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TITLE

PLANT EXPANSION TASK FORCE
TECHNICAL FEASIBILITY AND R & D EFFORTS

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PLANT EXPANSION TASK FORCE
TECHNICAL FEASIBILITY AND R & D EFFORTS

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INTRODUCTION

The Expansion Study Task Force has evaluated several cases of Hanford reactor operation at power levels considerably higher than is presently obtained in the six older reactors. These higher power levels result in more rigorous operating conditions of temperature, heat flux, neutron flux, hydraulics, reactor control, etc. The purpose of this document, the various components of which were prepared by Process and Reactor Development Sub-Section personnel, is to assess the technical feasibility of operation under the proposed conditions, and to delineate those specific areas of development effort which may be necessary to provide adequate support for an expansion program.

SUMMARY

Of the three methods which may be considered in evaluating potential power level increases for the Hanford Reactors, including increasing the coolant flow rate, increasing the bulk outlet temperature, and operating with net steam generation, only that of flow rate increases appear feasible now. The other methods would require considerable extension of technology, particularly in the areas of fuel jacket corrosion at bulk outlet temperatures much above 105°C, and of fuel element heat transfer and hydraulics. Because the required technology has not been developed sufficiently for those conditions of more rigorous service, this study has principally considered only that case of increasing power level through an increase of flow rate.

Operation of the reactors at significantly higher levels (20-50%) would impose severe service conditions on numerous reactor components, particularly the graphite and fuel elements. However, it is the considered conclusion of this study that power level increases of this magnitude are feasible without sacrifice of reactor safety, although a decrease in the useful pile life may result, and numerous changes would be required in process conditions, operating procedures, and reactor plant components. Each technical area presents its own problems and limitations, the severity of which would require diligent development efforts during the next two year period to assure success of the program. The principal technical factors influencing and affected by the power level increases are summarized below.

Graphite Contraction

During the immediate future at existing power levels it is anticipated that the top center of the graphite stack will continue to contract at a rate of approximately 0.35-0.5 inch per year. The graphite at the front and rear of the reactor exhibits a negligible contraction rate. Thus, over a period of time the graphite channel will become seriously distorted, making difficult the charging and discharging of fuel elements. The use of 8" fuel elements in the top tube rows of B, D, and F reactors will become increasingly difficult over the coming year at existing power levels. The use of 4" elements might delay these difficulties until 1969. Some difficulties may also be encountered in control rod operation, although to a lesser degree. Operation of the reactors at the current levels would result in a depression of the top center of the graphite moderator of approximately 10 inches within the next 20 years. The rate of graphite contraction is a function of power level, and perhaps also graphite temperature. A power level increase of 25 percent, plus reasonably pessimistic assumptions regarding the temperature dependence of graphite contraction, lead to a calculated 10 inch contraction within the next 12 to 14 years. There may be several methods of overcoming the charge-discharge problems arising from graphite distortion. In general, these methods encompass the straightening of process channels by an appropriate graphite drilling procedure, and the use of appropriate tube-fuel designs.

Estimates of a useable process channel lifetime have been made based upon existing contraction rate data. These values, as indicated below, should not be regarded as firm, rather they are typical. The extent of tube distortion varies depending upon the location of the tube in the reactor, the procedure for graphite drilling has been postulated, but not yet demonstrated, and considerable design flexibility exists in the choice of ribless tubes and self-supported fuel elements. If the use of 4" self-supported fuel elements were to be adopted, and appropriate channel straightening could be accomplished at the time of tube replacement, it is estimated that the useful service life of process tube channels located at the top of the reactor would be on the order of 12-15 years, assuming the reactor is operating at 120% of present power levels.

Process Tube Corrosion and Wear

Aluminum tube corrosion data indicate that the minimum service life of ribbed and ribless process tubes is 3 years and 5 years, respectively, during operation at 95°C bulk outlet. The life of the average tube, operating at near 95°C tube outlet temperature, would be approximately 9 years for ribbed tubes and 15 years for ribless tubes. Operation of the reactor at 105°C bulk outlet would reduce the useful service life by a factor of 2 or 3. The corrosion rate of Zircaloy process tubes is essentially zero, so that Zircaloy tubes would, for all practical corrosion purposes, remain serviceable for the lifetime of the reactor.

An unknown for both aluminum and Zircaloy smooth bore process tubes is the galling or wear which may be encountered during the charge-discharge of self-supported fuel elements. It is possible that this could materially decrease effective tube life, particularly for aluminum. A development program will be necessary to evaluate the seriousness of this problem.

The combined limitations of process tube corrosion rate and wear, and graphite contraction will determine the ultimate life of the process tube. For approximately 150-200 tubes in an ill-defined, upper portion of the reactor, the graphite contraction would indicate a useful tube channel life up to 15 years, depending upon location, the tube-fuel geometry, and the success in straightening the channels. It may be desirable to provide these upper process channels with aluminum tubes, since the aluminum corrosion rates also indicate a process tube life in the range of 3 to 15 years. For the middle and lower portions of the reactors, where the magnitude of graphite contraction is less, and the frequency of required tube replacement depends upon corrosion effects only, it may be desirable to utilize Zircaloy tubes.

Utilization of Zircaloy Process Tubes

The incentive to retube the reactors with Zircaloy process tubes arises from the high strength, very low corrosion rates at temperatures to 600°F, and the low neutron cross section of Zircaloy-2. Except for tube replacement occasioned by graphite contraction considerations, or the possibility of Zircaloy hydrogen embrittlement caused by a fuel failure, it is anticipated that Zircaloy tubes would virtually eliminate all future tube replacement. An evaluation of the mechanical and chemical properties of Zircaloy-2, the results of reactor experience with this material, and the fact that development efforts have demonstrated that production quantities of the tubes can be fabricated, indicated that Zircaloy tubes would be completely acceptable for reactor service. It is probable that the costs of Zircaloy tubes in initial production quantities will be in the range of $1,000 each and will decrease as production increases. This indicates that, in light of the graphite contraction problem, economic considerations will probably establish the justification for use of Zircaloy as tubing material.

Fuel Performance

It is improbable that substantially higher power levels could be sustained without major improvements in fuel performance. It is anticipated that improvement will be achieved through the utilization of appropriately designed self-supported fuel elements. The temperatures of the fuel element materials prevailing at 95°C bulk outlet temperature with the re-designed fuel should permit satisfacory performance of X-8001 aluminum-clad fuel bonded by the Al-Si process insofar as material deterioration or interaction is concerned. However, the ability to obtain and maintain an Al-Si bond with an I & E fuel element having a large diameter internal hole may be difficult, and may thus place a limitation on the geometrical design of the element. In addition, a significant increase in the perfection of the fuel will probably be required. Development of an alternate fuel cladding process through the pilot plant stage to provide a back-up element with significantly improved characteristics and a higher potential for routinely manufactured perfection is, therefore, considered highly advisable.

Problems associated with irradiation behavior of fuel elements include core cleavage, corrosion, and distortion. It has been concluded that at the specific power of fuel elements considered in this study, cleavage failures although they might occur, probably will not pose a major problem with I & E elements. Hot-spot failures and accelerated corrosion of fuel jackets should be reduced appreciably by the self-supports. Based on fuel jacket corrosion only, goal exposures of 900-1000 MWD/T appear feasible for a 95°C bulk outlet temperature. At a 105°C bulk outlet temperature, it would probably be necessary to reduce the goal exposure to 600-700 MWD/T. Fuel distortion and process channel distortion present a complex problem when considering stuck charges. However, it is anticipated that metal stability characteristics are such that the degree of warp manifested in 8-inch fuel elements at the higher power levels will not significantly increase the problem.

New fuel designs frequently lead to new performance problems, such as the potential Al-Si bond failures for I & E fuel elements of larger diameter. In-reactor performance of the existing size I & E self-supported fuel elements has been completely satisfactory, although the limited number of fuel elements irradiated (approximately 1000) does not permit an accurate statistical evaluation of performance. It is anticipated that to demonstrate adequate performance of any new fuel element designed for more rigorous service, an extensive in-reactor testing program will be required.

Heat Transfer and Coolant Flow Considerations

Of the basic engineering factors limiting reactor power level, the problem of adequate fuel element cooling at all times is a factor of prime importance in considering reactor design changes and increased power levels. It is axiomatic that the reactors must be operated to ensure no fuel burnout (i.e., no melting) during equilibrium operation at full level, and for the abnormal events of sudden flow loss, failure of pumping power, or inadvertent power surges.

Fuel melting during equilibrium operation due to subcooled burnout is improbable at the power levels studied, and would not present an operational limit. With appropriately designed rear fittings, and with the selection of a sufficiently high front header pressure, the use of the Panellit gauges and the application of the present Instability Limits would provide adequate flow monitoring protection at the higher power levels. Adequate flow monitoring protection can probably be achieved with front header pressures as low as 250 psig, although pressures in the range of 350-400 psig are more desirable. Preliminary studies indicate that some means must be provided to assure adequate fuel element cooling during the transient shutdown following the failure of a process tube front connector assembly. The method considered is to pressurize the rear cross header system to permit rear-to-front cooling during the short transient, and it is indicated that rear header pressures in the range of 40-80 psig will be required at the higher power levels studied. This pressure will be dependent upon the tube power and upon the design of fittings and the fuel-tube geometry. Redesign of front and rear fittings should be scoped not only for hydraulic considerations, but also for ease of maintenance and continuity of operation.

Recent studies indicate that during the summer months at the power levels which have just been attained, the coolant backup capacity is inadequate to provide acceptable levels of emergency cooling. Action is indicated now if the emergency cooling criteria are not to be exceeded during the summer months, and further power increases will require additional increases in the back-up capacity. Increases in the flow rate or capacity in such systems as the High Tanks, Export Water System, and/or the Pump Flywheels will be required. A preliminary evaluation of this problem area indicates that a power level increase of 40 percent will require a 40 percent increase in the High Tank flow rate, and for short time periods a 60-70 percent increase in the Export Water flow rate. These capacity increases are substantial, and deserve further detailed study.

Reactor Control and Nuclear Safety

In evaluating the status of nuclear safety, it has been assumed that reactor operation at the higher power levels would not be conducted outside the existing safety criteria with respect to Water Plant Reliability, Speed of Control, and Total Control. Also modifications will not be made such that these criteria cannot be met. Further it was assumed that at the higher power levels, the probability for a nuclear incident to occur would be no greater than at existing levels. In order to meet these requirements, physics calculations indicate that certain control changes will be necessary, depending upon the final design chosen.

In general it may be stated that fuel design modifications should be in the direction of larger rather than smaller fuel elements in order to reduce the need for Supplementary Control Systems to satisfy the criteria for Total Control and Speed of Control. If larger fuel elements are not utilized, substantial additional control capacity will be required at the B, D, DR, & F Reactors at the higher power levels in order to meet the Total Control criteria. This additional control could be in the form of additional vertical safety channels or temporary poison columns. If process tube channels were overbored approximately 200 mils, and the large fuel elements were utilized, the need for the additional control would be significantly decreased. The Speed of Control criteria could probably be satisfied with existing control systems, except at very high power levels (2500-2900 MW) utilizing small fuel elements. The conclusions pertaining to the need for additional control are based on technical considerations. However, whether this control should be provided in the form of additional rods or temporary poison is a matter for economic consideration.

Physics calculations indicate that no unsurmountable difficulties will be experienced during normal reactor operation with control of startup transients or with transient cycling during normal equilibrium operation. However, non-equilibrium control losses could be reduced about 40% by installing approximately eight additional Horizontal Control Rods at C and H Reactors, and fourteen rods at B, D, DR and F Reactors. This change would also permit reducing the front-to-rear flux peaking by 15 to 25 percent.

The above safety considerations have assumed that improved nuclear instrumentation will be installed at all reactors, including power rate-of-rise, octant monitors, gamma compensated log N ion chambers, and the zone temperature monitors. Other process instrumentation systems should be improved or modified as necessary to assure continuing efforts in securing increased reliability and process protection.

Radiological Considerations

Criteria have recently been proposed which would limit the amount of phosphorus-32 and hexavalent chromium which may be dumped into the Columbia River. During periods of low river flow rates, the amount of phosphorus-32 returned to the river may exceed the proposed criteria, and methods for its reduction or elimination are probably required independent of the Expansion Program. Also the amount of hexavalent chromium in plant drinking water may exceed the proposed criteria during certain low water periods under present operating conditions. This latter problem is probably not serious, even with increased coolant flow rates, since the dichromate concentration in process water could probably be reduced to 1.5 ppm for short periods during the late summer months. The above do not take into account the imponderable effects of public opinion on the amount of radioactivity which may be discharged to the Columbia River.

Rapid deterioration of the biological shield due to overheating of the masonite probably will not occur if appropriate precautions are taken. Data indicate that if a gas atmosphere containing 90% helium is maintained, the shielding at B, D, C, and F could withstand a power level increase of approximately 50%. The increase at DR and H would be somewhat less. If a layer of black mint pieces were placed in the outermost lattice and the gas atmosphere contained at least 75% helium, the reactor level could be doubled at B, C, D, and F without serious shielding deterioration.

Personnel exposure rates are expected to increase as the power level is raised. The percentage exposure increase will be roughly 1.2 times the percentage power level increase. Experience indicates that the major fraction of exposure is received from the discharge area, the charge area, and the wash pad. Radiation exposure limit of 3R per year is not to be exceeded.

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Research and Development Programs

Redesign of the reactor systems for operation at the proposed higher power levels will require significant Research and Development efforts, beyond that required for continuity of operation, particularly in the areas of graphite contraction, fuel performance evaluation, physics, and heat transfer. It will be necessary to place greater emphasis on several of these problems than at present. The timing of the Expansion Program and the process development work needed for the NPR must be such that adequate manpower and facilities are available for both programs when required.

STUDY BASES

Potentially three basic possibilities exist for increasing the power levels of the reactors. These include the following:

a. Maintaining the bulk outlet temperature at 95°C and increasing the coolant flow rate.

b. Pressurizing the coolant effluent system and operating with a bulk outlet temperature substantially above 95°C. This could be accompanied by simultaneous flow rate increases.

c. Pressurizing the coolant effluent system and operating with net steam generation from individual process tubes and in the bulk effluent stream.

These three methods are listed in order of increasing complexity. Increasing the coolant flow rates is rather straight forward and represents only a modest extension of technology and engineering. Rapid corrosion of aluminum jacketed fuel elements would be experienced at temperatures above approximately 105°C bulk outlet, and would present a serious limitation on fuel performance unless major improvements were made in aluminum corrosion, or unless an alternate fuel jacket material were used. Generation of steam requires high coolant exit temperatures, approximately 140°C, and requires that substantial heat transfer development work be performed before successful application to the reactor system could be assured. Because the required technology for corrosion, heat transfer, and hydraulics has not been developed sufficiently for the more severe service, this study has primarily considered only that case of increasing power levels through an increase of coolant flow rate.

The parameters utilized in this study are those outlined by L. W. Lang in HW-60786 RD,[1] and were supplemented by three additional cases, i.e., 95°C bulk outlet and flow rates of 120,000, 135,000, and 150,000 gpm. For reference purposes, these cases are summarized below:

Case	Bulk Outlet °C	Flow Rate, gpm
A	Existing	80,000
B	95	85,000
C	95	95,000
D	95	105,000
E	105	105,000
F	105	95,000
G	95	120,000
H	95	135,000
K	95	150,000

[redacted]

Fuel dimensions, heat transfer data, graphite temperatures, conversion ratio data, and other physics data were calculated for each case using IBM programs prepared by R. J. Shields and R. E. Hall assuming pressure drop across the active section of the tube of 200, 300, and 400 psi. In addition, it was assumed that the reactors could be retubed with (a) aluminum tubes of existing dimensions, (b) zircaloy tubes in existing tube channels, or (c) zircaloy or aluminum tubes having an increase in o.d. of 200 mils (this latter assumes overboring of the graphite channel).

One important ground rule of the study was that it would be permissable to replace the pumps in the 190 Building, but that the horsepower requirements of the replacement pumps must remain unchanged. Thus in order to achieve the flow rates indicated above, the hydraulic resistance of the piping and reactor system must be reduced. The following flow and pressure data were supplied by R. T. Jaske and N. F. Fifer which meet this requirement:

Top of Riser Pressure, psig	Reactor Flow, gpm
510	95,000
450	105,000
350	120,000
280	130,000

Using these bases as outlined above, studies have been made to evaluate the problems and technical considerations associated with graphite contraction, corrosion, fuel performance, heat transfer and hydraulics, reactor control and nuclear safety, radiological engineering, and shield deterioration. The results of these studies are discussed in the following sections.

GRAPHITE DISTORTION AND OLD REACTOR LIFE

Since the phenomenon of continued contraction of graphite under neutron irradiation at temperatures above about 200°C was substantiated about 1953 or 1954, the useful life of a reactor has been viewed as limited by the inability to charge fuel elements into distorted process tubes. This is, of course, not a life until a fixed date. Life of existing reactors could extend until it is no longer economic to take corrective action to permit charging and discharging fuel elements. It has been assumed, for present purposes, that a reactor life will extend until it is no longer possible to charge 4" fuel elements into a substantial number of tubes in the upper part of the reactor and it is no longer possible to ream, broach or drill to straighten process tube channels and to retain reasonable structural integrity of the graphite moderator. More particularly it is presumed that structural integrity of the moderator will be maintained so long as in straightening process tube channels no cutting takes place outside of the graphite blocks in which the tube channel exists. Further, it is presumed that other changes can be made readily, such as modification of horizontal control rod tips or of vertical safety rods that may be required as the graphite continues to contract and distort. This study has not considered the economic aspects of making such changes in the graphite stack.

Recent manometer traverses of process tubes in the top center of the old reactors indicate the vertical distortion from the original straight tubes to be as tabulated below:

Reactor	Inlet Hump (About 9' in) inches	Center Depression inches	Outlet Hump (About 9' from end) inches
B	+0.9	-1.5	+0.4
D	+1.4	-1.5	+0.5
F	+1.3	-1.4	+1.1
DR	+0.5	-1.3	+0.4
H	+0.4	-2.0	+0.4
C	+0.3	-1.5	+0.2

At present power levels the center depressions in the several reactors are dropping at approximate rates tabulated below:

Reactor	Rate of Drop of Center Depression Inches/yr.
B,D,F	0.35
C	0.4
DR	0.45
H	0.5*

(*A period of rapid rate of drop of the center of H between 1956 and 1958 contributes to the high rate figure for this reactor. The reason for the rapid rate is not yet apparent. It is possible that with continued operation at the present power level, the sustained rate of drop for the H Reactor after 1958 may be shown to be between o.4 and 0.45 inch/yr.)

