REACTOR KINETICS
AND
REACTOR SAFETY STUDIES

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ABSTRACT

The goals and accomplishments in reactor kinetics and reactor safety studies of the Atomics International Reactor Safety Program are reviewed. Present directions and future requirements are presented.

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I. INTRODUCTION

It is the purpose of this document to report on the area of the Atomics International Reactor Safety Program where nondirected research has been and is being pursued. Accomplishments in this area can benefit all reactor safety and development projects. The goals can be suitably expressed by the purpose and need for Project 4589, Reactor Kinetics:

"In order to achieve economic nuclear power without compromising the need for safety it is vitally important to be aware of potential problems in the design and operation of future reactor systems as well as to improve the understanding of present reactors. The objective of this project is to investigate the general area of reactor dynamics to develop a fundamental understanding and in order to seek out potential problems concerning the safety of reactor systems. Theories will be formulated where none exist and, with present theories, tested or verified by experiments or application of previous measurements where feasible. This work will not be limited to specific goals but will look toward potential problems to provide direction for the work.

"It is also important to have realistic models of the dynamic behavior of reactor systems in order to understand their basic safety. It is the intent of this project to develop models which can lead to an understanding of the dynamic behavior of reactors and to use these models for parametric safety studies. Computer programs will be written to provide tools for the use of these models and to allow the theories developed under this project to be used for hazard analysis by engineering personnel. Current practices in analyzing the safety of reactors will be scrutinized and improved where possible."

It is the intent of this MEMO to document past accomplishments and show how present and future research will contribute to the reactor development program at AI and throughout the industry.

II. PAST ACCOMPLISHMENTS

Past accomplishments are in such diverse areas as numerical analysis, reactor noise, bubble growth, code development and poisons to enhance reactor safety. All are related to the overall goals of the project as presented in section I and have in addition contributed to other areas of reactor physics. Let us briefly review the accomplishments before discussing present and future research.

A. NUMERICAL ANALYSIS

Of fundamental importance is the ability to follow the time behavior of a reactor including the various physical phenomenon which effect the behavior. Although one can solve the space independent equations (including delayed neutrons) analytically it is in general not possible to do so if feedbacks are included; hence the necessity for a numerical method. As recent as 1957 the best* numerical method available was Runge-kutta. This method is severely limited in the size of the time step because of the existence of a solution of the form $e^{-\beta t/l}$ where β/l is of order 10^{-2} - 10^{-6} . Thus thousands of calculations were required to follow reactor transients.

1. AIREK Equations

During FY '58 E. R. Cohen developed a numerical procedure which was not so limited¹. This procedure is a generalization of Runge-kutta accounting explicitly for the exponential nature of the solutions and for the first time provided the industry with a fast reliable method for reactor kinetics calculations. As a result the first AIREK code² was developed. This code was used in one form or another for several years and was recoded in 1962 (AIREK 3)³ to take advantage of new programing and graphical display techniques⁴.

Special versions of AIREK were used in the KEWB program as tools to verify the physical models postulated to explain radiolytic gas formation, nucleate bubble production, and other feedback mechanisms^{5,6}. These tools were invaluable in developing an understanding of the phenomenology of KEWB.

*Fastest method for a given accuracy.

AIREK 3 became the standard engineering tool for space independent reactor kinetics (in addition to analog computation) at AI, throughout the country and in other parts of the world. The equations solved by AIREK 3 are the usual space independent equations¹.

The code computes about 300 time points per second per differential equation on an IBM 7090. For example a problem with 6 delayed neutron groups, one feedback equation, $\beta = 10^{-4}$ 90% step reactivity insertion required 66 time steps (with a maximum step size of 0.2 seconds) to follow the transient for 10 seconds. The computer time was 1.5 seconds for calculation, 1 second for printed output and 4 seconds for graphical output; the maximum error in the results was less than 0.1%. Thus AIREK 3 will follow rapid transients with a minimum of computer time. Unlike other codes, AIREK 3 also has the ability to take large (~ 3 seconds or more) time steps when following slow transients.

2. Quasi-Linear Equations

Although the AIREK equations served as useful integration equations and overcame many deficiencies of the other methods, the equations are deficient in two respects: (1) A general error analysis is not possible and (2) for those special cases where it is possible to analyze the error, it can be shown that the method reduces to a generalization of Runge-kutta but requires 5 intermediate evaluations rather than 4 for an error of order h^4 . Consequently additional work was supported by the project to develop a more efficient numerical procedure for integrating the reactor kinetics equations (1-4).

