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AUTH.NAME AUTHOR AFFILIATION
UTLEY,E.E. Carolina Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
SCHWENCER,A. Operating Reactors Branch 1

SUBJECT: Forwards info re potential for steam generator water hammer at PWRs w/feedrings that discharge from bottom,in response to NRC 790918 request.

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TITLE: Steam Generator Feedwater Flow Instability (Water Hammer

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November 16, 1979

FILE: NG-3514(R) SERIAL NO.: GD-79-2945

Office of Nuclear Reactor Regulation
Attention: Mr. Albert Schwencer, Chief
Operating Reactor Branch No. 1
United States Nuclear Regulatory Commission
Washington, D.C. 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261
LICENSE NO. DPR-23
RESPONSE TO REQUEST FOR INFORMATION - STEAM GENERATOR WATER HAMMER

Dear Mr. Schwencer:

In response to your request of September 18, 1979, for information regarding the potential for steam generator water hammer at pressurized water reactors with feedrings that discharge from the bottom, Carolina Power & Light Company submits the enclosed information.

If you have any questions on this subject, do not hesitate to call.

Yours very truly,

E. E. Utley

E. E. Utley
Executive Vice President
Power Supply & Customer Services

EEU/jcb

Enclosure

A012 5/1/1

411 Fayetteville Street • P. O. Box 1551 • Raleigh, N. C. 27602

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P 7911230

NRC Comment 1.1

Describe the expected behavior of steam generator water level as a result of reactor trip from power levels greater than 30% of full power. Include actual plant measurements of steam generator level and other available related data such as feedwater flow and auxiliary feedwater flow.

Response

See Attachments 1, 2 and 3.

NRC Comment 1.2

Provide the number and causes of loss of feedwater events during the operational history of the plant. You may refer to material submitted previously.

Response

The following is a list of the reactor trips caused by loss of feedwater or related occurrences by date, event, cause and power level.

			POWER LEVEL	
DATE	EVENT	CAUSE	MWe	% THERMAL
9/26/70	Turbine & Rx Trip	Loss of mainfeed pump due to low suction pressure	0	12%
9/26/70	Same as above	Same as above	Same	
9/29/70	Rx Trip	Lo Level + SF/FF mismatch "B" SG	100	24%
10/1/70	Rx Trip	Same as 9/26/70	149	31%
10/29/70	Rx Trip	Lo-level & SF/FF mismatch "C" S/G Cycling MSIV	Normal	
10/25/70	Rx Trip	Lo-lo level C-SG-Loss of Main Feed Pump	0	5%
11/26/70	Rx Trip	"C" SG Lo-level with SF/FF mismatch Loss of Power -- Loss of main feed pump	52	18%
1/12/71	Rx Trip	Lo-lo "A" SG level	30	12%
1/28/71	Rx Trip	Lo level & SF/FF mismatch frozen reg. controller	452	71%

DATE	EVENT	CAUSE	POWER LEVEL MWe	POWER LEVEL % THERMAL
2/28/71	Rx Trip	Lo-lo level in "A" SG Loss of feed reg. valve linkage problem	735	99%
3/11/71	Rx Trip	"B" SG Lo-level & SF/FF mismatch loss of reg valve due to I&C work being performed	715	100%
1/16/72	Rx Trip	Lo-lo SG Level - Due to failure of AFW discharge valve to open. Level being controlled by AFW pump.	0	
1/16/72	Rx Trip	Same	0	
7/7/72	Rx Trip	Lo Level - SF/FF mismatch Loss of feedpump	670	99.9%
7/8/72	Rx Trip	Lo-Lo level stuck reg valve	50	12%
9/26/72	Rx Trip	Lo level with SF/FF mismatch "B" SG - Reg. Block valve would not open	80	21%
8/26/73	Rx Trip	Same as 9/26/72	0	3%
4/1/75	Rx Trip	Lo level & SF/FF mismatch Closure of "A" FW Reg. Valve	730	100%
4/1/75	Rx Trip	Same as above - Broken wire on solenoid valve	720	100%
4/5/75	Rx Trip	Lo-Lo Level due to I/P converter failure on S/G Reg. Valve	440	60%
5/30/75	Rx Trip	SG "B" Lo level & SF/FF due to Inst. Air loss	711	100%
8/27/75	Rx Trip	SG "B" Lo level & SF/FF - Lost "S" condensate & "B" feed pump	649	100%
12/17/75	Rx Trip	Lo-Lo level SG #3 loss of "A" condensate pump due to faulty switch	600	89%

