THERMAL TRANSIENT BEHAVIOR
OF THE NEUTRON SENSOR
OF A REACTOR SAFETY DEVICE

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ABSTRACT

The thermal transient behavior of the fuse component of a reactor safety device has been investigated in the KEWB I reactor. The fuse is a neutron flux sensor which, at excessive flux levels, will trip the safety device and cause reactor shutdown.

Experimental results on the majority of reactor transients were in good agreement with calculations. However, on the fastest transients, experimental results were obscured by spurious effects during those intervals when the radiation level was high.

It was not possible to demonstrate reactor shutdown with the safety devices because the inherent shutdown mechanism of the KEWB reactor was too efficient. The integrated neutron flux levels attained with the fastest transients were insufficient to cause the fuse to trip starting from ambient temperatures. Nevertheless, those tests have shown that the fuse component design is adequate for the high power level reactors for which it is intended. A successful trial run was made to demonstrate fuse tripping on transient neutron flux by heating the fuse to about 20° C above ambient temperature prior to the beginning of the transient.

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I. INTRODUCTION

An ultimate goal of nuclear reactor design is the achievement of an inherently safe reactor. Although the probability of a serious incident is considered to be rather small for present reactors, as long as the probability does exist, the question of reactor safety will be of considerable importance.

A normal reactor control system incorporates safety features such that under adverse circumstances the control rods are returned to the reactor core to effect a shutdown. Maximum safety can be obtained by having a control system with many fail-safe features. However, so many interlocks can be included that the system becomes quite complicated and a source of excessive down-time of the reactor due to component failure. The use of independent safety devices offers a solution compromising neither reactor safety nor reactor operating time. Most circumstances may be handled by the control system while the ultimate safety is provided by the independent safety system.

Although ultimately the majority of reactors may be designed for power purposes, at present the greatest need for safety devices is in research reactors. Many of these have large amounts of excess reactivity included in the core to take care of losses caused by experiments and fuel burnup. Moreover, the programming of these reactors is usually not constant, with each startup affected by different experiments and configurations. In addition, due to the variety of applications of such experimental machines, there is always the possibility of a hazard due to unforeseen circumstances. These considerations make the need of safety devices for research reactors most urgent.

A program to develop safety devices for reactor protection is in progress at Atomic International. This report is concerned with the in-pile testing of a key component of one such safety device designed for application to research reactors. In this component for the safety system, the neutron flux sensor, neutron flux is converted into heat by the process of fission of uranium-235. The heat is used to melt solder and release a gas, stored under high pressure, which actuates the system.

II. DESCRIPTION OF SAFETY DEVICE

Early in the program, a survey was made to determine the number and relative safety of the various types of research reactors. The survey revealed that the most popular, but the least safe, were the enriched uranium-fueled, light-water moderated reactors.¹ These included the swimming pool type such as the Bulk Shielding Facility Reactor and the tank type such as the Materials Testing Reactor. These reactors have very short prompt neutron lifetimes. Hence, a given excess reactivity (greater than prompt critical) causes them to have a very short period of flux increase.²

In view of these findings, the safety device was designed to fit into an empty fuel or reflector position of a reactor of the BSFR or MTR type. The device, shown in Fig. 1, is a completely unitized system called the Double-Diaphragm Safety Device.³ It is composed of a high-pressure storage chamber outside the reactor core and a low-pressure receiver chamber within the core separated by an intermediate chamber. The walls of the intermediate chamber contain rupturable diaphragms which remain intact as long as the pressure in the intermediate chamber is maintained. Boron trifluoride is kept at high pressure in the storage chamber and helium is maintained at about half that pressure in the intermediate chamber. A release tube, whose terminal point lies well within the core region, leads from the intermediate chamber into the receiver. An enriched uranium-loaded cap is sealed to the end of the tube by means of a low-melting-point solder. Under normal operating conditions, the heat generated in the uranium is drained from the solder to the outside wall of the device. In the event of a reactor excursion, however, the excessive heat generation in the uranium causes the solder to melt. The cap then blows off and relieves the helium pressure in the intermediate chamber so that an excessive pressure differential is produced across the first diaphragm. The two diaphragms, bursting sequentially, allow the boron trifluoride to expand into the receiver. The introduction of this poison into the reactor core produces the desired reactivity change.

