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GENERAL ELECTRIC

HANFORD ATOMIC PRODUCTS OPERATION - RICHLAND, WASHINGTON

TITLE

A REVIEW OF
HANFORD POWER LEVEL LIMITS

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A REVIEW OF HANFORD POWER LEVEL LIMITS

by

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I. INTRODUCTION

As for any nuclear reactor, the operation of the Hanford reactors is governed by many technical and administrative considerations. With the completion of current project action, later this year, certain of the technical limitations on reactor power level will be relieved, notably at the K Reactors. At that time, the technical and engineering limitations will ~~permit operation of some reactors at power levels which exceed~~ administrative power level limits, imposed by the Atomic Energy Commission, that are currently in force.

The following report discusses the various technical factors governing reactor power levels and compares these with the currently imposed administrative power level limits.

II. SUMMARY

The table below summarizes the power level status for the various Hanford reactors:

Table I

Hanford Reactor Power Level - Megawatts

<u>Reactor</u>	<u>Present Administrative Limit</u>	<u>Potential Winter 1962-63 Peak</u>
Old (B, D, F, DR, H)	2090	2050-2145
C	2310	2470
KE, KW	4400	4870

The potential peak power level at each reactor is limited by bulk outlet temperature restrictions (95 C), and by available coolant flow rates (except at the K Reactors where Project CGI-883 will provide a significant increase in flow). No significant changes in procedures, reactor physics characteristics or mechanical equipment other than improved coolant backup facilities (Projects CGI-844 and 905) and Project CGI-883 at 100-K are associated with attainment of the power levels noted in the table.

The rate at which reactor power levels can be increased when on bulk outlet temperature limits is limited by the rate at which the inlet water temperature decreases in the fall of the year. This built-in governor limits the maximum power level rise to about five per cent over a month's span.

Technical factors affecting the safety and continuity of operation of the reactors are discussed. The various control and coolant system safety criteria are spelled out and compliance with them discussed. The effect of fission product inventory is examined and shown to be small for those long-lived isotopes which contribute to long-term off-site hazards.

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III. DISCUSSION

A. Factors Limiting Reactor Power Level

As for any nuclear reactor, the operating power levels of the Hanford reactors are governed by a number of technical and administrative considerations. Paramount among the technical factors affecting the operating power level of the reactors are, of course, those considerations involving the nuclear safety of the reactors. These considerations, which deal with the potential for the release of a portion of the reactors' fission product inventory, can be stated in terms of maintaining adequate nuclear control of the heat generation process and of providing adequate removal capacity for the heat generated. Other important technical factors affecting the reactor operating power level involve those considerations necessary to assure efficient, economical and sustained operation of the reactors as production facilities. Finally, during the past several years, contacts with the Advisory Committee on Reactor Safeguards, its predecessor groups, and other AEC review authorities have resulted in the establishment of certain specific restrictions, referred to as administrative limits, which have been imposed upon the Hanford reactor power levels. The factors affecting Hanford reactor power level and the means of judging limiting conditions are summarized in the following tabulation:

Table II

Factors Affecting Hanford Reactor Power Level

1. Technical Factors

a. Reactor Safety	Nuclear Control	Speed of Control (Coolant Loss) Total Control
	Heat Removal	Sub-cooled Heat Transfer Burnout Emergency Transient Heat Transfer Burnout Backup System Capacity
b. Continuity of Operation	Effluent System Damage	Bulk Outlet Temperature
c. Economics and Reactor Life	Heat Removal Rate Fuel Performance Tube Replacement Moderator Damage	Coolant System Capacity Fuel Exposure and Power Corrosion Rate Graphite Contraction Graphite Oxidation

2. Administrative Factors

Administrative Limits	Fission Product Inventory	Reactor Power Level
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Within the prime requirement for the prevention of major release of fission products, the technical factors are closely interrelated and are largely controlled by economic considerations. Limits expressed for the technical factors must be qualified and considered provisional because each limit can be increased by appropriate modifications to the reactor plants. Reactor power levels limited by bulk outlet temperature, currently the most restrictive of the safety and continuity limits, are variable in nature, being dependent upon the temperature of the water supplied as the reactor coolant. As will be later shown, the power level potential of some reactors, as limited by bulk outlet temperature, will exceed their currently specified administrative limit during the coming winter (colder inlet water) season.

