FAULT ANALYSIS OF NUCLEAR REACTORS

By

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ABSTRACT

The operation of nuclear power stations results in the accumulation of large quantities of potentially dangerous fission products within the fuel. Absolute certainty of fission product retention within the reactor system cannot be accomplished. In the past the safety of power reactors has been assessed in a qualitative fashion on the basis of the "maximum credible accident" approach. Experience has shown that there was no straightforward way of discriminating between "credible" and "incredible" faults, so the scientific way of dealing with this situation is to investigate the whole spectrum of faults in a quantitative manner. This has become the premise behind the statistical fault analysis method which therefore assesses the total environmental hazard due to all definable initiating events to which a relevant nuclear power plant might be subject during its life. This Thesis first describes the conventional fault analysis methods together with its shortcomings and the methods of calculation used for magnox and advanced gas cooled reactors (AGR). A brief review of the alternatives is followed by a detailed description of the Author's APC compound model which incorporates existing analytical models for fault consequence evaluation and adds to them quantitative assessments of reactor design, operating and maintenance procedures, and engineered safeguards. This model is directly applicable to AGR's and has a possibility of straightforward extention to any gas cooled graphite moderated reactor.

In order to illustrate an application of the model a statistical fault analysis study was carried out which comprised three characteristic AGR faults: failure of supply to all circulators, medium size penetration failure and reactor start-up fault. Summated outcomes of the study in terms of the probability distributions of curie release of iodine - 131 are compared with Cave's criteria. Finally, another form of application of the compound model, in terms of sensitivity analysis is illustrated.

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(1) INTRODUCTION

(1) Nuclear power is showing advantages over other forms of energy as a cheaper source of electricity. In addition, air pollution problems are virtually non-existent; thus in the sphere of public health nuclear power offers a substantial improvement by reduction of the amount of noxious gases discharged into the atmosphere during normal operation of fossil fuel power stations. However, the operation of nuclear power reactors results in the accumulation of large quantities of potentially dangerous fission products within the fuel.

(2) The consideration of safety has played a vital part in the design of nuclear power plants, mainly because of the large potential radiological hazard to the public or to the operating staff as a result of a major fault. Less important, but still significant, is the protection of the large capital investment involved in the plant structure, which might suffer damage following a major fault, and which might be inaccessible for repair due to local internal radioactivity levels even if there were no release of radioactivity. Absolute certainty of fission product retention within the reactor system cannot be accomplished; the essential problem is to arrive at an appropriate balance between safety and economy so that on the one hand the potential advantages of nuclear power are not lost by the increase in cost associated with an oversafe design, but on the other hand radiological risk is kept acceptably low.

(3) In the past the safety of power reactors has been assessed in a qualitative fashion; in the sense that it has not been customary to express in mathematical terms all the possible consequences of every conceivable fault condition. Instead, the designer has provided inherent or engineered safeguards which were intended either to make the occurrence of a potentially dangerous fault condition extremely

unlikely or, if it should nevertheless occur, to bring the reactor to a safe condition. Following established engineering principles, the designer has provided the most comprehensive safeguards against accidents which could cause serious harm and less elaborate precautions against accidents which could, at the worst, cause some temporary inconvenience. As a consequence, the extent to which additional safety features should be incoprorated in a nuclear power plant design has been a matter of subjective engineering judgement, the designer and safety assessor reaching a compromise at some point where the risk associated with each potential fault was thought to be reduced to an insignificant level. For convenience, faults reduced to this level have been regarded as "incredible". Other potential fault conditions could not reasonably be regarded as incredible because, for example, they involved a human operator or an outside agency. In each group of such unlikely but possible fault conditions the most potentially hazardous one from the point of view of harm to the public, was taken to be the maximum credible accident (MCA). Measures were taken to reduce the consequences of this MCA to a level which would be considered reasonable by the licensing authorities; the consequences of any lesser accident of this type would then also be acceptable.

(4) Experience has shown that there was no straightforward way of discriminating between "credible" and "incredible" faults, the "incredible" ones were sometimes made up of a combination of individually not very unlikely events and the "credible" would actually be exceedingly improbable. The scientific way of dealing with this situation is to investigate the whole spectrum of faults in a quantitative manner. Consequently, the basic premise behind the statistical fault analysis method (SFAM) is non-discrimination between the various faults to which a particular station might be subject in its expected life.

This results in the full safety evaluation comprising a spectrum of events with associated consequences and probabilities of occurrence. A general hazard evaluation method must comprise models describing both the statistical aspects of the fault occurrence and the consequences in terms of fission product release. In addition, owing to the cumulative nature of radiological risk assessment a model to perform summation of independent risks must be included. These requirements are satisfied by a compound model which appears to be a satisfactory executive tool for the implementation of the statistical fault analysis concept.

(5) The SFAM with a compound model as its executive tool is generally applicable to any power or research reactor type. However, numerous alternatives would have to be incorporated in a specification of the universal compound model to reflect differences in fault consequences. In addition, the nuclear power plant components vary conspicuously from one reactor type to another. If, however, various reactor types are grouped into: light water, gas cooled graphite moderated, heavy water and sodium cooled reactors; then specification of semi-universal compound models becomes feasible. A compound model developed by the Author is directly applicable to advanced gas cooled reactors and has a possibility of straightforward extention to any gas cooled graphite moderated reactor.

(6) Early gas-cooled graphite moderated reactors had one natural uranium fuel pin per fuel element, encased in a finned can of a relatively low melting point megnesium alloy called magnox, and were known as magnox reactors. The advanced gas cooled reactor or AGR differs from the magnox reactors in that uramium dioxide fuel in stainless steel cans is arranged in a cluster of fuel pins in each element, which includes a graphite sleeve. The coolant flows in the moderator penetrations and on the outside of the sleeve in the opposite direction to that in the main fuel channel; this is called a re-entrant

coolant flow arrangement. The AGR operates at higher fuel temperatures and power ratings than the magnox reactors, and the re-entrant coolant flow arrangement permits relatively lower graphite moderator temperatures.

(7) Several studies have been published on the subject of various safety aspects of the AGR, e.g. references 1-5. These studies discuss the safety philosophy of the AGR and present various qualitative analyses of its behaviour. A paper by Cave and Holmes attempts a simple statistical fault analysis for the AGR and draws attention to the problem of defining quantitative criteria. Farmer,[6] Adams,[7] and Laurence[8] have proposed quantitative safety standards to which siting considerations can be related; of these the first two are based on releases to the atmosphere of iodine-131; they relate the accepted probability of a release to its magnitude; that of Laurence stipulates limits for the probabilities of particular events. In addition, Farmer has illustrated the concept of the probability associated risk by a fault study based upon a statistical fault analysis method discussed in the GRASP/R1 report.[9] Stanners[10] has also given an account of a statistical approach to reactor fault analysis and has discussed methods of comparing the expected release with some permissible target release. The American paper by Mulvihill[11] gives a full account of a probabilistic methodology directly applicable to light water reactors.

(8) At various stages of development of this Thesis work, the Author had been publishing several papers[12-15] on the subject of statistical fault analysis applied to AGR's. This final Thesis write-up incorporates all the major points discussed in the quoted papers for the sake of completeness and integrity of the text. It first outlines the philosophy and the methods of calculation used for the AGR conventional fault analysis studies and shows that these methods follow consistently the techniques developes for the fault transient studies of magnox reactors. It then describes the form of a SFAM in terms of the APC compound model,

comprising fault consequence models, which include reactor core transient evaluation, can failure mechanism and radioactivity release, fission product retention and radiological risk assessment submodels, together with a fault probability model to evaluate the probabilities of initiating events and an overall summation model. A brief comparison between the described method of statistical fault analysis and some other methods is followed by a resume of a possible general classification of AGR faults. Emphasis is placed upon illustration of the APC compound model application. This took the form of a fault study embodying an analysis of three characteristic potential AGR accidents: failure of supply to all circulators, medium size penetration failure and reactor start-up fault. Summated outcomes of the study in terms of the probability distributions of curie release of iodine-131 were compared with Cave's criteria. Finally, another form of the compound model application in terms of the sensitivity analysis was illustrated.

(II). CONVENTIONAL FAULT ANALYSIS METHOD

(II.1) Philosophy

(9) The existing method of assessing reactor hazards as exemplified by the analyses in safety reports required by the INI (Inspectorate for Nuclear Installations, the licensing authority in the UK) or the AEC (Atomic Energy Commission in the US) for construction and operating permits, evaluates only credible faults in terms of fission product release to the environment. Probability of the fault occurrence is not a factor except in establishing credibility. Credibility is not defined on a numerical basis but rather on a basis of judgement; the designer and safety assessor reaching a compromise at some point where the risk is thought to be reduced to an insignificant level. For convenience faults reduced to this level have been regarded as 'incredible' and some particular fault which could lead to a large release of fission products if uncontrolled is contemplated sufficiently likely to warrant detailed consideration of its effects is regarded as the 'maximum credible accident' (MCA). The transient behaviour of the reactor core following this fault is then analysed in detail to enable decisions to be made as to what safety equipment is necessary to limit the release of fission products to the atmosphere to some amount determined by the characteristics of the site. The extent to which other credible faults are analysed is not necessarily the same; thus there is no certainty that a common standard of safety has been reached in all areas of design. Moreover, until a basic design has been repeated several times with little variation it is not obvious what particular fault should be regarded as the MCA. For steel pressure vessel magnox reactors designed in the UK and France; for light water reactors

designed in the US, Germany, Sweden and other countries; and for heavy water reactors in Canada, Sweden, the UK and France, the convention has become firmly established that the MCA is complete rupture of the largest external duct or pipe of the primary coolant circuit. For concrete pressure vessel magnox reactors in the UK and France and advanced gas cooled reactors in the UK, the MCA is either a small size penetration failure or a lifted safety valve.

(10) The philosophy behind this credible/incredible approach is probably best described in the section 'accident analysis' of Reference 16 and the section 'fault studies' of Reference 17. In both documents guidance on the specification of faults is given, including a very generalized list of typical cases that ought to be analysed in a typical light water or gas cooled reactor design, respectively. Once the list of credible faults has been defined, for a design under consideration, reactor kinetic computer codes are used to evaluate the reactor core temperature transients in selected fuel channels following a fault onset. This information serves as a basis for the fuel and cladding failure assessment. In the case of a cladding failure some of the fission products are released to the coolant circuit and then attenuation mechanisms operate to reduce the activity to be released to the atmosphere. In view of the fact that very few credible faults lead to fission product release, the techniques for evaluating cladding failure and fission product release are on a very elementary level, especially in comparison with very comprehensive reactor kinetic computer codes.

(II.2) Methods of Calculation used in Fault Transient Studies for Magnox Reactors

(11) The methods of calculation must provide adequate estimates of various core material temperatures, power output level, neutron flux level and distribution, control rod movements, core reactivity changes, controller response, controller constants, xenon poisoning, reactor trip setting etc. The spatial distribution of neutron flux and reactor core temperatures is strictly three-dimensional, but it is possible to reduce the number of physical dimensions by assuming some symmetry and/or by averaging properties in one or more directions. Estimates of varying degrees of accuracy are required at different stages of a reactor's history and hence it is possible to conceive of a whole range of valuable kinetic models varying from the simplest point calculations to a complex multidimensional. Therefore, the methods of calculation can be classified in two main ways; first, by the number of physical dimensions included in the mathematical model of the reactor, and secondly by the mathematical technique adopted for the solution of a set of non-linear partial differential equations. Further variations include different ways of representing reactor neutron kinetics, heat transfer, reactivity controller, and the various auxiliary equations.

(12) Fault transient studies for magnox reactors were directed towards ensuring that the hottest can in any fuel channel of the reactor did not approach melting point too closely in any credible fault condition, since this could lead to substantial radioactivity release into the coolant and also to uncertainty about subsequent heat transfer processes in that channel even if the reactor were shutdown. Early studies [18]

considered the time behaviour of core average fuel, can and moderator temperatures, and assessed the behaviour of maximum can temperatures by using the ratio of maximum to average value that occurred in the steady state. It was soon found, however, that during major fault transients, and in particular during reactivity faults associated with inadvertent withdrawal of a large group of control rods, the ratio did not stay constant and even the position of the maximum temperature might vary axially during a transient. This was most evident in the fuel region of the channel mainly because movement of a large control rod group could cause substantial changes in the axial neutron flux shape, and so of the heat generation with consequent large changes in axial temperature profiles. Thus, the simplest model, so called 'point' model, proved to be inadequate for fault studies but still remained useful for preliminary surveys and initial optimizations of the control system.

(13). Space-and-time variant transient models were developed for magnox reactors to study the behaviour of complete reactor core channels. Three dimensional reactor kinetic programs such as STAB[19] or SKIP[20] use elaborate finite difference techniques for the solution of the spatially dependent differential equations. Apart from some analyses related to asymmetric reactivity faults not very many fault transient studies have been performed with them, mainly due to lengthy preparations of data and astronomical costs involved. One of the two possibilities for two dimensional models, the (R, θ) model, is not suitable for fault transients as the variations in axial direction (direction of coolant flow and control rod motion) are averaged; the necessity of having an (R, z) model is diminished by the fact that the radial flux shape is flattened, resulting in a large number of channels behaving similarly. Consequently axial programs with one or more radial channels have become a standard tool for assessing fault transient behaviour of the reactor core. Typically[21], a core average channel is represented, which is arranged to provide correct feedback of temperature, control rod motion, and xenon reactivities to the neutron kinetics simulation, with simultaneous calculation of the heat transfer processes and temperatures in a channel having the hottest can temperatures in the core. The kinetics of fission neutron heating and fission product decay heating is represented, and the kinetics of release of heat from the graphite in certain conditions of the Wigner energy, which is associated with graphite crystal lattice deformation and is stored during neutron irradiation. For conditions after a depressurisation of the reactor pressure vessel, when air ingress into the core may be conceivable, heat production from oxidation of the graphite is also represented. Heat transfer by convection, conduction, or radiation, is represented conventionally. The CEGB KINAX program[22] computes the behaviour of an additional fuel channel; this channel might for instance, simulate the behaviour of a typical reactor trip channel.

(14) The precision of available data places an upper limit on the accuracy of a prediction of a system's behaviour. Having accepted this fact, an appreciation of the value of reactor kinetic codes require an understanding of the scope of the required data. One way of grouping the data would be as follows:

(a) heat transfer - thermal capacities and conductivities of the constituents of a lattice cell, convective and radiative heat transfer coefficients;

(b) reactor geometry and performance - core height, diameters of the lattice constituents, fuel length, can length, steady state channel ratings, coolant mass flows, coolant inlet and outlet temperatures, etc;

(c) heat generation - neutron flux shapes, heat production shapes, distribution of neutron and fission product power between fuel and moderator regions, fission product power fractions, graphite oxidation rates;

(d) neutron kinetics and diffusion - delayed neutron fractions and decay constants, mean neutron life time, temperature coefficients of reactivity, xenon poisoning reactivity coefficients, iodine and xenon fission yields, cross sections, migration areas, K_{∞}, etc;

(e) control rod motion - control rod reactivity worths, vertical rod positions, rod speeds, trip reactivity insertion characteristic;

(f) controller - thermocouple response and thermocouple time constants, servo loop characteristics, control rod mechanism time constant, control signals, etc;

(g) other - release of Wigner Energy, trip margins, etc.

(II.3) <u>Methods of Calculation used for AGR Fault Transient Studies</u>

(15.) Similar space-and-time variant models to those used for magnox reactors are used to study the behaviour in fault conditions of AGRs, although the AGR models are more complicated owing to the multiple fuel pin geometry, the re-entrant coolant flow arrangement and the fact that limitations imposed by the first generation computers have been removed. The AGR version of STAB is known as REST[23] while an AGR SKIP version is being developed by the CEGB. Much the same as for magnox reactors, the axial programs have become a standard tool for assessing the transient behaviour of reactor core channels. Several AGR axial reactor kinetic codes are in use in Britain. In view of the fact that the AGR fuel cycle equilibrium core contains fuel channels covering a wide spectrum of irradiations and consequently ratings, the three channel programs, e.g. KINAGRAX[24], or REAXITRAN[25] have gradually become a standard tool for assessing fault transient behaviour. Typically, the channels can represent: new fuel or hot channel, core average channel which provides suitably averaged thermal reactivity feedbacks to the core neutron kinetics, and reject fuel or trip channel. This fact does not reduce the value of two channel programs like DAGGER[26] for simulations of the hot channel transient behaviour. For a specialized purpose, single channel programs are satisfactory; for example FLIRT[27] has been particularly written with a view to investigating multigroup neutron energy effects (up to five neutron energy groups can be represented) while LEAK[28] was written with a view to analysing transients following a depressurisation fault (with the underlying assumption that the reactor would be tripped quickly following a fault onset).

(16) The complex AGR reactor core arrangement necessitates a very elaborate heat transfer representation. Various assumptions have been used in order to simplify the heat transfer model representation; these assumptions vary from one program to another. Typically, fuel cluster

arrangement is simulated by considering one pin in each of the concentric rings of pins as being representative of all the pins in that particular ring. Full radiative heat transfer treatment is incorporated. The equations include terms representing ring-to-ring and ring-to-sleeve radiation for the fuel element. The radiation from inner sleeve to outer sleeve and from thence to bulk moderator is also included. The inner and outer sleeves are treated separately, receiving and losing heat by convection and radiation; heat is transmitted across the inter-sleeve gap by conduction and radiation. Major characteristics of the re-entrant coolant flow are also represented. Neutron kinetic and diffusion, fission product heating and xenon poisoning equations are practically identical to those used in magnox reactor kinetic codes. The same applies to phenomena characterizing the behaviour of graphite, e.g. Wigner energy release or graphite oxidation. Either analogue or digital controller actions can be simulated. Implicit finite difference methods are used for both space and time integration. Interdependence of neutron power and temperatures is treated by iterating between the heat transfer and neutron kinetic equations updating appropriate differences in between the iterations. A more detailed description of the Author's program, being a good example of an AGR kinetic program, is offered in the ensuing chapter.

(III) STATISTICAL FAULT ANALYSIS METHOD

(III.1) Meeting of Safety Evaluation Requirements

(17) Cave [29] has considered the problems of safety evaluation from three different aspects:-

(a) definition of objectives, i.e. the provision of sufficient guidance to designers and operators as to the hazards to the public, station personnel and plant which are acceptable in fault conditions;

(b) preliminary assessment and design optimisation, i.e. the assessment of the cost of meeting the safety requirements in a particular reactor system or in a variation of an existing system so that effort is not wasted in developing a design which is unlikely to be economic if the safety requirements are met;

(c) final assessment, i.e. the detailed assessment of the safety of the reactor which is being built to ensure that the safety requirements are being met, taking into account the design, siting, fabrication and the operating intentions.

These stated aspects, worded identically or similarly, represent the essence of safety evaluation. It is extremely difficult to argue that the present qualitative approach has provided any satisfactory criterion for safety in fault conditions. In general, Licensing Authorities are reluctant to commit themselves to specific quantitative criteria for safety which relate the risk of a particular fault to its consequences.

This situation leaves designers to interpret vaguely worded qualitative criteria as best they can; experience shows that the interpretations can vary widely, particularly in estimating the reliability needed in plant and structures whose failure could lead to large releases of fission products.

(18) The statistical fault analysis method via its executive tool in terms of the compound model overcomes the technical deficiencies experienced in the qualitative approach and meets the requirements of safety evaluation by:-

(a) incorporating reliability analysis as a means of assessing the performance of different engineering structures and safeguards;

(b) enabling optimisation studies to be carried out aimed at achieving balance between reactor safety and economy, as it can determine the effect on the hazard of variation in reactor design including the engineering safeguards;

(c) enabling sensitivity analysis or parameter survey studies to be performed, both of which are extremely important since many inputs are known only to an accuracy of one or two orders of magnitude;

(d) permitting uncertainties in parameter input data values by accepting probability distributions rather than discrete values only.

(III.2) Block Diagram of APC Compound Model

(19) To the best of the Author's knowledge two complete compound models have been described in the available literature: an American model directly applicable to light water cooled and moderated reactors [11] and a UK model by the Author, named APC model, which is directly applicable to advanced gas cooled reactors and has a possibility of straightforward extension to any gas cooled graphite moderated reactor. A comparison between the two models is offered following a detailed description of the APC model.

(20) The scheme of the APC compound model is shown in block diagram form in Fig.1. The constituents of the model are the following submodels: reactor transient behaviour, can failure mechanism and activity release, fission product retention, fault probability, overall summation and radiological risk assessment. The reactor transient program evaluates the core temperatures and in particular the fuel and can temperature histories in selected fuel channels following a fault onset. This information serves as a basis for the fuel and cladding failure assessment. In case of a cladding failure the fission products are released to the coolant circuit, and then retention mechanisms operate to reduce the activity which is likely to be released to the atmosphere. Statistical aspects are added to the fault consequence submodels by allowing randomness to some of the input parameters, i.e. by representing them in the form of probability distributions. The part of the model dealing with statistical relationships between the initiation of faults and equipment failures, and malfunction of safety devices, is represented in a fault probability

submodel. Summation of the effects arising from all the individual initiating events is exercised in an overall summation submodel. Curie release and radiological risk are correlated in a radiological risk assessment submodel.

(III.3) <u>Reactor Transients Submodel - REAXITRAN Program</u>

(21) APC use the REAXITRAN program for conventional fault transient evaluation of advanced gas-cooled reactors. REAXITRAN is written in the EMA computer code. It has recently been translated into FORTRAN IV by the Brown Boveri Mannheim. Several documents describe various aspects of the program: APC/R1181[25] is known as a program write-up; APC/R1264[30] offers the program listings and flow diagrams; APC/TN1343[31] gives the recent program modifications; APC/R1381[32] summarises the joint APC/CEGB correlation exercise between REAXITRAN AND KINAGRAX [24] which resulted in a very good agreement between the two programs. The comparison concerned itself with a severe depressurisation fault. For the sake of completeness an abridged description of the program is offered in this chapter.

(22) For any mathematical model intended to simulate an actual physical system, a number of assumptions have to be made. This is, in general, a very delicate task. A qualitative assessment of the numerical importance of different physical phenomena has to be performed, bearing in mind the various uses that will be made of the representation. The channel diagram given in Fig. 2 shows the representation in the REAXITRAN program of typical AGR lattice geometry in simplified cylindrical form. Further to this representation, the REAXITRAN equations are characterised by the following principle assumptions and/or requirements:

(a) It is assumed that the flow of heat in the solids is in the radial direction only.

(b) Allowance is not made for the radial variation of main coolant and sleeve temperatures within the channel.

(c) The thermal capacities associated with all the solids are assumed to vary with temperature only in the steady state. The thermal capacities of the coolant streams are ignored.

(d) The radial heat flow out of the channel is ignored.

(e) The distribution of heat generated between the rings of fuel pins and the moderator regions is assumed to be constant along the channel. In other words, the effect of the reflectors is ignored.

(f) Neither axial variation nor variation between the channels is represented of the flow functions, which describe the variation of the convective heat transfer coefficients with flow.

(g) Reactor kinetic representation is similar to that in MIDAX[21]. Neutron kinetic behaviour is represented by point equations with six delayed neutron groups. The axial distribution of fission product power has the same shape as the axial neutron flux distribution, and the neutron flux shape is recomputed at required time intervals by solving the steady state solution of the diffusion equation. The relevant neutron flux shape is then used to give the point neutron kinetic equations. The neutron flux representation is, therefore, fully space-and-time variant to any required accuracy.

(h) The reactivity controller is of continuous (analogue) type represented by second-order measuring lag, double lead/lag phase advance, controller characteristic with dead band and first order mechanism lag.

(i) Up to six control rod groups (zone, trim/override and four coarse rod groups) are included and may be moved independently and simultaneously. Zone rods can be trimmed either manually or automatically.

(j) Provision is made for the tabular representation of the following disturbances: flow, reactivity, coolant inlet temperature, demanded coolant outlet temperature and coolant bypass temperature. In addition reactivity disturbances can be specified by movement of control rod groups. The flow disturbance history may differ for the re-entrant coolant passages from that in the main coolant channel, thus allowing the simulation of stagnation or reverse flow phenomenon if required. Manually or automatically initiated reactor trip can be simulated either by reactivity addition only or by control rod motion over axial mesh points.

(k) All the initial temperature distributions are specified on input together with some other basic information, and from this information the heat transfer and heat transport terms are computed.

(l) Allowance is made for the effects of xenon poisoning, Wigner energy release and graphite oxidation.

(23) The program is written in a manner comprising fourteen independent routines named as follows: heat generation, neutron kinetics, xenon poisoning, fission product heating, heat transfer, Wigner energy, flow, flux shape, reactivity controller, reactivity balance, disturbance, graphite oxidation, printing and graph plotting. An additional steering chapter is used for all manipulations with the

routines. The REAXITRAN program equations are listed in Appendix I, which assumes that, for the sake of convenience, all the equations can be grouped into: heat transfer, neutronic, controller and reactor trip equations.

(24) The total heat generation is subdivided between the neutron and the fission product power. The assumed distribution of neutron and fission product power between the fuel and the moderator regions is shown in Fig. 3. The fuel stringer comprises up to four rings of clad fuel pins. A full radiative heat transfer treatment is incorporated; the equations include items representing ring-to-ring and ring-to-sleeve radiation for the fuel stringer. The radiation from inner sleeve to outer sleeve and from thence to the main moderator is also fully treated. The main coolant region is bounded by the inner surface of one of two graphite sleeves which are separated by a stagnant gas region. The annulus region between the outer surface of the outer sleeve and the main moderator brick contains coolant entering at the top of the reactor core. The annulus coolant together with the re-entrant passage coolant flows downwards; at the bottom of the reactor core they mix with the bypass coolant, and then the mixture flows upwards along the fuel channel forming the so-called stringer flow. Wigner energy release is represented by a step reduction in graphite specific heat when the local graphite temperature exceeds the initial steady state temperature by more than a margin specified in the input data. The magnitude of the graphite specific heat drop is an axially dependent input quantity, while the temperature margin is axially invariant. The exothermic oxidation of graphite is represented to give rise to the heat generation terms at each point in all fuel channels. In the flow routine, the normalised convective heat transfer flow functions (can to coolant, inner sleeve

to coolant, outer sleeve to annulus coolant, moderator to re-entrant coolant) are computed on the basis of linear interpolation between the points fed on input. Subsequently, the relevant flow dependent heat transfer terms are updated.

(25) Well established point neutron kinetic equations are used as part of the space-and-time variant neutron flux representation. The effective statistically (neutron flux squared) weighted reactor reactivity is assumed to equal the sum of statistically averaged reactivities due to temperature, xenon and control rod movement effects. The source term is equal to zero except during start-up runs. The trip reactivity term is only a reactivity addition and is not associated with particular simulated rod movement. Fission product heating is represented as the sum of four terms. If, however, the sum of the fractions is smaller than unity, then in effect there is a fifth term of zero time constant, each term being related to the normalised neutron power by a first order linear differential equation. The standard xenon and iodine equations are represented with full axial variation and the local reactivity effect is computed at each mesh point. The flux shape routine is based upon that employed in the MIDAX program.[15] The initial values of K_∞ are computed at each mesh point such that the axial diffusion equation represents the steady state flux at each mesh point in the average channel including the end points. The axial flux shape remains unchanged for a specified number of time steps; meanwhile the reactivity changes accumulate, resulting in a modified value of K_∞. The eigen-value is given a value such that the diffusion equation represents again a steady state flux shape. The average channel being representative of the reactor as a whole, is used to compute the local, axially variant, reactivity changes. To represent the movement of control rod tips over the axial mesh points, a cubic representation of local reactivity effect with distances has been assumed.

(26) Gas outlet thermocouple response in the analogue controller is that of a second order system, thus enabling the representation of effects such as neutron scatter plug, thermal inertia, and radiation from the surrounding materials. The measured temperature is compared with the demanded one and the error signal fed to a double lead/lag phase advance analogue controller with deadband. The mechanism response is then represented by a first order lag. Trimming action is automatically initiated when the zone rods move outside the specified range. The trimming characteristic allows for a deadband. Allowance for a trip margin and servo reset trip setting is made in the simulation of a reactor trip initiated by a measured gas outlet temperature signal.

(27) Implicit finite difference methods have been used both for space and time integration. The Crank-Nicolson formula is used to replace differential equations by difference equations:

$$T\,(n + \Delta n) = T\,(n) + \frac{\Delta n}{2}\,[T'(n) + T'(n + \Delta n)]$$

involving first derivatives only and with no second difference correction. Since $T'(n + \Delta n)$ is not known till $T\,(n + \Delta n)$ is, the solution of the difference equations is obtained by an iterative scheme. In all, there are four iterative loops in the program. One inner loop solves the heat transfer equations, the second inner loop solves the neutron flux shape diffusion equation, the third inner loop solves the neutron kinetic equations and the outer loop encompasses the whole system of equations. The principle that the direction of integration should coincide with the direction of flow to avoid numerical instabilities has been adopted in the program. For solution the heat transfer equations are divided between those relating to the 'hot' region and those relating to the 'cold'. The 'hot' region embraces the fuel stringer and extends as far as the outer sleeve. The 'cold' region comprises the re-entrant coolant passages and the bulk moderator. The process of solving 'cold' region equations followed by 'hot' region equations is interrupted when successive estimates of the coolant outlet temperature or peak can temperature in a selected control channel satisfy a convergence criterion of the form:

$$\frac{T^{(i)} - T^{(i-1)}}{T^{(i)}} < \epsilon$$

where index i indicates the iteration count.

(III.4) Cladding Failure Mechanisms

(28) In AGRs pressure within the cans is always below the coolant system pressure in normal operation. Thus, if coolant pressure is retained, there is no tendency for the cans to fail. A few cans might fail owing to reduced mechanical properties of the canning material, but in general, in fault conditions not resulting in a significant drop in coolant pressure, failure of cans at temperatures below 900°C is extremely unlikely. At higher temperatures, intrusion into small interpellet gaps becomes progressively more likely up to the melting of 20/25 Nb stainless steel cans at about 1437°C. In the present AGR designs, can melting is a clear-cut safety limit for faults occurring in a pressurised reactor. There is, however, also an economic metallurgical limit intended to prevent cans from undergoing significant material property changes and thereby prejudicing their attaining target irradiation and casting doubts on their behaviour in a subsequent possible depressurisation accident. This material damage is both time and temperature dependent, so the cladding failure model assesses the cumulative effects of different periods of time at different temperatures from a notional graph such as the one displayed in Fig. 4 for illustrative purposes.

(29) In the event of a fault leading to loss of coolant pressure the primary can failure mechanism is stress-rupture. The proportion of cans expected to fail is assessed from a knowledge of:

(a) the hoop stresses in the can (tangential cladding stress is a function only of effective can thickness and inside diameter and of the pressure differential across the can wall);

(b) the cumulative damage characteristics of the can as a function of stress, temperature and irradiation;

(c) the temperature/time histories of the cans following a fault onset.

The data on bursting of AGR cans under fault conditions vary from one experiment to another. The information is invariably presented in terms of a series of curves of hoop stress against mean time to failure at different can temperatures; e.g. Fig. 5 reproduced from Reference 33. The experimental data can be fitted to an analytical expression. For any given value of stress and temperature there is a scatter about a mean value of time to failure. Analysis of these results on a statistical basis then permits an estimate to be made of the number of cans expected to fail as a function of the mean failure time. The scatter of individual values about the means expressed in terms of standard deviation is the inherent variability in the stress rupture properties of the type of pins tested. In a transient it is necessary to consider the failure behaviour of the cans under changing conditions of hoop stress and temperature. The currently favoured method[34] suggests evaluation of the fractional mean time to failure over short time intervals during which hoop stress and can temperature are assumed constant. Summation of the fractional mean times to failure over all intervals is then carried out, leading to the cumulative fraction, and the expected proportion of cans failing is read from the can failure probability curve.

(30) Rhodes, et al[35] have suggested that, for temperatures up to 950°C and hoop stresses from 5000 - 10000 psi, the experimental data on bursting of AGR cans can be fitted to an equation of the form:

$$\log_{10} t_m = A + B \log_{10} (HS) + C/T$$

where t_m - mean time to rupture (min)

HS - hoop stress (psi)

T - can temperature (°K)

A,B,C - constants

In addition, the scatter of individual values of $\log_{10}(t_m)$ about the means was calculated to have a standard deviation of 0.262. This estimate resulted in the probability of failure curve reproduced in Figure 6. The above mentioned information served as a basis for the author's APC 1380 EMA program, which computes : mean times to failure, used fractions of rupture life, cumulative fraction of rupture life, and probabilities of failure.

(III.5) Fission Product Inventories and Pressures

(31) Nearly 200 different isotopes are produced by the fissions of uranium and plutonium. However, only certain isotopes are important as health hazards and a limited number of isotopes contribute to the gas pressure generated by the fission products. The fission products are in general coupled through beta decay as follows: $A_{N1} \rightarrow A_{N2} \rightarrow A_{N3} \rightarrow A_{N4} \rightarrow A_{stable}$, where all the isotopes in the chain have the same mass number but different atomic numbers. The rate of change of the various atoms of the decay

chain is represented by the set of coupled linear differential equations of first order given in Appendix II. The solution of these differential equations is rather cumbersome.[11] For approximate activity assessment calculations the formulae suggested in Reference 36 and reproduced in the Appendix II have been used.

(32) The most significant fission products from the point of view of district hazard are those which are volatile, that is the rare gases, the halogens, caesiums, telluriums, rutheniums, strontiums, bariums and ceriums. All these will contribute to the external whole body dose from the cloud which, if large, may result in acute radiation sickness and may induce leukaemia. Persons in the path of the active cloud as it drifts downwind may inhale particles which have become active as the result of the condensation of volatile fission products, as well as volatile compounds such as methyl iodide. Such inhaled fission products will make an important contribution to the whole body dose and some will also irradiate specific organs, for example there will be an important dose to the thyroid gland if radioiodines are inhaled. Many workers [1,6,7] have used criteria based on iodine-131 which would be responsible for 70% of the dose to the thyroid from iodines and telluriums at reactor shutdown and 80% after 12 hours.[37] However, it has been pointed out[38] that some of the data relating thyroid dose to the incidence of thyroid carcinoma are uncertain. Also, since thyroids are likely to be ablated by doses of some tens of thousands rems, the justification for a linear relationship between release of iodine-131 and the incidence of thyroid carcinoma is doubtful, at least in the higher dose range. For the fully urban sites, releases of other isotopes will probably have to be taken into account.

(33) Of the various fission products some remain in the UO_2 lattice as metals, some form oxides and others form solid solutions in the UO_2. The rare gases, krypton and xenon, and the halogens which are in the vapour phase at the prevailing temperatures, will escape to some extent to the free space provided within the can. The mechanisms of the movement of these isotopes through the UO_2 crystal lattice are extremely complex and are not yet fully understood. In Reference 2 the following mechanisms have been reviewed:

(a) recoil from the free surfaces of the fuel of primary fission products;

(b) momentary recoil-induced evaporation of UO_2, which allows fission products to reach a free surface;

(c) diffusion of atoms or extremely small gas bubbles through the grains to boundaries;

(d) sweep processes by which fission products are conveyed by moving grain boundaries and dislocations;

(e) release of gas bubbles on grain boundaries by intergranular cracking and linking of bubbles on boundaries to form microcrack paths to free surfaces;

(f) at temperatures of from 1200 to 1400°C sintering of cracks formed by contraction due to fuel cycling, may begin.

Mathematical solutions of Fick's law with appropriate boundary conditions represent the rate of diffusion of fission products from UO_2.[39] The Booth model briefly described in Appendix III[40] is used to evaluate the diffusional release of fission gas. Other mechanisms than diffusion are only partially understood. For mathematical convenience the sweep process can be simply represented (e.g. equation 5 of Appendix III). The total pressure of the fission products in the void volume is the sum of the partial pressures of the nuclides that have escaped from the fuel.

(34) Improved versions of the UKAEA SUPATRICE 1 program[41] are used for estimations of the gas pressure within the cans and the amount of active iodine escaping from the fuel. The only fission gases assumed to be present in significant quantities are the stable isotopes of xenon and krypton and the active isotope krypton-85. The diffusional release of fission gas occurs from every isothermal fuel zone. The release takes place within three temperature ranges to which different activation energies are assigned. The second fission gas release mechanism based on the sweeping of grain boundaries through the fuel occurs in three temperature ranges above a specified minimum temperature. The fuel pin is divided lengthwise into up to 20 zones and radially into a maximum of 50 isothermal annular zones. The rating, fractional fission gas release and can temperature are calculated for each section and the temperature is calculated for the centre and for each annulus of each section; the total fractional fission gas release is then found. The current method of estimating the fraction of the total iodine which is free of the UO_2 lattice is to apply a factor to the calculated fraction of the fission gases.

(II.6) Fission Product Retention Mechanisms

(35) In normal operation there would be no free fission products in the coolant, but if, as a consequence of a fault condition, the coolant became contaminated, containment barriers would be provided by the following:

(a) physico-chemical effects within the can;

(b) surface adsorption processes within the pressure circuit;

(c) concrete pressure vessel;

(d) reactor building with its associated ventilation system;

(e) iodine adsorption plant and arrangements for discharging the coolant through a high stack;

(f) adsorption processes within the vent pipes.

(36) Since iodine is a very reactive element, chemical reactions with other fission products, uranium, oxygen or the can material itself are expected to take place.[42] Consequently, the can retention factor (ratio of the amount of iodine-131 present in the can-fuel interspace to the amount escaping from the can in event of its failure) can be defined. Collins[43] suggests that a factor of 3 could be used for pins which fail in an AGR depressurisation accident without exceeding 1100°C.

(37) It is expected that part of the iodine released into the coolant after failure of the can would be removed by surface adsorption on the moderator, boiler tubes and other structures. Experimental work in the Windscale AGR[44] shows that a considerable retention of iodine takes place on the internal platework of the reactor pressure circuit and especially on the finned tubes of the boiler if the coolant is being recirculated continuously. Iodine-131, released into the Windscale AGR from deliberately or accidentally failed fuel pins,was found principally in the form of methyl iodide.[45] The majority of the results indicate that this iodine has a half-life of 2.5 mins which corresponds to a few passes round the primary circuit. Methyl iodide which was injected separately into the circulating coolant was found to have a similar half-life. The ratio of the steady state iodine-131 released from the fuel to that remaining gas borne was found to be 5000. The quoted half-life corresponds to a deposition velocity of 0.02 cm/sec to the cooler parts of the heat exchangers where iodine was found to have been deposited. It is suggested by R. Collins[44] that the rate of plate-out is independent of mass transfer rate in the gas phase and is dictated entirely by a surface process. On this basis a simple rate equation can be defined (Appendix IV). This formula is fully valid in

a closed circuit. However, in a loss of coolant fault it is conceivable that some activity would escape before plate-out has had time to be fully effective and a lower retention factor may apply, as it cannot be assumed that the contaminated coolant would pass over suitable surfaces before being lost from the system. Recent investigations[46] suggest that the deposition velocity for iodine is dependent on the linear gas velocity and is approximately proportional to (gas velocity)$^{1/3}$ at low flow rates.

(38) In conjunction with its various steel closure components the prestressed concrete pressure vessel of a typical AGR provides an enclosure for the reactor core. Thus, for faults in an intact pressure circuit the pressure vessel functions as a conventional containment. Iodine released into the coolant would circulate within the vessel where a substantial proportion will be removed by surface adsorption, and the remainder may escape slowly with the coolant lost from the vessel by its normal leakage. From commissioning tests with the first Oldbury reactor it appears that a leakage rate of less than 0.5% per day can be expected at full pressure. The leakage rate depends on the pressure and the temperature in the containment and on the nature of leaks.

(39) In fault conditions which begin with a substantial breach of the pressure circuit, complete depressurisation could be expected to take place before any substantial release of fission products from the fuel had occurred. Subsequently, if some fuel pins had ruptured, leakage of coolant from the breach might carry fission products to areas of the reactor building whence they could escape to the open air. It would be expected that deposition of iodine on bare concrete and, to a smaller extent, even on painted walls, floors and ceilings would take

place. This adsorption in terms of the deposition velocity has been discussed.[47,48,49] The analytical treatment applied for analysis of iodine adsorption processes in the pressure circuit is also valid in this case.

(40) In the event of a channel fire or meltout significant quantities of volatile and particulate fission products would be released to the coolant circuit. The gaseous effluent treatment plant is intended to remove particulates and iodines from the reactor coolant during blowdown, but it is possible to clean up the contaminated coolant gas by recirculation and also by driving it through the plant and up the stack. The plant consists of an absolute filter and a bed of activated charcoal. Iodine adsorption beds can be extremely effective in removing all forms of iodine.[50] The effectiveness of the beds is defined in terms of the logarithm of the decontamination factor divided by the stay time. The stay time is the volume of the bed divided by the volumetric flow rate of the gas. Almost any decontamination factor could be achieved theoretically, provided that the cost of increasing the volume of the beds can be accepted and that the beds are designed so that bypassing of the charcoal could be kept to insignificant proportions. A routine programme of testing and maintenance is required to ensure that the original decontamination factor is available. Recent tests of CEGB plant efficiency of reaction for methyl iodide indicate efficiencies varying from 96.7% to 99.98%.[51]

(41) For pressurized reactor faults resulting in lifting of the safety valves, the gas after leaving the vessel passes through the safety valves and particulate filters and up the long vent pipe.

Experiments performed at Windscale[52] show that CO_2 containing iodine, blown through the pipes, gives significant iodine plateout, the retention factor depending on the time spent by the gas in the pipe.

(III.7) Fault Probability Submodel

(42) A nuclear power station, like any complex engineering structure, consists of numerous structural, mechanical, electrical, electromechanical, pneumatic and electronic components. No component is infallible, and there is always a finite chance of the component failing or of the design limits being exceeded. A component or a group of components may fail or malfunction so that a critical function is lost. This leads to a fault which may result in the release of fission products from the fuel. Each fault is uniquely defined by the chain of events, the first item in the chain being the loss of some vital function caused by equipment failure or operator error. The remaining events in the fault chain are associated with the operation of safety devices, which may be defined as all equipment which reduces the plant to a safe state, mainly automatically, if a dangerous condition should arise. Proper operation of a safety device in the majority of cases prevents further development of a fault. The assignment of a probability of failure for critical equipment groups and safety devices requires a reliability assessment. The most commonly used definition of reliability is that the reliability of a system is the probability that it will perform a required function under specified conditions without failure for a specified period of time. As this definition implies, reliability assessment is based on detailed knowledge of the conditions of the system used and its expected behaviour. There are two

basic prerequisites necessary to estimate reliability, relevant and/or realistic statistical models and applicable statistics. If both these prerequisites are met, reliability analysis becomes a powerful tool for assessing nuclear safety.

(43) The Author has written a program, named STATMOD, which at first evaluates the conditional probability of a fault given an initiating event, taking into account a pre-assessed performace of engineered safeguards associated with overcoming the effects of an initiating event. In other words the program evaluates the statistical relationships between:-

(a) a finite number of initiating events;

(b) independent engineered safeguards associated with overcoming the effects of initiating events;

(c) a finite number of faults.

The basic mathematical background used for formulating the model has been reviewed elsewhere.[12] It contains elementary probability theory information which can be found in References 53-55, for instance. To derive the relationships presented in Appendix V the following assumptions had to be made:

(a) Failures are random; this implies that the Poisson distribution law with a constant average failure rate is valid. Underlying this assumption of randomness is that all components are designed and used in so-called 'useful life' phase of the bathtub process.[56]

(b) Failures are independent unless otherwise stated; this implies that failures cannot propagate, i.e. a failure of a component affects only that particular component and nothing else.

(c) There are no compensating failures; two failures cannot cancel each other.

(d) A fault will occur if one engineered safeguards complex associated with overcoming the effects of an equipment failure or an operator error, fails to operate correctly.

(e) Repair of devices during the station life is perfect (whenever a device is replaced, inspected or tested it is restored to perfect working condition). This implies that the station life can be divided into a number of equal time intervals during which the working equipment is on trial, resulting in the case of repeated independent trials with two possible outcomes for each trial.

(III.8) Overall Summation Submodel

(44) Fault consequence submodels (blocks 1-3 in Figure 1) evaluate the value for curie release C_k given initiating event E_i, while block 4 estimates probabilities of faults F_j occurring given E_i. When combining the outputs of blocks 3 and 4 the STATMOD treatment (Appendix VI) comprises four conceivable cases:

(a) all input parameters in the fault consequence submodels are discrete and fault sequence diagram analysis results in a single release;

(b) all input parameters in blocks 1-3 are discrete and fault sequence diagram analysis results in a number of releases;

(c) some input parameters in the fault consequence submodels are random;

(d) some input parameters in blocks 1-3 are random and fault sequence diagram analysis results in a number of releases.

In the first case the probability that the curie release is C_k is necessarily 0 or 1. In practice, some fault consequence input data parameters may be random in their nature and thus are more adequately represented by a probability distribution than by a discrete value. From the computational effort point of view the number of random input data parameters should be kept to a minimum. In other words, variance of other parameters will have to be ignored. If α input data parameters are allowed to be random and each is represented by β discrete values, to determine the probabilities P (C_k/F_j) it is necessary to exercise fault consequence submodels, α^β times. Each time a different value of curie release could be obtained. The resulting curie releases are grouped according to the nearest curie release decade. The number of releases in each of the decades divided by total number of runs determines the probabilities of having curie release C_k given F_j. If the fault sequence diagram analysis indicates that a fault may lead to a number of releases, then weighting factors should be associated with various releases. These weighting factors are in fact probabilities attributed to different sequences if the fault should occur, and as such their sum should not exceed unity. In some cases additional reliability analysis calculations may have to be performed before these factors are finally specified.

(45) The ultimate aim of statistical fault analysis is to assess the total environmental hazard due to all definable initiating events to which a nuclear power plant might be subject in its life by summating the effects arising from all the individual initiating events. The cumulative nature of the risk assessment is best represented by a probability distribution. STATMOD utilizes the properties of generating functions which are a special case of characteristic functions[12] to sum the effects arising from all the individual initiating events, resulting in an overall probability distribution of the curie release (appendix VI). S is a dummy variable used to form generating functions. The coefficients and exponents of S only are manipulated in the program. The generating functions are represented in the power series form (see equation 6); index K, denoting the number of curie decades, is specified on the program input. Because of the equal spacing between C_k and C_{k-1} on the logarithmic scale ($\log_{10} C_k - \log_{10} C_{k-1} = \log_{10} C_{k-1} - \log_{10} C_{k-2}$), whenever the polynomials of the form shown in the Appendix 6 are multiplied together or raised to the power of n in STATMOD, only the coefficients with exponents $C_o - C_k$ are retained either for the further computations or for the purpose of outputting. Having determined the overall probability distribution of curie release, STATMOD also computes:

(a) the mean curie release over a prescribed fraction of the station life;

(b) the probability that the curie release is greater or less than a specified curie release decade over the station life.

