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REACTOR COMPONENTS STUDY

INTRODUCTION

Reactor operating conditions are formally established to conserve pile life. A major part of the 2000 Research and Development effort is oriented to develop fundamental information required to properly establish these limiting operating conditions. This report reviews the current condition of reactor components believed important to continued productive life of the reactors and discusses the current status of technologies important in estimating the future productive life of the reactor complex. These evaluations are conducted continuously and this report is one of a series of status reports.

SUMMARY

Graphite

Major problems concerning the life of the graphite stack are (1) radiation induced expansion in the fringe zones which leads to curvature in the process tubes; (2) graphite oxidation which leads to deterioration in mechanical strength; (3) contraction of central zone graphite which accentuates process tube curvature; (4) mechanical damage to the graphite stack occurring from ruptured fuel element removal, tube replacement, or other maintenance; (5) radiation induced stored energy.

The limits in Process Standards, together with present operating procedures, recognize these problems and arbitrarily target a graphite lifetime of the order of 20 to 25 years from the startup of a given reactor. Our control of the graphite problem, of course, can only be as good as our current understanding of the technology. The older reactors represent our only large scale means of piloting degrees and types of deterioration; therefore, the state of the technology must be and is continuously reassessed.

The technical parameters in each of the problems listed above except for that of contraction are understood, although the application of the problems to the complex operating conditions of the stack are difficult and the application is made in what is believed to be a conservative fashion.

In the case of contraction, the phenomenon is not well understood technically. The magnitude of the effect is such that no immediate operating problem is of concern. The phenomenon could lead to prime future problems. Work is under way monitoring the effects in Hanford reactors and obtaining the effects under accelerated service conditions by means of MTR experiments.

Shields

Major problems concerning shields come from the deterioration of the biological shield. These effects are monitored and remedial measures are in effect in the form of fringe poison loadings which appears to be a workable method for controlling shield temperatures. Supplementary shielding, as a rather expensive maintenance item, appears to be a feasible approach to shield deterioration if the need should develop.

Other Reactor Components

Other reactor components are replaceable and deterioration is reflected in the maintenance required to assure a safe workable reactor complex. These components do not represent a basis for a limit to reactor life if one assumes that the monies necessary for maintenance are available.

The possibility of a major incident which could render a reactor completely unproductive is not treated in this report and is considered to be of low probability.

DISCUSSION

I. GRAPHITE

The characterization of the effects of radiation on graphite has been the subject of intensive research and development programs at HAPO since the beginning of plant operation. For the past few years the purpose of the programs has been to define the operating conditions which would permit a life for the graphite stack of the order of 20 to 25 years from startup. As a result of all studies to date it has been determined that graphite deterioration or damage could result from four principal sources, namely (1) chemical reaction of the graphite with the reactor atmosphere components resulting in loss of mechanical strength of the graphite blocks, (2) growth and contraction of the graphite resulting from irradiation instability, (3) mechanical damage arising from maintenance operations or other causes for shifting of the graphite blocks, and (4) the release of stored energy accumulated during irradiation which could result in excessive graphite temperatures and melting of internal reactor components. These factors are discussed in the following paragraphs:

A. Oxidation and Graphite Strength

Graphite oxidation results from the chemical reaction between graphite and the reactive components which may be contained in the reactor gas atmosphere, including carbon dioxide, oxygen and water vapor. Oxidation results in a loss of strength of the graphite, and the rate of oxidation must be limited to assure that graphite stack integrity will be maintained for an extended period of time.

The chemical reactions are strongly temperature dependent, and the graphite temperatures and gas atmosphere composition have been carefully controlled in order to maintain chemical reaction rates at tolerable levels. The reactor atmosphere is supplied principally with two types of gases - carbon dioxide and helium. At low graphite temperatures, carbon dioxide is permitted to be the principal component of the gas atmosphere, because the reaction rate between graphite and carbon dioxide is very low at the lower temperatures. At high graphite temperatures, helium is used as the primary reactor feed gas because it is chemically inert and has a high thermal conductivity. At intermediate temperatures, mixtures of carbon dioxide and helium are permitted, and the graphite temperature limit is established on the basis of the percentage of helium present. The maximum temperature at which a given gas mixture can be used has been established so that the oxidation rate is suitably controlled.