Under present operating conditions the elevations of the humps are remaining fairly constant. At B and F Reactors the height of the inlet humps has been lowered about an inch over the past few years by annealing of the expanded graphite through use of longer fuel element charges, positioning the charges off center slightly toward the inlet face, and through use of selected control rod patterns, thus skewing the flux toward the inlet face. A lesser amount of annealing has been achieved at D Reactor through the use of a longer fuel element charge. It appears that practically the extent of annealing obtainable by these means alone may now have been achieved.

It is, of course, not the extent of elevation or depression of the process tube which prevents passage of a fuel element. It is rather the sharpness of the bend or curve at the junction between two graphite tube blocks or at the location of a transverse fracture of a tube block which prevents the passage of fuel elements. One calculates, and it has been verified by experiment, that the bend in a B, D, F ribbed process tube at the junction of two new tube blocks becomes sufficiently sharp to prevent passage of a standard 8" fuel element when the tube blocks become out of line by about 0.4 inch per foot of length. A standard 4" element cannot pass through a bend between new tube blocks greater than about 0.7 inch per foot.

When a bend of 0.4 inch per foot develops as it did at the inlet hump in process tube 4676-F in June, 1957, and with some wear of the ribs in the tube, it was not possible to pass a 9" probe (the standard 8" fuel element is 8.87" long) beyond the inlet hump. At that time that process tube was removed, the tube broached with a 16" to 18" long broach, as is standard procedure before inserting a new process tube, and a new process tube inserted. With the straightening of the process tube channel achieved with the broach, it was immediately possible to pass a 12" probe past the inlet hump. It is still possible over two years later to charge 8" fuel elements into 4676-F even with the continued drop of the top center of the reactor totalling about 0.7 inch, because of the 0.6 inch annealing of the inlet hump that has intentionally been achieved as described above.

It should also be mentioned that in addition to the distortion of process tubes in the vertical direction, there is a horizontal component of distortion which is of larger magnitude the further the process tube location is from the vertical center plane of the reactor. The variation of displacement horizontally through the process tube or channel is quite similar to the variation vertically. The same factors contribute in both directions. The greatest horizontal displacement outward from the vertical center plane exists at the same position along the process tube as the inlet and outlet humps; the greatest displacement inward, at the approximate mid-point of the length of the process tube. Because the side faces of the graphite moderator stack are not fixed in position as is the bottom face, horizontal distortion can take place in both directions on each side of the vertical center plane. The degree of horizontal distortion is therefore less than the degree of vertical distortion in the top part of the reactor.

At F Reactor over the last two years there have been several instances when it has not been possible to discharge and charge 8" fuel elements in process tubes in the upper part of the reactor using the normal charging procedures. In these instances, the exposed fuel elements have been forced from the tubes using special procedures, the process tube removed and a new process tube inserted after broaching of the tube channel. As described for 4676-F this has again made it possible to charge and discharge 8" fuel elements in these tubes. There have been a few instances similar to this at B Reactor but, as yet, none at D Reactor.

What is described above shows that the condition has been reached at F Reactor in which the tube channels and process tubes of the upper part of the reactor are sufficiently distorted and bent that more and more frequent instances of inability to discharge and charge 8" fuel element will occur. Each instance can be relieved through the described procedure of tube replacement including broaching. It is a matter of economic and operating consideration as to whether to accept these instances or whether to revert to the use of 4" fuel elements which cost more per unit of uranium, take somewhat longer to charge, contribute to a slightly lower plutonium conversion and which, because of the greater number of fuel elements that would be used, may contribute to a slightly higher fuel rupture frequency. An evaluation of this situation is presently being made for F and B Reactors. The evaluation will likely be applicable to D Reactor in six months to a years time.

Experiments are currently being conducted to demonstrate the benefits in straightening the bends in a process tube channel that can be achieved using a specially designed drill about 5' long rather than the to-date standard 22" long broach. As indicated above, in the upper part of B, D, and F Reactors after a standard broaching of a channel, a second straightening would be necessary in 2 to 3 years to continue use of 8" fuel elements in tubes in that part of the reactor. Satisfactory application of the long drill will condition a process tube channel so that a second straightening operation would not be necessary for 4 or 5 years to continue use of 8" fuel elements.

[redacted]

It is estimated that at present power levels, use of standard 4" fuel elements (4.43" in length) after it is no longer possible to discharge and charge 8" fuel elements in a distorted process tube in the upper part of the reactor will permit satisfactory charging and discharging for an additional 8 years. Successful use of the longer drill at that time would recondition the tube channel so that 4" fuel elements could be used for perhaps another 12 years.

In considering the use of self-supported fuel elements in ribless tubes and making calculations of the lengths of time over which continuing distortion of tube channels can be tolerated without requiring corrective action, it is recognized that specific dimensions of the element and the supports, as well as the inside diameter of the process tube, must be considered. It is also necessary to consider the conditions under which the collapse of the supports shall be or can be accommodated. Combinations of certain dimensional arrangements and allowable conditions will permit appreciably more process tube bending than can be accommodated with present fuel elements in ribbed tubes. (Certain combinations and conditions can be more restrictive than present elements in ribbed tubes. For example, self-supported elements with 80 mil high supports, 1 1/16" in length at the quarter or seventh points of the element with the outside diameter of the supports 15 mils less than the inside of the process tube will permit only about three-fourths of the bending that can be accepted with present elements in ribbed tubes if the supports are not to be collapsed somewhat in the fuel charging operation.)

The data in Table I have been prepared assuming the use of self-supported fuel elements which will permit one and a half times as much bending of the process tubes as can be accommodated at present. It should be noted that the data in the table are presented with reference to the extent of distortion in the upper center part of the reactors. Corrective actions indicated at the various approximate dates will be necessary only in the upper parts of the reactors, say, the upper fourth or third. Time periods for satisfactory use of the channels in the lower two thirds on three fourths of the reactor are much longer. The data can be used to indicate the timing for various programs. It might be decided to go to the use of 4" fuel elements in perhaps the top 10 to 15 tubes in a reactor. Satisfactory operation could continue at present power levels until about 1969. If at that time the process tube channels were overbored using a long style drill, retubed with larger diameter ribbed tubes for use with self-supported fuel elements, and operation resumed at a 20% increase over the present power level, satisfactory operation with 8" elements could be achieved for an additional 4 to 6 years - to 1973-1974. Another 8 to 10 years could be obtained at that time by going to the use of 4" self-supported elements. This would be into 1981-1984. Thus the larger process tubes installed in this schedule would have a useful life of 12 to 15 years. During this period of time it might be necessary to use 4" fuel elements somewhat below the top 15th row of tubes if those tubes had been installed at the same time as the top rows. Following 1981-1984, a second drilling of the process tube channels in the top rows with the same long drill used previously could be made to extend the operating years still further.

The schedule described above is anticipated to be feasible even if the graphite should still be contracting at the same rate as at present per unit of energy dissipated in the graphite. Such continued contraction to the 1984 date indicated above would lead to a depression of the top center of the graphite moderator of over 10". This represents approximately a 3% linear contraction

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TABLE I

LIMITING DATES FOR CHARGING OF FUEL ELEMENTS IN B, D, AND F REACTORS

Estimated dates when particular fuel elements cannot be discharged and charged in a significant number of tubes in the uppper part of the reactors.(1)

	PRESENT POWER LEVEL			120% POWER LEVEL AFTER 1960		
Fuel Element and Tube	Date	Increment in Years(2)	Date After One Straightening(3)	Date	Increment in Years(2)	Date After One Straightening(3)
8" Ribbed Tube	1961	4-5	1965	1960+	3-4	1963
4" " "	1969	12-15	1981	1966	9-12	1975
			Date After Straightening in 1962			Date After Straightening in 1962
8" Self-Supported	----	6-7	1968	----	4-5	1967
4" " "	----	18-21	1980	----	12-15	1974

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1. Assumes rate of contraction of graphite per unit of time is proportional to the power level and the temperature of the graphite in °C. The temperature of the graphite in °C is assumed to be proportional to the power level. The rate of contraction, therefore, is assumed to be proportional to the square of the power level.

2. Increment in years available through straightening tube channel with long drill.

3. A second straightening with a long drill is considered feasible. This will extend the operational data another increment as indicated. Under some schedules a third straightening might be feasible.

[redacted]

over the height of the reactor. While there is no direct experimental evidence at hand as yet of saturation of graphite with respect to contraction damage, no samples having received a sufficiently great exposure to neutron irradiations, certain considerations of the structure and properties of the polycrystalline graphite bars suggest that contraction may not proceed beyond 2 1/2 to 4%. Further it might be expected that as the full amount of contraction possible is approached, the rate of contraction per unit of energy dissipated in the graphite may become less. To the extent that such a condition prevails and to the extent that perhaps the rate of contraction is less than proportional to the temperature of the graphite (see footnote to tabulation), then the dates of the tabulation and the schedule presented as an example are conservative. It may be about one year before experimental data can be obtained to permit evaluating these two factors more firmly.

The comments above have been made specifically with respect to B, D, and F Reactors. In two major respects the graphite moderators in DR, H, and C Reactors differ from that in B, D, and F in relation to contraction. The graphite in DR, H, and C was not operated for an extended time at low temperatures as was the case for the graphite in B, D, and F and in which case there was a tremendous growth of the graphite; and, the tube block arrangement in H and C incorporated trunion blocks, undercut tube blocks, and in C, overboring of tube channels, neither of which factors is included in the arrangement in B, D, and F. Because of these factors the problem due to graphite distortion in DR, H, and C are not so advanced as in B, D, and F, even though the total exposure of the graphite in all of the reactors is about the same to date. Especially the tube curvature at C Reactor is very gentle. When difficulties do materialize they will be of a different nature than in B, D, and F. For example, it appears that restriction to the passage of fuel elements in DR and H (and probably C as well) will first develop in the center depression of the process tube rather than at the inlet or outlet humps which are not so high as those at B, D, and F. The means by which the tube channels at DR, H, and C could be straightened when it becomes necessary can be largely the same as those which can be used at B, D, and F. Some modifications will undoubtedly be necessary at H and C due to the presence of trunion blocks.

ALUMINUM TUBE CORROSION

A summary of anticipated corrosion rates of aluminum process tubes has been prepared by N. R. Miller. It was assumed that (1) the "R" value for ribless tubes containing self-supported elements is 1.05, and for ribbed tubes, 1.15; (2) 30 mils of aluminum would be corroded from the tube before replacement is required; (3) water mixing elements are used in the ribbed tubes to minimize top-of-annulus corrosion rates; and (4) no allowances were made for wear of tubes due to fuel charging or for tube external galvanic corrosion.

The corrosion data are summarized below, expressed as anticipated tube life:

	Bulk Outlet Temp-°C	
	95	105
A. Maximum Tube Outlet Temperature	(Tube Life-Calendar Months)	
1. Ribbed Tubes	34	16
2. Ribless Tubes	61	34

B. Average Tube Outlet Temperature	Bulk Outlet Temp.-°C 95	105
1. Ribbed Tubes	110	33
2. Ribless Tubes	180	55

These data indicate that with a bulk outlet temperature of 95°C, the life of an aluminum ribbed tube would be in the order of 3 to 9 years, and for a ribless tube, 5 to 15 years, depending upon the service conditions. These of course, discount external tube corrosion and effects of galling during fuel charging operations. If the reactor were operating at 105°C bulk outlet, the tube service life would be reduced by a factor of 2 or 3.

An evaluation has also been made of the dates when major retubing efforts will be required at the reactors. These data are indicated below:

Reactor	Pessimistic	Probable	Optimistic
B	7/62	1/63	7/63
C	10/62	4/61	9/61
D	1/62	7/62	1/63
DR	7/63	1/64	7/64
H	7/62	1/63	7/63
KE		7/62	
KW		7/62	

Because of the rather severe external corrosion of process tubes at F Reactor the retubing process is rather a continuing operation and is not included above. Recently the B, D, DR, and H Reactors have been utilizing water mixing elements to minimize top-of-annulus corrosion rates of second generation process tubes. If these mixers prove to be capable of reducing TOA temperatures of tubes two years old (or more) to bulk outlet temperatures or lower, the optimistic expectation should be realized. The present condition of the process tubes at C Reactor is not accurately known. This will be established within the next two or three months, and a more realistic estimate of life expectation can then be made.

One factor in evaluating the life of ribless tubes is the effect of galling or wear during the charging and discharging of self-supported fuel elements. At present the extent of such wear is unknown. A program has been initiated to evaluate the effect of fuel element weight, the effect of size and configuration of the support, and the influence of lubricants or waxes.

ZIRCALOY PROCESS TUBES

The incentive to retube existing HAPO reactors with Zircaloy process tubes arises from the high strength, very low corrosion rate at temperatures to 600°F, and low neutron cross section of Zircaloy-2. Based on these characteristics and foreseeable reactor service conditions, Zircaloy process tubes would have a service life equal to or greater than the life of the reactor. It is anticipated that replacement of aluminum tubes with Zircaloy would virtually eliminate all future tube replacement occasioned by corrosion. Further, because of the strength and corrosion resistance, a thinner wall process tube can be utilized, thus permitting flexibility to select a larger coolant annulus or a larger fuel element, whichever is more desirable.

The status of the Zircaloy tube program has been reviewed by D. H. Curtiss,[16] and is summarized in the following paragraphs.

A. General Applicability of Zircaloy-2 as Process Tube Material

The prime advantages of Zircaloy lies in the high strength, very low corrosion rate at temperatures to 600°F and low neutron cross section. An extensive review of the various physical and chemical properties of Zircaloy has been made by G. E. Zima.[29] The properties will be only briefly summarized here.

The ultimate strength of Zr-2 ranges from 64,000 psi to 100,000 depending upon fabrication history and amount of cold work. For existing reactor application where coolant temperatures of 5-20°C at tube pressure of 450 psi at B, D, F, DR, and H Reactors, hoop stress in the tube at the inlet is about 11,000 psi at B, C, F, DR, & H, and 9,000 psi at C Reactor. Thus, from an ultimate strength standpoint, the stress in the tube is a factor of 6-12 less than ultimate strength of the tube. Secondary creep rates at these stresses and temperatures are not measureable in the laboratory.

Extrapolating existing corrosion data, to obtain a decrease in wall thickness of 1% in 20 years requires a service temperature of approximately 350°C. At temperatures up to 150°C corrosion is negligible.

Galvanic corrosion, as observed with aluminum tubes in wet graphite, has not been observed in laboratory experiments. No galvanic corrosion was observed on the H Loop process tube or the KER process tubes.

Abrasion or corrosion associated with vibration will occur with Zircaloy in a manner similar to the old C slug chattering problem. Insofar as chattering slugs are concerned, Zircaloy is only moderately better than aluminum.

Hydrogen embrittlement and even total degradation of the metal can occur at temperatures in excess of 400°C. At temperatures envisaged in old reactors, essentially no reaction with pile atmosphere is expected. However, fuel failures of a type in which the uranium metal is in contact with the Zircaloy may promote local hydrogen embrittlement resulting in tube replacement. Self-supported fuel elements may ease this problem. This is an area requiring additional development efforts to define the magnitude of embrittlement by this mechanism.

Irradiation effects are limited to the mechanical properties. Irradiation at postulated existing reactor coolant temperatures will increase the mechanical strength and yet retain required ductility.

In-reactor experience with large tubes has been limited to H Loop, KER, C Reactor (3 tubes), D Reactor (1 tube), and the Chalk River NRX Loop. The tubes in H Loop, operated at 200°C for 13 months before catastrophic fuel element failure burst the tube and required tube removal. Examination indicated no adverse property changes except at point of fuel failure. KER has operated successfully at temperatures in excess of 200°C for more than a year with no adverse changes. A zirconium tube was installed in 3586-D in July 7, 1955, and has been in successful operation to date. Some difficulty was encountered in maintaining an adequate seal on the rear face nozzle arrangement and for a short period

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of time water leaked into the reactor similar to a VanStone leak in conventional aluminum tubes. Modifications of the seal arrangement corrected this situation and subsequently no problems have been encountered.

Three Zircaloy-3 tubes were installed at C Reactor to obtain additional operating experience. One tube was removed because of rear VanStone leak, which resulted from a poorly formed flange, and one was removed as a result of a stuck rupture. The stuck rupture was discharged at 7,000 pounds and the tube was then removed because of potential damage.

In summary, the properities of zirconium and zircaloy are quite suitable for utilization as process tubes. The effects of irradiation and operating experience indicate that zircaloy tubes will perform satisfactorily under projected environmental conditions.

B. Process Tube Procurement

The current status of the procurement of ribbed process tubes (BDF size) is discussed in detail in HW-61186, "Summary Report on BDF Zircaloy Process Tube Procurement". To summarize briefly, four contracts for ribbed tubes have been placed for a minumum of 75 tubes each for a total of 300. Because fabrication of these ribbed tubes has not yet been reduced to standard production procedures, the fabrication of the tubes will be completed in two lots by each vendor, the first lot being a pilot for the entire order. The delivery of the first lot of tubes from each company ranges from September to November, 1959. Completion of the orders ranges from January to July, 1960. Further, it is estimated that the final price of the tubes on these initial orders will be approximately $1500/tube. This price is expected to be reduced in subsequent production orders to $1000/tube or less.

Two fixed price orders have been placed for the delivery of approximately 50 tubes each of smooth bore tubes. The tubes have C Reactor OD dimension of 1.767" max. The ID is 1.681" ± .005" with a minimum wall of .035". The orders were placed with Harvey Aluminum Co. and Bridgeport Brass Company for $1029/acceptable tube. Initial delivery is expected in November, 1959 with final delivery in January, 1960.

C. Equipment Development and On-Pile Hardware

In converting existing reactors to the use of Zircaloy process tubes, there are some major development problems in equipment and hardware. In the following section, these items are discussed.

1. Fuel Element Charging Equipment and Procedures. For Zircaloy process tubes using the conventional ribbed geometry-supported I & E fuel element, there is little or no need for change or modification in existing procedure that is uniquely dictated by the use of Zircaloy tubes. However, for the smooth bore or vestigical ribbed tubes using self-supported fuel elements a major change in the charging machine is required. Two prime features must be incorporated into the charging machine. These are 1) the need for more careful handling to insure against damage of the supports on the fuel element, and 2) the ability to index the fuel element such that it can be charged into tubes with full or partial ribs.