(III.9) Radiological Risk Assessment

(46) This model has not been fully elaborated, mainly because it is entirely dependent upon the adopted risk assessment criterion. Possible criteria are those suggested by Cave[1], Adams and Stone[7], Farmer[6], Beattie, et. al.[57], Doron and Albers[58] and Stanners[10]; they all compare the expected release with some permissible target release. Cave's criterion relates curie release of iodine-131 against probability of event during reactor life (30 years); in addition it differentiates between the summed risk over all faults and the risk due to individual faults. Adams has proposed a link between the safety index (Σpc = sum of the probability curie release of iodine-131 when all possible faults and sequences are embodied)and the distance from the site which is based on the specification of an acceptable individual risk (thyroid carcinoma being dominant, and average atmospheric conditions relevant to the particular site being taken into account). Farmer's criterion relates curie release of iodine-131 as abscissa against events per programme (1000 reactor years) as ordinate. A criterion proposed by Beattie, et. al. assumes that the probability of various curie releases is controlled by a curve of the kind discussed by Farmer and suggests another curve relating upper limits of probability to numbers of cases of thyroid cancer. This curve, called the f-N curve (f=frequency, probability per unit time; N=number of cases), is a characteristic of the population distribution round the site, its meteorology, and the probability - release control curve of the reactor. The definition of Doron's mean

annual severity (MAS) term coincides with that of Adams's safety index; however, Doron has fully illustrated his concept on an example (loss of coolant accident in a PWR) and pointed out that the approach is extremely useful in comparing various branches, or various accidents, and in trying to find the weak points in the design of a reactor safeguards system. Stanners has suggested an integrated weighted curie release method and proved that all the criteria proposed were readily comparable.

(47) The Author has found Cave's criteria extremely useful and straightforward tools especially for quick design assessments, and consequently has adopted these criteria for the purposes of this Thesis. Having done so, the need for having a radiological risk assessment model had disappeared. However, if one is puzzled by the meaning of curie release associated with an accident a simple dose/curie release relationship such as the one suggested by Adams, could be straightforwardly applied.

(48) For fully urban sites the harm which could be caused by the release of fission products to the atmosphere was discussed[41] under three headings, namely:

(a) acute whole body effects due to radiation sickness;

(b) delayed whole body effect, notably leukaemia;

(c) delayed effects on specific organs such as the thyroid.

If, at some stage of development of fault analysis, a division like this gets accepted as a basis it becomes clear that releases other than iodine-131 will have to be taken into account. All releases would have to be grouped and blocks 2 and 3 of the APC compound model would have to be modified to analyse the behaviour of the fission products other than iodine-131 within the reactor system. Block 3 would be outputting curie releases associated with various groups of fission products and the overall summation model would sum these various curie releases. A radiological risk assessment model would then link curie releases of various isotopes to their radiobiological effects. Alternatively the radiological risk assessment model can become another fault consequence submodel (Figure 7) and the summation parameter could be some biological parameter, say injury. The radiological risk assessment submodel in this context would have to be fairly elaborate. One straightforward possibility would be to incorporate the APC CLOUDOS [59] program either in its present form or more likely in a modified form taking perhaps some equations from the model specified by Stauber.[60]

(III.10) Comparison between the American and APC Compound Model

(49) A detailed comparison between the two compound models could be a subject for a separate study. Such a study would be very beneficial from the technical point of view, as it would reveal the differences between the light water and the gas cooled reactors insofar as methods of calculation, engineering standards, nature of reactor systems, and computer programming methods used are concerned. Perhaps, it could even give a hint at differences in basic economies of the two reactor systems. A brief comparison which follows is meant only to outline the basic differences rather than penetrate deeply into the subject.

(50) Having comprehended the major difference between the two models, that they simulate different reactor systems, the constituents of the two models will be now reviewed. Block scheme of the American model is given in Exhibit 1.[11] It contains eight submodels: blocks 1,2,7,8 perform analogue functions as blocks 1,2,4,5 in Figure 1; blocks 3 and 4 perform similar functions like block 3 in Figure 1; block 5 collects the information from previous blocks and contains an atmospheric dispersion submodel the latter function is anticipated for block 6 of Fig.1 (see para's 46-48); block 6 comprises physical damage submodels. Two ways in which noxins (this means something that can cause damage to something useful) can cause damage are accounted for:

(a) by making it necessary to restrict or eliminate activities that are useful and desirable referred to as 'loss of function';

(b) by making it necessary to perform other activities that are not especially useful or desirable referred to as "damage from obligatory tasks'.

To summarise, apart from simulating the behaviour of different reactor systems, there are two additional essential differences:

(a) APC model does not comprise physical damage submodels;

(b) APC model, in its present form is geared to produce a probability distribution of curie release of iodine-131 while the American model can be used to evaluate a probability distribution of either dosage or damage as a function of position with respect to the site.

(51) The reactor transient submodels have nothing in common, mainly because different coolants and reactor core arrangements dictate entirely different heat transfer processes. Neutron kinetic representations should not be very different, however, it appears that the REAXITRAN neutronics is far more flexible and complete than that employed in AIREK.[61] Both compound models assume that stress-rupture is the basic cladding failure mechanism and the analytical treatments are quite similiar; however, the American model does not discuss cladding failure mechanisms in a pressurised reactor. Fission product release submodels perform more or less the same functions to much the same degree of complexity. The fission product inventory sub-model is far more sophisticated and detailed in the American model. As an example, the CURIE-DOSE-THUNDERHEAD programs[62] deal with four fission product groups: helogens, noble gases, caesium, and the rest of the inventory is treated as a whole, while the APC model in its present form treats only iodine-131.The same is valid for the fission product deposition submodels, but these submodels are hardly directly comparable as they simulate different reactor systems. There is a close resemblance in the formulation of blocks 7 and 8 of the American model and blocks 4 and 5 of the APC model respectively, inasmuch as the basic probabillty theory applications are concerned. However, the differences lie in the postulations:

(a) a distinction made in the American model that faults may result from equipment failures, operator errors or natural phenomena has been abandoned at a later stage in the development of the APC model;

(b) the American model assumes that no fault occurs if at least one safety device works, while APC model assumes that no fault occurs if all safety devices work;

(c) the American model truncates the generating functions to four-term ones, while APC model truncates them to eight-term ones (another difference caused by the choice of summation parameters).

Lastly, the American model is fully computerised as an entirety, while the links between the constituents of the APC model have not been established. The integrated form offers tremendous advantages for the production runs, while the non-integrated form offers extra flexibility for introducing inevitable improvements.

(III.11) Comparison between Statistical Aspects of the APC Compound Model and some other SFAMS proposed in the U.K.

(52) This subject has been discussed in References 13 and 14. For the sake of completeness, several conclusions will be reproduced in this paragraph. In general, it was shown that all other methods can be treated as special cases of the Author's method. The method proposed by Cave[1] assumes that for each primary fault which necessitates operation of the automatic protective system the reactor either shuts down correctly or the fault is uncorrected and leads to core meltdown. The method described in Reference 9 assumes that all input parameters to the fault consequence submodels are discrete and presents the outcome in terms of mean curie release as a point on a curie release/probability diagram. Stanners[10] has started from the same basic premise that events

influencing the course of a fault should be represented by probability distributions rather than by single discrete values, but the mathematical developments of the same premise differ and as a consequence this approach does not result in a probability distribution of curie release as a function of time.

IV CLASSIFICATION OF FAULTS

(IV.1) Identification of Procedures

(53) Following the basic premise of statistical fault analysis method that in any nuclear reactor no definable fault is so improbable that it can be disregarded, the first step is to compile a generalised list of faults so that the program computes the risk in terms of fission product release summed over all faults. Each fault is uniquely defined by a chain of events. The initial item in a chain is a loss of some vital function or an operator error. The first stage in defining a generalised list of faults is to compile a list of faults relevant to a reactor type under consideration. The second stage is to compile a list of critical components by analysing in detail the design under consideration and deducing how groups of components may fail to danger. For equipment groups thus selected, specific faults can be attributed. Reactor operating and maintenance procedures should be reviewed to see whether definable errors could lead to initiation of faults. Reactor design should then be scrutinized to determine what additional faults could be caused by external faults. Finally the performance of engineered safeguards should be assessed; the definition encompasses devices whose function is to prevent the occurrence of a fault as well as those designed to mitigate the consequences if a fault does occur. The assignment of failure rates to critical components and engineered safeguards requires a reliability assessment.

(IV.2) Generalised List of Faults for AGR's

(54) The subject of a generalised list of faults for an AGR has been dealt with by several authors, e.g. Cave,[1] Jarvis,[63] Worthington[64], etc. In an AGR, structural damage or radioactive release are in general only possible following excessive overheating of the reactor core and particularly of the fuel. Broadly, faults which can lead to

such overheating can be classified into two main divisions. Firstly, there are faults due to increase of heat generation following inadvertent core reactivity increases; these are therefore called reactivity faults. Secondly, there are faults associated with decrease of heat removal from the core due to a reduction in coolant mass flow, and these are called coolant circulation faults. Coolant circulation faults can also be divided into two classes, those associated with partial or total failure of the coolant circulation through the core or through individual fuel channels, with the system still pressurised, and those where the coolant mass flow is reduced because of inadvertent system depressurisation with or without the coolant circulation equipment still in operation. When this division of faults is applied it may result in a more specific list, such as now given.

Reactivity faults

Symmetrical uncontrolled withdrawal of coarse, trim/override or zone rods

Asymmetrical uncontrolled withrawal of individual rods

Fuel stringer withdrawal or insertion

Substantial water ingress due to burst boiler tubes.

Coolant circulation faults

(a) Faults causing breach of pressure circuit

Failure of pressure vessel itself

Failure of small size penetrations

Failure of medium size penetrations

Failure of large size penetrations

Failure of fuelling/control unit standpipes.

(b) Faults in pressurised reactor

Loss of supply to one or more circulators

Spurious operation of gas circuit isolating valves

Failure of gas circulator outlet duct or casing

Core bypass due to structural failures within the pressure circuit

Incorrect gag setting

Channel flow starvation due to mechanical blockage, bypassing, or inadvertent gag closure

Loss of natural circulation.

To this list out-of-core faults should be added; fuel handling and fuel ejection faults.

(55) Core overheating can in fact also occur due to inlet coolant temperature increases, which may follow malfunction in the secondary circuit; but the resulting fault conditions are not very acute. The following brief list of faults might be deemed adequate.

Secondary coolant faults

Loss of main feed pump

Spurious operation of feed regulators

Failure of feed or steam pipes

Turbine/generator trips.

(56) At a later stage in the analysis faults external to the reactor ought to be defined. It is extremely difficult to identify them to the same precision as other faults, but an effort should be made, and the following list could serve as guidance:

External faults

Natural phenomena (floods, earthquakes, wind carried objects)

Missiles originating within (mainly rotating machinery in particular turbine alternator runaway and circulator overspeed) or outside the station

Sabotage

Fire

Aircraft crashes

Shock wave effects.

(IV.3) Generalised List of Critical Components

(57) The second stage of analysis is completely related to a concrete design under consideration. In looking for critical components which form the equipment groups, the following division could be very helpful:

(a) Pressure retaining and structural components

(b) Primary and secondary coolant systems

(c) Reactor automatic protective systems

(d) Reactor emergency cooling systems.

As regards the reactor operation, all modes of reactor operation should be accounted for, namely:

(a) Shutdown

(b) Start-up

(c) Part-load

(d) Full-load.

(V) APPLICATION OF THE APC COMPOUND MODEL TO AN AGR - STAGE 1 FUNDAMENTALS AND STEADY STATE CALCULATIONS

(V.1.) Brief Description of the AGR Design under Consideration

(58) The fuel (UO_2 pellets) is contained in stainless steel cans sealed at each end to form fuel pins. Thirty-six pins are arranged in three ringed clusters within supporting graphite sleeves. Fuel stringers comprise eight clusters held by a central stainless steel rod. The core is made up of graphite columns containing either fuel or control rods.

(59) The complete reactor unit is housed within a cylindrical concrete pressure vessel with the core at the centre, the boilers spaced around it and the circulators under the boilers. An inner pressure cylinder in the form of a bell divides the space into two regions. The circulator delivers the gas into an annular space between the cylinder and the radial neutron shield. Here it flows upwards into the free space between the dome of the pressure cylinder and the reactor core, where it is divided into two streams. About 50% flows in a downward direction through the annular gap between the radial shield and core restraint cylinder into the space below the reactor core, whilst the other 50% flows through the top shield and the intermediate spaces between graphite blocks of the moderator into the area below the reactor core. After mixing the recombined total flow passes upwards through the fuel stringer channels, taking up heat from the fuel pins and charge tubes, and is discharged through ports into the plenum outside the pressure cylinder; from there it passes down through the boilers, giving

up heat to the secondary loop, and back to the circulator inlet.

(60) The circulators are driven by synchronous motors through fluid couplings so that variation of gas mass flow is obtained by variation in circulator speed. The motors derive their source of electrical supply from Unit Boards. In addition to the main motor, a 20%/80% pony motor for use during shutdown and after fault conditions and a very low speed (5%) pony motor (barring motor) for use immediately following a reactor trip are provided. The 20%/80% speed pony motor power supplies are taken from the station electrical system backed up by a supply taken from boards fed by automatically starting diesel generators. The 5% speed pony motors are also supplied from these boards. Each circulator is associated with its own lubrication system.

(61) There are five groups of control rods. Shutdown capacity is provided by three groups of boron steel rods which are fully withdrawn during normal operation. Grey rods divided between the zone and trim/override groups are partially inserted into the core during normal operation. The zone rods are on automatic control with the purpose of retaining the gas outlet temperature within the prescribed limits.

(62) The reactor automatic protective system is based upon the use of a primary and secondary guard line system. Reactor tripping functions relevant to the faults analysed in this Thesis are incorporated into the secondary guard lines. Primary guard lines provide protection in the shutdown condition. Tripping two out of three of the secondary guard lines results in operation of the common final break contactors and the individual control rod clutch break relays. The clutches on both black and grey groups of control rods are de-energised and all rods drop into the core. The trip channels are connected into each of the three secondary guard lines. Auto reset gas outlet temperature

trip amplifiers are arranged in nine groups of four each connected in two out of four coincidence. Circulator speed trip channels protect against high rate of change of average circulator speed, excessively low average speed and excessive deviation of any one circulator speed from the average. Protection is provided by three speed-sensing devices on each circulator feeding corresponding trip amplifiers. The three circulator speed trip channels for each circulator are connected in 2/3 (two out of three) coincidence into each guard line. Reactor coolant rate-of-change of pressure protection is provided by four sets of reactor pressure transducers feeding trip amplifiers. Trip signals are initiated for rates of fall of coolant pressure in excess of the set rate, and for coolant pressures below a preset minimum limit. The trip channels are connected in 2/4 coincidence in each guard line. The shutdown amplifier channels provide back-up to the coolant gas outlet protection for various reactivity excursions. They are connected into each of the guard lines in 2/3 coincidence and give a trip signal under margin and high power conditions. High coolant pressure trip channels provide a third line of protection against fault conditions in a pressurized reactor. The signals from four pressure switches are connected into each of the guard lines in 2/4 coincidence.

(V.2.) Selection of Illustrative Faults and associated Initiating Events

(63) It was thought that the best way of illustrating the method would be to conduct a small statistical fault analysis study. For practical reasons this study had to comprise a fairly small number of faults. From the AGR conventional fault analysis experience it was renowned that some faults illustrate very well the behaviour of AGR's under fault conditions. The following faults have been distilled from a generalized list such as the one presented in Chapter IV: failure of

supply to all circulators, medium size penetration failure and reactor start-up fault. These faults illustrate respectively the reactor core behaviour:

(a) in a pressurized reactor at full power following a fairly high frequency fault resulting in a loss of heat removal capacity;

(b) in a depressurized reactor at full power following a very low frequency non-conventional fault leading to a severe loss of heat removal capacity;

(c) in a pressurized reactor at very low power following a low frequency multiple fault resulting in excessive rates of increase of neutron power.

(64) The type and frequency of failures of the gas circulator drives for the AGR greatly depend on the overall system design, on the method of dealing with equipment outage, and on the number and type of circulators for a particular design. For the design under consideration the initiating event is either a fault leading to a loss of the unit boards or a multiple fault to the circulator drives.

(65) The use of a concrete pressure vessel, dependent for pressure strength on a multiplicity of steel tendons, provides a high degree of inherent structural integrity against sudden failure. Such failures have therefore very low probability of occurrence. Since failure of the concrete pressure vessel itself is considered to be of very low probability, the only potential sources for a reactor depressurisation are the vessel penetrations. They can be in general classified as large, medium and small size. In this context, medium size penetration was assumed to be of 36 in. diameter; this orifice size may be

associated with a boiler penetration, for instance. In the conventional fault analysis medium size penetration failure is considered to be incredible.

(66) The black rod groups are not capable of movement until the secondary guard line is reinstated after a reactor shutdown. This in turn depends on several other conditions being satisfied, such as minimum gas pressure in the circuit and adequate flow to ensure satisfactory operation of the coolant gas outlet rate trips. Interlock systems designed to safety circuit standards ensure that the rod groups are withdrawn in the correct sequence and prevent excessive rates of reactivity addition to the core. During the initial stages of start-up the amount of rod withdrawal that can occur without the intervention of the operator is limited by an automatic system which ensures that the reactor period cannot reduce below a pre-determined level. Failure of this period freeze system, which is controlled by the data processing system, would cause unrestricted withdrawal of Group 3 of black rods (withdrawals of Groups 1 and 2 does not result in reactor supercriticality). At some partial insertion of Group 3 the reactor would reach the thermal region and the zone control autopermissive interlock would be cleared. The zone rods would be then free to move and one can postulate a false signal from the data processing system or some malfunction in the zone control loop. In both cases the zone rods would be driven out of the core. The initiating event is defined as the period freeze interlock failure following a malfunction in the zone control loop.

(V.3) Basic Assumptions behind the Simulations

(67) For the sake of convenience it has been assumed that the

faults always take place in a fuel cycle equilibrium core. An AGR fuel cycle equilibrium core contains fuel channels of varying degree of irradiation, from newly loaded channels to reject channels, a newly loaded channel being on average approximately 15% more highly rated than the half-irradiated channel, and the reject channel being on average 15% less rated than the half irradiated channel. [65] In view of the fact that REAXITRAN can simulate the transient behaviour of up to three fuel channels, it has been postulated that all the fuel channels can be divided into three equal groups: newly loaded, half irradiated and reject fuel. All the channels within a group behave identically in the transients. Pessimistic higher ratings have been then associated with these channels; the rating of a newly loaded channel was assumed to be that of the peak channel and the rating of a reject channel was identified with the rating of the time and space averaged channel which in the calculations represented the driving channel. Further to that a mean channel irradiation of 2800 MWd/t was associated with a newly loaded channel rating, 9000 MWd/t with a half-irradiated channel, and a mean reject irradiation of 18300 MWd/t with the average channel rating. Having built in these pessimistic factors in the assumptions, it was thought justified to ignore possible random effects in the can temperature transients (the majority of the systematic effects have been included in the steady state core temperature evaluations).

(V.4). Basic REAXITRAN Input Data Requirements

(68) Any REAXITRAN set of input data consists principally of:

(a) bulk reactor quantities;

(b) axially variant channel independent quantities;

(c) axially variant channel dependent quantities;

(d) disturbance/time histories;

(e) program control parameters.

Bulk reactor quantities are assumed to be axially invariant in the program and are relevant to the reactor as a whole. Some axially variant quantities do not vary from one channel to another, and some are only required for the driving channel. Variables not relevant to the simulations of the faults described in paragraph 63, like xenon poisoning parameters, for instance, will not be discussed. Disturbance/ time histories are to be presented together with a fault description. Technicalities concerning running of the program were thought to be of no interest. The notation used will be that of Appendix I.

(69) The REAXITRAN simulation of the active core assumed nine axial mesh points; that is, the mesh points were situated at the end of each fuel element and at the bottom and top core reflector boundaries. The mesh intervals were therefore constant and equal to the fuel element pitch length. Relevant details of the core are given in Table 1 and basic fuel stringer properties are summarized in Table 2. The can was assumed to lie over the UO_2 pellets only, thus the quantities L_1 and L_2 were 1 and 0.9294 respectively. Since REAXITRAN does not distinguish between graphite main bricks and interstitial bricks, the data presented in Table 3 relate to the whole of the moderator. The distribution of neutron and fission product power between fuel pins and moderator components (see Fig. 3) is presented in Table 4. The fission product decay curve four term fit is given in Table 5. This fit is valid up to a time of 4000 sec; for one run exceeding 4000 sec. it was replaced by another fit valid until 100,000 secs. Core mean weighted delayed neutron fractions and decay constants used are summarized in Table 6.

In calculations the three group fission rates in U-235, U-238, Pu-239 and Pu-241 were considered. Channel powers (see para. 64) used are given in Table 7. Cluster radiation constants using an 11-surface model [66] are quoted in Table 8. The rod grouping details, group reactivity worth and speeds are quoted in Table 9. In the normal operating reactor only grey rods will be in the active core. These rods were designed to operate at an insertion of 62.5%. The reactor controller and reactor trip parameters used are summarized in Table 10. The temperature rise required to initiate the release of Wigner energy in all moderator components was taken to be 50°C.

(70) Axial variation of channel independent variables is presented in Table 11. The neutron flux shape was normalised to the peak of unity. For the sake of pessimism the reactivity coefficients quoted relate to the inner reactor core region. The KINAGRAX program was used for evaluation of a set of steady state temperatures for the three selected channels. The axial variations of temperatures in the peak channel are illustrated in Fig. 8.

(V.5). Iodine -131 Inventories

(71) With reference to Eq. 5 of Appendix II, the iodine activity is a function of rating and fission yield. The iodine yield depends upon the isotopic content of fissionable materials in the fuel, which in turn varies with irradiation. Ignoring Pu-241, the relative contents of U-235 and Pu-239 against irradiation are displayed in Fig. 9. In the literature, [36] iodine yields of 3.1% for U-235 and 3.77% for Pu-239, respectively, are quoted. Application of linear weighting formula results in the iodine yield versus irradiation graph, presented in Fig. 10. Assumptions discussed in para. 67, regarding channel ratings and mean irradiations, result in a distribution of fuel stringer ring

irradiations quoted in Table 12.[67] In this table and in other forthcoming tables, fuel assemblies 2-7 only have been presented; the reason being the fact that the fuel assemblies 1 and 8 make little or no contribution to any subsequent follow-up calculations. From the information contained in Fig. 10 and Table 12 it was straight-forward to construct Table 13, showing iodine yields in fuel stringer rings of fuel pins. The mean ring ratings quoted in Table 14 have been derived from an REAXITRAN output (REAXITRAN outputs heat production terms as a part of the steady state printout information). After a conversion from MW/TeU into MW (1.288 kg of uranium per pin), iodine activities quoted in Table 15 have been obtained. The approximate sum over all fuel stringers yields a quantity of 3.9×10^7 Curies for the total iodine-131 inventory in the fuel cycle equilibrium reactor core. If required, equilibrium inventories of some other fission products can be evaluated similarly.

(72) Referring to paragraph 34, improved versions of the SUPATRICE program can be used for an evaluation of free iodine-131 inventories in free spaces within the cans. At the time when this work had been carried out considerable difficulties were being experienced in attempting to run the SUPATRICE programs on the London ATLAS digital computer. In addition, considerable uncertainties exist as regards the fission gas release rates used as input data for the SUPATRICE programs. These two reasons have influenced the Author to adopt a simpler approach. Reference 68 suggests a curve relating estimated fractional release of iodine-131 under equilibrium conditions to mean pin rating. Using the mean ring ratings, corresponding fractional releases have been estimated from this curve which is for the sake of completeness reproduced in Fig. 11. These fractional releases are quoted in Table 16

and the corresponding inventories of free iodine-131 in Table 17.

(V.6). Hoop Stresses

(73) With reference to the definition discussed in para. 29, and assuming the mean coolant pressure of 450 psia and the mean can dimensions of d/2c = 23.8 (ribs smeared uniformly over the can surface giving an effective increase in the can thickness), the corresponding linear can hoop stress versus pressure differential curve is displayed in Fig. 12. The SUPATRICE programs compute the can internal pressure which is directly proportional to the fission gas release rates (see eq. 6 of Appendix III). For the reasons already discussed in para. 72, the Author adopted a "simpler limit" approach suggested by Pearson[69]. This approach results in the assumption that the peak can pressure from gaseous fission products at the end of fuel burn-up equals the coolant pressure and that the distribution of pressures among cans at any time is then proportional to the mean burn-up inside the cans. The resulting can internal pressures are quoted in Table 18.

(VI) APPLICATION OF THE APC COMPOUND MODEL TO AN AGR - STAGE 2 RESULTS OF THE STATISTICAL FAULT ANALYSIS STUDY

(VI.1) Failure of Supply to all Circulators

(a) Reliability assessment of the initiating event

(74) 11 kV unit boards are tied to the main generator and to the 400 kV grid switching station busbars via T-offs from the main connections between the generator and generator transformer and the unit transformer. One unit board supplies two of the four gas circulator main motors, and the second supplies the remaining two gas circulator main motors. Loss of the unit boards can be caused by an electrical fault, by a missile or by mechanical damage. There are various conceivable electrical faults, the following predominant ones would each result in the loss of the unit boards:

(a) cable fault in the 400 kV line from the generator transformer;

(b) generator transformer fault;

(c) fault in either unit transformer;

(d) generator fault;

(e) faults in the interconnection between the above mentioned units;

(f) failure of protection on the unit boards.

When these individual faults are combined with possible multiple faults, a combination of failures that would cause failure of all circulators is obtained. This results in the reliability block diagram displayed in Fig.13. Each of the blocks is in itself very complex and would require an elaborate study to make an accurate assessment of its failure rate. For the purpose of this work it was deemed adequate to associate a failure rate with the complete block. Table 19, compiled from the available statistics, shows tentative failure rates in terms of events/year which can be attributed to various individual faults. Owing to the diversity employed, failure rates associated with the multiple nonavailability of all circulators are negligible in comparison with the derived rate of 0.23 events/year. In order to have multiple nonavailability of all circulators, either all services to fluid couplings must fail, all cooling water services must fail, structural failures of drive components must occur in all circuits, or some combination of these failures must occur. The combinations of failures are shown in Fig. 13.

(b) Fault development and sequence diagram

(75) Given the loss of supply to all circulators, the circulator speed reduces in accordance with the circulator rundown characteristic displayed in Fig.14. This characteristic depends on primary (main motor, coupling parts) and secondary (blower plus shaft) inertias. The core mass flow reduces accordingly. The automatic control system should see any changes in the gas outlet temperature. Against this fault the automatic protective system provides three lines of protection: circulator speed, gas outlet temperature and high coolant pressure trip channels. The fourth line of protection is the operator action. It is sufficient for one line of protection to be effective to cause a reactor trip. The reactivity insertion characteristic following a reactor trip is displayed in Fig. 15. Initiation of a reactor trip at full power produces a kind of rundown in neutron and

fission product powers shown in Fig. 16. Following the gas circulator rundown the speed is held at 5% by the shutdown cooling system consisting of the very low speed pony motors. In addition to the shutdown cooling system, there is a possibility that the natural circulation can be sustained. During core flow changes convective heat transfer coefficients vary with flow in a manner illustrated in Fig. 17.

(76) The initiating event and possible fault developments having been defined, the stage in the analysis is reached when construction of a fault sequence diagram depicting explicitly modes of the plant behaviour following the fault onset becomes a predominant problem. A fault sequence diagram given in Fig. 18 could be deemed adequate to represent various alternatives for the reactor system behaviour. It does not comprise all possible modes of the plant behaviour, but it actually represents what is hoped to be a suitable compromise that might lead to an optimum solution. The plant behaviour, given the failure of supply to all circulators, is by and large governed by the performance of automatic control, automatic protective and shutdown cooling systems.

(c) REAXITRAN temperature transients

(77) Once defined, the fault sequence diagram specifies all possible combinations for which the temperature transient behaviour assessment is required. When the fault sequence diagram analysis is being performed for the first time it is advisable to produce the temperature transient assessments for all possible sequences. If however, some preliminary knowledge exists, then a number of sequences can be ignored as the small differences in temperatures would not be

recognizable by the existing crude can melting criterion. In addition the reactor steady state temperatures are well below the can melting limit, thus allowing ample margins for the transient temperature excursions. The preliminary knowledge related to the example under consideration has not been fully exploited in order to avoid diminishing its illustrative value. Nevertheless, some evident factors have been taken into account, e.g. if the core temperature transient behaviour following the trip initiated by the second line of protection is satisfactory, then a similar case incorporating the first line of protection need not be analysed; or if the reactor is tripped by the first line of protection the controller response is negligible.

(78) The temperature transients associated with the sequence 0-2-4-8 are displayed in Fig. 19. In accordance with the investigations reported in Reference 70, the preset trip margin of 40°C initiates the reactor trip after approximately 15 secs. The circulator shaft speed trip signal would come through very quickly and initiate the reactor trip in approximately 2 secs. The controller response on the time scale of 15 secs is relatively small and consequently does not affect much the initial transient peak. On these grounds the sequences 0-1(2)-3-8 and 0-1-4-8 need not be analysed. The terminology used means that the sequence may branch from 0 to 1 or 2, then to 3 and 8. The temperature transients associated with the sequence 0-2-4-9 are displayed in Fig.20. For the reasons just explained the sequences 0-1(2)-3-9 and 0-1-4-9 need not be analysed. In terms of the core coolant mass flow, the natural circulation branch 9 is only a special case of the branch 8 when only one out of four low speed pony motors is operational.

(79) No cans melt (melting temperature 1437°C) following the sequence 0-2-5-9 as shown in Fig. 21. The effect of the controller (sequence 0-1-5-9)is displayed in Fig. 22. In both cases a significant proportion of cans would exceed the currently accepted can damage criterion. This means that these cans would be prone to failure in subsequent operations of the reactor and consequently some financial penalty might be involved. The sequences 0-1(2)-5-8 need not be analysed as the can damage would be mainly caused by the excessive temperatures reached in the initial transient peak. The APC 1322 four plenum (reactor core, charge tube region, boiler, diagrid region) computer program[71] was used for evaluation of the mean coolant pressure transient following the fault onset. Assumptions of the over-pressure trip margin of 10% and an instrumentation delay of, say, 10 secs [72] yield an overall trip delay time of 40 secs as shown in Fig. 23.

(80) The temperature transients associated with the no trip sequence 0-2-7-9 are displayed in Fig. 24, and the corresponding axial profiles of inner ring can temperatures in Fig. 25. The negative fuel temperature reactivity coefficient acts to avert the can melting. It partially succeeds in the initial transient peak, but soon after that, owing to the positive moderator temperature reactivity coefficient, the overall temperature reactivity coefficient becomes positive, resulting in the divergent reactor core behaviour. The sequence 0-2-7-8 results only in a relative delay of the massive can melting stage. Operation of the controller (sequence 0-1-7-9) is sufficient to prevent the can melting in the initial transient peak, as shown in Fig. 26, however here again, as the transient progresses the overall temperature reactivity coefficient becomes positive leading to the divergent reactor

core behaviour. Much the same is valid for the sequence 0-1-7-8; the time scale is only different so that the can melting stage is further delayed by approximately 45 mins. The sequence 0-1-6-9 is characterized by a significant portion of cans exceeding the can damage criterion, while the sequence 0-2-6-9 results in partial reactor core meltdown. An analysis of the sequences 0-1(2)-6-8 would not have altered the final results.

(81) All sequences incorporating branch 10 yield similar results which are typical for the behaviour of an AGR in zero flow conditions. After a dip caused by the beneficial radiation dominated condition, the can temperatures, influenced by the moderator temperature rise, gradually increase until they arrive at the melting stage. This stage, as shown in Fig. 27, is reached after several hours. To illustrate the differences in the initial part of the transient between this envelope case and a practical case, the temperature transients associated with the sequence 0-2-4-10 are displayed in Fig. 28.

(82) Apart from risk and damage-free sequences, all other sequences can now be grouped into those resulting in can damage and those resulting in can melting. Sequences leading to can damage have no safety implications, but may have severe financial penalties. A financial criterion has not yet been defined, so at present can damage assessment is only educational to illustrate a possible trend in the future. The safety valves would lift in all can melting sequences, thus releasing gaseous fission products to the atmosphere. In some cases they would lift, reseat and lift again. Sequences containing the no trip branch and sequences containing the coolant flow stagnation branch eventually result in almost complete reactor core meltdown. For the sequence 0-2-6-9, the cans which have exceeded the melting temperature after a trip delay of 3 mins are marked in Table 20. These cans contain 8.05×10^6 Ci of iodine-131 when the summation over the whole core is

performed. For the purposes of this Thesis, an assumption was made that when one fuel assembly melts it does not influence behaviour of the others in the same channel.

(d) Reliability assessment of engineered safeguards

(83) The fault sequence diagram analysis has unambiguously shown that the plant behaviour following the failure of supply to all circulators is principally governed by the performance of automatic protective and emergency cooling systems and to a much smaller extent by that of the automatic control system. In fact the automatic control system is not an engineered safeguards system; however, in some cases it mitigates the fault consequences and therefore can be considered as such (see definition in paragraph 53).

(84) Fig. 29 displays an abbreviated reliability block diagram for the automatic protective system in which the guard lines incorporating the gas outlet temperature trip amplifiers, circulator shaft speed trip amplifiers, and coolant gas pressure switches, are omitted. As shown in Appendix VII, the redundancy logic arrangements applied are such that the integrity of the system is virtually dependent upon reliability of the output stages. In other words reliability of the system is mainly dependent upon the contactors and inverters. In addition the probability of the control rod drop failure is small owing to the fact that the failure of any three adjacent rods to drop would possibly lead to local criticality.[73] It is currently believed that a probability of 10^{-3} is attributable to the individual rod drop failure resulting in a probability of 10^{-9} for the failure of three rods

to drop. When this quantity is multiplied by a small number of three adjacent rod combinations the final probability lies in between 10^{-7} and 10^{-8}.[74] The automatic protective system is not an active system so fail to danger faults would not show immediately and therefore can be classified as unrevealed faults. In fact these faults remain in existence until the next proof check when the fault is immediately repaired. It is usually assumed that these faults are exponentially distributed with respect to time so that the cumulative probability of failure is a function of the elapsed time from the last proof check on the appropriate equipment when the equipment was known to be working correctly. Inverters are usually tested at 3-monthly intervals while the contactors can be tested only at the prolonged plant maintenance shutdown periods (approximately 2 yearly intervals).[75] When these proof check times are combined with tentative failure rates of 0.0015 faults/year for contactors and 0.001 faults/year for inverters, the overall probability of failure per demand of 1.3×10^{-5} is obtained.

(85) The initiating signal to switch the very low speed pony motors (barring motors) comes from the guard lines and is fed to the motor via a contactor. If the electrical supplies are live the clutch would engage the motor. The essential supply system consists of a system based on four diesel generators. The system has two voltage levels, namely 3.3 kV and 415 V. The 3.3 kV voltage level consists of four switchboards termed diesel generator boards. These are normally supplied from the station 11 kV system via either an essential auxiliary transformer or a station auxiliary board. The four diesel generators are connected one to each board. They are automatically started when a 'no-volts' signal appears on the diesel

generator boards. From each 3.3 kV diesel generator board, a 415 V essential services board is fed via a transformer. From this information the reliability functional block diagram shown in Fig. 30 has been drawn. Table 21 quotes predominant faults which would each result in the failure of a barring motor to start. In the absence of any statistical data, failure rates have been qualitatively assessed and a possibility of an operator action ignored.
A tentative probability of 1.4×10^{-2} was associated with the failure of one barring motor to start; hence a probability of failure of all barring motors to start would be 3.8×10^{-8} (example of independent trials). As regards the natural circulation, it is currently believed that it would be sustained providing the boiler evaporation end point does not drop or, in other words, providing the supplies for flooding the boilers were available.[76] With reference to paragraph 78, probability of sustaining the natural circulation should be accounted for; however, in view of the facts that readily available statistical data were not at hand and that the quoted probabilities were thought to be satisfactory, no analysis has been done. A further pessimism was introduced by assuming that one barring motor would be out for maintenance. This resulted in a probability of failure of 2.7×10^{-6} per demand attributable to all three barring motors to start.

(86) For the automatic control system the safe condition is when the zone rods are driven into the core to reduce the core reactivity. These rods are dropped into the core when the automatic protective system functions, and once this happens the control system ceases to influence events. The reactor is divided into five sectors; each one is supplied with a gas outlet temperature loop controlling one zone

rod. The block diagram representing the analogue equivalent of the zone control loop is given in Fig. 31. Faults which affect the automatic control system adversely are those which prevent the rods from being driven into the core when called upon. The temperature demand signal block is in a circuit common to all five loops. Other blocks are repeated in all five loops, and consequently the faults occurring in them affect only single zone rod movement. The control system is an active system operating on a semi-continuous basis to alleviate neutron flux peaking effects in addition to controlling the reactor power level. The faults causing spurious rod drive would show immediately, thus they can be classified as revealed faults. On this basis all faults leading to single zone rod malfunction can be ignored and only faults in the temperature demand signal block resulting in rod withdrawal, and loss of mains supplies resulting in rod freeze need to be considered. They may not be damaging to the reactor system unless they occur simultaneously with the initiating event. Bearing in mind the expected frequency and duration of the initiating event a probability of control system failure of, say, 10^{-3} is definitely on the pessimistic side. There is no need to calculate this probability more accurately since it did not constitute an item in the set of basic STATMOD data.

(e) Fault probabilities

(87) Given the initiating event (E_1) the REAXITRAN results indicate that the fault (F_1), in terms of the fission product release to the pressure circuit, would occur if either the automatic protective system fails to trip the reactor or a small percentage of the full load coolant circulation cannot be restored. Hence the conditional probability of fault F_1 on the hypothesis of E_1 is:

$$P(F_1/E_1) = q\ (S_1) + q\ (S_2)$$

where $q\ (S_1)$ stands for the probability of failure to shut the reactor down and $q\ (S_2)$ stands for the probability of failure to restore the coolant circulation when the reactor is in the shutdown condition. For the assessed values of $q\ (S_1)$ and $q\ (S_2)$, $P(F_1/E_1)$ becomes 1.3×10^{-5} and this also fixes the value of $P(F_0/E_1) = 1-1.3 \times 10^{-5}$.

(88) Given the average failure rate estimate of 0.23 events/year, Table 22 summarizes the probabilities of any particular number of initiating events as a function of time during the station life assuming that the maximum number of events was 8.

(VI.2) Medium Size Penetration Failure

(a) Reliability assessment of the initiating event

(89) A satisfactory explanation for the modes of failures in service of steel pressure parts of prestressed concrete pressure vessels has not been found in the available literature. Consequently, there was no alternative but to assume that failures occur at random. These steel components are not subject to significant irradiation and are all subject to inspection and/or retest while in service. These factors have been taken into account by Cave[1]; for medium size penetrations probability of failure of 10^{-6} in 30 years service has been quoted. When this probability is multiplied by a number of medium size penetrations per reactor and divided by 30 years, a quantity of approximately 10^{-6} events/year has been obtained.

(b) Fault development and sequence diagram

(90) Given the pressure vessel penetration failure, the reactor

rapidly depressurises through the orifice with the core flow initially associated with the flow dynamics of the depressurisation. Calculation of the core flow history and pressure decrease were made taking into account the full gas circuit dynamic characteristics using the APC 1170 computer program.[77] Fig. 32 shows the time variation of reactor pressure following the initiating event. What has been said in paragraph 75 as regards the response of the automatic control and automatic protective systems is also valid in this case. The lines of protection are of course different; reactor coolant rate-of-change of pressure transducers and channel gas outlet thermocouples represent two lines of protection against a depressurisation fault. Operator trip action represents the third line of protection. Effectiveness of either line of protection results also in activation of the intertripping devices causing the circulator trip. In this case the core flow is in addition to flow dynamics of the depressurisation influenced by the rundown characteristic of the circulators as shown in Fig. 33. Following the gas circulator rundown the speed is held at 20% by the emergency cooling system consisting of the low speed pony motors. The design has ensured (an interlock arrangement) that the high speed (80%) pony motors cannot be activated before the reactor pressure becomes critical, i.e. 75 secs after the fault onset. Thereafter, the operator is in a position to activate the high speed pony motors (operator action).

(91) A fault sequence diagram displayed in Fig. 34 represents reasonably well various possible alternatives following the initiating event. Here again it does not comprise all possible modes of the plant behaviour, which is principally governed by the performance of

automatic control, automatic protective, inter-tripping and emergency cooling systems. The delayed trip branch has been omitted owing to the fact that in a depressurised reactor it would produce much the same results as the no trip branch.

(c) REAXITRAN temperature transients

(93) The basic principles described in paragraph 77 have been preserved. In addition the fact that, for depressurisation faults generally, detailed knowledge of transient can temperatures for some sequences is required in order to assess the exact number of cans bursting, has been recognized.

(94) The temperature transients associated with the fast trip operator action sequence 0-1(2)-3-6-8-9 are displayed in Fig. 35. A brief analysis had shown that the rate-of-change of pressure signal would activate the reactor trip circuits in approximately 3 secs and on this time scale the branches 1 and 2 become undistinguishable. It was assumed that the operator would activate high speed pony motors at 15 mins. The corresponding axial profiles of inner ring can temperatures in the newly loaded channel are exhibited in Fig. 36. Given the initiating event, the rate of gas outlet temperature rise is similar to that displayed in Fig. 19. As a consequence, here again, the slow reactor trip was initiated after approximately 15 secs. The temperature transients attributed to the sequence 0-2-4-6-8-9 are exhibited in Fig. 37 and the corresponding axial profiles of inner ring can temperatures in Fig. 38. The controller operational sequence 0-1-4-6-8-9 resulted in small can temperature differences which did not justify a separate presentation. Fig. 39, associated with the

sequence 0-1(2)-3-7 illustrates a pre-judged conclusion that the failure of the inter-tripping system to operate is beneficial from the fault transient behaviour point of view. The gas circulation maintained is sufficient to prevent any can in the core from being damaged.

(95) The no trip sequences closely resemble those following the failure of supply to all circulators (Figs. 24, 25, 26). The coolant flow differences affect only the time scale of the reactor core meltout. For the sake of illustration, the temperature transients attributed to the sequence 0-2-5-7 are presented in Fig. 40. Sequences incorporating the coolant flow stagnation branch yield similar results which are typical for the behaviour of an AGR in zero flow conditions as already discussed in paragraph 81 and shown in Fig. 27. The "trip but no operator action sequences" 0-1(2)-3(4)-6-8 yield temperature transients exhibited in Fig. 41. Owing to a small coolant flow, a great majority of fuel cans remain at elevated temperatures for an extended period of time but do not reach a massive melt out stage.

(96) Apart from risk and damage free sequences, all other sequences can be grouped into those resulting in complete reactor core meltdown, those resulting in a significant number of cans failing and those resulting in a small number of cans failing. Sequences incorporating either zero flow branch or no trip branch result in complete reactor core meltdown. Sequences incorporating trip but no operator action branches result in a significant number of cans bursting, while sequences embodying trip and operator action branches lead to either a small number of cans bursting (slow trip sequences) or a very small number of cans failing (fast trip sequences).

(d) Determination of expected number of cans failing and free iodine activities

(97) Following the initiating event some seconds elapse before the cans become subjected to hoop stresses. These times, worked out from Fig. 32 and Table 18, are presented in Table 23. Owing to a fast rate of depressurisation, soon after these times have been reached, the cans become subjected to the respective ultimate hoop stresses quoted in Table 24. In between the times quoted in Table 23 and 72 secs when the reactor is fully depressurized, the cans are subjected to changing conditions of hoop stress and temperature. In order to illustrate the time variation of hoop stresses they have been calculated 25 secs after the fault onset, i.e. when the coolant pressure was reduced to 117 psia. The stresses obtained are summarized in Table 25. Now that the variations of can hoop stress and temperature with time are known and the APC 1380 program can be used, the stage is reached when one must fix the time intervals over which can temperatures and stresses can be assumed constant. For each time increment the time to failure is assessed for the corresponding stress and temperature from the appropriate stress rupture data; the obtained fraction is then a measure of the fraction of rupture life used during the chosen time increment. Summation of fractional mean times to failure obtained in this way over all increments results in a cumulative fraction (see paragraph 29). To arrive at an acceptable time increment from the accuracy point of view, several time increments had been selected and the results compared. It was found that, for this kind of a transient, 25 sec increments were adequate in the initial stages, while 50 sec and even 100 sec increments were acceptable in the subsequent stages. Ideally one should take average

values over a time interval, however, the Author proved that it was adequate to take values at the end of a time increment without significantly jeopardising overall accuracy of the results.

(98) Complete can temperature/time histories for the fuel assemblies 2 - 6 of the newly loaded channel associated with the sequence 0-1(2)-3-6-8-9 are exhibited in Table 26. From the can failure point of view the most vulnerable cans were those positioned in fuel assembly 4 and in particular inner ring ones. Consequently, cumulative fractions of the rupture life should first be calculated for the cans in fuel assembly 4. For practical purposes it was appropriate to assume that: (a) no cans fail in the newly loaded channel owing to small hoop stresses; (b) in the absence of any experimental data cans subjected to temperatures below 800°C were not likely to fail irrespective of the corresponding hoop stresses. Hence Table 27 contains only the can temperatures in the reject channel which exceed 800°C during the transient. The cumulative fraction for the outer ring of 0.0473 was found to be below the threshold for failure (threshold 0.1). This fact means that there would be no failures in any outer ring of fuel pins. A small proportion of the cans in inner and middle rings of fuel assemblies 3, 4 and 5 are likely to fail as shown in Table 28. Similar calculations have been performed for the half irradiated channel whose temperature/time histories are displayed in Table 29. Finally, the ultimate probabilities of can failures for the whole sequence 0-1(2)-3-6-8-9 are summarized in Table 30.

(99) Much the same procedure has been repeated for the sequence 0-1(2)-4-6-8-9. Here again, it was found that practically no cans fail in the newly loaded channel whose complete can temperature/time

histories are shown in Table 31. The number of time intervals had been reduced in view of the experience gained from the previous example. The can temperatures in the reject channel exceeding 800°C are quoted in Table 32, while the corresponding fractions of rupture life used are presented in Table 33. The ultimate probabilities of can failures for the sequence 0-1(2)-4-6-8-9 are summarized in Table 34. Tripping on the second line of protection meant in terms of the can failure that the cans use up more of the rupture life.

(100) The trip but no operator action sequences 0-1(2)-3(4)-6-8 are characterized by the fact that the cans spend long enough at elevated temperatures to justify the assumption that a large proportion of the cans is likely to fail irrespective of the hoop stresses. One may argue that the cans in the bottom and top fuel assemblies should not fail in view of the fact that they have not been subjected to excessive temperatures. For the sake of consistency with the computations performed for the trip and operator action sequences it was assumed that no cans fail in these two fuel assemblies and that all other cans in fuel assemblies 2-7 fail.

(101) Having established actual probabilities associated with can failures, it became straightforward to calculate free iodine activities available for release simply by multiplying the free iodine inventories in failed cans by the failed can population. On this basis fast trip operator action sequence yielded 9 Ci, slow trip operator action sequence 1.4×10^3 Ci, and trip but no operator action sequence 8.5×10^4 Ci.

