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HANFORD SAFETY DEVICE
SUMMARY REPORT

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.

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HANFORD SAFETY DEVICE
SUMMARY REPORT

BY

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ABSTRACT

The Hanford Safety Device Program was set up to study and test self-contained safety systems of high, inherent reliability, and to apply at least one of them to the Hanford reactors.

A complete laboratory and in-pile test program was conducted on an experimental model which was designated Mark II. An extension of this design, designated Mark IV, has essentially the same features but is better suited to standard production fabrication techniques.

Both designs feature a reactivity change which is effected by releasing an absorbent gas, boron trifluoride, from a small chamber into a larger chamber. When the flux exceeds a preset level, excessive heat generated in a U^{235} heater melts a fusible plug, and causes the release of the gas.

The program has been completed with the design and testing of these and other safety systems which offer back-up protection for the Hanford reactors.

I. INTRODUCTION

Early in 1952, the Atomic Energy Commission requested a study of "inherently reliable" reactor shutdown devices. After the presentation of two conceptual designs, active development of a reactor safety device was begun by Atomics International, a Division of North American Aviation Inc., in November of that year.

The program consisted of conceptual analysis and experimental work leading to one or more safety devices, the design, development and testing of the devices, and the application of at least one of them to the Hanford reactors.

Some general requirements of the safety device were that it should be:

1) Extremely reliable
2) Directly triggered by an increase in neutron flux above a prescribed level
3) Composed of a minimum of mechanical gadgetry and electrical circuitry
4) Bomb and earthquake proof
5) Sabotage proof
6) Fail-safe.

In addition, specific application to the Hanford reactors required that the device must:

1) Be applicable to both existing and future reactors
2) Be faster than existing Hanford safety rods
3) Provide absorption for 2.5 per cent Δk
4) Result in a minimum loss of production
5) Trigger on 30 to 50 per cent increase in power level over a prescribed level
6) Be insensitive to ± 10 per cent variation in power level.

II. INITIAL PROGRAM

A. DESIGN CONSIDERATIONS

The conceptual designs proposed before active development work began were based on the use of thimbles into which poison, in the form of boron carbide or boron trifluoride, could be introduced by gas pressure. Poison introduction was to be initiated by the rupture of a diaphragm of a pressurized vessel, within which the poison was stored in a frangible container. The rupturing mechanism would consist of a toggle-spring arrangement, having a very small spring constant near the "set" position. Two methods were proposed for triggering the toggle spring past its dead-center position. One of these was a bellows arrangement utilizing a fission-heated gas as a driving agent. The other consisted of a two-position, bimetallic element to be heated past the snap-over point by a coating of fissionable material. These devices were to be assembled as units and inserted into unused process tubes.

Considerable work had been done elsewhere, with good reproducibility, on the rupture of aluminum diaphragms in which rupture was initiated by pressure increase. However, the rupture of pressure-loaded diaphragms by plunger initiation proved to be unreliable. The use of this type of trigger in safety element designs was therefore discontinued (Ref. 1, pp 7-9).

A number of self-activated reactor safety devices had been reported in the literature, but a review of these revealed serious failings in most, such as long response times, lack of stability against creep, or vulnerability to radiation damage. In the initial work on this program, several types of trigger mechanisms with many modifications were considered (Ref. 2). Study indicated that metallic differential-expansion elements would be unsatisfactory because of the response times involved. The solder-joint type of trigger element seemed feasible where the ambient thermal flux level was greater than $5 \times 10^{11} \text{cm}^{-2} \text{sec}^{-1}$. Gas-expansion elements might be feasible for mechanisms with small energy requirements, such as for making electrical contacts. Consideration was given also to the use of momentum transfer of fission fragments, thermopiles using fission heat, radiation-damage elements, and ionization-sensitive devices. These ideas were set aside for a variety of reasons. The magnitude of the effect of momentum transfer requires too delicate a design. Thermopiles, in general, are not easily adaptable to fail-safe designs. Dependence upon radiation damage effects did not look promising because of long response time; and ionization-sensitive devices seemed to suffer from inconsistent breakdown point.

An investigation was started on a new means of introducing poison into the reactor. This was the injection, into the coolant passages, of a poison-carrying material that would coat out upon the coolant tube with coolant either present or absent.

B. FIRST EXPERIMENTAL SAFETY ELEMENT

1. Design

The first design (Ref. 3, pp 5-8) upon which the experimental program could be based was chosen utilizing the following concepts and mechanisms:

a) The use of fission heat to power the trigger device

b) The weakening or melting of a solder element as a trigger

c) The use of a pressurized gas as a driving agent

d) The use of a poison-bearing coating material

e) The use of the reactor process tubes for poison introduction

f) The use of a fuel-shielded poison to enable storage of the safety devices in the high flux regions of the reactors.

The device was to function in the following manner. The solder plug in the trigger would fail at some predetermined flux level, allowing the high-pressure gas to expand into the poison-storage chamber. Here the increased pressure would rupture the retaining diaphragm and force the poison material into the coolant passages where its neutron absorption would be effective in shutting down the reactor. Approximately 500 safety elements would be used in a reactor loading, giving increased reliability and minimizing the adverse effect of failure in a single unit.

Creep in the solder trigger and obtaining satisfactory coating materials were thought to be two major problems of this system. Accordingly, the experimental program was begun with the investigation of these two phenomena (Ref. 3, pp 9-12).

2. Solder-Plug Tests

Solder-plug tests consisted of lifetime determination as a function of temperature, load, plug diameter, plug length, and plug material. Except for preliminary tests, all experiments were made with 0.070-inch diameter plugs of 1/8-, 1/4-, 3/8-, and1/2-inch lengths. Preliminary results showed the lifetime to increase approximately exponentially with length and to decrease approximately exponentially with diameter (Ref. 1, pp 6-7).

The possibility of a safety device designed to be stored in high flux regions of reactors gave rise to interest in solders with higher melting points than the 183° C for tin-lead eutectic. Short-term shear-strength tests were made at elevated temperatures with pure tin, tin-silver eutectic, and tin-antimony eutectic solders. Shear strength was found to be approximately 25 per cent higher in tin-silver and tin-antimony solders at their operating temperatures (defined as two-thirds of the scale between coolant temperature and the melting point of the solder) than for the tin-lead solder at its operating temperature (Ref. 4, pp 7-8).

Selection of tin-silver eutectic (96.5 per cent tin - 3.5 per cent silver) solder to be used in the safety device was based on lifetime and shear tests. Tin-silver plugs, 0.070-inch diameter, and 0.5-inch long, pressurized to 1500 psi showed no creep after 1100 hours at 210° C, which is only 11 degrees below the melting point (Ref. 5, pp 7-8). Lifetime tests continued primarily with 0.25-inch long, 0.035-inch diameter tin-silver plugs for a period of 4225 hours without evidence of creep (Ref. 6, p 7).

3. Poison Coating Studies

A major experimental program based on the first prototype safety device design was the poison coating investigation. Experiments were performed to determine the feasibility of plating the inside of the aluminum process tubes with poison-metal coatings or with poison pigments in an organic carrier.

Static tests were begun with cadmium fluoroborate, a solution which dissolves aluminum oxide and plates cadmium on aluminum metal simultaneously. Although good coats were obtained, the process was too slow. However, a plating rate of 4 mg/cm^2 sec was obtained with cadmium chloride solution in dilute hydrofluoric acid. This would be sufficient to produce reactor shutdown within one second (Ref. 3, p 10).

