INDEX

CF # 66-8-96

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

October 4, 1966

Wanda

To: N. T. Bray

Enclosed are copies of MSR memos issued May through August: MSR-66-9 through MSR-66-28.

RB Briggs

R. B. Briggs

alg

RBB

X-847

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 6, 1966

MSR-66-9

To: R. E. Thoma

From: B. J. Sturm

Subject: The Dimensional Change of M.S.R.E. Coolant on Freezing

In the operation of the Molten Salt Reactor Experiment one is concerned over possible effects on the radiator by accidental freezing of the coolant therein. A large expansion on freezing may bulge, burst or otherwise mechanically damage the tubes holding the salt. Contracting on freezing would presumably be less damaging, but even this effect should not be neglected. In this memorandum the available information is used to calculate the dimensional change of the coolant on freezing.

As dimensions circumscribe a volume, and volume is inversely related to density; a dimensional change can be described as a density change. The density of the molten coolant and its temperature coefficient have been measured by several investigators.[1,1a] Two of these expressions for density are

$$d\ (g/cc) = 2.296 - 0.000482\ t\ (°C) \quad \text{and}$$

$$d\ (g/cc) = 2.193 - 0.000382\ t\ (°C).$$

At 457°C, the liquidus temperature of the coolant, these expressions give density values of 2.07 and 2.02 g/cc respectively.

The composition of the coolant (34.0 mole % BeF_2, 66.0 mole % LiF) is almost exactly that of Li_2BeF_4, therefore, the density of the Li_2BeF_4 crystal can be used for that of the solid coolant. The density of Li_2BeF_4 at room temperature calculated from reported[2] crystal structure data is 2.168 g/cc, The temperature coefficient of the density of the crystal has not been reported, but a reasonable estimate can be made from that of another fluoride, CrF_3. In the structures of both CrF_3 and Li_2BeF_4 the fluoride ion is by far the largest ion, so is by far the largest contributor to the temperature coefficient. Based on the change in lattice constants between room temperature and 300°C,[3] CrF_3 has a dimensional coefficient of 0.000047 Å/Å/°C. Applying this coefficient to the Li_2BeF_4 crystal, the density at 457°C is 2.041 g/cc. The density of the solid

may be represented by the expression

$$d\ (g/cc) = 2.175 - 0.000294\ t\ (°C).$$

This expression and that used by Piper[1a] for the liquid coolant are plotted on a graph.

The density of the molten coolant at 457°C as obtained from either of the two formulas above differs by no more than 1.5% from that of solid Li_2BeF_4 at the same temperature. Within the likely limits of experimental error -- probably less than 2% -- liquid and solid M.S.R.E. coolant have the same density. Accordingly, if a dimensional change not exceeding 2% is too small to damage the M.S.R.E. piping, no direct damage should be caused by freezing the coolant. Remelting of the salt, however, could be damaging if considerable local overheating occurred. This would raise the temperature of some molten coolant causing it to expand while still confined by plugs of solid salt. Carefully controlled heating should avoid this situation. The temperature coefficient of the melt is so small that, within the limits of a 2% volume change, a rise of 87°C, bringing the temperature of the melt to 544°C, would be allowed.

A solid usually has significantly greater density than does its melt. The low density of solid Li_2BeF_4 is due to the 2 Å void along the c axis[2] of the crystal. Its room temperature density is 5% less than the 2.279 g/cc expected from the sum[4] of the molar volumes of the constituents, LiF[5] and BeF_2.

This lack of dimensional change of Li_2BeF_4 on changing state is consistent with results of experiments in which metal capsules containing M.S.R.R. fuel[6] (predominantly Li_2BeF_4) were subjected to numerous thermal cycles in which the fuel was remelted and then allowed to solidify. Most fluorides when subjected to such conditions caused considerable bulging of the capsules, but in experiments with fuel there was no detectable bulging (less than 0.001").[7]

As "bulk" densities of numerous compounds differ substantially from their theoretical densities, the application of the crystal density to the solid M.S.R.E. coolant has been questioned. Bulk density figures are empirical and depend on the manner in which the measurement is obtained.[8] They are influenced by the size and shape of the crystals as these factors affect the ability to pack together. Size and shape are affected by the conditions under which the material was produced and by subsequent treatment. When a melt, which does not change volume on solidification, crystallizes within a given confined space, the crystals would be expected to form in such sizes, shapes, and arrangements so as to fit within these confines.

References

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(1a) H. B. Piper, ORNL, private communication, May 6, 1966.

(2) J. H. Burns and E. K. Gordon, Acta Cryst. 20, 135-8 (1966).

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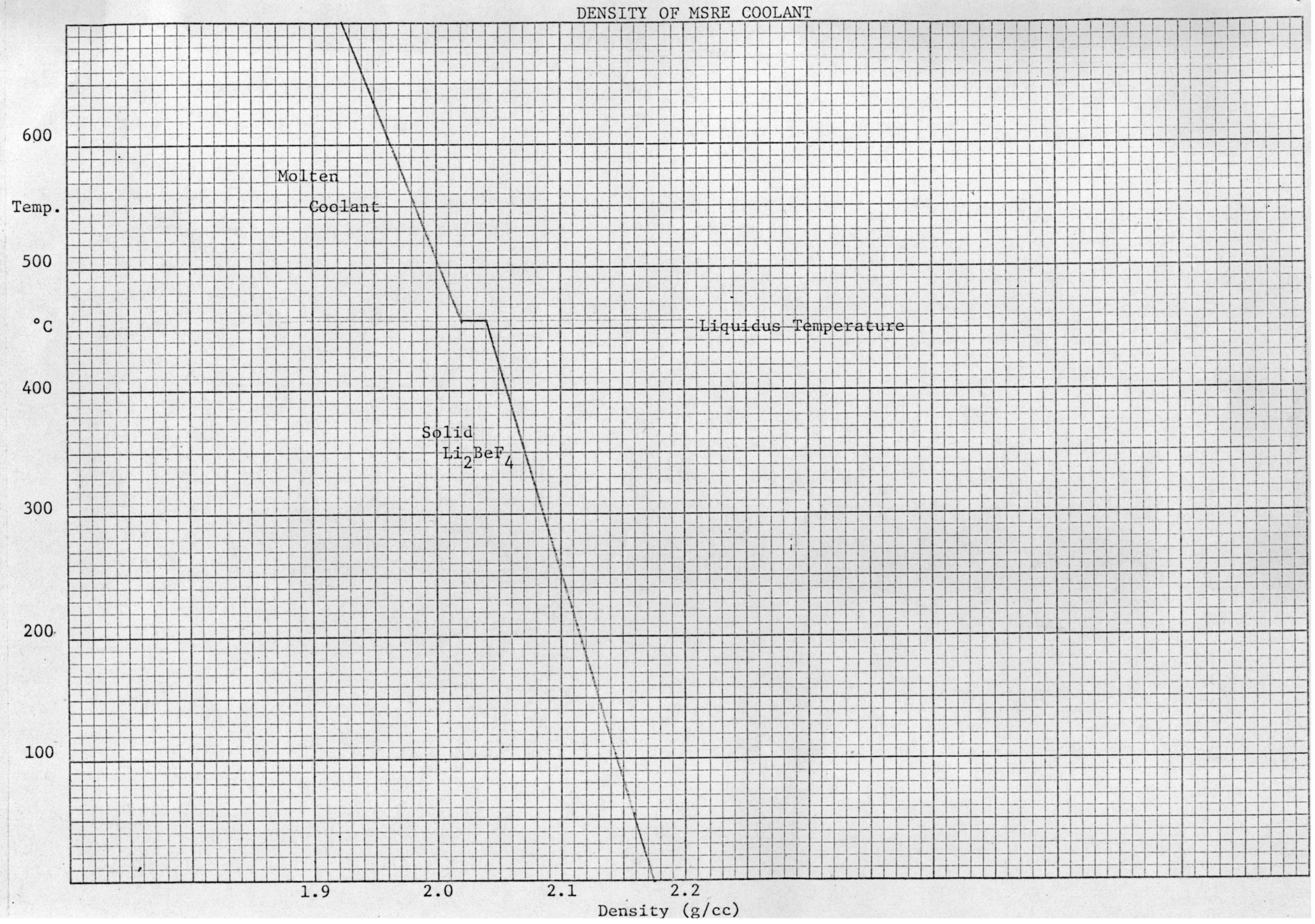
(4) S. I. Cohen and T. N. Jones, "A Summary of Density Measurements on Molten Fluoride Mixtures and a Correlation for Predicting Densities of Fluoride Mixtures", ORNL-1702 (July 19, 1954; declassified Nov. 2, 1961).

(5) W. Kwasnik, "Fluorverbindungen", Ch. 4 of Handbuch der präparativen Anorganischen Chemie, (ed. by G. Brauer), 2nd ed., Ferdinand Enke Verlag, Stuttgart, Germany, 1960, p. 220.

(6) F. F. Blankenship, "Effects of Radiation on the Compatibility of M.S.R.E. Materials," Molten Salt Reactor Program Semi-annual Progress Report, July 31, 1964, ORNL-3708, pp. 252-287.

(7) S. S. Kirslis, ORNL, Private communication, May 4, 1966.

(8) A. T. Parsons, "Specific Gravity", in Thorpe's Dictionary of Applied Chemistry, vol. XI, 4th ed., Longmans, Green and Co., N. Y., 1954, p 7.



R. E. Thoma May 6, 1966

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INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 10, 1966

To: R. E. Thoma

MSR-66-10

cc: R. B. Briggs✓
R. B. Gallaher
Dunlap Scott, Jr.
H. F. McDuffie
C. M. Boyd
A. D. Horton
J. C. White

From: A. S. Meyer, Jr.

Subject: Hydrocarbons in Seal Purge of MSRE Pumps

The helium from the 524 line of the MSRE was analyzed as a possible source of hydrocarbons which contribute to plugging of the main charcoal traps. Since the pump oil is specified to have a low vapor pressure (< 0.5 microns at room temperature) a low concentration of hydrocarbons was anticipated; therefore, the hydrocarbons were concentrated by trapping prior to the analysis. A flow of approximately 45 cc/min of the helium from the 524 line was passed through a 5-ft x 38-in OD unpacked trap which was cooled with liquid nitrogen to concentrate higher hydrocarbons, then through a cooled molecular sieve trap to catch methane. The flow was continued for about 10 hours until about a cubic foot of sample had passed. When relatively high concentrations of hydrocarbon were found in this sample an equivalent sample was taken from the molten salt pump operating at Y-12 test stand. This sample was taken in the same traps and all conditions were identical except for a higher pressure (7 psig vs 2 psig) during the sampling of the Y-12 helium. The only probable effect of this increased pressure would be to produce more efficient trapping of the contaminants.

After the traps were warmed to room temperature and allowed to stand for 24 hours for mixing, 6-cc samples of the contents were analyzed by resolving the hydrocarbons on a 10-ft x 1/4-in column of Silicone-550, which separates the hydrocarbons roughly in order of boiling point, and an 8-ft x 1/4-in column packed with silica gel modified with squalane, which retains unsaturated hydrocarbons preferentially. Both columns were operated isothermally at room temperature for a few minutes then programmed to 125°C. Hydrocarbons in the column effluents were measured with a sensitive flame ionization detector which responds as a "carbon counter." That is the molar response of any hydrocarbon is proportional to its number of carbon atoms within ± 10%. All computations could then be made in comparable units designated as methane equivalents, the molar or volume concentration that would be obtained by converting each compound to an equivalent concentration of methane. Separate samples were also analyzed by passing them through a mercuric perchlorate column at 60°C then through the same chromatographic columns. The mercuric perchlorate column absorbs all unsaturated hydrocarbons, but passes saturates unchanged (at least through C_8). Identifications and tentative identifications were made by comparing observed retention times and temperatures with those of available known compounds. The results are appended in Table I.

Proximate analyses were also made in which the gas samples, with and without the olefin absorber, were resolved on a silicone column programmed to 200°C. No attempt was made to identify individual compounds but the results (Table II) are reported grouped according to approximate carbon number. These analyses are comparable to those made on charcoal traps from Reactor Chemistry Division experiments. For the MSRE 524 line sample liquid hydrocarbons which might have been present as aerosols were determined as follows: The exhausted trap was cooled in liquid nitrogen and evacuated, and a measured quantity of carbon disulfide was distilled into it. After a lengthy procedure to assure equilibration the carbon disulfide was withdrawn and analyzed. The quantity of liquid hydrocarbons found was negligible, < 10 ppm above C_{10}, and was incorporated with the results reported in Table II.

For the MSRE samples, gas from the second trap, which was packed with molecular sieve, was analyzed and no methane (< 1 ppm) was found. The only contaminants were oxygen and nitrogen in concentration corresponding to about 2 cc of air, a quantity which could have easily been introduced in the fittings during the assembly of the traps.

The accuracy of these analyses is subject to some question. The reproducibility of well defined chromatographic peaks is better than 5% and with a few exceptions all but trace components in these samples agreed within 10% between the measurements made on the two columns. The concentrations reported were calculated on the assumption of quantitative trapping and a concentration factor of 100 based on the assumption that the traps were sealed with helium at liquid nitrogen temperature. Both these assumptions are not valid and would yield opposite biases. The factor of incomplete trapping is the more important and the results are probably somewhat lower than the actual concentration, perhaps as much as 25%. It is suggested that in any future sampling one or more unconcentrated samples be taken for comparison.

The composition of the 524 line sample is qualitatively different from that of samples derived from the thermal cracking of pump oil in that predominant concentration of C_5—C_7 unsaturates are present. Most of these unsaturates do not correspond to any known compounds available for comparison and some are retained stronger by the silica gel column which may indicate polyunsaturated compounds or acetylenic hydrocarbons. Such an identification is questionable since no similar compounds have been found either as lighter hydrocarbons or in other samples. The sample obtained from the Y-12 pump bears little resemblance to the MSRE sample.

The high concentration of unsaturates is characteristic of thermal and/or radiation cracking but is not conclusive. Dunlap Scott has suggested that these unsaturates may be present in the original pump oil and have been depleted in the Y-12 pump during its extended periods of its operation. This might be proved by the analysis of a sample of the pump oil, but would probably require a preliminary vacuum distillation to concentrate the lighter hydrocarbons.

A. S. Meyer, Jr.

A. S. Meyer, Jr.
Analytical Chemistry Division

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Table I. Volatile Hydrocarbons in Seal Purge of MSRE Pumps

Hydrocarbon ( ) tentative identification	Concentration, vpm Methane Equivalent	
	MSRE 524 Line (Sample 7090)	Y-12 Pump
methane	< 0.01	-
ethane	0.5	0.2
ethylene	0.5	0.2
propane	2.4	0.02
propylene	34.0	0.2
i-butane	0.1	-
n-butane	0.8	-
butene-1	1.2	0.03
i-butylene + trans-butene 2	7.6	0.05
cis butene -2	-	0.02
i-pentane	0.2	-
n-pentane	1.2	0.04
pentene T	1.1	0.02
cis and trans pentene 2	1.9	0.04
(isomeric pentene)	43.0	0.02
(n-hexane)	~2	-
(isomeric hexane)	4.3	-
(hexene-T)	20	-
unidentified unsaturate	20	-
" "	33	-
" "	22	-
(isomeric hexenes)	2.2	-
(isomeric heptanes)	2.2	
unidentified unsaturate	~2	1.3
(n-heptane)	5	-

Table II. Hydrocarbons in Seal Purge of MSRE Pumps

Approximate Carbon No.	Concentration, vpm - Methane Equivalent			
	MSRE 524 Line (Sample 7090)		Y-12 Pump	
	Total HC	Unsaturate	Total HC	Unsaturates
$< C_6$	166	143	2.5	0.5
C_6	48	40	3.1	3.0
C_7	16	5	0.3	ND
C_8	17	< 1	0.4	ND
C_{9-10}	12	ND[a]	0.4	ND[a]
$> C_{10}$	17	ND[a]	ND	ND[a]
	276	189	6.7	3.5

[a] Could not be determined because saturated hydrocarbons retained by olefin absorber.

5-19

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 16, 1966

MSR-66-11

To: R. B. Briggs

From: P. N. Haubenreich

Subject: Persons Qualified to Operate the MSRE at Power

Last year I reported to you the names of those who were fully qualified to operate the MSRE during the zero-power nuclear experiments (in MSR 65-35 and MSR-65-51). Subsequently all operations personnel were given further training with particular emphasis on operation of the reactor at high power. Most of this training came in a one-week lecture session in October, 1965 and in simulator practice in November, 1965. After the training and before the beginning of power operation, each operator was given a written examination. After the papers were graded, each operator was examined orally. The following persons demonstrated that they are qualified in every respect to carry out the duties of Shift Engineer or Shift Technician during any stage of MSRE operations, including operation at high power.

Shift Engineer	Shift Technician
J. L. Crowley	W. H. Duckworth
R. H. Guymon	J. D. Emch
P. H. Harley	E. H. Guinn
V. D. Holt	T. G. Hill
T. L. Hudson	C. C. Hurtt
A. I. Krakoviak	L. E. Penton
H. R. Payne	W. E. Ramsey
H. C. Roller	R. Smith, Jr.
R. C. Steffy	S. R. West
W. C. Ulrich	J. E. Wolfe

Whenever the reactor is being operated at power, the reactor control room is in charge of a person who is certified as fully qualified for this responsibility. Persons on the operating crew who have not been so certified are assigned other duties for which they are qualified.

Paul N. Haubenreich

Paul N. Haubenreich

PNH:al

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INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

May 19, 1966 MSR-66-12

TO: Distribution

FROM: R. Blumberg

SUBJECT: Recent Repair of MSRE Sampler-Enricher Drive Motor

On April 29 during a testing operation, the teleflex cable drive motor quit working with about 15 ft of cable extended. An electrical check showed a complete open circuit to two of the motor leads. The reactor was drained and in the following five working days the failed unit was removed, repaired and replaced. A description is given here of the trouble, the day-to-day work involved with fixing it and some conclusions on related matters such as decontamination, the use of the decon cell, contamination control and the administration of radioactive maintenance work.

Daily Log and Description of Work

Friday, April 29. While the latch was being lowered toward the pump bowl with about 15 ft of cable reeled out, the motor stopped and would not respond. An electrical check was made by Clyde Mathews and two open circuits were found. The reactor was drained and put into a standby condition.

Saturday, April 30. Dick Gallaher and Ray Carnes working with the manipulator hand and the transfer tool, through the open door of area 1C pulled up the 15 ft of cable and finally the latch and got it into the 3-A area. The operational and maintenance valves were then closed. The remainder of the day was spent in the preparing the decon cell to receive area 1-C, working out a procedure for the transfer, and in removing shielding, and piping connections.

Monday, May 2. Area 1-C was removed and transferred to the decon cell in a 55 gal drum, with several plastic bags. The decon cell was set up as a gas mask C-zone and remained that way for the rest of the operation. A radiation survey was made to help in planning the remainder of the repair. After some cleaning the upper lid was removed and 3 disconnected leads were found broken or burned off at the underside of one of the six connectors in the upper lid. Electricians ground out the broken connector.

Tuesday, May 3. A new connector was welded in, the leads re-soldered to the pins and extensive electrical checkout of the gear was done. The outside of the box, cable and latch were cleaned with damp swabs while the crane picked the unit out of the drum. An attempt was made to straighten out some kinks in the teleflex cable. This did not work out too well as the cable was too hot to handle directly. Epoxy was poured into the connector on the under side of the upper lid to act as mechanical support and electrical insulation.

Wednesday, May 4. The cable was cleaned in a garbage can full of soap suds by scrubbing with hard bristle brushes. This treatment brought the radiation level down considerably so that a millwright was able to satisfactorily straighten out the teleflex cable using hand pliers, while protected with 3 pair of rubber gloves. The dirty plastic bags were replaced with clean ones and the unit was returned to the drum where the upper lid was reconnected.

Thursday, May 5. Using a long cable strung from the sampler control panel, the electrical gear was checked functionally and found to be in order. The unit was replaced and leak checking commenced. Since the system proved satisfactory, the work was complete.

Discussion

1. Radiation Levels. The cable and latch read 3 and 60 R/hr at 1 in. respectively. This was after being swabbed w/damp rags. After the soap and water washing, these were down by a factor of six. The radiation was soft and attenuated greatly with distance. Gamma scan analysis of samples of the wash water and contaminated blotter paper indicate that the radiation comes mainly from Ba, La-140, Te, I-132, Ru-103, Sr-89 and Ce-141. Undoubtedly these levels are close to the upper limit for the kind of direct repairs that were done here.

2. Decontamination. From this and other experiences with salt contaminated equipment, we can say that wiping is not very effective but that washing is effective. It appears that decontamination requires large quantities of solvent and perhaps long dissolution times, and may be of considerable use in future situations involving salt contaminated parts.

3. Decon Cell. This cell has been used for the pump rotary element inspection, revision of the graphite sample hold down assembly, off-gas system valves and filters, and lately for the sampler-enricher. It has proven very effective in contamination control. Windle Ogg said it was very much like a walk-in hood. However in the future we shall need better and more shielding due to higher and higher levels of radiation. Thus far we have gotten along with some concrete blocks, and some 4 in. slabs of lead from the sampler-enricher. It may be feasible to install in the cell a frame that would support the old HRT lead bridge that we now have on hand. This has a 4 in. thick floor and a 2 in. thick side piece. Early in the process of any repair job, usually when time is at a premium, we are faced with a decision. How shall we set up the decon cell to provide shielding to make virtually direct repairs. This depends on many factors; the size of the equipment, radiation level, type of repair which may or may not be known and the available shielding. Fortunately there are not too many different shielding configurations to choose between. However, the decision must be made and the cell set up before going on any further. It is worth noting that thus far most of the shutdown time is spent in the process of effecting a repair, rather than in removing or replacing from the installed position. Since we do not have spares to immediately replace most of our equipment, it appears that repair activities in the decon cell will become very important with respect to how quickly we can get the reactor back in operation.

4. Contamination Control. Each time that we handle contaminated equipment there is bound to be a spread of contamination, leading to a general increase in background radiation. The effect can range all the way from a nuisance to an intolerable and expensive situation. Some of this is unavoidable. However, most of it can be avoided by strict adherance to C-zone and related procedures, set up by our H.P. men. It may be worthwhile to assign a technician to help traffic flow, maintain C-zone boundaries and provide assistance for those who are working in the cell. We must also learn to use plastic bags and the many types of containers that are available. During the work on the sampler, Woody Spencer was able to have made 2 special sizes of plastic bags on very short notice. One was used to transport the unit from the sampler to the decon cell and the other was used to test out the cable drive motor.

Administration. Remote maintenance jobs are subject to a great many uncertainties such as how hot will it be, how long will it take, and since it is usually a failure of some sort, what actually needs to be done to effect a repair. Coupled with these uncertainties is the fact that many different skills are required at one time or another. In this past operation, the following people were involved: the equipment specialists (Gallaher and Carnes), remote maintenance group (Blumberg and Shugart), P and E supervision, 8 different craft skills, operations, health physics, procurement and fabrication (Spencer and Anderson) and management. While this job was accomplished fairly smoothly, it is felt that improvements could have been made. The following are suggestions which may help.

1. Make definite plans. These should be written whenever time permits and should contain enough details to indicate in advance, who (what craft) will do each step and whether any special equipment or information is required. The purpose of the written plan being to utilize lead time to the best possible advantage.

2. Communicate with everyone as soon as a course of action has been established. Even the cleaners like to know a little bit more than just what they have to do. Also communication must work two ways. Those who know what is to be done must tell others, and those who don't know should go find out. One way to accomplish this communication is simply by walking around to the people involved and telling or asking them.

Conclusions

1. The recent maintenance work on the sampler-enricher was a fairly successful operation.

2. The loss to the MSRE program in schedule delay was balanced in part by a gain in experience and education in maintenance practices.

3. Repair work in the decon cell may be the critical path in our future operations.

4. The uncertainties in our maintenance needs requires our keeping a flexible position.

R. Blumberg

RB:job
Attachment

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INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

April 7, 1966

To: R. B. Briggs MSR-66-13

From: J. H. Shaffer

Subject: Production of ^{233}U-Bearing Fuel Concentrate for the MSRE

A preliminary survey has been made of processing techniques and approximate costs for the production of sufficient ^{233}U-bearing fuel concentrate mixture to accommodate the planned replacement of ^{235}U with ^{233}U in the fuel of the MSRE. Presumably, ^{233}U as its tetrafluoride salt will be combined with ^{7}LiF to form the binary eutectic mixture, ^{7}LiF-$^{233}UF_4$(73-27 mole%). Additions of this mixture to the barren fuel solvent in the MSRE will follow the general procedure used during its initial fueling with ^{235}U. A total requirement of 60 kg of ^{233}U has been estimated by P. N. Haubenreich and includes a 20% excess over the probable amount needed to achieve criticality in the MSRE.

Although the ^{235}U-bearing fuel concentrate for the initial fueling of the MSRE was prepared in our fluoride production facility, ^{233}U must be handled in facilities that are specifically designed for alpha containment and gamma shielding. Increased radiation safety requirements for alpha containment result from the very short half-life of ^{233}U relative to that of ^{235}U. Since ^{232}U is an isotopic impurity of ^{233}U that is currently available, the gamma ray attenuation of its daughter activities (notably the 2.6 Mev gamma from ^{208}Tl) is of some concern. Shielding requirements would be influenced by the actual concentration of ^{232}U in the material allotted for this program and by the time period required for the complete refueling operation.

Uranium-233 is an artificial isotope that is produced by neutron bombardment of thorium-232. Since the material is obtained as a by-product of nuclear reactor operations, an initial purification step is required to remove extraneous fission products, fuel contaminants, and corrosion products. Thus, the necessity of this processing step is

independent of the ^{232}U content of the material. Operational costs for this production phase, if conducted in an existing ORNL solvent extraction system, may not exceed $100,000 (R. E. Brooksbank, Chemical Technology Division) for the 60 kg ^{233}U charge if handling operations can be minimized.

A second preparatory phase of the production operation is required to convert the uranyl nitrate product of the solvent extraction process to anhydrous uranium tetrafluoride. This operation will also require alpha containment and perhaps some shielding to cope with ^{232}U daughter activities. Material must also be handled in small batches to conform with nuclear safety requirements. Although this conversion is handled routinely on ^{235}U in the Y-12 plant, facilities for processing ^{233}U in quantities required by this program are nonexistent. However, operational costs have been estimated on the basis of small batch preparations by G. B. Marrow, Y-12 Development Division, at about $33,000 for the proposed 60 kg of ^{233}U. This production requirement, together with other anticipated commitments of the Y-12 plant, may provide sufficient incentive for the construction of a ^{233}U handling facility at Y-12.

The blending of ^{7}LiF with $^{233}UF_4$ as the binary eutectic mixture, its melting and purification by treatment with anhydrous HF and H_2, and its transfer into containers and fuel enriching capsules for loading into the MSRE can be accomplished by procedures previously employed during the initial MSRE fueling operations. However, because of additional costs for alpha containment and gamma shielding, this operation can be more economically performed in existing facilities within the Chemical Technology's ^{233}U pilot plant. This production can be conducted concurrently with the solvent extraction processing phase with a minimum personnel requirement. As a second alternative, this production phase may be combined with $^{233}UF_4$ preparation at the Y-12 plant. In either event, operational costs should not exceed $22,000. This amount is approximately three times the cost incurred during the preparation of

the ^{235}U-bearing concentrate mixture to allow for increased costs of remote handling procedures and smaller batch sizes that will be dictated on the basis of the reduced critical mass of ^{233}U.

