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DESCRIPTION:

Ltr furnishing info re Refueling outage & reporting several plant conditions.....W/Attached Figs 1 thru 6.

ENCLOSURES:

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ACKNOWLEDGED

PLANT NAMES: H. B. Robinson Units2

FOR ACTION/INFORMATION 6-1-73 AB

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Carolina Power & Light Company

May 25, 1973

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Serial: NG-73-24

Mr. John F. O'Leary
Directorate of Reactor Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

50-261

Dear Mr. O'Leary:

H. B. ROBINSON UNIT NO. 2
LICENSE DPR-23
REFUELING OUTAGE

During the recent (March 16 - May 16, 1973) Robinson Plant refueling outage several plant conditions were revealed which are appropriate for report either in accordance with Technical Specifications and Bases or as items of mutual interest. The broken Rod Control Cluster and the Safety Injection Pump performance are reported in accordance with Section 6.6.3 of the Technical Specifications, and the remaining items are reported for your information and to formalize previous verbal communications.

SAFETY INJECTION PUMP PERFORMANCE

On April 23, 1973, during the refueling outage, a test of the high head safety injection system was performed. The purpose of the test was four-fold: 1) To measure the pump shutoff head (at miniflow) of the safety injection pumps as normally done during the periodic test of the pumps, 2) To measure the flow delivery capability of the system under the condition of essentially zero reactor back pressure, 3) To demonstrate that flow is delivered down each individual cold leg injection branch line, and 4) To demonstrate the effect of the pump miniflow on the system delivery capability.

The results of this test were as follows:

1. The system performance for the minimum safeguards case used in the safety analyses was verified by the test. System delivery at the high reactor back pressures as determined by the test was slightly greater than that assumed in the safety analysis (see figure 1). This represents a slight conservatism, when compared to the safety analysis, in the speed at which boron injection tank contents would be swept into the reactor during a steam break accident.

2. The required flow to the reactor through the three cold leg injection lines from two operating pumps was demonstrated by the test for each pair of pumps.

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3. The small effect of pump miniflow was demonstrated during the acceptance runs with two pumps operating since the system delivery changed only slightly when the miniflow was terminated. Back calculation to the shutoff head of the pumps indicated that the miniflow was satisfactory and very close to the design value.

4. The total developed head for each of the three safety injection pumps, while running on miniflow, was above the minimum acceptance value of 1425 psig and therefore is considered to be in accordance with the Technical Specification requirement. This result verifies the results obtained during normal Periodic Tests.

The test results also indicated that several data points for each pump fell slightly below the pump characteristic submitted to the AEC Division of Reactor Licensing in Carolina Power & Light Company's letter, Docket No. 50-261, dated August 12, 1970, (see figures 2 through 5). For this reason, we are reporting this information to the Division of Reactor Licensing.

Since the overall system performance for the minimum safeguards case used in the safety analyses was verified by the test, there are no unreviewed safety problems associated with the apparent change in pump characteristics. Therefore, Carolina Power & Light Company will continue to operate H. B. Robinson Unit No. 2 in accordance with the approved Technical Specifications. Normal Periodic Tests of the safety injection pumps will be continued in order to verify satisfactory operability and performance. It is our intention to conduct further system tests and to inspect at least one safety injection pump during the next refueling shutdown.

Westinghouse has requested Worthington, the pump vendor, to review the results of the H. B. Robinson tests. We will forward additional comments and recommendations for resolution of this matter as soon as available.

ROD CONTROL CLUSTER (RCC) VANE BREAKAGE

On April 4, 1973, during core unloading, a vane from the RCC in fuel assembly A-19 was discovered separated from the spider hub during operations in the RCC change fixture (see figure 6). The following special inspection and evaluation was conducted immediately upon discovery of the problem:

1) A visual inspection of all RCC's in RCC change fixture during withdrawal of the control rod from the fuel assembly to determine if any other control rods were similarly affected with separated vanes.

2) A visual inspection of each RCC Spider Assembly within fuel assemblies in the spent fuel storage rack. Each vane attachment was inspected closely for indications of separation. Integrity of the vanes was reassured by condition of tack weld at top side of each vane.

3) A detailed visual inspection of the separated vane, the associated spider, and rodlets.

4) Radiation measurements of rodlets from separated vane and associated control rod assembly to determine approximately when during plant operation the separation might have occurred.

5) A visual inspection using boroscope inside the control rod guide tube of the failed control rod for anomalies resulting from or contributing to failure.

