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N-REACTOR HAZARDS SUMMARY REVIEW
(Phase Ib, Production Only)
Volume 2
REACTOR AND REACTOR AUXILIARY
SYSTEMS

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N-REACTOR HAZARDS SUMMARY REVIEW
VOLUME II
REACTOR AND REACTOR AUXILIARY SYSTEMS

(Sections 3, 4 and 5)

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3. NUCLEAR PROCESSES AND CHARACTERISTICS

3.1 Basic Physics Processes

H-Reactor is a converter—the principal nuclear processes are the transmutation of U^{238} to Pu^{239} and the fissioning of U^{235} . The heat released in the fissioning process may ultimately be used to produce electrical energy, and the reactor and fuel were designed to permit this.

Fission neutrons are born in the reactor fuel with an average energy of two Mev. To be useful as chain carriers, they must be slowed to thermal energies. The principal slowing down process for neutrons in H-Reactor is elastic scattering with the carbon nuclei of the reactor moderator. The cooling water also contributes to the moderating process.

In a graphite moderated reactor, the ^{multiplication factor} ~~multiplication factor~~ is maximized at a particular carbon-to-uranium ~~ratio~~ ratio. If the C:U ratio is decreased sufficiently below the optimum reactivity point, the chain reaction ~~cannot~~ cannot be sustained with natural uranium. The C:U ratio at H-Reactor is 33 (compared to 75 at the K Reactors) and a uranium enrichment to 0.947 per cent U^{235} by weight is required to sustain the chain reaction with reasonable excess reactivity for accommodation of operational reactivity transients.

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Unlike the multiplication factor, the absorption of neutrons in U^{238} increases as the O:U ratio is decreased. At H-Reactor, the relatively low O:U ratio of 33 produces the relatively high resonance neutron capture probability of 0.17.

With the low O:U ratio, the water coolant in the core is worth more as a moderator than as a poison. Its removal reduces the moderating efficiency and increases the resonance capture probability. (The resonance capture probability in the dry H lattice is 0.23). Therefore, loss of coolant results in a reduction in reactivity. Conversely, a cold water injection at equilibrium reactor power level (which would increase the amount of water in the process tubes), and a flooding of the graphite structure can add reactivity.

The relative nuclear reaction rates are functions of the concentrations of each material present, the temperatures of these materials, and the average energy (temperature) of the neutrons. The dynamic equilibrium among the various competitive nuclear reactions will change as new materials (fission products and transuranic elements) are formed and as the temperature of core materials changes throughout a transient. The principal operating reactivity effects are those associated with the buildup of Pu^{239} and burnout of U^{235} , buildup of fission products Mo^{135} and Sr^{149} , and heating of moderator, coolant and uranium.

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3.2 Lattice Characteristics*

3.2.1 Description of Lattice

Lattice dimensions are given in Table 3.1 and a cross-sectional view of the lattice is shown in Figure 3-1.

TABLE 3.1

LATTICE DIMENSIONS

Spacing	8-in. horizontal x 9-in. vertical	
Equivalent Cell Diameter (inches)	9.575	
Fuel Element Diameter (inches)	<u>Outer</u>	<u>Inner</u>
OD Clad	2.406	1.249
OD Fuel	2.354	1.166
ID Clad	1.764	0.438
ID FUEL	1.819	0.492
Process Tube Diameter (inches)		
OD	3.258	
ID	2.708	

3.2.2 Cold, Clean, Green Lattice Parameters

The cold, clean, green lattice parameters for the H-reactor have been determined experimentally for a prototypic lattice.⁽¹⁾

*Several special terms are commonly used to describe lattice characteristics. Among them:
 Cold - Graphite, coolant, and fuel at or near room temperature
 Hot - Graphite, coolant, and fuel at or near equilibrium operating temperature.
 Clean - No short-lived, fission product poison present.
 Dirty - Short-lived fission products present.
 Green - Fuel at low or zero total exposure.
 Ripx - Fuel at or near goal exposure.

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3.2.2.1 Measurements Made in the Physical Constants Testing Reactor (PCTR)

Two lattice arrangements were studied. One was as exact a mockup of the actual lattice as possible; the other was a "condensed" lattice in which the voids in the graphite stack were removed but the same graphite-to-uranium ratio was maintained. The lattice parameters were measured in each arrangement to evaluate the neutron streaming effect caused by the large void areas.

3.2.2.2 Exponential Pile Measurements

The material buckling for the mockup and condensed lattices was measured in an exponential pile. The measurement of the material buckling and k (in PCTR) enabled a quantitative determination of the migration area. The migration area deduced for the condensed lattice is $300\frac{1}{2}\text{cm}^2$; for the mockup lattice, 646 cm^2 . The ratio of the two values is 2.15 which is in good agreement with the ratio predicted theoretically.

3.2.2.3 Best-Value Lattice Parameters

The availability of experimental parameters for a lattice almost identical to the H lattice makes it possible to normalize theoretical lattice parameters and determine best values for the actual H lattice. The best value cold, clean, green lattice parameters for H-Reactor with 0.947 wt per cent U^{235} enriched fuel are given in Table 3.2.

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TABLE 3.2
BEST-VALUE, COLD, CLEAN,
GREEN LATTICE PARAMETERS*

<u>Parameters</u>	<u>Value</u>
k_{∞}	1.0778
η	1.4184
ϵ	1.0411
ρ	0.8291
ρ	0.8003
ρ_{∞}^2	115.6 μ b

*No cross-cooling tubes included. Effect would reduce k_{∞} to 1.0752.

3.3 Core Characteristics

3.3.1 Description of Core and Loading

The H Reactor core is comprised of 1004 lattice units which can be increased to 1072. This core is pierced by process channels, steam vents, horizontal control rod channels, the Ball 3X Safety System channels, and the graphite moderator cooling tubes. These void systems reduce the core graphite density from a nominal 1.7 g/cm³ to an effective 1.3 g/cm³.

The physical dimensions of the core are set by the lattice geometry and the ~~min~~ charge length. The core width is 22.7 ft, the core height is 24.0 ft, and

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the core length is 35.0 ft. The corners are truncated as shown in Figure 3-2, resulting in a core equivalent radius of 12.6 feet. This roughly cylindrical arrangement contains 692.0 tons of graphite and 366.5 tons of uranium.

The core is surrounded radially by a reflector of high-purity graphite which is infinite in effect. The front and rear "reflectors" are composed of graphite plus other material (e.g., process tube, water, and fuel spacers).

3.3.2 Slowing Down and Diffusion of Neutrons

As one consequence of the relatively low C:U ratio, the ratio of fast-to-thermal flux is estimated to be $2.50^{(2)}$, a relatively high number compared to existing graphite-moderated, reactors.

The lattice migration area, M^2 , deduced from an exponential experiment is 646 cm^2 ; this is separable into a Fermi Age (τ) of $\sim 480 \text{ cm}^2$ and a Diffusion Area (L^2) of $\sim 160 \text{ cm}^2$ at room temperature. At operating temperature M^2 increases by ~ 10 per cent.

3.3.3 Neutron Streaming

The large void fraction in the reactor core allows preferential neutron passage along the void channels. This "streaming" affects various lattice parameters, in particular τ and L^2 and thus reactor leakage and multiplication. If the graphite were

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For the fully-loaded, green reactor of about 1.060 at room temperature. The effective

Both three-group and modified one-group calculations indicate a k_{eff}

τ = Prompt Age

L^2 = Lattice diffusion length squared, and

B_g^2 = Billo geometric buckling,

k_∞ = infinite lattice multiplication,

k_{eff} = effective pile multiplication,

where:

$$k_{eff} = \frac{k_\infty}{1 + B_g^2 L^2} \approx \frac{k_\infty}{1 + B_g^2 \tau} \approx \frac{k_\infty}{1 + B_g^2 (L^2 + \tau)}$$

by the equation:

Concomitant loading, the effective and infinite lattice multiplication factors are related

neutron multiplication area and the characteristics of the reflector. For a uniform, homogeneous

The neutron leakage in a reactor is a function of the reactor size, the

3.3.4. Neutron Leakage - Loss

and leakage.