Data have been obtained on both irradiated and unirradiated graphite from which the variation of graphite compressive strength as a function of oxidation may be determined. These data indicate that for a uniform ex-reactor oxidation of 10% there is a loss of approximately 50% in the compressive strength for both irradiated and unirradiated graphite (See Fig.1). Further oxidation to 25% results in a loss of strength by a factor of 6. Spot checks on tensile and shear strengths indicate a similar decrease in strength with oxidation. The maximum graphite temperature limits which are imposed by oxidation considerations are based primarily on the above strength data for unirradiated graphite.

The compressive loading of the bottom layers of the graphite stack is of the order of 100 psi while the compressive strength of the graphite even after 25% oxidation is ~1000 psi. It is thus evident that there is little danger of "stack collapse" even after such excessive oxidation.

However, to prevent gross damage to the graphite lattice during maintenance operations for removal of stuck rupture fuel elements, tube removal, or gross stack shifting, the loss of tensile and shear strength as a result of oxidation is of serious concern. Thus, to assure that graphite strength will be maintained at an acceptable level, the graphite temperature and gas composition are maintained at conservative values which limit the oxidation rate to approximately 2% per 1000 pile operating days⁽²⁾ (See Figure 2). It should be noted here that graphite strength is not the only factor in establishing the ultimate life of the graphite stack. Other effects, graphite growth and contraction, appear to be equally serious, and are discussed under the section on Distortion.

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**FIGURE 1 - IRRADIATED GRAPHITE COMPRESSIVE
STRENGTH VS UNIFORM OXIDATION**

Figure 2

GRAPHITE TEMPERATURE LIMITS
Vs.
GAS COMPOSITION

Remainder CO₂ plus not more than 5% other gases

Oxygen ≤ 0.05% by Volume

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It is estimated that the graphite in the BDF reactors has experienced a total oxidation of about 1% as a result of the operating conditions to date, and the oxidation in the newer reactors is less. Operational problems are encountered in maintaining low concentrations of oxygen and water vapor in the gas atmosphere, particularly in those cases involving water leaks into the graphite stack and when the reactor is opened to permit maintenance, such as process tube or control rod replacement.

Other than deleterious effects on the graphite strength, oxidation of the graphite presents no unusual hazards or production loss. With 10% graphite burnout at K reactors, the over-all reactivity would decrease approximately 0.5%, and the conversion ratio would increase slightly. At the older reactors the reactivity may increase or decrease, depending upon the extent of burnout, and the conversion ratio would also increase.

1. Oxidation Studies

Monitoring studies are performed on graphite samples exposed for various lengths of time in the reactors to evaluate burnout under existing operating conditions.⁽³⁾ One tube in each reactor is used for continuous monitoring of burnout. Graphite samples from the operating reactors provide a valuable source of data for the evaluation of the immediate moderator conditions, but they do not permit a systematic study of damage effects at other, more stringent conditions. To obtain these additional data, experiments are being performed at Hanford to evaluate graphite oxidation at temperatures up to 750°C with gas atmosphere containing 5-25% carbon dioxide.

Experimental studies have indicated that, whereas oxidation decreases the strength of the graphite, irradiation increases graphite strength. The resulting net strength is the important parameter to be used in evaluating the integrity and potential life of the graphite stack. The current fundamental graphite studies include evaluation of these parameters which will define the strength characteristics of the graphite components. These studies, together with graphite strength data obtained from graphite samples removed from the reactors, provide the basis for establishing operating limits which would permit a reactor life of the order of 20-25 years from startup.

