HW--36621

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GENERAL ELECTRIC COMPANY

May 6, 1955

U. S. Atomic Energy Commission
Hanford Operations Office
Richland, Washington

Attention: D. G. Sturges, Chief
Operations Division

Gentlemen:

RECOMMENDATIONS BY ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Reference: 1) HAN-58364, "Hanford Production Reactors," C. R. McCullough, dated February 25, 1955.

2) Letter, D. G. Sturges to A. B. Greninger, subject as above, dated April 13, 1955.

Our general comments regarding each of the recommendations made by the Advisory Committee on Reactor Safeguards following the December 6 and 7 meeting (reference 1) are transmitted in the nature of an interim reply as per your request (reference 2). It is noted that the recommendations primarily define areas of reactor safety into which the committee recommends that future studies be channeled and that these areas are consistent with programmed extensions of the work presented at the December 6 and 7 meeting; minutes of this most recent meeting have been documented as HW-36226, with twenty copies forwarded to your office.

The general comments contained herein include a brief summary of our current status on each of the items suggested by the Committee with estimates of the

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GENERAL ELECTRIC COMPANY

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24. HM Parker
25. RH Beaton
26. WK Woods - PF Gast
27. WK MacCready
28. RO Mehann
29. OC Schroeder
30. JH Warren
31. HP Shaw - AB Carson
32. WJ Ozeroff
33. PH Reinker
34. JF Music - GC Fullmer
35. RL Dickeman
36. 300 File
37. Yellow File

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expected timing for submission of a more comprehensive discussion. The very general nature of the suggestions is such as to require the accomplishment of a substantial development effort in some cases before a complete reply can be given, and in these instances we will submit periodic progress reports as significant phases of the investigations are completed.

Comments on Committee Recommendations

1. Sequence of Events Following Loss of Water

The Committee recommends that "The present studies of the sequence of the events following the loss of water should be continued including cases where it is assumed the safety and control rods do not function, and where it is also assumed the ball safety system does not function. These studies should include considerations of nuclear run-aways, chemical reactions, and the effects of fission product heating."

Material presented to the Committee at the December 6 and 7 meeting indicated that a severe power excursion would not be experienced in the event pressure was lost from the coolant system during operation at elevated (500°C) graphite temperatures; this was found to be true even though pessimistic assumptions were made regarding the rate of water ejection. Significant power excursions would probably be experienced if pressure were instantaneously lost from a pile operating at 1000 KW per tube with reduced graphite temperatures because a reduced beneficial reactivity loss resulting from the positive graphite temperature coefficient being reduced to zero is realized. The major points required to more completely evaluate the seriousness of this situation are itemized:

a. The power excursions must be integrated and the integrated energy developed in the excursion taken into account in the rate of water loss prediction. Small excursions will be relatively unaffected by this refinement but the magnitude of a severe excursion will be increased.

b. The negative metal temperature coefficient of reactivity has been ignored in a deliberate effort toward conservatism. The inclusion of this retarding effect will tend to moderate a severe excursion.

c. The graphite temperature coefficient of reactivity has been assumed to be zero in the dry pile. Recent determinations of the graphite temperature coefficient of reactivity in the dry lattice were made during the KW pile start-up over the temperature range of 20°C to 180°C. These recent experiments qualitatively confirm earlier measurements at H pile made over a smaller temperature interval and show that the dry pile possesses a temperature coefficient of reactivity which is quite negative over the temperature

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interval studied when loaded with natural uranium fuel. The dry pile coefficient will probably be less negative as the plutonium concentration in the fuel increases with exposure and, also, may be less negative at elevated temperatures with virgin uranium although it appears constant over the range studied at KW pile. The excursions were calculated assuming a constant dry pile graphite coefficient of zero which appears to be conservative. Theoretical evaluation of the moderator coefficient is currently underway as is an experimental determination of the wet pile coefficient as a function of fuel exposure and moderator temperature at KE pile; this latter measurement may yield some further insight into the dry pile situation as well. Also, one of the first experiments planned in the Physical Constants Test Reactor this summer is an attempt to measure lattice temperature coefficients as a function of temperature and plutonium concentration in the fuel. It is hoped that the combined theoretical and experimental approach will permit the degree of conservatism inherent in the method of treating the moderator temperature coefficient of reactivity upon water loss to be evaluated.

d. Additional experimental work will be performed on the Heat Transfer single tube mockup which electrically simulates fission heating, but preserves all other salient features of a high power process tube, to further solidify our concepts regarding boilout mechanisms and the times required. The effect of uranium penetration into the aluminum jacket at elevated temperatures upon the component melting times in event of water loss is also being determined.

In addition to refining the understanding of the basic effects taking place upon water loss,we are determining the conditions under which the ball 3X system would limit an excursion. The case of a complete nuclear run-away of a Hanford reactor including chemical effects, has been considered by Dr. Mark Mills and we would tend to give somewhat lower priority to a reassessment of this problem.

