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THE HEAVY WATER COMPONENTS TEST REACTOR: SAFETY SYSTEMS, FUEL FAILURE DETECTION, AND STANDBY CONDITION

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May 1966

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AIKEN, SOUTH CAROLINA

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ABSTRACT

The three papers in this report give details of the following aspects of the Heavy Water Components Test Reactor (HWCTR):

1. Concept and performance of the safety systems that were utilized in the reactor.

2. Fuel failure detection systems that were used, and the design concept for a failure location system that was proved in prototype form.

3. Measures that were taken to place the reactor system in standby condition, for possible future restart, after the original operating program was terminated on December 1, 1964.

GENERAL INTRODUCTION AND CONTENTS

The Heavy Water Components Test Reactor (HWCTR) was built at the Savannah River site to test fuel and mechanical components that possessed potential applicability to power reactors on the basis of the overall technology that had been developed for the heavy-water-moderated, liquid-heavy-water-cooled production reactors at Savannah River. The HWCTR operating program was terminated on December 1, 1964, as a result of the AEC's decision to abandon development on the liquid-heavy-water-cooled concept and undertake a new program based on organic coolants.

This report incorporates three separate papers that were prepared to describe separate phases of the HWCTR program. The papers are separately entitled:

PERFORMANCE OF THE HWCTR SAFETY SYSTEMS

by

C. P. Ross

SUMMARY

Safe operation of the HWCTR depends on control mechanisms that provide a defense in depth against accidents. These control mechanisms include basic design requirements, testing procedures, administrative requirements, operating procedures, alarm circuits, and safety systems. Safety systems are not used for routine control of the reactor, but must be tested frequently to ensure reliability. The following safety systems are discussed in this paper: (1) safety rods, (2) poison injection, (3) pressure relief, and (4) containment.

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INTRODUCTION: ROLE OF SAFETY SYSTEMS

Du Pont operated the HWCTR (Heavy Water Components Test Reactor) from March 1962 to December 1964. The reactor was operated primarily to test fuel elements as part of the AEC's development program on power reactors moderated and cooled by D_2O. It was placed in standby condition when this program was redirected toward D_2O-moderated reactors cooled by organic liquids. The HWCTR is a pressure vessel reactor designed for 1500 psi and 315°C, cooled and moderated by D_2O, and pressurized by helium (Figure 1).

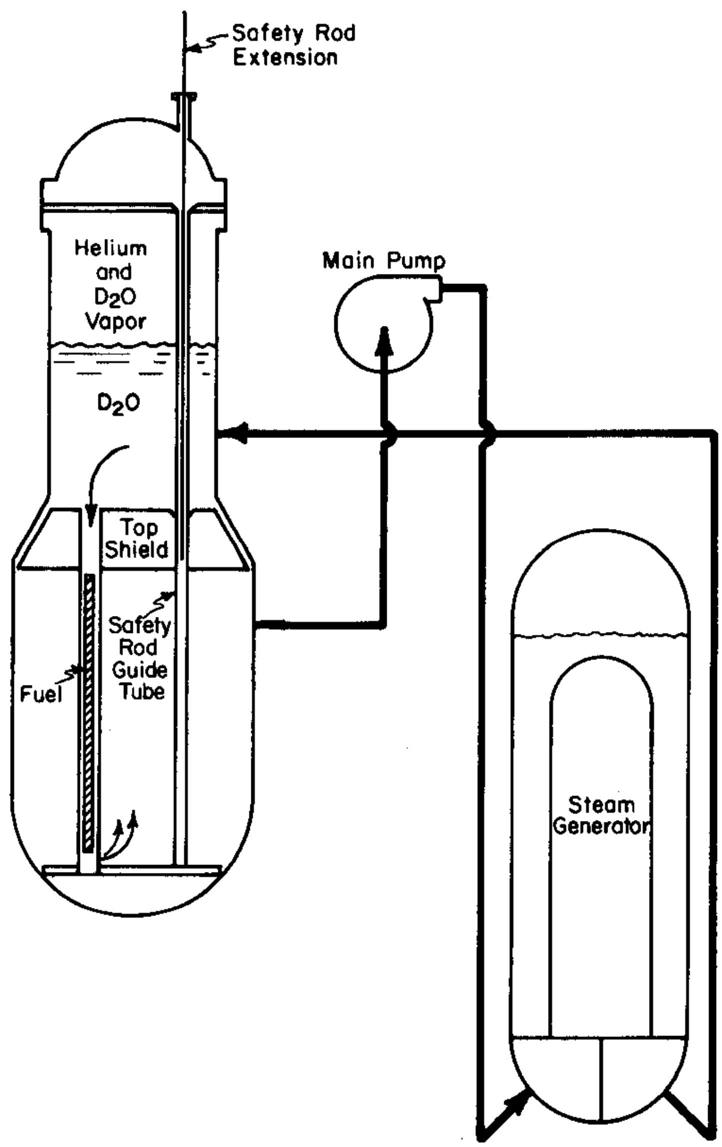


FIG. 1 HWCTR FLOW DIAGRAM

Safe operation of the HWCTR depends on control mechanisms that provide a defense in depth against accidents that could injure people or damage equipment.

These control mechanisms start with the basic concept of the reactor, and include the following:

- Design for negative power and temperature coefficients of reactivity.
- Test procedures to demonstrate that each system and each item of equipment is adequate for the specified process conditions.
- Mechanical limits on rate and amount of rod withdrawal and steam valve opening.
- Automatic alarm and scram circuits to inform the operator of abnormal conditions and to shut down the reactor, if necessary.
- Engineered safeguards to limit the extent of incidents that cannot be protected against entirely by safety circuits, such as large leaks of coolant or loss of electric power.
- An organizational structure that clearly defines responsibilities and provides for review of the operation at high levels of responsibility.

The safety systems described in this report are essential parts of the overall approach to safe operation. The safety systems are not relied on to control the reactor during normal operation, but adequate testing must establish that they will act reliably if needed. Figure 2 shows where each of these safety systems is intended to stop the progression of an incident through the various barriers to off-site release of fission products.

The HWCTR was operated under pre-established limits by trained operators and supervisors following written procedures. Abnormal operation was detected by surveillance of indicating instruments and by alarm circuits. Written emergency procedures for each alarm circuit provided step-by-step action for identifying and correcting the abnormality. Thorough analysis and reporting of all unusual incidents was required, and these reports were audited twice a year for trends that were not evident from analysis of individual incidents.

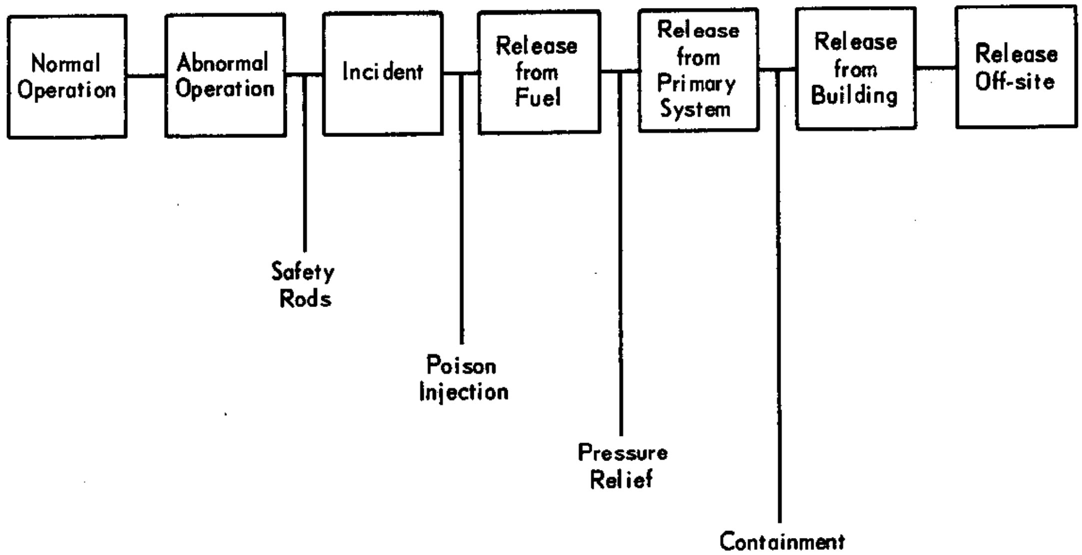


FIG. 2 INCIDENT PROGRESSION AND SAFETY SYSTEMS

Backing up the procedural controls were automatic safety circuits designed to scram the reactor by dropping safety rods before the safety limit of any variable was reached. The primary limits on reactor operation were on the D_2O coolant temperature, the burnout safety factor of the fuel, and the pressure in the primary system.[1]

A system for injecting a soluble neutron poison was provided to ensure that the reactor could be scrammed if the safety rods failed to drop.

Other engineered safeguards were provided to ensure continued cooling of the fuel to remove decay heat for extended periods of time even under the most adverse circumstances. A pressure relief system ensured integrity of the primary system; an emergency cooling system was provided for the fuel; and backup power and cooling water supplies gave additional confidence that decay heat could be removed.

Finally, a containment building was provided to minimize radiation hazards off the site even in the event of a complete loss of coolant from the primary system followed by destruction of the core.

The following sections provide information that might be useful to other designers or operators. These sections describe the design features, testing procedures, and possible areas of concern for these safety systems: safety rods, poison injection, pressure relief, and containment.

