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HAZARD SUMMARY REPORT FOR THE ARGONNE AGN-201 REACTOR

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ARGONNE NATIONAL LABORATORY
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HAZARD SUMMARY REPORT FOR THE ARGONNE AGN-201 REACTOR

Edited by

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Nuclear Science and Engineering

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HAZARD SUMMARY REPORT FOR THE ARGONNE AGN-201 REACTOR

by

K. C. Ruzich and W. J. Sturm

I. INTRODUCTION

The AGN-201 Reactor was designed and built by Aerojet-General Nucleonics, San Ramon, California. The reactor was intended for use in education, research, medical diagnosis, and industrial process control. The design criteria were intended to provide low cost, maximum safety, portability, and high sensitivity. A large number of AGN-201 reactors are presently being used for nuclear education at universities throughout the United States.

The AGN-201, Serial Number 108, was installed at the International School of Nuclear Science and Engineering, Argonne National Laboratory, in the Summer of 1957. The reactor was then operated for several months under Aerojet-General Nucleonics supervision by special arrangement between this company, Argonne National Laboratory, and the Atomic Energy Commission. In November of 1960, approval for operation of the reactor under Argonne supervision was obtained from the AEC. The reactor serves as a training and research facility at the International Institute of Nuclear Science and Engineering. It is operated by trained members of the staff, and the Argonaut Responsible Reactor Supervisor directs the operation of the reactor as part of the Institute's reactor training program.

The AGN-201 reactor is located in Building D-24 in the east section of the Laboratory. This building also contains a number of experimental training facilities including three exponential assemblies. Figure 1 is a diagram of the building floor plan. Locating the reactor in this building, which is adjacent to the Argonaut Reactor Building, makes the training reactor facilities of the International Institute a compact unit (see Figure 2).

A number of additional modifications have now been made on the reactor both for safety purposes and convenience of operation. Locks have been placed on the glory hole to prevent unauthorized insertion or removal of materials. Two locks are present on the thermal column tank bolts to prevent unauthorized removal and subsequent access to the core. A read-out meter for period measurement has been installed. Electrical changes have been made to prevent calibration of a flux indicator during operation of the reactor. This was required because, previously, calibration of the indicators rendered the high- and low-level trips on as many as two detecting channels inoperative during the calibration. A positive current source has been installed for the purpose of checking out the current-reading instruments prior to reactor operation. A neutron source drive mechanism has been placed in access port number two. This mechanism allows the reactor operator to drive the neutron source out of the reactor automatically when prescribed during the course of a reactor operation.

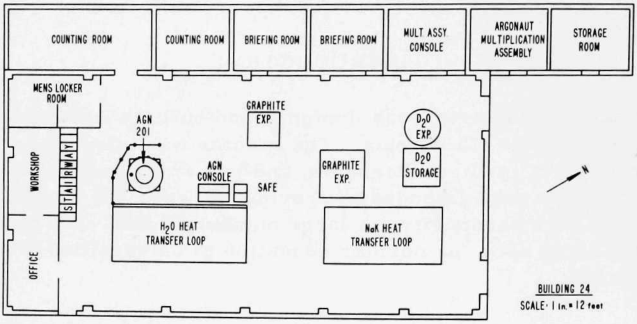

Fig. 1. Floor Plan of Building D-24

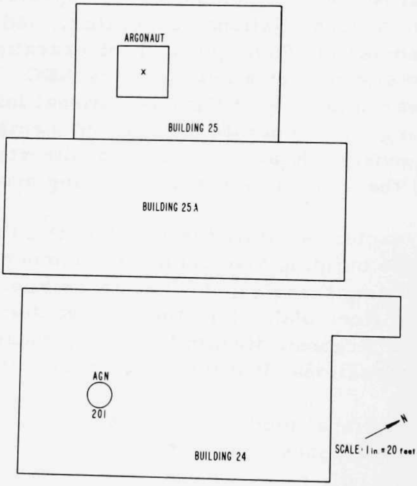

Fig. 2. IINSE Reactor Buildings

II. SUMMARY OF AGN-201 HAZARDS REPORT*

The Hazards Summary Report for the AGN-201 reactor by the Aerojet staff covers the essential characteristics of the facility. The following is in part a summary of that document. Only the pertinent topics are summarized, and no additions or modifications are made. Since this report will be distributed to individuals participating in the Institute, the summary of the pertinent topics has been made as complete as possible.

A. Physical Description of the Facility

The AGN-201 is an industrially manufactured, homogeneous, thermal reactor. It is polyethylene moderated, graphite reflected, and shielded by lead and water. The reactor operates at a power level of 100 milliwatts. The amount of fuel available restricts the reactivity which can be loaded into the reactor to 0.25% at normal operating temperature.

1. Reactor Unit (Figure 3)

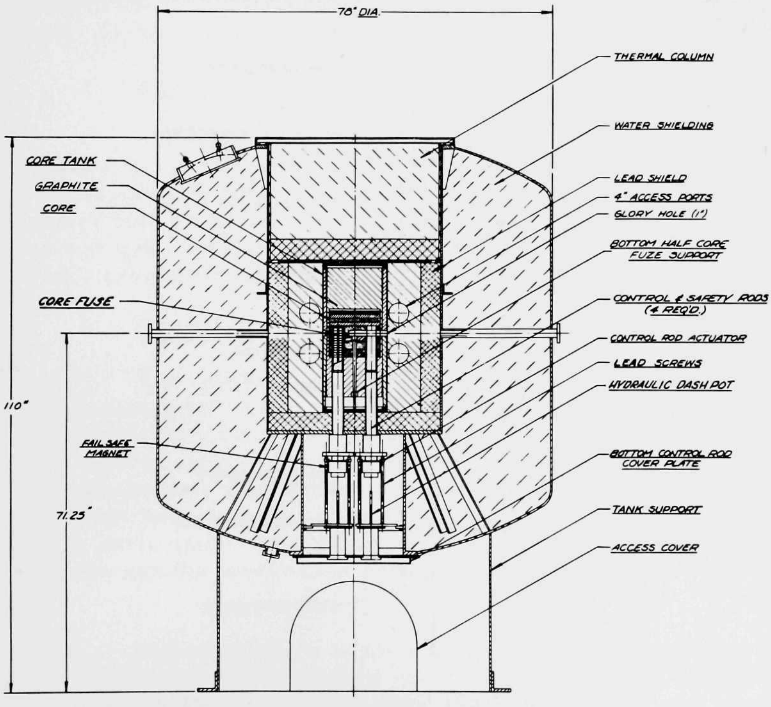

Fig. 3. AGN-201 Reactor Unit

*Aerojet-General Nucleonics Staff, Hazards Summary Report for the AGN-201 Reactor, Report Number 23, Revised 1 April 1959.

a. Core

The AGN-201 core is made up of a series of circular discs formed from a mixture of polyethylene and UO_2. The core contains 20% enriched UO_2 and has a critical mass of approximately 650 gm of U^{235}. The core configuration is approximately a 25 x 25-cm right cylinder. Each of the four bottom discs has four holes, two for safety rods and two for the control rods. The glory hole, which is a $\frac{15}{16}$-in.-ID through hole, passes through the center of the core.

b. Core Tank

The core and part of the graphite reflector are contained in a gas-tight aluminum (65-mil) tank. The tank has re-entrant thimbles in its base into which four control and safety rods are inserted. The core tank may be considered to be made of an upper and lower section, separated by an aluminum baffle passing through the fuel cylinder in the same plane as the glory hole. Access to the core is gained through the detachable top and bottom cover plates. The core tank and its contents are sketched in Figure 4.