According to this preliminary survey, the production of sufficient ^{233}U-bearing fuel concentrate mixture to accommodate the planned replacement of ^{235}U with ^{233}U in the fuel of the MSRE will require about 10 weeks of production time at a cost of about $155,000. This production plan is contingent upon Y-12's construction of a suitable processing facility or upon the availability of commercial processors of ^{233}U.

J. H. Shaffer

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INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

June 1, 1966

MSR-66-14

To: P. N. Haubenreich

From: Stanley Cantor

Subject: Specific Heats and Thermal Conductivities of MSRE Salts

Specific Heats

Coolant Salt

The specific heats for liquid coolant as given in the MSRE Design Data Sheets, ORNL CF-64-6-43, correspond to data taken on a 68-32 mole % $LiF-BeF_2$ mixture reported in MSRP Prog. Rept. March 1 - Aug. 31, 1961, ORNL-3215, p. 131. The MSRE coolant, which is 66-34 mole % $LiF-BeF_2$, would be expected to exhibit a slightly lower (less than 1%) specific heat. I, therefore, suggest that the equation for specific heat be revised to read $0.172 + 3.31 \times 10^{-4}\ t\ (^{o}F)$. Assuming that the specific heats on the 68-32 mixture had the usual experimental uncertainties of about 6%, then it should be understood that the revised equation for the coolant yields specific heats that have uncertainties of approximately 7%.

Fuel Salt

At present the Design Data Sheets do not contain any specific heat information for the MSRE fuel salt. I suggest that

$$0.48 \pm 0.05 \frac{\text{Btu}}{\text{lb-}^{\circ}\text{F}}$$

be used as an estimate until measured values or until improved methods of estimation become available. The above estimate was obtained by a method described in ORNL-3913, pp. 39-30.

Thermal Conductivities

The only positive statement that can be made at present is that the thermal conductivity of coolant as given in the Design Data Sheets is much too high. The number given is 3.5 Btu-ft/ft^2-hr-$^{\circ}$F and is of unknown origin. For the thermal conductivity of the fuel salt, 3.5 also seems to be used for most calculations.

My own estimates, in part based on older thermal conductivity measurements, are

2.8 Btu-ft/ft^2-hr-$^{\circ}$F for the coolant salt

2.6 Btu-ft/ft^2-hr-$^{\circ}$F for the fuel salt.

However, these may be much too high. J. W. Cooke has just prepared a CF memo (currently in press) in which he discusses measurements for composition 71.2 - 23.0 - 5.0 - 0.8 mole % $LiF-BeF_2-ZrF_4-UF_4$; he obtains values between 0.75 to 0.92 Btu-ft/ft^2-hr-$^{\circ}$F. These results are between one-third to one-fourth of the expected values. These measurements

would also imply that previous ORNL measurements of thermal conductivities of molten fluorides were erroneously high.

Because of the large uncertainties that presently exist in measured thermal conductivities of fluorides, any estimated thermal conductivities of MSRE fuel and coolant salts would be quite unreliable.

Stanley Cantor

Stanley Cantor

SC/pah

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File

June 6

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

June 3, 1966

To: Distribution MSR-66-15

Subject: The Sampling Schedule for the MSRE Surveillance Specimens

Scope

We recommend that the sampling of the surveillance specimens for the MSRE be made when the reactor produces approx 10,000 Mw-hr and again when approx 50,000 Mw-hr of work is attained. This schedule is governed by the potential decreases in some of the desirable mechanical properties of INOR-8 as a result of irradiation by thermal neutrons. For this reason, INOR-8 is stressed more than graphite in this memorandum. Details on the INOR-8 and graphite specimens are not given except through references. The purpose is to present the sampling schedule mentioned above and to explain why it is recommended.

Nomenclature and Locations

Graphite and INOR-8 (Hastelloy N) surveillance specimens for the MSRE[1] are located at three different locations. These are:

1. The Reactor Core Specimens (5081 and 5085).[2]
2. The Reactor Core Control Specimens (5081 and 5085).
3. The Reactor Vessel Specimens (5065 and 5085).

The Reactor Core Specimens are located longitudinally along a line approx 3.6 in. from the longitudinal center line of the core. They consist of three separate stringers, S, R, and L, of graphite and INOR-8 specimens as shown in Fig. 1. Matching stringers, SC, RC, and LC for the Reactor Core Control Specimens are located in the controlled test rig in the high bay area of Building 7503. These controls are exposed to molten fluoride salts and temperature conditions similar to those of the reactor, but they are not subjected to any radiation.

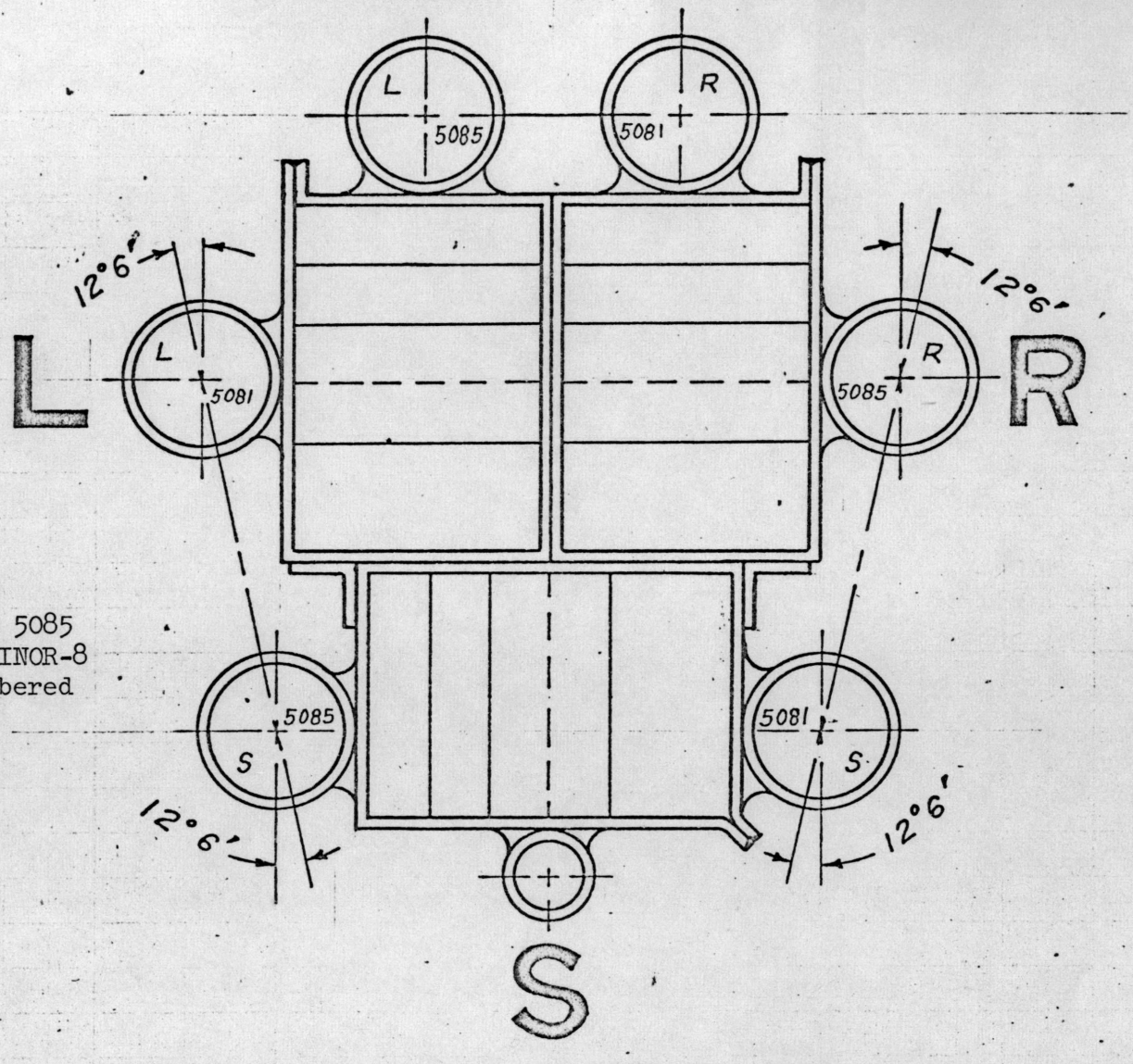
Three sets of the Reactor Vessel Specimens, V1, V2, and V3, are located approx 4.5 in. outside the reactor vessel wall. These are INOR-8 specimens without any graphite.

---

[1] MSR Program Semiann. Progr. Rept. Aug. 31, 1965, ORNL-3872, pp. 87-92.

[2] The numbers in parentheses are the heat numbers of the INOR-8 specimens; they are not part of the titles.

7-10-65 WHC



Note:
The numbers 5081 and 5085 are heat numbers for the INOR-8 test specimens in the numbered positions.

Fig. 1. A Transverse Cross Sectional View 20 in. Below the Top of the Reactor Core Specimen Assembly.

Sampling Schedule

The potential changes in the mechanical properties of the INOR-8 govern our sampling schedule. The schedule is based upon: (1) irradiation studies on INOR-8 made by the Mechanical Properties Group of the Metals and Ceramics Division, (2) the evaluation of the service life of the MSRE,[3] (3) neutron flux values for the core specimens position calculated by J. R. Engel,[4] and (4) neutron flux values at the reactor vessel wall, the Reactor Vessel Specimens locations.[5] These flux values are plotted in Fig. 2. Table 1 is the sampling schedule that we recommend. On this basis, the first sampling should be made sometime within the first half of August, 1966. This takes into account the power operations accumulated by the MSRE to date and assumes continuous operation at 6.5 to 7.5 Mw power levels until August 1, 1966.

Discussion and Future Sampling Schedules

The last three columns in Table 1 are maximum values corresponding to the maximum flux regions near the horizontal midplane of the reactor core. The flux gradients potentially enable us to examine a range of conditions. For example, the ratio of the maximum to the minimum thermal neutron flux along the Reactor Core Specimens is approximately two. Therefore, the INOR-8 specimens from the upper and lower ends of the stringer S for the first sampling should enable us to get data approaching the 10,000-hr level of service information sought by R. B. Briggs.[6] The specimens from the central part of the stringer that were in the maximum flux region should provide data for the 20,000-hr service, the currently extrapolated safe service life for the reactor.[6]

The maximum thermal neutron dose on the reactor vessel lags far behind both of these; see the last column in Table 1. These additional data should help in the further evaluation of the service life of the MSRE while the dose in its INOR-8 is still low enough to have little effect.

None of the Reactor Vessel Specimens should be removed during the first sampling because the thermal neutron dose on these is too low to produce changes that would provide significant data.

To this point, sampling has been concerned only with the removal of specimens from their test sites. This is only the first half of each sampling operation. The last half of each involves the replacement of the specimens removed with new specimens. The replacements will normally be from MSRE grade or improved

---

[3] R. B. Briggs, Effects of Irradiation on Service Life of MSRE, ORNL-CF-66-5-16 (May 4, 1966).

[4] Private communication from J. R. Engel of the Reactor Division on October 8, 1965.

[5] Private communication from H. B. Piper of the Reactor Division on May 24, 1966.

[6] R. B. Briggs, Effects of Irradiation on Service Life of MSRE, ORNL-CF-66-5-16 (May 4, 1966), p. 21.

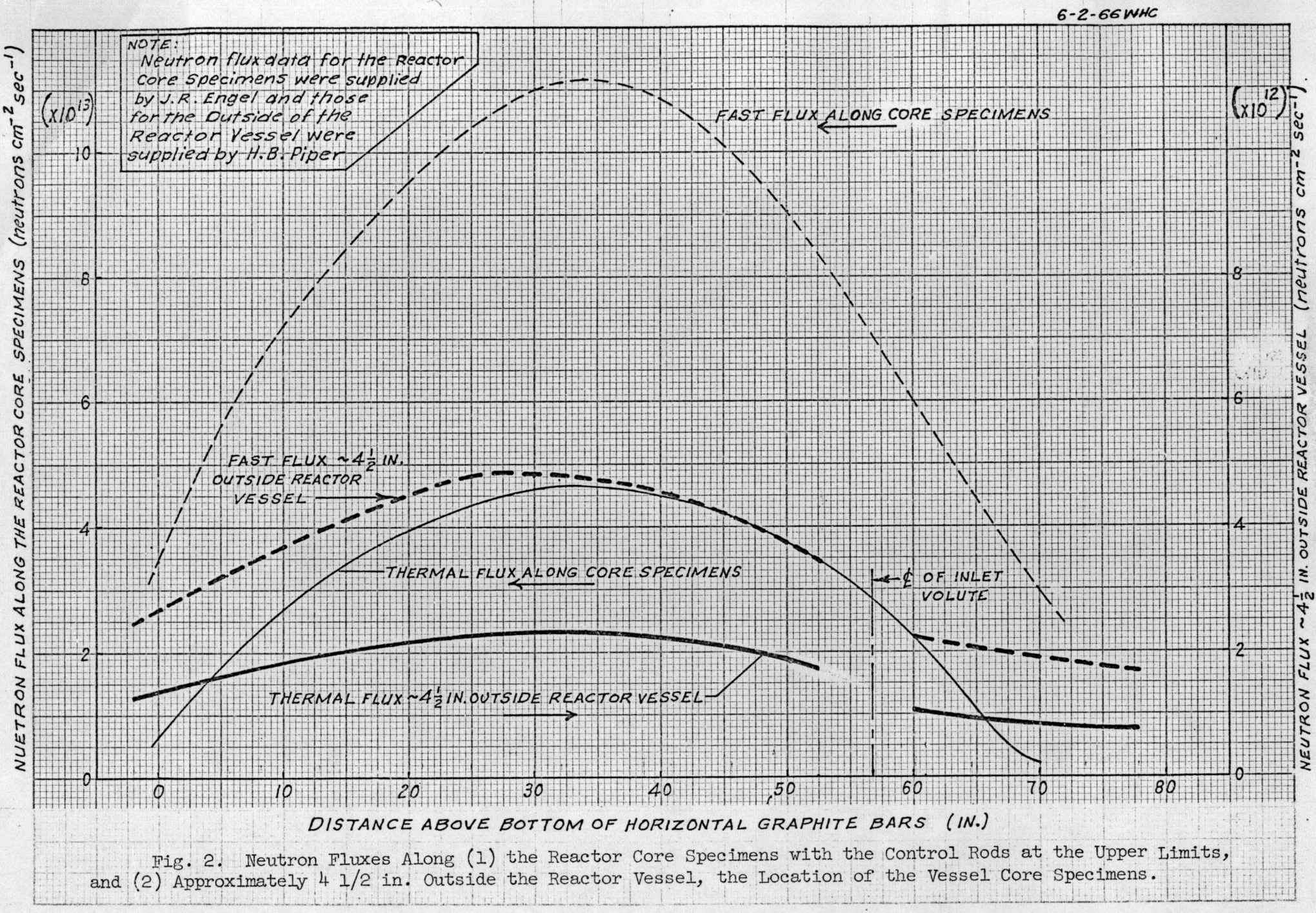


Fig. 2. Neutron Fluxes Along (1) the Reactor Core Specimens with the Control Rods at the Upper Limits, and (2) Approximately 4 1/2 in. Outside the Reactor Vessel, the Location of the Vessel Core Specimens.

Table 1. Sampling Schedule for the MSRE Surveillance Specimens

Sampling No.	Reactor Mwhr[a]	Stringer		Thermal Dose on Stringer, nvt	Equivalent Operating Time for Vessel at 10 Mw, hr	Actual Dose on Vessel, nvt (thermal)
		Core[b]	Vessel			
1	10,000	S		1.7×10^{20}	20,000	8.3×10^{18}
2	50,000	R		8.5×10^{20}	100,000	4.1×10^{19}
			V1	4.1×10^{19}	5,000	4.1×10^{19}
		S1		6.6×10^{20}	80,000	4.1×10^{19}

[a]The integrated power levels at which specimens are pulled can vary ± 5% from the indicated figure.

[b]A matching control stringer is also removed from the controlled test rig with each Reactor Core Specimens sampling.

grades of graphite plus various heats of experimental alloys. The replacement stringers will be identified with the original designation and a numerical suffix. For example, during the first sampling, the S stringer will be removed and replaced with stringer S2. Then, S2 will be removed in the second sampling and replaced with S3.

The second sampling is determined by the time required for the various specimens to acquire thermal neutron doses that will produce significant and measurable changes. Obviously stringers R and S1, respectively, are sources of additional data on the original materials and the first data on the new materials. The Reactor Vessel Specimens, V1, was removed at this time to determine the effects of both nitriding and thermal dose. The latter is sufficient to cause measurable changes, but not severe changes with respect to the service life of the MSRE.

The sampling schedule given in Table 1 shows the desirable situation in that the thermal neutron dose accumulates much faster in the INOR-8 surveillance specimens than it does in the INOR-8 associated with the reactor vessel. For this reason, we have shown only two samplings. We believe that future sampling schedules should be guided by the data obtained under this current schedule.

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6-10-66

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

June 9, 1966

MSR-66-16

To: P. N. Haubenreich

From: R. E. Thoma

Subject: Fuel and Coolant Sampling Schedule for Initial Period of MSRE Run No. 7

The cumulative results of chemical analysis of MSRE fuel and coolant samples have provided assurance that throughout the previous operating periods the salts have remained in very pure condition and are of the correct composition. Confirmation that purity and composition are maintained must, of course, be continued in the next period of sustained power operation. A schedule of chemical analysis is recommended, therefore, which is a compromise of the annoying desire for continuous monitoring with the need for economic prudence.

Initially, fuel salt specimens should be obtained from the pump bowl on alternate days. Every fourth sample should be submitted for oxide analysis. As discussed previously, the possibility exists that fluorine produced in the frozen samples may exclude the possibility of determining the oxide concentration of the fuel salt if fluorine oxidizes the oxide ion rapidly. Oxide values, originally reported to be approximately 100 ppm, have fallen to approximately 50 ppm during the approach to full power. I will indicate special test procedures which concern the question of determining possible errors in future analyses from evolving fluorine.

It is anticipated that the results of the first five to ten analyses will indicate that the composition of the fuel salt has remained constant and that no marked increase in corrosion of the fuel salt containment vessel has occurred. Hopefully, the results will be so uniform as to permit the sampling frequency to be decreased temporarily to twice a week and ultimately to one sample every few days.

Coolant salt sampling should be carried out at one-week intervals in order to provide indication that the coolant circuit is not sustaining corrosion and that fuel salt is not leaking into the coolant system.

The aggregate costs of the analyses required by the sampling schedule recommended here is approximately $65,000 for a twelve-month period. These costs, although moderately expensive, enable us to acquire much worthwhile chemical information about the condition of the reactor salts. One serious limitation of such analyses is that they indicate the condition of the salt only at the moment of sampling. Clearly, one of the major incentives for improvement of analytical methods is the need for on-line methods for determining the concentration of uranium, zirconium, chromium, and oxygen in the MSRE salts.

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R. E. Thoma

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INTRA-LABORATORY CORRESPONDENCE
OAK RIDGE NATIONAL LABORATORY

June 15, 1966

MSR-66-17

To: Distribution

From: P. N. Haubenreich

Subject: In-Situ Heat Treatment of Reactor Vessel Closure Weld

Preface

This work was done last fall and the writeup was practically completed shortly thereafter. Although this memo is late in being issued, I think it is worth issuing as a convenient record of the details.

Introduction

While the reactor was shut down, being prepared for initial power operation (about a year after the reactor vessel was installed), tests of INOR-8 from the heats used in the vessel showed that the closure weld must have poor mechanical properties in the as-welded condition. Specifically, the rupture life and ductility were unacceptably low. The test results showed that these properties could be greatly improved by heat-treating the weld for a reasonable period at 1400°F. Although this is well above the normal operating temperature of the MSRE (1200°F), it appeared attainable, and calculations showed that heating the reactor vessel to this temperature involved no harmful thermal stresses. Therefore, we decided to heat-treat the closure weld in place. The duration at 1400°F was specified as about 100 hours.

Description of Treatment

The reactor vessel heaters are contained in nine groups of vertical tubes, as shown in Fig. 1. There are gaps where the fuel inlet and outlet pipes and the fuel drain line leave the enclosure. The uneven distribution of the heaters leads to unavoidable variation in temperature around the vessel (when it contains no salt, of course). This can be partially corrected by operating R-1 heaters at higher current than R-2 or R-3. (Further subdivision of the heater controls was not practical.)

The heat treatment took place after the fuel system had been open for installation of the core specimens and for removal and replacement of the fuel pump rotary element. Therefore the system was

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purged for about 40 hr with dry helium before the heatup was started. Beginning on September 26, 1965, the fuel system and portions of the coolant piping in the reactor cell were heated gradually, reaching 1200°F on September 28. The reactor vessel temperatures were brought on up until the hottest point was near 1400°F on that date. After some adjustments were made to reduce the variation in temperature around the vessel, the temperatures were brought on up until the lowest temperature (TE-R-49) in the ring of thermocouples just below the closure weld was at 1400°. Temperatures approached steady state on September 30. Figure 2 shows the circumferential temperature profiles just below the closure weld and just below the distributor ring at 1500 on that date. The variation around the vessel amounted to 62°F. (A more complete picture of temperatures on the reactor vessel and heater settings at that time are given in the table at the end of this memo.)

Figure 3 shows the temperature history of TE-R-1 and TE-R-49, the hottest and least hot thermocouples in the ring below the closure weld. Taking into consideration the effect of the temperature before reaching "steady state" we judged that the equivalent of 100 hours at 1400°F had been reached by noon, October 3. Beginning at that time the reactor and piping were cooled down in preparation for the containment leak testing.

Before the reactor cell was closed, we inspected the penetration of Line 103 through the thermal shield for possible effects of the unusually high reactor temperature. We found nothing of any consequence.

Paul N. Haubenreich

Paul N. Haubenreich

PNH:al

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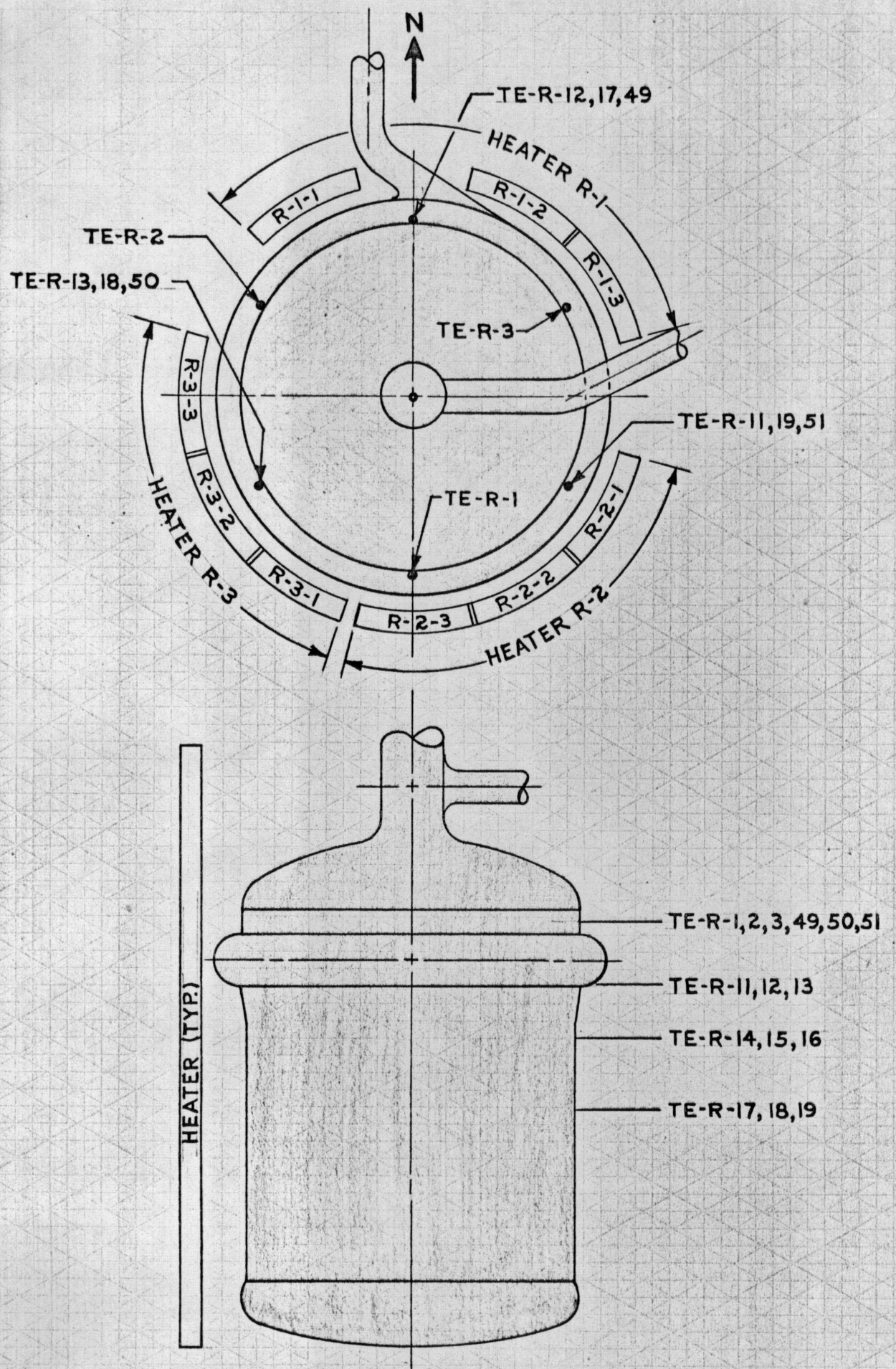