SAFETY RODS

Automatic protection against exceeding safety limits was provided by safety circuits that scrammed the reactor by disengaging the electromagnetic clutches of six safety rods. The set point of each circuit was checked prior to each startup. Written bypass rules stated the conditions required prior to bypassing any circuit, a time limit on the bypass, and the backup automatic protection and surveillance required while the circuit was bypassed. The rods dropped by gravity into guide tubes that were reduced in diameter over the lower three feet to snub the rods hydraulically, as shown in Figure 3.

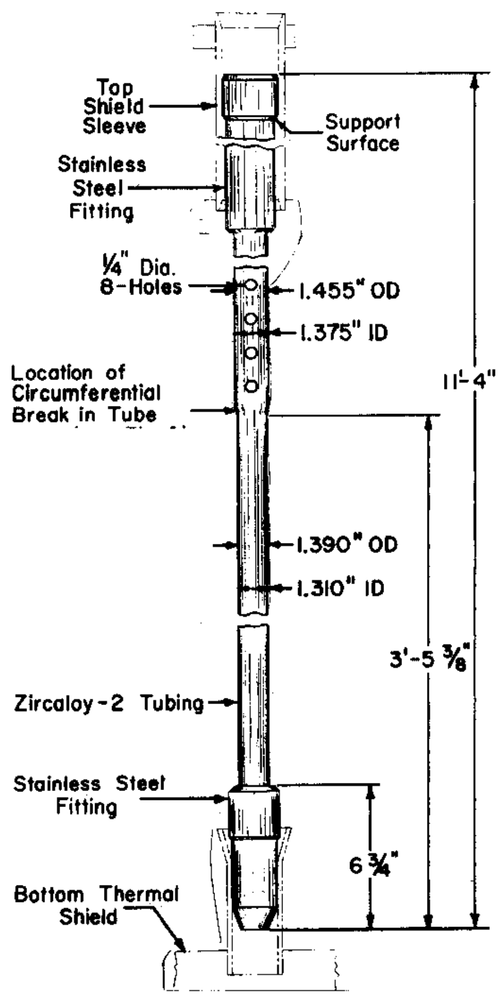
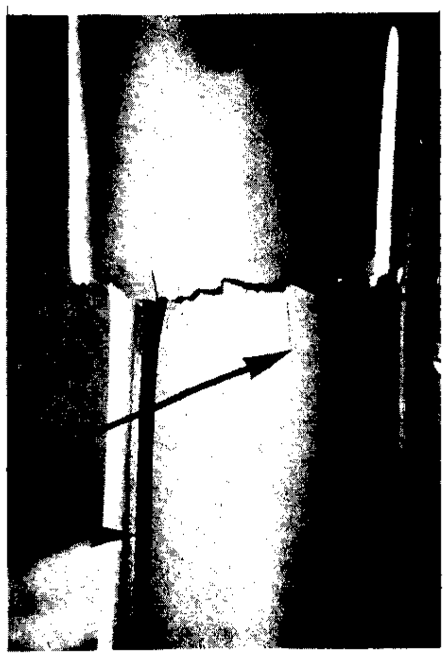


FIG. 3 HWCTR SAFETY ROD GUIDE TUBE

The mechanical and electrical performance of the safety rod drive system, use of 17-4 PH stainless steel parts, and measurement of drop times were discussed in detail by Kiger and Kale.[2,3] Drop times were measured before each startup to ensure that all rods would be 90% inserted in <1.5 seconds after a scram signal. The reactivity was required to decrease by 6% $(\Delta K/K)_{eff}$ two seconds after a scram. The reactivity worth of the safety rods was required to be at least 1% $(\Delta K/K)_{eff}$ greater than the reactivity held in temperature coefficients. The safety rods plus one of the 12 control rods could override both temperature and xenon decay, and hence could keep the reactor subcritical under all conditions.

One safety rod problem with serious potential was the failure of four of the six Zircaloy-2 guide tubes after nearly three years of service. The failures consisted of longitudinal splits starting at the transition to the reduced diameter portions of the tubes (Figure 4). Splitting was caused by a combination of high residual stress in the necked down area and fatigue due to cumulative tensile stress pulses when hydraulic snubbing occurred.[4] The failures were detected when safety rods dropped into the core too rapidly during routine drop tests with the reactor shut down. These unexpected discoveries during tests to find possible slow drop times emphasized the need to be able to test every safety

Diameter Reduction From 1.375" ID To 1.310" ID



Circumferential Break

Longitudinal Cracks

Material: Zircaloy-2, drawn and annealed

FIG. 4 FRACTURES IN SAFETY ROD GUIDE TUBE system at regular intervals to find unforeseen difficulties. The photograph in Figure 4 shows that it might have been possible for a safety rod to drop onto the cracked bottom section, and hence not fully into the core, if the failures had not been discovered during the required drop tests. Replacement guide tubes were ordered, with a longer transition section (1 inch, rather than 1/4 inch) leading to the hydraulic snubbing section, to reduce the stress pulse. Fabrication specifications were improved to reduce residual stresses and susceptibility to deuterium pickup.

Examination of all six original guide tubes after removal from the reactor revealed the following additional contributing factors to the failures.[5]

a) The D_2 content of the Zircaloy was high (100-300 ppm), especially in the reduced diameter sections of the rods, showing that the pickup of D_2 produced by corrosion was as high as 50% of the total produced.

b) Radially oriented platelets of zirconium hydrides had formed near the outer surface of the Zircaloy.

c) The cracks had started at least several weeks before they propagated to a size sufficient to affect the drop times.

To determine the possible effect of material composition on susceptibility to failure, some similar (but larger diameter) housing tubes of low-nickel Zircaloy-2 that had been in the reactor for the same period as the guide tubes were also examined. The significant result of the comparison was that the D_2 pickup of the low-nickel tubes was a maximum of 12% of the total produced. In postirradiation ductility tests, these low-nickel tubes could be completely collapsed without cracking.

The data indicate that Zircaloy-4, or low-nickel Zircaloy-2, is superior to Zircaloy-2 for service at conditions similar to those in the HWCTR; i.e., temperature 200-250°C, pD 10-11 (measured at 25°C), chlorides less than 0.1 ppm, and oxygen less than 0.1 std cc/kg D_2O (0.14 ppm).

Another problem with serious potential arose after about two years of service. On separate occasions, about two months apart, a safety rod failed to drop by gravity on a scram signal but was driven in automatically by the drive mechanism. On each occasion the overrunning cam clutch, used to drive in a rod that fails to fall, was found to be cracked. After the first occasion, the clutch was replaced and subsequent rod drop times were satisfactory. After the second occasion, all clutches were examined and a third cracked clutch was discovered.

The cracks apparently resulted from high stress concentrations in the corners of a keyway in the clutch, since the corners were cut with almost no radii. New clutches with 60-mil radii in the keyway corners were installed, tested through several cycles of torquing, and inspected. No cracks appeared then or in subsequent periodic inspections.

POISON INJECTION

Backing up the safety and control rods was a system to inject a neutron poison solution directly into the reactor core. This system was required to be operable before raising any rods, and was intended to be actuated manually if the neutron flux failed to decrease after a scram signal. The reactivity worth of the solution was the same as that of the control rods, a minimum of 24% $(\Delta K/K)_{eff}$. If uniformly distributed in the full 7000 gallons of D_2O, the poison would keep the reactor subcritical under all conditions.

The system was required to be tested at least once a year during an outage, by replacing the poison solution with D_2O and timing an injection. Injection of poison was never required during reactor operation.

As the system was originally designed and built, poison solution was delivered rapidly from a storage tank pressurized by helium from an external source. It was planned to adjust the helium pressure for the proper injection time depending on whether the reactor was pressurized or unpressurized, and to determine the proper helium pressures experimentally during prestartup tests. Tests of this arrangement before initial critical showed that it would not deliver the poison properly under certain combinations of reactor pressure and flow. On several occasions with the reactor unpressurized, some D_2O was blown out of the tank and the main D_2O circulating pumps became gas bound by large volumes of helium.

The solution to these problems was to pressurize the storage tank by connecting its gas space to the reactor gas space, to elevate the storage tank about 50 feet above the reactor for the proper driving force, and to install an orifice in the injection line to provide the proper injection time. The revised system is shown schematically in Figure 5. Injection was started by piercing two rupture discs in the injection line.

