1971. As a proposal, this document is to provide an outline of the current and contemplated development programs and their pertinence, to provide an introduction to a reactor concept being studied including its merits, and to present plans for an orderly execution of a reactor project.

A preliminary reactor concept, with many attractive features, consists of a small plutonium "booster" reactor producing 3 to 10 megawatts of average power in conjunction with a charged particle neutron source. The relative merits of the proposed system as well as a general discussion of such pulsed systems is given in Appendix A*. The combination of a reactor ("booster") and a charged particle neutron source provides a facility of great flexibility as a reactor technology tool because of the wide range of pulse frequencies, power-pulse heights, and pulse widths that can be produced. Pulse frequencies variable from 0 to 200 pulses per second and widths variable between 10^{-7} and 10^{-5} seconds are representative with corresponding peak-power pulse-heights (3,000 MW) within the limits imposed by average power and shut down shock conditions. In the neutron-spectrometer mode of operation, the reactor is operated as a subcritical neutron pulse amplifier -- amplifying short neutron driving pulses from the charged particle source. This operational mode would also be advantageous for materials test purposes but other operational modes would be available as well for testing materials at higher power per pulse. High neutron gains and corresponding high reactor pulse power are obtained by means of rotating reactivity controls. The average reactivity of the reactor is maintained well below delayed critical, but the rotating reactivity permits short time operation between delayed and prompt critical for large amplification of the source pulse.

* The appendices enclosed will frequently be of a preliminary nature or an outline of documents for later presentation. For example, Appendix B will finally result in a definitive conceptual proposal and construction data sheets. Others will be issued as reports and publications such as the results of the moderator development studies. A schedule for the completion of the various Appendices is given in Appendix C.

INC-C-10003

Fig. 1 A cutaway view of a concept of the RPTF reactor and moderator system.

Schematic representations of the preliminary RPTF are shown in Figures 1, 2, and 3 and more system details are developed in Appendix B. Figure 1 is a cutaway view of the reactor and moderator system. The 8" x 8" cylindrical reactor core consists of fuel and liquid sodium coolant. A central tube through the core provides access for the electron or other charged particle beam and a plutonium target. The target cooling is contained in the particle beam tube allowing independent maintenance of the target. The upper half of the central tube also provides test sample access from the reactor top. Irradiation space is also available in the elliptical containment vessel. The moderators for neutron spectroscopy sources are to be cooled to 80°K with liquid nitrogen. Liquid He cooling to obtain lower temperatures could easily be incorporated if studies indicate sufficient merit but liquid nitrogen temperatures are adequate for large performance gains. Two one-meter radius rotating reactivity controls, which are in phase, are shown as they approach the reactor to provide the cyclic reactivity control. To begin with, studies are being made including fuel as a part of the rotating reactivity control. The reactor-pulse-frequency can be varied by operating the two rotating reactivity controls at different frequencies. This approach is desirable because the pulse characteristics are relatively unaltered compared to a single rotor control when repetition rates are varied. A horizontal section of the RPTF reactor concept and moderator is shown in Figure 2. The drawing shows the rotating reactivity controls in the position for maximum reactivity. To allow for clear presentation of the rotor position, only a few of the fuel pins of a total array throughout the core are shown. The relative positions of the reflector, B¹⁰ decoupling shield, and moderator are also shown. Figure 3 shows the relative vertical positions of the charged particle target, reactor core, moderators, sample holders and rotating fuel. It also gives the direction of flow for the reactor, moderator, sample and target coolants.

In Figure 4, neutron beam tube configurations and a larger view of the rotational system are shown. Multiple beam access to a common moderator surface is obtained by voiding space around the reactor. The voided space also provides great flexibility in the test utilization of the reactor, and would even allow a complete change of the reactor system. Safeguard advantages are also inherent in the voided concept. The voided region is to be cooled by gaseous helium. The helium system will provide an additional radioactivity containment.

A schematic of the neutron source and reactor building of the RPTF is shown in Figure 5, and another aspect of the facility flexibility is illustrated. This aspect is represented by a separate charged particle target cell and points up the fact that the charged particle source and the reactor can be operated independently. The reactor design allows for operation as a "pulsed reactor" in a super-prompt critical mode in contrast to the "booster" or neutron amplifier mode. At the same time, the electron beam can be utilized to study heat and energy transfer problems separately -- for example. The normal target power density from such an electron beam is at least 10 times that of the reactor itself, and can be increased (~25-30 MW/liter) by focusing the beam to a smaller size.

Fig. 2 A horizontal section through the center of the RPTF reactor core and moderator.

Fig. 3 A vertical section through the center line of the RPTf reactor core and moderator.

Fig. 4 A horizontal view showing typical neutron beam tube configurations and relative positions of the rotating fuel with respect to the RPTF reactor core.

Fig. 5 A plan view of the charged particle neutron source and reactor building.

II. WHY – A UNIQUE REACTOR TECHNOLOGY TOOL

A. Nuclear Materials Technology Tool (See Appendix D for Details)

Historically, the major users and motivators for repetitively pulsed reactors or boosters have been the experimenters using neutron time-of-flight methods. But at high average powers such reactors also have the capability of exciting acoustical and kinetic motion in test samples in addition to heat. It is to be argued that such high powered pulsed reactors are needed to provide methods of studying reactor dynamic problems. The information to be obtained is required to establish "factors of safety" for the fast reactor power program. The "factor of safety" can be obtained from calculations of the phenomena leading to the "onset of failure" and are to be called "unusual events" in contrast to "credible accident", because in many cases they are more concerned with reliability in contrast to serious accident. This information will assist in the design against events leading to accidents.

The problems to be studied are correctly called "equation of state" phenomena, but the phrase is too often associated with explosive systems so it will not be used again. Instead the "state" region to be studied will be designated as mechanical transient and shock threshold where energy produced in the reactor is coupled to acoustical and kinetic energies of mechanical motion as well as to heat. Measurements on test samples in the RPTF will provide transient mechanical response data which will test the dynamical computational methods used to predict such dynamical properties of reactors. Because the conductivity, heat capacities, stress, and other thermodynamic properties are usually unknown, the RPTF tests will provide unique information about such quantities. The unique aspects involve irradiation of reactor fuel test samples wherein the data are obtained under pulsed irradiation conditions. Also, the ability to screen materials in an accelerated neutron-pulse transient-stress environment would provide assurance that the tested materials would not encounter vibration damage problems after long term usage in steady state reactors.

In order to develop the assertions made above, a discussion of the power pulse properties of the RPTF in comparison with the single burst "Godiva" type reactors will be undertaken. These reactor types are chosen because the reactor periods, pulses, and spectra are similar to the RPTF. Operating at an average power of 3.3 MW with pulses at a frequency of 200 pps, the RPTF power per pulse is 16.5 kilojoules or 5×10^{14} fissions per pulse. Operating at 20 pps, 5×10^{15} fissions per pulse are produced. The latter pulse power as shown in Table I^[1] is 1/4 the fissions per pulse obtained by Godiva II. Likewise the 200 pps operation produces 1/40 the maximum fission per pulse of Godiva II. Because the shutdown mechanism for the Godiva II is caused by mechanical expansion and because it is shock limited above 2×10^{16} fissions per pulse, it is concluded that the RPTF will be well into the pulse energy range for exciting mechanical energy. The available range is, however, below the shock limit. Beyster and Russell ^[2] have demonstrated such mechanical vibrations as shown in Figure 6 which also shows a representative time scale for the transient mechanical waves. While the pulse widths used for the data in Figure 6 were 5 μ seconds the same as the 200 pps RPTF mode (1/40 Godiva power), the peak power was 1/20 the corresponding RPTF case.

Fig. 6 The bottom figure shows the surface velocity produced by the introduction of an electron beam pulse onto a U-235 spherical critical assembly on the General Atomic L-band linac. The top two figures show the linac electron pulse and fission power pulse, respectively.

TABLE I

COMPARISON OF PULSE PROPERTIES

	pps	Pulse Width	Fissions/ Pulse	Flux Intensity at Peak of Pulse
RPTF Case I	200/sec	5 μ sec	5×10^{14}	1×10^{18} n/cm ² sec
RPTF Case II	20/sec	50 μ sec	5×10^{15}	1×10^{18} n/cm ² sec
Godiva II	2000/year	40 μ sec	2×10^{16}	5×10^{18} n/cm ² sec

It should again be stressed that the mechanical and acoustical excitation capability shown above will take place during high power operation with an average central flux of greater than 1×10^{15} n/cm²sec and exterior flux greater than 1×10^{14} n/cm²sec. Corresponding peak central fluxes are 1×10^{18} for both operation modes. Unique material sample tests can therefore be carried out in a high power reactor environment.

For the single pulse burst (repetition 1/hr) reactors like Godiva, there are two points of direct interest as follows: a) the maximum pulse power is determined by a safe limit beyond which destructive damage occurs to the system in a shock like way where the reactor is distorted and fractured, b) the reactor lifetime is limited due to metallurgical phase change of the metal because of work hardening (such damage may also limit the peak power chosen for operation). While these limits are being increased by careful design considerations to dissipate the mechanical energy and by recourse to different alloys, the demonstration of limits caused by mechanical and material damage is important to note.

For such reasons in the design of the RPTF itself, there must be developed materials which can stand the transient operation for a very much larger number of pulses at high average power. For example, a few hours or days of RPTF operation would provide equivalent nvt's to the total lifetime of one of the single burst systems. The pulse lifetimes of single burst systems observed to date are less than 10,000 pulses. To operate the RPTF at 30 MW would require an extrapolation of technology indeed, since the power per pulse would considerably exceed those of the Godivas III

An approach to the study and solution of the very high transient power density problems is provided by the existing electron beams used to produce such source neutrons at many installations in the country. While confined to a small volume, the peak and average target power densities are equivalent to or higher than the highest proposed in the RPTF dependent upon the size of the beam spot on the target. Current operating experience with these target systems takes place with at least 10 times the proposed reactor power densities for long periods and high repetition rates equivalent to those desired for the RPTF. There are, therefore, examples of materials withstanding the high power density shock type of operation. This is accomplished by using high temperature materials such as tantalum, by increasing the cooling surface area, and by using short water cooling paths allowed by the small volume. Because the electron beams are pulsed, a required tool for studying such transient mechanical design problems of the RPTF is available.

While the pertinence of such mechanical transient studies to the RPTF design is easily illustrated as above, the relevance to general reactor dynamics justifies further discussion. Continuing in the vein of pulsed systems, the fission energy is deposited in a time fast compared to the time required to attain thermal heat equilibrium. Because of the resulting transient temperature conditions, a transient thermal stress in addition to the normal static thermal stress is generated. To the extent that fuel and coolant are free to move, the transient stresses generate mechanical motion. The stress forces are balanced by inertial forces and opposing stresses from positioning grids, etc. To the extent that the fuel is free to move or is bound, bending or bowing results with possible accompanying vibration. Generally the heat equilibrium time constants are determined by conduction and heat capacity and are slow compared to the mechanical expansion. An important point is, however, that the two have different time constants, and phase differences exist which may be coupled to vibrational modes.

The rate of fission energy deposition in the single burst "Godiva" like reactors becomes high and times are also short compared to the velocity of sound. Therefore, the expansion which determines shutdown and which determines the negative temperature coefficient is delayed by the time required for such expansion to reach the surface. For example, a typical time of travel for the expansion wave is given by the reciprocal sound velocity of 10μ seconds per centimeter. The reactor shuts itself down because the neutron leakage is increased as the reactor expands, so the delay before expansion has an important bearing upon the size of the power pulse that can be generated. Because the neutron lifetimes or the neutron generation times are short compared to the expansion time, more energy can be obtained in the pulse before shutdown. As the power pulse becomes larger, the expansion energy forms into a shock wave which eventually breaks up the system at the limiting power. The destruction is more analogous to collision damage being relatively mild as contrasted to explosive damage.

In this sense, these small fast reactor systems are really the safest because large excursions are not possible since the neutron-power coupling to mechanical energy is relatively rapid. Consequently, the system comes apart before a large amount of energy can be generated. In contrast, larger reactor systems which depend upon thermal expansion of the coolant for a negative temperature coefficient may have long coefficient delays. The large thermal energy storage capacity can result in a much larger destructive event. The basic argument is to point up the importance of the time delay or phase information to the problem of predicting the onset of an "unusual event".

While the small fast reactor systems are inherently safer against large excursions, they have more problems with dynamic stability. The very rapid responses and high frequency neutron responses pose larger problems due to rapid neutronic coupling to mechanical motion. In contrast, the thermal reactor neutronics have long neutron lifetimes which will integrate out high frequency dynamic-mechanical effects. However, to the extent that an appreciable number of the fission reactions take place with high energy neutrons, short time coupling problems must also be examined for thermal reactors.

In standard steady state operation of power systems, the opportunity for such phased coupling to exist is minimized. And the engineer carefully designs to damp out such instabilities and applies sophisticated control systems tests which examine the frequency response and phase properties by means of oscillation and noise correlation techniques. However, these can only examine small variation and fluctuations around steady state and do not provide information about larger excursions.

A generally valid question which can be asked is "How can such transient effects be generated in a steady state power system?" This question is in contrast to the pulse systems where they are reactivity or source generated directly and deliberately. One obvious situation where such questions are important would be the reactor response to a deliberate shock by an alien from a projectile or a shock wave created from an explosive. While studies about such questions can be made in the single burst reactors such as the PBF, FRAN, and TREAT, phase differences between acoustical or shock waves and the energy deposition are more difficult if not impossible to observe because of the slow rates of energy release. Such phase differences could easily be observed in the RPTF. Also, differences to be expected due to radiation embrittlement would be more easily detected in an RPTF.

Those materials which perform well in the RPTF would be expected to be very immune from transient stress damage in steady state reactor systems. And while such vibration enhanced radiation damage effects are not expected because the design has minimized their probability of existence, considerable assurance of reliability and integrity would exist if the materials were shown to be resistant to such damage in the RPTF.

In principle, calculations and predictions of the dynamic properties are restricted because much of the basic physical data has not been measured. For materials which will have been subjected to a large (nvt) radiation history such information will be slow and difficult to obtain. For irradiation dynamic-stress-hardening damage, the RPTF would provide a testing ground for the physical models used to predict such properties and for testing the calculational models. Thus, though the sample sizes which can be irradiated are small, valuable and necessary advanced technological information will result.

In order to perform meaningful experiments within the limits imposed by the RPTF reactor, development of miniaturized instrument techniques and methods will be used for measurements of transient pressure and temperature, for example. These will also be of general interest to steady state reactors. Thus, an RPTF would also play a significant role in the general reactor instrument development problem.

Since the technological area under discussion is of vital importance to the reactor development programs, a significant effort will be made to further develop the merits of this research area to provide a more quantitative description of the measurement and test capabilities. Engineering tests utilizing available linacs and fast transient reactors will be undertaken early in FY 1968. Unfortunately, the available linac facilities are not well designed or well located for carrying out such testing programs. For example, the testing of plutonium fuels as an electron beam target or in a Godiva system will require extra attention to hazard control.

B. Neutron Time-of-Flight Sources (See Appendices E and F)

Neutrons with thermal energies (0.025 eV) have proven to be powerful research probes in providing information about the energy states, positions, and motions of atoms in molecular systems. The large potential of new knowledge to be gained has resulted in an increasing demand for higher flux research reactors. However, with the increasingly higher flux performance the fuel costs become a limiting cost in the operation of such steady state research facilities, and the research programs find it increasingly difficult to justify research reactor fluxes above 10^{15} - 10^{16} n/cm²sec. This higher operations expense is imposed by the increasingly rapid rate of fuel utilization.