(e) Reliability assessment of engineered safeguards

(102) As the fault sequence diagram analysis has shown, the plant

behaviour following the medium size penetration failure is principally governed by the performance of automatic protective and emergency cooling systems and by operator action. The automatic control system influences the events to a very small extent, while a malfunction of the intertripping system is beneficial from the fault transient point of view. The reliability assessment of the automatic protective system has already been described in paragraph 84 and that of the automatic control system in paragraph 86, while that of the intertripping system need not be performed.

(103) Following the gas circulator rundown (inter-tripping system operational), the circulator speed is held at 20% by the low speed pony motors which are supplied from the station electrical system backed up by automatically starting diesel supplies. The initiating signal to switch the low speed pony motors comes from the guard lines and is fed to the motor via a contactor. If the electrical supplies are live, the clutch would engage the motor. Apart from the fact that the low speed pony motors are supplied from the 3.3 kV voltage level (very low speed pony motors are supplied from the 415 V voltage level), the circuits are either identical or alike and therefore the reliability functional block diagram presented in Fig. 30 as well as the assessments discussed in paragraph 85 can be considered to preserve their validity. The effect of starting less than four low speed pony motors has been investigated and proved to have a relatively small influence on the can temperatures. At the time when the circulator speed is held by the pony motors, the reactor is already fully depressurised and the core is subjected to predominant radiative heat transfer processes. A small coolant flow effect is very beneficial as it prevents the can meltout but is not very dependent on the number of

low speed pony motors running.

(104) An interlock system in the design ensures that the high speed pony motors cannot be activated before the reactor pressure has reached a critical value, i.e. 75 secs after the fault onset. Thereafter, the operator is in a position to activate them. However, it is extremely unlikely that he would make his decision very rapidly in view of the fact that at present his operating manual based upon the credible/incredible approach does not even discuss possible actions following a medium size penetration failure. Instead, a procedure following a small size penetration failure would be described recommending a time scale of 15 mins to activate the high speed pony motors. The operator would probably have this time at the back of his mind and consequently it is reasonable to postulate that 15 mins is the most likely time of his action. It is of course possible that he would act before or equally likely after this time. Consideration of all possible times of the operator action is desirable, and an attempt had been made in that direction but the ultimate outcome in terms of fission product release did not vary significantly enough to justify an expansion of this analysis. It was extremely difficult to associate any probability with an operator action, and in particular with the action when the operator under pressure arising from unknown and unexpected circumstances is expected to make a right decision. The Author had decided to adopt an initial speculative probability of 0.9 for the correct action. The reason behind suggesting this high probability of success was a very generous time scale for the action which allows even enough time for consultations with the engineers.

(f) Fault probabilities

(105) Given the initiating event (E_2), the REAXITRAN results unambiguously show that the fault (F_2), in terms of the fission product release to the pressure circuit, can be only avoided by tripping the reactor and by activating the high speed pony motors as soon as possible (75 secs). Only a speculative probability can be associated with this event (see paragraph 104). The following probabilities were assumed as initial ones:

$$P(F_2/E_2) = 0.99$$

$$P(F_o/E_2) = 0.01$$

(106) For the quoted failure rate estimate of 10^{-6} events/year, Table 35 displays the probabilities of any particular number of initiating events. It was postulated that the maximum number of events was 6.

(VI.3) Reactor Start-up Fault

(a) Reliability assessment of the initiating event

(107) A tentative reliability functional block diagram associated with the period freeze system is displayed in Fig. 42. Calculations resembling those summarized in Appendix VII yield the following expression for the probability of the period freeze interlock failure:

$$p(s) = p\left(j + \ell + f_1f_2 + f_1f_3 + f_2f_3\right)$$

since $f_1f_2 + f_1f_3 + f_2f_3 \ll j + \ell$

the overall probability of failure reduces to $p(s) = p(j + \ell)$; in terms of frequencies it reduces to

$$f(s) = 1.5 \times 10^{-3} + 10^{-4} = 1.6 \times 10^{-3} \text{ faults/year}$$

To this frequency a frequency of malfunction in the zone control system should be superimposed. In the absence of any information, a very pessimistic frequency of 0.1 faults/year was assumed. Since the initiating event is a multiple fault composed of the two faults, the frequencies multiply resulting in an overall tentative frequency of 1.6×10^{-4} faults/year for the start-up fault.

(b) Fault development and sequence diagram.

(108) Given that the reactor is brought up to criticality by extraction of Group 3 and that at near criticality the zone group begins to withdraw together with Group 3, the reactor response in terms of neutron power and statistically weighted total reactivity is displayed in Fig. 43. Failure of the automatic control system is the underlying assumption, therefore no response from it can be expected. There are effectively three lines of protection capable of providing a signal to the automatic protective system, i.e. shutdown amplifiers measuring neutron flux, gas outlet temperature trip amplifiers and high coolant pressure switches. The start-up fault develops rapidly, and hence it is probably realistic to assume that the operator (delayed) trip does not provide another effective line of protection. The reactor trip initiates the circulator trip, and following the gas circulator rundown (Fig. 14) the circulator speed is held at 5%

by the shutdown cooling system consisting of very low speed pony motors (barring motors).

(109) A fault sequence diagram exhibited in Fig. 44 adequately represents various possible alternatives following the initiating event. The plant behaviour is by and large governed by the performance of the automatic protective system. The intertripping system behaviour only influences rate of decrease of the reactor core temperatures following the reactor trip. In view of the fact that the fission product power terms are very small during reactivity excursions in the course of the reactor start-up stage, at which the fault was postulated, performance of the heat removal systems becomes immaterial.

(c) REAXITRAN temperature transients

(110) From past experience it was well known that the start-up fault develops very rapidly. Fig. 45 associated with the no trip sequence 0-4-6 illustrates this preconceived notion. Axial plots of the outer ring centre fuel temperatures are displayed in Fig. 46. The negative fuel temperature coefficient of reactivity succeeds only inasmuch as it reduces the divergence rate of the reactor core temperatures.

(111) Having acquired a notion of the time scale of events, the question arose whether all three trip signals could terminate this fault before significant damage to the fuel had occurred. The sequence 0-2-6 was simulated first because REAXITRAN contains a subroutine which automatically trips the reactor as soon as the measured gas outlet temperature in the trip channel reaches the trip setting temperature. The temperature transients depicting this sequence are shown in Fig. 47. From Fig. 45 it became evident that the shutdown

amplifiers represented the first line of protection. A servo-reset rate of, say, 10%/min would terminate the transient in its infancy before the peak can temperature had reached 400°C; thus, the sequences 0-1-5(6) need not be analysed. The APC 1322 program which has already been mentioned was used for an evaluation of the mean coolant pressure transient following the start-up fault. Allowing for an instrumentation delay and postulating a trip margin of 10%, the trip initiation point was reached at the time when the gas outlet temperature has exceeded 475°C (see Fig. 45). This means that the high pressure level protection represents in fact the third line of protection against the start-up fault. Even this relatively slow trip signal would terminate the fault without any fuel damage, and hence likewise the sequences 0-1(2)-5(6) the sequences 0-3-5(6) need not be analysed.

(112) Apart from the no trip sequence 0-4-6, which results in complete reactor fuel meltdown, all other sequences are damage and risk free.

(d) Reliability assessment of engineered safeguards

(113) The fault sequence diagram analysis has shown that the plant behaviour following the initiating event is governed by the performance of the automatic protective system only. Despite the fact that the automatic protective system provides different lines of protection against the start-up fault, the overall probability of failure per demand of 1.3×10^{-5} remains unchanged (see paragraph 84).

(e) Fault probabilities

(114) Given the initiating event (E_3), from the REAXITRAN results it can be unambiguously concluded that the reactor start-up fault

(F_3) in terms of the fission product release to the pressure circuit, would only occur if the automatic protective system has failed to trip the reactor on the time scale of approximately 1 min. Hence,

$$P(F_3/E_3) = q\ (S_1) \simeq 1.3 \times 10^{-5}$$

$$P(F_0/E_3) = 1 - 1.3 \times 10^{-5}$$

(115) For the tentative failure rate estimate of 1.6×10^{-4} faults/year, Table 36 summarizes the probabilities of any particular number of initiating events as a function of the station life, the maximum number of events was fixed at 7.

(VII) APPLICATION OF THE APC COMPOUND MODEL TO AN AGR - STAGE 3 OVERALL SUMMATION OF RELEASES AND SENSITIVITY ANALYSIS

(VII.1) Iodine Retention

(116) The results discussed in Chapter 6 have indicated that several fault sequences yield iodine -131 releases from the fuel. All these sequences together with the retention mechanisms to which released iodine is likely to be subjected on its way to the atmosphere, are summarised in Table 37. In this table the following abbreviations have been used: AC - failure of supply to all circulators; DP - medium size penetration failure; and ST - start-up fault. In addition to the iodine retention mechanisms described in section III.6, fuel and can meltouts have been taken into account. Once released from the fuel, the route that iodine follows is mainly influenced by:

(a) the plant design and layout;

(b) nature of the initiating event;

(c) prevailing conditions of reactor core temperatures and coolant pressures.

Given the fault sequences quoted in Table 37, the following applies:

(a) released iodine does not pass through an adsorption plant on its way to the atmosphere;

(b) the concrete pressure vessel acts only as a part of the pressure circuit and not as a conventional containment;

(c) in a pressurised reactor iodine leaks out from the reactor system via safety valve vent pipes, and in a depressurised reactor via a failed penetration through the pressure vessel.

In the case of a single channel meltout, for instance, the concrete pressure vessel acts as a containment and released iodine passes through an adsorption plant before reaching the high stack.

(117) Iodine released in the sequences resulting in the fuel meltout, i.e. AC1, DP1 and ST1, is subjected to the UO_2 meltout retention factor. A factor of 1.0846 has been suggested in reference 78. To the best of the Author's knowledge, nobody has yet predicted precisely what happens to a fuel stringer following the can meltout (the sequences AC2, AC3, DP2 result in complete or partial can meltout). From the point of view of iodine release the worst assumption is that UO_2 melts as well. This assumption is obviously too pessimistic, so in the absence of any information a tentative factor of 2 was assumed for the calculations. With reference to paragraph 36, iodine released in the sequences DP3, DP4 and DP5 is subjected to a retention factor of 3.

(118) The simple rate equation depicting the iodine adsorption processes in the pressure circuit is not valid for the sequences quoted in Table 37. Good mixing of the gas in the primary circuit cannot be accomplished before it escapes through the vessel penetrations (the safety valves in this context are treated as any other penetrations). However, before escaping, a portion of the gas should pass over suitable deposition surfaces; boiler tubes in particular. It is reasonable to assume that the gas would make several passes through the boilers before escaping via the safety valve penetrations because of their small size and to much lesser extent because of their position below the boilers. The same assumption cannot be applied to the gas escaping through a medium size penetration. A single pass was postulated for the latter case (this results in a factor of 1.4) and a number of passes for the former case to justify a speculative factor of 10.

(119) The gas escaping via the safety valve penetrations passes through the filters and up the long vent pipe. It accelerates as it travels up the pipe and the retention factor depends on the time spent in the pipe. For the calculated travel time of 2 secs, a tentative retention factor of 2 has been assumed. The gas escaping via a medium size penetration passes through the reactor building where iodine becomes subjected to the adsorption processes discussed in paragraph 39. The reactor building volume, $V = 6.5 \times 10^4\ m^3$; corresponding surface area, $S = 6 \times 10^4\ m^2$ and mean deposition velocity of 3.8×10^{-4} m/sec,[48] give the half life for the removal of iodine by deposition of $T_{\frac{1}{2}} = \frac{0.693}{\lambda} = 1974$ secs or approximately 33 mins. This means that a retention factor of 2 applies if the gas stays in the building effectively for 33 mins. If however it stays for, say, 20 mins the retention factor would be 1.52; for 10 mins it would be 1.23 and for 2 mins 1.07. It was thought that the first case would be nearest to the real situation.[79]

(VII.2) Summation of Releases

(120) All input parameters to the fault consequence submodels have been assumed to be discrete so far and the fault sequence diagram analyses have resulted in a number of curie releases; these facts correspond to the case (b)(see paragraph 44). An underlying assumption behind that approach us that input data uncertainties are small and therefore can be ignored. One can argue that the set of REAXITRAN input data complies reasonably well with this assumption, but the same cannot be said for the iodine retention factors, for instance. Consequently, an alternative would be to assume that all input data are discrete apart from some iodine retention factors which are better represented by probability distributions. Both approaches have been applied to the fault sequences summarized in Table 37.

(121) All discrete input data approach yields two different magnitudes of releases for the AC sequences, four for the DP sequences and a single release for the ST sequences (see Table 37). An initiating event generally results in a single outcome, therefore all releases once having been grouped should be weighted against each other. In the absence of any supplementary reliability analysis results, guessed factors quoted in Table 38 have been fed into the STATMOD run number 12. The resultant probability distributions of curie release of iodine -131, for 5 and 30 yrs of station life operation, are displayed in Fig. 48. At 10^6 Ci the curves violate Cave's criterion. A brief analysis of the part of STATMOD output containing the generating functions has shown that AC sequences contribute most to the violation of the criterion. The weighting factors were tentatively changed (see Table 38), but this fact has not altered the results significantly as shown in Fig. 48.

(122) For the reasons explained in paragraph 44, a number of input parameters represented by probability distributions rather than by discrete values should be kept to a minimum. From Table 37 it became obvious that the pressure circuit retention factor influences most the magnitude of iodine releases to the atmosphere; therefore it was decided to vary this factor and keep the others constant. A range of 10-100 was assumed to be valid in a pressurized reactor and 1.4-8.4 in a depressurized reactor. An upper limit of 100 in a pressurized reactor was based upon the fact that the safety valves having been lifted might reseat and thus allow a significant number of passes round the pressure circuit to be made. It was further decided to represent the random pressure circuit retention factor with the three discrete values; this resulted in a number of releases summarized in Table 39. These releases can be now grouped either by:

(a) applying simple linear weighting formula (see Appendix VI), or

(b) attributing pre-assessed probabilities to various fault sequences. The two approaches yielded the probabilities quoted in Tables 41 and 42 and inputted to the STATMOD runs 14 and 16, respectively. The outputted probability distributions of iodine release are displayed in Fig. 49. The latter approach yielded a smaller risk; nevertheless at 10^6 Ci Cave's criterion was breached again.

(123) Apart from determining an overall probability distribution of curie release, the STATMOD outputs mean curie releases and probabilities of releases being less/greater than a specified decade (see paragraph 45). The STATMOD R16 has yielded the probabilities quoted in Table 42 and the value of 0.251 Ci/year for the mean release over the 30 yrs. The ICRP regulations specify a maximum breathing air concentration of iodine -131 for the general population of 3×10^{-10} μCi/cm^3. A reactor releasing an average of 28.3 m^3/sec of air to the atmosphere with the above concentration of I -131 would release about 0.27 Ci/year.[58]

(VII.3) Sensitivity Analysis

(124) Given a fault sequence resulting in iodine release to the atmosphere, actual magnitude of the release is dependent upon quantities associated with hundreds of input data parameters to the constituents of the compound model. From the magnitude of release point of view, some input data variations have small or even negligible effect but on the other hand even small variations in some of the input data parameters are distinctly measurable. In terms of their relative contribution to the magnitude of the outcome all input data parameters can be classified into, say: highly sensitive, moderately sensitive, fairly insensitive and highly insensitive. Having identified highly sensitive parameters, one

can suggest changes in the design which could result in a reduced radiological risk. The APC compound model and in particular the STATMOD program are ideally suited for the purpose of conducting this, so called, sensitivity analysis.

(125) It has already been shown that ultimate releases are very much affected by the iodine retention processes, hence some iodine retention factors qualify to be classified as highly sensitive input data parameters . In Table 37 a factor of 10 (STATMOD R12) for the pressure circuit retention was replaced by a factor of 100 (STATMOD R13); the probabilities obtained are compared in Table 43. Violation of Cave's criterion at 10^6 Ci can be avoided providing the gas is given a chance to make a significant number of passes over the suitable surfaces. Alternatively, iodine absorbers can be provided on the safety valves. The radiological risk level was not affected much when the frequency of the initiating event was reduced to 0.12 events/year (STATMOD R15) as shown in Table 43; thus this input parameter can be classified only as a moderately sensitive one.

(126) Another highly sensitive input parameter if, not the most sensitive one, is the probability of the automatic protective system to shut the reactor down when called upon to do so. The reliability analysis presented in Appendix VII has deliberately ignored a possibility of the common mode failures [80] for the three main reasons:

(a) the phenomenon is not yet fully understood;

(b) no mathematical formulation has yet been offered;

(c) the suggested rates of 10^{-2} - 10^{-3} faults/year yield alarming results.

At this stage in the analysis their effect can be taken into account.

Assuming that the exponential laws are valid, the quoted failure rates yield probabilities of failure of 10^{-3} - 10^{-4}. These probabilities then become predominant and the assessed probability of random failures of 1.3×10^{-5} can be practically ignored. For the sake of illustration the STATMOD R12 was repeated with a probability of 10^{-3} instead of 1.3×10^{-5}; the probabilities of releases obtained (STATMOD R17) quoted in Table 44 are obviously totally unacceptable. One possible way out of this situation would be to design two entirely independent automatic protective systems. In such case a failure probability of 10^{-3} can be tolerated as the overall probability would be 10^{-6}. This case has been investigated (STATMOD R18) and the results are displayed in Table 44 and Fig. 50. Cave's criterion was breached again because the probability of failure (assumed value 2.7×10^{-6}) of the emergency cooling system dominated the hazard. Referring to paragraph 86 it would not be unrealistic to replace 2.7×10^{-6} by, say, 10^{-7} (STATMOD R19). This improves the situation significantly (Fig. 50), but the danger of violation of the criterion had not yet been removed. The next run (STATMOD R20) assumed that both crucial probabilities were equal to 10^{-7}. Breaching of the criterion for all faults is now avoided (see Fig. 50), but full satisfaction cannot yet be claimed because the probability associated with 10^{6} Ci release due to the failure of supply of all circulators is still above the individual fault risk line. The point of full satisfaction was reached when both probabilities were assumed to be 10^{-8} (STATMOD R21).

(127) The DP sequences contribute most to the values of the overall probability distributions at 10^{4} and 10^{7} Ci (see Fig. 50). These two points are well below Cave's criterion. It was thought that this was due to a very low fault frequency of 10^{-6} events/year. This frequency

was increased to 10^{-5} events/year in STATMOD R22 and the other parameters preserved the values specified for the STATMOD R12. Obtained probabilities of 5.94×10^{-5} and 2.97×10^{-9} respectively were again well below the criterion for individual faults. A further increase to 10^{-4} events/year (STATMOD R23) resulted in the probabilities of 5.92×10^{-4} and 3.03×10^{-8} respectively which both breached the criterion for individual faults.

(VIII) SUMMARY AND CONCLUSIONS

(VIII.1) Summary

(128) This Thesis first describes the conventional fault analysis method (CFAM) together with its shortcomings and the methods of calculation used for Magnox reactors and AGR's. A brief review of an alternative, in terms of the statistical fault analysis method (SFAM) is followed by a suggestion that the compound model proposed can be used as an executive tool for the implementation of the statistical fault analysis concept. A detailed description of the Author's APC compound model follows, which incorporates existing analytical models for fault consequence evaluation of AGR's and adds to them quantitative assessments of reactor design, operating and maintenance procedures and engineered safeguards. In its present form the end product of the model is a probability distribution of curie release of iodine-131 as a function of time during the station life; thus the model offers a means for assessing safety of gas cooled graphite moderated reactors in quantitative terms. It is also clearly indicated how the present version of the compound model can be modified so that it can cope with high temperature gas cooled reactor systems, where it is possible that the release to atmosphere of strontium isotopes or active rare gases could be the dominating hazard in some fault conditions. A comparison between the APC and the American compound models is followed by a general classification of faults relevant to AGR's.

(129) In order to illustrate an application of the APC compound model concept a small statistical fault analysis study was conducted which embodied three characteristic AGR faults; failure of supply to all circulators, medium size penetration failure and reactor start-up fault.

The ultimate summated outcomes of the study are compared against Cave's release criteria. Finally another form of application of the compound model, in terms of sensitivity analysis, is illustrated.

(VIII.2) Conclusions

(130) The forthcoming conclusions can be grouped into those relating to the SFAM in general and its applications as discussed in this Thesis and those that have been deduced from the outcomes of the statistical fault analysis study which has been performed.

(131) The accuracy and reliability of the results of any analytical investigation are highly dependent on:

(a) the validity of the assumptions made in formulating the mathematical models simulating actual physical processes, and

(b) the availability and validity of input data information.

Some critics of the SFAM argue against adoption of the method because of the apparent lack of available statistics concerning the performance of various reactor components, and in particular structural components. In addition, certain limitations in the application of the SFAM listed below may have resulted in a hesitant attitude of the licensing authorities:

(a) the lack of understanding of fission product transport under fault conditions, and in particular of iodine retention mechanism;

(b) the inadequate understanding of gas dynamics associated with the release of iodine;

(c) the lack of reliable information on can failure mechanisms and fission product movements within the can.

(132) It is undeniable that it is virtually impossible to establish probability numbers which have an absolute meaning for the majority of structural components, e.g. concrete pressure vessel, internal pressure dome and cylinder, core support structure, boiler shield wall, etc. This is simply because the failure of such components is so highly dependent on individual circumstances, which are so seldom similar, that meaningful statistics are just not available and probably will not be available for some years to come. Few truly relevant statistics have been compiled for operator actions, and such statistics usually deal with very simple and predictable situations. Pessimistic probabilities of initiating events and pessimistic iodine retention factors can convert a tolerable fault into one which is utterly unacceptable. It is needless to say that it would be ideal if all input data quantities were known precisely; the amount of work would be then tremendously reduced, and it would be much easier to arrive at conclusions evident even to non-experts. However, as the situation is not like this, the method presented in this work accepts the fact that some input parameters are known, and probably will be known for some years to come, only to an accuracy of one or even two orders of magnitude; it offers an option to represent these parameters in the form of probability distributions rather than using discrete values. For practical reasons a number of quantities represented by probability distributions should be kept to a minimum. In the statistical fault analysis study which has been performed only iodine retention factors were represented by probability distributions, as illustrated in Chapter VII. Possible

optimistic tones in the outcome due to either underestimation of initiating event frequencies or possible imperfections in the formulation of mathematical models are counteracted by a pessimistic treatment of some essential reactor core parameters, e.g. the treatment of channel powers, fission product gas pressures and fractional release rates as described in Chapter V. Additional pessimism is built in as the method does not allow for either prolonged plant shutdowns or possible reactor write-offs following the occurrence of a major fault.

(133) A frequent criticism of the SFAM is its computational cost and an associated man power-effort involved. Undeniably, owing to the thoroughness of SFAM or perhaps to the superficiality of the CFAM, the amount of work will be initially significantly increased. However, even in the first safety report based upon the SFAM, the total number of possible sequences can be drastically reduced simply by applying common sense and existing experience. An analysis of the start-up fault (Chapter VI) could serve as a very good illustration. In addition, the first safety report will show that some low frequency initiating events (medium size penetration failure for instance) contribute very little to the ultimate probability distribution of release, so it will be probably justified to omit these faults from a list of faults to be analysed in a forthcoming safety report relevant to the same reactor type. Some of these faults might even belong to the types usually analysed in safety reports based upon the CFAM.

(134) Having dealt with some criticisms of the SFAM, it is worthwhile now to summarise its main straightforward advantages over the CFAM as seen by the Author:

(a) It incorporates reliability analysis as a means of assessing the performance of different engineering structures, systems and safeguards, as illustrated in Chapter VI. Reliability analysis has already become common place in the aero-space and communications industries.

(b) It enables iterative optimisation studies to be carried out between different variants of the same reactor system, aimed at achieving a balance between reactor safety and economy, as it can determine the effect on the hazard of variation in reactor design, including the engineered safeguards. The sensitivity analysis presented in Chapter VII serves as an example how this can be done.

(c) Starting with the premise that particular quantitative safety criteria are to be met, it stimulates parameter survey studies aiming at defining the performance required from important areas of the plant and detecting weaknesses at the outset of the development of a station design. Later in the design these studies provide a kind of safety feedback to any proposed changes in the design. In this respect the compound model becomes a tool similar to various critical path analysis programs based upon the PERT (program evaluation and review technique) concept. Results discussed in Chapter VII illustrate to some extent this aspect of the SFAM.

(d) It enables comparisons to be made between the risk being generated by the growing number of nuclear power stations and socially acceptable risks for which the statistics are readily available, e.g. road injuries, death from influenza, injuries caused by aircraft crashes or railway accidents, fatalities of all kinds from gas, etc.

(e) In addition to reliability analysis it widely opens doors to other new fields of investigation, e.g. fission product transport under accident conditions, fields which may never be fully understood otherwise.

(f) It provides a means of criticising present designs and provides guidance as to how the present impeccable nuclear safety record can be maintained in the future despite the increasing numbers of nuclear power stations. This aspect is best illustrated in the following paragraph.

(135) From the outcome of the limited statistical fault analysis study which is presented in this Thesis, and which has been carried out only for the sake of an illustration of the SFAM developed by the Author several important conclusions relevant to AGR's currently under construction, are suggested:

(a) The plant behaviour following the majority of conceivable initiating events is mainly dependent on the performance of automatic protective and shutdown cooling systems; this fact suggests that detailed reliability analysis studies should be predominantly orientated towards better understanding of the characteristics of the constituents of which the two systems are composed.

(b) The overall hazard is dominated by relatively high frequency initiating events, such as the ones leading to the failure of supply to all circulators, coupled with either malfunction of the automatic protective system or to a smaller extent of the shutdown cooling system.

(c) Detailed studies may show a difficulty in meeting the presently proposed safety criteria, which place emphasis on avoiding large releases of iodine-131 of the order of 10^5-10^7 Ci, owing to high probability of failure per demand of the automatic protective system. Even if the common mode failures are not taken into account, the random probability of failure assessed in the present indicative study of approximately 10^{-5} is too high, as illustrated in Chapter VII.

(d) If the present safety record is to be maintained, the indicative results suggest that more detailed consideration might well be given to the possibility that future nuclear power plant designs should be modified, possibly by duplicating the automatic protective system, in order that the probability of failure per demand can be reduced to 10^{-7}-10^{-8}; and by putting iodine absorbers on the safety valves in order to augment the iodine retention factors against faults in a pressurised reactor. A permanent solution would be to redesign the fuel in order to ensure the overall negative coefficient of reactivity throughout the reactor core life.

All these conclusions may be influenced to some extent by the limited nature of the study, and also by intrinsic pessimisms such as those discussed in paragraph 132.

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APPENDIX I

REAXITRAN PROGRAM EQUATIONS

HEAT TRANSFER EQUATIONS

Heat Generation Terms in Fuel Stringer Rings

$$H_{ui} = \{\alpha_{ui} P_n + \mu_{ui} P_{fp}\} \frac{\Phi \, \Omega \, E}{L_1 \, L_2} \qquad i = 1,---N \tag{1}$$

Heat Generation Terms in Moderator Regions

Inner sleeve:

$$H_{s1} = \{\alpha_{s1} P_n + \mu_{s1} P_{fp}\} \, \Phi \, \Omega \, E \quad + O_{s1} \tag{2}$$

Outer sleeve:

$$H_{s2} = \{\alpha_{s2} P_n + \mu_{s2} P_{fp}\} \, \Phi \, \Omega \, E \quad + O_{s2} \tag{3}$$

Bulk moderator:

$$H_m = \{\alpha_m P_n + \mu_m P_{fp}\} \, \Phi \, \Omega \, E \quad + O_m \tag{4}$$

Fuel Stringer Temperatures

Centre uranium:

$$T_{uci} = T_{ui} + H_{ui}/U_{uci} \tag{5}$$

Mean uranium:

$$q_{ui} \frac{\partial T_u}{\partial t} = H_{ui} - U_{uai} (T_{ui} - T_{ai}) \tag{6}$$

Surface uranium:

$$T_{usi} = T_{ai} + U_{usi} (T_{ui} - T_{ai}) \qquad i = 1,---N \tag{7}$$

Can temperatures:

$$q_{ai} \frac{\partial T_{ai}}{\partial t} = L_1 U_{uai} (T_{ui} - T_{ai}) - h_{aic} f_a (T_{ai} - T_c) -$$

$$\sum_{\substack{j=1 \\ i \neq j}}^{N} R_{aij} \{(T_{ai} + 273.2)^4 - (T_{aj} + 273.2)^4\} - R_{ais}$$

$$\{(T_{ai} + 273.2)^4 - (T_{s1} + 273.2)^4\} \quad i = 1, ---N \tag{8}$$

Coolant temperatures

Main coolant:

$$W_c J_c \frac{\partial T_c}{\partial z} = L_2 f_a \sum_{i=1}^{N} h_{aic} (T_{ai} - T_c) + h_{sc} f_{s1} (T_{s1} - T_c) \tag{9}$$

Annulus coolant:

$$- W_{ac} J_{ac} \frac{\partial T_{ac}}{\partial z} = h_{sac} f_m (T_{s2} - T_{ac}) + h_{mac} f_m (T_{ma} - T_{ac}) \tag{10}$$

Re-entrant passage coolant:

$$- W_{rc} J_{rc} \frac{\partial T_{rc}}{\partial z} = h_{mrc} f_m (T_{mr} - T_{rc}) \tag{11}$$

Boundary condition at mixing point:

$$T_c = \frac{1}{J_c W_c} \{J_{ac} W_{ac} T_{ac} + J_{rc} W_{rc} T_{rc} + J_b W_b T_b\} \tag{12}$$

Graphite Temperatures

Inner sleeve:

$$G_{s1}\, q_{s1}\, \frac{\partial T_{s1}}{\partial t} = H_{s1} - U_{ss}\,(T_{s1} - T_{s2}) - h_{sc}\, f_{s1}\,(T_{s1} - T_{c}) +$$

$$L_2 \sum_{i=1}^{N} R_{ais}\,\{(T_{ai} + 273.2)^4 - (T_{s1} + 273.2)^4\} -$$

$$R_{ss}\{(T_{s1} + 273.2)^4 - (T_{s2} + 273.2)^4\} \qquad (13)$$

Outer sleeve:

$$G_{s2}\, q_{s2}\, \frac{\partial T_{s2}}{\partial t} = H_{s2} + U_{ss}\,(T_{s1} - T_{s2}) - h_{sac}\, f_{m}\,(T_{s2} - T_{ac}) +$$

$$R_{ss}\,\{(T_{s1} + 273.2)^4 - (T_{s2} + 273.2)^4\} -$$

$$R_{sm}\,\{(T_{s2} + 273.2)^4 - (T_{ma} + 273.2)^4\} \qquad (14)$$

Inner edge of main moderator:

$$U_{mam}\,(T_{m} - T_{ma}) - h_{mac}\, f_{m}\,(T_{ma} - T_{ac}) +$$

$$R_{sm}\,\{(T_{s2} + 273.2)^4 - (T_{ma} + 273.2)^4\} = 0 \qquad (15)$$

Mean main moderator:

$$G_{m}\, q_{m}\, \frac{\partial T_{m}}{\partial t} = H_{m} - U_{mam}\,(T_{m} - T_{ma}) - U_{mmr}\,(T_{m} - T_{mr}) \qquad (16)$$

Outer edge of main moderator:

$$U_{mmr}\,(T_{m} - T_{mr}) - h_{mrc}\, f_{m}\,(T_{mr} - T_{rc}) = 0 \qquad (17)$$

Mean Fuel and Moderator Temperatures

Fuel:

$$T_U = \sum_{i=1}^{N} F_i \, T_{ui} \tag{18}$$

Moderator:

$$T_M = F_{s1} + F_{s2} \, T_{s2} + F_m \, T_m + F_a \sum_{i=1}^{N} T_{ai} \tag{19}$$

Reactivity Change due to Temperature Effect

$$\delta k_T = a_u \, T_U + a_m \, T_M - \delta k_T \, (0) \tag{20}$$

Wigner Energy Terms

$$G_{s1} = \begin{cases} 1 \; ; & T_{s1} - T_{s1} \, (0) < T_{ws1} \\ w_{s1} ; & T_{s1} - T_{s1} \, (0) \geqslant T_{ws1} \end{cases} \tag{21}$$

$$G_{s2} = \begin{cases} 1 \; ; & T_{s2} - T_{s2} \, (0) < T_{ws2} \\ w_{s2} ; & T_{s2} - T_{s2} \, (0) \geqslant T_{ws2} \end{cases} \tag{22}$$

$$G_m = \begin{cases} 1 \; ; & T_m - T_m \, (0) < T_{wm} \\ w_m \; ; & T_m - T_m \, (0) \geqslant T_{wm} \end{cases} \tag{23}$$

Graphite Oxidation Terms

$$O_{s1} = \eta_{s1} \, q_{s1} \, B_{s1} \, \gamma \, K \, e^{-b/(T_{s1} + 273.2)} \tag{24}$$

$$O_{s2} = \eta_{s2} \, q_{s2} \, B_{s2} \, \gamma \, K \, e^{-b/(T_{s2} + 273.2)} \tag{25}$$

$$O_m = (\eta_{ma} + \eta_{mr}) \, q_m \, B_m \, \gamma \, K \, e^{-b/(T_m + 273.2)} \tag{26}$$

NEUTRONIC EQUATIONS

Normalised Neutron Power:

$$\frac{dP_n}{dt} = \frac{\delta k_{SW} - \beta}{l} \quad P_n + \sum_{j=1}^{6} \lambda_j D_j + S \tag{27}$$

Delayed Neutron Group Concentrations:

$$\frac{dD_j}{dt} = \frac{\beta_j}{l} \; P_n - \lambda_j D_j \qquad j = 1,2,3 \ldots 6 \tag{28}$$

Reactivity Balance:

$$\delta k_{SW} = \delta k_{SWT} + \delta k_{SWX} + \delta k_{SWZ} + \delta k_{SWTO} + \sum_{j=1}^{4} \delta k_{SWBj} + \delta k_{TR} \tag{29}$$

Normalised Fission Product Power:

$$\frac{\partial p_{fpj}}{\partial t} = p\, \Psi_j \, P_n - \delta_j \, p_{fpj} \qquad j = 1,2,3,4 \tag{30}$$

$$P_{fp} = p'(1 - \sum_{j=1}^{4} \Psi_j) \;\; P_n + \sum_{j=1}^{4} \delta_j \, p_{fpj} \tag{31}$$

Iodine and Xenon:

$$\frac{\partial I}{\partial t} = - \lambda_I I \;\; + y_I \, \Sigma_f \;\; \varphi \tag{32}$$

Iodine and Xenon: - Continued

$$\frac{\partial X}{\partial t} = \lambda_I I - \lambda_X X - \sigma_X X \varphi + y_X \Sigma_f \varphi \tag{33}$$

Neutron Diffusion:

$$M^2 \frac{d^2\Phi}{dz^2} + [(K_\infty - 1) + \delta k + \nu] \Phi = 0 \tag{34}$$

Reactivity Balance:

$$\delta k = \delta k_T + \delta k_X + \delta k_Z + \delta k_{TO} + \sum_{j=1}^{4} \delta k_{Bj} \tag{35}$$

Reactivity Changes due to Control Rod Movement:

$$\delta k_l = \rho_l (Z_l) - C_l (Z_l (0)) \tag{36}$$

$$\rho_l = \begin{cases} 0 \; ; & z_l \leqslant -1 \\ (0.25 z_l^3 - 0.75 z_l - 0.5) C_l ; & -1 < z_l < 1 \\ -C_l \; ; & z_l \geqslant 1 \end{cases} \tag{37}$$

$z_l = k - m Z_l$ $\qquad l = Z, TO, B_j \; j = 1,2,3,4$

Statistically Weighted Reactivity Changes

$$\delta k_{SWl} = \frac{\sum_{k=0}^{m} \delta k_l \Phi^2}{\sum_{k=0}^{m} \Phi^2} \tag{38}$$

$l = T, X, Z, TO, B_j \qquad j = 1,2,3,4$

CONTROLLER AND REACTOR TRIP EQUATIONS

Measured gas outlet temperature

$$T_{COm} + (\tau_1 + \tau_2) \frac{dT_{COm}}{dt} + \tau_1 \tau_2 \frac{d^2 T_{COm}}{dt^2} = T_{CO} + r\tau_2 \frac{dT_{CO}}{dt} \tag{39}$$

$$\tau_l = u_l + v_l (1 - W_c) \qquad l = 1, 2 \tag{40}$$

Double lead lag/phase advance:

$$\epsilon = T_{COm} - T_{COD} \tag{41}$$

$$x + (\tau_5 + \tau_6) \frac{dx}{dt} + \tau_5 \tau_6 \frac{d^2 x}{dt^2} = \epsilon + (\tau_3 + \tau_4) \frac{d\epsilon}{dt} + \tau_3 \tau_4 \frac{d^2 \epsilon}{dt^2} \tag{42}$$

$$\tau_l = u_l + v_l (1 - W_c) \qquad l = 3, 4, 5, 6 \tag{43}$$

Controller characteristic:

$$\frac{d\theta}{dt} = \begin{cases} 0; & |x| \leqslant d \\ -\,\mathrm{SIGN}(x) \dfrac{|x| - d}{s - d} V_{Zmx}; & d < |x| \leqslant s \\ -\,\mathrm{SIGN}(x)\, V_{Zmx}; & |x| \geqslant s \end{cases} \tag{44}$$

Mechanism lag:

$$\theta = Z_Z + \tau_7 \frac{dZ_Z}{dt} \tag{45}$$

Trimming action:

$$\Delta = Z_Z - \grave{Z}_{ZR} \tag{46}$$

$$\frac{dZ_{TO}}{dt} = \begin{cases} 0; & |\Delta| < d' \\ \mathrm{SIGN}(\Delta) \dfrac{|\Delta| - d'}{\varkappa}; & |\Delta| > d' \end{cases} \tag{47}$$

Trip initiation:

$$\frac{dT_{TR}}{dt} = \begin{cases} \dfrac{dT_{COm}}{dt} \; ; & \dfrac{dT_{COm}}{dt} < \zeta \\ \\ \zeta \; ; & \dfrac{dT_{COm}}{dt} \geqslant \zeta \end{cases} \tag{48}$$

$$T_{TR}\,(0) = \Delta T + T_{COD} \tag{49}$$

NOMENCLATURE

Symbols

a	reactivity coefficients
b	activation energy/universal gas constant
d	controller half dead band
d'	half dead band for trim/override rods
e	natural logarithm
f	flow functions (variation of convective heat transfer coefficients with flow)
h	convective heat transfer terms
k	current index for axial mesh points
k_∞	infinite neutron multiplication factor
δk	reactivity change
l	mean neutron life time
m	upper mesh point
p	$P_{fp}(0)/P_n(0)$
q	thermal capacities
r	attenuation factor
s	half controller saturation band
t	time
u	components of controller time constants
v	components of controller time constants
w	specific heat relative drops due to Wigner energy effect
x	controller action signal
y	fission yields
z	axial distance from rod tip in units of ΔZ
B	graphite reactivities
C	control rod worth
D	delayed neutron group concentrations
E	channel ratings per unit length
F	weighting functions
G	Wigner energy terms
H	heat generation terms
I	iodine concentration
J	mass flow specific heat product
K	heat of graphite oxidation reaction
L_1	total fuel length/total can length
L_2	total can length/total channel length

NOMENCLATURE - cont'd

Symbols

M^2	axial migration area
N	number of fuel stringer rings
O	heat generation terms due to graphite oxidation
P	normalised powers
R	radiative heat transfer terms
S	neutron source term
T	temperatures
ΔT	trip margin on gas outlet temperature
U	conductive heat transfer terms
V	control rod speeds
W	normalised mass flows
X	xenon concentration
Z	fractional withdrawal of control rods
(0)	steady state values
α	fractions of neutron power
β	yield fraction of delayed neutron groups
γ	factor for air/CO_2 mixture
δ	coefficients associated with fission product groups
ϵ	controller error signal
ζ	servo reset trip
η	fractions of graphite volume oxidising
θ	signal to control rod motor
φ	absolute neutron flux
$\varkappa$	proportionality constant
λ	decay constants (delayed neutrons; xenon; iodine)
μ	fractions of fission product power
ν	eigenvalue
ρ	reactivity
σ	microscopic cross section
τ	time constants
τ_1	thermocouple lag time constant
τ_2	thermocouple surrounding material time constant
τ_3	first lead time constant
τ_4	second lead time constant
τ_5	first lag time constant

NOMENCLATURE - cont'd

Symbols

τ_6	second lag time constant
τ_7	mechanism lag time constant
Φ	axial neutron flux shape
Ψ	fractional constants for fission product power groups
Δ	error signal for trim/override rods
Σ	macroscopic cross section
Ω	ratios between axial neutron flux shape and axial heat production shape

Subscripts

a	can
b	bypass coolant
c	main coolant
f	fission
i	current index for fuel stringer rings
j	current index for fission product groups; delayed neutron groups; bulk rod groups; fuel stringer rings
l	current index for zone, trim/override and bulk rods; xenon and temperature effect
m	main moderator
n	neutron
u	uranium
w	due to Wigner energy release
ac	annulus coolant
aic	can to main coolant
aij	can ring to ring
ais	can ring to sleeve
as	can to sleeve
fp	fission product
ma	main moderator inner edge
mr	main moderator outer edge
mx	maximum
mac	moderator to annulus coolant
mam	moderator to inner moderator edge
mmr	moderator to outer moderator edge
mrc	moderator to re-entrant coolant
rc	re-entrant coolant
s1	inner sleeve
s2	outer sleeve
sc	inner sleeve to main coolant
sm	outer sleeve to main moderator
ss	inner sleeve to outer sleeve
sac	outer sleeve to annulus coolant
ua	uranium to can
uc	centre uranium
us	surface uranium

Subscripts - cont'd

B	bulk rods
I	iodine
M	mean moderator
T	temperature effect
U	mean fuel
X	xenon
Z	zone rods
CO	coolant outlet
COD	coolant outlet demanded
COm	coolant outlet measured
SW	statistically weighted
SWB	statistically weighted bulk rods
SWT	statistically weighted temperature effect
SWX	statistically weighted xenon effect
SWZ	statistically weighted zone rods
SWTO	statistically weighted trim/override rods
TO	trim/override
TR	trip
ZR	reference insertion of zone rods

APPENDIX II

FISSION PRODUCT ACTIVITY EQUATIONS

Fission Product Decay Chain

$$\frac{dN_1}{dt} = S_1 - \lambda_1 N_1 \qquad (1)$$

$$\frac{dN_2}{dt} = S_2 + \lambda_1 N_1 - \lambda_2 N_2 \qquad (2)$$

$$\vdots$$

$$\frac{dN_j}{dt} = S_j + \lambda_{j-1} N_{j-1} - \lambda_j N_j \qquad (3)$$

Approximate Expressions for Fission Product Inventories

$$A_j = 8.4 Y_j H \left\{ 1 - \exp\left(-\lambda_j t\right) \right\} \qquad (4)$$

When the radioactive mean life is short in comparison with the irradiation time the activity rapidly reaches the equilibrium value of:

$$A_j = 8.4 Y_j H \qquad (5)$$

NOMENCLATURE

A_j - activity of isotope j

H - rating

N_j - number of atoms of jth member of decay chain (j = 1,2...)