B. Distortion

Distortion of the graphite stacks continues to be a problem of real concern at all reactors. The instability of graphite under neutron irradiation has resulted in varying degrees of stack deterioration.^(4, 5, 6, 7) During early operation of the older reactors at low temperatures, the entire graphite stack expanded, with the top center portions of the graphite approximately 4 inches above the top fringe graphite. As graphite temperatures were increased the central graphite annealed and contracted at a faster rate than the fringe, resulting in a profile in the various graphite layers similar to a meteor crater. (See Figure 3) The edges are raised above the base level while the center is depressed below it. Graphite distortion results in distortion or damage to reactor components, such as bending of the process

**FIGURE 3 - TYPICAL EXPANSION OF
ORIGINAL REACTOR**

tube, flattening of the tube and gunbarrel, and fracturing of the graphite blocks at the entry of the gunbarrel. Such graphite component distortion results in operational difficulty in charging fuel elements, replacement of process tubes, movement of horizontal rods, etc.

B, D, and F reactors suffer considerable "crater" distortion of the stack. The stack at the B Reactor has suffered the worst distortion, with a variation of approximately 3 inches in elevation. B and F reactors presently require 4 inch fuel elements in the upper portion of the stack to facilitate fuel charging in those process channels having the greatest distortion. The other reactors have little graphite expansion in the fringes, but are undergoing contraction in the center. Laboratory measurements have been made on samples exposed at various typical conditions and show contraction to range between .05 to .1 per cent per 1000 MWD exposure depending on the temperature of irradiation, for temperatures in excess of 300°C. This represents a contraction rate of approximately C.3 to 0.6 inch per year, which substantiates the data obtained from measurements of the reactor stack contraction, as shown on Figures 4 and 5. Distortion of this type will cause contraction of individual graphite blocks causing gaps to occur between the blocks, resulting in higher graphite temperatures. Control rod binding and fuel element charging difficulties may also be encountered in the future. Thus contraction of the stack may ultimately limit the practical operating life of a reactor.

The mechanism of the growth and contraction of the graphite is not completely understood. However, considerable data are available which permit the formulation of practical reactor operating limits for control of distortion and oxidation. At HAP0, stack deterioration is controlled by maintaining the graphite at temperatures which will neither cause excessive growth nor excessive oxidation. As shown in Figure 2, the present graphite temperature is 650°C, and may be permitted only with an essentially inert helium gas atmosphere to prevent excessive oxidation of the graphite. The lower maximum graphite temperature limit has been established to prevent excessive growth from occurring in the fringes. Because the maximum graphite temperature, which generally occurs in the central zone of a reactor, must be maintained at temperatures in excess of 450°C to prevent fringe zone expansion, contraction of the central zone will occur. Thus it is recognized that stack contraction will continue under present operating conditions. A limited amount of high temperature data indicate that the present contraction rate can be expected to continue until graphite temperatures of 800-850°C are reached. The following table summarizes present and anticipated conditions for forecast power levels.

<u>Reactors</u>	<u>Present Conditions</u>	
	<u>Nom. % He</u>	<u>Nom. Max. Graphite Temp.</u>
B, D, F, DR, H	50-55%	560
C	65%	565
K's	90%	585
<u>Forecast Estimates (FY-63 and beyond)</u>		
B, D, F, DR, H	100%	650
C	100%	700
K's	100%	800

Figure 4

Graphite Distortion at C, F and H Reactors

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Figure 5

Effect of Skewed Flux at F Reactor

Vertical Traverse - 4676-F

The rate of graphite distortion at the K Reactors has not yet been accurately established. However, it has recently been observed that a significant separation has occurred between the graphite blocks in the near and far fringe regions at both reactors. The cause of this separation cannot be explained on basis of mechanical or radiation damage as no large mechanical forces have been applied to the moderator stack in the vicinity of the affected region during maintenance operation. Efforts are being made to evaluate corrective and preventative measures. The effect on the productive life of the K Reactors of this anomalous stack distortion is currently not known.