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Programs for developing and demonstrating this equipment have been initiated and a prototype package modification to an existing charging machine has been developed.

Further refinement of this initial prototype is in progress.

2. Tube Installation and Removal Equipment and Procedure. Since the advent of mass replacement of the aluminum process tubes, there has been a continuing program to improve tube removal techniques. Development work in general has been directed toward 1) increasing the number of tubes removed per day, and 2) simultaneously decreasing the personnel exposure rate.

   Current techniques and equipment will permit routine replacement at a rate of approximately 40 tubes per day providing the tubes are located in a block. For the demonstration tests at C and F Reactors, such removal techniques are satisfactory, but may be inadequate for mass change-out of all process tubes, where a rate of 100 tubes/day may be required. A different approach than currently used needs to be developed to meet these projected requirements.

   Because of the expected long life of the Zircaloy tubes, routine removal is not anticipated. Rather, removal will probably be restricted to problem tubes. Because of the very high radiation level of discharged Zircaloy tubes, any equipment which is remotely operated must be extremely reliable. Work is in progress on this item. The initial approach is the use of a guillotine and hydraulic push pole.

3. VanStoning Equipment. Equipment has been developed which is applicable to VanStoning both aluminum and Zircaloy process tubes. Because of the highly cold worked condition of the Zircaloy, an annealing step is generally required, and the equipment has been developed for this step. Trial runs in the laboratory have successfully demonstrated the ability to anneal, to trim and remove ribs, and to VanStone. It would appear on the basis of these preliminary results that precise cutting to length prior to tube installation is not required. Work is in progress to determine the feasibility of eliminating the annealing prior to VanStoning. Initial laboratory results are very encouraging that annealing may be eliminated, and two Zircaloy tubes for testing purposes have successfully been installed at the C Reactor with no annealing required.

4. Rupture Removal Equipment. There appears to be no need for modifying existing rupture removal techniques for conventional I & E fuel elements in ribbed tubes as a result of utilization of Zircaloy tubes. However, for self-supported fuel elements some modifications may be required. In general, those factors depending upon tube strength will be somewhat better because of the greater strength of the Zr-2 tubes. Further, it is felt that severely stuck self-supported fuel elements in ribless Zr-2 process tubes will, if anything, be somewhat easier to remove. Studies are in progress to determine what, if any special techniques and/or equipment may be required.

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5. Decontamination. Current decontamination agents (namely Turco 4306-B) are not compatible with Zircaloy. There are thus two decontamination problems: 1) to attain the lowest radiation level for removal of the aluminum tubes in the changeout operation and 2) develop suitable technique and agents for Zircaloy tubes. Programs for item one have been in progress for some time. To date, rear-face-only decontamination has been successfully demonstrated and resulted in lowering rear face radiation levels to 15 mr/hr. Through pile decontamination has not been tried on a full pile because of the adverse affects in tube and fuel jacket corrosion. However, highly encouraging single tube runs have been made. On the basis of results to date, it appears that Turco 4306-B can be used in through pile application to reduce rear face levels to 5-10 mr/hr in the event that a full pile Zr-2 tube changeout in accomplished in one shutdown.

   Work is in progress to develop a decontamination system which can be used after major Zr-2 tube installations. Currently some 12 candidate decontamination agents are under study and it is expected that techniques and agents will be ready for use by early CY-61.

FUEL ELEMENTS

In an expansion program of the magnitude being considered in this study, assurance of adequate fuel performance is an important key in evaluating the feasibility of operation under more rigorous conditions. Not only will the fuel operate under higher neutron and heat flux levels that are currently encountered, but it will be of a size and geometry not yet experimentally tested. Also, the mechanical problems of manufacturing and of handling the fuel elements will be somewhat different from those existing now. It can be concluded from these studies that new areas of technology must be developed for the design, manufacture and handling of fuel elements which will not limit the reactors production rate through performance failure. Several areas of required technology and development are currently being investigated and are discussed in the following sections.

A. Fuel Design Considerations

Depending upon final technical, economic, and engineering analyses, the fuel element-tube configuration to be used in the reactors may be of several types, including the following:

1. I & E fuel in ribbed aluminum or Zircaloy tubes.

2. I & E fuel elements in larger ribbed aluminum or Zircaloy tubes resulting from process channel overboring.

3. Self-supported I & E fuel elements in vestigal ribbed aluminum or Zircaloy tubes.

4. Self-supported I & E fuel elements in ribless aluminum or Zircaloy tubes.

5. Self-supported I & E fuel elements in larger aluminum or Zircaloy tubes resulting from process channel overboring.

Each of these cases represent fuel elements of different dimensions, the fuel element for existing ribbed aluminum tubes being the smallest, and for Zircaloy ribbed tubes in an overbored channel being the largest. Not only do the fuel elements contribute to changes in conversion ratio and reactor control, but also to coolant hydraulics, methods of sustaining the feature of self-support, fuel charging and handling etc.

Using the conditions established for the study cases, P. A. Carlson has calculated typical I & E self-supported fuel element sized utilizing an IBM program prepared by R. E. Hall which would provide nearly equal fuel jacket corrosion rates on both the inner and outer surfaces. The results are shown in Table II. The designs assume eight supports per eight-inch fuel element, and were calculated to correspond to the available pressure drop for the active zone as provided by N. Fifer. It was assumed that a minumum practical annulus for I & E slugs is 0.080 inch annulus for this case would have the dimensions of 1.64" x 0.39" and 1.69" x 0.40" for the 1.800" and the 1.850" tube sizes, respectively. These designs have a tube pressure drop of about 250 psi at tube flow rates corresponding to the 95,000 gpm case.

It is evident from Table II that at the higher flow rates, oveboring is required to obtain fuel elements of larger size than currently used in the old reactors. These designs are conservative in that eight supports of current design were assumed per fuel element. Also, the supports were assumed to be randomly distributed in the annulus along the tube length. It should be possible to reduce the pressure drop taken up in the supports either by eliminating two supports per element, by redesign of the support, or by alignment of the supports in the tube. The latter may be required if present poison spline procedures are required. Rough estimtes indicate that these changes may add a maximum of .010 inch to the diameter of the fuel elements shown in Table II.

It should be noted that the procedures used to calculate these designs were based on experimental data from only one self-supported geometry in the range of the small tube diameters. However, they are believed to be a fair approximation of actual dimensions. Additional verification of the analytical technique is expected to be gained from flow lab tests now beginning.

The results of these design calculations indicate that the final fuel design will emerge as a balance among the factors of 1) available pumping power, 2) decision for overboring process channels, 3) desire for improved conversion ratio, 4) the amount of tube pressure drop available, and 5) characteristics of fuel performance. Larger fuel elements have the characteristics of providing an increased conversion ratio, reduced total control problems, increased heat transfer area, increased thermal stresses, increase pumping requirement (and/or reduced coolant flow rate), and required overboring of process channels. These items and the irradiation performance of typical fuel elements must be thoroughly investigated before final selection of a fuel element design can be accomplished.

B. Irradiation Behavior

Problems involved in irradiation behavior of fuel elements include those of core cleavage, corrosion, and distortion. These are discussed below.

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TABLE II
TYPICAL FUEL ELEMENT DESIGN

RBORE S	TUBE ID INCHES	FUEL OD INCHES	FUEL ID INCH	ANNULUS THICKNESS INCH	CORROSION RATE MILS/MO	TUBE POWER-KW	$S_{I\&E*}$ / Ss	$Q_{I\&E*}$ / Qs
	1.600	1.436	0.365	0.082	3.4	1387	0.46	3.03
	1.650	1.490	0.370	0.080	3.8	"	0.47	2.96
	1.800	1.660	0.373	0.070	3.7	"	0.51	2.74
	1.850	1.715	0.375	0.067	3.4	"	0.52	2.68
	1.600	1.404	0.390	0.098	4.8	1533	0.47	3.24
	1.650	1.460	0.394	0.095	5.1	"	0.49	3.15
	1.800	1.632	0.398	0.084	4.6	"	0.53	2.87
	1.850	1.690	0.398	0.080	4.5	"	0.55	2.80
	1.600	1.330	0.450	0.135	9.6	1747	0.45	3.88
	1.650	1.396	0.445	0.127	8.0	"	0.48	3.63
	1.600	1.570	0.452	0.115	7.3	"	0.54	3.24
	1.850	1.630	0.450	0.110	7.0	"	0.56	3.11
	1.600	1.23	0.6	0.185	12.8	1900	0.31	6.05
	1.650	1.33	0.5	0.160	13.6	"	0.44	4.35
	1.800	1.50	0.5	0.150	11.2	"	0.50	3.80
	1.850	1.56	0.5	0.145	12.0	"	0.53	3.58

indicated tube power
er.

cated geometry.

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1. Core Cleavage. Analysis of the core cleavage problem encountered at HAPO with solid fuel elements is complicated by lack of knowledge of the mechanical properities of uranium as a function of irradiation. It is known that prior to irradiation and at low exposures (up to approximately 200 MWD/T) uranium is relatively ductile. However, as irradiation continues, embrittlement evidently occurs and core cleavage sometimes occurs under adverse stress condition. One technique of analysis developed by O. E. Adams[14] is to assume that the uranium behaves elastically. It is then possible to calculate relative steady state stresses in fuel cores and to calculate ratios of specific power at which maximum tangential stresses are equal in solid and I & E fuels. Results of such calculations are shown in Table II.

   Even though it is known that elastic stress theory does not accurately predict stress levels for these cases, it provides a suitable tool for comparative purposes. It can be concluded that at the specific power levels of the fuel studied, cleavage failures should not be a major problem. However, I & E fuel cleavage failures should not be discounted at the higher power levels, since solid failures occurred at tube powers as low as 700 kw/tube. It is anticipated that cleavage failures for I & E elements may again appear at tube powers in the range of 1500-2000 kw.

2. Hot Spot Failures. Corrosion of the aluminum cladding of fuel elements normally occurs by two methods, uniform corrosion and accelerated corrosion. Uniform corrosion is most easily envisioned as a chemical dissolution of the aluminum cladding; whereas accelerated corrosion usually involves an intergranular attack mechanism. For C-64-F alloy aluminum, uniform corrosion rates below about 7 mils per month have in the past been tolerable; however, a finite incidence of hot-spot type fuel failures made higher power operation appear uneconomic.

   Correlations[15] have been developed, based on solid fuel performance relating hot-spot failure incidence to specific power (indirectly tube power), surface or outlet temperature, and irradiation time. These correlations indicate that fuel column rupture potential increases as the 5th power of irradiation time, the 8.7th power of maximum surface temperature, and the 8.3th power of maximum specific power. The extension of this analysis to I & E geometries has been found to fit past experience, and currently goal exposures are being established as a function of power to attempt to reduce HAPO failure incidence.

   Investigations into the causes and occurrences of hot-spots have indicated that fuel misalignment does cause hot-spot formation; and has shown that self-supported fuel elements significantly reduce the incidence of hot-spot flow patterns. This reduction may be either the result of fuel centering per se or the associated decrease in maximum surface temperatures. It is logical to assume from these data that misalignment is a necessary condition for hot-spot formation; however, a test is currently scheduled to remove any doubt from this statement. If in fact misalignment is a necessary condition for failure, use of a fuel geometry which will increase the degree of assurance of alignment within the process tube should permit operation without the classic hot-spot failure at uniform corrosion rates up the the limit of the cladding alloy in use.

3. Fuel Element Distortion. Warp and growth of existing fuel elements as a result of irradiation have not been found to be a direct function of specific power, however, random distortion of fuel elements does occur in the current production fuels. It is anticipated that irradiation of fuel at conditions shown in Table II will, at best, not result in a decrease in the degree of warp, but will probably increase the effect. Since the degree of tolerable warp is reflected in stuck charges and hot-spot failure incidence, and since this degree is a definite function of length, the full core length must be chosen to assure no intolerable warp problem. It appears that fuel of the current length is warrented for the cases in this study.

C. Self-Supported Fuel Elements

Operation of the reactors at the higher power levels will require considerable dependence upon improved fuel performance. It is anticipated that the self-supported fuel designs will meet this requirement. Following is a discussion of characteristics and problems occassioned by the use of self-supported fuel elements.

1. Annular Flow Area. Even though it is possible to design a ribbed tube to center the fuel elements, no assurance of fuel alignment can be obtained unless a third rib is introduced, and this is undesirable for fuel charge-discharge reasons in the Hanford reactors. The self-supported fuel element, having collapsible supports, provides a high degree of assurance of alignment, since collapse of the supports would be required to obtain misalignment. The collapsibility feature essentially guarantees tolerable alignment features (unless the supports are broken off during charging operations), and should significantly reduce or eliminate ruptures caused by hot spots. It is desirable to provide as large an area as possible in the process tube to accommodate a warped or ruptured fuel element. In a ribbed tube this area is limited to the upper annulus area due to the fixed ribs. For a ribless tube with self-supported elements, the area for expansion or warp is almost the entire annulus area provided the supports are collapsible. This feature is important in facilitating the removal of stuck ruptures and badly warped slugs.

2. Flexibility. With a ribbed process tube, changes in the reactor flow resistance cannot be made simply by changing fuel size without affecting flow balance and eccentricity of the elements in the process tube. This problem does not exist with self-supported elements and fuel size changes to take advantage of increased pumping plant capacity or to change the pile flow resistance may be made without adverse fuel performance effects.

3. Use of Poison Splines. One problem introduced by the self-supported fuel element is the operational problem of supplementary poison control. The use of randomly oriented fuel supports in the process would preclude the use of splines in the outer annulus area of that tube. Alternative solutions include poison columns for continuous charge-discharge, placing the spline in the hole of the I & E fuel element, or providing process channels with special ribbed tubes to be used only for poison control.

4. Galling of Ribless Process Tubes. There has been some speculation that the fuel supports may cause serious galling of the process tube during the charge-discharge operation. In an aluminum tube the wall thickness could be substantially reduced by such galling, and in a Zircaloy tube the protective oxide layer would be removed and perhaps permit accelerated corrosion. Proper design of the fuel supports may alleviate the problem, if it exists. No specific data are presently available to assess the nature of this problem and an area of investigation is indicated.

D. Testing of Self-Supported Fuel Elements

To date nine solid and thirty I & E self-supported fuel charges have been irradiated to goal exposures of 500 MWD/T and 900 MWD/T, respectively, in thirteen ribless aluminum process tubes at B Reactor. No hot-spot patterns have been observed on any of the discharged elements, and no failures have occurred in any charge to date. Comparison of these results to observations of regular fuel from the B Reactor indicate a factor of 10 reduction in hot-spot incidence at better than a 95% confidence level. It has been further demonstrated that under the test conditions the collapsible support has adequate strength, does not introduce corrosion or erosion problems of its own, and probably promotes water mixing in the coolant annulus. Experience obtained in the KER facility, however, indicates that fuel jacket corrosion under the supports may be a potential problem.

A test is currently being conducted at high specific power, high surface temperature to compare the relative rupture resistance of self-supported and regular fuel operating at similar maximum surface temperatures. This test is scheduled to run until failure of a self-supported element and will provide valuable rupture data, high corrosion rate data, and support damage data. Future testing will be directed toward evaluation of alternate types of supports, and a demonstration test of up to 100 tubes at C Reactor. This test is scheduled for reactor charging starting in January 1960, and two complete cycles of fuel from these tubes should be available for evaluation by October, 1960.

To date a sufficient number of self-supported fuel elements have not been irradiated to permit an accurate statistical evaluation of their performance. It is therefore desirable to increase the number of self-supported charges so that a reasonably accurate performance evaluation can be made.

E. Fuel Jacket Corrosion Behavior

Calculations have been made to determine the probable fuel jacket corrosion rates expected under the various cases of the Water Plant Expansion Study. The calculations were made on the following basis:

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1. Slug Sizes, tube flow rates, flow split between the hole and annulus channel, tube power, and heat generation rates were taken from Shield"s study.

2. It was assumed that with self-supported elements the maximum "R" value, i.e. ratio of the maximum Δt across any part of the annulus to the bulk annulus Δ t would be 1.05. With ribbed aluminum tubes, the maximum "R" value was taken to be 1.15.

3. No decrease in corrosivity of plant process water was assumed.

4. It was assumed that 20 mils of jacket could be "safely" removed before fuel element failures became prevalent.

5. The use of X-8001 aluminum cladding was assumed.

From these assumption, the following maximum exposures were calculated:

TABLE III
ALLOWABLE EXPOSURES BASED ON UNIFORM JACKET CORROSION

Case	Active Zone ΔP, psi	Allowable Exposure MWD/T: Al Tubes w/no Overbore	Zr Tubes w/no Overbore	Al Tubes 200 Mils Overbore	Zr Tubes 200 Mils Overbore
A	200	1370	1260	1130	1038
	300		1280		1080
	400		1310		1060
B	200	1340	1180	1150	1050
	300		1180		1050
	400		1200		1050
C	200	1270	1320	1090	1025
	300		1240		1065
	400		1265		1030
D	200	1260	1260	1120	1020
	300				1020
	400		1250		1050
E	200	650	700	600	560
	300		680		580
	400		680		570
F	200	650	690	600	890
	300		680		610
	400		730		620
G	200	1150	1220	1080	1040
	300		1210		990
	500		1210		990

[redacted]

H	200	1290	1210	990	960
	300		1170		940
	400		1170		970
K	200	1040	1150	990	930
	300		1110		930
	400		1140		930

This study may be summarized by stating that except for the two cases E and F, whose bulk outlet temperature is 105°C, the goal exposure may be carried to as least 900 MWD/T without experiencing limiting conditions arising from the jacket corrosion of self-supported elements.

There are several items to be noted which are not readily apparent from the above figures.

1. Fuel Element Design. The fuel elements in the above cases were designed so that annulus and hole exit water temperature would be equal. A more common basis has been to equalize hole and annulus jacket corrosion rates. The practical results of the Shields' type design is that under a given condition of tube pressure drop, flow, power generation and inlet temperature, the "Shield" type element will exhibit higher maximum corrosion rates than an element based on equal hole and jacket corrosion rates. In the particular cases under consideration, annulus corrosion rates exceed hole corrosion rates with ribbed tubes and the reverse holds for self-supported elements. This then is an indication that the exposures shown in the figures are somewhat conservative if it is to be assumed that the actual fuel elements will be designed on the alternative basis.