Fig. 1. Double Diaphragm Safety Device

III. PURPOSE OF IN-PILE TESTS

The empirical determination of the thermal transient behavior of the fuse component or neutron sensor of the system, was considered essential to the establishment of the feasibility of the system as a reactor safety device. The in-pile tests were conducted in the KEWB I transient facility.⁴ This reactor offered a convenient means for a limited investigation of the thermal transient behavior of the fuse component. In addition, there was a possibility that the tests would demonstrate the feasibility of the system as a safety device. These tests are considered to be preliminary to higher power tests to be conducted in the SPERT I reactor,⁵ since SPERT I is more representative than KEWB I of the type of reactor for which the safety device is designed.

The fuse behavior was to be determined by observing temperature in various parts of the fuse for a series of reactor transients. Trip time, response time (which is defined as that interval between reactor power reaching trip level and solder temperature reaching its trip level) and maximum power overshoot can be derived from the curves of fuse temperature and reactor power as functions of time.

Preliminary calculations indicated a possibility that there would be sufficient heat generation in the fuse to cause device triggering on the severest KEWB transients. Subsequent testing and more detailed calculations showed that the device would not trip during the time of increasing reactor power and that, if tripping did occur, it would be after the power pulse was over.

IV. CALCULATIONS

The safety device fuse component, shown in Fig. 2, consists of a series of layers: 1) the Monel cap, 2) the solder, 3) the heater of uranium dioxide in a copper matrix, and 4) the stainless-steel thermal pad.³ In the complete safety device, this assembly is attached to a copper block from which heat flows through the device wall to the reactor coolant. In the transient test models, this copper block was unnecessary because there was no steady-state heat to dissipate.

Fig. 2. Safety Device Fuse Component

The functions of the various parts are as follows: the thermal pad provides a temperature gradient to allow the fuse ambient temperature to be near the solder melting point under steady-state reactor power, a condition which is desirable for optimum fuse response time; the heater provides the heat for maintaining elevated fuse temperature and for melting the solder; the cap retains the pressure in the intermediate chamber until the solder melts.

The laminar construction of the fuse simplifies calculations of fuse thermal behavior. Since the layers are thin, the heat loss at the edge of the fuse can be neglected. Consequently, heat flow can be assumed to be unidirectional and the problem reduced to that of one-dimensional, multiregion heat transfer. At the moderate temperatures encountered in the problem, radiation and conduction losses to the air may also be neglected. Convection losses to the air at the face of the cap are taken into account, however. The fuse geometry, simplified for the purpose of calculations, is shown in Fig. 3.

Fig. 3. Approximation of Fuse Geometry for Heat Transfer Calculations

The analysis is based on solution of the differential equation for heat conduction in one dimension,

$$\frac{\partial \theta}{\partial t} = a \frac{\partial^2 \theta}{\partial x^2} + \frac{N \sigma \beta \phi(t)}{\rho c}$$

where

θ = temperature ($^{\circ}\text{C}$)

t = time (sec)

x = distance (cm)

ρ = density (gm cm^{-3})

c = heat capacity ($\text{cal gm}^{-1} \text{ } ^{\circ}\text{C}^{-1}$)

a = thermal diffusivity ($\text{cm}^2 \text{sec}^{-1}$) = $K/\rho c$

K = thermal conductivity ($\text{cal cm}^{-1} \text{ } ^{\circ}\text{C}^{-1} \text{sec}^{-1}$)

N = atomic density of uranium-235 (atoms cm^{-3})

σ = microscopic fission absorption cross section (cm^2)

$\phi(t)$ = neutron flux ($\text{cm}^{-2} \text{sec}^{-1}$)

β = conversion factor (calories per fission)

The boundary conditions are: 1) at each interface the temperature is continuous, 2) at each interface the net heat flux is zero, 3) at the cap boundary, heat is lost by convection to the air, and 4) at the steel tube boundary, the temperature remains constant. The initial condition for the problem is that at time zero the temperature is uniform throughout.