λ_j - decay constant for jth member

S_j - rate at which atoms of jth member are being produced by fission

t - time

Y_j - fission yield of isotope j

APPENDIX III

FISSION PRODUCT TRANSPORT EQUATIONS

Transient Fission Product Diffusion

$$\frac{dC}{dt} = D\nabla^2 C - \lambda C + B \tag{1}$$

For diffusion in a sphere with spherical geometry

$$\frac{dC}{dt} = \frac{D}{r}\,\frac{d^2C}{dr^2} + B - \lambda C \tag{2}$$

Diffusion Coefficient

$$D^1 = D_{RT} \exp\{-E/R(T + 273.2)\} \tag{3}$$

Fission Gas Release by Diffusion

$$f_d = 4\sqrt{(D^1\theta/\pi)} - 3/2\ (D^1\theta) \tag{4}$$

Fission Gas Release by Grain Boundary Sweeping

$$f_s = f_{mx} \left\{1 - \frac{1}{S\theta}\left[1 - \exp(-S\theta)\right]\right\} \tag{5}$$

Fission Product Pressure

$$P = \frac{\sum_{j=1}^{n} Y_j f_j L}{6.02 \times 10^{23}} \times \frac{R\ (T + 273.2)}{V} \tag{6}$$

Nomenclature

Symbols

B -	source of fission products
C -	fission product concentration
D, D^1 -	fission product diffusion coefficients
E -	Fission product activation energy for diffusion
f -	fractional release
L -	total number of fissions up to θ
λ -	decay constant
P -	fission product pressure
r -	radius
R -	gas constant
S -	sweep parameter
t -	time
T -	fuel temperature
θ -	irradiation time
V -	total void volume
Y -	fission yield

Subscripts

d -	diffusion
j -	current index for fission products
mx -	maximum
n -	number of fission product groups
RT -	reference temperature
s -	sweep

APPENDIX IV

IODINE RETENTION EQUATIONS

Iodine Concentration in Adsorption Processes

$$\frac{dI}{dt} = \frac{\nu S}{V} I \tag{1}$$

$$I(t) = I(o) \exp(-\lambda t) \tag{2}$$

Half Life

$$T_{\frac{1}{2}} = \frac{0.693}{\nu} \times \frac{V}{S} \tag{3}$$

Pressure vessel leakage rate

$$L(t) = L_{mx} \left\{ \frac{R(T+273.2)}{R_{mx}(T_{mx}+273.2)} \times \frac{1-(p_b/p)^2}{1-(p_b/p_{mx})} \right\}^{\frac{1}{2}} \tag{4}$$

Nomenclature

b	-	reactor building
I	-	iodine concentration
I(o)	-	iodine concentration at t=0
L	-	leakage rate
λ	-	removal constant
mx	-	maximum
ν	-	deposition velocity
p	-	pressure
R	-	gas constant
S	-	surface
T	-	gas temperature
$T_{\frac{1}{2}}$	-	half life (=0.693/λ in secs)
t	-	time
V	-	removal volume

APPENDIX V

FAULT PROBABILITY MODEL EQUATIONS

Conditional probability of fault F_j on the hypothesis of initiating event E_i

$$P(F_j/E_i) = \sum_{m=1}^{M} q\,\{S_m(E_i)\}\,.\;P\,\{F_j/S(E_i)\} \qquad (1)$$

Conditional probability of no fault on the hypothesis of initiating event E_i

$$P(F_o/E_i) = 1 - \sum_{m=1}^{M} q\,\{S_m(E_i)\}.\left[1-P\,\{F_o/S(E_i)\}\right] \qquad (2)$$

Probability distribution of the number n of initating events E_i in time

$$P(n,\theta_{E_i}) = \frac{\left(\theta_{E_i}t\right)^n \exp\left(-\theta_{E_i}t\right)}{n!} \qquad (3)$$

Nomenclature

E_i - initiating event (i varies from 1 to I)

F_j - faults (j varies from 0 to J)

n - number of initiating events (n varies from 0 to N)

q - probability of failure

p - probability

$S(E_i)$ - safety devices associated with overcoming the effects of initiating event E_i

θ_{E_i} - average failure rate associated with initiating event E_i

t - time

I,J,N,M - prespecified indeces on program input

APPENDIX VI

OVERALL SUMMATION MODEL EQUATIONS

Probability of curie release taking on value C_k given E_i

$$P\ (C_k/E_i) = \sum_{j=o}^{J} P\ (C_k/F_j) .\ P\ (F_j/E_i) \qquad (1)$$

Probability of having C_k given F_j assuming random values for input parameters

$$P\ (C_k/F_j) = \frac{\text{number of releases in } C_k \text{ decade}}{\text{total number of runs}} \qquad (2)$$

Generating function associated with E_i

$$g\ (S/E_i) = \sum_{k=0}^{K} P\ (C_k/E_i)\ S^{C_k} \qquad (3)$$

Generating function associated with n failures of E_i

$$g(S/E_i, t) = \sum_{n=0}^{N} P\ (n, \theta_{E_i}) \cdot \left\{ g\ (S/E_i) \right\}^n \qquad (4)$$

Generating function associated with all initiating events

$$g\ (S/E,t) = \prod_{i=1}^{I} g\ (S/E_i, t) \qquad (5)$$

Generating function in power series form

$$g\ (S/E,t) = \sum_{k=0}^{K} p_k S^{C_k} \qquad (6)$$

Probability that curie release is less than C_y for $C_k \leqslant C_y$

$$P\left[\, C(t) < C_y \,\right] = \sum_{k=1}^{y} p_k \tag{7}$$

Probability that curie release exceeds C_x

$$P\left[\, C(t) \geqslant C_x \,\right] = 1 - \sum_{k=0}^{x} p_k \tag{8}$$

Mean curie release due to all initiating events

$$M(E,\ t) = \frac{d}{dS}\left\{ g\ (S/E,t)\right\}_{S=1} = \sum_{k=0}^{K} p_k\, C_k \tag{9}$$

Nomenclature:

C	-	curie release
E_i	-	initiating events (i varies from 1 to I)
F_j	-	faults (j varies from 0 to J)
g	-	generating functions
k	-	index for curie release decades (k varies from 0 to K)
M	-	mean curie release
n	-	number of initiating events
p_k	-	probability that curie release is C_k
S	-	dummy variable used to form generating functions
x, y	-	predetermined curie release decades
θ_{E_i}	-	average failure rate associated with E_i
I, J, K, N	-	prespecified indeces on program input

APPENDIX VII

CALCULATION OF AUTOMATIC PROTECTIVE SYSTEM RELIABILITY

$$p(s) = p(r) + p(z) \tag{1}$$
$$p(z) = p(y_1) \cdot p(y_2) \tag{2}$$
$$p(y_1) = p(c_1) + p(x_1) + p(x_2) \tag{3}$$
$$p(y_2) = p(c_2) + p(x_3) + p(x_4) \tag{4}$$
$$p(x_1) = p(i_1) + p(g_1) \cdot p(g_2) + p(g_1) \cdot p(g_3) + p(g_2) \cdot p(g_3) \tag{5}$$

$$p(x_4) = p(i_4) + p(g_1) \cdot p(g_2) + p(g_1) \cdot p(g_3) + p(g_2) \cdot p(g_3) \tag{8}$$
$$p(x_1) \simeq p(i_1) \tag{9}$$

$$p(x_4) \simeq p(i_4) \tag{12}$$
$$p(y_1) = p(c_1) + 2p(i) \tag{13}$$
$$p(y_2) = p(c_2) + 2p(i) \tag{14}$$
$$p(z) = p^2(c) + 4\,p^2(i) + 4\,p(c) \cdot p(i) \tag{15}$$
$$p(s) = p(r) + p^2(c) + 4\,p^2(i) + 4\,p(c) \cdot p(i) \tag{16}$$
$$p(c) \simeq \theta_c t_c; \qquad p(i) \simeq \theta_i t_i \tag{17}$$

$t_c = 2$ years $\quad t_i = 3$ month

$\theta_c = 0.0015$ faults/y; $\quad \theta_i = 0.001$ faults/y

$$p(s) = 9 \times 10^{-6} + 3 \times 10^{-6} = 1.3 \times 10^{-5} \tag{18}$$

Nomenclature

c	-	contactor
i	-	inverter
g	-	guard lines
p	-	probability
r	-	rod clutch
s	-	total output signal
θ	-	mean failure rate
t	-	elapsed time
x	-	output signal from inverters
y	-	output signal from contactors
z	-	output signal

TABLE 1

REACTOR CORE DATA

Item	Units	Value
core heat output	MW	1457.8
mean fuel rating	MW/TeU	9.63
active core height	cm	829.2
number of mesh points	-	9
mesh interval	cm	103.65
fuel element pitch	cm	103.65

TABLE 2

FUEL ELEMENT DATA

Item	Units	Value
UO_2 weight/unit length of pin	gm. cm^{-1}	15.16
can weight/unit length of pin	gm. cm^{-1}	1.525
inner sleeve weight/unit length of cluster	gm. cm^{-1}	54.68
outer sleeve weight/unit length of cluster	gm. cm^{-1}	176.9
UO_2 density	gm. cm^{-3}	10.65
can density	gm. cm^{-3}	7.98
inner sleeve density	gm. cm^{-3}	1.78
outer sleeve density	gm. cm^{-3}	1.85
UO_2 specific heat at 0^oC	kWsec/gm^oC	2.96×10^{-4}
UO_2 specific heat linear temperature coefficient	$^oC^{-1}$	9.89×10^{-5}
can specific heat	kW sec/gm ^oC	5.02×10^{-4}

MODERATOR DATA

Item	Units	Value
weight/unit length, end of life	gm. cm^{-1}	1573
density of bulk moderator	gm. cm^{-3}	1.816
graphite specific heat at $0^{o}C$	kWsec/gm^{o}C	9.588×10^{-4}
graphite specific heat linear temperature coefficient	$^{o}C^{-1}$	1.367×10^{-3}

TABLE 4

FUEL AND MODERATOR NEUTRON AND FISSION PRODUCT POWER FRACTIONS

Component	Power Fraction	
	Neutron, α	Fission Product, μ
Fuel Pins		
inner ring	0.1313	0.1085
middle ring	0.2948	0.2438
outer ring	0.5227	0.4324
Moderator		
inner sleeve	0.0029	0.0012
outer sleeve	0.0076	0.0031
bulk	0.0426	0.175
Total power	0.917	0.083

TABLE 5

FISSION PRODUCT POWER TERMS

Group	Fraction, γ	Decay constant, δ
1	0.1815	0.28
2	0.2572	0.0364
3	0.2271	0.00397
4	0.3342	0.00013

TABLE 6

DELAYED NEUTRON DATA

Group	Delayed neutron fraction, β	Decay constant λ, sec^{-1}
1	0.000182	0.0128
2	0.001146	0.0314
3	0.001017	0.123
4	0.002137	0.322
5	0.000762	1.36
6	0.000183	3.65
Total	0.005427	-

TABLE 7

CHANNEL POWERS

Item	Channel power, kW
driving channel	3573
peak channel	4600
half irradiated	3860

TABLE 8

RADIATION CONSTANTS

Item	Value, kW/cm $^{o}K^{4}$
R_{ss}, inner sleeve to outer sleeve	2.09×10^{-13}
R_{sm}, outer sleeve to moderator	2.50×10^{-13}
R_{a1s}, inner ring to inner sleeve	1.012×10^{-14}
R_{a2s}, middle ring to inner sleeve	3.125×10^{-14}
R_{a3s}, outer ring to inner sleeve	1.017×10^{-13}
R_{a12}, inner ring to middle ring	2.106×10^{-14}
R_{a13}, inner ring to outer ring	8.458×10^{-15}
R_{a23}, middle ring to outer ring	3.312×10^{-14}

TABLE 9

CONTROL RODS

Rod group	Rod type	Group worth, Niles	Rod speed fractions of active core per 1000 secs
coarse group 1	black	2.2	0.718
coarse group 2	black	2.8	0.718
coarse group 3	black	3.4	0.718
zone	grey	0.8	1.692
trim/override	grey	1.9	0.718

TABLE 10

REACTOR CONTROLLER AND TRIP PARAMETERS

Item	Units	Values at full load
thermocouple response attenuation factor	-	0.53
thermocouple lag	sec	2
thermcouple surrounding material time constant	sec	120
controller phase advances	sec	10
controller lags	sec	1
deadband	^{o}C	1
saturation band	^{o}C	21
control rod mechanism lag	sec	1
trip margin on gas outlet temperature	^{o}C	40
servo reset trip rate	$^{o}C/min$	10

TABLE 11

AXIAL VARIATION OF CHANNEL INDEPENDENT VARIABLES

Mesh point	Heat production shape/ neutron flux shape Ω	Neutron flux shape Φ	Fuel reactivity coefficient a_u mN/°C	Moderator reactivity coefficient a_m mN/°C	Migration area M^2 [cm^2]	Innder sleeve specific heat fraction G_{S1}
9 (top)	1.08	0.0716	-1.148	5.20	1066	0.917
8	0.9302	0.2747	-1.115	6.00	1059	0.91
7	0.9272	0.4768	-1.092	6.40	1054	0.88
6	0.9819	0.6967	-1.072	6.20	1051	0.867
5	1.0486	0.9049	-1.063	5.40	1050	0.89
4	1.0284	0.9971	-1.082	6.10	1049	0.9
3	1.0139	0.9144	-1.139	6.45	1049	0.873
2	0.9896	0.6185	-1.242	6.20	1051	0.867
1 (bottom)	0.92	0.161	-1.404	5.45	1064	0.867

TABLE 12

FUEL STRINGER IRRADIATIONS IN MWd/t

FUEL ASSEMBLY / FUEL CHANNEL	2			3			4			5			6			7		
	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3
NEWLY LOADED CHANNEL	3484	3903	4617	3698	4143	4900	3443	3858	4563	2723	3051	3609	1894	2122	2510	1135	1272	1505
HALF IRRADIATED CHANNEL	11216	12568	14865	11864	13293	15723	11093	12429	14701	8874	9944	11761	6264	7000	8279	3778	4233	5007
REJECT CHANNEL	16739	18755	22183	16049	17892	21270	13977	15661	18524	15950	17872	21139	16315	18281	21623	15340	17188	20330

TABLE 13

FUEL STRINGER IODINE YIELDS IN %

FUEL ASSEMBLY / FUEL CHANNEL	2			3			4			5			6			7		
	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3
NEWLY LOADED CHANNEL	3.16	3. 17	3.18	3.165	3.175	3.185	3.16	3.17	3.18	3.145	3.15	3.165	3.135	3.14	3.145	3.12	3.125	3.13
HALF IRRADIATED CHANNEL	3.25	3.265	3.29	3.26	3.275	3.30	3.25	3.265	3.29	3.23	3.24	3.26	3.195	3.205	3.22	3.165	3.17	3.18
REJECT CHANNEL	3.31	3.335	3.37	3.30	3.32	3.36	3.28	3.30	3.33	3.30	3.32	3.36	3.305	3.325	3.36	3.295	3.305	3.35

TABLE 14

FUEL STRINGER RATINGS IN MW/TeU

FUEL ASSEMBLY / FUEL CHANNEL	2			3			4			5			6			7		
	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3
NEWLY LOADED CHANNEL	16.54	18.60	21.98	18.43	20.73	24.48	16.95	19.07	22.52	13.22	14.87	17.56	9.09	10.23	12.08	5.31	5.97	7.05
HALF IRRADIATED CHANNEL	13.88	15.61	18.44	15.46	17.39	20.54	14.23	16.00	18.90	11.09	12.48	14.74	7.63	8.59	10.14	4.45	5.01	5.92
REJECT CHANNEL	12.72	14.31	16.90	14.17	15.93	18.83	13.04	14.67	17.32	10.17	11.44	13.51	7.00	7.87	9.29	4.09	4.59	5.42

TABLE 15

FUEL STRINGER IODINE INVENTORIES IN KILO-CURIES

FUEL ASSEMBLY / FUEL CHANNEL	2			3			4			5			6			7		
	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3
NEWLY LOADED CHANNEL	3.4	7.7	13.6	3.8	8.6	15.2	3.5	7.9	14.0	2.7	6.1	10.8	1.9	4.2	7.4	1.1	2.4	4.3
HALF IRRADIATED CHANNEL	2.9	6.6	11.8	3.3	7.4	13.2	3.0	6.8	12.1	2.3	5.3	9.4	1.6	3.6	6.4	0.9	2.1	3.7
REJECT CHANNEL	2.7	6.2	11.1	3.1	6.9	12.3	2.8	6.3	11.3	2.2	4.9	8.9	1.5	3.4	6.1	0.9	2.0	3.6

TABLE 16

FUEL STRINGER IODINE FRACTIONAL RELEASES IN %

FUEL ASSEMBLY / FUEL CHANNEL	2			3			4			5			6			7		
	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3
NEWLY LOADED CHANNEL	0.07	0.14	0.50	0.13	0.30	1.60	0.08	0.16	0.57	0.031	0.047	0.087	0.017	0.02	0.027	0.013	0.013	0.013
HALF IRRADIATED CHANNEL	0.036	0.054	0.13	0.053	0.086	0.27	0.037	0.06	0.16	0.022	0.029	0.046	0.014	0.016	0.02	0.013	0.013	0.013
REJECT CHANNEL	0.030	0.037	0.079	0.037	0.06	0.16	0.031	0.043	0.086	0.02	0.024	0.035	0.013	0.015	0.017	0.013	0.013	0.013

TABLE 17

FUEL STRINGER FREE IODINE INVENTORIES IN CURIES

FUEL ASSEMBLY / FUEL CHANNEL	2			3			4			5			6			7		
	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3
NEWLY LOADED CHANNEL	2.38	10.78	68.0	4.94	25.8	243.2	2.80	12.64	79.8	0.84	2.84	9.40	0.32	0.84	2.0	0.14	0.31	0.56
HALF IRRADIATED CHANNEL	1.05	3.56	15.34	1.75	6.36	35.64	1.11	4.08	19.36	0.51	1.54	4.32	0.22	0.58	1.28	0.12	0.27	0.48
REJECT CHANNEL	0.81	2.29	8.77	1.15	4.14	19.68	0.87	2.71	9.72	0.44	1.18	3.12	0.20	0.51	1.04	0.12	0.26	0.47

TABLE 18

CAN INTERNAL PRESSURE IN PSIA

FUEL ASSEMBLY / FUEL CHANNEL	2			3			4			5			6			7		
	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3
NEWLY LOADED CHANNEL	70.7	79.2	93.6	73.4	84.2	99.5	69.8	78.3	92.7	55.4	61.7	73.4	38.2	43.2	50.9	23	25.8	30.5
HALF IRRADIATED CHANNEL	229.5	255.2	301.5	240.8	269.6	319.1	225	252	298.4	180	201.6	238.5	126.9	141.8	167.9	76.6	85.9	101.6
REJECT CHANNEL	339.8	380.3	450	325.4	365	431.6	283.5	317.7	375.8	323.6	362.7	428.9	330.8	370.8	438.8	311.2	348.7	412.4

TABLE 20

ALL CIRCULATOR FAILURE. SEQUENCE 0-2-6-9. CAN MELTING.

FUEL ASSEMBLY	2			3			4			5			6			7		
FUEL CHANNEL	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3
NEWLY LOADED CHANNEL				x	x		x	x	x	x	x							
HALF IRRADIATED CHANNEL							x	x										
REJECT CHANNEL							x											

TABLE 19

ALL CIRCULATOR FAILURE

INITIATING EVENT FREQUENCIES

Type of fault	Events/year
generator fault	0.010
generator transformer fault	0.075
unit transformer fault	0.075
400 kV cable fault	0.033
fault on interconnection	0.017
failure of protection	0.020
mechanical damage	-
missile	-
total	0.230

TABLE 21

BARRING MOTORS

FAILURE PROBABILITIES

Type of fault	Probability
contactor fault	$p_1 = 3.10^{-3}$
clutch fault	$p_2 = 10^{-2}$
loss of supplies	$p_3 \quad 10^{-3}$
total	$1.4.10^{-2}$

TABLE 22

ALL CIRCULATOR FAILURE

PROBABILITIES OF NUMBER OF EVENTS

Number of Events	Probabilities					
	5 yrs	10 yrs	15 yrs	20 yrs	25 yrs	30 yrs
0	3.17,-1	1.00,-1	3.17,-2	1.01,-2	3.18,-3	1.01,-3
1	3.64,-1	2.31,-1	1.10,-1	4.62,-2	1.83,-2	6.95,-3
2	2.09,-1	2.65,-1	1.89,-1	1.06,-1	5.26,-2	2.40,-2
3	8.03,-2	2.03,-1	2.17,-1	1.63,-1	1.01,-1	5.52,-2
4	2.31,-2	1.17,-1	1.87,-1	1.88,-1	1.45,-1	9.52,-2
5	5.31,-3	5.38,-2	1.29,-1	1.73,-1	1.67,-1	1.31,-1
6	1.02,-3	2.06,-2	7.44,-2	1.32,-1	1.60,-1	1.51,-1
7	1.67,-4	6.77,-3	3.66,-2	8.69,-2	1.31,-1	1.49,-1
8	2.75,-5	2.59,-3	2.48,-2	9.51,-2	2.22,-1	3.86,-1

TABLE 23

TIMES AFTER FAULT ONSET WHEN CANS BECOME SUBJECTED TO HOOP STRESSES IN SECS

FUEL ASSEMBLY / FUEL CHANNEL	2			3			4			5			6			7		
	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3
NEWLY LOADED CHANNEL	34	32	30	34	30.5	27.5	34	32	29	39	36	33	46	43	40	58	57	50
HALF IRRADIATED CHANNEL	13	10.5	7.5	12	9.5	6	13	10.5	7.5	17	15	12	23.5	22	18.5	32	30	27.5
REJECT CHANNEL	5	3	0	6	4	1	8.5	6	3	6	4	1	6	3.5	0.5	6.5	4.5	1.5

TABLE 24

ULTIMATE HOOP STRESSES IN PSIA

FUEL ASSEMBLY / FUEL CHANNEL	2			3			4			5			6			7		
	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3
NEWLY LOADED CHANNEL	1350	1550	1895	1415	1670	2035	1335	1530	1875	985	1135	1415	575	695	880	214	281	393
HALF IRRADIATED CHANNEL	5120	5740	6830	5400	6095	7260	5260	5665	6760	3950	4475	5345	2690	2900	3670	1490	1711	2085
REJECT CHANNEL	7760	8720	10380	7410	8355	9950	6430	7235	8620	7380	8305	9880	7545	8495	10115	7073	8203	9482

TABLE 25

INTERMEDIATE HOOP STRESSES AT 25 SECS IN PSIA

FUEL ASSEMBLY / FUEL CHANNEL	2			3			4			5			6			7		
	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3
NEWLY LOADED CHANNEL	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
HALF IRRADIATED CHANNEL	2677	3289	4391	2946	3632	4810	2570	3213	4317	1500	2013	2892	236	590	1211	0	0	0
REJECT CHANNEL	5303	6266	7925	4960	5902	7473	3950	4777	6159	4917	5848	7890	5088	6040	7659	4622	5514	7031

TABLE 26

Time (Sec.)	Can Temperatures (°C) Fuel Assemblies 2			3			4			5			6		
	R1	R2	R3	R1	R2	R3	R1	R2	R3	R1	R2	R3	R1	R2	R3
25	617	637	667	724	745	776	788	807	833	807	822	842	797	806	819
50	637	657	683	746	764	788	807	823	839	822	833	845	806	813	819
75	660	679	700	769	786	801	828	840	847	837	846	848	816	820	818
100	680	697	713	791	804	810	847	855	852	851	856	850	824	825	816
150	713	726	732	825	831	823	876	876	857	872	869	849	837	831	809
200	739	749	745	852	851	832	899	891	861	888	879	849	845	834	804
250	761	766	755	873	867	838	916	902	865	901	886	849	852	836	800
300	780	780	762	890	879	844	930	911	868	910	891	851	857	837	798
350	794	791	767	904	887	848	940	917	871	918	896	852	860	838	797
400	806	799	771	914	894	852	948	922	873	924	899	854	863	839	796
450	815	805	775	922	899	855	954	926	876	928	902	856	865	840	796
500	822	810	777	928	903	858	958	929	878	931	904	858	867	840	796
550	828	814	780	932	906	860	962	931	880	934	905	859	868	841	797
600	833	817	782	936	908	862	964	932	882	936	907	861	869	841	798
650	837	819	783	938	910	864	966	934	883	938	908	862	870	841	798
700	839	821	784	940	911	865	967	934	884	939	908	863	870	842	799
800	843	823	786	942	913	867	968	935	886	940	909	864	871	842	800
900	845	824	787	943	913	868	968	935	886	940	910	865	871	843	801
1000	828	808	774	925	897	855	953	922	877	929	901	860	865	839	800
1100	785	769	741	883	860	825	917	892	853	904	880	845	851	829	797
1200	729	718	698	825	808	783	867	848	818	866	848	820	829	812	788
1300	681	674	660	773	762	744	819	806	783	828	814	791	805	792	772

Depressurisation. sequence 0-1(2)-3-6-8-9. Temperature/time histories in newly loaded channel.

TABLE 27

Time (Sec.)	Can Temperatures (°C) Fuel Assemblies														
	2			3			4			5			6		
	R1	R2	R3	R1	R2	R3	R1	R2	R3	R1	R2	R3	R1	R2	R3
25									809			815			
50									813		805	816			
75								809	816	806	814	817			
100							811	820	819	816	820	814			
150							834	835	819	831	828	809	803		
200				803	803		850	845	820	842	833	804	808		
250				820	814		864	853	820	850	836	801	812		
300				833	822		874	859	821	856	838		814		
350				844	829		882	863	822	861	839		816		
400				852	834		888	866	823	864	840		816		
450				859	838		893	869	824	867	841		817		
500				864	841		897	871	825	869	841		817		
550				868	843		900	872	826	870	842		816		
600				871	845	800	902	873	827	871	842		816		
650				874	846	801	903	874	827	872	842		815		
700				875	847	802	904	874	828	872	842		815		
800				877	849	804	905	874	828	872	842		814		
900				879	850	805	905	874	829	872	842		812		
1000				866	838		894	864	821	864	838		806		
1100				835	811		867	841	803	846	822				
1200							828	807		820	801				

Depressurisation sequence 0-1(2)-3-6-8-9. Temperature/time histories in reject channel.

TABLE 28

Time Interval (Sec.)	Can Temperatures (°C) Fuel Assemblies														
	2			3			4			5			6		
	R1	R2	R3	R1	R2	R3	R1	R2	R3	R1	R2	R3	R1	R2	R3
0-25				-	-		-	-	.0002	-	-		-		
25-50				-	-		-	-	.0004	-	.0005		-		
50-75				-	-		-	.0007	.0008	.0003	.0007		-		
75-100				-	-		.0003	.0011	.0010	.0005	.0009		-		
100-150				-	-		.0014	.0021	.0020	.0019	.0025		.0006		
150-200				.0006	.0009		.0026	.0032	.0021	.0030	.0031		.0008		
200-250				.0012	.0014		.0046	.0044	.0021	.0041	.0035		.0009		
250-300				.0021	.0020		.0068	.0055	.0022	.0052	.0037		.0010		
300-350				.0033	.0026		.0092	.0073	.0023	.0064	.0039		.0011		
350-400				.0045	.0032		.0115	.0082	.0024	.0071	.0041		.0011		
400-450				.0060	.0038		.0138	.0088	.0025	.0081	.0042		.0012		
450-500				.0073	.0043		.0160	.0092	.0026	.0087	.0042		.0012		
500-550				.0085	.0046		.0178	.0095	.0027	.0090	.0044		.0011		
550-600				.0095	.0051		.0192	.0099	.0027	.0094	.0044		.0011		
600-650				.0107	.0053		.0199	.0099	.0028	.0097	.0044		.0011		
650-700				.0111	.0055		.0206	.0099	.0056	.0097	.0044		.0011		
700-800				.0240	.0118		.0428	.0198	.0058	.0194	.0088		.0020		
800-900				.0259	.0124		.0428	.0198	.0044	.0194	.0088		.0019		
900-1000				.0157	.0076		.0286	.0134	.0019	.0142	.0070		.0014		
1000-1100				.0046	.0025		.0103	.0054	.0008	.0071	.0038				
1100-1200							.0022	.0011		.0024	.0016				
SUM				.1350	.0731		.2704	.1492	.0473	.1456	.0787		.0175		

Depressurisation sequence 0-1(2)-3-6-8-9. Fractions of rupture life in reject channel.

TABLE 29

Time (Sec.)	Can Temperatures (°C) Fuel Assemblies														
	2			3			4			5			6		
	R1	R2	R3	R1	R2	R3	R1	R2	R3	R1	R2	R3	R1	R2	R3
25									817		804	823			803
50								805	822	803	814	825			802
75							807	819	827	815	823	825		801	
100							823	831	830	826	831	824	804	805	
150					802		847	848	832	843	840	820	814	808	
200				819	818		866	860	834	855	846	817	820	809	
250				837	831	803	880	869	835	865	850	814	825	809	
300				851	840	807	892	876	837	872	854	813	828	809	
350				863	848	809	901	881	838	878	856	813	830	808	
400				872	853	812	908	884	840	882	858	813	831	808	
450				879	857	814	913	887	841	885	859	814	832	807	
500				884	861	816	917	890	843	887	860	814	833	807	
550				888	863	818	920	891	844	889	861	815	833	806	
600				892	865	820	922	892	845	891	862	816	833	806	
650				894	867	821	923	893	846	892	862	816	833	805	
700	801			896	868	822	925	894	847	892	862	817	833	805	
800	805			898	869	824	926	894	848	893	863	818	832	804	
900	808			899	870	825	926	894	848	893	863	818	832	804	
1000				885	858	816	913	883	840	884	856	815	826		
1100				850	827		883	858	820	864	840	805	813		
1200				803			842	821		836	817				
1300							801			806					

Depressurisation, sequence 0-1(2)-3-6-8-9. Temperature/time histories in half irradiated channel.

TABLE 30

DEPRESSURISATION SEQUENCE 0-1(2)-3-6-8-9. PROBABILITIES OF CAN FAILURE IN %.

FUEL ASSEMBLY / FUEL CHANNEL	2			3			4			5			6			7		
	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3
NEWLY LOADED CHANNEL	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
HALF IRRADIATED CHANNEL	-	-	-	0.01	-	-	2	0.05	-	-	-	-	-	-	-	-	-	-
REJECT CHANNEL	-	-	-	0.05	0.01	-	1.6	0.075	-	0.075	0.01	-	-	-	-	-	-	-

TABLE 31

Time (Sec.)	Can Temperatures (°C) Fuel Assemblies														
	2			3			4			5			6		
	R1	R2	R3	R1	R2	R3	R1	R2	R3	R1	R2	R3	R1	R2	R3
25	718	748	792	849	880	926	919	947	985	926	948	978	896	910	930
50	739	765	798	872	896	923	939	959	977	943	958	973	908	917	924
75	762	785	807	896	915	925	961	973	973	959	968	966	918	922	917
100	781	800	812	915	927	924	977	981	966	971	973	958	925	923	907
150	807	818	814	940	940	916	996	987	952	985	975	941	932	921	888
200	827	831	815	957	947	911	1008	990	943	992	974	930	935	917	875
250	842	839	814	969	952	907	1015	990	937	997	972	922	936	912	865
300	853	845	814	977	954	905	1019	989	933	998	969	917	936	908	858
400	866	851	813	984	954	902	1021	986	928	998	964	910	932	900	849
500	873	853	813	985	953	900	1018	981	924	994	959	906	927	894	844
600	875	852	811	983	949	898	1014	977	921	989	954	903	922	889	840
700	874	850	810	979	946	896	1009	972	918	983	949	899	916	884	837
800	872	848	808	975	942	893	1003	967	915	978	944	895	911	879	834
900	869	845	806	970	937	890	998	962	911	972	939	891	907	875	831
1000	847	825	789	947	917	874	977	945	898	956	926	882	895	867	826
1100	800	783	753	900	875	840	936	910	870	926	900	864	876	852	819
1200	740	728	707	838	821	794	882	862	832	884	864	835	850	832	806
1300	689	681	667	783	771	753	831	817	794	843	827	804	823	808	787

Depressurisation sequence 0-1(2)-4-6-8-9. Temperature/time histories in newly loaded channel.

TABLE 32

Time (Sec.)	Can Temperatures (°C) Fuel Assemblies														
	2			3			4			5			6		
	R1	R2	R3	R1	R2	R3	R1	R2	R3	R1	R2	R3	R1	R2	R3
25				810	838	880	874	900	936	879	899	927	853	867	885
50				828	851	876	891	911	932	893	908	921	863	871	878
75				847	864	875	909	922	927	906	914	914	871	874	870
100				862	873	871	921	928	920	914	917	904	876	874	860
150				882	882	861	927	932	905	924	916	886	880	870	841
200				895	887	854	947	933	894	929	913	873	882	865	826
250				905	890	849	953	933	886	932	910	863	881	860	815
300				911	891	845	956	931	881	933	906	856	880	854	807
400	809			918	891	840	958	927	874	931	899	847	875	845	
500	816			919	888	837	955	922	869	927	893	841	870	837	
600	818			918	885	835	951	916	865	921	887	837	863	831	
700	819			915	882	833	946	911	861	916	882	833	857	825	
800	818			911	879	831	941	906	857	911	877	830	851	820	
900	816			907	875	829	936	902	853	906	873	826	846	815	
1000	800			890	860	817	920	888	842	893	863	819	836	808	
1100				855	829		888	861	821	871	845	808	821		
1200				807			846	824		841	820		802		
1300							805			811					

Depressurisation sequence 0-1(2)-4-6-8-9. Temperature/time histories in reject channel.

TABLE 33

Time Interval (Sec.)	Can Temperatures (°C) Fuel Assemblies 2			3			4			5			6		
	R1	R2	R3	R1	R2	R3	R1	R2	R3	R1	R2	R3	R1	R2	R3
0-25				.0004	.0019	.0174	.0007	.0130	.0821	.0064	.0196	.0935	.0023	.0064	
25-50				.0009	.0032	.0149	.0032	.0194	.0716	.0108	.0272	.0758	.0037	.0074	
50-75				.0019	.0054	.0144	.0123	.0286	.0602	.0173	.0337	.0592	.0051	.0083	
75-100				.0034	.0076	.0124	.0189	.0353	.0471	.0230	.0375	.0413	.0061	.0083	
100-150				.0145	.0213	.0168	.0659	.0811	.0551	.0656	.0725	.0426	.0142	.0118	
150-200				.0235	.0257	.0127	.0927	.0840	.0369	.0781	.0651	.0261	.0153	.0096	
200-250				.0338	.0288	.0105	.1134	.0840	.0274	.0866	.0585	.0177	.0148	.0076	
250-300				.0419	.0298	.0089	.1253	.0784	.0227	.0897	.0507	.0135	.0118	.0054	
300-400				.1076	.0597	.0146	.2677	.1364	.0348	.1674	.0786	.0189	.0194	.0076	
400-500				.1114	.0534	.0129	.2424	.1143	.0288	.1456	.0630	.0148	.0148	.0060	
500-600				.1076	.0477	.0119	.2121	.0923	.0246	.1180	.0504	.0126	.0117	.0048	
600-700				.0967	.0426	.0110	.1792	.0675	.0211	.0989	.0418	.0107	.0092	.0038	
700-800				.0839	.0381	.0101	.1513	.0648	.0180	.0828	.0346	.0094	.0076	.0031	
800-900				.0726	.0327	.0093	.1275	.0560	.0154	.0692	.0298	.0080	.0051	.0023	
900-1000				.0390	.0184	.0056	.0731	.0335	.0099	.0430	.0202	.0060	.0027		
1000-1100				.0102	.0066		.0229	.0120	.0042	.0188	.0099	.0038	.0012		
1100-1200				.0014			.0045	.0027		.0058	.0036				
1200-1300							.0008			.0016					
SUM				.751	.423	.183	1.714	1.004	.556	1.129	.697	.454	.145	.092	

Depressurisation sequence 0-1(2)-4-6-8-9. Fractions of rupture life in reject channel.

TABLE 34

DEPRESSURISATION SEQUENCE 0-1(2)-4-6-8-9. PROBABILITIES OF CAN FAILURE IN %

FUEL ASSEMBLY / FUEL CHANNEL	2			3			4			5			6			7		
	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3	Ring 1	Ring 2	Ring 3
NEWLY LOADED CHANNEL	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
HALF IRRADIATED CHANNEL	-	-	-	17	3	0.04	82.5	42	10	3	0.45	0.03	-	-	-	-	-	-
REJECT CHANNEL	-	-	-	32	8	0.2	80	50	17	57	29	9	0.08	0.01	-	-	-	-

TABLE 35

MEDIUM SIZE PENETRATION FAILURE

PROBABILITIES OF NUMBER OF EVENTS

Number of events	Probabilities					
	5 yrs	10 yrs	15 yrs	20 yrs	25 yrs	30 yrs
0	9.99995,-1	9.99990,-1	9.99985,-1	9.99980,-1	9.99975,-1	9.99970,-1
1	4.50,-6	9.99,-6	1.50,-5	1.99,-5	2.50,-5	3.00,-5
2	1.25,-11	4.99,-11	1.12,-10	1.99,-10	3.12,-10	4.50,-10
3	2.08,-17	1.67,-16	5.62,-16	1.33,-15	2.60,-15	4.50,-15
4	2.60,-23	4.17,-22	2.11,-21	6.67,-21	1.63,-20	3.37,-20
5	2.60,-29	8.33,-28	6.33,-27	2.67,-26	8.14,-26	2.02,-25
6	2.17,-35	1.39,-33	1.58,-32	8.89,-32	3.39,-31	1.01,-30

TABLE 36

START-UP FAULT

PROBABILITIES OF NUMBER OF EVENTS

Number of events	Probabilities					
	5 yrs	10 yrs	15 yrs	20 yrs	25 yrs	30 yrs
0	9.993,-1	9.985,-1	9.976,-1	9.970,-1	9.963,-1	9.955,-1
1	7.49,-4	1.50,-3	2.25,-3	2.99,-3	3.74,-3	4.48,-3
2	2.81,-7	1.12,-6	2.53,-6	4.49,-6	7.00,-6	1.01,-5
3	7.02,-11	5.62,-10	1.89,-9	4.49,-9	8.76,-9	1.51,-8
4	1.32,-14	2.11,-13	1.06,-12	3.36,-12	8.21,-12	1.70,-11
5	1.98,-18	6.32,-17	4.79,-16	2.02,-15	6.16,-15	1.53,-14
6	2.97,-22	1.58,-20	1.80,-19	1.01,-18	3.85,-18	1.15,-17
7	2.65,-26	3.38,-24	5.78,-23	4.33,-22	2.06,-21	7.38,-21

TABLE 37

IODINE -131 RELEASES. DISCRETE RETENTION FACTORS

Fault sequence	iodine -131 activity in curies available for release	Iodine retention factors								Release rounded off to the nearest curie decade
		fuel meltout	can meltout	processes within cladding	pressure circuit	pressure vessel	safety valve vent pipes	iodine adsorption plant	reactor building	
AC1 (no trip)	3.9×10^7	1.0846	-	-	10	-	2	-	-	10^6
AC2 (flow stagnation)	3.9×10^7	-	2	-	10	-	2	-	-	10^6
AC3 (delayed trip)	8.05×10^6	-	2	-	10	-	2	-	-	10^5
DP1 (no trip)	3.9×10^7	1.0846	-	-	1.4	-	-	-	2	10^7
DP2 (flow stagnation)	3.9×10^7	-	2	-	1.4	-	-	-	2	10^7
DP3 (trip no operator action)	8.5×10^4	-	-	3	1.4	-	-	-	2	10^4
DP4 (slow trip operator action)	1.4×10^3	-	-	3	1.4	-	-	-	2	10^2
DP5 (fast trip operator action)	9	-	-	3	1.4	-	-	-	2	10
ST1 (no trip)	3.9×10^7	1.0846	-	-	10	-	2	-	-	10^6

TABLE 38

DISCRETE RETENTION FACTORS

GUESSED WEIGHTING FACTORS

Fault sequence	Release (curies)	Weighting factor (STATMOD R12)	Weighting factor (STATMOD R15)
AC1/2	10^6	0.5	0.1
AC3	10^5	0.5	0.9
DP1/2	10^7	10^{-5}	10^{-5}
DP3	10^4	0.20	0.20
DP4	10^2	0.24	0.24
DP5	10	0.56	0.56
ST1	10^6	1	1

TABLE 39

IODINE -131 RELEASES. RANDOM RETENTION FACTORS

fault sequence	iodine -131 activity in curies released to pressure circuit	iodine retention factors					releases rounded off to the nearest curie decade		
		pressure circuit			safety valve pipes	reactor building	R1	R2	R3
AC1	3.6×10^7	10	50	100	2	-	10^6	10^5	10^5
AC2	1.95×10^7	10	50	100	2	-	10^6	10^5	10^5
AC3	4×10^6	10	50	100	2	-	10^5	10^4	10^4
DP1	3.6×10^7	1.4	4.2	8.4	-	2	10^7	10^6	10^6
DP2	1.95×10^7	1.4	4.2	8.4	-	2	10^7	10^6	10^6
DP3	2.83×10^4	1.4	4.2	8.4	-	2	10^4	10^3	10^3
DP4	4.7×10^2	1.4	4.2	8.4	-	2	10^2	10^2	10^1
DP5	3	1.4	4.2	8.4	-	2	0	0	0
ST1	3.6×10^7	10	50	100	2	-	10^6	10^5	10^5

TABLE 40

RANDOM RETENTION FACTORS. LINEAR WEIGHTING FACTORS

fault sequences	probabilities of releases - STATMOD R14							
	0	10^1	10^2	10^3	10^4	10^5	10^6	10^7
AC	-	-	-	-	0.222	0.555	0.222	-
DP	0.133	0.133	0.133	0.133	0.067	-	0.267	0.133
ST	-	-	-	-	-	0.667	0.333	-

TABLE 41

RANDOM RETENTION FACTORS. GUESSED WEIGHTING FACTORS

fault sequences	probabilities of releases - STATMOD R16							
	0	10^1	10^2	10^3	10^4	10^5	10^6	10^7
AC	-	-	-	-	0.333	0.5	0.167	-
DP	0.373	0.267	0.16	0.134	0.067	-	6.7,-6	3.3,-6
ST	-	-	-	-	-	0.667	0.333	-

TABLE 42

STATMOD R16. PROBABILITIES OF RELEASE BEING LESS OR GREATER THAN C_K

release	probabilities at 30 yrs	
	less	greater
0	-	1.17547,-4
10^1	0.999890	1.09618,-4
10^2	0.999895	1.04867,-4
10^3	0.999899	1.00887,-4
10^4	0.999960	3.96790,-5
10^5	0.999996	3.45589,-6
10^6	0.999999	1.73177,-7
10^7	0.999999	1.72638,-7

TABLE 43

STATMOD RUNS 12,13,14. PROBABILITIES OF RELEASE

release curies	probabilities at t=30 yrs		
	STATMOD R12	STATMOD R13	STATMOD R15
10^4	5.94,-6	5.54,-6	5.94,-6
10^5	4.94,-5	4.95,-5	2.81,-5
10^6	4.95,-5	1.22,-9	2.82,-5
10^7	1.47,-9	6.50,-10	9.66,-10

TABLE 44

STATMOD RUNS 12,17,18. PROBABILITIES OF RELEASE

release curies	probabilities at t=30 yrs		
	STATMOD R12	STATMOD R17	STATMOD R18
10^1	1.66,-5	1.65,-5	1.66,-5
10^2	7.13,-6	7.09,-6	7.13,-6
10^3	1.18,-10	1.18,-10	1.18,-10
10^4	5.94,-6	5.91,-6	5.94,-6
10^5	4.94,-5	3.14,-3	1.16,-5
10^6	4.95,-5	3.14,-3	1.17,-5
10^7	1.47,-9	1.04,-7	5.74,-10

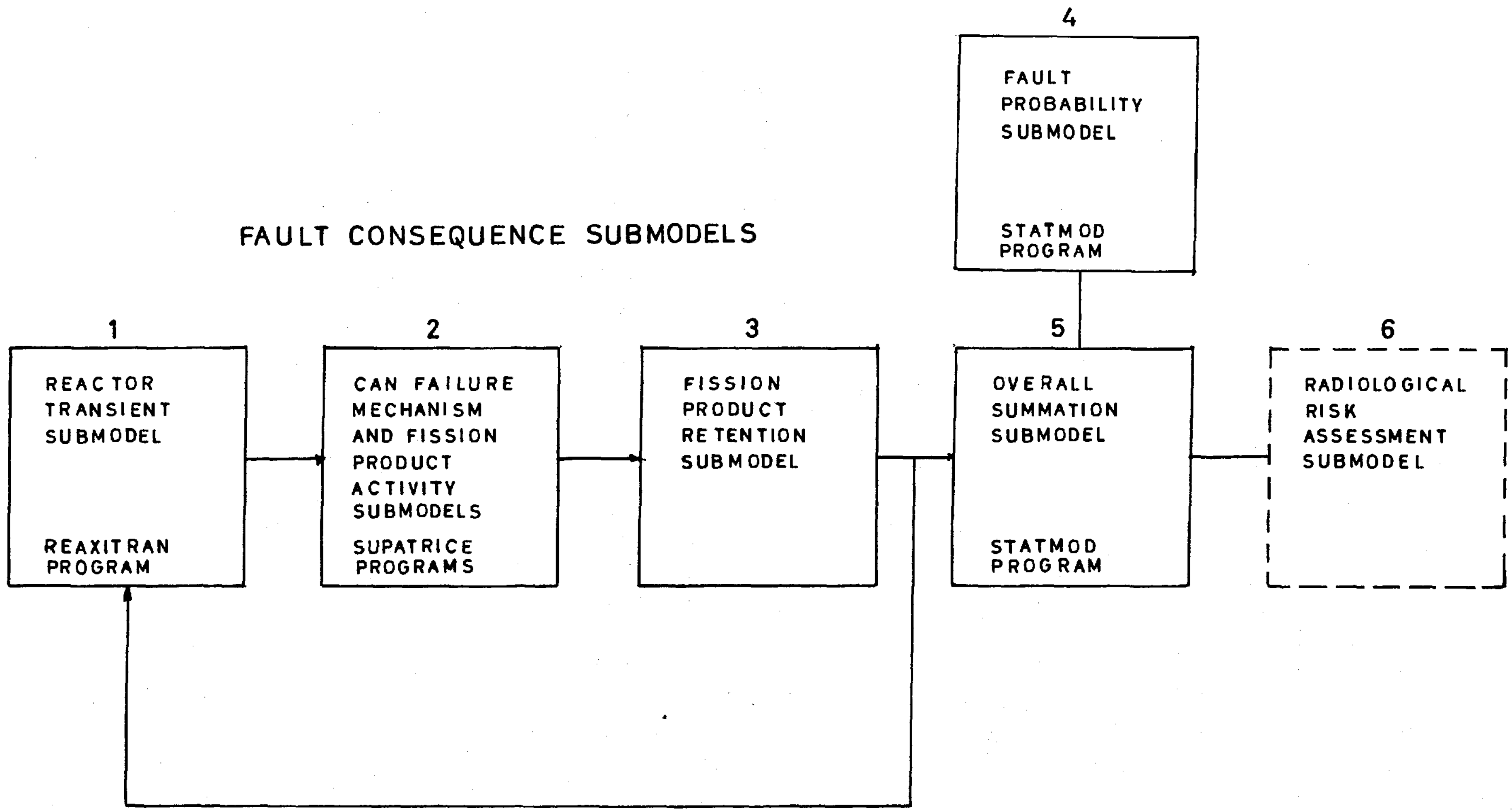


FIG.1 BLOCK SCHEME OF THE APC COMPOUND MODEL

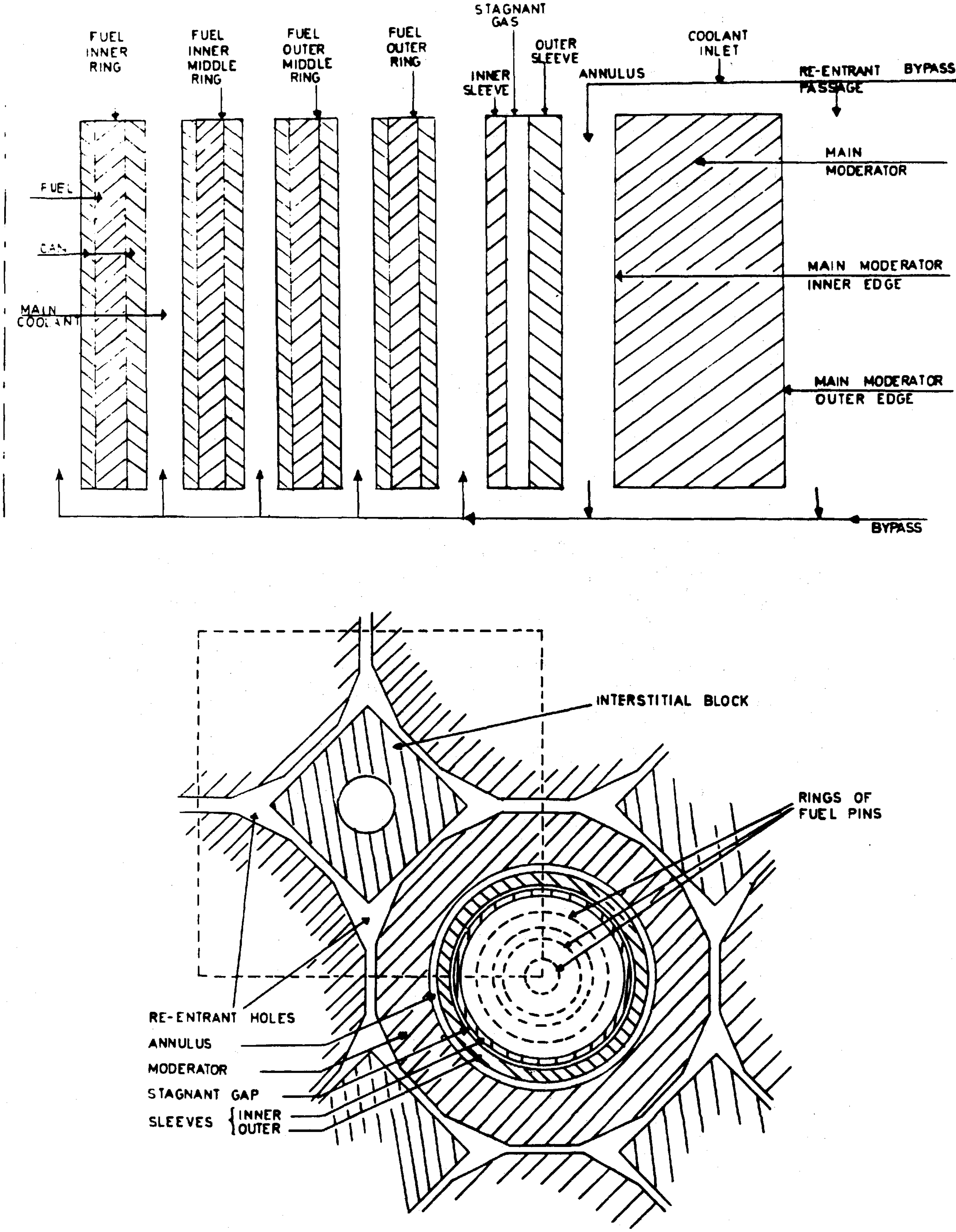


FIG. 2 SIMULATED AND ACTUAL CORE ARRANGEMENT OF AN AGR.