1. Distortion Studies

Routine monitoring of the vertical height of the graphite stack is performed to determine distortion trends at all reactors. Graphite samples are also being irradiated in the MTR to evaluate the effects of high flux and high temperature on graphite contraction. Mechanical strength of the graphite is a sensitive function of graphite oxidation, as discussed previously. It has been determined that the present graphite stacks are sufficiently strong to permit a large reduction in graphite strength before trouble would be expected from bulk collapse. However, some components of the stack, e.g., trunnion blocks and graphite keys, are inherently weak, and can be broken in tube and stuck rupture removal processes. Graphite strength as a function of the irradiation and temperature conditions in the reactor is continuously monitored to furnish information used in establishing maximum tube and ruptured fuel discharge forces. Tests are being conducted to evaluate more accurately the effects of higher temperature operation on reactor life. In addition to these tests, methods of annealing and mechanically eliminating the existing stack distortion are being studied. Equipment has been developed for the purpose of renovating badly distorted process channels to assure continued, efficient operation.

Measurements are continuing to determine the effects of irradiation on the thermal conductivity of graphite, and to evaluate graphite thermal conductivity at temperatures in the range of 750-1100°C. Thermal conductivity is a factor which affects the graphite temperature, and which, in turn, affects other more important properties such as dimensional stability and oxidation rate.

C. Stored Energy

The release of energy stored in the graphite matrix as a result of irradiation at various temperatures is not an operational or safety problem at the Hanford reactors.⁽⁸⁾ Stored energy builds up as a direct function of in-pile irradiation and as an inverse function of exposure temperature. For a nominal exposure temperature of 30°C the accumulation of stored energy appears to saturate at 630 cal/gram after an exposure of 5000 MWD/CT. By contrast, stored energy saturates at about 18 cal/gram at an exposure temperature of 400°C. (Figure 6) In order for the energy to be released in a self-sustaining reaction, the stored energy annealing spectrum must exceed the specific heat curve. Such a stored energy spectrum is obtained only under low temperature irradiation conditions - below 150°C and at relatively short exposures. At the B, D and F reactors the temperature of irradiation in the filler blocks of the fringe zones

**FIGURE 6 - TOTAL STORED ENERGY BUILDUP
AS A FUNCTION OF EXPOSURE TEMPERATURE**

has been in the range of 200-300°C. Thus, it is expected that the filler blocks in the fringe zone graphite do not have a release spectrum which exceeds the specific heat curve, and spontaneous energy release will not occur. The total stored energy in the filler blocks in the fringe zone graphite does not exceed approximately 50-75 cal/gram, and in the central zone approximately 30 cal/gram. The magnitude of the temperature increases expected from the release of this stored energy are not adequate to cause excessive graphite burnout or component melting. Continuous monitoring of the graphite is performed to determine the total stored energy in the various reactors. By removing samples from the moderator at specific locations and determining the total stored energy and the release spectrum of the samples, the stored energy conditions of the reactor stack can be evaluated.⁽⁹⁾

As in the case of distortion, the control of stored energy has been through the control of graphite operating temperatures, and it has not been necessary to anneal the graphite stack by deliberately triggering the stored energy reaction.

D. Mechanical Damage

Mechanical damage to the graphite stack could occur during ruptured fuel element removal procedures or other maintenance procedures. However, this type of damage has, to a large extent, been carefully controlled, so that ultimate reactor life is virtually unaffected by this type of damage. However, as noted earlier, it is important that graphite strength does not become reduced to the extent that normal tube maintenance cannot be accomplished without excessive graphite damage. Continuous programs are conducted to monitor and evaluate mechanical damage, and to ascertain the occurrence of anomalous conditions as may be encountered, as at the K reactors.

II. MAINTENANCE

Aside from the graphite, essentially all other components of the reactor complex are subject to maintenance or replacement. As the reactors and water plants become older, the types of maintenance problems will vary. Several examples may be cited. If the graphite is operated at temperatures in the range of 650°C or above, then replacement or continuous maintenance of gas seals may be required. Aluminum process tubes are subject to corrosion and must be replaced periodically. Pump impellers, water flow instrumentation, and water plant components are subject to deterioration because of stringent service requirements. Thermo-couples which monitor the reactor process conditions are subject to failure due to high temperature service or mechanical damage, and must be replaced or repaired frequently. Instrumentation must be maintained in a state of first class repair.