B. Safety Limits

1. Nuclear Control

The operation of the Hanford reactors must conform to two control criteria on nuclear safety. These criteria have resulted from extensive studies and relate reactor operation to the response and capacity of the safety control systems. Complete statements of the criteria have been included in the appendix to this report. The presently adopted interpretations of the general statements of the criteria have been discussed with the Advisory Committee on Reactor Safeguards and the AEC, and have been found to be acceptable bases for safety analysis.

a. Speed of Control

Specifically the speed of control criterion is interpreted as follows:

The energy generated in the excursion shall be insufficient to cause the surface temperature of the fuel elements at any point in the reactor to reach the melting point of uranium until the power generation at that point returns to its pre-accident equilibrium value.

Until recently, only the total power level was considered in the interpretation of the criterion; however, recent studies, made possible by the purchase of additional analog equipment, have allowed the power level to be defined on an approximately local basis. This has led to the adoption of the newer, more restrictive interpretation of the criterion.

The maximum power level admissible under the criterion varies with reactor type, fuel element geometry, reactor temperature, fuel exposure, and safety system response time. The following table compares the limiting power levels imposed by the speed-of-control criterion for the indicated operating conditions and the presently approved maximum power levels.

Table IIIPower Level Limits

<u>Reactor</u>	<u>Speed-of-Control Limit</u>
B, D, F	2760 MW
DR	2220 MW
C	3000 MW
H	2580 MW
KE, KW	>6000 MW

In arriving at the speed-of-control limits of Table II, the following conditions were assumed.

Table IVAssumed Operating Conditions

<u>File</u>	<u>Flux Squared Weighted Graphite Temp. - °C</u>	<u>Flux Squared Weighted Exposure (MWD/Ton)</u>	<u>Normal VSR* Response, Sec.</u>
B, D, F	300 to 500	0 to 500	2.3
DR	300 to 500	0 to 700**	2.3
C	300 to 500	0 to 500	2.1
H	300 to 500	0 to 700**	2.3
KE, KW	300 to 500	0 to 500	1.9

* The normal VSR response time (time between opening of the Panellit summation relay and 90% insertion of the VSR's) is considered representative of properly operating rods as installed in DR, C, H, KE and KW. The normal response time for B, D and F reactors is considered representative of the response times after the completion of a design change in progress.

** DR and H reactors are currently employing the "block" discharge scheme, which can result in higher fuel exposures than the "continuous" discharge.

Table III applies to the reactors loaded with the present fuel element geometries. Any deviations from these geometries can result in different power level limits. For instance, C reactor loaded with the CVI (overbore) elements would lose reactivity on loss of coolant. This would have the effect of raising the power level limit imposed by the speed-of-control criterion.

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b. Total Control

The presently adopted interpretation of the total-control criterion is as follows:

A prompt, critical state is never permitted. Criticality, short of prompt, is allowed in the reactor following loss of coolant only after a time when it is obvious that emergency procedures to save the reactor from complete meltdown are no longer possible. In the case of the present reactors, with aluminum process tubes, the time at which the reactor complex has reached the melting point of aluminum is considered the time after which a supercritical state cannot be allowed.

shouldn't this be 'before'?

Meeting the total-control criterion in itself has no limiting effect on reactor power. Complying with the criterion involves at most some loss of production efficiency. The present use of splines or poison columns at all eight reactors to supplement the limited and fixed capacity of the Ball 3X systems enables all reactors to meet their total-control requirements satisfactorily at all times. This ability would not be changed at higher power levels.

2. Heat Removal

a. Sub-Cooled Heat Transfer Burnout

If the fuel heat generation rate is increased under constant conditions of coolant flow and pressure, a point is reached where nucleate or surface boiling occurs at the surface of the fuel elements. With further increases in the heat generation rate, film boiling is eventually initiated and the fuel element surface becomes covered with a vapor blanket. Since the heat transfer coefficients through the steam blanket are very much less than through the water film, the heat generation rate may then exceed the rate of heat removal. The temperature of the fuel element surface then rises rapidly and may exceed the melting temperature of the fuel element cladding. This phenomenon is known as sub-cooled heat transfer burnout when the bulk temperature of the coolant is below its saturation value at the time of failure of the fuel cladding. A review of the potential for sub-cooled burnout at equilibrium power levels in the Hanford reactors indicates that the operating heat flux densities are a factor of two to three times lower than the conservatively estimated heat flux densities at which film boiling leading to burnout could occur.⁽¹⁾ Using as a limiting value 70 per cent of the film boiling initiating heat flux, the reactor power level limits are far in excess of current or projected capabilities. This is illustrated in the following table:

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Table V

Reactor Power Level and
Burnout Heat Flux Relationships

<u>Reactor</u>	<u>Current</u> <u>Administrative</u> <u>Limit - MW</u>	<u>Percent of</u> <u>Sub-cooled</u> <u>Burnout Limit</u>	<u>Reactor Power</u> <u>Level at Burnout</u> <u>Limit - MW*</u>
B, D, F, DR, H	2090	38	3900
C	2310	46	3500
KE, KW	4400	53	5830

* Burnout limit taken as 70 percent of film boiling heat flux.

