226
11-8

2335

ANCR-1082

UC-80

ATRC SAFETY ANALYSIS REPORT SUPPLEMENT-1 OF IDO-16950-1

D.W. Knight and N.C. Kaufman

Aerojet Nuclear Company

NATIONAL REACTOR TESTING STATION

Idaho Falls, Idaho — 83401

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

DATE PUBLISHED—SEPTEMBER 1972

PREPARED FOR THE

U. S. ATOMIC ENERGY COMMISSION

IDAHO OPERATIONS OFFICE UNDER CONTRACT AT(10-1)-1375

DISCLAIMER

Printed in the United States of America
Available from
National Technical Information Service
U. S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22151
Price: Printed Copy $3.00; Microfiche $0.95

ANCR-1082

Reactor Technology
TID-4500

ATRC SAFETY ANALYSIS REPORT

SUPPLEMENT-1 OF IDO-16950-1

D. W. Knight and N. C. Kaufman

AEROJET NUCLEAR COMPANY

Date Published - September 1972

PREPARED FOR THE U. S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
UNDER CONTRACT NO. AT(10-1)-1375

CONTENTS

FIGURES

TABLES

1.0 INTRODUCTION

Technical Specifications for the Advanced Test Reactor Critical Facility (ATRC) have recently been prepared. During this preparation, all of the ATRC Operating Limits and associated analyses were reconsidered. Where justifiable, some requirements were made less restrictive. Where necessary to characterize more completely current operational practice and current safety philosophies, additional limits and requirements were specified. However, the majority of the requirements of the new Technical Specifications derive from considerations and analyses in the original safety analysis for ATRC[1]. Moreover, no new accident potentials or safety issues were identified.

This document has been prepared to supplement the original ATRC Safety Analysis Report (SAR)[1]. It provides the basis for those revised requirements and conclusions incorporated in the new Technical Specifications. The areas considered in the report are:

(1) Maximum credible accidental reactivity insertion.

(2) Reactor operating power level and trip level for the Neutron Level Subsystem.

(3) Permissible ramp reactivity insertion rates.

(4) Hold-down reactivity requirements.

(5) Shutdown reactivity requirements.

(6) Definition of major incident

(7) Unreliability requirements of Neutron Level and Safety Rod Subsystems.

2.0 SAFETY ANALYSIS

2.1 Maximum Credible Accidental Reactivity Insertion

The maximum credible accidental reactivity insertion at ATRC was originally considered to result from the rupture of a special high-pressure loop. During such an accident, the entire flux trap annulus was assumed to be voided[1,2]. Measurements have since shown that the reactivity effect of voiding the high-pressure loop may, under some conditions, be larger than the 1.20$ originally postulated. Thus, the operation of the high-pressure loop was restricted to require the use of an aluminum filler piece in the flux trap annulus. Under these conditions the reactivity effect of voiding the loop can be no more than one dollar ($\sim$0.75% $\Delta k/k$)[2].

With the large voiding effect thus prevented, an analysis of possible credible accidents at ATRC was conducted. No credible accidents were identified which would introduce more than one dollar in reactivity in the reactor. However, in order to be conservative, the maximum credible accidental reactivity insertion is now considered to result from an unspecified accident which would introduce 1.10$ into the reactor in a stepwise manner.

2.2 Reactor Operating Power Level and Trip Level

The ATRC Operating Limits specified a maximum power level and a trip level. The Technical Specifications now make no distinction, because the reactor can be operated at any power level up to the trip level. To prove the validity of this assertion, three parameters which might limit the operating power level of the ATRC were investigated:

(1) Radiation level at the surface of the ATRC canal.

(2) The temperature of the ATRC fuel element hot spot.

(3) The relationship between operating power level and the total energy released during an accident.