The result of this additional investigation of numerical methods was the development of the quasi-linear equations⁷ which represent a generalization of the mean value approach to the Runge-kutta equations. The quasi-linear equations of comparable accuracy to the AIREK equations have several advantages. Firstly, only four intermediate evaluations are required. Secondly, generation of only one exponential function is required. Thirdly, the equations reduce exactly to the Runge-kutta equations for certain special cases.

Numerical verification of the efficacy of the method has not as yet been accomplished due to limited funding during FY 1962-63. It is

planned to test the method during the present fiscal year (1964) and possibly to use it as the basis of the Dynamic Simulator (see section III-A) currently under development.

B. REACTOR NOISE ANALYSIS

During the period FY 1959-61, theoretical investigations^{8,9,10} were carried out as to the possibility of using the random fluctuations of a power reactor operating at steady state to obtain dynamic information for that reactor. The result of these studies was the formulation of a theory which is the basis of what is now a standard experimental technique: noise analysis.

It is a well known characteristic of reactors that at steady operation, measurements of the power level reveal the presence of small, apparently random fluctuations about the mean power level. These fluctuations, appropriately called power noise, represent the response of the reactor to two types of stimuli. These may be classified as (1) Thermal Fluctuations (2) Nuclear Fluctuations. The former are temperature dependent and barring quantum considerations disappear at absolute zero while the latter are insensitive to temperature and are peculiar to reactors.

In 1958 Moore suggested that since the power noise represented the response of the reactor to a stimulus, however complex, the characteristics of the noise were governed by the kinetics of the reactor and in particular, the noise spectrum was shown to be related to the reactor power transfer function. It was proposed therefore that the power transfer function could be determined by a measurement of the power noise spectrum. This suggestion was explored by Cohn¹¹ and was found to give excellent results. In particular, Cohn was able to determine the ratio of the effective delayed neutrons fraction β to the prompt neutron lifetime ℓ , for a variety of reactors. These experiments were repeated¹² with comparable success. Since, β/ℓ usually determines the high frequency end of the noise spectrum, Cohn's method was designed to be effective in the range 10-500 C.P.S. and the low frequency spectrum was not explored by these measurements. These low frequencies are of interest because they are usually determined by the thermo-mechanical feedback mechanisms

in the reactor. Low frequency spectral measurements have been made¹³ with comparable success and appear to be most helpful in (a) verification of theoretical feedback models (b) evaluation of physical parameters involved in these mechanisms.

The result of this work, that the noise spectrum is proportional to the square modulus of the transfer function, although strictly true only for white noise sources can be shown to be approximately true for all sources and is the principal result of the theory. Experimentally this suggests evaluation of the constants in the transfer function by fitting to an experiment. Basically there are two methods; either by calculation of the auto-correlation function or by direct frequency analysis of the signal. The first method has been used by Hirota (1960) and the second by Cohn, Lundholm and Griffin, and Cummins. In practice, the auto correlation analysis of the noise record is perhaps most convenient for the lower frequencies while the direct frequency analysis is most convenient for the high frequencies although with special equipment either method can be extended into the range of the other.

It was pointed out that since reactor noise is characteristic of the reactor in which it appears, it should be possible to determine kinetic parameters from power noise measurements.

That this is indeed possible was shown by Cohn for a number of reactors. Using a tunable band pass filter, Cohn was able to obtain the high frequency tail of the power noise spectrum, whose shape is determined by the ratio of the effective delayed neutron fraction β to the prompt neutron lifetime ℓ . The results agreed within the experimental error with those obtained from (a) a $1/v$ poison measurement, (b) the deHoffmann counting statistics method, and (c) measurements using Rossi techniques.

Although the high frequency tail of the spectrum is determined by β/ℓ , the low frequency portion is more complicated and should provide an additional means for obtaining other kinetic parameters.

In order to make more detailed studies of power noise, it is first necessary to develop a formalism by means of which the noise transfer

function corresponding to any given kinetic model can be calculated. With this end in mind, a very general kinetic network consisting of a set of n coupled branches in n dependent variables was considered.

The results show that from any kinetic model, the corresponding macro-stochastic equations for the noise can be formulated. These equations are satisfied by the various noise correlation functions and can be solved for the various noise spectra. It has been suggested that these spectra should be power dependent. Making use of this power dependence together with the frequency dependence of the spectra should permit determination of kinetic parameters. In principle, more of such information is obtainable from noise measurements than from pile oscillator techniques, because the latter activates only one loop of the system, while the former can activate all loops.

The method of noise analysis has proved to be a significant contribution to the field of reactor kinetics. The method is useful in determining the shape of the transfer function and the lifetime. In particular the method extends the frequency range in which such measurements are possible and adds to our experimental capability.

