

B + W 205 - Decay Heat

Removal after small-break

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1. Introduction

The Nuclear Systems Analysis Section recently conducted a qualitative examination of the decay heat removal problem associated with recovery of a Combustion Engineering System 80 pressurized water reactor (PWR) plant from a very small break loss-of-coolant accident (LOCA). A very small break LOCA is one for which the steam generator must remove a significant portion of the decay heat; otherwise, reactor coolant system depressurization occurs since the break is too small to facilitate the transport of all decay heat to the surcore. In this class of LOCA, depressurization rates are relatively slow (when compared to those normally analyzed as small breaks) and there might lead to inadequate makeup rates from the high pressure injection pump. An ongoing qualitative consideration of this concern

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now presents a similar decay heat removal problem during post-LOCA recovery which needs to be resolved for a Babcock & Wilcox 205-Gen-Assembly PWR such as the Ballefonte Nuclear Plant. This problem and its associated background material are the subject of this report.

2. Physical Arrangement and Characteristics

The physical arrangement of the reactor coolant system for a typical 205-Gen-Assembly plant such as Ballefonte is shown in Figure 1. The plant has two pass-through steam generators each with a single hot leg piping connection (33 inch) at the top and two cold leg piping connections (32 inch) at the bottom. A reactor coolant pump is provided in each cold leg. The pressurizer is attached to one of the hot leg pipes by a 24 foot long swing line (14 inch).

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The reactor vessel has two corresponding hot leg piping manifolds and four cold leg manifolds.

The plant elevations corresponding to various points in the reactor coolant system are given in Table 2. The reactor coolant system characteristics of interest are given in Table 2.

3. Predicted Post-LOCA Modes of Decay Heat Removal

The conditions predicted to be associated with each quasi-steady-state post-LOCA mode of decay heat removal are detailed in Appendix A. These conditions are for a very small break (initially $< 0.05 \text{ ft}^2$) for which the steam generator cannot remove a significant portion of the decay heat; otherwise, reactor coolant system repressurization occurs since the break is too small to accommodate the transport of all decay heat to the generators. Tables 1-6

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as well as on loss of H_2 set point, as
primary action, maximum core cooling (core
temp) response using cold leg injection,
isolation of main steam line attachments, and
steam generator at saturation conditions corresponding
to the lowest safety valve set point.

Modes 1-7 given in Appendix A define
the conditions which are thought to exist in
the reactor coolant system to provide for decay
heat removal during the relatively slow
changing post-loss period. These conditions
are summarized in Table 3.

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The retention of water in the condenser is a function of the transition from natural circulation to convection during Mode 3 may be a problem because of the time delay incurred while waiting for the water level in the U-tube region of each hot leg to rise and in the steam generator tubes to drain to below the secondary side water level. In very small tubes, this could take several minutes during which time there is no significant removal of heat from the system. In addition, for certain operating situations, the initial secondary side water level may be very low thus requiring additional time before the water level can communicate with an effective condensing surface.

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An additional concern associated with Mode 3 is the possibility of quick isolation by operator action, thereby postponing indefinitely the transition to decay heat removal by Mode 4 without assurance that natural circulation or some other effective means of decay heat removal can be re-established in time.

One alternative to natural circulation might be full depressurization and subsequent two-phase fluid release through the pressure relief valves. However, the available full pressure working capacity may not be commensurate with the short term peak rate required to remove all of the decay heat through the valves. In addition, the relief valves have not been qualified for this severe service. Another alternative might be to open the pressure relief valve

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to replace the isolated tank with a new path for decay heat removal. However, only one relief valve is normally provided and it is not qualified for this severe service as suggested with on-site power or safety-grade controls.

Operation in Mode 4 appears reasonable to achieve although the master level will be unaware as to what is happening in the master vessel level. Also, the presence of a pressure level is not an indication that adequate core coverage is being achieved.

Operation in Mode 5 may be uncertain. The onset of Mode 5 cannot be predicted or easily controlled since there is no level information available to the operator. The refueling of the reactor that the pump will disrupt normal steam flow from the core to the

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steam generator tubes are well head to an
intermittent rising of the hot leg piping as the
steam is forced through. However, these
expansion should be readily accommodated in
the vertical hot leg pipes. It appears
reasonable that adequate decay heat transport
will be achieved during this portion of the
transition period. The real concern will
develop when refilling of the vertical hot leg
pipes increases the cold leg cross section in
the steam generator tubes to above the
secondary side water level. At this time, decay
heat removal by steam condensation inside the
tubes will ~~cease~~ and the system will start
to depressurize. ^{Not used but be substantially}
^{depressed} ~~the~~
In order to re-establish natural circulation by
decay heat removal as high as Mod 6, it is
necessary that ~~the~~ ^{the} ~~5~~ ⁵ be terminated
refilling the ~~5~~ ⁵ ~~leg~~ ^{leg} of the hot leg

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spring with liquid and establishing the density gradient required for natural circulation. It is not clear how this can be accomplished. There could be a problem with significant accumulation of non-condensable gases in the U-tube region. If enough non-condensable gases have accumulated, natural circulation cannot be re-established (unless the gases can be removed by some means).