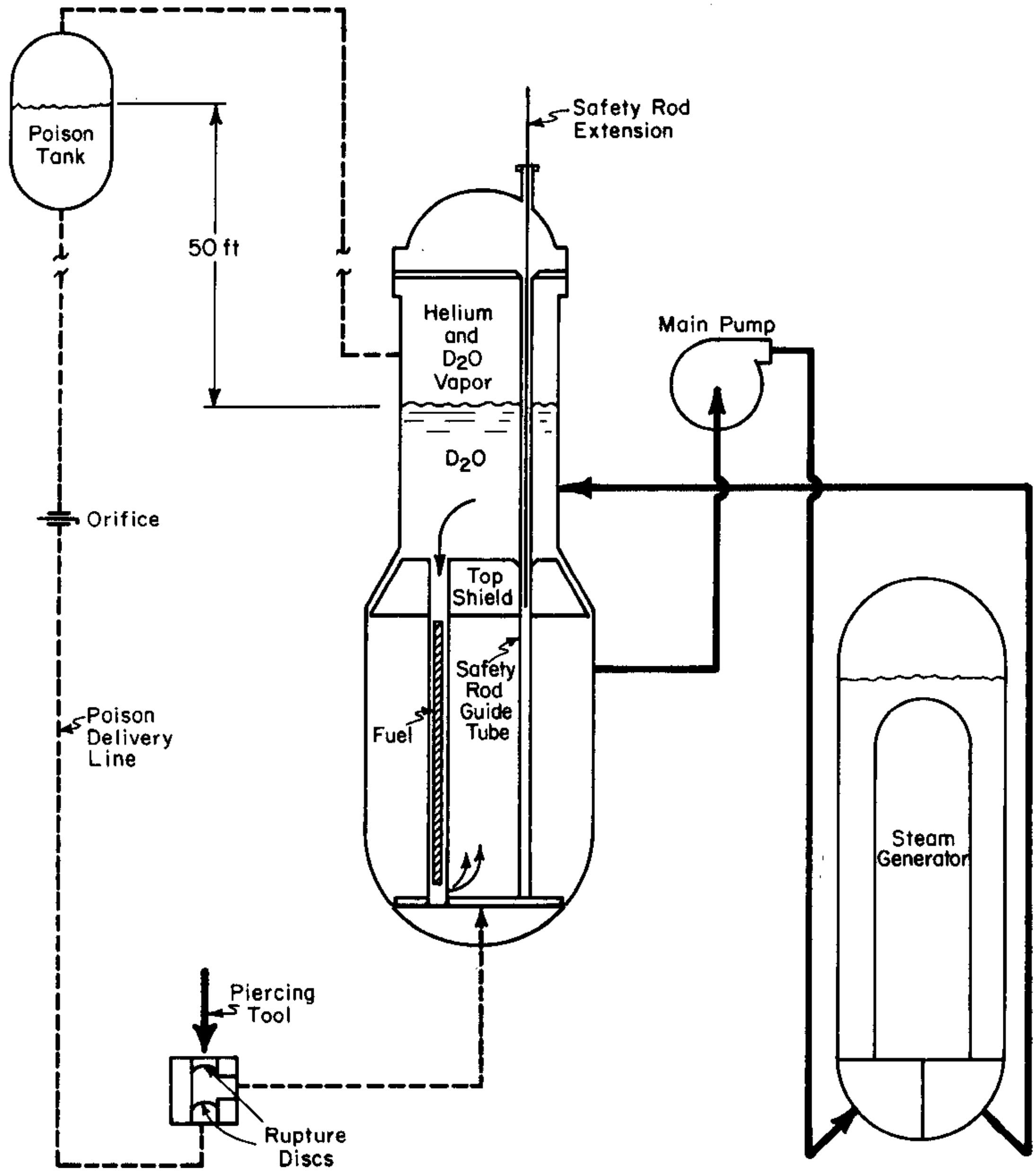


FIG. 5 POISON INJECTION SYSTEM

Although the operation of this system is simple in principle, considerable effort went into design and testing to ensure that the poison would get to the right place at the right time and stay there. A solution of potassium tetraborate, $K_2B_4O_7 \cdot 4H_2O$, in D_2O was chosen as the poison on the basis that it would remain soluble in suitable concentrations under storage and reactor conditions. The worth of the solution was determined by tests against a boric acid standard in a test reactor.

With the pressure equalized between the reactor gas space and the poison storage tank, the remaining variable that affected the injection time was the flow of coolant through the reactor. About half the system pressure drop is across the core, or between the reactor gas space and the moderator space where the poison is injected. Therefore the flow in the reactor directly affects the driving force for injection. The amount of poison that gets into the reactor also depends on the coolant flow, because the level of poison solution remaining in the delivery line depends on the pressure at the delivery point in the moderator space. At full flow (AC motors on), all the poison is injected and some helium circulates through the core. At shutdown flow (DC motors only), the level in the poison delivery line remains a few feet below the D_2O level in the reactor.

Injection time must be significantly longer than the recirculation time of the D_2O to ensure that enough poison is always in the reactor core. For short injection times the poison could be swept completely out of the core and permit a power excursion. Table I shows the amount of poison injected and compares its injection time to the D_2O circulation time for full flow and shutdown flow, and Figure 6 shows the amount of poison in the core region and its reactivity worth as a function of time after injection.

TABLE I

Poison Injection Characteristics

	Type Flow	
	Full (AC motors)	Shutdown (DC motors)
Total poison injected, gallons	73	60
Total injection time, minutes	3.6	3.9
Total D_2O circulation time, minutes	0.6	2.0

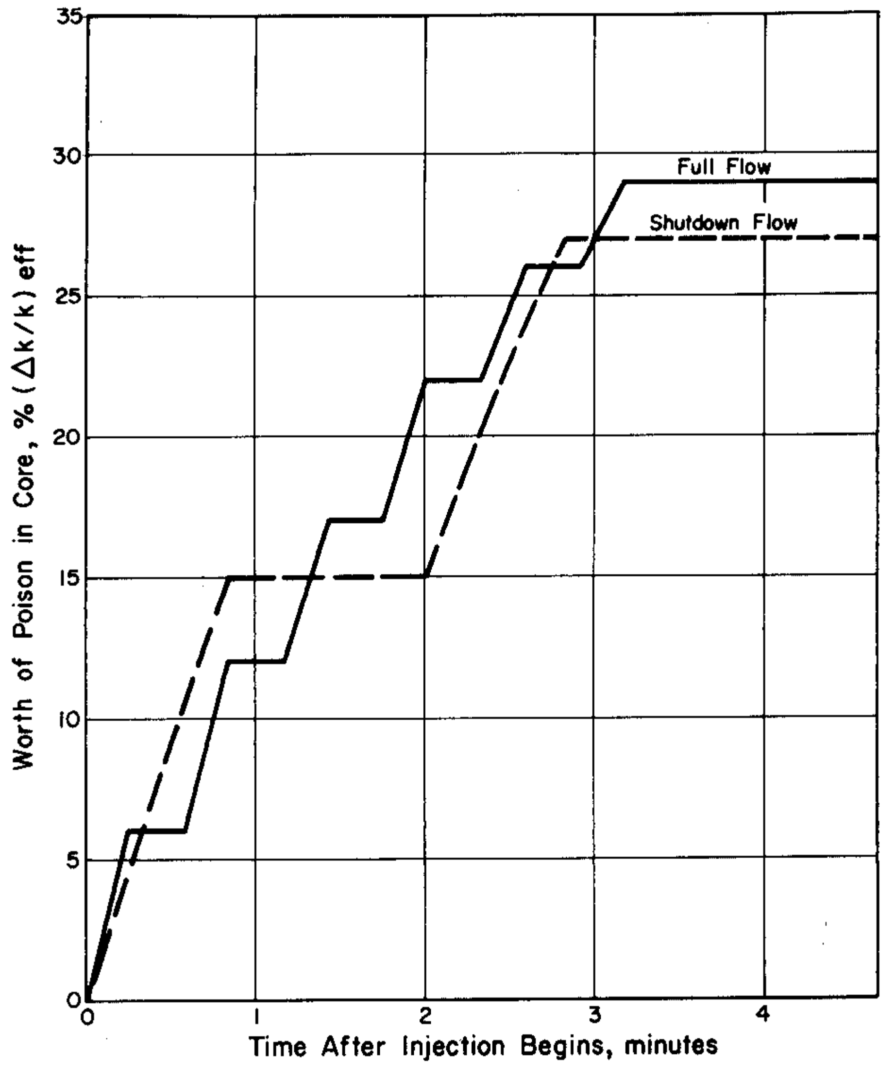


FIG. 6 WORTH OF POISON INJECTION SYSTEM

After the poison was injected in tests at full flow, the gas blown into the reactor increased the indicated level by 8 inches, and reduced the flow by about 4%. There was no fluctuation in flow to indicate gas binding of the D_2O circulating pumps. Absence of gas binding was demonstrated by a test in which the AC motor on one pump was shut down, then restarted. Flow was re-established immediately.

PRESSURE RELIEF

Original equipment of the HWCTR included two sets of relief valves on the primary system. The first comprised two pairs of high capacity gas relief valves, connected to the reactor gas space and interlocked so that one of each pair was always on line. The second was a pair of smaller liquid relief valves, one of which was always on line and set at a lower pressure than the gas relief valves (see Figure 7). Both sets of valves were equipped with leak detectors downstream, and switches to scram the reactor if a valve opened. The liquid relief valves were downstream of a heat exchanger that cooled the D_2O to 100°C or less, and their effluent was directed to the main storage tank in the basement of the reactor building.