DATE	EVENT	CAUSE	POWER LEVEL MWe	POWER LEVEL % THERMAL
10/2/76	Rx Trip	SF/FF mismatch to level loss of feed pump due to faulty temperature indication	315	80%
1/3/78	Rx Trip	Lo-Lo level on "C" S/B - MSIV closed Bad Solenoid	180	100%
10/16/78	Rx Trip	"C" S/G Low level with SF/FF mismatch	682	100%
6/3/79	Rx Trip	"B" S/G SF/FF mismatch with low level - shutdown trip		

NRC Comment 1.3

Provide the number and causes of loss of off-site power events during the operational history of the plant.

Response

There have been two occurrences which constitute partial losses of on-site power. No total losses of off-site power have occurred.

These two instances where a partial loss of on-site power occurred are described in the following reports previously submitted:

June 4, 1975, Letter to Mr. Norman C. Moseley from Mr. E. E. Utley
SUBJECT: IE Inspection Report No. 50-261/75-1 Item I.B.

INCIDENT REPORT NO. 21, Low DC Voltage "B" Battery - Reactor Trip, March 14, 1971 - April 7, 1971 Letter to Dr. Peter A. Morris from Mr. E. E. Utley.

NRC Comment 2

If administrative controls have been adopted to limit the flow of auxiliary feedwater for the purpose of reducing the probability of water hammer, show when they were adopted and give the answers to Items 1.1, 1.2, and 1.3 for before and after such controls were established.

Response

No administrative controls to limit flowrates to prevent water hammer exist.

NRC Comment 3

If administrative controls have been adopted to limit the flow of auxiliary feedwater for the purpose of reducing the probability of water hammer, show that an adequate water inventory and flow will be maintained to accommodate all postulated transient and accident conditions.

Response

No such administrative controls exist as identified in response to Comment 2.

NRC Comment 4

If auxiliary feedwater flow in your facility is not at present initiated automatically for normal and accident event, present your evaluation of whether automating the actuation of auxiliary feedwater might increase the probability of inducing steam generator water hammer. One of the signals that would automatically initiate the flow of auxiliary feedwater would be the steam generator low water level. This set point should be above the top of the main feedwater sparger to reduce the probability of steam generator water hammer.

Response

The auxiliary feedwater system is automatically initiated at present from the following sources:

1. All "SI" (safety injection) signals.

2. Loss of AC power or undervoltage providing there is no "SI" signal to E1 and/or E2 emergency busses (black-out sequence).

3. Trip of both main feedwater pumps and lo-lo level in one of three steam generators initiates automatic starting of both motor driven pumps, trip of both main feed pumps and lo-lo level in two of three steam generators initiates an automatic start of the steam driven pump.

NRC Comment 5

Describe the means that will be used to monitor for the occurrence of steam generator water hammer and possible damage from such an event. Include all instrumentation that will be employed. Describe the inspections that will be performed and give the frequency of such inspections.

Response

Plant Modification #490 provided instrumentation for two feedwater lines, as previously described in Section 7.0 of the July 10, 1979 letter to Mr. Albert Schwencer from Mr. E. E. Utley. Data gathering has been completed and no further action is planned. Data collected during a reactor trip from 30% power was included with no apparent evidence of water hammer observed.

NRC Comment 6

Describe the reporting procedures that will be used to document and report water hammer and damage to piping and piping support systems. Such reports were requested in our letter to you dated September 2, 1977.