Dynamic tests were initiated to investigate the process of plating cadmium from high-velocity water streams. It was found that for very high water velocity across aluminum surfaces, some 70 to 80 per cent of the plated metal washed away. However, most of the metal which still adhered at the end of the first second tended to remain during subsequent flow. Platings of 2.2 mg/cm^2 were obtained with deposition times of less than one second (Ref. 1 pp 9-10).

A more representative simulation for a reactor process tube was produced by running hot water through an aluminum tube to produce a heavy, hydrous oxide film. Presence of the film caused the plating efficiency to be less than desired for reactor control. To produce a satisfactory coat, film removal rates had to be increased or contact time lengthened. This problem, coupled with the increased success of the organic coatings (q. v.), caused discontinuance of this approach (Ref. 7, p 16).

The organic coatings were the vehicles for conveying poison from the storage chamber to the process tubes. Some of the poison "pigments" considered for this application were oxides of gadolinium or samarium; others were a mixture of lanthanides in the form of soluble organometallic compounds or finely pulverized oxides (Ref. 3, pp 10-11).

A wide variety of organic materials was used unsuccessfully in attempts to coat aluminum under water. These included wet-surface enamels, silicon resins, nylon, and conventional vehicles. While the materials would adhere to aluminum in still or slowly-moving water, they would not in a fast-moving stream of water (Ref. 1, p 10).

The first organic coatings which gave signs of promise were mixtures of four major constituents: 1) a resin, Beckacite; 2) a vehicle, Hercolyn; 3) a coagulant, ethyl cellulose; and 4) a solvent, a mixture of ethyl and isopropyl alcohols. Upon replacing the alcohols by phenol, coating characteristics improved and coatings were obtained which appeared satisfactory. The best coatings were obtained with the following approximate composition: Hercolyn, 45 per cent; phenol, 37.5 per cent; Beckacite, 15 per cent; and ethyl cellulose, 2.5 per cent (Ref. 4, pp 15-16).

The early experiments were conducted with the various components at different temperatures. It was believed that results could be improved if the coating, water, and surface to be coated, were all at the same temperature.

Experiments were performed in the range 40° to 90° C in a simulated process tube. The surface was type 3S aluminum with a film of hydrous aluminum oxide. The coatings were optimum at 62° C, reasonable at 60° and 65° C, and poor at 55° and 70° C. Although the composition could have been modified to shift the optimum temperature, the small useful range was difficult to apply to production reactors (Ref. 8, pp 41-42).

4. Process-Tube Explosion Study

A series of controlled explosions were performed in a Hanford-type process tube to determine the stresses caused by sudden poison injection or by chamber rupture. The tube was filled with water and the diaphragm of the pressure chamber was ruptured by applying gas pressure. Rupturing the diaphragm of a water-filled chamber duplicated normal poison injection, while rupturing the diaphragm of a gas-filled chamber simulated failure of the high-pressure gas chamber. Under these simulated operating conditions there was no measurable deformation of the tube, even with burst pressures of up to 2400 psi (Ref. 7, pp 23-25).

C. SAFETY DEVICE PERFORMANCE CALCULATIONS

It would be desirable for the poison stored in an active fuel channel to cause a minimum of interference with normal operation. Calculations were performed to determine the compositions of poison-bearing safety elements that would be equivalent to an ordinary fuel-bearing slug in its effect on pile reactivity. The two types of material considered were slightly-enriched uranium, an alloy of aluminum, and 93.4 per cent U^{235} enriched uranium (Ref. 3, p 12). Results showed that an internally-black, 1.36-inch OD, 0.25-inch wall fuel element would effectively replace a normal fuel element if it was made of 1.8 per cent U^{235} or 14 weight per cent fully-enriched uranium aluminum alloy. A similar element would replace a standard spiking slug if the composition of the annular fuel shell was 9.1 per cent U^{235}, or 27.7 per cent fully-enriched, uranium aluminum alloy (Ref. 1, pp 11-12, and Ref. 9).

The operation of safety circuits which activate the vertical and horizontal rods of a Hanford reactor was compared analytically with the performance of the proposed poison-injecting safety elements. The reactivity of the safety devices was assumed to be a linear function of time with -2.5 per cent reactivity addition in 0.5 second.

The calculations were performed with a number of different initial conditions. Steady-state operation at full power was assumed with a step-function addition of reactivity of two per cent, and with a linear addition of two per cent over a time interval of one second. A start-up accident arising from inadequate or omitted temporary poison after an extended outage was also considered. Total energy releases for the bursts were calculated to be about three times as great without the safety elements as with them (Ref. 7, pp 25-27).

It was concluded from the results of the study that if the safety devices were capable of introducing negative reactivity as assumed, they would provide a superior safeguard against reactor accidents initiated at or near full operating power. Although the devices were less flexible in protection against start-up accidents, they would constitute a reliable "last resort" in the event that all other safety systems were inoperative (Ref. 10).

Calculations were made to compare the relative advantages of various uranium-loading geometries for the sensing elements. Geometries considered were solder, film, and uniform loading. Response time and reactor overload were studied as a function of reactor period. The following conclusions were drawn from the work (Ref. 7, pp 7-11):

1) Uranium-loaded solder produces the fastest response.
2) For a given mechanical strength, a stainless-steel wall has a faster response to thermal transients than walls made of either aluminum or zirconium.
3) A steel wall must be less than 1/8-inch thick for a thermal relaxation time of less than one second.
4) Film loading is almost as fast as solder loading and does not suffer from possible radiation damage affecting the solder plug life.

D. CADMIUM CONDENSATE DEVICE

A different method of altering the neutron-absorbing area of a fuel element-size device was suggested and demonstrated. The device utilized the volatility of cadmium at a relatively low temperature. It consisted of a mixture of cadmium and a pyrophoric material in a combustion compartment in a relatively large, evacuated chamber. Compressed oxygen, upon being released by a suitable neutron-flux-sensitive trigger, ignited the pyrophoric material, and the heat of this combustion vaporized the cadmium. The resultant cadmium vapor then condensed on the walls of the chamber.

The pyrophoric materials considered were aluminum, magnesium, uranium, zirconium, and barium metals, all in a finely divided state. Preliminary experiments with aluminum and magnesium indicated that, at room temperature, spontaneous combustion would not be initiated by oxygen. Uranium powder, however, was used successfully to vaporize the cadmium (Ref. 4, pp 9-15).

III. PROTOTYPE BORON TRIFLUORIDE SAFETY ELEMENTS

A. DESIGNS

In January 1954, the decision was made to shift emphasis from the poison-coating approach to some system requiring a less extensive development program. However, the system was to allow for poison storage within the reactor core and this would result in some production sacrifice.

The designs considered were based on achieving a reactivity loss by a redistribution of poison from a small-area configuration to a large-area configuration. Poisons considered were cadmium metal redistributed by vaporization, cadmium amalgam, boron carbide or gadolinium oxide in powder form, and enriched boron trifluoride or helium-three.

An enriched-uranium-powered solder trigger element was used in some form in all the designs. The use of gaseous poison appeared to require the least amount of development, so this was given the highest priority. Of the two gases, boron trifluoride was chosen because of the unavailability of the rare isotope of helium.

Two boron trifluoride elements were designed - a small one having the external dimensions of a standard fuel slug and a long one which could extend entirely through the reactor. The small element had enriched boron trifluoride stored at one end at about 1200 psi in a cadmium-lined chamber. A release tube led from the chamber to the fuse element located a few inches away from the depressed-flux region around the storage chamber. A weighted "web" was used to increase the effective density to reduce "chattering" of the element in the turbulent water stream of a process tube. The heater tube, located at the end of the release tube, was a 0.035-inch ID tube made of about 20 weight per cent, fully-enriched uranium oxide in copper. The tube was sealed with a half-inch-long plug of tin-silver eutectic solder which had a melting point of 221° C. A stainless-steel pad served as a thermal resistance and provided a predictable temperature difference between the coolant stream and heater tube.