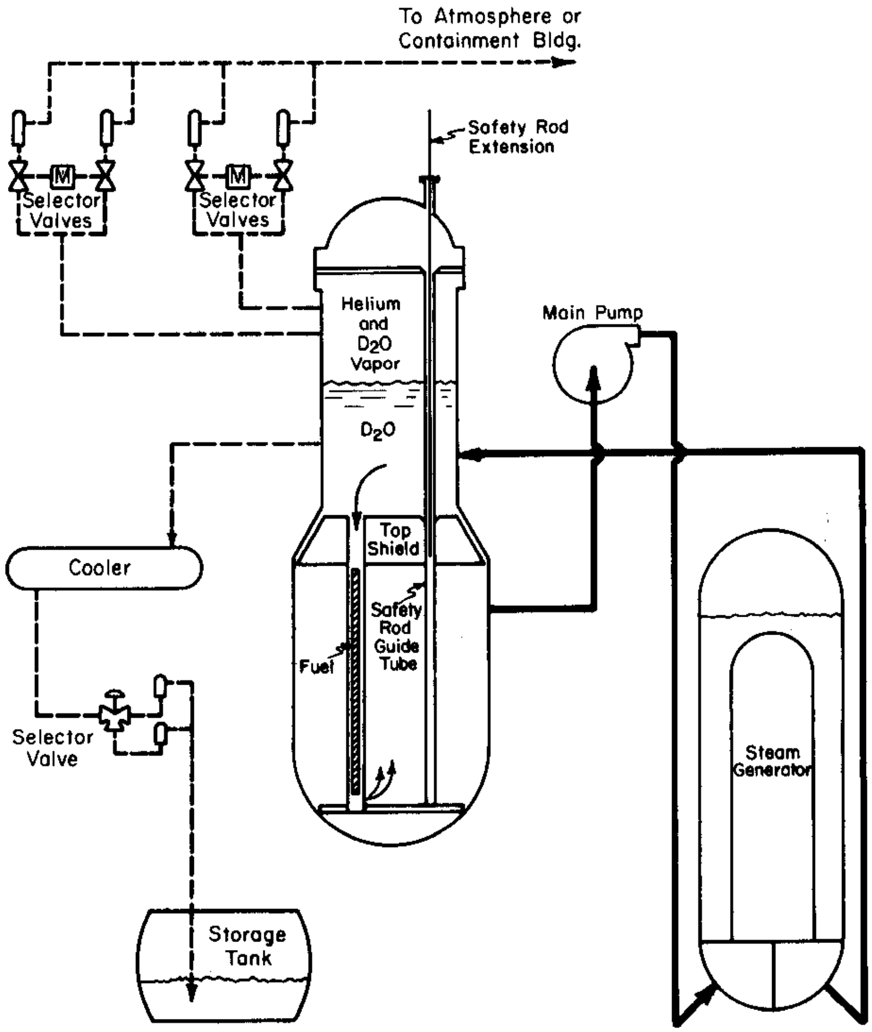


FIG. 7 PRESSURE RELIEF SYSTEMS

It was recognized that the gas relief valves could depressurize the reactor rapidly, and thereby permit boiling to occur. The liquid relief valves were provided to minimize the probability of the gas relief valves opening. Before the HWCTR was shut down, plans were being made to replace the gas relief valves with another pair of liquid relief valves. This change would eliminate the possibility of depressurizing the reactor rapidly, and would allow operation at higher pressures and temperatures because only a single relief setting would be required.

The two on-line gas relief valves were designed to pass steam at a rate equivalent to 60 MW (nominal full reactor power) at 10% overpressure (1650 psi). This design was adopted in 1958 to conform with the ASME Code with Special Nuclear Cases in effect at that time. This code requirement was based on coping with an accident in which all other heat removal capacity is lost at a constant heat generation rate, as in conventional boilers. A revision to this section of the code recognizes that in nuclear reactors the heat generation rate is a function of the system temperature and hence the heat removal rate. The revised code also allows the inclusion of the temperature coefficient of reactivity in the power calculations after heat removal capacity is lost. When this factor is taken into account it is calculated that gas relief valves are not needed to prevent excess pressure in the primary system of the HWCTR due to expansion of the moderator. Emergency cooling must be supplied eventually to prevent the vapor pressure of the D_2O from exceeding the design pressure.

Relief of reactor pressure was never required during operation. Tests at reactor hydraulic conditions showed that settings could be made within ±50 psi of the desired pressure. These tests also showed that the valves relieved less than 150 gpm, rather than the design value of 230 gpm, because of the pressure drop in the relief lines. However, the flow was more than adequate to cope with these new design criteria:

a) Loss of heat removal capacity

b) Malfunction of helium addition system

c) Malfunction of D_2O makeup system

Relief systems are not effective against reactivity addition incidents. For the assumed maximum ramp reactivity increase of 5×10^{-4} $(\Delta K/K)$/sec (two rods driving out), liquid relief capacity of about 500 gpm would be required to prevent overpressure from moderator expansion. With any relief system, overpressure would occur in about 30 seconds from the vapor pressure of the D_2O.

However, even with no relief, 10% overpressure (1650 psi) would not be reached until 20 seconds after this ramp addition began. The poison injection system is intended to be tripped manually within 15 seconds if the neutron flux is not decreasing, and would stop the pressure increase.

Figure 8 shows the pressure increase from D_2O expansion after a reactivity increase, as well as from loss of heat removal capacity and maximum addition of helium and D_2O. Pressure increases were calculated as if the safety circuits did not function.

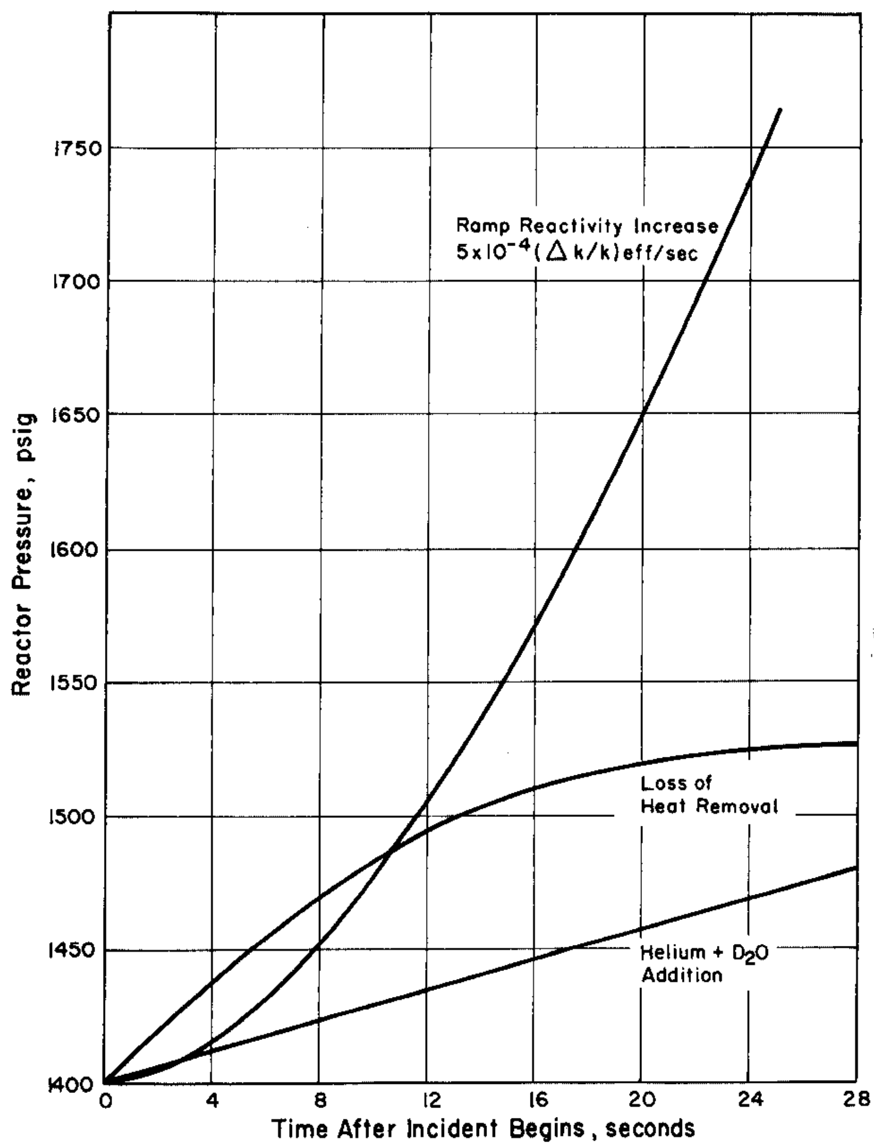


FIG. 8 PRESSURE TRANSIENTS
(no relief or scram)

CONTAINMENT

The HWCTR containment building is steel above grade and prestressed concrete below.[8] This building is compared to a typical all steel building in Figure 9, and the joint between the steel and concrete is shown in Figure 10. The concrete is lined with a thermal-setting plastic reinforced by fiberglass fabric to minimize leakage. Because of the large distance from the Savannah River Plant boundary, it was estimated that a leakage rate of about 1% of the building contents per day would be allowable. Although the sought-for leakage rate of 1% per day at 24 psig was not achieved with the building in an operable condition, the addition of iodine adsorbers reduced calculated off-site doses to acceptable values.

