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FROM: Carolina Power & Light Co. Raleigh, N.C. 27602 E.E. Utley	DATE OF DOC 7-23-75	DATE REC'D 7-24-75	LTR XX	TWX	RPT	OTHER
TO: Mr. George Lear	ORIG 3 signed	CC 37	OTHER	SENT NRC PDR XX SENT LOCAL PDR XX		
CLASS / UNCLASS XXX / PROP INFO	INPUT	NO CYS REC'D 40		DOCKET NO: 50-261		

DESCRIPTION: Ltr re our 5-15-75 ltr.... furnishing info in re to potential for occurrence of secondary system fluid flow instability(water hammer) at H.B. Robinson Unit 2 Plant & trans the following:

Do Not Remove ACKNOWLEDGED

PLANT NAME: H.B. Robinson Unit 2

ENCLOSURES: Figure I indicates the relative position of the feedwater ring within the span of the narrow range level indicator...

Figure II is entitled Westinghouse Nuclear Service Division Tech Bulletin that has Fig. 1 thru 4 re operation of Nuclear Power Plant equipment.....with attached drawing ~~wikix~~ which will aid in draingage characteristics for the m main & auxiliary feedwater piping...(40 cys)

FOR ACTION/INFORMATION DHL 7-25-75

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Regulatory Docket File

July 23, 1975

DOCKETED USNRC JUL 24 1975 MAIL SECTION DOCKET CLERK

FILE: NG-3514 (R)

SERIAL: NG-75-1124

RECEIVED JUL 24 1975 U.S. NUCLEAR REGULATORY COMMISSION Regulatory Mail Section

Mr. George Lear, Chief
Operating Reactors Branch No. 1
Division of Reactor Licensing
Office of Nuclear Reactor Regulations
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

50-261

Dear Mr. Lear:

H. B. ROBINSON UNIT NO. 2
LICENSE NO. DPR-23
RESPONSE TO NRC REQUEST FOR ANALYSIS TO DETERMINE
POTENTIAL OCCURRENCE OF SECONDARY SYSTEM FLUID FLOW INSTABILITY

In response to your request of May 15, 1975 for information regarding the potential for occurrence of secondary system fluid flow instability (water hammer) at H. B. Robinson Unit No. 2, we submit the following information:

The steam generators installed in H. B. Robinson Unit No. 2 are illustrated in Figure I. The figure indicates the relative position of the feedwater ring within the span of the narrow range level indicator. During normal operation the steam generator level is programmed to operate at 52% of narrow range span from 20% to 100% reactor power. Below 20% of rated power, the level is programmed to drop linearly to 39% of narrow range span at zero power. Figure I also illustrates these programmed levels and their relation to feedwater ring position. Table I below summarizes the various setpoints associated with the narrow range steam generator instrument.

TABLE I - STEAM GENERATOR NARROW RANGE LEVEL SETPOINTS

Level Setpoint	Function
75%	High Level Turbine Trip/Feedwater Pump Trip
60%	High Level Alarm
35%	Low Level Alarm
30%	Low Level Concurrent with Steam/Feed Flow Mismatch - Reactor Trip
15%	Low Low Level Alarm/Reactor Trip

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As indicated in Figure I, the lowest level reached during operation before action is initiated (15%) is below the center line position of the feedwater ring (18%). However, this is of no apparent consequence as discussed below.