These components were sealed in either a stainless-steel or zirconium can which formed the pressure-tight chamber for the poison in its expanded state. The entire assembly was then clad with a standard fuel slug can.

The reactivity loss per element in the stored state and expanded state was calculated to be 3/4 and 2-1/4 inhours respectively. This meant that about 650 elements would be required to afford the desired safety control of 2-1/2 per cent Δk (Ref 4, pp 20-22).

The long boron trifluoride element was considered as an alternate or supplementary design. This system was composed of a high-pressure storage chamber containing boron trifluoride at 2000 psi, and a low-pressure receiver. An intermediate chamber, containing helium at 1000 psi, was located between these two chambers. The walls of the intermediate chamber contained diaphragms rupturable at 1500 psi. A release tube led from the intermediate chamber to an enriched-uranium-powered solder plug trigger element. Upon failure of the solder plug, the helium pressure was released, creating an excessive pressure differential across the first diaphragm. The diaphragms would burst sequentially, allowing the poison gas to expand into the receiver chamber to produce the required reactivity change (Ref. 4, p 23, and Ref. 5, pp 9-24).

B. LONG BORON TRIFLUORIDE ELEMENT (DOUBLE-DIAPHRAGM) TESTS

A series of tests were performed to measure the reaction times and to determine critical pressures and dimensions of the double-diaphragm system. A test rig was set up which consisted of the storage chamber with pressure lead, and an intermediate chamber with pressure lead and a connecting, sealed glass tube. Pressures in the chambers were 1000 psi and 500 psi respectively for the storage and intermediate chambers. Coined diaphragms were held in place by threaded fittings. The gas in the intermediate chamber was released by breaking the glass tube. Metallic silver paint on the glass tube was part of the electronic starting circuit (when the tube was broken, the timer started). The timer was stopped by a second signal, effected by the blast of gas breaking a thin aluminum foil stretched across the outlet orifice.

Orifices of various sizes were placed in the fitting which held the glass tube to simulate heater tubes of different diameters. Connecting tubes of different lengths were placed between the orifice and the intermediate chamber to simulate remote location of the trigger element. Delay time was found to be a decreasing function of orifice diameter and an increasing function of release tube length. The minimum delay time, extrapolated to a release tube of zero length and an orifice diameter of 0.070 inch, was found to be three milliseconds (Ref. 11, pp 16-19).

C. ASSEMBLY DETAILS

The design of the prototype safety element called for solder joints between the stainless-steel resistance pad and the copper-uranium dioxide heater tube, and between the pad and steel chamber walls. It was established that uranium dioxide was readily dissolved by the molten hard-solder fluxes customarily used. This meant that the hard-soldering between heater tube and pad would have to be done in a hydrogen furnace without flux to prevent leaching the uranium dioxide from the copper compact. The use of a plating material was indicated and after unsatisfactory results with copper and silver, a satisfactory bond was obtained with a nickel-plated compact using grade SS silver solder in a hydrogen furnace (Ref. 11, pp 11-13).

The stainless-steel resistance pad necessarily had to be brazed to the stainless-steel wall subsequent to filling the solder plug. Consequently, the brazing operation had to be performed in such a way that heat did not reach the solder plug. However, efforts using copper chill blocks, water cooling by submersion, and drip techniques were unsuccessful (Ref. 5, p 10).

As an alternate assembly method, standard soft-solder techniques were tried. Excellent results were obtained joining stainless steel to stainless steel with orthophosphoric acid liquid flux and 60-40 commercial grade lead-tin solder. The flux was warmed gently and allowed to react with the surface oxide until it turned green. The pieces were then brought together and the jointing accomplished (Ref. 5, pp 10-12).

In the spring of 1954, arrangements were made to have Sylvania Electric Products fabricate the uranium dioxide-copper heater elements. Using powder metallurgy techniques, fabrication consisted of compacting the tubes in quarter-length units which were sintered under optimum conditions of 1050° C in a wet hydrogen atmosphere for 16 hours. The tubes were coined on a tin-plated inner copper tube with tin washers between units and then heat treated to obtain a good bond (Ref. 8, p 7; Ref. 11, pp 8-10; and Ref. 12).

D. FUSE THERMAL CONDUCTANCE TESTS

Thermal conductance measurements were made on simulated safety element trigger assemblies to determine if their thermal characteristics could be consistently maintained. The variable features that were investigated were the heater tube-to-pad braze joint, the pad-to-chamber wall soft-solder joint and the wall-to-cladding shrink-fit joint. It was determined that the amount of interference fit was unimportant (Ref. 8, pp 14-16). Soldering techniques, however, did affect the thermal conductivity. Of utmost importance was the maintenance of good, homogeneous bonds that were free from voids or gas pockets. Although presence of a small fillet along the sides of the pad did not appear harmful, it was important to keep the fillet uniform in size from unit to unit (Ref. 6, p 8). It was concluded that consistent manufacturing methods could reduce the variation in thermal conductance to within two per cent of the average.

E. REACTIVITY TESTS

Although it is evident that safety elements utilizing a change in geometry of poison would cause a change in reactivity in a reactor, accurate calculations on the quantitative effect were necessary. Since data upon which such studies could be based did not exist, it was proposed to obtain these data by slug reactivity measurements in the Hanford 305 test pile.

The measurements were made to determine the effects of independently varying the following parameters of the absorbing material: macroscopic cross section (gas pressure), length, diameter, and location within the test slug. The simulated safety elements were contained in standard fuel element cans finished to the standard fuel slug length. The neutron-capture cross section was varied by filling the appropriate parts of the test elements with mixtures of sand and boron carbide. Unpoisoned regions were filled with aluminum plugs.

All samples were located in the center of a column of natural-uranium fuel elements in the 305 test pile. The difference in reactivity between test and normal loading was measured by the relative positions of the calibrated control rod at criticality. Perturbation of flux along the column was measured via gold foils attached to the elements and adjacent slugs (Ref. 5, pp 24-27, and Ref. 13, pp 27-38).

IV. MARK II AND MARK III SAFETY ELEMENTS

A. DESIGNS

In the spring of 1954, an evaluation study on the relative advantages of several safety element design changes was started. The reactivity data obtained from the Hanford test pile were used to determine the boron trifluoride charging pressure and high-pressure chamber size which would yield the optimum design.

The reactivity absorption of a triggered element would be almost entirely due to its boron content, while the stored-state reactivity loss varied with boron trifluoride pressure and low-pressure chamber material. The use of higher pressures lowered the stored-state absorption by allowing the poison to be confined to a smaller volume. The substitution of zirconium for stainless steel in the low-pressure chamber gave a marked decrease in the stored-state absorption. The analysis showed that several times as much boron was required to produce the maximum reactivity change as the maximum reactivity ratio. Since it was concluded that the minimum possible stored-state reactivity loss should be obtained, the design was set for the maximum reactivity ratio (Ref. 11, pp 25-28, and Ref. 13).

The design of an improved model of the boron trifluoride safety element, designated the Mark II (Fig. 1) was completed in the summer of 1954. The design features were selected to yield an element with maximum reactivity ratio, within the limitations imposed by the use of stainless steel, and reasonable gas pressures. The element was to absorb about 0.7 and 2.5 inhours in the stored and triggered states, respectively. This would yield a reactivity ratio of about 3.6 and would provide a reactivity change of about 1.8 inhours. For two per cent Δk reactivity control, 610 elements would be required for a reactor loading.