With axes) the overall pile reactivity would be increased by 0.8 per cent due to decrease

"General" over the core (relating the same number of graphite atoms but filling all

geometric buckling may then be determined from a rearrangement of the previous equation.

$$B_G^2 = \frac{k_{\infty} - k_{\text{eff}}}{k_{\text{eff}} (\lambda^2)}$$

Assuming "best values" of the parameters as given in Table 3.2 and a

k_{∞} for the fully loaded reactor of 1.060, the total geometrical buckling is 22.1

microbarns
~~microbarns~~ (1 $\mu\text{b} = 10^{-6} \text{ cm}^{-2}$).

The front and rear reflector augmentation at N-Reactor as estimated from a cosine flux distribution and the longitudinal buckling component is 7.3 μb ; this indicates a radial buckling component of 14.8 μb and a radial reflector augmentation of ~240 centimeters.

The large reflector savings are due to the predominantly "fast" neutron spectrum in the core; epithermal neutrons emerging from the core into the reflector are quickly thermalized. In the thermal state, there is a good probability that the neutrons will be transferred back to the core. There is actually a net gradient of thermal neutrons from the reflector inward to the core.

It should be noted that the reflector augmentation as defined by the geometrical buckling is extremely sensitive to the total pile reactivity; an error 1/100

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of only 0.2 per cent $\Delta k/k$ in the determination of k_{eff} can result in an error of over 60 cm in the estimate of δ , the reflector savings. Best estimates for the effective reactor size (in terms of neutron boundaries) are

Effective radius = 625 cm

Effective length = 1160 cm

These dimensions are based upon $k_{eff} = 1.060$ at room temperature.

3.3.5 Core Power Distribution

The reflector augmentation results in a high flattening efficiency in the reactor, even without the smoothing effect of control rods or other flattening agents. Theoretical flattening efficiencies of 80 to 95 per cent are calculated for routine operation, with actual efficiencies (including fine structure) of 70 to 90 per cent. The best estimate for the flattening efficiency is 90 per cent. Radial power distributions for various flattening efficiencies are plotted in Figure 3-3.⁽³⁾ The axial power distribution will follow an approximate cosine shape with a peaking upstream of pile center due to the higher coolant temperature towards the outlet end of the reactor.

3.4 Reactivity Transients

3.4.1 Temperature Coefficients of Reactivity

Temperature changes affect reactivity by changing the nuclear properties

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and the physical properties of the materials. The fuel temperature and coolant temperature coefficients of reactivity are negative; the graphite temperature coefficient is positive. Other physical effects, such as fuel expansion, are too small to produce any significant temperature coefficient.

The integral graphite temperature reactivity effect is shown in Figure 3-4, as a function of local graphite temperature. The slope of the curve is the differential coefficient, $1/k \, dk/dT$. To obtain a reactor-averaged coefficient, the time-averaged weighted graphite temperature distribution over the active volume of the reactor is volume-averaged. The graphite coefficient is positive for the exposures shown up to 1400 kld/ton; moreover, the differential coefficient becomes more positive with increasing exposure. The hot differential coefficient is less than half the cold value.

The integral coolant temperature reactivity effect is shown in Figures 3-5 and 3-6 as a function of the inlet coolant temperature. The magnitudes of the differential coefficients (slopes) are constant with temperature for zero power (Figure 3-5); but increase with temperature for the full power case (Figure 3-6). Thus, Figure 3-5 depicts reactivity behavior with coolant temperature during a reactor startup. At zero power, the coolant temperature is constant along the length of the tube. Figure 3-6 shows the reactivity effect of varying the coolant inlet at full reactor power. In

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this case, the average coolant temperature would be approximately the mean of the inlet and outlet temperatures. (The graphite and coolant temperature coefficients are temperature-independent).

The reactivity effect associated with increasing the fuel temperature (Doppler effect) is shown in Figure 3-7.⁽⁴⁾ The differential Doppler coefficient is dependent on the square root of the absolute fuel temperature. The curve in Figure 3-7 is plotted against the average fuel temperature, and is based on an experimental determination in the PCRR using dry H-Reactor fuel in the H lattice. The coefficient can be expressed as:

$$\frac{1}{\beta} \frac{d\beta}{dT} = -5 \times 10^{-4} (T^{\circ}K)^{-1/2} / \text{°K}.$$

2.4.2 Exposure-Dependent Effects

The reactivity effects associated with exposure include the effects of U^{235} -depletion; Am^{239} , Pu^{240} and Pu^{241} -formation; and fission product formation. The net reactivity effect of U^{235} -burnout and plutonium formation is initially positive. This increase results from the higher fission cross section and neutron yield of Pu^{239} compared to U^{235} . The net exposure effect, including all fission products, is negative. The following ~~subsequent~~ sections deal with the various exposure-dependent effects on reactivity.

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of the amount of nitrogen captured, then decreases to zero due to known decay.

The known nitrogen initially increases due to the decay of the precursor ^{14}C and the

value of the nitrogen exposure of 50 kg/ton (2.5 days). After several minutes,

after a reactor starting, nitrogen exposure increases the equilibrium

3.4.2.2 Nitrogen Exposure

positively gain to an average exposure of 700 kg/ton of 10^{-3} .

models (but including nitrogen, nitrogen holding, and scattering), indicate a cold, long term

formation and assumption of an average cross section of 50 barns per nitrogen isotope

exposure will be a compensating factor. Calculations based on ^{14}C -nitrogen and nitrogen

higher level enrichment will tend to decrease the long term gain, but the higher nitrogen

positive to nitrogen level up to an exposure of 500 kg/ton . In addition, the

in the other nitrogen isotopes, the cold, long-term values to

the temperature dependence.

at nitrogen, since the temperature dependence of these exposure effects is small.

and also, called the long term gain. The long term gain is given near zero for the

nitrogen formation (other than nitrogen and scattering) and for present purposes could be zero

The effect of nitrogen formation, ^{14}C -nitrogen and nitrogen

3.4.2.1 Nitrogen Exposure (1952)

typical startup and shutdown xenon transient is shown in Figure 3-8 for an average fuel exposure of 700 kld/ton. The exposure dependence of the equilibrium operating xenon poisoning is shown in Figure 3-9. The equilibrium xenon in R-Reactor will be worth 2.7 to 3.1 per cent $\Delta k/k$.

3.4.2.3 Samarium Poisoning

Sm^{149} poisoning reaches an equilibrium value of about 6.5×10^{-3} ~~microkld/tonne~~ $\Delta k/k$ 100 kld/ton exposure. After each reactor shutdown, the samarium poisoning increases due to the decay of its precursor, Eu^{148} . The equilibrium value is regained by the time an additional exposure of 100 kld/ton is reached after each startup. The equilibrium samarium is shown in Figure 3-9 as a function of fuel exposure.

3.4.2.4 Neptunium Holdup

Calculations of the reactivity gains in a reactor due to plutonium buildup usually assumes instantaneous formation of plutonium. The fact that the formation of Pu^{239} is delayed through the 2.33 day, beta-decay of Np^{239} makes it necessary to correct the calculated reactivity gains. This is most conveniently done by assigning a reactivity "loss" for the neptunium holdup effect. The holdup loss at equilibrium is about 4×10^{-3} . The equilibrium neptunium holdup effect is shown in

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Figure 3-9 as a function of exposure.