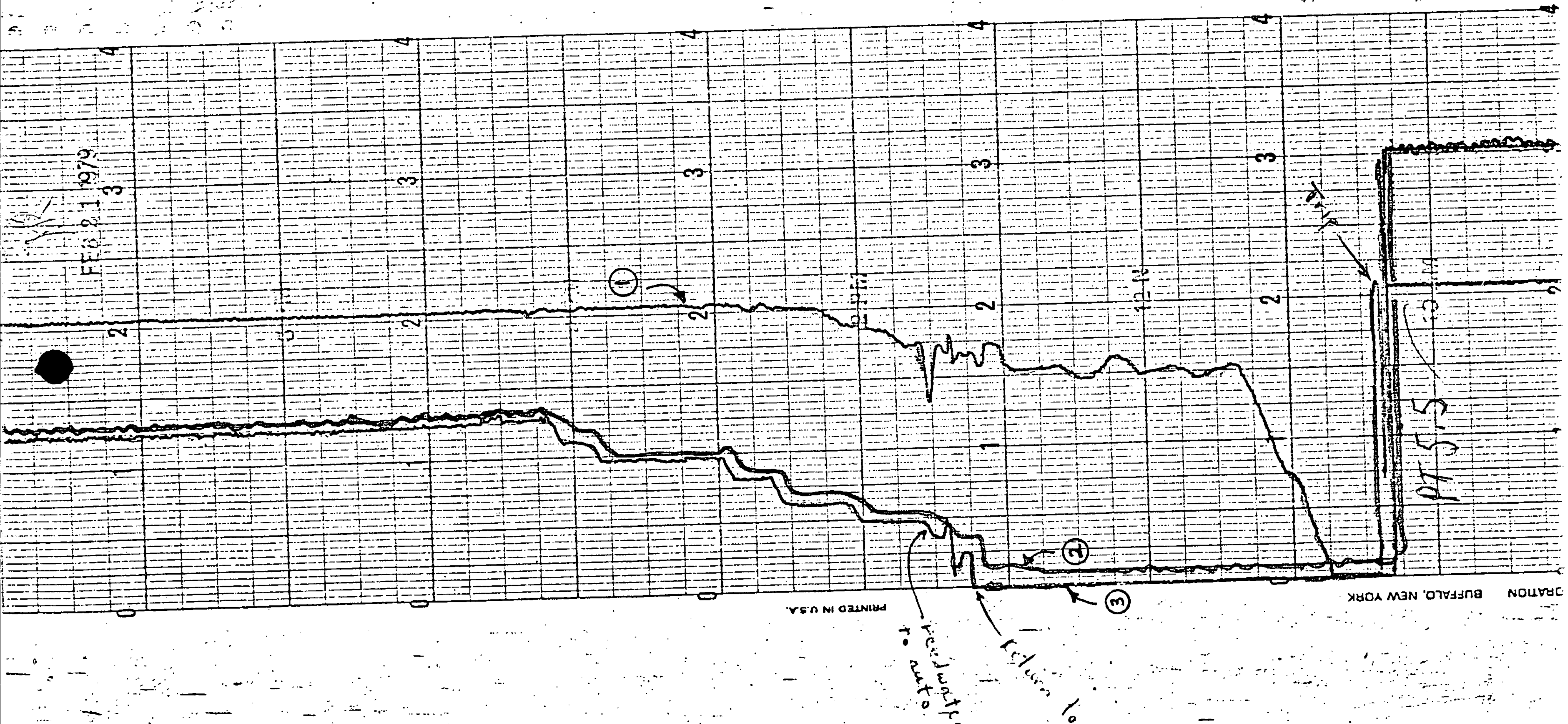
Response

As required by HBR Technical Specification 6.9.2.a(7) if, as a direct result of an occurrence of water hammer, a plant shutdown is required a report shall be issued by telephone within 24 hours and later confirmed by written correspondence no later than the first working day following the event with a written follow-up report issued within two weeks. Otherwise, an incident of water hammer will constitute an operating experience and will be handled as such.