To the extent that the research programs can be carried out by means of neutron time-of-flight techniques, pulsed reactor sources provide a mechanism for significant improvements in performance before the corresponding fuel consumption operating cost limits are approached. Simply, the reactor is only operated when neutrons are desired, therefore increased performance by the duty cycle factors (10 to 2000) are obtained for the same average power density and equivalent operating costs of present steady state research reactors. A large market for pulsed reactors will be created when their development has progressed to the point that they are economical and practical. This market will exist because large research fields are opened providing significant insight into the properties of materials. Current AEC laboratory programs would be provided with orders of magnitude improvements in measurement capability and university programs could have equivalent and perhaps better capability than is now available in AEC laboratories.

Realization of the full benefit of the idea of operating the reactor only when needed cannot be attained because of the steady state flux-peaking or "flux-trapping" concept. The reason why this cannot be attained is because the flux enhancements imply a neutron storage concept requiring a long neutron lifetime. This long neutron lifetime is incompatible with the idea of removing them on demand. Such considerations emphasize the difference in philosophy which arises for the design of neutron moderators for pulsed sources. Also it has been recognized that the moderator design problems for such pulsed systems were not generally understood and effort should be made to determine the best methods of approach.

A measurement program to determine the importance of various moderator parameters for optimizing the design of repetitively pulsed reactors was undertaken because it was fairly clear that large gains in performance (an order of magnitude or more) could be obtained by proper design. These are discussed in Appendix F. However, the arguments presented to this effect were not sufficiently clear to be convincing and the choice was not clear about the relative merits of "pulsed reactors" which are operated supercritically and are pulsed by purely mechanical rotating reactivity and neutron sources in a subcritical mode. Basically, the important difference between the two systems for the moment is that the pulsed reactors are restricted to pulse lengths greater than 35 μ seconds, while the boosters can be operated with pulses in the range of 0.1 to 10 μ seconds. Thus, unless the thermal neutrons can be removed efficiently from the reactor in a time shorter than 35 μ seconds, there would be no need to include the charged particle neutron source.

A quantitative discussion for a basis of comparison for the merits of pulsed neutron sources for neutron time-of-flight experiments was presented by Michaudon[3], in which he shows that a valid comparison can be made on the basis of a "figure of merit" given as follows:

$$F = \frac{N_m}{\tau^2}, \quad (1)$$

where

N_m = moderator efficiency

and

τ = resolution width of the moderator pulse-reactor system.

It can also be shown that a reasonably valid figure of merit for comparison of repetitively pulsed reactor systems can be obtained by a figure of merit given in the equation as follows:

$$F_p \propto \frac{P_d M}{\tau^2}, \quad (2)$$

where

P_d = power density of the fast reactor source

M = leakage from reactor

and

τ = time-width of the neutron pulse from the moderator source at energy E including the broadening caused by the fast reactor.

A discussion and examples of the application of Equation (2) are given in Appendix G. The major conclusion is as follows:

The current state of the moderator design studies show a distinct advantage to be gained by using booster reactors driven by charged particle neutron sources. This conclusion is based first upon the fact that the optimum number of thermal energy neutrons would be obtained using the same moderator arrangement for both pulsed reactor and booster, and second upon the fact that neutron pulse widths much shorter than 35 μ seconds are observed from these maximum intensity configurations without decreasing the peak neutron pulse intensity. For equivalent average reactor power densities the present RPTF concept operating with 5-6 μ seconds reactor pulses would have a "figure of merit" a factor of 25 better than a 50 μ second pulsed reactor for 0.05 eV neutrons. The 0.05 eV energy occurs at the peak of the most useful thermal neutron energy range. This energy range is the critical energy range for neutron inelastic scattering studies particularly for phonon dispersion relation determinations and for neutron diffraction studies. Thus, a major point in support of the booster systems is based upon this "figure of merit" improvement at room temperature thermal neutron energies.

A simple argument to illustrate this short pulse advantage is as follows:

Moderator time-widths of less than 9 μ seconds are observed, when corrected for measurement resolution, from moderator configurations producing the highest peak intensities at the detector. When combined with the fast reactor neutron driving pulse of 5 μ seconds, by taking the square root of the sum of the squares of the two combining pulse widths, a τ^2 for Equation (2) of 10 μ seconds is obtained. However, the pulse width of the SORA reactor is 50 μ seconds; therefore the minimum ratio of resolution widths should be 5. Because the advantage goes as the ratio squared, the RPTF would have a figure of merit improvement of 25. Because the SORA is designed for a lower power density, the advantage would actually be larger.

For pulsed cold neutron sources at energies below 0.001 eV, a clear distinction between the merits of the "booster" and pulsed reactors cannot yet be made, and it appears possible that the pulsed reactors would be preferable. However, the following points are also germane to the current RPTF proposal.

a) Operation in the pulsed reactor mode has been planned for. However, to date very little study of the particular response of the system in this mode has been made, so detailed properties cannot be reported at this time. Under any circumstances, the current reactor configuration has been chosen to optimize the shorter pulse operation. But, the RPTF can utilize whichever mode is appropriate for cold neutron experiments and hence should not be ruled against because of an argument based upon "cold neutrons" alone.

b) There are important indications that the short moderation or slowing-down time advantages can be extended to much lower neutron energies than the 0.01 eV limit which has already been demonstrated. These indications are provided by the theoretical calculations underway to explain the experimental results and observation that the front edges of the time pulses are probably limited by experimental resolutions at all energies observed. The expectation is that the moderators can be chosen which will yield cold neutrons very rapidly. In addition, several methods are demonstrated for shortening the time widths of the neutron pulses by chopping off the "die away" tails. These shorter pulses appear to be possible with corresponding (in the figure of merit sense) little loss in peak neutron pulse intensity. The developments toward shorter pulses have the promise that one cooled moderator source can be used for all neutron energies. While perhaps not optimum, one moderator can be sufficiently close to optimized to allow the reactor design to contain the corresponding simplification.

c) Operation for cold neutron research requires a much lower repetition frequency because of the long flight times needed for the low neutron velocities. The RPTF concept provides for low repetition frequencies for this purpose as well as for the accompanying higher power per pulse for materials testing purposes. It is to be recalled that in this mode peak fission rates of 10^{20} fissions/sec or 5×10^{15} total fissions per pulse are proposed. The corresponding power per pulse or power density per pulse is 1/4 the maximum to be obtained with Godiva II, which may not be attainable.

At this point, the chosen limit of the average power density of 0.5 MW/liter is to be re-emphasized because higher performance could certainly be obtained by increasing it. To some extent, the limit has been chosen because the proposed

performance is restricted by the "state of the art" for electron-neutron source systems. This limit exists because the present study includes a commercial neutron source machine reasonably available. However, there are also large reservations about the practical realization of higher power densities because of the severe problems posed by shock shutdown limitations and by transient radiation materials fatigue effects. This opinion should be taken in the light that this would be a first device of the type which is already a large extrapolation of current technology.

A figure of interest to the fatigue problems is the average volume per fission per pulse which is the volume which has 1 fission on the average per pulse in the reactor core. This is a cubic volume with 2.3 microns on the side for the 200 pps operation and 1.1 microns for 20 pps. These volumes are even smaller in the fuel when taking into account the dilution due to coolant and particle sizes or peak to average power. These are dimensions somewhat smaller than the fission fragment ranges, and the probability is high that several fission products traverse any specific volume 1 micron on a side during each reactor pulse. Such considerations point dramatically to the very real possibilities of rate dependent damage effects which are probably well represented by the limited material (nvt) lifetimes observed with Godiva systems. Refractory materials might change properties significantly with each pulse. It is because of such considerations that a liquid fuel has been considered most promising even though it might transfer large transient stresses to the fuel containment walls, but the latter would be alleviated by the 5-7% bubble containment and free surfaces. It is also anticipated that baffles can be used to design around such problems if necessary.

While the 200 pps operation appears less limited by power per pulse considerations, it is not clear that the shock waves may be more easily induced because of the steeper pulse wave fronts or faster rise time. There is, however, experimental evidence that the stress yield points are higher for higher frequency shocks. This subject is under debate, and such discussions only serve to emphasize the experimental character of the transient reactor operation being proposed.

III. WHERE AND HOW IN SUMMARY

Because high-average-power, pulsed reactors are fundamentally a new reactor type from the point of view of reactor operating experience, the first experimental reactor of this type most probably should be sited well away from centers of population. Such a pulsed reactor will require material performance under very unusual operation conditions calling for fuel development and testing under simulated and actual fission pulsing conditions. And, many new sets of operational procedures will be needed. Until reliability has been proven and operational experience demonstrated, remote siting would provide an additional assurance of public protection consistent with past AEC practice in carrying out its responsibilities.

Because interest has been expressed by several research and development groups in obtaining reactors of this type, the first facility should probably represent a cooperative planning and development program. To the extent that such altruistic objectives can be realized, efforts will and have been made to implement such cooperative efforts. In addition, industrial participation will be encouraged because of their potential contributions and because the knowledge and experience gained will be of great value to the power industry. Specifically, a market for such reactors is anticipated once they have been demonstrated.

IV. WHEN AND HOW – PROJECT AND DEVELOPMENT

A. Schedules

In order to complete construction on the project in FY 1973, an exacting schedule must be adhered to, and a valid reason for the early completion must be given. The reason given is, of course, "completion in time to contribute significantly to the DRDT fast reactor programs". Therefore, the presumption made here is that a valid need by the fast reactor program has been established, as outlined in Section I. Otherwise, completion of the project should be phased to provide a continuation of current MTR programs if the MTR is phased out of operation. This would provide an experimental test of this new type of pulsed research reactor before such reactors are constructed near population centers, and would provide an advanced facility for the continuation of INC development programs. Early completion of the project would also be desirable for reactor experiment and program reasons as well.

A bar graph showing a proposal schedule for the starting and completion dates of the development and project programs is given in Figure 7. To a significant extent, this must be considered preliminary and only representative of the problems which are posed by the transition from a development program into a project status. In the past, those problems which would provide the biggest performance returns have been pursued, but in the future all problems must be examined and solved in detail. A philosophy of a detailed examination of engineering problems and a complete detailed design of the system must evolve from the past philosophy of an examination of those areas which would yield the largest returns in the concept of such a system. An intermediate phase is now under way in which a clear distinction has been made in favor of the booster compared to the pulsed reactor, but significant gains can still be made by additional optimization. At the same time engineering examination of characteristic properties, limits, and critical design areas are also being undertaken.

Referring to Figure 7, moderator optimization and the problems posed by such transient operation have been discussed in some detail earlier in this report. Reactor Concepts as a distinct entry separate from Conceptual Design is made to emphasize a large degree of latitude of exploration of more attractive configurations and to point up the fact that the final concept should likely be based upon a choice of many or several concept proposals. The system presented here does not necessarily represent a final or best concept from Idaho Nuclear Corp. Engineering Problems as an item is included to cover the large areas of standard development and testing problems which are necessary to complete such a project. This item requires continuing expansion and will become more detailed as the project matures. The distinction between Fuel Development and Testing is to emphasize the need for an extensive testing program with unusual requirements. Control Systems include the rotating reactivity systems, transient instrumentation peculiar to the RPTF, and the usual complement of controls and instruments needed for a small liquid metal cooled fast reactor. It can be anticipated that important reactor performance gains can be made by improvements in the neutron source performance. The current concept only involves a small extrapolation of the "state of the art" of commercially available neutron sources. However, significant attention to the capability and evaluation of the

Figure 7

design parameters of such sources must be carried out by the staff. Procurement of the electron-neutron source will be from the industry with a minimum of development -- provided that another type of charged particle source is not found more desirable. Experience with existing and new neutron source systems will provide the best base for such evaluation. In the performance of critical experiments, utilization of existing facilities at ORNL is planned.

In the project area of Figure 7, noteworthy points are the early procurement of the neutron source and reactor fuel. The neutron source will be needed early to provide fuel test capabilities for the higher power energy transient studies. The early instigation of fuel procurement is in anticipation of problems in establishing a source of supply and the development of specifications and inspection methods. Some procurement, before testing is complete, serves to emphasize the need to have very close liaison between the development and fabrication functions because of the experimental nature of the new pulsed reactor which will place unusual demands on the fuel.

The area of project management and coordination is essentially a proposal that this function be performed by Idaho Nuclear Corporation. The program and project is planned to utilize the full capabilities of the AEC and reactor industry as they can be applied most efficiently to the RPTF. Because of the experimental nature of the project, extra attention is justified in reaching objectives as balances against development requirements. Because of the additional operational complications, early acquaintance and planning by operations personnel will be a necessity.

A block diagram of the functional aspects of the RPTF project is shown in Figure 8. This represents the complex interaction that exists between the development program, the area of design optimization and the anticipated applications of the facility. Closer liaison between the management decision function and the reactor users should be undertaken by expanding contacts with other AEC and university laboratories.

In Figure 8, the management function would be best carried out in the blocks to the right of the Title I and II blocks. In addition to the liaison and design decision functions above, responsibilities for contracts, scheduling, specifications and inspection, systems design descriptions, safety analysis, coordination and reporting, and acceptance and planning are proposed.

Additional development program areas are represented as evolving from the design problems resulting from the iterative design studies shown as loops. The functions in which effort has been expended by the INC are enclosed in heavy lines and those anticipated for future activity are enclosed in heavy dashed lines. The other functions must of course be included, but are left open with light lines because expectation of involvement of other organizations. The X'ed blocks with continuous lines are those for which appendices with appreciable documentation are included, and those with dashed X's require future expanded documentation for which schedules are shown in Appendix C. The shaded areas are those regions for which critical engineering design and research and development problems exist in the concept presented. Expansion of an evolution of such a functional block diagram will provide an outline of the project development.

Fig. 8 A block diagram of the functional aspects of the RPTF project.

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B. Current Programs and Planned Programs by INC

To date the INC pulsed reactor development program is to be classified as "decision oriented development research". The objective has been to scope the properties of such systems to determine their merits and limits and to explore those systems which are anticipated to yield the greatest gains in performance. For example, the moderator studies provide a basis for deciding between boosters and pulsed reactors. Some of this approach will continue until a final concept has been chosen. These development programs also satisfied a criteria of fairly broad general interest.

Major emphasis has been placed in the area of moderator studies where order-of-magnitude performance gains appeared to be possible. The current results are discussed in Appendix F where clearly short pulse advantages are demonstrated. The first measurements were intended to investigate the important parameters in determining and adjusting pulse widths, and all of those observed to date were found to be of importance. Methods which are available for providing narrower pulses and eliminating "die away" tails are: 1) reduction of moderator temperature below that for which neutrons are desired. For this purpose, the Maxwellian equilibrium distribution must not overlap the energy, 2) utilization of both homogeneous and heterogeneous poisons, 3) utilization of materials with high hydrogen densities and high inelastic scattering cross sections for very low energy neutrons and by varying the moderator dimensions. Important gains in neutron peak yields as well as in the pulse width can be gained by the thickness variation.

Although important pulse width improvement have been demonstrated, the subjects listed below remain to be worked out: a) a careful determination of the optimum thickness for a given material, b) a study of target moderator geometry effects, c) a study of the effect of materials in the vicinity of the moderators, and d) the determination of an absolute scale of the relation between the neutron flux at the reactor surface and the neutron leakage from the moderator as a function of energy.

Several of the above items can best be determined by converting thermal fluxes to fission neutron spectra by complete absorption of the thermal neutrons in fuel plates. This technique better simulates the reactor leakage spectrum and allows easier control of variables. It can be accomplished by using either thermal column neutrons or pulsed Cockroft-Walton 14 MeV moderated sources. Since it does not provide detailed neutron pulse time-distribution, cross checks with the Rensselaer linac will be made to test critical points. While the major portion of this program will be completed in FY 1968, a remaining portion will persist to confirm earlier results. Many refinements using actual reactor systems such as the SORA critical experiments at ORNL and the U-235 sphere at GA will be profitable. Additional gains of about a factor of two or less are expected whereas another factor of 10 is not anticipated.