The upper limit for the reactor power level was found to be controlled by accident conditions. The limiting power level and trip level, based on accident considerations, were established in two steps. The first step consisted of studying the relationship between total energy release and initial power level, for a fixed trip level. The results of this study are shown in Table I. It can be seen that the higher the power level the greater the total energy release from an accident. Moreover, the maximum energy release is seen to result from an initial power level only slightly below the trip level.

TABLE I

TOTAL ENERGY RELEASE FROM THE MAXIMUM CREDIBLE ACCIDENT AS A FUNCTION OF THE INITIAL POWER LEVEL

Initial Power Level	Total Energy Release Above Trip Level (MW-sec)
0.25 mW	0.72
5.0 kW	3.0
11.5 kW	6.2

Safety Rod Worth - 5.3$

Trip Level - 12 kW

Reactivity Insertion (Step) - 1.10$

Safety Rod Drop Time - 800 msec

Safety Rod Release Time - 25 msec

β/ℓ - 210 sec^{-1}

The second step was to determine the initial power level and trip level that would result in a total energy release of approximately 14 MW-sec. An energy release of 14 MW-sec would give a protective margin for a minor incident of 100% (a total energy release above the trip level of 28 MW-sec would result in minor fuel plate deformation and melting). The results of the calculations are shown in Table II. From these results a maximum trip level of 26 kW can be permitted to limit the energy release to 14 MW-sec.

TABLE II

TOTAL ENERGY RELEASE FROM STEP ACCIDENT AS A FUNCTION OF INITIAL POWER LEVEL AND TRIP LEVEL

Initial Power Level (kW)	Trip Level (kW)	Total Energy Release Above Trip Level (MW-sec)
14.5	15	8.3
19.5	20	10.7
26.5	27	14.3

Safety Rod Worth - 5.3$

Safety Rod Drop Time - 800 msec

Safety Rod Release Time - 25 msec

Reactivity Insertion - 1.10$

β/ℓ - 210 sec^{-1}

Reactor behavior during all postulated reactivity accidents was calculated using the spatially independent IREKIN Code[3]. The model used did not include expected negative thermodynamic feedback resulting from temperature increases and void formation. It thus considerably overestimated energy release for a given accident. The parameters used in the calculations are safety limit values or, as in the case of β/ℓ, a value which would give a larger energy release than if measured values were used (the largest measured value of β/ℓ is 185 sec^{-1}[4].

The radiation level at the canal surface above the reactor and the temperature of the fuel element hot spot do not present a hazard at a power level of 26 kW. The radiation level above the reactor at a power level of 26 kW would be approximately 13 mR/hr[5]. The temperature of the hot spot in an ATRC fuel element at a power level of 26 kW under natural convection conditions would be less than 100°F. These calculations were made using the method developed for the BSR-II Reactor[6]. A peaking factor of 3.6 was used for the latter calculations[4].

2.3 Ramp Reactivity Insertion Rates

Ramp reactivity insertion rates have been reevaluated,based on a trip level of 26 kW. The results of the analyses are shown in Table III.

TABLE III

STUDY OF ATRC RAMP ACCIDENTS

Ramp Rate ($/sec)	Total Energy Release Above Trip Level (MW-sec)
0.13	0.63
0.15	1.30
0.20	6.90
0.257	51.00

Safety Rod Worth - 5.3$

Trip Level - 26 kW

Safety Rod Drop Time - 800 msec

Safety Rod Release Time - 25 msec

Initial Power Level - 0.25 mW

β/ℓ - 210 sec^{-1}

Interpolating from these results, a ramp reactivity insertion rate of 0.22 $/sec will produce 14 MW-sec of energy, which corresponds to the same protective margin that exists for the maximum step reactivity insertion (1.10$).

The energy calculations, which were made using the IREKIN Code, are considered very conservative. Not only were negative thermodynamic feedback effects neglected, but the calculations contained the following additional conservatisms:

(1) The calculations assumed the safety rods scrammed from their upper limits, even though a moment before the scram they were at positions of maximum differential reactivity worth (middle of safety rod withdrawal stroke).