2. Type of Cladding. It has been demonstrated that aluminum alloy C-64-F experiences an accelerated corrosion attack at high temperature and heat flux. This rapid corrosion attack would probably preclude the use of this material for power increases obtained with a 95°C bulk outlet and a reactor flow rate of 100,000 gpm. Although the X-8001-F alloy has demonstrated superior corrosion resistance at the higher temperatures, difficulties have been encountered with groove pitting and bond integrity. It would be necessary to reduce the severity of these problems before the X-8001-F would be considered as an acceptable cladding material. A continued intensive fuel cladding development program will be required to achieve the more rigorous service conditions postulated in this study.

3. Inlet Water Temperature. Shields assumed a 12°C inlet water temperature and this temperature was used in all corrosion calculations. At a given tube power, allowable exposures would be roughly doubled during winter months and halved during late summer months.

4. Validity of Corrosion Rate Calculations. Corrosion rates were calculated from a semi-theoretical equation which seems to fit observations of jacket corrosion rates reasonably well. In some cases, however, the combined conditions of heat flux and surface temperature calculated for the various cases somewhat exceeds our experience and the required extrapolation is not certainly accurate.

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In particular, extrapolation required for cases H, K, E, and F were progressively more severe and the calculated results are therefore progressively less certain. This is particularly unfortunate inasmuch as the allowable exposures for these cases appear to be more and more borderline. Happily, though, tests to be initiated soon to evaluate self-supported elements will produce corrosion data which can be utilized to test the required extrapolations.

5. Possible Improvement in Process Water Quality. It is possible that the process water could be made less corrosive to aluminum by a reduction of process water pH below 7.0. Such a reduction in corrosivity is not assured and the effects of such a change on other materials plus added chemical costs would have to be determined.

EQUIPMENT AND PLANT MODIFICATIONS FOR SELF-SUPPORTED FUEL ELEMENTS

In addition to fuel performance characteristics at the higher operating levels, several equipment changes will be required to facilitate handling the self-supported fuel elements. The reactor changes to be considered include charging machines, nozzle modifications, discharge chute modifications, and new tools to facilitate charge-discharge. These topics are not discussed here since it is assumed that others are working on these details. Also, some changes will be required in the 300 area canning facilities. These changes are summarized briefly in the following paragraphs.

A. Process Considerations

1. Canning Cycle. Since the mass of uranium in the proposed fuel designs are quite different from the normal production fuel elements, it may be necessary to develop new canning cycles which allow the core to reach the proper canning temperature prior to assembly. At present, the development of different canning cycles is a cut-and-try process which may require as much as 6 months lead time to achieve the optimum for one geometry, once the final size has been determined. With the heavier elements, a longer cycle (perhaps 50-55 seconds) may be required, thus reducing throughput.

2. Canning Yield. Experience has shown that the introduction of a new fuel model may require an extended learning period (perhaps in excess of 6 months) during which the yield is appreciably reduced. During the introduction of the I & E fuel element, the yield was between 40 and 60 percent for several months

3. Process Unknowns. Only a limited amount of work has been done on large ID fuel elements. However, there is some reason to believe that a geometry effect may exist with respect to spire assembly. As the spire size increases there may be problems assoicated with brittle bonding not now encountered. In addition, it is necessary to develop the proper relationship between the spire-core and can-core annulli to promote wetting of cans and spires. If proper wetting cannot be achieved by a modification of the annulli, it may be necessary to investigate the use of fluxes in either the canning bath or duplex. Programs of this nature are involved and probably would require 6 to 9 months of development effort.

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As the OD of fuel elements increases, larger sleeves are required. If the dimensions of the sleeves are increased, it is likely they will float in the Al-Si bath unless the wall is made thicker. This, of course, upsets the heat transfer rate and may require a different canning cycle.

4. Self-Supports. By the time this program is ready for production quantities, the feasibility of the self-support program will have been established. In support welding, no problems are foreseen as a result of the larger diameter fuel elements.

B. Equipment Modification

1. Acme Gridley Lathes. The maximum OD which can be held in the collets of the present lathe is 1.496". Anything larger will require at least one additional lathe at a cost of an estimated $50,000.00 and a delivery time of perhaps 12 to 16 months.

2. Welders. The maximum OD the automatic welders will receive is 1 3/4 inches. Modification of the welders will be relatively inexpensive but require 6 months for retooling.

3. Baskets. Fuel elements in excess of 1.500" will require new baskets for pickling of component parts, canning and etch, and autoclave equipment. Cost may be about $50,000.00.

4. X-Ray Trays. Fuel elements in excess of 1.500 inches may require new trays for a total cost of $50,000.00. Six months will be required to design and fabricate suitable equipment.

5. Nondestructive Test Equipment. Modification of the test heads, tracks and feed mechanism for fuel elements in excess of 1.500 inches would require approximately 6 months and perhaps $30,000.00.

6. Miscellaneous Equipment. Approximately $20,000.00 would be required for miscellaneous equipment such as modifications to conveyors, gages, inspection stations, etc.

7. Aluminum Components. The lead time required for production quantities of aluminum components is about 5 months after the purchase order is written. An additional 2 to 3 months may be needed for development orders requiring new prints, tools and testing prior to acceptance. Small scale developmental orders may be available in 3 to 4 months.

8. Steel Sleeve. A lead time of about 4 to 5 months is usually allowed for receipt of steel sleeves.

9. Uranium. 4 to 5 months lead time on uranium cores is required for other than normal sizes.

Due to less reduction required during the rolling of some of the larger sized rods, there may be some change in the metallurgical properties of the uranium. The degree and seriousness of this problem can not be evaluated at present.

10. Storage Space. If the expansion program is justified for all reactors, additional storage space may be required during the transition from one product model to another because of high inventories required for aluminum and uranium components.

C. Reactor Testing

Since no allowance has been made in this study for unusual problems which may arise as a result of proposed changes, irradiation testing of a limited quantity of fuel elements of the selected geometry is recommended. Among the items to be studied would be bond deterioration, unusual temperature conditions, and uranium. Two reactor cycles would give a fairly good feel for any unusual problems not now foreseen.

HEAT TRANSFER CONSIDERATIONS

Of the basic engineering factors limiting reactor power levels, the requirement of adequate fuel element cooling at all times is a factor of prime importance in considering reactor design changes and increased power levels. In most Hanford process tubes, as the flow is reduced at constant power generation, a point is reached where the flow rate drops to a very low rate soon after the instant of change, and slug melting may result if the reactor continues to operate. To prevent this damage, process variables for each tube must be controlled, and in the event of a serious flow reduction or power surge, the reactor must be shut down rapidly. To allow the initiation of a shutdown, instrumentation sensing the tube outlet water temperature and pressure at certain points in the flow passage is provided. Of these, primary reliance is placed on the pressure instrumentation which is in the scram circuit. The requirement is that the pressure instrumentation will detect conditions of inadequate coolant flow and automatically initiate a reactor shutdown. With the decreased rate of heat generation immediately following the scram, adequate cooling flow should be available to prevent slug jacket melting and gross damage even though the condition causing the inadequate coolant flow prevails.

The point at which continued operation would cause serious overheating of the fuel elements and the response of the instrumentation to these conditions is affected by changes in tube power, flow rates, and flow resistances.

The following sections discuss the effects of the design changes and power levels on individual tube burnout protection. Both sub-cooled and flow instability burnout during accidental transient operations are considered.

A. Study Bases

The flow, power, and pressure data on which this study is based are shown in Table IV, and were provided in part by the Reactor Design Modification Unit.

TABLE IV
EXPANSION STUDY BASIS

Reactor Flow	Tube Flow	Tube Power	Front Header Pressure	Active Zone Δ p	Panellit Pressure	Bulk Outlet
95,000	52.7	1390	495	362	258	95°C
105,000	58.3	1530	432	292	228	95°C
120,000	66.7	1750	328	186	184	95°C
130,000	72.2	1900	254	106	145	95°C

[redacted]

95,000	52.7	1550	495	334	262	105°C
105,000	58.3	1710	432	275	236	105°C
120,000	66.7	1960	328	153	186	105°C
130,000	72.2	2110	254	93	152	105°C

Obtaining these flow rates and pressures assumes 7-pump operation with present pump motors and redesigned pumps, front and rear fittings similar in size to those at the K Reactors, existing nozzles and gas seals, existing piping, and new front and rear headers. At each set of operating conditions aluminum and Zircaloy process tubes of both present OD and increased OD (+0.200 inch) are considered.

B. Sub-Cooled Burnout

Fuel melting may occur at conditions where the local bulk coolant temperature is below the saturation temperature if the heat flux is sufficiently high. This phenomenon is termed sub-cooled burnout. At equilibrium conditions the approach to sub-cooled burnout in the Hanford reactors is very low because the cosine heat generation distirbution along the charge length means that theheat flux is low at the position of near saturated coolant conditions. Present estimates, believed to be conservative, indicate that the maximum attainable heat flux at the point of maximum sub-cooled burnout potential in a tube of K Reactor dimensions if of the order of 1.5×10^6 BTU/hr - ft^2. The point of maximum burnout potential is determined by plotting the normal operating heat flux and maximum attainable heat flux (determined by local thermodynamic properties describing the coolant) as a function of position in the process tube. The closest approach to burnout assuming a cosine power generation occurs about seven feet from the outlet of the active charge. At present K Reactor power levels the maximum equilibrium heat flux at this point is about 500,000 BTU/hr - ft^2 indicating a safety factor of 3. Increasing the power levels to the maximum values considered in Table IV decreases this safety factor to abour 2.

During accidental flow reductions, critical flow occurs in the rear fittings causing pressurization of the tube and suppression of the start and degree of boiling. Thus, although the coolant temperature in the tube is rising, the saturation temperature also goes up which delays attaining sub-cooled burnout conditions at some point along the charge length of relatively high heat flux. The result is that flow instability and the probability of net boiling burnout at the end of the charge is the more critical situation. Thus, sub-cooled burnout per se is not believed to represent a limit to achieving the conditions of Table IV. This conclusion is supported by limited laboratory data on the K system which appears to substantiate existing empirical techniques used to evaluate the maximum attainable heat flux. However, several factors peculiar to the Hanford case not taken into account in the analysis or in the experiments might affect the burnout process. Chief among these is an eccentricity of the coolant channel around the fuel element and skewed flux patterns. Also, the maximum attainable heat fluxes for the normal case is determined inexactly. A program is scheduled for the Thermal Hydraulics Laboratory to determine the limit to steady-state heat flux in all the Hanford geometries and to investigate the effects of non-uniform conditions. This program is expected to shown significant results during the coming year.

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C. Flow Instability

1. Present Instability Limits. Protection against flow instability and fuel melting in the Hanford reactors is achieved through individual tube pressure instrumentation. The requirement is that the instrumentation will detect conditions of inadequate cooling flow and will shutdown the reactor automatically. With the decreased rate of heat generation immediately following the scram, adequate cooling flow should be available to prevent slug jacket melting and gross damage even though the condition causing the inadequate flow prevails. The present flow protection criteria is not expected to prevent damage in the event of a complete and instantaneous flow reduction, since the reactor scram which would be immediately initiated would minimize damage, whereas the low reactor power levels required to ensure complete flow protection are not justified on an economic basis. Present philosophy is based on obtaining a reactor scram before or at the point at which unstable flow conditions (i.e. self-induced flow reduction) occur in the process tube. This philosophy is based on an extensive backlog of laboratory data which have shown that a large degree of safety exists between the onset of flow instability in the tubes and film boiling. Limits are applied in the form of outlet temperature restrictions by first determining the degree of flow reduction which just causes flow instability, and second, the response of the pressure sensing Panellit gauge to this flow change. Low and high trip pressure settings are applied so that in the event the tube is plugged either upstream or downstream of the Panellit pressure tap, an automatic scram will take place before or at the point of flow instability.

   In the case of plugging downstream of the Panellit tap, a high trip is arbitrarily set to occur after a 30% or greater flow reduction. These trip settings are not affected by flow instability because both the plugging and two-phase pressure drop is downstream of the gauge tap. This incremental flow reduction corresponds to a outlet water temperature rise of about 30°C. Unstable flow occurs in the old pile geometry at about 160°C (or equivalent enthalpy) and so the outlet water temperature limit is at 130°C.

   Instability conditions occur at approximately identical conditions if the plugging is upstream of the Panellit tap, but the response of the gauge to this plugging is markedly different. The difference is caused by the fact that the upstream plugging is followed by pressurization resulting from boiling in the rear fittings. The Panellit gauge pressure first falls to a value depending on the amount and rapidity of the flow reduction, and then proceeds towards the high trip as increased boiling and pressurization occurs in the rear fittings. The criterion of Panellit low trip specifications is that a low trip is intended to occur in the event of rapid upstream plugging. However, a limitation on the permissable trip span is applied because the high trip will then be capable of providing adequate protection and backup for the low trip in most tubes.

   The minimum permissable low trip pressure is that pressure at which the flow just goes unstable. Outlet temperature limits are imposed at or below the high trip temperature limit depending on the range in pressure between the normal Panellit pressure and the low trip pressure. With

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smaller Panellit pressure spans, the outlet temperature limit is higher because the sensitivity of the gauge to a low trip is increased. In practice most tubes have trip settings which will initiate a scram before the flow reduces to the point of instability. This is possible because the maximum limits which could be set now are higher than of current interest.

2. Effect of Expansion Study Changes on TAI Limits. The currently specified instability limits for the old reactors are valid for front cross-header pressures greater than 450 psig(2) and are strongly influenced by critical flow in the rear Parker fittings. Table IV shows that under the expansion study ground rules, increased flow and power levels correspond to decreased supply pressures and enlarged rear fitting. The effect on currently specified instability limits of decreased supply pressures for the old reactors is in the direction of decreasing both the high trip outlet water temperatures and the limits determined by the low trip pressure. The decreased supply pressures also, in general, mean lower venturi throat pressures which decreases the driving force which sends the gauge toward the low trip and this also results in lower limits. Instability limits using present empirical techniques(3) were determined for the cases in Table IV to determine the feasibility of using the existing panellit system and protection philosophy at the new operating conditions, and to locate points where adequate protection is not provided by the existing system requiring a change in philosophy or instrumentation or where limits are too restrictive for practical operation. A comparison of present old pile high trip outlet water temperature limits with those calculated for the expansion cases are shown in Table V.

TABLE V
INSTABILITY LIMITS - CURRENT SPECIFICATIONS

Case	Reactor Flow	High Trip Outlet Water Temperature Limit
Present	85,000	130°C
I	95,000 gpm	130°C
II	105,000 gpm	130°C
III	120,000 gpm	120°C
IV	130,000 gpm	110°C

As indicated in Table V, increased flow and power corresponding to decreased supply pressures result in decreased outlet temperatures limits. Also the rate of transition from flow instability to fuel melting increases with tube power and, thus, the response time of the Panellit gauge becomes increasingly more critical as the power is raised. With respect to the high trip, both events that occur with downstream plugging are in the right direction and a faster response to a high trip is obtained. With a flow reduction caused by an event upstream of the Panellit tap, the Panellit pressure first falls to a value depending on the amount of the flow reduction and remains at this level until boiling, which is occurring downstream, pushes the gauge toward the high trip. If the Panellit gauge response time is not adequate, a low pressure trip may not occur. Under present operating conditions, the initial decrease in Panellit pressure corresponding to the flow decrease prevails for about two seconds after the flow reduction occurs. At higher tube powers, this time may be decreased because of more rapid pressurization due to boiling at the end of the tube, and the gauge may not respond in time to assure reaching the trip point. The requirement is that the gauge must respond to the pressure indicative of a flow change during the shortest period in which the indication prevails. This problem may become critical at higher tube powers and will have to be verified experimentally before these levels are considered routinely. Also, lower supply pressures mean lower ventui throat pressures which further reduces the sensitivity of the gauge to obtain a low trip. Thus, increased power level associated with decreased panellit pressures mean decreased limits or more restrictive trip spans to provide the necessary gauge sensitivity. The latter method of course increases the probability of unwanted scrams due to gauge oscillations.

This analysis indicates that limits for Cases I and II based on existing empirical techniques will be no more restrictive than those in present use at the K Reactors, but that tube outlet water temperature limits are reduced for Cases III and IV. This may not be serious in Case III, but may prevent the attainment of power levels corresponding to a 95°C bulk outlet water temperature in Case IV unless improvements were made to the orificing efficiency and the present Panellit gauges replaced by a direct flow measuring instrument. However, even in Cases I and II some of the conservatism currently present in the existing old pile limits will be removed. The reason for this is that critical flow occurs in the Parker fitting in the present old pile geometry which causes a rapid pressurization of the tube under conditions of flow reduction or power surge, suppression of the start and degree of boiling, and delays the onset of film boiling with its attendant rapid slug surface temperature rise. The delay in the onset of film boiling coupled with the large Panellit pressure increase makes possible the use of the Panellit high trip in the event a sudden upstream plugging does not affect a low trip. In the expansion study cases, less restrictive fittings are used so that the critical pressure is less than the saturation pressure. Thus, boiling progresses farther into the process tube during a given transient. If the pressure is low enough, high steam qualities may result in the process tube, and fuel burnout may occur before "unstable flow" is observed. Laboratory experiments with the K system (also proposed for expansion study cases) have shown that this double protection (i.e., high trip backup) is available on tubes operating above 700 kw. It has also been demonstrated that under certain conditions fuel element burnout can be obtained in tubes below 700 kw without experiencing flow instability or getting a high pressure trip. A similar situation would be present in the old piles equipped with K fittings and should be recognized and avoided by design of the rear fittings. A possible solution is to provide the proper flow distribution by adjusting the area of the rear fittings and using venturis in the front fittings of all tubes to obtain the desired Panellit pressure. The cross-sectional flow fitting could be designed so that all tubes exhibit similar boiling curves and provide each tube with similar high trip protection.[4-10]

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It has been concluded in these studies, that in the range of flow rates and pressures seriously being considered, the Panellit gauges will offer adequate protection. However, an agressive development program must be pursued to accurately define the operating conditions under which the existing flow monitoring system fails to function as required.

3. <u>What Can be Done to Increase Limits</u>. The question arises as to what steps must be taken to attain the power levels represented by the high flow cases and still provide adequate protection against fuel melting. Presently specified instability limits reflect conservative assumptions in application to the reactors, principally from the standpoint that laboratory data have shown that a significant delay in the transition from incipient instability to burnout is provided by pressurization in the rear fittings. However, even the present philosophy does not ensure against fuel melting in cases of complete and instantaneous flow reduction, but depends on the decreased rate of heat generation immediately following the reactor scram to minimize damage. A philosophy which would permit higher outlet water temperatures implies acceptance of a greater degree of risk in case of incidents of this type, and further increases the probability of fuel melting during lesser transients.