FIG. 1 REACTOR VESSEL HEATER & THERMOCOUPLE LAYOUT

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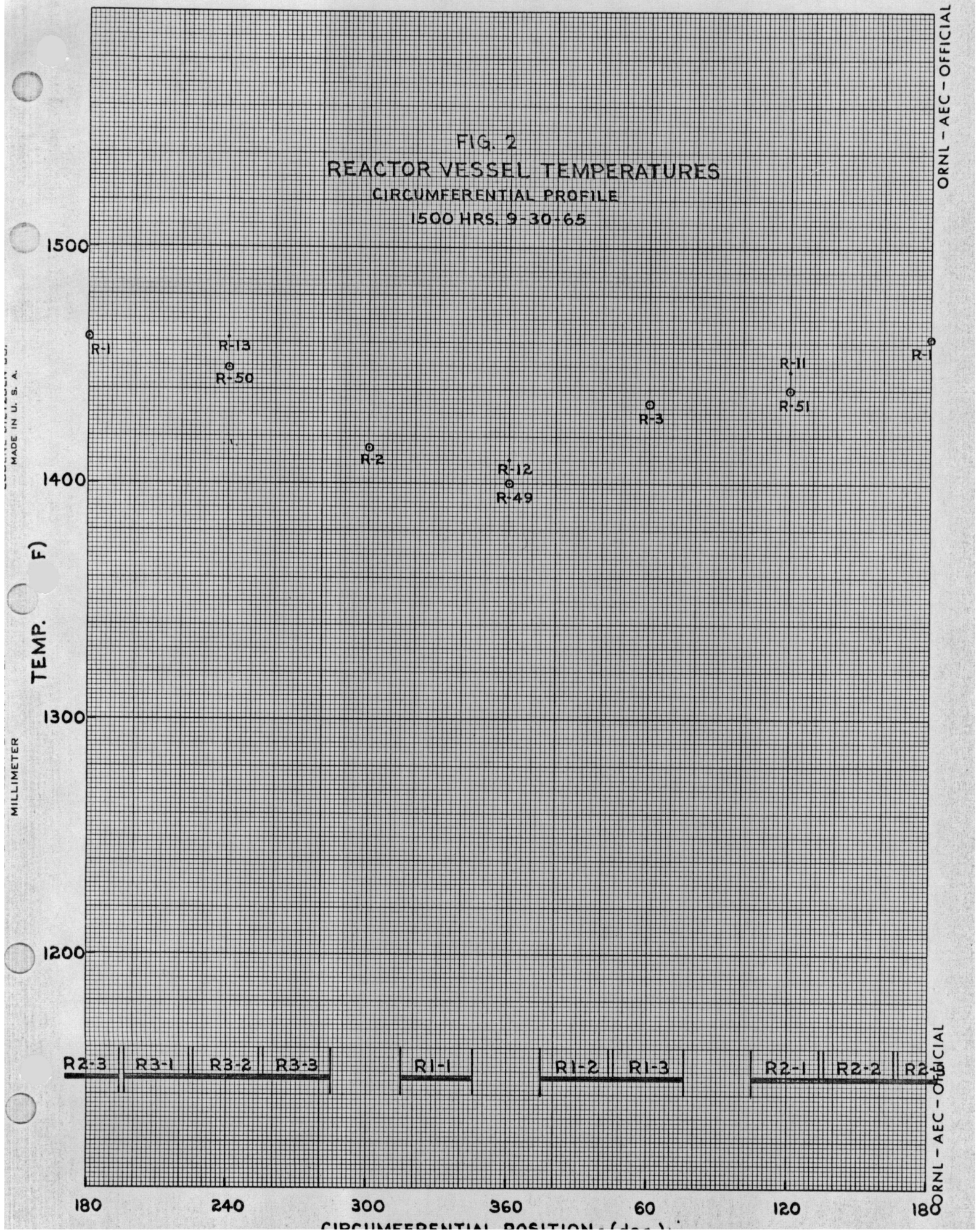
FIG. 2
REACTOR VESSEL TEMPERATURES
CIRCUMFERENTIAL PROFILE
1500 HRS. 9-30-65
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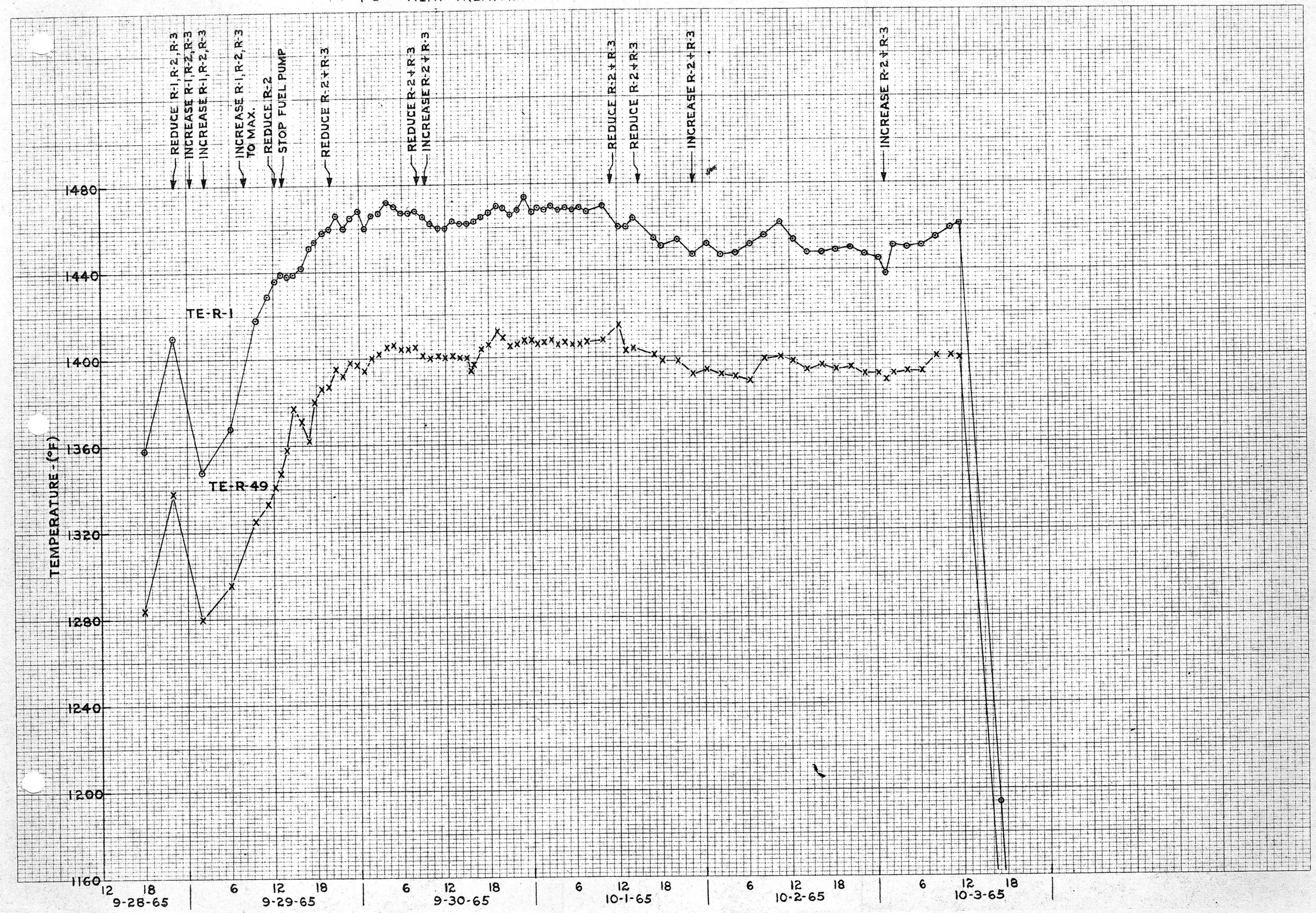
1500
1400
1300
1200
TEMP. F)
R-1
R-13
R-50
R-2
R-12
R-49
R-3
R-11
R-51
R-1
R2-3
R3-1
R3-2
R3-3
R1-1
R1-2
R1-3
R2-1
R2-2
R2-3
180
240
300
360
60
120
180
CIRCUMFERENTIAL POSITION - (deg.)
ORNL - AEC - OFFICIAL
MILLIMETER
MADE IN U. S. A.

FIG. 3 HEAT TREATMENT OF REACTOR CLOSURE WELD



STEADY-STATE TEMPERATURES AND HEATER SETTINGS

(1500, September 30, 1965)

THERMOCOUPLE LOCATIONS

z(in.) \ θ	0	60	120	180	240	300
2	49	3	51	1	50	2
13 1/2	12	-	11	-	13	-
22 1/2	-	16	-	14	-	15
34 1/2	17	-	19	-	18	-
46 1/2	-	22	-	20	-	21
58 1/2	23	-	25	-	24	-

INDICATED TEMPERATURES (°F)

Z(in.) \ θ	0	60	120	180	240	300
2	1400	1434	1440	1462	1449	1415
13 1/2	1410	-	1448	-	1462	-
22 1/2	-	1437	-	1471	-	1414
34 1/2	1403	-	1443	-	1454	-
46 1/2	-	1422	-	1445	-	1392
58 1/2	1376	-	1408	-	1408	-

HEATER SETTINGS

Heater	R1-2	R1-3	R2-1	R2-2	R2-3	R3-1	R3-2	R3-3	R1-1
Location (°)	15-45	45-75	105-135	135-165	165-195	195-225	225-255	255-285	315-345
Amps	25	24	24.4	23.5	24.6	24	21.5	27.5	22.5

(Note: θ is angle in degrees clockwise from north, z is distance below closure weld.)

ORNL - AEC - OFFICIAL

B. Briggs

MSR 66-18

July 5, 1966

To: R. B. Briggs

From: Roy C. Robertson

Subject: MSBR Steam System Performance Calculations

Introduction

The Molten Salt Breeder Reactor study[1] employed a 3,500 psi-1,000°F/1,000°F once-through steam system similar to that used in the TVA's Bull Run steam station.[2] This system was analyzed for the case of 700°F feedwater temperature (Case A), involving a reheat steam preheater and mixing heating of the feedwater, and for 580°F feedwater, (Case B), requring th addition of one more high-pressure stage of feedwater heating. The reasons for these two feedwater temperatures are discussed in the reference report.[1] The flowsheets for the two cases are shown in Figures 1 and 2, following.

The properties of the fuel, blanket and coolant salts and the performance of the boiler-superheaters. steam reheaters, reheat steam preheaters, etc., are described in the reference report.[1]

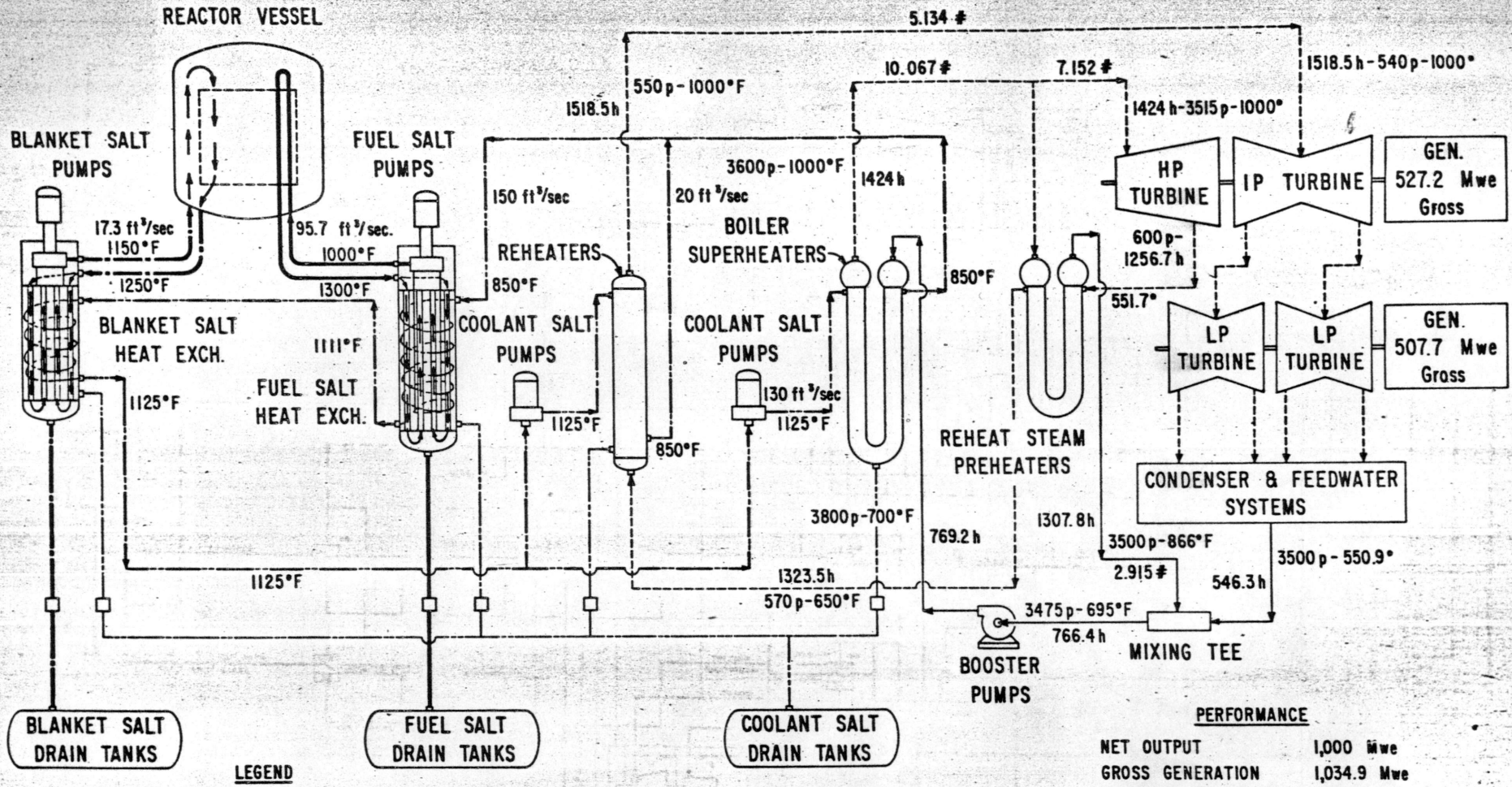
Performance

The energy balances made to determine the thermodynamic performance of the steam power system for both Case A and B were made using the simplified flowsheet shown in Figure 3. The simplifications were primarily concerned with the handling of the computations and such minor items as the turbine shaft seal steam, and have negligible effect on the values obtained for the overall efficiency of the cycle.

The performance data for both the A and B cases is summarized in Table I, following. The steam flowrates and properties at the various state points in the cycles are shown in Tables II and IV.

---

1. P. R. Kasten, et al, Design Studies of 1;000 Mw(e) Molten-Salt Breeder Reactors, ORNL-3996, July, 1966.
2. Engineering Data TVA Steam Plants, Supplements 1 & 2, Technical Monograph No. 55, Vol. 3, Tennessee Valley Authority, Knoxville, Tenn., June, 1963.



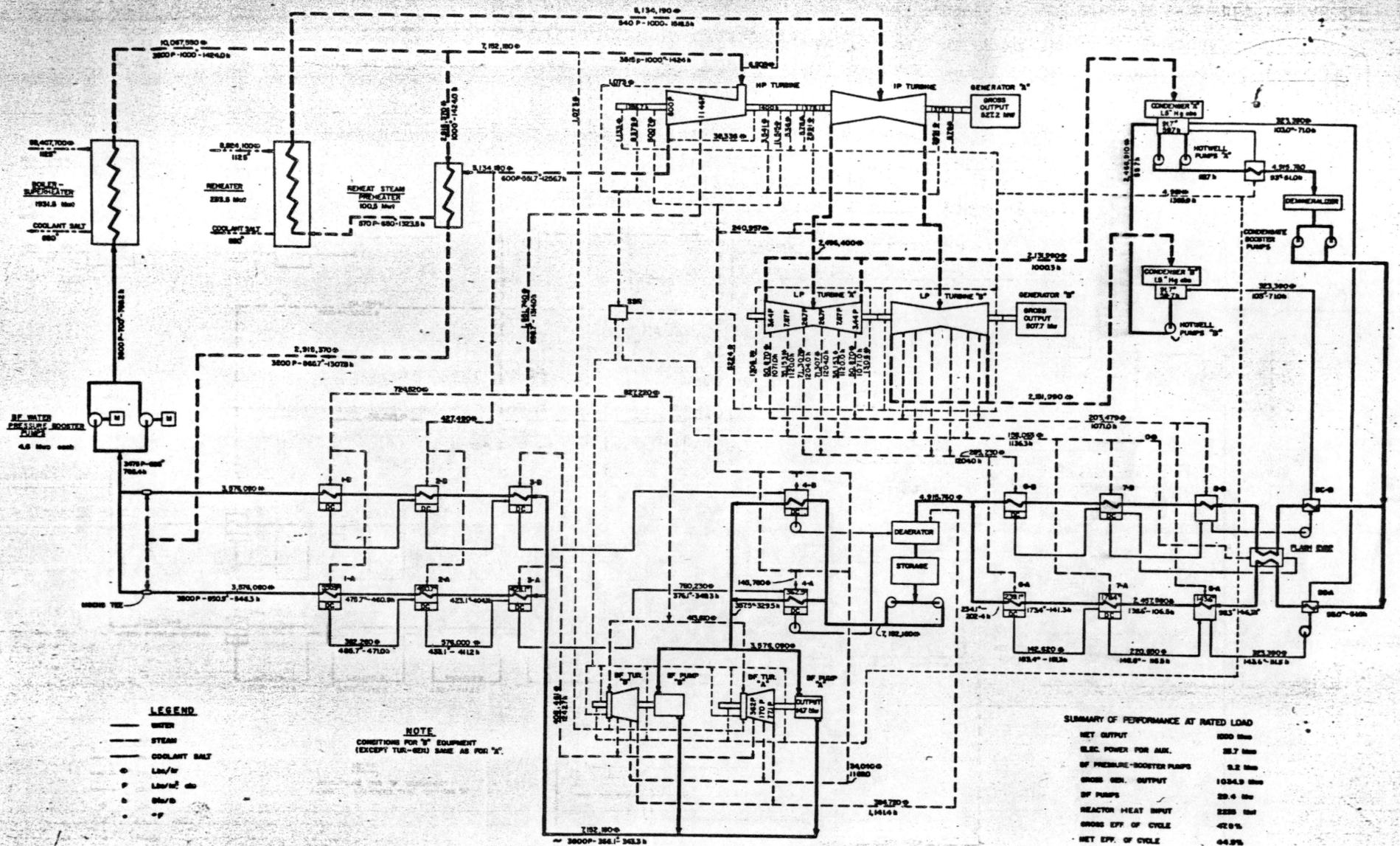
LEGEND

FUEL
BLANKET
COOLANT
STEAM
H_2O
#_____ 10^6 lb/hr
p_____ psia
h_____ Btu./lb.
°_____ °F
—□— Freeze Valve

MOLTEN SALT BREEDER REACTOR FLOW DIAGRAM

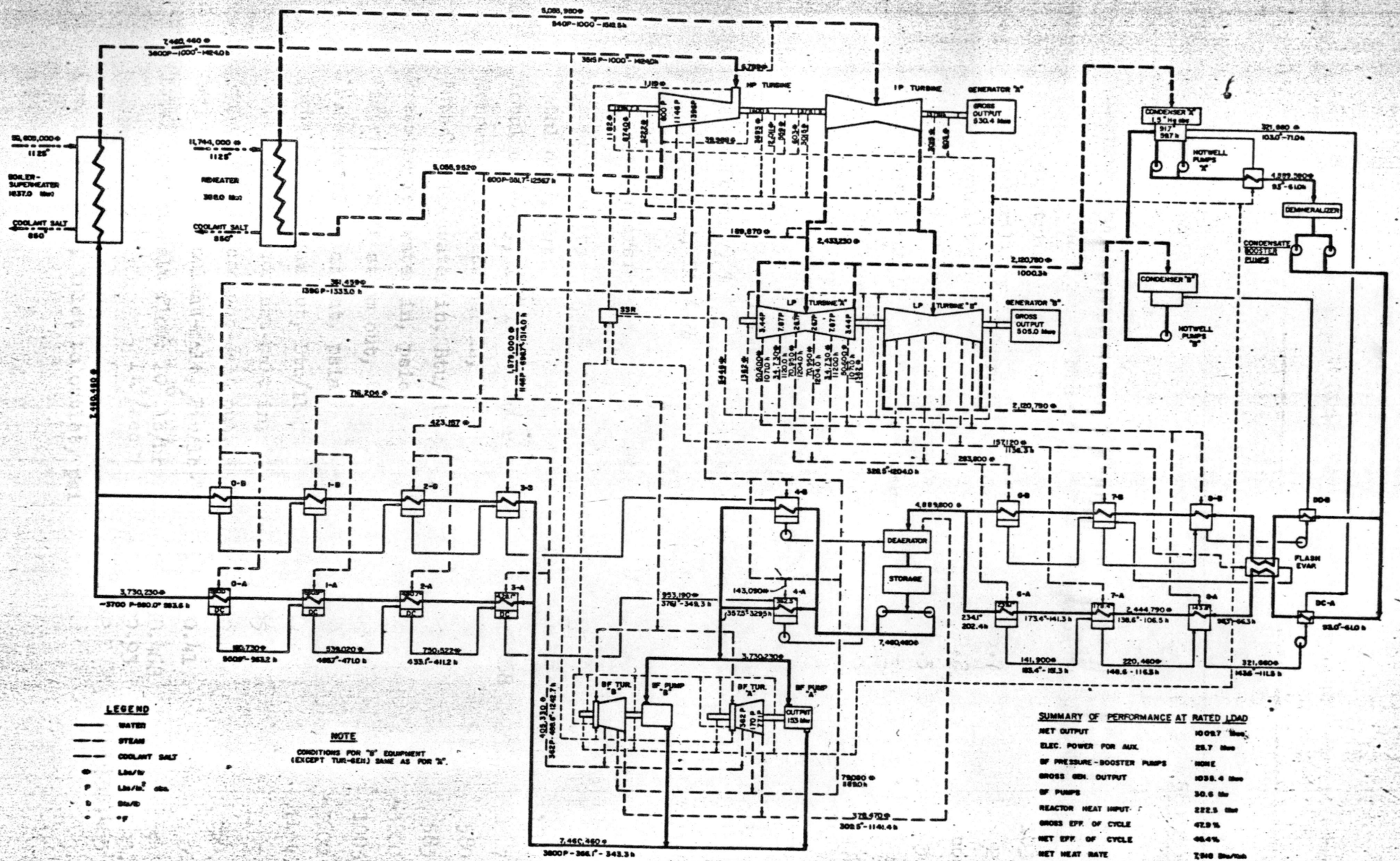
PERFORMANCE

NET OUTPUT	1,000 Mwe
GROSS GENERATION	1,034.9 Mwe
BF BOOSTER PUMPS	9.2 Mwe
STATION AUXILIARIES	25.7 Mwe
REACTOR HEAT INPUT	2,225 Mwt
NET HEAT RATE	7,601 Btu/kwh
NET EFFICIENCY	44.9 %



MSBR STEAM POWER CYCLE

Fig. 1. MSBR Steam System Flowsheet for 700°F Feedwater (Case A).



MSBR STEAM POWER CYCLE

Fig. 2. MSBR Steam System Flowsheet for 580°F Feedwater (Case B).

Table I. MSBR Steam-Power System Design and Performance Data.

	700°F Feedwater	580°F Feedwater
General Performance:		
Reactor heat input, Mwt	2225	2225
Net electrical output, Mwe	1000	1009.7
Gross electrical generation, Mwe	1034.9	1035.4
Station auxiliary load, Mwe	25.7	25.7
BF pressure-booster pump load, Mwe	9.2	none
BF pump steam-turbine power output, Mw	29.3	30.6
Flow to turbine throttle, 10^6 lb/hr	7.152	7.460
Flow from superheater, 10^6 lb/hr	10.068	7.460
Gross efficiency: (1034.9 + 29.3)/2225 = %	47.83	
(1035.4 + 30.6)/2225 = %	-	47.91
Gross heat rate, Btu/kwh	7136	7124
Net efficiency, %	44.9	45.4
Net heat rate, Btu/kwh	7601	7518
Boiler-Superheaters:		
Number of units	16	16
Total duty, Mwt	1931.5	1837.0
Total capacity, 10^6 lb steam/hr	10.068	7.460
Temperature feedwater in, °F	700	580
Enthalpy feedwater in, Btu/lb	769.2	583.6
Pressure feedwater in, psia	~3800	~3800
Temperature of steam out, °F	1003	1003
Pressure of steam out, psia	~3600	~3600
Enthalpy of steam out, Btu/lb	1424.0	1424.0
Temperature of coolant-salt in, °F	1125	1125
Temperature of coolant-salt out, °F	850	850
Avg. sp. heat coolant-salt, Btu/lb-°F	0.41	0.41
Total coolant-salt flow: 10^6 lb/hr	58.468	55.608
" ft^3/sec	129.93	123.57
" gpm	58,316	55,463
Coolant-salt Δp, inlet to outlet, psi		

	700°F Feedwater	580°F Feedwater
Boiler Feedwater Pumps:		
Number of units	2	2
Centrifugal pump:		
Number of stages	6	6
Feedwater flow rate, 10^6 lb/hr total	7.152	7,460
Required capacity, gpm	8,060	8,408
Head, approximate ft	9,380	9,380
Speed, rpm	5,000	5,000
Water inlet temperature, °F	357.5	357.5
Water inlet enthalpy, Btu/lb	329.5	329.5
Water inlet specific volume, ft^3/lb	~0.01808	~0.01808
Steam-Turbine drive:		
Power req'd. at rated flow, Mw each	14.66	15.30
Power, nominal hp each	20,000	20,000
Throttle steam conditions, psia/°F	1070/700	1070/700
Throttle flow, lb/hr each	413,610	431,400
Exhaust pressure, approx. psia	77	77
Number of stages	8	8
Number of extraction points	3	3
Boiler Feedwater Pressure Booster Pump:		
Number of units	2	none
Centrifugal pump:		
Feedwater flow rate, 10^6 lb/hr total	10.067	
Required capacity, gpm each	9,500	
Head, approx. ft	1,413	
Water inlet temperature, °F	695°F	
Water inlet pressure, psia	~3,500	
Water inlet specific volume, ft^3/lb	~ 0.03020	
Water outlet temperature, °F	~ 700	
Electric motor drive:		
Power req'd. at rated flow, Mw each	4.587	
Power, nominal hp each	6,150	

	700°F Feedwater	580°F Feedwater
Steam Reheaters:		
Number of units	8	8
Total duty, Mwt	293.5	388.0
Total capacity, 10^6 lb steam/hr	5.134	5.056
Temperature steam in, °F	650	551.7
Pressure steam in, psia	~ 570	~ 600
Enthalpy steam in, Btu/lb	1323.5	1256.7
Temperature steam out, °F	1000	1000
Pressure steam out, psia	~ 540	~ 540
Enthalpy steam out, Btu/lb	1518.5	1518.5
Temperature coolant-salt in, °F	1125	1125
Temperature coolant-salt out, °F	850	850
Avg. sp. heat coolant-salt, Btu/lb-°F	0.41	0.41
Total coolant salt flow, 10^6 lb/hr	8.884	11.744
ft^3/sec	19.742	26.098
gpm	8,861	11,714
Coolant-salt Δp, inlet to outlet, psi	~ 60	~ 60
Reheat Steam Preheater:		
Number of units	8	none
Total duty, Mwt	100.45	
Total capacity, 10^6 lb heated steam/hr	5.134	
Temperature heated steam in, °F	551.7	
Temperature heated steam out, °F	650	
Pressure heated steam in, psia	~ 580	
Pressure heated steam out, psia	~ 570	
Enthalpy heated steam in, Btu/lb	1256.7	
Enthalpy heated steam out, Btu/lb	1323.5	
Total heating steam, 10^6 lb/hr	2.915	
Temperature heating steam in, °F	1000	
Temperature heating steam out, °F	866	
Pressure heating steam in, psia	3515	
Pressure heating steam out, psia		

CASE A - FEEDWATER AT 700°F, REHEAT STEAM AT 650°F

The steam conditions in the MSBR cycle are almost identical to those in the Bull Run system except that a reheat steam preheater, a mixing tee on the feedwater line, and a feedwater pressure-booster pump have been added.

The flow rates are not the same in the two cycles in that the MSBR system has a net output of 1,000 Mwe whereas the Bull Run plant delivers about 891 Mwe. Further, the MSBR main feedwater pumps need not deliver water at more than about 3,800 psia as compared to the ~4,200 psia required in the Bull Run system. This latter change effects the energy balances sufficiently to prevent direct use of "scaled-up" Bull Run cycle data.