(2) The scram resulted when the power level reached the Safety Limit (26 kW). No credit was taken for operator action or any other subsystem action.

2.4 Hold-Down Reactivity

To prevent inadvertent criticality during subcritical experiment or fuel element changes, sufficient hold-down reactivity must be available to allow for loading errors and uncompensated reactivity changes. The largest single uncompensated change in reactivity that can be conceived is approximately 1.5$ (fuel element insertion in high-worth position). Applying a safety factor to account for two successive uncompensated changes and adding conservatism to account for an error equivalent to 1.0$, the minimum hold-down reactivity required is 4.0$ ($\sim$3.0% $\Delta k/k$).

2.5 Shutdown Reactivity

The amount of shutdown reactivity necessary to mitigate an accident cannot be determined without considering the rate at which the shutdown reactivity is inserted into the reactor. The studies in Section 2.2 have shown that the maximum credible accident can be acceptably mitigated if -5.3$ is inserted into the reactor with a delay of 25 msec and a drop time of 800 msec and if the neutron level subsystem trip level is 26 kW. In those studies it was assumed that the safety rods fell under gravity with a constant acceleration of 9.4 ft/sec^2. A curve showing the position of the safety rods as a function of time is presented in Figure 1, while the reactivity worth of the safety rods as a function of position, that was assumed in the calculations, is shown in Figure 2.

For a step or ramp reactivity accident the largest credible positive reactivity that would require compensation is 1.10$. Applying a safety factor of two (compensation would be required for 2.20$) and assuming a minimum stroke worth of 5.3$, no more than 20 inches of safety rod insertion, as shown in Figure 2, will be required to mitigate conservatively a step accident of 1.10$ or a ramp accident with a reactivity insertion rate of 0.22 $/sec. Allowing an additional 9 inches to assure that all 20 inches are above the shock absorbers (and thus in free-fall) the safety rods must be 29 inches withdrawn prior to criticality. As shown in Figure 2, 29 inches of withdrawal will provide the required reactivity.