3.4.2.5 Overall Long Term Reactivity Effects

The total, long term, exposure-dependent, reactivity effects are given in Figures 3-10 and 3-11. Figure 3-10 illustrates the transients obtained by assuming a uniform fuel exposure throughout the reactor; Figure 3-11 illustrates the more realistic assumption of a "continuous" discharge scheme, where fuel exposures are uniformly distributed from zero to the goal exposure on the graph. On both figures, the cold, long term gain curve represents the long term reactivity behavior of the cold, xenon-free reactor which, however, accumulates exposure under normal operating conditions. The hot curve represents the long term behavior of the hot operating reactor, which thus includes the exposure-dependence of xenon poisoning. Both curves include the effects of plutonium buildup, U^{235} burnout, neptunium holdup, samarium buildup, and other fission product poisoning. Since both the cold and hot curves are normalized to zero reactivity⁶ change at zero exposure, the absolute values of the overall temperature coefficient and xenon poisoning are not shown; but the combined affects of variations in these parameters with exposure is illustrated by the deviation of the hot from the cold curve as exposure progresses.

3.4.3 Operational Reactivity Transients

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The various temperature and exposure reactivity contributions have

been combined to determine typical reactor startup and shutdown reactivity transients. To facilitate the calculation of operational transients, the coolant and doppler coefficients are combined in more usable form. Since coolant and fuel temperatures are not separable in practice, the coefficients are expressed first in terms of the effect of inlet coolant temperature changes on the average coolant temperature and average uranium temperature, and also in terms of the effect of power changes on the average coolant and average uranium temperature within the core. Table 3.3 thus gives the integrated effects of changes in coolant temperature, graphite temperature, and reactor power from the fully shutdown condition up to equilibrium. Table 3.4 gives the differential temperature and power coefficients for changes near the equilibrium operating point.

TABLE 3.3

INTEGRATED REACTIVITY CHANGES, COLD TO HOT

	<u>Green</u>	<u>700 Mw/T Pile Average</u>
Graphite	+2.8%	+3.0%
Coolant Inlet*	-2.4	-2.0
Power**	<u>-1.2</u>	<u>-1.1</u>
Net	- .8%	- .1%

*Includes uniform changes in uranium temperature due to coolant temperature changes.

**Includes coolant and uranium temperature changes associated with power changes (the effects of the ΔT across the reactor).

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TABLE 2.1

DIFFERENTIAL TEMPERATURE AND POWER COEFFICIENTS

	<u>Graphite</u>	<u>700 123/F Pile Inlet</u>
Graphite	$+0.0011\%/^{\circ}\text{F}$	$+0.0010\%/^{\circ}\text{F}$
Coolant Inlet*	$-0.0096\%/^{\circ}\text{F}$	$-0.0079\%/^{\circ}\text{F}$
Power**	$-0.0003\%/1\%$	$-0.00026\%/1\%$

*, ** - See footnotes, Table 3.3

3.4.3.1 Startup Reactivity Transient

A reactivity transient for a startup from an infinite outage (zero xenon) is shown in Figure 3-12. The initial rapid reactivity loss of about 2.5 per cent results from increasing the primary loop inlet temperature from shutdown conditions to approximately the normal operating inlet temperature (about 370 F). This increase is accomplished while the reactor power level is held at about ten per cent. After two hours, the power level is increased to equilibrium, thus increasing the average coolant and uranium temperatures and causing a further reactivity loss of about 1.0 per cent. Reactivity then increases due to graphite heating, until at about five hours the increasing negative ^{poison}~~power~~ effect of xenon buildup overcomes the positive graphite temperature ^{reactivity} effect; and the reactivity subsequently decreases until the xenon reaches an equilibrium value.

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3.4.3.2 Normal Shutdown Transient

Typical reactivity shutdown transients are given in Figure 3-13 for the reactor loaded with green fuel and for the reactor loaded with fuel at an average exposure of 600 Mxd/ton. The initial equilibrium reactivities are those expected with 0.947 per cent enriched fuel-actually the reactivity carried in control rods at equilibrium will be nearer 1 per cent, so the entire set of curves can be moved down about 1 per cent. To be consistent with the startup transient, the reactivities are shown as they would be expected to be with the full 0.947 per cent enrichment.

The assumed power reduction in Figure 3-13 is 2 1/2 per cent per minute. The initial, small reactivity peak is due to the initial cooling of the fuel and coolant. Reactivity then decreases for about an hour as the graphite cools and xenon builds up. During this period, the average coolant temperature is held constant at about 336 F. Further cooling of the coolant and graphite to shutdown temperature (100F) and further xenon buildup produces the transient shown between 1 1/2 hours and 10 hours after starting down. After 10 hours, the xenon decay becomes predominant and the reactivity increases to the cold pile values of 5.9 and 5.1 per cent, respectively, for the green and ripe (700 ^{MWD/T Average} ~~MWD/T Average~~ exposure) cases.

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3.4.3.3 Scram Transients

The reactivity transients associated with a reactor scram and with scram recovery are also shown on Figure 3-12. The initial reactivity increase is due to the combined effect of reducing the coolant and uranium temperatures to about normal inlet inlet temperature (~370 F). The subsequent rapid reactivity decrease is due to graphite cooling and xenon buildup. The reactivity stops decreasing, and begins to increase, when the reactor power has been increased far enough (to about 75 per cent of equilibrium) so that the graphite temperature again begins to increase and heat begins to come out. The reactivity then slowly approaches its normal equilibrium value.

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interlock and ⁵ power setback trips aid in fulfilling this function. The system does enhance reactor safety in that it minimizes the frequency at which the reactor alarm circuits are "tested" with true fast periods, by providing a visual indication of approach-to-trip conditions.

4.1.2 System Description

Two independent channels of low-level K_{eff} neutron flux monitors are provided. Each channel utilizes a fission chamber detector which is located in the top reactor shield near the reactor centerline as shown in Figure 4-1.

The penetration of the chambers into the reactor shield is remotely adjustable, thus permitting the useable range of the instrument (nominally 5 decades) to be adjusted downward for maximum sensitivity, or upward to insure overlap with the intermediate range flux monitoring system.

Each fission counter feeds into a log count rate amplifier which gives a five-decade log count rate meter and recorder, and thence into a period monitor. The log count rate recorder incorporates contacts which automatically initiate chamber withdrawal when a full-scale reading is approached.

The period monitor output is displayed on a period meter which is calibrated from a positive 10 second/a ^{to} negative 100 second period.

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signals from the two independent channels are "auctioneered," and the fastest period is displayed in a single meter at the reactor control console. The period reactor output is monitored by two independent electronic trip switches. The trip switch initiates power setback action (controlled, slow, rod insertion) if the rising period becomes shorter than the pre-set trip point (probably 30 seconds but adjustable between 10 and 60 seconds). The other trip switch actuates the fast-rod-withdrawal interlock circuit, ^{Thus} ~~the~~ inhibiting control rod withdrawal at the fast (30 rods per minute) speed if the rising period becomes shorter than the preset trip point (probably 40 seconds but adjustable between 20 and 60 seconds). Fast control rod withdrawal is also inhibited by a contact on the log count rate recorder which trips at count rates less than 10 counts per second, indicative of possible channel malfunction.