If this cannot be established, it will be necessary to assume that the water level inside the steam generator tubes can be re-established at a suitable level below that on the secondary side in order to provide for continuation of buoy heat removal.

Whether the primary side may have to be terminated is such a questionable question in view of the absence of level information. Furthermore, since the lithium system performs

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one safety function, it is in single track configuration and is operated only in off-site power. If the track was isolated earlier, it is not clear that the required section can be achieved. If the track has not been isolated, it may be able to provide the required section in time. Although valves have been provided for a direct drain to the sump to control post-LOCA foam buildup, these valves are not designed to operate at the high pressure associated with the very small track LOCA. If decay heat removal cannot be achieved, full vaporization and subsequent two-phase flow release through the pressure relief valve will occur. If makeup has been terminated, this operation will result in a loss of liquid inventory and eventual establishment of an appropriate liquid level inside the

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Steam generator tubes for decay heat removal.
If makeup has not been terminated, the safety
relief will continue to be subjected to two-phase
or slug flow to achieve decay heat removal. The
safety valves have not been qualified for this
service.

4. Accident Investigation Results

Information concerning a Babcock &
Wilcox analysis for the small break ECCS
for a B3W 225-Fuel-Assembly plant is
included as Appendix B. This analysis
is for a 2.25 square foot pump discharge
break which is near the upper size limit
for a very small break break but at the
lower calculation limit for the ECCS analysis
of small breaks. It is based on the usual
ECCS evaluation rules.

The results shown in Figure 4-34 of

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Appendix E documents that the reactor vessel level will drop to the top of the core within 10 minutes following the break, thus leading into the predicted eventual occurrence of the first step modes of operation outlined in Appendix A. The results show that the core will remain virtually erect at all times. The NRC evaluation of the BSW small break LOCA analysis is included as Appendix C.

5. Areas of concern

The transition from natural circulation to core boiling during Mode 3 is of concern because of the time delay involved while waiting for the primary side water level inside the steam generator tubes to drop so below the secondary side water level and thereby establish a primary side

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concerns surface for area heat removal. It would be an added concern if the joints along side the tank 3.2 thereby disrupt a timely achievement of the required primary side draw.

The transition from core boiling to natural circulation during Mode 5 is also a concern because reflooding of the reactor coolant system will eventually increase the steam generator tube level to above the secondary side water level. At that time, core heat removal will be drastically reduced. It is not clear that natural circulation can be re-established by continuing the reflooding process, particularly if large amounts of non-condensable gases are present. If natural circulation is not re-established, it will be necessary to reduce the water level inside the steam generator tubes to below the secondary side water level to re-establish

in condensing surface. If this well requires
reworking in terminating the makeup (as
mentioned idea) and waiting for the water
to drain through the break. Again, it would
be an additional concern if the operator should
release the break thereby disrupting a timely
achievement of the required primary side drain.

Adding to these concerns is the
uncertainty associated with unknown vessel
level, the adequacy of emergency operating
instructions and operator training for the event,
and the consequences of the unstable slug
flow conditions which are predicted to
develop in the piping and safety valves as a
consequence of certain operating situations. These
very small break LOCA considerations appear to
be generic for pressurized water reactors although
it may be more severe for the BWR 205-Gen-2 design.
Next item of the one-through steam generator configuration.

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3. Effects of pressure

A loss of effects power was assumed in preparing this report. If the very small heat should not be recognized by an effects power loss, the effect of continued operation of the water coolant pumps would need to be evaluated. The loss of system pressure at a high average water coolant temperature might lead to unacceptable pump or seal damage if the operator does not manually trip the pumps within a few minutes. Physical damage might be induced by excessive vibration or severe flow instability. In addition, it is not clear what effect the flow instability will have on the maintenance of an adequate core flow of decay heat removal. The large heat output of the pumps might cause steam binding in the pump and a resulting

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from interception. These search warrant
giving coverage were not investigated.