ATTACHMENT 1

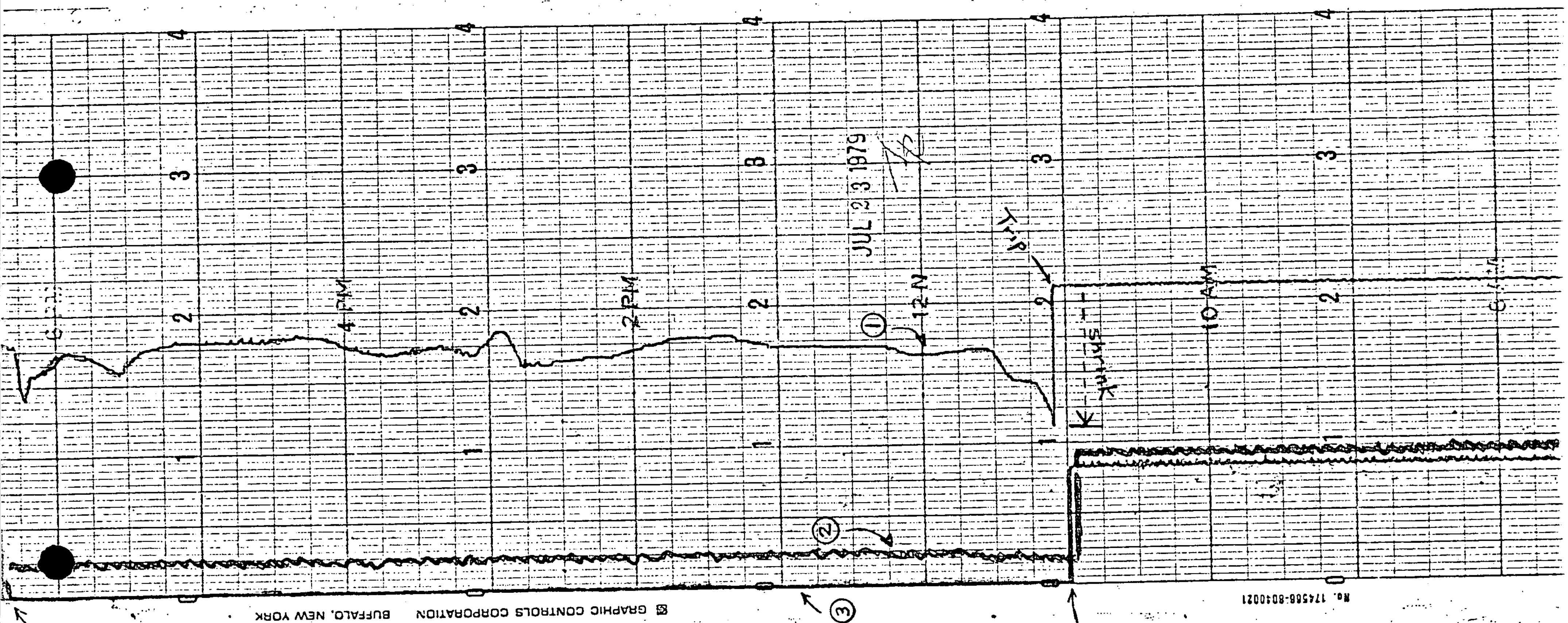
TRIP FROM 100% Rx Power at 1029 on 2-21-79

① ——— 'A' S/G NARROW RANGE LEVEL (0-4 ON SCALE = 0-100%)
② ——— 'A' S/G STEAM FLOW (0-4 ON SCALE = 0-4 lbs./HR.x10^6)
③ ——— 'A' S/G FEED FLOW (SAME AS STEAM FLOW)



TRIP from 30% Rx Power - at 1103 on 7/23/79

ATTACHMENT 2



(0-4 ON SCALE = 0-100%)	①	= 'A' S/G narrow range level
(0-4 ON SCALE = 0-4 lbs/HR x10^6)	②	'A' S/G steam flow
(SAME AS STEAM FLOW)	③	A S/G main feed flow