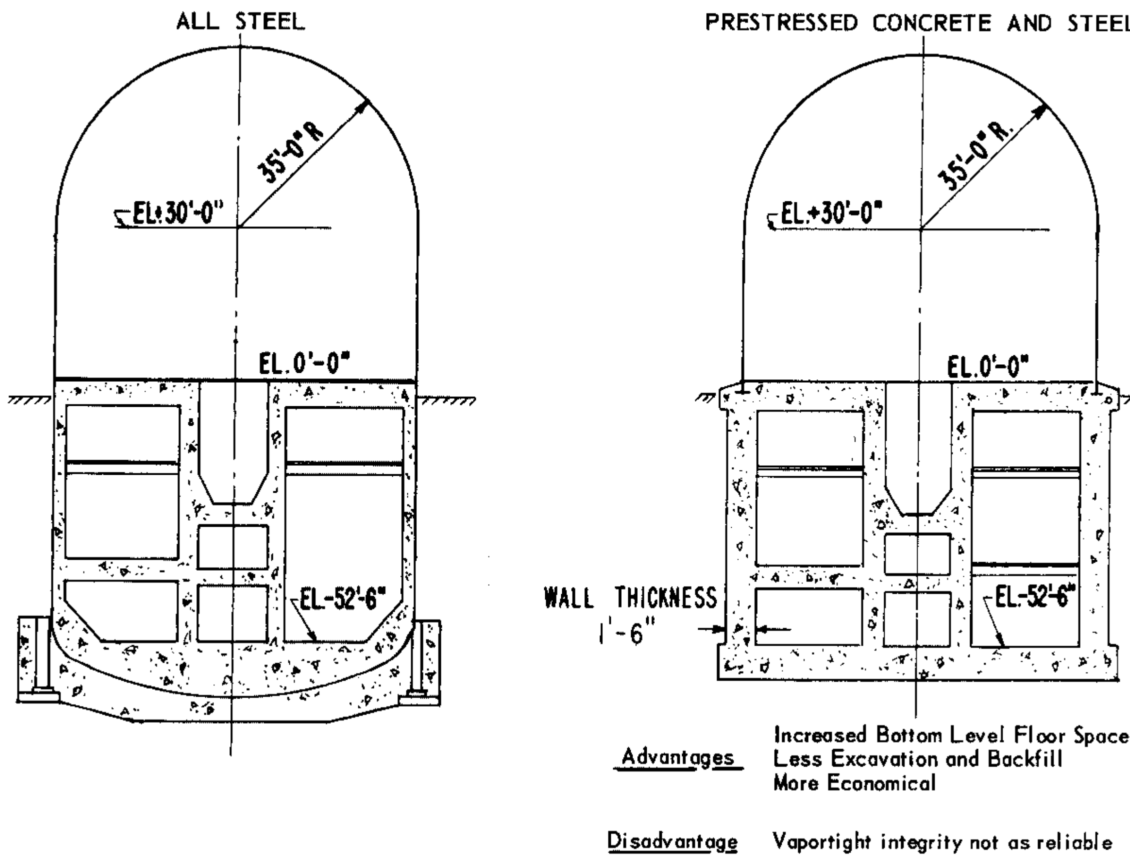


FIG. 9 COMPARISON OF CONTAINMENT VESSELS

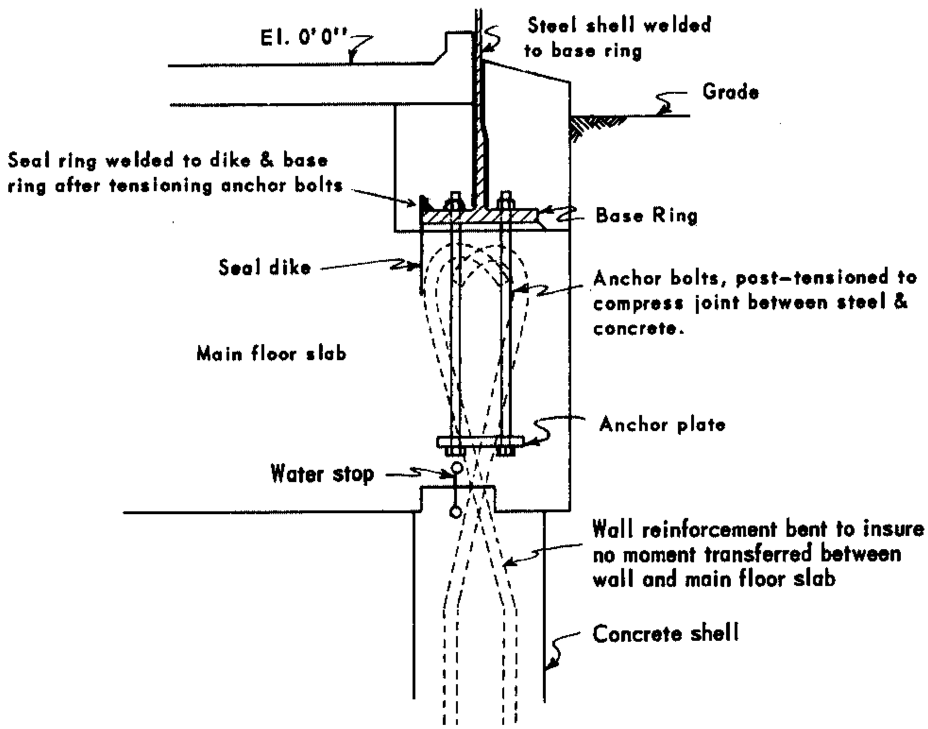


FIG. 10 MAIN JOINT OF CONTAINMENT BUILDING

Under any of the following circumstances, the building was required to be in a condition such that containment would be initiated automatically by a stack activity signal:

- Reactor critical
- D_2O temperature above 100°C
- Irradiated fuel being charged or discharged
- Lattice elements being moved whose complete removal or insertion would cause the reactor to be subcritical by less than the worth of six safety rods.

A deluge system, consisting of 15,000 gallons of water stored in the dome with connections to nozzles throughout the building, was provided to reduce the pressure and temperature in the building by condensing steam after a loss-of-coolant accident. This system could be activated manually, but was also designed to be operated automatically by a high pressure signal after containment was established. It was tested each year by actuating each valve in the system separately. Nozzles were inspected individually.

After a strength test of the building at 29 psig, but before most of the equipment was installed, the first leak test was run at 24 psig using the reference tank method, with suitable corrections for temperature and humidity changes.[7] The average leak rate was determined by statistical analysis of data taken over a 36-hour period. The measured leak rate of about 0.6% per day was judged satisfactory. Subsequent tests were run at a reduced pressure of 5 psig to prevent damage to instruments, and the results scaled to 24 psig on the basis of compressible laminar flow through the leak paths. This scaling method was verified by one set of consecutive tests at 5 and 12 psig. In most tests at 5 psig, operating personnel entered the building to locate leakage points. Leaks were detected by sound when all mechanical equipment was shut down. Also, during each test "Freon"* injection and a halide leak detector were used to locate leaks through electrical conduits.

When the building was completed and all penetrations made, measured leak rates were equivalent to 2 to 3% per day at 24 psig, even after those leaks that were found were eliminated. Many of the building penetrations were in the concrete walls and main floor slab, and it is believed these accounted for most of the leakage. Soap bubble tests of the exposed main joint revealed some leakage.

Because the leakage rate was higher than predicted, and because the calculated off-site doses after a core meltdown were on the borderline of acceptability based on conservative calculations, it was decided to take advantage of the development of carbon beds for iodine adsorption at Savannah River. If the contents of the building were circulated through carbon beds after an incident to remove iodine, only the noble gas fission products would contribute significantly to off-site doses. Four such adsorbers were installed, and circulation of the building air was continuous when the containment system was activated. In calculating the effect of iodine removal, credit was taken for only one of the adsorbers operating.

With iodine adsorbers installed, calculation of off-site doses from a ground level release established that a leakage rate equivalent to 7.6% per day or less at 24 psig would limit off-site whole body doses to less than 25 rad and thyroid doses to less than 30 Rem. The most recent leak test of the building indicated a leakage rate of 2.5% per day at 24 psig.

* Trademark of E. I. du Pont de Nemours & Co. for fluorinated hydrocarbons.

BIBLIOGRAPHY

1. L. M. Arnett, et al. Final Hazards Evaluation of the HWCTR. USAEC Report DP-600, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1962).

2. S. H. Kale. HWCTR Control-Rod and Safety-Rod Drive Systems. ASME Publication 63-WA-257, Presented at Annual ASME Winter Meeting, Philadelphia, Pa., November 17-22, 1963. TID-19001.

3. E. O. Kiger and S. H. Kale. Performance of HWCTR Safety Rod and Control Rod Drive Systems. USAEC Report DP-971, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1965).

4. D. F. Babcock. Heavy Water Moderated Power Reactors. Progress Report May - June 1964. USAEC Report DP-915, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1964).

5. D. F. Babcock. Heavy Water Moderated Power Reactors. Progress Report March - April 1965. USAEC Report DP-975, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1965).

6. H. W. Bellas. HWCTR - SRP Composite Concrete-Steel Containment Vessel, Engineering Considerations. USAEC Report DP-599, E. I. du Pont de Nemours & Co., Engineering Dept., Wilmington, Del. (1961).

7. S. H. Kale, et al. Leak Testing of the HWCTR Containment Building. USAEC Report DP-968, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1965).

AN INSTRUMENT FOR LOCATING FAILED FUEL ELEMENTS IN THE HWCTR

by

V. D. Vandervelde

SUMMARY

A prototype of a new fuel failure location instrument was tested in the HWCTR for eight months prior to termination of reactor operation. This prototype, the fixed window gamma monitor (FWGM), had a photomultiplier probe which monitored the sample stream from the effluent of each test element in sequence. An energy window from 0.05 to 0.15 Mev was determined to be a satisfactory compromise between excessive background count rate and absolute sensitivity. This interval included the neptunium and xenon activity peaks.