Due to the chopped cosine power distribution which occurs along the normal Hanford reactor process tube, sub-cooled burnout would be initiated at a point along the fuel charge. Thus, slightly exceeding the heat flux required for film boiling would result in the failure of one or two fuel elements in the column - perhaps with jacket melting - but would not lead to general fuel column meltdown.

b. Individual Process Tube Emergency Transient Protection

During normal reactor operation, the cooling provided to a process tube is sufficient to remove large quantities of heat while maintaining the fuel element surface temperature only slightly higher than the temperature of the cooling water itself. With an accidental reduction in coolant flow rate or an increase in power level beyond certain limits, these excellent heat transfer conditions no longer prevail. If the relationship between heat generation rate and flow rate becomes too unbalanced, coolant boiling occurs in the process tube and rear fittings. The pressure loss in a process tube with steam-water flow is greater than for water alone; and the flow of water to the process tube is reduced, thereby causing more steam to form and causing additional self-aggravated reduction of coolant flow. The final result may be a very low flow rate to the tube involved which can cause excessive heating and possible fuel element cladding melting. This self-aggravating flow decrease is called "flow instability," and it is a requirement that this condition be detected by individual tube coolant pressure instrumentation that will automatically initiate a reactor shutdown in time to prevent jacket melting.

To prevent this damage, the flow to each process tube is monitored continuously by measuring the pressure in the throat of a venturi or orifice with a Panellit pressure gage and comparing this pressure against a set pressure. Disturbances in flow are indicated by either a high pressure or low pressure reading. This instrumentation is set

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to scram the reactor whenever inadequate flow conditions are detected, and, except for extreme cases of total flow interruption, in a sufficiently short time that during the shutdown transient, the reduced coolant flow is adequate to cool the fuel and to avoid fuel cladding melting.

The maximum tube outlet temperature permitted by the current philosophy of instability limit administration is 130 C. This particular maximum is somewhat arbitrary, and is based upon high pressure trip requirements. Electrically heated process tube laboratory data consistently show that the instability limits as presently applied are conservative. The same method of limit administration has been found to adequately protect the simulated fuel charge under transient conditions when the initial heat generation rates are approximately twice those prevailing in reactor fuel charges.

c. Coolant System Capacity

Adequate cooling of reactor fuel elements is the paramount consideration of nuclear reactor safety. As an outgrowth of reactor safety reviews, there evolved, in 1959, a statement of the fundamental criteria for Hanford reactor coolant supply systems. These criteria are used as guides for the determination of the adequacy and reliability of the coolant supply systems of the existing reactors. The criteria have been recorded in several documents, and are included for ease of reference as an appendix to this report. Basically, the criteria call for three independent, reliable sources of cooling; a primary system to provide cooling for all phases of reactor operation and shutdown; a secondary system capable of providing adequate shutdown cooling indefinitely, assuming instantaneous loss of power to the primary system; and a last-ditch system capable of providing shutdown flow indefinitely, assuming instantaneous loss of power to the primary system and concurrent failure of the secondary system to provide its rated flow. The capacity and conformance of the various reactor coolant systems to the criteria are discussed below.

1) Primary Systems

The primary coolant systems of all reactors have high integrity and are adequate, under the criteria, for all phases of reactor operation.

At the K Reactors, Project CGI-883, Increased Process Water Flow, 100-K Area, ⁽²⁾ is nearing completion and will expand the water pumping facilities in 100-KE and KW to provide an increased primary system flow rate. The work includes six additional 181-KE and KW pumps, all new low lift pumps, upgrading the low lift pump motors from 900 HP to 1500 HP, modification of all high lift pump cases to accommodate new increased capacity impellers, major electrical modification to the distribution system to accommodate the increased electrical load, and installation of new type reactor rear face crossheader expansion joints.

The flow and power level capacity during the coming 1962-1963 cold water season for each reactor is tabulated below:

Table VI

Hanford Reactor Flow and Power Level Potential
1962-63 Winter Peak

<u>Reactor</u>	<u>Flow - gpm</u>	<u>Power Level*- MW</u>
B	90,500	2145
D	88,000	2085
DR	87,900	2080
F	90,100	2135
H	86,400	2050
C	104,300	2470
KE, KW	205,000**	4870

* Inlet temperature 5 C, bulk outlet temperature 95 C.