6) A review of plant operating history for operating irregularities that may have had a relationship to failure. This included CRDM and rod scram irregularities.

The results of this inspection were as follows:

1) Only one RCC Assembly was involved. No anomalies were observed in the inspection of remaining RCC Assemblies.

2) Radiation measurements revealed the rodlets on the separated vane three feet from the top were at a level of 1.1×10^5 R/hr compared with 600 mr/hr for the rodlets on the main control rod assembly. This result indicates the separated vane with the two rodlets was in the full scram position in core for a significant period of time while the remaining rodlets with spider assembly were in the normal low flux position, withdrawn from the core. Calculations indicate the period of time was about 8000 full power hours of plant operation.

3) No anomalies were observed in the control rod guide tube. There were no significant marks observed on the control rod or in the guide tube that would indicate rod jamming from debris.

4) No event in plant operating history could be related to the failure incident.

5) No anomalies were visible in the braze joint.

The most probable failure mechanisms are:

1) less than full strength on the braze joint on the RCC Assembly and

2) abnormal loading on the RCC Assembly due to the presence of debris.

There were no positive indications as to the cause of failure. Since no markings indicating interference or impact were present on the failed vane, it appears that the joint was not overstressed because of jamming from debris. On the other hand, visual examination of the braze joint showed no indication of insufficient braze.

Although there was no evidence of the presence of debris, this does not exclude the possibility that debris existed earlier in cycle I operations. Radiation levels on the failed vane indicate that the failure occurred early in the first cycle. However, extensive operating time has been accumulated with the other control rods, and the probability is good that if other rods were similarly affected, they would have exhibited problems.

A similar problem occurred at another plant during first core cycle. In that case two RCC's were involved and the cause after hot cell examination of the failed joint was attributed to a manufacturing defect. The corrective action in June, 1970, was to replace the two affected RCC Assemblies. There has been no indication of any problem since that time.

In any event this failure did not cause damage to the control rod guide tube. The RCC was replaced with a spare assembly from on-site spares, and the new RCC was test dropped five times at hot shutdown conditions with satisfactory results. This incident is considered an isolated one, and no further action is deemed necessary.

The effect on core physics parameters both during cycle I operation and for future operation is negligible. The integral worth of the two-rodlet vane is estimated to be less than 0.02% $\Delta\rho$. During cycle I, this would have caused a slight flux depression in the immediate area of the rodlets. This effect would be local to the fuel assembly containing the rodlets and was not observable as core flux tilt or on the nearest moveable in-core detector 1½ assemblies away. The assembly, A-19, (Region I) contained a fixed in-core detector which was out of service. Any reduction of fuel burnup and potential flux peaks are of no concern since Region I fuel was removed from the core and will not be reused.

STEAM GENERATOR GIRTH WELD

During the inservice inspection performed in March, 1973, an ultrasonic indication was found in the channel head to tube sheet weld of the "A" steam generator. The indication was inspected and evaluated by the vendor, Westinghouse - Tampa Division. The indication has the form of a slag line 22 inches long having a cross section contained within a circle of 0.080 inch diameter. The extent of this indication was compared to a similar indication found on a 44 series steam generator of another utility. The comparison showed that indication to be more heavily loaded and of a worse geometric configuration than the indication on the Robinson steam generator. Analyses presented by Westinghouse in the report "Evaluation of Ultrasonic Indication Reported During In-Service Inspection on Steam Generator "A" at Carolina Power & Light Company H. B. Robinson Plant No. 2" show that the indication is of no structural consequence and that the steam generator could be returned to service without adverse safety implications. This indication will be under scrutiny at future inservice inspections.

AUXILIARY FEEDWATER THERMAL SLEEVES

Our correspondence of February 9, 1973, reported the discovery and repair of leaks and cracked welds in two of the three Auxiliary Feedwater (AFW) connections to the main feedwater lines. Subsequent engineering study by Ebasco Corporation revealed that the probable cause of this condition was thermal shock resulting from initiation of relatively cool AFW flow into hot main feedwater headers. To remedy this condition thermal sleeves designed by Ebasco were fabricated and installed during this refueling outage.

The existing saddles and nozzles were removed, and the 16-inch main feedwater header opening prepared for the thermal sleeve to main header weld. Welding was done using the TIG welding process for the root pass and manual shielded metal arc for the remainder of the weld. Upon completion of these welds the 4-inch auxiliary feedwater header was rewelded to the thermal sleeve. All welds made during this repair were radiographed. The thermal sleeve to 16-inch header weld was given a post weld heat treatment after which the welds were again radiographed. In addition to 100% radiography the repair was successfully leak tested prior to operation. It is anticipated that this repair is a permanent solution, and no further problems are expected.