The uranium-powered fuse assembly and the stainless-steel can and cladding remained the same as in the prototype design. The storage chamber was changed to have minimum surface for a cylinder, that is, a right circular cylinder with height equal to diameter. Because of the excess wall thickness in this design, slots were milled into the wall to reduce parasitic absorption and to allow better poison distribution upon triggering. The web was to be made of aluminum with a lead core and was to serve the purposes of increasing the compressive strength of the element and of increasing the effective density to prevent chattering in the process channel (Ref 8, pp 20-23).

The design of a safety element affording a possibly better reactivity ratio was also completed. This design, designated Mark III, was essentially the same as the Mark II with zirconium substituted for the stainless-steel components, giving less stored-state absorption. The storage chamber consisted of a zirconium shell with an inner capsule of stainless steel to provide adequate strength (Ref. 6, pp 19-20).

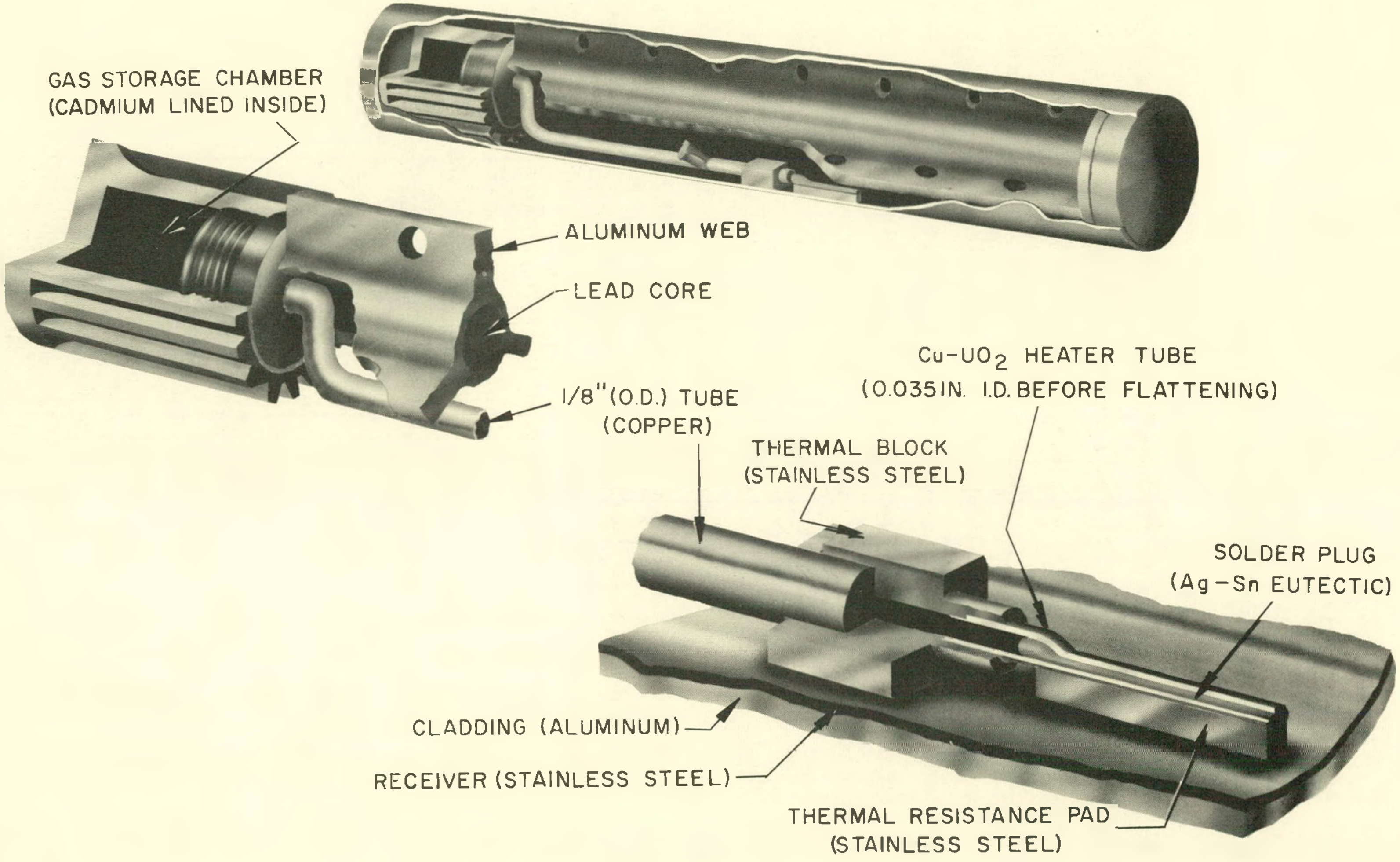

Fig. 1. Mark II Safety Element

B. MEASUREMENT OF THE RELATIVE EFFECTIVENESS OF BORON TRIFLUORIDE

The relative effectiveness of boron trifluoride was measured experimentally in the WBNS (Water Boiler Neutron Source, a low power, homogeneous, solution-type research reactor at Atomics International) by the danger coefficient method. The measurements were designed to determine the relative effectiveness of boron trifluoride and the boron carbide-sand mixtures that were run in the Hanford 305 test pile. Abnormalities in early measurements, caused by condensation of boron trifluoride, resulted in a somewhat extensive search for impurities (Ref. 8, pp 23-26 and Ref. 11, pp 20-22). After resolving the difficulties, it was determined that the correction factor, defined as the ratio of boron-10 atoms in boron carbide to that in boron trifluoride for the same reactivity effect, was 2.4 (Ref. 8, pp 36-39).

Additional reactivity measurements were made in the Hanford 305 test pile utilizing actual Mark II and Mark III safety elements. The control ratio was determined by measuring the reactivity loss of each element in the stored and triggered states. They were placed individually in the test pile for the stored measurement, removed, triggered by heating in an oven, replaced in the pile, and the reactivity loss again determined.

The control ratios measured were somewhat lower than expected due to excessive stored-state loss. The ratios were 3.02 and 3.01 for the Mark II samples and 3.84 for the Mark III sample (Ref. 14, pp 7-8).

C. BORON TRIFLUORIDE GENERATION AND LOADING

The use of enriched boron trifluoride as the reactor poison in the Mark II design necessitated assembly of a gas generation and loading system for filling the devices. The gas, generated by heating CaF_2-$B^{10}F_3$ powder to about 350° C, was cycled through a dry ice-acetone trap to remove high-boiling, condensable vapors. The gas was further purified to remove the more permanent gases such as oxygen by repeated sublimation of boron trifluoride from one liquid-nitrogen trap to another while the less-readily-condensable gases were drawn off by a vacuum pump (Ref. 11, p 20).

Boron trifluoride was stored by condensing the gas in a stainless-steel pressure vessel, using liquid nitrogen. For loading, the gas was admitted from the storage chamber to the evacuated loading system, which consisted to a system of valves, gages, and connecting pipes. When the proper temperature and pressure conditions were reached, the fuse assembly was sealed, under pressure, by a crimp-braze technique (Ref. 6, pp 25-28 and Ref. 15, pp 28, 32-34).

The system has been used successfully to load many test assemblies and has been the base design for an enlarged system for loading production-lot quantities of safety devices (Ref. 16).