The second major effort will be to develop techniques for target and fuel element tests using electron beam energy deposition methods. In particular instrumentation methods will be evolved for determining shock effects in nonfuel or nonradioactive liquid metal targets. Such information should assist in the evaluation of shock effects in the RPTF. Irradiations of materials for radiation damage tests will also be instigated if the information cannot be obtained from

concurrent pulse reactor and other reactor programs with special emphasis upon mechanical transient testing.

Because a more definitive conceptual design will be issued in FY 1968, reactor parametric studies will be continued to provide a better definition of reactor properties and their pulsing characteristics. Experimental data from the critical experiments of SORA under dynamic and pulsed source conditions and the GA U-235 sphere will be used to provide moderator neutron leakage yield normalizations. Also the effect of the moderator as a reflector to the reactors will be investigated to observe the amount of the fast reactor pulse broadening.

A cost vs benefit analysis to determine the merits of higher performance charged particle neutron sources will be instigated. Hydrodynamic calculations of dynamic heat and stress problems will be undertaken in conjunction with experimental measurements to investigate the transient mechanical properties. Mechanical and heat removal design studies on the mechanical reactivity rotor will be investigated. These rotor cooling problems are presently considered to have the severest engineering problems. They are also considered to be very critical in obtaining optimum performance. Considerable relaxation in the requirements could be tolerated for the highest frequency booster mode of operation, but this relaxation is not considered desirable because of its many side effects. Poorer high frequency response and poorer background reduction between pulses are examples of results of compromises in rotor performance. Heat removal and radiation damage problems for the cold moderator will be more extensively studied, and plans will be made for some development work to explore operational problems.

V. WHY - PROGRAM USAGE AFTER CONSTRUCTION

(See Appendix E for Details)

A. General

New technological advances necessarily imply risks because the unknown defies accurate predictions, but criteria for investment in advanced facilities still must be established for making decisions as to the relative merits of alternatives. Yields can be more specifically estimated for applications which are directly pertinent to well defined objectives. With this in mind, a considerable portion of this document has been devoted to scope the applicability of the facility to a specific development area of relatively direct and vital interest to the reactor programs. The RPTF is believed to have a much broader potential for discovery and development and many other fascinating possibilities exist.

In general, the operation experience, the technological support of industry, and the investment in training will all add benefits to the nuclear industry. Successful operation of the RPTF in the pulsed mode would definitely lend confidence to the operation of steady state fast reactors. Because successful operation of RPTF is imperative, exhaustive and conservative engineering approaches to its design and operational problems is imperative.

The specific materials programs to be carried out are to be carefully phased with reactor development needs, and Appendix E to be presented late in FY 1968 will provide more definitive plans. Quantitative estimates about the data to be obtained and the properties to be investigated are intended. These program plans will evolve as the appropriate LMFBF development programs become more definite, and cooperation with other laboratories is solicited.

As a facility for use in applied and basic neutron research, a case for significant intensity and resolution gains has been made. A corresponding presentation and discussion of the merit for the long range research programs contemplated will be detailed in the forthcoming Appendix E. Some general comments will, however, be given here.

The need to discuss the RPTF in the sense of a national facility or a step toward a national facility appears worthy of discussion because of "exciting prospects" that the RPTF can be used to explore new and vast fields of research. The size of effort is, of course, very modest compared to high energy physics programs. Those to be mentioned have large merit, in the Weinberg sense, because they could quickly yield practical value.

One area about which several suggestions have been made requires the phasing of some transient phenomenon such as pressure, magnetic fields, temperature, chemical reaction, phase change -- with reactor pulsing so as to study the transient change of state by diffraction and neutron inelastic scattering. Generally, rate-dependent phenomena which have rates compatible with the

different pulse timing regions should be examined to determine the sensitivity of the neutron techniques for supplying information and determining the usefulness of the information to be obtained. Such applications are probably unique and the advantages would be expected to have a "peak-flux-squared" type of dependence.

A second area worthy of national consideration is the possibility of marked contributions by pulsed neutron diffraction to the structure of organic and biological materials. Although the classical methods have superior prospects because of a larger tradition of refinement, it is already clear that complementary advantages would be available because of much higher resolutions for the largest spacings. This would provide perhaps improved methods for observing long range effects.

The intention is to encourage RPTF utilization by many groups, and methods similar to those employed in the use of the high energy physics facilities should perhaps be explored. If the assumption is made that important national goals are involved, it readily follows that such considerations should be examined.

Two other areas in which potentially new and unique capabilities for research purposes should be explored involve those cases in which the phenomena to be investigated depend upon the square of the peak fluxes or power densities. An example of such a problem is that of neutron-neutron scattering which has been described before [4]. Another distinct possibility for interest has been implied earlier in this proposal where fission damage effects are postulated to be rate dependent. Further examination of such ideas will bring in concepts of time coincidence, fission fragment cross section, time-dependence and cross sections of the temperature of energy spike of the fragment track, etc.

B. Proposed INC Program -- General Philosophy

As envisioned, the INC program involves a nuclear science and technology "mix" involving both basic and applied or development programs. As is presently the case, both program areas are concerned with the same subject materials but have different objectives. A close coupling between the two, however, provides extra returns to the project areas because they are thereby maintained at "state of the art" levels in knowledge and instrumentation. At the same time the basic programs are more closely oriented toward the technology needs. Examples of returns from the latter are: a) the demonstration of the importance of fission-interference phenomena to thermal reactors and to the nuclear Doppler coefficient, b) the "handbook" usage of gamma ray spectrum catalogues as a standard of the field running the gamut from isotope usage to reactor shielding and gamma heating calculations. In both cases, the long range returns are yet to be realized because of the general importance of such information to long term reactor economies in terms of burnout predictions and the control of fissile materials.

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APPENDIX A

GENERAL PROPERTIES OF REPETITIVELY PULSED SYSTEMS

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APPENDIX A

General Properties of Repetitively Pulsed Systems

I. INTRODUCTION

Methods of providing intense neutron sources for research and development uses are being considered by various laboratories in different countries around the world [1, 2, 3-A]. One very promising way of obtaining significant gains in neutron intensity, for a variety of applications, over those presently available in research and testing reactors is to have the reactor operate in the pulsed mode. This is to say that the reactor is on for only a short period of time and essentially off for the remaining time. This has a particular advantage, for example, in time-of-flight experiment in that neutrons are produced only when they are needed. In order for a steady state reactor to provide the same relative intensity for an experiment of the above type, it would have to operate at very high power levels, and the operating and construction costs would be substantially more than a reactor operating in the pulsed mode. Since a pulsed reactor can provide a very intense neutron source, at relatively low cost compared to an equivalent steady state reactor, the characteristics and potential of these reactors as well as the problems associated with them need to be investigated.

In Section II, the concept of a reactor as a neutron amplifier is discussed. The application to several modes of pulsed neutron amplifiers is treated in Section III. Examples of such pulsed reactor systems are given, and the appropriate solutions to the point-reactor-kinetics equations describing certain of these systems are developed. In Section IV, possible pulse characteristics are discussed which cover the operational region of the Repetitively Pulsed Test Facility (RPTF) as a "booster" with and without repetitive reactivity insertion. Sensitivities of the reactor type to changes in neutron lifetimes, delayed neutron fractions, and the eta (neutrons produced per neutron absorbed) are calculated for the reactor fuels U-235, Pu-239, and U-233. The results of the study are summarized in Section V.

II. REACTORS AS NEUTRON AMPLIFIERS

The effective multiplication factor for a reactor can be defined as follows:

$$k(t) \equiv \frac{\text{rate of release of prompt plus potential neutrons at time } t}{\text{rate of loss of actual neutrons by absorption and leakage at time } t}$$

Here "potential neutrons" are those delayed neutrons which will be produced by the precursors which are made. In the usual way, for a reactor with no external source, if

$k(t) > 1$, the neutron population increases with time and the reactor is super-critical

$k(t) = 1$, the neutron population neither increases nor decreases with time and the reactor is delayed critical, and

$k(t) < 1$, the neutron population decreases with time and the reactor is subcritical.

A specific value of $k(t)$ of interest in the supercritical range is the prompt critical value. At prompt critical the reactor would be critical on prompt neutrons only.

In a subcritical reactor (constant k) with an external steady state source, there is amplification of the source due to the fissions taking place in the fuel. The total gain can be defined as the number of neutrons produced per neutron introduced. For a subcritical reactor with a steady state source, this total gain or subcritical multiplication factor is $1/(1-k)$. In addition to total gain, when a subcritical reactor is pulsed with an external source, it is desirable and meaningful to talk about the prompt gain, that is, the number of prompt neutrons produced per neutrons introduced. For a pulsed subcritical reactor the prompt gain more correctly the prompt amplification is given by $k/(1-\beta^*)k$, where β^* is the effective delayed neutron fraction. In addition, the concept of prompt gain can be extended to k values just below prompt critical because as far as prompt neutrons alone are concerned, the reactor is subcritical. A plot of total and prompt gain is shown in Figure 1-A. As can be seen, when k approaches 1, the total gain becomes infinite while the prompt gain is finite. However, the large total gains imply either a steady state or equilibrium situation. Since the prompt gain is dependent on the delayed neutron fractions there is a significant difference in the prompt gain for the various fuels at a fixed k . This implies, for example, that for the same gain, less power would have to be removed from a reactor fueled with Pu-239 than one fueled with either U-233 or U-235. It can also be implied from these curves that a large prompt gains (but $k < 1$) the pulse characteristics would not be too good because of the large total gain in comparison to the prompt gain. Also, in order to get prompt gains larger than about 480 for Pu-239, 370 for U-233 and 155 for U-235, then k has to be larger than 1 (obviously for only a short period of time).

III. MODES OF PULSE PRODUCTION

There are a number of modes in which a repetitively pulsed (multiple pulses per second) reactor can operate. By specifying that the reactor be pulsed many times a second, such pulsed reactor types as SPERT[5-A] and Godiva[6-A] will be excluded. Thus, the modes of interest for repetitive-pulse operation are:

- (1) repetitive pulsing of a subcritical (constant k) reactor by an external source
- (2) vary k between two values which are less than delayed critical and simultaneously pulse the reactor with an external source when k is around its maximum value

Fig. 1-A A comparison is made of the actual prompt neutron gains for U-233, U-235, and Pu-239 with the total gains for a reactor operating with varying effective multiplication factors.

- (3) vary k from a value below delayed critical to some value above prompt critical
- (4) vary k from a value less than delayed critical to some value between prompt and delayed critical and simultaneously pulse the reactor with an external source when k is around a maximum
- (5) vary k from some value below delayed critical to a value above prompt critical and simultaneously pulse the reactor with an external source around maximum k .

A. Examples of Repetitively Pulsed Reactors

Various reactors have either operated or have been proposed to operate in most of the above modes. Examples of systems utilizing the first mode of operation are the Harwell booster [7-A] and a reactor proposed by Fluharty et al [8-A]. The Harwell system is an operating system and consists of a subcritical U-235 fueled reactor that is pulsed by an electron linear accelerator. The prompt gain of the reactor is approximately 10 and electron pulse widths of a few tenths of a microsecond are used to pulse the reactor at a maximum repetition rate of 750 pulses per second. The leakage neutrons from the reactor are moderated to the 1 eV-100 KeV energy region, and these neutron pulses, which are on the order of 0.5 microsecond, are then used for various experimental purposes.

As was seen in Figure 1-A, for a subcritical reactor with a constant k , when the effective multiplication factor is increased to get larger prompt gains, the total gain of the system increases even more. Thus, the fraction of energy that is generated in the prompt portion of the pulse becomes less and less and the pulse yield characteristics of the reactor degenerates. In order to circumvent this problem of poor pulse characteristics at high gains, Fluharty et al proposed a subcritical liquid Pu reactor in which the fuel was circulated and the delayed neutron precursors removed. With the delayed neutron precursors removed, a larger fraction of energy would be produced in the prompt portion of the pulse and the pulse yield characteristics would correspondingly improve. This particular concept was a proposal only, and the system was never built. The reactor was to be pulsed by a linear accelerator and the fuel itself was to be the target for the accelerator. One advantage of such a flowing liquid fueled reactor would be that high power and high power densities could be achieved.

The Harwell Super Booster [9-A] is a reactor that has been proposed to operate in the second mode. That is to say that the super booster would be a subcritical assembly driven by an electron linear accelerator. The reactivity of the reactor would be modulated in phase with the pulsing from the linac so as to maintain the highest possible prompt multiplication at the moment of the pulse, and to minimize the steady background produced by delayed neutrons. The reflected reactor would be fueled with PuC and cooled with Na or NaK and would have an average power of 9 MW. The reactor would be pulsed by rotating a reflector piece.

The only reactor that has been operated in mode three is the Russian reactor, IBR[10-A, 11-A], located in the Dubna Institute. The reactor consists of Pu rods, separated by a gap. Within this gap moves the outer edge of a wheel 1.1 meters in diameter, into which a U-235 disc has been pressed. A second smaller wheel rotates past the outer surface of the reactor. Both wheels have variable speeds and the reactor goes super prompt critical only when both inserts are in their most reactive positions. The small wheel rotates at a submultiple of the larger wheels' speed and this allows variable pulse repetition frequency with a minimum of pulse distortion. The average power is about 6 Kw and it is anticipated that it will be raised to 30 kW. The IBR has also been operated in conjunction with a microtron[12-A]. This reduced the width of the pulses from 36 to 3 or 4 μ sec.

A second reactor which would operate in mode three is the SORA reactor [13-A, 14-A]. The SORA reactor is a EURATOM effort and has been under design since 1962. It has been proposed to build it at the ISPRA center in Italy. Unlike the IBR, SORA is only in the design state and it would use a reflector cycling to oscillate reactivity. The reactor would be initially fueled by a U-Mo alloy enriched to approximately 93% in U-235. The reactor would be pulsed by repetitively passing a Be reflector piece past a reactor surface. The pulsating device has a diameter of 1.8 meters and tip velocity of about 300 m/sec. The reactor would have an average power of 0.6 MW and average power density of 0.1 MW/l. It would produce pulses having a half width of 50 μ sec.

Brookhaven National Laboratory has been studying a U-233 fueled reactor which is called the Advanced Pulsed Reactor[15-A]. This reactor would be fueled with a U-233 alloy and cooled with Na. It would operate in mode three and is proposed to have an average power of 30 MW and an average power density of 1.5 MW/l. It would have two pulsation devices and produce neutron bursts having pulse widths at half maximum of 60 μ sec.

Several laboratories have investigated reactors which could operate in mode four. Brookhaven has also studied a reactor which is called the Advanced Booster[15-A]. This reactor would be pulsed both by varying the reactivity as well as by pulsing with a linear accelerator. The fuel would be a U-233 alloy and the coolant would be Na. Reactivity cycling is via a rotating reflector and the pulsating device would have a diameter of 1.27 meters. The reactor would produce pulses having a width of 6.5 μ sec and is proposed to have an average power and power density of 9 Mw and 1.5 MW/l respectively.

General Atomics Laboratory has been studying a reactor system which is called an Accelerator Booster Fast Reactor [16-A, 17-A]. The operation of this device would fall into mode four in that the reactivity of the reactor would oscillate from a value between delayed and prompt critical to some value below delayed critical. Many methods of reactivity variation such as rotating a slotted reflector to cycling fuel in and out of the reactor were investigated. Primarily gas cooled reactors were investigated; however, Na and NaK cooled reactors were studied as well. A reactor power of 0.5 Mw having a power density of 0.185 MW/l with gas cooling has been proposed.

The Repetitively Pulsed Test Facility (RPTF), which is the concept proposed in this report, would operate in mode four. The reactor would have an average power of 3.3 MW and average power density of 0.55 MW/l. It would produce pulses having widths of 5.4 μ sec. It should be noted that any system which can operate in mode four conceivably could operate in any of the five modes listed.