CHARCOAL VENTILATION FILTER TEST

Change No. 16 to Robinson Plant Technical Specifications requires (in part), for the charcoal filters in the spent fuel building exhaust and containment purge systems, the demonstration of a removal efficiency for methyl iodide of 98%, and, after July 1, 1973, of 99.5%. The initial test of the charcoal sample from these new beds was conducted by MSA Corporation prior to refueling and indicated an efficiency for methyl iodide of approximately 98%. In anticipation of July, 1973, requirement of 99.5% and since new, unused charcoal can be expected to produce optimum efficiency, Carolina Power & Light Company arranged a retest of this carbon utilizing the specific test conditions set forth in the Technical Specifications. This test was conducted on April 16, 1973, in Richland, Washington, under conditions of 24°C, 70% relative humidity, 40 ft/min face velocity, and 1.5 mg/m^3 methyl iodide concentration. It was witnessed by two Carolina Power & Light Company engineers and a private consultant. The test resulted in a methyl iodide removal efficiency of 98.57%.

There is serious doubt that commercially available charcoal, new or used, can perform to the 99.5% efficiency under the conditions required by the Technical Specifications. Even under less humid conditions there is little evidence that this efficiency may be consistently attained. Certainly, the normal reduction in efficiency through usage in ventilation air systems would require that these beds be completely replaced yearly at exorbitant costs.

Carolina Power & Light Company intends to further investigate this problem with the objective of obtaining more usable specifications which meet the fuel handling accident safety analysis and Safety Guide 25.

Yours very truly,

E. E. Utley
Vice-President
Bulk Power Supply

DBW:mp
Attachments

cc: Mr. C. D. Barham
Mr. N. B. Bessac
Mr. B. J. Furr
Mr. D. V. Menscer

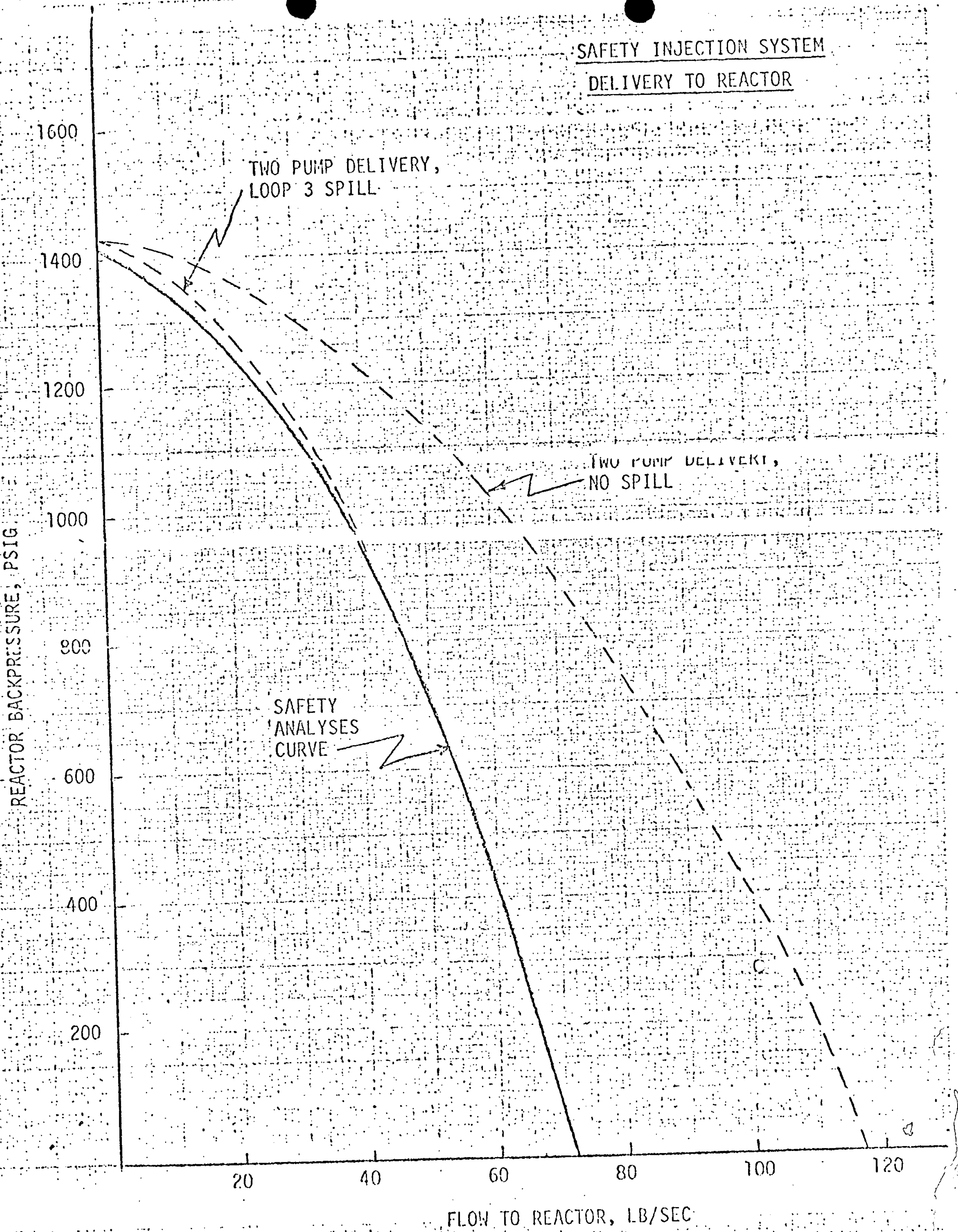
SAFETY INJECTION SYSTEM
DELIVERY TO REACTOR
TWO PUMP DELIVERY,
LOOP 3 SPILL
TWO PUMP DELIVERY,
NO SPILL
SAFETY
ANALYSES
CURVE
REACTOR BACKPRESSURE, PSIG
1600
1400
1200
1000
800
600
400
200
20
40
60
80
100
120
FLOW TO REACTOR, LB/SEC

