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INSTRUMENTATION STUDIES - STATUS REPORT

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INSTRUMENTATION STUDIES - STATUS REPORT

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I. INTRODUCTION

The purpose of this study is to obtain a model, represented by an analog or digital computer program, which gives the best representation of the reactor and its safety instruments, and using this model to perform analyses which will lead to the establishment of criteria for present and future safety instrumentation systems. Qualitative aspects of this study are discussed in Reference 1. It may seem that developing a computer program is part of, but not one of the purposes of the study. However, this study will be continuing; thus if a good model is arrived at now, one must only continue to upgrade the model as new techniques become available. Then when a new instrumentation system is to be analyzed one can proceed directly with the analysis without first having to decide which is the best model to use.

Until recently the means for performing calculations to provide data of a type from which an analysis could be made and criteria established for complete safety instrumentation systems has not been available at Hanford. Instruments were analyzed using the best available methods and then adding factors of conservatism to cover the uncertainties. With the equipment now available geometrical effects can be considered as well as the various lag time interrelationships; thus a more accurate simulation of the reactor and instrumentation together can be obtained. These calculations lead to a refined criteria for the reactor functions to be covered and reduce these functions to limiting parameters for the safety instrumentation.

The purpose of this document is to relate the work which has been done to date on the study. It includes the assumptions which are used, a discussion of the analog and digital computer programs, and a summary and discussion of the preliminary results obtained. As mentioned above the study is meant to be a continuing one, and as significant results are obtained they will be published. One word of caution: the results presented here represent the present state of knowledge of the authors on the subject and thus are only preliminary. Nevertheless, an attempt is made to state numerically the minimum degree of protection of particular systems on the basis of results to date. Although these results must be stated over-conservatively lacking the more refined computer runs yet to come, they will provide a frame of reference for necessary interim management decisions on project action and plant maintenance.

II. SUMMARY AND CONCLUSIONS

Two computer programs were used in making calculations for this study; these programs refer to the one and six-node models. For the one-node model the standard reactor kinetics, heat transfer, and instrument delay equations were programmed for the analog and digital computers. From this model the effects of full-pile reactivity insertions could be calculated. The six-node model contains the time-dependent diffusion equations, and heat transfer and instrument delay equations. This model was programmed for the analog computer only and was used to calculate the effects of local reactivity insertions.

It is quickly apparent from this study that three types of protection must be provided - period, rate, and level. Period and rate systems are

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appropriate at low and intermediate levels to ensure that the high level range is not entered at a rate too fast for assurance of protection by the high level system.

From the results obtained to date there are several conclusions which can be drawn concerning the range of tube power over which high level instrumentation systems, in the old reactors, such as the Zone Temperature Monitor (ZTM), Octant Flux Monitor, etc., would meet the criteria established for this study. For the six old reactors the worst credible accident considered initiating in the intermediate and high level ranges was the simultaneous withdrawal of three rods at B, D, DR, and F reactors and four rods at C and H reactors. These numbers relate to the number of rods two operators can conveniently move simultaneously. It was assumed that the rods could be withdrawn in 40 seconds. For this case either an Octant Flux Monitor system or the Beckman Flux Monitor system (with 110 per cent trips and a one second instrument delays) meet the criteria for a range of tube powers from 0 to 1300 kw. For 130 per cent trip the criteria would not be met for tube powers between 1000 and 1300 kw. The latter assumption (130 per cent trip) is more realistic considering the high amount of procedural reliance in the use of either flux system with percentage trips.

The Zone Temperature Monitor with a fixed trip at 125 C, meets the criteria only over a small range of tube powers, 1160-1320 kw, which may be too small to be adequate. If the number of rods which can be withdrawn simultaneously is assumed limited to two or the rod speed is assumed to be cut in half, the results are as follows: the Zone Temperature Monitor meets the criteria over a range of tube powers of from 700 to 1320 kw. This is an adequate range for a high level instrumentation system and the protection is procedure-free.

For the Octant Monitor or Beckman flux system (with 110 per cent trips and one second instrument delays) the criteria would be met, in the case of reduced withdrawal rates, for tube powers between 0 and 1300 kw. In this case the criteria would again not be met for tube powers between 1050 and 1300 kw for 130 per cent trip.

The above results indicate that if the rod system withdrawal speed were reduced the Zone Temperature Monitor would meet the established criteria over an adequate high level range, whereas the ability of a percentage trip instrument to meet the criteria would be unduly dependent on procedure. The above conclusions relative to the older reactors should apply qualitatively to K reactor high-level protection.