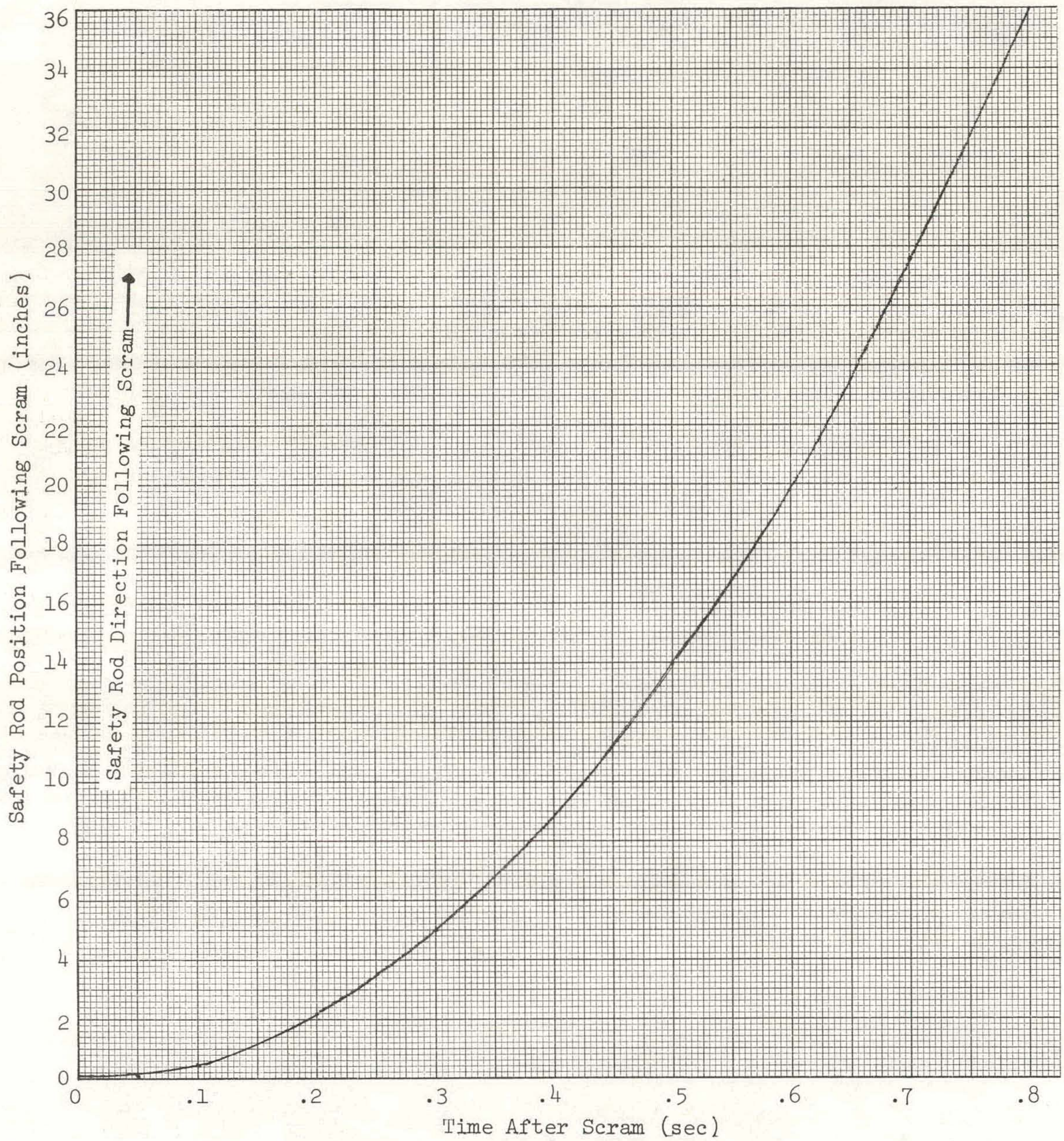


Fig. 1. Position of ATRC safety rods as function of time following a scram.