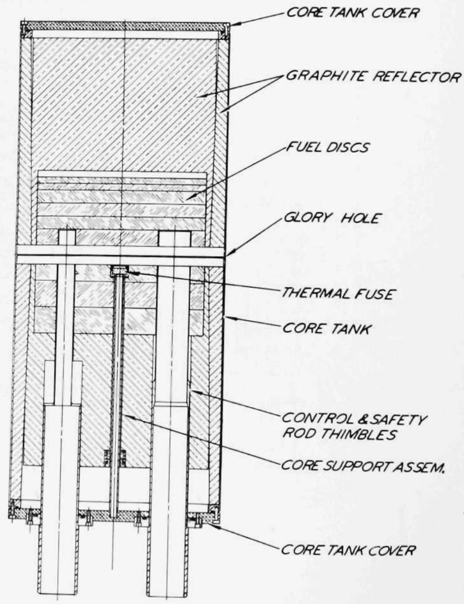

Fig. 4. AGN-201 Core Tank and Contents

c. Fusing System

The lower section of the core tank contains one-half of the core material as well as a cylindrical section of graphite reflector. This half of the assembly is supported by an aluminum tube hanging from a polystyrene fuse link which, in turn, is supported by an aluminum rod which is screwed into the bottom cover plate of the core tank. The polystyrene fuse, which supports a load of about 15 kg, is designed to soften at 100°C. The fuse has a fuel density of 108 mg/cm^3, which is twice that of the core. Because of this high fuel density and because of the central position of the fuse, it is expected that the fuse will soften before the rest of the core. In the event of an accidental runaway, the lower section of the core will drop 2 in. to the bottom of the core tank. The separation of the core reduces the reactivity by 5 to 10 percent, hence rendering the reactor subcritical.

d. Reflector

The reflector consists of 20 cm of high-density graphite on all sides of the core. Holes are provided for the glory hole, the two safety rods, the two control rods, and the four access ports.

e. Reactor Tank

The lead shield, reflector, and core are enclosed in and supported by a $\frac{5}{16}$-in. wall steel tank. A removable top cover is provided. This tank acts as a secondary container for the core tank assembly and, with the glory hole and access ports closed, is gas tight. The upper portion of the reactor tank contains a removable "thermal column tank" which can be filled with graphite or water.

f. Shielding

Ten centimeters of lead, which is contained within the reactor tank, completely surround the graphite reflector. This serves as a gamma shield for the core. The water tank is the third and outermost tank. It is constructed of steel and is $6\frac{1}{2}$ ft in diameter. When filled, the tank contains approximately 1,000 gal of water and affords 55 cm of shielding for the fast neutrons. In order to limit the production of capture gammas in the water, boric acid can be added at a concentration of 6 gm/liter. This will reduce the gamma level at the surface of the tank by approximately one-half.

Radiation levels at the surface of the shielding have been thoroughly measured at 100-milliwatt operation. Assuming a dosage of 7.5 mrem/hr as the weekly tolerance for 40-hr exposure, it was found that the only position above tolerance was around the tank skirt (base of the reactor). The exposure was found to be 120% of tolerance, but, since full body irradiation at this location would be difficult to obtain, this was not considered a problem. If graphite is placed in the thermal column tank instead of water, a high radiation level does exist at the top of the reactor (730% of tolerance).

g. Safety and Control Rods (Figure 5)

The AGN-201 has two safety and two control rods. Three of these, the two safety and the coarse control rods, are identical in design although their functions are different. Each contains about 14.0 gm of U^{235} in the form of fuel sealed in aluminum capsules and operates in a manner such that the reactivity is increased as the rod is inserted. The amount of reactivity each rod controls is nearly proportional to the amount of contained fuel. A rod containing 14.2 gm of U^{235} controls about 1.6% reactivity. The fine control rod is smaller in diameter and is loaded normally to control about 0.15% reactivity.

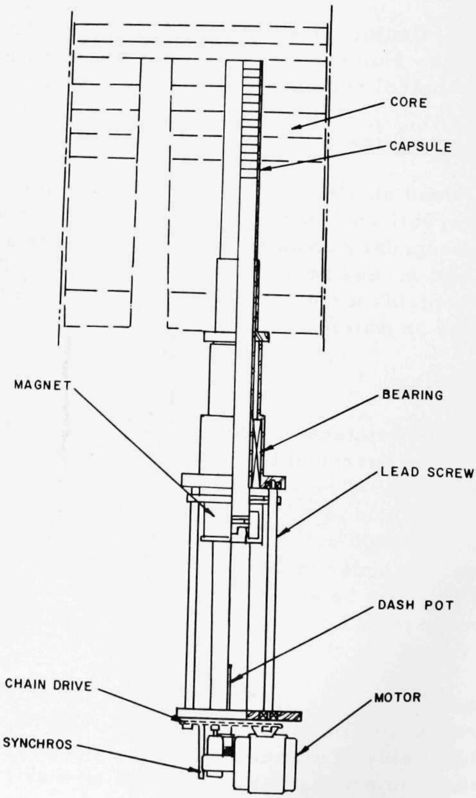

Fig. 5. AGN-201 Control Rod Assembly

The rods are driven into the core by reversible DC motors through lead screw assemblies which are controlled by switches at the control console. The lead screws are coupled to the coarse and safety rods through an electromagnet. This allows decoupling when the scram signal is received. The fine rod is driven in like manner but without the magnetic coupling. The total distance of travel of each rod is 25 cm. In the out position, the active fuel in the rod is just inside the lead shield and partially in the graphite reflector. The maximum rate of travel inward is 0.46 cm/sec. The speed of travel yields a maximum reactivity change of 3×10^{-4} sec^{-1} for the coarse rod. The safety system is a "fail safe" design in that the scram signal opens the holding magnets, allowing the rods to be accelerated outward by both gravity and spring loading. The total withdrawal time is estimated to be 150 ms for the safety and coarse rods. The fine rod is automatically driven out until it is in its outermost position.

2. Reactor Instrumentation and Controls

a. Instruments

The reactor has two BF_3 ionization chambers and one cadmium-covered BF_3 proportional counter, all of which are located in the water tank just outside the lead shield. These detectors are connected, respectively, to a logarithmic micromicroammeter, a linear micromicro-ammeter, and a pulse amplifier and count rate meter located on the reactor console. Each indicator is connected to a sensitrol relay for high- and low-level trip purposes. Any one of the three neutron-flux indicators may be selected to be recorded on a strip chart recorder.

b. Scram System

The flux-level scram is described as follows. The three instrument outputs are each fed to a sensitrol relay which is set at desired high and low scram levels. When these levels are reached, the relay closes and is held by its permanent magnet. This action interrupts the current to the safety and coarse control rod magnets and transfers the magnet power supply to actuate scram alarm. The rods drop into their down position, the scram light appears on the console, and an alarm bell rings. An annunciator light also appears, which indicates which instrument initiated the scram.