III. COMPARISON OF INSTRUMENTATION SYSTEMS

A. Ultimate Instrumentation Systems

The reactor flux and temperature monitoring systems are used to prevent the exceeding of a high temperature limit designed so as to prevent occurrence of a nuclear hazard. In general, no monitoring system would protect against excessive temperatures in the case of a flow loss. For particular cases though, flux or temperature monitoring systems could

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give a trip upon flow loss and minimize the consequences. For example, a flow loss that would result in a power excursion (due to a positive void coefficient of reactivity) would be "seen" by a flux instrument. A temperature system would "see" a general flow reduction whereas a flux monitoring system wouldn't for the case of a negative short-term temperature coefficient of reactivity.

Complete reactor level instrumentation has been visualized on the basis of ultimate temperature and ultimate flux monitoring systems. "Ultimate" refers to automatic tube temperature protection over the entire range of operation. The ultimate flux or temperature monitoring system will include fixed high-level, linear rate-of-rise, and nuclear period trips. The high-level and linear rate-of-rise portions of the temperature and flux monitoring systems would have to have representative geometrical coverage. Adequate geometrical coverage for the high-level-flux system would require the use of a number of in-core chambers with the appropriate trip circuits, whereas a grid of outlet temperature detectors would provide good geometric temperature coverage.

A high-level flux monitoring system used for preventing excessive temperature has an inherent problem of correlating flux and temperature; this correlation would vary with flux distribution for limited detector geometrical coverage. An in-core flux system with adequate geometrical coverage and appropriate signal addition arrangements might nearly eliminate the variation in the correlation between flux and temperature due to the flux distribution. Assuming the calibration of the in-core system were independent of flux distribution, there would remain the problem of absolute calibration of the high-level system; the absolute calibration could be done occasionally at some moderate power level. The absolute calibration of the high-level system would depend on such things as fuel loading and location of chambers relative to rods and supplementary control. Assuming that the calibration of the ultimate flux system could be independent of flux distribution and the precision of the absolute calibrations were five per cent, the maximum allowable reactivity ramp at a given level for the ultimate high level flux system could be as much as a factor of two greater than for an ultimate high-level temperature system; this difference is due to the fact that the time constants and delay times of a flux system would be shorter than those for a temperature system.

B. Pseudo - Ideal Systems

The "ultimate" system as described above represents a maximum temperature or flux monitoring system. An instrument system or combination of systems short of the "ultimate" may be quite adequate, however, if automatic response is fast enough to prevent the exceeding of the stated protection level for the range of reactivity ramps physically attainable.

For a temperature monitoring system, such as the Zone Temperature Monitor, to meet the criteria, it must give good geometrical coverage; i.e., one could not monitor say four temperatures in the reactor and hope to operate the reactor under the present criteria without imposing unreasonable restrictions on operation. Thus in considering a temperature system

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as a safety device one is forced to approach the characteristics of an ultimate temperature system. For a flux monitoring system the amount of geometrical coverage can be reduced further than for a temperature system, the exact amount as yet undefined, and still give good protection. Although one can get by with less geometrical coverage for a flux system the system does have several disadvantages which will be pointed out.

As the number of monitoring points is reduced the output of the flux monitor chambers must be representative of a large number of tube outlet temperatures. If the flux distribution were constant there would be no problems involved; one could set a fixed trip according to the maximum peak tube power. The flux distribution is not constant, however, and thus the fixed trip setting will vary according to the flux distribution. If the setting is not varied then a situation such as the following could occur. Supposing the fixed trip is set for a flattened flux distribution and during the operating cycle one gets a cosine distribution. Now for the same peak temperature in the center the flux in the fringe, and thus chamber readings, are reduced and one is actually further below the trip point and thus can tolerate less of a ramp. To adjust the trip settings as a function of flux distribution would be an untenable situation involving too much procedure. One other point about a fixed trip system with limited geometrical coverage is that the trips would have to be set high enough to allow flux distribution variations which occur during operation, and the higher the trip setting the smaller the allowable ramp for a given power level.

If one used a percentage trip in the case above, the trip would be set a fixed amount, say 10 per cent above the current reading and one would supposedly be "safer." Setting of this kind of trip requires procedures and dependence on operators setting trips correctly, however, and does not give true automatic protection. It may be possible to build a device which adjusts the trips automatically; i.e., as the flux increases, and thus the instrument reading increases, the trip would be automatically increased. One feature which would have to be incorporated in such a device would be the ability to distinguish between various rates of increase; i.e., for fast rates the trip would be held constant, but for slow trends the trip would be increased or decreased depending on the direction of the trend. Another feature which the device must have is an ultimate trip setting beyond which no further increases would be allowed. This scheme, although numerically "safer" than the fixed trip case, still suffers from poor correlation between instrument reading and outlet tube temperature and would be more difficult to build fail-safe.