No reactor system has specifically been proposed to operate in mode five.

B. Point Reactor Kinetics Equations and Solutions

The general point reactor kinetics equations are:

$$\frac{dN(t)}{dt} = \left\{ \frac{(1-\beta^*)}{\ell(t)} k(t) - 1 \right\} N(t) + \sum_i \lambda_i C_i(t) + S(t) \quad (1-A)$$

$$\frac{dC_i(t)}{dt} = \frac{k(t)\beta_i^* N(t)}{\ell(t)} - \lambda_i C_i(t)$$

where

$N(t)$ = the total number of neutrons in the reactor at time t

$k(t)$ = the effective neutron multiplication factor at time t

$\ell(t)$ = the prompt neutron lifetime at time t

$C_i(t)$ = the average concentration of type i neutron precursors at time t

λ_i = the probability per unit of type i neutron precursors will decay

$S(t)$ = the rate at which neutrons are introduced from external sources at time t

β_i = the fraction of i type delayed neutrons per fission

$$\beta = \sum_i \beta_i$$

and

$\beta^* = \lambda \beta$ = the effective delayed neutron fraction.

In repetitively pulsed reactor operation a very essential criteria for successful operation (time-independent pulse amplitudes) is that the time averaged precursor concentration remains constant, or

$$\frac{dC_i(t)}{dt} = 0$$

then the $\lambda_i C_i(t)$ term is set equal to $\bar{k}\beta^*\bar{N}/\bar{\ell}$ where the average quantities indicated are averaged over one period^[18-A, 16-A]. Then Equation (1-A) becomes:

$$\frac{dN(t)}{dt} = \left\{ \frac{(1-\beta^*) k(t)-1}{\ell(t)} \right\} N(t) + \frac{\bar{k}\beta^*\bar{N}}{\bar{\ell}} + S(t) \quad (2-A)$$

Sometimes the following definitions are made

$$\rho(t) = \frac{k(t)-1}{k(t)} \quad \text{and} \quad \Lambda(t) = \frac{\ell(t)}{k(t)}$$

and Equation (2-A) rewritten as

$$\frac{dN(t)}{dt} = \left\{ \frac{\rho(t)-\beta^*}{\Lambda(t)} \right\} N(t) + \frac{\bar{k}\beta^*\bar{N}}{\bar{\ell}} + S(t) \quad (3-A)$$

where $\Lambda(t)$ is called the neutron generation time at time t and $\rho(t)$ is called the reactivity at time t . Also, $K(t)$ is sometimes defined as $\rho(t)/\beta^*$ and $\Lambda^*(t)$ is set equal to $\Lambda(t)/\beta^*$. Then Equation (3-A) becomes

$$\frac{dN(t)}{dt} = \left\{ \frac{K(t)-1}{\Lambda^*(t)} \right\} N(t) + \frac{\bar{k}\beta^*\bar{N}}{\bar{\ell}} + S(t) \quad (4-A)$$

where $\Lambda^*(t)$ is the reduced neutron generation time and $K(t)$ is the reactivity in dollars at time t . In this work, Equation (2-A) will be used and rewritten as

$$\frac{dN(t)}{dt} = -\alpha(t)N(t) + \frac{\bar{k}\beta^*\bar{N}}{\bar{\ell}} + S(t) \quad (5-A)$$

where

$$\alpha(t) = \frac{\Delta k_p(t)}{\ell(t)}$$

and

$$\Delta k_p(t) = 1 - (1-\beta^*)k(t)$$

Now,

$$\frac{N(t)}{\ell(t)} = \text{the neutron loss rate at time } t$$

and

$$N_L(t) = \left\{ \frac{n-k(t)}{n} \right\} \frac{N(t)}{\ell(t)} = \text{the neutron loss rate by leakage and non-fuel absorptions at time } t.$$

Also

$$N_p(t) = \frac{k(t)N(t)}{\ell(t)} = \text{the neutron production rate at time } t$$

and

$$P(t) = \frac{\delta F_{\text{I}} k(t) N(t)}{v \ell(t)} = \text{the power production rate at time } t$$

where

η = the average number of neutrons released per absorption in the fuel

ν = the average number of neutrons released per fission

E_f = the energy released per fission

and δ = a conversion constant.

In order to obtain analytical solutions for $N(t)$ from Equation (2-A) it is usually assumed that $\ell(t)$ is a constant, ℓ_o . Now, with $N(t) = N_s(t) + N_b(t)$ and $S(t) = S_s(t) + S_b(t)$, Equation (5-A) can be rewritten as follows:

$$\left. \begin{aligned} \frac{dN_s(t)}{dt} &= -\alpha(t) N_s(t) + S_s(t) \\ \frac{dN_b(t)}{dt} &= -\alpha(t) N_b(t) + \frac{\bar{k}\beta^*\bar{N}}{\ell_o} + S_b(t) \end{aligned} \right\} (6-A)$$

where

$N_s(t)$ = the number of prompt neutrons associated with the external source at time t

$N_b(t)$ = the number of neutrons associated with the delayed neutrons at time t

$S_s(t)$ = rate at which source neutrons are introduced at time t

and $S_b(t)$ = rate at which background (spontaneous fission etc) neutrons are introduced at time t

In the most general type of repetitively pulsed reactor operation (mode four), the effective multiplication factor is kept at some constant value below delayed critical. The reactor multiplication is then varied in some fashion and when it is at its maximum value, neutrons from an external source are introduced for a short period of time. The multiplication of the reactor is then reduced to its constant value and the entire sequence is repeated in a periodic fashion. The following time sequence will be used for the remainder of the discussion:

where

$t = 0$ is the time when $k(t)$ changes from its constant value

$t = t_1$ is the time at which the external source is turned on

$t = t_p$ is the time when $k(t)$ reaches its maximum value

$t = t_2$ is the time at which the external source is turned off

$t = t_k$ is the time when $k(t)$ returns to its constant value

and $t = T$ is the time at which the entire sequence is repeated.

If it is assumed that

$$\begin{aligned} S_s(t) &= S_s & t_1 < t < t_2 \\ &= 0 & \text{everywhere else} \end{aligned}$$

and

$$S_b(t) = S_b \quad 0 \leq t \leq T$$

then, analytical solutions to Equation (6-A) can be obtained for two cases of assumed variations of $k(t)$. The two assumed methods of varying $k(t)$ are:

Case 1 (reactor is always subcritical)

$$k(t) = k_o \quad 0 \leq t \leq T$$

Case 2 (step change in $k(t)$)

$$\begin{aligned} k(t) &= k_m \quad 0 < t < t_k \\ &= k_o \quad \text{everywhere else} \end{aligned}$$

Now, for Case 1

$$\begin{aligned} N_s(t) &= 0 \\ N_s(t) &= \frac{S_s}{\alpha} \{1 - e^{-\alpha(t-t_1)}\} \\ N_s(t) &= \frac{S_s}{\alpha} \{1 - e^{-\alpha T_s}\} e^{-\alpha(t-t_2)} \\ N_b(t) &= \frac{1}{\alpha} \left\{ \frac{k\beta^* \bar{N}}{\ell_o} + S_b \right\} \end{aligned} \quad \left. \begin{aligned} 0 &\leq t \leq t_1 \\ t_1 &\leq t \leq t_2 \\ t_2 &\leq t \leq T \\ 0 &\leq t \leq T \end{aligned} \right\} \quad (7-A)$$

where

$$\alpha = \frac{\Delta k_p}{\ell_o}$$

$$\Delta k_p = 1 - (1 - \beta^*) k_o$$

$$T_s = t_2 - t_1$$

and

$$T \text{ is such that } e^{-\alpha(T-t_2)} \rightarrow 0.$$

Also

$$N_L(t) = \left\{ \frac{\eta - k_o}{\eta \ell_o} \right\} N(t) \quad 0 \leq t \leq T \quad (8-A)$$

and

$$P(t) = \frac{\delta E_f k_o}{v \ell_o} N(t) \quad 0 \leq t \leq T \quad (9-A)$$

In addition, for $S_b = 0 : 0$

$$\text{Total Gain} = \frac{\frac{k_o}{l_o} \int_0^T N(t) dt}{S_s T_s} = \frac{k_o}{1-k_o}$$

and

$$\text{Prompt Gain} = \frac{\frac{k_o}{l_o} \int_0^T N_s(t) dt}{S_s T_s} = \frac{k_o}{1-(1-\beta^*)k_o}$$

$$\text{Prompt Leakage Gain} = \frac{\left\{ \frac{\eta-k_o}{\eta l_o} \right\} \int_0^T N_s(t) dt}{S_s T_s}$$

Also, for Case 2 [16-A, 17-A]

$$\left. \begin{aligned} N_s(t) &= 0 & 0 \leq t \leq t_1 \\ N_s(t) &= \frac{S_s}{\alpha_1} \{1 - e^{-\alpha_1(t-t_1)}\} & t_1 \leq t \leq t_2 \\ N_s(t) &= \frac{S_s}{\alpha_1} \{1 - e^{-\alpha_1 T_s}\} e^{-\alpha_1(t-t_2)} & t_2 \leq t \leq t_k \\ N_s(t) &= \frac{S_s}{\alpha_1} \{1 - e^{-\alpha_1 T_s}\} \{e^{-\alpha_1(t_k-t_2)}\} e^{-\alpha_2(t-t_k)} & t_k \leq t \leq T \\ N_b(t) &= C \left\{ \frac{1}{\alpha_1} - \left(\frac{1}{\alpha_1} - \frac{1}{\alpha_2} \right) e^{-\alpha_1 t} \right\} & 0 \leq t \leq t_k \\ N_b(t) &= C \left\{ \frac{1}{\alpha_2} + \left(\frac{1}{\alpha_1} - \frac{1}{\alpha_2} \right) [1 - e^{-\alpha_1 t_k}] e^{-\alpha_2(t-t_k)} \right\} & t_k \leq t \leq T \end{aligned} \right\} (10-A)$$

where

$$\alpha_1 = \frac{1-(1-\beta^*)k_m}{l_o} \quad \text{and} \quad \alpha_2 = \frac{1-(1-\beta^*)k_o}{l_o}$$

Therefore,

$$\bar{k} = \frac{1}{T} \{k_m t_k + k_o (T-t_k)\}$$

$$\bar{N} = \left\{ \frac{S_s}{\alpha_1} \left(T_s - \left[\frac{1}{\alpha_1} - \frac{1}{\alpha_2} + \frac{e^{-\alpha_2(T-t_k)}}{\alpha_2} \right] [1 - e^{-\alpha_1 T_s}] e^{-\alpha_1(t_k - t_2)} \right) \right.$$

$$\left. S_b \left\{ \frac{T}{\alpha_2} + \left(\frac{1}{\alpha_1} - \frac{1}{\alpha_2} \right) t_k - \left(\frac{1}{\alpha_1} - \frac{1}{\alpha_2} + \frac{e^{-\alpha_2(T-t_k)}}{\alpha_2} \right) \left(\frac{1}{\alpha_1} - \frac{1}{\alpha_2} \right) [1 - e^{-\alpha_1 t_k}] \right\} / D \right.$$

$$D = T - \frac{\beta^* \bar{k}}{\ell_o} \left\{ \frac{T}{\alpha_2} + \left(\frac{1}{\alpha_1} - \frac{1}{\alpha_2} \right) t_k - \left(\frac{1}{\alpha_1} - \frac{1}{\alpha_2} + \frac{e^{-\alpha_2(T-t_k)}}{\alpha_2} \right) \left(\frac{1}{\alpha_1} - \frac{1}{\alpha_2} \right) \right.$$

$$\left. [1 - e^{-\alpha_2(T-t_k)}] \right\}$$

$$C = \frac{\bar{k} \beta^* \bar{N}}{\ell_o} + S_b$$

and

T is such that $e^{-\alpha_2(T-t_k)} \rightarrow 0$.

Also,

$$N_L(t) = \frac{N(t)}{\ell_o} - \frac{k(t) N(t)}{\eta \ell_o} \quad 0 \leq t \leq T \quad (11-A)$$

$$P(t) = \frac{\delta E_f k(t) N(t)}{\nu \ell_o} \quad 0 \leq t \leq T \quad (12-A)$$

In addition,

$$\text{Total Gain} = \frac{\int_0^T \frac{k(t) N(t)}{\ell_o} dt}{S_s T_s + S_b T} = \frac{A + B}{S_s T_s + S_b T}$$

where

$$A = \frac{S_s}{1 - (1 - \beta^*) k_m} \left\{ k_m^* T_s - \left[\frac{k_m}{\alpha_1} - \frac{k_o}{\alpha_2} + \frac{k_o e^{-\alpha_2(T-t_k)}}{\alpha_2} \right] (1 - e^{-\alpha_1 T_s}) e^{-\alpha_1(t_k - t_2)} \right.$$

$$B = \frac{C}{\ell_o} \left\{ \frac{k_o T}{\alpha_2} + \left(\frac{k_m}{\alpha_1} - \frac{k_o}{\alpha_2} \right) t_k - \left(\frac{k_m}{\alpha_1} - \frac{k_o}{\alpha_2} + \frac{k_o e^{-\alpha_2(T-t_k)}}{\alpha_2} \right) \left(\frac{1}{\alpha_1} - \frac{1}{\alpha_2} \right) (1 - e^{-\alpha_1 t_k}) \right\}$$

and

$$\text{Prompt Gain} = \frac{\int_0^T \frac{k(t) N_s(t) dt}{\ell_o}}{S_s T_s} = \frac{A}{S_s T_s}$$

$$\text{Prompt Leakage Gain} = \frac{\int_0^T \left\{ \frac{N_s(t)}{\ell_o} - \frac{k(t) N_s(t)}{\eta \ell_o} \right\} dt}{S_s T_s}$$

C. Pulse Widths

One way of defining an "effective pulse width" from a reactor is as follows [16-A, 17-A]:

$$\tau = \frac{\int_0^T N_s(t) dt}{N_s(\text{max})}$$

In some cases the full width of the pulse at half maximum is quoted for the reactor pulse width. In discussing experimental applications of the reactor pulse, the usual definition of the pulse width is twice the square root of the variance, σ^2 , where

$$\sigma^2 = \frac{\int_0^T N_s(t) (t - \langle t \rangle)^2 dt}{\int_0^T N_s(t) dt}$$

However, in this report τ will be used when discussing reactor pulse widths.

IV. COMPARISON OF FUELS FOR REPETITIVE PULSED REACTORS

Since one of the primary functions of pulsed reactors is to feed relatively short bursts of neutrons into moderators, only fast reactors need be considered because of their small neutron lifetimes.

In order to illustrate the degradation of the prompt pulse characteristics at high gains of subcritical (constant k) reactors, repetitively pulsed with an accelerator, calculations were made for various prompt gains using U-235, U-233 and Pu-239 as fuel. As a first approximation it was assumed that the prompt neutron lifetimes for the various reactors with different fuel were the same. The

effects of different delayed neutron fraction, number of neutrons released per fission, etc, can then be seen. It should be mentioned that only approximate attempts were made to define effective η 's, β 's, etc, for the appropriate cases. As can be seen in Table I-A, as the prompt gain is varied from 50 to 150, the prompt pulse yield characteristics become poorer. One infers this from the ratio of the peak power to average background power and the fraction of energy in the prompt pulse. This effect is most pronounced for U-235 and least for Pu-239. In addition, the average power increases as the prompt gain is increased, and it can become prohibitively large as indicated by the U-235 reactor at a prompt gain of 150. For the same prompt gain the Pu-239 fueled reactor produces a lower power than either the U-233 or U-235 fueled reactors. The advantage of having a fuel with a large η is indicated in Table I-A by the prompt leakage gain and leakage rate parameters. Thus, keeping in mind the assumptions made, it can be concluded that there is a definite advantage of having a subcritical reactor as described above fueled with Pu-239 instead of U-235. Plutonium also is an improvement over U-233 but the advantage is not as dramatic as it is for U-235. It should be noted that advantages in pulse yield characteristics as indicated by the peak-to-average background (signal-to-noise) ratio can only be completely realized if a moderator would not spread the reactor pulse out in time in the thermalizing process. This would be definitely true for neutrons of energies greater than 0.1 eV and, depending on moderator optimization, may be realized at lower neutron energies as discussed in Appendix F.