C. POISON MIXTURES TO ENHANCE REACTOR SAFETY

It was suggested that the addition of poisons to the moderator of thermal reactors could increase the negative temperature coefficient and at the same time reduce the temperature defect* thus improving safety. The former is accomplished by introducing a poison having a resonance in its absorption cross section above the normal operating temperature. Thus if the reactor goes on an excursion the moderator temperature would increase and raise the average neutron temperature toward the resonance energy of the poison. This increase in spectrum produces additional parasitic captures, reduces the thermal utilization and consequently the reactivity. On the other hand reduction of the temperature defect requires a poison whose cross section decreases as the temperature is increased from ambient to operating temperature.

The requirements on the temperature variation of the poison cross sections appear at first to be mutually exclusive. It would, to say the

*Reduction in reactivity due to increase from ambient to operating temperature e.g. burnable poison.

least, be extremely fortuitous to find a poison having such characteristics for any given operating temperature. Consequently a study¹⁴ was first begun to find poisons which would increase the negative temperature coefficient. The results showed that cadmium or erbium satisfy this requirement; however, the resulting reactivity penalty must be about 5% at normal operating conditions for significant improvement.

Based on the results of the first study, a search ensued to find binary poison mixtures having the characteristic that the cross section decreases from ambient to operating temperature but increases above operating temperature. The results of the latter study¹⁵ showed that mixtures of gadolinium-europium and cadmium-gadolinium among other each possessed the desired characteristic. Thus it was established that reduction in temperature defect is possible simultaneous with an improvement in stability against positive reactivity insertions.

D. BUBBLE GROWTH

1. KEWB

In an attempt to understand the shutdown mechanisms of a homogeneous water boiler theories were postulated to explain bubble formation and were compared to experiment. These bubbles could be radiolytic gas, vapor, or nucleate (fission) bubbles. Several questions related to bubble formation and growth were pursued in order to gain a fundamental understanding of these processes.

A wave equation was derived for the pressure field governing bubble growth and the Green's function found¹⁶. Several examples were pursued in an attempt to correlate with KEWB power traces. Under the assumption of fixed bubble radius the problem of the diffusion of radiolytic gas from a constant source was solved¹⁷. The following complementary problems were investigated by Flatt¹⁸: (1) The gas pressure is constant; the solution is assumed to expand sufficiently rapidly to alleviate any inertial pressures. (2) The pressure in the solution is a given function of time so that the bubble radius is the unknown. A fission nucleation model was postulated to explain the inertial pressure¹⁹. This model assumes that fission nucleation of a "hard" bubble compresses the

surrounding solution with consequent pressure rise; further the diffusion of radiolytic gas from the solution into the bubble causes the gas pressure to increase.

Application of these theories to KEWB transients provided fundamental insight into the phenomenon of inertial pressure and bubble growth as a mechanism for shut down.

2. Bubble Growth on a Heated Plate

Studies of bubble growth on a heated plate were initiated in an attempt to determine the effect of viscosity on latent heat transfer in a bubble²⁰.

An analytic solution was obtained for the equilibrium radius and temperature field of a vapor bubble²¹ in an attempt to advance knowledge of sub-cooled nucleate boiling and to predict coolant expulsion during a transient²¹. The model uses a Plesset-Zwick solution while the bubble is in the boundary layer and a Bankoff-Miksen solution outside.

Approximate solutions were found for the Navier-Stokes equations for viscous flow about an expanding or contracting bubble and were extended by non-dimensionalizing temperature. These solutions indicate that the integral of the vertical velocity of an expanding bubble does not exist. This may mean that the flows about expanding and collapsing bubbles are qualitatively different.

These studies have added to our understanding of sub-cooled boiling and our ability to predict the voiding mechanism in reactor transients.

E. MULTICHANNEL FLOW STABILITY

The theory of multichannel two-phase flow stability was developed in 1959^{22,23} to provide an understanding of the propagation of channel choking and subsequent core meltdown. This theory provides the conditions for which propagation of boiling from one channel to another will or will not occur and thus has an important place in reactor hazard studies. The ultimate result of propagated channel choking is core meltdown and pressurization with the possibility of breaching the primary containment.

An analysis was made of coolant boiling in sodium graphite reactors (SGR) with application to SRE²⁴. The results showed that boiling propagation in SGR's is only a contributing factor in core meltdown. Propagation of boiling will be self sustaining only if the coolant in the affected channels is already boiling or near boiling and will not occur if the reactivity insertion is terminated or if widely differing temperature conditions exist among the channels. Finally, localized boiling or vapor formation cannot lead to propagation of boiling and ultimate core meltdown.