4.1.3 System Operation

The subcritical monitoring system will be operated by first inserting both chambers to obtain count rates greater than 10 cps with all rods inserted, thus clearing the fast rod withdrawal interlock. Rods will then be withdrawn to approach critical. The subcritical count rate will increase by a factor of at least 10 before critical is reached. As this occurs, one of the chambers will be manually

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~~the~~ withdrawn, so that ~~the~~ count rates on the two channels differ by about a factor of 10. As rod withdrawal continues, both channels will indicate an apparent rising period which will become shorter as critical is approached, finally activating the fast-rod-withdrawal interlock. Rod withdrawal will then continue at the slow speed (6 feet per minute per rod) until a supercritical rising period no shorter than 30 seconds is established. As the count rate on each channel reaches about 10^5 cps, the channel will be automatically withdrawn until a predetermined limit point is reached, which will be adjusted to yield a count rate of about 100 cps. While the channel is being withdrawn, the period meter will indicate a falling period; however, since the channels will be staggered in count rate, the other channel will continue to register the true period. If an excessive number of control rods are withdrawn, thus achieving a rising period shorter than 30 seconds, the power setback circuit will be actuated, inserting a pre-determined rod pattern ∇ until the period is again slower than 30 seconds.

4.2 Intermediate Range Flux Monitoring System

4.2.1 System Description

Three independent channels of intermediate range neutron flux monitors are provided. Each channel utilizes a gamma-compensated

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detector located in the top reactor shield near the reactor centerline as shown in Figure _____. The position of each chamber will be manually adjusted to make the channel sensitive to neutron flux levels equivalent to a reactor power level of about 100 watts. Reactor kinetics studies indicate that this sensitivity combined with a 15-second period trip setting and the response characteristics of the reactor safety system, would be adequate to limit a reactor startup excursion resulting from the maximum credible positive reactivity ^{Ramp} ~~change~~ (generated by control rod withdrawal), so that no fuel or coolant temperature increase would be detected.

The signal from each chamber will be fed to a log level amplifier with a useful range of ~~XXXXX~~ seven decades. This amplifier will also be capable of being driven beyond its ^{use} ~~useful~~ range without damage, thus permitting continuous operation of the intermediate range instrumentation during equilibrium operation, and assuring its availability for period trip protection in the event a power setback reduces the reactor power level to within the useful range. The log level amplifier output is fed to log level meters, to a recorder, and to a period amplifier.

Each log level amplifier signal output is monitored by three independent electronic trip switches. Each perform a different function

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- 1) One trip switch will be set at about .01 MW reactor power level, and will actuate the fast-rod-withdrawal interlock in two-out-of-three coincidence with the similar trip switches from the other two channels.
- 2) Another trip switch will be set to trip the reactor safety circuit at a power level of about two percent of equilibrium. This trip level is chosen to be above the minimum sensitivity of the high level flux monitoring system. Again, a two-out-of-three ~~XXXX~~ coincidence will be used.
- 3) The third trip switch will be adjusted to about .1 MW reactor power level, and utilized in conjunction with a time-delay relay to provide capability of detecting an abnormally high power level after a reactor scram, and initiate the backup (Ball 3X) safety system action. Two-out-of-three coincidence will be used.

Each of the three log level amplifier outputs will be fed to a period monitored^{ing} amplifier. The output of the period amplifier will be displayed on a meter calibrated from a positive period of ten seconds⁵ ~~to~~ a negative period of 100 seconds. A remote, "auctioneered," meter at the reactor control console will display the fastest period from the three channels. Each period amplifier output will be monitored by two independent electronic trip ~~detectors~~ ^{One of these}

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trip switches trips the power setback circuit/^{if} the rising ~~period~~ should become shorter than the preset trip point (probably 25 seconds, but adjustable between 10 and 40 seconds). Two-out-of-three coincidence from all three channels is utilized. The other trip switch would trip the reactor safety circuit and scram the horizontal rod system. This trip is also adjustable between a 10 and 40 second period, and will be set at about a 15 second period. Two-out-of-three coincidence is again employed.

4.2.2 System Bypass Provisions

Key-locked bypass switches are provided for each trip function, but not for each individual trip contact. That is, the output of a two-out-of-three logic circuit may be bypassed, but not the individual channels which are used in the coincidence circuit. Thus, removal of a single channel from service, and instrument failures which cause spurious trips, will count as one trip in the coincidence circuitry, and a trip from either of the remaining two channels would complete the trip function. The use of the key-locked bypass switches is restricted by administrative control; and, in general, is allowable only during reactor shutdown to permit equipment maintenance. An exception to this practice is described in Section 4.2.4.

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4.2.3 System Accuracy and Response

The primary function of the intermediate range nuclear instrumentation is to provide an indication of the exponential rate of power increase at low (.01 to about 100 MW) power levels. A secondary function is to provide a reasonably accurate indication of the true power level at about two percent of equilibrium power. At this level, the high level flux monitoring system ~~(instrumentation)~~ (Section 4.3) will assume the monitoring and reactor safety functions, and a comparison of the ~~two~~ indications from the two systems will be made to verify operability of the high level system.

The accuracy of the log level signal derived from the compensated ion chambers in representing true reactor power level is primarily dependent on the power (flux) distribution within the reactor core, which in turn depends on the inserted control rod pattern, ^{fuel} ~~rod~~ loading pattern ~~(if non-uniform)~~ (if non-uniform), and exposure distribution in the loaded fuel. Under normal operating conditions, these factors are expected to introduce an uncertainty in the log level indication of about 20 percent. However, it is credible that improper control rod positioning could introduce flux distortions which would introduce uncertainties in the log level indications greater than a factor of two. ~~The degree of accuracy is, however,~~

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~~it is credible that improper control rod positioning could introduce flux~~
~~distortions which would introduce uncertainties in the log $\bar{P}$ level indications~~
~~greater than a factor of two. This degree of accuracy is, however, acceptable,~~
adequate)
^ since the useful range of the instrumentation is limited by the scram trip
setting at an expected level of about two percent of equilibrium, where a factor of
two error is acceptable. At this level, the distortion would be detected by other
reactor instrumentation (tube outlet temperature, high level flux monitors, and
incore flux monitors). The proper functioning of the period indication derived
from the intermediate range instrumentation would not be inhibited by such a
power distortion, although the minimum useful sensitivity of the instrumentation
could be reduced by the degree of the distortion, or, say, by a factor of two.
Reader kinetics calculations indicate that increasing the power level at which a
period trip from the intermediate range instrumentation is effective by a factor
of two, or even of ten, would not substantially decrease the degree of reactor
protection afforded. In the latter case, with a worst-credible startup excursion,
a coolant temperature increase of about 20 to 30 F would be observed.

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channel will thus not affect the remaining two channels.

With a.c. power through independent circuit breakers; an internal fault in one

Each of the three channels of instrumentation is separately supplied

4.2.4 System Reliability

as described in Section 4.3.

protected against excessive power rates by the high level flux monitoring system, within the useful range of the instrumentation. At higher power levels, the reactor

of seven (e^2). This increase is not significant at the low reactor power levels

during which time the reactor power would increase by no more than a factor

setting and a ten second rising period, the trip would occur within 20 seconds,

second or two periods, whichever is longer, e.g., with a 15 second period trip

rising periods shorter than the trip setting a trip will be obtained within one

The response of period instrumentation trips is such that for all

trip period by more than .005."

of the indicated period in seconds shall not deviate from the reciprocal of the

specified in terms of the reciprocal period in seconds; that is, "the reciprocal

two seconds compared to a true reactor period of 20 seconds. The accuracy is

The period indicated by the period monitor will be within about

accuracy of 2%

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designed to be fail-safe; power supply failure results in a trip indication.

Since the three detectors are located close together near the ^{center} of the top reactor shield, they will "see" essentially the same local reactor flux changes.

An analog comparison circuit is provided which will continually compare the magnitude of the three signals, and annunciate a downscale deviation in any single channel.

4.2.5 System Operation

The intermediate range flux monitoring system is expected to be "on" at all times; it will be removed from service for maintenance only, and the $\frac{1}{2}$ trip functions would be bypassed only if maintenance is to be performed in conjunction with ~~REACTOR~~ maintenance or testing of the reactor safety circuit. The only exception to this general rule is the trip function which monitors the log ~~level~~ level signal, and which is set to scram the reactor at about two percent of equilibrium power level. It is necessary that this trip be bypassed to reach equilibrium power level. An interlock is provided which prevents ~~RESETTING~~ resetting the reactor safety circuit, and thus prevents rod withdrawal, to insure that this trip is unbypassed prior to each reactor startup.