However, over the years, tube outlet temperature limits have been extended with increasing knowledge from the initial concept of excess header pressure to the present philosophy which operates on the basis that a reactor trip will be initiated at the onset of unstable flow conditions in the process tube. The laboratory data which has made this transition possible also shows that a sufficient safety factor exists between the onset of unstable flow and fuel burnout, indicating that even less restrictive limits may be possible, and yet maintain a sufficient degree of safety.

An extension of the existing philosophy would be based on tripping before burnout. To establish such a philosophy requires laboratory work to accurately define where burnout occurs and the technical limitations required to prevent fuel melting under various conditions. Also an evaluation and acceptance of greater risk in operation of the reactor under the new philosophy would be required.

An alternate approach, suggested by John Batch, would be to start with the existing knowledge developed in the laboratory and develop a header-to-header system optimized for heat transfer. This is justified on the basis that ultimate reactor limits will be determined to a large extent by heat transfer considerations. The present system was developed primarily from material considerations. It may be possible, for example to develop a practical coolant system in which unstable flow does not occur over the entire range of possible flows to burnout. Flow protection may then be applied over a wider range of Panellit trips since gauge response would not be an overriding factor. Or it may be desirable or required to abandon pressure instrumentation in favor of flow meters set to trip at specified outlet water temperatures. A laboratory program aimed to optimize the flow system for heat transfer could be completed within the two year limit suggested by the study. The program would aim to specify a flow system to provide optimum heat transfer with self-supported I & E fuel elements in overbored process tubes. To establish limits, burnout points would have to be determined as well as determining where to trip to prevent damage under the various transients expected in the reactor. Adequacy of present pressure instrumentation would have to be checked, and if required, improved flow monitoring instrumentation developed.

D. Front Fitting Failures

The most probable cause of complete loss of supply pressure to a tube during reactor operation is the failure of a front face connector. While such an incident would immediately initiate a scram, the power reduction is not immediate because of the time required for insertion of the rods, and the continuing post-scram decay heat. The only coolant available after such an incident is that of hot water forced back through the tube by the pressure in the rear header. This problem has been investigated in the laboratory to determine the amount of rear header pressure required to prevent serious damage to the reactor components upon loss of a front connector.[4] The results which are of a preliminary nature indicate that a tube of the K geometry operating at 1000 kw can survive the transient without melting if the rear header pressure exceeds 30 psig. The rate of transient is increased at higher tube powers, and a higher minimum header pressure would be required to prevent serious damage at power levels above 1000 kw.

Two methods are suggested to cope with the problem of front fitting failures. The first is to provide adequate rear header pressurization to obtain sufficient reverse flow during the shutdown transient. The second is to develop a front connector assembly which demonstrates performance reliability considerably greater than the existing flexible connectors.

No experimental data have been obtained to determine the amount of rear face pressurization which would be required at the higher tube powers. Estimates based on limited laboratory data indicate pressures in the order of 40-80 psig may be required depending on the power level, and that some decrease in fuel element diameter would be necessary, perhaps as much as 0.025 inch at the higher flows to permit the desired transient flow rate. It thus appears that the required pressurization would be costly in terms of equipment, loss of coolant capacity, decreased conversion ratio, and an increase of supplementary control to satisfy the Total Control Criteria. It should be emphasized that no firm technical data have been developed upon which to specify this type of coolant requirement. However, the distinct possibility exists that some level of pressurization will be necessary, and sufficient funds should be included in this study for that purpose. A detailed laboratory program is being conducted to assess the nature, consequences and transient coolant requirements for this type of coolant failure. A program is also being conducted by Equipment Development Sub-Section to develop improved connectors which do not carry the same high degree of risk as the present flexible connectors.

COOLING SYSTEM BACK-UP

A study has been completed by S. S. Jones, HW-62861,[12] which evaluates the capacity and adequacy of the secondary and last ditch cooling systems at the present and higher power levels. The following is a summary of that study.

In determining the required system changes for the expansion study it is necessary to employ certain basic criteria to serve as a guide for establishing system capacities, etc. The study of the reactor back-up cooling systems makes use of the reactor cooling safety criteria recently adopted by the Irradiation Processing Department and adds certain specific requirements listed below:

1. The reactor bulk outlet temperature shall not exceed 90°C at 40 or more seconds after a scram.

2. The reactor water flow shall not be less than that required to assure no boiling of the coolant, or in no case less than 2000 gpm.

3. The reactor water plants shall always have the water storage requirements as listed in Water Plant Standards, HW-27155.[13]

The flow decay for the present system was obtained experimentally. This information was extrapolated to the study cases G and K using realistic flywheel decay conditions. The required flow, based upon the requirements of 90°C bulk temperature limit and assurance of no boiling was obtained by reference to the reactor heat-output characteristics at various times after shutdown. The adequacy of the secondary and last ditch cooling systems was then determined by a direct comparison of the required and expected flow decays, shown in Figures 1 to 6. A study of the secondary cooling system, i.e., the steam turbine pumps that are in parallel with the electrically driven process pumps, shows this system to be adequate for all the plant expansion cases as long as assurance is given that at least 6 steam driven pumps will be placed in full operation within three minutes after the loss of electric power.

The last ditch system provides reactor cooling in the remote event that event that both electric and steam power are lost at an area. There are two components to this system: the hi-tanks and the export line. An important aspect of their adequacy is the temperature of the water in these systems when called upon to cool the reactor. Experimental measurements made during the summer of 1959 indicated hi-tank temperatures as high as 37°C and export temperatures up to 27°C, and are shown in Figure 7. A temperature of 30°C has been assumed for this adequacy study on the basis that only minor modifications are required to maintain hi-tank temperatures below 30°C, and that export temperatures would never exceed this value. The flow characteristics of the present last ditch system were extrapolated to the more severe study cases and the flow adequacy determined. The results are summarized as follows:

			PRESENT HI-TANK			PRESENT EXPORT		
Study Case	Flow 1000 gpm	Power Level mw	Flow Avail. 1000 gpm	Flow Required 1000 gpm	% Adeq.	Flow Avail. 1000 gpm	Flow Required 1000 gpm	% Adeq.
A	80	1800	11	12	92	3.8	4.5	85
G	120	2600	13.5	17	79	4.5	7.2	63
K	150	3300	14.7	22	67	5	9.8	51

This indicates a last ditch system of questionable adequacy at present conditions during the summer months and definitely inadcquate for the plant expansion program, and should serve to emphasize that modifications will be required of the last ditch system independent of an expansion program.

Four basic ways of improving this last ditch system reliability have been studied.

1. Minor modifications to increase system flow 10 to 30 percent and maintain temperature $\leq$30°C.

2. New or additional pump flysheels.

3. New hi-tanks with increased elevation.

4. Large increase in export system reliability and capacity.

The first method is required for the existing operating conditions (Case A) and should provide adequate cooling during the summer months. The second method would help in study Case A, but is of questionable value in the other two cases. Methods (3) and (4) are interrelated since the hi-tank size and flow rate determine when the export must take-over and the lower the export capacity requirements. This is indicated in the following tabulation which suggests various ways of achieving adequate last ditch cooling:

Study Case	FULL LEVEL Flow 1000 gpm.	FULL LEVEL Power mw	NEW HI-TANKS Flow 1000 gpm.	NEW HI-TANKS Cap. 1000 gals.	OLD HI-TANKS Revised	OLD HI-TANKS Flow 1000 gpm.	OLD HI-TANKS Cap. 1000 gals.	EXPORT New	EXPORT Revised	EXPORT Flow 1000 gpm.
A	80	1800	No	No	Yes	12	600	No	Yes	5
G-1	120	2600	No	No	Yes	17	600	Yes	---	9
G-2	120	2600	17	300	Yes	15	600	Yes	---	7
G-3	120	2600	17	800	No	13	600	No	Yes	5
K-1	150	3300	22	330	Yes	18	600	Yes	---	9
K-2	150	3300	22	800	No	14	600	Yes	---	7

For Case G at a flow rate of 120,000 gpm, there is a significant choice between hi-tank capcity increase and modifications to the export system, this selection ranges from no extra hi-tank and doubling the export capacity to more than doubling the hi-tank capacity with only minor export system modification. At the present time our export system data is of questionable accuracy. Basically, the study has indicated the need for improving our knowledge of this system and for funds in a design scope for its modification. The export requirements will be double the values shown in the foregoing Table for the B-C and D-DR Area. Thus, for Case G it might be desirable to install large hi-tanks at H and F and use the present export systems with minor modifications. At the B-C and the D-DR Reactors the much larger export flow may force the economics towards small or no additional hi-tank modifications and a completely new export system.

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This study also included an analysis of the water storage requirements of the old reactor areas. The results indicated that our present storage capacity is adequate at all the old reactor areas as long as the reactor flow is throttled according to reactor requirements during any isolation period.

REACTOR CONTROL AND NUCLEAR SAFETY

It has been assumed in these studies that no component or system changes would be made which would decrease the assurance of safe operation of the production facilities. It is anticipated that all equipment modifications or procedural changes will embody, wherever possible, provisions for increasing the assurance of safe operation over that which existed before the changes are made. Reactor plant changes will be considered on the basis that such changes fall within the established safety criteria, namely the criteria for Water Plant Reliability, Speed of Control and Total Control. These criteria are discussed in the Hazards Review for the existing reactors, HW-61580.(17)

A. Nuclear Safety

The probability of occurrence of an event which would lead to a major fission product release is not increased as power levels are increased, providing the supporting equipment is not taxed beyond its capacity. Compliance with adequate safety criteria will prevent this eventuality. Considerable R&D work will be required to establish and verify operating limits for components and systems as they are modified from time to time as power levels are increased, and to determine what further modifications will be required in order to permit further increases.

The probability of lesser fission product releases such as might occur during discharge operations is not increased during normal operation since the fission power is reduced to zero before discharge.

The minimum time for remedial action is not significantly changed by increases in power level. For example, at current levels the time required for aluminum process tubes to melt in the event of coolant loss at full power would be about 80 minutes. Increasing the power level by 40% would reduce this time to about 40 minutes.(28) The change would hardly be of operational significance.

The fission product inventory will increase as the power levels are increased. However, the greatest increase will be in the short lived fission products which reach saturation levels early in the fuel exposure period. The percentage increase in the very short lived fission products will be nearly proportional to the power level increase. The longer lived fission products would be less sensitive to power level and more sensitive to residence time (exposure level). At 100 days after shutdown the % increase in gross fission product activity would be less than one third the percentage increase in power level. For example, a 40% increase in power level would yield an increase of gross fission product activity about as follows:

1 hour after shutdown	~35%
100 days after shutdown	~10%

[redacted]

[redacted]

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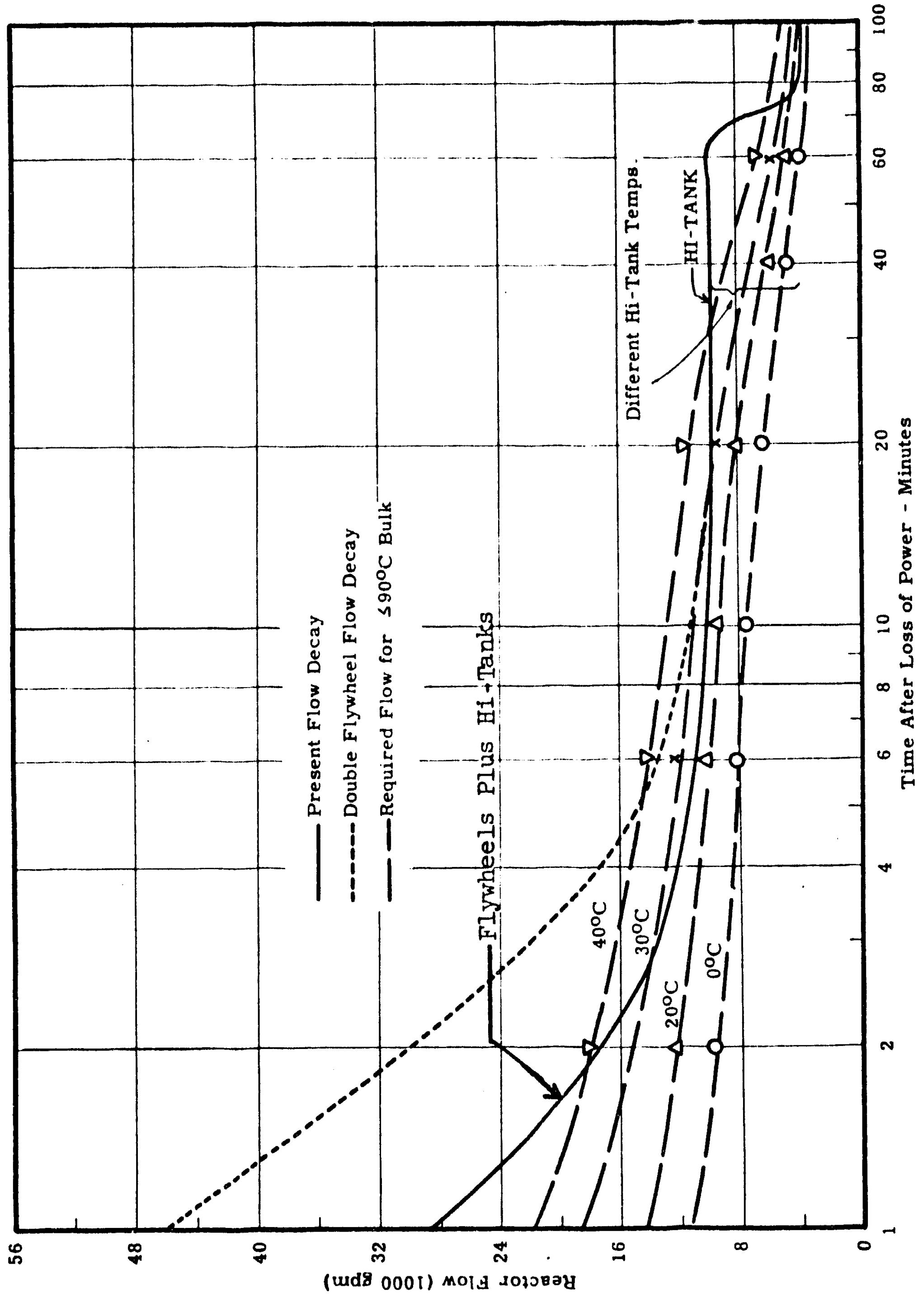

Figure 1. Flywheel to Hi-Tank Conditions for 80,000 gpm Equilibrium Flow Thru B, D, DR, F, & H Reactors (1800 MW, $t_i = 12^{o}C$

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AEC-GE RICHLAND WASH.

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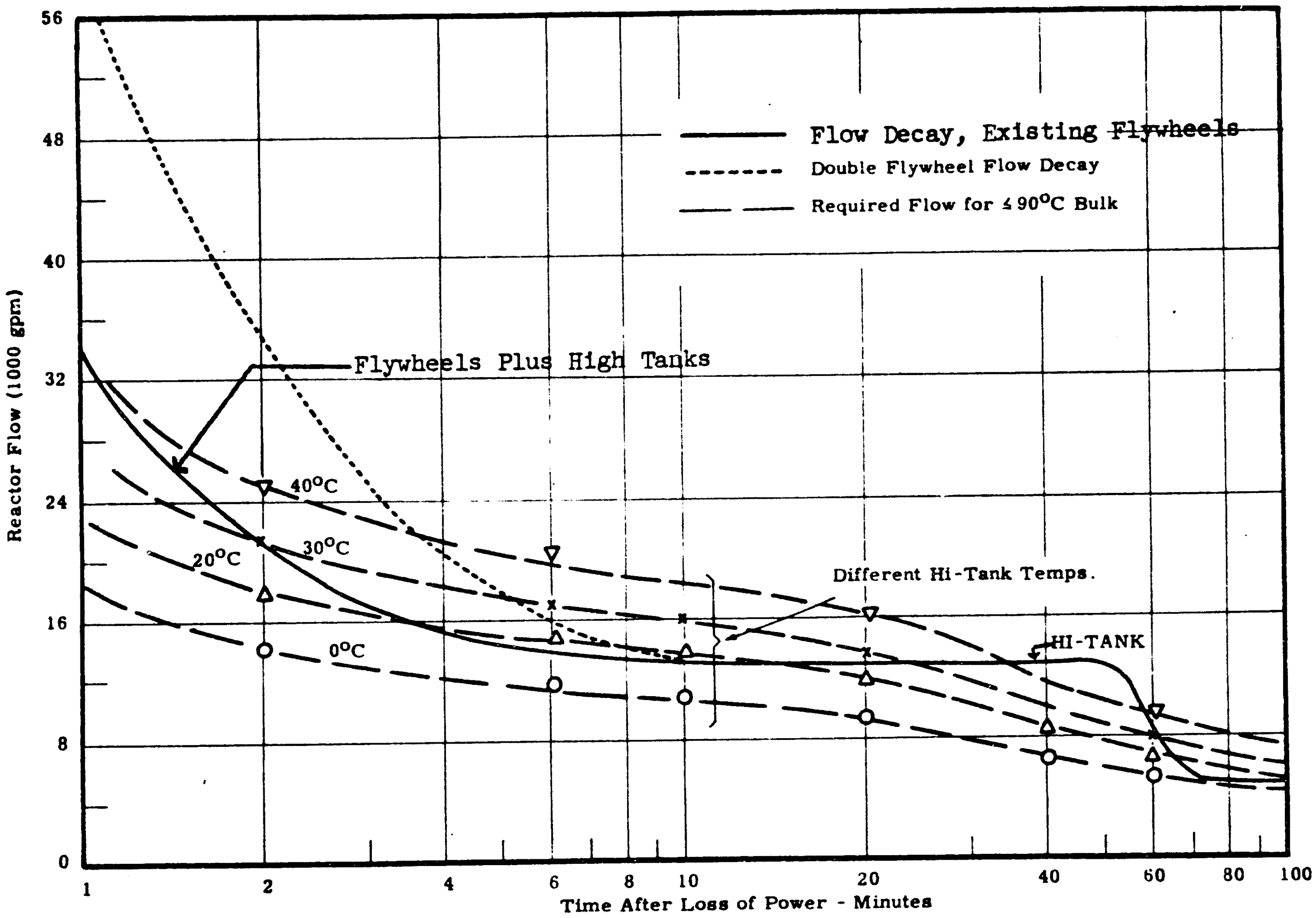