A plot of typical power pulses as calculated with Equation (7-A) for these subcritical reactors is given in Figure 2-A. The prompt gain for each of the reactors is 100. However, as was noted previously, for the same gain, less power would have to be removed from Pu-239 than either U-235 or U-233. This is exemplified in the peak power and average background power quantities for they are significantly higher for U-235 than Pu-239. Again, it can be seen that Pu-239 is better in all respects than U-233 but the differences are not as large as for the U-235, Pu-239 comparison.

In order to illustrate the effect on reactor pulse shape characteristics of varying the prompt neutron lifetime, calculations were performed for a subcritical reactor (constant k) fueled with Pu-239 and repetitively pulsed with an external source. It was assumed that η and β^* remained fixed even though ℓ_0 changed. A plot of total loss rate by leakage and non-fuel absorption as calculated with Equation (8-A), is given in Figure 3-A for a prompt source gain of 250. Prompt neutron lifetimes of 5×10^{-8} , 1×10^{-8} , 5×10^{-9} , and 1×10^{-9} sec were used in the calculations. As can be seen in Figure 3-A a reactor with a lifetime of the order of 10^{-9} sec would have very good high frequency response. A reactor with such good response characteristics would show a very definite advantage in the high energy region (where thermalization effects are negligible) and the advantage would go as the peak intensity over the square of the pulse width. However, a reactor with such a short lifetime is probably not attainable (the LAMPRE reactor which was a fast system had a lifetime of 8×10^{-9} sec) but it does serve to indicate what kind of lifetimes would give a reactor with very good response characteristics. Practically speaking, typical lifetimes being quoted for repetitively pulsed reactors lie in the range of $1 - 4 \times 10^{-8}$ sec. To a point, large gains can be made in lowering the neutron lifetime. This is indicated in Figure 3-A where τ is 14.6 μ sec for an ℓ_0 of 5×10^{-8} sec and 5.0

TABLE I-A

COMPARISON OF SOURCE PULSED SUBCRITICAL (CONSTANT k) REACTORS
FUELED WITH DIFFERENT FISSILE MATERIALS

Comparison of Source Pulsed Subcritical (Constant k) Reactors Fueled with Different Fissile Materials

($S_s = 9.5 \times 10^{17}$ neutrons/sec, $\ell_c = 1 \times 10^{-8}$ sec, $T_s = 4$ μ sec, Rep. Rate = 200 pps)

Parameter	U-235			U-233			Pu-239		
	$\beta^*=0.00241, \nu=2.46, \eta=2.13$			$\beta^*=0.00267, \nu=2.51, \eta=2.33$			$\beta^*=0.00204, \nu=2.93, \eta=2.69$		
Prompt Gain	50	100	150	50	100	150	50	100	150
Prompt Leakage Gain [†]	27	54	80	29	58	86	32	64	95
Total Gain	74	279	3396	56	136	250	56	126	216
Average Total Power (MW)	0.71	2.58	37.4	0.55	1.30	2.39	0.47	1.07	1.83
Peak Power/Av. Bkg. Power	2650	683	48	8111	3368	1741	11 002	4786	2637
% Prompt Energy in Pulse	68	36	3.9	84	73	60	90	80	69
Average Prompt Leakage Rate [†] (neutrons/sec)	2.07 $\times 10^{16}$	4.07 $\times 10^{16}$	6.06 $\times 10^{16}$	2.21 $\times 10^{16}$	4.39 $\times 10^{16}$	6.55 $\times 10^{16}$	2.46 $\times 10^{16}$	4.83 $\times 10^{16}$	7.21 $\times 10^{16}$
Effective Pulse Width (μ sec)	4.0	4.1	4.3	4.0	4.1	4.3	4.0	4.1	4.3

[†] Maximum possible

Fig. 2—A Details are presented of the relative power pulse shapes for U-233, U-235, and Pu-239 for the same gain and neutron lifetimes. Note that the peak amplitude of the U-235 is highest because the pulse is added on to a higher background, and the peak to background ratio is far superior for Pu-239.

Fig. 3-A The effect of variation in the prompt neutron lifetime on the reactor pulse shape is illustrated. The areas under each pulse are the same because the gains are the same, while the instantaneous or rate gains at the peaks are noticeably depressed for longer neutron lifetimes.

μsec for an ℓ_0 of 1×10^{-8} sec. However, in decreasing ℓ_0 to 5×10^{-9} sec τ was reduced to $4.2 \mu\text{sec}$. Thus, if the advantage can be taken of a high frequency response system then small neutron lifetimes are desirable, and to a point, large gains can be made in reactor pulse shape characteristics by lowering ℓ_0 . Also, it is desirable to have as large an η as possible because this helps to optimize the neutron leakage.

To see the magnitude of the effect of η on the neutron leakage term, calculations were performed for a Pu-239 fueled subcritical reactor where ℓ_0 and β^* were assumed to be constant and only η varied. Now this is not necessarily a good assumption for all cases because η and ℓ_0 are definitely coupled to each other. Nevertheless, Table II-A shows the results of the calculations for a prompt gain of 250. Values of η were calculated for an assumed average neutron energy and a value of the maximum average neutron leakage rates were computed. As can be seen in Table II-A the effect on the leakage rate is not too great. A larger effect is observed in going from one fuel to another as indicated in Table I-A. Thus the variation in η is greater when η changes because a different fuel is used than when η changes because the neutrons' energy changes.

As was seen in Table I-A, at large gains the pulse yield characteristics of a reactor get poorer when the reactor is always subcritical (constant k). A way of retaining good pulse yield characteristics at high gain is to vary $k(t)$ in some periodic fashion. A typical way of doing this would be to let $k(t)$ be some value between delayed and prompt critical for a short period of time and below delayed critical for the majority of the time (mode four operation). Also $k(t)$ could be varied in such a fashion to fulfill either mode two, three, or five conditions. This type of operation would allow higher prompt gains to be attained than those available with a constant subcritical reactor unless the technique such as proposed by Fluharty is used.

In order to study reactor pulse characteristics when $k(t)$ is varied, the equations derived in case 2 will be used as a first approximation. In this case, $k(t)$ was assumed to be a step function. In some instances this may be a reasonable representation of how $k(t)$ varies when it is changed by rotating either fuel or reflector as the pulsating mechanism. In general a more realistic variation of $k(t)$ would be as a parabola [13-A, 14-A, 18-A]. However, with a step change in $k(t)$, an analytical solution to Equation (6-A) can be found as was shown in Section III. Only reactors operating in modes two or four will be calculated using $k(t)$ as a step function. In general, the assumption of a step change in $k(t)$ will give pessimistic results for such reactor parameters as average power and pulse width, and optimistic results on such quantities as gains and number of leakage neutrons. Nevertheless, the assumption of a step change in $k(t)$ should allow comparisons to be made of the relative pulse yield characteristics and reactor parameters for reactors fueled with U-235, U-233, and Pu-239. In order to see the advantage of varying $k(t)$ versus not varying it, Table III-A lists some reactor characteristics for a Pu-239 fueled reactor operated at a prompt gain of 250.

As can be seen in Table III-A, by oscillating $k(t)$, the pulse yield characteristics and reactor operating parameters are greatly improved. When the effective multiplication is constant, the average reactor power is larger than

Table II-A

Sensitivity of Neutron Leakage to Variations in η

($S_s = 9.5 \times 10^{17}$ neutrons/sec, $\ell_o = 1 \times 10^{-8}$ sec, $T_s = 4 \mu\text{sec}$,
 $\beta^* = 0.00204$, Rep. Rate = 200 pps)

Av. Neutron Energy (MeV)	η	Av. Prompt Leakage Rate [†] (neutrons/sec)
1.0	2.93	1.26×10^{17}
0.75	2.85	1.24×10^{17}
0.50	2.72	1.21×10^{17}
0.40	2.69	1.20×10^{17}
0.25	2.63	1.18×10^{17}

[†]Maximum possible

Table III-A

Effect of the Introduction of Cyclic Reactivity Variation

Pu-239

($S_s = 9.5 \times 10^{17}$ neutrons/sec, $\ell_o = 1 \times 10^{-8}$ sec, $\beta^* = 0.00204$,
 $T_s = 4 \mu\text{sec}$, $t_k = 750 \mu\text{sec}$, Rep. Rate = 200 pps)

$k(t)$	Constant	Step	Step
Reactivity Swing	0	\$ 5.0	\$10.0
Prompt Gain	250	250	250
Total Gain	510	311	292
Average Total Power (MW)	4.33	2.64	2.48
Peak Power/Av. Bkg. Power	959	4065	5877
% Prompt Energy in Pulse	49	80	85
Effective Pulse Width, τ (μsec)	5.0	5.0	5.0

when $k(t)$ is oscillated. The peak to background as well as the fraction of energy in the prompt pulse is larger for the cases when $k(t)$ is varied compared to when it is constant. Thus, dramatic improvements in pulse yield characteristics at relatively high gains can be expected when $k(t)$ is oscillated in some fashion. Also, as indicated by the fact that all the τ 's are the same, then τ is independent of t_k as long as t_k is very large compared to the reactor pulse relaxation time. However, the signal to noise ratio should be affected somewhat. These points will be discussed in more detail a little later.

Typical power pulse shapes, as calculated using Equation (12-A), from reactors fueled with U-235, U-233, and Pu-239 with a step change in reactivity of \$5.00 are shown in Figure 4-A. The reactors were all assumed to have the same lifetime and a prompt gain of 200. Note the significantly higher background powers associated with U-235 when compared with U-233 and Pu-239 and also how the power level changes when $k(t)$ is varied at 0 and 200 μ sec. The plots were made for a t_k of 200 μ sec, that is, the step change in $k(t)$ lasted for 200 μ sec, for ease of plotting only.

To investigate more fully the effect of t_k on the pulse yield characteristics of a reactor operating in either mode two or four, calculations were made for step changes in $k(t)$ lasting 500 and 1000 μ sec. The fuel was assumed to be Pu-239, and the computations were performed for several prompt neutron lifetimes. The results are shown in Table IV-A. As can be seen in Table IV-A, for a fixed ℓ_0 , changing t_k from 500 to 1000 μ sec has no effect on τ . However, other pulse characteristics as indicated by the peak to average background ratio and the fraction of prompt energy in the pulse become poorer. Note the effect (as was also indicated in Figure 3-A) of the dramatic improvement in pulse shape characteristics up to a point when ℓ_0 is varied and t_k is held constant. It can be concluded for reactors operating in modes two or four that the pulse characteristics are not too sensitive (not as sensitive as reactors operating in modes three and five) to the length of t_k . Also, the lifetime can have a very significant effect on the reactor pulse shape characteristics and a small ℓ_0 is preferred.

In order to further illustrate the desirability of having a repetitively pulsed reactor fueled with a fissile material having a low β^* and a high η , calculations were made for reactors fueled with U-235, U-233, and Pu-239 and all operating at an average power of 9 MW. The results are shown in Table V-A. The relative advantage of one fuel to another is somewhat harder to depict from the results shown in Table V-A than they were when the prompt gains were all the same. For instance, the τ for U-235 is less than the τ for either U-233 or Pu-239. However, the U-233 and Pu-239 reactors have more gain and better peak to average background ratios than U-235. Even though the U-233 and Pu-239 fueled reactors have equivalent peak to average power ratios, presumably due to the amplification of delayed neutrons, the advantage of the Pu system is reflected in the prompt leakage rate values.

Additional studies which should be made include calculations in which $k(t)$ is allowed to vary as the rotating reactivity arms vary the reactivity of the system. This will also allow reactors operating in the super prompt critical range to be studied realistically. Specific reactor systems need to be investigated in detail in order to assign effective values to such parameters as η , ℓ_0 , and β for use in studying a reactor's dynamic response.

Fig. 4 A plot is shown of the power pulse shape as a function of time over one period for the same gain and neutron lifetime for U-233, U-235, and Pu-239. The relative effects of the different delayed neutron fractions are illustrated for a step reactivity insertion for 200 μ sec.

Table IV-A

Effect of Varying Time of Reactivity Introduction and Neutron Lifetime

 $(S_s = 9.5 \times 10^{17}, \beta^* = 0.00204, \eta = 2.69, T_s = 4 \text{ } \mu\text{sec}, \text{Prompt Gain} = 300,$

Rep. Rate = 200 pps, Reactivity Swing \$5.0)

t_k (μsec)	<u>500</u>			<u>1000</u>		
ℓ_o (sec)	5×10^{-8}	1×10^{-8}	5×10^{-9}	5×10^{-8}	1×10^{-8}	5×10^{-9}
Av. Total Power (MW)	3.16	3.16	3.16	3.35	3.35	3.35
Av. Background Power (MW)	0.61	0.62	0.62	0.80	0.81	0.81
Peak Power/Bkg. Power	1215	3797	4796	927	2898	3660
% Prompt Energy in Pulse	81	80	80	76	76	76
Effective Pulse Width, (μsec)	17.1	5.4	4.3	17.1	5.4	4.3

Table V-A

Comparison of Fuels for Reactors Operating at the Same Av. Total Power

 $(S_s = 9.5 \times 10^{17} \text{ neutrons/sec}, \ell_o = 1 \times 10^{-8} \text{ sec}, T_s = 4 \text{ } \mu\text{sec},$ $t_k = 750 \text{ } \mu\text{sec}, \text{Rep. Rate} = 200 \text{ pps}, \text{Reactivity Swing} = \$5.0)$

Parameter	U-235	U-233	Pu-239
	$\beta^* = 0.00641$ $\nu = 2.46$ $\eta = 2.13$	$\beta^* = 0.00267$ $\nu = 2.51$ $\eta = 2.33$	$\beta^* = 0.00204$ $\nu = 2.93$ $\eta = 2.69$
Prompt Gain	422	584	685
Prompt Leakage Gain [†]	222	333	430
Total Gain	937	943	1061
Av. Total Power (MW)	9	9	9
Peak Power/Av. Bkg. Power	634	1016	1011
% Prompt Energy in Pulse	45	62	65
Av. Prompt Leakage Rate [†] (neutrons/sec)	1.69×10^{17}	2.53×10^{17}	3.27×10^{17}
Effective Pulse Width, (μsec)	6.5	8.0	9.0

[†]Maximum possible

V. CONCLUSIONS

There is a definite degradation in the relative pulse yield characteristics at large neutron gains for subcritical reactors with a constant reactivity. A way of lessening the effects of poor yield is to have a fuel with a small delayed neutron fraction. Hence reactors fueled with Pu-239 would have better pulse yields at high gains than either U-233 or U-235. In order to have a fast responding system, a short prompt neutron lifetime is essential.

Large gains in pulse shape characteristics can be made if, for example, the lifetime is reduced from about 5×10^{-8} to 1×10^{-8} sec. Further reduction in the lifetime would not change the pulse characteristics as dramatically for the time of the driving pulse studied here, but they would improve drastically for shorter driving pulses. To the extent that the reactor pulse is not spread out in time, the advantage of lowering the lifetime goes as the peak power intensity/ (driving pulse width)². For neutron energies for which the moderator time broadening of the pulse is negligible, the advantage provided by shorter neutron lifetimes would vary as $(1/\ell_0)^2$, because the utilization of shorter driving pulses require the short lifetimes. On the basis of current moderator studies, such gains would be realized for neutron energies above 0.3 eV, and indications are that this limit may be lowered to the thermal energy range. A fuel with as large an η as possible is desirable.

