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DATE: June 7, 1955

SUBJECT: Design Review Committee Report on the
Homogeneous Reactor Test (HRT).

TO: J. A. Swartout

FROM: C. J. Borkowski

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To: J. A. Swartout
From: C. J. Borkowski

Date: June 7, 1955
Subject: Design Review Committee Report
on the Homogeneous Reactor
Test (HRT).

The members of the HRT Design Review Committee have examined, by no means completely, various aspects of the HRT including, engineering design, equipment layout, piping drawings, control and instrumentation, reactor kinetics and some phases of soup chemistry.

In general the committee has been impressed with the quality and thoroughness of the engineering design for the HRT. Conservative safety factors well in excess of those used in industrial practise insure safety in operation.

Since the pressure vessel will be welded after the reactor core is installed it will be impossible to make any repairs or replacement of the reactor core in the event of even minor damage to the core. As there is no spare pressure vessel the HRT program could be delayed six to ten months should the reactor core be damaged.

The committee realizes that a great deal of thought has been given to the dump operation and that experiments are still underway to permit a safe dump with a minimum dependence on an instrument dump. Since the results of these later experiments have not been reviewed by the committee, the present comments on the dump system are tentative.

The HRT is a good reactor in all respects and should outperform the HRE.

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Engineering Design Review of HRT

Sylvan Cromer

I have reviewed the design of the Homogeneous Reactor Test from the standpoint of engineering, and have also discussed the design basis used with Mr. Briggs and Mr. Beall. The planning and layout have been carefully done. The safety factors used are well in excess of those generally used in industrial practice, and appear to be adequate to cope with any feasible condition. Operation of this reactor for an extended period should provide a great deal of useful information on the type of problems that may be encountered with a large reactor of this type, and will possibly enable one to make an order-of-magnitude estimate on the cost of maintenance.

The reactor pit, designed for containment of radioactive material in case of a catastrophe, has been carefully engineered and should be adequate to take care of any situation that may arise. In the future, should changes in the design be made that would materially change the inventory of hot, high pressure liquid in either the core or blanket systems, the pit design will obviously have to be re-evaluated.

It is unfortunate that, under conditions approximating those expected in the reactor, more basic information on corrosion is not available at this time. While the corrosion rates observed will probably not cause trouble in the test reactor, extreme care must be exercised in extrapolating optimistic results obtained from this test to full-scale power producers. The importance of continuing an aggressive research and development program in parallel with the reactor test cannot be overemphasized.

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Review of HRT Engineering Layout and Chemistry

F. L. Culler

The design of the HRT was reviewed from the standpoint of operability and safety. My comments deal primarily with piping, engineering layout and minor phases of the chemistry.

First, a general comment about the quality of the work and the care with which the design has been prosecuted. In reviewing a rather small part of the total information that has been prepared for the design and operation of the HRT, I was impressed with the quality, care and thoroughness that has been exercised in the design. From my limited survey, I found nothing that would limit the operation of the machine or would provide a serious safety hazard. I am also impressed with the supporting development work for this design.

Some of the things on which comment may be desirable follow:

- (1) Precipitation of uranium from core or blanket by hydrogen peroxide formed by radiation decomposition of water may be a problem in spots in the reactor circuit where liquid is trapped or moves very slowly and/or where the temperature of the solution drops below a given temperature for each power level in the solution. Two possibilities where such conditions may exist were noted:
 - (a) The small annulus between the inner and outer shells of the gas separator. R. B. Briggs reports that there will be a flow of reactor fluid through this to prevent precipitation.
 - (b) The baffled expansion assembly for core and blanket (and

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particularly the blanket) at the top of the reactor provides a possible trap for solution if purge fluids are not supplied under sufficient pressure to maintain flow. During our review meeting, Briggs stated that purge liquid will be pumped through the baffle chambers for both core and blanket systems. He also pointed out that the gamma flux at this point will be materially lower than in the reactor core itself, thus reducing the possibility of precipitation.

- (2) At the time of our review meeting, the decision as to where ice seals were to be applied had not been made. Therefore, this comment is offered as a suggestion. Lines 126 and 226, which are vents from the high pressure to the low pressure systems, appear to require frozen seals.
- (3) I considered the possibility of shock from sudden closure of the radiation detector actuated valves, and I concluded that no serious shock could result from rapid closure.
- (4) In my opinion, the most important safety device in the reactor circuit is the liquid level indicator on the pressurizer. The safety of the machine is dependent upon the constant and proper function of this unit since the expansion space necessary for sudden increases in activity is to be maintained by the liquid level indicator on the pressurizer. There is nothing wrong with the concept of the device, but such gadgets are always subject to failure. I would like to suggest duplication of the liquid level controller.