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4.3 High Level Flux Monitoring System

4.3.1 System Description

The high level system consists of twelve independent primary flux monitoring channels. Each channel utilizes an ion chamber detector. Six chambers are located above the reactor core, and six below, as shown on Figure 5-2. The penetration of each chamber into the flux shield is manually adjustable, and the detectors will be adjusted to make the channel sensitive to a reactor power level of about 0.5 percent of equilibrium. This will assure adequate overlapping coverage with the intermediate range instrumentation. The range of the system will extend to about 20 percent above the normal equilibrium level. Each channel consists of an ion chamber, amplifier and independent power supply, electronic trip switch, linear meter, and recorder. The output of all twelve channels is also printed out by a twelve-point recorder at the reactor control console.

The electronic trip switch will be set to scram the horizontal rod safety system at about 120 percent of the normal equilibrium level. A two-out-of-twelve coincidence logic circuit is employed to minimize the possibility of spurious reactor scrams caused by component failures. Each of the twelve individual channel recorders contains an adjustable micro-

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These switches will be set to trip at about 10 to 15 percent above the normal signal; a two-out-of-twelve trip logic circuit will be employed, and will actuate the power setback circuit.

The output current of the twelve individual amplifiers is averaged by four independent electronic averaging circuits. The four resultant identical signals are utilized as follows:

1) One average signal is used to drive a meter and recorder.

2) Each of the remaining three average signals is monitored by three

independent, electronic trip switches. One set of three trip switches

(one trip switch monitoring each average signal) will be set to sense

the reactor primary safety circuit (horizontal rods) at about 110 percent

of normal equilibrium level using a two-out-of-three coincidence logic.

The ^{Second} ~~remaining~~ set of three trip switches will be set at about ²⁵ ~~115~~ percent

of equilibrium level and used in conjunction with a two-delay relay to

provide a backup ball-3% trip should an abnormally slow rate of flux

decay after scram be detected.

The third set of three trip switches will be set at a convenient level between 100 percent and 110 percent of the equilibrium signal, and used in two-out-of-three coincidence to trip the power setback circuit.

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СЛУШАЮЩИЕ: ПО ВОПРОСУ ПОДПИСАНЫ ОСТАВЛЯЮТ ПРАВО, ПРИНЦИПИАЛЬНО ЗАДАТЬ ВОПРОСЫ-ПОСЛЕДСТВИЯ

Այս 30 օրվա ընդամենը օդ ընթրվում են հազարից քսանհինգ հազար-հինգ

4.3.2 System Reports Provision

reached almost as positive prior rates (about 250 Ht/min).

connected in two-out-of-three coincidence and will be set to activate the power

ԵՄԼԻ ԱՆՆ ՍԵՐՈՒՄԱՆ ԴԻՆԶ ՄՈՒԼԻՍ-ՇԵՐԵՄԻ ՄԵՐԵՆՈՒՄԻՆԻՍ ԿՐԵՆՈՍՈՆԱՆ ԵՐԵՎԱՆ

(Note) A 1000 watt heater would heat about 300 ml./minute, the three power

TRANSMISSION) APPROXIMATELY 600 METERS OF TRANSMISSIONS BEING-TO-BE-ONE IN

at a ratio between 150 and 400 lb./cubic ft. These three trip switches are connected

...the ... of ...

SECRET - ALL INFORMATION CONTAINED HEREIN IS UNCLASSIFIED EXCEPT WHERE SHOWN OTHERWISE

(continued) Attached are 52 unclassified reports which were not previously (continued)

TO THE PRESIDENT/6 MONTHS REVIEW OF THE 1980-1981 YEAR: 1. International Year of the

'STUNTERS ARE TO BEAT OUT' OF THE MARKET BY THE NEW 'STUNTERS'

NE PASSEZ PAS CEUX DE LA MONTAGNE TEMPERATURE SANS QUE VOUS NE SOYIEZ PAS

channels in a reactor quarter (top-front, bottom-rear, etc.) can be concurrently bypassed, nor no more than two of the four channels on one side of the reactor (left or right) can be bypassed concurrently. Inadvertent (or deliberate) excessive bypassing beyond these limits will initiate a time-delayed (about 30 to 60 seconds) scram of the primary safety system (horizontal rods).

It is expected that one or a few individual high level channels/ scram trips will be bypassed for maintenance purposes during reactor operation, with excessive bypassing prohibited by the ^{logic} ~~the~~ arrangement described above. The two-out-of-three scram trip logic circuits on the average of the twelve channels will not be bypassed during reactor operation. The two-out-of-three scram-trip logic circuit on the three power rate channels will be ~~xxx~~ routinely bypassed once equilibrium level is attained, because of the sensitivity of rate-type circuits to short ^{few} ~~few~~ (cycle) power line transients which would not otherwise require a reactor scram. Mandatory procedures will, however, require that this trip be unbypassed if the power level is significantly (amount to be determined) reduced below the equilibrium level. Further, an interlock is provided in the safety circuit reset circuit to prevent resetting the safety circuit, and thus prevent withdrawal of control rods, unless the power rate trip is unbypassed.

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The power setback circuit is not considered to be a reactor safety circuit; while the bypassing scheme recommended will in general be the same as that required for the scram trips, enforcing of power setback bypass restrictions will be at the option of the plant operating management.

4.3.3 System Accuracy and Response

The high level flux monitoring amplifiers are designed to be accurate to within plus or minus one percent of full-scale, with a permissible drift in any 1000 hour period of plus or minus one-quarter percent. The electronic trips switches are ~~HM-76930~~ settable to an accuracy of better than one percent of full scale, with a permissible drift in any 1000 hour period of less than one percent. The accuracy of the averaging circuits is within two percent of the true average of the twelve current input, with a permissible drift of plus or minus one-quarter percent. The electronic trip switches which monitor the average signals are settable to an accuracy of better than one percent of full scale, with a permissible drift of plus or minus one-quarter percent. The accuracy of the indicated power ^{change,} rate-of-~~HM-76930~~ compared to the true rate-of-change of the input signal to the rate circuits, is equivalent to plus or minus about 50 %/minute, which includes rate circuit drift, gain stability, and the accuracy of the calibrating ramp generator.

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System Response

The time constant of the trip circuits of the twelve individual high level channels, in responding to a step change in signal current from the chamber, is 20 milliseconds. The time constant of the signal averaging circuits in responding to a step change in the time average output current from the twelve high level amplifiers is 20 milliseconds. The time constant of the power rate circuits is one second. The time constant is defined as the time required for a circuit to indicate, or respond to, 63 percent of a step change in the input signal.

4.3.4 System Reliability

Each one of the twelve high level flux monitoring channels is furnished AC power through a separate circuit breaker; thus, a fault in the circuitry of one channel which opens a breaker would not disable other channels. Further, the loss of AC power would result in opening all of the trip contacts, and is thus fail-safe. Similarly, each of the three average trip circuits, and each of the three power rate circuits, is separately powered, and their trip contacts are fail-safe upon loss of power.

An analog comparison circuit compares each of the twelve individual channel signals to an average signal; a significant difference between the signals

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(adjustable, and to be established) will be annunciated, and the faulty circuit is identified. ~~XXXXXXXXXX~~ This analog comparison thus will detect spurious down-scale signal shifts in the individual channels from any cause.