** Post-CGI-883 flow rate.

2) Secondary Systems

The secondary cooling systems of all reactors are separately powered systems which accelerate on loss of power to the primary system. Energy stored in the flywheels of the primary pumps provide a coast-down transient supply that provides cooling until the secondary supply system is able to pick up the cooling load.

The secondary systems of the six older reactors meet the criterion for independence and have ample capacity. At C Reactor, the primary and secondary power supplies drive a common shaft and a common pump, but this mechanical coupling has been judged acceptable. The reliability of the older reactor secondary systems is acceptable for conditions not involving earthquake or storm damage to above-ground steam lines.

As constructed, the reliability of the K Reactor secondary system is high, even though compliance with the criterion of independence from the primary system could not be assured. Project CGI-844, 100-K Area Coolant Backup, ⁽³⁾ which provides independently powered steam driven pumps in the secondary system, is nearing completion. This project action will better assure compliance with the Reactor Cooling Criteria.

The flow and power level capacities of the various reactor secondary coolant systems are tabulated below:

Table VII

Adequacy of the Secondary Cooling System

<u>Reactor</u>	<u>No. of Steam Pumps</u>		<u>Critical Time When Pumps Take Over Load* (Seconds)</u>	<u>Calculated Reactor Flow at Critical Time 1000 gpm</u>		<u>Inlet Coolant Temp. t_i °C</u>	<u>Heat Output Decay Factor F_p</u>	<u>Max. Power Level for Compliance w/Criteria MW</u>
	<u>Scheduled to Pick Up Cooling Load</u>	<u>Assumed to Pick Up Cooling Load</u>		<u>Flow at Critical Time</u>	<u>Season of the Year</u>			
B, D, DR, F, H	7	5	100	21.0	Winter	5	.130	3600
				21.8	Summer	20		3100
C	8	6	190	29.0	Winter	5	.113	5725
				29.8	Summer	20		4850
K's	3**	2	60	40	Winter	5	.137	6500
					Summer	20		5400

* The point of maximum reactor bulk outlet temperature (= 90 C) during the shutdown transient.

** Post-Project CGI-844, one steam driven and two low lift pumps powered by emergency generators scheduled to pick up load.

3) Last Ditch Systems

As noted above, the last ditch systems are to provide cooling in the event of loss of primary pumping power and concurrent failure of the secondary system to provide adequate cooling. As with the transition to the secondary system, the flywheel decay of the primary pumps provide coolant flow until the last ditch system is able to take over the cooling load.

The last ditch system of the older reactors (high tanks and export system) are deficient both in capacity and reliability. Project CGI-905, Rev. 1, "Interim Modifications for Improved Coolant Backup - B, C, D, DR, F and H Reactors," (4) is under way to revise the system where significant improvement can be gained in capacity or reliability. The project will provide for automatic starting of steam driven export pumps upon loss of normal electric pumping power, automatically restrict flow to the 200 Area reservoirs when electrically driven export pumps are dropped from the export system, install hydraulic accumulators on the export system surge suppressors

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to better assure valve actuation at the proper time after the electrically driven pumps have stopped, relocate high tank sensing points to achieve earlier introduction of export water into the reactor, provide corrosion protection in the high tanks and lines, and provide temperature monitoring of high tank water and cooling of stored water during the summer months. Upon completion of this project, later this year, the modified older reactor last ditch systems will be sufficient in capacity for current levels of operation, but the system will remain deficient in reliability as measured by the Reactor Coolant System Criteria; because the system is not independent of the secondary coolant system of the reactors, its dual function as a last ditch system and a source of supply to the 200 Areas reduces its reliability as a backup system, the failure of any one of several components in any area can render the complete system inadequate, and because the system is difficult to test to assess its reliability.⁽⁵⁾

The last ditch system of the K Reactors has also been found deficient, when compared to the criteria, in that the system capacity is inadequate under certain conditions of operation.⁽⁶⁾ As noted above, Project CGI-844, "100-K Area Coolant Backup," is nearing completion and will modify the existing high pressure crosstie line and install a diesel driven pumping station which discharges into the modified crosstie. Upon the completion of this work, the deficiencies of the 100-K Area coolant backup systems will be corrected, better assuring their compliance with the Reactor Coolant System Criteria.

The capacity of the various reactor last ditch coolant systems, after the completion of project action now in progress, may be found in Table VIII.