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- (5) In laboratory bomb and out-of-pile circulating loop studies, it has been indicated that the rare earth fission products and other components of the fuel solution will plate out on areas of the reactor circuit that are at temperatures higher than the fluid. This fact, in the longrun, may be an advantage for chemical processing, but in the HRT metal walls, valves, pumps, et cetera, should not have surfaces that are hotter than the circulating fluid, nor should the solution be allowed to rise more than a few degrees above the core average temperature except under controlled conditions. The zirconium core vessel will provide an excellent surface for plating provided the temperature is sufficient to maintain the wall above ambient.
- (6) Assuming that fission products and corrosion products precipitate as suspended particles, which may be the case, it will be necessary to eliminate pockets from the core and blanket systems. If the precipitated fission products are small in size and possess low settling rates, concentration probably will not occur in areas of low flow rate. But once such concentration occurs it is highly probable that a hot spot will develop, followed by plating of fission products at the hot spot.
- (7) Although the use of blow-out patches followed by spring-loaded relief valves is rather common practice in plants which handle dangerous or valuable chemicals, there is danger that an undetected pinhole leak in the rupture disc may allow the pressure on the discharge side of the disc to build up to system pressure over a period of time.

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- (7) continued
Under such a circumstance the blowout patch will not relieve under its design conditions, but will hold to higher pressures with the maximum pressure being somewhere below the sum of the settings of valve and disc. A bleed line between the rupture disc and the relief valve will prevent pressure build-up.
- (8) Returning again to the precipitation of corrosion products and fission products on metal surfaces that are at temperatures above that of the soup - very recent work in Chemical Technology Division has shown that rare earths and corrosion product deposits on stainless steel surfaces through which as little as 0.3 to 0.5 kw per square foot of heat is transferred. The calculated temperature (not measured) of the stainless steel surface under such a circumstance is only 2 - 5° C above the fluid temperature. These experiments were done in a static system; they are now being repeated using zirconium. Deposits formed containing mixed fission products and corrosion products are relatively difficult to redissolve in soup after they have been formed.
- (9) To support these static experiments, data from Loop IA (CTD) have shown that essentially all of the neodymium added as neodymium sulphate to the circulating loop was plated out in the loop heat exchanger. No precipitated particles of neodymium sulphate appeared in samples withdrawn from the loop.
- These data indicate that this deposition should be considered in the reactor. The most obvious hot spot is the pressurizer where the temperature will be maintained at 35° C to 50° C higher than in the
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- (9) continued
reactor, and the boiling of soup, slightly diluted by fresh water
(10 gph), must occur to maintain overpressure of 750 lbs. or more.
Solids deposition in the pressurizer appear to be certain with wall
temperatures indicated in ORNL CF 54-9-25, HRT Pressurizer Heater.
Solids deposition will cause a drop in overall heat transfer coefficient,
possible restriction and plugging of pressurizer heated tubes, and
most important, may block inlet and outlet tubes to the liquid level
controller. Solids deposition on the float mechanism for the liquid
level controller may impair its operation, a function vital to the
safety of the homogeneous reactor since the control of the reactor
its negative temperature coefficient depend primarily upon the
maintenance of 60 liter vapor volume close to the reactor cell.
- (10) The problem of solids in the liquid level controller for a blanket
of ThO_2 slurry would appear to be more serious than for the core,
although with a natural uranium blanket solids deposition will be
a problem similar to that in the core.
- (11) The second likely surface for fission and corrosion product deposition
is the core vessel wall, although deposition on Zr has not yet been
confirmed, because of gamma heating of the core vessel. Fission
products and corrosion products of high cross section would thus be
retained in the core, making the neutron losses higher than the
predicted 5.6%. We have not yet determined whether or not uranium is
occluded or precipitated in the films which deposit on hot surfaces,
but such a possibility must be recognized.

- (12) Another spot where fission products and corrosion products might precipitate is in the circulating pumps since there will be a slight rise in temperature in passing through the pumps.
- (13) The HRT will not operate under conditions that make fission product deposition a serious consideration. Approximately one week of operation of the HRT at 5 MW is required to build up to FP saturation (20 mg/1000 g D₂O) assuming that the volume of core solution is 500 liters (based on rare earth production of ~1.36 per day). Thus, the rate of production of rare earths is not sufficient to be of serious consequence. However, with the TBR, 320 MW core, the rate of production would be 81 grams per day, which if deposited in a critical area such as in the pressurizer might be restrictive. These numbers are for rare earths only; corrosion products and other fission products will precipitate as well. If 1.36 grams of rare earths per day were deposited on the core vessel, the reactivity decrease from rare earths alone would be approximately $1.2 \times 10^{-3} \% K$ per MWD of operation, based on $\phi_{wall} = 1/2 \phi_{core}$.
- The reactor circuit possibly should not be designed to operate using a fuel solution saturated with fission products and corrosion products. This, in turn, defines the necessity of operating the chemical plant under process conditions that will keep fission product concentrations below saturation for the hottest spot in the reactor circuit.