The FWGM properly identified the location of two oxide fuel element failures that occurred after its installation. The first failure occurred during a power ascension from 25 to 50% of full power, and the second occurred approximately three hours after a reactor shutdown from full power. The FWGM signal increased a factor of four above background during the first failure, and a factor of ten above background during the second failure. The favorable performance of the instrument during these fuel failures was a satisfactory demonstration that it should be effective for locating fuel element failures under normal operating and shutdown conditions.

Design was completed for an improved failure location instrument which incorporated the FWGM electronic system with a new, improved concept for monitoring integral sample lines at operating pressure.

CONTENTS

LIST OF FIGURES

INTRODUCTION

The fuel elements of the HWCTR were cooled with D_2O flowing downward through the assemblies, past monitor pins, and into the bulk moderator (see Figure 1). The moderator coolant was pumped into the steam generators and back into the reactor. Further details of the HWCTR are given in DP-600.[1]

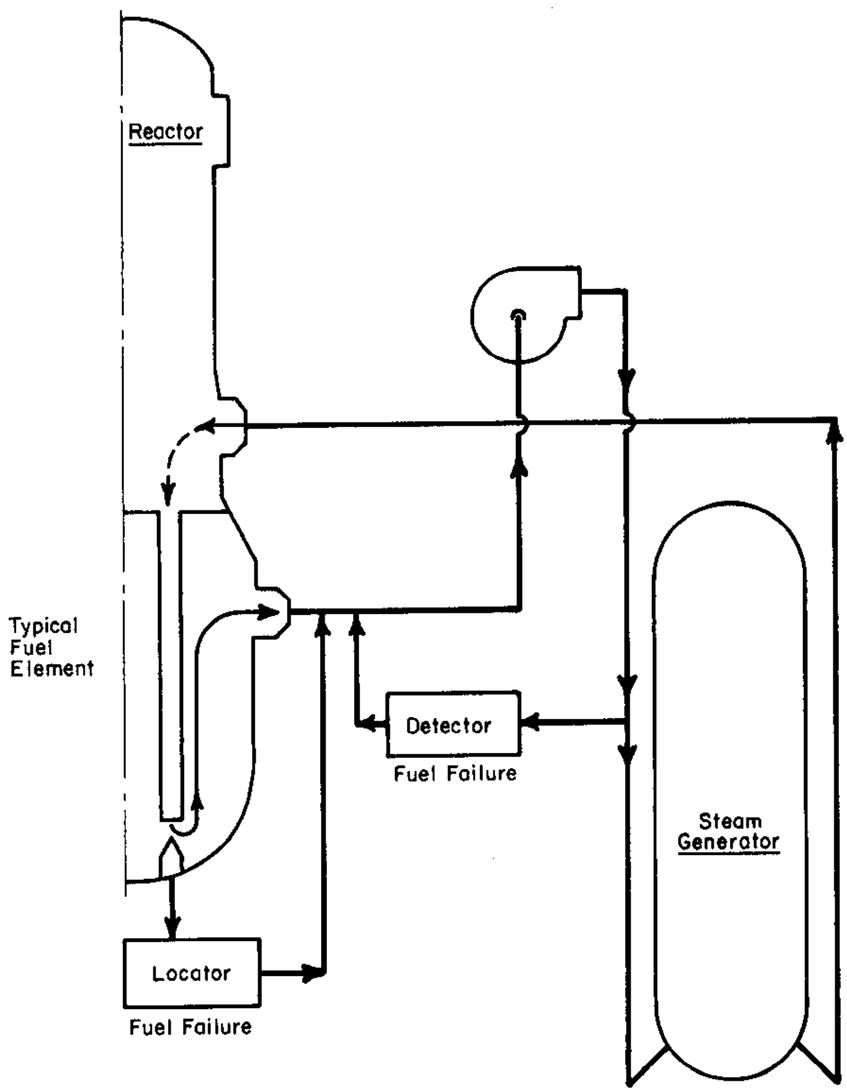


FIG. 1 HWCTR FLOW DIAGRAM

The HWCTR as originally designed and built was equipped with several instrument systems to detect the existence and position of failed fuel elements.[2] Considerable redundancy was built into this set of systems because of the necessity to shut down the high temperature reactor promptly when metallic uranium elements ruptured and because of the lack of proven performance of any single system. In general, the systems that had high sensitivity for detecting fission products from failed fuel had slow time responses and those that responded rapidly had low sensitivity. Actual experience demonstrated that the fast-acting systems had sufficient sensitivity to provide a signal in adequate time to allow the reactor to be shut down before a fuel failure had progressed significantly. The addition of uranium to the system was never more than a few grams.

The principal deficiency of the set of systems was the inability of the position detector to respond adequately when the fuel element failed at zero or low reactor power.

LOW ENERGY GAMMA MONITOR (LEGM)

A sample of the D_2O coolant was diverted from the process stream at the discharge of the circulating pump, passed through a 15-minute delay coil, and then passed across a probe scintillator crystal. From the probe assembly the sample was directed to a sediment pot, where the lowered velocity permitted particulate matter to settle out. The sample stream was then directed through a flowmeter and returned to the process stream at the pump suction. The sediment trap and flowmeter were normally bypassed during operation and were used only when calibrating the system.

The instrumentation scanned and recorded continuously the activity in the energy range from 0.03 to 0.3 Mev, which encompasses fission product ^{239}Np (0.1 Mev) and ^{133}Xe (0.08 Mev). The 15-minute delay coil prevented these low energy levels from being masked by short-lived products, such as ^{16}N (7.4-second half-life), ^{19}O (29-second half-life), ^{17}F (66-second half-life), and ^{27}Mg (10-minute half-life).

The pulses generated in the photomultiplier tube of the probe assembly were amplified in a linear amplifier and directed to a scanning pulse height analyzer. The scanning window of the analyzer permitted signals in the 0.03 to 0.3 Mev range to be directed to a count rate meter and recorder. The analyzer scanned this energy range, up and down, in 7.32 minutes. Signal level to the recorder was adjusted so that normal activity peaks in the 0.1 Mev region of the chart did not exceed 20% of full scale, thus providing 80% of full scale for observation of increases in activity.

This system had slow response due to the sample delay time, but it had a high signal-to-noise ratio.

GAS FISSION PRODUCT MONITOR (GFPM)

A 10-gpm purge stream was taken from the reactor vessel, cooled to 30°C, and then depressurized into a purification hold tank. A 100-cm^3/min sample of the gases evolved in this tank was pumped through an electrostatic precipitator that consisted of a cylindrical tube maintained positive with respect to a central moving wire electrode. The gas stream was then returned to the low pressure gas system.

The daughters of radioactive gas atoms were electrostatically precipitated onto the moving wire. The activity of radioactive daughters was detected as the wire passed in front of a detector. The radioactive daughters of fission product xenon and krypton were detected in this manner. A particular advantage of this system was that ^{41}A, the major gaseous background activity when the blanket gas was contaminated with air, was not detected since its daughter, ^{41}K, is stable.

The response of this system was similar to the LEGM - there was about a 15-minute delay in the sample arriving at the detector. Again, the slow response was offset by the high sensitivity provided by good signal-to-noise ratio.

DELAYED NEUTRON MONITOR (DNM)

A coolant sample line from the discharge of the circulating pump was directed through a H_2O reservoir. A boron-coated ion chamber in the reservoir detected the presence of delayed neutrons from ^{87}Br (54-second half-life) and ^{137}I (22-second half-life). The H_2O served both as a coolant and as a thermalizing medium between the sample line and the ion chamber. Transient time to the detector was about 60 seconds; this transient time minimized photoneutron activity from short-lived nonfission product gamma emitters (i.e., ^{16}N) acting upon the D_2O in the sample line. From the reservoir, the sample was directed to a sediment pot, through a flowmeter, and then returned to the process stream at the pump suction. The sediment trap and flowmeter were normally bypassed during operation and were used only for flow calibration.

The short sample transient time enabled this instrument to respond rapidly to large fuel failures, but its sensitivity was lower than that of either the LEGM or GFPM.

SCANNING LIQUID PHOTONEUTRON MONITOR (SLPM)

This system served as the only locator-detector among the original failure detection systems. Samples of the effluent from each of the ten test fuel positions were selected sequentially by a multiport valve, which routed the selected stream through tubing in a reservoir of pure D_2O. A boron-coated ion chamber in the reservoir detected photoneutrons caused by fission product gamma activity through the reaction $D(\gamma,n)H$. The D_2O thus acted both as a photoneutron source and as a thermalizing medium. Sample transient time to the reservoir was about 60 seconds; hence some delayed neutrons were also detected by the probe.

The mechanical arrangement of the multiport valve permitted a continuous stream from each of the test assemblies to pass through the valve, with the sample stream from one of the ten positions being diverted through the D_2O reservoir. A flow transducer, connected across a section of the sample line through the reservoir, measured the differential pressure as an indication of flow through the sample line. The combined samples were then returned to the suction of the circulating pumps.

The electronics for this system were identical to those for the DNM, with the addition of the fluid switch control circuits. The signal from the flow transducer was utilized to re-index the fluid switch automatically after each scan. The same signal was also displayed on the strip chart recorder as an indication of the position being measured. The scanning time for the fluid switch was adjustable between 4 to 15 seconds per position.