Operating occurrences likely to cause level, and therefore the steam/water interface, in a steam generator to drop below the feed ring can be divided into two general categories. These categories are: (1) shrink of the generator level as a result of a sudden decrease in boiling activity, and (2) steam flow exceeding feed flow by a significant amount causing the level to boil down. The first could be experienced following a turbine trip from full power. Strip chart recorders for Unit No. 2 have revealed that, subsequent to a turbine trip at full load, the indicated level has dropped momentarily below 18% of narrow range span which is the approximate position of the feedwater ring centerline. The level was then returned to normal using main or auxiliary feed. The latter category, in which the generator simply boils down, could occur as a result of either of two conditions. These are loss of feedwater flow and excessive steam flow. Either could be caused by a number of situations. A loss of feedwater flow has occurred at H. B. Robinson as a result of inadvertent feed regulating valve closure. This loss of flow caused the level in the affected steam generator to drop below the low level alarm setpoint, and with the coincident feed and steam flow mismatch a reactor/turbine trip occurred. As a result a low level condition was compounded by shrink, and the affected generator level went below narrow range span. Again, level was restored using auxiliary feed.

Concerns for the above conditions, which resulted in uncovering the feedwater ring, rest with the potential for water hammer. A water hammer could occur as a result of steam entering an emptied feedwater ring and its associated piping. At H. B. Robinson there is no record of any water hammer experience associated with the feed lines to the steam generators. Some infrequent water hammer has occurred in other portions of the secondary system due to valve malfunctions, but these occurrences were not related to the steam generators. As mentioned above, conditions causing the feed ring to become uncovered have occurred, however, with no apparent water hammer resulting.

This lack of occurrence of water hammer under the above mentioned conditions is attributed to the piping arrangement in the feedwater system. A recent technical bulletin from the steam generator vendor (Westinghouse NSD-TB-75-7) describes the water hammer phenomenon and discusses the probable causes, effects and methods of minimizing occurrence. As a result of a test program conducted by the vendor, observations indicate that the probability of damage from water hammer can be kept to a minimum by the elimination of long horizontal pipe runs at the feedwater inlet nozzle. This minimizes the length of pipe which can drain into the steam generator when the level drops below the feedwater inlet and reduces the magnitude of a water hammer slug, thereby reducing the amplitude of the resulting pressure wave. Figure II was taken from the technical bulletin and illustrates various acceptable piping arrangements as recommended by the vendor. The arrangement at

H. B. Robinson can be best described as a combination of arrangements 1 and 4 of Figure II where dimension "A" is no less than 20 feet and "G" is approximately $3\frac{1}{4}$ feet. Therefore, based upon this technical bulletin, the as-built piping in the feedwater system is in a preferred configuration for the purpose of reducing the effects of water hammer, and no changes to system design are planned.

Because of the favorable operating history and apparent lack of flow instability occurrences, no specific analyses employing dynamic forcing functions have been performed on the feedwater or auxiliary feedwater piping system. A study was initiated subsequent to the Indian Point Unit No. 2 piping failure of 1973 to investigate the probability of a similar occurrence at H. B. Robinson. The findings of that study are reflected in this response. In addition, no test program has ever been initiated regarding the uncovering of the feedwater lines and resulting water hammer occurrence. The initiation of such analyses or test programs to further investigate these flow instabilities are not warranted at this time. However, it has always been a policy at H. B. Robinson to keep abreast of new developments in any field concerning safe operation of the reactor plant, and it is assured that the subject of flow instabilities is no exception.

With respect to a design loss of coolant accident (LOCA), the resulting pressure wave effects from an accompanying water hammer are expected to be of no consequence to the accident analysis. This is demonstrated by the fact that the uncovering of the feedwater ring and inlet nozzle has occurred on occasion as described above but with no apparent formation of a significant pressure wave. Therefore, it is inconceivable that a LOCA would produce conditions appreciably different from these already experienced.

Attached, as requested, are drawings which will aid in the performance of an independent analysis of drainage characteristics for the main and auxiliary feedwater piping. This study and past investigations have revealed no indication of potential flow instability problems.