Review of HRT Equipment, Equipment Layouts, and Piping Drawings

H. E. Goeller

In my review of the HRT design, I have studied primarily equipment, equipment layouts and piping drawings and memos. Mr. G. W. T. Kearsley assisted me in making short study of the steam system. I asked him to study this phase of the design because of his considerable experience in the design and operation of steam systems at Y-12, the Idaho Chemical Processing Plant and elsewhere.

In reviewing equipment designs, I have made a list of the several items which I thought questionable. I am sure that in most cases the items and ideas questioned have been studied in considerable detail and that there are probably very good reasons for detailing the design as it is already done.

1. Weld neck flanges -- ring joints (C-18200). It appears that the same material is used for both flanges and rings thus making a difficult seal. It would reduce bolt stresses and probably improve seating if gold plating was used on the rings; however, the cost is considerable, and its use would definitely be limited to critical locations.

2. Valves -- In the 1/2" size hand operated bellows seal valves it appears that the shaft is free to rotate thereby placing the bellows under rotational strain. If possible, the shaft should be keyed as in the larger valves to prevent this rotation. The use of nylon or other organic material in the packing above the bellows on the air operated process valves may result in sticking of the valve stems if radiation damage swells the packing. It is assumed that the valve seats

on all valves are ground after the weld nipples are installed since such welding will warp the valve bodies.

3. HRT Liquid Level Controller. (D-18145) The float system used in the controller has the inherent disadvantage in that if the float leaks the control fails.

4. Lapp Diaphragm Head Assembly (D-19973). I noticed the use of backup rings in welding on the outer head rings, which in this case should not be objectionable because the space involved would never contain process fluid except in the case of a leak. However, I'd like to point out that backup rings should not be used where contact with process solution normally occurs because of the difficulty of equipment decontamination. I don't believe I noticed their use in any other design.

5. Gas Separator (E-18181). It appears that the only longitudinal structural restraint to the Ti vanes and liner is the 1/4" gas outlet. Although longitudinal stresses are normally balanced, damage to either set of vanes might result in a large thrust force. It might be advisable to install a welded ring restraint, as this could also be designed to prevent the sudden (but small) expansion in the line at the Ti section exit.

6. Reflux Condenser (E-18219). The spacing of the cooling surfaces appears inadequate if there is a vapor surge, since there is a nearly free path to the off gas connection. However, the probability of such a surge is highly unlikely.

7. Entrainment Separator (E-18237). The York stainless steel mesh filters may corrode out after long use but should be satisfactory otherwise. Although there is no definite proof, it appears very likely that similar filters used in

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the ICPP intercycle evaporators did corrode severely since the Ni and Cr content of evaporator product in early operations there were quite high.

8. Equipment Layouts

a. Access to the outer dump tanks looks rather difficult. It might be advisable to put in roof slabs rather than a solid roof to permit their removal if this ever becomes necessary.

b. In a final plant it is recommended that no equipment be located over other equipment, to permit optimum underwater maintenance. However, since the HRT is experimental in nature, this is not a serious problem.

c. If the water level during maintenance is only to the bottom of the slabs there may be insufficient shielding of the highest equipment such as the cold traps. Although they are normally not exposed to process solutions a misoperation might permit them to become too hot to maintain easily. If water level can be raised a few feet above the bottoms of the slabs, no problem exists.

d. It is assumed that in a HR power unit the equipment would be compacted into a smaller area. For the HRT the extra space is undoubtedly valid for experimental reasons.

9. The following notes on Steam System for HRT were made by Mr. G. W. T. Kearsley.

Dwg. H-37725 (Heat Exchanger (Boiler))

Probably is O.K. but effect of trying to liberate 20,000 to 30,000 lbs steam per hour from surface of boiler water only about 24 sq ft area should be reviewed carefully.

Dwg. DKPL-7601-A-1 Flow Sheet

1. The purpose of the pressure relief valves (set at 1350 lb) located

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on the continuous blow-down line from the boilers is not understood. (Note that steam pressure relief valves are to be set at 1500 lb.).