Figure 2. Flywheel to Hi-Tank Conditions for 120,000 gpm Equilibrium Flow Thru B, D, DR, F, & H Reactors (2600 MW, $t_i = 12^{o}C$)

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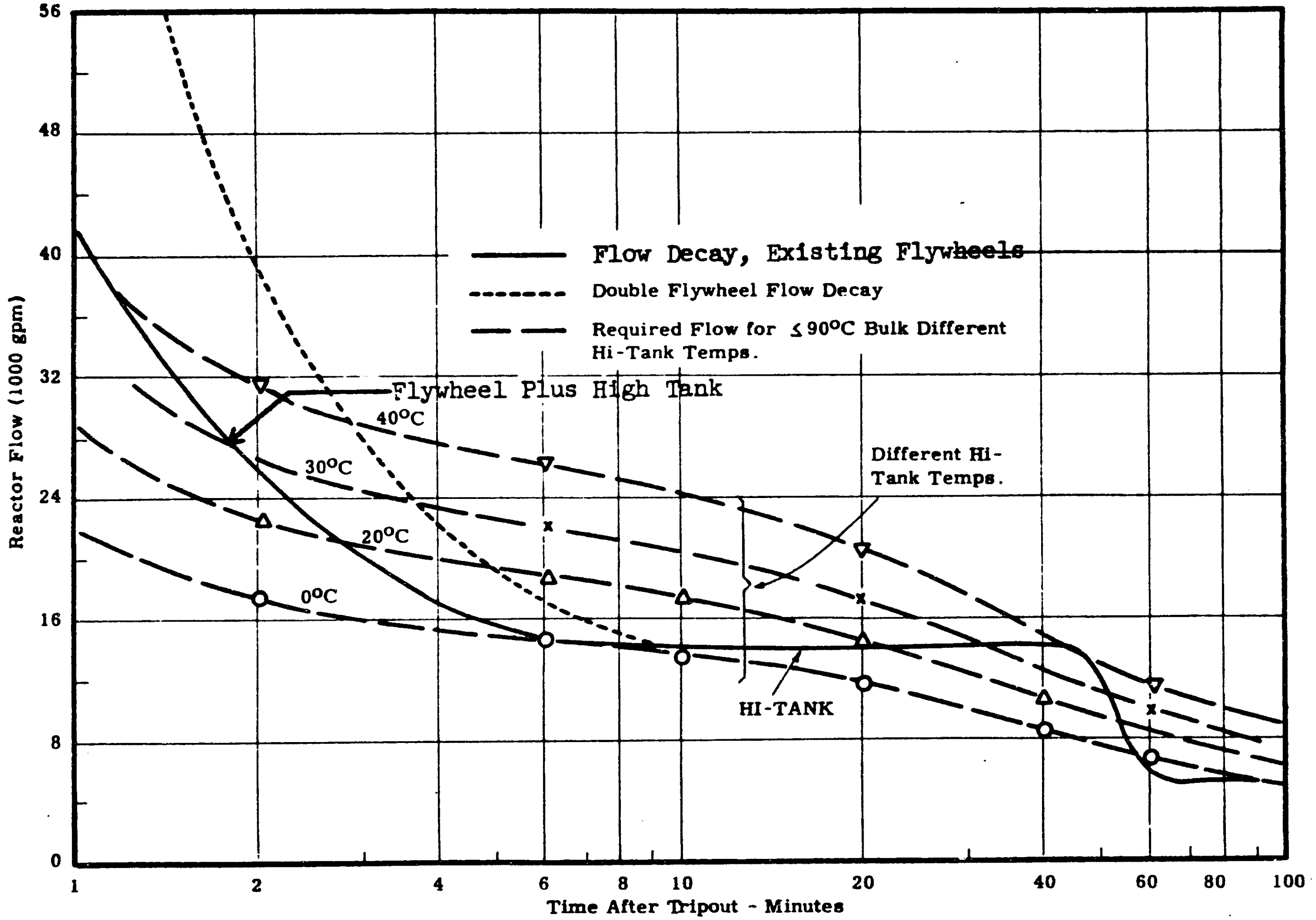

Figure 3. Flywheel to Hi-Tank Conditions for 150,000 gpm Equilibrium Flow Thru B, D, DR, F, & H Reactors (3300 MW)

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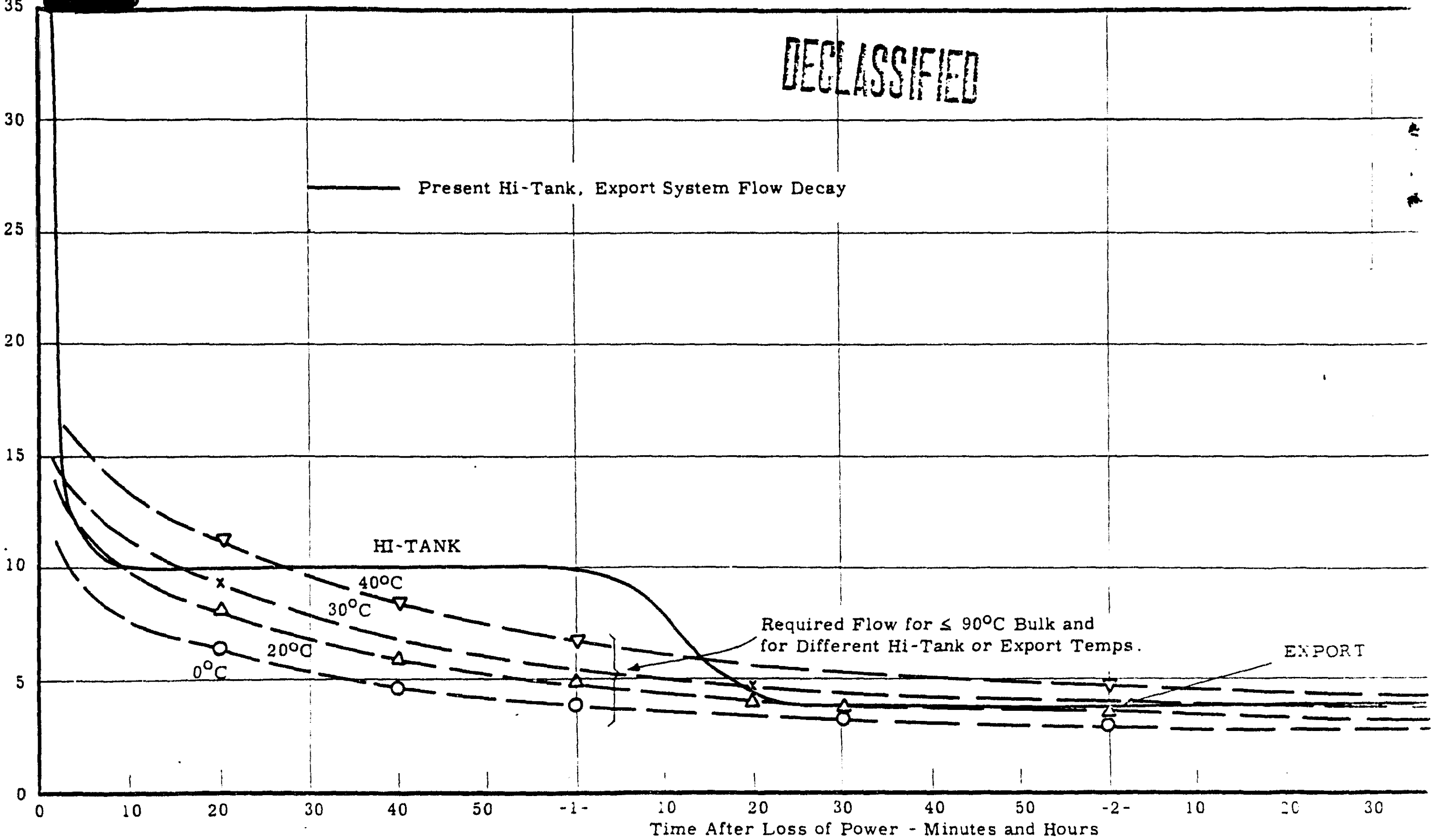

Figure 4. Hi-Tank to Export Conditions for 80,000 gpm Equilibrium Flow Thru B, D, DR, F, & H Reactors (1800 MW) [redacted]

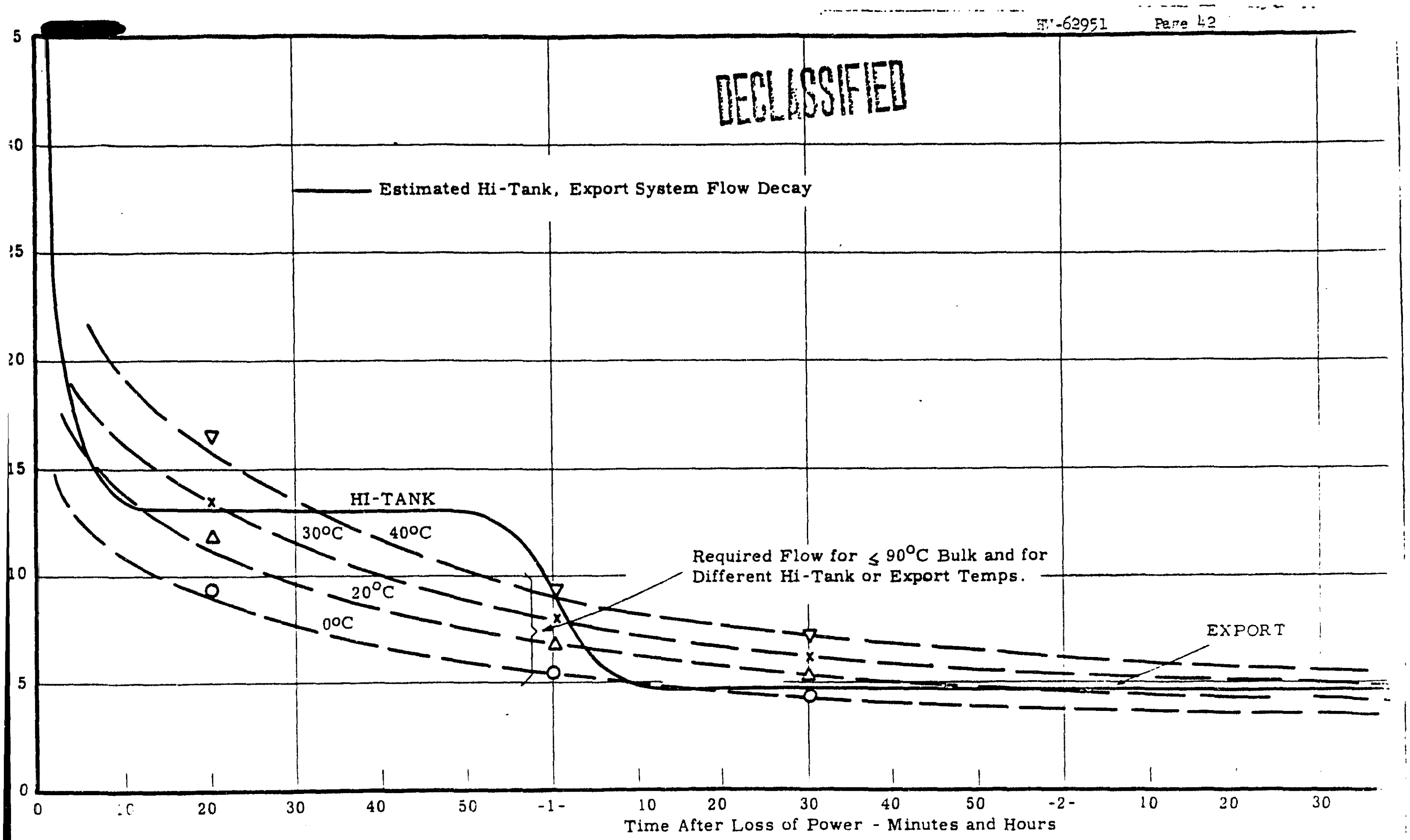

Figure 5. Hi-Tank to Export Conditions for 120,000 gpm Equilibrium [redacted]
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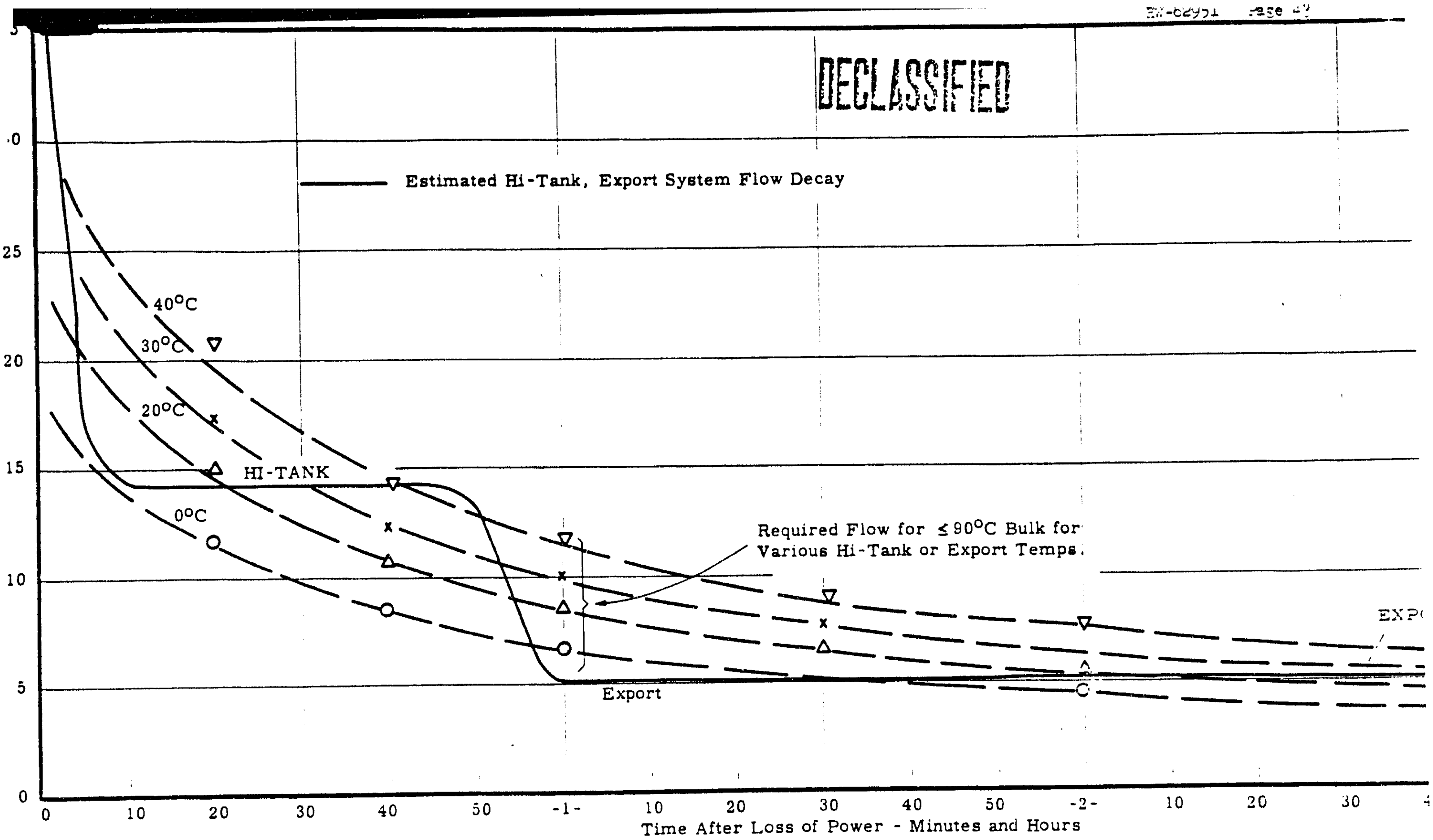

Figure 6. Hi-Tank to Export Conditions for 150,000 gpm Equilibrium Flow Thru B, D, DR, F, & H Reactors (3300 MW)

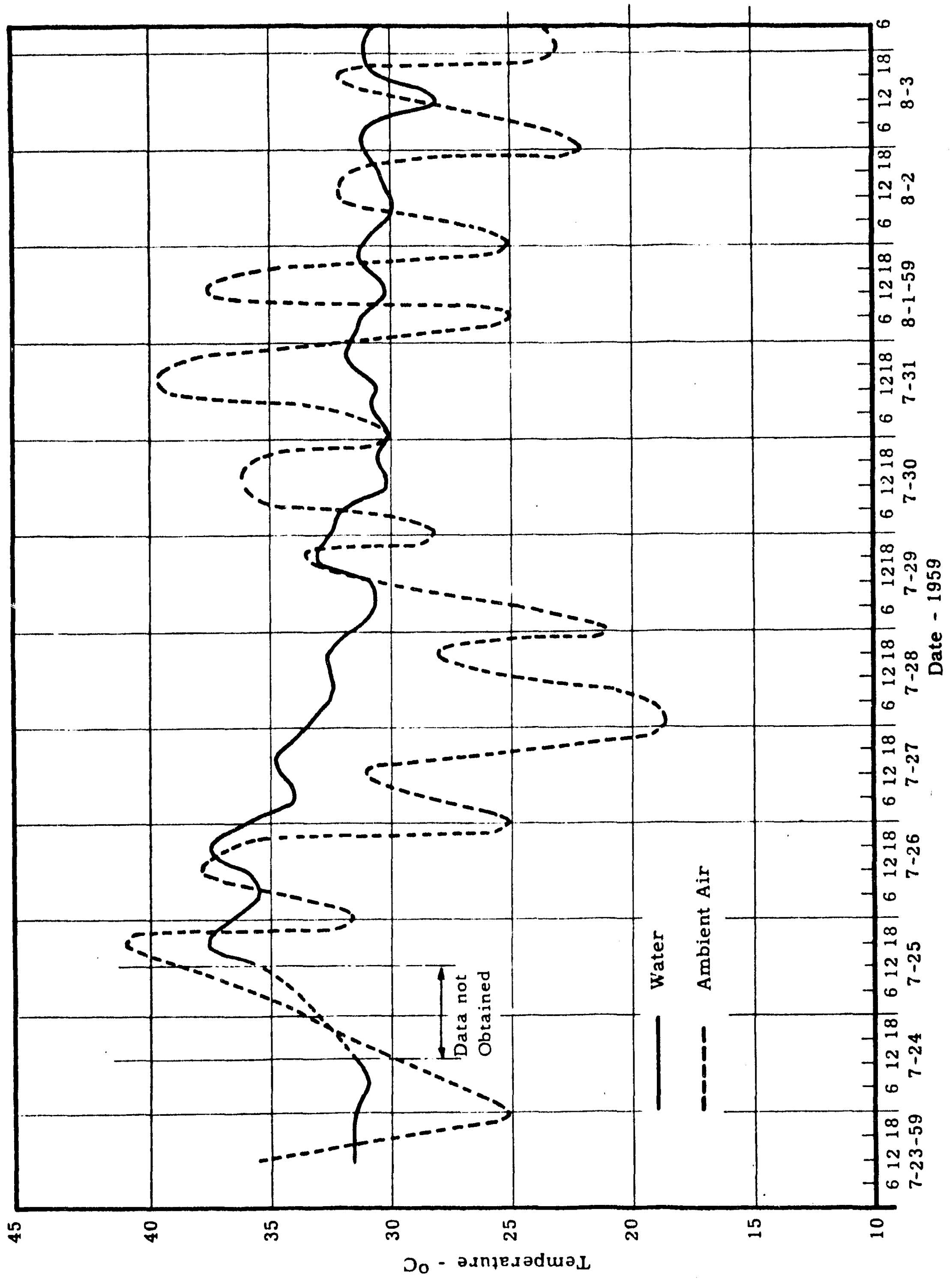

Figure 7. DR-Near Hi-Tank Temp.