Yours very truly,

E. E. Utley
Vice President
Bulk Power Supply

JC:dwh

cc: Messrs. N. B. Bessac
P. W. Howe
J. A. Jones
R. E. Jones
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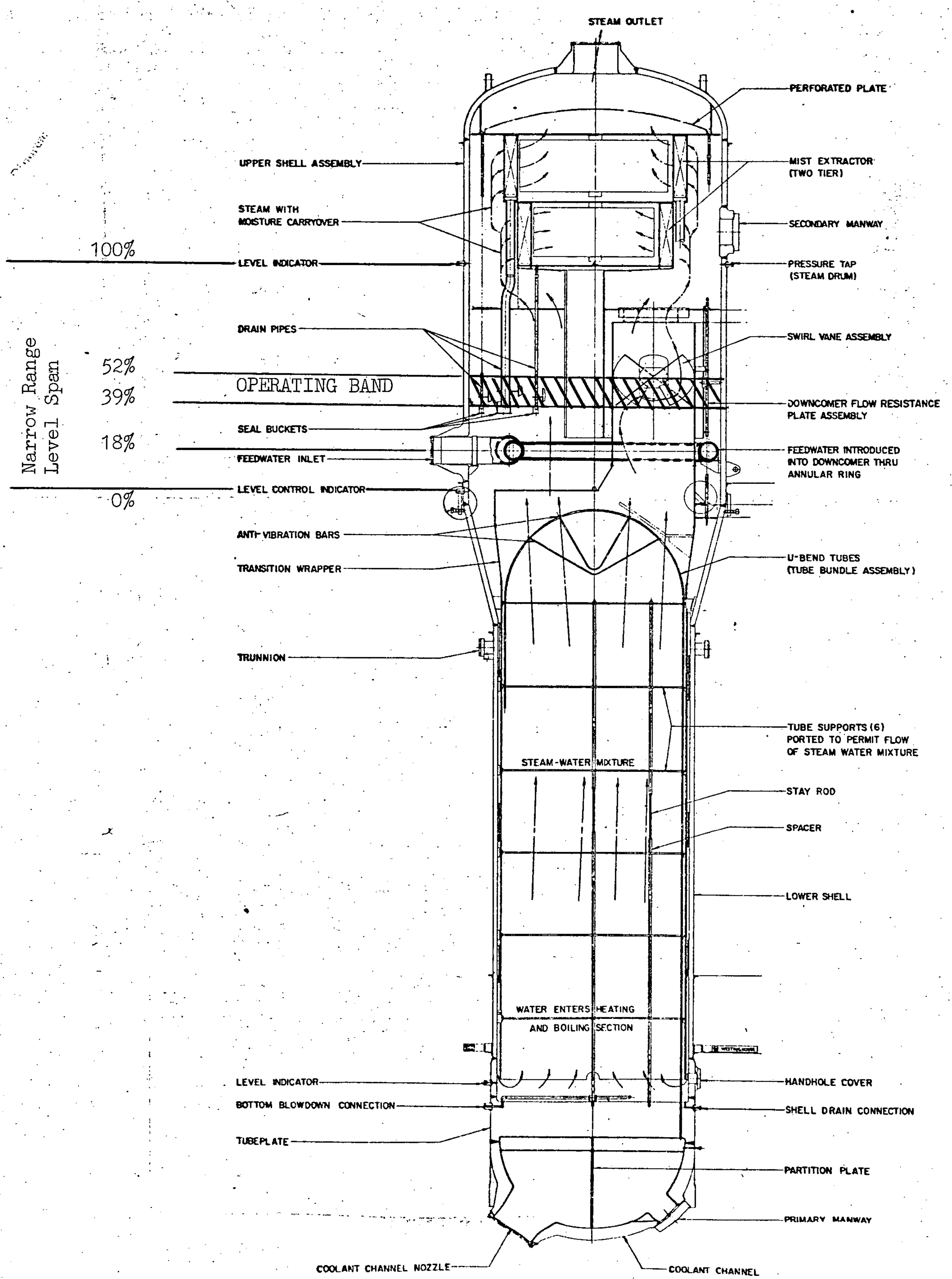