The equation with the proper constants and boundary conditions is written for each region. Essentially, the equations are solved simultaneously by the PACE computer* and temperature change in each region plotted as a function of time. A list of the constants for each region is given in Table I. A typical set of curves for an actual KEWB transient is shown in Fig. 4.

V. APPARATUS AND PROCEDURE

Although the safety device was designed for reactors such as the MTR or BSFR, the test assemblies had to be made to fit the central exposure facility of the KEWB I reactor. The test devices were therefore made with an outside diameter of 1.6 inches, rather than the three-inch square size of swimming pool type reactor fuel elements. The central exposure facility of the KEWB I reactor also contains the poison rod used to initiate reactor transients.⁴ Consequently, the devices were designed to be inserted only to the center of the core leaving the other half of the central exposure facility for the poison rod. Since the safety devices occupied a location symmetrical to that occupied by the poison rod, they were designed to control as much reactivity as the rod.

The length of the devices was chosen from considerations of storage of the poison away from the core before tripping, and proper poison concentration in the receiver chamber after tripping. The first 20 inches of length of the devices constituted the receiver and intermediate chambers. The poison in the storage chamber was located about 14 inches away from the core. Over-all length of the devices was 30 inches. A handling extension was used to facilitate insertion and removal of the devices from the reactor. The components of one device are shown in Fig. 5.

* PACE, Precision Analog Computer Equipment, manufactured by Electronic Associates, Inc.

TABLE I
PHYSICAL CONSTANTS FOR THERMAL CALCULATIONS

Region No.	Region	Material	Limits		Density (gm/cm ³)	Heat Capacity (cal/gm-°C)	Thermal Conductivity (cal/cm-°C-sec)	Thermal Diffusivity (cm ² /sec)	U ²³⁵ Density (atoms/cm ³)
1	Cap	K-Monel	-0.018	-0.005	8.47	0.127	0.045	0.042	0
2	Solder	Tin-Lead-Bismuth	-0.005	0	8.8	0.060	0.10	0.189	0
3	Heater Coat	Nickel	0	0.005	8.90	0.112	0.22	0.22	0
4	Heater	Copper-Uranium Dioxide	0.005	0.030	9.00	0.086	0.08	0.103	6.2 x 10 ²¹
5	Pad Coat	Copper	0.030	0.035	8.96	0.092	0.94	1.14	0
6	Pad	Stainless Steel	0.035	0.074	8.04	0.12	0.033	0.034	0
7	Release Tube	Dilute Stainless Steel	0.074	5.164	8.02	0.12	0.004	0.0042	0

Fig. 4. Typical Analytical Temperature-Time Curves

Fig. 5. Safety Device Components, KEWB Model

Fig. 6. Installation of Safety Device in KEWB

The two assemblies used to observe fuse transient behavior were similar to the safety devices except that: 1) the intermediate chamber was simulated by a small storage chamber which contained no rupturable diaphragms, and 2) thermocouples were mounted in the solder layer and on the cap of the component.

The experimental arrangement for the tests is shown in Fig. 6. The fuse component was located within one inch of the center of the core where the neutron flux is greatest. The instrumentation cable lead to the recording apparatus in the control building, located 200 feet from the reactor.⁴

Data from the test were recorded on a 36-channel recording oscillograph as galvanometer deflections vs time. Part of a typical oscillograph record is shown in Fig. 7.