III. PRESENT DIRECTIONS

With the availability of additional funding in FY 1964 to support the reactor kinetics project for the first time since FY 1961, three major tasks in reactor dynamics were pursued. The first task is a theoretical investigation in space dependent kinetics to determine what dynamical information contained in the dispersion law can be measured. This work may lead to the development of continuous on-line criticality monitor. The remaining tasks were designed to develop tools with which to meet the goals of the project as described in section I and at the same time provide engineers involved in reactor dynamics with usable tools for design and hazards studies.

A. DYNAMIC SIMULATION

In order to provide useful tools for understanding the dynamic behavior of reactors, several digital codes were developed which are extensions of AIREK 3 to incorporate the phenomenology of heat transfer, fluid flow, decay heat, fuel melting, and coolant boiling. These tools will be used to test postulated models for fuel melting, boiling, pressurization, etc., and to obtain an understanding of the dynamical behavior of real reactor systems.

1. AIREK 3A

For lack of a better name, we will refer to the first simulator we developed as AIREK 3A²⁵ because of the fact that the AIREK 3 code forms some 90%. AIREK 3 is a general space independent reactor kinetics code allowing for 15 delayed neutron equations and 50 feedback equations.

Many kinds of feedback can be represented by the formulation. However the equation will not handle a distributed heat transfer model. The equation describing the behavior of prompt neutrons is a non-linear, variable coefficient, first order differential equation while the remainder are linear, constant coefficient, first order differential equations. The method used to solve these equations is that developed by E. R. Cohen¹ - essentially a 5th order correct method which we shall refer to as the Geneva-58 method. Since the equations are first order, the problem is

an initial value problem i.e., given conditions at time t find the solution at a later time $t+h$ (where, hopefully, h is not too small). Incorporated in AIREK 3 is an automatic interval switching technique such that the step size h , is changed (doubled or halved) through the calculation so as to bound the error. Intervals are recomputed automatically if the error exceeds the upper bound. The output is limited to digital printout and graphical display of the neutron level, reactivity, inverse period, 6 delayed neutron groups (if wanted), and 4 feedbacks.

AIREK 3A differs significantly from its predecessor in only two ways although other differences exist. Of most significance is the generalization of the feedback equations. A new form of the feedback allows for all the previous generality plus the representation of a nodal heat transfer representation including both conduction and convection. One could follow one channel with 5 axial and 10 radial nodes or 3 channels each with 3 axial and 4 radial nodes.

The second major change is that up to 25 of the feedbacks can be printed and/or displayed according to input options. Furthermore, the feedbacks can be grouped on the CRT frames in an arbitrary fashion.

Other differences include flow decay, flow dependence on the film coefficient, decay heat production, and scrams based on power, temperature, and/or period with reactivity based on rod drop through gravity.

The experience gained in developing and using AIREK 3A has led to the development of a far superior dynamic simulator which is now being checked out.

2. AIREK 3B

In order to provide temperatures typical of catastrophic excursions in a SGR for the fission product retention program, AIREK 3A was modified to incorporate crude models for fuel melting, nucleate boiling and vaporization of coolant²⁵. Both the average channel (reactivity feedback) and the hot channel (representing 12 central elements) are followed in this study. Three axial nodes are used but only 2 radial; fuel and coolant.

The following phenomena are considered: burn-out, nucleate boiling and bulk boiling. If the heat flux exceeds burn-out in the hot channel the fuel is assumed to fall according to gravity and the reactivity varies as the importance of the point to which the rod has fallen. If the temperature of the coolant in either the Av or hot channel reaches 1570°F, the film coefficients are replaced by nucleate boiling coefficients. If the temperature of the coolant in either the Av or hot channel reaches the boiling point (1616), the temperature is held constant until sufficient time has elapsed that a heat flux corresponding to 90% of the latent heat of vaporization has passed from the fuel to the coolant and at that time the channel is assumed voided and a positive reactivity step is inserted. At such time later that 100% of the latent heat of vaporization has transferred to the coolant the fuel rod is assumed to fail as in the burn-out case. Either bulk boiling or burn-out in the average channel terminates the problem. The cases ran were banked rod withdrawal from full power at rates corresponding to 10, 5, 2, 1, 0.5 and 0.1 \$ per second.

3. A Dynamic Simulator

One of the pitfalls in developing a general code of any type is that special calculations and subroutines somehow become included, usually because of a requirement to complete the code by a given date. Although AIREK 3A was intended to be general, several special features are included e.g., only certain nodes are used for scrams, special printing features, and the special form of the flow decay equations. The justification for these non-general features is, of course, the pressure of time.