To detect non-safe failures in the electronic trip circuits which ~~LM~~ could inhibit trip action, each trip circuit will be pulsed at 1.2 second intervals with a signal of trip magnitude. The duration of the pulse is 200 microseconds, which is too short to allow the final trip relay to open. The pulse is reflected by the trip relay coil and the reflected pulse is detected; absence of a reflected pulse would indicate a fault in the trip circuit, and would be annunciated.

4.3.5 System Operation

The high level flux monitoring system will be "on" at all times during reactor operation/~~XXXXXXXXXX~~ The trip logic circuits and the bypassing logic circuit described in 4.3.1 and 4.3.2 make it impossible to establish safety circuit continuity, and thus withdraw rods for startup, unless at least eight high level channels are unbypassed and no more than one of these eight is tripped or removed from service. In addition, an interlock in the reset circuit requires that the power rate trip system be ~~by~~ unbypassed before the reset circuit operates.

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The high level flux monitoring system will only be sensitive to power levels greater than about 0.5 percent of equilibrium power; thus, there will be no meaningful signal presented by this system during the initial phase of reactor startup. The high flux trip on the intermediate range instrumentation performs the function of providing a check point in power level. This trip will not be bypassed until meaningful signals are observed on all operable high level channels.

The provision of power rate circuits and trips eliminates the necessity of range-changing in the high level flux monitoring circuits. In effect, the power rate trips prevent a rapid approach to the high level trip setting. The rate of rise will be limited by the rate trip settings, and in the high level trip settings are exceeded while the power is increasing at a rate slower than the power rate trip setting, there will be essentially no level signal due to the rapid response of the safety system. One exception to this general rule exists: During cold startups, while the total reactor flow rate is approximately 50 percent of normal, the two-out-of-three coincidence logic of the high level flux monitoring signal will be reduced by a factor of 10. This arrangement prevents, by gram action, any accidental or erroneous

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percentage as a function of the existing power level which is being maintained.

Process standards will reflect this by providing for a variable trip setting

trip setting could be permitted while still providing adequate protection.

increase were stopped at a level considerably below normal equilibrium, a wider

accidental power excursions. It is, of course, obvious that if the power

maximum possible positive reactivity ramp, provides adequate protection against

kinetic studies indicate that this trip setting, even in conjunction with the

signal be first set to not greater than 110 percent of the existing signal.

rates, it is necessary that the two-out-of-three trips on the average flux monitor

automatic trip protection against approaching the high level trips at excessive

planned power increase is completed. However, since this bypassing removes the

power limits and fluctuations, it is desirable to bypass these trips as soon as the

since the power rate excursions are sensitive to momentary

equilibrium.

increased to normal, this trip will be reset to ~~set~~ 110 percent of normal

steaming operation is established and the primary loop pressure and flow are

which, if unchecked, could exceed the reduced heat removal capacity. Once

4.4 In-Core Flux Monitoring System

4.4.1 General

The in-core monitoring system is not connected to the reactor safety (scram) circuits and it is not depended upon for reactor protection. The primary function of the system is to provide information to the reactor operator concerning the front-to-rear distribution of power.

4.4.2 System Description

Twenty-five independent channels of in-core monitors are provided. Each channel utilizes a small fission chamber which will be located in a side-to-side graphite moderator cooling tube. Installation is made through a compression-type Tee fitting and the chamber is in direct contact with the cooling water. Penetration depth is controlled by attaching the compression ferrule of the fitting to the chamber cable at a fixed distance from the chamber.

Power to operate the chambers is furnished by two 0-150 volt AC power supplies, each capable of supplying power to all of the chambers.

Current through the chambers will be indicated by 25 indicating meters in the Control Room. Range switches are provided to shunt part of the current around the meters. The switches will have eleven positions for sensitivity adjustment of each measuring system. Ranges will be

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adjustable from 50 to 1000 micro-amperes, full scale. At full reactor power about 750 micro-amperes will flow through the circuit.

A resistor of approximately 1000 ohms is in series with each indicating meter. The voltage drops across these resistors provide signals to two mt multipoint recorders in the control room. Zero suppression circuitry is provided for each channel.

4.4.3 System Operation

The in-core monitoring chambers will be located in positions to provide a front-to-rear flux level distribution pattern. At startup each circuit will be set on a high sensitivity range. As power level increases and water loading approach 100 percent the sensitivity will be decreased by means of the channel position range switches. It will be possible to expand the readable scale of any channel by suppressing the reading with the zero suppression resistor and placing the range switch in a more sensitive position.

4.5 Traveling Wire Flux Mapping System

4.5.1 General

The traveling wire flux mapping system is not connected to the reactor safety (scram) circuits. The function of the system is to obtain maps of

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the front-to-rear flux distribution at nine different places in the reactor.

The wires are located in a position which permits correlation of readings with the in-core monitoring system, and thus calibration of the flux monitor channels.

The two systems provide information to the reactor operator which assists him in maintaining optimum flux distribution.

4.5.2 System Description

Nine channels of ϕ flux mapping are provided by the traveling wire flux mapping system. Each channel is comprised of a flexible wire drive unit, a tubing path into the reactor (rear-to-front), and a wire storage container. Five radiation detectors are provided for the nine channels. The above equipment is located outside and inside the shielding wall behind the reactor. The motor control equipment and radiation measuring and recording instruments are located in the control room.

The motor drive equipment will permit manual or automatic insertion of special titanium wires into the reactor where they will be allowed to "soak" for a predetermined time, and then slowly withdrawn past the gamma sensitive detector. The wire is inserted into a water cooled thimble which extends through the active section of the reactor.

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The five scintillation-type gamma detectors will be read out sequentially by a single channel of instrumentation. A single power supply with individual voltage adjusting potentiometers for the detectors is located in the control room. A multirange amplifier and recorder complete the instrumentation system. The recorder is provided with an integrator for summing the area below a plotted curve.

4.5.3 System Operation

The traveling wires will be stored on reels behind the shielding wall and the system is de-energized until a traverse is desired. Power will then be turned on, the timers set for the desired "soak" time, (approximately ten minutes), and switches set up for automatic operation. The first wire will be drawn from storage past its gamma detector to obtain a record of the residual activity of the wire on the recorder chart. This step may be omitted. The wire will then be driven into the reactor at a high speed and allowed to "soak" for a predetermined time. During the soak period the second wire will be drawn past its gamma detector and the residual count taken. After the "soak" period, the first wire will be withdrawn rapidly from the reactor. As the activated portion of the wire approaches the detector

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For reading. The remaining values will be added in the same manner.

have been inserted into the table in the manner and at the proper time.

be recorded on the single ion strip chart recorder. The second value recorded will

be added to a predetermined starting speed. The resulting value will

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5.0 Nuclear Control and Safety Circuitry

5.1 General

The reactor is provided with a horizontal rod system which performs the functions of reactor control and is also the primary reactor safety system. The reactor is also provided with a second, or backup, safety system, the Ball-3K system. The following sections describe the electrical circuitry which is provided to actuate the horizontal rod system for reactor control, and the relay logic circuits which transmit trip information from the safety sensors to the rod scram drive and the Ball-3K trip system.

5.2 Horizontal Rod Electrical Control and Safety Circuits

Each horizontal rod circuit is equipped with a key-locked, five-position, "Service Selector Switch" which establishes the primary function of each rod. These functions are:

- a. Manual
- b. Power Set Back
- c. Safety
- d. Withdrawal
- e. Off

Figure 5-1 presents the control circuit associated with each switch position. The functions and operation of the rods in each of these services is described below.

5.2.1 Manual

The operational control of a reactor demands a large amount of flexibility in the selection and actuation of horizontal rods. The following functions have been provided:

- a. In addition to the horizontal rod scram speed, two control speeds are provided, a fast insertion and withdrawal speed of 30 feet per minute.