Because of the extensive maintenance programs which have been carried out at the reactors since startup it can be stated that the integrity and reliability of the reactor components has not decreased during the past years. Because of the higher levels of experience which have been achieved many of the components have much greater reliability than those originally installed. It is anticipated that maintenance considerations will not seriously affect the lifetime of the reactor systems if one assumes that adequate money will be available for maintenance when required.

Following is a list of maintenance and replacement projects which could be required at the various reactors in the foreseeable future to assure continuity of production at an acceptable level of operating efficiency.

1. Rebuilding of water plant filters.
2. Repair of existing underground raw water pipes when leaks appear.
3. High tank inside painting and cathodic protection.
4. 183 Building valve repair.
5. Replacement of 190 tank bottoms.
6. Dredging of 181 pump house forebays.
7. Effluent system repairs.
8. Pump impeller replacement in the 190 and 183 Buildings.
9. Reactor thermocouple replacement.
10. Tube replacement.
11. Graphite boring to reduce process channel curvature.
12. Replacement of gas seals on gunbarrels and vertical safety rods.
13. Replacement of inlet and exit process tube fittings (uncertain).
14. Operational charge-discharge equipment maintenance.
15. Freeing of stuck gunbarrels.
16. Ball 3X and safety circuit renovation.

Other plant improvement projects are not included, such as increasing water plant capacities, installation of improved nuclear instrumentation, installation of operational charge-discharge systems, and installation of reactor containment systems. Projects such as the latter are also scheduled, but they are designed to increase the production capacities or to improve upon the nuclear safety of the reactor systems.

III. SHIELDING

Studies to detect deterioration of the Hanford shields and of associated radiation leakage and structural damage have been under way for more than ten years. This section summarizes these studies and the observations and conclusions which have been made, and briefly describes the protective methods which have been developed and envisioned in order to ensure that shield deterioration does not become a limiting problem at Hanford.

A. Background

1. Description

Primary shielding of a Hanford reactor consists of two parts: an inner "thermal" shield nearly a foot in thickness, and a surrounding "biological" shield approximately four feet thick. In contrast to the high density iron thermal shield whose purpose is to eliminate the bulk of the heat-producing energy, the biological shield contains large amounts of low-density hydrogenous material to provide thorough moderation of intermediate energy neutrons and, thereby, ensure that no "windows" exist which will permit excessive numbers of these biologically damaging neutrons to escape. Furthermore, moderation of these neutrons deep within the shield will permit them to be captured as thermal neutrons well back in the shield, thus providing more attenuation of secondary gamma radiation from this source. Figure 7 shows a typical section of a top biological shield in one of the older Hanford reactors. Layers of iron are alternated with layers of high-

FIGURE 7

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TYPICAL PLATE WIDTHS BETWEEN T-STEMS TOP SHIELD

Figure 7

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density masonite of nearly equivalent width. For convenience in meeting construction tolerances, each layer of iron actually was made up of half-thickness slabs of iron, and each slab consisted of two pieces tack-welded to each other and to the iron "T-seam" at either side of each four-foot shield section. Thus, the masonite was used as a supporting material when the shield was being constructed, but tack-welding of the steel plates no longer would require this masonite structural support as long as the tack-welds remain intact.

2. Deterioration of Masonite

Experimental studies have demonstrated that in the ranges of fluxes and temperatures encountered in the Hanford biological shields, temperature is a much greater contributor to masonite deterioration than is radiation leakage. Out-of-pile studies indicate that a 15 to 20°C increase in masonite temperature will reduce its structural "half-life" by a factor of two. For example, masonite would lose half of its compressive strength in approximately two years in a temperature field of 100°C, whereas it would lose the same fraction in the order of two weeks at 200°C. Loss of hydrogen is expected to occur at about the same rates.