All was not lost, however, as much was learned and much can be used directly in the new dynamics simulator which is presently in the checkout stage. This section then will serve as the specifications.

Foremost is the need for more feedback equations; the plans at present call for 100 with any 50 edited i.e. printed and/or displayed. Little or no increase in running time per equation will result and in fact we could allow several hundred, but 100 is felt to be a practical upper limit.

The reactivity calculation has been generalized so that any feedback variable can be tested for reactivity scram, each with a unique delay time. Start-up problems can be handled such that until sensible power generation, 0.1% full power, say, the feedback equations will not be integrated since no change is expected; this feature reduces the running time of start-up problems by 75%. CRT displays of neutron level, inverse period, delayed neutrons and reactivity up to the time of sensible power can be provided (on an option) in addition to the normal displays. Flow variation can take several forms: (1) simple exponential as in AIREK 3A, (2) a table provided in the input data, (3) defined by the external loop equations. The special output routine can be extended so that specified linear combinations of feedback variables can be edited. The interval switching procedure will test all feedback variables in addition to power.

This simulator is being built of small modules or subroutines so that future modifications or additions can be made as better physical models are available. An initial temperature calculation can be incorporated for use on an input option. An input generation routine is being written for optional use to compute the conduction and convection coefficients, for an nxm nodal system with built-in tables of specific heat, density, conductivity and viscosity for the common fuel, structure and coolant materials.

B. REACTOR SITING

In order to assess the potential hazards of reactor excursions and/or other incidents in which fission products are released from the primary containment a reactor siting code, AISITE, was written²⁶.

AISITE is a computer program developed to satisfy the need for a fast, reliable method of studying accidents and determining siting criteria for reactors. It is based on methods developed by the Division of Licensing and Regulation of the United States Atomic Energy Commission (USAEC)²⁷. Basically the code computes dose versus distance for assumed values of engineering parameters and meteorological conditions, and finds the

critical distance factors (exclusion, low population and population center distances) based on tolerance doses to the important organs.

The major features of the code are described below.

1. Reactor Inventory

The formulation and assumptions are those of reference 27, with the provision for cooling of the fission products.

2. Isotope Release

In contrast to reference 27, AISITE allows multiple containment (up to 4 levels). Isotopes are grouped according to halogens, bone seekers, gases, solids, Na^{24} or gross fission products; each group has a separate release fraction and clean-up rate.

The equations are based on assumed exponential release, i.e., the rate of leakage at time t depends on the amount present at time t . More general dependence is needed and in particular linear release (rate of release is proportional to the initial amount) should be treated explicitly rather than the present stop-gap technique.

3. Isotope Inhalation

The formulation for the transport and subsequent diffusion of the cloud is that of Pasquill²⁸ rather than Sutton's which is used in reference 27. In addition provision is made in the code for release through a stack and the cloud of fission products is allowed to decay as it moves downwind. Meteorological conditions are specified as input; type F stable dispersion has been used most commonly to date.

4. Direct Dose from Building

The direct dose from the building is calculated using the following assumptions:

a. Dose point is sufficiently distant that the building is a point source;

b. Inventory of isotopes does not decrease with time, i.e., source from building is constant - this assumption vastly overestimates the close-in dose; and

c. No shielding by the building or inner containers although attenuation in air is accounted for - this assumption overestimates the close-in dose by 10-10,000.

5. Direct Dose from Cloud

The formulation is that of reference 29, considers only gammas and assumes a semi-infinite cloud. This model underestimates the close-in dose and is in need of improvement.

6. Exclusion Area

This distance is found by searching for the largest distance at which an organ receives its tolerance dose (300 rad for the thyroid or 25 rad for any other organ) in a two-hour release.

7. Low Population Boundary

This distance is found in the same manner as the exclusion area except the time of release is taken to be 30 days.

8. Low Population Center Distance

This distance is defined to be 1.3 times the low population boundary distance.

Other features of the code include built-in libraries of isotope decay constants, average gamma energies, yields, absorption coefficients in air, mass of organs, and others. Addressable data is used so that multiple cases can be run where only changes are specified. The code automatically varies any parameter (building leak rate, filter efficiency, clean-up rate, etc.) and finds the critical distance factors. Graphical displays of dose versus distance for any organ and of the critical distances versus the parameter are available.

AISITE has found many applications in the time it has been available. Hazards analysis of a general nature have been performed for a typical SGR as well as for DON, SRE-PEP and others. Accidents such as core melt-down, sodium fires, fuel element fires, organic fires and steam generator ruptures have been studied. Many other kinds of accidents can be studied with the present code.