C. Continuity of Operation

Effluent system boiling, because of the potential widespread damage to the reactor involved, represents the major limiting consideration other than the true safety limits discussed above. The effluent systems of the existing reactors are all relatively weak as piping systems. Should effluent boiling occur, which results in pressurization of the system, there is a strong potential for the failure of the system. Considerable reactor outage time would be required for the repair and restoration of such a failure.

Considerable effort has been devoted to the study of the hydraulic characteristics of the effluent piping systems, to the analysis of the various strengths and weaknesses of the system, and to the characteristics of transient effects arising from power level surge or partial flow loss and their possible damage consequences. As a result of these studies, the Bulk Outlet Temperature Limit has been established at a point below the effluent system saturation point by a margin judged to allow for credible operational effluent temperature surges. This margin is not

Table VIII

Capacity of Last Ditch Coolant Systems
Post-Projects CGI-844 and CGI-905

Reactor	Coolant Temp., °C		Temp.	High Tanks				Time Grove Valve Opens Mins.	Export		Time to Drain High Tank Mins.	Export		Heat Decay Factor	Ex-port Flow	Assur.	
	Outlet	Inlet		Assur. Flow gpm	Time of Take-Over Mins.	Heat Decay Fact.	Max. Power		High Tank Flow	Net High Tank Flow		Time of Export Take-Over	Heat Decay Factor			Ex-port Flow Fact.	Max. Power
B, D, F, DR, H	90	5	7	11,000 (10,800) Avg.	4.5	.106	2300	15	13	7.0 (6.8) Avg.	75	90	.036	6.3	1.10	5.7	3500
	90	20	23	11,200 (11,000) Avg.	4.5	.106	1850	15	13.2	7.2 (7.0) Avg.	69	84	.037	6.4	1.10	5.8	2800
C	90	5	7	16,500 (16,000) Avg.	5.5	.102	3500	13	11.2	10.5 (10) Avg.	48	61	.046	8.5	1.08	7.9	3800
	90	20	23	17,000 (16,500) Avg.	5.5	.102	2950	13	11.7	11.0 (10.3) Avg.	46	59	.048	8.7	1.08	8.1	3000

<u>Last-Ditch System</u>							
Coolant Temp., °C	Outlet	Inlet	Assur. Flow gpm	Time of Take-Over Mins.	Heat Decay Fact.	Max. Power	
KE, KW	90	5	32,000	1.1	.132	5400	
	90	20	32,000	1.1	.132	4550	

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great enough to prevent effluent system damage under all credible cases of temperature surge, but rather takes into qualitative account the relative probability of such surges so as not to unnecessarily restrict the production capacity of the reactors.

Currently, the Bulk Outlet Temperature Limit is specified as 95 C for all reactors. This limit is presently the most restrictive of the technical factors limiting reactor power level. The power level capability, during the coming cold inlet water season, as limited by bulk outlet temperature, may be found in Table VI, above.

D. Reactor Life and Economics

Currently, the graphite of the moderators in the reactors is contracting. The elevation of the top center of the reactor graphite is decreasing at a rate of approximately 0.35 inch per year at B, D, and F Reactors, 0.4 inch per year at DR, H and C Reactors, and 0.8 inch per year at the K Reactors.⁽⁷⁾ Higher reactor power levels may tend to increase the contraction rate somewhat, which would make remedial action appropriate at a slightly earlier date. The graphite is being vigilantly monitored, and methods for straightening channels have been developed. It is not anticipated that these effects will prevent attaining a 20-year life for the reactors.

Economic considerations to some extent determine reactor power level, in that power level and fuel exposure are somewhat interchangeable. The same is true to a much lesser extent for considerations involving the rate at which aluminum process tubes require replacement. The recent highly favorable fuel performance and improved ability to efficiently monitor and replace process tubing reduce the impact of these considerations as a determinant of maximum reactor power level.

E. Administrative Limits

The power levels at which the reactors are operated assume critical importance because of two compelling reasons. First, assuming a reasonable proficiency as regards operating continuity, the power level is a direct measure of the reactor's productive output; and second, being an expression of the rate of energy release within the reactor, the power level is frequently referred to when considering matters of nuclear safety. In the past few years, some rather substantial increases in power level have been achieved, and at the same time, the safety status has not been compromised in any way. These increases in power level, and consequently plant production, are chiefly the fruits of successful research and development effort, coupled with safety evaluations and occasional administratively imposed power level limits to assess and maintain the safety condition of the reactors.