2. Drain lines 307 and 308 could, under certain conditions, feed water to the drums. Why not install traps and return the condensate from the drums to the feed water storage tank?

3. It looks like the samplers, as shown, will sample a mixture of boiler water and drum condensate, or drum condensate, but never boiler water alone.

4. If boilers are operated at 1200 psia, then the feed water should be heated to a higher temperature than 212° F.

5. On warm up boilers should be drained before steam is admitted to them. They could be wrecked by turning live steam into them when partly filled with cold water.

6. Direction of flow arrows seem reversed on cooling water lines 390 and 391 to auxiliary heat exchanger.

Dwg. CKPL7602-B-1

1. There is something wrong at the turbine condenser if it requires water velocity above 10 ft/sec and effluent is up to 160° F. Possibly too many passes are used.

2. The steam killer would be more effective if a trap were added in the condensate line and full steam temperature maintained in the coils.

Dwg. CKPL7602-B-3

Cooling Tower Pump shown as delivering from 716 to 378 GPM under different loadings. The water flow over the cooling tower should be about constant to keep all surfaces wet and avoid crudding up with lime.

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10. In reading over ORNL-1834 "HRT-Summary Report for the Advisory Committee on Reactor Safeguards" I found two minor errors which should be added to the errata.

- a. Page 63 -- Underline Pressurizers
- b. Page 83, line 5 -- Change fuel solution to blanket solution.

HRT Design Review - Controls

E. P. Epler and S. H. Hanauer

Since the absence of conventional nuclear safety devices is predicated on the nuclear stability of the HRT (which we do not discuss here), the dump operation is evidently to be used to reduce quickly the pressure in the reactor, and hence to minimize the consequences of a leak. Failure to dump, when a dump is needed, could have serious consequences; dumping without cause, and especially without control, can have equally serious results.

The conventional approach to such a problem is to seek reliability in numbers; that is, one provides several safety mechanisms, operation of any one of which will achieve the desired result. Ideally, "separate mechanisms" implies complete independence: two or more channels of information, control, and actuation with no essential interconnections. With an increase in equipment to achieve these ends, however, comes an inescapable burden of additional effort to keep the system working, and additional process down time due to safety equipment malfunction. The two requirements of safety and operational reliability cannot simultaneously be met without large systems of safety equipment. Viewed in this light, the HRT dump system leans, in our opinion, heavily toward the achievement of operating continuity, and the safety aspect has perhaps been unduly minimized.

Our first question concerns the advisability of providing only one dump line. The experience of the HRE with respect to dump lines was not good, and we are concerned that a single plugged line, or a single sticking dump valve

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could result in at least a ruptured core tank, and at worst a whole lot more. That there are considerations of leakage, and of simultaneous plugging from the same cause (non-independence) we recognize, but we suggest re-appraisal of this matter.

We next consider the single dump line and valve, and the presently designed dump control system. It is unfortunate that the controlled dump is so dependent on the operation of one device; viz, the derivative controller. If, for some reason, this instrument is inoperative, the dump cannot be initiated, and the operator has the dubious choice of using HV 152 to attempt manual control, or initiating the uncontrolled dump. Inasmuch as two instrument channels are needed in any case for a controlled dump, we suggest as one solution of this problem that instruments and controls be arranged with sufficient symmetry that either of two instruments may initiate the dump, perhaps at the option of the operator. In the case where one instrument was apparently misbehaving, primary control could be transferred, and the ΔP contacts could be allowed to keep the differential pressure from getting out of hand. We recognize that in this case the differential pressure control would be quite coarse, but the alternative might be an uncontrolled dump.

Testing the dump circuits and controls is a necessary operation, and provision for this is proper. We submit, however, that an operation such as this requires careful programming, an alarm to warn the operator, and an override to restore the channel to service instantly and automatically if a bona fide dump signal should occur. As the system is presently designed, two hand-operated valves (H-152 and -252) transfer control from the dump instruments to manual

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controllers. The manual controls are to be adjusted until they effectively call for closed dump valves, after which manipulation of a test switch will permit checking the dump control valves. The hand valves, the manual controllers, the test switch, and the controller pneumatic indicator are in the control room; the indicator gauges for testing the pneumatic controls are on the second level. The consequences of inadvertently leaving the hand valves in the "manual" position (or, worse, leaving just one in this position) are obvious. It is impossible for one man to check out the system, and the console operator cannot check the status of his dump system. One possible method of meeting these objections would be replacing the hand valves with solenoid valves, and modifying the test switch to four-position, spring-return, as follows:

Position 1: Normal

2: Transfer control to manual controllers, and sound alarm

3: Test Position 1

4: Test Position 2

There should also be an override to restore control to the automatic dump if a dump signal occurs. We further suggest grouping all dump test equipment in the same location.