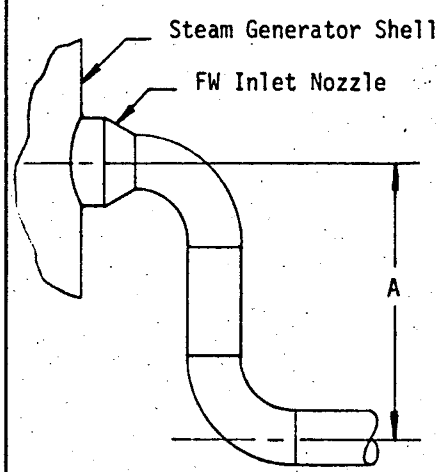
FIGURE [illegible] - Schematic Flow Diagram

P.O. Box 2728, Pittsburgh, PA 15230

SKETCH SHEET
FORM 28577

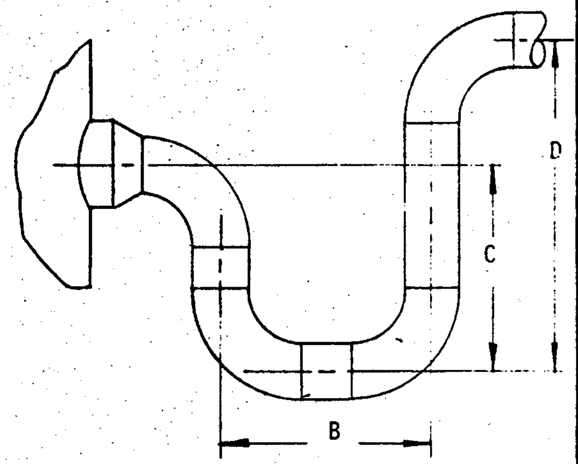
Attachment to NSD-TB-75-7

WESTINGHOUSE ELECTRIC CORPORATION



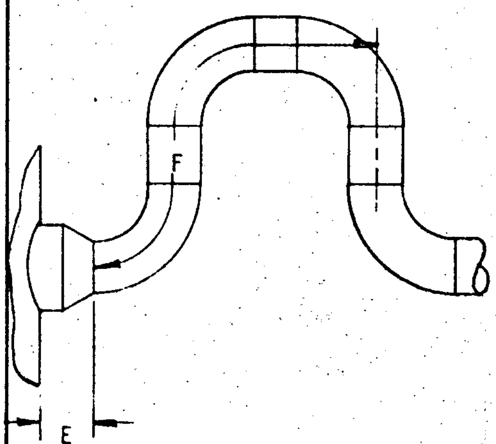
Preferred arrangement.
"A" - to be not less than 24 inches.

Figure 1



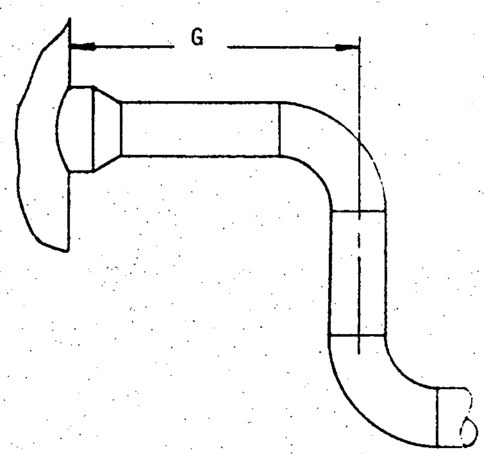
Recommended when arrangement of Figure 1 cannot be attained.
"C" - to be not less than 36 inches.
"B" & "D" - no requirement established.

Figure 2



Recommended only when arrangements of Figures 1 & 2 cannot be attained.
"E" + "F" not to exceed 8 feet.