The most practical way of retaining good pulse yield characteristics at high gain is to periodically vary $k(t)$. This reduces the background neutrons and puts a larger fraction of the energy produced in the prompt portion of the pulse. Again, if the advantage can be retained in the thermalization process, the most desirable fuel for a reactor would be Pu-239. This fuel has a distinct advantage over U-235 and to a lesser extent, over U-233 as well.

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APPENDIX B
RPTF CONCEPT

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APPENDIX B

RPTF Concept

Characteristics of the RPTF concept are listed in Table I-B. The reactor parameters are based upon the LAMPRE experiment using molten iron-plutonium eutectic contained in tantalum-walled cylindrical fuel pins. For the average power of 3.3 megawatts, the fuel system has a demonstrated reliable reactor operating experience, being well below thresholds of corrosion problems. Also the operating-temperature ranges are below thresholds of stainless-steel embrittlement problems allowing their usage for the reactor tanks.

The physics numbers have been obtained by interpolations and modest extrapolations of LAMPRE criticality data, and the critical mass may be a little large. For the pulse reactor (PR) mode, the numbers involve larger extrapolations. Based upon step function rotary reactivity insertions 750 μ sec long, the booster would yield 75% of the power to the pulse. This would yield a pulse peak emission rate 2500 times larger than the between pulse emission rate. The two rotors consist of three paddle arms one meter in radius spinning at 4000 rpm.

Appendix B is due for completion on March 1, 1968.

TABLE I-B

Preliminary RPTF Characteristics

<u>Operating Characteristics</u>	<u>Pulsed Reactor Mode</u>	<u>Booster Mode</u>
Core size (liters)	6	6
Average power (MW)	3.3	3.3
Energy in pulse (MW-sec)	0.12	0.012
Pulse repetition rate (pps)	20	200
Peak power (MW)	2,250	2,250
Effective power pulse width (μsec)	50	5.4
Average prompt leakage rate (neutrons/sec)	1.4×10^{17}	1.4×10^{17}
<u>Physics</u>		
Core loading (Kg Pu-239)	50	50
Prompt neutron lifetime (sec)	1×10^{-8}	1×10^{-8}
Prompt multiplication	293	293
Effective η	2.65	2.65
Rotating reactivity worth (est.)	\$5.0	\$5.0
Effective β	0.0034	0.0034
Wheel radius (meters)	1.0	1.0
<u>Neutron Source Characteristics</u>		
Average power (kW)		100
Voltage (MeV)		100
Pulse width (μsec)		4
Pu-239 target yield (neutrons/sec-MW)		9.5×10^{15}
Neutron yield (neutrons/sec)		9.5×10^{17}

Mechanical, Thermal and Hydraulic - Scaled from "LAMPRE I Final Design Status Report", LA-2833

APPENDIX C
RPTF DEVELOPMENT PROGRAM SCHEDULES

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APPENDIX C

RPTF Development Program Schedules

A major portion of the current content of this appendix is included in the body of the proposal, and redundancy of repeating those work schedules here is avoided. The fact that Appendix C in itself is a duplication and therefore possibly confusing has been discounted because of the wish to place emphasis upon the continuing and evolving nature of this function. Basically, the reporting and scheduling function will consist of expansions of the bar graphs and functional block diagrams presented in the main report. Pertinent techniques and other systems techniques are to be adopted when appropriate. Considerable interest exists at INC in adapting information retrieval techniques using computers with display console presentation for management purposes. It is reasonably probable that these will be available for project usage.

Figure 1-C is a bar graph schedule for the completion of finished documents providing basic support for the proposal. Because these require great technical detail, they can be best written as documents coherent in themselves. For example the Moderator Optimizing report (Appendix F) will also serve as a first draft for the document to be submitted for publication. Appendix D which is to describe the proposed R & D programs using the RPTF can be best prepared using reactor and moderator input data so as to provide the most quantitative information about experimental capabilities for the variety of instrument types under consideration.

FIGURE 1-C

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APPENDIX D

POTENTIAL FOR TRANSIENT "EQUATION OF STATE" PROGRAM USING THE RPTF

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APPENDIX D

Potential for Transient "Equation of State" Program Using the RPTF

This appendix is intended to provide a quantitative definition of the information which can be obtained using the RPTF as a materials study facility. The realization of its potential for testing has resulted from an examination of the problems associated with its design. Such an examination will pay double dividends because of its pertinence to the RPTF and to the question of value for reactor programs in general. This study is due to be completed by January 1, 1968.

One observation that may have significance is the value of subcritical operation using a charged particle neutron source for materials test problems. One problem concerns the feedback from the test sample to the pulsed reactor where a positive coefficient from the sample, which might result from melting could be handled easily by turning the neutron source off to avoid any reactor damage. At the same time, it is desirable to have positive sample feedback because of the sensitivity of the reactor response to subtle changes taking place in the sample. By the use of short-time-width driving pulses, several degrees of additional information becomes potentially available because of the added sensitivity to different time scale factors. Pulse integrals, pulse tails, and the background level between pulses will all provide information of interest about the test sample. For example changes in the delayed neutrons would imply changes in the nuclear fission reaction process which would perturb the power level between pulses. There is concern at the present time that the Doppler coefficient might be smaller for melted UO_2 for example. In the RPTF, melting would not be expected to result from a single pulse but would result from decreased cooling, and the average subcritical point of operation would be set to keep any positive feedback coefficient in the subcritical operation mode.

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APPENDIX E
GENERAL R & D PROGRAMS USING THE RPTF

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APPENDIX E

General R & D Programs Using the RPTF

As a research and development tool the RPTF would have a very large potential of usage, and the problem of optimization becomes excessive for simultaneous usage where all users have optimum experiments. For example, the lower repetition rates needed for very cold neutron velocities are not compatible with higher repetition rates desired for thermal and epithermal neutron purposes. Likewise, there are many other resolution questions in addition to those of the energy resolution provided by TOF methods which require examination before advantages can be made quantitative for a specific spectrometry system. A reasonably detailed survey of these areas is being undertaken, but is scheduled to use the results of the moderator optimization studies in order to yield more quantitative results. The detailed development of the measurement merits and the basic and applied information to be obtained is scheduled for completion on January 22, 1968.

The INC programs are conceived to consist of three areas of approximately equal magnitude as follows: a) an applied materials development program, b) an applied nuclear and condensed water cross section program in the area of the "physics of reactors" in contrast to the usual "reactor physics" concept where the field is confined to reactor neutron problems, and c) a basic research program to be carried out in collaboration with universities. As proposed this program would only utilize 40-60% of the facility capability allowing for many other programs from a national base.

*Materials testing programs are to be developed separately in Appendix D.

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APPENDIX F
MODERATOR OPTIMIZATION

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APPENDIX F

Moderator Optimization

I. Introduction

Limits on the flux obtainable for neutron sources from steady state reactors are being reached because of the excessive cost of fuel required. Also restrictions are being imposed on the power density of such systems because of the heat transfer, heat transport and core materials. For time-of-flight (TOF) experiments the beams from steady state systems are chopped so that only 0.1 to 0.5% of the beam neutrons are used, and it appears advantageous to operate the reactor only when neutrons are desired. If the neutrons can be generated only on demand, pulsed reactors having the same average power as steady state systems would have effective fluxes 200 to 1000 times higher.

For these reasons studies are underway to learn how to design an optimum beam source moderator for pulsed reactors, and some initial results are reported here because they are expected to be of general interest. The basic motivation for building pulse reactors is to provide enhanced "effective neutron source fluxes" for neutron TOF spectroscopy primarily in the thermal neutron energy region.

While clearly spelling out the motivation, the practical pulsed reactor system is not expected to yield effective fluxes as high as the ideal. Because the neutrons can be produced only approximately on demand, there are time uncertainties in the neutron emission time, and the reactor and experiments need to be designed to take advantage of the neutron time-intensity function that the moderator yields from the neutron source. In this sense the moderator operates on the incoming neutrons to produce a source neutron pulse with a peak intensity and a width in time. The neutron time-intensity distribution or the pulse from the moderator provides the input information for the design of neutron TOF spectrometers. An optimization problem results in which the pulsed reactor, moderator, and experiments are variable[1-F].

A major objective here has been to determine the delta source moderator response functions for a variety of moderators. When these are folded or convoluted with the pulse-time shape of the reactor source, the reactor beam neutron-time-distribution can be obtained for experiment design. By iterating the above process an optimum reactor-moderator-experiment design can be reached. In order to determine the delta function response of moderators, the Linear Electron Accelerator at Rensselaer Polytechnic Institute (RPI) was used to provide very short pulses of fast neutrons impinging on various moderators. The neutron-leakage-time distributions were measured as a function of time and energy after introduction of an approximate fast neutron delta function source from the electron accelerator.

In these comparisons of different configurations we have adopted Michaudon't [2-F] "figure of merit" ($F = \phi_p/\tau^2$). Here ϕ_p is the peak intensity and τ is the full width at half maximum (FWHM) in microseconds. Since the TOF technique is basically one of energy determination, a widely accepted basis for comparison of the relative advantages of systems can be made by a calculation of F .

In these survey measurements, the philosophy has been to observe a relatively large number of parametric variables in order to determine the relative importance of each in determining an optimum moderator system. To accomplish this objective, sacrifices were made in the precision of the data. The data collection time was adjusted to give counting statistics barely sufficient to measure peak intensities and pulse widths. Particular care was taken to assure proper normalization between experimental runs for the different samples and accelerator operating conditions. However, the determination of absolute yields per input fast neutron, measurement of source-moderator geometrical factors, and the measurement of the effects of surrounding materials remain to be accomplished.

II. EXPERIMENTAL CONDITIONS

Data were taken simultaneously with a disk chopper and neutron diffraction spectrometer using a beryllium crystal. Time-of-flight techniques were used to obtain time dependent spectrum measurements from which energy dependent time distributions were obtained. Figure 1-F shows the experimental arrangement with the chopper and crystal spectrometer on opposite sides of the medium and at right angles to the electron beam. The linac was pulsed at 150 pps and either a 2.5 or 4.5 μ sec electron pulse width was used. The electron energy was 45-60 MeV and the peak current ranged between 400 to 650 mA. The electron target material was tantalum. The chopper and crystal spectrometer viewed opposite sides of the moderator with viewing surfaces parallel to the incident fast neutrons as illustrated in Figure 2-F. The crystal spectrometer always looked at the surface whereas the chopper looked at either the surface or center of the medium. The size of the moderating medium parallel to the incident fast neutron source was either 10" x 10" or 4" x 4" and the thickness perpendicular to the beam varied between 1" to 3". In most cases the 4" x 4" medium was investigated.

The chopper rotor was spun at 9000 rpm with a burst width of 7.2 μ sec. The basic time channel was 8 μ sec which gave a time resolution of 2.5 μ sec/meter. When looking at late times in the neutron time distribution the 19 μ sec chopper burst was used. The collimation for the chopper was designed to view a 1/2" diameter re-entrant hole into the center of the medium. The flight path from the chopping plane to a 2-1/2" end window BF₃ detector was 3.17 meters. A detailed description of the chopper system has previously been published [3,4,5,6-F].

For the crystal spectrometer the (0002) planes of a Be crystal were used because of the many higher order diffraction peaks that are available and the planes of the crystal were parallel to its face. The mosaic structure of the crystal led to a broadening of 6' of arc. A crystal angle of 45° was used and the overall flight path was 3.4 meters. The area of the moderator that the spectrometer looked at was 1/2" x 2", and always viewed the surface of the medium. A further source of instrumental resolution broadening was the total reflection of neutrons from the Ni septa of the Soller slit collimator. The

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Fig. 1—F Disc chopper and crystal spectrometer experimental setup for the pulsed neutron thermalization experiment at RPI.

SAMPLE CONFIGURATION

Fig. 2—F Sample configuration for pulsed neutron thermalization experiment.

critical angle for total reflection varies from 2' to 15' of arc in the energy range of 0.63 eV to 0.012 eV. The nominal angular divergence of the collimator was 2' of arc.

The collimation between the crystal and detector was maintained at 1/2". A single 5/8" dia. He³ counter with 10 atmospheres pressure was used as the neutron detector. These conditions gave a time resolution, for the crystal spectrometer, of 1 to 4 μ sec/meter in the energy region of 0.63 eV to 0.012 eV. High density concrete block was placed between the moderator and the He³ detector to help shield the detector from the gamma flash and also attenuate the background neutrons seen by the He³ detector. With this system it was possible for several energies to be observed simultaneously with the crystal set at a single angle. Under these conditions 7 orders and 3 forbidden half orders (by multiple Bragg scattering^[7-F]) were observed which gave 10 different time-distribution leakage spectra in the energy region from 0.012 to 0.63 eV. Similar time dependent neutron spectra measurements using a crystal as the monochromator have been performed by Whittimore^[8-F] and Sinclair and Williams^[9-F].

Thermalization properties of several moderating materials were measured at liquid nitrogen (LN) temperature. Figure 3-F is a cutaway view of the LN cryostat. A table designed to support and position the moderating medium and the electron target is shown in Figure 4-F. The position of the target and neutron shield for the LN cryostat which holds the medium are shown in their relative positions when taking data. Because of the radiation hazard of the electron target, the cryostat and B₄C shield were mounted on rails so that one could roll them back from the target when changing samples. A dowel pin was used to position the cryostat when it was in the data position.

The boral plate and B₄C above the moderator served to shield the sample from the neutrons that were thermalized in the LN reservoir. The moderating materials were always surrounded by cadmium and boral to provide a thermal neutron shield. The aluminum blocks between the sample and LN provided a conductor for heat transfer. The cryostat was surrounded by 2" of B₄C and the chopper looked at one side of the moderator and the crystal spectrometer at the other face.

A Digital Electronics PDP-7 computer was used for controlling the accelerator trigger and storage of the time-of-flight chopper data whereas the time-of-flight data for the crystal spectrometer were stored in two TMC 1024 channel analyzers. Figure 5-F shows the timing relationship of the accelerator, chopper, and crystal spectrometer. The timing sequence for the crystal spectrometer was relatively simple. The timing for this analyzer was controlled from the accelerator trigger pulse.

The chopper rotated at 9,000 rpm, which gave a basic period of 6,666 μ sec. The timing sequence for the chopper was as follows: A magnetic pickup signal from the chopper disc provided the reference time signal for the whole sequence. The fixed time delay between the magnetic pickup signal and the center of the chopper pulse was measured to be 700 μ sec. The accelerator trigger pulse was phased with respect to the chopper opening by means of a computer-controlled delay generator; each accelerator delay setting corresponded to a

Figure 3-F Cutaway view of the liquid nitrogen cryostat.

Fig. 4—F Table for supporting and positioning the moderating medium and electron target with their relative positions shown when taking data.

Fig. 5-F The timing relationship of the chopper signal, accelerator trigger, chopper time-of-flight analyzer, and crystal spectrometer time-of-flight analyzer.

particular phase time t_c between the fast neutron source pulse and the chopper opening. By programming changes in the accelerator delay over a range of preselected values, one could measure the spectra existing at the chopper for various times t_c after the source pulses. The chopper background was obtained by delaying the accelerator trigger until after the chopper had closed, as indicated by the dotted pulses. There is a delay called the injection delay between the accelerator trigger and the fast neutron production in the target. Time zero for the fast neutron pulse is assumed to be in the center of the fast neutron burst. This timing can be calculated from the various delays and checked by observing which time channels corresponded to the gamma flash for the chopper and crystal spectrometer data. The center of the chopper pulse is zero time of flight for the chopper data; therefore the chopper analyzer was triggered from a fixed delay which was controlled from the magnetic pickup signal.