Scrams also result from low level of shielding water, low reactor temperature (falls below 16°C), earthquake, main power failure, or manual scram.

c. Startup

The main switch and circuit breaker inside the rear of the console is closed. The operator turns on an ignition-type lock which closes the power switch. The rod carriages are then driven out to their down positions if necessary. If the shielding water level and temperature are up, and the earthquake switch closed, the "Interlocks OK" indicator lights. In addition, the interlock light indicates that the cable connections to the rods are correctly connected. If none of the sensitrol relays are closed, the operator may then energize the holding magnets for the safety and coarse control rods. First, the number one safety rod is raised to its upper position. Next, the number two safety rod is fully inserted. After these steps have been completed, the control rods may be moved. Both the coarse and fine control rods have two speeds of insertion. The slow speed is available for convenience in reproducing rod positions and adjusting reactor power. The control positions are indicated to the nearest 0.01 cm on the console indicator.

B. Nuclear Characteristics and Safety Considerations

1. Safety Considerations during Nuclear Runaway

To evaluate the safety characteristics of the AGN-201 Reactor, a nuclear excursion resulting from a 2% instantaneous increase of reactivity is considered. The reactor would have a period of about 10 ms. The excursion would last from 200 to 220 ms, at which time the average temperature rise of the core (approximately 70°C) would be sufficient to stop the reactor because of core expansion. The temperature at the center of the core would rise to about 110°C. Since the fuel material is exposed to about 5 megarep of ionizing radiation during fabrication, the polyethylene will not melt below about 200°C. During the excursion a peak power of about 54 Mw is reached, and the total energy released is 1.7 Mjoule. It is expected that all the fission products released would be contained in the core and reactor fluid-tight metal tanks. To insure that the system does not remain in a near-critical state (if there is also a failure to scram), the thermal fuse which melts at 100°C will drop the lower half of the core to the bottom of the core tank, so that the reactor becomes subcritical. The total radiation dose to a person next to the reactor would be approximately one rem. If a loss of shielding water preceded the excursion, personnel next to the reactor would receive an exposure of about 200-300 rem of fast neutrons.

The total elapsed time between a neutron-induced signal from an ion chamber and a 2% decrease of reactivity from the resulting scramming of the safety rods may be as long as 300 ms. This breaks down to about 250 ms for the electronic circuitry and 50 ms for the necessary safety rod travel. Periods in excess of 30-50 ms will be adequately arrested by the scram system. Periods of this magnitude are initiated by a reactivity increase of about one percent.

2. The Criticality Experiment

Three people, to include one reactor operator and the Reactor Supervisor,* will be the minimum number of personnel required for conducting a criticality experiment. The Supervisor shall have the overall responsibility for the safe conduct of the experiment. All the data, taken for the purpose of determining how much nuclear material is to be added to the core, shall be processed independently by two people. The addition of material at each step shall not exceed one-half of the estimated remaining amount required for criticality, except at the final steps where 5 to 6 gm of U^{235} has been taken as the maximum safe amount to add.

The experiment is initiated by loading the core with dummy fuel (pure polyethylene). A small neutron source is brought near each of the flux-monitoring detectors to demonstrate operability and to cause a scram. Additional neutron detectors are placed in the access ports and water tank. The source is placed in the glory hole near the core, and the positions and sensitivities of the additional detectors, as well as the exact location of the source, are adjusted to insure reliable counting rates when high multiplications are achieved during the approach to critical. After proper positioning, an unmultiplied count for the system is taken from each instrument.

The approach to critical is carefully followed. For each step, three multiplied counts from each neutron detector are taken:

(1) all rods out;
(2) control rods out, safety rods in;
(3) control rods in, safety rods in.

The ratio of multiplied to unmultiplied counts gives the neutron multiplication, the reciprocal of which is plotted versus mass of U^{235} at each step. Extrapolation to a reciprocal multiplication value of zero gives a series of increasingly more accurate estimates of the critical mass as the core is built up.** The initial loading for the multiplication experiment is the complete lower half of the core.

After criticality has been reached, it is necessary to calibrate the power level, calibrate the control and safety rods, measure the temperature coefficient of reactivity, and evaluate the shield.†

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*At ANL, this person is designated the "Responsible Reactor Supervisor."

**For a further discussion on the multiplication experiment, see: Glasstone and Edlund, The Element of Nuclear Reactor Theory, Van Nostrand (1952), page 221.

†In A. T. Biehl et al., Elementary Reactor Experimentations (Oct 1957) are given the methods used to calibrate or measure these factors.

3. Operation of the AGN-201 Reactor

a. General Safety Rules

(1) Two people shall be present when the reactor is started up, one of whom must be a qualified reactor operator.

(2) A log book shall be kept of all reactor operations. A pre-startup check list shall also be employed.

(3) The neutron source shall be in the reactor at least during each startup procedure.

(4) The reactor shall not be operated if any of the instruments or controls are not functioning properly.

(5) Excess reactivity available shall be limited to 0.25 percent, which gives rise to a period of about 15 sec.

b. Normal Startup Procedure

(1) Operator signs the log book, unlocks the main power switch and the control console, and starts the check list.*

(2) A visual inspection of the reactor is made. Each relay is checked and set at scram level, and a neutron source is brought near each neutron detector to initiate a test scram.

(3) Meter readings are recorded for each of the instruments with source in the reactor and out, and are compared with the previous startup data.

(4) Safety rods are inserted one at a time and scrammed by means of the manual scram button.

(5) The safety rods are first inserted, then the coarse control rod is slowly driven into the core, while the counting rate is carefully monitored.

(6) The fine control rod is used for the final criticality adjustment. Meter readings are recorded periodically.

(7) The reactor is shut down by setting the scram power level less than the actual power. This serves to test the high-level-trip safety system. Meter readings are taken; the reactor electrical power switch at the console and the main power switch are turned off and locked.

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*Aerojet-General Nucleonics, The AGN-201 Reactor Manual Operation of the Reactor (1957).

C. Summary of Appendices

1. Assembly Drawing of the AGN-201 Reactor

Cf. Reactor Hazards Summary Report for the AGN-201 Nuclear Reactor, by the AGN staff, AGN-23, Revised April 1, 1959.

2. Core Fabrication Procedures for the AGN-201 Reactor

Cf. Reactor Hazards Summary Report for the AGN-201 Nuclear Reactor, by the AGN staff, AGN-23, Revised April 1, 1959.

3. Preliminary Test - Control and Safety Rods*

ABSTRACT

In the rod scram test it was found that the control rod scrammed essentially as calculated in the scram design calculations. The overall scram time was found to be approximately 150 ms, with the first 5 in. of travel taking 88 ms. Curves give the complete results in graphical form.

a. Purpose

It was the purpose of the control rod scram test to determine the scram rate of the control rod assembly itself.

b. Equipment and Wiring

Figure 6 shows the electrical wiring of the test setup. A general description of the equipment follows:

The control rod assembly and slide wire assembly were rigidly bolted in a vertical position to a jig fixture. The control rod assembly (control rod, magnet, motor drive, dash pot, springs, capsule, thimble, etc.) was complete, including a wood cylinder used to mock up the polyethylene and graphite components in the capsule. The slide wire assembly consisted of an insulating rod used to support the nichrome slide wire and two brass strips used as a slide to grip the slide wire. The slide, bolted to and insulated from the magnet plate, was lined up so that it would move down the slide wire and pick off a voltage proportional to its distance from the starting point. This voltage was placed on the vertical scale of the oscilloscope. Thus the voltage vs time curve obtained on the scope gave the displacement vs time curve desired.