FIGURE 1

SAFETY INJECTION PUMP PARAMETERS

SAFEGUARDS RE-EVALUATION

[REPLACES FSAR FIG. 6.2-7]

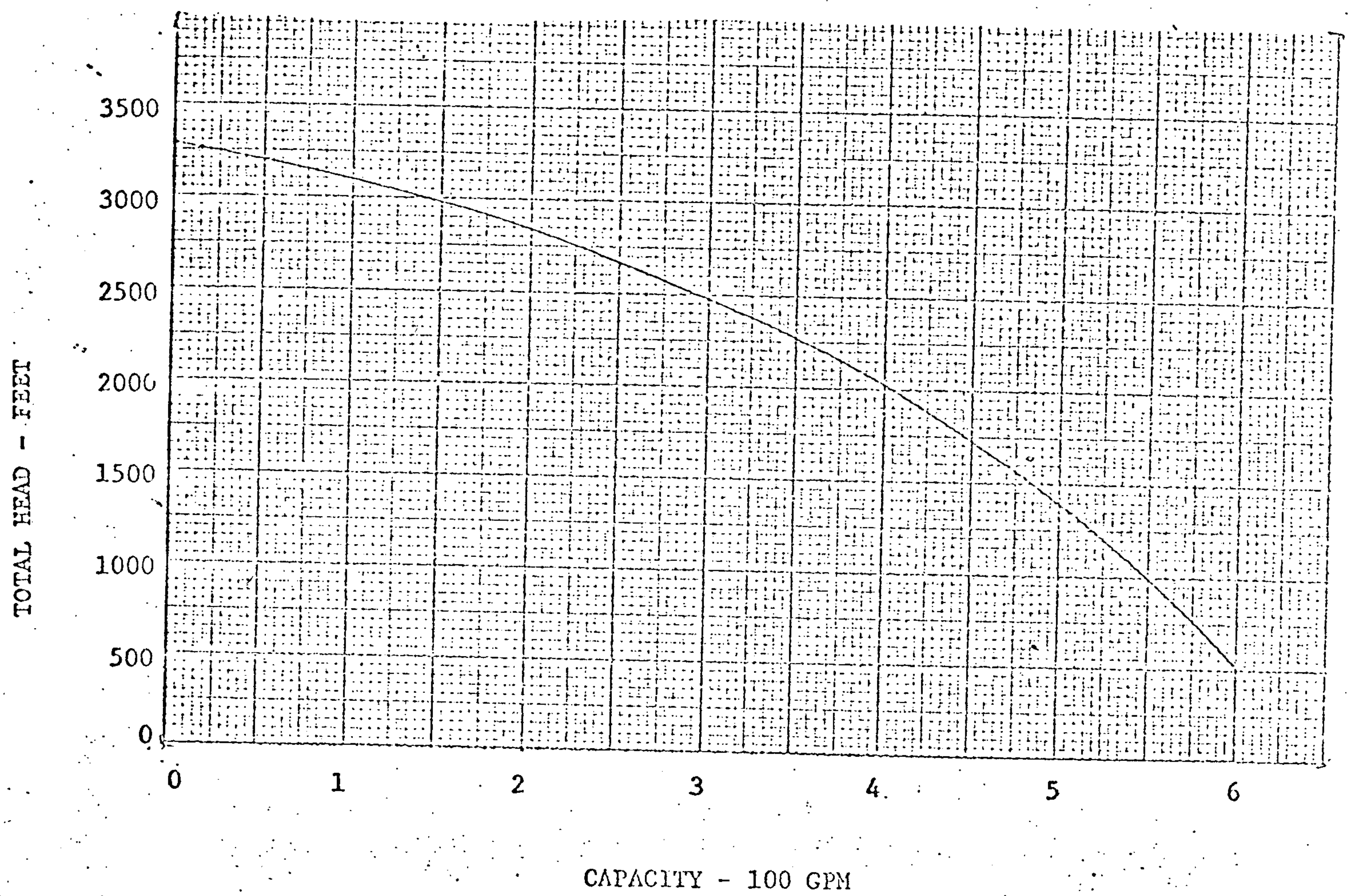

FIGURE 2

SAFETY INJECTION PUMP PARAMETERS

SAFEGUARDS RE-EVALUATION

SAFETY INJECTION PUMP "A"