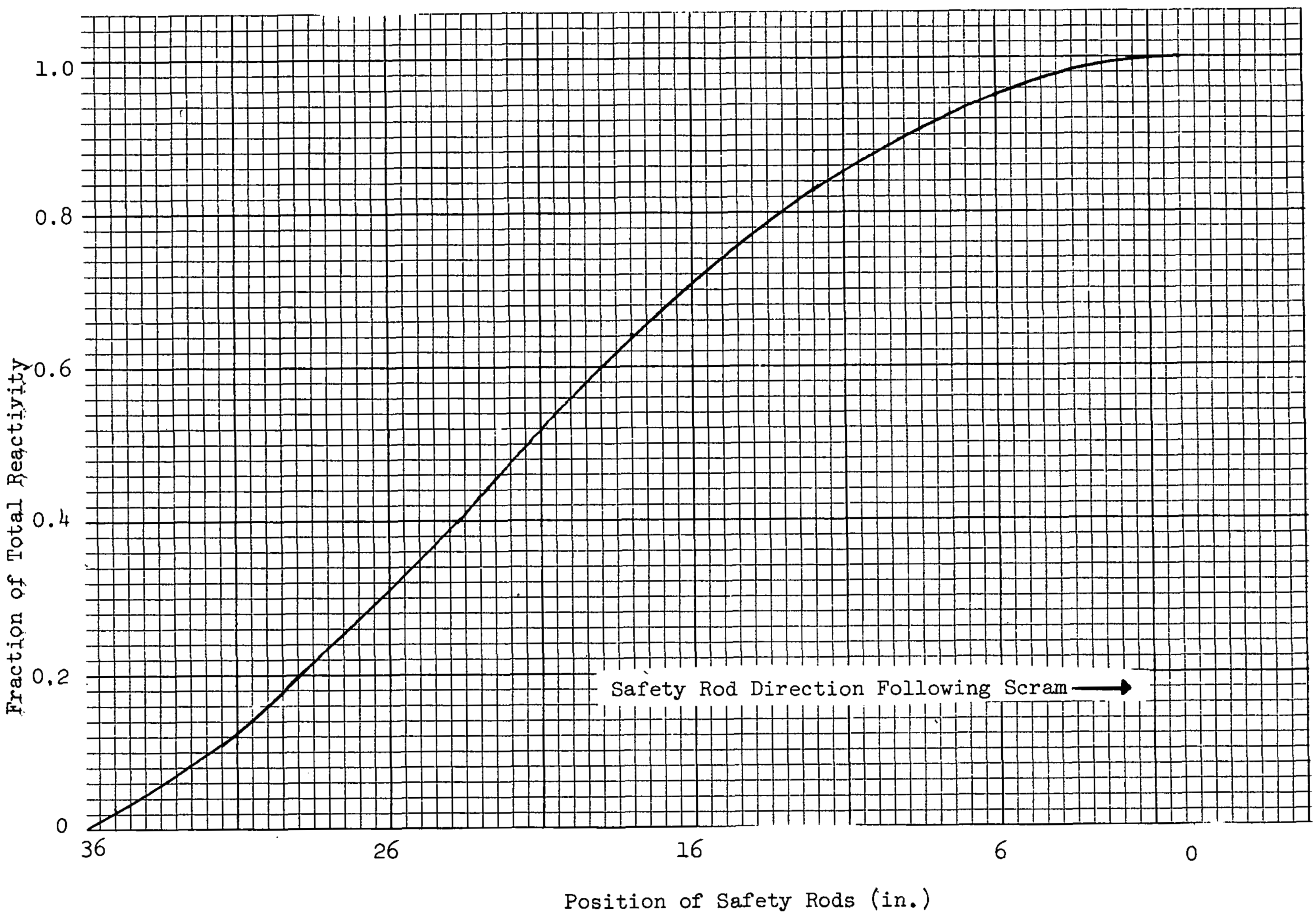


Fig. 2. Reactivity of safety rods as a function of position.

2.6 Major Incident

A major incident at ATRC has been defined as an incident for which there may be damage to the permanent reactor components (eg, grid plate, control rod drive mechanisms). However, it must be feasible to repair or replace the damaged components. The damage to fuel elements may be extensive. Radiological doses to personnel resulting from release of fission product inventory caused by melting of fuel plates may be greater than outlined in AECM 0524 but will be less than requirements of 10CFR100.

Because the ATRC does not have a pressure vessel or a containment structure and because the reactor structural components (including the pool structure) can be repaired or replaced, the limiting criteria for a major incident will be the exposure of personnel to ionizing radiation. An example of a major incident would be the insertion of sufficient reactivity to cause fuel plate melting and the release of fission products to the ATRC building and eventually to the environment. For calculational purposes a major incident has been arbitrarily defined as the melting of the equivalent of one fuel element. Calculations have shown it would require a total energy release of approximately 60 MW-sec to melt the equivalent of one fuel element (melting would occur at the core hot-spots). Radiological doses that would be received by ATRC personnel and off-site personnel, if the equivalent of one fuel element from a 24-MWd ATR core melted, are shown in Table IV.

TABLE IV

RADIOLOGICAL DOSES TO ATRC AND OFF-SITE PERSONNEL RESULTING FROM MELTING OF A FUEL ELEMENT

	ATRC Personnel	NTRS Boundary
Thyroid Dose	232 rem	3.2 rem
Whole Body Dose	0.84 rem	0.3 mrem
Bone Dose	0.83 rem	0.01 rem

The dose calculations in Table IV were made using the following assumptions:

(1) The ATRC core was preirradiated in the ATR and had generated 24 MWd of energy.