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*See Reactor Hazards Summary Report for the AGN-201 Nuclear Reactor, by AGN staff, AGN-23, Revised April 1, 1959.

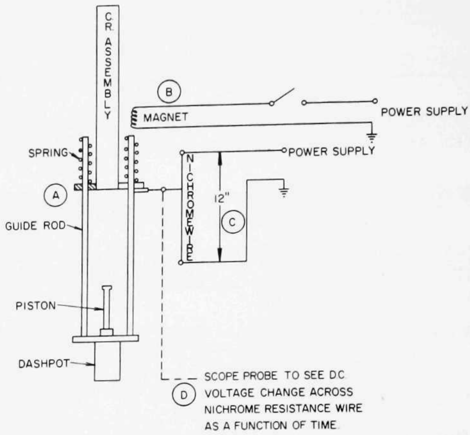

Fig. 6. Wiring Sketch (Rod Scram Test)

c. Discussion

The scram data presented in Figures 7 and 8 have been corrected for the transient effect of the collapsing magnet field and were verified by the fact that successive runs gave reproducible results. From analysis of this curve, it is apparent that magnet delay is approximately 20 ms. This delay was obtained when the magnet was operating at 35 ma or 120% of the minimum current required to support the control rod at the upper stop. The first 5 in. of travel is reasonably approximated by a constant 5-g acceleration. The last 5 in. of travel show the deceleration effect of the dash pot.

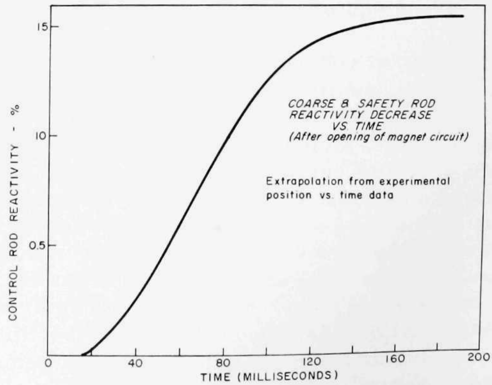

Fig. 7. Coarse and Safety Rod Reactivity Decrease vs Time

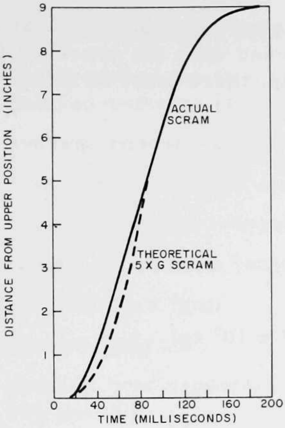

Fig. 8

Coarse and Safety Rod Scram Characteristic

d. Conclusion

The data described above give evidence that the initial design specifications have been approximated. However, the flexibility of the present design allows the use of a stronger spring in the control rod assembly, which will be incorporated to give a faster scram time.

4. Nuclear Behavior of Core in an Accidental Runaway

a. Evaluation of Energy Released during a Nuclear Runaway*

ABSTRACT

An evaluation of the energy released in a nuclear runaway accident in the AGN-201 has been made. For a 2% step increase in reactivity, about 1.7 Mjoules is calculated to be released, which is sufficient to heat the core about 71°C. The peak power is about 54 Mw.

(1) Introduction

One of the principal problems in evaluating the AGN-201 reactor is the extent of the energy generated in an accidental nuclear runaway. In this problem, the assumption is made that a step increase in reactivity is imposed upon the system and the only source of limiting the excursion is the negative temperature coefficient. A discussion of the problem is given in The Reactor Handbook - Volume I - Physics, and this general procedure is followed in this analysis.

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*See Reactor Hazards Summary Report for the AGN-201 Nuclear Reactor, AGN-23, Revised April 1, 1959.

As a first approximation, consider that a 2% step increase above delayed critical is imposed upon the reactor. In order to bring the reactor to just critical again, there must be a temperature increase of

$$\Delta u = \Delta k / C_T = \frac{0.02}{3.6 \times 10^{-4}} = 55°C \quad ,$$

or an energy liberation of

$$\Delta E = (\Delta u)\ Mc$$

$$\Delta E = 55 \times 12000 \times 0.55 = 3.7 \times 10^5 \text{ cal}$$

$$= 1.46 \text{ Mjoules} \quad .$$

A dynamic analysis indicates that approximately 1.7 Mjoules is actually released in such an accident.

(2) Analysis of Accident

Considering the time-dependent behavior of the neutron density, including one average group of delayed neutrons, one obtains for the time-dependent diffusion equations:

$$\dot{n} = \frac{P_0 - C_T\theta}{\ell} n - \frac{\beta}{\ell} n + \bar{\lambda} C \tag{1}$$

$$\left.\begin{aligned} \dot{C} &= \frac{\beta n}{\ell} - \lambda C \\ P = \dot{E} &= \Sigma_f n v \epsilon (\mathrm{Vol}) \end{aligned}\right\} \tag{2}$$

$$\dot{\theta} = \frac{\dot{E}}{\rho c (\mathrm{Vol})}$$

$$= \frac{\Sigma_f v \epsilon n}{\rho c} \tag{3}$$

where

n = neutron density (n/cm^3)

P_0 = excess reactivity (dimensionless)

C_T = temperature coefficient of reactivity ($°C^{-1}$)

θ = temperature rise, ($°C$)

ℓ = effective neutron lifetime (sec)

β = fraction of delayed neutrons (dimensionless)

$\overline{\lambda}$ = reciprocal of the average mean lifetime of the 6 groups of delayed neutrons (sec^{-1})

C = average concentration of delayed neutron precursors

Σ_f = macroscopic fission cross section (cm^{-1})

v = average thermal neutron velocity (cm/sec)

ϵ = energy per fission (watt-sec/fission = joules/fission)

M = mass of core (gm)

ρ = density (gm/cm^3)

c = specific heat capacity (watt-sec/gm-°C = joules/gm-°C = cal/gm-°C)

Vol = core volume (cm^3)

E = energy (watt-sec or joules)

P = power (watts)

The solution to the coupled nonlinear differential equations (1), (2), and (3) yields the neutron density (and thus the power and energy), the temperature, and the delayed neutron precursor density as a function of time. Since only a first integral of the equations can be obtained analytically, a numerical finite difference method will be used in which equations (1), (2), and (3) become

$$n_{i+1}(t) = n_i(t) + \Delta t\left[\left(\frac{P_0 - C_T\theta_i}{\ell}\right) n_i(t) - \frac{\beta}{\ell} n_i(t) + C_i(t)\,\overline{\lambda}\right]$$

$$C_{i+1}(t) = C_i(t) + \Delta t\left[\frac{\beta}{\ell} n_i(t) - \overline{\lambda}\, C_i(t)\right]$$

$$\theta_{i+1}(t) = \theta_i(t) + \Delta t\left[\frac{\Sigma_f v \epsilon}{\rho c}\right] n_i(t) \quad .$$

These can be solved as functions of time once initial values for n, C, and θ are chosen. The initial values and other pertinent constants in the case of the AGN-201 operating at 100 milliwatts with a 2% step increase in reactivity inserted are:

$t_0 = 0$

$\theta_0 = 0$

$C_0 = 1.185 \times 10^{+4}$ atoms/cm^3

$C_T = 3.6 \times 10^{-4}$/°C

$$\beta = 0.0075$$

$$\Sigma_f = 0.074 \text{ cm}^{-1}$$

$$v_s = 2.22 \times 10^5 \text{ cm/sec}$$

$$\epsilon = 76.6 \times 10^{-13} \text{ cal/fission}$$

$$\epsilon = 32.1 \times 10^{-12} \text{ watt-sec/fission}$$

$$n_0 = \frac{(P/Vol)}{\Sigma_f v \epsilon} = 1.58 \times 10 \text{ neutrons/cm}^3$$

$$P_0 = 0.020$$

$$\ell = 10^{-4} \text{ sec}$$

$$\bar{\lambda} = 0.1 \text{ sec}$$

$$\rho c = 2 \text{ watt-sec/cm}^3\text{-°C} = 0.478 \text{ cal/cm}^3\text{-°C}$$

$$\frac{\Sigma_f v \epsilon}{\rho c} = 2.64 \times 10^{-3} \text{ cm}^3\text{-°C/sec}$$

(3) Discussion of Results

(a) Assumptions

It will be recalled that in the above analysis the following assumptions were made:

1) At time equals zero, a 2% step increase in reactivity was inserted with the reactor at 100 mw.

2) At time zero, the energy in the core was negligible in comparison with the energy liberated during the accident. There was no heat removed from the core during the excursion. These are both very reasonable assumptions for the AGN-201.

(b) Results

The numerical solutions (see Figures 9, 10, and 11) to equations (1), (2), and (3) yield 54.4 mw for the peak power at t = 204 ms, a total energy release of 1.71 Mjoules and a temperature rise of 71.3°C, as compared with the crude preliminary values of 55°C and 1.1 Mjoules arrived at in Section 1.

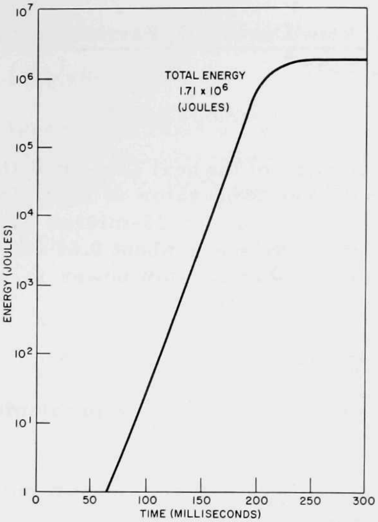

Fig. 9. Energy vs Time (Power Excursion)

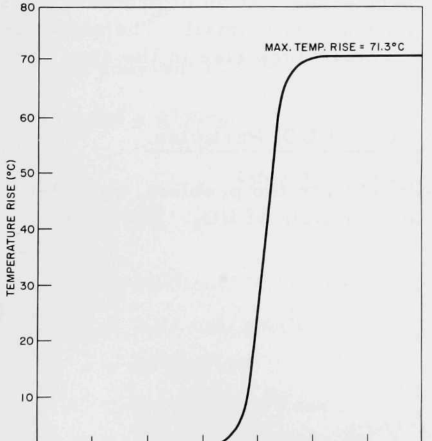

Fig. 10. Temperature vs Time (Power Excursion)

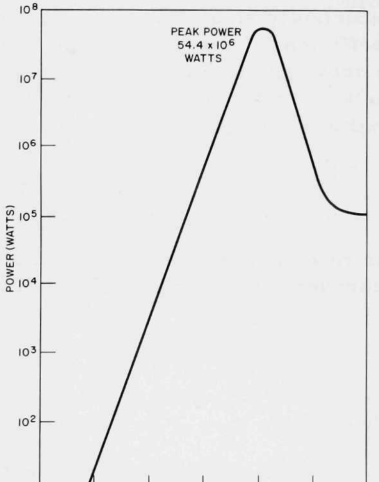

Fig. 11. Power vs Time (Power Excursion)

b. Heat Flow Out of UO_2 Particles into Polyethylene Moderator*

ABSTRACT

An evaluation of the heat flow out of the UO_2 particles into the polyethylene moderator is made for the AGN-201 reactor. It is shown that for 20-micron particles the heat flows into the polyethylene in about 0.69 ms. The temperature rise of the core at 100-mw power is calculated to be 0.044°C in the steady state.

(1) Introduction

At least two heat flow problems arise in the AGN-201 reactor; these are

(i) the rate of flow of heat out of the UO_2 particles and into the polyethylene core, and

(ii) the steady-state temperature of the core at a 100-mw power level.

The first problem is important in connection with the evaluation of the nuclear runaway accident, for the particle size must be sufficiently small so as not to offer any delay in the negative temperature coefficient of reactivity. The object of the investigation of problem (i) is to show that 20-micron particles are satisfactorily small. The object of the second problem is to show that the temperature rise in the core is negligibly small.

(2) Rate of Heat Flow Out of UO_2 Particles

As a first approximation to the problem, consider the rate of flow of heat out of a spherical particle of UO_2. The time-dependent heat flow equation is then

$$C\rho\frac{\delta u}{\delta t} = k\nabla^2 u + S \qquad (4)$$

where

C is the specific heat (cal-gm^{-1}/°C)

ρ is the density (gm/cm^3)

u is the temperature (°K)

*See Reactor Hazards Summary Report for the AGN-201 Nuclear Reactor, AGN-23, Revised April 1, 1959.

k is the thermal conductivity (cal/cm-°C-sec)

S is the source (cal/cm^3-sec) .

The source S has been taken as uniform in space and a delta function in time at t = 0. An approximate solution may be easily found by assuming S(r) is in the normal mode; then, by separation of variables,

$$u(\vec{r}, t) = u(\vec{r})\, T(t) \tag{5}$$

and Eq. (4) becomes

$$\frac{1}{T}\frac{\delta T}{\delta t} = \alpha^2 \frac{\nabla^2 u}{u} = -B^2 \tag{6}$$

where

$$\alpha^2 = k/C\rho \quad .$$

The solution must then be of the equations

$$\frac{\delta T}{\delta t} + B^2 T = 0 \tag{7}$$

and

$$\nabla^2 u + \frac{B^2}{\alpha^2} u = 0 \quad . \tag{8}$$

The time constant is then

$$1/B^2 \simeq R^2/\pi^2\alpha^2 \quad .$$

Time for Diffusion in UO_2 Particles

The constants for UO_2 are

$$k \simeq 10^{-3}\ cal/cm^1\text{-}^\circ C\text{-}sec$$

$$C = 0.03\ cal/gm\text{-}^\circ C$$

$$\rho = 10\ gm/cm^3$$

$$\alpha^2 \simeq 3 \times 10^{-3}\ cm^2/sec$$

$$R = 10\mu = 10 \times 10^{-4}\ cm \quad .$$

Thus,

$$1/B^2 \simeq 0.034 \times 10^{-3}\ sec = 0.034\ ms \quad .$$

Time for Diffusion in $(CH_2)_n$ about Particles

The amount of uranium per cm^3 is 250 mg, contained in 284 mg UO_2. The volume ratio is then 34.2 to one, or the elementary cell around each UO_2 is approximately of 32.8-μ radius. Using

$$k = 8 \times 10^{-4} \text{ cal/cm}^1\text{-}^\circ\text{C-sec}$$

$$C = 0.55 \text{ cal/gm-}^\circ\text{C}$$

$$\rho = 0.92 \text{ gm/cm}^3$$

$$R = 3.28 \times 10^{-3} \text{ cm}$$

$$\alpha^2 = 1.58 \times 10^{-3} \text{ cm}^2/\text{sec}$$

for the polyethylene, we obtain

$$1/B^2 = 0.69 \times 10^{-3} \text{ sec} = 0.69 \text{ ms} \quad .$$

Discussion of Diffusion in UO_2 and $(CH_2)_n$

Thus we see that the heat gets out of the UO_2 particles (of 20-μ diameter) in about 0.034 ms and then diffuses into the $(CH_2)_n$ in about 0.69 ms. Thus most of the delay is due to the poor thermal conductivity of the $(CH_2)_n$. Equations (7) and (8) could be solved to yield a slightly more accurate answer, but the additional effort is probably not justified by the small increase in accuracy.