D. GAS-EXPANSION TESTS

Tests were conducted to determine the rate of expansion of boron trifluoride from the high-pressure chamber into the low-pressure chamber. Experimental data were recorded on an oscillograph, using thermocouples and strain gages for temperature and pressure, respectively, in each chamber.

An analytical expression derived for the flow of a perfect gas through an orifice, and checked experimentally with helium, also showed good correlation with boron trifluoride when empirical pressure-volume relationships were introduced into the equation.

Calculations based on the experimental data were made to determine the rate of reactivity change of the safety element due to the expansion of the gas from the storage chamber. These showed that the element would be 90 per cent effective within 150 milliseconds after the solder plug had failed (Ref 15, p 11, and Ref. 17).

E. IN-PILE TESTS

1. Boron Trifluoride Irradiation

Dry boron trifluoride is not seriously corrosive to any of the materials it would contact in the safety device. However, it was necessary to demonstrate that irradiation during a normal cycle would not increase the corrosion rate. For this purpose, an experiment was designed to irradiate materials of the safety device in a boron trifluoride atmosphere (Ref. 18). These were aluminum, copper, cadmium, tin-silver eutectic solder, Type 347 stainless steel, zirconium and grade SS silver solder (Ref. 8, p 40).

The reaction $B^{10} (n, \alpha) Li^7$, as measured by helium content, occurred at about the rate predicted and radiation damage, which produces boron and fluorine, appeared to proceed at approximately the same rate (Ref. 15, p 32 and Ref. 19 pp 25-26). After completion of the pressure measurements, the capsules were washed several times with hot water. Conclusions based on the analysis of the washings were that the lithium from the (n, α) reaction combined completely with the free fluorine from radiation decomposition. The remaining free fluorine apparently combined principally with zinc and iron.

Calculations for the corrosion in a safety element in a Hanford reactor for a full cycle were made. They indicated that there would be negligible corrosion insofar as it might influence the operation or integrity of the device (Ref. 20, pp 7-9).

2. Solder Plug Irradiation

An experiment was designed to determine the possible adverse effect of irradiation upon the solder plugs (Ref. 5, pp 14-15). Four trigger assemblies in a common mounting were to be pressurized to 2000 psi and held at 210° C during irradiation. Enriched-uranium loading was set such that electrical heating had to be supplied to override fluctuations in fission heat and provide temperature control (Ref. 6, pp 10-13).

A wiring error was responsible for the premature melting of the fuses early in the in-pile test preparations, so that essentially no data were obtained (Ref. 20, p 7). The test elements were not rebuilt because it was believed that the worth of the experiment would not justify the magnitude of the effort.

3. Fuse Assembly Response Time

The response time of a reactor safety element is one of its most important features. The best way to obtain this information is from conditions closely approximating actual operation in a neutron flux excursion. An experiment was designed utilizing excursion simulation by transportation of the trigger assemblies through the exponentially-decreasing flux region of the MTR reflector (Ref. 7, p 12).

Assuming a truly exponential flux region in the MTR, simulation of a reactor excursion would require driving the test element at a constant speed toward the core. A driving mechanism was designed to approximate the required conditions, providing insertion distances of 1-1/2 to 15 inches in 10, 5, 1, and 1/4 seconds, and of 1-1/2 to 7 inches in 0.1 second (Ref. 5, pp 12-14).

The test capsule contained four fuse elements, all of which were instrumented with thermocouples. In addition, one fuse element was connected to a storage chamber containing helium under high pressure. Melting of this latter fuse would demonstrate trigger action when the melting point was reached (Ref. 6, pp 8-10).

The response time for the trigger is the difference in time between trip level of reactor power and trip level of fuse temperature. It was found, as expected, that this time delay was an increasing function of reactor period. It was also dependent upon the thermal relaxation time of the trigger element, which was found to be 2.5 seconds. This was a slightly longer time than predicted analytically (Ref. 15, p 7). However, the qualitative behavior was as expected and the feature of inherent operation on excess neutron flux was effectively demonstrated (Ref. 21).

F. SUPPLEMENTARY TRIGGER ELEMENT STUDIES

1. Thin-Slab Approximation (Ref. 22)

It was found desirable to determine the safety element time response by laboratory methods to supplement the inpile measurements. A direct means of experimental simulation of the thermal transients in a trigger assembly was developed. The system, called the "thin-slab approximation," consisted of a thin cross section of a trigger assembly, enlarged by a factor of ten to yield experimentally-observable temperature changes. The thin slab was heated by allowing the radiant energy from an incandescent lamp to fall upon the blackened heater-tube portion of the slab. Small thermocouples were fastened to the other side to obtain temperature-transient data (Ref. 6, p 17 and Ref. 15, pp 13-17).

Preliminary measurements to establish techniques were made at atmospheric pressure utilizing a Brown potentiometer as a recorder. Collimation of the beam was obtained by a condensing lens and a system of baffles. The transient was initiated by opening or closing a shutter interposed in the light beam (Ref. 19, pp 11, 14-19).

The accuracy and magnitude of the preliminary results were seriously affected by convection currents set up as the slab was heated. For this reason, a vacuum test chamber was constructed to minimize convection losses. It was found that the thermal relaxation time was indeed a function of the pressure (Ref. 20, pp 9-13).

In addition to convection effects, it was observed that the time constant was dependent upon the temperature of the test section, indicating perhaps the influence of radiation losses. This problem was approached by two methods: by making a thicker slab, to reduce the surface-to-volume ratio and hence decrease the percentage of heat lost by radiation, and by using transients with smaller temperature changes.

A different technique was used for measuring the temperature transients, as the recording system was inaccurate for small changes. Basically, the new system was to record the time when the galvanometer of a laboratory potentiometer passed through the zero point, indicating a thermocouple voltage equal to the potentiometer setting. The process was repeated throughout the transient, yielding thermocouple voltages at specified times. Using very small temperature swings and the thicker slab, the equivalent thermal relaxation time of the Mark II fuse was found to be 1.12 seconds (Ref. 14, pp 9-13).

It was demonstrated that the convection and radiation losses were essentially eliminated, by experimental runs with the thicker element, in the range of 1 to 18 microns and temperature changes from 1° to 50° C (Ref. 23, pp 5-6).

2. Electrical Analogs (Ref. 24)

A two-dimensional electrical analog was constructed to supplement the analytical calculations and thermal measurements of the Mark II fuse assembly. It was felt that the electrical analog method could act as a bridge between a few physical measurements and theoretical calculations and could be used to extend both. The analog consisted of a two-dimensional array of resistors and capacitors, the values of which were determined from the physical properties of the various materials. In the system, voltage and current were analogous to temperature and heat flow and electrical resistance and capacitance simulated thermal resistance and capacitance (Ref. 15, pp 11, 13).

The measurement technique, for determination of the time delay between heater-tube heat generation and fuse temperature, consisted of monitoring the input current to the analog and the voltage in the fuse region of the analog. These traces were presented simultaneously on an oscilloscope screen, photographed, and the time between the two traces measured (Ref. 19, pp 7-11).

The thermal relaxation time was measured in a similar fashion. The input current to the analog was changed instantaneously and the curve of voltage in the analog versus time obtained. The curves were very nearly exponential and the determination of their slope gave the thermal relaxation time. This was 1.27 seconds for the Mark II fuse assembly analog, which agreed with theoretical calculations but not with the inpile experiments (Ref. 14, pp 13, 15-16 and Ref. 20, pp 13-17).