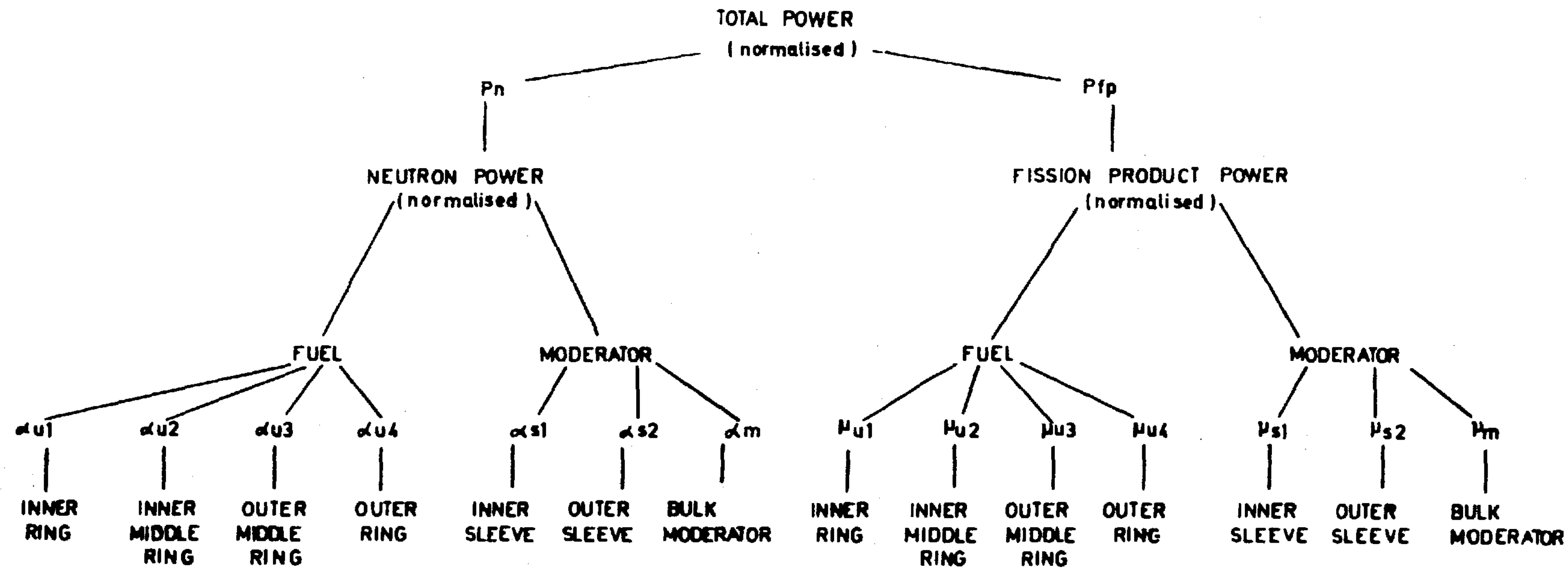


FIG. 3 DISTRIBUTION OF NEUTRON AND FISSION PRODUCT POWER BETWEEN FUEL AND MODERATOR COMPONENTS.

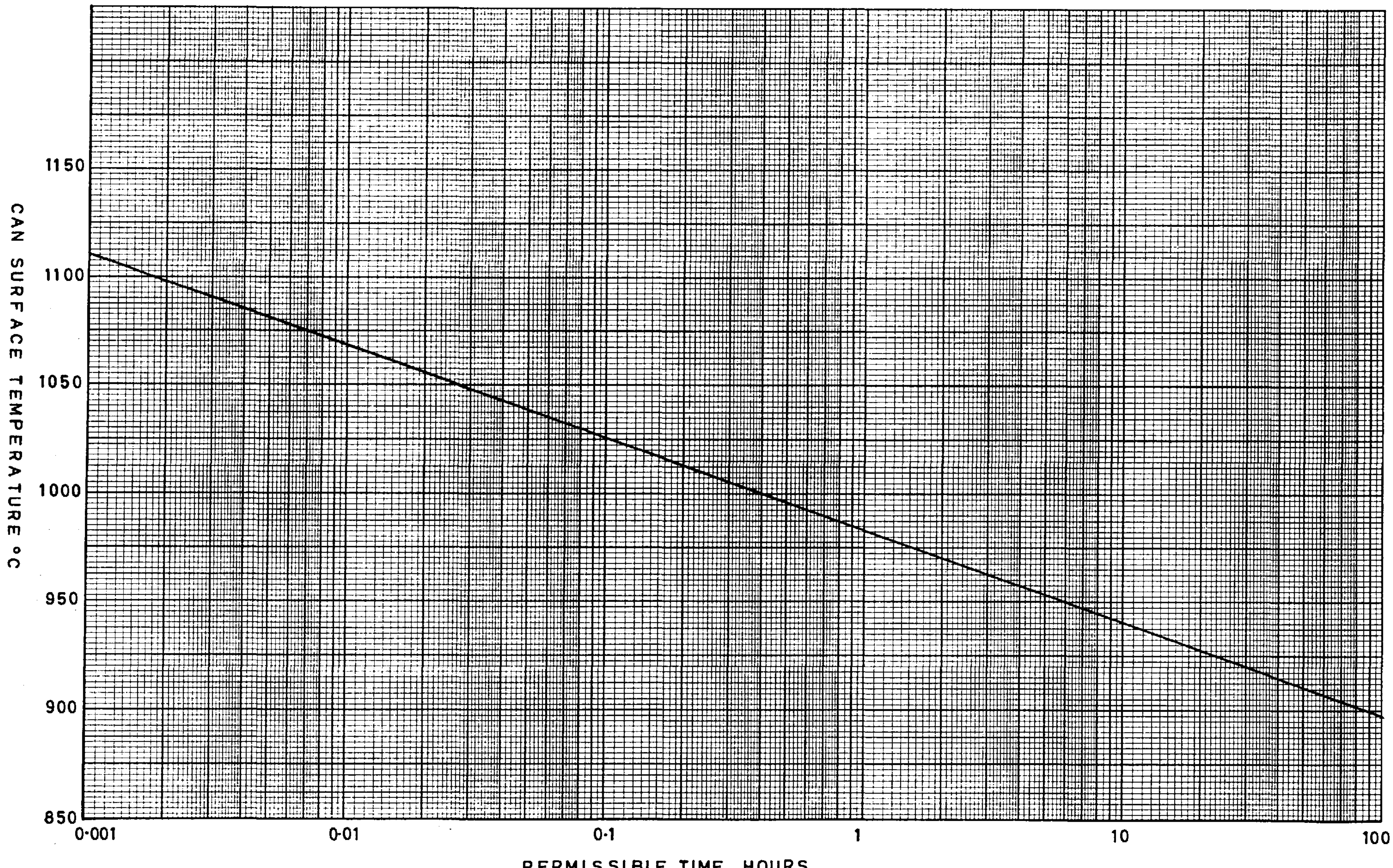


FIG. 4 CAN TIME-TEMPERATURE LIMITS IN PRESSURISED REACTOR

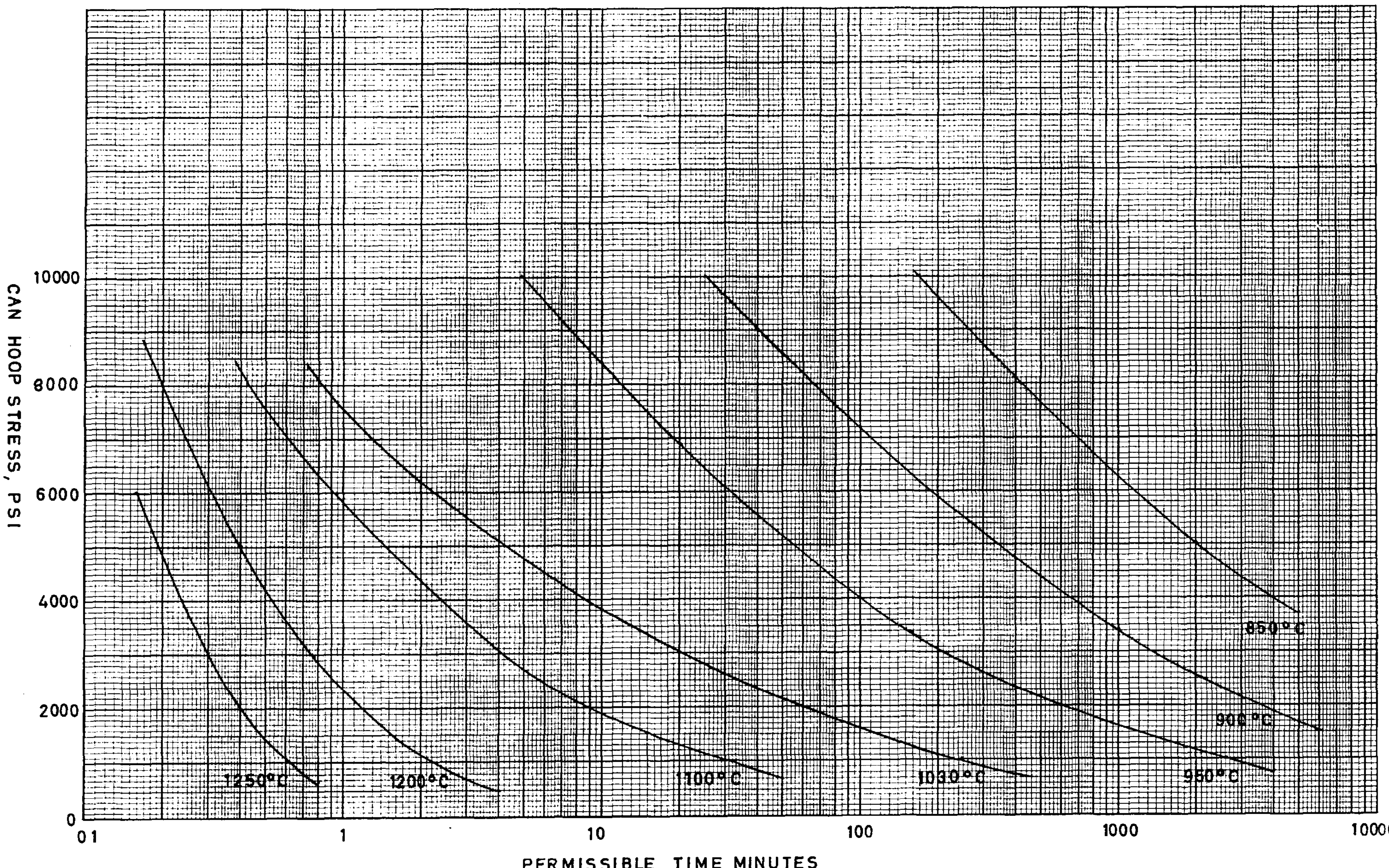


FIG. 5 CAN STRESS-TEMPERATURE-TIME LIMITS IN DEPRESSURISED REACTOR

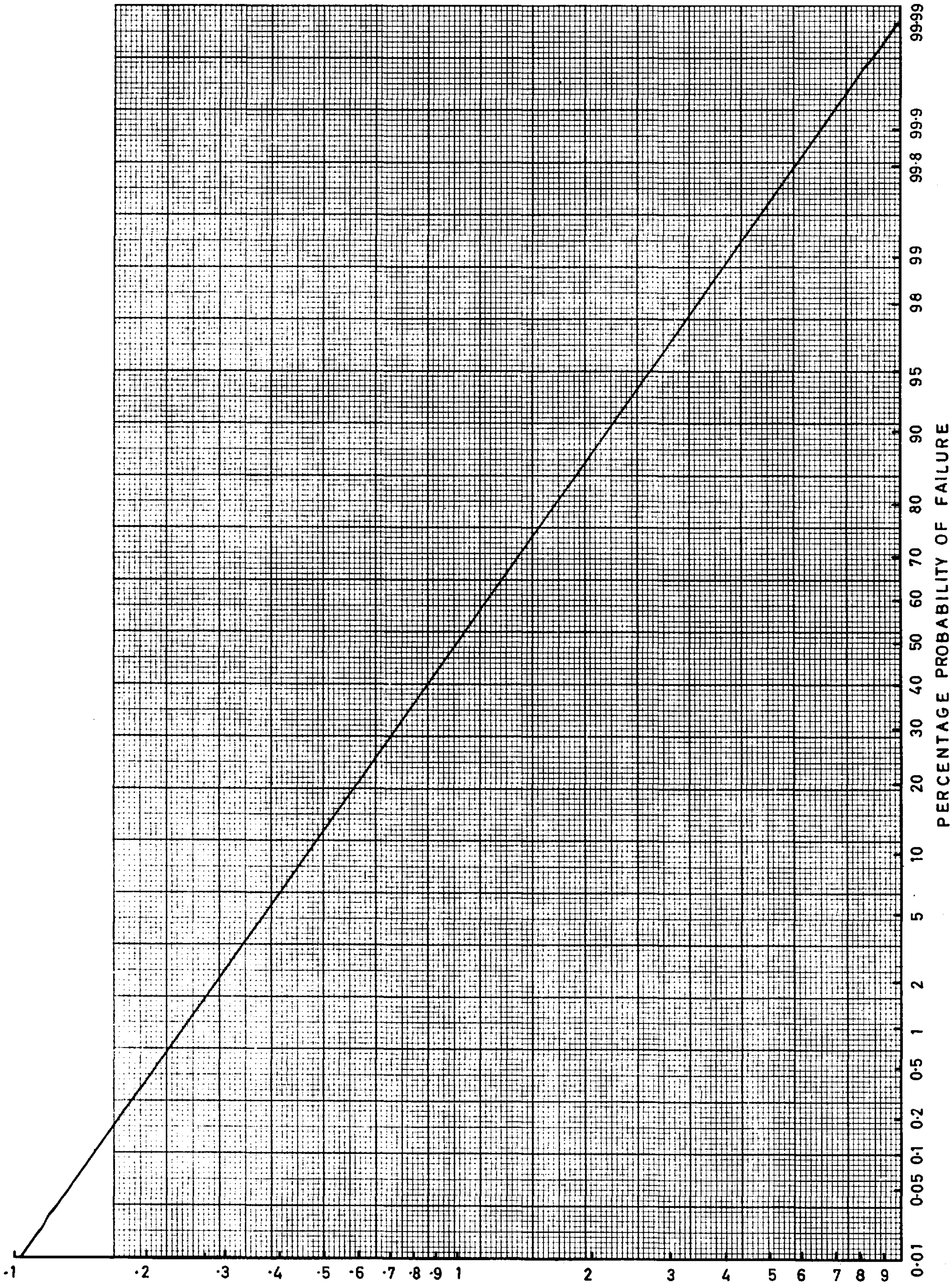


FIG.6 PERCENTAGE PROBABILITY OF FAILURE AS A FUNCTION OF FRACTION OF MEAN TIME TO FAILURE

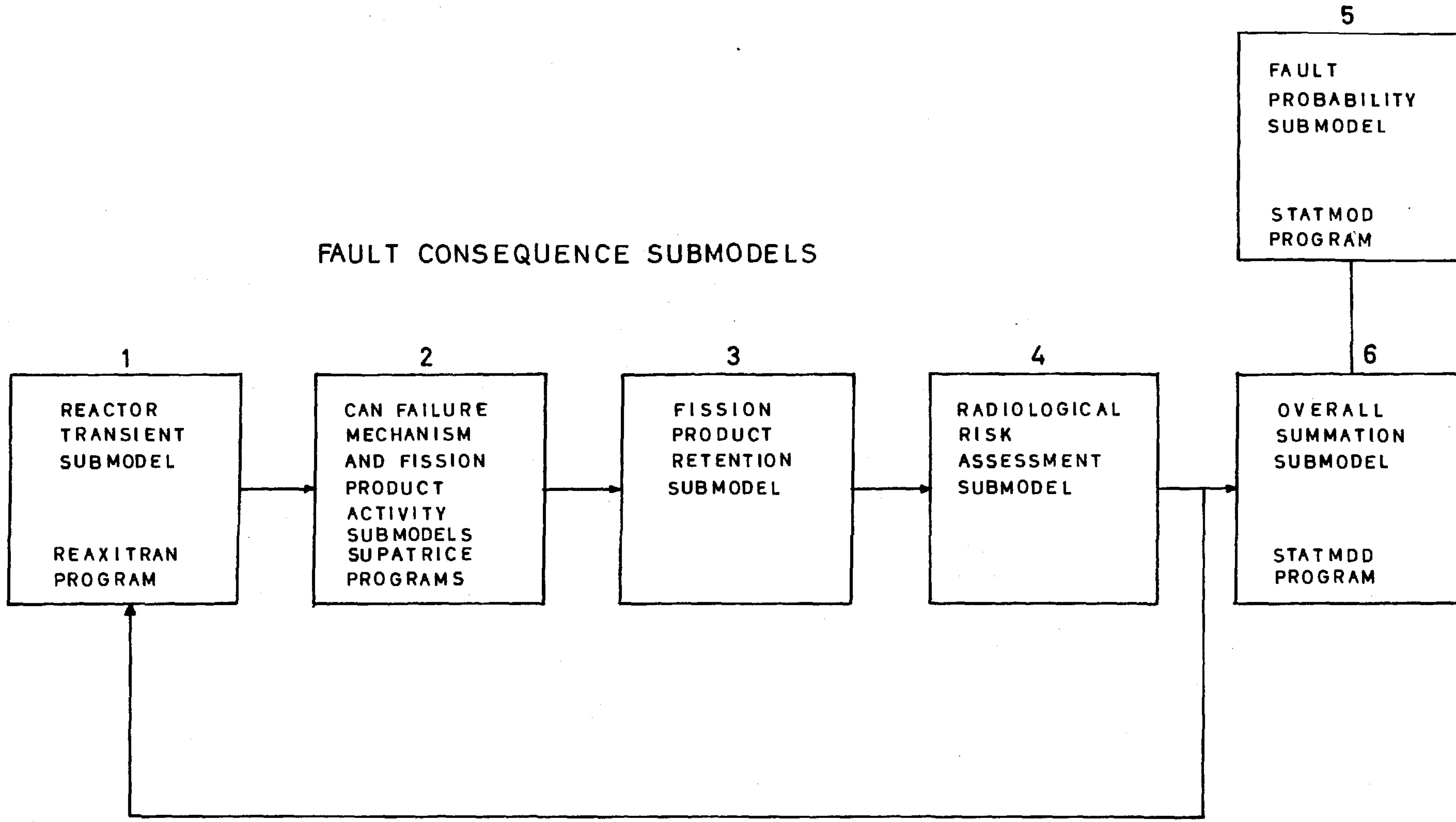


FIG.7 BLOCK SCHEME OF THE ALTERNATIVE APC COMPOUND MODEL

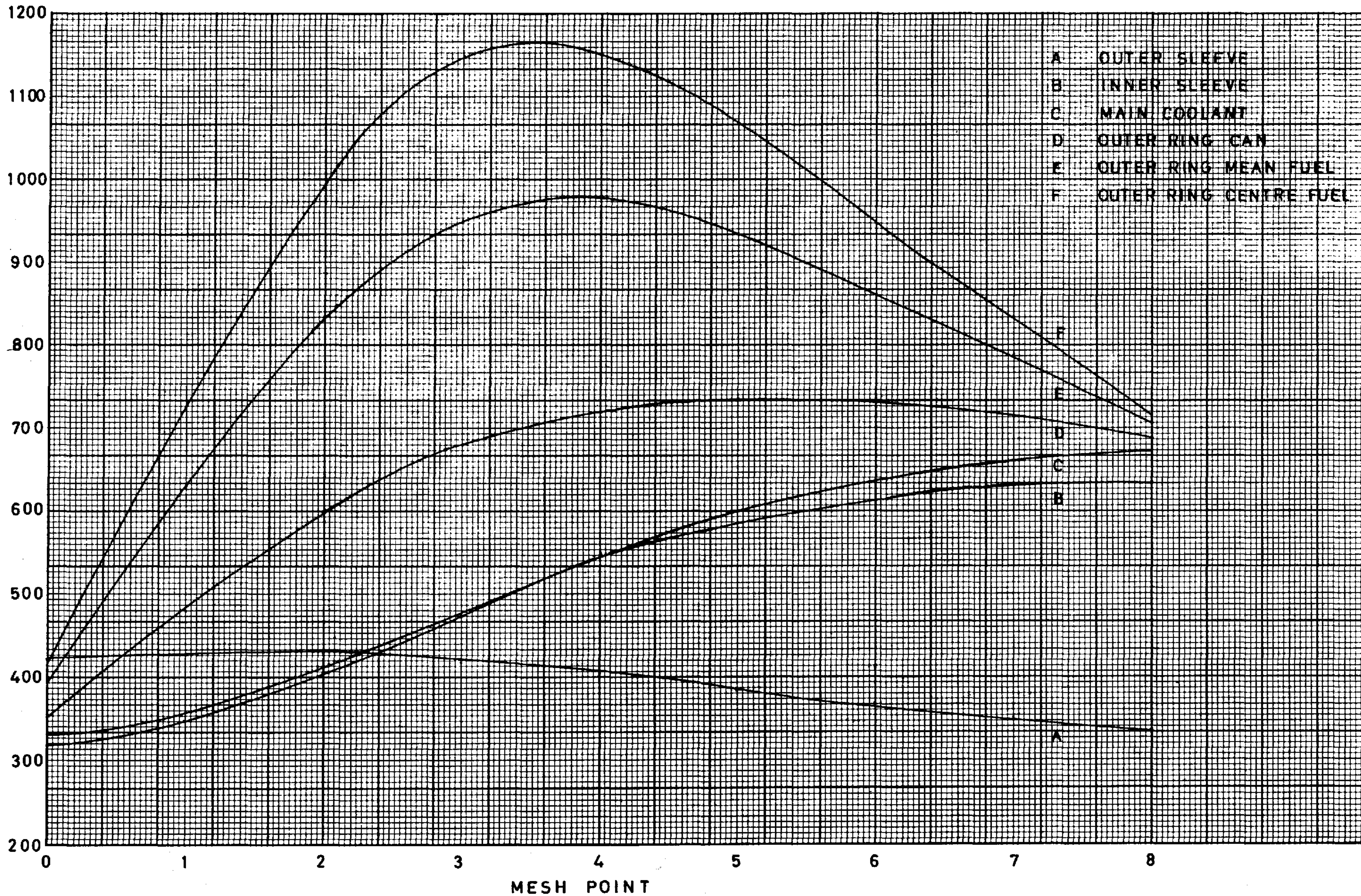


FIG. 8 TYPICAL AGR. AXIAL PLOTS OF TEMPERATURES IN NEWLY LOADED CHANNEL AT FULL POWER

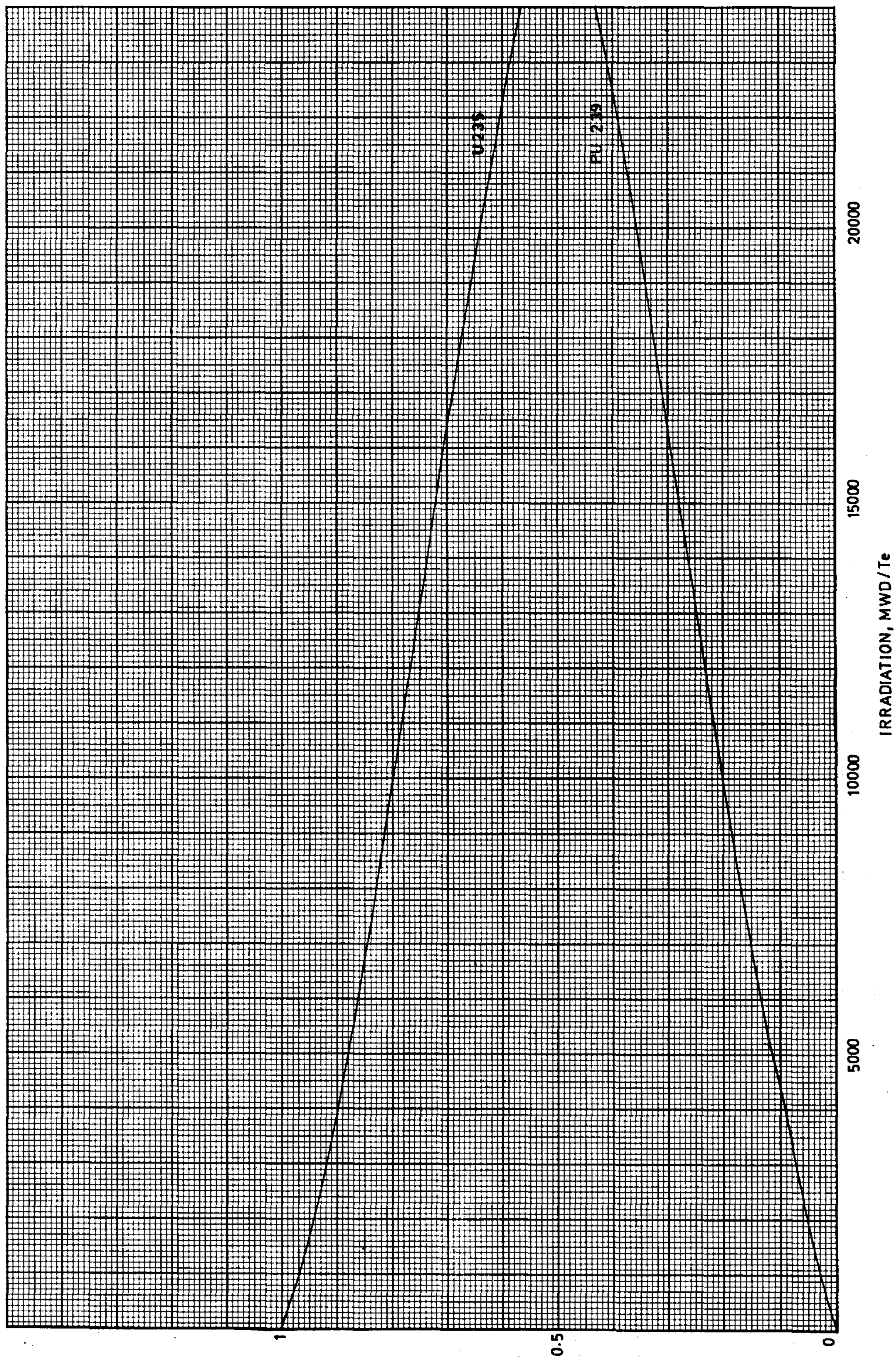


FIG.9 RELATIVE CONTENTS OF FISSIONABLE ISOTOPES vs IRRADIATION IN TYPICAL AGR TYPE FUEL.

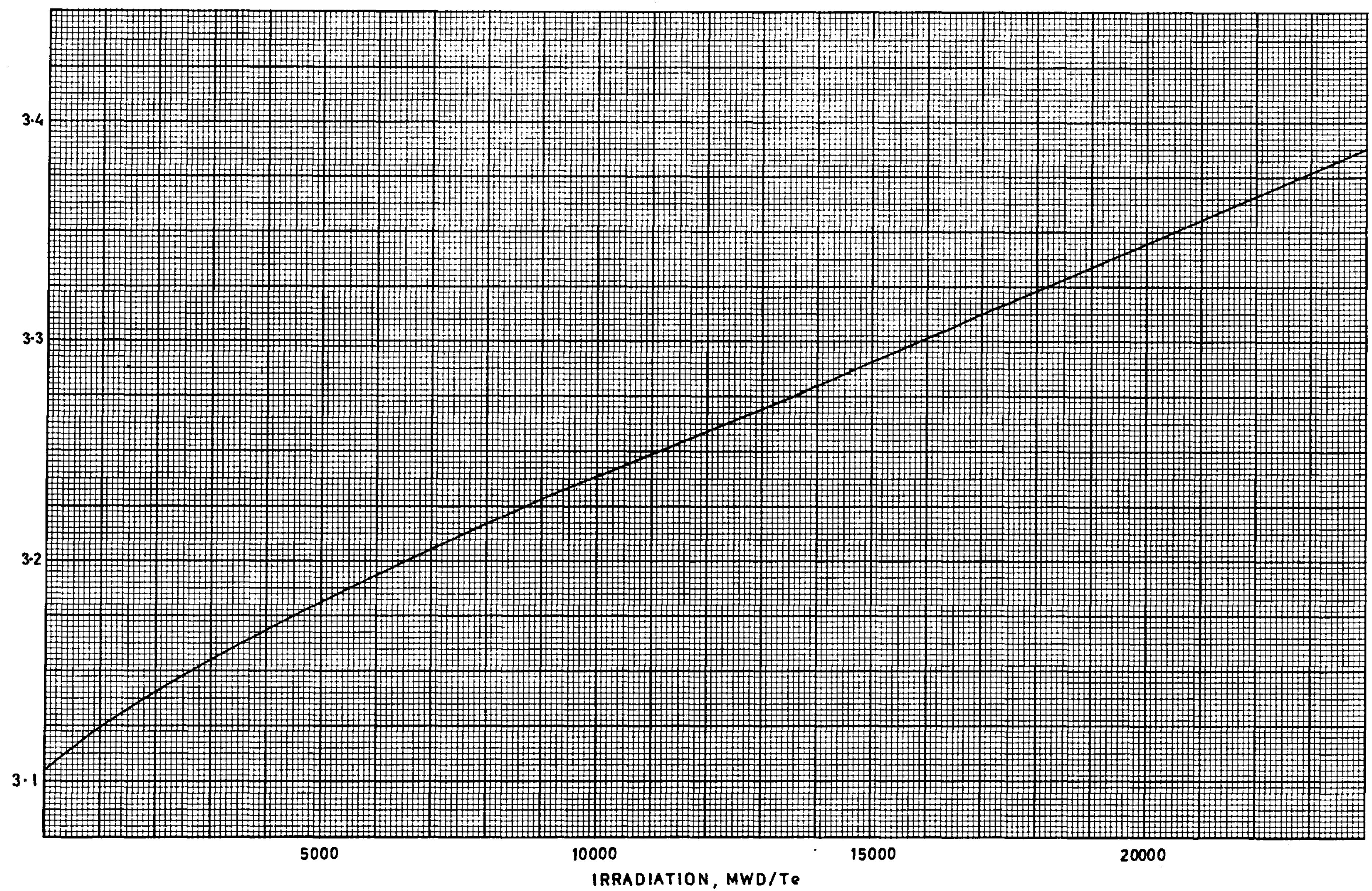


FIG.10 IODINE YIELD vs. IRRADIATION IN TYPICAL AGR TYPE FUEL.

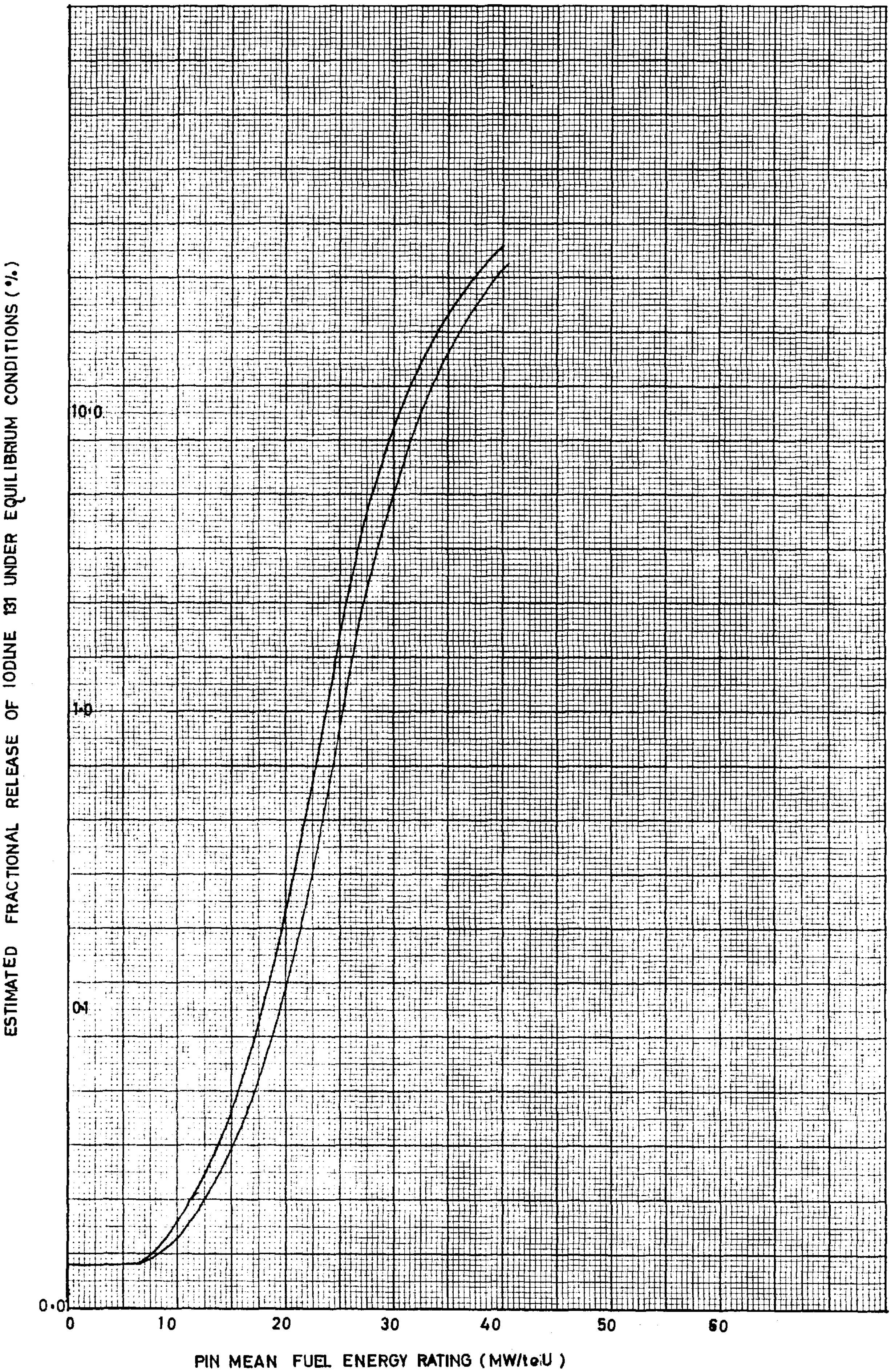


FIG 11 INDICATIVE FRACTIONAL RELEASE OF IODINE UNDER EQUILIBRIUM CONDITIONS AGAINST MEAN PIN RATING

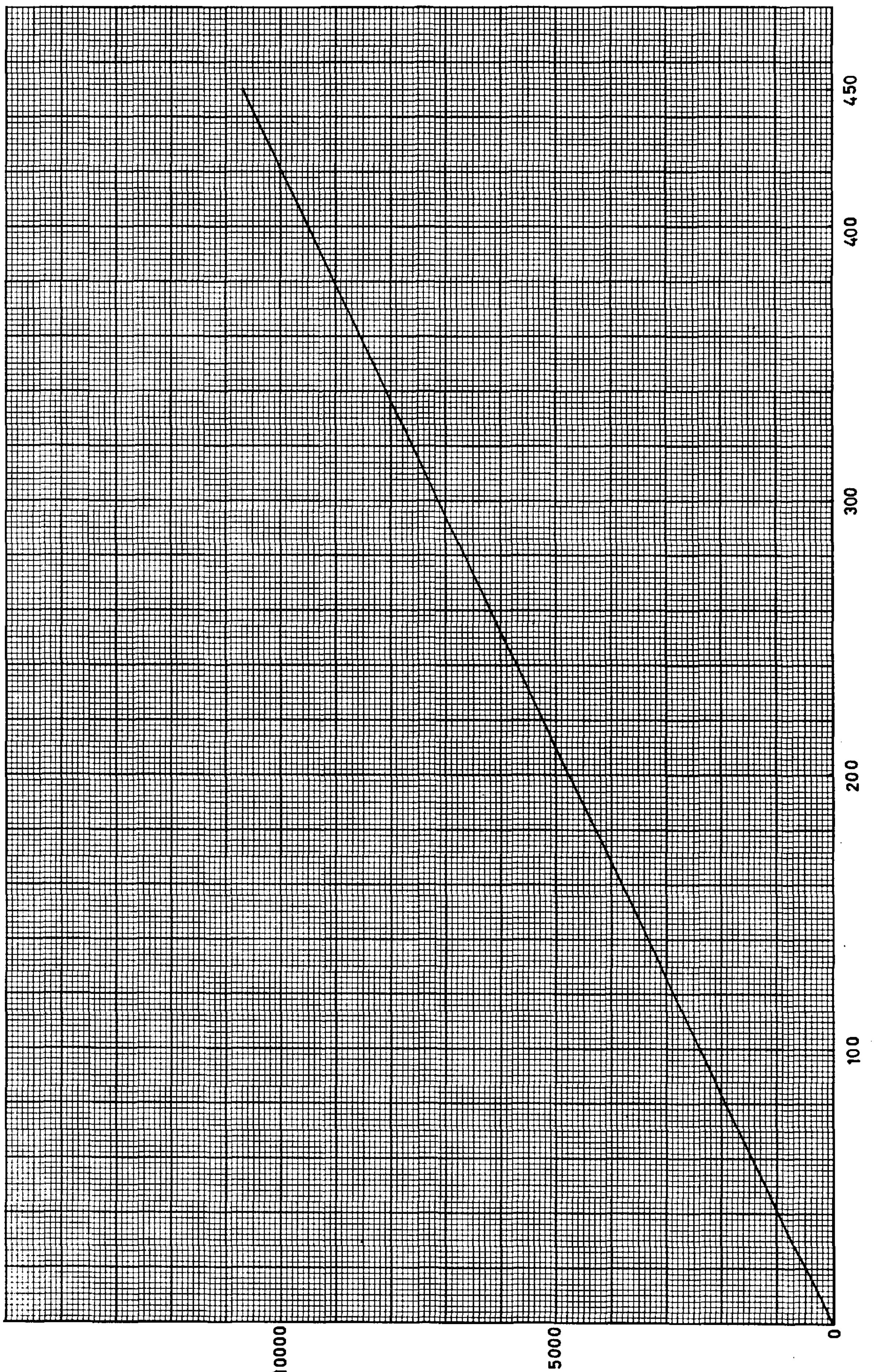


FIG.12 TYPICAL AGR. CAN HOOP STRESS vs. PRESSURE DIFFERENTIAL

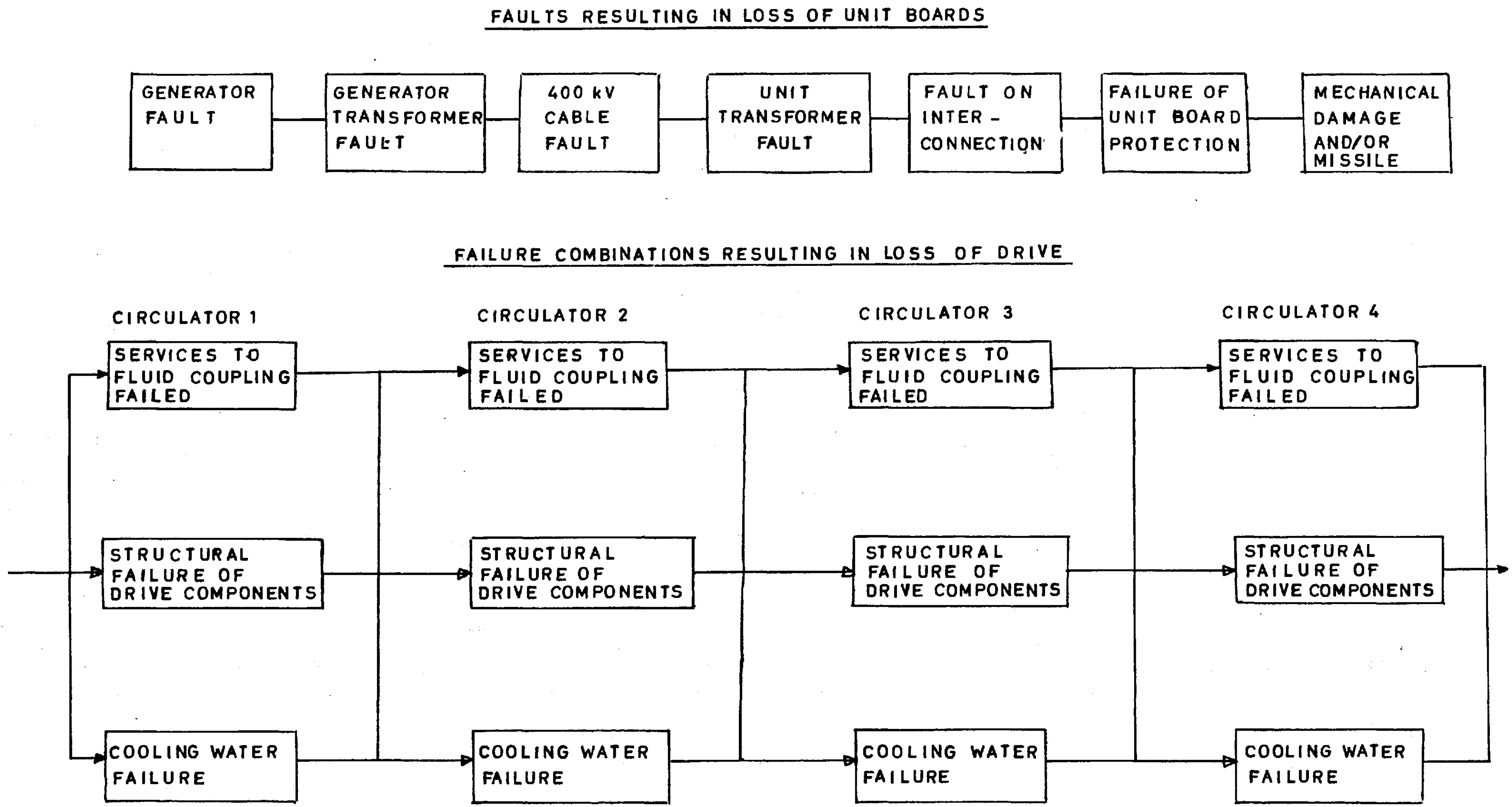


FIG.13 TENTATIVE RELIABILITY FUNCTIONAL BLOCK SCHEME ASSOCIATED WITH FAILURE OF SUPPLY TO ALL CIRCULATORS.