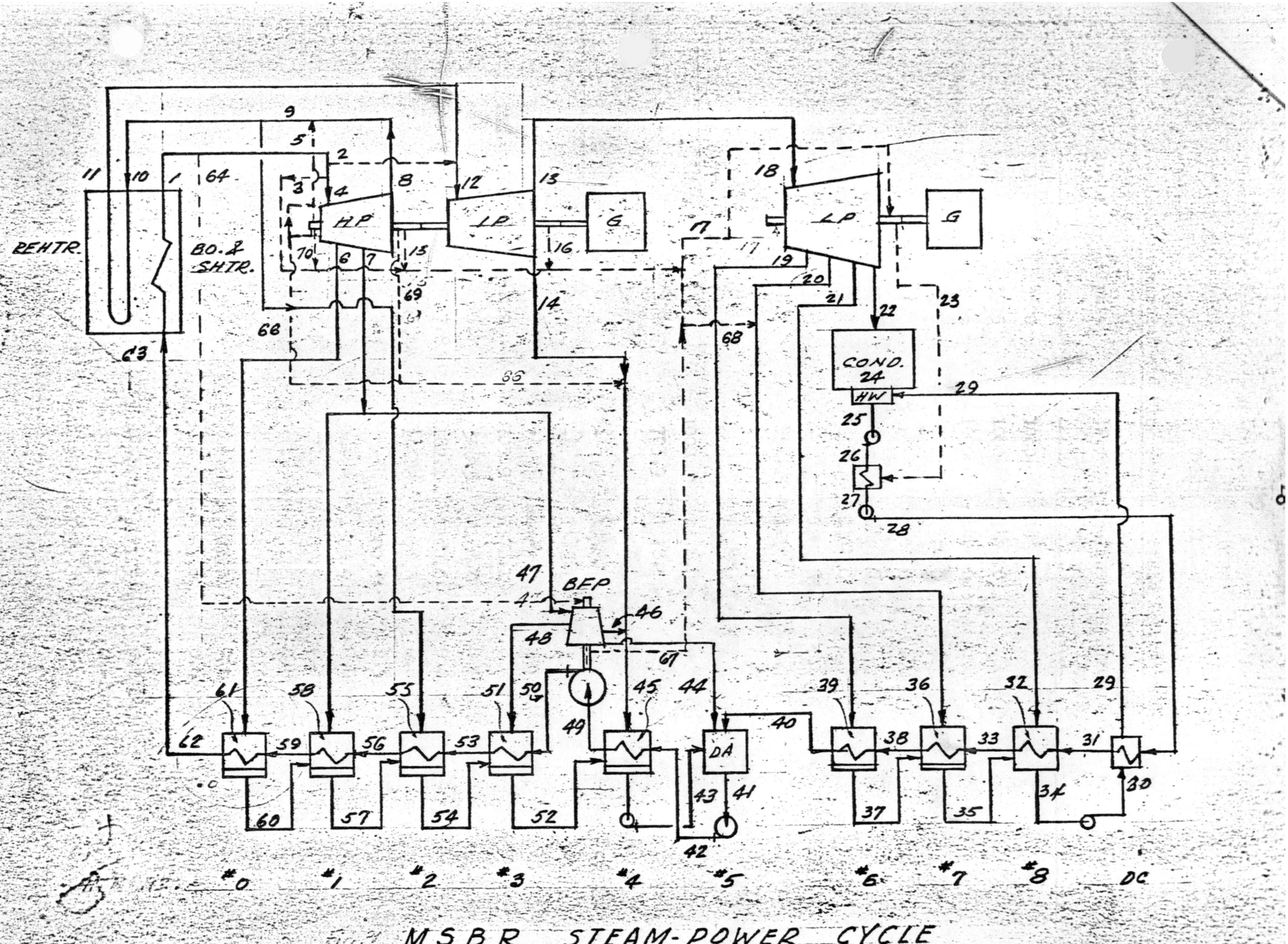
To facilitate the analysis of the MSBR steam-power cycle the system was approximated by the simplified arrangement shown in Figure 3. The simplifications, primarily concerned with showing one heater rather than two parallel ones and with the handling of the turbine seal bleed-offs, have negligible effect on the overall system performance.

The properties and flow rates at each of the state points are shown in Table II. For convenience, the cycle was analyzed on the basis of 1 lb of steam leaving the boiler-superheater, point "1", and all other flows are expressed as a proportion of this value of w_1. The performance of the conventional portion of the system, i.e. that corresponding to the Bull Run cycle and that of Case B, following, was determined first and then the reheat steam preheater and feedwater tempering added as a last step. Heater No. 0, shown in Figure 3', is used in the Case B analysis and will be discussed in that section of the performance calculations.

At Heater No. 0:

This heater not used in Case A. $h_{62} = h_{59} = h_{63}$

$$w_6 = w_{61} = w_{60} = 0$$



MSBR STEAM-POWER CYCLE

Figure 3

Table II

Properties and Flow Rates at State Points in Power Cycle Case A

State Point	p psia	°F	Btu/lb	Flow Rates lbs/lb w_1	lb/hr
1	3600	1000	1424.0	1.0000	7,152,180
2	3515	1000	1424.0	0.000906	6,508
3	3515	1000	1424.0	0.000151	1,073
4	3500	1000	1424.0	0.99879	7,143,526
5	-	-	1400.0	0.00536	38,336
6	1396	-	1333.0	*	0
7	1146	698.7	1314.0	0.21696	1,551,737
8	600	-	1255.6	0.77226	5,523,342
9	~ 600	551.7	1256.7	0.77762	5,561,678
10	~ 600	-	1256.7	0.71785	5,134,192
11	-	1000	1518.6	0.71785	5,134,192
12	540	1000	1518.5	0.71876	5,140,701
13	~ 172	705.9	1378.1	0.68410	4,892,806
14	-	-	1378.1	0.03369	240,957
15	-	-	1256.7	0.001328	9,512
16	-	-	1378.1	0.000966	6,938
17	-	-	-	0.000725	5,222
18	~ 172	705.9	1378.1	0.68410	4,892,806
19	26.7	328.5	1204.0	0.03988	285,229
20	7.8	182.1	1120.0	0.01966	140,612
21	3.44	146.8	1071.0	0.02845	203,479
22	1.5" Hg	-	1000.3	0.59618	4,263,987
23	-	-	1355.9	0.000664	4,720
24	1.5" Hg	91.7	59.7	0.59618	4,263,987
25	-	-	-	0.68661	4,910,759
26	-	-	59.7	0.68661	4,910,759
27	-	-	61.0	0.68727	4,915,479
28	-	-	61.0	0.68727	4,915,479
29	-	103.0	71.0	0.09043	646,772
30	-	143.6	111.5	0.09043	646,772
31	-	98.3	66.3	0.68727	4,915,479
32	3.17	143.6	1071.0	0.02845	203,479

Table II, Cont'd. Case A

State Point	psia	°F	Btu/lb	Flow Rates lbs/lb w_1	lb/hr
33	-	138.6	106.5	0.68727	4,915,479
34	-	-	111.5	0.09043	646,772
35	-	148.6	116.5	0.06198	443,292
36	7.24	178.4	1136.3	0.02210	158,063
37	-	183.4	151.3	0.03988	285,229
38	-	173.4	141.3	0.68727	4,915,479
39	24.6	239.1	1204.0	0.03988	285,229
40	-	234.1	202.4	0.68727	4,915,479
41	71.7	304.5	274.2	1.0000	7,152,180
42	-	304.5	274.2	1.0000	7,152,180
43	-	314.5	284.6	0.25894	1,851,985
44	77.1	309.5	1141.4	0.05379	384,716
45	-	-	1295.0	0.04076	291,523
46	170	368.4	1189.0	0.00476	34,044
47	1146	698.7	1314.0	0.11566	827,221
48	362	488.8	1242.7	0.05711	408,461
49	-	357.5	329.5	1.0000	7,152,180
50	~ 3800	366.1	343.3	1.0000	7,152,180
51	337	428.1	1242.7	0.05711	408,461
52	-	376.1	349.3	0.21818	1,560,463
53	~ 3775	423.1	404.3	1.0000	7,152,180
54	-	433.1	411.2	0.16107	1,152,002
55	570	480.7	1256.7	0.05977	427,486
56	~ 3750	475.7	460.9	1.0000	7,152,180
57	-	485.7	471.0	0.1013	724,516
58	1053	550.9	1314.0	0.1013	724,516
59	~ 3700	550.9	546.3	1.0000	7.152,180
60	1325.8	560.9	563.2	*	0
61	1325.8	580.0	1333.0	*	0
62	~ 3700	580.0	583.6	1.0000	7,152,180
63	3700	580.0	583.6	1.0000	7,152,180
64	3515	1000.0	1424.0	0.000151	1,073
66	-	-	1400.0	0.00161	11,515
67	-	-	1141.4	0.000151	1,073
68	-	-	1358.3	0.00244	17,451
69	-	-	1256.7	0.000695	5,007
70	-	-	1400.0	0.000574	4,077

At Heater No. 1:

$$w_{58}(h_{58} - h_{57}) = w_1(h_{59} - h_{56})$$

$$w_{58}(1314.0 - 471.0) = w_1(546.3 - 460.9)$$

$$w_{58} = w_{57} = 0.1013\, w_1$$

At Heater No. 2:

$$w_{55}(h_{55} - h_{54}) + w_{57}(h_{57} - h_{54}) = w_1(h_{56} - h_{53})$$

$$w_{55}(1256.7 - 411.2) + 0.1013\, w_1(471.0 - 411.2) = w_1(460.9 - 404.3)$$

$$w_1(460.9 - 404.3)$$

$$w_{55} = 0.05977\, w_1$$

$$w_{54} = w_{55} + w_{57} = 0.16107\, w_1$$

At Heater No. 3:

$$w_{51}(h_{51} - h_{52}) + w_{54}(h_{54} - h_{52}) = w_1(h_{53} - h_{50})$$

$$w_{51}(1242.7 - 349.3) + 0.16107\, w_1\,(411.2 - 349.3) =$$

$$w_1(411.2 - 343.3)$$

$$w_{51} = 0.05711\, w_1$$

$$w_{52} = w_{51} + w_{54} = 0.21818\, w_1$$

At Heater No. 4:

$$w_{45}(h_{45} - h_{43}) + w_{52}(h_{52} - h_{43}) = w_1(h_{49} - h_{42})$$

$$w_{45}(1295.0 - 284.6) + 0.21818\, w_1(349.3 - 284.6) =$$

$$w_1(329.5 - 274.2)$$

$$w_{45} = 0.04076\, w_1$$

$$w_{43} = w_{45} + w_{52} = 0.25894\, w_1$$

At Heater No. 5: (Deaerator):

$$w_{40} = w_1 - w_{43} - w_{44} = 0.74106\, w_1 - w_{44}$$

$$w_{44}(h_{44}) + w_{40}(h_{40}) + w_{43}(h_{43}) = w_1(h_{41})$$

Heater No. 5, cont'd.

$$w_{44}(1141.4) + 202.2(0.74106\ w_1 - w_{44}) + 0.25894(284.6) = 274.2\ w_1$$

$$w_{44} = 0.05379\ w_1$$

$$w_{40} = 0.68727\ w_1 = w_{33}$$

At Boiler Feedpump:

$$w_{47}(h_{47} - h_{48}) + (w_{47} - w_{48})(h_{48} - h_{46}) + (w_{47} - w_{48} - w_{46})(h_{46} - h_{44}) + w_{64}(h_{64} - h_{67}) = \frac{w_1\ (h_{50} - h_{49})}{\eta_{BR}},$$

where η_{BR} is the proportion of the pump input work that goes into the pumped fluid as an enthalpy increase. This is assumed to be the same as the pump performance in the Bull Run cycle, giving a value of η_{BR} of 0.986.*

The value of w_{64} and other seal steam and bleed-offs were taken as the Bull Run cycle quantity times the ratio of the MSBR output to the Bull Run plant output, or 1.1320. These values are shown in Table II and are not explained further.

$$w_{47}(1314.0 - 1242.7) + (w_{47} - w_{48})(124217 - 1189.0) + (w_{47} - w_{48} - w_{46})(1189.0 - 1141.4) + w_{64}(1424.0 - 1141.4) = w_1(343.3 - 329.5)$$

This reduces to:

$$172.60\ w_{47} - 101.30\ w_{48} - 47.6\ w_{46} = 13.9526\ w_1$$

$$w_{48} = w_{51} = 0.05711\ w_1$$

Thus,

$$w_{47} = 0.11435\ w_1 + 0.2758\ w_{46}$$

Also,

$$w_{47} = w_{44} + w_{48} + w_{46} = (0.05379 + 0.05711)w_1 + w_{46}$$

Therefore,

$$0.11435\ w_1 + 0.2758\ w_{46} = 0.1109\ w_1 + w_{46}$$

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* From the TVA Bull Run plant data, Dwg No. 47K1110-2:

$$805{,}000(1314.0 - 1242.7) + (805{,}000 - 351{,}200)(1242.7 - 1189.0) + (805{,}000 - 351{,}200 - 112{,}100)(1189.0 - 1141.4) = \frac{6{,}335{,}200(344.8 - 329.5)}{\eta_{BR}}$$

Boiler Feedpump, cont'd.

$$w_{46} = 0.00476\ w_1$$

$$w_{47} = 0.11566\ w_1$$

$$w_7 = w_{58} + w_{47} = 0.21696\ w_1$$

$$w_{14} = w_{45} - w_{46} - w_{66} - w_{69} = 0.03369\ w_1$$

$$w_{68} = 0.00244\ w_1$$

At Heater No. 6:

$$w_{39}(h_{39} - h_{37}) = w_{38}(h_{40} - h_{38})$$

$$w_{39}(1204.0 - 151.3) = w_{38}(202.4 - 141.3)$$

$$w_{38} = w_1 - w_{44} - w_{43} = 0.68727\ w_1$$

$$w_{39} = 0.03988\ w_1 = w_{37} = w_{19}$$

At Heater No. 7:

$$w_{36}(h_{36} - h_{35}) + w_{37}(h_{37} - h_{35}) = w_{33}(h_{38} - h_{33})$$

$$w_{36}(1136.3 - 116.5) + 0.03988\ w_1(151.3 - 116.5) = 0.68727\ w_1(141.3 - 106.5$$

$$w_{36} = 0.02210\ w_1$$

At Heater No. 8:

$$w_{32}(h_{32} - h_{29}) + w_{35}(h_{35} - h_{29}) = w_{33}(h_{33} - h_{28})$$

$$w_{35} = w_{36} + w_{37} = 0.06198\ w_1$$

$$w_{32} = 0.02845\ w_1$$

$$w_{34} = w_{32} + w_{35} = 0.09043\ w_1$$

Other Flow Rates:

$$w_{20} = w_{36} - w_{68} = 0.02210\ w_1 - 0.00244\ w_1 = 0.01966\ w_1$$

$$w_4 = w_1 - w_{64} - w_3 - w_2 = 0.99879\ w_1$$

$$w_8 = w_4 - w_5 - w_{66} - w_{70} - w_6 - w_7 - w_{15} - w_{69} = 0.77226\ w_1$$

Other Flow Rates, cont'd.

$w_9 = w_8 + w_5 = 0.77762\ w_1$

$w_{10} = w_{11} = w_9 - w_{55} = 0.71785\ w_1$

$w_{12} = w_{11} + w_2 = 0.71876\ w_1$

$w_{13} = w_{18} = w_{12} - w_{14} - w_{16} = 0.68410\ w_1$

$w_{19} = w_{39} = 0.03988\ w_1$

$w_{21} = w_{32} = 0.02845\ w_1$

$w_{23} = 0.000664\ w_1$

$w_{22} = w_{18} - w_{19} - w_{20} - w_{21} - w_{23} + w_{17}) = 0.59618\ w_1$

$w_{28} = w_{31} = w_{33} = w_{27} = w_{40} = 0.68727\ w_1$

$w_{25} = w_{26} = w_{28} - w_{23} = 0.68661\ w_1$

$w_{22} = w_{26} - w_{29} = 0.59618\ w_1$ (checks with above)

$w_{29} = w_{30} = w_{34} = 0.09043\ w_1$

$w_{40} = w_{13} + w_{67} + w_{70} + w_{15} + w_{16} + w_3 = 0.68727\ w_1$ (checks with above)

Turbine Work:

	Btu/lb w_1
High-pressure turbine:	
$w_4(h_4 - h_5) = 0.99879\ w_1(1424.0 - 1400.0) =$	23.971
$(w_4 - w_5 - w_{66} - w_{70})(h_5 - h_6) = 0.99125\ w_1(1400.0 - 1333.0) =$	66.414
$(0.99125\ w_1 - w_6)(h_6 - h_7) = 0.99125\ w_1(1333.0 - 1314.0) =$	18.834
$(0.99125\ w_1 - w_7)(h_7 - h_8) = 0.77429\ w_1(1314.0 - 1255.6) =$	45.218
Intermediate-pressure turbine:	
$w_{12}(h_{12} - h_{13}) = 0.71876\ w_1(1518.5 - 1378.1) =$	100.914
Total for unit	255.351

Turbine Work, cont'd. Btu/lb w_1

Low-pressure turbine:

$w_{18}(h_{18} - h_{19}) = 0.68410\ w_1(1378.1 - 1204.0) =$ 119.03

$(0.68410\ w_1 - w_{19})(h_{19} - h_{20}) = 0.64422\ w_1(1204.0 - 1120.0) =$ 54.114

$(0.64422\ w_1 - w_{20})(h_{20} - h_{21}) = 0.62456\ w_1(1120.0 - 1071.0) =$ 30.603

$(0.62456\ w_1 - w_{21})(h_{21} - h_{22}) = 0.59611\ w_1(1071.0 - 1000.3) =$ 42.145

Total for low-pressure turbine 245.892

Total for both turbines 501.243

The internal efficiency is taken to be the same as that in the Bull Run unit, or 98.527%.* The gross output of the turbine generator unit per lb of steam leaving the steam generator is then:

Gross output/ lb w_1 = 501.243 x 0.98527 = 493.860 Btu/lb w_1

Reheat Preheater, Feedwater Tempering and Pressure-Booster Pump:

The feedwater leaving Heater No. 1 at 550.9°F and about 3500 psia is mixed with the 865.7°F/3500 psia heating steam from the reheat steam preheater to raise its temperature to about 695°F. An electric-motor driven pressure-booster pump then raises the feedwater pressure to about 3800 psia.** The reheat steam leaving the high-pressure turbine exhaust at about 551.7°F and ~ 600 psia is heated by 1000°F/3515 psia throttle steam in a shell-and-tube heat exchanger to 650°F before it enters the reheaters. The arrangement is shown on the flowsheet, Figure 1, and in the simplified diagram, Figure 3.

At the pressure booster pumps the entering conditions are 695°F, 3475 psia and 766.4 Btu/lb and the discharge pressure is 3800 psia. From Fig. 4 Keenan and Keyes(1), the isentropic Δh is 2.1 Btu/lb. Assuming an

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* From the TVA Bull Run plant data, Dwg. No. 47K1110-2:

$$\eta = \frac{913,832}{913,832 + 13,660} = 0.98527$$

** The pressure-booster pumps could be steam turbine driven to effect a small gain in plant efficiency. This refinement was not considered necessary for the MSBR study.

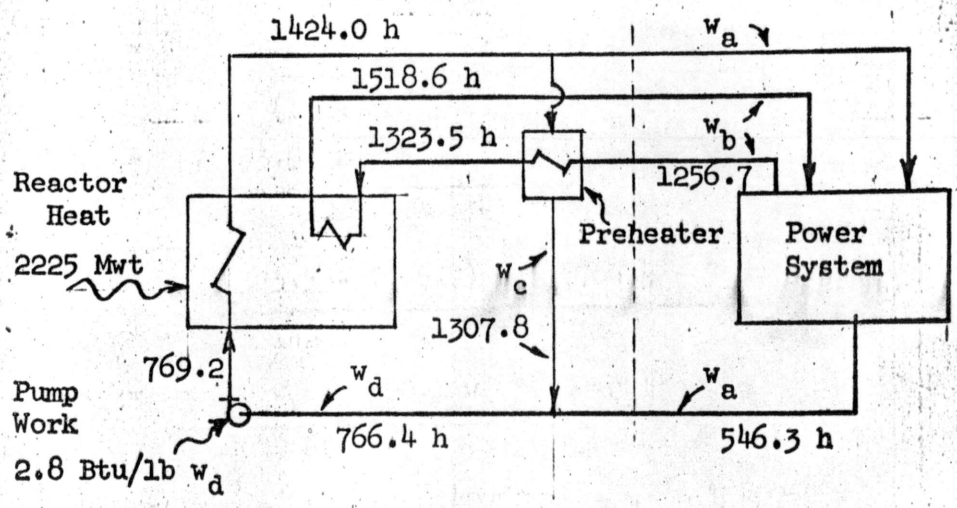

Figure 4 . . Simplified Flow Diagram of Reheat Steam and Feedwater Tempering Arrangement.

efficiency of 75%, the actual Δh through the pump is 2.80 Btu/lb. (If the electric drive motor is 90% efficient, 3.11 Btu/lb is chargeable against the gross generated electrical output of the plant, as will be covered subsequently). Thus, 2.80 Btu/lb of fluid pumped is returned to the cycle and results in a temperature rise of about 5°F and a feedwater temperature to the boiler-superheater of 700°F.

Referring to the simplified diagram, Figure 4 the enthalpy of the heated steam leaving the reheat steam preheat is 1323.5 Btu/lb at the assumed 650°F, 570 psia conditions. The enthalpy entering the boiler-superheater is 769.2 Btu/lb at 700°F, 3800 psia, and subtracting the pump work gives 766.4 Btu/lb after the mixing point. (Only about 2% of the pump input work is lost and this correction was not made).

The state points in Figure 3 and Figure 4 are related, as follows:

$$w_a = w_1$$

$$w_b = w_{10} = 0.71785\, w_1$$

$$w_a = \frac{(2225 \times 1000 \times 3413) + 2.8\, w_d}{(1424.0 - 546.3) + w_b(1518.6 - 1256.7)}$$

$$w_a = \frac{7{,}593.93 \times 10^6 + 2.8\, w_d}{1065.704}$$

$$w_c = \frac{w_b(1323.5 - 1256.7) + w_a(766.4 - 546.3)}{1424.0 - 766.4}$$

$$w_c = \frac{66.80\, w_b + 220.1\, w_a}{657.60} = 0.40762\, w_a$$

$$w_d = w_c + w_a = 0.40762\, w_a + w_a = 1.40762\, w_a$$

$$w_a = \frac{(7{,}593.93 \times 10^6) + 2.8(1.40762\, w_a)}{1065.704}$$

w_a = 7,152,180 lb/hr

w_c = 2,915,372 lb/hr

w_b = 5,134,192 lb/hr

w_d = 10,067,552 lb/hr

Boiler Feedwater Pumps:

Power required = $w_a (h_{50} - h_{49})$

$$= \frac{7{,}152{,}180\ (343.3 - 329.5)}{3413 \times 1000 \times 0.986} = 29.328 \text{ Mw total for two}$$

Capacity, each of two = $\frac{7{,}152{,}180 \times 0.01808 \times 7.48}{2 \times 60}$ = 8,060 gpm

Approximate head = (3800 - 200) x 144 x 0.01808 = 9,380 ft

Feedwater Pressure-Booster Pumps:

Power required = $\frac{10{,}067{,}552 \times 3.11}{3413 \times 1000}$ = 9.174 Mwe total for two

Capacity, each of two = $\frac{10{,}067{,}552 \times 0.03020 \times 7.48}{2 \times 60}$ = 18,952 gpm

Approximate head = (3800 - 3475) x 144 x 0.03020 = 1,413 ft

Gross Output Each Generator:

$$\text{HP-IP Turbine-Generator} = \frac{255.351 \times 7,152,1\,0 \times 0.98527}{3413 \times 1000} = 527.223 \text{ Mwe}$$

$$\text{HP-IP Turbine-Generator Internal Losses} = \frac{535.105 \times 0.015}{0.985} = 8,029 \text{ kw}$$

$$\text{LP Turbine-Generator Output} = \frac{245.892 \times 7,152,1\,0 \times 0.98527}{3413 \times 1000} = 507.693 \text{ Mwe}$$

$$\text{LP Turbone-Generator Internal Losses} = \frac{507,693 \times 0.015}{0.985} = 7,731 \text{ kw}$$

Gross Efficiency and Output:

Taking credit for the boiler feedwater pump turbine output,

$$\text{Gross electrical output} = \frac{493.860 \times 7,152,1\,0}{3413 \times 1000} = 1034.91 \text{ Mwe}$$

$$\eta_{gross} = \frac{1034.91 + 29.328}{2225} = 47.8\ \%$$

Net Efficiency and Output:

The assumed auxiliary electric load for the MSBR plant is 25.7 Mwe.

$$\text{Net electrical output} = 1034.91 - 9.17 - 25.7 = 1000 \text{ Mwe}$$

$$\eta_{net} = \frac{1000}{2225} = 44.9\%$$

$$\text{Net heat rate} = 3413/0.449 = 7601 \text{ Btu/kwh}$$

Boiler-Superheater:

The heat transferred from the coolant-salt in the boiler-superheater is:

$$Q = \frac{10,067,552\ (1424.0 - 769.2)}{3413 \times 1000} = 1931.51 \text{ Mwt}$$

The required coolant-salt flow is: *

$$W_{cs} = \frac{1931.51 \times 3413 \times 1000}{0.41(1125 - 850)} = 58,467,698 \text{ lb/hr}$$

$$= \frac{58,467,698}{3600 \times 125} = 129.93 \text{ ft}^3/\text{sec}$$

$$= \frac{58,467,698 \times 7.4805}{60 \times 125} = 58,316 \text{ gpm}$$

* Based on assumed coolant-salt properties of $c_p = 0.41$ Btu/lb-°F, and an average density of 125 lbs/ft^3.

Reheater:

The heat transferred in the reheater is:

$$Q = \frac{5,134,192\ (1518.6 - 1323.5)}{3413 \times 1000} = 293.49 \text{ Mwt}$$

The required coolant-salt flow is:

$$w_{cs} = \frac{293.49 \times 3413 \times 1000}{0.41(1125 - 850)} = 8,884,087 \text{ lb/hr}$$

$$= \frac{8,884,087}{3600 \times 125} = 19.742 \text{ ft}^3/\text{sec}$$

$$= \frac{8,884,087 \times 7.4805}{60 \times 125} = 8,861 \text{ gpm}$$

Preheater:

The heat transferred in the preheater is:

$$Q = \frac{5,134,192\ (1323.5 - 1256.7)}{3413 \times 1000} = 100.49 \text{ Mwt}$$

Pipe Sizes:

The MSBR steam piping generally follows the practices used in the Bull Run plant. The 1000°F main steam and hot reheat lines use 2-1/4% Cr - 1% Mo Croloy steel, the cold reheat line uses A155 steel, and the feedwater line (after the deaerator) A106 steel. The allowable stresses at the average line temperatures, the assumed velocities, etc., are shown in Table III.