A test was run to determine the thermal conductivity of the copper-uranium dioxide matrix of the experimental heater tubes. This was done to determine the validity of the value used in calculations of the electrical analogs. A relative measurement, comparing the temperature difference across a copper tube and across a copper-uranium dioxide matrix tube carrying the same heat flow, was taken. The particular compact contained very little uranium dioxide; hence, the measurement was primarily a measure of the decreased density caused by powder metallurgy techniques (Ref. 14, pp 14, 17-20).

3. Simulated Fuse Assembly Response Tests

A further, more direct investigation into the Mark II fuse thermal relaxation time was then performed. This consisted of thermal measurements made with a simulated fuse assembly identical with that used for the in-pile measurements with the following exceptions:

a) The copper-uranium dioxide tube was simulated by a brass tube

b) There was no aluminum cladding.

Neither of these differences should have caused appreciable error.

The assembly was first warmed to an equilibrium of about 70° C. The transient was then initiated by directing a stream of ice water onto the chamber wall, in the fuse vicinity. Data of temperature versus time, via thermocouples, were obtained on a recording potentiometer. The thermal relaxation time was found to be 1.47 ± 0.15 seconds, a value between the in-pile determination of 2.5 seconds and the analytical and analog values of 1.25 to 1.29 seconds (Ref 14, pp 20-22).

Electrical analogs of the three thin-slab experimental sections (q. v.) were made for correlation purposes. Thermal relaxation times for the three different geometries were 1.04, 1.06, and 1.07 seconds, which agreed reasonably well with the thin-slab thermal measurement. Time delay as a function of reactor period was also determined (Ref. 14, pp 22-25).

G. ENGINEERING TESTS

1. Mark II Tests

After completion of the Mark II design, a number of devices were fabricated for test and demonstration purposes. Most of the samples were for engineering tests to determine the applicability of various design features. The tests were for impact, vibration, pressure and axial compression (Ref. 6, pp 17-18).

The storage system pressure-test assemblies consisted of only those parts which were to contain high pressure. These were the cadmium-plated storage chamber and the fuse assembly. The latter consisted of release tube, fill tubes, stainless-steel block, copper (instead of copper-uranium dioxide) heater tube, and tin-silver solder plug. The samples were pressurized with oil as the hydraulic fluid. The results indicated that the system was good to about 15,000 psi, well above the anticipated storage pressures, and that the weakest component was the release tube. In the event of failure poison would be released into the receiver, indicating a fail-safe design (Ref. 15, pp 19-25 and Ref. 19, pp 19-22).

The receiver pressure assemblies consisted of the receiver, a dummy storage chamber, and a pressurizing tube. They were held at 660° C, the melting point of aluminum, and pressurized to failure. The four samples failed in the region of 1200 to 1400 psi (approximately four times the expected loading of 300 to 400 psi) Ref. 15, pp 25-27).

The compression test samples were complete except for the fuse assembly. The operation of removing fuel slugs from a Hanford reactor process tube may require the application of forces as high as 10,000 pounds; consequently, the safety device must be able to withstand this load. Hanford specifications are for maximum expansion on the diameter of 0.040 inch at 10,000 pounds load. The three safety device samples did not reach this distortion until the load was 16,000 pounds (Ref. 15, pp 27-30).

Fabrication, storage, transport, and loading operations of the reactor safety device involve a considerable amount of handling. It was, therefore, necessary to demonstrate that there were no personnel hazards inherent in the design. A test was designed to investigate the integrity of the device when given a severe impact.

Six devices, oriented to strike on a corner, were dropped approximately 95 feet onto a concrete block. All elements withstood the drop, including one which failed internally, in the sense that the boron trifluoride was contained within the elements (Ref. 14, pp 25-27).

It was desirable to test the safety elements under conditions at least as severe as those in a process channel for a time comparable to the fuel cycle of six months. The conditions that could lead to failure in the device were 1) vibration due to water flow; 2) water hammer, 3) elevated ambient temperature, and 4) thermal cycling. The effects to be studied were 1) creep on the solder plug, 2) general corrosion of components, 3) bond stability, and 4) integrity of the high pressure system. Temperatures at which the devices were held during vibration were: 65° C, effective ambient temperature in reactor; 160° C, effective fuse operating temperature; and 210° C, a practical maximum temperature at which the fuse could be tested.

The devices held at 65° C and remained intact for 700 hours; however, the others failed at some point in the storage system. It was believed that excessive pressures caused by the elevated temperatures were the cause of the failures. Inspection of the devices showed one fuse leak, one block-and-fuse-junction leak, one copper-tube rupture, and one crimp-seal leak; two leaks were undefined. Corrosion was slight or nonexistent, all but one of the fuses were satisfactory, and all thermal bonds were good (Ref. 25, pp 10-12).

During a second run of rebuilt assemblies, more leaks developed. These were traced to junctions in the copper release tube where they had been reworked. It was decided that extensive reworking should be avoided, so the assemblies were again rebuilt utilizing new parts wherever possible (Ref. 23, p 2).

The tests were resumed under conditions designed to be more severe than expected in a process tube, but the test apparatus failed to withstand these test conditions. If there had been more time on the project, it might have been advisable to analyse the process tube conditions and perform a more conservative test. In spite of the equivocal results, it was believed that the design was not necessarily unsatisfactory, but would need more testing to be proved (Ref. 26).

2. Mark III Tests

The engineering test schedule for the Mark III design was, in general, the same as for the Mark II. Some tests could be eliminated or simplified because of applicable information from the Mark II tests, but differences in the mechanical properties of zirconium from stainless steel necessitated further testing. The engineering test schedule was for impact, vibration, pressure, and axial-compression.

Three Mark III storage chambers were successfully pressurized hydraulically with oil to about 25,000 psi, indicating a satisfactory design (Ref. 19, pp 22-23).

Three Mark III impact test assemblies were dropped a distance of 95 feet onto a concrete block. Although all the assemblies withstood the first drop, when they were dropped again, two failed internally. However, the released gas was contained within the receiver. Both failures were in the copper release tube, which had been weakened by reworking. Since the release tube had been demonstrated to be satisfactory in the Mark II tests, it was assumed that the Mark III design was also mechanically adequate in this respect (Ref. 27, pp 5-8).

The receiver pressure-test assemblies consisted of the receiver, dummy storage chamber, and a pressurizing tube. They were held at 660° C, the melting temperature of the aluminum cladding, and pressurized to failure. These chambers were unsatisfactory; they failed in the range of 260 to 420 psi, which is approximately the same as the expected loading (Ref. 20, pp 27-29).

Two more test assemblies with twice the wall thickness were also tested. Their failure points were 500 and 590 psi, which is less than twice the expected loading. This test confirmed the earlier conclusions that zirconium was not applicable for the safety device (Ref. 14, pp 28-30).

Interest in this design waned and the remaining test program was curtailed.

H. PRODUCTION PILE TEST

In the spring of 1954, a series of production pile tests was planned for the boron trifluoride safety element. The primary objective of these tests was to determine the enriched-uranium loading for the heater tube to provide the required temperature drop across the resistance pad. Other objectives were: to observe the behavior of the boron trifluoride under irradiation with regard to pressure build-up, and to determine the effect, if any, on reactor operation of the triggering of a single element (Ref. 5, p 27).

The methods of measuring the desired quantities were as follows: 1) The temperature drop across the pad for various loadings was to be determined via thermocouples located in the fuse assembly which were to be monitored by a recording potentiometer. 2) The effect of radiation on the boron trifluoride was to be obtained from pressure measurement by a transducer located outside the reactor and coupled to the system by a capillary tube. 3) The effect on reactor operation caused by triggering a single device was to be measured by existing reactor instrumentation (Ref. 14, pp 7-8).