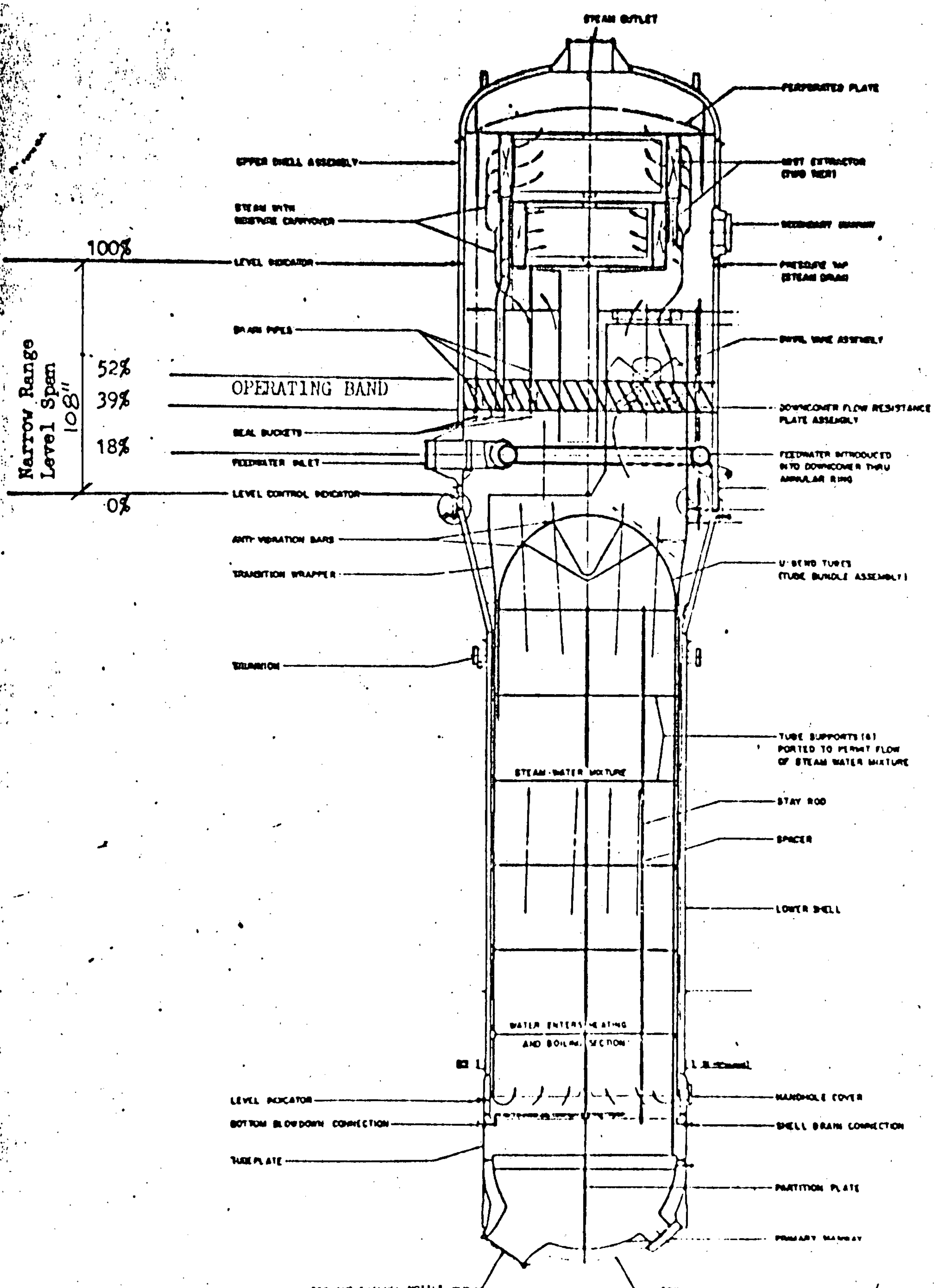


FIGURE 2 - Schematic Flow Diagram

ATTACHMENT 3