The flow rates are so large that in most cases the streams must be divided into parallel lines. Since the number of parallel paths required for a particular stream is not the same throughout the length of the run, representative points were selected and the number of parallel pipes at these particular places in the lines are those shown in Table III.

Table III

MSBR Steam-Power System Pipe Sizes

	Steam Leaving Boiler-Superheater	Cold Reheat to Pre-Heater	Cold Reheat to Re-Heater	Hot Reheat Leaving Reheater	Feedwater to Boiler-Superheater	Heating Steam to Preheater	Heating Steam from Preheater
Flow rate, 10^6 lb/hr	10.086	5.069	5.069	5.069	10.086	2.915	2.915
Pressure, psia	3600	600	570	540	3800	3600	3500
Temperature, °F	1000	552	650	1000	700	1000	866
Specific volume, ft^3/lb	0.1988	0.8783	1.0699	1.5707	0.0293	0.1988	0.1643
Total vol. flow, 10^3 cfm	33.445	78.894	90.0	132.148	4.929	9.618	7.949
Assumed velocity, 10^3 ft/min	10-12	5.8-7.4	5.8-7.4	15.4	15.4	1.1	1.1
Total flow area, $in.^2$	481.5	177.3	1756.0	1235.7	657.2	138.5	114.5
Number pipes	8	2	8	8	8	2	2
Estimated ID, in.	8.76	33.6	16.7	14.0	10.2	9.4	8.52
Pipe material	A335 Gr P-22	A155 Gr KC-70	A155 Gr KC-70	A335 Gr P-22	A106 Gr C	A335 Gr P-22	A335 Gr P-22
Allowable S, lb/in^2 at °F	7,800	15,750	15,750	7,800	16,600	7,800	7,800
Calc. wall thickness, in.	2.62	0.675	0.37	0.49	1.259	2.09	2.04
Nominal pipe size, OD	14	35	18	16	12	12	12
thick., in.	3	0.6875	0.5	0.5	1.312	2	2
Actual velocity, 10^3 ft/min	11.9	6.0	5.7	13.5	1.12	11.5	9.52

The following estimate of the nominal sizes of the cold reheat lines between the preheater and the reheater is illustrative of the method used:

Steam conditions: 570 psia, 650°F, 1.0699 ft^3/lb

5,069,600 lb/hr

Assumed velocity, say, 7,400 ft/min.

Flow area required:

$$A = \frac{5,134,192 \times 1.0699}{60 \times 7,400} = 12.2 \text{ ft}^2$$

If the flow is divided into eight pipes, the ID of each is:

$$ID = \sqrt{\frac{12.2 \times 144 \times 4}{8 \times \pi}} = 16.7 \text{ in.}$$

(OD = about 18 in.)

Assuming the allowable stress for A155-Grade KC-70 steel at 650°F to be:15,750 psi[3]:

$$\text{Wall thickness} = \frac{PD}{2S + 2yP} = \frac{650 \times 18}{2 \times 15,750 + 2 \times 0.4 \times 650} = 0.37 \text{ in.}$$

The terms in the above equation are as explained in ref. 3, p. 13.

On the basis of this estimate, 18-in. OD piping is assumed and the wall thickness is taken as 1/2 in. The actual steam velocity in this case becomes:

$$V = \frac{5,069,600 \times 1.0699}{17^2 \times \pi/4 \times 60} = 5,700 \text{ ft/min}$$

Coolant Salt Temperatures

At Fuel-Salt Heat Exchanger:

$Q = 0.95 \times 2225 = 2113.75$ Mwt

This is based on arbitrary assumption that 95% of the reactor heat is dissipated in the fuel-salt heat exchanger and 5% in the blanket-salt exchanger. Using the salt properties shown in the reference report:[1]

$$t_{out} - t_{in} = \frac{2113.75 \times 3413 \times 1000}{67.352 \times 10^6 \times 0.41} = 261.25°F$$

$$t_{out} = 850 + 261.25 = 1111.25°F$$

At Blanket-Salt Heat Exchanger:

$Q = 0.05$ a $2225 = 111.25$ Mwt

$$t_{out} - t_{in} = \frac{111.25 \times 3413 \times 1000}{67.352 \times 10^6 \times 0.41} = 13.75°F$$

$$t_{out} = 1111.25 + 13.75 = 1125°F$$

Fuel-Salt Circulation Rate

$Q = 2113.75$ Mwt (See above) note

$$w = \frac{2113.75 \times 3413 \times 1000}{0.55 \times (1300 - 1000)} = 43.723 \times 10^6 \text{ lb/hr}$$

$$= \frac{43.723 \times 10^6}{3600 \times 127} = 95.673 \text{ ft}^3/\text{sec}$$

$$= \frac{95.673 \times 60 \times 7.48}{4} = 10{,}734.5 \text{ gpm capacity each of 4 pumps.}$$

Blanket-Salt Circulation Rate

$Q = 111.25$ Mwt (See above)

$$w = \frac{111.25 \times 3413 \times 1000}{0.22 \times (1250 - 1100)} = 17.26 \times 10^6 \text{ lb/hr}$$

$$= \frac{17.26 \times 10^6}{3600 \times 277} = 17.31 \text{ ft}^3/\text{sec}$$

$$= \frac{17.31 \times 60 \times 7.48}{4} = 194.18 \text{ gpm capacity each of 4 pumps.}$$

CASE B - FEEDWATER AT 580°F, REHEAT STEAM AT 550°F.

The Bull Run cycle is easily adapted to supplying feedwater at 580°F and the reheat steam, normally at ~ 550°F in the Bull Run cycle, needs no tempering for Case B in the MSBR study. The feedwater heating is accomplished by the addition of a ninth stage of feedwater heating, using heating steam from an additional extraction point on the high-pressure turbine. The condensate drain would be cascaded back through the next three heaters to the deaerating heater.

The new ninth, or "top", heater has been given the rather unconventional designation of "Heater No. 0" in order that the other eight heaters will bear the same numbers as those in Case A and in the Bull Run heat balances. This ninth heater has been assumed to have a "zero approach" in that the heated feedwater leaving it is at 580°F, essentially the same temperature as the condensing steam on the shell side.* The TVA has studied the economics of using this zero approach for the last, or top, stage of feedwater heating, a study further complicated by the fact that this heater may be cut out to obtain peak turbine output, and despite the higher capital cost for the extra surface required, have employed a zero approach unit for Heater No. 1 in the Bull Run plant. In the MSBR study, the zero approach was retained for the No. 1 heater and also used for the No. 0 unit as well. An economic analysis to justify this assumption was not within the scope of this initial MSBR study.

As in Case A, the Case B study was made on the basis of the simplified flowsheet shown in Figure 3 , and was analyzed on the basis of 1 lb of steam leaving the boiler-superheater, with all other flow rates expressed as proportional to this w_1. The conditions at each state point are summarized in Table III, following.

At Heater No. 0

$$w_1(h_{62} - h_{59}) = w_6(h_{61} - h_{60})^{**}$$

$$w_1(583.6 - 546.3) = w_6(1333.0 - 563.2)$$

$$w_6 = 0.04845\, w_1 = w_{60} = w_{61}$$

* Heaters Nos. 2 through 8 allow a 5% Δp in the extraction steam line between the shell side of the heater and the turbine stage, a 10°F Δt between the condensate drain and the incoming water, and a 5°F approach of the heated feedwater to the saturation temperature on the shell side.

** The value of h_{61} was determined by assuming a 5% Δp and by using the expansion curve for the Bull Run turbine.

Table IV

Properties and Flow Rates at State Points in Power Cycle Case B

State Point	psia	°F	Btu/lb	Flow Rates lbs/lb w_1	Flow Rates lb/hr
1	3600	1000	1424.0	1.00000	7,460,457
2	3515	1000	1424.0	0.000906	6,789
3	3515	1000	1424.0	0.000151	1,119
4	3500	1000	1424.0	0.9988	7,451,504
5	-	-	1400.0	0.00536	39,988
6	1396	-	1333.0	0.04845	361,459
7	1146	698.7	1314.0	0.21165	1,579,006
8	600	-	1255.6	0.72913	5,439,643
9	~ 600	551.7	1256.7	0.7345	5,479,706
10	~ 600	-	1256.7	0.6777	5,055,952
11	-	1000	1518.6	0.6777	5,055,952
12	540	1000	1518.5	0.67868	5,063,263
13	~ 172	705.9	1378.1	0.6523	4,866,456
14	-	-	1378.1	0.02545	189,869
15	-	-	1256.7	0.001328	9,922
16	-	-	1378.1	0.000966	7,237
17	-	-	-	0.000725	5,446
18	~ 172	705.9	1378.1	0.6523	4,866,456
19	26.7	328.5	1204.0	0.03804	283,796
20	7.8	182.1	1120.0	0.01862	138,914
21	3.44	146.8	1071.0	0.027131	202,402
22	1.5" Hg	-	1000.3	0.56854	4,241,568
23	-	-	1355.9	0.000664	4,923
24	1.5" Hg	91.7	59.7	0.56854	4,241,568
25	-	-	-	0.6547	4,884,361
26	-	-	59.7	0.6547	4,884,361
27	-	-	61.0	0.6554	4,889,583
28	-	-	61.0	0.6554	4,889,583
29	-	103.0	71.0	0.08623	643,315
30	-	143.6	111.5	0.08623	643,315
31	-	98.3	66.3	0.6554	4,889,583
32	3.17	143.6	1071.0	0.027131	202,402

Table IV Cont'd - Case B

State Point	psia	°F	Btu/lb	Flow Rates lbs/lb W_1	lb/hr
33	-	138.6	106.5	0.65538	4,889,434
34	-	-	111.5	0.08623	643,315
35	-	148.6	116.5	0.05910	440,913
36	7.24	178.4	1136.3	0.02106	157,117
37	-	183.4	151.3	0.03804	283,796
38	-	173.4	141.3	0.65538	4,889,434
39	24.6	239.1	1204.0	0.03804	283,796
40	-	234.1	202.4	0.65538	4,889,434
41	71.7	304.5	274.2	1.00000	7,460,457
42	-	304.5	274.2	1.00000	7,460,457
43	-	314.5	284.6	0.29389	2,192,554
44	77.1	309.5	1141.4	0.05073	378,469
45	-	-	1295.0	0.03836	286,183
46	170	368.4	1189.0	0.01060	79,081
47	1146	698.7	1314.0	0.11565	862,802
48	362	488.8	1242.7	0.05433	405,327
49	-	357.5	329.5	1.00000	7,460,457
50	~ 3800	366.1	343.3	1.00000	7,460,457
51	337	428.1	1242.7	0.05433	405,327
52	-	376.1	349.3	0.25553	1,906,371
53	~ 3775	423.1	404.3	1.00000	7,460,457
54	-	433.1	411.2	0.20120	1,501,044
55	570	480.7	1256.7	0.05672	423,157
56	~ 3750	475.7	460.9	1.00000	7,460,457
57	-	485.7	471.0	0.14450	1,078,036
58	1053	550.9	1314.0	0.0960	716,204
59	~ 3700	550.9	546.3	1.00000	7,460,457
60	1325.8	560.9	563.2	0.04845	361,459
61	1325.8	580.0	1333.0	0.04845	361,459
62	3700	580.0	583.6	1.00000	7,460,457
63	3700	580.0	583.6	1.00000	7,460,457
64	3515	1000.0	1424.0	0.000151	1,119
66	-	-	1400.0	0.00161	12,011
67	-	-	1141.4	0.000151	1,119
68	-	-	1358.3	0.00244	18,204
69	-	-	1256.7	0.000695	5,222
70	-	-	1400.0	0.000574	4,252

At Heater No. 1

$$w_{58}(h_{58} - h_{57}) + w_6(h_{60} - h_{57}) = w_1(h_{59} - h_{56})$$

$$w_{58}(1314.0 - 471.0) + 0.0845\ w_1\ (563.2 - 471.0) = w_1(546.3 - 460.9)$$

$$w_{58} = 0.0960\ w_1$$

$$w_{57} = w_{58} + w_{60} = 0.0960\ w_1 + 0.04845\ w_1 = 0.1445\ w_1$$

At Heater No. 2

$$w_{55}(h_{55} - h_{54}) + w_{57}\ (h_{57} - h_{54}) = w_1(h_{56} - h_{53})$$

$$w_{55}(1256.7 - 411.2) + w_{57}(471.0 - 411.2) = w_1(460.9 - 404.3)$$

using $w_{57} = 0.1445\ w_1$, above

$$w_{55} = 0.05672\ w_1$$

At Heater No. 3

$$w_{51}(h_{51} - h_{52}) + w_{54}(h_{54} - h_{52}) = w_1(h_{53} - h_{50})$$

$$w_{54} = w_{55} + w_{57} = 0.05672\ w_1 + 0.1445\ w_1 = 0.2012\ w_1$$

$$w_{51}(1242.7 - 349.3) + 0.2012\ w_1(411.2 - 349.3) = w_1(404.3 - 343.3)$$

$$w_{48} = w_{51} = 0.05433\ w_1$$

At Heater No. 4

$$w_{45}(h_{45} - h_{43}) + w_{52}(h_{52} - h_{43}) = w_1(h_{49} - h_{42})$$

$$w_{52} = w_{51} + w_{54} = 0.05433\ w_1 + 0.2012\ w_1 = 0.2555\ w_1$$

$$w_{45}(1295.0 - 284.6) + w_{52}(349.3 - 284.6) = w_1(329.5 - 274.2)$$

$$w_{45} = 0.03836\ w_1$$

Heater No. 5 (Deaerator)

$$w_{44}(h_{44}) + w_{40}(h_{40}) + w_{43}(h_{43}) = w_1(h_{41})$$

$$w_{43} = w_{45} + w_{52} = 0.03836\ w_1 + 0.2555\ w_1 = 0.2939\ w_1$$

$$w_{40} = w_1 - w_{43} - w_{44} = 0.7061\ w_1 - w_{44}$$

Heater No. 5, Cont'd.

$$w_{44}(1141.4) + 202.4(0.7061\, w_1 - w_{44}) + 0.2939\, w_1(284.6) = 274.2\, w_1$$

$$w_{44} = 0.05073\, w_1$$

$$w_{40} = 0.7061\, w_1 - 0.05073\, w_1 = 0.65538\, w_1$$

Boiler Feed Pump

$$w_{47}(h_{47} - h_{48}) + (w_{47} - w_{48})(h_{48} - h_{46}) + (w_{47} - w_{48} - w_{46})(h_{46} - h_{44}) + w_{64}(h_{64} - h_{67}) = \frac{w_1}{\eta_{BR}}(h_{50} - h_{49})$$

where η_{BR} is the proportion of the pump input work that goes into the pumped fluid as a enthalpy increase. This is assumed to be the same as the pump performance in the Bull Run cycle, and is: (Data is from Figure _____)

$$805{,}000(1314.0 - 1242.7) + (805{,}000 - 351{,}200)(1242.7 - 1189.0) + (805{,}000 - 351{,}200 - 112{,}100)(1189.0 - 1141.4) = \frac{6{,}335{,}200(344.8 - 329.5)}{\eta_{BR}}$$

$$\eta_{BR} = 0.986$$

then:

$$w_{47}(1314.0 - 1242.7) + (w_{47} - w_{48})(1242.7 - 1189.0) + (w_{47} - w_{48} - w_{46})(1189.0 - 1141.4) + w_{64}(1424.0 - 1141.4) = \frac{w_1\,(343.3 - 329.5)}{0.986}$$

this reduces to:

$$172.6\, w_{47} - 101.3\, w_{48} + 47.6\, w_{46} = 13.954\, w_1 .$$

But, $w_{48} = w_{51} = 0.05433\, w_1$,

and, $w_{47} = w_{44} + w_{48} + w_{46} = 0.05073 + 0.05433\, w_1 + w_{46}$

$$w_{47} = 0.10506\, w_1 + w_{46}.$$

Thus, $172.6(0.10506\, w_1 + w_{46}) - 101.3(0.05433\, w_1) + 47.6\, w_{46} = 13.954\, w_1$

$$w_{46} = 0.01060\, w_1$$

$$w_{47} = 0.10506\, w_1 + 0.01059\, w_1 = 0.11565\, w_1.$$

Also: $w_7 = w_{58} + w_{47} = 0.0960\, w_1 + 0.11565\, w_1 = 0.21165\, w_1$

$$w_{14} = w_{45} - w_{46} - w_{66} - w_{69}$$

The bleed from the steam seals, w_{66}, w_{69}, etc., is taken as a

direct proportion of that in the Bull Run cycle. This ratio was determined from a preliminary rough calculation (not shown) of Case B and the throttle flow rate for the MSBR was found to be 1.118 times that in the Bull Run system.

The steam seal and bleed steam flows used in the simplified flow sheet of Figure 3 approximate the effect of this steam on the cycle performance but are not necessarily representative of the actual flow paths. The correct arrangement is shown in the flow diagram for Case B, Figure 2.

Thus: $w_{14} = (0.03846 - 0.01060 - 0.00161 - 0.000695)\, w_1 = 0.02545\, w_1$

Heater No. 6

$$w_{39}(h_{39} - h_{37}) = w_{38}(h_{40} - h_{38})$$

$$w_{39}(1204.0 - 151.3) = w_{38}(202.4 - 141.3)$$

$$w_{39} = 0.05804\, w_{38}$$

But, $w_{38} = w_{40} = w_1 - w_{44} - w_{43} = (1.0000 - 0.05073 - 0.2939)\, w_1$

$$w_{38} = 0.6554\, w_1,$$

and, $w_{39} = 0.05804(0.6554)\, w_1 = 0.03804\, w_1$

$$w_{19} = w_{37} = w_{39}$$

Heater No. 7

$$w_{36}(h_{36} - h_{35}) + w_{37}(h_{37} - h_{35}) = w_{33}(h_{38} - h_{33}).$$

Using, $w_{37} = 0.03804\, w_1$ (above)

$w_{33} = w_{40} = 0.6554\, w_1$

$$w_{36}(1136.3 - 116.5) + 0.03804\, w_1(151.3 - 116.5) = 0.6554\, w_1(141.3 - 106.5)$$

$$w_{36} = 0.02106\, w_1$$

Heater No. 8

$$w_{32}(h_{32} - h_{29}) + w_{35}(h_{35} - h_{29}) = w_{33}(h_{33} - h_{28})$$

$$w_{35} = w_{36} + w_{37} = 0.02106\, w_1 + 0.03804\, w_1 = 0.05910\, w_1$$

$$w_{33} = w_{40} = 0.6554\, w_1$$

Heater No. 8, Cont'd.

$$w_{32}(1071.0 - 71.0) + 0.0591\ w_1(116.5 - 71.0) = 0.6554\ w_1(106.5 - 61.0)$$

$$w_{32} = 0.02713\ w_1$$

$$w_{21} = w_{32}$$

$$w_{30} = w_{29} = w_{34} = w_{32} + w_{35} = 0.02713\ w_1 + 0.0591\ w_1 = 0.08623\ w_1$$

Turbine Shaft Seals

$$w_{68} = w_3 + w_{15} + w_{16} + w_{67} + w_{70} + w_{17}$$

$$w_{68} = (0.000151 + 0.001328 + 0.000966 + 0.000151 + 0.000574 - 0.000725)w_1$$

$$w_{68} = 0.00244$$

This flow does not check exactly because of the simplifying assumptions made in the analysis. The effect of this discrepancy on the calculated performance is insignificant.

Other Flow Rates

$$w_{20} = w_{36} - w_{68} = 0.02106\ w_1 - 0.00244\ w_1 = 0.01862\ w_1$$

$$w_4 = w_1 - w_{64} - w_3 - w_2 = (1.000 - 0.000151 - 0.000151 - 0.000906)w_1$$

$$w_4 = 0.99879\ w_1$$

$$w_8 = w_4 - w_5 - w_{66} - w_{70} - w_6 - w_7 - w_{15} - w_{69}$$

$$w_8 = (0.99879 - 0.00536 - 0.00161 - 0.000574 - 0.04845 - 0.21165 - 0.001328 - 0.000695)\ w_1 = 0.7291\ w_1$$

$$w_9 = w_8 + w_5 = 0.7291\ w_1 + 0.00536\ w_1 = 0.7345\ w_1$$

$$w_{10} = w_9 - w_{55} = 0.7345\ w_1 + 0.05672\ w_1 = 0.6778\ w_1$$

$$w_{11} = w_{10}$$

$$w_{12} = w_{11} + w_2 = 0.6778\ w_1 + 0.000906\ w_1 = 0.6787\ w_1$$

$$w_{13} = w_{12} - w_{14} - w_{16} = 0.6787\ w_1 - 0.02545\ w_1 - 0.000966\ w_1$$

$$w_{13} = 0.6523\ w_1$$

$$w_{18} = w_{13}$$

$$w_{22} = w_{18} - w_{19} - w_{20} - w_{21} + w_{17} - w_{23}$$

$$w_{22} = (0.6523 - 0.03804 - 0.01862 - 0.02713 + 0.000725 - 0.000664)w_1$$

$$w_{22} = 0.56854\ w_1$$

$$w_{33} = w_{31} = w_{28} = 0.6554\ w_1$$

$$w_{26} = w_{28} - w_{23} = 0.6554\ w_1 - 0.000664\ w_1 = 0.6547\ w_1$$

$$w_{25} = w_{26}$$

$$w_{22} = w_{26} - w_{29} = 0.6547\ w_1 - 0.08623\ w_1 = 0.56854\ w_1$$

This value checks favorably with the flow rate of $0.56854\ w_1$ obtained for w_{22} above.

$$w_{40} = w_{13} + w_{67} + w_{70} + w_{15} + w_{16} + w_3$$

$$w_{40} = 0.6523 + 0.000151 + 0.000574 + 0.001328 + 0.000966 + 0.000151)\ w_1$$

$$w_{40} = 0.6554\ w_1$$

This value of w_{40} checks with the previously-obtained flow rate of 0.6554, see Heater No. 8.

Turbine Work

	Btu/lb w_1
High-Pressure:	
$w_4(h_4 - h_5) = 0.9988\ w_1(1424.0 - 1400.0) =$	23.971
$(w_4 - w_5 - w_{66} - w_{70})(h_5 - h_6) = 0.99126\ w_1(1400.0 - 1333.0)$	66.414
$(w_4 - w_5 - w_{66} - w_{70} - w_6)(h_6 - h_7) =$ $0.9428\ w_1\ (1333.0 - 1314.0) =$	17.913
$(w_4 - w_5 - w_{66} - w_{70} - w_6 - w_7)(h_7 - h_8) =$ $0.7312\ w_1\ (1314.0 - 1255.6) =$	42.700
Intermediate Pressure:	
$w_{12}\ (h_{12} - h_{13}) = 0.6787\ w_1(1518.5 - 1378.1) =$	95.287
Total for hp-ip turbine	246.285

Turbine Work, Contd.

	Btu/lb w_1
Low-Pressure:	
$w_{18}(h_{18} - h_{19}) = 0.6523\ w_1(1378.1 - 1204.0) =$	113.500
$(w_{18} - w_{19})(h_{19} - h_{20}) = 0.6143\ w_1(1204.0 - 1120.0) =$	51.601
$(w_{18} - w_{19} - w_{20})(h_{20} - h_{21}) =$ $0.5956\ w_1\ (1120.0 - 1071.0) =$	29.184
$(w_{18} - w_{19} - w_{20} - w_{21})(h_{21} - h_{22}) =$ $0.5685\ w_1\ (1071.0 - 1000.3) =$	40.193
Total for Turbine B	234.478

Total internal work of turbine-generator unit:

$$246.285 + 234.478 = 480.763 \text{ Btu/lb } w_1$$

Gross output of turbine-generator unit:

$$480.763 \text{ (internal eff.)} = 480.763 \times 0.98527 = 473.68 \text{ Btu/lb } w_1$$

The above value of internal efficiency is the same as that in the Bull Run unit, or

$$\eta = \frac{913,832}{913,832 + 13,660} = 0.98527$$

Heat Input to Cycle

$$Q = w_1(h_1 - h_{62}) + w_{10}(h_{11} - h_{10})$$

$$= 1.000\ (1424.0 - 583.6) + 0.6777\ w_1\ (1518.6 - 1256.7) = 1017.89 \text{ Btu/lb } w_1$$

Efficiencies and Outputs

$$\text{Gross generated output} = \frac{473.681 \times 2225}{1017.89} = 1035.42 \text{ Mwe}$$

Net output = 1035.42 - 25.7 = 1009.7 Mwe

This assumes that the auxiliary load is the same as in Case A, or 25.7 Mwe.

Net Efficiency: $\eta = 1009.7/2225 = 45.38\ \%$

Flow Rate at Boiler Outlet

$$w_1 = \frac{2225 \times 3413 \times 1000}{1017.89} = 7,460,457 \text{ lb/hr}$$

Other Flow Rates

All other mass flow rates shown in Table IV and on Figure 2 were determined by taking the flow at the point, expressed in terms of w_1, times the above value of w_1, or 7,460,457 lb/hr.

Division of Work Between Turbine A and Turbine B

HP & IP Shaft = $\frac{246.29 \times 0.98527 \times 7,460,457}{3413 \times 1000}$ = 530.423 Mwe gross

LP Shaft = $\frac{234.976 \times 0.98527 \times \quad}{3413 \times 1000}$ = 504,994 Mwe gross

1035.42 Mwe gross

Heat Transferred in Boiler-Superheater and Reheater

Boiler-Superheater: $Q = \frac{7,460,457\ (1424.0 - 583.6)}{3413 \times 1000}$ = 1837.0 Mwt

Reheater: $Q = \frac{5,055,952\ (1518.6 - 1256.7)}{3413 \times 1000}$ = 388.0 Mwt

Coolant-Salt Flow

Boiler-Superheater: $w = \frac{7,460,457\ (1424.0 - 583.6)}{0.41\ (1125 - 850)}$ = 55,607,699 lb/hr

$= \frac{55,607,699}{3600 \times 125}$ = 123.57 ft^3/sec

$= \frac{55,607,699 \times 7.4085}{60 \times 125}$ = 55,463 gpm

Reheater: $w = \frac{5,055,952\ (1518.6 - 1256.7)}{0.41(1125 - 850)}$ = 11,744,158 lb/hr

= 26.1 ft^3/sec

= 11,714 gpm

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INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

July 12, 1966

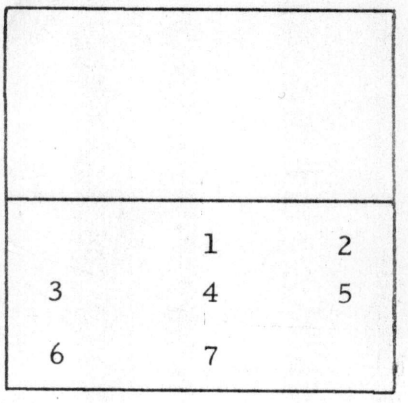
MSR-66-19

To: R. E. Thoma

From: H. A. Friedman

Subject: Segregation of $LiF-BeF_2-UF_4$ Fuel Salt on Freezing in Drain Tanks

An experiment was conducted for the purpose of determining whether fuel salt mixtures of the type proposed for the Molten Salt Breeder Reactor would undergo serious segregation of uranium-bearing phases under very slow static cooling. The results of a previous experiment[1] with the MSRE fuel mixture, $LiF-BeF_2-ZrF_4-UF_4$ (65.0-29.1-5.0-0.9 mole %), indicated that very little segregation takes place in this four-component fuel mixture. The salt, $LiF-BeF_2-UF_4$ (66.1-33.4-0.5 mole %), was slowly cooled at the top surface from the liquidus past the solidus at the rate of 0.363°C/hr under a protective blanket of He gas purified with a charcoal trap at liquid nitrogen temperature. The frozen ingot was then core sampled. The experiment was duplicated as nearly as possible. The results are summarized in Table 1. In the second run the salt became wet after cutting open the container, and the values of the resulting analysis were low. These results were corrected in the last columns of the table. The following figure indicates the position of the core samples.