2. Jamming of the Ball 3X System

The Committee recommends that "further consideration be given to the possibility of jamming the balls in the safety system with or without the rods present because of the small clearances involved."

Numerous full scale safety rod and ball 3X drop tests have been conducted and at no time have we observed any evidence of 1) the balls bridging so as to impede their fall, 2) the balls retarding the insertion time of the safety rod when both are dropped simultaneously. As a minimum, we then conclude that the probability of retarded control insertion on other than a small percentage of the total individual elements in the system is small. Considerable plant experience has also accumulated in "scramming" the installed ball 3X systems without any evidence that a portion of the balls failed to enter the pile.

In the event system designs are modified, the possibility of jamming will certainly be investigated.

The details of our test results and plant experience are being accumulated in document form and will be submitted to the Atomic Energy Commission about the middle of June.

3. Margin Between Control Strength and Potential Reactivity Available

The Committee noted an apparent "small margin between the effectiveness of the control and safety systems and the possible increases in reactivity" and recommends that "consideration be given to means of increasing this margin of safety."

The requirements of a control system fall in two categories: 1) the requirement for fast acting control which will quickly compensate the excess reactivity potentially available, and 2) the requirement for total control to maintain the reactor sub-critical under all reasonable conditions. The fast acting control must compensate the maximum reactivity gain upon complete loss of water from the reactor as well as any additional excess reactivity which might be in the system at the time of water loss; this accumulated requirement should never exceed 1000 inhours in a Hanford pile. The strength of the safety rod systems at all piles except KE and KW is approximately 1500 inhours or greater. At KE and KW piles the maximum reactivity to be gained upon loss of water is about 600 inhours and the strength of the fast acting controls will exceed 1000 inhours. We conclude, therefore, that a substantial margin of safety exists between the requirement for fast acting control and the maximum excess reactivity potentially available to the system.

The requirement for total control to maintain the reactor sub-critical is established as that required to compensate loss of water with complete xenon decay. In this application the ball 3X systems are known to be effective since speed of response is not critical. The strength of the ball 3X system at all piles will approximate or exceed 1800 inhours whereas the maximum reactivity which can potentially be realized ranges from 1350 to 1600 inhours depending upon the pile, its power level, and the type of loading; the very large potential reactivity gains associated with melting large quantities of enriched slugs and redistributing U-235 throughout the lattice are not considered here in control requirements. The margin of safety shown is increased further in practice in that neutron absorbers, viz, cadmium or lithium, are temporarily loaded in selected channels during extended outages to partially compensate for xenon decay. Also, neutron absorbing spline, i.e., cadmium or boron encased in flexible aluminum strips designed to be inserted between the slug and process tube wall, are also available and used. A mechanical system to insert and withdraw boron containing splines is currently under development to increase the control capacity of the horizontal control systems for operational facility in transient control; when developed, these systems will, of course, also be available as alternative methods of partially compensating xenon decay during extended reactor outages.

In summary, we conclude that appreciable margins of safety are available in the Hanford control systems except in situations involving the large scale melting of aluminum U-235 alloy slugs. We are actively developing the technology and economics associated with utilizing slightly enriched uranium in applications involving enrichment to reduce or, perhaps, eliminate the large reactivity gains potentially accompanying melting of the enriched fuel. We will submit a review of the status of this development during the middle of June.

4. Release of Fission Products Upon Fuel Melting

The Committee recommends that "information on the rate of release of fission products from melted fuel elements as a function of temperature and as related especially to possible Hanford conditions should be compiled and evaluated in relation to the hazards of these reactors."

We are remaining current as regards the development of information bearing on the rate of release of gross as well as selective fission products at elevated fuel temperatures. Consideration is also being given to the mechanism of release with emphasis placed on defining the possibility of neutron absorbers being liberated while the slug is in solid form; we are cognizant of the work of Dr. I. B. Johns in this regard. Experiments are planned in which gas evolution rates will be determined as a function of both temperature and state of uranium fuel elements which have been previously irradiated. It is expected that these studies will enable us to better define the rate at which fission products might escape from the reactor in the event cooling was lost and could not be re-established.

The experimental phase of this study is just beginning and it is not expected that results warranting significant conclusions will be obtained before January 1, 1956. The information now available in this field will be compiled and evaluated in light of Hanford conditions concurrently with the experimental study.

5. Reactor Fuses

The Committee recommends that "the reactor fuses program as outlined should be continued and accelerated if possible. A recommendation should be submitted by Hanford on the feasibility of loading one reactor of their selection with reactor fuses. This feasibility study should include the research, pre-testing, design, and procurement required before such a loading could be made. The cost, time scale, and production losses of such a program should also be included."

It is expected that the program for developing an in-pile safety fuse will be carried forward to include the in-pile testing at Hanford of three fuse elements; the timing on this program is established by progress at North American Aviation in providing the experimental assemblies. Also we believe that the irradiations scheduled by North American Aviation at Arco in support of the fuse development program should be completed and the data analyzed as scheduled, prior to entering the Hanford phase of the in-pile testing. The

current schedule for the experiments to be done at Hanford, as developed with North American Aviation personnel, calls for the initial irradiation to begin July 15, 1955, the second irradiation about August 15, 1955 and the third and final irradiation about December 15, 1955. The final irradiation will be in the nature of a life test of the element and may continue over a six month period.