3. Early Prediction of Shield Temperatures

The temperature of the shield is dependent upon two factors: rate of production of heat, and manner of removal of heat. The heat produced in the shield is directly proportional to the radiation leakage to the shield which in turn would be expected to be directly proportional to the power generation rates of nearby uranium columns. The removal rate of the heat is dependent upon the effective conductivity of the coolant system: the heat paths, the gradients, and the effective area of the coolant channel. Base conditions change with the local coolant temperature as determined by amount of flow and by seasonal river temperature.

An equation was set up with terms for each of the above effects, and constants for the equation were determined which would best describe observed thermal shield temperatures.

The biological shield forms a thick insulator with a temperature gradient from near-thermal shield temperatures at the inside to temperatures somewhat above air temperature at the surface. Data from test hole plug thermocouple facilities especially designed for the purpose were used to relate observed thermal shield temperatures to adjacent biological shield temperatures. Thus, a crude but basically sound method was developed prior to the start of Project CG-558 for estimating biological shield temperatures as a function of pile operating conditions. In combination with the out-of-pile temperature deterioration data, shield temperature traverses were predicted, and their effects integrated over time to predict the manner and rate of masonite deterioration in the future.

4. Early Prediction of Shield Attenuation Loss

Anticipated rates of masonite deterioration indicated that in some cases on the order of half of the hydrogen would be lost from the

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biological shield. Losses of this magnitude, particularly in a non-uniform manner, would cause large losses in attenuation efficiency which could not be accurately calculated. Therefore, experimental attenuation measurements were made using slabs with masonite removed in amounts corresponding to the previously estimated temperature damage conditions. Several such experiments were carried out to simulate various stages of shield burnout. In addition to the observed changes in gamma attenuation and in thermal, epi-thermal, and fast neutron fluxes, theoretical calculations were made to determine the biological effect of the resultant intermediate neutron energy spectrum; a lucite "flux integrator" consisting of additional moderator was also used as a check on the theoretical calculations. The data from these attenuation tests thereby provided a method of estimating the increase in biological dose to be expected as a function of future operating conditions.

5. Installation of Monitoring Devices Under Project CG-558

Two types of monitoring provisions were made in conjunction with this project: thermocouples were installed in each iron layer in the expected maximum temperature regions on the far side and top of each reactor, and sample wells were placed in the outer masonite layer of the far-side of each reactor to permit determining the neutron flux as a function of time with foils not exposed to the background scatter normally present on the experimental level. The arrangement of thermocouple and test well monitors is shown schematically in Figure 7.

B. Observations and Conclusions

1. Description of Deteriorated Masonite

Masonite chips have been removed from test holes and vertical rod holes when step-plugs have been removed during the past several years. The residue material is charred in appearance, slightly shrunk in size, and brittle in texture. Approximately two-thirds of the original masonite remains as a graphite-like substance after the hydrogen has all been evolved; this observation is important in postulating structural effects should the tack-welds of the iron layers fail.

2. Post CG-558 Temperature Observations

Figure 8 shows recent temperature data from the far-side and top shields at each reactor. Three important observations have been made during post CG-558 operation:

- (a) Top shield temperatures have been more severe than expected. Neutron streaming up the VSR holes and subsequent scattering into the top shield is apparently a significant factor.
- (b) The DR and H biological shields experience considerably higher temperatures than do the older piles, presumably because of their thinner reflectors and their larger gas gaps between the thermal and biological shields.

FIGURE 8

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EQUILIBRIUM BIOLOGICAL SHIELD TEMPERATURES

FIGURE 8

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- (c) The conductivity of the pile gas atmosphere has a strong effect on shield temperatures. It is presumed that the coolant pipes are no longer mechanically bonded to the thermal shield and that the pile gas is thus an important part of the heat removal path.