Since loss of instrument air prevents a controlled dump, the pressure loading on the dump valves must be obtained from the same instrument air supply. In order to insure that this will really be the case, considerable care will have to be taken with the physical layout and protection of the interconnecting pneumatic lines.

Provision for orderly revision of control interlocks is an excellent idea.

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The jumper board installed for this purpose should have the following features:

1. Limited Access. The operator should have permission, from the senior engineer or higher authority, to make changes in the control interlocks. This supervision should be enforced with something like a lock-and-key arrangement.
2. Alarms. Interlocks jumpered out should be clearly and unequivocally indicated to the console operator.

The detailed design of the jumper board was not reviewed.

HRT Instrumentation

C. A. Mossman

The writer has reviewed the HRT Instrumentation and believes that the following are worth further consideration by the review committee.

The relatively low allowable differential pressure between the fuel and blanket system seems to me to be the limiting factor of the reactor design. According to the most recent data the drop from 2000 psi to 1000 psi during a reactor dump occurs in the first few seconds as the liquid drops out of the pressurizer. The approach is to control the rate of drop in pressure of the fuel system and to match this with the pressure in the blanket system. To do this, a fairly complicated pneumatic control system is required which uses dump valves that must also throttle. In my opinion the fuel dump valve is a questionable item as a pure on-off valve. Furthermore as a throttling valve I would expect it to have an even shorter life. The control system required to give a safe dump can probably be made to work satisfactorily but to depend on it working correctly every time is, in my opinion, living dangerously. I believe that the approach to the problem should be to design the system so that a dump can be made free flow; that is, the system is so designed that a safe dump requires no control equipment. The fact that the bursting pressure of the core is two to three times that of its collapsing pressure is worth serious consideration in this problem. Furthermore collapsing pressure is highly unpredictable while bursting pressure is predictable by non-destructive tests. From this it would seem that an effort should be made to utilize this fact, particularly during a dump,

by designing the system so that the core runs at a pressure higher than the blanket.

The most recent test data shows that the pressure vs time curves for the two systems will be somewhat like figure No. 1. If we break the curve into two parts I believe we can see a logical solution. The first part, the drop from 2000 psi to 1000 psi is the fuel and blanket dropping out of the pressurizer, and the volume of liquid which each valve must pass is approximately equal. The second part of the curve is the drop from 1000 psi to zero which is the removal of the blanket and fuel from the rest of the high pressure system. Since the fuel and blanket were at a temperature producing 1000 psi a curve shape as shown on figure No. 1 results. The volume of liquid, in this part of the dump, which each valve must pass is the ratio in volumes between the two systems, namely 4 to 1, blanket to fuel. From the above it would seem that we should not attempt to use one set of valves for both sections of the dump but rather should have a set of valves for each. Since the existing letdown valves for the fuel and blanket system are presently of about equal capacity it is suggested that they be considered for this application.

My proposal is as follows:

- (1) Use the letdown valves to drop the pressure from 2000 psi to 1000 psi. This would take approximately 1 to 2 minutes and would slow down to a safe rate this critical part of the dump. The trim of these valves may have to be changed to equal percentage type to get a reasonable dump time.
- (2) Use the present dump valves only for the drop from 1000 psi to 0.

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- (3) Resize the letdown and dump valves so that with a free dump the core runs at a slightly higher pressure than the blanket. This would enable us to use the additional pressure safety factor and if throttling is required it would be done by the less critical blanket dump valve.
- (4) Provide an instrument control system to (a) throttle the blanket valves only if the core pressure exceeded the blanket by say 200 psi, (b) stop the dump if the collapsing pressure exceeds say 50 psi.

The trim used in the letdown and dump valves in both the fuel and blanket systems is, to me, highly questionable. It should be stated at the start that this is a case of one man's opinion vs another and there is little experimental data to back up either case. However consider the following.

The design and selection of valve trim for applications such as these is a difficult problem and one for which there is little information to draw upon, and the only safe way is operation under actual conditions. Much time has been spent evaluating corrosion rates of various metals for the components of the reactors. Most of this work has been at the relatively low velocity of 100 ft/sec or less. From this data we know that the corrosion rate of Type 317 stainless steel goes up rapidly beyond 50 ft. per sec. Corrosion rates at the 500 ft/sec velocities which will exist in the valve are only speculation. Added to this is the problem of erosion which goes hand in hand with corrosion. In this case we have some information to draw upon. Erosion data is available for the valves used in continuous blown-down service for high pressure boilers. These valves represent a problem, similar to our letdown valves. The most successful valves

for this application have a smooth contoured body and Stellite (or equivalent) trim. The corrosion rate for Stellite is known to be high in fuel above 150° C but this would apply only to the fuel dump valve. In this case it is anybody's guess whether the combined effect of corrosion and erosion is worse with Type 347 stainless steel or with Stellite. It should be mentioned that all of the valves on liquid throttling service in the HRE had Stellite to Stellite trim, and to my knowledge there were no failures due to corrosion or erosion. Listed below are the four valves under discussion with the trim materials to be used.