The feasibility of loading one of the Hanford reactors with safety fuses as a full pile demonstration experiment can not be fully evaluated until the performance of the reactor fuses in-pile has been demonstrated. However, a preliminary study will be prepared which will include the research, pre-testing, design, and procurement which now appears to be required before such a loading could be made. The cost and production losses resulting from this full pile test program will be developed assuming adequate performance of the fuse element. This preliminary feasibility study can be completed October 1, 1955 and will be followed by a more comprehensive study following the in-pile fuse testing program at Hanford which may extend to June 1, 1956. The order of magnitude of cost and production loss and a review of salient operational problems yet to be solved before fuse utilization in even a limited program could be considered feasible were outlined at the December 6 and 7 meeting with the Committee.

6. Flow Re-establishment Problems

The Committee recommends that "studies should be made of the possibility and consequences of restarting normal water flow after interruption."

The majority of the conditions visualized as possibly resulting in flow interruption will involve a major failure in the coolant piping system; under these conditions normal flow cannot be re-established, and therefore, supplementary cooling systems which are either completely independent of existing cooling systems or which utilize only a portion of the existing systems have been under study. In the limited cases in which normal flow could be re-established following interruption we believe, on the basis of information now available, that the proper course of action is to re-establish flow as quickly as possible. Failure to re-establish or provide cooling from a supplementary source will inevitable lead to melting and vaporization of the aluminum, fuel, and control elements in the reactor, whereas prompt action in re-establishing cooling when possible will maintain the integrity of the control systems and in most cases, the fuel elements if the coolant can be properly distributed. The potential hazards associated with introducing water into a reactor at elevated temperature include high pressure steam formation and potential energy release from chemical reactions.

The study of the rate of steam formation and release and probable implications regarding the integrity of the reactor and/or shield structures is currently in progress; preliminary results show that if cooling were re-established before extreme temperature conditions exist the steam formed would be vented without toppling the shields. The sequence of events in the extreme cases are now being evaluated. It would appear that in the majority of the cases in which cooling was interrupted and could be re-established the necessary action to re-establish could be taken before extreme in-pile temperatures were reached.

We are also currently evaluating the published data regarding the rate of chemical reaction and the energy release which might be expected as high temperature water, aluminum and uranium reacted. However, the details of such reactions appear strongly dependent upon the specific conditions present and very little of the existing data applies directly. We plan to perform a limited amount of experimentation at Hanford on this subject and it is expected that a more complete discussion of both the steam release and the chemical reaction problems can be submitted by January 1, 1956.

7. History of the Hanford Operation

The Committee recommends that "a study and documentation of the history of the operation of the Hanford production reactors with particular regard to equipment failures, operational errors, and analysis of scrams is suggested if this is practical. This suggestion has the intent of trying to discover from such a review indications of possible future difficulties or the lack of same."

A history of Hanford operating experience containing the type of information requested by the Committee is currently being assembled for documentation. We expect to submit this compilation to the Atomic Energy Commission before June 15; copy coverage for the Committee will be provided.

8. Operation of K Reactor Safety Rods

The Committee requests that "a review of the mechanism and operation of the K reactor safety rods should be presented to the Committee. This review should include time of travel, position indication, whether releasable during rise, consequences of mechanism failures, etc."

The information requested by the Committee describing the detailed performance of the K reactor safety rods is currently being asembled into a single compilation from several sources. This documented review will be forwarded to the Atomic Energy Commission before June 15.

9. Reactor Safety Committee At Hanford

The Committee recommends that "the advantages of a permanently organized reactor safety committee at Hanford should be considered."

The General Electric Company is organized to place complete responsibility for all phases of its operation at the appropriate levels within its line organization. The responsibility for reactor safety at the Hanford Atomic Products Operation is clearly assigned within the line organization in a manner consistent with overall company policy. We are also familiar with the committee or staff type of organization and have had considerable experience with it in the past. However, our experience has convinced us that for our case assignment of responsibility and the discharge of assigned responsibility is more effectively carried out through explicit assignments within the structure of the line organization.

GENERAL ELECTRIC COMPANY

D. G. Sturges

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Reactor safety is considered to be a prime responsibility at Hanford; the programs in this field are actively supported, continuing programs which are given serious consideration in the assignment of personnel, funds, and facilities.

We believe it beneficial that the technical personnel contributing to reactor safety studies also have responsibility for contributing to other areas of the Operation; in this manner, the breadth of experience is increased and the perspective and interest of responsible individuals contributing to the programs are better maintained. We are continually alert for more effective ways to discharge our responsibilities, but conclude that for our situation the disadvantages of the staff or permanent committee type of organizational structure to discharge the reactor safety responsibility outweigh the advantages.

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