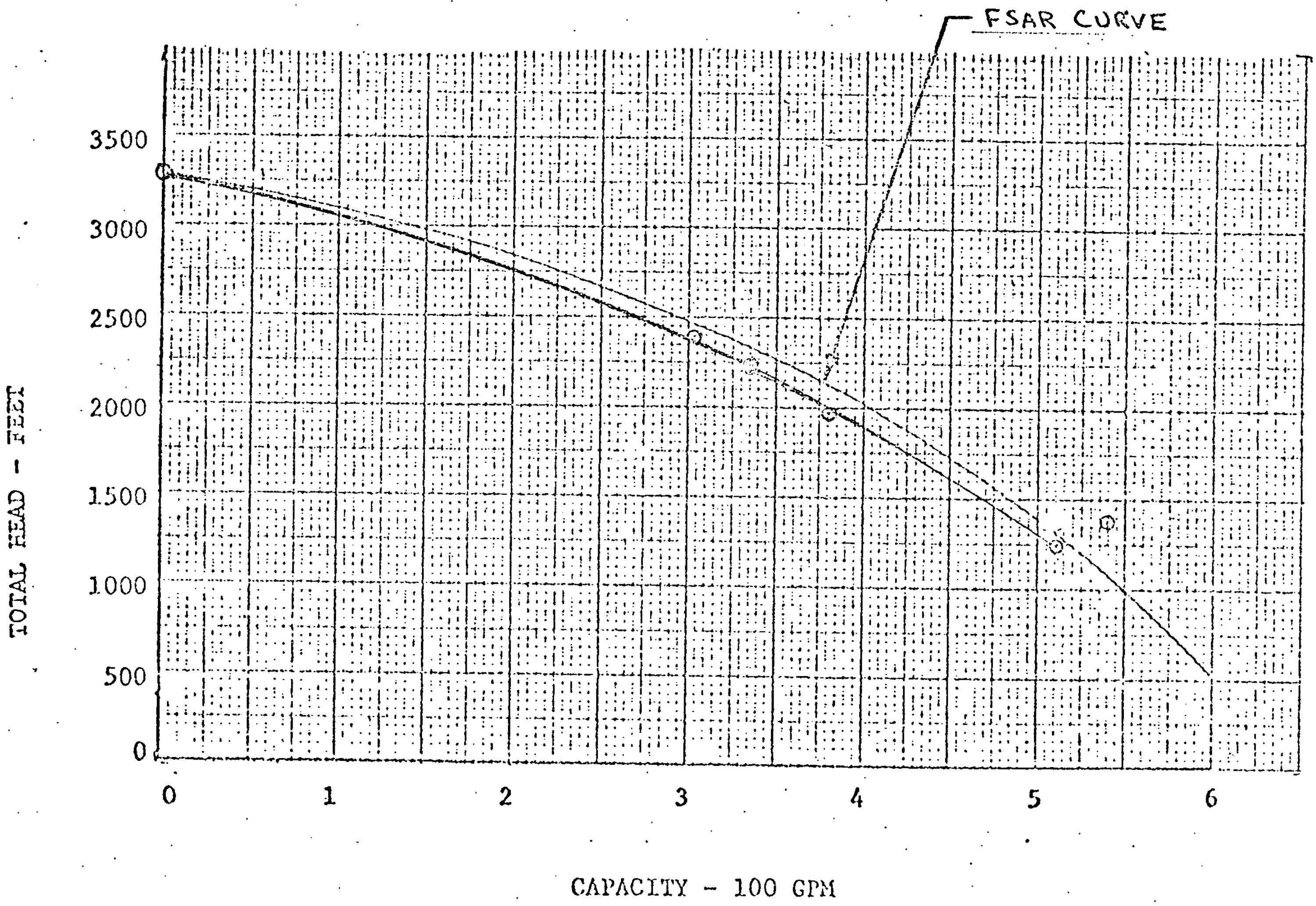

FIGURE 3

SAFETY INJECTION PUMP PARAMETERS

SAFEGUARDS RE-EVALUATION

SAFETY INJECTION PUMP "B"