Although the salts for both runs came from the same batch, the lithium in the first run tends to be greater and the beryllium less than in the second run. Very little segregation of the salts occurred in either run. A slightly higher concentration of uranium was found at the bottom part of each ingot than the top.

If the ingot froze under equilibrium conditions with no entrapment of liquid in the boundaries between the crystals, one would expect to find $2LiF \cdot BeF_2$ at the upper part of the ingot since this phase crystallizes first at approximately 440°C and has a lower density than the liquid. Next a mixture of $2LiF \cdot BeF_2$ and $LiF \cdot UF_4$ should precipitate simultaneously at approximately 425°C, $2LiF \cdot BeF_2$ rising and $LiF \cdot UF_4$ being more dense should settle to the bottom. Finally, $2LiF \cdot BeF_2$, $LiF \cdot UF_4$, and a small amount of BeF_2 freeze together at the eutectic temperature of 350°C.[2] Theoretically, the uranium should be concentrated at the bottom; in actual practice this does not appear to occur since only a slight amount of segregation was found.

In another related experiment $PbCl_2$ was separated from a LiCl-KCl eutectic liquid by a fractional crystallization technique, the salt being lowered from a hot zone to a cooler one.[3] Vigorous stirring was necessary to prevent entrapment of the liquid phase — separation did not occur without this stirring. The conditions in this experiment can be considered similar to the chloride separation except the cooling front was accomplished by slowly lowering the temperature of the furnace instead of lowering the container.

These experiments cannot be construed to represent actual cooling a radioactive salt, since in the laboratory experiment the cooling rate was controlled by supplying heat to the sides and cooling at the top. The salt from a reactor will be of greater quantity as well as being internally heated by decay fragments. To duplicate the cooling of a salt after radiation more closely the cooling rate of a large mass should be controlled by adding heat internally, possibly by induction heat with several susceptors in the center of the melt. The crucible containing the salt would have to be a ceramic, possibly ZrO_2.

References

1. R. E. Thoma and H. A. Friedman, "Segregation of the MSRE Fuel Salt in Freezing in Drain Tanks," MSR-65-15 (March 17, 1965).
2. L. V. Jones et al., J. Am. Ceram. Soc. 45, 79 (1962).
3. Reactor Chem. Div. Ann. Progr. Rept. Dec. 31, 1965, ORNL-3913, p. 45.

H. A. Friedman

HAF/pah

Attachment

cc: Distribution

Table 1. Analysis of Samples

Position	Sample 1 (wt %)			Sample 2 (wt %)					
	Li	Be	U	Li	Be	U	Li*	Be*	U*
1	13.49	8.71	2.77	10.95	9.27	2.49	11.83	10.01	2.69
2	13.66	8.57	2.93	10.95	9.13	2.62	11.90	9.92	2.85
3	13.58	8.67	2.41	10.93	9.39	2.41	11.75	10.09	2.59
4	14.15	8.66	2.92	10.74	9.13	2.92	11.72	9.96	3.19
5	14.04	8.62	3.18	10.86	8.92	2.88	11.94	9.81	3.17
6	13.63	8.58	3.49	10.77	8.99	4.42	11.56	9.68	4.76
7	13.64	8.56	4.37	10.64	9.20	3.81	11.46	9.91	4.11
Starting Salt	13.13	8.63	3.43						

*Corrected values.

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INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

July 12, 1966

MSR-66-20

To: Distribution

From: MSRE Staff

Subject: MSRE Shutdown Caused by Short in Component Coolant Pump Power Wiring (June 27, 1966)

Summary

(R. H. Guymon)

The MSRE was shut down and drained at 0310, June 27, 1966, due to a phase-to-phase short in the electric wires to component coolant pump #1. The surge in current tripped Breaker "S" between the TVA feeder and Bus G-3. (Breaker H, supplying CCP-1 from Bus G-3 did not trip.) Bus G-3 also supplies power to the coolant pump, the health physics instruments, and other equipment. When the coolant pump stopped, control circuits caused a load scram and stopped the fuel pump which, in turn, caused a rod scram. The reactor had been operating at full power, and when circulation stopped, the radiator outlet temperature cooled below the interlock at 980°F, which automatically initiated a coolant drain. The fuel drain valve had started to thaw as soon as CCP-1 stopped. Before the trouble could be diagnosed, power restored, coolant circulation restarted, and the standby component coolant pump put on line, the drain valves thawed.

After the trouble was traced to CCP-1, inspection showed the heavy magnesia-insulated cable penetrating the dome had been damaged beyond repair. No replacement cable was available, so modifications to use 3 lighter cables were designed. While the changes were being made and the dome was being closed and tested, the coolant and fuel systems were refilled. At 0630 on July 1, operation at full power was resumed, 99 hours after the failure.

Symptoms and Operator Action

(P. H. Harley)

There were many simultaneous annunciations. Lost CP, FP, CCP-1, scrammed rods and load. All personnel monitors became inoperative. Also lost TF-1, CCC-1, RCC-1, DCC, and FOP-1. The outstanding annunciations were Flux >12 Mw, rod scram, safety channel tripped. The audible annunciator alarm could not be silenced.

The Shift Supervisor was the only one in the Control Room at the time of failure. Two operators were sampling fuel and one was in main office at the Xerox machine.

The Shift Supervisor announced the failure over the P.A. to summon aid and started looking for the source. Because there were so many alarms, they were not much help in pinpointing the trouble. However, from past experience it was judged most likely to be an instrument power supply failure.

First, all instrument power breakers in the ACR were checked. All appeared normal.

Second, the safety channels were checked but could not be reset because the fuel pump was off and flux was >15 kw.

An operator was dispatched to check the 48-v power supply and reported low voltage and battery being discharged. He could not start the standby MG set because its power supply was dead. This led to the discovery that Breaker S to Bus 3 was open. Attempted to start Diesel-Generator 3 but noticed no indication that it started. (The voltage was apparently slow in increasing.) Had an operator call shift electrician.

The Shift Supervisor started to Diesel House but before he got there, heard DG-3 running. He returned to the control room, and sent an operator to Diesel House. Restarted equipment operating from Bus 3 except for Coolant Pump and Component Coolant Pump No. 1. Did not start the coolant pump because the fuel and coolant drain had started by this time. Attempted to start CCP-1, but gave up after DG-3 speed decreased alarmingly when CCP-1 start button was pressed. Instead, started CCP-2. Had electrician check Breaker S and then parallel and transfer load from DG-3 to TVA.

Several hours later (about 0730) an attempt was made to start CCP-1 but nothing happened when start button was pressed.

Analysis of Electrical System Performance

(T. F. Mullinix)

Because all equipment on Bus 3 restarted successfully except CCP-1 and that would not start, we decided that this was the most likely cause of the trouble.

Testing from CCP-1, Breaker H compartment indicated an open circuit between the breaker and the CCP-1 motor, but no ground. Testing at the junction just outside the dome showed good cable back to Breaker H, but an open circuit inside the dome.

In the junction box inside the dome, the MI cable seal had disintegrated and what was left showed evidence of considerable arcing and fire. A phase-to-phase fault had obviously occurred.

The question arises, "Why did Breaker S open and not Breaker H?" Presumably the magnitude of the fault was so great that the time delay characteristics of Breakers H and S did not coordinate and the entire bus was cleared.

As in most such cases, the cause of the fault was destroyed when the fault occurred. The only thing that can be said with any degree of certainty is that there was an insulation breakdown at or in the MI cable seal. There is no mention of epoxy in the construction drawings and it is not normal practice to put epoxy in these seals. However, an ORNL electrician who participated in the installation as electrical foreman for the contractor recalls that epoxy was placed in the bell before the cover was silver-soldered in place. This was done on oral instructions from the G E & C electrical engineer following the job at that time.

The fault could conceivably have been caused by any of the following:

1. improper original installation which broke down with time,
2. broken ceramic insulator post caused by vibration,
3. moisture collecting on seal terminals,
4. voltage breakdown within the seal caused by epoxy filling.

Repairs have been made as described elsewhere in this report with the following cautions.

1. No epoxy in the MI cable seals.
2. Glass taping and moisture proofing (as near as possible) of all conductors and joints.

High megger readings were observed after the repairs were made.

Inspection and Repairs

(B. H. Webster)

After the trouble was determined to be inside the CCP-1 dome, the dome was removed and inspection begun as follows:

A smear was taken from inside the dome by Health Physics. Because of tritium (from the moisture condensed in the dome from the cell atmosphere), it was determined that protective clothing would be required.

On visually inspecting the inside of the dome, we found a black, sooty substance covering the bottom of the dome, the oil-to-water heat exchanger and piping, also the Greenfield sheath and the electrical junction box just inside the dome. This black substance and the odor of burned insulation led us to believe, at first, that the motor was burned out.

The cover was removed from the junction box inside the dome and we found all three electrical leads were burned such that they had separated completely. This apparently occurred inside the ceramic seal at the end of the 3-conductor #4 MI cable. (Photographs are attached to the copy of this memo in the MSRE files.) We found, on further examination, that two of the wires were welded together inside the MI cable which explained the earlier belief that we had only one open circuit.

After finding the situation as described above, a search was begun for new 3-conductor #4 MI cable and seals. All local sources were exhausted after about 4 hours and we cast about for alternate methods of rewiring the motor. Hugh Felts suggested we use three 3-conductor No. 10 cables and tie the ends together as was done on the fuel circulating pump motor. This method was soon agreed upon and work was started immediately to procure this material and make sketches (attached) for installation. In order to install the two additional cables, it was necessary to drill holes in the bottom of the dome and weld in two 3/4" pipe nipples as shown in MSRE Sk. 522. Welding was chosen in preference to threaded fittings because of strength considerations.

After the repairs were completed and the dome was closed up, the dome was pressure tested at 50 psig. The leak rate was not measured accurately but was low. The pressure was bled down to 30 psig and the fittings were soap-checked for leakage. Minor leaks around the threads were stopped by applying epoxy with the dome at a negative pressure.

The maintenance people learned of the mishap at 0800, Monday, June 27. The dome was ready for pressure testing at 1200 hrs, Thursday, June 30. During this period, craft time on this job amounted to 144 manhours straight-time and 100 manhours overtime.

Comments and Recommendations

P. H. Harley, Shift Supervisor on Duty

No attempt was made to block the coolant drain because it was believed to be safer to try and locate the trouble. The suspected source, CCP-1 was not found until ~ 0730, 4 1/2 hours after the failure when the log was checked and it was noticed that CCP-1 had been on at time of failure. (It was originally thought that CCP-2 had been on.)

A few minutes diagnosis, before any operator action was taken, might have indicated the source of the trouble sooner. The fuel drain could have been prevented by getting CCP-2 on immediately.

Attempting to start duplicate equipment would have given an earlier indication of the trouble instead of checking instruments.

Since there was freezing in the radiator as noticed during the refill, it may be safer to let a coolant drain proceed than try to stop it. This might cause a more serious freeze up. However, getting circulation started immediately might have prevented the freezing and stopped the coolant drain.

An indication that the diesel generator had started or was trying to start might have let us get the Bus 3 equipment back on in time to prevent the drains. On-Off lights at the start button would have helped.

I think the continual alarm came from the computer each time a fast scan started. The annunciator noise could have been eliminated by pulling the relay. Maybe we don't need the audible alarm here.

Colored markers for the operating equipment, where there is a standby unit, would let the operator know which unit stopped.

During periods when all operators are busy and away from the control room, especially fuel sampling where contamination hazard is involved, are when operations are more vulnerable to delays during emergencies. (Persons are delayed in leaving the contamination area until they check that they are not contaminated.)

R. H. Guymon, Operations Chief

Failures of this sort are bound to occur. The handling of these requires quick analysis and action by the operator. Since most of the circuitry is designed to fail safe, safety is usually not in jeopardy. However, many on-the-spot decisions must be made while trying to alleviate the immediate difficulty. For instance, should the drain be delayed with the possible hazard of freezing salt in the radiator or elsewhere in the system, or should the drain be allowed to continue with the consequences of delaying the project and thermal cycling the system.

Since many different electrical failures give similar indications, they are the most difficult situations to analyze. An attempt will be made to make up a procedure or check list which will cover most situations and which could be done essentially without thinking on all electrical failures. This could include starting diesels, checking instrument power circuit breakers, starting equipment, etc. Further training of each operator could improve the speed of doing the routine items and leave more time for thinking about the unusual.

T. F. Mullinix, G E & C Engineer

Epoxy should not have been used in the seal.

During the next shutdown, an investigation of the wiring on CCP-2 should be made.

B. H. Webster, Maintenance Chief

We should immediately order some 3-conductor #4 MI cable and fittings for future use if necessary.

P. N. Haubenreich, Department Head

This incident interrupted power operation for four days. If blame is to be placed, it must be with the original installation of the CCP-1 wiring.

When the failure occurred, it was not evident that CCP-1 was the culprit. If it had been, the operators probably could have gotten coolant circulation going and prevented the fuel and coolant drain. It is worthwhile trying to improve troubleshooting procedures; but shutdowns cannot always be avoided when important equipment fails, even by the best efforts of the operators.

The time required to get the system back into operation was minimized by the intensive efforts of the Plant and Equipment forces, and they are to be commended.

We still don't know for sure exactly why the short occurred, so we are guessing at what should be done about CCP-2 wiring. Certainly we should inspect it during the long shutdown and perhaps we should change the epoxy-filled seal.

Action Taken

(By 7/8/66)

1. We have ordered enough cable and fittings to replace the wiring on both CCP's.

2. We have scheduled an inspection of CCP-2 wiring during the next planned shutdown.

3. We now keep a round, red magnet on the control board beside the symbol for the unit in operation in all cases where there are duplicate units and only one operates at a time.

4. A quick checklist for troubleshooting electrical failures was prepared by T. L. Hudson from suggestions from each crew. This is being reviewed by crews now.

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INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

July 19, 1966

To: R. E. Brooksbank
P. N. Haubenreich
J. H. Shaffer

MSR-66-21

Subject: Study of Use of U^{233} in MSRE

Our tentative plans for FY 1968 call for changing the fuel in the MSRE from the present moderate concentration of uranium obtained by use of U^{235} of intermediate enrichment to a low concentration of uranium containing mostly U^{233}. Some of the problems and costs of preparing the enriching salt were investigated by McDuffie, Shaffer, and others, but little else has been done. We need a thorough study of the desirability and the problems of making this change of fuel. A committee consisting of P. N. Haubenreich, chairman, J. H. Shaffer, and R. E. Brooksbank is appointed to study the proposal, to prepare plans for carrying it out, to estimate the cost, and to include the information in a report. Assistance of other members of the Molten-Salt Reactor Program will be provided as required by the committee.

The report should include the following.

1. The advantages and disadvantages of making the change from the standpoint of the information we can obtain about the nuclear characteristics, engineering, and operation of the MSRE and of future molten-salt reactors.

2. Advantages and disadvantages of making the change from the standpoint of information that we can obtain about the chemistry of molten-salt reactor fuels and their compatibility with graphite and structural materials.

3. A complete plan and schedule for preparing the new fuel salt, processing the fuel now in use, and starting up the reactor again.

4. An estimate of the added costs to the program of making the change. This should include the costs of the new batch of fuel and of returning the used fuel to production channels. It should not include the costs of processing and other work done at the MSRE by people who normally work there. Mention should be made of loss in operating time if this is an important factor.

5. Recommendations concerning the desirability of making the change.

The report is needed by October 1, 1966.

R B Briggs

R. B. Briggs

RBB:alg

cc: G. M. Adamson
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INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

July 22, 1966

To: F. F. Blankenship
J. L. Crowley
J. H. DeVan
H. F. McDuffie
D. Scott

MSR-66-22

Subject: Use of Sodium Fluoroborate Salt in MSRE

Our tentative plans for the MSRE call for replacing the lithium fluoride - beryllium fluoride coolant with a sodium fluoroborate salt in FY 1968. The purpose of making the change is to demonstrate on a large scale in an operating reactor the salt we propose for the coolant systems of large power breeders. However, the problems of using this material in the MSRE have not been investigated, and plans have not been made for obtaining the salt or putting it in the reactor system. A committee composed of H. F. McDuffie, chairman, F. F. Blankenship, J. H. DeVan, J. L. Crowley, and D. Scott is appointed to investigate the use of sodium fluoroborate in the MSRE and to include the results of their investigation in a report.

The report should contain:

1. a discussion of the advantages and disadvantages of using sodium fluoroborate in the MSRE coolant system,

2. a discussion of the minimum research and development needed before the salt could reasonably be used in the MSRE,

3. plans, a schedule, and an estimate of the costs of the research and development work,

4. plans, a schedule, and an estimate of costs of producing salt for the research and development program and for the reactor,

5. recommendations concerning the desirability of making this demonstration in the MSRE.

The report is needed by October 1, 1966.

RBB

RBB:alg

R. B. Briggs

cc: G. M. Adamson
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7-26

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

July 25, 1966

MSR-66-23

To: R. B. Briggs

From: R. E. Thoma

Subject: Program for Examination of MSRE Surveillance Specimens

Plans for detailed examinations of MSRE core specimens have been formulated from the joint recommendations of staff members from the Reactor Chemistry, Metals and Ceramics, Analytical Chemistry, and Operations Divisions.[1] The principal objective of the first examinations by the Reactor Chemistry Division is to ascertain the extent to which the noble metal fission products of moderate cross section, Ru, Mo, Rh, Pd, Nb, and Tc, have deposited on the graphite moderator. Deposition of I and Te will also be examined. We will give primary attention to molybdenum because it can form a stable carbide and because its radiochemistry permits its use as an indicator in samples that have not decayed too long. Niobium is also a possible carbide former. The radiochemical properties of the noble metal fission products and their possible fate in the reactor have been described by S. S. Kirslis and F. F. Blankenship[2-4] in recent correspondence. Their appraisal of these phenomena serves as a basis for examining

the MSRE specimens according to the plan described below. Samples of graphite will be taken from near the top, middle, and bottom of the core.

1. Photography

As the graphite sample holder is disassembled, photographs will be taken of all exposed parts. It is expected that photographs which show smooth, clean, salt-free graphite surfaces will serve as an impressive display of the integrity of the moderator.

2. Gamma Scans

A measurement of the gamma activity along the entire length of the graphite sample assembly is precluded because of time. Also, variation in the areas of exposed surface and in the quantities of adherent salt that contribute to the gamma activity would interfere with interpretation of the data. Instead, it is planned that a complete gamma spectrum will be made of selected sections of Samples VA1, Y2, and VH5, removed from the fuel inlet, center, and outlet areas of the sampler channel.

3. Metallography

This technique affords a comparison of the surface of the graphite samples with as-received graphite specimens and with control specimens that have been exposed to molten fluoride salt without radiation. Although surface films will be

sought, it seems unlikely that the carbides can be formed on the graphite in amounts that could be detected by this method; if all the molybdenum which has been formed to date were deposited in the channels of the graphite, the surface concentration could not exceed 30 to 50 micrograms/cm^2.

4. Autoradiography

Autoradiographs will be made by E. L. Long, Jr. His results will be used along with those obtained from x-radiographic examinations to determine whether or not fuel salt has penetrated or adhered to the sample surfaces and to obtain a profile of radioactivity in the graphite. The map of radioactivity will be useful for selecting sites for core drilling.

5. X-Radiography

This method has proved to be one of the best means for ascertaining whether uranium-bearing phases have penetrated graphite surfaces. Radiographs of cross sections of each of the three graphite samples from the MSRE will be made by R. W. McClung's group.

6. Spark Spectroscopy

It may be possible to obtain spark spectroscopy data from several short traverses of the graphite surfaces if the gross activity of the specimens is not too high. No facilities exist for performing spark spectroscopy experiments with very active specimens. This technique affords its greatest

possible relevance in the current examinations if it becomes necessary to describe the extent of heterogeneity of coatings on the graphite.

7. X-Ray Diffraction

Diffraction analysis is a unique method for establishing the identity of nonmetallic films. It is anticipated that the technique will have limited application in the present series of tests because of sensitivity limits. The minimum film thickness required for registry of diffraction lines is 5000 to 6000 Å, which is presumably much greater than will be found in MSRE specimens. The method should therefore be used either if layer films are thicker than 6000 Å or if a tour de force of analyses is desired.

8. Radiochemical Analyses

The most definitive analyses which should afford quantitative answers to the question "How much molybdenum is deposited per cm^2 of graphite surface at various locations in the reactor?" are expected to be obtained from radiochemical analyses of thin sections of the graphite sample. In general, analyses will be made from specimens which are removed from the graphite surfaces by milling off layers as thin as 1 mil and collecting the powder. Radiochemical separations will be made of the dissolved specimens. Aliquots of the solutions should be submitted to C. E. Lamb for fluorometric analysis of uranium. Core samples of the graphite may

also be obtained and submitted to the same treatment as the milled specimens. Details of sampling procedures to be employed with these samples are discussed in a memo from J. G. Morgan to E. M. King, July 25, 1966; this memo provides the flowsheet for the handling of the samples.

9. Mass Spectrometric Analysis

A method of determining the concentration of fission product molybdenum in MSRE graphite specimens has been devised, and its use in the current program of examinations is proposed. A description of the method of analysis is given below:

Accumulation of molybdenum isotopes in graphite could lead to the buildup of substantial neutron poison in proportion to the fission yield. The rate of approach to saturation will be determined by the reciprocal of the cross section and flux, such that the time required to attain 50% saturation for a cross section of 1 barn at a flux of 10^{15} is 21.3 years. The molybdenum isotopes in question are

Isotope	Halflife	Fission Yield (%)	Cross Section (barns)	Years to Half-Saturation
92	stable	0	<0.3	>70
93	6.9 hr	0	-	-
94	stable	0	-	-
95	stable	6.4	13.4	1.6
96	stable	low	1.2	17.6
97	stable	6.2	2.1	10.3
98	stable	5.9	0.4	53
99	67 hr	6.1	-	-
100	stable	6.5	0.5	42

In 1.6 years, ^{95}Mo would be capable of absorbing 3.2 neutrons per 100 fissions. It would substantially inhibit the attainment of breeding if it were retained entirely on core graphite pieces. Other isotopes (e.g. ^{97}Mo) could also absorb undue quantities of neutrons.

The only radioactive fission product which is useful in determining the fate of fission product molybdenum is ^{99}Mo; it decays to ^{99}Tc with a 67-hr halflife. Radiochemical analyses should therefore be made to determine the ^{99}Mo and ^{99}Tc.

Since the isotopic composition of stable molybdenum products differs significantly from natural molybdenum (see table below), the isotopic dilution method may be used to determine the absolute quantities of both natural and fission product molybdenum recovered from MSRE graphite specimens.

	Abundance (%)	
Isotope	Natural	Fission Product (6 months at 10^{14}, 100 d cool)
92	15.88	-
93	-	-
94	9.12	-
95	15.70	19.6
96	15.50	~0.002
97	9.45	26.0
98	23.75	25.9
99	-	Unstable, 67-d halflife
100	9.62	28.3

An aliquot of solution from dissolved pieces of MSRE graphite could be spiked with a known amount of ^{92}Mo (or possibly another) isotope. The absolute amounts of natural and fission product molybdenum present can then be calculated from the results of mass spectrographic determinations of the relative abundance of the various molybdenum isotopes.

Since the isotopes in question are all stable, there is no urgency because of decay. The dissolution for radiochemical determination can be used (as much as can be made available). Mo-92 will be added so as to ensure homogeneity of the solution. The molybdenum can be separated if desired by precipitation or other procedures.

The maximum yield of molybdenum isotopes is estimated as follows: at 300 Mwd, approximately 1.26 moles of U have fissioned, producing 1.26 x (0.064 + 0.062 + 0.059 + 0.65) moles of molybdenum, or 1.26 x (0.25) = 0.415 moles Mo, equivalent to about 41 grams. There are 10^6 square centimeters of surface, so that we anticipate an absolute maximum of 41 micrograms/cm^2. We expect to achieve sensitivity to at least 1% of this level or perhaps for the respective isotopes to 0.1 microgram/cm^2. We should have available the yield of several cm^2 of graphite. A spike containing from a hundred to one thousand micrograms or so of ^{92}Mo should be appropriate.

John Sites believes that the above approach is feasible and will obtain the isotopic material. Ed Wyatt stated that

he normally added molybdenum as a carrier before counting and could readily use a ^{92}Mo solution and make the indicated separation for mass spectrographic as well as radiochemical determination.

With modifications as required this approach appears to be capable of determining whether the MSRE graphite specimens contain significant quantities of molybdenum.

The examinations proposed here will be carried out by the staff of the Operations and Analytical Chemistry Divisions in close cooperation with representatives of the Reactor Chemistry and Metals and Ceramics Divisions. The interleaving of efforts is expected to involve less detailed written instructions for hot-cell and analytical operations than is necessary in less intensive efforts.

R. E. Thoma

R. E. Thoma

RET/pah

cc: Distribution

References

1. Analytical Chemistry Division: E. I. Wyatt and A. S. Meyer.
Reactor Chemistry Division: F. F. Blankenship, E. G. Bohlmann, E. L. Compere, W. R. Grimes, S. S. Kirslis, H. F. McDuffie, J. G. Morgan, and R. E. Thoma.
Operations Division: S. E. Dismuke, E. M. King, and A. A. Walls.
Metals and Ceramics Division: W. H. Cook and C. D. Scott.

2. S. S. Kirslis and F. F. Blankenship, "Fate of Noble Metal Fission Products," Intra-Laboratory Correspondence, January 7, 1965.

3. S. S. Kirslis, "Radiochemical Analyses of MSRE Fuel Salt Samples," MSR-66-7 (March 18, 1966).

4. S. S. Kirslis, "Detailed Program of MSRE Fission Product Analyses," Intra-Laboratory Correspondence (May 11, 1966).

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INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

July 18, 1966

MSR-66-24

To: P. N. Haubenreich

From: R. B. Gallaher

Subject: Recovery of Capsule from Sampler-Enricher

On July 11, 1966, while attempting to insert the key of an empty sample capsule into the drive unit latch, the manipulator slipped, jerking the capsule assembly in Area 1C. The capsule dropped onto the gate of the operational valve, about 14-1/2 inches below the bottom of the access port. Recovery of the capsule was necessary before another sample could be isolated.

The immediate action taken was to tag "Do Not Operate" to the operational and maintenance valves to prevent the capsule from dropping into the pump bowl. The circuit breakers in the power supply to the valve motors were opened and tagged out. The sample that had just been isolated was removed from the sampler.

The keys on all sample capsules had been changed from brass to nickel-plated mild steel a few months ago so that a magnet could be used for recovery. However, an attempt to do so had not been tried. Therefore, equipment and testing were required.