The fast response time and lower sensitivity associated with the DNM were also inherent in the design of the SLPM.

The SLPM was ineffective for locating a failure at reactor powers less than 20% full power, because the increase in count rate from fission products was within the normal variation of the background count rate. Several failures, especially in uranium oxide fuel, were detected either during a startup or after shutdown. Laboratory analyses of D_2O samples taken from the fuel effluents were used to locate these failures. Often, more than one series of samples had to be taken and, since each series was preceded by a one- to five-hour period of no-flow time to allow fission products to accumulate in the flow channel of the failed element, this technique extended reactor downtime.

EVALUATION OF FOUR INITIAL SYSTEMS

The LEGM proved to be the most reliable of the several systems for indicating failures of both metal and oxide fuels. The disadvantages for using a duplicate of this system for locating failures are:

- The long delay time (15-20 minutes) from the reactor. Although this delay time is in part responsible for the high sensitivity of the instrument, it is not compatible with the prompt location of failures.
- The long gamma energy scanning time (7.3 minutes).

A scanning instrument yields details of the gamma energy peaks, which aid in separating background from fission product signal, but any practical scanning time would require long intervals between the monitoring of individual test fuel elements.

Tests were performed with a portable pulse height scanning analyzer (similar to the LEGM) to monitor the SLPM sample lines. The results indicated that the samples after a 1-1/2 minute delay (for the background activity of ^{16}N to decay) produced a broad gamma energy peak centered near 0.1 Mev. Such an energy peak could be monitored with a fixed-energy window (from ~0.05 to ~0.14 Mev), and the total count rate of the peak would not be sensitive to electronic "drift" (up to 0.01 Mev) as long as the width of the window stays constant.

FIXED WINDOW GAMMA MONITOR (FWGM)

A prototype fixed window gamma monitor was installed and calibrated. The probe, consisting of a thin NaI crystal, photomultiplier, and preamplifier, was placed near the effluent sample stream of the SLPM multiport valve. This FWGM arrangement properly identified the location of the two oxide fuel elements which failed after installation. Table I is a summary of the failures in the HWCTR. The last two failures were identified by the FWGM and were typical of those that had preceded them. This experience was a realistic demonstration that a locator of this type can be a valuable aid in reactor operation when prompt location of fuel element failure is important.

TABLE I

Instrument Response to Fuel Failures

Element[a]	Reactor Power, MW	LEGM, % Increase	SLPM, cps Background	SLPM, cps At Failure	FWGM, cps Background	FWGM, cps At Failure
A	36	>1460	5	13[b]	-	-
A	39	56	Inoperable		-	-
B	0	>400	>1	>1	-	-
B	1	2250	>1	>1	-	-
C	10	550	3	9[b]	-	-
C	37	100	15	15	-	-
B	0	200	>1	>1	-	-
C	20	420	7	21[b]	180,000	790,000[c]
B	0	100	>1	>1	3,000	30,000[c]

(a) A = Natural uranium metal
B = Natural uranium oxide
C = 1-1/2% enriched uranium oxide
(b) Failure located with SLPM.
(c) Failure located with FWGM.

The senstivity of a device that monitors several samples is dependent upon the relative increase in activity of one sample above the neighboring samples. Activity increases due to reactor power changes or other whole system changes tend to affect all samples together, while the activity in a sample with fission products is seen as a rise above all others. Figure 2 shows the FWGM activity chart at the time of failure of a slightly enriched uranium oxide element. This failure occurred as a small, short-lived burst of activity during a power ascension. The reactor was shut down about 80 minutes later, after the activity increase was confirmed by the other fuel failure detectors.

The activity, recorded on linear chart paper, was from a logarithmic count rate meter whose scale normalizes to the chart at 33% = 10^4 cps, 67% = 10^5 cps, and 100% = 10^6. The square wave at the upper 2% of the recorder chart was made by a pulsing pen used to record the position of the multiport valve by moving up or down as the valve changed position. The valve positions are indicated on Figure 2 with dashed lines to the activity trace to indicate the activity being recorded at the various switch positions. The multiport valve was set to rotate once each 3 minutes.

The activity of position 3 (circled on chart) shows a sudden rise at the beginning of a power ascension (Figure 2) while the other activities show a slower rise in proportion to the power increase. About 12 minutes later a smaller (but still significant) rise in activity of point 3 may be noted.

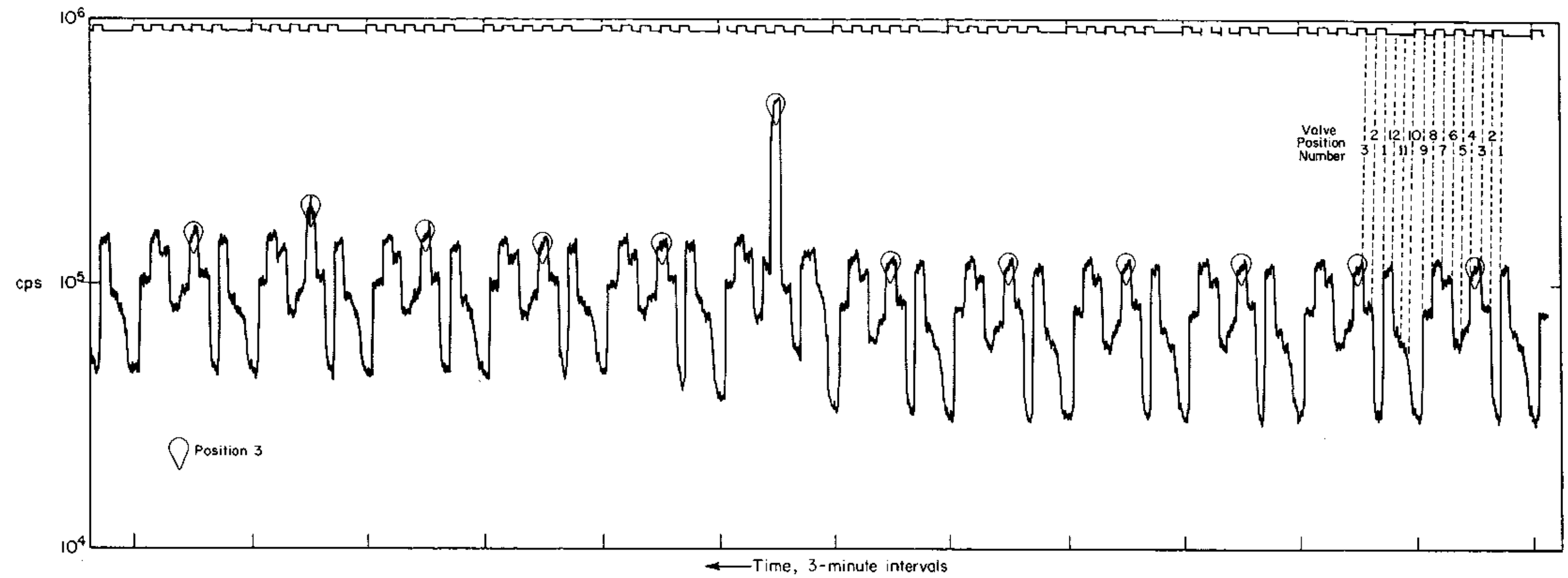


FIG. 2 FWGM RECORDER CHART DURING FAILURE ON MAY 30, 1964

Operation of the multiport valve became erratic a few months prior to HWCTR shutdown, and plans were made for an activity sampling selector which would not depend upon valve operations in the high pressure moderator system. The design of the system was complete and fabrication was in progress when operation of HWCTR was terminated. Figure 3 shows a cross section and isometric view of the proposed system, which consisted of up to 24 stationary, continuously flowing, shielded sample lines; a rotating lead shield (~0.2 rpm) with a single hole aimed at the samples in sequence; and a stationary probe which feeds its signal to the FWGM.