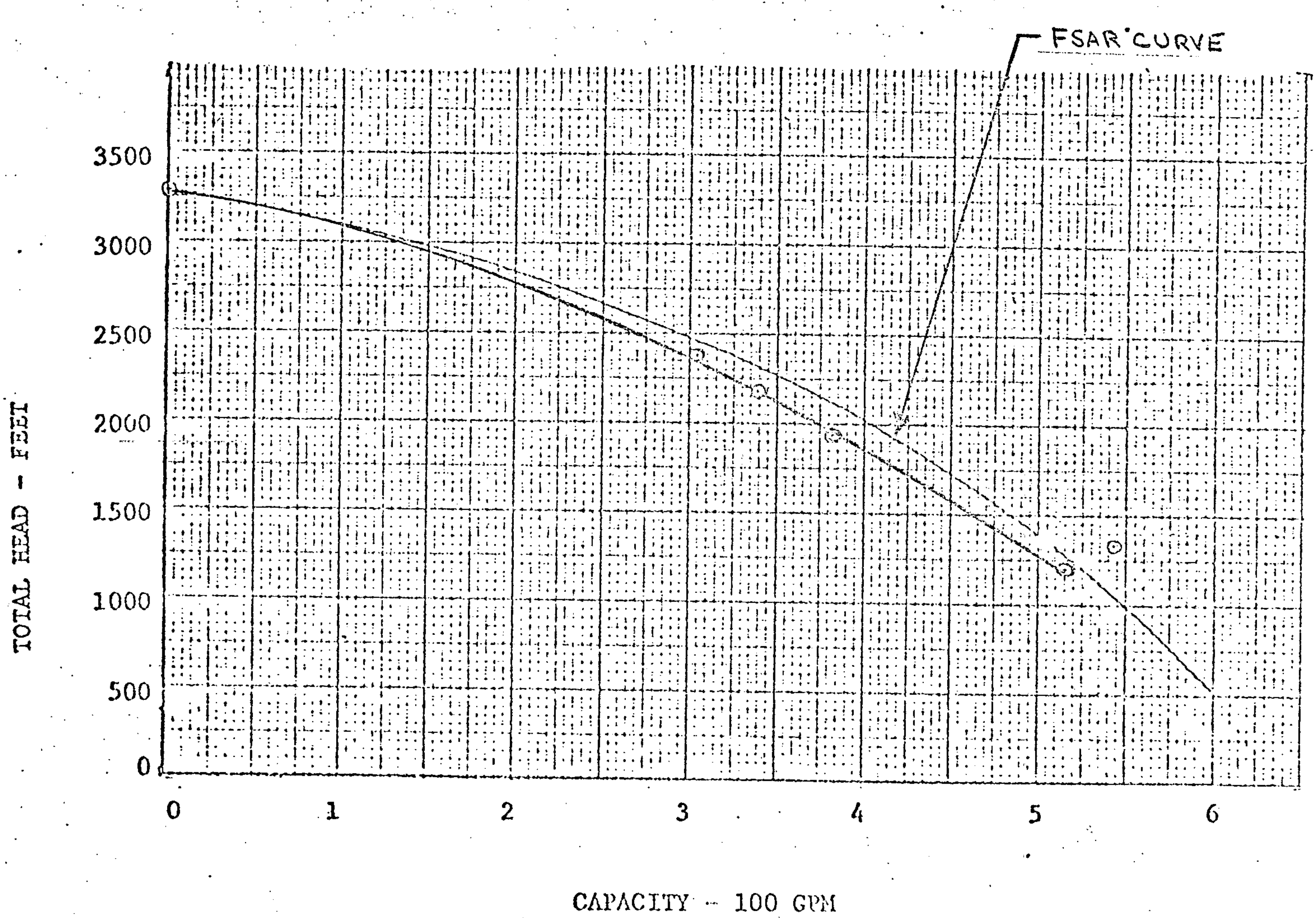

FIGURE 4

SAFETY INJECTION PUMP PARAMETERS

SAFEGUARDS RE-EVALUATION

SAFETY INJECTION PUMP "C"