The fission heater in the first assembly to be placed in-pile generated more heat than was expected. This made it necessary to conclude the experiment before the reactor reached full power. After establishing the fuse temperature for a given reactor power, the fuse was triggered by applying power to the heater mounted on the fuse tube. The poisoning effect on reactor operation caused by the fuse triggering was not detected.

The experiment indicated that the succeeding test assemblies would have to be modified because: 1) the heat generation was higher than expected, 2) the enriched uranium loading in the heater tube was 50 per cent higher, and 3) flux levels in the reactor were expected to be higher (Ref. 27, pp 14-17).

Appropriate modifications were made in the fuse assembly to reduce the operating temperature. The modified assembly was installed in the reactor and the power level increased. Early in the experiment the device triggered prematurely.

The fuse and pad thermocouples indicated correct design of the fuse assembly for, even after poison release, the fuse temperature was nearly at the correct level. However, excessive pressure was generated in the high-pressure system, caused presumably by higher temperatures than anticipated in the gas-storage chamber. Possible sources of heat generation in the region of the stored gas were local gamma heating and the (n, α) reaction on the B^{10} in the gas itself. A careful analysis showed that the former would have to be 2-1/2 times as large as predicted (0.65 watt/g, rather than 0.25 watt/g) in order to account for the indicated temperature distribution (Ref. 26). However, it was known from other experimental evidence that the neutron flux penetrating the cadmium shield around the storage chamber could produce the indicated pressure if certain assumptions were made as to temperature distribution within the stored gas (Ref. 28).

The experiment would have been continued with increased heat drain provided in the assemblies, but termination of the program precluded this effort.

V. MARK IV DOUBLE-LENGTH SAFETY ELEMENT

A number of Mark II and Mark III safety elements had been individually fabricated for engineering and performance tests. It was felt that the complicated and expensive techniques needed to make these laboratory-type devices warranted a study of design changes which could yield a device more amenable to production fabrication. Accordingly, in the spring of 1956, work was begun on such a device incorporating certain improvements which could be applied to the Mark II if desired.

A double-length element, with a single storage chamber containing twice the poison of a Mark II element, was considered because it would have the advantage of reducing considerably the surface-to-volume ratio of the stored poison. Hence, it should produce less stored-state absorption than two Mark II elements. The principal features of the double-length element, designated the Mark IV (Fig. 2), were as follows (Ref. 25, pp 7-10):

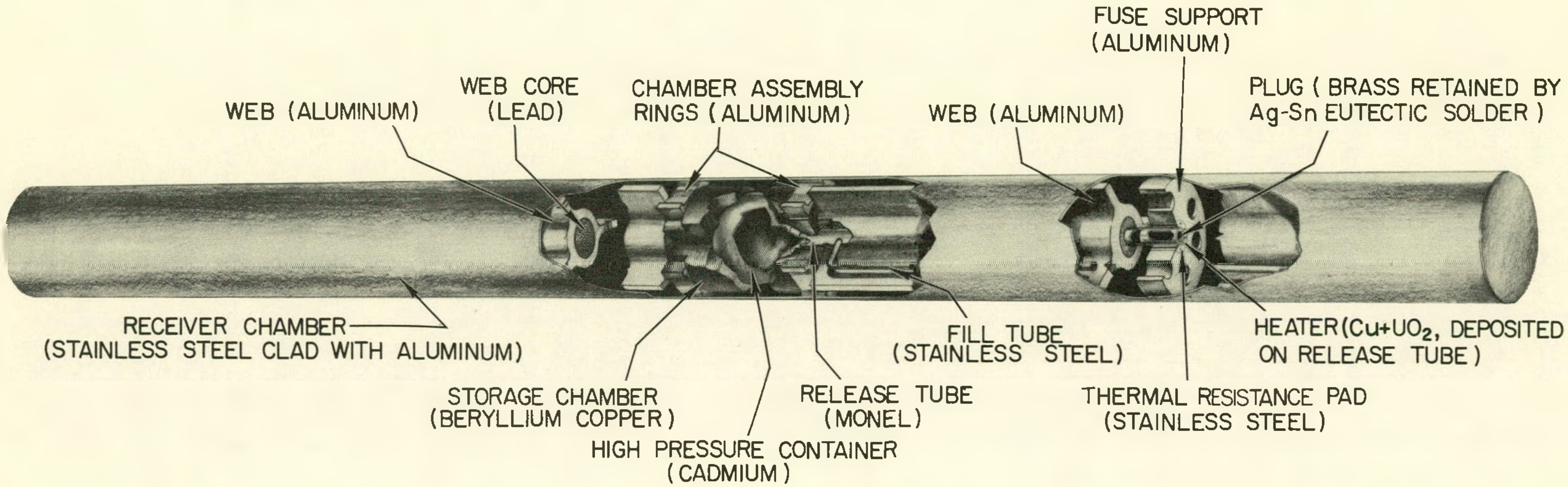

Fig. 2. Mark IV Safety Element

1. The high-pressure container was to be a cadmium liner within a beryllium-copper chamber; the chamber would provide only mechanical support.
2. The fuse was to be integral with a straight release tube located on the axis of the element.
3. The block (as on a Mark II) was eliminated because the use of a Monel release tube made axial conduction almost negligible.
4. The pad was to be annular, providing simplified assembly and eliminating fuse hot spots.

Tests were conducted to determine the applicability of a draw-cladding technique to clad the element. Effective cladding with a good thermal bond was accomplished when the proper control of dimensions was maintained. Measurements indicated that interference fits of about 0.0005 inch were obtained (Ref. 27, p 13).

Solder plugs for the Mark IV were fabricated and tested. The plugs contained a hollow, metal insert to reduce the heat capacity of the plug and facilitate complete blow-out. The plugs were pressurized for short durations at 5000 psi and leak checked at 1500 psi (Ref. 27, p 14).

Several engineering tests, such as described under Mark II and Mark III, would be required in order to prove the design adequacy of the Mark IV; these were not performed before curtailment of the program. Many of the improvements in fabrication methods, however, could be adapted to the Mark II design with little or no further testing required.

VI. EXTERNAL POISON SAFETY SYSTEM

The proposed trigger mechanism for an external-poison safety system consisted of: 1) a pressurized tube plugged with solder at its lower end; 2) a convection cooling loop filled with NaK: 3) tubes to bring cooling water in and out; 4) a protective sheath; and 5) blocks connecting the tubes to the protective sheath. In the reactor, the protective sheath was to reach equilibrium at the graphite moderator temperature. Heat from the protective sheath was to be conducted through the heater block into one side of the convection loop, and cause thermal expansion of the NaK. The resulting density unbalance was to create circulation within the loop, providing cooling for the fuse (Ref 25, pp 12-14).

This trigger would be used with a double-diaphragm system (described in Section IV-B) which would propel a neutron-absorbing powder into the existing vertical safety-rod channels. Compared with other existing externally-stored safety devices, the system could be faster, simpler, and more nearly fail-safe. The concept was not followed beyond the study stage (Ref. 27, pp 17-25).

VII. CONCLUSIONS

The goals of the Hanford Safety Device Program consisted of conceptual analysis and experimental work leading to one or more safety systems, their design, development and testing, and the application of at least one of them to the Hanford reactors. These requirements have been satisfied by the accomplishments of the program. The Mark II safety element has undergone a complete laboratory and in-pile test program. It is felt that this design with certain Mark IV modifications, or the Mark IV design, can be used successfully in an ultimate or back-up safety system for the Hanford reactors.