The chopper measures the energy spectra for a variety of phase times whereas the crystal spectrometer measures the leakage-time spectra for a group of energies. The neutron emission time for the chopper data is equal to the phase time t_c , the time between the fast neutron pulse and the chopper opening, minus the drift time of the neutrons (the time that it takes the neutron of a given energy to travel from the viewing surface of the moderator to the chopping edge of the rotor). Figure 6-F is a diagram showing the relationship between the phase time, phase time corrected for drift time, neutron time of flight, and counts/channel. After correcting for the drift time one gets the true emission time of the neutrons which is represented by the dashed line. The emission time is the time between which the fast neutron enters the moderating medium and the thermalized neutron leaks out of the system. After making corrections in the chopper data for the drift time, one obtains the time spectra by plotting the corrected data at a given energy. This corresponds to a constant time of flight for the different emission times, and is represented by the line perpendicular to the time-of-flight axis. If the time distributions are plotted at energies corresponding to those observed with the crystal spectrometer, the two sets of data can be compared. A list of the materials and experimental conditions under which these materials were investigated is shown in Table I-F.

III. DATA

In addition to the neutron drift-time corrections, the chopper data were corrected for detector efficiency, beam attenuation because of absorbing or scattering materials in the beam, and flight path differences due to the relative position of absorption of the neutrons of different energy in the end window BF₃ detector. Nineteen different runs were made on different materials of different size and temperature and all of these measurements were arbitrarily normalized to the 3" room temperature water run. Three different methods of normalizing the chopper and crystal spectrometer data were considered as follows:

- 1) The areas under the chopper background curves from 200 to 1320 μ sec were compared, since the background counting rate is proportional to the

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Fig. 6-F Diagram showing the relationship between the phase time, phase time minus drift time, neutron time of flight, and count rate.

TABLE I-F
SAMPLES AND EXPERIMENTAL CONDITIONS

Sample	Size* (inches)	Temperature	Accelerator Pulse Width (μ sec)
Water	4 x 4 x 3	Room	2.5
Water	4 x 4 x 1	Room	2.5
Water + Boric Acid	4 x 4 x 3	Room	2.5
Water + Cadmium Sulfate	4 x 4 x 3	Room	2.5
Polyethylene	10 x 10 x 3	Room	4.5
Polyethylene	4 x 4 x 3	Room	2.5
Polyethylene	4 x 4 x 1-1/2	Room	2.5
Polyethylene	4 x 4 x 3	Liquid Nitrogen	2.5
Polyethylene	4 x 4 x 1	Liquid Nitrogen	2.5
Polyethylene + Cd Sheet [†]	4 x 4 x 3	Room	2.5
Polyethylene Pellets	4 x 4 x 3	Room	2.5
Lucite	4 x 4 x 3	Liquid Nitrogen	2.5
Zirconium Hydride Powder	4 x 4 x 3	Room	2.5
Zirconium Hydride Powder	4 x 4 x 1	Room	2.5
Zirconium Hydride Powder	4 x 4 x 3	Liquid Nitrogen	2.5
Solid Ammonia	3 x 4 x 3	Liquid Nitrogen	2.5

* Time dependent measurements were always made perpendicular to the 4" x 4" or 10" x 10" surfaces.

[†] The cadmium sheet was placed parallel to and 1 cm inside the 4" x 4" surface viewed by the crystal spectrometer.

accelerator neutron production. A typical background for one of the chopper runs is shown in Figure 7-F.

2) A 3 mil gold wire in a 4" cube of polyethylene was irradiated by the target for the duration of each run. The ratio of the count rates, after corrections for decay time, were then compared.

3) Ratios were taken of the number of gamma flashes or accelerator pulses. The first two methods agreed very well whereas the accelerator pulses gave factors which differed by as much as 2.4.

The crystal spectrometer normalization factors were obtained from the chopper background data (Method 1 above) by making corrections for the number of spectra observed by the chopper for each run. Sets of raw data for the 4" x 4" x 3" light water are shown in Figures 8-F and 9-F. These data show the diffraction peaks for all of the energies that were observed with the crystal spectrometer. The first figure shows the data for the three lowest energies on both a linear and semi-log plot. The data are plotted on the same horizontal scale, but for comparison purposes, both semi-log and linear scales are used on the vertical axis. Each scale has its merits and disadvantages; the semi-log plot preferentially distorts the data but does indicate quite dramatically when the neutron energy distribution becomes exponential indicating an equilibrium Maxwellian distribution. The linear scale does not distort the data and is a better indication of the "figure of merit", however it is quite difficult to tell when the neutron spectrum reaches equilibrium. The eight higher energy diffraction peaks are shown in Figure 9-F. The 0.0796, 0.156 and 0.258 eV peaks are the half or forbidden orders.

In order to get true neutron-leakage-time spectra the following operations were performed on the raw data: (1) normalization of all samples to the same number of incident neutrons, (2) background subtraction, (3) subtraction of the die-away tail of one energy from an adjacent energy, (4) conversion from time channels to a time-of-flight base, (5) subtraction of the time of flight for a particular energy neutron from the observed time of flight to obtain the time distributions on a real time base, (6) correction for energy dependent flight paths in the detector, and (7) corrections for counter efficiency, crystal reflectivity, and air scattering in the flight path.

The effect of applying the first five of the above corrections to the raw data for the 4" x 4" x 3" light water sample can be seen in the two left hand plots of Figure 10-F. The effects of applying all of the above corrections are reflected in the two right hand plots of Figure 10-F. The correction for the crystal reflectivity, detector efficiency, and air scattering from one diffraction peak to the other was accomplished by comparing both the ratios of the peak heights of the chopper emission-time-distribution spectra and the area under the time distributions over a given time interval. It should be noted that when all of the corrections are applied and when the neutron spectrum is in equilibrium ($\tau \approx 35 \mu\text{sec}$), the relative intensities correspond reasonably well to that of a Maxwell-Boltzman distribution.

Fig. 7-F A typical background for one of the chopper runs with the shaded area as the portion of the curve used for obtaining the normalization factor for the individual runs.

Fig. 8-F Raw data on a linear and semi-log plot showing the three lowest energy diffraction peaks from the (0002) planes of a Be crystal for a 4" x 4" x 3" light water sample at room temperature.

Fig. 9-F Crystal spectrometer raw data for the 4" x 4" x 3" light water sample at room temperature, showing the eight highest energy diffraction peaks observed with (0002) planes of a Be crystal.

Fig. 10-F Neutron-leakage-time distributions from the surface of a 4" x 4" x 3" light water room temperature sample, both uncorrected and corrected for detector efficiency and crystal reflectivity.

A comparison of a 4" x 4" x 3" and a 4" x 4" x 1" light water sample at room temperature as shown in Figure 11-F, serves to illustrate the effect of varying the thickness of the moderator. The pulse width of the 1" sample is narrower but the peak intensities are down. Figure 12-F is a further illustration of the effect of varying only the moderator thickness. Here, 4" x 4" x 3" and 4" x 4" x 1-1/2" polyethylene samples are compared. It appears that the 3" thickness is not the optimum thickness for these 4" x 4" room temperature samples and that the peak intensity and pulse width can be improved by going to samples which are less than 3" thick. These data also indicate that the optimum thickness for water is different from that for polyethylene. These conclusions are further illustrated in the relative "figures of merit" for each energy, peak intensity/(FWHM)², (ϕ_p/τ^2) which is also listed in the figure. The "figure of merit" was arbitrarily set equal to unity for the light water sample at all energies; thus one cannot compare the "figure of merit" of one energy to that of another for a particular sample.

The differences in neutron pulse characteristics for samples of light water and light water homogeneously poisoned with boric acid and cadmium sulfate are shown in Figure 13-F. For boric acid poisoning, a significant shortening of the neutron pulses was obtained without too severe a reduction in the peak intensity. However, the amount of cadmium poisoning was much too small to show a significant effect. In fact, neither of the poison concentrations were optimized to produce the most desirable neutron-leakage-time distributions. This method of pulse shaping needs to be investigated further for both room temperature and cold moderators.

The effect on the neutron time distributions of varying only the temperature of the moderator is shown in Figure 14-F. In this figure, 4" x 4" x 3" polyethylene samples at room and LN temperatures are compared. Lowering the temperature has very little effect on the peak intensities, however, it does have a dramatic beneficial effect on cutting off the tails of the distributions, and thereby shortening the pulse widths. The most dramatic improvement was for the 0.05096 eV time distribution. Figure 15-F shows the two lowest energies for the above samples, plotted over a longer time and on both linear and semi-log scales. The higher intensity and the longer relaxation time, emphasized by the semi-log plot at 0.01274 eV, are due to the peak of the Maxwellian being shifted to lower neutron energy for the cold sample. Further enhancement of the pulse characteristics from a LN cooled sample can be made by poisoning the sample.

The neutron time distributions from different moderators for various experimental conditions as compared to a light water sample are shown in Figures 16-F to 25-F. Specifically, in Figure 16-F one can see the more attractive pulse characteristics of solid ammonia as a moderator compared to light water. Also, the dramatic effect of heterogeneously poisoning a sample is demonstrated in this figure. Figure 17-F is a replot of the two lowest energies for the above samples over a longer time and on both linear and semi-log scales. In particular one can see the advantage of heterogeneously poisoning a cold sample to sharpen up the pulse at low energies. For example, the solid ammonia sample could probably be heterogeneously poisoned in such a way that its peak intensity would be comparable to light water but its pulse width would be much narrower. (It should be noted that the heterogeneously poisoned polyethylene sample was 3" thick and had a sheet of cadmium 1 cm in from the

Fig. 11—F Neutron leakage-time distributions from the surface of 4" x 4" x 3" and 4" x 4" x 1" light water samples at room temperature.

Fig. 12-F Neutron-leakage-time distributions from the surface of 4" x 4" x 3" and 4" x 4" x 1-1/2" polyethylene samples at room temperature.

Fig. 13—F Neutron-leakage-time distributions from the surface of unpoisoned light water sample and from a similar sample 4" x 4" x 3" homogeneously poisoned with boric acid and cadmium sulfate.

Fig. 14—F Comparison of the neutron-leakage-time distributions as a function of moderator temperature for 4" x 4" x 3" polyethylene samples.

Fig. 15—F Neutron-leakage-time distributions for the two lowest energies of the 4'' x 4'' x 3'' polyethylene sample at room and liquid nitrogen temperatures, plotted on both linear and semi-log scales.

surface viewed by the crystal spectrometer. This configuration has not yet been optimized for peak intensity nor "figure of merit".) The longer relaxation time for the room temperature water sample compared to the solid ammonia sample at LN temperature may be due to the fact that the water sample was 4" high while the ammonia sample was approximately 3" high.

Two more samples are compared with the 3" light water sample in Figure 18-F. The 1" light water is a comparison of sample thickness only as in Figure 11-F. The 4" x 4" x 1" polyethylene sample shows the general advantage of a low temperature sample. If one can assume that the 1" light water and polyethylene samples have essentially the same characteristics, one notes that there is a more dramatic difference in peak intensities between these two 1" thick samples than there was between two comparable 3" thick samples as shown in Figure 14-F. This intensity difference appears to be a thickness effect correlated to the distribution of epithermal neutrons, and is not considered to be due to temperature except for the die-away tails.

In Figure 19-F, it can be seen, for the two intermediate energies shown, that lucite shows the general advantage of a low temperature sample. However, its pulse characteristics are not as good as the same size polyethylene sample at LN temperature.

Cold powdered zirconium hydride appears to be an inferior moderator to light water. This sample was not expected to be a favorable material for either intensity or pulse width for neutron energies below the 0.13 eV vibrational state. It was tested because it is a material which has an extreme variation in the inelastic scattering cross section, being high above 0.13 eV and dropping off very rapidly below this value. Figures 20-F and 21-F show the pulse characteristics of the 4" x 4" x 3" and 4" x 4" x 1" zirconium hydride samples at room temperature as compared to light water. Essentially identical results were obtained at LN and room temperatures for the same zirconium hydride samples; in neither case is zirconium hydride as good as light water.

Figure 22-F is a comparison of the neutron-leakage-time distributions for light water, polyethylene, and polyethylene pellets, all at room temperature. The pellets were measured to illustrate the effect of changing material density to correspond to the lower density of the powdered zirconium hydride. As expected, the polyethylene pellets did not reveal any improvement in the neutron pulse characteristics.

The two lowest energies are plotted on linear and semi-log scales in Figure 23-F. From these data one observes that the light water and solid polyethylene 3" samples show about the same characteristics. The light water is a little faster system and the polyethylene has a little higher intensity.

The next set of data is a comparison between the 4" x 4" x 3" and 10" x 10" x 3" polyethylene samples at room temperature. Figure 24-F shows the four energies on a linear plot and Figure 25-F displays the two lowest energies on linear and semi-log scales. These data show that the 4" x 4" x 3" moderator has a higher neutron yield and is also a faster system. One should keep in mind however, that the neutron distribution for the 10" x 10" sample was

Fig. 16-F Neutron-leakage-time distributions for solid ammonia and heterogeneously poisoned polyethylene samples, compared to light water.

Fig. 17-F The two lowest energies of the time spectra for the solid ammonia and heterogeneously poisoned polyethylene samples compared to light water on both linear and semi-log plots.

Fig. 18—F Comparison of the neutron-leakage-time distributions for 4" x 4" x 1" and 4" x 4" x 3" light water at room temperature and 4" x 4" x 1" polyethylene at liquid nitrogen temperature.

Fig. 19-F Neutron-leakage-time spectra for lucite and powdered zirconium hydride at liquid nitrogen temperature compared to light water at room temperature.

Fig. 20—F Comparison of the surface-neutron-leakage-time distributions of 4" x 4" x 3" light water, 4" x 4" x 3" zirconium hydride, and 4" x 4" x 1" zirconium hydride samples at room temperature.

Fig. 21-F Neutron-leakage-time distributions for the two lowest energies of the 4" x 4" x 3" light water and zirconium hydride and 4" x 4" x 1" zirconium hydride samples plotted on linear and semi-log scales.

Fig. 22—F Comparison of the surface-neutron-leakage-time distributions of the 4" x 4" x 3" light water, polyethylene pellets, and polyethylene samples.

5" from the front face of the moderator upon which the fast neutrons were incident; whereas the 4" x 4" sample was viewed 2" from the front surface. From flux distribution measurements it has been observed that the flux peaks at about 1" from the surface in the direction of the fast neutrons. Therefore, one would expect the 4" sample to have a higher intensity than the 10" when measured at their respective centers. It has also been demonstrated from varying the thickness of the moderator that when viewing the system perpendicular to the fast-neutron source that 2" is about the optimum thickness for polyethylene. These results are also in general agreement with integral measurements made at BNL[10-F] which were performed under slightly different conditions.

IV. CONCLUSIONS

A list of experimentally observed emission-time pulse widths is presented in Table II-F and from these the relative "figures of merit", normalized to the 4" x 4" x 3" light water sample at room temperature, are presented in Table III-F. Because the relative detection efficiencies of the crystal spectrometer at different energies are still being obtained from chopper data, the "figure of merit" comparisons always refer to the water referenced to the same energy. Since these efficiencies are not given, comparisons of "figures of merit" between different energies are not meaningful. For the purposes of determining the significance of the pulse width measurements, a list of resolutions is presented in Table IV-F. Error analyses of the accuracies of the height and width determinations are not yet completed. The standard deviation for a repeat value of the width determinations is about 1 μ sec. The peak heights are accurate to about 10% at the higher energies and 1-3% for lower energies.

Much higher counting rates were observed with the crystal spectrometer because a larger surface source area was utilized and because the running time was continuous. (In contrast the chopper only records data for one specific delay time after the linac burst and for the fraction of total time given by the reciprocal of the total number of delay phase times covered. Thus, the chopper data were cycled between various delays.) Because of the favorable counting rates for the crystal spectrometer, sample run times were reduced, and frequently the chopper data did not contain sufficient statistical accuracy to permit an adequate calibration of the crystal order efficiencies from the chopper data.