IV. ANALYTICAL STUDY PROGRAM

A. Assumptions

In performing any nuclear safety study one must first determine what to protect against and then establish limits and criteria which provide this protection. For this study it was determined that we would never allow an incident to proceed which would lead to a gross release of fission products.

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Fission products will be released if melting of the fuel element jacket occurs. Jacket melting implies excessive fuel temperatures. Excessive fuel temperatures can occur in two ways, one due to nuclear excursion and the second due to loss of coolant water. For this study it has been assumed that there is a continuous supply of coolant and that water loss is covered by the speed of control criterion which is discussed elsewhere.^{/2/} Thus, the goal of this study must be to define the conditions necessary to prevent the occurrence of a nuclear excursion which would lead to excessive fuel temperatures.

The nuclear behavior of the reactor, when boiling occurs in a large number of tubes in the active zone, is extremely difficult to describe due to complications in describing the two-phase flow state, and because of the positive void coefficient of the Hanford reactors; i.e., if we allow boiling to occur in the active zone we will add to the excursion, and the amount of added reactivity is difficult to calculate. Furthermore, the degree of protection afforded by instrumentation during such conditions cannot be accurately defined. For these reasons, the ultimate or upper outlet-water-temperature limit assumed in this study was 150 C at the six older reactors and 145 C at the K's.^{/3/} In making all calculations the transient must be turned around before this limit is exceeded to be satisfactory.

The second assumption made was that the axial flux distribution, for a given integrated column power, has only a second-order effect on the boiling limit: the basis for this assumption is that the pressurization of the Hanford reactors is sufficiently high that boiling limits based on equilibrium observations are relatively insensitive to longitudinal flux distribution. When this assumption is made one can use a one-node model to describe the temperature effects in the axial direction. At the present time, the authors are unsure of the limitations of this assumption with respect to dynamic conditions such as, flow and detection lag times, and geometry effects of rod insertion. To determine how good it is an investigation is now underway to describe the front-to-rear temperature effects by a nodal analysis and to determine how good is the assumption of a one-node model in the longitudinal direction. If an analysis shows the assumptions to be nonconservative, one may be able to analyze the nodal model results and determine a safety factor for correcting existing program results. Another path which one may follow would be to alter existing programs so that they would include a nodal front-to-rear representation of the temperature effects.

A third assumption made is that a nodal model of the reactor in the top-to-bottom direction is required to adequately describe the kinetic behavior of the reactor and the various instrumentation systems being studied. Because of the size of the Hanford reactors, instrument location or geometrical coverage of the reactor is an important consideration. Excursions due to local insertions of reactivity, such as from flushing poison columns and/or pulling rods, could continue sufficiently to cause damage if the safety instrumentation were placed in a position such that local distortion effects could result in an appreciable time lapse before the excursion was detected and safety action started. Using a nodal mode one can study these effects to determine the maximum

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ramp which can be tolerated as a function of instrument location and response characteristics, flux distribution, power level, and location where reactivity is inserted. The nodal model also describes the insertion of the vertical safety rods more accurately.

The most accurate description of the problem would be provided by a three-dimensional nodal model. The computing equipment required to do this problem, however, would be enormous. The present analog facility does not have the necessary equipment, and it is doubtful if there is enough equipment to do the problem adequately on the IBM 7090. For the present time the study is being carried out using a one-dimensional 11-node model. The choice of 11 nodes was arrived at by comparison of results from one and six-node models.

The final assumption made is that all instrumentation systems considered provide automatic protection. This can be stated also as assuming that the instrumentation systems are independent from reliance on procedures during operation (such as trip setting adjustments, range-changing, and switching on). If the proper functioning of the instrument relies on procedure during operation, the probabilities of malsettings and altered response characteristics must also be considered.

It is felt that a statistical reliability analysis should be made when considering various instrumentation systems, but that this analysis should not be part of this study of functional requirements.

B. Basic Computational Methods

Various analog and digital programs have been developed for the instrumentation study. The first programs developed were one-node models. The results from the one-node model programs indicated more elaborate calculations were needed which would take into consideration geometrical effects. The one-node programs give satisfactory results for uniform reactivity ramps and are extremely useful for obtaining a feeling for the problem. One-node results are also used to check specific results of a model with more nodes. Each program will be briefly described.