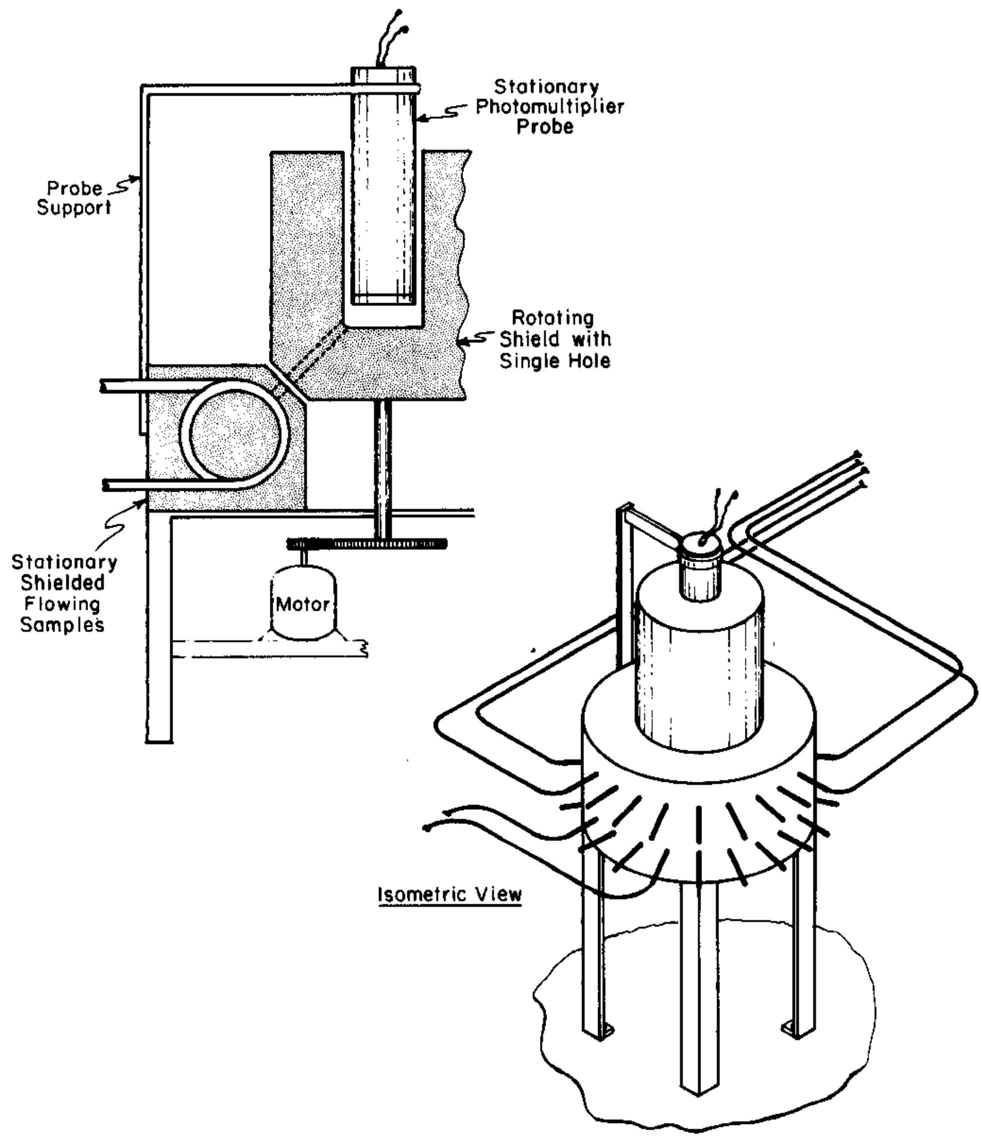


FIG. 3 PROPOSED ACTIVITY SAMPLE SELECTOR

BIBLIOGRAPHY

1. L. M. Arnett, D. Randall, C. P. Ross, and B. C. Rusche. Final Hazards Evaluation of the Heavy Water Components Test Reactor. USAEC Report DP-600, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1962).

2. E. O. Kiger. Fuel Failure Detection in the Heavy Water Components Test Reactor. Presented at the ANS Reactor Operations Division Meeting, October 21-23, 1963 in Ottawa, Canada. Trans. Am. Nucl. Soc. 6 (Supplement), 48-9 (1963).

STANDBY CONDITION OF THE HWCTR

by

L. M. Arnett

SUMMARY

The HWCTR was shut down December 1, 1964 and placed in a standby condition. The system was drained, vacuum dried, and filled with dry nitrogen at 5 inches H_2O pressure. The heavy water moderator was processed through an ion exchange bed, upgraded, and stored in drums. Rotating equipment was filled with new lubricating oil and left at ambient conditions. The floors in the building were cleaned. The whole facility is inspected once a week. Maintenance, operating, and technical know-how was recorded.

About six months would be needed to reactivate the HWCTR — 115-man-months for supervisory personnel, and 280 man-months for operators and mechanics.

CONTENTS

INTRODUCTION

The Heavy Water Components Test Reactor (HWCTR) was operated from March 1962 until De ember 1, 1964, for the U. S. Atomic Energy Commission to test fuel and other reactor components as part of the research and development program on power reactors that are moderated and cooled with heavy water.[1] Operation of the HWCTR was terminated because of the decision to redirect the work on this reactor concept to reactors that are organic-cooled. However, the facility is in standby condition for possible reactivation should new interests arise.

LAY-AWAY PROCEDURES

Following the decision to place the reactor in standy condition, all of the fuel assemblies and the two neutron sources were removed from the reactor and stored under water at another Savannah River Plant facility. All other reactor components, including control and safety rods, long-term corrosion coupons, and a rod containing gamma ion chambers, were left in the core.

The D_2O from the reactor was circulated through a deuterized mixed-bed (DOD) ion exchange resin to remove the lithium ion that was present during operation to maintain the pD at 10.7 ±0.5. After this treatment, the D_2O was returned to the reprocessing plant, upgraded to 99.75% D_2O, and stored in drums.

All filters and ion exchange beds in the reactor purification system and the spent fuel basin system were removed and buried in the same manner that exhausted items were handled during operation.

After the D_2O was drained from the reactor system, the high pressure portion was vacuum dried to remove traces of D_2O; the low pressure system was drained and air dried. The whole system was closed, as during normal operating conditions, and filled with dry nitrogen at a pressure of about five inches of H_2O. This pressure is maintained from a bank of cylinders and requires about 50 scf or nitrogen per day. The adherent coating or magnetite on the carbon steel piping and equipment is expected to remain essentially unaltered under these conditions.

The H_2O was drained from the secondary coolant system, and most of the system was left at ambient condition. The standpipe, into which the untreated well water is pumped, had a rust coating which indicated a corrosion rate of about 7 mils per year. The feedwater pumps and their steam turbine drives were in excellent condition. The shell sides of the steam generators and the purge cooler were filled with nitrogen. New tubes will be required for che steam generators if the facility is reactivated because the tubes are severely pitted on the steam side. The pitting occurred during a period of operation when the lack of control of the oxygen content of the feedwater went undetected.(2)

After all fuel elements were removed from the spent fuel basin, the basin water was circulated through the purification system of filters and ion exchange resin to remove dissolved and particulate materials. The water was drained and the basin was hosed once. The basin was closed with a tarpaulin and a wooden cover. Care will be required to avoid the possible spread of airborne activity whenever this cover is removed.

All rotating equipment was left at ambient condition except that lubricating systems were drained and filled with new oil containing "Rust-ban 623."* These systems will be operated about twice per year to circulate the oil.

The electrical power system remains activated for selected containment building heating, ventilating, and lighting circuits. The bank of storage batteries that served as a source of emergency power has been left on trickle charge from a motor-generator so that no capacity will be lost.

The 15,000-gallon deluge tank in the top of the containment building was drained, sandblasted, and painted to arrest an existing corrosion condition.

Although the exterior of the containment building under the insulation is gradually corroding, no preservative measures are being taken because no serious impairment of strength is expected in less than three to five years. The outer coating over the polystyrene blocks that provide thermal insulation is cracked, and rain water has access to the steel building. The metal surface under the insulation is to be examined periodically to ensure that the building does not seriously deteriorate.

Heated air is circulated in the containment building and the control house to limit corrosion and deterioration of equipment from high relative humidity. Incoming air is heated by duct heaters and lights in the buildings to maintain an inside minimum temperature of 75°F, or higher, to provide a temperature in the buildings at least 10°F above the dew point. This procedure is estimated to maintain a relative humidity below 50 to 60%. The net circulation of air through the containment building is 2000 scfm, and an internal circulation of 6000 scfm is provided by six pedestal fans.

* Trademark of Humble Oil and Refining Co.

The floors in the containment building were cleaned until the contamination of transferable beta and gamma materials was less than 500 cpm, as measured at one inch with a Geiger counter of 30 mg/cm^2 wall thickness. The radiation in the building is generally less than 5 mR/hr. At some isolated locations, all of which are clearly marked, the radiation is as high as 1 R/hr. These locations all result from entrapment of activity in specific places in piping and equipment. Health Physics personnel survey the building once each month to assure that the general radiation levels have not increased and that the material from the known hot spots has not been transferred to new locations.

The whole facility is inspected once each week. The condition of the ventilating system, the temperatures in the buildings, and the amount of nitrogen in the storage bank are recorded. New nitrogen cylinders are added when needed.

The maintenance, operating, and technical staffs prepared comprehensive documents containing the know-how pertaining to the facility at the time of shutdown. Included were the peculiarities of individual items of equipment and systems, and recommended revisions and additions that would improve the performance and usefulness of the facility. This information should be very helpful to any new staff that restarts the facility.

REACTIVATION

To reactivate the HWCTR in the near future would require about six months. The manpower estimates for this period are 115 man-months for supervisory personnel and 280 man-months for operators and mechanics. The manpower and cost estimates exclude any provision for the development of the new research and development program and any special materials and equipment that such a program would require.

BIBLIOGRAPHY

1. L. M. Arnett, D. Randall, C. P. Ross, and B. C. Rusche. Final Hazards Evaluation of the Heavy Water Components Test Reactor (HWCTR). USAEC Report DP-600, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1962).

2. J. M. McKibben. Corrosion of Equipment in Heavy Water Components Test Reactor. USAEC Report DP-964, E. I. du Pont de Nemours & Co., Savannah River Laboratory, Aiken, S. C. (1965).

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The material on safety systems and standby condition was released previously. It was presented by Ross and Arnett respectively at the ANS meeting at Jackson Hole, July 1965, and submitted but subsequently rejected for publication in Nucleonics and/or Nuclear Applications.

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Author: L. M. Arnett, C. P. Ross, and V. D. Vandervelde

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S. W. O'Rear

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