A straight 3/8" diameter by 6-1/4" long magnet was selected. A 3/16" diameter hole was drilled in one end. All sharp edges were removed. A brass key was attached to the magnet using 1/32" diameter wire rope. The assembly was 12-1/2 inches long. Another brass key was attached with a separate wire to make an assembly 48 inches long. The short loop was used to attach the magnet to the drive unit latch. The long loop was to assist in recovering the magnet in case the drive unit could not withdraw it easily. Brass keys could be handled near the magnet without difficulty.

Several tests were made using the chemical processing sampler as a test stand to demonstrate that the magnet would attract the mild steel key strongly enough to lift the capsule. The stainless steel walls were only mildly magnetic and did not prevent free movement of the magnet.

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Just prior to starting the recovery attempt, Area 1C and Area 3A were purged to remove the gaseous activity which might be present. The removal valve was opened to allow the magnet to be dropped into Area 3A. Some difficulty was encountered since the top shield is covered with hot rolled steel which is magnetic. Also the cast stainless steel of the ball valve caused some trouble by attracting the magnet some.

After the magnet was pushed into Area 3A, the removal valve was closed and the access port was opened. Using the manipulator, the magnet was attached to the drive unit latch using the small loop. The long loop remained in Area 3A. Next, the drive unit was inserted about 6 inches and then withdrawn without difficulty. When the magnet was removed from Area 1C, the capsule was attached to it. With the magnet lying in Area 3A but still attached to the latch, the capsule key was pulled free of the magnet. The capsule and key were moved to one side of Area 3A. The magnet was removed from the latch. After attaching the capsule to the latch, the access port was closed.

To remove the magnet from Area 3A, the removal valve was opened and a hook inserted through it into Area 3A. One of the wires on the magnet was looped over the hook. The hook with the magnet attached was withdrawn into a plastic container. The radiation level of the magnet was 10 R/hr at 2 feet. It was quickly removed from the high-bay area. Some contamination was dropped on the top of the sampler.

Should it be necessary to recover another capsule, two changes in the above procedure should be made. First, the magnet should be placed inside a non-magnetic tube before inserting into Area 3A. Then the tube and magnet could be lowered into Area 3A without catching on the sides. Second, the magnet should be withdrawn into a shielded cask to reduce personnel exposure.

R. B. Gallaher

RHG:al

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INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

July 29, 1966

TO: R. B. Briggs MSR-66-25

FROM: W. H. Cook

SUBJECT: Designations and Orientations for MSRE Surveillance Specimens

This is to confirm oral discussions with various members of Reactor Chemistry, Metals and Ceramics, Analytical Chemistry, and Operations Divisions concerning designations and orientation of the first MSRE graphite surveillance specimens that are to be examined.

The designations of the specimens to be examined immediately are listed in Table 1. Some interest has been expressed in additional samples, P2, P18, P19, and P22; so, these have been included in the table.

The lattice bar material, bar 1559, is slightly more porous than the core graphite. Bar 635 is considered typical of the core graphite. Bar 1229 represents material in the core that shows some slight evidence that its degree of graphitization is not quite as good as that represented by bar 635.

The locations and orientations of the reactor core specimens are shown in Figs. 1 through 7. These are self-explanatory. The control specimens (see Table 1) are located and oriented in the same way in the Controlled Test Rig. The control specimens were not exposed to any radiation. They were exposed in nonflowing (stagnant) salt with temperature gradients and pressures approximately matching those of the MSRE.

Table 1. Designations of Graphite Specimens of the MSRE Reactor Core and Control Specimens[a]

Stringer	Specimen Designation	Machined from Bar No.	Comment
RS1(CS1)	VA1(VA4)	1559	Lattice bar
	Y2(Y5)	635	Core bar
	VH5(VH7)	635	
	P2(P22)[b]	1229	Core bar
	P18(P19)[b]	1229	

[a]The designations enclosed in parentheses are those for the control specimens.

[b]Additional samples that may be included in the examinations.

W. H. Cook

WHC:lmp

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7-28-66 WHC
7-10-65 WHC

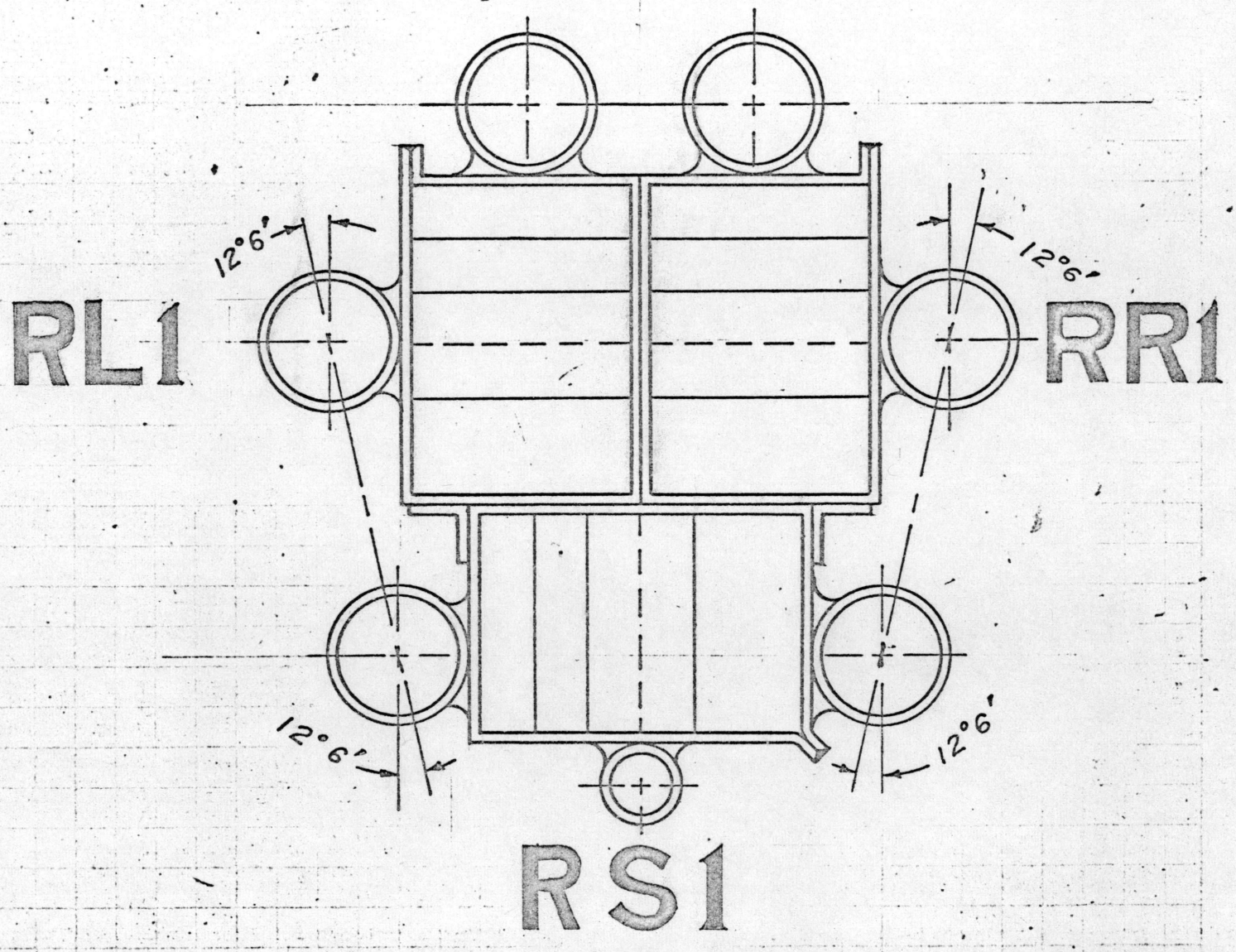


Fig. 1. Top View of MSRE Reactor Core Specimens of Stringers RS1, RR1 and RL1

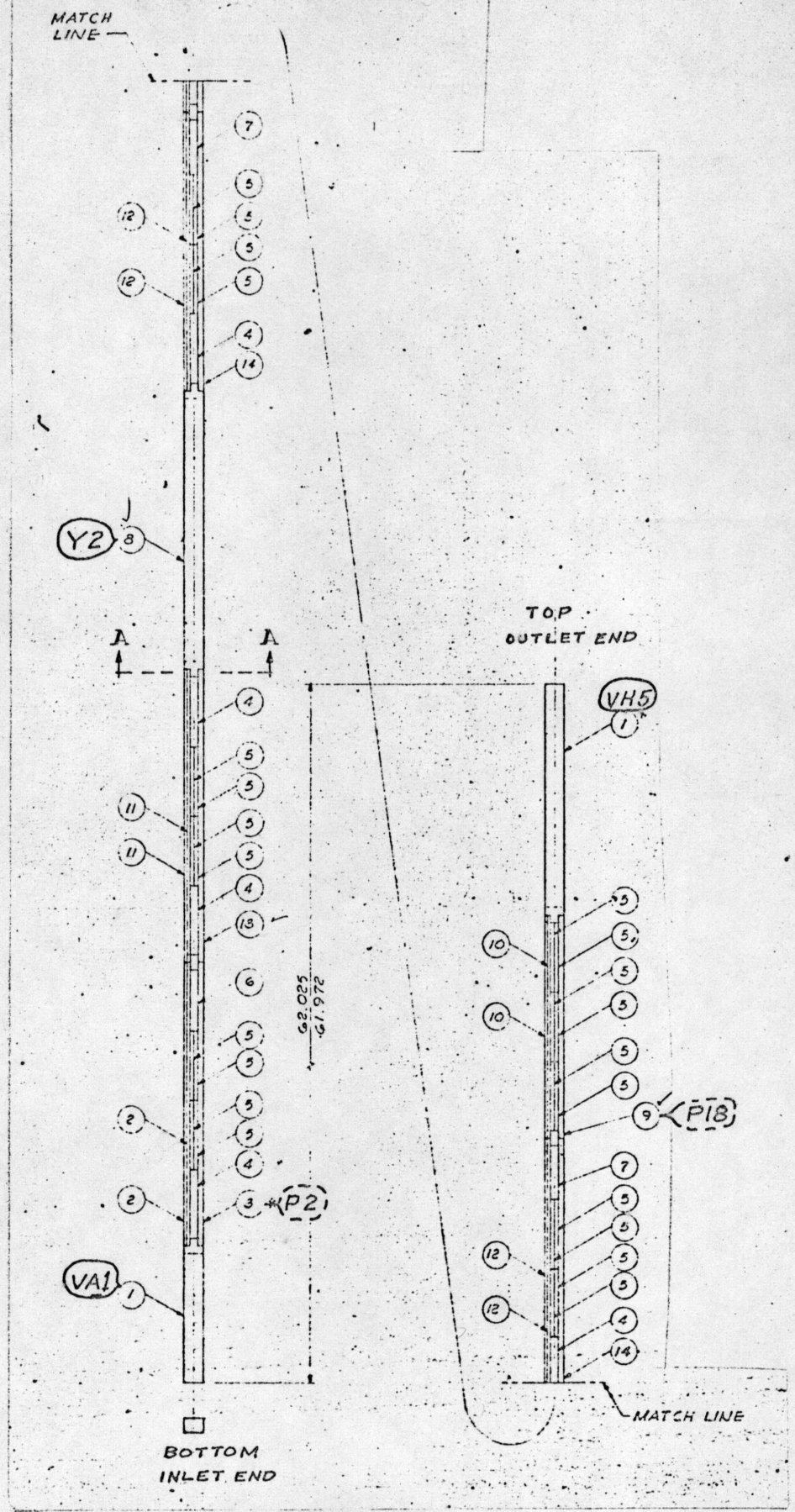
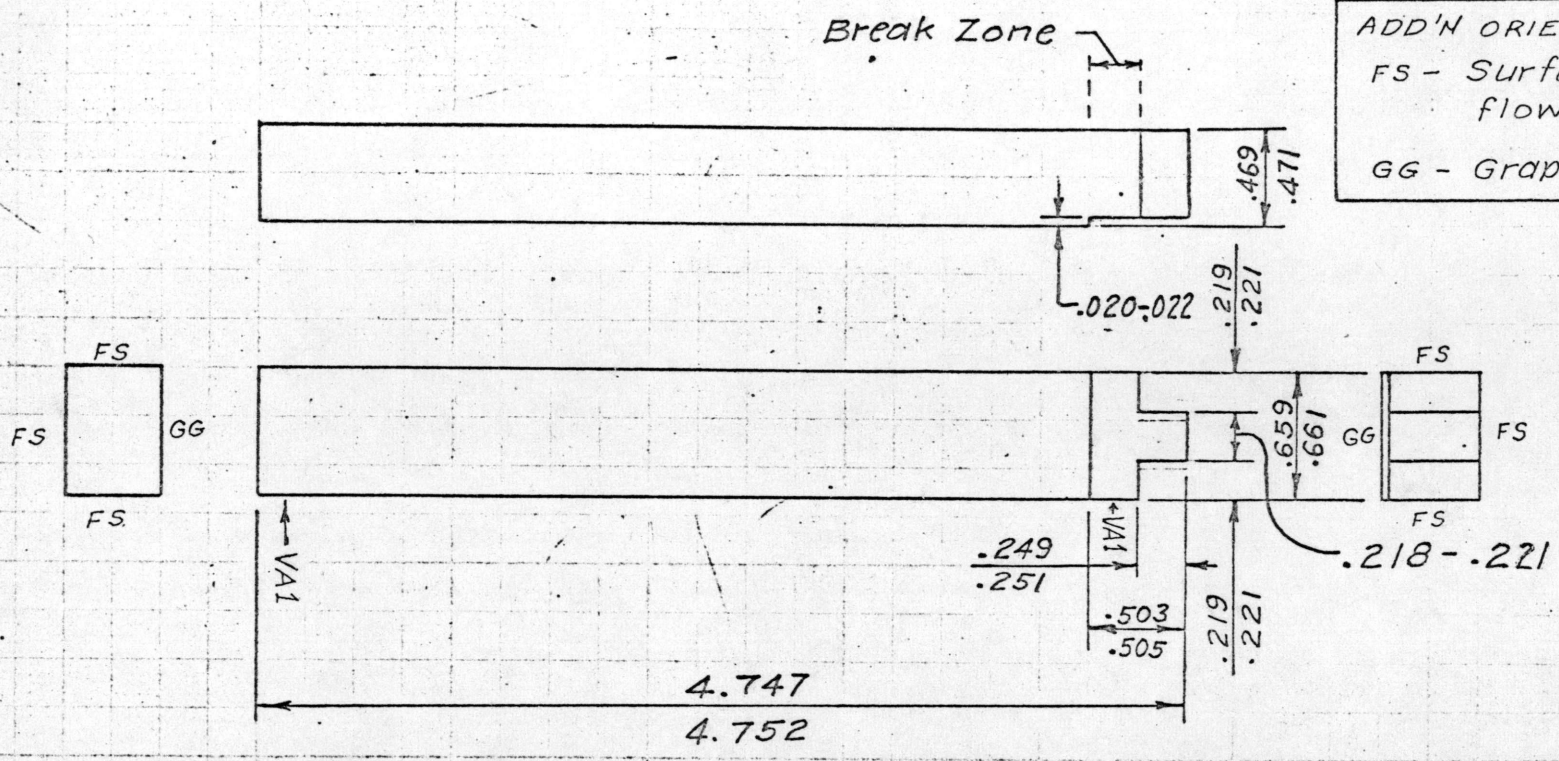


Fig. 2. STRINGER RS1

7-28-65 WHC
6-4-65 WHC –

ADD'N ORIENTATION NOTES:
FS – Surface exposed to flowing salt
GG – Graphite to Graphite.



SECTION A } OF STEM
~~SECTION H~~ }

NOTES:

For all surfaces, the roughness height shall not exceed 32 microinches (rms).

~~QUAN. REQ'D~~

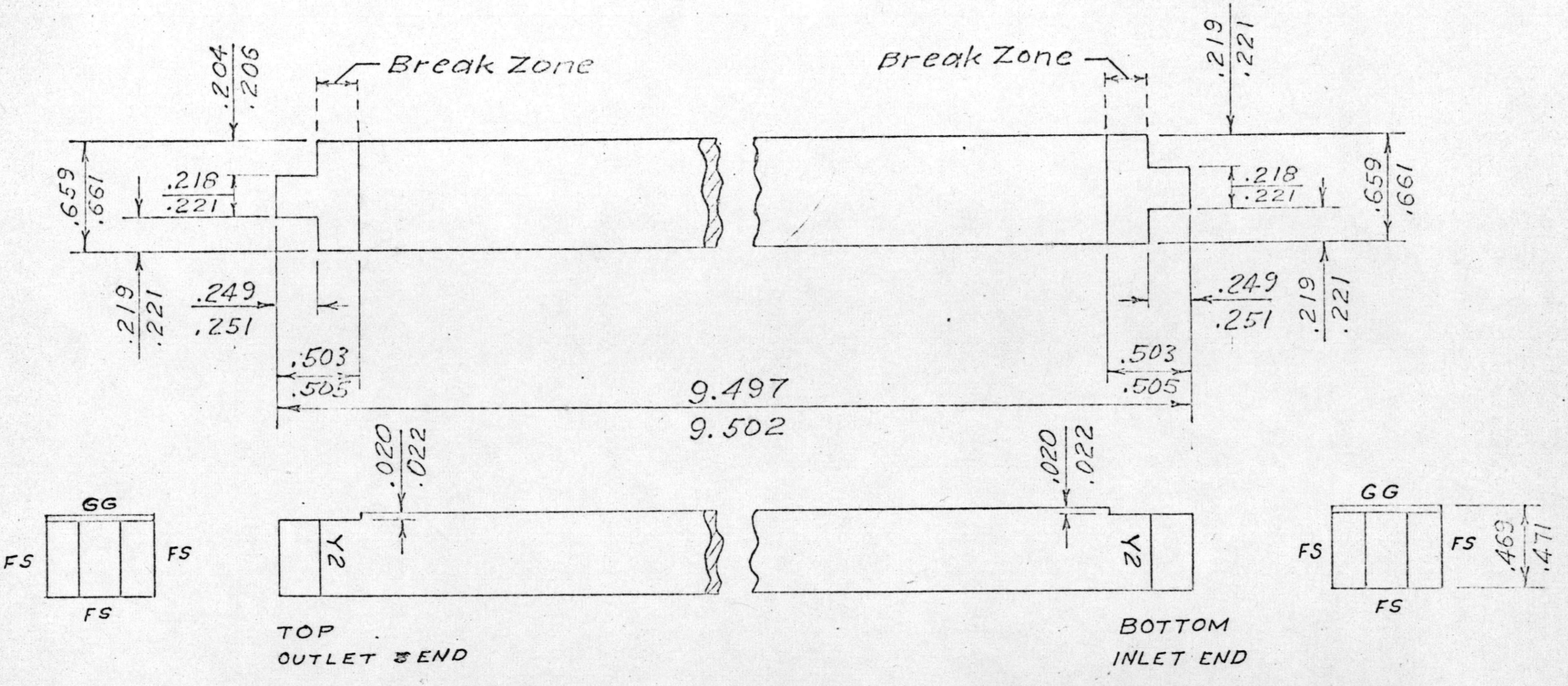
BAR NO. 1559
GRADE CGB
LATTICE BAR MAT'L

$.015^{+.000}_{-.010}$ Break at 45° only on the longitudinal corners within the indicated "Break Zone."

Fig. 3. ~~FIXED~~ SPECIMEN VA1 (Length ∥ Length of Original Bar)

FROM STRINGER RS1

7-28-66 WHC
6-4-65 WHC



SECTION D OF STEM

ADD'N ORIENTATION NOTES:
FS - Surface exposed to flowing salt.
GG - Graphite to Graphite;

~~QUAN. REQ'D~~

NOTES: For all surfaces, the roughness height shall not exceed 32 microinches (rms).

$.015^{+.000}_{-.010}$ Break at 45° only on the longitudinal corners within the indicated "Break Zone."

BAR NO. 635
GRADE CGB

Fig. 4. SPECIMEN: Y2

~~PIECE~~

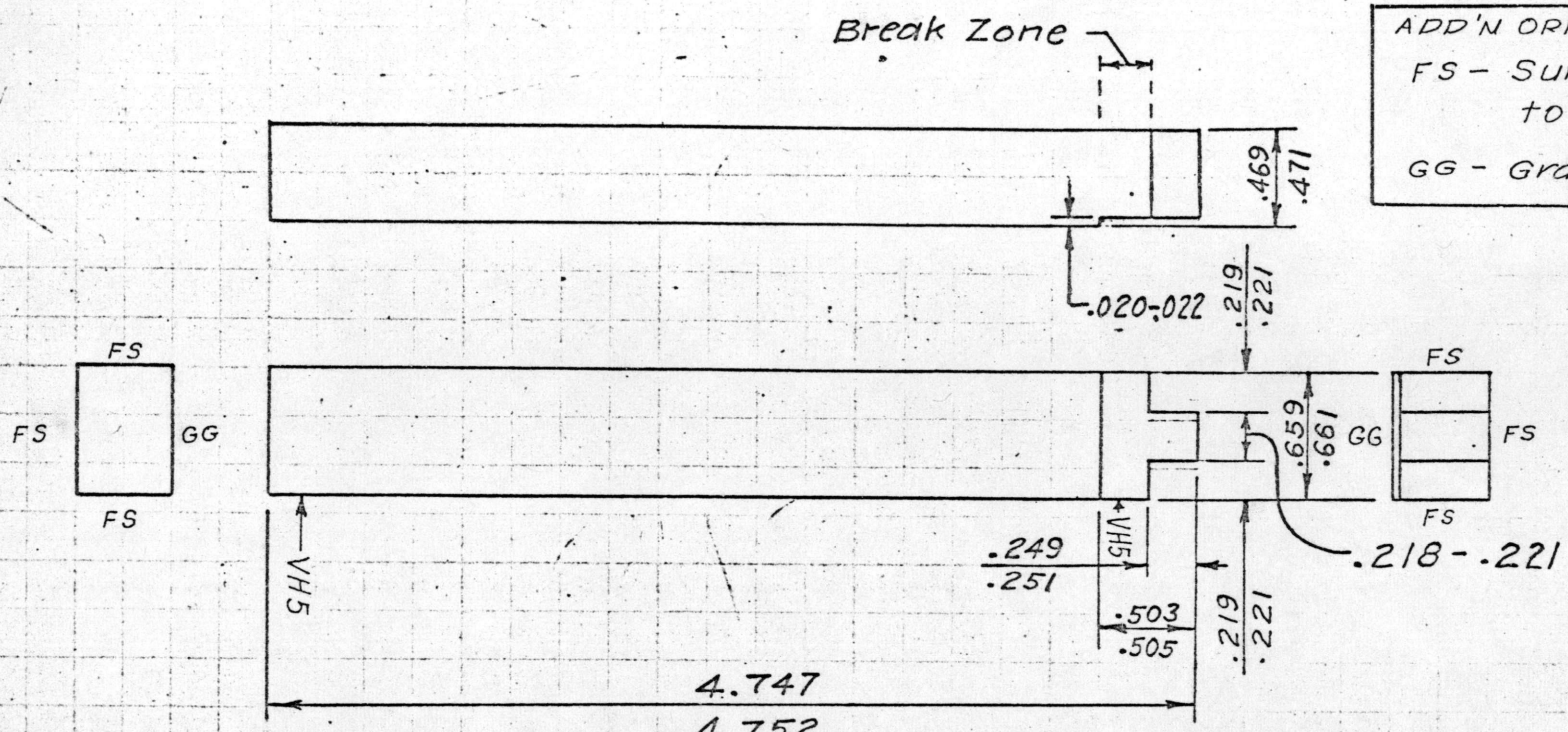
~40m

Length = Length of Original Bar

FROM STRINGER RS1

7-28-66 WHC
6-4-65 WHC –

ADD'N ORIENTATION NOTES:
FS – Surface exposed to flowing salt
GG – Graphite to Graphite



~~SECTION A~~ } OF STEM
SECTION H }

NOTES:

For all surfaces, the roughness height shall not exceed 32 microinches (rms).

$.015^{+.000}_{-.010}$ Break at 45° only on the longitudinal corners within the indicated "Break Zone."

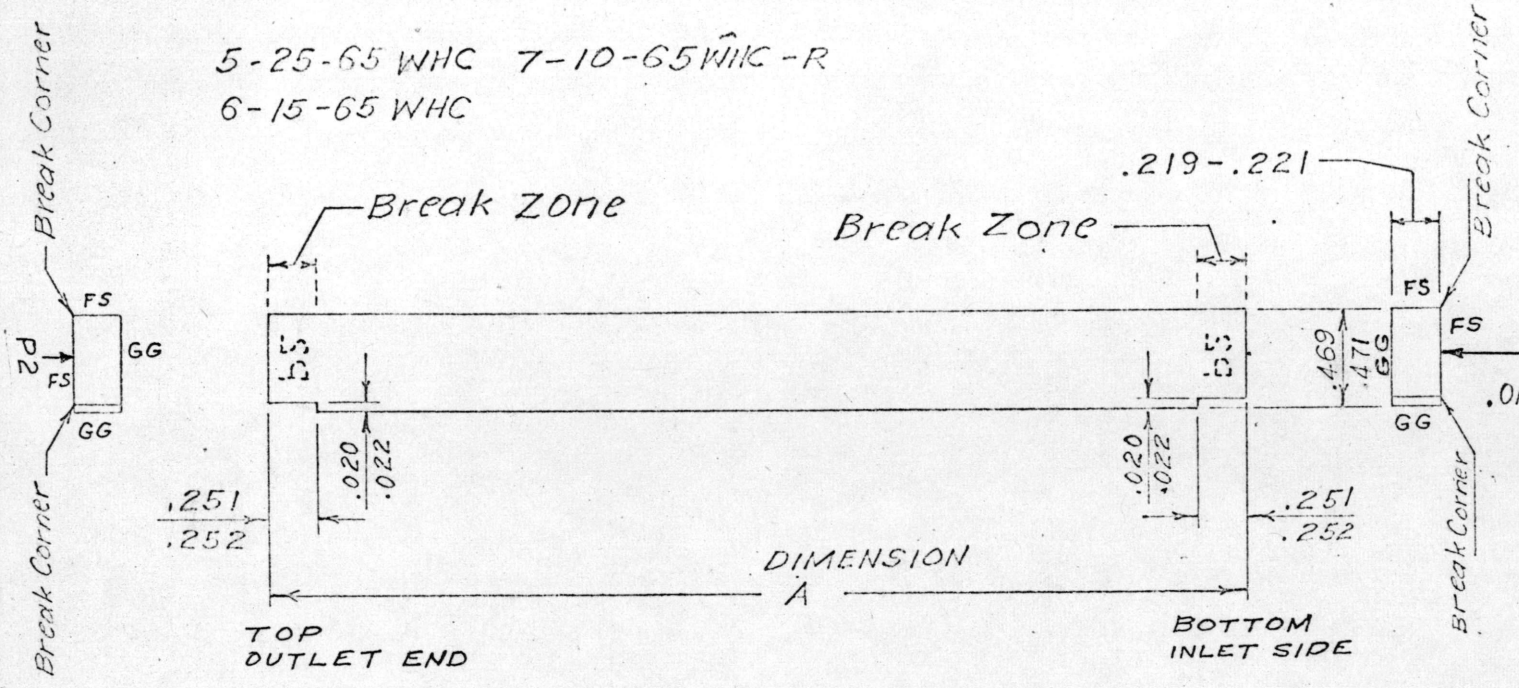
~~QUAN. REQ'D~~

BAR NO. 635
GRADE CGB

Fig. 5. ~~DWG.~~ SPECIMEN VH5

FROM STRINGER RS1

5-25-65 WHC 7-10-65 WHC-R
6-15-65 WHC



Notes:
For all surfaces, the roughness height shall not exceed 32 microinches (rms).