Figure 3

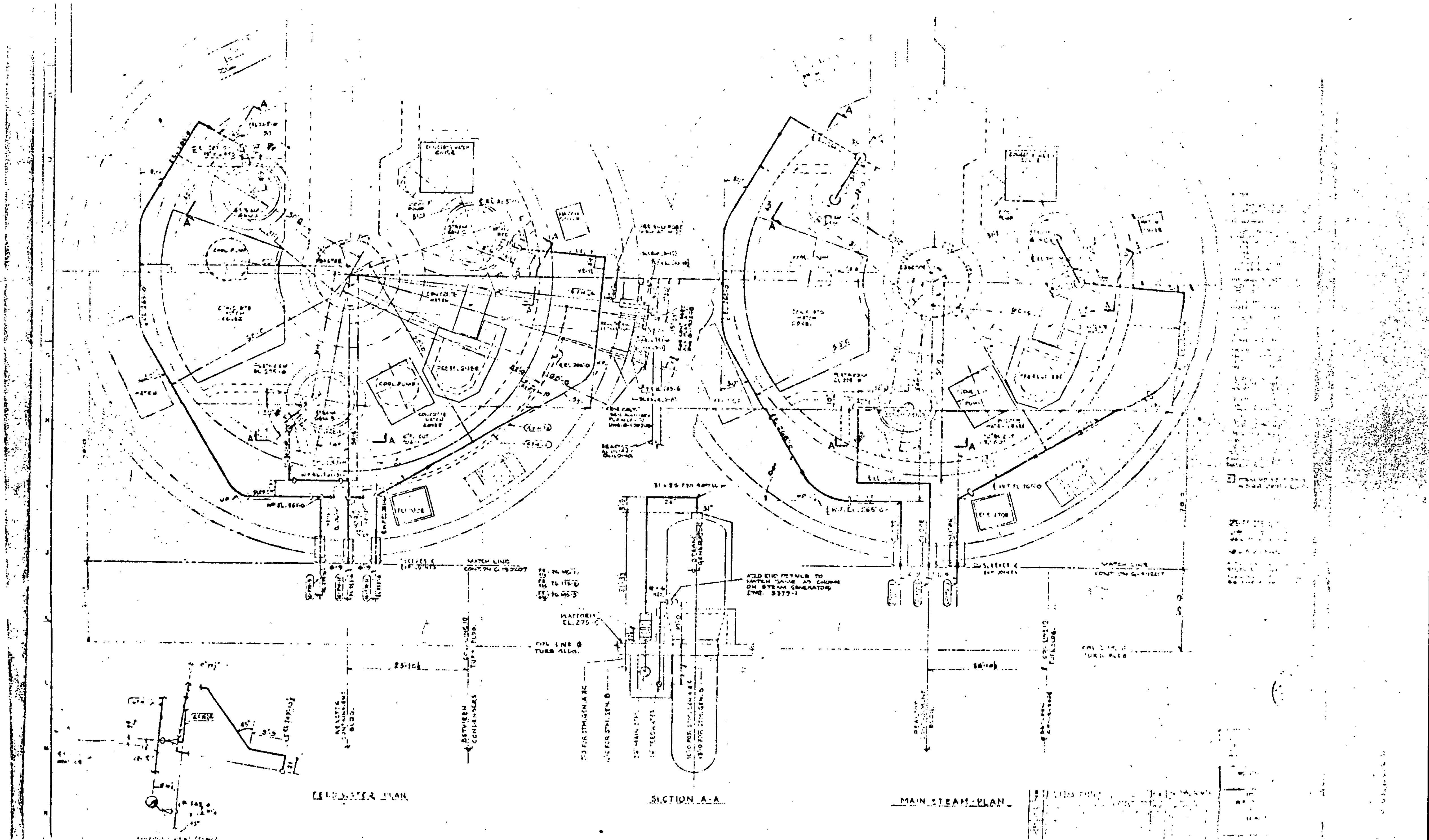


Similar to Figure 1.
"G" - to be minimum dimension.
Not to exceed 8 feet.

Figure 4

SSE-1257

FIGURE II


FEEDWATER PLAN
SECTION A-A
MAIN STEAM PLAN
BETWEEN CONDENSERS