(2) The fuel elements had cooled for five days prior to insertion in the ATRC. The postulated accident occurred immediately after the elements had been placed in the reactor.

(3) It took a maximum of ten seconds for personnel to evacuate the ATRC building after the accident.

(4) The quantity of fission products released into the building was fractionated according to TID-14844[7] (ie, one percent of the solids, 50 percent of the halogens, and 100 percent of the noble gases). The release was assumed to diffuse instantaneously throughout the building.

(5) The meteorological parameters were those associated with a short-term release, ie, a Hilsmeier-Gifford σ_y and a Markee σ_z. The long distance diffusion was characterized by Class F (inversion) weather with a 2-m/sec windspeed.

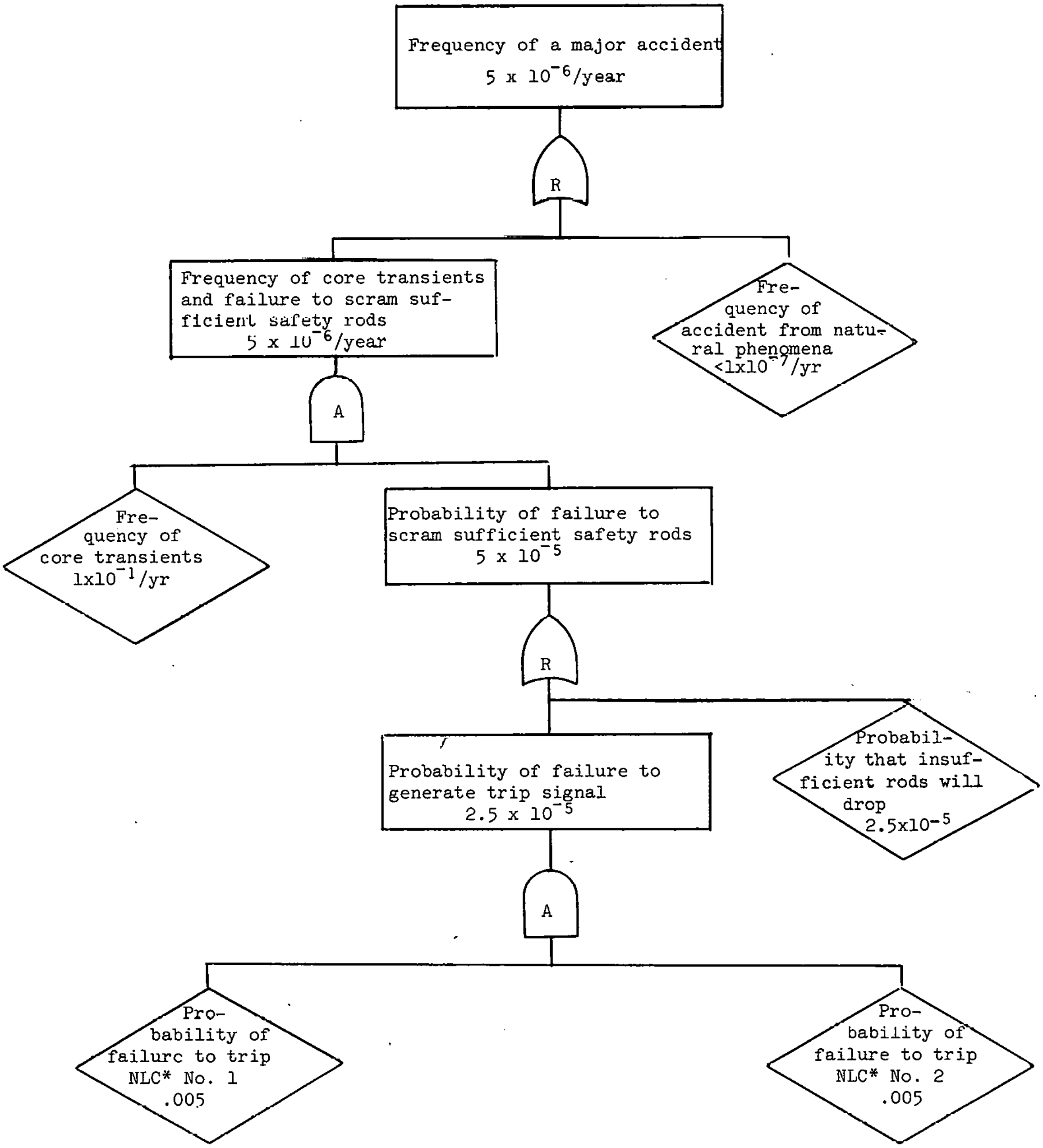
The bone dose is the result of inhaling ^{90}Sr. The 10CFR100 doses for accident conditions are 300-rem thyroid dose and 25-rem whole body gamma dose. The IDO guideline value bone dose is 150 rem.