The consequences of a potential fission product release due to operational error or loss of coolant will be significantly reduced after CG-791 - Reactor Confinement is completed. For fission product release accidents involving less than several process tubes the offsite consequences may be negligibly small. For major accidents the consequences will be greatly reduced, unless the building walls are destroyed or fission products cannot be directed through the filters.

Already approved equipment modification projects will further reduce reactor hazards, independent of power level. Thus we may be safer at higher power levels with these modifications than we are now. These projects include

CG-707	Subcritical Nuclear Instrumentation
CG-806	Reactor Nuclear Instrumentation
CG-817	Cross-header Differential Pressure Instrumentation
CG-791	Reactor Confinement

B. Reactor Control

Survey studies were made by W. S. Nechodom and D. E. Simpson(27) to evaluate the reactor control problems which may be manifest in terms of operational physics, total control, or speed of control. Using the study cases outlined by Lang, it was assumed that reactor power levels would be increased and that alternate process tubes would be utilized, including the following: (a) use of present aluminum process tubes, (b) use of Zircaloy replacement tubes of current outside diameter and (c) use of Zircaloy or aluminum tubes of 200 mils greater outside diameter. Each case involving new process tubes would require fuel element rediesign.

Data for all cases indicate that no unusual difficulty will be experienced during normal reactor operation with control of startup transients, shutdown reactivites, or normal equilibrium operation. However, the results do indicate that there will exist an increased need for supplementary control to minimize reactivity cycling, and that temporary poison charges or an increase in the capacity of the Ball 3X System will be required at the four oldest piles to satisfy existing Total Control Criteria.

For comparison with Nechodom's study the status of the available control in the reactors may be summarized as follows:

Safety System	B, D, F, DR	C, H	K
	(Control maintained in System, milli-k)		
HCR	15	25	20
VSR	44	70	44
Balls	57	85	56

Nechodom's physics program calculated the cold, dry k_{eff} for several representative cases, then normalized to compare with existing reactor data. The results summarized in Table VI indicate that for existing size tubes at the higher power levels, the cold, dry pile reactivities would be considerably higher than at present, and are due primarily to the use of small fuel elements and higher enrichments. While it would be possible to operate the reactor under the total control limitations imposed by this higher dry reactivity, the reactor operating efficiency would be reduced by the need for additional supplementary total control to satisfy the Total Control Criteria. In the 200 mil overbored cases, the cold, dry reactivities are only a few mill-k higher than the existing base case, and satisfying the Total Control Criteria would require only a small increase in supplementary control beyond that now used.

The foregoing discussion applies only to the four oldest reactors, and not to the C and H Reactors since the latter two have adequate vertical rod and ball 3X capcity to satisfy the Total Control Criteria for the proposed power level increases.

The studies performed by Nechodom indicated that the control response would be adequate at the higher power levels, and that no safety system modification would be required to satisfy the speed of control criteria for the cases most feasible under this study.

A further study has been made by Bowers, Parkos, and Montague to evaluate the need for increased operational control capacity to reduce production losses resulting from non-equilibrium operation and flux cycling. This study concludes that non-equilibrium losses could be reduced about 40 percent by installing approximately eight additional Horizontal Control rods at C and H, and fourteen at B, D, F, and DR. This change should also permit reducing the front-to-rear peaking by 15 to 25 percent.

There are several methods by which the Total Control Criteria could be satisfied for operation at the higher power levels at B, D, F, and DR, no additional control being required for C and H Reactors.

These include the following:

1. <u>Temporary Poison Columns</u>. This is currently a standard method of meeting the criteria, although substantial losses would be incurred by the necessity of shutting down the reactor to discharge poison after adequate xenon build up.

2. <u>Poison Column Discharge During Operation</u>. To minimize the shutdown and non-equilibrium losses incurred in (1) above, Operational Charge-Discharge or Poison Column Displacement procedures could be utilized. The feasibility of these methods has been demonstrated, but the mechanisms and procedures would require continued development for routine operational use.

3. <u>Installation of Additional Vertical Safety Channels to Increase the Vertical Rod and Ball 3X Capacity</u>. At the four older reactors, this would require vertical drilling through the top shield complex and the graphite stack. Adequate control is dependent upon fuel design and power level, although the installation of an additional 12-15 vertical safety channels may be indicated.

The Horizontal Control Rod System that would produce the desirable reduction in non-equilibrium losses and increase pile control ability consists of about 25 rods, representing an increase of 16 rods at B, D, F, and DR, and 10 rods at C and H Reactors. This would eliminate the need for supplementary control except for flattening.

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TABLE VI

OLD PILE EXPANSION STUDY - PHYSICS PARAMETERS

Fuel Element OD	Fuel Element ID	Fuel Element Uranium wt/ft.	Feed Enrichment % U^{235}	At 800 MWD/T 235 B.O.	Pu B.U.	Pile Power Level
45	.310	10.0 #/ft	.714	812	684	1550 MW
	.349	9.31	.731	824	665	2080
	.297	10.18	.720	813	680	2080
	.367	9.02	.734	825	660	2300
	.311	9.96	.721	813	674	2300
	.409	8.30	.741	833	649	2950
	.347	9.39	.723	817	666	2950
	.364	9.91	.717	817	679	2300
	.309	10.86	.708	806	691	2300
	.405	9.18	.722	825	668	2950
	.343	10.28	.709	810	683	2950
	.339	14.05	.729	804	724	2080
	.285	14.96	.732	798	734	2080
	.352	13.76	.727	805	719	2300
	.298	14.74	.729	797	730	2300
	.391	13.00	.725	809	712	2950
	.331	14.13	.725	799	722	2950
	.321	13.34	.734	805	712	2300
	.357	12.69	.732	808	704	2950

.ses

.uminum = 1.609"
.rconium = 1.659"
.ore Aluminum = 1.809"
Zirconium = 1.859"

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TABLE VI (Cont'd)

	Cold Clean Dry Reactivity Normalized and Corrected For Rod Strength	Total Control Position	Cold Clean Wet Reactivity Normalized and Corrected For Rod Strength	Normalized Cold Wet to Dry Reactivity Change	Calculated Hot Wet to Dry Reactivity Change	Speed of Control Position
1.	+ 44	Presently just meet criterion with occasional supp. poison	+ 19 mk	+ 25 mk	+ 15 mk	OK
2.	60	(Exceed criterion	25	35	20	OK
3.	54	(without supplemental	27	27	15	OK
4.	62	(control, resulting	23	39	26	OK
5.	55	(in considerable pro-	26	29	21	OK
6.	64	(duction loss in meeting	17	47	49	Unsatisfactory
7.	56	(criterion by use of (supplementary poison	22	34	43	Unsatisfactory
8.	59	(	25	34	25	OK
9.	52	(	27	25	19	OK
10.	61	(	20	41	48	Unsatisfactory
11.	53	(	24	29	42	Unsatisfactory
12.	47	(	31	16	7	OK
13.	44	(Slightly worse from	34	10	4	OK
14.	49	(standpoint of total	30	19	13	OK
15.	44	(control criterion	31	13	10	OK
16.	50	(than base case (1),	26	24	37	Possible Marginal
17.	46	(resulting in slightly (more production loss	29	17	32	" "
18.	49	(than present in meeting	29	20	14	OK
19.	50	(criterion with supple- (mental poison	25	25	36	Possibly Marginal

Modification of the Horizontal Rod System presents difficulties also, since at the four older reactors the rod room is directly over the control room, and it would probably be necessary to construct a second rod room on the far side of the reactor. Alternatives, such as liquid poison systems, would require considerable development before an operational system could be achieved.

NUCLEAR AND PROCESS INSTRUMENTATION

This study has assumed that improved nuclear instrumentation will be installed at all reactors consistent with the HAPO position of constantly improving the status of nuclear safety. The installation of the instrumentation included in the Plant Improvement Program, HW-62862,(18) must be completed, including the power rate-of-rise, octant monitors, gamma compensated log N ion chambers, and the zone temperature monitors. Other instrumentation systems, including the Panellit pressure monitor system, should be improved or modified as necessary to assure continuing efforts in securing increased reliability and process protection.

WASTE DISPOSAL AND RADIOLOGICAL PROBLEMS

Waste disposal criteria have been studied on the basis that the occurence of river flow rates 72% of normal would not cause the effects of waste disposal to exceed limits. Since the criterion forthe average body burden of phosphorus-32 could be exceeded under existing reactor operation, it may be necessary to reduce the output of this radioisotope independent of an expansion program. Provision to reduce the output of other radioisotopes will be required for most cases where the bulk outlet temperature is 105°C or higher. For reactor flow rates exceeding 100,000 gpm it may be necessary to reduce sodium dichromate concentrations as low as 1.5 ppm during periods of low river flow. No limit has been established as to the amount of heat which may be dissipated to the Columbia River.

A study was made by R. B. Hall to evaluate radiological limitations which may be encountered at higher flow rates, increased effluent temperatures, and higher power levels. Radiological criteria have been proposed, and although not yet accepted as "official", they should serve as a guide for the Expansion Study. These criteria are summarized below:

A. Modifications shall be so designed that radioisotopes which are discharged to the river or which reach the river after discharge of wastes to the soil can be controlled during a year when the river flow rate is normal such that:

   1. The combined effects of the radionuclides introduced from all reactors would result in an annual average concentration of radionuclides in the river at Pasco of no more than 3.6% MPC*.

   2. The combined effects of the radionuclides introduced from all reactors would result in an annual average concentration of radionuclides in the drinking water at any reactor area of no more than 7.2% MPC*.

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*MPC-Maximum Permissible Concentration of radioisotopes in drinking water. The values referred to are those for continuous occupational exposures as stated in Appendix A of the Radiation Protection Standards (HW-25457 Rev. 1).

P32

[redacted]

3. It would be improbable that the annual average body burden of phosphorus-32 for any individual would exceed 0.36 microcuries.

B. The release of hexavalent chromium ion to the river shall be so controlled that the concentration in drinking water will not exceed 0.05 ppm and the monthly average concentration in river water will not exceed 0.02 ppm.

C. Reasonable effort to limit the increase of river temperature due to the operation of the reactors during August and September is recommended. However, the present technical basis will not support the establishment of a limit on the amount of heat that may be added.

The criterion for the release of radioisotopes to the river was based on Radiation Protection Standards 3.1 and 7.2, on the assumption that river flow rates 72% of normal could be expected with reasonable frequency and on the premise that limits should not be exceeded when this low flow rates exists.

The criterion for the release of hexavalent chromium was based on a Public Health Service Water Standard and on recommendations of the Aquatic Biology Operation. The Public Health Service Water Standard, which is followed at HAPO, states "hexavalent chromium in excess of 0.05 ppm shall constitute grounds for rejection of the supply." This is interpreted by Industrial Hygiene to mean that 0.05 ppm hexavalent chromium shall be the maximum acceptable for drinking water at any time.[19] R. F. Foster established a limit of 0.02 ppm hexavalent chromium in river water due to its deleterious effect on juvenile fish.[20] A monetary value of about $60,000 per year can be placed on these fish. That is, if the juvenile fish between the reactors and the confluence of the Columbia with the Yakima and Snake Rivers were destroyed, the commerical fishing industry might suffer a annual loss of about $60,000. This value cannot be considered firm, but is quoted in order to give a basis for judgement.

The criterion for the release of heat is based on the opinion of the Biology Operation that temperatures over 20°C at Bonneville Dam could result in epidemic disease among salmon.[21] The usual high temperature period occurs in late August and early September which coincides with the time that a large run of Chinook Salmon are in the river. The value of this particular salmon run has been estimated to be $5,000,000 annually. Again the accuracy of this number is not guaranteed, but it is included to permit judgement of the problem.

A. <u>Effects on Environs</u>

1. <u>General</u>. The criterion for phosphorus-32 is being exceeded at this time, and it is expected that the operation of Priest Rapids Dam may cause the criterion for hexavalent chromium in drinking water to be exceeded during a winter when nighttime flow rates are reduced to 36,000 cfs.

   Release of effluent water at or near the shoreline except at high river flow rates could cause 100 Area drinking water to exceed these criteria and biological concentration of radioisotopes that could cause islands of foam and algae around Richland boat docks to be a radiation exposure problem.

[redacted]

2. Assumptions. For the purposes of forecasting the effects of proposed changes the following assumptions were made:

   a. Both 105-KE and 105-KW operating with flow rates of 188,000 gpm, with a 93° bulk outlet temperature limit and at 80% time operated efficiency.

   b. Normal river flow rates of 105,000 cfs annual average, 100,000 cfs average during August and September, 60,000 cfs average during minimum flow periods, and 36,000 cfs daily minimum flow during minimum flow periods.

   c. Uniform distribution of effluent water in the river except for considerations of 100 Area drinking water.

   d. Consumption of whitefish flesh at a uniform rate of 0.5 lb/week by a successful fisherman between October 1 and the spring freshet. Fish caught in the vicinity of Ringold.

   e. Hexavalent chromium is neither lost nor reduces to the less toxic trivalent chromium between the point of injection and the point of use.

3. Phosphorus-32. Figure 8 illustrates the potential body burden of phosphorus-32 under various operating conditions. The validity of the extrapolation may be questionable, but it is felt that these values would not be high by more than 25%. It can be seen that even the base case exceeds the criteria. Phosphorus-32 may be obtained from either fish or waterfowl, but because of the large number of migrant waterfowl only the uptake from fish was considered.

   There are two possible remedies for this situation, neither of which has been proven. Treatment of the effluent water by passing it over a bed of aluminum turnings might reduce the output of phosphorus-32 by a factor of 2. Even less is known about the other possibility which would be to substitute CO_2 for sulfuric acid for pH control. Some of the phosphorus-32 comes from the $S^{32}(n,p)\ P^{32}$ and the $P^{31}(n,\gamma)\ P^{32}$ reaction. A development program is indicated to establish the source and elimination of the phosphorus-32 from the effluent coolant.

4. Plant Drinking Water. Hexavalent chromium in plant drinking water may exceed the criterion for drinking water under present operating conditions. This forecast is based on the assumption that none of the hexavalent chromium is lost or reduced and on an estimate of the fraction of B, C, KW and KE effluents that would be taken into 100-D at low river flow rates.[22] Efficient operation of Priest Rapids Dam will probably drop the flow rate to 36,000 cfs during the night while releasing a daily average of 60,000 cfs. The reduction of hexavalent to trivalent chromium and loss of chromium in filter plants has not been studied. Routine analyses of 100-D and 100-F drinking water have shown hexavalent chromium in concentrations slightly above the detection limit of 0.005 ppm on a few measurements when the river flow rate was low.

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5. <u>Radioactivity in River Water</u>. Figure 9 illustrates the effects of various conditions relative to the radiological criterion for river contamination. The choice of Pasco as the point of interest does not ignore the possibility of navigation in the river past the reactors, but does assume that recommended radiological controls would be enforced.(23,24) The method of extrapolation has been successfully used in the past and is, therefore, expected to be reasonably accurate.

6. <u>Hexavalent Chromium in River Water</u>. Figure 10 illustrates the concentration of hexavalent chromium in the river under various operating conditions assuming no loss and no reduction to trivalent chromium which is less toxic. An average river flow rate of 60,000 cfs was assumed. Since the deleterious effect on the fish is dependent upon persistence of these concentrations for a month or so, and since an average flow rate of about 60,000 cfs is required for the hydroelectric facilities to meet their commitments, these curves represent the most probable condition. If it is desirable to consider the effects of lower flow rates, it can be done on this graph by lowering the line representing the criterion by the fraction of 60,000 cfs that is being considered.

7. <u>River Temperature</u>. The actual temperature of the river represents a balance between the forces which tend to increase the temperature such as heat absorbed directly from the sun, return flow of irrigation water, and the operation of industrial facilities including the reactors, and those forces which tend to remove heat from the water such as evaporation and heat transfer to the soil, the effect of the additional heat added will invluence the balance but will not be felt as a simple addition to the temperature some distance downstream.

The deleterious effect on the salmon run is caused by a virulent strain of bacteria. The probability of an epidemic of disease among the salmon increases with temperature over 20°C and with the length of time that such temperatures are maintained. The river temperature at the reactors normally exceeds 20°C for a few days every year, and in 1959 epidemic disease was reported when temperatures between 19°C and 21.5°C presisted for nearly ten weeks.

The complexity of the problem makes it impossible to state a limit on the amount of heat that the reactors may add to the river without establishing similar limits on other contributors over which HAPO has no control.

8. <u>Rupture Products</u>. Control of the release of fission products due to fuel element ruptures would not be required in the usual case.(25) However, protection against the unusual case is indicated. Andersen, et.al., (26) discuss the consequences of failure of several elements in the same channel in some detail. The effects of increasing production levels on the probability of such an occurrence have not been evaluated. It seems unlikely that HAPO would wish to incur the cost of decontamination and unfavorable public opinion which would result even though the consequences are described as being troublesome and not (technically) a major hazard to individuals in the environs.