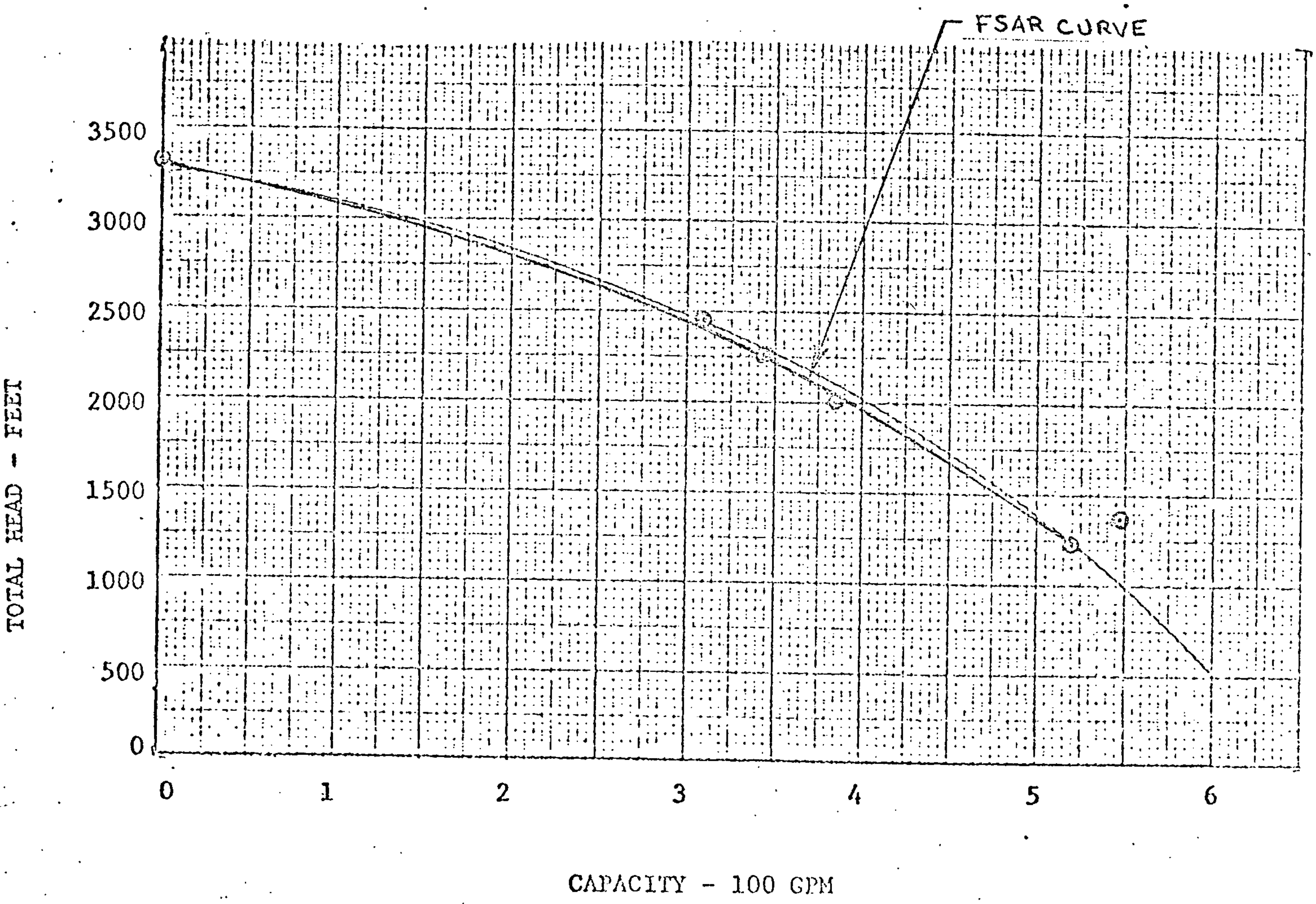

FIGURE 5