1. ANALOG A

The ANALOG A program is a one-node model containing the standard, six, delayed-neutron group simulations. The fuel temperature is calculated as a function of power level, inlet water temperature, outlet water temperature and time. A negative reactivity feedback, due to the metal coefficient, is calculated as a function of fuel temperature. The outlet water temperature is calculated as a function of fuel temperature, inlet temperature and time. The program includes a VSR (Vertical Safety Rod) negative reactivity function dependent upon time after a trip signal. The VSR trip signals are derived from a percentage increase in power level, a fixed power level trip and a temperature trip. The temperature trip is obtained from a simulation of a ZTM system. This simulation has provision for instrument and water lag times and time

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constants for the RTD (Resistance Temperature Detector) and the water downstream of the fuel charge. The ANALOG A Program is used to find the maximum allowable reactivity ramps for a given initial power level and outlet water temperature limit as a function of the various trip settings, time constants, lag times, and VSR functions.

2. TRIP 02^{4/}

TRIP 02 is an IBM-7090 digital program and contains the same equations as the ANALOG "A" program except for the two differential equations simulating the RTD and the water downstream of the fuel. The TRIP 02 program has one additional feature. For a given limit, TRIP 02 will iterate and find the maximum allowable reactivity ramp.

3. TRIP 04^{4/}

TRIP 04 is an IBM-7090 digital program and includes all the equations and features of the TRIP 02 and ANALOG A program. In addition, TRIP 04 contains four differential equations for simulating a nuclear or temperature rate-of-rise system. TRIP 04 also has a provision for including equations simulating a nuclear period system. The rate-of-rise simulation has a provision for varying the time constant of the circuit, and the temperature signal is obtained from the simulated output of an RTD.

4. ANALOG B

The ANALOG B program is a three-delay-group, six-region, one-dimensional analog simulation. The one dimension is in the top-to-bottom direction of the reactor. Each region contains space and time-dependent three-delay-group equations developed by R. E. Miller.^{15/} Each region also contains the same heat transfer, time lag equations and delay times as the ANALOG A program. The vertical safety rod equation is a function of time and space in the ANALOG B program. The ANALOG B program calculates parameters as a function of both space and time. The previous mentioned programs calculate parameters only as a function of time. The ANALOG B program has the following features.

- a. A reactivity ramp can be inserted in any of the six regions of the reactor.
- b. The initial flux distribution can be varied and calculated as a function of space and time for a reactivity ramp in any region.
- c. The fixed power level trips, the percent of initial power level trips and temperature trips can be obtained from each region.

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C. Equations

The following heat transfer equations are used in the instrumentation study programs.^{5,6/}

$$L (AC \rho)_M \frac{dT_m}{dt} = w - L u (\bar{T}_m - \bar{T}_w) \quad (1)$$

where:

L = length of uranium charge

A = cross-sectional area of uranium

C = specific heat of uranium

ρ = density of uranium

w = power

u = heat transfer coefficient

$\bar{T}_m$ = average metal temperature

$\bar{T}_w$ = average water temperature

$$\bar{T}_w = \frac{T_o + T_{in}}{2}$$

T_o = the outlet water temperature

T_{in} = the inlet water temperature

t = time

$$\frac{L(AC\rho)_w}{2} \frac{dT_o}{dt} = L u (\bar{T}_m - T_w) - F C_1 (T_o - T_{in}) \quad (2)$$

where:

F = flow

C_1 = conversion constant

$(LAC\rho)_w$ = are same as parameters mentioned above except the values are for water.

$$\frac{1}{\lambda_1} \frac{dT_1}{dt} = T_o - T_1 \quad (3)$$

$$\frac{1}{\lambda_2} \frac{dT_{RTD}}{dt} = T_1 - T_{RTD} \quad (4)$$

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where:

T_1 = water temperature at the RTD

T_{RTD} = temperature output of RTD

$\frac{1}{\lambda}$ = time constant

The standard reactor kinetic equation for the one-node, six delay group programs are as follows:

$$\frac{dn}{dt} = \frac{(\rho - \beta)n}{l^*} + \sum_{i=1}^6 \lambda_i C_i \quad (5)$$

$$\frac{dC_i}{dt} = \frac{\beta_i n}{l^*} - \lambda_i C_i \quad (6)$$

where:

n = neutron density

ρ = total excess reactivity

β = fraction of fissions neutrons that are delayed

β_i = fraction of delayed neutrons in the i^{th} group

C_i = concentration of the i^{th} group of delayed neutron emitters

λ_i = decay constant of the i^{th} group of delayed neutron emitters

l^* = neutron mean lifetime

t = time

The reactivity/7/ due to the metal coefficient is given by

$$(\Delta K)_m = K_1 \sqrt{T_m + 273} \quad (7)$$

where:

K_1 = a constant

T_m = average uranium temperature in °C

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The space and time-dependent reactor kinetic equations are as follows: /5/

$$\frac{dN_1}{dt} = \frac{N_1}{l_0} \left\{ \frac{(1+\Delta K_1 - \beta)(1+L^2 B^2) [1+M^2(B_x^2+B_y^2)] - [1+2M^2(B_x^2+B_y^2)]}{1+M^2(B_x^2+B_y^2)} \right. \\ \left. - \frac{2M^2}{\Delta Z_1 \Delta Z_{1-1}} \right\} + \frac{2M^2}{l_0} \frac{N_{1T}}{[\Delta Z_1 + \Delta Z_{1-1}]} + \frac{2M^2 N_{1-1}}{l_0 [\Delta Z_1 + \Delta Z_{1-1}]} + P e^{-\tau B^2} \left(\sum_{j=1}^3 \lambda_j C_{j1} \right) \quad (8)$$

$$\frac{dC_{j1}}{dt} = \frac{1+L^2 B^2}{P e^{-\tau B^2}} \frac{B_j N_1}{l_0} - \lambda_j C_{j1} \quad (9)$$

where:

N_1 = neutron density in i^{th} region

t = time

l_0 = neutron mean lifetime

K_1 = K_{eff-1} of i^{th} region

B = fraction of fission neutrons that are delayed

L^2 = diffusion length

B^2 = geometrical buckling

B_x^2 = geometrical buckling in x direction

B_y^2 = geometrical buckling in y direction

ΔZ = node spacing

P = resonance escape probability

$e^{-\tau B^2}$ = fast neutron non-leakage probability

τ = fermi age

λ_j = decay constant of the j^{th} group of neutron emitters

C_{j1} = concentration of the j^{th} group of delayed neutron emitters in the i^{th} region

β_j = fraction of delayed neutrons in the j^{th} group

$M^2 = L^2 + \tau$

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The equations in TRIP 04^{8,9} for nuclear and temperature rate of rise are as follows:

$$\frac{dv_1}{dT} = \frac{-\alpha}{\tau_1} T_4 - \frac{v_1}{\tau_1} \quad (10)$$

$$\frac{dv_2}{dT} = \frac{\alpha}{\tau_2} T_4 - \frac{v_1}{\tau_2} - \frac{v_2}{\tau_2} \quad (11)$$

$$\frac{dv_3}{dT} = \frac{N}{\tau_3^2} - \frac{v_3}{\tau_3} \quad (12)$$

$$\frac{dv_4}{dT} = \frac{N}{\tau_3 \tau_4} - \frac{v_3}{\tau_4} - \frac{v_4}{\tau_4} \quad (13)$$

$$\alpha = \frac{C}{1} \quad (14)$$

where:

τ = time constant

N = neutron density

v_1 = an intermediate variable

v_2 = indicated rate (thermal)

v_3 = an intermediate variable

v_4 = indicated rate (nuclear)

C = a constant

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V. RESULTS OF CALCULATIONS

A. Discussion of Graphs

Graphs are presented which are typical of the results to date. The graphs illustrate the relation between trip settings, initial level, delay times, time constants, and reactivity. The graphs in general present a limit curve plotted as a function of initial tube power and maximum allowable reactivity ramp. The area to the left of the curve is the region of allowable operation within the 150 C criterion. When comparing curves, one must keep in mind that the reactivity ramps for the one-node model, are given in terms of full pile reactivities and the ramps for the nodal models are in terms of local reactivities. The region of interest on the curves is for reactivity ramps less than 1 mk/sec, since credible horizontal-control-rod reactivity withdrawal rates fall less than 1 mk/sec.

Figures 1, 2, 3, 4, 9, 10, 11, 12 are for a fixed, water-temperature trip system such as a zone temperature monitor. These curves are one-node model calculations. The delay time is the time between trip and the dropping of the VSR's. The t_1 and t_2 are time constants for equations 3 and 4. The time constant t_1 has a value of about 1.5 seconds and 2.5 seconds for the old reactors and K's respectively. ^{16/} All curves are for a 21 C inlet water temperature.

Figures 5, 6, 15, and 16 are one-node model results for a flux trip set at various percentages above the initial flux. Figures 7, 8, 13, and 14, are one-node model results for a fixed flux trip corresponding to the power level indicated on the curve. The curves show the effects of varying the trip point and the instrumentation delay. The curves are representative of a Beckman or octant system for a uniform ramp over the pile.

Figures 17 through 21 show results of the six-node model calculations for the BIF Reactors. The curves also contain one-node results for comparison purposes. A fairly distorted flux distribution is assumed; the ratio of the tube power at the top of the central zone to tube power at the bottom of the central zone is about two to one. The ECT for such a skewed flux is only about eight per cent less than the ECT for a cosine flux. The curves for the skewed flux were calculated from analog data to correspond to the same local ramp and node spacing as used for the cosine and flat flux distribution.