The successful completion of the Water Plant Expansion Projects CG-558 and CG-600 established the base for power level increases at all reactors,

save the K Reactors, in 1957. The K Reactors had not yet exploited their full power potential. The first significant event following this was the successful development and adoption of the Internally and Externally Cooled Fuel Element (the I&E fuel element). Because the potential for power level increase was so substantial, and for other reasons, the AEC imposed power level and fuel element exposure limits quantitatively equal to previously established maximums for each of the three reactor classifications (old, C and K) until the safety implications, particularly with regard to the possibility for fission product release to the plant and adjacent areas, could be evaluated. These limits, imposed in August of 1958, limited the reactor power levels to 1440, 1740 and 3140 MW at the old, C and K Reactors, respectively, until late January, 1959. On the latter date, and contingent upon the beginning of establishing certain confinement features at the reactor buildings, these limits were increased to 1800, 2000 and 4000 MW. Within the ability to do so, the power level potential afforded by the I&E fuel element and increased reactor cooling flow rates was exploited; and by mid-1959, it appeared that the winter cold water period might permit exceeding the new power level limits at the C and older reactors. Being advised of this fact, and after due consideration, the AEC further and temporarily increased the appropriate limits to 1900 and 2100 MW in time to take advantage of the colder winter river water. The limit for the K Reactors was not increased at that time. These interim limits were confirmed in mid-1960. Finally, with the installation and tie-in of the reactor confinement and building exhaust filter equipment, the power level limits were further increased a uniform ten per cent on January 10, 1961, providing their present values of 2090, 2310 and 4400 MW for the older, C and K Reactors, respectively.

For ease of comparison, the various technical and administrative limits are retabulated below:

Table IX

Hanford Reactor Power Level Limits - MW

<u>Reactor</u>	<u>Speed of Control</u>	<u>Burnout</u>	<u>Secondary Backup</u>	<u>Last Ditch Backup</u>	<u>Bulk Outlet</u>	<u>Administrative Limit</u>
B	2760	3900	3600	2300	2145	2090
D	2760	3900	3600	2300	2085	2090
F	2760	3900	3600	2300	2135	2090
DR	2220	3900	3600	2300	2080	2090
H	2580	3900	3600	2300	2050	2090
C	3000	3500	5725	3500	2470	2310
KE, KW	>6000	5830	6500	5400	4870	4400

Assumptions: Cold water inlet, winter 1962-63 - 5 C.
Projects CGI-844, 883 and 905 complete.

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It may be seen from the table above that the bulk outlet temperature is the most restrictive technical power level limit, and that the technical capability of the K Reactors considerably exceeds the administrative limit, that the capability of C Reactor somewhat exceeds the administrative limit, and that the technical and administrative limits for the remaining reactors are approximately equal.

F. Operation at Higher Power Levels

Should consideration be given to further relaxation of the administrative limits, it is appropriate to consider the effects of reactor operation at higher power levels.

1. Effects of Higher Power Levels on the Potential for Release of Fission Products

a. Probability of Occurrence

The probability of occurrence of an event which would lead to a major fission product release (such as loss of coolant) is not increased as power levels are increased, provided the supporting equipment is not taxed beyond its capacity. Compliance with adequate safety criteria will prevent this eventuality.

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.

b. Minimum Time for Remedial Action

The time available for remedial action in the event of coolant interruption at full power is reduced as the power level at equilibrium is increased. However, since the time to melt process tubes at current levels is about 25 to 30 seconds while at highest contemplated levels may be reduced about 5 seconds, the decreased time is hardly of operational significance.

c. Fission Product Inventory

Increasing power levels at a constant average exposure of fuel yields a maximum increase in the very short-lived fission products which reach equilibrium values during the exposure. The minimum increase occurs for the very long half-lived fission products whose growth depends more strongly on exposure time than on power level.

For a typical reactor case, an increase of as much as 20 per cent in power would result in the following increases in gross fission product activity:

<u>Time After Shutdown</u>	<u>% Increase in F.P. Activity</u>
1 hour	18
1 day	15
10 days	10
100 days	6

2. Methods of Achieving Higher Power Levels

When administrative power level limits are relaxed, power level increases will be controlled fundamentally by bulk water outlet temperature limits. Bulk outlet temperature limits are highly seasonal as indicated in Figure 1. The reactors would be expected to operate at powers determined by the available flow rate and the difference in temperature between the inlet water and the allowable bulk outlet temperature. As the inlet temperature falls from summer to winter, the power level would gradually increase. Thus, increases in power level would be gradual. They would be adequately monitored, and made within established safety criteria prudently administered.

3. Changes in Operating Procedures

Current operating procedures are designed to assure safe and efficient operation of the reactors within mandatory operating limits established by competent technical personnel. The operating procedures are not dependent upon power-level or operating limits, but vary as new equipment is installed.