The following modifications have been proposed and are being implemented:

1. Improvement of Direct Dose Calculation

a. At present the direct dose from the building is computed without accounting for the shielding effect of the (multiple) containers; this overestimates the close-in doses.

b. The source of direct dose is taken to be constant but should be decreased by what escapes.

c. Provision should be made for the direct dose from sodium.

2. Release Phenomenon

a. Derive and code correct formulation for linear release rates as from fires.

b. Provide for a general release rate function.

c. Consider hold-up of products emitted by stack.

3. Cloud

a. Improve direct dose calculation from overhead cloud.

b. Improve cloud model.

c. Provide for β^- s.

C. SPACE DEPENDENT KINETICS

The general problem of the space dependent kinetics for a linear system has been considered with respect to the following question. What is there, analogous to the transfer function, which determines the stability of a space dependent system as well as the space-time response of that system to arbitrary stimuli? It was found that there exists a function, the so-called dispersion function, which determines (a) the space-time stability and (b) the response to an arbitrary stimulus. This work was reported at the "Conference on Reactor Stability and Control," University of Arizona, March 1963³⁰.

The analogy between the dispersion function and the transfer function was further pursued in an investigation into space-dependent noise. It was shown that the space-dependent noise of a reactor was governed by the dispersion function in the same way that the transfer function governs noise in small reactors. This work was reported to the ANS in November 1963³¹.

Having shown the existence and importance of the dispersion law, investigations into how it could be measured were initiated. Recognizing

that core access in a power reactor is limited, it was decided to confine all artificial stimulation of the reactor external to the core. Thus, the relation between the space-time response and the dispersion law when excited by sources on the reactor boundary was investigated and it was shown that the conventional pulsed neutron experiment and neutron wave experiment both investigated the dispersion law. It became clear that although in principle both experiments were mutually complementary, the wave experiment held greater flexibility. This flexibility originates in the fact that what the pulse experiment accomplishes by a change in buckling the wave experiment accomplishes by a change in source frequency. The theory of the wave experiment was extended to the general linear system, and it was shown that the wave experiment could be used to determine dispersion laws (a) in the presence of modal contamination and (b) with small signal to noise ratios.³²

Although access problems dictated the location of artificial sources to be outside the reactor, there are natural sources e.g. noise within the reactor and the possibility of dispersion function measurement using reactor noise was investigated. These results were reported at the "Conference on Reactor Noise," University of Florida, November 1963, in which several suggestions for experiments are included³³.

The problem of neutron wave propagation or neutron wave optics was further studied, in the general linear system, and it was shown that there can exist frequencies, called exceptional frequencies to which the medium is particularly transparent (or opaque). The search for these exceptional frequencies in a reactor, within age-diffusion theory, was undertaken and what is believed to be a new criterion of criticality has emerged viz. that state of affairs for which the reactor is completely transparent to the lowest spatial mode in the limit as the frequency approaches zero. This could be of use in the approach to criticality in a large reactor. This work is in the course of being documented.

An alternate mode of neutron wave propagation has been investigated. Rather than propagation of the neutron signal in the form of a monochromatic wave it can also be transmitted in the form of a short burst of thermal neutrons. A pulsed neutron experiment investigates the time

decay of the tail of such a burst long after it has passed. This experiment will investigate the space and time behavior of the whole wave packet as it proceeds through the reactor. Such an experiment is potentially more versatile than the wave experiment because (a) it contains information over the whole frequency spectrum rather than one frequency at a time and (b) it gives velocity information that is not present in a conventional wave experiment. The principal difficulty e.g. extraction of information from the data has been overcome and it has been shown that not only the dispersion law but its derivatives may be measured. The sharper the burst the more derivatives are measurable. The results of this latest investigation have been communicated informally to Perez at the University of Florida who plans to perform this experiment in graphite using the same equipment he used in a series of wave experiments, thus offering excellent experimental comparison of the two techniques. The analysis of the neutron wave-packet experiment is being documented.

IV. FUTURE PLANS

Fiscal Year 1964 was a period of preparing tools, surveying methods, reviewing current practices in hazard analysis, and research in space dependent kinetics. The next step in the development of reactor kinetics is twofold: (1) use the tools to understand the dynamic behavior of reactors, and (2) improve the tools by formulating physical models which will bring calculations into agreement with experiment. The results of this two-pronged approach will be a better understanding of reactor safety, the ability to predict behavior, and tools which can be used by engineers in hazards and design studies.