$.015^{+.000}_{-.010}$ Break at 45° only on indicated longitudinal corners within the indicated "Break Zone."

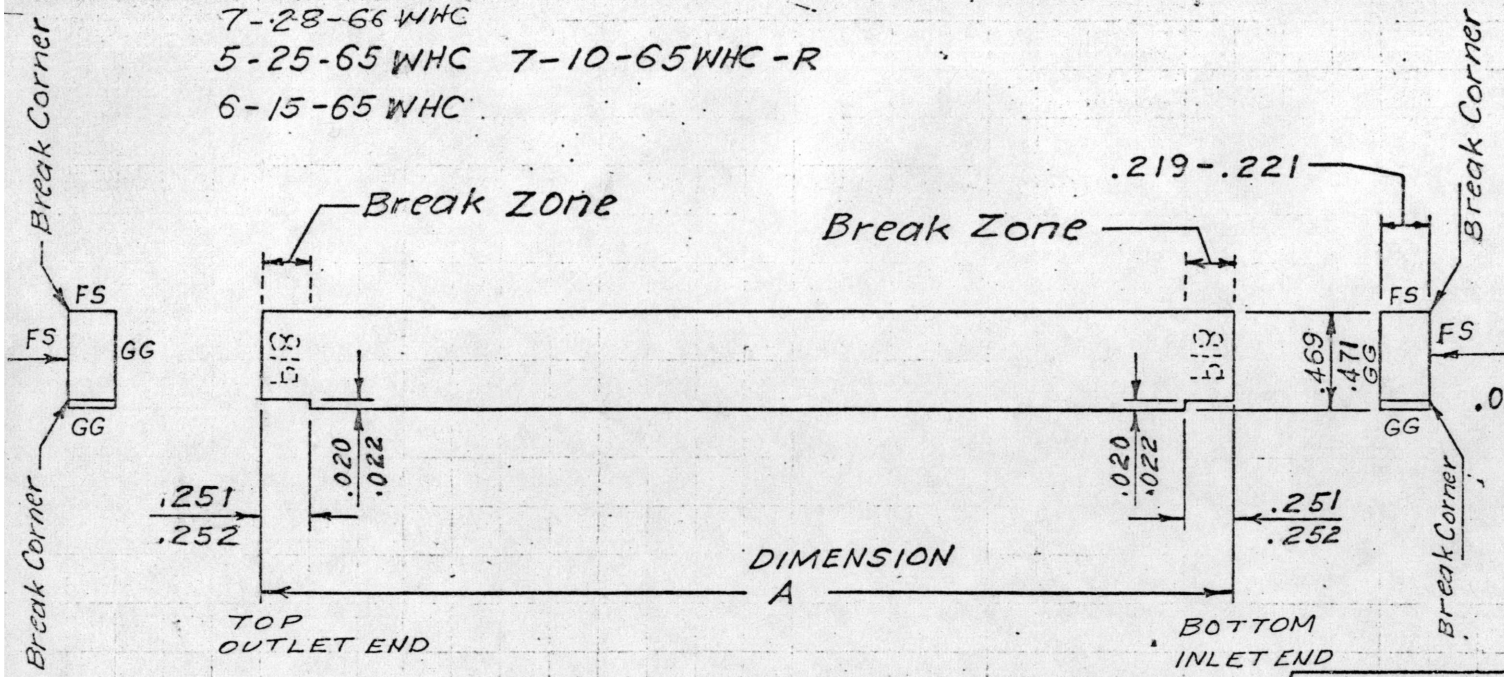
ITEM	DIMENSION A	QUANTITY REQ'D
~~P1~~ P2	9.247 9.252	

ADD'N NOTES ON OPIENTATION:
Specimen Markings are on side away from viewer in dwg.
FS- Surface exposed to flowing salt.
GG- Graphite to Graphite

BAR NO. 1229
GRADE CGB

Fig. 6. SPECIMEN ~~P1~~ P2 (Length ll Length of Original Bar)
STEM
FROM STRINGER RS1

7-28-66 WHC
5-25-65 WHC 7-10-65 WHC-R
6-15-65 WHC



Notes:
For all surfaces, the roughness height shall not exceed 32 microinches (rms).

Break at 45° only on indicated longitudinal corners within the indicated "Break Zone".

ITEM	DIMENSION A	QUANTITY REQ'D
~~B~~	~~9.247~~ ~~9.252~~	
~~C~~	~~9.497~~ ~~9.502~~	
~~E or F~~	~~8.997~~ ~~9.002~~	
~~G~~ P	7.247 7.252	

ADD'N NOTES ON ORIENTATION:

Specimen markings are on side away from viewer in dwg.

FS- Surface exposed to flowing salt

GG - Graphite to Graphite.

BAR NO. 1229
GRADE CGB

STEM

Fig. 7. SPECIMEN ~~DWG.~~ P18 (Length ≈ Length of original Bar)

FROM STRINGER RS1

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

August 15, 1966

To: C. K. McGlothlan MSR 66-26

Subject: Modifications Made to MSRE Radiator Door Assembly

The purpose of this letter is to list some of the modifications made to the MSRE radiator door assemblies. These changes were "phased-in" with other scheduled plant operations and changes.

The original door design did not have sufficient insulation facing the heat source to prevent warpage of the door frame. A new door assembly was designed and built that provided this insulation as well as a new gasket arrangement. The seal surface was taken off the door and relocated on the radiator facing. It consisted of two 3/4" wide Johns-Manville Thermocore #C-397 square asbestos packing strips. After experimental trials the "T" section, which had been designed to seal against the packing, was given more bearing area against the packing by welding on 1/8" x 1" wide metallic strips.

In the original design when the door was raised, the metal sealing surfaces scraped across the gasket causing rapid damage to the packing. The new design was arranged so that the door on the downward descent engaged four "throw-out" cams so that the door descended at some clearance away from the gasket, then engaged the cams and moved in to sealing position. In the first trials it was discovered that insufficient clearance existed in the vertical cam tracks so further revision had to be made to compensate for an accumulation of dimensional errors. It was also discovered that increased door weight necessitated the addition of shock absorbers; this was suspected early in the design and no serious delays were experienced in adding them. The shocks chosen were Hannafin "Softstop" 4" stroke, 1-1/8" diam bore, 20,000 lbs capacity. Insulated door hoods had been installed in the penthouse to provide a barrier to the heat flow into the penthouse; however, no attempt was made to make these hoods leak tight. A synchro-indicating device had been added to each 2" door drive shaft to indicate the door positions; however, this has not proven completely reliable as shown in the case when a door hung in the open position and the synchro indicated that it was closed.

The major problems that existed in November 1965, seemed to be:

1. The doors needed to be moved away from the radiator to eliminate the possibility of hanging up.

2. The cams and cam tracks needed to be modified to affect reliable operation and positive sealing of the doors.
3. Hoods needed to be widened to accommodate new door location.
4. Door limit switches needed to be added for reliable door positioning (100% open).
5. An "anti-snarl" (later called a "loss of tension") device was needed for the wire rope cables.

The cam tracks were modified to move the doors 1/2" further away from the face of the radiator. This gives approximately 1" total clearance. At the same time, the trip locks and modified brackets were moved. A new cam follower bracket was designed; the new cams (1-7/8" diam) gives 1/8" total clearance in the track. The end rollers were positioned to prevent side motion from jamming the doors. Some structural and sheet metal modifications were necessary. The door hoods were widened 1-1/2" for better door clearance. Limit switch assemblies (two per door) were added above the doors to prevent over-running and possible jamming of the doors. Each assembly has two limit switches - one to stop the door at 100% open position and the other 1" above. Above the high position, "hard stops" for the doors are located. An overload trip for stopping the drive motor will be installed to work in conjunction with these stops. A cable "loss of tension" device was designed and fabricated. These devices (one per each wire rope cable or a total of eight) will indicate if a door has jammed or if a cable breaks. The control circuits have not been designed as yet.

At the completion of the above modifications the heat balance of the radiator system was affected. Two major difficulties were encountered: one was the excessive temperature rise in the penthouse just above the radiator; the other a positive pressure in the penthouse (supposed to be negative with respect to high bay pressure)

Efforts to correct the above were as follows:

1. The heater leads junction boxes located just over the insulation above the radiator were relocated to the east wall of the penthouse. TA wire (thermoplastic asbestos, 194°F) was pulled from the junction boxes to the beaded heater leads coming out of the radiator. Connections were made which split both connectors at a point about 2' above the insulation. The beaded leads were spread out to get maximum air flow thru. A duct extension (10" x 14") was added to the lower stack exhaust opening in the south wall. It extends down to just above the radiator insulation and pulls air (2000 cfm) thru the south beaded lead openings.

A ten inch round flexible duct (Flexiaust GlasHose - 300°F rating) was attached to a plate that covered one half of the inlet of each coolant cell cooler. The inlets to the flexible ducts are positioned one at the northwest beaded lead outlet and other at the thermocouple junction box. Flow thru these ducts is about 500 cfm each.

2. Heat leakage thru the door hoods was reduced by welding up all cracks that were accessible, by patching holes, by adding asbestos boots at the wire rope cable openings, and in some places by stuffing insulation down in openings.

Even with the above modifications the penthouse pressure was positive with respect to the high bay pressure which is controlled in the range -0.1 to -0.3 inches of water. The flow thru the temporary hatch opening was 7000 to 8000 cfm. A duct was installed from the inlet of the south annulus blower (10,000 cfm rating) to the coolant drain cell, which in turn is open to the penthouse. Air flow from the coolant cell area with both annulus blowers on was 7440 cfm. The flow to the stack was 5500 cfm. The penthouse pressure varied from -.014 to -0.28 inches of water. However, when one main blower was turned on, the pressure rose to -0.04 inches of water; and when both main blowers were turned on, the pressure was slightly positive (the best that could be read with the available instrument).

A crossover duct, 1 sq ft area, was installed at the top of the inlet door hood to the top of the outlet hood to relieve the pressure in the inlet hood.

Ambient temperatures in the penthouse with the radiator hot and the reactor subcritical were averaging about 90°F at the drive shaft assembly level, 100°F just above the top of the radiator insulation, and 600°F four inches down in the beaded lead insulation. With the reactor at 1 mw, the temperature ranges were 100 to 150°F at the drive shaft level, 130 to 145°F at the floor elevation and 700 to 850°F in the beaded leads.

J. C. Feeman

JCF:am

Attachment

cc: E. J. Breeding
R. B. Briggs
P. N. Haubenreich
B H. Webster
file (JCF)

RADIATOR DOOR REFERENCE DRAWINGS

"AS BUILT" - APRIL 1966

E-DD-D	55510	Radiator Door Assembly
E-DD-D	55511	Radiator Door Frame Sub-assembly
E-DD-D	55512	Radiator Door Insulation
E-DD-D	55513	Radiator Door Frame Details and Parts
E-DD-D	55514	Radiator Door Gasket Seal Frame Details
E-DD-D	55515	Radiator Door Insulation Sub-assembly Details
E-DD-D	55516	Radiator Door Cam Bar Housing Col. - Enclosure - Upstream, downstream
E-DD-D	55517	Radiator Door Cam Bar Housing Col. - Enclosure - Upstream, downstream
E-DD-D	55518	Radiator Door Roller Trip Assembly & Details - Top - Bottom
E-DD-D	55519	Radiator Door Enclosure Gasket Installation & Sect.
D-DD-D	49581	Radiator Door Hood Addition - Crossover Duct
D-DD-D	49582	Radiator P.H. Hood Det. for E & W P.H. Coolers
E-DD-D	49583	Radiator Door Cam Followers & Roller Bracket Assembly & Details
E-DD-D	49584	Radiator Door Upper Limit SW Support
E-DD-D	49585	Radiator Door Upper Limit SW Housing
E-DD-D	49586	Radiator Door Upper Limit SW Support Plan
D-KK-A	49589	Radiator P.H. Flexible Ducts to E & W Cooler Return
D-KK-A	49590	Radiator P.H. Containment Ducts Extension Plan and Sect.
E-DD-D	49594	Radiator Door Stop Bracket & Shock Absorber
E-DD-D	49595	Wire Rope Cable Retainer & Limit Switch Assembly
E-DD-B	40446	Radiator Door Cam Lock Details
E-DD-B	40448	Radiator Door Cam Bar & Housing Col. Assembly
D-DD-C	40450	Radiator Door & Lifting Arrangement - Plan
E-DD-D	40470	Radiator Enclosure General Assembly, Sheet 1
E-DD-D	40471	Radiator Enclosure General Assembly, Sheet 2
E-DD-D	40480	Radiator Air Duct Flange Assembly
D-DD-D	40481	Radiator Enclosure Plating Details
E-DD-B	40488	Bottom Tie Beam Sub-assembly
D-DD-C	40498	Radiator Door Lifting Detail Support St. & Drive Shaft
E-DD-D	40746	Radiator Insulation Assembly
E-DD-C	41476	Radiator Door Lifting Detail Assembly & Detail at Clutch Assembly and Door Limit SW Loc.

E-DD-D	56277	Radiator Perimeter Duct (Upstream and Downstream)
E-DD-D	56278	Radiator Perimeter Duct (Upstream and Downstream) Detail
E-DD-D	56279	Radiator Door Limit Switches
E-KK-A	49577	Air Duct From Basement to Annulus Blower Fan, No. 2 - Loc & Det.

INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

August 18, 1966

MSR-66-27

To: Distribution

From: C. H. Gabbard

Subject: Salt Spill from MSRE Surveillance Test Rig on 12/2/65

One accomplishment of the MSRE operation which is so obvious that we may overlook it is the successful containment of the molten salt and the radioactive gases. During operation to date there has been no leakage of salt or radioactive gas from the reactor. Actually, the greatest probability of a salt release was not during operation, but while the salt was being charged into the reactor. Over 25,000 lb of salt was transferred in the molten state from 98 containers into the reactor system with no more than a few cc of salt oozing from tubing fittings on a few occasions. There has been one spill of salt in the reactor building, and that is the subject of this memo.

A small quantity of fuel salt containing beryllium and depleted uranium salts was spilled from the Surveillance Test Rig during the initial fill of the rig from its drain tank. The rig and drain tank had been heated to 1000°F while under vacuum. The test rig was then helium filled, purged, and then vented to atmospheric pressure. The drain tank was then pressurized to transfer the salt in the drain tank up into the actual test rig. The level in the rig was to be adjusted between two conductivity level probes, which were 1 inch apart, by adjusting the drain-tank pressure. The displaced helium in the test rig was to vent continuously to the ventilation stack as the filling progressed. The rig was overfilled as evidenced by both level probes being lighted and by visual observation of high temperatures just above the heater insulation. Pressure was vented off of the drain tank to lower the level in the test rig, but both the level probes remained lighted. The high temperature on the pipe wall receded below the thermal insulation indicating that the level had actually been lowered. The operator believed that he had vented sufficient pressure from the drain tank to lower the level below the probes and that a false level indication existed because salt had bridged the gap between the probe and the pipe wall. In an effort to clear the level signal, the operator removed the upper probe. After about 15 seconds, hot salt began running out of the opening and the operator quickly reinserted the probe to stop the salt flow. The remaining pressure was then vented from the drain tank and all

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the salt was drained out of the test rig into the drain tank. All the heaters were turned off and the test rig and drain tank cooled to room temperature. The probe was then again removed because freezing salt had prevented its complete reinstallation. The open fitting was then capped off.

The operator had been wearing a face mask and a respirator when the spill occurred, and there was no personal injury to him or his assistant who was also working in the area. The high bay was immediately evacuated and the affected area was roped off. Industrial Hygiene checked the area and found that there had been no general spread of contamination and that the contamination was localized in the immediate vicinity of the test rig. Normal work was resumed in the high bay the following morning except for the roped-off area around the test rig.

Prior to starting the actual decontamination of the rig, the contaminated area was enclosed in a plastic tent, and the equipment was sprayed with a plastic coating to seal in the contamination. Although there was apparently no actual damage to the equipment, the salt cleanup required the removal of the thermal insulation, the heater wiring, and the thermocouple leads. The decontamination of the rig was completed with Health Physics and Industrial Hygiene supervision. Airborne contamination was avoided by wetting the equipment and contaminated surfaces.

The operators did not believe the salt that remained in the rig or the internal surfaces of the rig had been exposed to atmosphere. Therefore after the rig had been decontaminated and repaired, it was heated, filled, and placed in operation. Operation of the rig was resumed on 12/23/65.

When the possible consequences of this spill are considered in regard both to personal injury and to damage to the experiment, the removal of the probe with the high salt level was an unwise and unsafe act. Preferably the salt should have been completely drained to the drain tank and the test rig and drain tank should have been vented prior to removing the probe. If this was not possible or was not desirable, the level should have been reduced sufficiently to insure that a spill was impossible, and the fill line should have been frozen to prevent an unexpected transfer of salt into the rig.

If we accept the removal of the probe as being a reasonable operator action under the existing circumstances, the exact cause of the spill is somewhat uncertain. The salt level could have been above the probe penetration at the time the probe was removed, or the level could have been lower initially with a subsequent transfer of salt from the drain tank. The 15-second delay between the probe removal and the salt spill would indicate that salt was not immediately available at the probe and that a transfer had occurred. The facts that the salt ran out relatively slowly and that the operator was able to reinsert the probe also indicate that a transfer was occurring from the drain tank.

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There are two areas in the rig design that probably contributed directly to the spill. First, the level probe penetrates the pipe wall with a relatively close clearance that is capable of trapping salt and shorting out the probe. This shorting is apparently a common problem because the operator had a high degree of confidence that the level indication was false and that he could clear the indication by removing and cleaning the probe. Second, the gage that indicated the test-rig pressure was located several feet away on the 1/4-inch offgas line. There was also a small valve in this line between the rig and the tee where the gage was connected. When the rig was being purged (as was the case), the pressure gage read somewhat lower than the actual pressure in the rig depending on the pressure drop through the line and valve. Removing the probe then vented this pressure, and the pressure in the drain tank increased the salt level about one foot per psi of pressure that was vented. This is the most probable explanation of what actually occurred.

The following recommendations are made in the hope of avoiding future accidents of a similar type.

1. Never open a system containing hazardous material to the atmosphere unless there is positive assurance that none of the material can escape. Procedures should be written, approved, and followed when performing such an operation.

2. The design of future level probes should be revised so that salt bridging between the probe and the pipe wall is normally impossible.

3. Pressure gages should be installed so that they are not subject to pressure-drop errors caused by a flowing stream. Complex valving systems should also be avoided where possible.

Any one of these three recommendations would probably have prevented the above spill.

C. H. Gabbard

C. H. Gabbard

CHG:al

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INTRA-LABORATORY CORRESPONDENCE

OAK RIDGE NATIONAL LABORATORY

August 30, 1966

MSR-66-28

To: Distribution

From: C. H. Gabbard

Subject: Trip Report to Joy Manufacturing Company
August 24, 1966

A meeting was held on August 24, 1966 with the Joy Manufacturing Company at New Philadelphia, Ohio, to discuss the repair of the MSRE Main Blowers. The meeting was attended by C. H. Gabbard and L. M. Haisten of Union Carbide and C. P. Jenkins, W. Bienko, and F. Pietro of Joy. The details of the repair and modification of the rotors, the delivery schedule, and overall cost of the repairs were discussed. The corrective measures proposed by Joy are as extensive as can be reasonably expected and appear to be adequate to eliminate any future failures. The delivery schedule as proposed by Joy was modified to ship two rebuilt rotary assemblies by September 16 for delivery in Oak Ridge by September 19. The third completely rebuilt unit would then be shipped at some later date. Joy agreed to better this schedule on the two rotor assemblies if possible. The original schedule called for shipment of the first unit on September 9 and the second and third units on September 23. Although an exact cost for the job was unavailable, Joy proposed to repair the two failed units at their costs of material, labor, and overhead and to repair the third unit at no cost. The overall cost was estimated to be about $10,000. Standard charges of $76.00 per day plus expenses would be made for the service man supervising the installation at the MSRE.

Joy still cannot give a satisfactory cause for the failures and they claim that 500 - 1000 units of this type are in satisfactory service. The design has existed since 1950. The design stresses of 6000 to 8000 psi are satisfactorily low. Joy stated that the rear hub castings had supported the entire centrifugal load on MG-3 in the area of the large continuous crack and this is proof that the design is structurally sound. Although no physical property tests had been run, they also stated that the base metal appeared satisfactory.

Both the front and rear hub castings are to be reinforced by the addition of gussets in the area of the blade sockets as shown on Figure 1. These gussets will reduce the bending moment at the base of the sockets where the cracking started and will distribute the load to

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the outer rim and to the central section of the hub. The gusset plates, which are cast of 40E aluminum (the hub material), will be welded in place. This reinforced design is similar to the design used on a larger 60" hub diameter fan. In addition to the hub reinforcement, the centrifugal blade loading will be reduced about 35% by using magnesium alloy AZ-91 for the blades instead of the 40E aluminum. The AZ-91 alloy has comparable tensile and yield strengths to the 40E alloy and has a higher endurance limit. The hubs and blades of essentially all of Joy's normal fan production are cast from these and two other alloys. As a final proof test, the rotors will be given an overspeed test at 2300 rpm for 15 minutes. The hubs will be dye-checked before and after the test to see if any cracks develop during the test.

Joy seems a little deficient in some areas of quality control and they seem to depend heavily on the overspeed test as a "catch all" to avoid failures in the field. Our blower rotors had been spin tested at 2300 rpm but they had not been dye-checked after the test. The normal procedure is a visual inspection for gross defects and a 30% overspeed test. Failures during the overspeed test are very rare. Ladle analysis of the various casting melts are not made. Joy depends on the "certified analysis" supplied by the alloy vendors for this. Joy uses prime metal for all melts and does not remelt scrap.

The installation, checkout, and operation of the repaired blowers in the MSRE were discussed. Joy did not think the adjustable motor mounts were necessary or desirable. Their representative would be responsible for obtaining the proper alignment on the floating shaft (0.001" per inch of coupling diameter). The axial clearance on the couplings should result in the flexible shims being visually flat. The adjustable mounts are more likely to lose their adjustment than a solid mount.

The Joy service man would also balance the rotors in place to 0.001" peak-to-peak displacement, and Joy felt that one vibration pickup on each bearing would be adequate. We could take reference readings when the machines were first balanced and watch for changes in the vibration thereafter. The pickups would be most sensitive in the horizontal plane because the blower mounting is more flexible, but other orientation would also be satisfactory. The vibration could increase to 0.004" before corrective action would be required.

Joy liked our proposed thermocouple installation to measure the bearing temperatures. The bearings should not exceed 200°F, and 150 - 160°F would be a more desirable operating temperature. We should be sure to use a grease that will operate at 200°F. Joy felt that the grease vent line on the bearings should not be used because a relatively high pressure would be required in the bearing to force the grease out through the vent. This would cause large amounts of grease to be forced through the bearing seals. The bearings should be lubricated once per month with 2 to 3 ounces of grease per bearing.

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A schedule of periodic inspections was discussed, but Joy felt that a rigid schedule was unnecessary. Instead, they recommended an inspection after the first month of operation and then inspections during our shutdown periods at our convenience. These inspections should include an alignment check and a visual inspection of the couplings, the blading, and the front rotor hubs. The hub inspection could be made through the front inspection plate, and removal of the upper casing half would be unnecessary.

In conclusion, the rebuilt units with the reinforced hubs and with the magnesium blades should operate satisfactorily without additional failures. Generally Joy's proposed repair, inspection, and testing procedures appear to be adequate. However, it is unclear whether the blades are included in the dyecheck inspections before and after the overspeed test. We should request (and insist) that the blades be included in these dyechecks. Some method of documentation is also required to determine if cracks propagate during the overspeed test. The castings will show some porosity, and it may be difficult to determine if the dye indications increase after the overspeed test.

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Attachment

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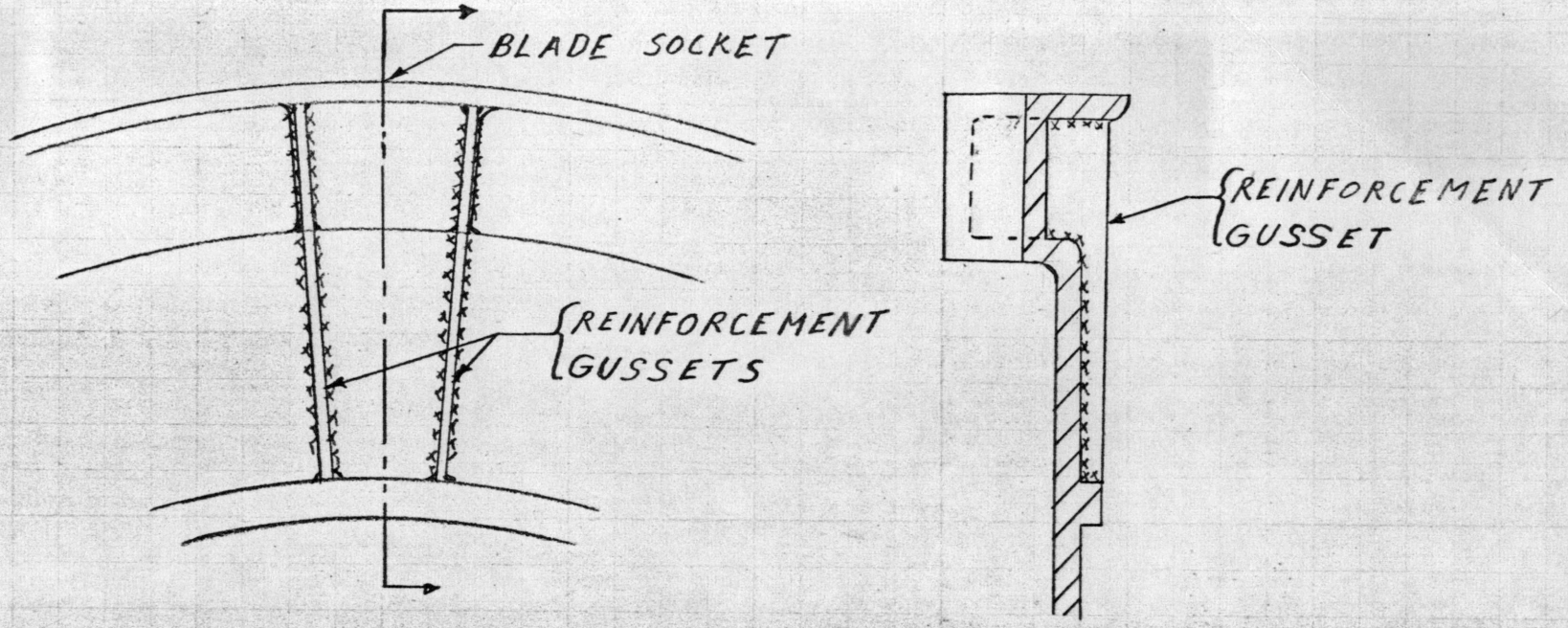

Figure 1

Hub Reinforcement for MSRE Main Blowers

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