(3) The Steady-state Temperature of the Core

An estimation of the core temperature may be made in an analogous manner by considering Eq. (4) and assuming a boundary condition at the surface of the core. This boundary condition may be approximated by assuming the carbon reflector to be a good heat conductor, and of infinite heat capacity, such that, at $r = R$, $u = u_1$ or room temperature. Then the steady-state equation is

$$k\nabla^2 u + S(r) = 0 \quad . \tag{9}$$

A simple solution may be found by assuming (as is very nearly the case) that

$$S(r) = S_0 \frac{u(r)}{u_0} \quad .$$

Then

$$u(r) = \frac{A \sin \nu r}{r} \quad , \tag{10}$$

where

$$\nu^2 = S_0/u_0k \quad .$$

By the boundary condition,

$$\nu^2 = \pi^2/R^2 = S_0/u_0k \quad .$$

Thus,

$$u_0 = \frac{S_0}{k}\left(\frac{R}{\pi}\right)^2 \quad , \tag{11}$$

where

$$S_0 = \frac{P}{V} = \frac{0.1}{12{,}000} = \frac{1}{12} \times 10^{-4}\,\text{watts/cm}^3 \text{ or } \frac{1}{48} \times 10^{-4}\,\text{cal/cm}^3\text{-sec} \quad .$$

With $k = 8 \times 10^{-4}$ cal/cm-°C and $R = 12.9$ cm,

$$u_0 = \frac{(12.9)^2}{48 \times 8 \times \pi^2} = 0.044°\text{C} \quad .$$

Thus the temperature rise of the core is 0.044°C at a power of 100 mw.

c. Temperature Coefficient of Reactivity*

ABSTRACT

The temperature coefficient of reactivity is calculated for the AGN-201 reactor to be -3.6×10^{-4} °C^{-1}. For a water boiler the comparable value is -3.0×10^{-4} °C^{-1}.

(1) Introduction

The temperature of the AGN-201 reactor will vary during normal operating conditions (due to variations in the ambient room temperature) as well as during an accidental nuclear runaway excursion. In both cases the change in temperature will cause a change in reactivity. The purpose of this section is to evaluate the magnitude of the temperature coefficient of reactivity.

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*See Reactor Hazards Summary Report for the AGN-201 Nuclear Reactor, AGN-23, Revised April 1, 1959.

As a first approximation, one would expect a value of about $-3 \times 10^{-4}\ ^\circ C^{-1}$, similar to a water boiler reactor, because of the similarity of the two types. This is reasonable since the coefficient of thermal expansion is not too dissimilar for water and polyethylene. In fact, the cubical coefficients of thermal expansion are:

Water @ 20°C = $2.07 \times 10^{-4}\ ^\circ C^{-1}$

Polyethylene @ 20°C = $5.4 \times 10^{-4}\ ^\circ C^{-1}$.

(2) Calculation Procedure

In order to calculate the temperature coefficient of reactivity, we follow the method of Glasstone and Edlund,* assuming the AGN-201 is a bare thermal reactor.

Nuclear Temperature Coefficients

The change in cross sections as the reactor heats causes a change in reactivity. Assuming $\sigma_a = \sigma_{a_0} \theta^{-1/2}$, where $\theta = T/T_0$, then

$$\frac{\delta \rho}{\delta \theta} \approx -\frac{B^2 L^2}{2k} \quad . \tag{12}$$

Since the U^{235} cross section varies nearly as $\nu^{-1.07}$, the variation due this change is approximately

$$\frac{\delta \rho}{\delta \theta} - \frac{k}{\nu}(\nu - k)x \quad , \tag{13}$$

where

$$x = \frac{1}{2}\left(1 - \frac{1}{1.07}\right)$$

Density Temperature Coefficients

The change in the density of the core will affect the leakage probabilities, which, in turn, changes the reactivity of the AGN-201. For an unconstrained system, the net change due to changes in L^2, τ, and increased size is

*S. Glasstone and M. C. Edlund, The Elements of Nuclear Reactor Theory, D. Van Nostrand Co., Inc., New York (1952)

$$\frac{\delta\rho}{\delta T} = -\frac{4}{3}\,\frac{k-1}{k}\,\alpha \tag{14}$$

where α is the cubical coefficient of expansion.

Evaluation of Constants

An approximate evaluation of the constants are as follows:

$$B^2 \approx (\pi/20)^2\ \mathrm{cm}^{-2}$$

$$k \approx 1.6$$

$$\nu = 2.46\ \mathrm{n/Fission}$$

$$L^2 \approx 1.07\ \mathrm{cm}^2 \quad .$$

These then yield, for the three partial temperature coefficients,

$$\frac{\delta\rho}{\delta\theta} = -0.825 \times 10^{-2} \quad ; \quad \frac{\delta\rho}{\delta T} = -0.28 \times 10^{-4}\,{}^\circ\mathrm{C}^{-1} \qquad 12$$

$$\frac{\delta\rho}{\delta\theta} = -2.0 \times 10^{-2} \quad ; \quad \frac{\delta\rho}{\delta T} = -0.67 \times 10^{-4}\,{}^\circ\mathrm{C}^{-1} \qquad 13$$

$$\frac{\delta\rho}{\delta\theta} = -8.1 \times 10^{-2} \quad ; \quad \frac{\delta\rho}{\delta T} = -2.7 \times 10^{-4}\,{}^\circ\mathrm{C}^{-1} \qquad 14$$

or a total of $-3.6 \times 10^{-4}\,{}^\circ\mathrm{C}^{-1}$

In order to check this, consider a water boiler reactor where the principal difference is in the coefficient of thermal expansion. By these same equations,

$$\frac{\delta\rho}{\delta T} = -0.28 \times 10^{-4}\,{}^\circ\mathrm{C}^{-1} \qquad 12$$

$$\frac{\delta\rho}{\delta T} = -0.67 \times 10^{-4}\,{}^\circ\mathrm{C}^{-1} \qquad 13$$

$$\frac{\delta\rho}{\delta T} = -1.1 \times 10^{-4}\,{}^\circ\mathrm{C}^{-1} \qquad 14$$

or a total of $-2.1 \times 10^{-4}\,{}^\circ\mathrm{C}^{-1}$

compared with a measured value of $-3 \times 10^{-4}\,{}^\circ\mathrm{C}^{-1}$. Thus one might expect the AGN-201 estimate of $-3.6 \times 10^{-4}\,{}^\circ\mathrm{C}^{-1}$ to be conservative (low) if anything.