The very small peak indicated in the
report was assumed to be located at the top
of a hill by GPS. Other peak locations were
not considered in detail. However, the peak
location is not thought to be a major influence
on the distance of the predicted meeting node
given in Appendix 2, i.e., there is no
one best way and location which will
result in the occurrence of each predicted
node.

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No. 1

Reactor Calant System Elevations

Component	Elevation (ft)
Pressurizer: Vessel High Point (inside)	659
Vessel Low Point (inside)	630
Hot Leg Connection @	643
Full Power Water Level	652
After Steam Water Level	645/634
Top of Heaters	641
Horizontal Surge Pipe @	625

Steam Generator: Top Tubesheet (inside face)	639
Bottom Tubesheet (inside face)	634

Reactor Vessel: Vessel High Point (inside)	645
Hot Leg Nozzle (high point)	633.6
Hot Leg Nozzle (low point)	630.4
Active Fuel (high point)	627.5
Active Fuel (low point)	615.5

3. Reactor Vessel

4. Reactor Vessel to Reactor

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Table 2

Reactor Isolation System Characteristics

Full Core Temperature (°F):	Core Entry	572
(3600 Mw)	Core Midplane	601
	Core Exit	630
	Pressurizer liquid	650
	SG Steam	603
	SG Feedwater	473
Full Pipe Pressures (psia):	RC Pump Discharge	2267
	Core Midplane	2235
	Core Exit	2205
	SG Steam	1075
Safety Valve Setting (psia):	Pressurizer	2500
	Steam Generator	1250/1225
Pressures (H ₂ O):	Pressurizer (Steam space)	1050
	Pressurizer (above reactor)	1200
	Pressurizer (total)	2250
	Reactor Vessel (above 1st row)	1550*
Actuation Pressures (psia):	Safety Injection Pump	1600
	Core Flooding Tank	600
	Reactor Steam	1945

* Safety Valve 12.50

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Table 3Post-Loop Modes of Decay Heat Removal

Mode	Condition	Press. Level	RV Level	System Press.	SG Tube Level
1	Natural Circulation ^b	Decr. P.	Full	Decr. P.	Full
2	Natural Circulation ^c	Incr. S	Decr. S ^d	Stable	Full
3	Transition	Incr. S	Decr. S ^e	Stable	Decr. S
4	Core Boiling	Stable	Decr. S ^f	Stable	Decr. S ^g
5	Transition	Incr. S ^h	Incr. S ⁱ	Decr. S	Incr. S
6	Natural Circulation	Full	Incr. S	Decr. S	Full
7	Forced Circulation	Full	Stable	Decr. S	Full

a. Nomenclature: Press. Level is pressurizer level.
 RV Level is reactor vessel level.
 System Press. is system pressure.
 SG Tube Level is level inside steam generator tubes.
 Decr. P. is decreasing rapidly.
 Decr. S is decreasing slowly.
 Incr. S is increasing slowly.
 Stable is no significant change.

b. With pressurizer pressure controlling.

c. With reactor vessel pressure controlling.

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1. Reactor vessel steam bubble has formed.

2. Reactor vessel level has reached top of hot leg nozzles.

3. While vertical hot leg pipes are empty.

4. Reactor vessel and steam generator tube level will reach a minimum and start to increase.

5. While vertical hot leg pipes are filling.

6. Stable after reactor vessel level has reached top of hot leg nozzles.

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Appendix A

Conditions Associated With Post-LOCA
Modes of Decay Heat Removal
For Very Small Break LOCA^{*2}

* Numbered notes are included at end of appendix

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Slide 1²Decay Heat Removal by Natural Circulation³With Pressurizer Pressure Controlling

- a. Pressurizer steam bubble pressure is above saturation pressure corresponding to reactor vessel top plenum effective liquid temperature and is superheated to pressurizer effective metal temperature.
- b. Pressurizer level and pressure are decreasing rapidly (steam bubble expanding) due to fluid loss through the break in excess of makeup capability.
- c. Pressurizer liquid is partially flashing to steam (thrusty cooling) in an attempt to maintain constant pressure as the steam bubble expands.

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d. Pressurizer steam bubble is superheating the pressurizer effective metal temperature thereby cooling metal.

e. Without an effective heat input other than from the pressurizer metal to compensate for the heat removed by flashing the pressurizer liquid and heat loss to the surrons, the pressurizer pressure is decreasing rapidly as the steam bubble expands.

f. Pressurizer steam bubble may expand until superheated steam is cooling and condensing in the circulating liquid at pressurizer connection to the hot leg piping.