The above factors have been considered in the postulation of future shield temperatures as shown in Figure 9. If helium is substituted for CO₂ to minimize graphite temperature increases as power levels are raised, maximum shield temperatures will probably not be much greater than at present and may be lower. Increases in level with proportionate increases in graphite temperature, as would be expected by maintaining current percentages of CO₂ or by substituting nitrogen for the present amounts of CO₂, may cause undesirably large increases in maximum shield temperatures.

3. Results of First Year's Attenuation Data

Counting data are taken during each equilibrium operating period from foils irradiated in the far-side test wells at the B, D, DR, F and H reactors. The first year's data have been plotted for each of these reactors with the following observations. Count rates are sufficiently low and the number of samplings sufficiently small that the data are not statistically meaningful. Best fits of data from the B, D, DR and F reactors actually indicate a negative slope, an extremely unlikely condition of a gain in attenuation efficiency. The H reactor data, which on the other hand appear more self-consistent, indicate an increased leakage of approximately four per cent. The year's increases in leakage which would have been predicted from previously derived methods ranged from approximately 14% at D reactor to approximately 40% at the H reactor.

4. Revised Burnout Predictions

The H reactor test well data trends were, therefore, used to derive new less-conservative constants for the equations relating shield burnout and leakage to operating conditions. Figure 10 shows the present hydrogen content of the more severely deteriorated shields as postulated by both methods. From the revised constants obtained from the H reactor foil test well data it is predicted that bulk shield leakage rates will increase by significantly less than a factor of one hundred during the next ten years if the maximum shield temperature is maintained below 160°C, and that leakage will increase by significantly less than a factor of ten during the next ten years if the maximum shield temperatures are maintained below 100°C. These values are not considered to be sufficiently severe to cause concern in the near future since they are of the same order of magnitude as the flux leakage presently encountered around test holes and vertical safety channels.

Front face deterioration is not considered to be a problem since the front biological temperature is not much above that of the inlet water.

5. Possibilities of Structural Failure

As pointed out initially, the biological shield is constructed in such a way that the masonite should not need to be relied on as a structural

FIGURE 9

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ANTICIPATED RANGE OF MAXIMUM SHIELD TEMPERATURE - ASSUMING EXISTING EXTENT OF FRINGE POISONING.

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FIGURE 9

FIGURE 10

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POSTULATED HYDROGEN DEPLETION IN SHIELD MASONITE

FIGURE 10

material. However, it is quite likely that tack-welds were broken in the top shields of the old piles during the early graphite expansion problems. It is also possible that tack-welds would break at the DR and H reactors where the top shield is not supported by the graphite stack. It was pointed out two years ago that the possibility existed for iron plate slippage as masonite layers deteriorated underneath and that such slippage might cause vertical safety rod binding of major proportions. It is currently believed that large scale rod trouble will not occur, at least without minor troubles providing a warning of several years. First, the brittle masonite carbon residue consisting of approximately two-thirds of its original thickness would still remain. Second, rivet-like pins were placed in the masonite at intervals of the order of a foot, and these pins, held in place by the masonite residue and the pressure of the plates above, should provide adequate compressive strength to support the iron layers above. Therefore, it is considered unlikely that major rod binding problems will result from shield deterioration during the next ten years, but it is quite possible that binding of individual rods will be experienced from time to time.

C. Protective Methods

1. Fringe Poison

Loading tests were performed during the KW startup program late in 1954 to check observed versus postulated flux patterns associated with a fringe poison pattern compensated for reactivity loss with nearby enrichment. The low-level flux pattern determinations were sufficiently promising that a fringe poison layer was placed in the operating H reactor early in 1955. Since that time "fringe poisoning" has become a standard operating method for reducing shield temperatures. Columns in the fringe layer on the far-side or the top are loaded with poison slugs, usually lithium aluminum alloy slugs, and sufficient enriched metal is loaded into nearby columns to compensate the reactivity loss due to the fringe poison and to hold up the flux levels inside the poison ring to near original values; the flux in the reflector is reduced by the fringe poison and so also then is the heat load on the thermal shield.