No basic changes in operating procedures are contemplated as a result of potential power level increases. However, installation of equipment under approved projects will necessitate some procedural changes to make the best use of this equipment. This equipment, however, will be installed and used independent of the requested relief from current power level limits.

Included in the appendix to this report are statements pertaining to the general philosophy of Hanford reactor operation.

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Figure 1

Seasonal Variation of Power Level
Based on Bulk Outlet Water Temperature Limits

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APPENDIXI. PHILOSOPHY OF OPERATIONGeneral

It is the objective of the General Electric Company at HAP0 to operate the reactors in a safe, prudent and productive fashion with safety being paramount and with due consideration being given to reactor life and operating continuity. Operation of reactors involves some unavoidable degree of risk of release of fission products. This risk is very small but is not zero. It does not appear to be possible at present to assign an absolute value to this risk; however, it is possible to evaluate qualitatively the changes in risk associated with changes in equipment, operating limits and procedures. Thus, if a given operational status has been established as having a "tolerable risk," changes can be limited to those which, based on sound engineering judgment and informed managerial decision, do not increase the risk of operation. Further, it is possible to establish criteria for equipment and operation which will aid in maintaining risk within tolerable levels.

II. GENERAL SAFETY GUIDE

No changes will be made which decrease the assurance of safe operation of the production reactors.

In practice, this means 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.

Changes in operating conditions are limited to "known safe" regions. The normal procedures for establishing new operating limits involve analytical studies, experimental verification in out-of-reactor facilities, small-scale in-reactor testing under carefully evaluated, controlled and monitored conditions, and a final technical evaluation. This final technical study is the basis for mandatory standards which, after written approval by responsible managers, establish new limits within which the reactors must operate.

III. REACTOR COOLANT SUPPLY CRITERIABases

Nuclear reactor safety requires that adequate cooling be maintained at all times. The damage, in terms of loss of life and impairment of health, loss of crops and long term evacuation of lands, which could result if the fission product inventory of a Hanford reactor were released has been estimated to be greater than one billion dollars. The fuel element itself is the most effective containment vessel devised, and fission products will not be released in dangerous quantities as long as adequate cooling is provided.

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The extreme potential liability which can accrue in the event of inadequate cooling justifies unusual measures to maintain the probability of cooling system failure at a very low value. At the same time, it must be recognized that the probability of failure cannot be reduced to zero; hence, regardless of the precautions taken, there will be a finite probability that a maximum potential liability accident will occur.

The ultimate reliability of a system or series of systems has not yet been expressed quantitatively in a satisfactory manner, nor has an acceptable level of cooling system reliability been adequately defined. On the other hand, the mandate, "the system shall not fail" as a basis or criterion, is equivalent to requiring that the system not be operated. Therefore, the extent of the unusual measure which will be required in a reactor plant to achieve an acceptable level of cooling system reliability is a matter of judgment.

Prudence requires that the failure probability be maintained at a very low level and that errors in judgment in this area be on the side of requiring "too much" rather than "too little;" normal engineering standards routinely applied to "low risk" engineering are not acceptable as bases for judgment in reactor cooling system design and operation.

The criteria presented below broadly and arbitrarily define cooling system characteristics and provisions which prudence would require of the cooling systems in any nuclear reactor containing a substantial fission product inventory. Failure induced by operator error, equipment failure, natural phenomenon (earthquake, lightning) is considered credible; acts of war and sabotage are not specifically considered here as appropriate bases for establishing cooling system requirements. In practice, considered evaluation of specialized situations may support responsible authorization of specific exceptions to the general requirements developed below.

Criteria

The normal reactor cooling system requirements shall be:

1. Three independent, reliable sources of cooling as follows:

a. Primary Cooling System

The primary cooling system will provide adequate reactor cooling during all phases of operation and shutdown. The design and operation shall be such that the secondary or last ditch backup systems are not called upon to provide the sole source of cooling during any phase of normal operation; and would not be expected to be required to function as the sole supply system during the life of the plant.

b. Secondary Cooling System

The secondary cooling system shall be independent of the primary system (common piping is acceptable) and capable of providing adequate

shutdown cooling indefinitely, assuming instantaneous loss of power to the primary system. Serious consideration should be given the operation of the secondary system at partial flow capability at all times and preferably at a flow capability sufficient to satisfy the full requirements in event of primary system failure without relying upon actuation or acceleration of equipment. That flow provided by the secondary system during normal operation may contribute to the primary cooling system requirements.

c. Last Ditch System

The last ditch cooling system must be independent of both the primary and secondary cooling system, including piping to the reactor manifold. It must be capable of providing shutdown flow indefinitely, assuming instantaneous loss of power to the primary system and concurrent failure of the secondary system to provide its rated flow.