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Mode 2

Decay Heat Removal by Natural Circulation³ With Reactor Vessel Pressure Controlling

- a. Steam bubble forms in reactor vessel plenum when pressurizer pressure decreases to below the saturation pressure corresponding to top plenum liquid temperature.
- b. Reactor vessel level starts decreasing (top plenum steam bubble expanding) due to fluid loss through the break in excess of makeup capability.
- c. Reactor vessel top plenum liquid is partially flashing to steam (chugging) in an attempt to maintain constant pressure as the steam bubble expands.
- d. Decay heat is providing an effective heat input

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to compensate for the heat removal by flashing the top glaucous liquid, thereby ensuring a liquid temperature and corresponding steam bubble pressure which follows closely the slowly changing effects of decay heat reduction, heat loss to the enclosure, and heat transport through the walls.

e. Pressurizer surge line is filled with non-quiescent liquid which is subcooled to below the reactor vessel top glaucous effective liquid temperature due to heat loss through the surge line in absence of heat gain & transport from the hot leg pipe.

f. Pressurizer steam bubble pressure is in equilibrium with slowly decreasing reactor vessel steam bubble pressure and is most

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likely contracting (pressure level slowly increasing)
as steam from the pressure bottle is
condensed on the pressure wall at a rate
which is commensurate with heat loss to
the environs.

Mode 3²

Decay Heat Removal During Transition From
Natural Circulation³ to Core Boiling⁴ Mode

- a. Reactor vessel level reaches top of hot leg
pipes and starts to extend into hot leg
due to fluid loss through the break
in excess of makeup capability.
- b. No additional fluid loss occurs, reactor
vessel steam starts to bubble through hot
leg pipes and accumulate in the U-bends
of the reactor coolant piping at the top of

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the steam generator.

c. Natural circulation through the reactor core and steam generator is disrupted.

d. Water in the reactor core starts to boil vigorously due to loss of natural circulation core cooling.

e. Reactor core exit temperature and corresponding reactor vessel steam bubble pressure start to increase in response to loss of natural circulation.

f. Steam bubbling through the hot legs to the high point U-bends increases the hot leg liquid temperature to the core exit temperature (no other immediate heat sinks for decay heat).

g. Pressurizer piping loop seal inhibits reactor steam entry into pressurizer.

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k. Pressure and increase with a commensurate water level decrease in U-head region of the hot leg piping as maximum steam bubble in pressure is compared to reactor vessel steam bubble pressure (Colder loop and water maintains a lower average water temperature in pressure).

i. Reactor vessel hot leg and cold leg pipe equilibrate to nearly the same pressure by vent valve actuation (maximum ΔP of 0.15 psi for vent valve opening or about 0.5 ft of hydrostatic head at 570°F).

j. Reactor coolant pump loop seal inhibits reactor steam entry into steam generators through cold leg piping.

k. Water level in the U-head region of each hot leg pipe decreases due to fluid

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loss through break in space of makeup capability.

(Water level in steam generator tubes will be slightly lower than in the hot leg pipe due to lower average temperature).

Mode 4²

Dray Heat Removal by Core Boiling

a. Water level in the vertical hot leg and corresponding steam generator tube elevation continues to increase due to fluid loss through the break in space of makeup capability.

b. Significant heat removal occurs by condensation inside the steam generator tubes when the water level in tubes drops below

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the secondary side water level (which is about the lower 20 to 50 percent of the total tube height).

c. Steam from the reactor core area accumulates in the reactor vessel upper plenum, batters into the hot leg piping, and breaks away at the water-steam interface in the vertical hot leg at a rate which is commensurate with the steam condensation rate in the steam generator and local heat loss to the surroundings.

d. Reactor coolant system pressure remains at the saturation pressure corresponding to the core exit temperature which is commensurate with the decay heat input, steam generator condensation rate, heat loss to the surroundings, and heat transport through the tank.

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e. Pressure level continues to increase as remaining steam bubbles in pressure coils and is compressed to reactant vessel steam bubble pressure while the hot water drawn from the hot leg pipe into the pressure loop coil piping is allowed to cool.

f. Pressure level ceases to increase as soon as the water level in the vented hot leg pipe is below the pressure loop coil nozzle attachment to the hot leg pipe (subsequent damage may not occur as will be limited because of the loop seal).