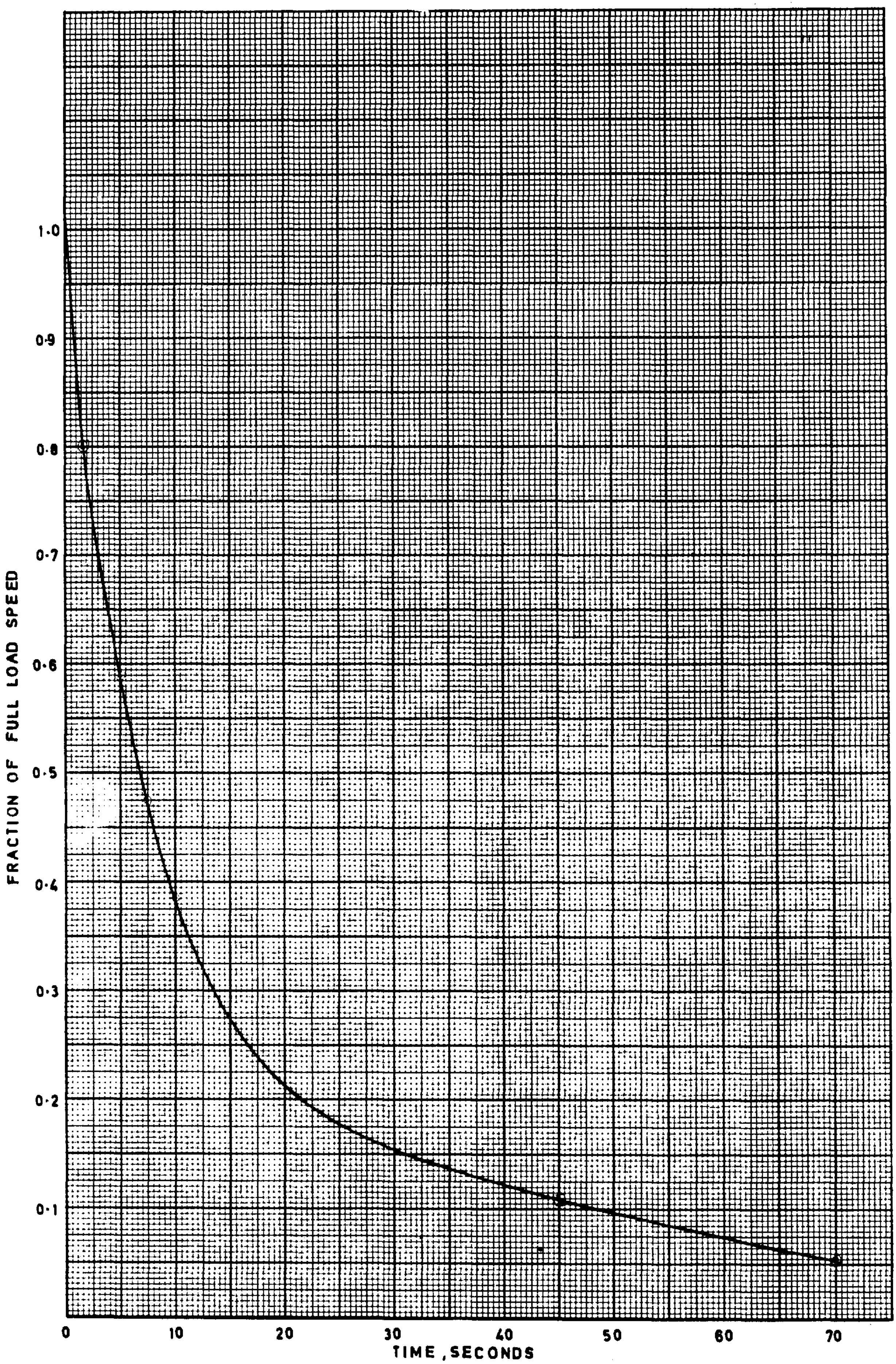


FIG. 14 TYPICAL AGR. CIRCULATOR UNISON RUNDOWN CHARACTERISTIC

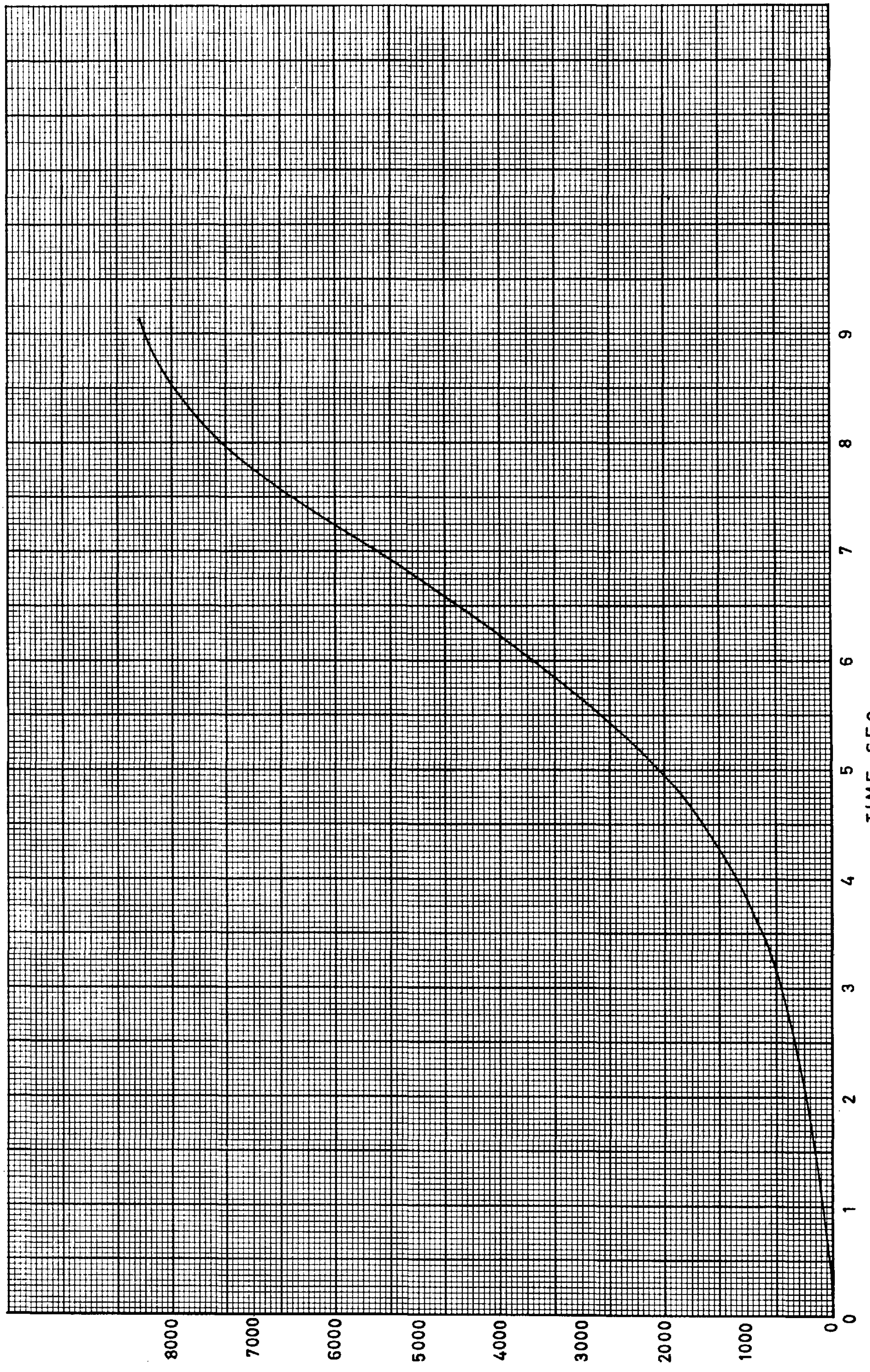


FIG. 15 REACTIVITY INSERTION CHARACTERISTIC FOLLOWING REACTOR TRIP

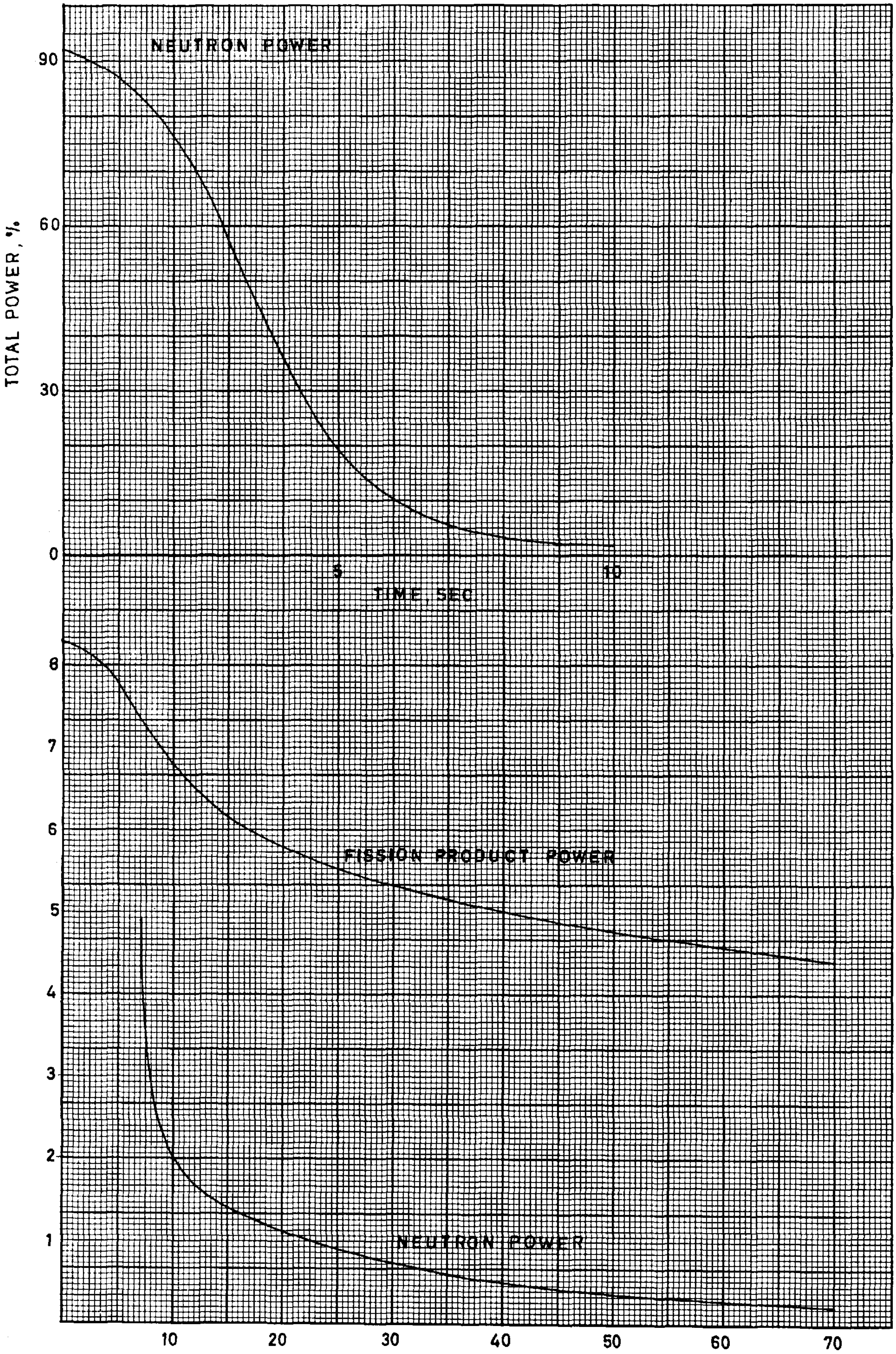


FIG.16 NEUTRON AND FISSION PRODUCT POWER HISTORIES FOLLOWING REACTOR TRIP AT FULL POWER

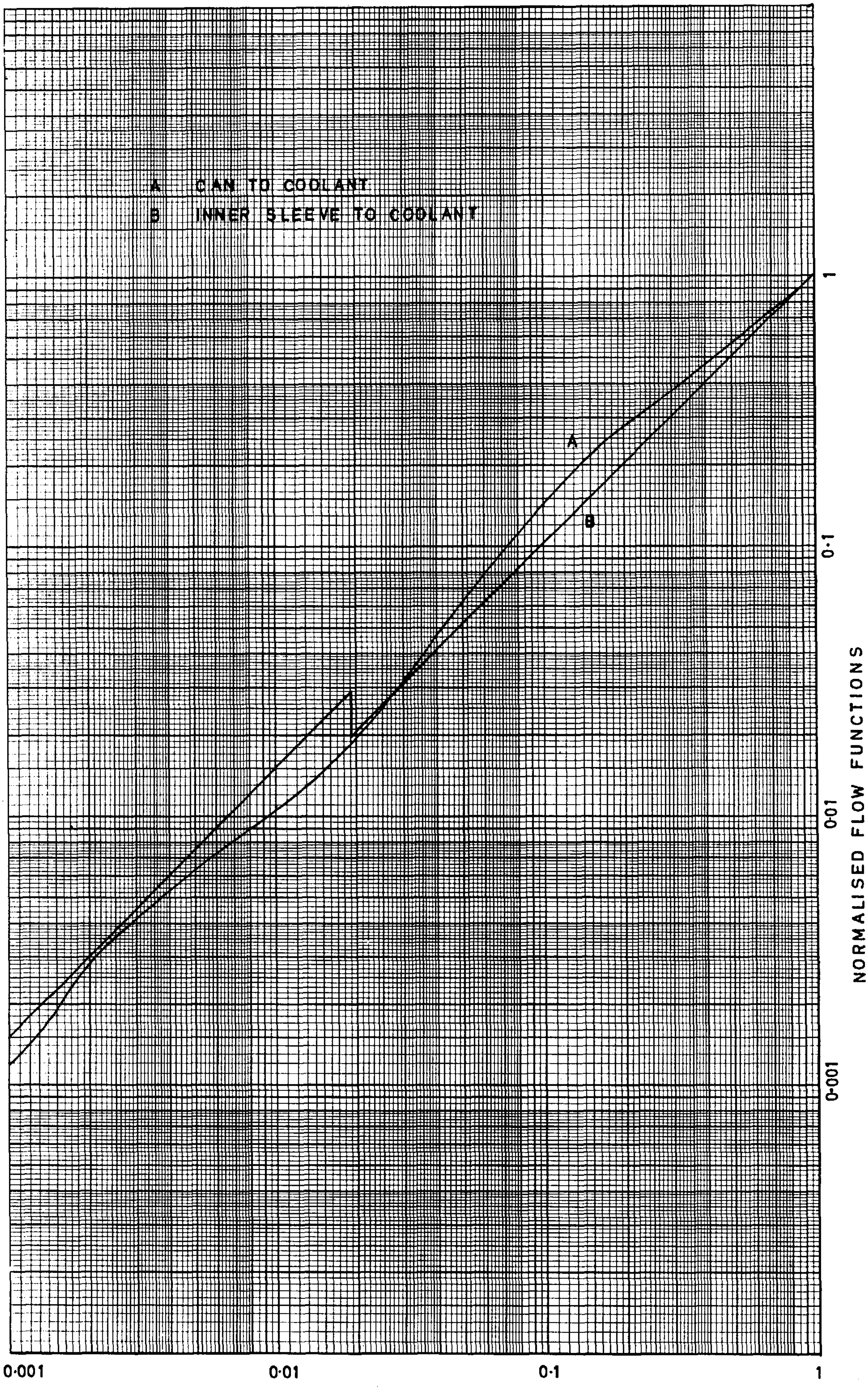


FIG. 17 CONVECTIVE HEAT TRANSFER COEFFICIENTS. VARIATIONS WITH FLOW

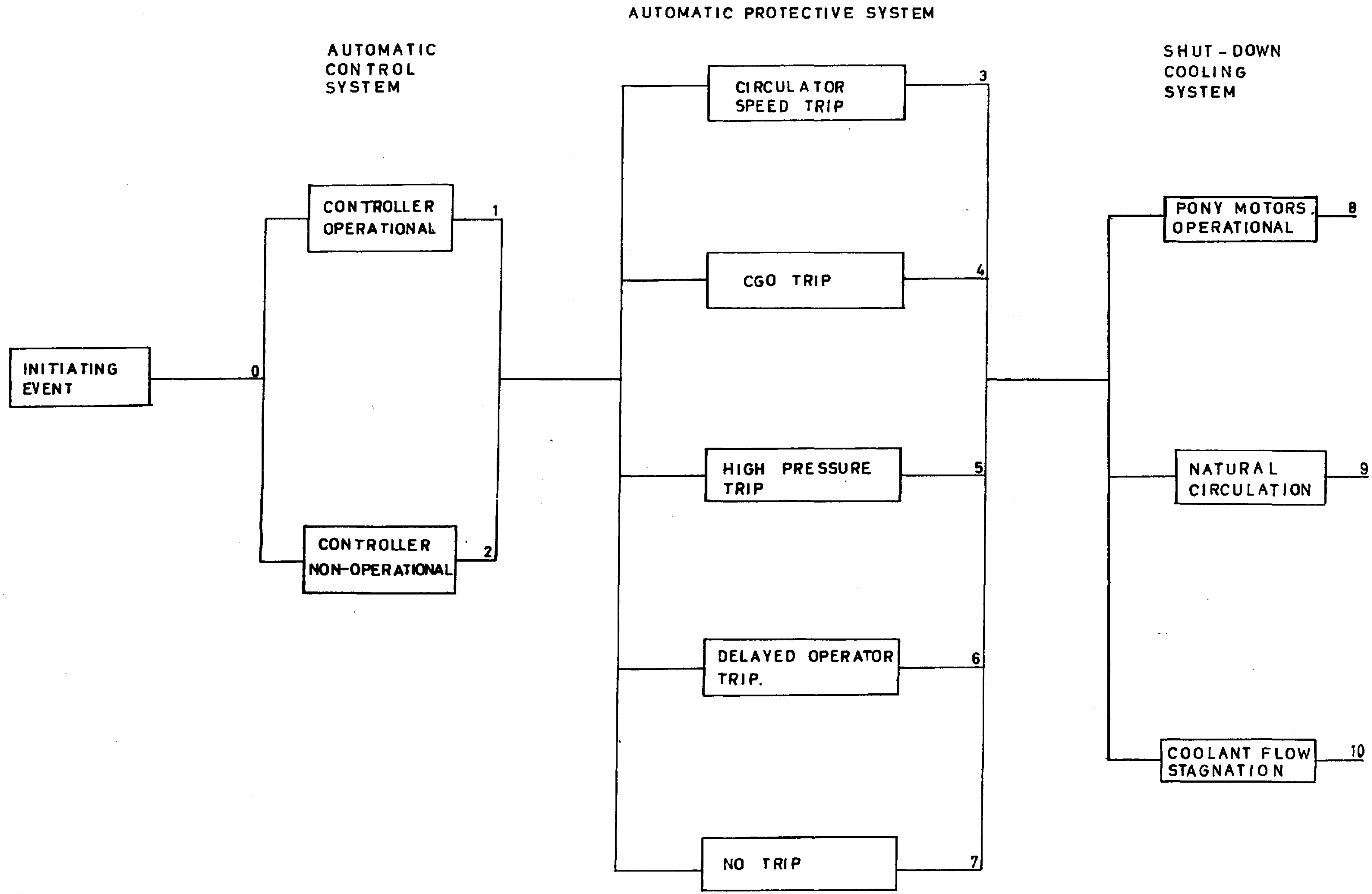


FIG.18 ALL CIRCULATOR FAILURE. FAULT PATH DIAGRAM.

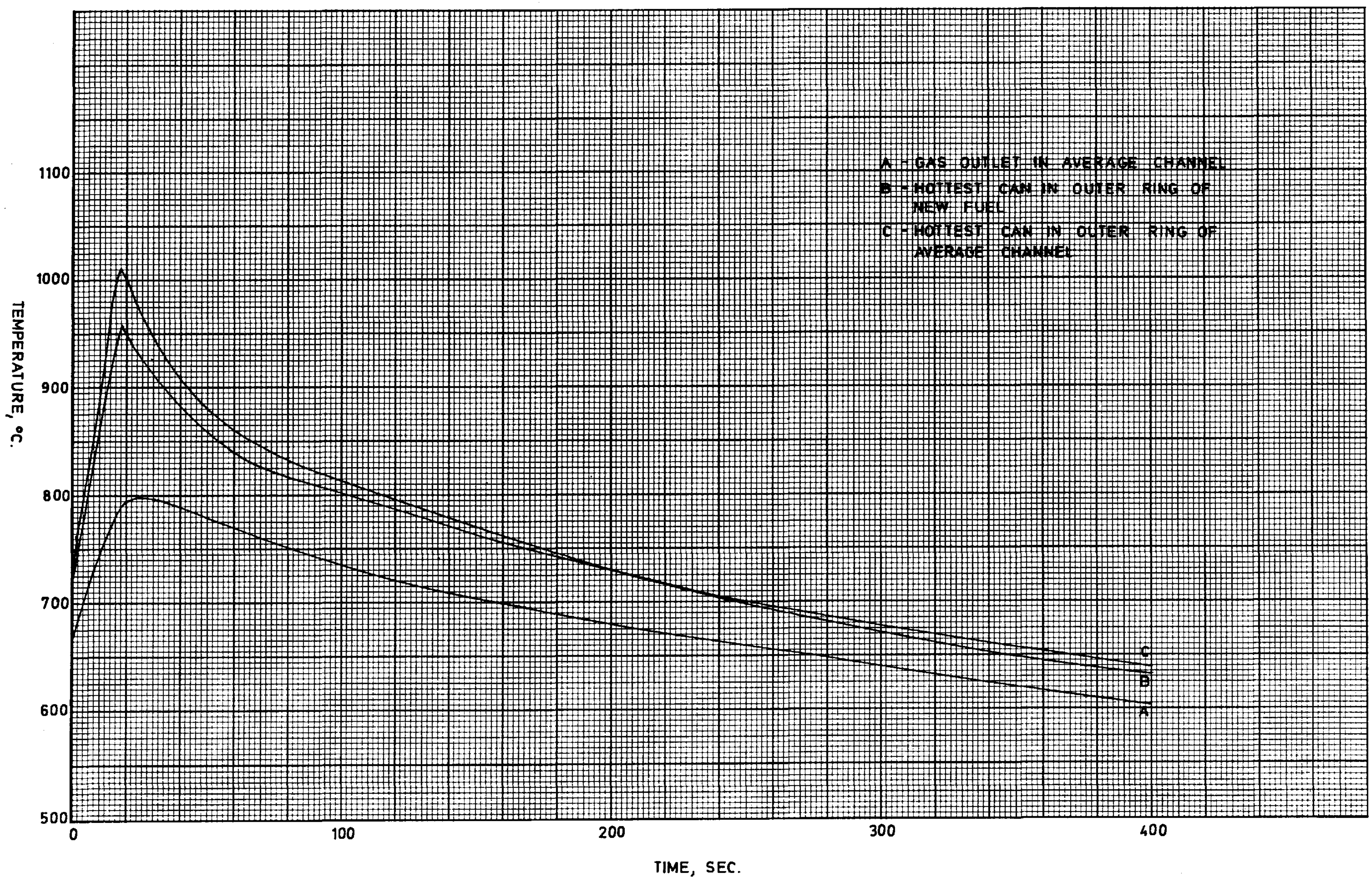


FIG.19 TYPICAL AGR. ALL CIRCULATOR FAILURE, TRIP ON MEASURED GAS OUTLET TEMPERATURE, PONY MOTORS OPERATIONAL. TEMPERATURE TRANSIENTS

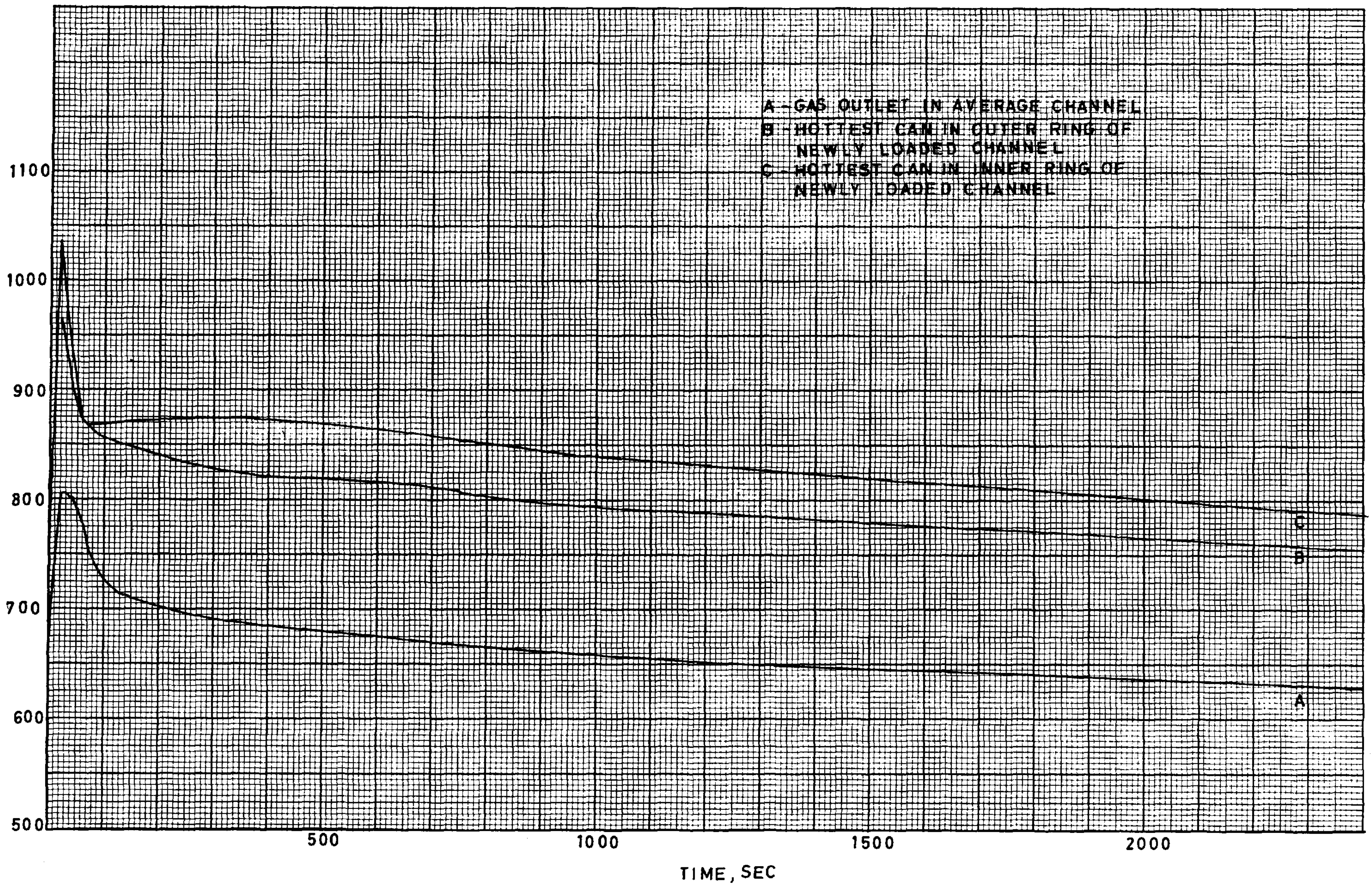


FIG.20 TYPICAL AGR. LOSS OF SUPPLY TO ALL CIRCULATORS. TRIP ON MEASURED GAS OUTLET TEMPERATURE. NATURAL CIRCULATION. TEMPERATURE TRANSIENTS.

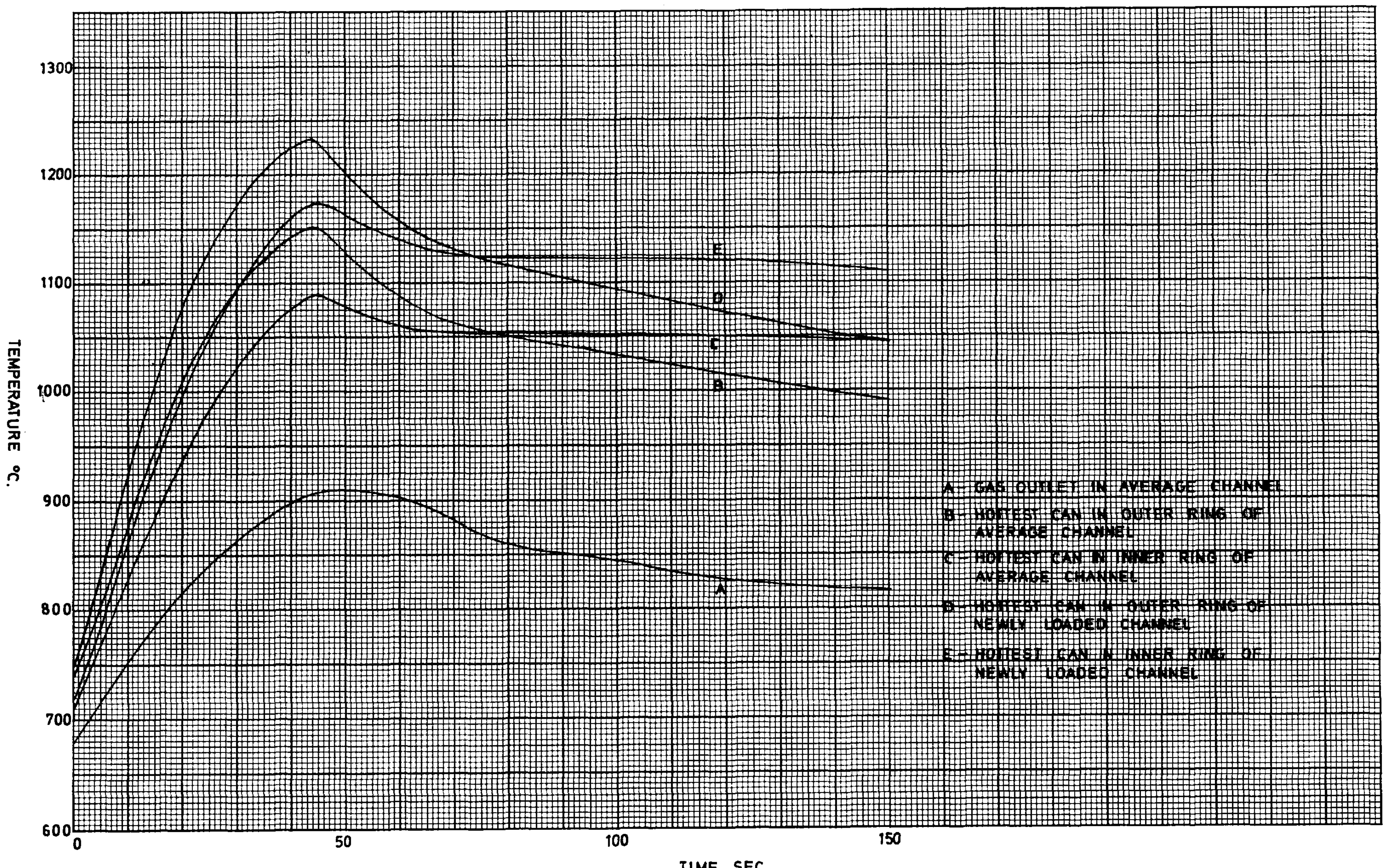


FIG.21 TYPICAL A.G.R. LOSS OF SUPPLY TO ALL CIRCULATORS.
REACTOR TRIP ON HIGH PRESSURE SIGNAL CONTROLLER
NON OPERATIONAL TEMPERATURE TRANSIENTS.

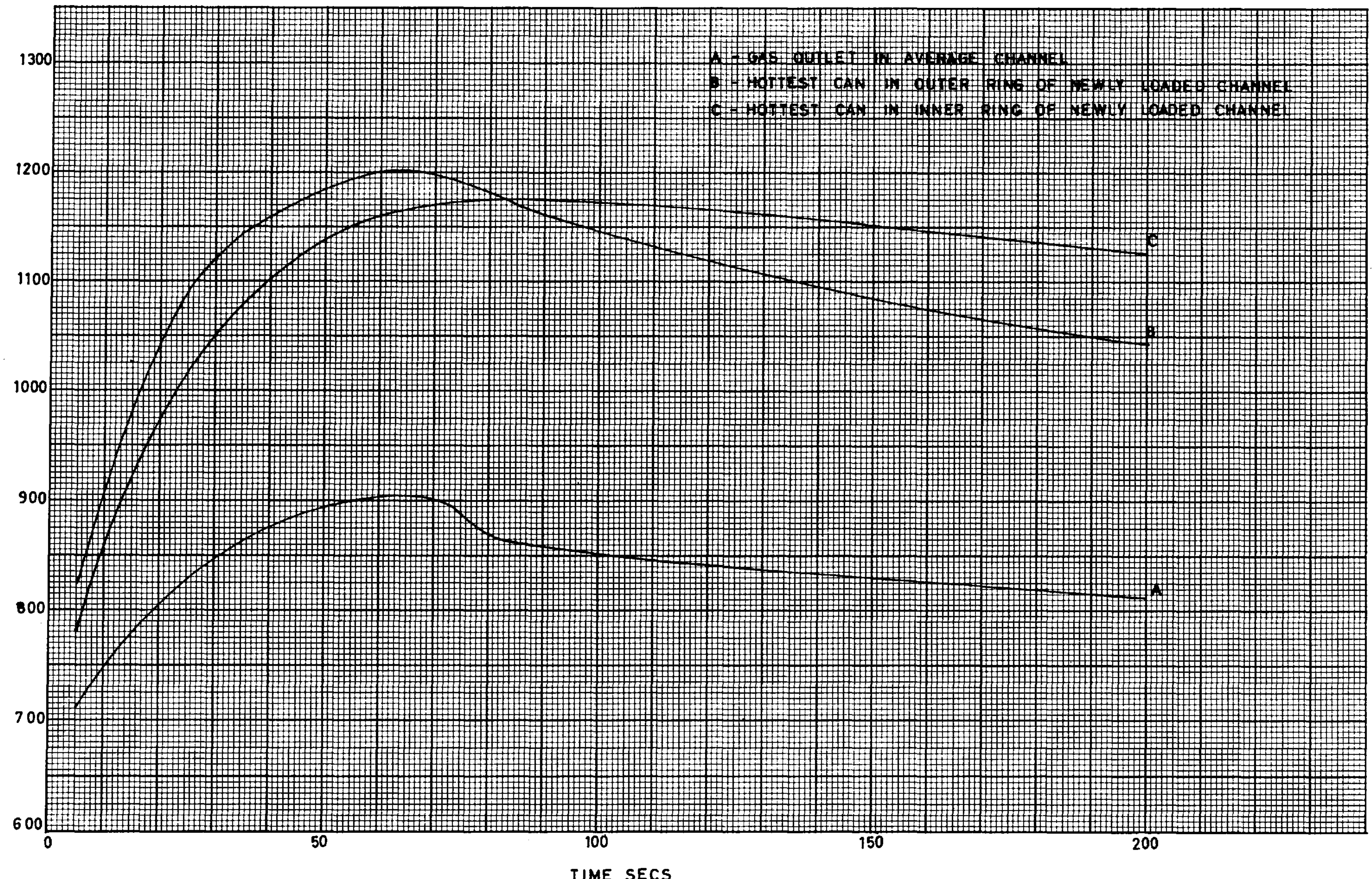


FIG.22 TYPICAL A.G.R. ALL CIRCULATOR FAILURE. REACTOR TRIP ON HIGH PRESSURE SIGNAL. CONTROLLER OPER-ATIONAL. TEMPERATURE TRANSIENTS.

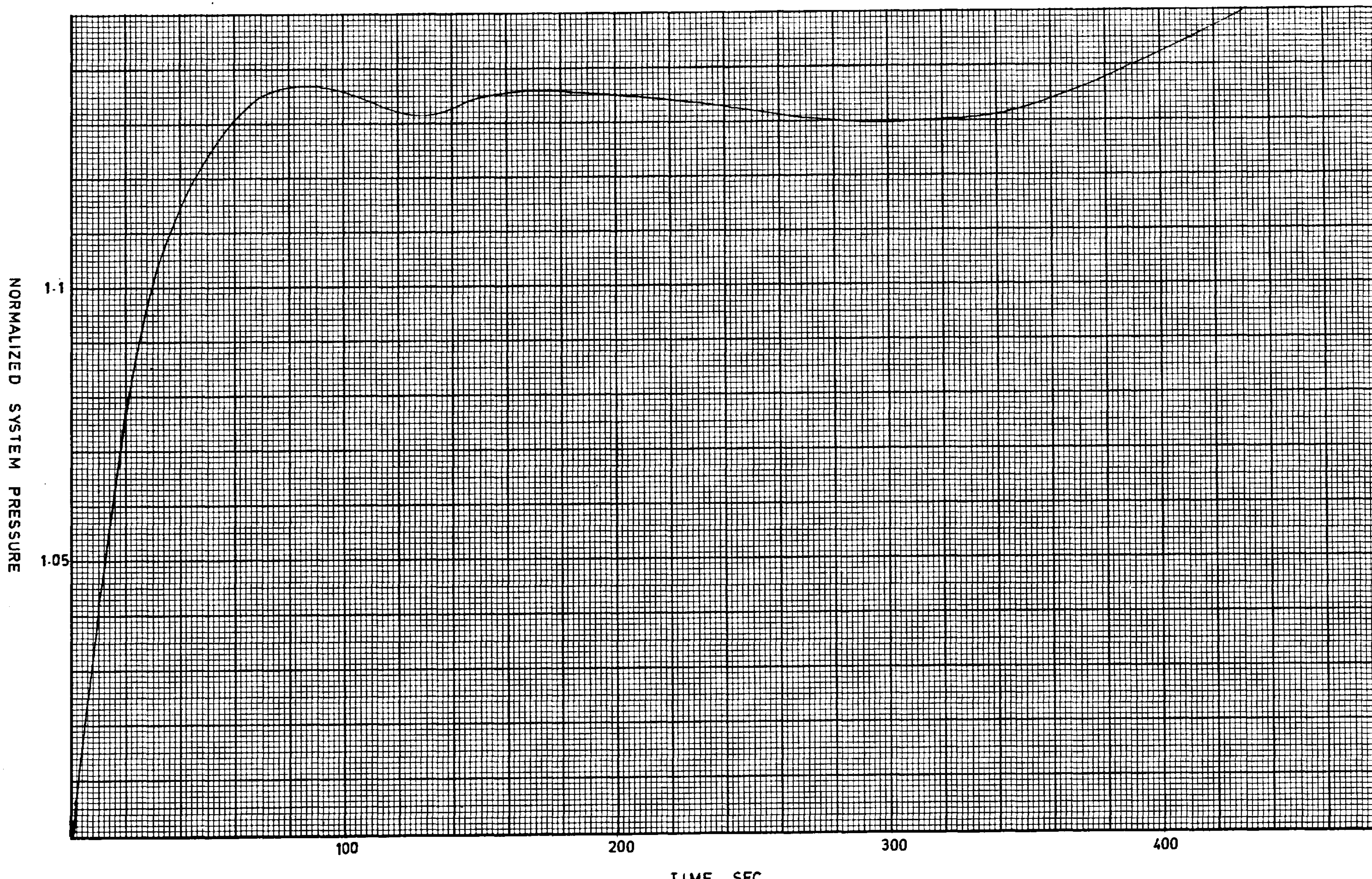


FIG.23 TYPICAL AGR. MEAN SYSTEM PRESSURE TRANSIENT FOLLOWING ALL CIRCULATOR FAILURE NON-TRIPPED CASE

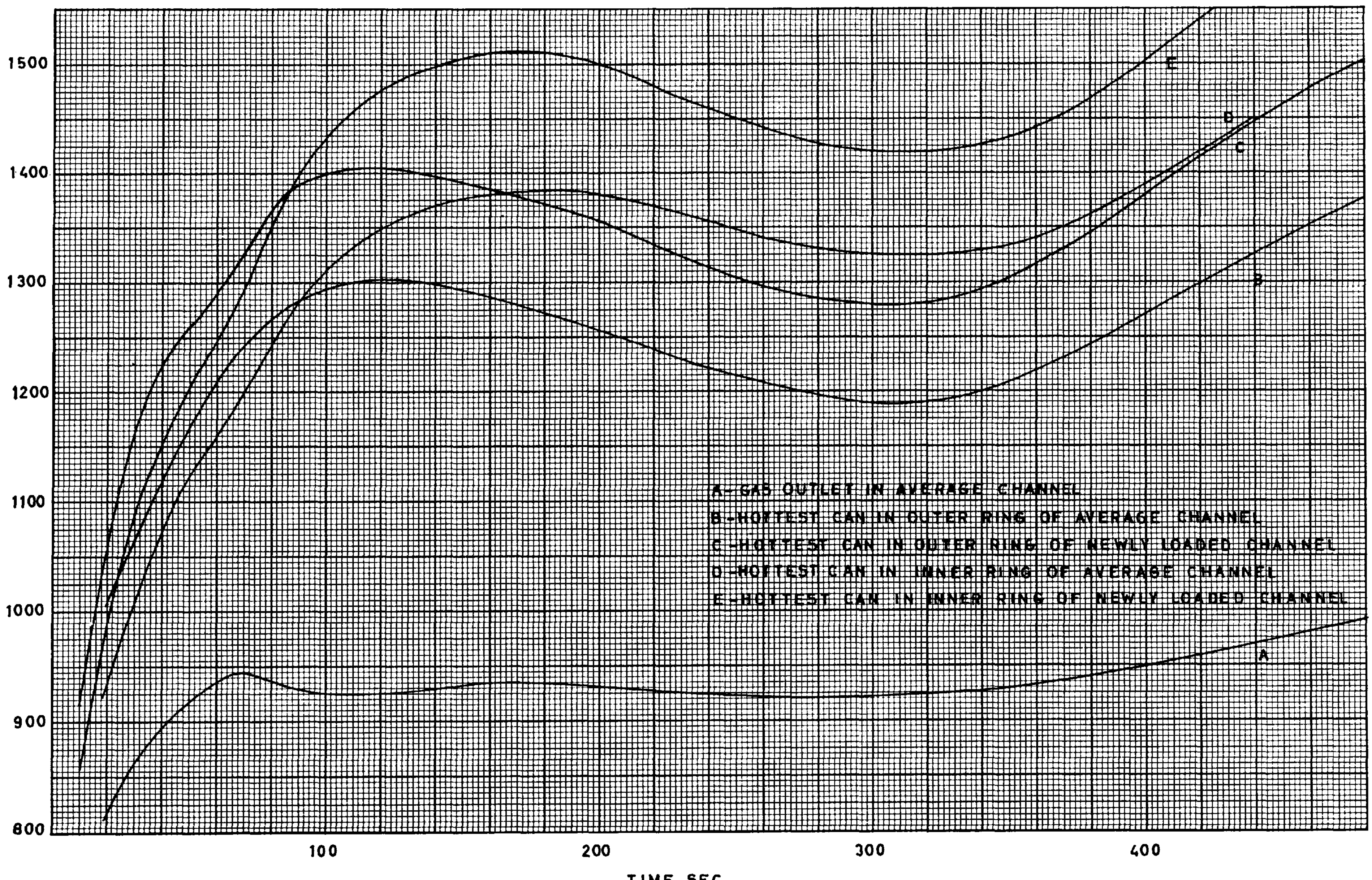


FIG. 24 TYPICAL A.G.R. ALL - CIRCULATOR FAILURE. REACTOR NON-TRIPPED. TEMPERATURE TRANSIENTS

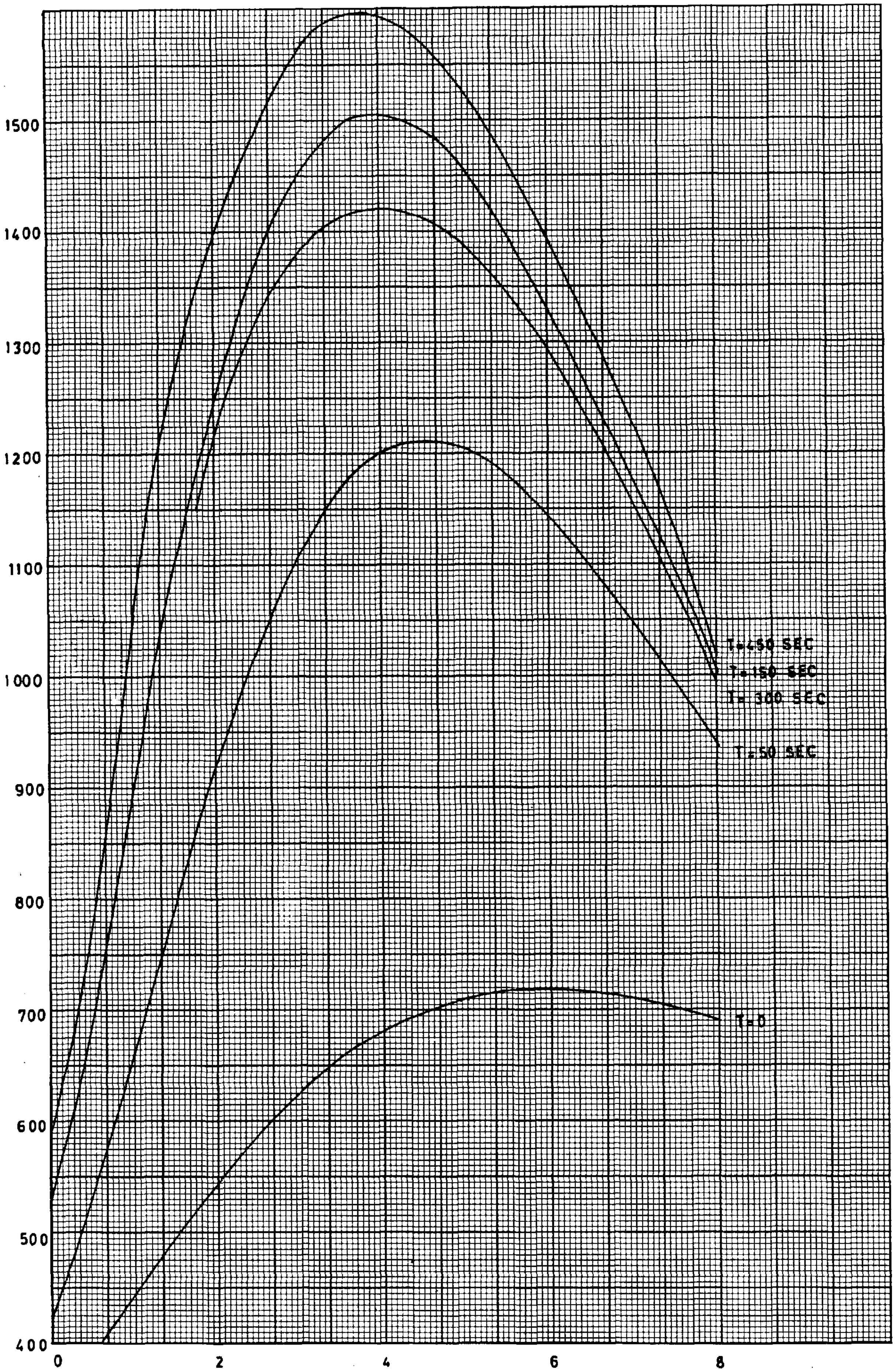


FIG.25 TYPICAL AGR. ALL CIRCULATOR FAILURE REACTOR NON-TRIPPED. AXIAL PROFILES OF INNER RING CAN TEMPERATURES IN NEWLY LOADED CHANNEL.