5. Radiation Levels of the AGN-201 Reactor

Figure 12 shows the measured radiation levels at different positions in the reactor assembly during a 100-mw operation.

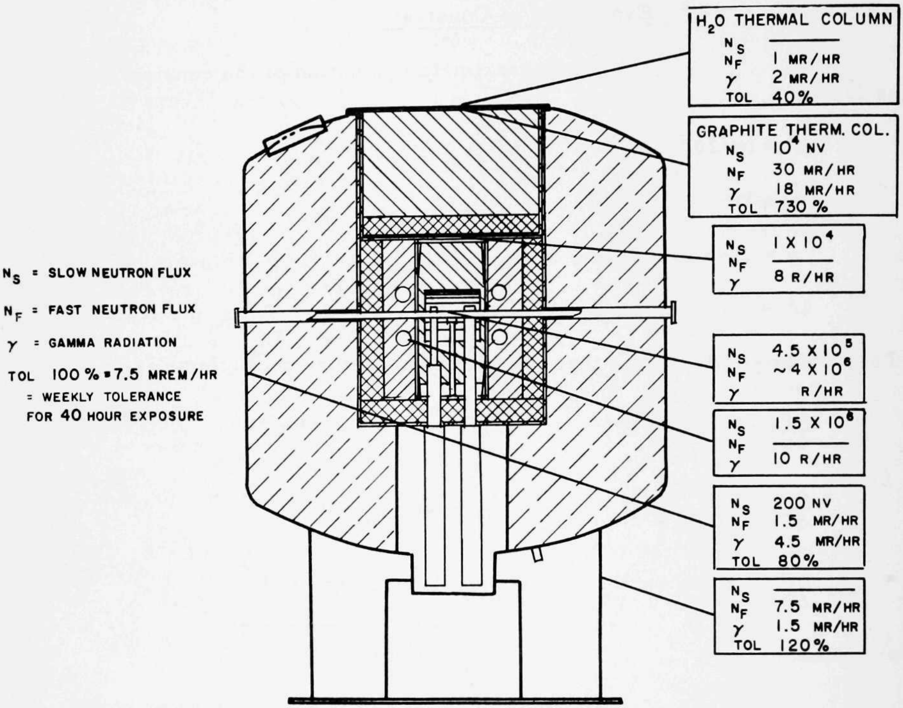

Fig. 12. Radiation Levels of the AGN-201 Reactor (100 mw Operation)

III. PERSONNEL RESPONSIBILITIES AND REQUIREMENTS*

A. Staff Organization and Responsibilities

The operating staff of the AGN-201 reactor consists of the following categories of individuals: Reactor Supervisor and Reactor Operator.

One Reactor Supervisor will be designated the Responsible Reactor Supervisor by the Director, IINSE. It will be his duty to coordinate the activities of the individuals involved in the operation of the facility.

The Reactor Supervisor is a staff scientist or engineer who is, by training and experience, capable of understanding the reactor, can exercise judgment as to the safety of its operation, and can assume the responsibility for changes in the reactor system. He is appointed a Reactor Supervisor by the Director, IINSE, in a letter to the Laboratory Director, which contains a statement of his pertinent training and experience.

The Reactor Operator is a Laboratory employee shown to be capable of operating the reactor according to the directions of the Reactor Supervisor. He is appointed a Reactor Operator by the Director, IINSE, in a letter to the Laboratory Director, containing the recommendation of the Responsible Reactor Supervisor. One Reactor Operator will be designated the Responsible Reactor Operator. It is this individual's responsibility to assure that all operations are approved by the Responsible Reactor Supervisor, that routine reactor operation is done within the precepts outlined in the Hazard Summary Report and the Operating Manual for the AGN-201, and that approved general laboratory procedures for work at a research facility be followed.

In addition to the above, there exists the category of observer, which includes all individuals learning the properties of the reactor or using the reactor as a tool, as well as those who may be merely observing in the more strict sense of the word. The numbers of these people and their activities shall be controlled by the Reactor Supervisor, consistent with the type of individual observing and the mode of operation of the reactor.

The safe operation of the reactor is the responsibility of the Reactor Supervisor, and it is his judgment that determines the action, if any, required in a given situation. He will review all operations and give the instructions to the Reactor Operator. The Reactor Supervisor will control the movement of fuel as well as of the pertinent reactor keys. He is expected to know and to follow the procedures and limitations set by the Summary Report on the Hazards of the AGN-201 Reactor. Any deviations

*Operating Manual for the Argonaut Reactor, ANL-6036 (August 1959).

from this report will be made only with the proper reviews. He should also keep the proper authorities informed of any matters related to the reactor which may be of interest or concern to the proper authorities, in accordance with the policy and practice guide of the Laboratory.

The safe operation of the reactor is also a responsibility of the Reactor Operator to the extent that he follows the operating instructions given him. It is also his responsibility to discontinue operations in the event that, in his estimation, an unsafe condition exists.

B. Review of Operations and Experiments

It will be the responsibility of the Responsible Reactor Supervisor to review each request for operation and to make a decision as to its safety or propriety. This decision may be referred to the proper authorities for review before the operation is performed; it may be made with the consultation of one or more of the Reactor Supervisors, or Laboratory scientists in the case of a somewhat new type of operation; or it may be made without consultation of other individuals in the case of a routine operation or developed experiment.

The operation of the reactor is performed only according to the directions of the Reactor Supervisor. The degree of control exercised may vary from being at the console during a type of operation which is new, to being aware of the status of the reactor in case of a routine operation. In all cases, the reactor is operated according to his instructions and by his consent. While the reactor is operating, it will be under the direct control of either the Reactor Supervisor or a Reactor Operator who is in such a location that he is aware of the status of the reactor at all times and can effect the proper action at any time.

Whenever work is being performed on or around the reactor, there shall be a minimum of two persons present, each to be aware of what the other is doing.

IV. REACTOR SAFETY EVALUATION

A. Characteristics of the System

1. During normal operation, negligible amounts of fission products are formed within the core and a large part of these are contained within the UO_2 particles.

2. The core and reactor gastight tanks are the primary and secondary seals which will retain the gaseous fission products released during a nuclear runaway.

3. The temperature coefficient of reactivity is negative and large in absolute value ($\sim -3.0 \times 10^{-4}\ {}^{\circ}C^{-1}$).

4. The amount of available excess reactivity in a normal loading of the core is restricted to about 0.25%.

5. The safety and control rod system is a "fail safe" design in that the scram signal opens the holding magnets, allowing the rods to be accelerated downward by both gravity and spring loading.

B. Causes of Hazards

1. Sabotage and Unauthorized Use of the Facility.

A well-informed saboteur presents a most effective means of destroying the reactor system. The main reliance must be placed upon the enforcement of Laboratory security regulations. The same is somewhat true for an individual who tries to operate the reactor without direct permission from the Responsible Reactor Supervisor. By installing locks on the critical components of the system and by adequate control of the keys to these locks, performance of an operation of this kind is made very difficult. It is, of course, assumed that any person on the reactor staff who has access to these keys is aware of his responsibilities and will never operate the reactor without authorization.

2. Accidental Operating Errors.

In general, accidental errors that occur during the normal operation of the reactor will be rectified before the resulting hazards arise. Interlocks insure that the proper procedure is followed during the startup of the reactor. Abnormal conditions caused by human error will automatically shut the reactor down. Scrams can be initiated by the following:

(a) Exceeding a maximum preset power level.