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and a slow insertion and withdrawal speed of 6 feet per minute.

b. The problem of rod selection for control, and rod actuation is somewhat complicated. The simplest approach would be to provide a rod-drive switch for each rod so that each rod might be operated independently. However, this would limit to two the number of rods which could be actuated simultaneously by one operator, patently too few when such a large number of rods are provided. The rods could have been combined into pre-determined groupings, all the rods in each group operating simultaneously. The concept is complicated in the H-Reactor, since the rods enter from both sides of the reactor. Thus, each group should include rods entering from both sides to maintain a balanced power distribution. Furthermore, a pre-determining of the rod-grouping during the design of the reactor was considered undesirable, since only limited data on the pile operating parameters such as temperature coefficients, individual rod control strengths, unperturbed flux distribution, etc., were available. All these factors would affect the desired number of rods in each group and hence the number of groups. It was therefore decided arbitrarily to establish the number of groups or gangs at six, but to permit any rod to be assigned to any of the six gangs. This provides the desired degree of flexibility in choosing rod patterns for reactor control.

The flexible-gang concept had one undesirable feature. A large number, or all, of the rods could be accidentally assigned to one gang and actuated simultaneously. Had the rod drives been electrical, further steps would have had to be taken to eliminate this feature. However, the

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rods are hydraulically driven, and the number of rods driven simultaneously is relatively limited by the capacity of the hydraulic supply system. This has the further advantage that the limiting effective withdrawal rates of a large number of rods are being moved is independent of the rod speed selected, being governed instead by the hydraulic pumping capacity.

In the final design, then, each rod is provided with a set of pushbuttons which designate to which of six gangs (or none) the rod is assigned. Each gang is actuated by a drive switch which permits driving the rods IN and OUT at either fast or slow speeds. Figure 5-2 is a simplified circuit diagram of the control circuit for one rod; the page opposite to this figure describes the circuit functions.

5.2.2 Power Setback:

The Power Setback System is designed as an aid to the reactor power operator in detecting and correcting potentially unsafe reactor parameters before the safety system detects the condition and corrects it by scramming the reactor. The Power Setback System is completely independent from the safety system (although some common signal sources are utilized) and safety system requirements (control strength, instrumentation time constants and trip points) are established without regard to the existence of the setback system.

The Power Setback Circuit (23 circuit) is a common application of safety circuit design philosophy; normally closed contacts of relays which are actuated by trips of sensors monitoring various parameters are placed in series with a breaker coil, which, when de-energized, initiates power setback action. Power setback action consists of utilizing timer relays to automatically apply power to the "IN-slow" solenoids of the rods selected for power setback; intermittent operation is utilized to effect a slow insertion rate.

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Rods will be selected for use in the power setback function in two ways. Firstly, every rod assigned to any group for manual control will be automatically assigned to the setback function; however, the operator may optionally cancel this setback assignment by a pushbutton. Secondly, by means of the Service Selector Switch, any rod may be permanently assigned to the setback function. When a power setback trip occurs, all rods assigned to the setback function by either method will be inserted until the trip clears itself. An interlock in the manual control circuit will prevent the withdrawal of any control rod when a power setback is underway. (Figure 5-2) By design, it will be physically impossible for the setback circuit to withdraw rods automatically. Figure 5-3 is a simplified circuit diagram of the Power Setback Circuit. Its connections to the manual control circuit are included in Figure 5-2. The page opposing the figure describes the circuit function. Parameters which are to be monitored for power setback trip actuation are:

- a. First period, subcritical
- b. Approaching fast period, intermediate power level
- c. Approaching excessive flux at high power level
- d. Approaching excessive power-rate-of-rise
- e. Approaching outlet bulk boiling
- f. High pressure main steam header
- g. Pressurizer high water level
- h. Primary loop very low pressure
- i. Primary loop very high pressure
- j. Zone monitor high temperature

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- 18. Evaporator high or low effluent coolant temperature
- 19. High average power level
- 20. Surge tank high liquid level

5.2.3 Safety Control

Any rod and any number of rods may be designated as safety rods by means of the Service Selector Switch. Once this function is selected, the rod will be disconnected from the manual control and setback functions when the conditions: 1) the manual CUF solenoid for the rod may be energized until the rod reaches the CUF limit switch, at which time its manual functions will be disconnected, and 2) an all-rod-insertion circuit is provided which will insert all rods, including safety rods, simultaneously after a planned shutdown. This circuit is provided to eliminate the necessity of scramming the safety rods into the reactor after a planned shutdown. (The circuit cannot be used for rod withdrawal). The all-rod-insertion circuit is shown in Figure 5-6 and its connections to the manual control circuit are shown on Figure 5-2.

The number and pattern of rods which will be designated as safety rods will be established and controlled through mandatory Process Standards. For example, the combined strength of the rods reserved for safety will be required to be sufficient to override any positive reactivity transient during shutdown from any operating conditions.

It is important to note that all horizontal rods which are withdrawn will act as safety rods when the reactor is scrammed. Only enough rods will be designated specifically as safety rods to ensure that the reactor

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will be shut down on demand. After shutdown, reactivity transients arising from neutron decay and reactor cooling will result in an increase in reactivity which will require that most of the rod system be inserted to maintain the reactor subcritical.

The Primary (II) Safety Circuit is a common application of safety circuit design philosophy. Two redundant strings of closed relay contacts are each placed in series with breaker relay coils. Contacts of the two strings of relays are placed in series in the power circuits which energize the main generator busbar solenoids. Thus, interrupting either redundant series string will lock the breaker. All sensors which trip the Primary (II) Safety Circuit have contacts in both strings.

Parameters which will be monitored for safety circuit trip actuation are:

- a. Fast period, intermediate power level
- b. Excessive neutron flux, intermediate power level
- c. Excessive neutron flux, high power level
- d. Excessive power-rate-of-rise
- e. Very high total power level
- f. Core monitor high temperature
- g. Outlet boiling
- h. Tube high flow
- i. Tube low flow
- j. Primary pump steam supply

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- k. Extremely high pressure, primary loop
- l. Evaporator secondary side extremely low liquid level
- m. Condenser circulating water, low pressure
- n. Ball (3X) safety system trip
- o. Pressurizer extremely low liquid level
- p. Moderator cooling tube low flow
- q. Emergency cooling water flow to reactor
- r. Rod low-water flow
- s. Electrical power failure
- t. Extremely low surge-tank liquid level
- u. Manual trip
- v. Battery failure

The safety circuit is designed to perform several auxiliary functions. Upon receiving a scram signal, it will initiate a programmed cut-back of cooling capacity in the secondary cooling system to maintain high primary loop temperatures, and it will start the diesel-driven emergency cooling pumps. It also provides auxiliary relays to perform a number of functions in the safety circuits. For example, auxiliary circuits are provided which fulfill the functions of safety system logic (two-out-of-three trips, etc.), reducing the output of many sensors to single relays whose contacts are included in the primary (1X) safety circuit strings. Circuits are also provided to count the rods which are turned off, to determine that all assigned safety rods are fully withdrawn, and to detect the failure of any rod to scram properly.

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Figure 5-4 is a simplified circuit diagram of the Primary (IX) Safety Circuit; the page opposing the figure describes its function. Figure 5-5 illustrates a representative portion of the auxiliary logic circuit, which determines that all safety rods are withdrawn, and is typical of such counting circuits.

5.2.4 Withdrawal

Occasions may arise when it will be desirable or necessary to withdraw a limited number of horizontal rods without energizing the scram solenoids by making up the Primary Safety Circuit. One such occasion would be loss of coolant to an inserted rod during operation. If the rod cannot conveniently be withdrawn during operation without unduly perturbing the power level or flux distribution, it will be necessary to scram the reactor and then withdraw the rod. The local graphite temperature surrounding the rod channel will probably be above the melting point of aluminum; thus, leaving the rod in place for more than several minutes could lead to rod damage. However, the rod is sheathed with titanium (1800 C melting point) so melting and diffusion into the stack will not be a problem.