<u>Valve</u>	<u>Temp.</u>	<u>Present Design</u>	<u>Suggested Design</u>
Fuel letdown	150°C	Plug-Stellite Seat-347 S.S.	Plug-Stellite Seat-Stellite
Fuel dump	285°C	Plug-347 S.S. Seat-347 S.S.	? ?
Blanket letdown	150°C	Plug-Stellite Seat-347 S.S.	Plug-Stellite Seat-Stellite
Blanket dump	285°C	Plug-Stellite Seat-347 S.S.	Plug-Stellite Seat-Stellite

The significant point of the above discussion is that there is essentially no data available to back up the selection of valve trim. To me, valves are every bit as important as pumps, pressurizers, heat exchangers, etc. and at least an equal amount of time and money should have been spent on their development. I would like to urge the review committee to recommend that a comprehensive valve test program be set up so that in the future we do not find ourselves in the position of guessing on a best design.

A large number of pneumatic instrument transmitters are being used in the HRT. Pneumatic transmitters are excellent for normal industrial applications. There

is however the threat that foreign material will plug the small openings in the nozzle or restricter of the transmitters. In this reactor, electric type transmitters have to be waterproffed because of the underwater maintenance approach. The waterproofing problem and cost were the two reasons given for the use of pneumatic transmitters. It is my opinion that electric transmitters can be made more reliable than the pneumatic type and should be the goal of the HRT instrumentation program. Cost for items such as these should be secondary to reliability. It is noted also that in the HRT most of the instrument transmitters are of a new and untested design. The above difficulty appears to be the result of a lack of a long range reactor instrumentation program.

Figure I

Comments on the Physics of the HRT with Special Reference to Reactor Kinetics

G. T. Trammell and T. A. Welton

The authors undertook, as part of the HRT design review effort, an examination of those aspects of the design which are most concerned with questions of reactor kinetics. It should be stated at the outset that this review makes no pretense at completeness. The aim was simply to make sure that the experience gained in the field of homogeneous reactor kinetics in the course of the work on the HRE was properly applied to the HRT.

Some general remarks are first in order. Our strong impression is that the HRT, from the kinetic aspect, is a very good reactor in all basic respects. It is cleaner, simpler, and better than the HRE, which ran without serious kinetic difficulty. The HRT is not much of an extrapolation, kinetically speaking, at least, from the HRE, and the authors feel that there is every reason to trust the extension of those kinetic concepts to the HRT which proved valuable for the HRE work. We further feel that this extension has been very competently performed. A number of spot checks of important numbers reveals only one important discrepancy, to be noted later.

Two extreme blanket compositions are assumed. The first will be designated the "slow" case, and has pure D_2O blanket. The second will be called the "fast" case and has 1350 g/kg ThO_2 . These are characterized by their prompt generation times, all other parameters pertinent to kinetics being essentially the same for both cases. We extracted plausible values from recent quarterlies:

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Slow: $\tau = 5.6 \times 10^{-4} \text{ sec}$

Fast: $\tau = 1.1 \times 10^{-4} \text{ sec}$

A reasonable temperature coefficient of reactivity seems to be:

$$\left(\frac{\partial k}{\partial T}\right)_p = -1.8 \times 10^{-3} \text{ } ^\circ\text{C}^{-1}.$$

The heat capacity of the core is expressed by a number, S

$$S = .76 \text{ } ^\circ\text{C}/\text{megawatt-sec.}$$

In terms of these numbers, the stability of the reactor against small, rapid, spontaneous power excursions is then ascertained by calculation of a quantity which plays the role of a squared frequency

$$\omega_N^2 = -\frac{S P_0}{\tau} \left(\frac{\partial k}{\partial T}\right)_p$$

and a quantity which plays the role of a damping coefficient

$$\lambda = \frac{\beta}{\tau} \quad (\beta = .005 = \text{fraction of delayed neutrons}).$$

Taking $P_0 = 5 \text{ megawatts}$, reasonable values seem to be:

$$\left. \begin{array}{l} \omega_N^2 = 12 \text{ sec}^{-2} \\ \lambda = 10 \text{ sec}^{-1} \end{array} \right\} \text{Slow}$$

$$\left. \begin{array}{l} \omega_N^2 = 63 \text{ sec}^{-2} \\ \lambda = 50 \text{ sec}^{-1} \end{array} \right\} \text{Fast}$$

In both cases, the system is highly overdamped, as was the case for the HRE, since $\lambda^2 \gg \omega_N^2$. This high damping makes extraneous any serious worry about instabilities which might be connected with transit time effects in the external loop. These conclusions are in agreement with much more careful calculations performed by members of the HRP physics group.