The pressure channel consisted of a standard strain gage bridge network, a balancing potentiometer, a battery, a Kintel d-c amplifier, and the galvanometer. The active arm and the temperature-compensating arm of the bridge were located on the storage chamber, while the other two arms and the balancing potentiometer were in a relay rack in the control room. The battery and the amplifier were mounted in the same relay rack; the galvanometer which was in the recording oscillograph was part of the regular KEWB instrumentation.

Each temperature channel consisted of a Chromel-Alumel thermocouple, reference junctions, copper extension wires, a Kintel d-c amplifier, and a galvanometer. The reference junctions, Chromel-copper and Alumel-copper, were at the pin connections at the end of the device, so the fuse temperatures recorded were relative to room temperature which remained constant during a run. The amplifier and galvanometer locations were the same as those for the pressure channel. The circuitry for the temperature and pressure channels is shown in Fig. 8.

VI. RESULTS

The recorded data were reduced using Benson-Lehner apparatus for reading and plotting. Reactor power and fuse temperature were plotted as functions of time with the calculated curves of temperature of an ideal fuse. From these

Fig. 7. Part of a Typical Oscillograph Record

Fig. 8. Fuel Test Circuitry

curves, peak power, total energy release, and peak fuse temperature were obtained. These quantities are listed in Table II.

The temperature vs time records for the less energetic transients were, in general, as expected. Although quantitative correlation was not obtained, qualitatively the agreement was good. The shape of the temperature curve and its location were accurately predicted analytically. A sample plot, showing temperature obtained experimentally with two fuse assemblies and the calculated temperature curve, is given in Fig. 9. The only apparent discrepancy is one of magnitude of temperature at a given time. In the example shown, the time after power peak required to reach 90 per cent of the maximum temperature was calculated to be 0.019 second and found to be 0.016 second and 0.015 second for the two experimental runs.

The major part of this difference is due to the fact that an ideal fuse was used for the calculations. A fairly small amount of excess solder in the fuse component can change appreciably the mass of that part of the fuse which operates at high temperature. Also, with these fuse test models, difficulty was encountered in producing a leak-tight fuse because of the thermocouples in the solder layer. Accordingly, solder was used quite liberally in these assemblies to effect the seal. The different temperatures achieved in the two fuses emphasize the fact that the exact fuse behavior is dependent upon assembly technique. Both had identical uranium loadings, caps, pads, and circuitry and yet there was about 20 per cent difference in transient temperature.

Although data obtained from the less energetic transients agreed well with calculations, those from the more energetic transients were not as expected. In the higher power transients a double hump appeared in the temperature curve, the first hump essentially in phase with reactor power, and the second one sometime after the power peak, with a pronounced minimum between. A run was made with a thermocouple located 2 inches from the fuse to see if the first peak was indeed thermal or if it was spurious. The hump appeared, indicating that the quantity being measured was not temperature due to the fission heating. However, it was believed that this spurious peak could be neglected since the remainder of the trace behaved in a normal manner. This effect was further investigated with thermocouples in air near the fuse and by using 0.010-inch iron-constantan couples in place of the