The withdrawal circuit keeps the scram solenoid energized from an auxiliary power circuit, bypasses various interlocks, and permits rod withdrawal by use of manual controls. Since, under these conditions, rods may be withdrawn without the capability of being scrambled by a safety circuit trip, the number of rods so withdrawn must be limited. A counting circuit is provided which will count the number of rods switched to WITHDRAWAL (or OFF - see below). If more than three rods are so switched, the counting circuit will remove the auxiliary power circuit from the scram solenoids, thus scrambling the withdrawn rods back into the reactor. This feature is not present during operation, however. With the Primary Safety Circuit made up (as is necessary

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to operate), the auxiliary power circuit and normal power circuit are in parallel. Thus, de-energizing the auxiliary power circuit by the counting circuit when more than three rods are placed on the WITHDRAWAL function will not result in a scram. However, the WITHDRAWAL or OFF rods will scram with the rest of the Horizontal Rod System if a safety circuit trip occurs.

Figure 5-6 is a simplified circuit diagram of the WITHDRAWAL circuit and its function is described on the page opposing the figure.

5.2.5 OFF

Circumstances may arise when a rod must be totally disabled for repair purposes, or to prevent the scrambling of a dry rod into the reactor from a safety circuit trip. An OFF position has been provided on the service selector switch to permit this.

Switching a rod to OFF will energize the scram solenoid from the same auxiliary power circuit described in the withdrawal function above. The switching will also register in the same counting circuit used to count rods switched to WITHDRAWAL. Thus, if more than a total of three rods be switched to either of these two functions, the auxiliary power circuit will be de-energized and the scram solenoids will either be energized by the normal power circuit if the safety circuit is made up, or the rods will scram. As long as three or fewer rods are switched to these positions, rods turned OFF will not scram when a safety circuit trip occurs.

Switching a rod to OFF de-energizes the manual control circuit for that rod and also removes it from the power setback circuit, thus effectively removing the rod from all control and safety circuits except the counting circuits.

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Figure 5-7 is a simplified circuit diagram of the rod OFF circuit. Its function is described on the page opposing the figure.

5.2.6 Interlocks

Various interlock circuits are provided to prevent certain types of rod manipulation unless specific conditions are satisfied. One interlock circuit will prevent the withdrawal of control rods unless all the rods designated as safety rods are first withdrawn. This circuit was described in Figure 5-5. Another interlock circuit will prevent withdrawal of rods at the fast withdrawal speed unless nuclear instrumentation indicates that the reactor is subcritical. This interlock appears as a contact in the manual control circuit of Figure 5-2.

Failure of these two interlocks would not constitute an unsafe condition. If control rods were withdrawn before the safety rods, the control rods would act as safety rods upon scram trip. The various sensor trip settings which monitor parameters such as rising periods, power rates, flux levels, temperatures, etc., which will appear in the Process Standards, will be adjusted to be compatible with the reactivity ramp which could be introduced by fast rod withdrawal, thus in effect ignoring the fast-rod-withdrawal interlock. This interlock is essentially an operator aid to prevent accidental fast rod withdrawal under conditions where the reactor might be scrammed as a result, with consequent lowering of operating efficiency.

A series of interlocks is also placed in the LX reset circuit which will prevent makeup of the Primary (LX) Safety Circuit, and hence reactor startup, unless a number of critical valves in the primary loop are properly positioned. Other interlocks in the LX reset circuit prevent reset action unless the following conditions are satisfied:

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- 1. Scram hydraulic accumulators pressurized
- 2. High flux trip on intermediate level nuclear instrumentation not bypassed
- 3. Power rate-of-rise trip not bypassed
- 4. Ball channel exit valves closed
- 5. Less than five ball hoppers locked
- 6. Graphite cooling water flow trips not bypassed
- 7. Emergency cooling water diesel system not bypassed.

5.3 Ball-RK Safety Circuit

5.3.1 General

The ball safety circuit is identical in concept to the horizontal rod safety circuit, with one exception; each hopper latching mechanism contains two independently-powered sets of pull-in and holding coils in a latching solenoid. Both holding coils must be de-energized by a ball safety circuit trip to drop the balls. Thus, spurious ball drops would not result from a failure of any single coil, relay, or power circuit. To preserve this protection against accidental ball drops, the redundant trip circuits (see Figure 5-9) are also powered by separate d.c. power supplies, and the logic circuits are powered through a diode matrix from both d.c. supplies so that either power source will keep the circuits energized. The result of this arrangement is that both trip circuits must be de-energized, thus de-energizing both hold-in solenoid coils in each hoppers, in order to drop the balls. Since this arrangement could result in a failure-to-trip if a critical relay would fail to trip on demand, each of the two ball trip circuits is interlocked with the horizontal rod trip circuit, so that a trip of either ball circuit will

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scram the horizontal rods. In this case, the ball-backup-to-horizontal-rods circuit (Section 5.4) will be actuated, and will initiate a partial or total ball drop if needed because of horizontal rod failure.

A simplified circuit diagram of the ball safety circuit is presented in Figure 5-9, and its function is described on the facing page.

5.3.2 Interlocks

The ball safety circuit is provided with lockout features to permit maintenance on individual channels during outages without the risk of inadvertent ball drops. Sufficient control to assure subcriticality must be inserted in the reactor from other sources (horizontal rods, xenon, or supplementary poisons) before ball lockout will be permitted. An interlock is provided to prevent resetting the horizontal rod safety circuit, and hence prevent rod withdrawal, if more than four ball hoppers are locked out. This interlock appears on Figure 5-4. An interlock is also provided to prevent resetting the horizontal rod safety circuit if any of the drain valves at the bottom of the ball channels are open. Another interlock prevents opening these drain valves once the horizontal rod safety circuit is reset.

5.3.3 Trip Systems

Only four automatic trips are included in the ball safety circuit. The sensors for two of these, the flux monitors, are described in Section 4. In addition to these, ball drops are initiated by seismoscope trips or by emergency cooling system actuation.

5.3.3.1 Seismoscopes

The seismoscope system consists of three pendulum-and-ring-type seismoscopes which are identical in design to those employed at the other Hanford

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production reactors. Each of the three seismoscope contacts will be adjusted to trip when an earthquake of intensity five or greater on the modified Mercalli scale occurs. The seismoscopes are arranged to trip the ball safety circuit through a two-out-of-three coincidence logic circuit (see Figure 5-9).

5.3.3.2 Emergency Cooling System

The circuitry actuating the emergency (last ditch) cooling system is described in Section 4 of Volume 3. Should an actuation of this emergency system be signalled, the ball (3K) safety system μ will also be tripped, causing a ball drop from all hoppers except those locked out.

The function of each trip system and its associated logic circuit is described more fully on Figure 5-9 and the facing page.

5.4 Ball-Backup-to-Horizontal-Rods Circuit

In most circumstances the ball safety system is considered as a backup to the horizontal rod safety system. Automatic backup is provided by two independent means:

a. If the reactor fails to shutdown rapidly after a horizontal rod scram, the condition will be detected by nuclear instrumentation, and the entire ball system tripped. Should events such as an earthquake occur which could significantly decrease the probability of a successful rod scram the entire ball system will be tripped. Should the emergency cooling system be actuated, and a large reactivity increase (approximately 3.6 percent k) is known to be imminent, the ball system will be tripped, in addition to the rod system, to give maximum assurance of adequate control capacity.

b. A second level of backup is provided by the ball-backup-to-horizontal rods circuit. If all but three of the horizontal rods are inserted, the control capacity is adequate to control the excess reactivity under all normal or accident conditions.

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