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g. Liquid from condensation inside the steam generator tubes is returned to the reactor vessel by gravity flow through the pump loop seals and cold leg pipes.

h. Reactor vessel level fluctuates about an elevation corresponding to the top of the reactor vessel hot leg recycle water level in the vertical hot leg loops to the horizontal hot legs.

i. Reactor vessel level decreases due to fluid loss through the break in excess of makeup capability until makeup capability speeds fluid loss through break.

includes cause of
break 3/2/66

j. Reactor core exit temperature and reactor vessel steam bubble pressure decrease in response to a decay heat reduction without

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a. Commensurate reduction in heat loss to the
surrounds and heat transfer through the break,
and eventually in response to a reduction in
steam generator pressure (and temperature) due to
heat loss to the surrounds decaying decay
heat transferred after the safety valves are
closed. (If generator action is considered, the
steam generator pressure may be reduced by
opening atmospheric dump valves and thereby
reduce the steam generator and reactor coolant
system temperatures).

b. Fluid loss through the break is decreasing
and makeup rate is increasing following
reduction in reactor vessel pressure.

1. Reactor vessel level starts to increase as
makeup capability exceeds fluid loss
through the break.

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12. Pressure level increases as pressure continues to rise.

Mode 5²

Decay Heat Removal During Transition From
Care Bailing⁴ to Natural Circulation³ Mode

a. Reactor vessel level continues to increase to top of reactor vessel hot leg riser and a water level appears in the vertical hot leg pipes as makeup capacity exceeds fluid loss through break.

b. Water level inside steam generator tubes tends to equalize (except for density difference) with water level in vertical hot leg pipes due to vent action (also due to gap between reactor upper internals and hot leg riser and a steam space coupling through the bypass valve

15
2
Steam Generator

... steam starts to bubble through the horizontal hot leg pipes and break away at the water-steam interface in the vertical hot leg pipes before reaching the water regions of the unfilled portion of the steam generator tubes.

d. Reactor vessel level fluctuates about an elevation corresponding to the top of the reactor vessel hot leg nozzles.

e. Excess heat removal is accomplished by condensation inside the steam generator tubes of the tube water level is sufficiently below the secondary side water level and the accumulation of non-condensable gases in the upper reaches of the steam generator tubes and in the U-bend region of the hot leg pipes down

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not inhibit an adequate condensation rate.

f. Pressure and increases as air remaining steam bubble in the pressure cools and water is drawn from the hot leg pipe into the pressure loop seal giving where it can cool.

g. Water level continues to increase inside steam generator tubes and raised hot leg pipes as makeup capability speeds fluid loss through leaks.

h. Decay heat removal by condensation inside the steam generator tubes essentially ceases when water level in the tubes becomes greater than on the secondary side of the steam generator.

i. Reactor core exit temperatures and corresponding water vessel steam bubble pressure start to increase in response to destruction of steam condensation.

g. Makeup water continues to fill the vented hot leg pipes and steam generator tubes and add to the pressure increase.

h. The U-bend region of the hot leg pipes contains steam non-condensable gases at saturation conditions corresponding to about reactor vessel steam bubble conditions.

i. The steam and gas bubble in the U-bend region continues to compress until the vented hot leg water level floods the U-bend region sufficiently to establish natural circulation as the bubble pressure reaches the set point of a pressure-safety valve.

m. If the ^{pressure} safety valve opens before natural circulation is re-established, the decay heat might be removed if the makeup flow rate at the safety valve opening

pressure in accordance with the design heat removal requirements. Note that since the pressurizer piping is below the vented hot leg water level, the safety valve will lose water and lost steam. Thus, the design heat must be removed in the form of sensible heat in water without taking credit for the heat of vaporization. Also note that the makeup rate from the high pressure pump is very low for this overpressure condition).

m. If the design heat removal rate is less than the generation rate, system pressure increases and additional safety valves open thereby rapidly expelling the system water inventory until the vented hot leg level drops below the pressurizer piping and the pressurizer water inventory is expelled.

with probable intermittent slug or two-phase flow instability) thereby permitting decay heat removal by core boiling and steam discharge through the safety valves.

Note 6²

Decay Heat Removal After Re-establishing Natural Circulation³ With Reactor Vessel Pressure Controlling

a. Natural circulation is assumed to be re-established with a reactor vessel level above the hot leg nozzles and with the vertical hot leg piping and steam generator tubes full of water.

b. Reactor vessel level continues to increase (top pressure steam bubble contracts) as makeup capability exceeds fluid loss through the tank.