A. SPACE DEPENDENT KINETICS

Studies in space dependent kinetics will continue. The work will be pursued with the object of developing an alternative experimental technique to the pulsed, modulated and noise measurements. This technique will determine the dispersion law into which is imbedded fundamental information on reactor stability. It is anticipated that the use of a spectrum of frequencies rather than a single frequency will provide information in one experiment. This would be advantageous, for example, in an on-line reactor monitor. Numerous experiments are feasible: the three dimensional diffraction grating experiment, the thermal pulsed experiment, multiregion wave measurements, etc. Each type of experiment would yield fundamental information on stability. Experiments will be planned and initiated to determine the dispersion relation from actual measurements in both space and time.

The first problem to be studied will be the transmission of a neutron wave (or wave-packet) through a medium in which is imbedded another medium having a different dispersion function. A study of the transmission as a function of frequency should yield a sequence of transmission resonances rather like the Ramsauer effect in quantum theory. A study of the shape of these resonances as well as the resonant frequencies should reveal information concerning the dispersion function of the imbedded medium. If such information is revealed, this experiment would be immediately useful in the investigation of reactor cores by neutron

transmission from the reflector through the core out into the reflector again. This problem represents the first departure from the study of homogeneous systems.

The extension to inhomogeneous systems will permit that which has been until now either impossible or possible to a very limited extent i.e. space-time stability testing of power reactors. It is foreseen that two classes of such tests can be performed: (1) a careful measurement of the dispersion law by neutron wave experiments when the reactor is first put into operation and at selected times during the subsequent life of the reactor, and (2) on-line monitoring of the dispersion law by wave packet experiments to be performed during operation. The class of tests described monitor the reactor for long term changes in the dispersion law due to burn-up, poisoning, radiation damage, etc., and also serve as standards against which to compare the results of the frequent on-line measurements. The wave packet experiments, although probably lacking the accuracy of the wave experiments, have the advantage of being able to give a quick survey of the entire frequency spectrum, and as such are ideally suited to on-line monitoring.

To date, experimental confirmation of both the wave³⁴ and wave packet³⁵ experiments in graphite are available. No reliable experimental results are available in homogeneous multiplying media. No results are available in inhomogeneous media - multiplying or non-multiplying. The extension to fast systems has not been attempted experimentally or theoretically.

B. DYNAMIC SIMULATION

The development of the dynamic simulator begun in FY 1964 will be continued. Emphasis will be placed on using the code to perform parameter studies to find the importance of each parameter in determining the dynamical behavior of the system. With this understanding, the code can then be modified to account more rigorously for variations in parameters with temperature, spectrum, pressure, etc.

1. Studies of Dynamical Parameters

Every review of hazards studies cites lack of knowledge of the basic parameters which go into the study as the major uncertainty^{36,37}.

Questions relating to these parameters and their effect on reactor dynamics will be investigated and resolved. Typical areas of investigation would include: the effect of uncertainties in lifetime, beta, heat transfer coefficients, etc., on the results of postulated accidents; the effect of burn-up on these parameters and its effect on safety; the importance of variation of these parameters during a transient due to change of temperature, spectrum, volume, etc.; and the best way to calculate these space independent parameters. The results of this study will provide insight to the importance of the errors and will help to set the reliability of future hazards studies. If, for example, the temperature variation of reactivity coefficients is significant in determining the result of a postulated accident, then the code will be so modified.

Studies of dynamical parameters will contribute to reactor safety in several ways:

a. The results of such studies on specific reactor systems give us intimate knowledge of the effect and uncertainties of design parameters on postulated accidents, thus leading to an overall understanding of their relationship with safety. Thus understanding will permit more intelligent reactor design and ultimately relieve the present requirement for overdesign of safety features because of lack of knowledge.

b. These studies will indicate where future research emphasis should be directed i.e. which parameters have little effect on the dynamical behavior or ultimately on the results of postulated accidents and which parameters are significant. If uncertainties exist in the method of calculating any of these or if only experiment will provide answer, then the way for future work is clear.

c. It will become obvious that the variation of certain of these parameters during a simulated transient must be taken specifically into account. At present it is the practice both in analog and digital dynamics studies to estimate average temperatures, pressures, spectra, etc., from which the parameters of the model (reactivity coefficients, conduction coefficient, lifetime, beta, etc.) are calculated. (Only the

film coefficient for heat transfer is allowed to vary in the typical hazard study.) In reality all parameters used vary either directly because of temperature change or indirectly because the space independent model dictates an average over the energy spectrum which in turn varies.

The result of such studies, then, will lead to improved knowledge of what is important and to the direction for future effort.