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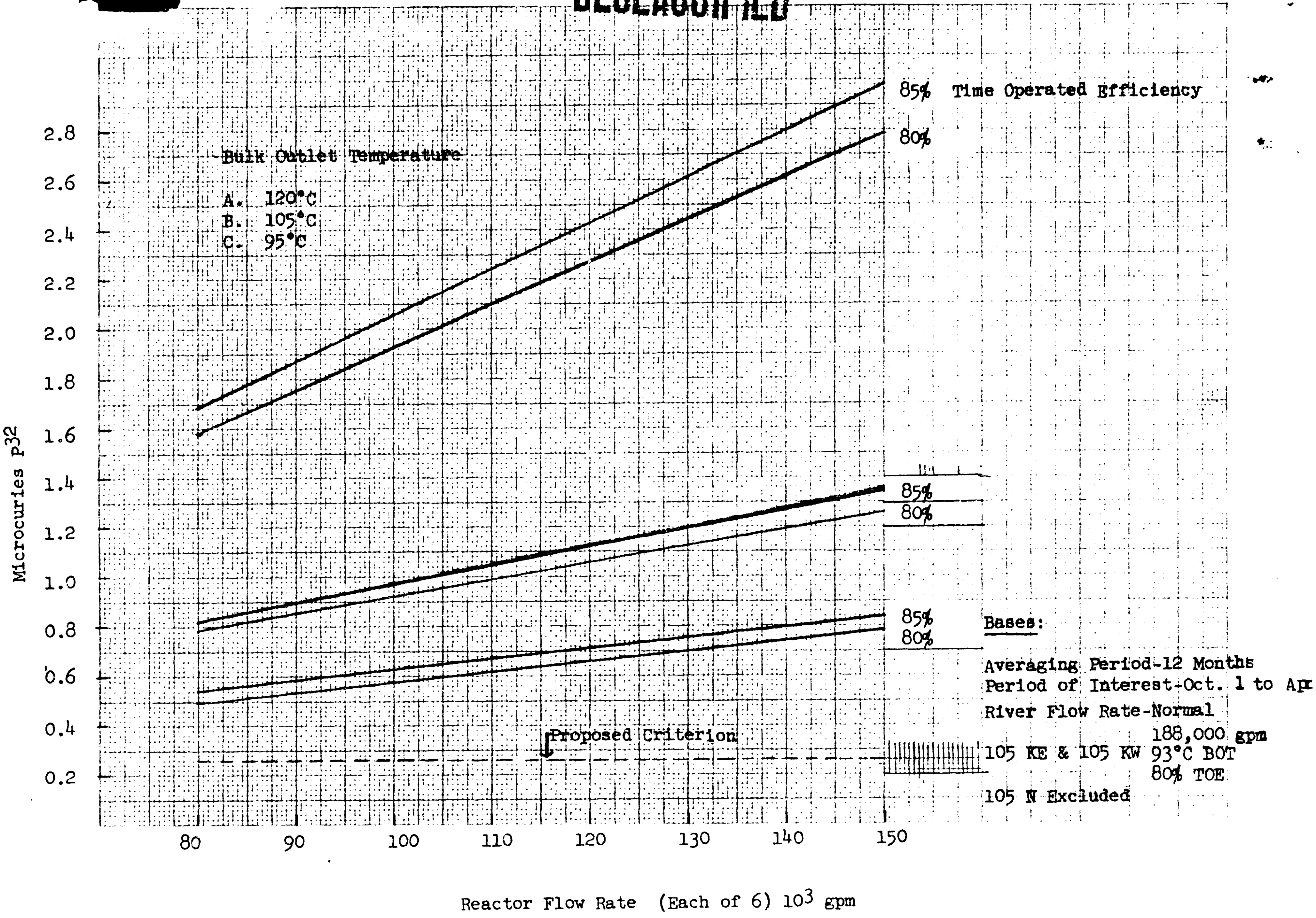

Figure 8. AVERAGE BODY BURDEN OF PHOSPHORUS-32

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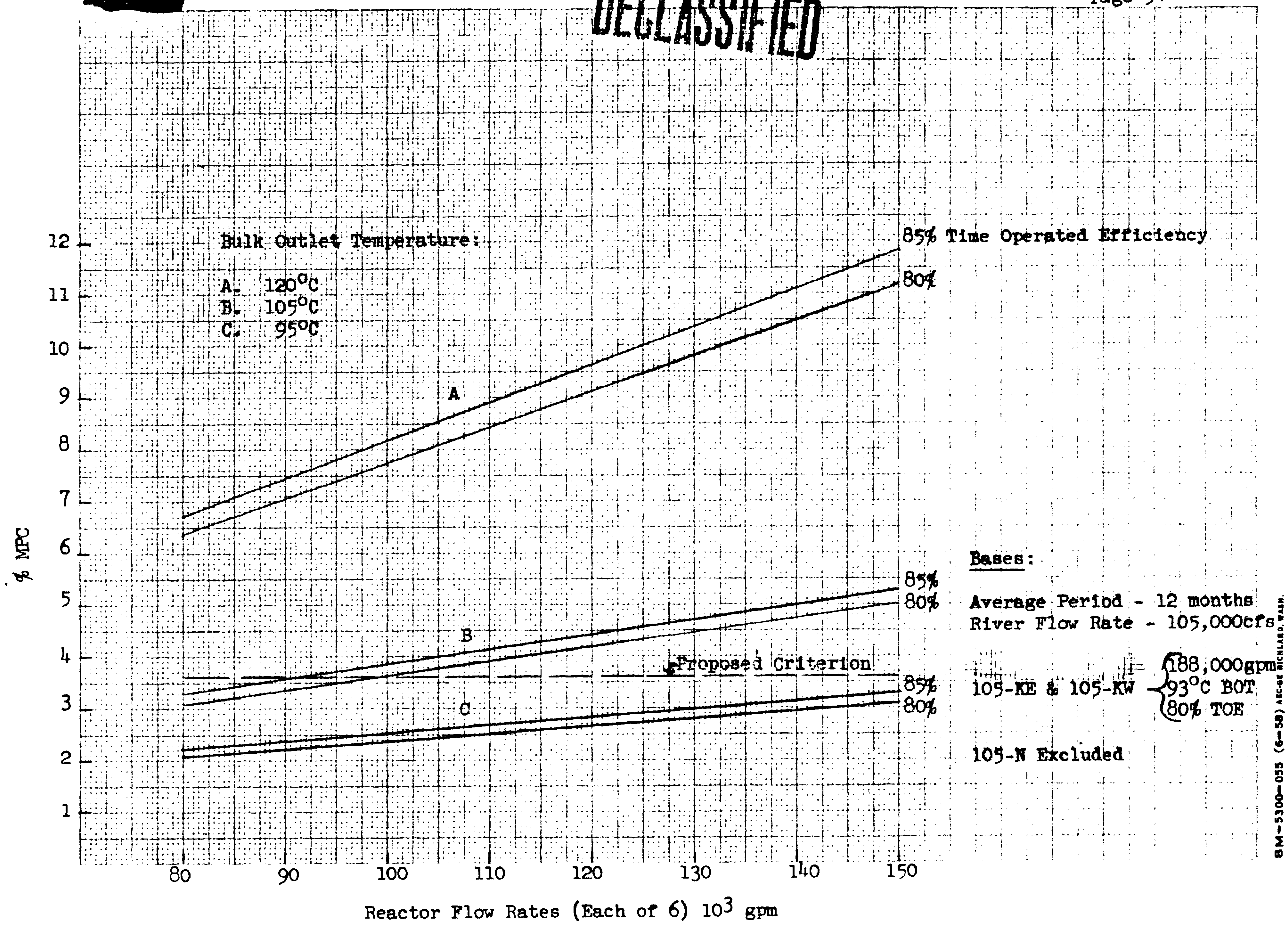

Figure 9. AVERAGE RADIOACTIVITY IN RIVER AT PASCO

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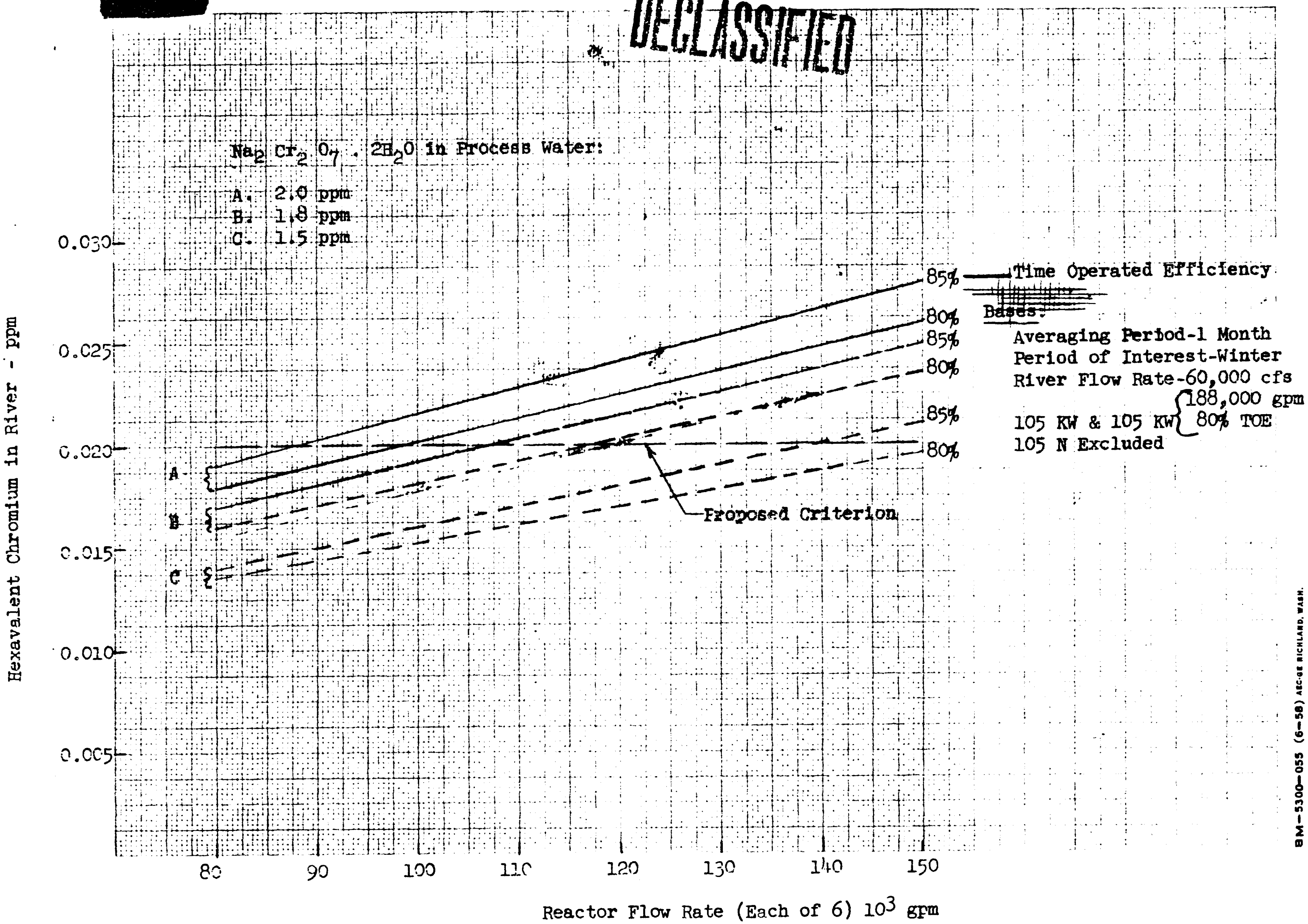

BM-5300-055 (6-58) AEC-GE RICHLAND, WASH.

Figure 10. HEXAVALENT CHROMIUM IN RIVER

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B. Personnel Exposures

Increases in reactor power levels will increase the radiation levels around the reactor and will contribute to an increase in Total Radiation Exposure received by operating and maintenance personnel. The results of a study performed by L. A. Carter indicates that the percentage exposure increase will be roughly 1.2 times the percentage power level increase. Previous studies have shown that the majority of monitor, operator, and maintenance exposure is received in the principal reactor locations as indicated below:

Location	Radiation Monitors	Operators	Maintenance
Rear Face	54%	42%	62%
Front Face	16	26	21
Wash Pad	20	22	8
Others	10	10	9

The effect of power level increases is summarized in the following paragraphs.

1. Discharge Area. If a power level increase is achieved by flow only, a proportional increase in discharge area background during outages is expected. If the power level increase is achieved by a bulk outlet temperature increase only, the radiation level would increase by a factor of approximately 1.25 times the power level increase.

2. Front Face During Operation. Increased power level will not significantly increase the rate of front shield deterioration since there would be no significant change in front shield temperature. The leakage flux through the shield is expected to increase in proportion with the power level.

3. Front Face During Outage. No significant change in front face background during outages is anticipated.

4. Washpad. No significant change in washpad background would be expected as a result of power level increases.

5. Other Locations. Less than 10% of personnel exposure is received in locations other than those listed above, and power level increases are not expected to significantly increase the exposure rates at those locations.

C. Reactor Shielding

The temperature of the masonite in the biological shielding must be maintained sufficiently low as to prevent deterioration and subsequent loss of usefulness. Existing shield temperature limitations were established on the basis that an increase of radioactive leakage by a factor of ten in the next ten years could be tolerated. Based upon this criterion, W. L. Smalley has estimated the magnitude of power level increases that could be tolerated without shield damage. It has been assumed in these studies that appropriate steps will be taken to control maximum shield temperatures including (1) Fringe poison will be charged as required up to the point where each reactor contains a full blanket of black mint loadings in the outside lattices of each side, top, and possibly the bottom; and, (2) Helium concentrations in the reactor gas will be increased as

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required and to the maximum extent possible considering reactivity requirements. The results of this study are summarized below, assuming the conditions as indicated:

1. Full fringe poison loadings in each of the outer lattice layers and 75 percent helium in the reactor gas:

   B, C, D, F - Approximately double present power level
   DR, H - Approximately 40 percent power level increase

2. Full fringe poison loadings in each of the outer lattice layers and 90 percent helium in the reactor gas:

   DR, H - Approximately 70 percent power level increase

For the DR and H Reactors the limiting temperatures will occur in the top shields. This condition is caused by neutron streaming through the top shields, and a built-in gas space between the top shield and the thermal shield which reduces heat conduction from the biological shield. The anticipated shield temperatures under the above conditions is shown in Figure 11.

If radiation dose rates external to the shielding increase more rapidly than anticipated, several methods may be available to restore shielding effectiveness, including the following:

1. Installation of external shield patches.(30)

2. Installation of auxiliary shield walls around the top of the reactor.

3. Filling of the voids formed within the shields by pumping grout or some other material into the deteriorated shield.

Also to reduce neutron streaming around the vertical safety rods, it may be possible to provide the VSR tips with neutron scattering materials which would be allowed to hang within the top reflector. In addition to increased radioactive leakage, consideration should be given to possible structural problems associated with the masonite deterioration.(31)

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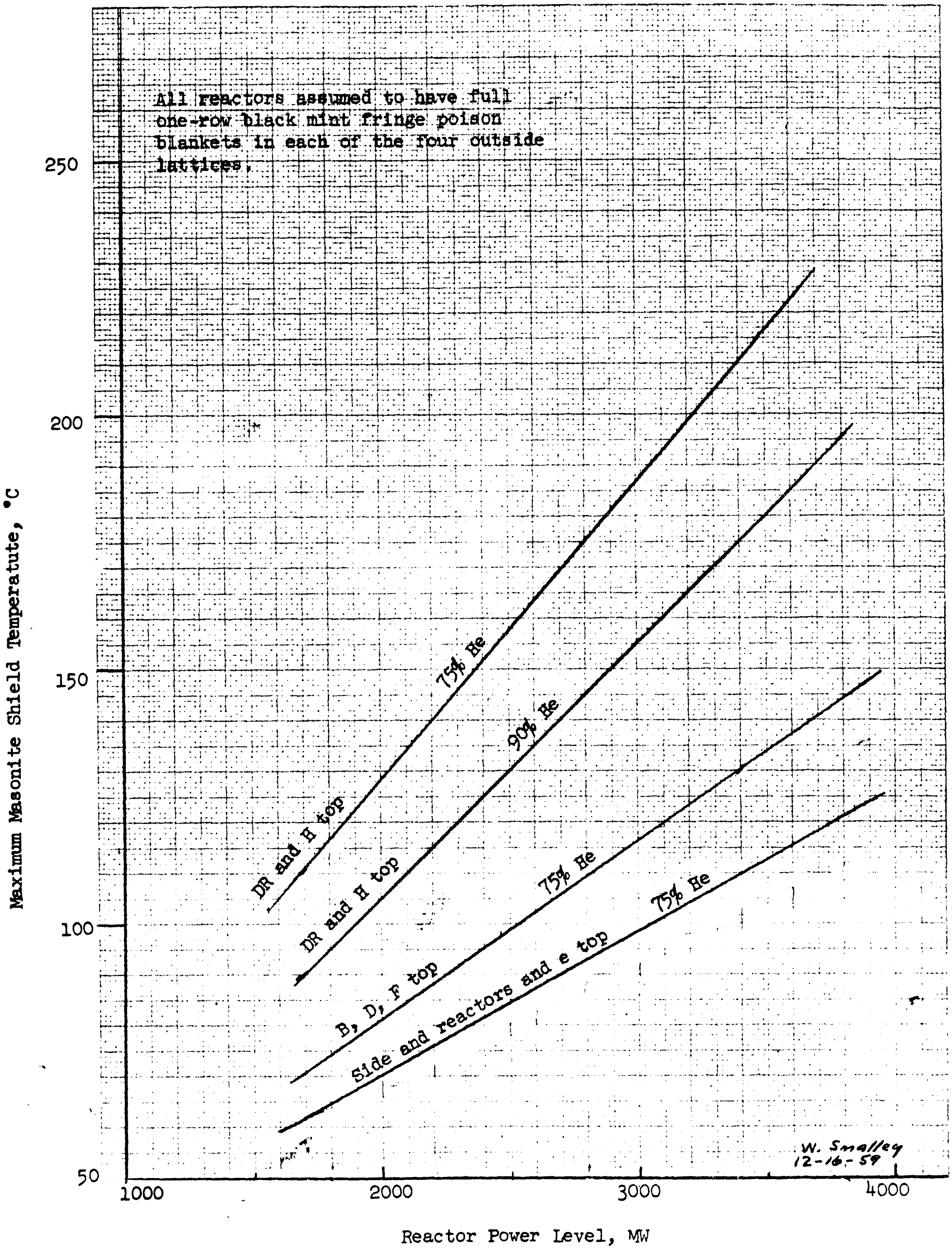

Figure 11. ESTIMATED MAXIMUM SHIELD TEMPERATURES

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