Figure 17 is for a temperature monitor with a fixed trip at 125 C. It is of worth to note that the allowable ramp for the skewed flux (Curve III) is shown less than the allowable ramp for the cosine flux (Curve II) at the same power level. Curve III was calculated from a different node spacing and in normalizing the results to the calculation of Curve II enough uncertainty may have been introduced to cause this unexpected result.

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Figures 18, 19, 20 and 21 are for percentage trips on initial flux. Curve I in each figure can be considered as a "worst" case for an octant system and Curve II a "worst" case for a Beckman system. Some of the curves show that the maximum allowable level even for zero ramp is close to the present operating power level range.

B. Fixed and Percent Trip Instrumentation

For either fixed or percent trips, there is some limiting power level above which operation would be unsafe since the allowable ramp is zero; i.e., a ramp of any size would cause one to exceed the limit before the transient could be turned around. The limiting power level decreases for the percent trip cases as the percent trip is increased. There may be a small range of rates for which the rate-of-rise would provide protection above the limiting power level. But the rate-of-rise wouldn't give protection for very fast ramps or ramps which result in a power rate less than the trip setting.

The allowable ramp, for a given percent trip case, increases as the initial power level decreases. The allowable ramp, for a fixed trip case, on the other hand, increases as the initial power level increases and approaches the trip level. A fixed trip case is in reality a percent trip for a given equilibrium power distribution. Due to variations in the power distribution at equilibrium, a fixed trip in terms of percent trip would probably be higher than a percent trip which is variable. For a given fixed flux or temperature trip, the maximum allowable ramp, for practical purposes, would depend on the relationship between the normal flux variations at equilibrium and the trip setting.

An increase in the fixed trip will decrease the allowable ramp for a given power level. Increasing the percent trip setting will decrease the maximum allowable ramp for a given initial level, and will decrease the maximum allowable level.

Consider the case where a flux monitor such as an octant system is operated as a fixed trip. Assume the fixed trip is set for 110 per cent of a flat equilibrium power distribution. Also assume that the equilibrium power level is the maximum allowed to protect for a given maximum credible ramp in a flat flux. Next, assume the pile power is below the equilibrium level, there is a cosine shaped flux distortion, and a small continuing linear power rise occurs at the initial pile power. If the power rate-of-rise is small, a power rate trip will not occur. The fixed flux trip would not give a trip until the tube powers in the center of the reactor were considerably greater than the tube powers for a flat equilibrium power distribution. This would most likely result in exceeding the temperature limit. In this case, a variable percent trip (varied by procedure or automatically) would probably provide more protection than the fixed trip.

A temperature monitoring device such as a ZTM has better geometrical coverage than an octant or present Beckman system. The ZTM operated as

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a fixed trip would give more protection than the octant system for the above mentioned case provided that the maximum allowable ramps were the same for both cases.

The one-node results show, for a given power level, that the maximum allowable ramp for a fixed flux trip is greater than the maximum allowable ramp for a fixed temperature trip. The results of the ANALOG B program, which considers the geometrical effects of the two systems, did show an improvement in the performance (in terms of maximum allowable ramp for a given level) of the ZTM relative to an octant system.

The curves show that small changes in delay times and time constants have a large effect on the maximum allowable ramp for a given initial power level. Consider the case of a fixed trip temperature monitoring device with and without a scanning device. If the system with the scanning device has an additional delay, such as 0.8 second, then the allowable ramp for a given level will be less than the allowable ramp without the delay.

The following sketch would be representative of the limit curves for a fixed trip and rate-of-rise system.

The intersection of the fixed trip and rate-of-rise curves would define the maximum allowable ramp. Adjusting the fixed trip and rate-of-rise trip would change the location of the curves and the maximum allowable ramp.

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C. Conclusions

1. B, D, DR, and F Reactors

The cases which have been considered for these reactors are both local and full pile insertions of reactivity which are linear functions of time, i.e., ramps. Octant, Beckman, and Zone Temperature instrumentation systems were considered. For those cases where local insertions of reactivity were considered, the effects of different flux distributions were also studied. Local reactivity insertions which would simulate the flushing of a full column of poison followed by a rod withdrawal have not been studied at this time.

For all of the data collected so far, the region of interest has centered on high level. In this region, the worst conditions occur for local insertion of reactivity. For this reason, only the results from the six-node model are summarized. The curves obtained from the one-node model, which represents the maximum reactivity insertions which can be tolerated when reactivity is inserted uniformly, are of academic interest only.