It is assumed that the pressure in the reactor vessel steam bottle pressure which responds to changes in the decay heat input, heat loss to the surroundings, and heat transport to the steam generator and through the break.

d. Pressure is filled with liquid but may contain a trapped ^{steam and} non-condensable gas bottle which is in pressure equilibrium with the reactor vessel steam bottle.

e. Pressure steam bottle contracts as steam is condensed on pressure walls at a rate which is commensurate with heat loss to surroundings.

44-7⁵Survey Heat Removal by Forced Circulation⁶Using Decay Heat Removal System

- a. Natural circulation has been previously established with reactor vessel level above hot leg nozzle and the pressurizer essentially filled with liquid.
- b. Reactor vessel level has been stabilized by adjusting mixing capability to equal fluid loss through the break. (In the long term, operator action may be used to re-establish a pressurizer steam bubble and install a stable pressurizer level.)
- c. Decay Heat Removal (DHR) system is in place to provide forced circulation decay heat removal from the reactor core.
- d. Reactor vessel top plenum is partially

solid reaction. Buttle at saturation pressure corresponding to top pressure or liquid temperature (unless pressure or pressure control has been established).

c. Reactor vessel pressure is below D.H.R. design operating pressure and decreasing at a rate commensurate with decay heat input, but loss to excessive heat transport through leaks, and heat removal by the D.H.R.

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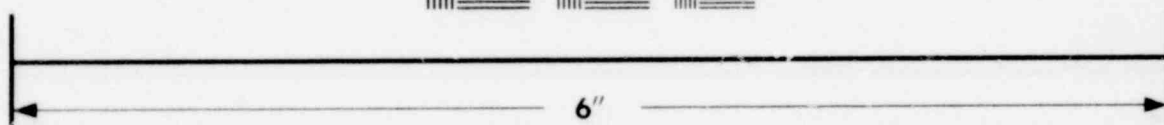
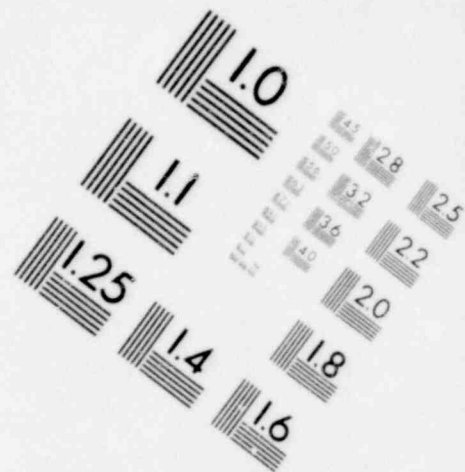
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tubes with the core exit temperature (also assumed to be reactor vessel top pressure effective liquid temperature), established by the inlet heat rate and natural convection heat transfer and heat transport coefficients.

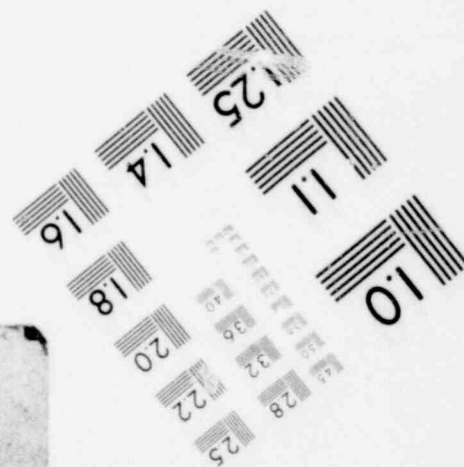
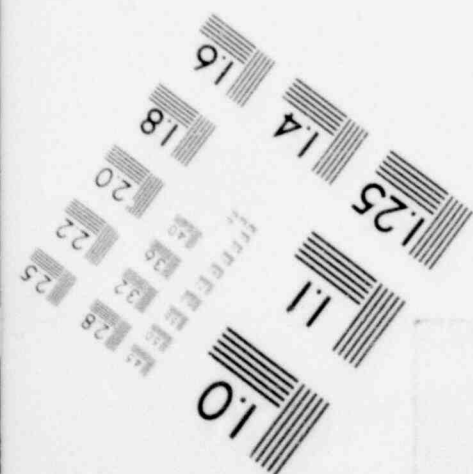
4. Steady heat removal in by boiling heat transfer on the core, saturated steam heat transport from the reactor vessel to steam generator, and condensing heat transfer inside the steam generator tubes.

5. Providing a very small break, loss of off-site power, operator actions permissive where required, closure of main steam isolation and atmospheric dump valves, and minimum core cooling (one train) response using SIF for heat removal and high pressure injection for refueling.