2. An independent system is defined as one with power sources and pumping systems such that failure or cause of failure in one cannot induce failure of the alternate systems.
3. Adequate cooling is defined, as a minimum, as that required to maintain reactor components, including fuel elements, at temperatures below the melting point. In practice, the objective of requiring levels of cooling capable of protecting against all reactor or fuel element damage will be prudent to adopt.
4. Failure of any single component in a system shall not be capable of rendering that system inadequate. In practice, certain piping and valves may be qualified as exceptions.
5. All engineering, construction, operation and maintenance shall meet or exceed requirements specified by applicable codes and standards.

Interpretation

The present interpretation of the meaning of "adequate heat removal" is that no gross coolant boiling shall be permitted in the heat generating zone of the process tube during equilibrium operation or during emergency shutdown transients.

IV. NUCLEAR CONTROL CRITERIA

Speed of Control Criterion

Basis

The early evolution of a criterion for the response time of a scram control system stemmed from the desire to limit the consequences of a nuclear excursion resulting from any credible accident. Since the total loss of

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coolant accident at equilibrium power represents the maximum credible accident (in terms of disastrous consequences) and, in addition, is the one accident in which some potential exists for a nuclear excursion due to the increase in reactivity upon loss of coolant, it is for this accident that the control system response must be adequate.

Reactor meltdown is inevitable if coolant is permanently lost. The purpose of the criterion is to insure reactor operation which would not aggravate the meltdown with respect to the release of fission products; i.e., uranium melting shall not occur while steam being forcibly ejected from the process tubing is able to sweep fission products outside of the reactor block.

The chief phenomena occurring during the few seconds following a loss of coolant incident at equilibrium are: 1) boilout of coolant and attendant reactivity effect; 2) trip and insertion of safety control rods and associated reactivity effect; and 3) heating of uranium and attendant reactivity effect. The magnitudes of these phenomena plus the relative rates at which each becomes effective determine whether or not a power excursion develops which results in fuel surfaces reaching melting temperatures before the excursion is terminated. The Speed of Control Criterion sets restrictions on the possible combinations of fuel exposure, power level, fuel type, safety system response times, etc.

Criterion

The operation of the Hanford reactors shall be such that in the event of a power excursion resulting from any credible combination of events, including complete and permanent loss of coolant, the response of the reactor safety control system shall be sufficient to insure that no energy state will develop during the excursion which would significantly increase the consequences of the accident.

Interpretation

The present interpretation is as follows:

The energy generated in the excursion shall be insufficient to allow the surface temperature of the hottest fuel element in the reactor to reach the melting point of uranium (1133°C). The excursion is considered terminated when the reactor power level returns to its pre-accident equilibrium value. Following this period, if coolant is not restored immediately, fuel melting will occur anyway, from heat generated by fission product decay.

Total Control Criterion

Basis

Existing Hanford reactors have under certain circumstances the property of being more reactive in the dry state than in the wet, as a result of

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significant overmoderation. Thus, in the disaster case of complete loss of coolant, there can occur an increase in over-all reactivity. Of equal or greater significance is the fact that fuel meltdown is inevitable, unless cooling water is restored. This is primarily a result of heat generation from fission product decay in the irradiated uranium fuel. The problem is further complicated by variations in fuel exposure and operating and shutdown flux distributions.

Early safety philosophy required that the dry reactor be maintained subcritical at all times. However, there is little incentive for requiring subcriticality after an appreciable number of process tubes have melted. Ensuing exposure in a reactor reaching the critical point after this time is an insignificant fraction of normal operational exposures, and the total fission product inventory will be only slightly increased. In keeping with this conclusion, current safety philosophy at Hanford requires that the following Total Control Criterion be satisfied.

Criterion

The operation of the Hanford reactors shall be such that in the event of any credible combination of events, the total control strength will be sufficient to insure that no reactivity state will develop which would significantly increase the consequences of the accident.

Interpretation

With respect to the maximum credible accident, the present interpretation of the criterion is that no reactivity state will be permitted to develop in the dry reactor (after the scram and any excursion which may have accompanied the accident) which would decrease the probability of restoring coolant to the reactor; i.e., a delayed neutron critical state is not permitted until after a significant number of process tubes have been rendered useless as coolant channels. (A prompt critical state is never permitted.)

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