The next question which requires investigation is that of the mechanical stability of the system composed of core plus pressurizer. Stated generally, the HRE experience was that at reasonable power densities and at prompt generation times characteristic of aqueous, thermal reactors, such stability could be achieved by placing the pressurizer as close as possible to the core and introducing damping by sending the main core circuit flow through at least part of the pipe joining core and pressurizer. The HRT design reflects these criteria qualitatively, but a numerical investigation leaves considerable doubt as to stability in the "fast" case.

For this investigation, several parameters are required:

$$\omega_H^2 = \frac{C^2}{V \int_0^L \frac{dx}{A(x)}}$$

$$\alpha = \frac{2 V p_f}{C^2 \rho A U} \omega_H^2$$

in addition to the ω_N^2 and λ previously defined. The quantity ω_H^2 is the squared circular frequency of the free oscillations of fluid between core and pressurizer, C is the speed of sound in the core fluid, V is the core volume, L is the total length of piping between core outlet and pressurizer free surface, $A(x)$ is cross-sectional area of pressurizer pipe as a function of distance from core outlet, α is the damping coefficient for free pressurizer-core oscillations, A is the cross-sectional area of the part of the pressurizer pipe common to pressurizer and main flow, p_f is the pressure drop in the steady state due to turbulent drag between core outlet and pressurizer surface, ρ is the core fluid density and U is the core outlet velocity. 133 027

A reasonable value for ω_H^2 is 3400 sec^{-2} , assuming no gas bubbles in the core. This is in agreement with previous work by members of the HRP physics group. Forgetting for the moment entrance and turning losses in the core outlet pipe, we take p_f from the mechanical design data as 22.5 pounds/ft^2 . This leads to a value $\alpha = 0.19 \text{ sec}^{-1}$, very much less than the value given by previous HRP work. The discrepancy is apparently compound. First, only about half the length of pipe between core and free surface actually carries the turbulent main flow and the previous figure seems to ascribe full turbulence to the stagnant leg. Second, our figure ignores the entrance loss at the core outlet. We feel that this has been seriously overestimated in previous work. For a straight section of pipe without entrance loss, of length L and diameter D ,

$$p_f = \frac{fL}{D} \frac{1}{2} \rho v^2 .$$

For the actual outlet conditions $f = 0.012$, $L = 57''$, $D = 3.5''$, so $p_f = 0.2(1/2 \rho v^2)$. For the entrance loss, we do not feel entitled to take more than $0.1(1/2 \rho v^2)$ since the entrance is quite rounded, and in addition the curvature of the sphere should aid the convergence of the flow. The situation is less clear at the tee where the pressurizer flow leaves the main stream. In previous work, a friction coefficient of nearly 2, as compared with 0.1 for the entrance, was assumed. We feel that this is certainly too large, but that its exact value is very much in doubt. It should be emphasized that we are not interested in the conventional pressure drop across a fitting, but rather in a more complicated parameter which describes the irreversible pressure loss accompanying the mixing of a slowly moving stream with a rapidly moving stream. It is not at

all clear that the usual turning loss coefficient has any connection with this number. In the absence of definite experimental information we would hesitate to assume more than $0.2(1/2 \rho v^2)$ for the tee contribution to p_f . Our result is that α is probably less than $\frac{5}{2} \times 0.19 \text{ sec}^{-1} \sim .50 \text{ sec}^{-1}$.

Now, for mechanical stability, α must be less than the nuclear anti-damping coefficient

$$\frac{\lambda \frac{\omega_N^2}{H}}{\omega_H^2 + \lambda^2}.$$

This is satisfied with a very adequate margin for the slow case, but for the fast case, the anti-damping coefficient is 0.52 sec^{-1} . The calculation is rough and the only valid conclusion is that more careful calculation, as well as a reasonable experimental determination of α , is very much in order. It is entirely possible that a suitable measurement of p_f may settle the question more simply.