The following general conclusions can, however, be made from the results:

a) The time required for the neutrons to reach thermal energies in hydrogenous materials is surprisingly short, and the time widths of the corresponding high energy initial value pulses are also short. The time-pulse shapes and the intensities of these slowing-down transients are found to be independent of the moderator temperature for the two temperatures observed. These are seen in their pure form for cold samples where the equilibrium tails are gone. This conclusion can be seen repeatedly for different materials at the two temperatures by the similarity or identity between rise times and peak intensities. It is concluded that the full transient pulse as seen at low temperatures actually exists

Fig. 23—F Neutron-leakage-time distributions for the two lowest energies of the 4" x 4" x 3" light water, polyethylene pellets, and polyethylene samples on linear and semi-log plots.

Fig. 24—F Comparison of the neutron-leakage-time distributions of a 4" x 4" x 3" polyethylene and 10" x 10" x 3" polyethylene samples.

Fig. 25—F Neutron-leakage-time distributions for the two lowest energies of the 4" x 4" x 3" polyethylene and 10" x 10" x 3" polyethylene samples plotted on linear and semi-log scales.

TABLE II-F
EXPERIMENTALLY OBSERVED PULSE WIDTHS (μsec)*

		<u>4" x 4" x 3" Samples</u>						
<u>Sample</u>	<u>Temp.</u>	<u>0.4586 eV</u>	<u>0.3185 eV</u>	<u>0.2038 eV</u>	<u>0.1147 eV</u>	<u>0.07963 eV</u>	<u>0.05096 eV</u>	<u>0.01274 eV</u>
Water	Room	5.5	5.1	6.4	10	20	37	62
Water + Cd Sulfate	Room	4.2	5.2	6.0	9.5	14	33	56
Water + Boric Acid	Room	4.8	4.8	6	9	15	26	42
Polyethylene Pellets	Room	3.8	5.5	6.5	12	18	38	65
Polyethylene	Room	4.6	4.6	6.1	11	19	37	66
Polyethylene	Liquid N ₂	4.6	4.5	5.4	7.5	11	15	75
Solid Ammonia	Liquid N ₂	3.5	4.4	5.0	6.0	8.3	11	43
Polyethylene + Cd Sheet	Room	2.9	3.6	4.3	5.8	9.3	12	24
Lucite	Liquid N ₂	9.7	5.8	6.5	9.0	11	18	92
Zirconium Hydride	Liquid N ₂	6.6	6.9	7.8	18	30	39	94
Zirconium Hydride	Room	4.9	5.6	8	19	25	35	78
		<u>4" x 4" x 1-1/2" Sample</u>						
Polyethylene	Room	4.3	4.5	5.4	9.0	14	28	50
		<u>4" x 4" x 1" Samples</u>						
Water	Room	3	4	5	9	14	20	35
Polyethylene	Liquid N ₂	4.5	4.2	4.7	7.5	9.0	15	54
Zirconium Hydride	Room	4.5	4.1	5	11	16	21	42
		<u>10" x 10" x 3" Sample</u>						
Polyethylene	Room	3.3	5.8	7.5	12	19	46	78

*Resolution effects have not been subtracted from these data.

TABLE III-F
EXPERIMENTALLY OBSERVED "FIGURES OF MERIT*" FOR VARIOUS MODERATORS**

		<u>4" x 4" x 3" Samples</u>						
Sample	Temp.	0.4586 eV	0.3185 eV	0.2038 eV	0.1117 eV	0.07963 eV	0.05096 eV	0.01274 eV
Water	Room	1.0	1.0	1.0	1.0	1.0	1.0	1.0
Water + Cd Sulfate	Room	2.0	1.1	1.2	1.1	2.3	1.3	1.2
Water + Boric Acid	Room	1.4	1.6	1.3	1.4	2.0	2.2	2.0
Polyethylene Pellets	Room	2.0	.91	.89	.63	1.1	.66	.67
Polyethylene	Room	1.4	1.7	1.4	1.2	1.3	1.1	1.0
Polyethylene	Liquid N ₂	1.5	1.9	1.7	2.4	3.9	7.0	1.7
Solid Ammonia	Liquid N ₂	2.9	1.7	1.8	2.7	5.3	10.8	5.4
Polyethylene + Cd Sheet	Room	4.0	2.2	2.1	2.8	3.7	6.1	3.5
Lucite	Liquid N ₂	1.9	1.0	1.2	1.4	3.9	4.2	.88
Zirconium Hydride	Liquid N ₂	.40	.29	.33	.22	.20	.25	.09
Zirconium Hydride	Room	.94	.56	.36	.20	.41	.37	.14
		<u>4" x 4" x 1-1/2" Sample</u>						
Polyethylene	Room	3.7	3.2	3.3	3.0	5.4	3.3	2.9
		<u>4" x 4" x 1" Samples</u>						
Water	Room	2.9	1.3	1.2	.77	1.4	2.0	1.7
Polyethylene	Liquid N ₂	1.8	2.1	2.4	2.3	6.4	6.5	2.4
Zirconium Hydride	Room	.49	.55	.43	.23	.33	.41	.20
		<u>10" x 10" x 3" Sample</u>						
Polyethylene	Room	.55	.55	.49	.57	.9	.43	.44

*The figure of merit is relative to the 4" x 4" x 3" water sample at room temperature and is defined as follows:

$$\text{Figure of Merit} \equiv \frac{\text{Peak Height}}{[\text{Full Width at Half Maximum}]^2}$$

**Resolution effects have not been subtracted from these data.

TABLE IV-F

Crystal Spectrometer and Chopper Resolutions

<u>Crystal Spectrometer</u>		
<u>Energy (eV)</u>	<u>Time Resolution (μsec)</u>	<u>Energy Resolution (μsec/meter)*</u>
0.01274	10.8	3.0
0.05096	4.4	0.88
0.07963	3.5	0.65
0.1147	3.1	0.47
0.2038	2.9	0.32
0.3185	2.8	0.25
0.4586	2.8	0.20
0.6243	2.8	0.17
<u>Chopper TOF Spectrometer</u>		
All energies	7.6	3.4 [†]

*For an accelerator pulse width of 2.5 μ sec.

[†]8 μ sec channel width and 7.2 μ sec chopper burst.

for all energies but the trailing edge is masked by subsequent up-scattering and events leading to thermal equilibrium. This well ordered slowing-down transient appears to have a shape similar to a chi-squared distribution function as demonstrated in the various size samples. Their shapes are relatively spatially independent.

b) The intensity of the surface leakage transient functions are surprisingly large and have a strong dependence upon the thickness of the material for the slab geometries tested. This thickness dependence also appears to be greater than can be accounted for by differences in hydrogen densities. The thickness for maximum intensity appears to lie between 1-1/2" and 2" for those materials where thickness was varied. It is expected that a major reason for this thickness dependence can be found by studying the fast source distributions. From gold and indium activation measurements one can conclude that the optimum thickness seems to be highly correlated with the episcadmium fluxes. A correlation between the distance at which this episcadmium flux peaks perpendicular to the source and the optimum thickness also appears to be present.

c) The slowing-down transient pulse widths (also observed times to peak) are shown to have a materials sensitivity greater than that resulting from hydrogen density. It is to be expected that they are scattering kernel sensitive. However, the short time observed for solid ammonia was anticipated because of the large total cross sections observed at long neutron wavelengths[11-F]. Evidence is therefore presented for such a correlation which could be anticipated from the reciprocity theorem. Liquid and solid methane have been suggested by Brugger and others to provide better possibilities based upon this approach.

These general conclusions should be taken in the light of first order terms and first of a kind experiments. Modifications and corrections are anticipated because there appear to be other effects of structure in the data. There are several unexplained phenomena which merit further investigation such as the difference in pulse shapes for the forbidden orders compared to the allowed orders. Usually this will not be important to the moderator optimization problems. It is planned with the next set of measurements to try to optimize the moderator thickness by taking measurements of several different thicknesses between one and three inches. The optimum thickness does not appear to be the same for polyethylene and water. Other things planned are more homogeneous and heterogeneous poisoned samples and studies of materials at liquid helium temperature. Another interesting thing to look at would be heterogeneous poisoned ammonia at low temperatures. Other materials that might be of interest would be cold methane and liquid hydrogen.

All variables and methods which have been examined are found to have important properties in modifying the pulse widths particularly in modifying the properties of the fundamental mode. Because the slowing-down transient is rapid, leakage or buckling in the sense of the fundamental, and poisons ($1/\nu$, non- $1/\nu$ such as Cd, or heterogeneous) except in their more extreme or limiting forms have surprisingly small effect upon the intensity of the initial transient for the energy range observed. In return they have large effects on the equilibrium "die-away" times, hence on pulse widths in the neutron temperature ranges

of the moderator. These studies show that a large variety of methods is available for obtaining improved moderator "figures of merit" for room temperature neutrons, and reactor sources which produce source pulses greater than 10 to 15 μ sec cannot take advantage of these improved "figures of merit".

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APPENDIX G
FIGURE OF MERIT DEVELOPMENT

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APPENDIX G

Figure of Merit Development

A fundamental relationship which gives the number of neutrons/cm²sec per channel width τ_2 at a distance d from a moderator source can be given as follows:

$$N(E) = \frac{J_o f \tau_1 A_1 \epsilon A_2 \tau_2}{4\pi d^3}, \quad (1-G)$$

where

J_o = the peak rate of the leakage neutrons leaving the reactor per cm² during a pulse

f = the frequency of pulses per second

τ_1 = the fast reactor pulse width

A_1 = the area or solid angle subtended by the moderator with respect to the reactor

ϵ = the moderator efficiency, ie, the relative effectiveness of the moderator

A_2 = the area of the moderator being viewed

and τ_2 = the channel width or time of observation at distance d .

The only difficult part to understand is the fact that the number observed per channel τ_2 depends inversely upon the distance d , and some of the concepts of moderator efficiency and area could be refined.

Following the usual development and multiplying the numerator and denominator by τ^3 , the following expression is obtained:

$$N(E) = \frac{J_o f \tau_1 \epsilon A_1 A_2 \tau_2}{4\pi \tau^3} \left(\frac{\tau^3}{d^3} \right). \quad (2-G)$$

Making the following specifications, definitions and qualifications, expression (3-G) is obtained:

A) Defining the resolution as being $\tau/d = R$, where τ is now the time uncertainty about the energy of the neutron that arrives during τ_2 .

B) Noting that the product $J_o f \tau_1$ is proportional to the reactor power density $P_d \left[\frac{\eta-1}{\eta} \right]$ where η is the effective neutrons produced per fission in the fast reactor source.

$$N(E) \propto \frac{P_d (\eta-1) \epsilon A_1 A_2 \tau_2 R^3}{\eta 4\pi \tau^3}. \quad (3-G)$$

In order to compare two reactor systems, an accepted practice is to examine their relative merits on the basis of comparative neutrons/cm²sec at an experimental distance d and for the same resolution. The ratios of the numbers of

neutrons/cm²sec is given in the expression as follows where primes are used to denote different reactors:

$$R = \frac{N(E)'}{N(E)''} = \frac{P_d'(\eta'-1)\epsilon'A_1'A_2'\tau_2'}{P_d''(\eta''-1)\epsilon''A_1''A_2''\tau_2''} \frac{(\tau'')^3}{(\tau')^3} \frac{\eta}{\eta'} \quad (4-G)$$

Leaving considerable latitude for discussion, it is assumed that $\epsilon'A_1'A_2' = \epsilon''A_1''A_2''$ and that $\frac{\eta-1}{\eta} = M$ where M is the leakage fraction, then Equation (4-G) can be simplified to the following expression:

$$R = \frac{P_d'M'\tau_2'}{P_d''M''\tau_2''} \frac{(\tau'')^3}{(\tau')^3} \quad (5-G)$$

In order to have equivalent resolutions it is also customary for $\tau_2'/\tau_2'' = \tau'/\tau''$, so that the usual expressions for comparison in terms of the figure of merit are obtained by the following equation:

$$R = \frac{P_d'M(\tau'')^2}{P_d''M''(\tau')^2} \equiv F'/F'' \quad (6-G)$$

where $F' = P_d'M'/(\tau')^2$, and $F'' = P_d''M''/(\tau'')^2$.

While still a simplification, the value of τ is assumed to be given by the following expressions:

$$\tau = (\tau_1^2 + \tau_m^2 + \tau_2^2)^{1/2}$$

where

τ_1 = fast reactor pulse width

τ_m = the delta function response of the moderator

and

τ_2 = the channel width as before for which the neutrons are observed

and

$$\tau = \tau_2 \left[\frac{\tau_1^2 + \tau_m^2}{\tau_2^2} + 1 \right]^{1/2} \quad (7-G)$$

There are many strategies that can be assumed, but for the sake of a comparison it seems most valid to assume that τ_1 and τ_m are fixed as presented by the reactor designer. With this assumption an optimum $N(E)$ in Equation (3-G) is obtained by letting $\tau_2^2 = \tau_1^2 + \tau_m^2$, so that Equation (3-G) can be

rewritten as given in Equation (8-G) and Equation (6-G) can be rewritten as given in Equation (9-G) as follows:

$$N(E) \approx \frac{P_d (M \epsilon A_1 A_2 R^3)}{4\pi 2^{3/2} (\tau_1^2 + \tau_m^2)} \quad (8-G)$$

and

$$R = \frac{P_d' M' (\tau_1'^2 + \tau_m'^2)}{P_d'' M'' (\tau_1''^2 + \tau_m''^2)} \quad (9-G)$$

Equation (9-G) is now in the form desired to compare different reactor moderator sources, but some discussion of the limits of validity are justified. Basically the pulsed reactors are limited by τ_1'' being fixed because of the slow pulses yielded by the mechanical rate of reactivity insertion. Once it has been established that τ_m'' can be made small compared to τ_1'' without loss of peak intensity, it becomes more advantageous to reduce τ_1'' . Previously for the booster systems it has been widely assumed that $\tau_m' > \tau_1'$. If the values of τ_m' can be reduced to be comparable with τ_1' or τ_1' can be reduced to equal τ_m' , more detailed considerations of the type made by Stevens^(1-G) become important. The above development is consistent with Stevens' conclusion of maximum source and reactor power. Except for the case where τ_m' can be reduced, the limiting factor is the power density and it is not useful to extract the reactor pulse width or the reactor neutron source width from the power density term.

Because τ_m' is energy dependent and becomes increasingly smaller as the neutron energy is increased, considerations such as those of Stevens are very important when considering higher neutron energies, and a resolution vs energy plot is required to tell the whole story. Efficient approaches to provide this information are being considered. For these cases where utilization of higher energies become important large returns can be obtained by an increase in the neutron power output of the charged particle neutron source. The enhanced neutron source performance would be particularly valuable to maintain shorter pulses for higher neutron energy spectroscopy. Detailed discussions of these matters are to be forthcoming in Appendix B.

A comparison of the "figure of merit" for the various reactor concepts based on Equation (9-G) is given in Table I-G. A 50 μ second reactor pulse was used as the standard for comparison with equal moderator pulse widths for both systems.

TABLE 1-G

Relative "Figure of Merit" Comparisons for Different Systems

	τ_1 (μsec)	τ_m (μsec)	P_d (MW/l)	n	Relative* FOM
RPTF	5	9	.55	2.65	14
BNL(APR)	6	9	1.5	2.5	33
SORA	50	9	0.10	2.5	0.10
BNL(PR)	60	9	1.5	2.5	1.1

* Relative to standard conditions for $P_d = 1$ MW/liter, $\tau_1 = 50$ μsec , $n = 2.5$.

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