The external poison storage concept, although not proved by engineering nor by in-pile tests, appears feasible after detailed study, and has the advantage of causing much less stored-state reactivity loss in the reactor.

In addition, a number of design concepts which originated and were investigated in the Hanford program are currently under development on other safety projects. These include the double-diaphragm system, which utilizes boron trifluoride with a uranium-powered solder plug as a trigger; and the cadmium condensate system which utilizes the heat of combustion of shredded aluminum to vaporize the metallic poison.

APPENDIX A
REACTOR SAFETY TOPICAL REPORTS

Number	Title	Author
NAA-SR-272	"Composition of a Fuel Rod with a Co-Axial Void Containing Poison", April 1, 1954 (Classified)	D. L. Hetrick
NAA-SR-285	"Kinetic Behavior of a Hanford Reactor and the Performance of Safety System," April, 1954 (Classified)	D. L. Hetrick
NAA-SR-1483	"The Experimental Determination of the Rate of Gas Expansion in a Mark II Safety Element", March 15, 1956 (Unclassified)	D. L. Condotta
NAA-SR-1501	"Safety Element Design Studies", July 1, 1956 (Classified).	C. W. Wheelock
NAA-SR-1782	"A High Pressure $B^{10}F_3$ Loading System for a Reactor Safety Element", February 1, 1957.	C. R. F. Smith
NAA-SR-1761	"The Transient Behavior of a Prototype Reactor Fuse in Simulated Reactor Excursions", March 15, 1957.	N. E. Huston & R. R. Eggleston
NAA-SR-1999	"Experimental Determination of the Radiation Decomposition of Boron-10 Enriched Boron Trifluoride and the Resultant Corrosion of the Containing Materials", Oct. 9, 1957.	W. K. McCarty
NAA-SR-2003	"Electrical Analog Applications to a Reactor Safety Device Program", Nov. 15, 1957.	C. C. Weeks
NAA-SR-2461	"Thin Slab Approximation of a Two-Dimensional Heat Flow Problem", to be published	T. H. Springer
NAA-SR-2462	"Production Pile Test of a Reactor Safety Device", to be published.	N. E. Huston & W. K. McCarty
SEP-210	"Fabrication of Heater Tube Fuses for the Hanford Safety Element", April 12, 1956, (Classified)	H. S. Kalish & S. I. Megeff

APPENDIX B
REACTOR SAFETY PROGRESS REPORTS

Number	Class	Period	Author
NAA-SR-225	C	Nov. '52 - Jan. '53	N. E. Huston
NAA-SR-250	C	Feb. - Apr. 53	N. E. Huston
NAA-SR-276	C	May - Jul. 53	N. E. Huston
NAA-SR-846	C	Aug. - Oct. 53	N. E. Huston
NAA-SR-982	C	Nov. '53 - Jan. 54	N. E. Huston
NAA-SR-1062	U	Feb. - Apr. 54	N. E. Huston
NAA-SR-1121	C	May - July 54	N. E. Huston
NAA-SR-1190	C	Aug. - Oct. 54	N. E. Huston
NAA-SR-1309	C	Nov. '54 - Jan. 55	N. E. Huston
NAA-SR-1448	C	Feb. - Apr. 55	S. H. Fitch & N. E. Huston
NAA-SR-1560	C	May - July 55	S. H. Fitch & N. E. Huston
NAA-SR-1568	U	Aug. - Oct. 55	S. H. Fitch & N. E. Huston
NAA-SR-1634	U	Nov. '55 - Jan. 56	S. H. Fitch & N. E. Huston
NAA-SR-1726	U	Feb. - Apr. 56	S. H. Fitch & N. E. Huston
NAA-SR-1810	U	May - Jul. 56	N. C. Miller
NAA-SR-1954	U	Aug. - Dec. 56	N. C. Miller
NAA-SR-2157	U	Jan. - Mar. 57	N. C. Miller
NAA-SR-2224	U	Apr. - June 57	N. C. Miller

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2. N. E. Huston, "Reactor Safety Quarterly Progress Report, November 1952 - January 1953," NAA-SR-225, April 14, 1953 (Classified).
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4. N. E. Huston, "Reactor Safety Quarterly Progress Report, November 1953 - January 1954," NAA-SR-982, July 1, 1954 (Classified).
5. N. E. Huston, "Reactor Safety Quarterly Progress Report, February-April 1954," NAA-SR-1062, October 1, 1954.
6. N. E. Huston, "Reactor Safety Quarterly Progress Report, November 1954 - January 1955," NAA-SR-1309, July 15, 1955 (Classified).
7. N. E. Huston, "Reactor Safety Quarterly Progress Report, August-October 1953," NAA-SR-846, April 1, 1954 (Classified).
8. N. E. Huston, "Reactor Safety Quarterly Progress Report, August-October 1954," NAA-SR-1190, April 15, 1955 (Classified).
9. D. Hetrick, "Composition of a Fuel Rod with a Co-Axial Void Containing Poison," NAA-SR-272, April 1, 1954 (Classified).
10. D. Hetrick, "Kinetic Behavior of a Hanford Reactor and the Performance of Safety System," NAA-SR-285, March 1, 1954 (Classified).
11. N. E. Huston, "Reactor Safety Quarterly Progress Report, May-July 1954," NAA-SR-1121, January 15, 1955 (Classified).
12. H. S. Kalish and S. I. Megeff, "Fabrication of Heater Tube Fuses for the Hanford Safety Element," SEP-210, April 12, 1956 (Classified).
13. C. W. Wheelock, "Safety Element Design Studies," NAA-SR-1501, March 3, 1955 (Classified).
14. S. H. Fitch and N. E. Huston, "Reactor Safety Quarterly Progress Report November 1955 - January 1956," NAA-SR-1634, July 15, 1956.
15. S. H. Fitch and N. E. Huston, "Reactor Safety Quarterly Progress Report February-April 1955," NAA-SR-1448, October 1, 1955 (Classified).
16. C. R. F. Smith, "A High Pressure $B^{10}F_3$ Loading System for a Reactor Safety Element," NAA-SR-1782, February 1, 1957.

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17. D. L. Condotta, "The Experimental Determination of the Rate of Gas Expansion in a Mark II Safety Element," NAA-SR-1483, March 15, 1956 (Classified).
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19. S. H. Fitch and N. E. Huston, "Reactor Safety Quarterly Progress Report May-July 1955," NAA-SR-1560, April 1, 1956 (Classified).
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21. N. E. Huston and R. R. Eggleston, "The Transient Behavior of a Prototype Reactor Fuse in Simulated Excursions," NAA-SR-1761, March 15, 1957.
22. T. H. Springer, "Thin Slab Approximation of a Two-Dimensional Heat Flow Problem," NAA-SR-2461 (to be published).
23. N. C. Miller, "Reactor Safety Quarterly Progress Report August-December 1956," NAA-SR-1954, August 15, 1957.
24. C. C. Weeks, "Electrical Analog Application to a Reactor Safety Device Program," NAA-SR-2003, November 15, 1957.
25. S. H. Fitch and N. E. Huston, "Reactor Safety Quarterly Progress Report February-April 1956," NAA-SR-1726, January 15, 1957.
26. N. C. Miller, "Reactor Safety Quarterly Progress Report April-June 1957," NAA-SR-2224, March 15, 1958.
27. N. C. Miller, "Reactor Safety Quarterly Progress Report May-July 1956," NAA-SR-1810, March 15, 1957.
28. N. E. Huston and W. K. McCarty "Production Pile Test of a Reactor Safety Device," NAA-SR-2462 (to be published).