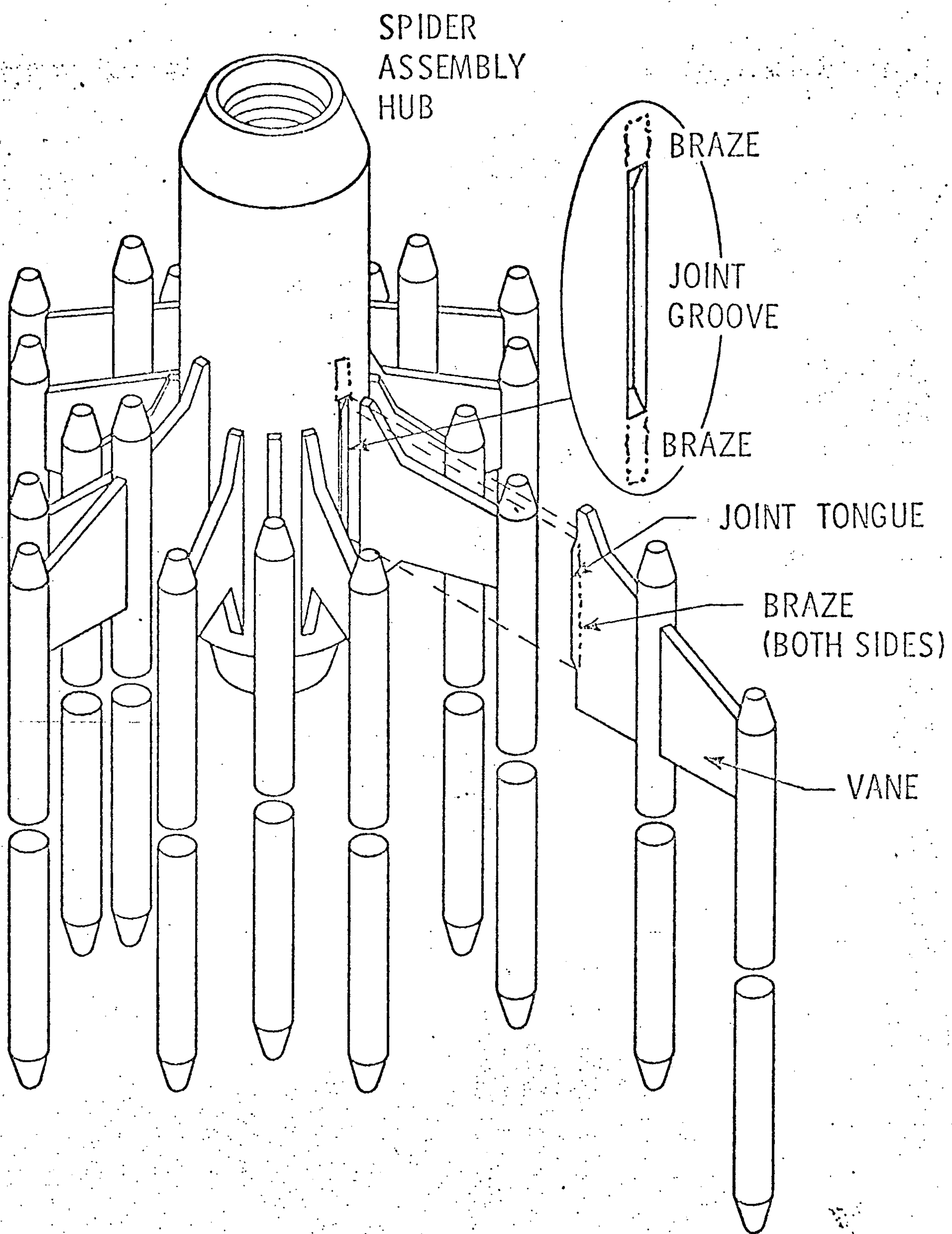

Figure 6. Control Rod Braze Joint Configuration