(b) Placing the reactor on a period which is less than about 12 sec.

(c) Lowering of the shielding water level.

(d) Loss of electrical power.

(e) Pressing the sensitrol reset button.

(f) Reaching a minimum preset power level.

(g) Disconnecting the electrical cables to the safety and control rods.

(h) Pressing the manual scram button.

The worst possible human error that can arise is the accidental insertion of fissionable material into the reactor. Entry to the core can be gained through the top of the reactor by the removal of the thermal column tank or through the 1-in.-diameter glory hole.

During normal operation the fuel loading is fixed, the thermal column water tank is installed, and the top cover plate is locked in place. Hence, it is extremely doubtful that accidental insertion of fissionable material would occur through this entrance. The glory hole is normally locked in any of three positions: closed, open, or closed with aluminum plugs in place. The accidental insertion of fissionable material into the core through this entry is therefore possible only when the lock is left open. The unavailability of fissionable material in the general area limits the chance of this occurring.

3. Equipment Failure.

As far as possible, all electrical and mechanical equipment has been designed so that an equipment failure will cause the reactor to shut down. In the event of an electrical power failure, the safety and coarse control rods, which are held in place by electromagnets, will be rapidly ejected from the reactor core. Any electrical cable failures will also scram the reactor. Since each of the safety and coarse control rods has a reactivity worth of more than 1%, any one rod can shut the reactor down under normal conditions.

There are three flux indicators which monitor the power level of the facility. Each is connected to a sensitrol relay for high- and low-level trip purposes. The reactor can be scrammed automatically even though as many as two trip circuits fail simultaneously. If all the trip circuits fail, a shutdown can still be initiated by actuating the manual scram.

A major problem would arise if all the rods failed to scram when the reactor power was rising. This event is very unlikely because of the "fail safe" design of the rods. The resulting hazards are analyzed below in Part C.

C. Hazards due to Accidental Operating Errors and Equipment Failure.

1. An increase in the radiation level can arise if the shielding plugs and covers are not in place. A visual inspection of the shielding is required in the checkout procedure. If the operator's inspection proves inadequate, the area gamma monitor will detect the error.

2. If the reactor is operated without water in the shielding tank at 100-mw power, the radiation level just outside the reactor tank will be about 10 mrem/hr of gamma rays and about 50 mrem/hr of fast neutrons.* Although the radiation levels are above the permissible level, the hazards are obviously far from acute. It is doubtful that the loss of the shielding water would not be detected during the checkout procedure. A shielding water level switch, included in the interlock system of the reactor, prevents operation of the reactor when the water level has dropped.

3. If the control rods are accidentally inserted after the operating power is reached, the power will rise with a period of less than 15 sec. The reactor would then scram after a preset upper power level on the logarithmic channel or count rate meter is reached. If the scram mechanism did not function properly, the reactor power would rise until the negative temperature coefficient reduced the reactor to a "just critical" state at some high power. When the rods are fully inserted, approximately 0.25% of reactivity is placed into the system. Since the temperature coefficient of reactivity is about $-2.5 \times 10^{-4}/°C$, the equilibrium temperature reached at the higher power level will be about 10°C over room temperature. This will correspond to a fission rate of approximately 10 watts, a factor of 100 above the normal operating power.** The radiation level at the least advantageous position adjacent to the tank would be approximately 900 mr/hr. This radiation level is indeed excessive, but it is inconceivable that it would go undetected for any length of time.

D. Hypothetical Maximum Accident

The accidental insertion of fissionable material into the core through the glory hole could produce a major accident. The hazards involved would be dependent upon the amount of fissionable material inserted, and the insertion speed of the material. The hypothetical maximum conceivable accident occurring, which could hardly be called an "accident," would be the insertion of Argonaut-type fuel. The volume of the glory hole through the core is about 115 cm^3. The volume of an Argonaut fuel plate, which contains about 20 gm U^{235}, is 106 cm^3. The reactivity worth of a gram of U^{235} ranges from 0.1% at the center of the core to 0.036% at the surface.**

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*A. T. Biehl et al., Elementary Reactor Experimentations (Oct 1957), p. 21.

**ibid., p. 99.

If it is assumed that the average worth of a gram of U^{235} is 0.06% and that it is possible to insert instantaneously a reshaped fuel plate, the induced reactivity will be approximately 1.2%. If a natural uranium rod was instantaneously inserted into the core, the induced reactivity would be about 0.93%. The accidental insertion of either of these materials seems doubtful, since they would first have to be reshaped to fit into the glory hole. However, their induced reactivities do have a bearing on the maximum reactivity that can be put into the system. As discussed in the next section, a 2% step increase of reactivity is chosen to determine the hazards of a nuclear runaway.

E. Evaluation of the Hypothetical Nuclear Runaway

An evaluation of a nuclear runaway accident in the AGN-201 Reactor has been made by the Aerojet staff. A 2% instantaneous step increase in reactivity was arbitrarily chosen. As seen from the previous section, insertion of this magnitude of reactivity is within the realm of possibility and should adequately describe the maximum power excursion.

Two assumptions are used as a basis for calculating the power generated in the accident.

1. At time equal zero, an ~2% step increase in reactivity is inserted with the reactor at 100-mw power.

2. At time zero, the energy in the core is negligible compared with the energy liberated during the accident, and there is no heat removed from the core during the excursion.

Some of the pertinent constants used in the calculation were:

1. Prompt neutron lifetime = 10^{-4} sec.
2. Reciprocal of the average mean lifetime of the six groups of delayed neutrons = 0.1 sec^{-1}.
3. Temperature coefficient of reactivity = $-3.6 \times 10^{-4}/°C$.
4. Specific heat capacity = 0.52 cal/gm-°C.
5. Core density = 0.92 gm/cm^3.

The time-dependent behavior of the neutron density, including one average group of delayed neutrons, is considered. A numerical finite difference solution of the three nonlinear differential equations (for neutron density, precursor density, and temperature) yielded a value of 54.4 Mw for the peak power at time equal to 204 ms and a total energy release of 1.71 megajoules. The resulting average temperature rise was 71.3°C, and the temperature rise at the center of the core was about 110°C. The total dose to a person standing next to the reactor was calculated to be about 1 rem. The prediction that the core does not melt and that the fission products are contained within the core and primary and secondary containers is reasonable, since polyethylene does not melt below 200°C. The power excursion is self-limiting because of core expansion due to the temperature rise. This is strongly dependent on the magnitude of the temperature coefficient of reactivity. The measured value of this coefficient is given as $-2.5 \times 10^{-4}\,°C^{-1}$,* whereas the value used in the calculation was $-3.6 \times 10^{-4}\,°C^{-1}$. A higher temperature could therefore be reached during the runaway, but it is estimated to be considerably less than that needed to melt the polyethylene.

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*ibid., p. 169.

ACKNOWLEDGEMENTS

Acknowledgement is made to Aerojet-General Nucleonics for their permission to reproduce certain figures and calculations presented in the following AGN publications:

1. Reactor Hazards Summary Report for the AGN-201 Nuclear Reactor by AGN Staff, AGN-23, Revised April 1, 1959.

2. Elementary Reactor Experimentations, by A. T. Biehl <u>et al.</u> (Oct 1957).