2.7 Protection System Reliability Studies

A preliminary risk tree analysis of the ATRC has been completed. The preparation of the risk tree, in general, followed procedures outlined by Romano Salvatori[8]. As new data become available the risk tree can be updated. The risk tree is shown in Figure 3.

The unreliability requirements of the Neutron Level Subsystem which resulted from the analysis (0.005 for each channel, and assuming a minimum of two channels) and upon which the specifications are based are reasonable and very close to unreliability requirements which are expected to be recommended in an IEEE standard. An unreliability of 0.005 can be achieved through the use of quality components, good engineering design, and the use of channel testing intervals based on fail-unsafe failure rate data. When the unreliability of each channel is maintained $\leq$ 0.005, the probability of failing to generate a trip signal will be $\leq 2.5 \times 10^{-5}$. Moreover, when the probability of more than two safety rods failing to scram is maintained at or below 2.5×10^{-5}, the probability of failure to trip sufficient safety rods to mitigate an accident will be $\leq 5 \times 10^{-5}$ (combined probability of failure to generate a trip signal and failure to scram at least three safety rods). To maintain the unreliability of the safety rod subsystem at 2.5×10^{-5} the unreliability of each of the five safety rods must be $\leq$ 0.014.

The frequency of core transients and the frequency of natural phenomena which would cause a major accident at ATRC are not well known. Therefore, best estimates of these values were used in the risk tree analysis. The frequency of reactor power transients has been set at 1×10^{-1}/year. The actual frequency of reactor power transients is believed to be lower. The frequency of natural phenomena which would cause a major accident at ATRC is considered to be extremely small and has accordingly been set at 1×10^{-7}/year.



*Neutron Level Channel

Fig. 3. Simplified ATRC risk tree

3.0 REFERENCES

[1] E. E. Burdick and J. W. Henscheid (eds.) Advanced Test Reactor Critical Facility Safety Analysis Report, IDO-16950-1 (July 1964).

[2] N. C. Kaufman and J. W. Henscheid, Private Communication.

[3] R. J. Wagner, IREKIN--Program for the Numerical Solution of the Reactor Kinetics Equations, IDO-17114 (Jan. 1966).

[4] J. W. Henscheid, N. C. Kaufman, and J. L. Durney, A Summary of Data from the ATRC Experiment Program, IN-1053 (Feb. 1957).

[5] J. O. Howard, G. M. Jacks, Advanced Test Reactor Final Shielding Design, IDO-24467 (Dec. 1963).

[6] E. G. Silver and J. Lewin, Safeguard Report for a Stainless Steel Research Reactor for the BSF (BSR-II), ORNL-2470 (Aug. 1958).

[7] J. J. DiNunno et al, Calculations of Distance Factors for Power and Test Reactor Sites, TID-14844 (1962).

[8] Romano Salvatori, "Systematic Approach to Safety Design and Evaluation", Transactions of IEEE Symposium on Nuclear Science, Vol. 18, No. 1 (Feb. 1971) pp 495-508.