Fringe poisoning is particularly effective for side shield protection, where alternate tubes of poison in the fringe layer achieve a reduction of approximately 25% in shield temperature rise. Fringe poisoning is less effective but nonetheless extremely valuable in the top of the pile where a solid layer may be necessary to achieve this same 25% reduction in shield temperatures.

Figure 11 shows the present fringe poison loadings. It is not feasible to attempt a much higher degree of fringe poisoning than presently exists in the B, DR and H top shields since streaming is now their major heat contributor. There is considerable room for control of shield temperatures on the far-side of the B, DR and H reactors and on both the top and sides of the other reactors, however.

Thus, fringe poisoning, whose conversion economics are not far different from natural uranium if a useful product is produced, is a workable method for controlling shield temperatures which is relied on in large part for averting future shield deterioration problems.

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FIGURE 11

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FRINGE POISON LOADINGS - AUGUST, 1958

☐ H
☐ B,DR

FIGURE 11

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2. Thermal Shield Coolant Flow

It was noted earlier that thermal shield coolant temperature was one of the factors entering into shield temperature prediction. There is only limited flexibility in this regard, however, as the flow through the thermal shield loop is already sufficiently great that the coolant temperature rise is of the order of only 5-10°C. Increases in thermal shield flow would, therefore, have an almost negligible effect on maximum shield temperatures.

On the other hand, continuity of thermal shield coolant is extremely important; local biological shield temperature rises would be expected to increase by a factor of 50% if alternate thermal loop tubes were out of service.

3. Protection From Additional Helium

Biological shield temperature data taken during large transitions in pile atmosphere composition indicate that shield temperature has a significant inverse relation to pile gas conductivity. As CO₂ concentrations were taken up at the B reactor during the Fall of 1957 in a high temperature graphite test, biological shield temperatures went up sufficiently that fringe poisoning was required. As the helium concentration was increased again during early 1958, B reactor shield temperatures came back down. Similar observations have been made in thermal shield temperatures at the KE reactor as gas composition changes were made during coefficient tests. Data from routine operation are not so conclusive, as smaller changes in gas composition are often masked by the long heating time lag of the shield during short periods of operation.

*Stress -
effect of heat
helium supply not
competitive with min
shield temperatures.*

Although not defined quantitatively by sufficient statistical data, the effect can be qualitatively explained on physical grounds in that heat removal paths are improved by the more conductive gas. Figure 9, incorporating our best estimates of the magnitude of this effect, illustrates the rather wide margin of its uncertainty but also indicates that increases in helium composition as pile power levels increase will probably keep maximum shield temperatures near their present levels under the projected power level conditions.

4. Supplementary Shielding

The alternative approach of supplementary shielding as existing shields would lose their attenuation efficiency was also considered in some detail during the CG-558 construction period. Methods considered ranged from simple shadow shield partitions to protect the front work area from top and far-side scattering to rather massive exterior slabs to retain original experimental level access during operation.

This approach appears to be mechanically feasible and may be regarded as a potential aid, but it appears expensive and inconvenient in comparison to moderate fringe poisoning during the next few years.

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FRINGE POISON - SHIELD PROTECTION

Following is the tabulated information which you requested for early this week:

Reactor	Present Numbers of Poison Columns		Future Expected Additions to Poison	Annual Throughput of E to Support Poison, Tons	
	Top	Side		Top	Side
B	20 Mint	10 Depl.	None	3	3
C	0	0	None	0	0
D	0	0	None	0	0
DN	20 Mint	10 Mint	None	12	0
F	0	0	(Possibly FY 1950)	(3)	(3)
H	30 Mint	3 Mint	None	15	5
KE	0	0	None	0	0
KU	0	0	None	0	0
Total Annual E Metal Throughput for Fringe Poison				49	

RESTRICTED DATA

This document contains restricted data as defined in Atomic Energy Act of 1946. Its transmission or the disclosure of its contents in any form to an unauthorized person is prohibited.

Total Annual E Metal Throughput Adjusted for
Possible Added Loadings and 25% level Increases 75

ORIGINAL SIGNED BY
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