2. Improvement of Dynamic Simulator

The dynamics simulator will be modified by the use of improved mathematical representation of physical models (e.g. coolant flow) and incorporation of improved physical models (e.g. fuel melting). Preliminary studies have indicated that it is impossible to obtain additional information regarding temperature profiles in fuel elements without the necessity of dividing the fuel into many radial nodes (which is costly). This will be done to improve the accuracy of the calculation at little added cost. Experience with both analog and digital representation of the transport of coolant indicates that instabilities often arise. Thus a relation between the number of axial nodes and coolant velocity will be determined to assure stability of the numerical model. The dynamic simulator will be modified to include models for fuel melting, burnout, expansion and the external heat transfer loop. These models will be checked-out against reactors for which dynamic data is available to insure accuracy.

Four general types of improvement are possible:

a. Incorporation of Parameter Variation with Time

Due to change in temperature, spectrum, etc., this type of improvement was discussed in the previous section.

b. Improved Numerical Representation of Physical Models

Certain of the commonly used numerical approximations are subject to errors. For example, Mason and Winson³⁸ discuss several numerical heat and mass transfer models and indicate their inherent errors.

c. Improved Programming Features

The utility of a computer code depends to a large extent on secondary features as well as on the validity of the mathematical and

physical models. Improving the integration equations in order to reduce machine running time will permit use of more sophisticated mathematical and physical models in comparable running time. Incorporation of a routine to generate the input data (material densities, specific heat, conductivity, etc.), increased flexibility and improved graphical display of output data all contribute to the usefulness of such a code.

d. Development and Programming of Physical Models

Extension of the simulator into the realm of coolant boiling and fuel melting is a necessity for a complete understanding of the consequences of postulated accidents to the reactor per se and to the public. The work with AIREK 3B indicates that physical phenomena as coolant buoyancy effects, time dependency of fuel melting and volumetric expansion are significant in determining the dynamic behavior of reactors during such catastrophies. Models for these effects must be reviewed and incorporated into the dynamic simulator. Calculations using the models can then be compared with experiment and discarded or improved.

Incorporation of the external loop equation is needed for representation of coolant delay times and for flow decay due to pipe rupture, loss of power, etc.

Incorporation of these improvements into the dynamic simulator will provide a tool capable of accurate prediction of reactor transient behavior which can be used as the basis of design, control, and hazards studies.

C. REACTOR SITING

The determination of reactor siting is the logical extension of the results of reactor dynamics studies with respect to the consequences of reactor accidents and ultimately public safety. Thus many of the remarks regarding the tool to study dynamic simulation of reactors and the use of such a tool are applicable to the tool for studying reactor siting, AISITE. Here we are concerned with the models which predict the dose as a result of a reactor accident in which fission products are released from the primary container and possibly into the atmosphere.

1. Parameter Studies (Siting Handbook)

As in the dynamical parameter studies, one of the basic questions we are asking is how sensitive are the results (exclusion distance, low population distance, etc.) to the uncertainties of the parameters. The results of these studies will indicate the degree of uncertainty of present studies and indicate the direction of future work. In addition to providing direction, studies of this nature provide insight into the effect that design decisions have on safety. For example, if doubling the cost of the building to reduce the leak rate results in marginal increase of exclusion and low population distance and hence in public safety, it is hardly worthwhile.

Thus, it is important for the reactor designer to understand how design changes will affect safety. Consequently, a reactor siting handbook will be prepared to which a designer can refer for such information. This document can contain the results of parameter studies for typical SGR's, BWR's, PWR's, etc., showing the effect of variation of building leak rate, filter efficiency, weather conditions, power level, stack height, etc., on the critical distance. Both graphical and tabular results will be included.

2. Improvement of Models

Comparison of results obtained using AISITE with experiment will indicate which models need improvement. Also sensitivity of the results to certain parameters of the model point out areas of potential difficulty in that improved models may be necessary.

D. MULTIREGION-MULTIGROUP KINETICS

No adequate means is at present available for analyzing the dynamic behavior of coupled (fast-thermal) cores, multiregion fast and multiregion thermal reactors. The usual space independent equations represent a reactor in which the flux shape and spectrum are sensibly constant during the transient and so are not suitable for many systems. Several documents have appeared recently in the literature^{39,40} indicating the growing interest and need for such a tool. The reported codes both use Euler's method (time discretion) for integration in time and are slow.

Several methods have been suggested^{41,42} in the past for solving the one-dimensional few or multigroup space-time diffusion equation using far better numerical methods.

The intent of this work is to provide a tool capable of solving problems in reactor dynamics to which the space independent kinetics formulation does not apply. Several of these problems have been mentioned earlier; examples of others are temperature discontinuities, spectral shift, reflected neutrons (effective seventh delay group), xenon oscillations, and others.

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