TABLE II
DATA SUMMATION

KEWB Run Number	Device Number	Reactor Period (sec)	Peak Power (Mw)	Energy Release (Mw-sec)	Peak Temperature (°C)	Calculated Peak Temp. (°C)
8F	4	17.6	0.0062	0.463	5.6	-
9F	4	8.52	0.0176	0.405	6.65	-
10F	4	4.48	0.0325	0.42	8.0	-
11F	4	2.62	0.0585	0.441	8.92	-
12F	4	1.27	0.124	0.383	10.75	-
13F	4	0.76	0.353	0.523	14.2	-
14F	4	0.236	0.773	0.548	17.25	-
19F	4	0.130	0.84	0.58	17.3	37
20F	4	0.031	8.5	0.86	30.0	51
21F	4	0.024	12.36	0.81	34.0	58
22F	4	0.012	38.	1.06	44.1	81
29F	4	0.0075	77.5	1.11	57.3	110
23F	4	0.0069	91.5	1.38	48.3	110
33F	4	0.0065	97.7	1.46	51.2	110
1F	3	18.2	0.0064	0.602	7.22	-
2F	3	9.1	0.0158	0.407	6.75	-
3F	3	3.84	0.042	0.431	9.25	-
4F	3	2.12	0.081	0.462	10.5	-
5F	3	0.96	0.245	0.629	13.6	-
6F	3	0.54	0.435	0.678	15.5	-
7F	3	0.199	1.156	0.881	18.9	-
24F	3	0.062	2.39	0.49	28.1	-
25F	3	0.033	7.27	0.67	38.3	51
26F	3	0.015	24.8	1.04	65.0	-
27F	3	0.0076	76.2	1.42	76.3	-
28F	3	0.0066	92.4	1.25	79.0	110

Fig. 9. Reactor Power and Fuse Temperature Curves for a Moderate Transient

0.003-inch Chromel-Alumel couples. Small humps were observed with the attached couples but they were much less pronounced. No humps were observed with the floating couples. However, oscillations were observed which were ascribed to convection currents set up in the air near the fuse.

A plot, typical of the more energetic transients, is shown in Fig. 10. For the similar transients shown, the two fuse test assemblies showed essentially the same behavior. Agreement with the calculation was poor since the theory predicted a single peak occurring shortly after the power peak.

The exact reasons for the discrepancies are not known as yet. However, there is evidence that there are two phenomena influencing the temperature reading in addition to the actual fuse temperature. The first is an effect on the thermocouples, not understood as yet, which gives rise to a spurious temperature indication essentially in phase with reactor power. Also, the thermocouples have a finite time constant which was manifested by the shift in the temperature peak to a later time and to a lower value.

Spurious thermocouple readings in high radiation fields have been observed at other laboratories, e.g., the Hanford Atomic Products Operation and the Special Power Excursion Reactor Test at the National Reactor Test Station. The problem was solved at these other sites by using other materials for thermocouples or by using thermocouples grounded at the metering junction. However, the cause of the false readings has not been determined.

The four safety devices were tested before the spurious nature of the first hump on the temperature trace was established. The temperature traces and the fact that both fuse test assemblies had lost pressure after the fastest transients indicated that the devices would trip. However, none of the devices tripped instantly on the most energetic transients available.

It was believed worthwhile to demonstrate a device triggering on neutron flux, if at all possible. Accordingly, an electrical heater was installed surrounding the fuse of one of the devices which raised the fuse temperature about 20° C above ambient. The transient was initiated and the device did fire on the temperature rise caused by the transient neutron flux. Trip time was approximately 400 milliseconds after the power peak.

Fig. 10. Reactor Power and Fuse Temperature Curves for a Fast Transient

VII. CONCLUSIONS

The thermal transient behavior of the fuse component of a reactor safety device was investigated in KEWB I reactor transients. Experimental results on the slower transients showed good agreement with calculations. However, on the fastest transients, the results were obscured by spurious effects during those intervals when radiation levels were high. This phenomenon has not been thoroughly investigated as yet, so these spurious effects remain unexplained.

It was not possible to demonstrate reactor shutdown because of the efficiency of the inherent shutdown mechanism of the KEWB reactor. However, a successful trial run was made to demonstrate fuse tripping on transient neutron flux by heating the fuse to about 20° C above ambient temperature at the beginning of the transient.

The KEWB test results indicate that the design of the fuse component of the safety device is adequate. With the satisfactory completion of these tests, the safety device is ready for more extensive, higher power tests in SPERT. The SPERT reactor offers higher power transients and a more representative environment for testing of this safety device, which was designed for heterogeneous, light-water-moderated reactors.

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