A completely safe procedure would be to change the present core pressurizer arrangement to reduce L considerably. At the same time, the turbulent leg of the pressurizer flow should be increased, if possible.

A minimum procedure would be to leave the design as it is with the rigid provision that the reactor shall not be carried to operation at any power greater than a watt or so without first performing a careful check of the actual α by dynamic tests in the completed system. It is otherwise entirely possible that the "breeder" (fast) case cannot be run at design power, which would be unfortunate.

The final check in a system of this type must be the verification that

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no foreseeable rate of increase of reactivity can damage the core. A rough check was made of the extensive pressure-rise calculations performed by members of the HRP physics group. A reasonably adequate calculation can be made by analogy with the simple analytical calculation of peak pressure following a sudden introduction of excess prompt reactivity. Such considerations have apparently been made by members of the HRP physics group, but we do not completely agree with their work. The following is therefore included to attempt to establish somewhat firmer contact. Our approximate treatment seems to agree reasonably well with their exact calculations, and we see no reason to question their exact results.

Suppose excess prompt reactivity is introduced at a rate b per second, and that prompt criticality is reached at power level P_0 . Then:

$$P(t) = P_0 e^{\frac{bt^2}{2\tau}}$$

where t is the time elapsed since reaching prompt criticality. This expression holds only up to a time of the order of that at which thermal expansion has essentially cancelled the reactivity increase. To find this time:

$$\begin{aligned} \dot{T} &= S P = S P_0 e^{\frac{bt^2}{2\tau}} \\ T &= \frac{\tau}{bt} S P_0 e^{\frac{bt^2}{2\tau}} \quad \text{approx.} \end{aligned}$$

Now, assume expansion at constant pressure (ω_H^2 large) and equate the reactivity decrease from thermal expansion to that inserted

$$\left| \frac{\partial k}{\partial T} \right| T = bt$$

$$bt = \frac{\tau}{bt} \left| \frac{\partial K}{\partial t} \right| S P_o e^{\frac{bt^2}{2\tau}}.$$

Defining $\omega_N^2 = \frac{S P_o}{\tau} \left| \frac{\partial k}{\partial T} \right|$, t is conveniently found as

$$t = \frac{\omega_N \tau}{b} n, \text{ where } n = \sqrt{\frac{4b}{\omega_N^2 \tau}} \sqrt{\log n}.$$

We neglect the frictional contribution to the pressure rise (which is generally satisfactory for similar situations) and calculate only the inertial contribution:

$$\delta p = \frac{c^2 S P_o}{\omega_H^2 \omega_H^2} \left| \frac{\partial \rho}{\partial T} \right| \left(\frac{b}{\tau} \right)^3 t^3$$

(all quantities in absolute units).

or:

$$\delta p = \frac{8 c^2}{\omega_H^2} \frac{d\rho}{dk} b^{3/2} \tau^{-1/2} (\log n)^{3/2}.$$

Experience in similar problems dictates that this result be arbitrarily reduced by a factor of two (or possibly more) to allow for the crudity with which the motion is represented. Take

$$c = 2400 \text{ ft-sec}^{-1}$$

$$\omega_H^2 = 3400 \text{ sec}^{-2}$$

$$\frac{d\rho}{dk} = 2.5 \text{ slug-ft}^{-3}$$

$$\tau = 10^{-4} \text{ sec (worst case)}$$

$$b = 0.015 \text{ sec}^{-1} \text{ (worst case)}$$

$$\sqrt{\frac{4b}{\omega_N^2 \tau}} = \frac{.7 \times 10^4}{\sqrt{P_o}} \quad P_o = \text{initial power in watts.}$$

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For various values of P_0 , $\log n$ is approximately given by:

$P_0(\text{watts})$	10^{-2}	10^0	10^2	10^4
$\log n$	13	10.5	8	6

Introducing the arbitrary factor of two, δp becomes (in psi):

$$\delta p = 22 (\log n)^{3/2}$$

from which we obtain the following results:

$P_0(\text{watts})$	10^{-2}	10^0	10^2	10^4
p (psi)	1000	750	500	320

If we require $\delta p < 200$ psi in order to guarantee core integrity, we see that only for initial power somewhat in excess of 10 kilowatts can the full ramp rate of 1.5% per second be tolerated. This is in qualitative agreement with the more detailed work of the HRP physics group, and we must concur in their recommendations as to allowed start-up procedures. We feel that it would be wise to go even slightly further and make sure that criticality is never reached by changing the pump speed, at least until the core-circuit flow as a function of time following pump current charge is known much better than at present. It probably should be required that the pump speed should not be increased so long as any marked temperature difference exists between core and heat exchanger.