- The heat loss is by forced convection
 through the reactor core and DTR heat
 exchangers using DTR pumps in the shutdown
 cooling mode of operation with the core exit
 temperatures (also assumed to be the reactor vessel
 top pressure effective liquid temperatures,
 established by the decay heat rate and forced
 convection heat transfer and transport coefficients.



MICROCOPY RESOLUTION TEST CHART



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Appendix B

South Brook L27K Analysis

by ~~Isaac~~ J. Wilcox

BAW-10074A, Rev 1
Topical Report
March 1976

MULTINODE ANALYSIS OF SMALL BREAKS FOR
BAW's 205-FUEL-ASSEMBLY NUCLEAR PLANTS
WITH INTERNALS VENT VALVES
- Revision 1 -

by

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Revision 1
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Topical Report BAW-10074A, Rev 1

March 1976

Multinode Analysis of Small Breaks
For B&W's 205-Fuel-Assembly Nuclear Plants
With Internals Vent Valves - Revision 1

1

R. C. Jones, B. M. Dunn, C. E. Parks

Key Words: Multinode Analysis, Nuclear Plant, Small Break

ABSTRACT

Multinode analyses were conducted for several small breaks in the reactor coolant system of B&W's 205-fuel-assembly nuclear plants with internals vent valves. The multinode blowdown code CRAFT was used to evaluate the hydrodynamics and transient water inventories of the reactor coolant system. The FOAM code was used to compute a swell level history for the core, and the THTAI-3 code was used to perform transient fuel pin thermal calculations. Curves showing the parameters of interest are presented. These results are well within the Final Acceptance Criteria.

1

1. INTRODUCTION

This topical report evaluates the effectiveness of the emergency core cooling systems for B&W's 205-fuel-assembly (FA) reactor designs with the following features:

1. Eight 14-inch-diameter internals vent valves.
2. Cross-connected LPI systems.
3. HPI system manifolded into each cold leg.

The plant is assumed to be operating at 3760 MWt, and the analysis is conservative for plants of similar design operating at a lower rated power. The analysis is also conservative for plants of similar design with Mark-C design fuel assemblies.

The report presents the results of an analysis of loss-of-coolant accidents resulting from small breaks in the reactor coolant system. (Small breaks are defined as those breaks in the reactor coolant system with less than 0.5-ft² leak area.) The CRAFT,¹ FOAM,² and THERMAL-3³ computer codes were used for the analysis.

A spectrum of ruptures is considered: 0.5-, 0.1-, 0.1-, and 0.05-ft² leak areas. Also, the report includes the double-ended rupture of the core flooding line (0.44-ft² leak area). These ruptures provide an appropriate spectrum for evaluation of the effects of small leaks.

2. SUMMARY AND CONCLUSIONS

The spectrum of small breaks analyzed in section 4 shows that the cladding will undergo only a moderate increase in temperature during small loss-of-coolant accidents. Because the peak temperatures are low, no potential for cladding swelling or metal-water reaction exists. Therefore, the core geometry is unchanged and will remain amenable to cooling. Long-term cooling is established once the injection rate matches the leak rate and the core is covered with a steam/water mixture.

The double-ended rupture of a core flooding tank line yields a maximum cladding temperature of 1636°F. Core geometry will be maintained in a coolable configuration. Long-term cooling is established by using the same reasoning as in the paragraph above.

For all breaks analyzed, the conditions of the Final Acceptance Criteria are met. Conformance of this analysis to the Final Acceptance Criteria is demonstrated in Appendix A of BAW-10104, Revision 1.³ Therefore, the design of the emergency core cooling system is adequate to control small loss-of-coolant accidents.

Results for the breaks considered are summarized as follows:

Break	Initial Temp, °F	Peak Temp, °F	Time at which long-term cooling is established, s
0.5 ft ² at pump disch	710	1178	200
0.3 ft ² at pump disch	710	710	400
0.1 ft ² at pump disch	710	710	2200
0.1 ft ² at pump suct	710	710	2550
0.05 ft ² at pump disch	710	710	2800
0.44 ft ² CF line break	710	1636	400

5 seconds of the accident. At 180 seconds, two-phase fluid flows through the break and results in a more rapid depressurization of the reactor coolant system. For the break at the discharge, this effect occurred at 200 seconds. After this time, the break at the suction depressurizes faster than the break at the pump discharge due to higher quality fluid existing in the system. The core flooding tank actuation pressure is reached at 775 seconds. At 1100 seconds, the LPI pump begins to inject water to the reactor vessel, resulting in an increase in the inner vessel inner volume. The auxiliary feedwater pump flow stops at 1190 seconds after the break occurs and causes an increase in the system pressure. Long-term cooling is established at 1550 seconds. The core remains covered by a two-phase mixture throughout the transient, and no cladding temperature increase will occur.