Curves in Figures 17-21 were used to obtain the results given below. The tables were obtained by using the worst conditions. For the Beckmans and ZIM this is found to occur when the flux is skewed to the top; the worst condition for the octant system occurs with a cosine flux distribution. The flux distributions referred to here are in the top-to-bottom direction. Although the case of the flushing of a poison column followed by the withdrawal of several rods has not been simulated, the most severe accident for B, D, DR, and F reactors which is credible, is considered to be the simultaneous withdrawal of three rods from a local region of the reactor. It is assumed that the full pile effect of each rod is two milli-k and for three rods six milli-k; shadowing is conservatively not considered. The local effect of these rods was obtained by using the 7090 buckling program. In the discussion which follows several statements will be made which use the word protected. Protected in this discussion means the transient will be turned around before the limit, which has been established for this study, is exceeded.

Table I gives the range of protection which various instruments provide, for various trip settings and delay times. For Table I it is assumed that the rods can be withdrawn in 40 seconds. At the present time the old reactors are operating at 1200 kw tube powers. With this in mind we see from Table I that with a 110 per cent trip and a one-second delay both the octant and the Beckmans provide protection in the high level range, the octant giving a greater amount of protection because of its better coverage. Notice, however, that for 130 per cent trip and a one-second delay neither the Beckman nor the octant provide protection for equilibrium tube powers. Even for zero second delay neither instrument gives adequate protection. So long as procedures are followed for the 110 per cent setting both the Beckman and octant give protection.

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If procedures break down the reactor may not be protected for this situation. Only the ZTM provides a limited range of protection, but this instrument relies less on procedure than does the other system.

If it is made physically impossible to move more than two rods and if it is assumed that the rods can be withdrawn in 40 seconds, the range of protection is extended as shown in Table II. Here again the octant and Beckman instruments provide adequate protection for 110 per cent trip and a one-second instrument delay, the range being somewhat greater because of the decreased ramp. For 130 per cent trip and zero or one second delay neither instrument gives protection for equilibrium tube powers. The range of protection provided by the ZTM has increased and again provides procedure-free protection in this range. The range of protection given here for the ZTM may be sufficient when the ZTM is coupled with a rate-of-rise and period instrument. Confirmation of this must await further studies.

Table III gives the range of protection when only one rod can be withdrawn in 40 seconds. The results are about the same as in Tables I and II. The octants and Beckmans give protection so long as procedures are followed but may not give protection when procedures break down. The range of protection of the ZTM has again been increased.

One other case of interest is the withdrawal of three rods but at half speed, i.e., it will take 80 seconds to withdraw a rod. Table IV summarizes these results and they are similar to those in Table III.

For the B, D, DR, and F reactors the Beckmans and an octant system provide adequate protection when three rods are withdrawn simultaneously provided procedures are followed. The ZTM gives only a limited range of protection which is not adequate. If the rod speed is cut in half or if it is made physically impossible to withdraw more than two rods the ZTM will provide adequate range of protection. The Beckmans and octant also provide protection if procedures are followed.

2. C and H Reactors

For C and H reactors the most severe accident which is credible in the equilibrium region is the simultaneous withdrawal of four rods from a region of the reactor. It is again assumed that the full pile worth of one rod is two milli-k and for four rods 8 milli-k. Table V summarizes the range of protection provided by the various safety systems assuming the rods can be withdrawn in 40 seconds. The results indicate, as for the B, D, DR and F reactors, that Beckmans and octant give protection which is adequate as long as procedures are followed. The range of protection for the ZTM is

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quite small and not adequate. For the withdrawal of three, two, and one rod in forty seconds the results are the same as those given in Tables I, II, and III respectively. The range of protection for withdrawal of four rods simultaneously but at half speed are given in Table VI.

For C and H reactors the ZTM gives adequate high level protection if the number of rods which can be withdrawn is limited to two or if the rod speed is cut in half. The octant and Beckman provide protection even in the worst case if procedures are followed but do not give protection for any of the cases considered if procedures break down.

3. K Reactors

To date no nodal model cases have been run for the K reactors; complete data will be obtained during the next set of analog runs. One may surmise from K reactor one-node calculation that the same general conclusions on high level instrumentation apply to both K and the old reactors; i.e., the protection provided by an octant or Beckman system is highly dependent on procedure, and the use of a ZTM system may require some restrictions on physical limitation of the HCR system withdrawal rates.