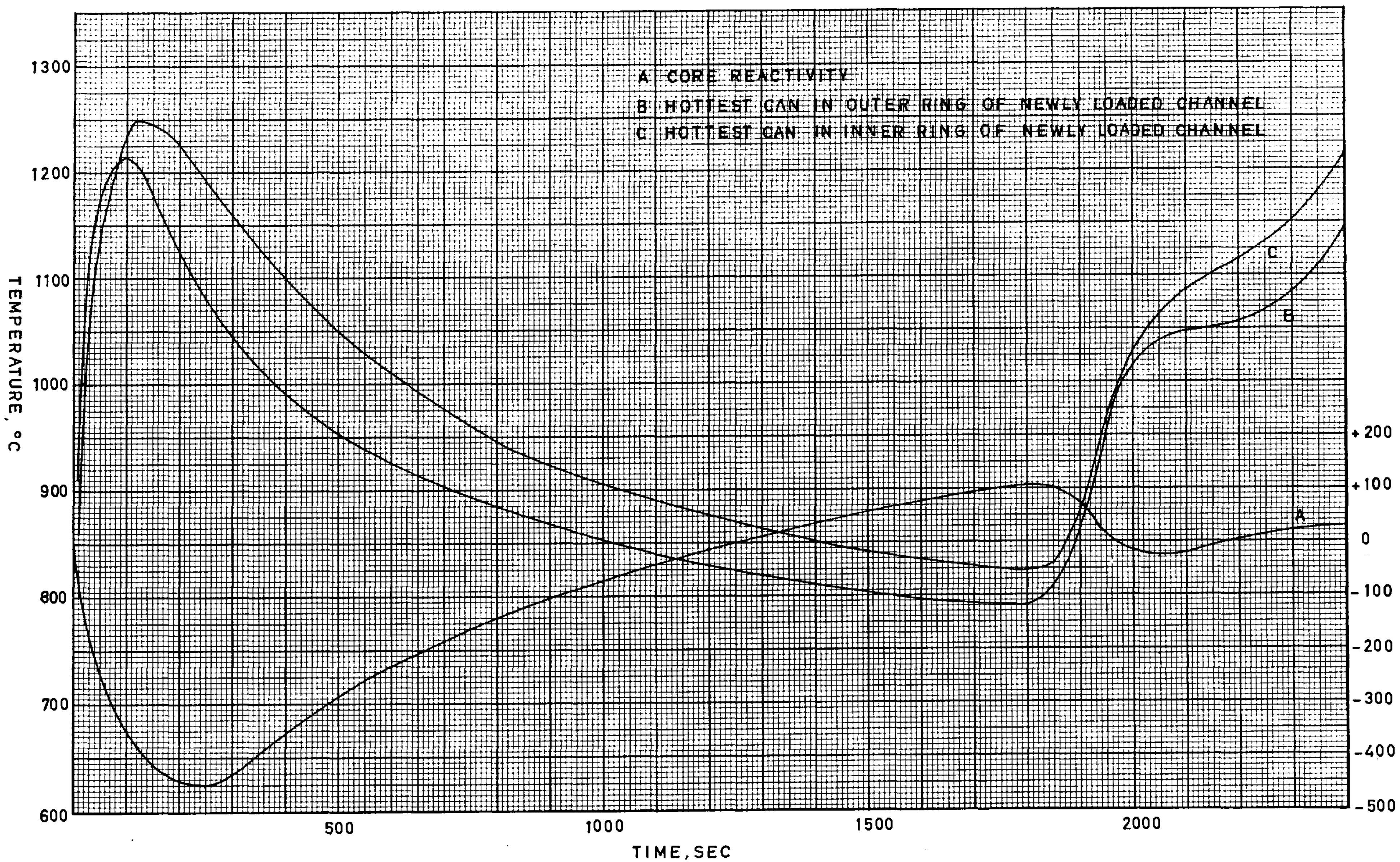


FIG.26 TYPICAL AGR. ALL CIRCULATOR FAILURE. REACTOR NON-TRIPPED. CONTROLLER OPERATIONAL. TRANSIENTS

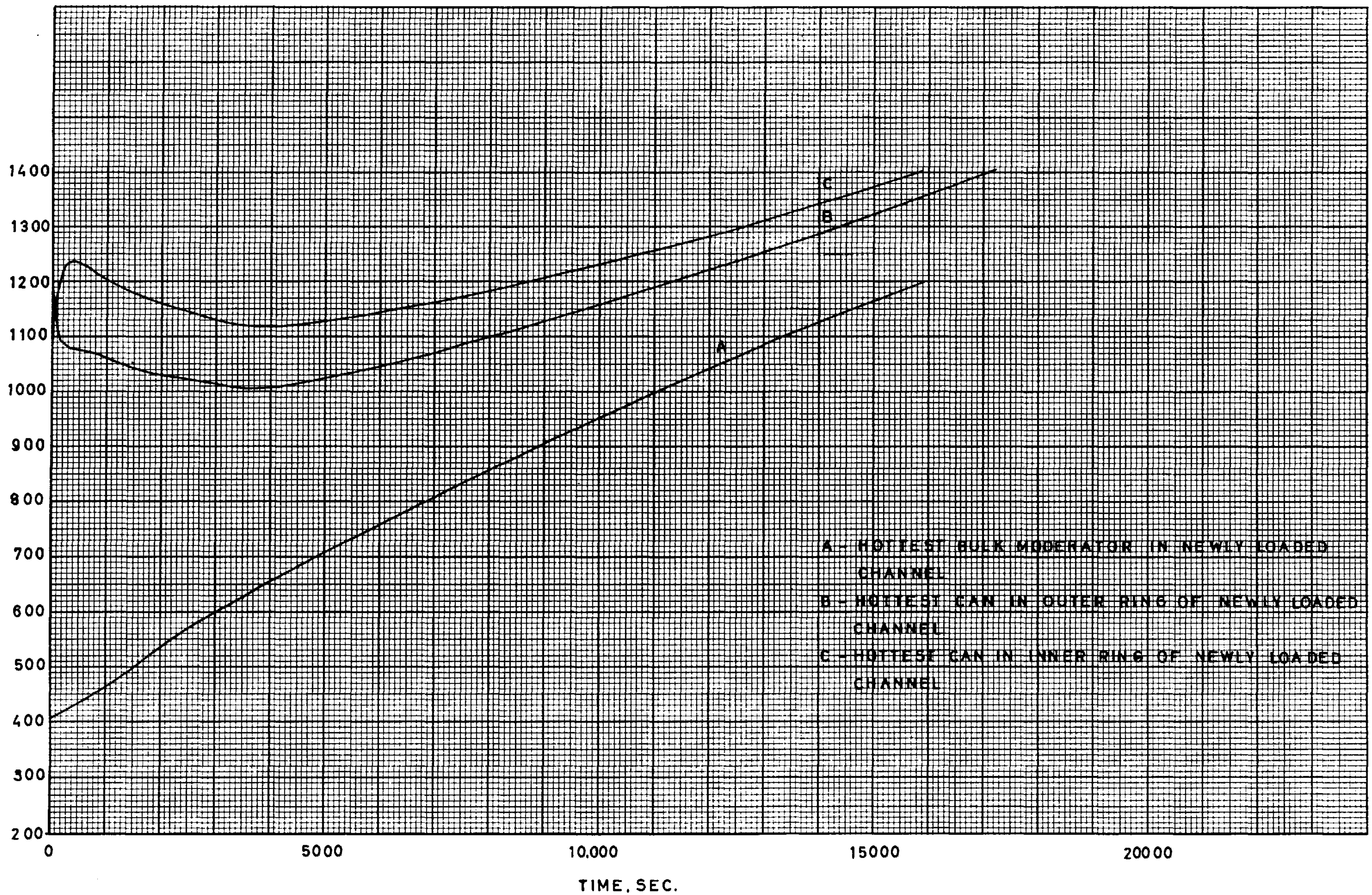


FIG.27 TYPICAL A.G.R. HYPOTHETICAL CASE (INSTANTANEOUS FLOW STAGNATION FOLLOWED BY REACTOR TRIP). TEMPERATURE TRANSIENTS IN NEWLY LOADED CHANNEL.

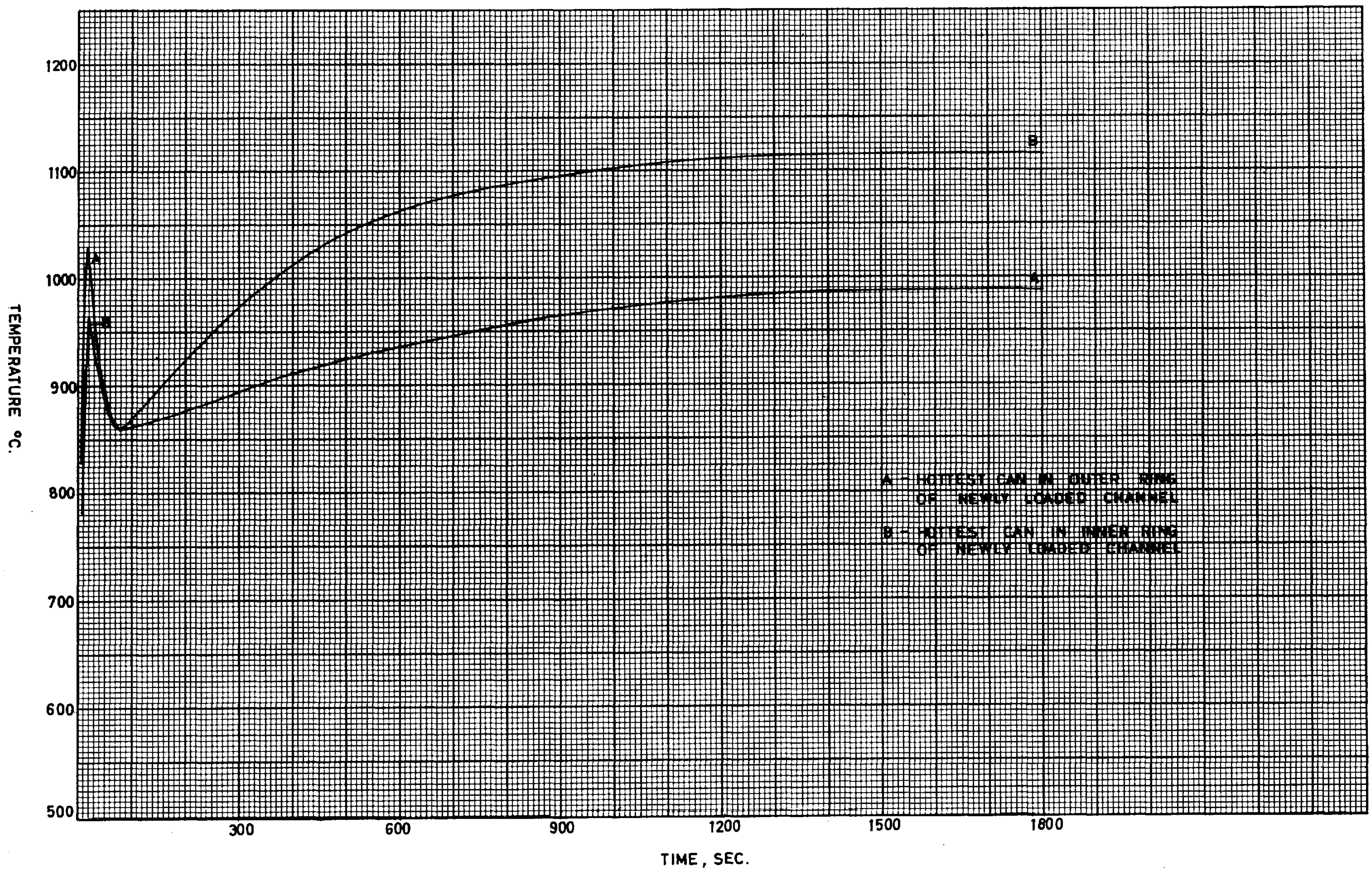


FIG. 28 TYPICAL AGR ALL CIRCULATOR FAILURE. TRIP ON MEASURED GAS OUTLET TEMPERATURE. FLOW STAGNATION. TEMPERATURE TRANSIENTS.

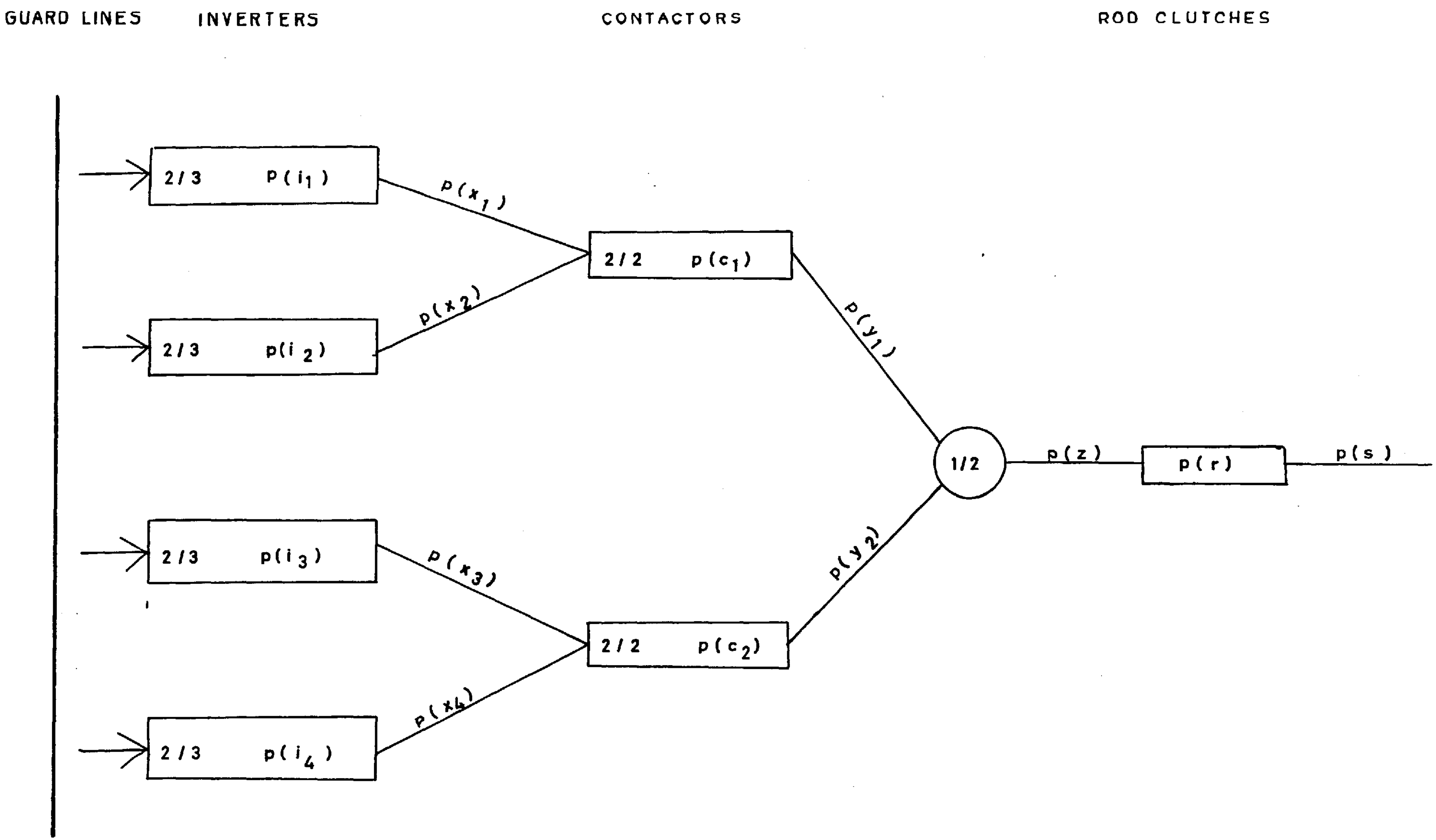


FIG.29 ABBREVIATED RELIABILITY BLOCK DIAGRAM FOR AUTOMATIC PROTECTIVE SYSTEM

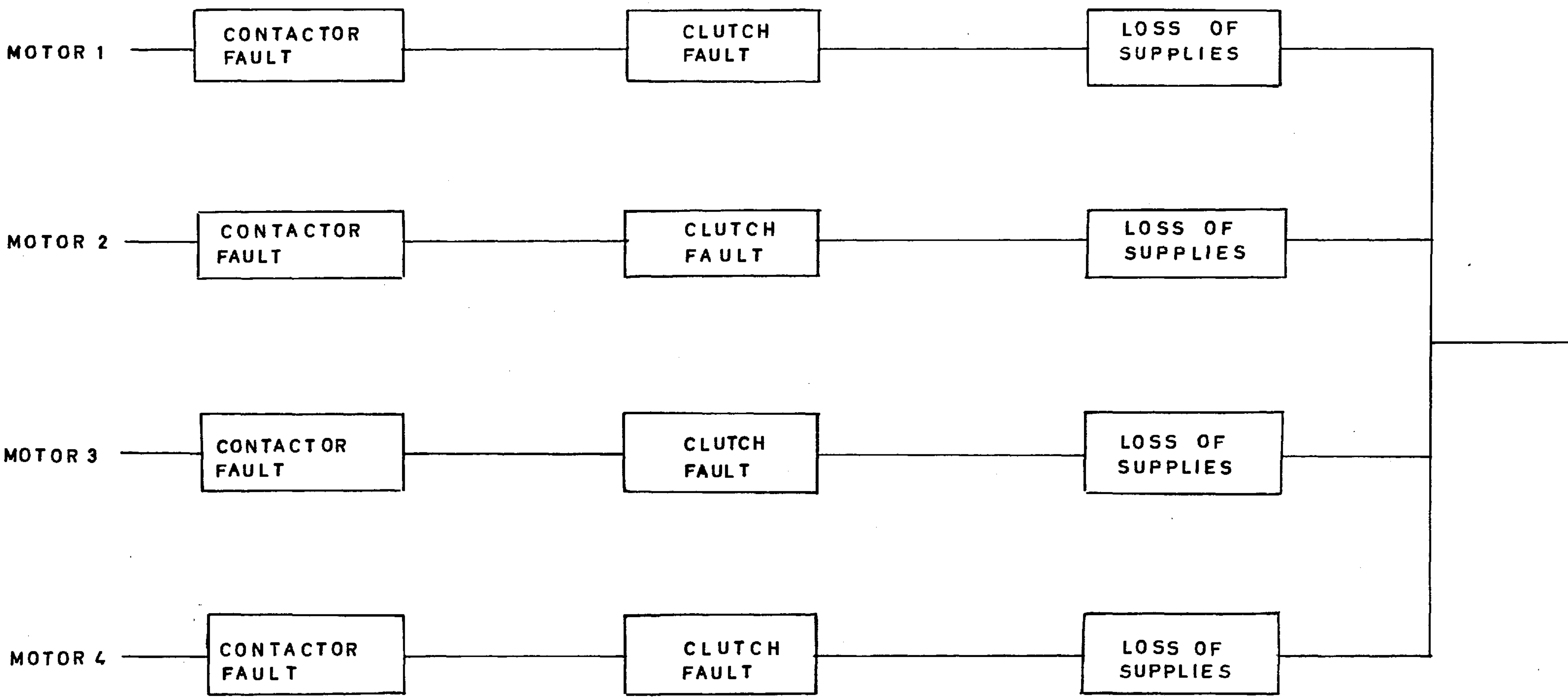


FIG. 30 EMERGENCY COOLING SYSTEM. TENTATIVE RELIABILITY FUNCTIONAL BLOCK DIAGRAM

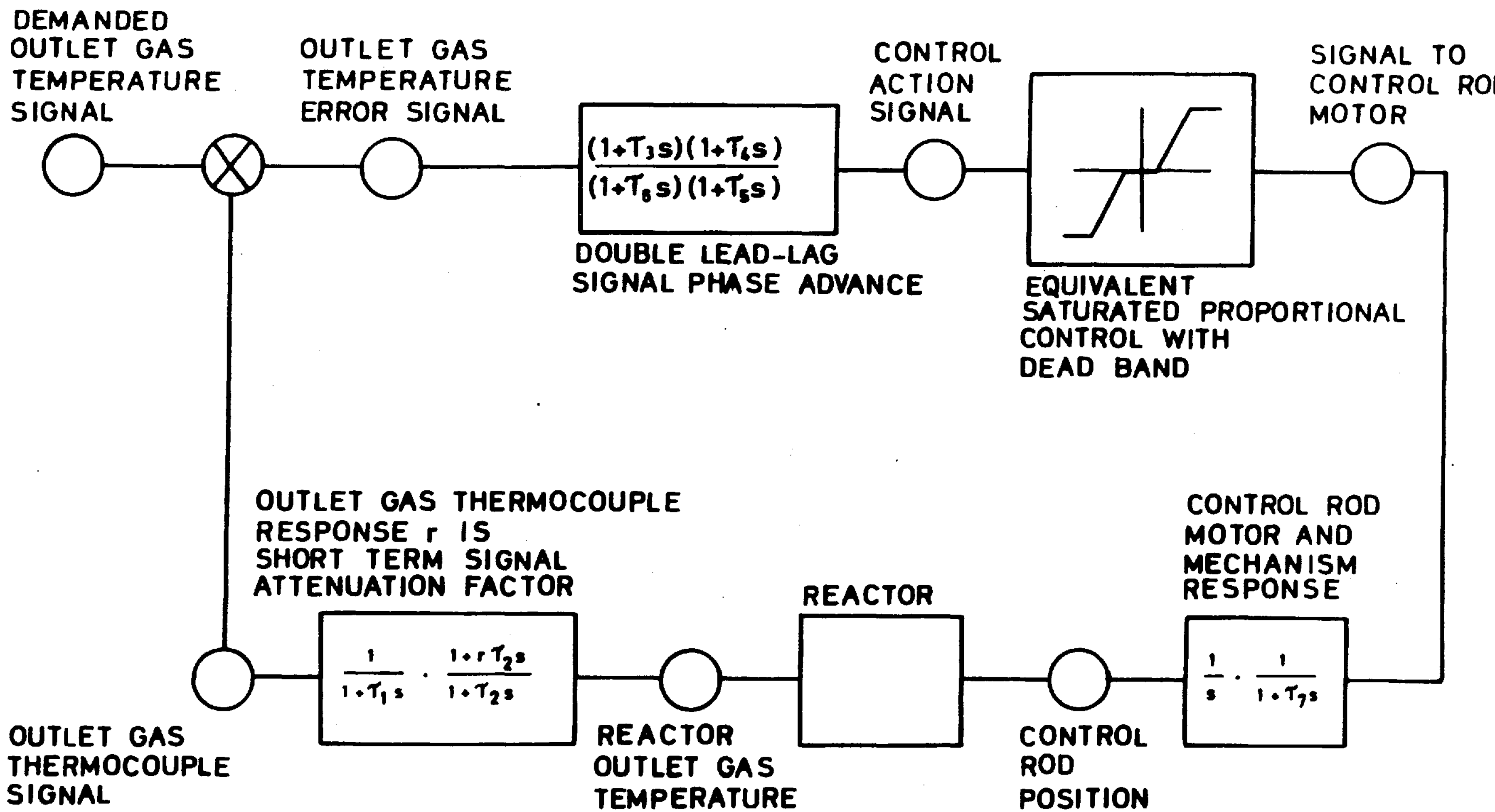


FIG.31 ZONE CONTROL LOOP BLOCK DIAGRAM

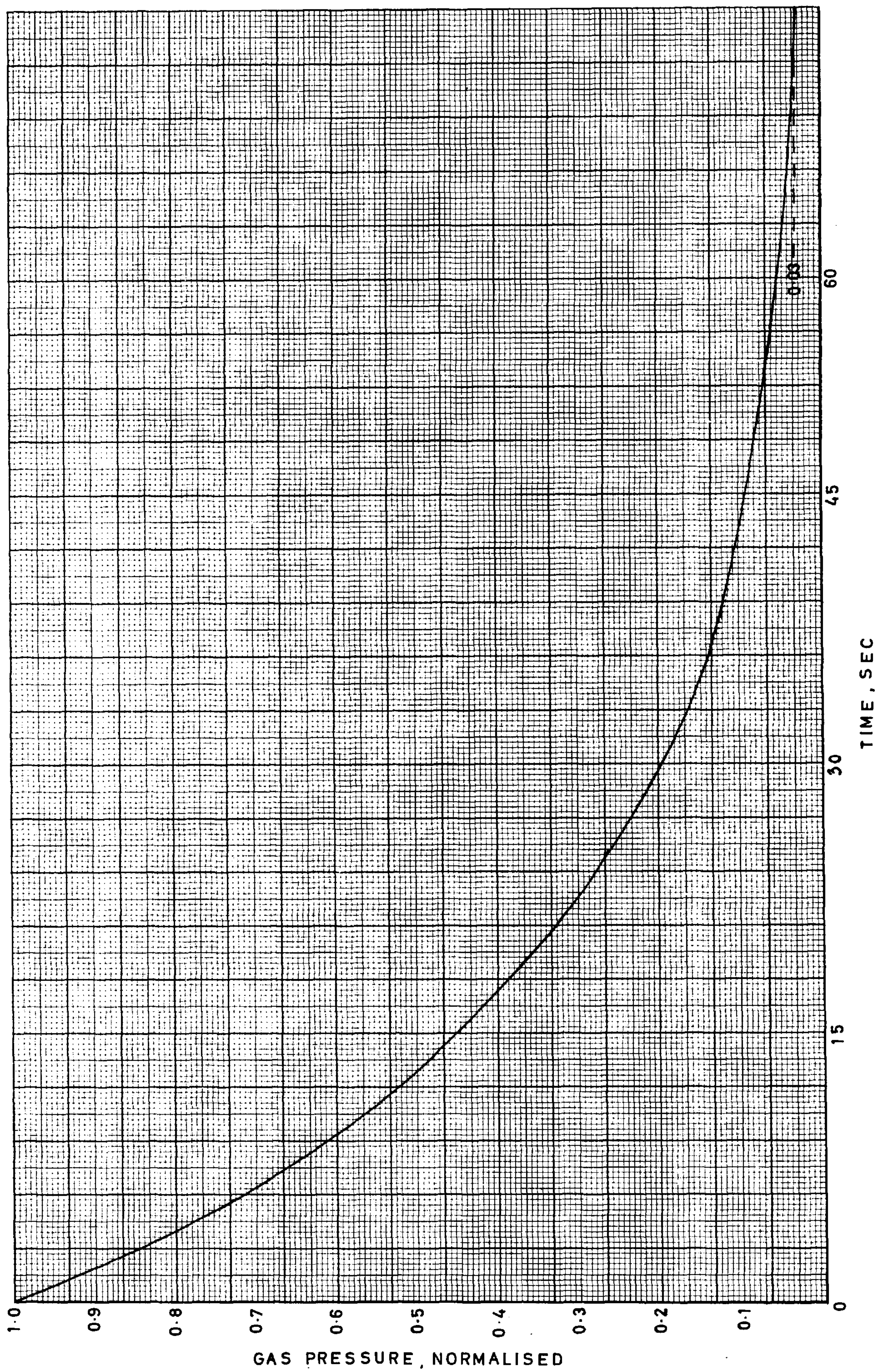


FIG.32 TYPICAL AGR. GAS PRESSURE vs TIME FOLLOWING MEDIUM SIZE PENETRATION FAILURE

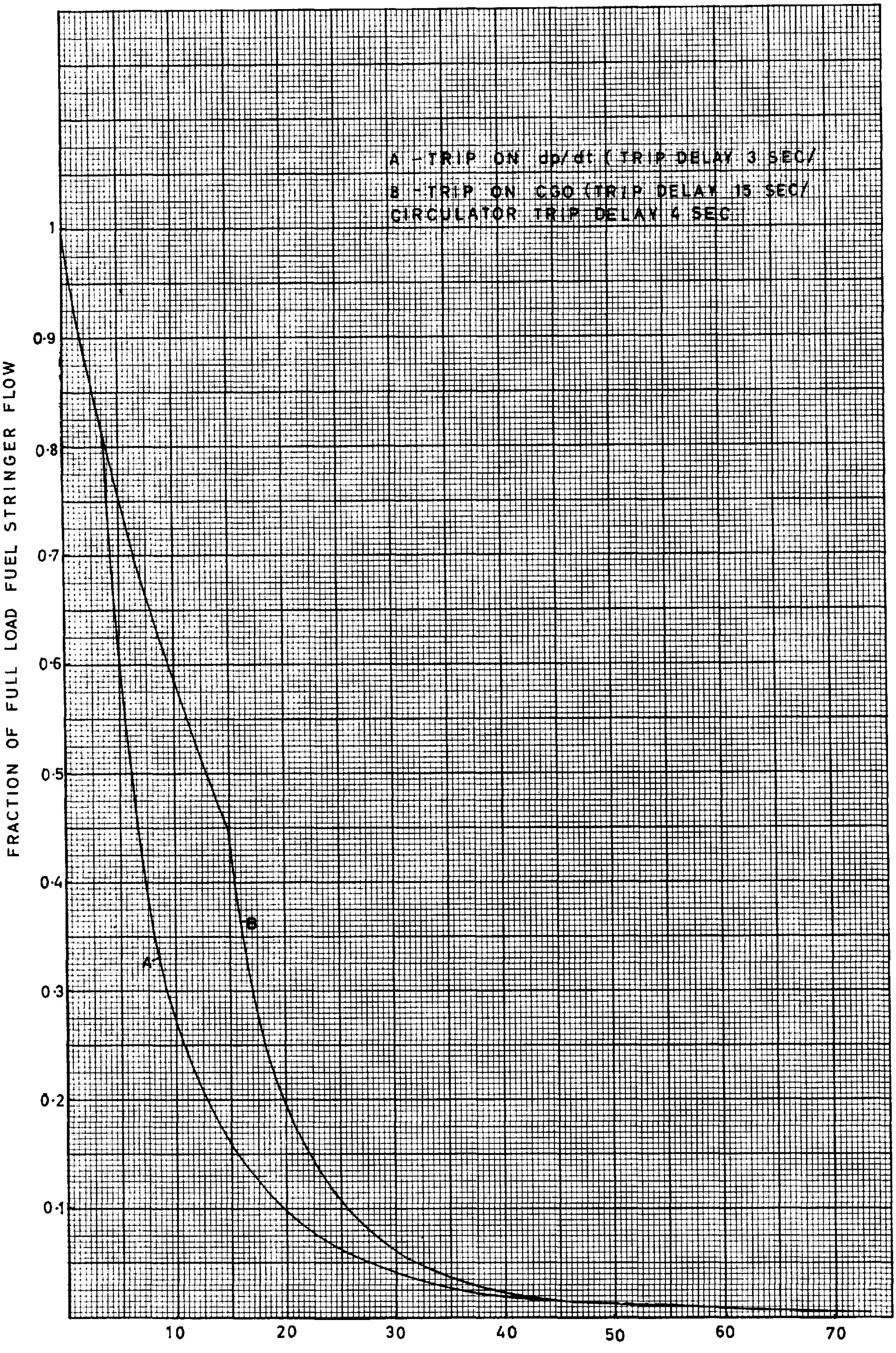


FIG.33 TYPICAL AGR. COOLANT FLOW HISTORIES FOLLOWING MEDIUM SIZE PENETRATION FAILURE. INTER-TRIP OPERATIONAL

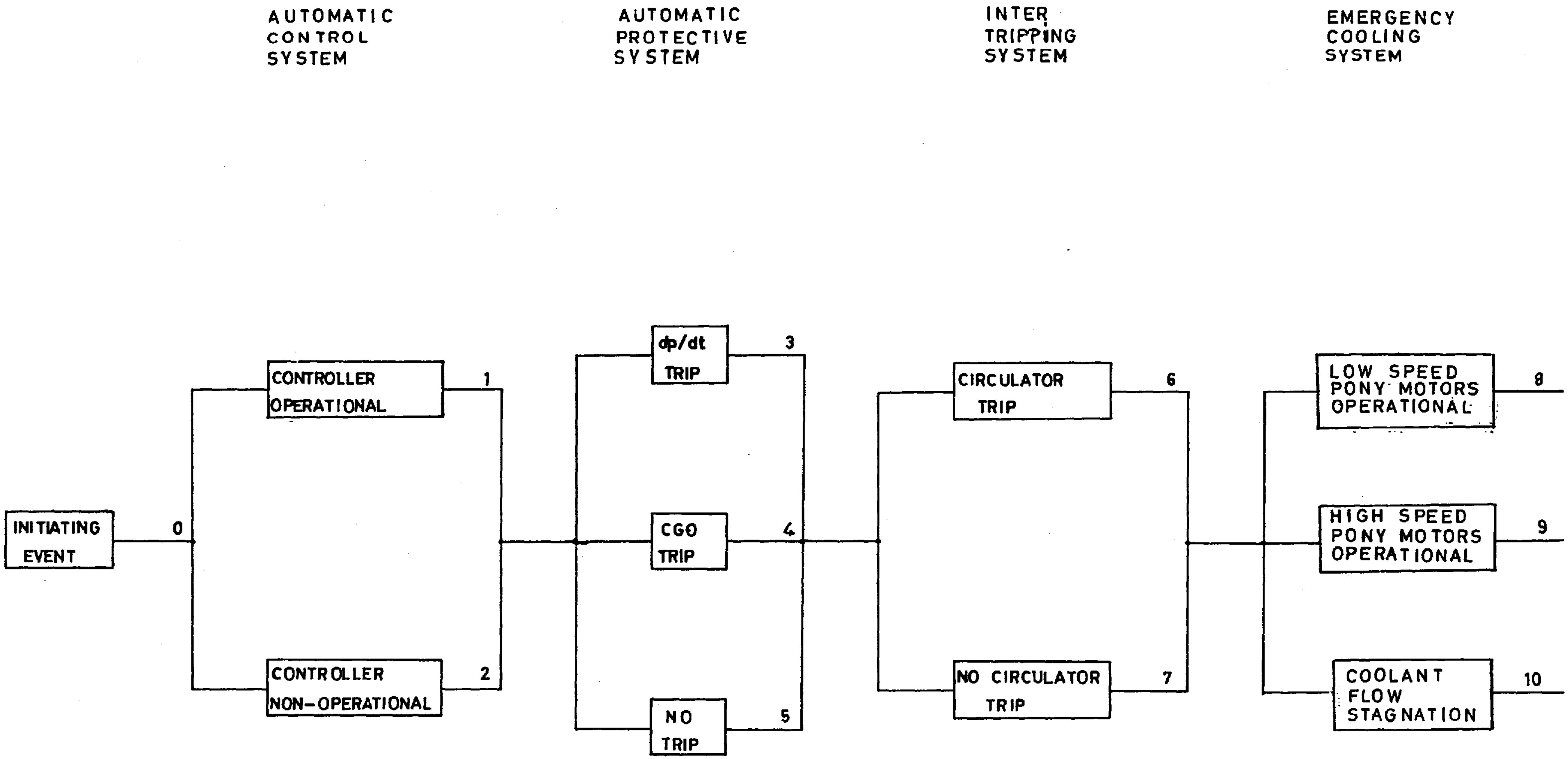


FIG. 34 MEDIUM SIZE PENETRATION FAILURE. FAULT SEQUENCE DIAGRAM

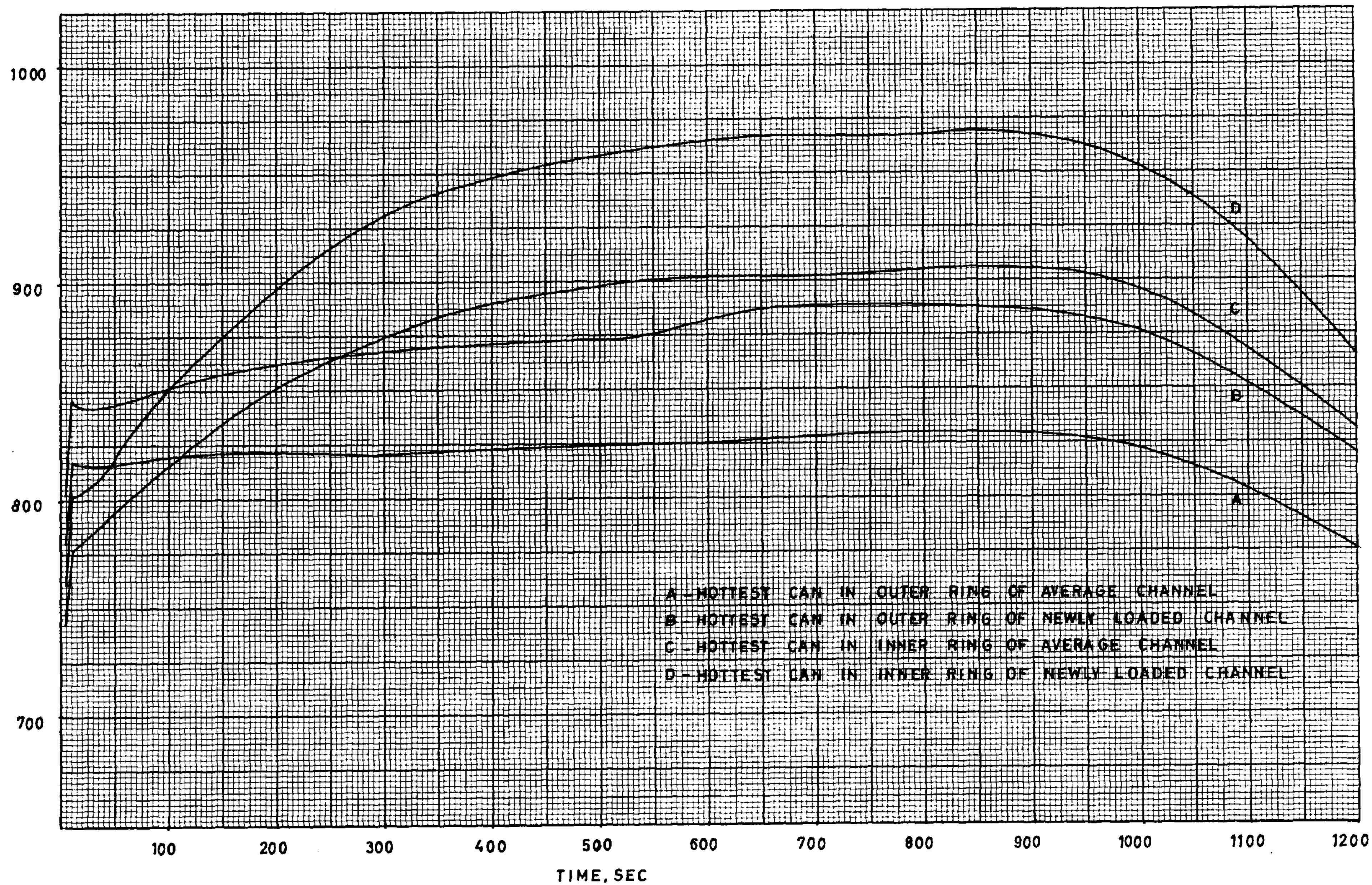


FIG. 35 TYPICAL AGR. DEPRESSURISATION THROUGH 36 IN. HOLE. REACTOR TRIPPED ON dp/dt SIGNAL. OPERATOR ACTION AFTER 15 MIN. TEMPERATURE TRANSIENTS.

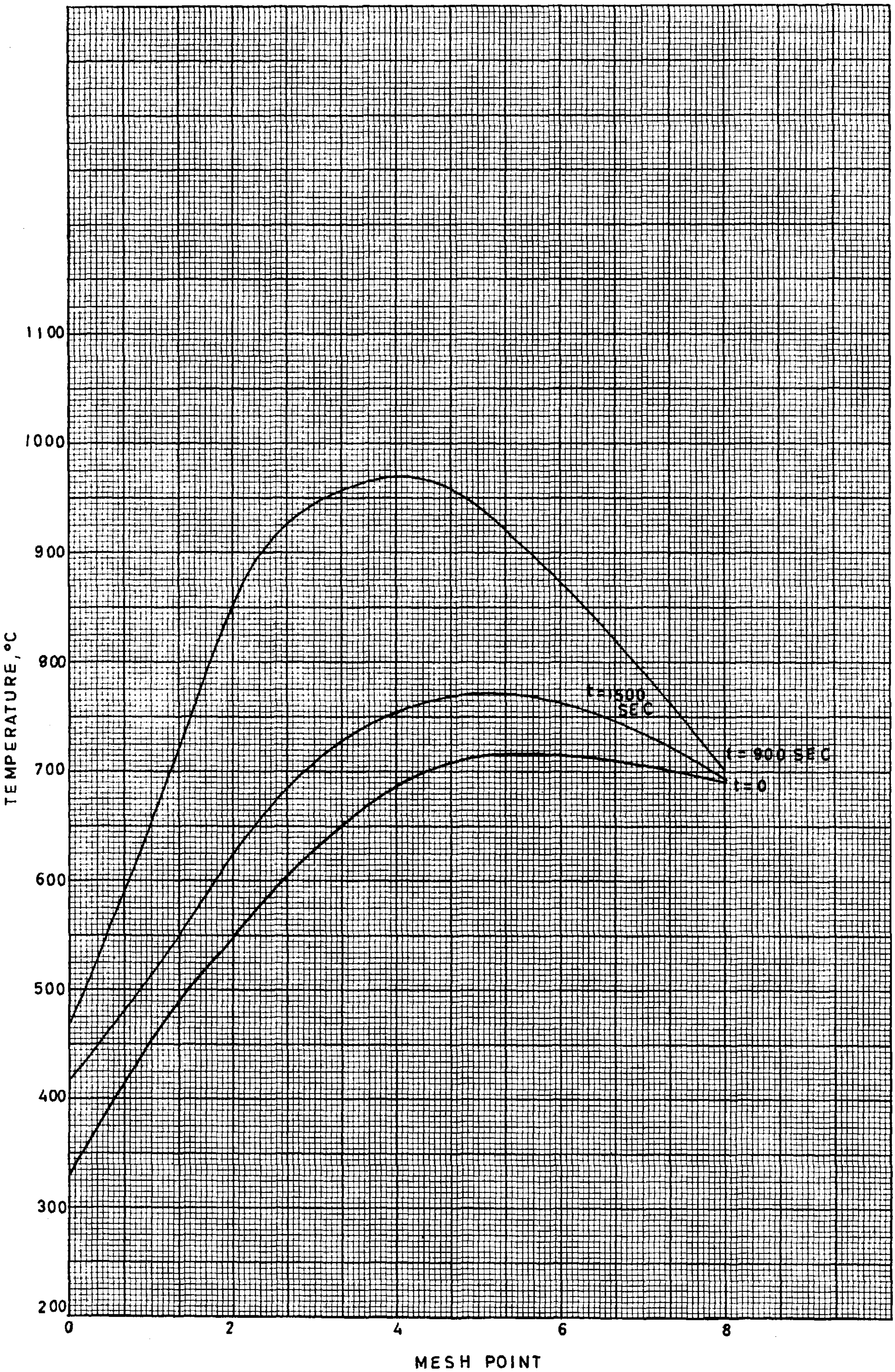


FIG. 36 TYPICAL AGR. DEPRESSURISATION THROUGH 36in HOLE. REACTOR TRIPPED ON dp/dt SIGNAL. OPERATOR ACTION AFTER 15 MIN. PROFILES OF INNER RING CAN TEMPERATURES IN NEWLY LOADED CHANNEL.

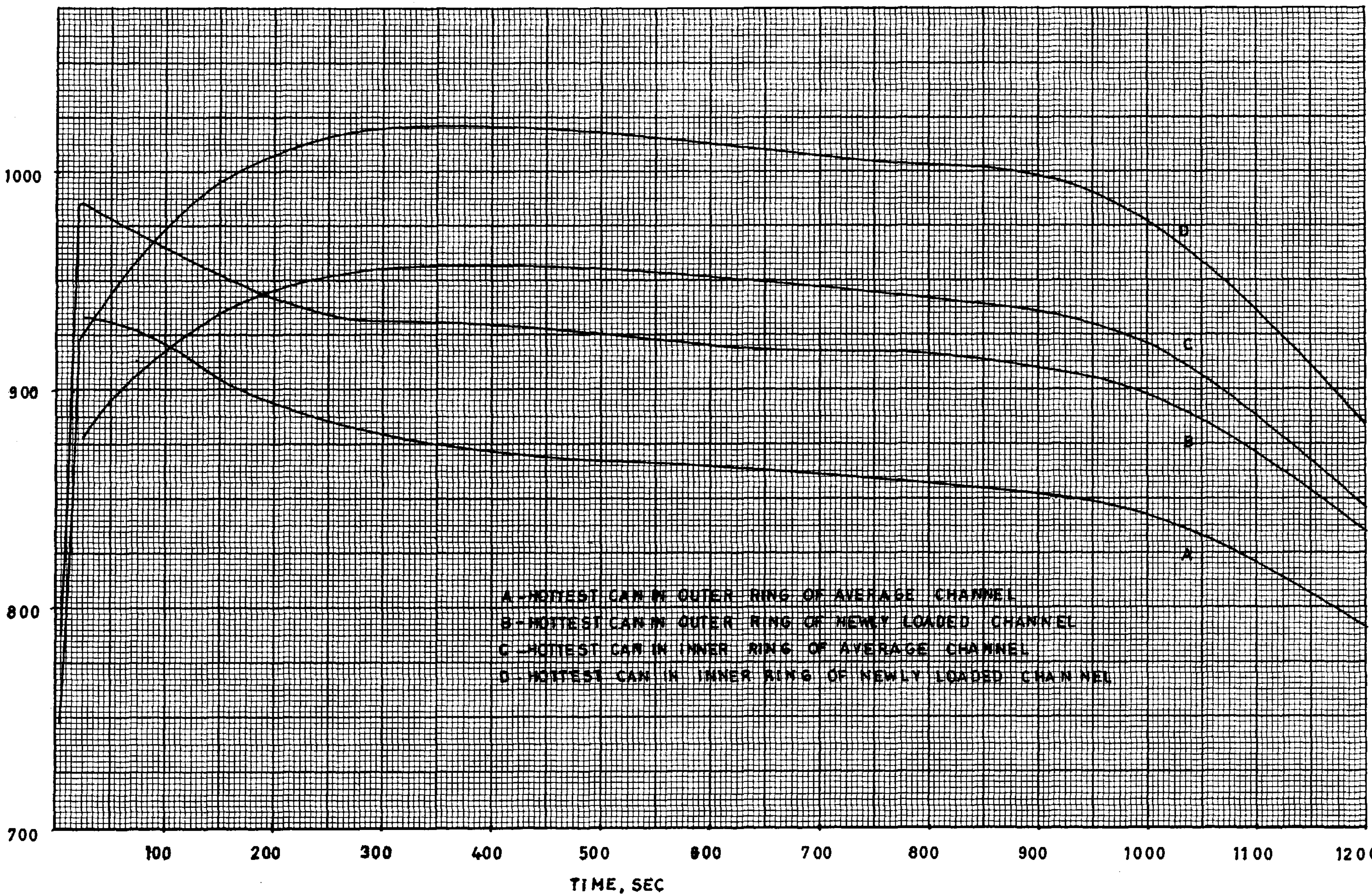


FIG.37 TYPICAL AGR. DEPRESSURISATION THROUGH 36 IN. HOLE. REACTOR TRIPPED ON CGO SIGNAL. OPERATOR ACTION AFTER 15 MIN. TEMPERATURE TRANSIENTS.

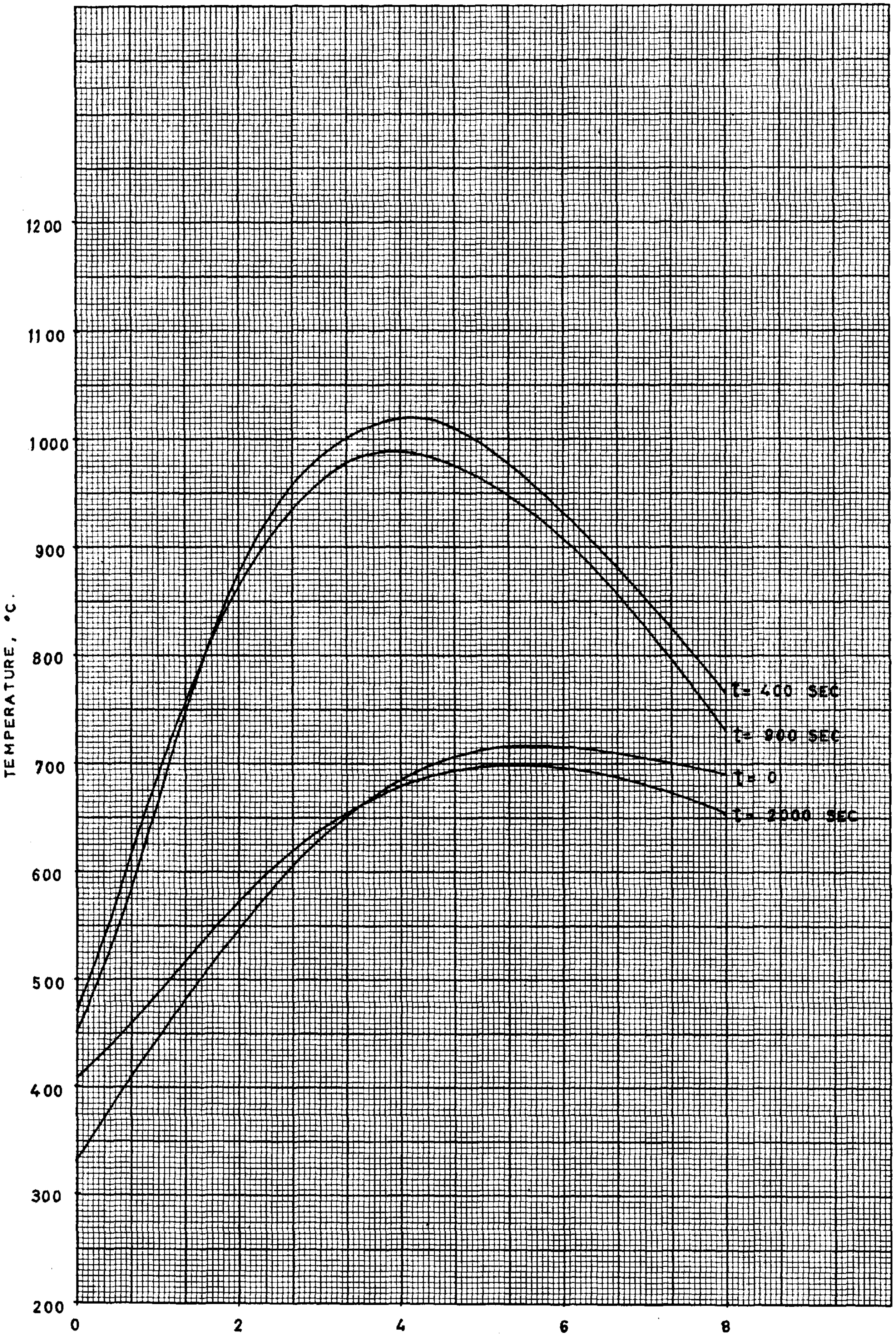


FIG.38 TYPICAL AGR. DEPRESSURISATION THROUGH 36 in HOLE. REACTOR TRIPPED ON CGO SIGNAL. OPERATOR ACTION AFTER 15 MIN. AXIAL PROFILES OF INNER RING CAN TEMPERATURES IN NEWLY LOADED CHANNEL.