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TABLE I

Three Rods at Full Speed
Local Ramp = 0.63 mk/sec

<u>System</u>	<u>Range of Protection in KW</u>
ZTM, 125° C Trip	1160 - 1320
Octant, 110% trip, 0 second delay	0 - 1440
Beckman, 110% trip, 0 second delay	0 - 1360
Octant, 110% trip, 1 second delay	0 - 1380
Beckman, 110% trip, 1 second delay	0 - 1260
Octant, 130% trip, 0 second delay	0 - 1180
Beckman, 130% trip, 0 second delay	0 - 1080
Octant, 130% trip, 1 second delay	0 - 1080
Beckman, 130% trip, 1 second delay	0 - 1000

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TABLE II

Two Rods at Full Speed
Local Ramp - 0.465 mk/sec

<u>System</u>	<u>Range of Protection in KW</u>
ZTM, 125° C trip	860 - 1320
Octant, 110% trip, 0 second delay	0 - 1440
Beckman, 110% trip, 0 second delay	0 - 1380
Octant, 110% trip, 1 second delay	0 - 1400
Beckman, 110% trip, 1 second delay	0 - 1280
Octant, 130% trip, 0 second delay	0 - 1180
Beckman, 130% trip, 0 second delay	0 - 1080
Octant, 130% trip, 1 second delay	0 - 1100
Beckman, 130% trip, 1 second delay	0 - 1020

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TABLE III

One Rod at Full Speed
Local Ramp = 0.28 mk/sec

<u>System</u>	<u>Range of Protection in KW</u>
ZTM, 125° C trip	460 - 1320
Octant, 110% trip, 0 second delay	0 - 1460
Beckman, 110% trip, 0 second delay	0 - 1370
Octant, 110% trip, 1 second delay	0 - 1430
Beckman, 110% trip, 1 second delay	0 - 1290
Octant, 130% trip, 0 second delay	0 - 1190
Beckman, 130% trip, 0 second delay	0 - 1090
Octant, 130% trip, 1 second delay	0 - 1110
Beckman, 130% trip, 1 second delay	0 - 1050

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TABLE IV

Three Rods at Half Speed
Local Ramp = 0.32 mk/sec

<u>System</u>	<u>Ramp of Protection in KW</u>
ZTM 125° C trip	390 - 1320
Octant, 110% trip, 0 second delay	0 - 1460
Beckman, 110% trip, 0 second delay	0 - 1370
Octant, 110% trip, 1 second delay	0 - 1430
Beckman, 110% trip, 1 second delay	0 - 1290
Octant, 130% trip, 0 second delay	0 - 1190
Beckman, 130% trip, 0 second delay	0 - 1090
Octant, 130% trip, 1 second delay	0 - 1110
Beckman, 130% trip, 1 second delay	0 - 1050

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TABLE V

Four Rods at Full Speed
Local Ramp = 0.76 mk/sec

<u>System</u>	<u>Range of Protection in KW</u>
ZTM, 125° C Trip	1220 - 1320
Octant, 110% trip, 1 second delay	0 - 1360
Beckman, 110% trip, 1 second delay	0 - 1250
Octant, 130% trip, 0 second delay	0 - 1160
Beckman, 130% trip, 0 second delay	0 - 1070
Octant, 130% trip, 1 second delay	0 - 1060
Beckman, 130% trip, 1 second delay	0 - 980
Octant, 110% trip, 0 second delay	0 - 1430
Beckman, 110% trip, 0 second delay	0 - 1350

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TABLE VI

Four Rod at Half Speed
Local Ramp = 0.38 mk/sec

<u>System</u>	<u>Range of Protection in KW</u>
ZTM, 125° C trip	660 - 1320
Octant, 110% trip, 0 second delay	0 - 1450
Beckman, 110% trip, 0 second delay	0 - 1370
Octant, 110% trip, 1 second delay	0 - 1420
Beckman, 110% trip, 1 second delay	0 - 1280
Octant, 130% trip, 0 second delay	0 - 1180
Beckman, 130% trip, 0 second delay	0 - 1090
Octant, 130% trip, 1 second delay	0 - 1100
Beckman, 130% trip, 1 second delay	0 - 1040

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LIMIT CURVES

Plots of the maximum allowable reactivity ramp, for which the criterion can be met, as a function of tube power. Effects of various parameters are also included. On curves where the flux distribution is specified the results were obtained from the six-node model.

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Curve I 0 second delay
 Curve II 1 second delay
 Curve III 2 seconds delay

FIGURE 15
 K REMOTE
 110% TRIP

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Initial Peak Tube Power in kw

FIGURE 17
BDF REACTORS
125° C TRIP
0 SECOND DELAY
 $t_1 = t_2 = 1.5$ sec

Curve I Flat flux in central zone, ramp in center
Curve II Cosine flux, ramp in center
Curve III Flux skewed to top, ramp in top
Curve IV One node model

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FIGURE 19
BDF REACTORS
110% TRIP
1 SECOND DELAY

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FIGURE 20
BDF REACTORS
130% TRIP
0 SECOND DELAY

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FIGURE 21
EDF REACTORS
130% TRIP
1 SECOND DELAY

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