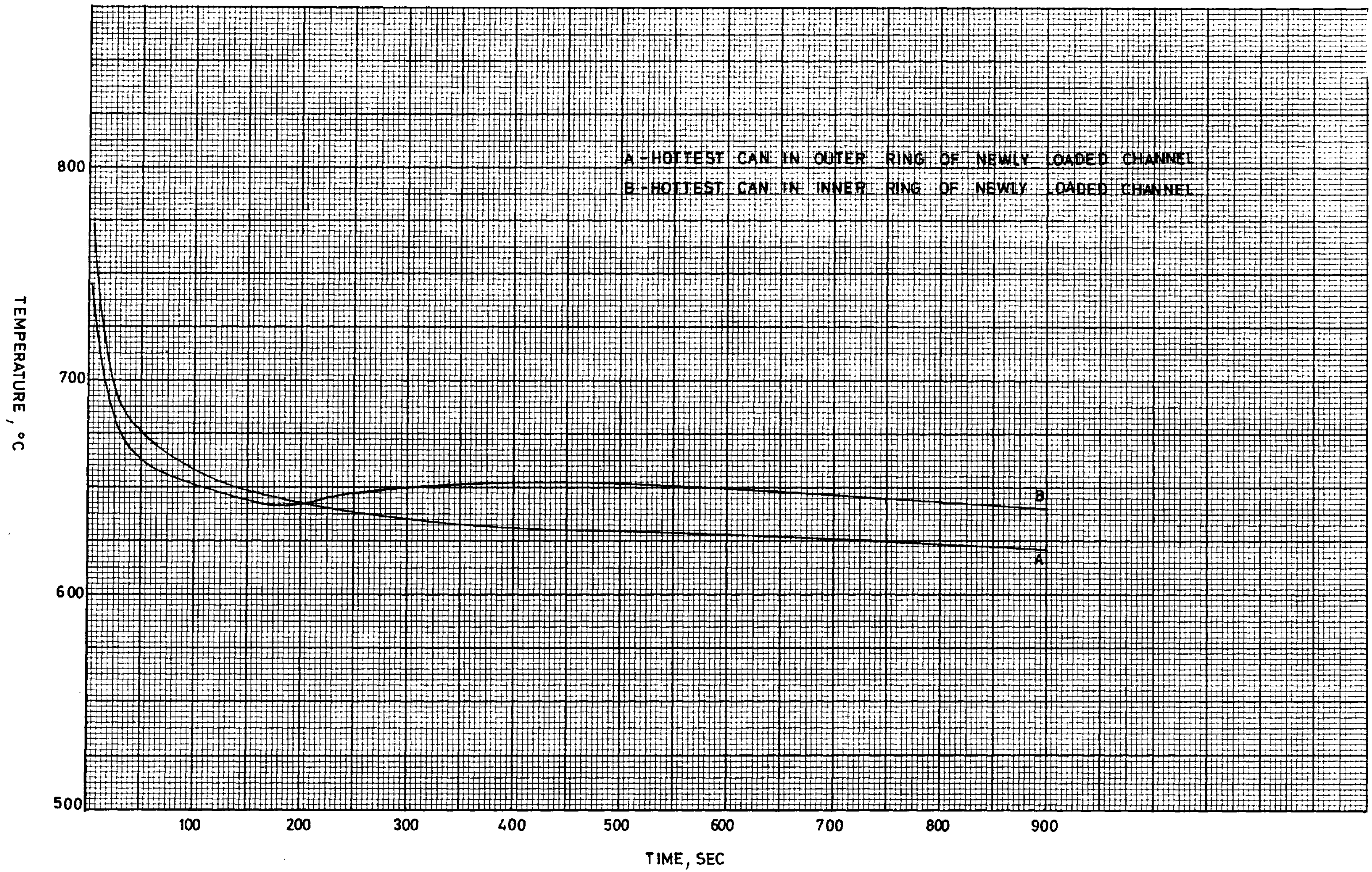


FIG. 39 TYPICAL AGR. DEPRESSURISATION THROUGH 36.in. HOLE. REACTOR TRIPPED ON DP/DT SIGNAL. NO CIRCULATOR TRIP. TEMPERATURE TRANSIENTS.

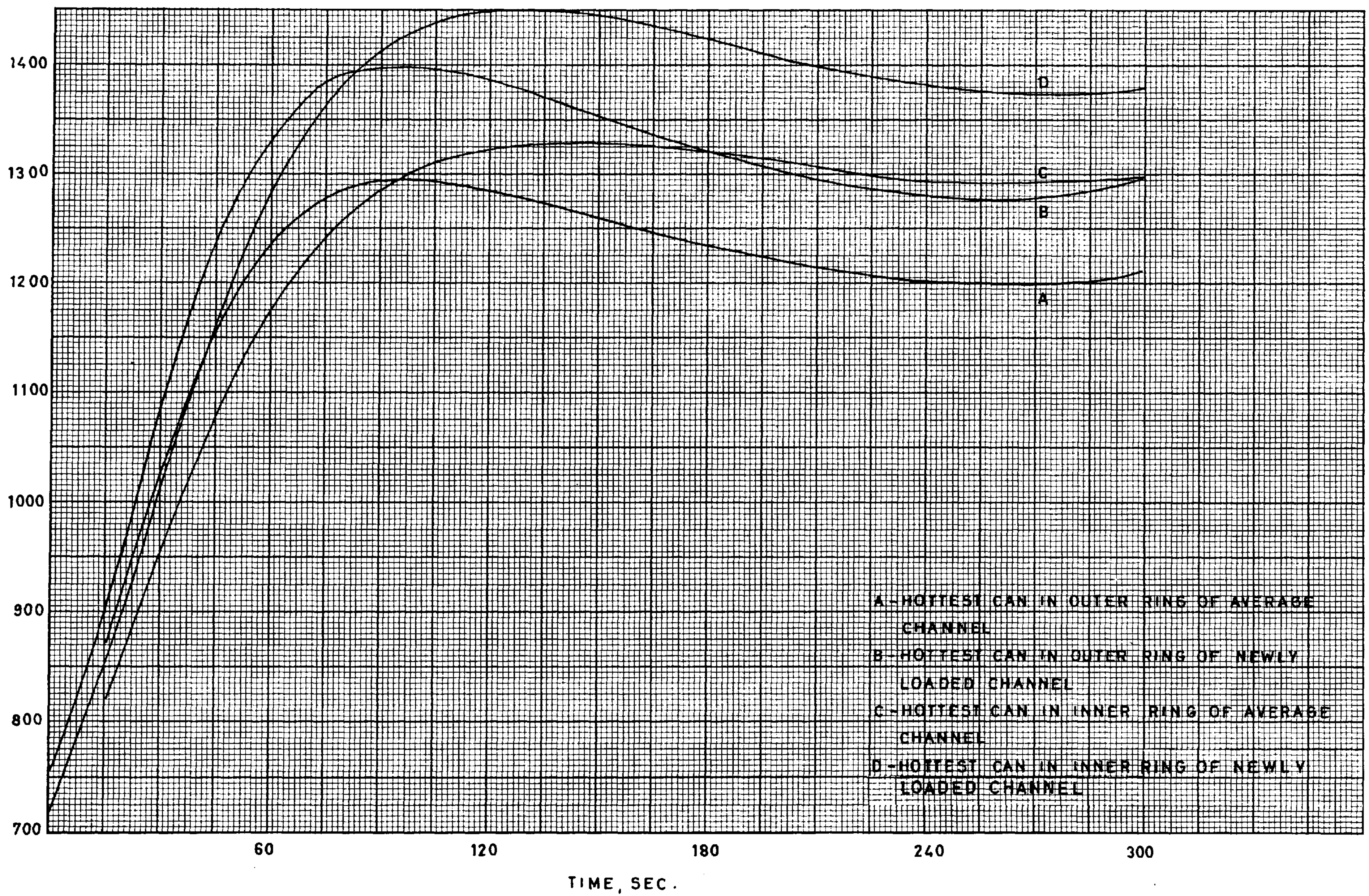
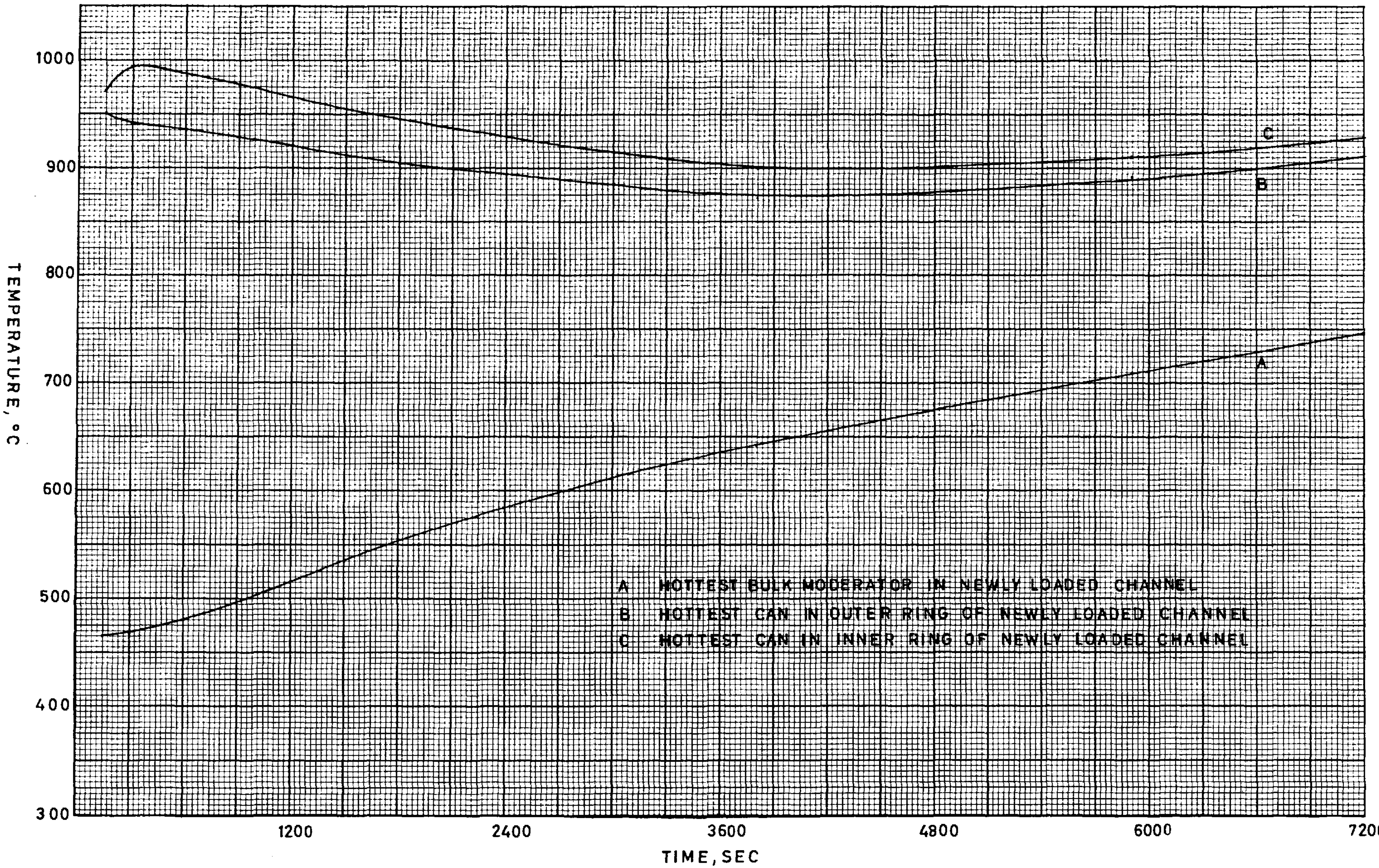


FIG. 40 TYPICAL AGR. DEPRESSURISATION THROUGH 36 in HOLE.
NO REACTOR TRIP . TEMPERATURE TRANSIENTS

FIG.41 TYPICAL AGR. MEDIUM SIZE PENETRATION FAILURE. REACTOR TRIPPED. NO OPERATOR ACTION. TEMPERATURE TRANSIENTS.



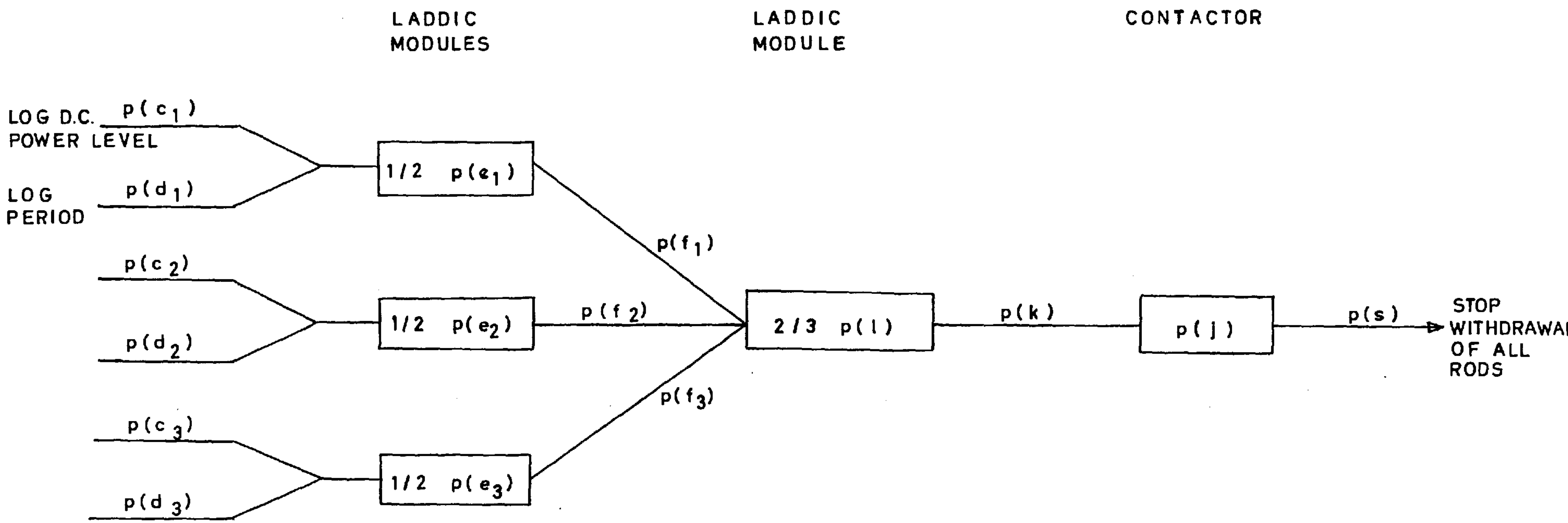


FIG.42 PERIOD FREEZE SYSTEM. TENTATIVE RELIABILITY
FUNCTIONAL BLOCK DIAGRAM

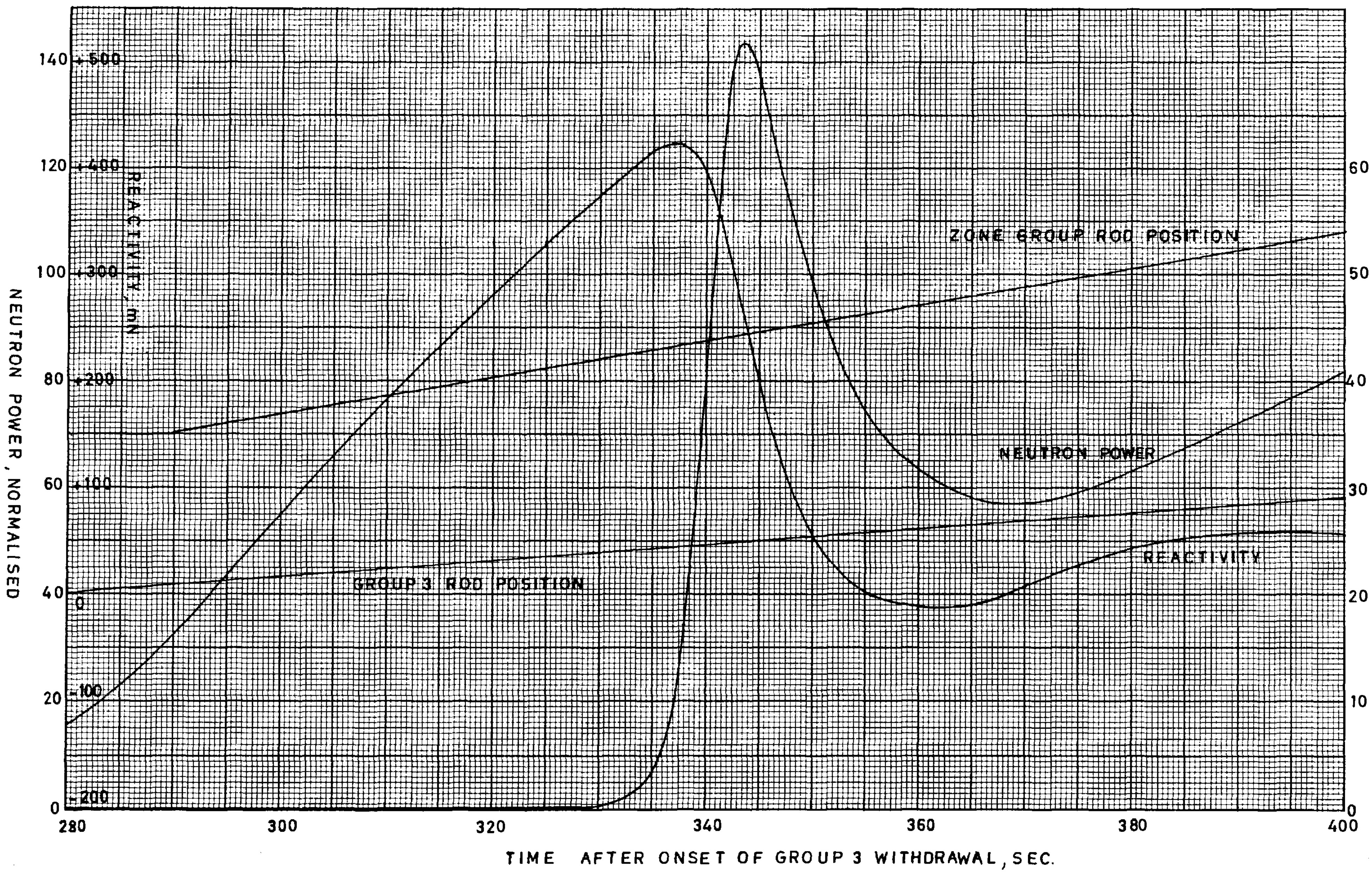


FIG. 43 TYPICAL AGR. START-UP FAULT. PLOTS OF NEUTRON POWER, TOTAL REACTIVITY AND CONTROL ROD POSITIONS

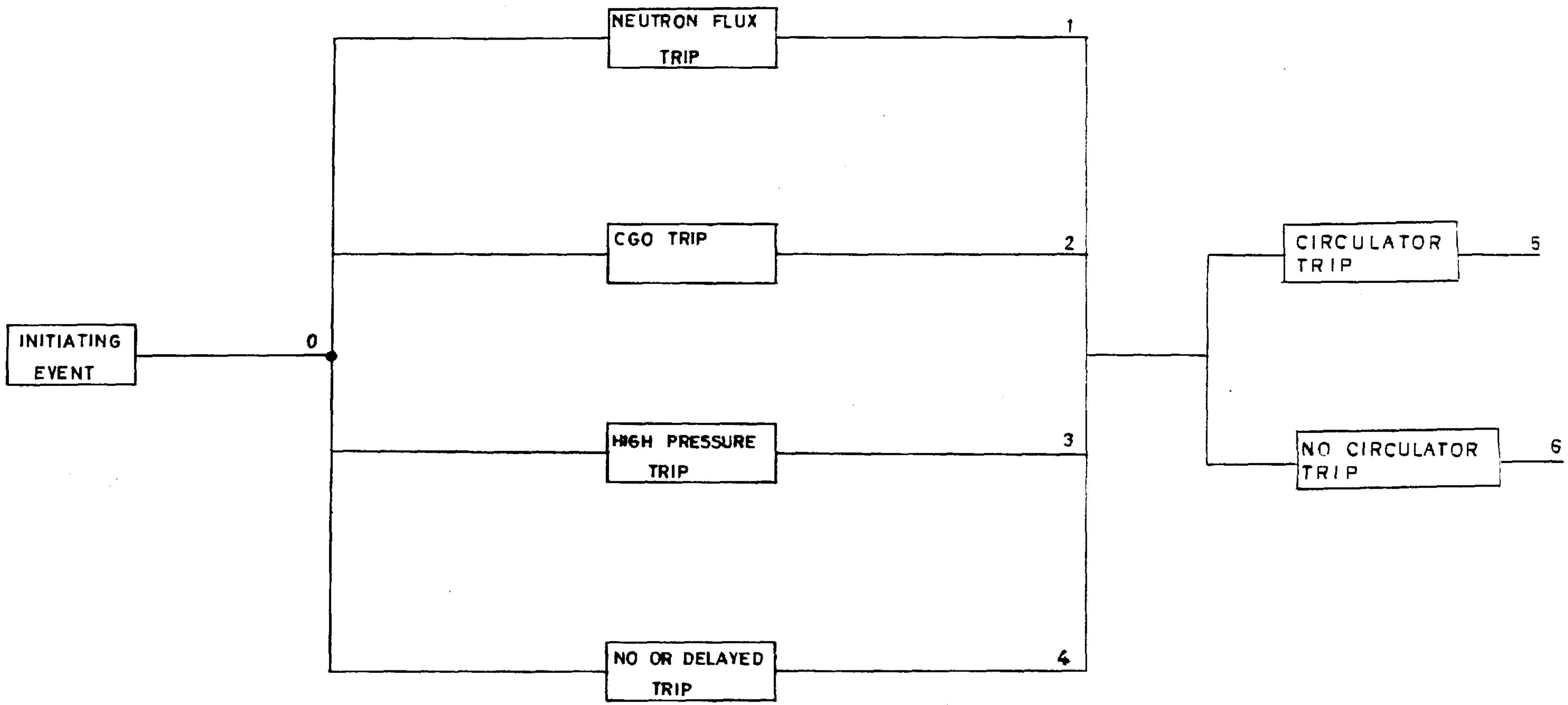


FIG. 44 START-UP FAULT. FAULT SEQUENCE DIAGRAM

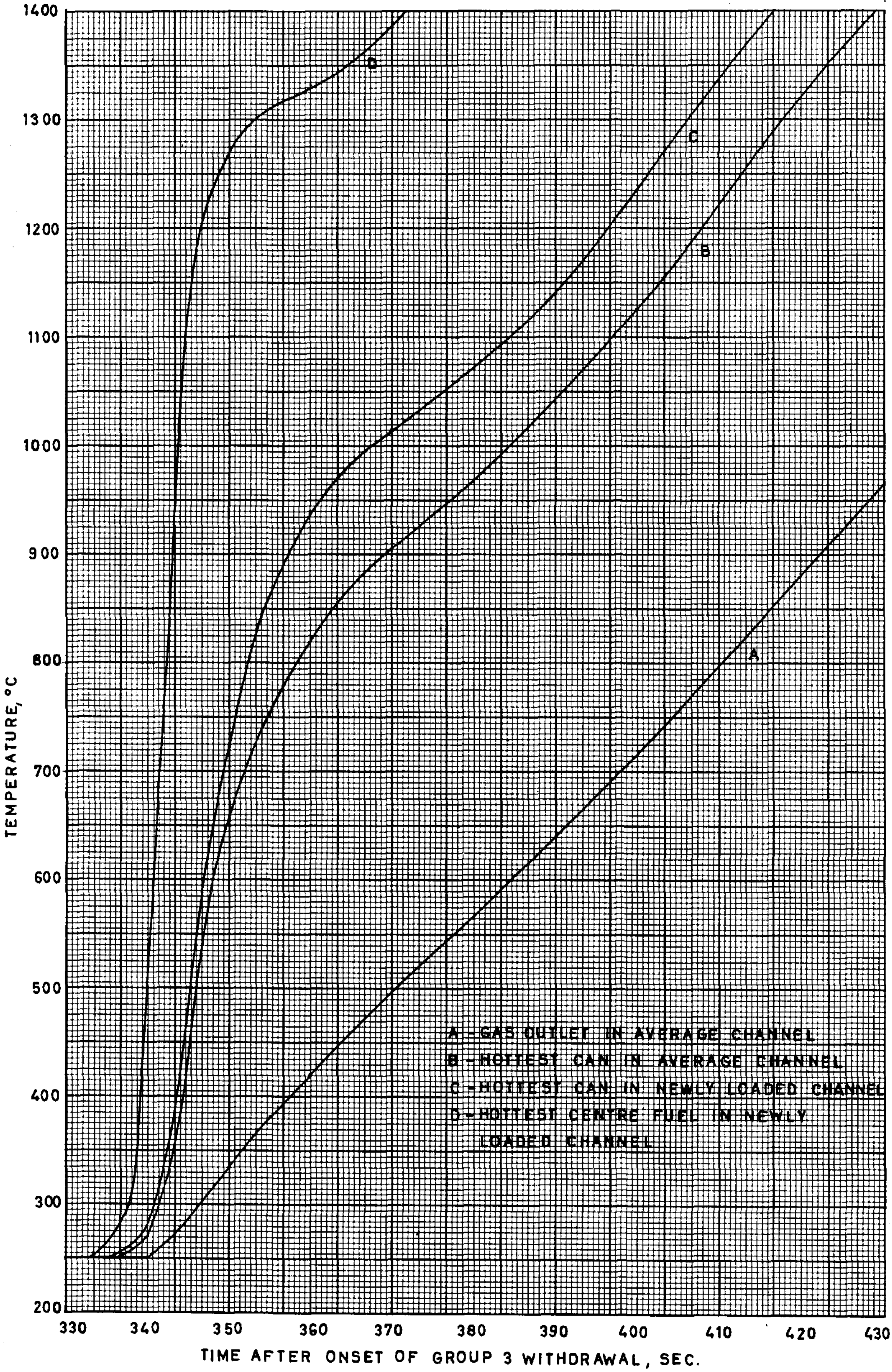


FIG.45 TYPICAL AGR. START-UP FAULT. NO REACTOR TRIP TEMPERATURE TRANSIENTS.

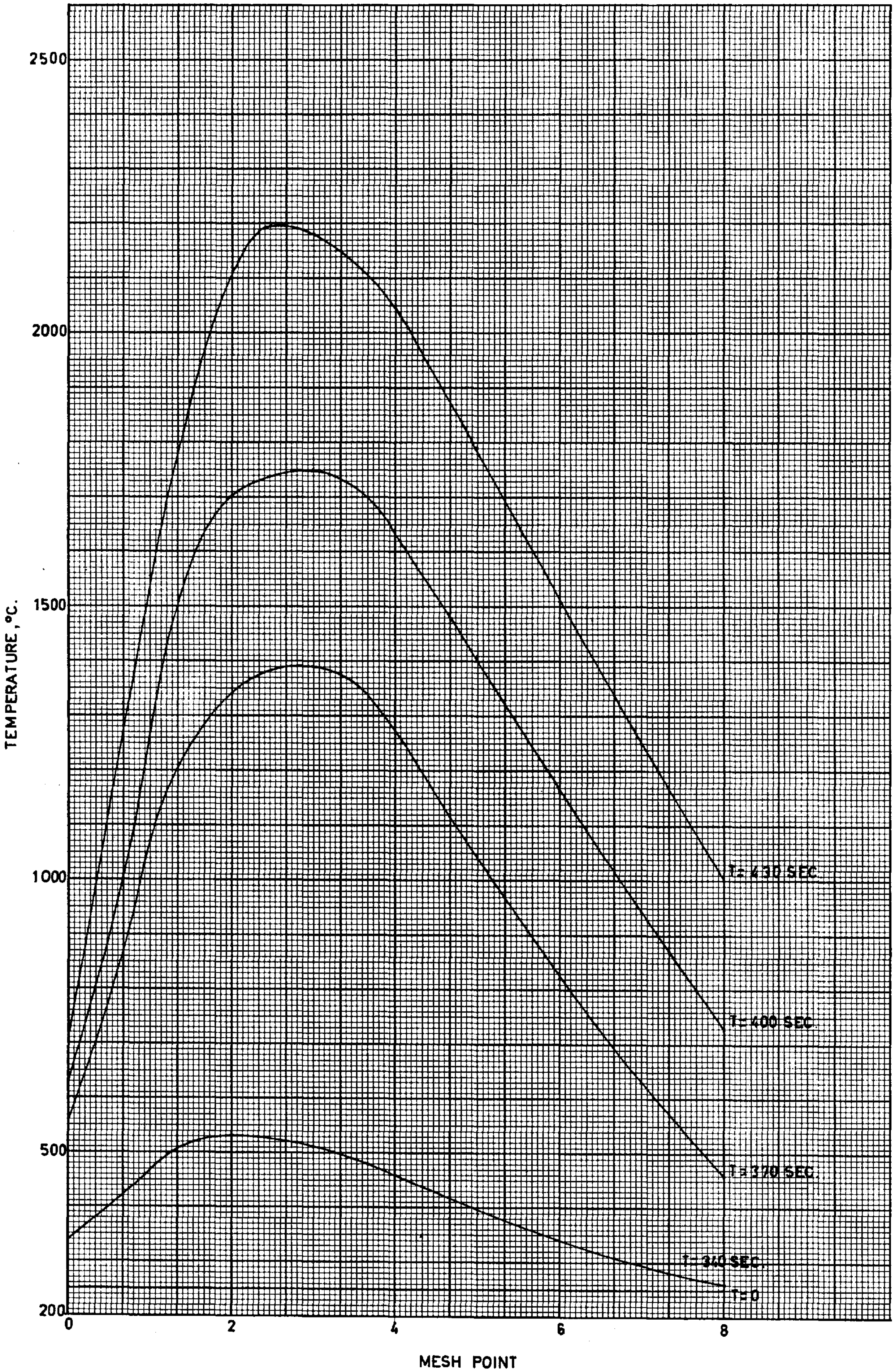


FIG. 46 TYPICAL AGR. START-UP FAULT. NO REACTOR TRIP
AXIAL PROFILES OF OUTER RING CENTRE FUEL
TEMPERATURES IN NEWLY LOADED CHANNEL.

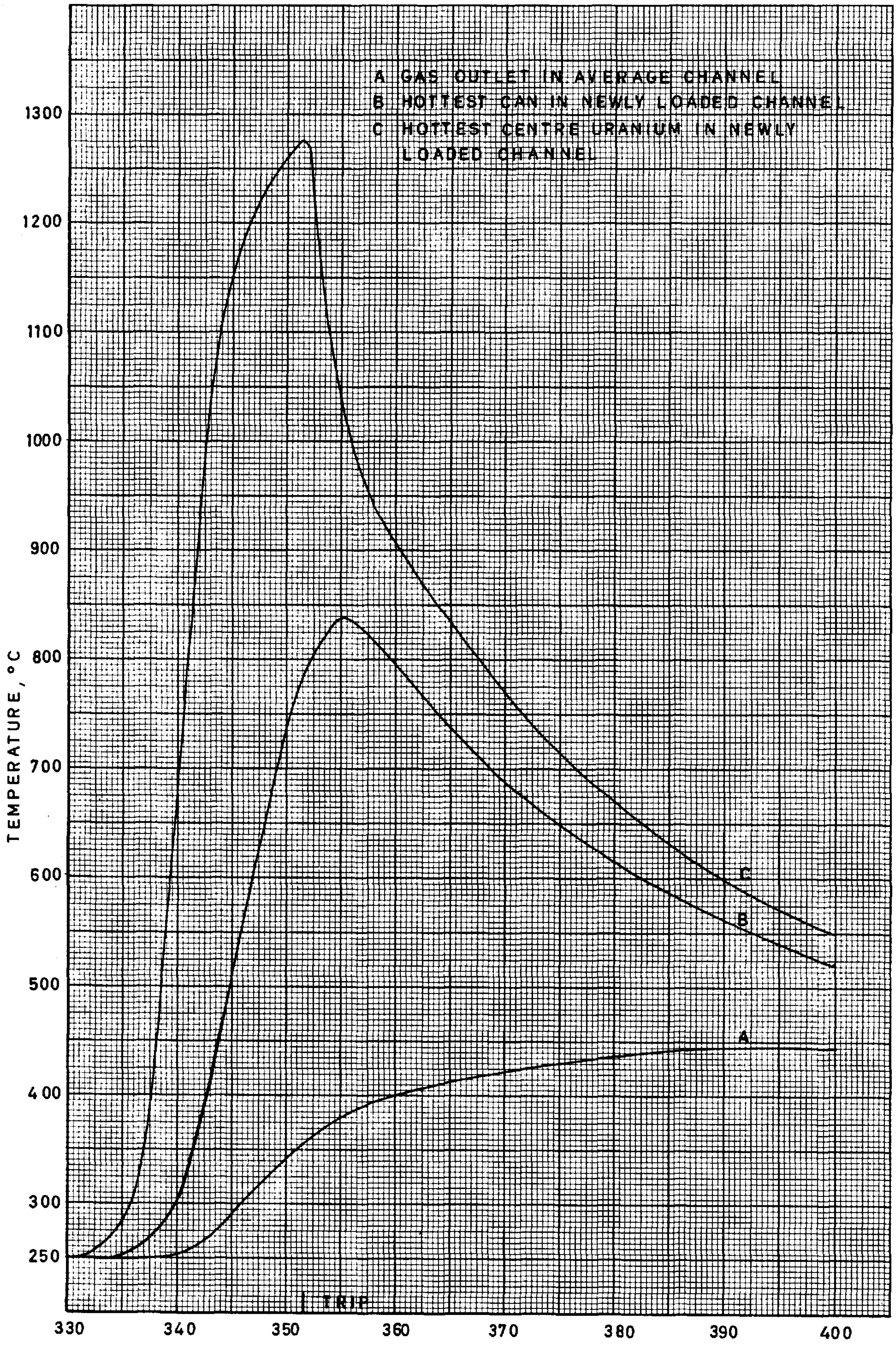


FIG. 47 TYPICAL AGR. START-UP FAULT. TRIP ON CGO. TEMPERATURE TRANSIENTS.

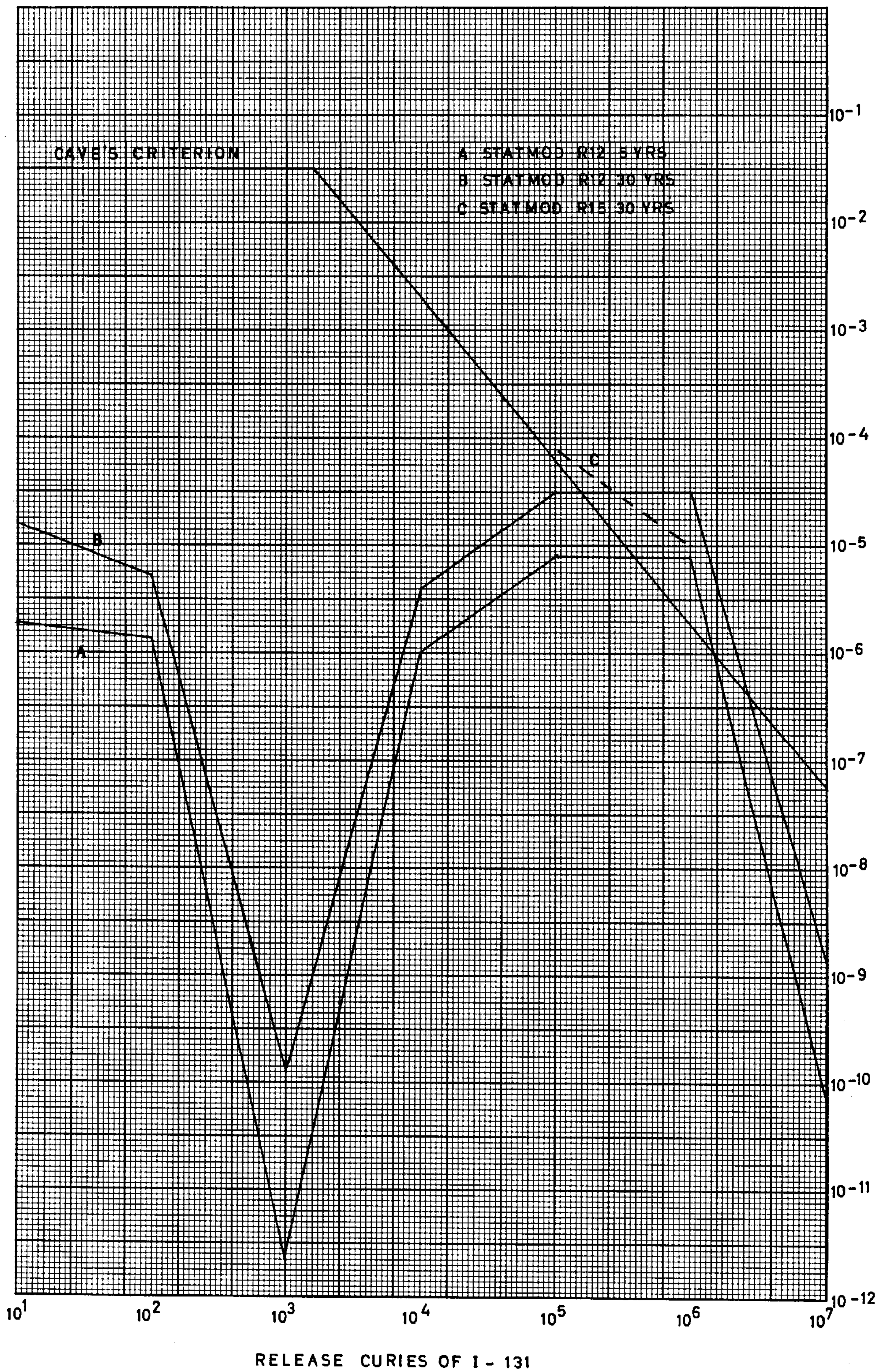


FIG. 48 PROBABILITY DISTRIBUTION OF IODINE CURIE RELEASE. FIXED RETENTION FACTORS. ASSUMPTIONS AS IN TABLE 37

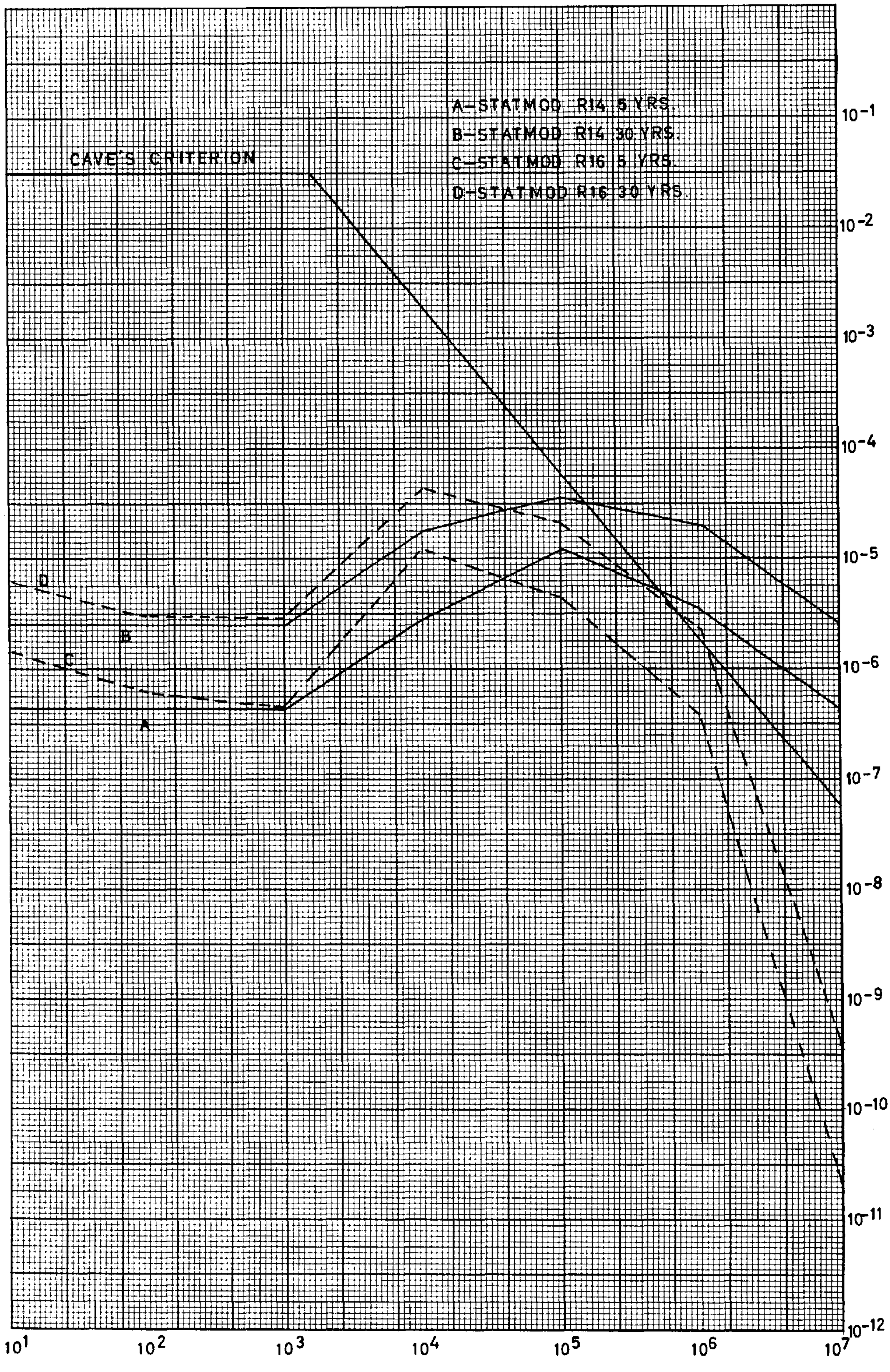


FIG. 49 PROBABILITY DISTRIBUTION OF IODINE CURIE RELEASE. RETENTION FACTORS RANDOM. ASSUMPTIONS AS IN TABLES 39,40,41

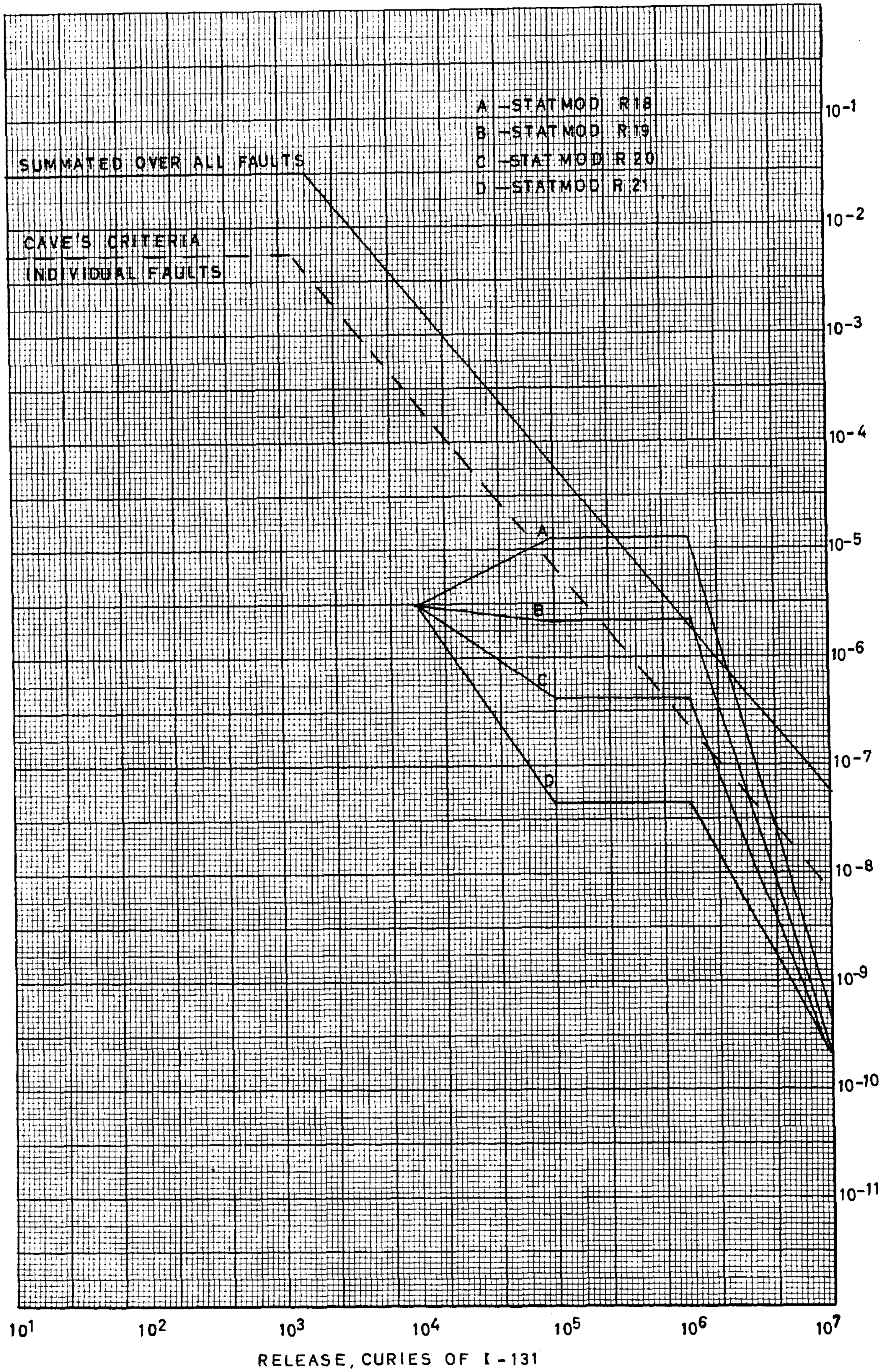


FIG. 50 PROBABILITY DISTRIBUTIONS OF IODINE RELEASE. STATMOD RUNS 18-21