As was shown, no major differences in response resulted from the change of location. The lower pressures that occur during the suction break result in increased pumped injection. Therefore, small breaks at the pump discharge are considered to be slightly worse than breaks at the pump suction.

4.7. 0.05-ft² Break at Pump Discharge

The 0.05-ft² break at pump discharge is the smallest break analyzed. This break responds basically in the same manner as the 0.1-ft² break at the pump discharge as shown in Figures 4-31 through 4-34. The reactor trip occurs 3 seconds after the break, the auxiliary feedwater pump shuts off at 1510 seconds, the core remains covered by a two-phase liquid throughout the accident, and the cladding temperature never exceeds its prebreak value. The results for this break show that the EPI pumps alone are sufficient to handle breaks of this size and smaller since the core flooding tanks inject water for only a short period of time and thus play an insignificant role in controlling the accident.

Figure 4-31. Core Pressure for 0.05-ft² Break at Pump Discharge

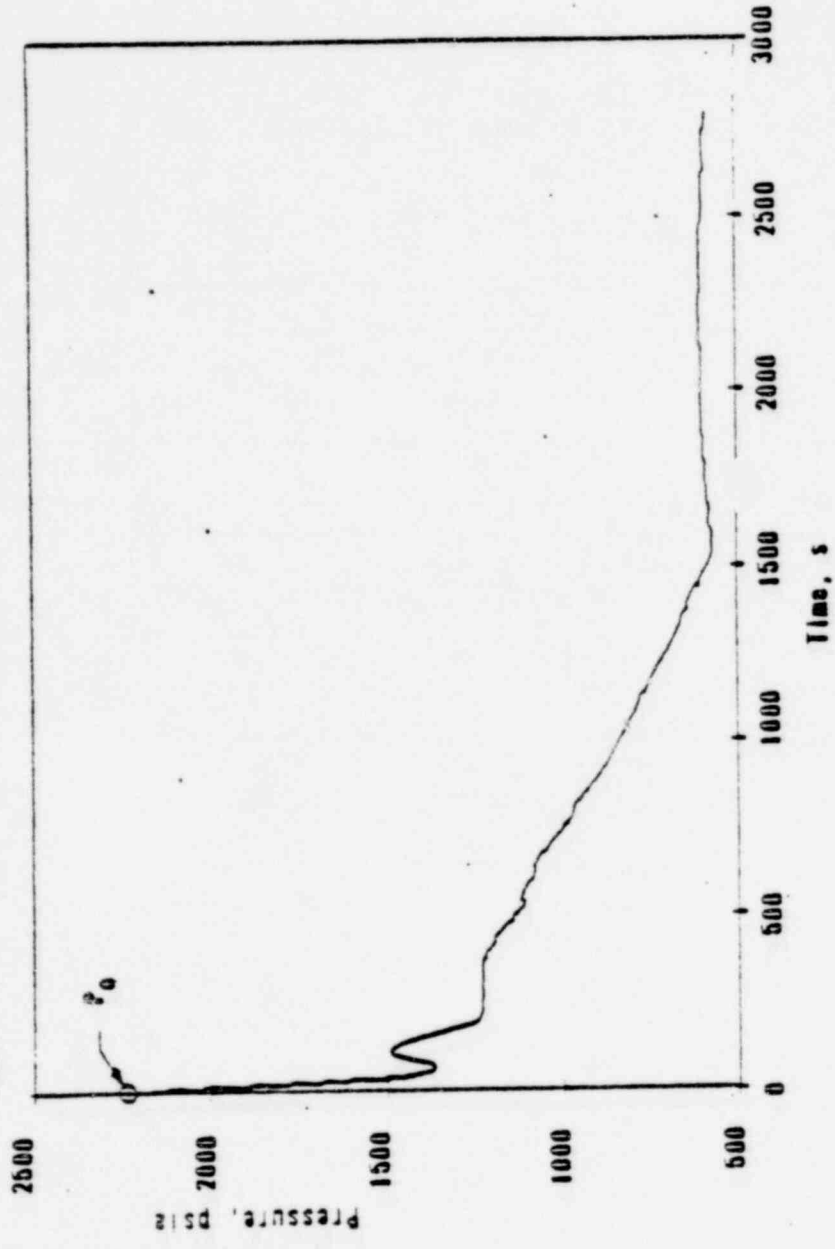
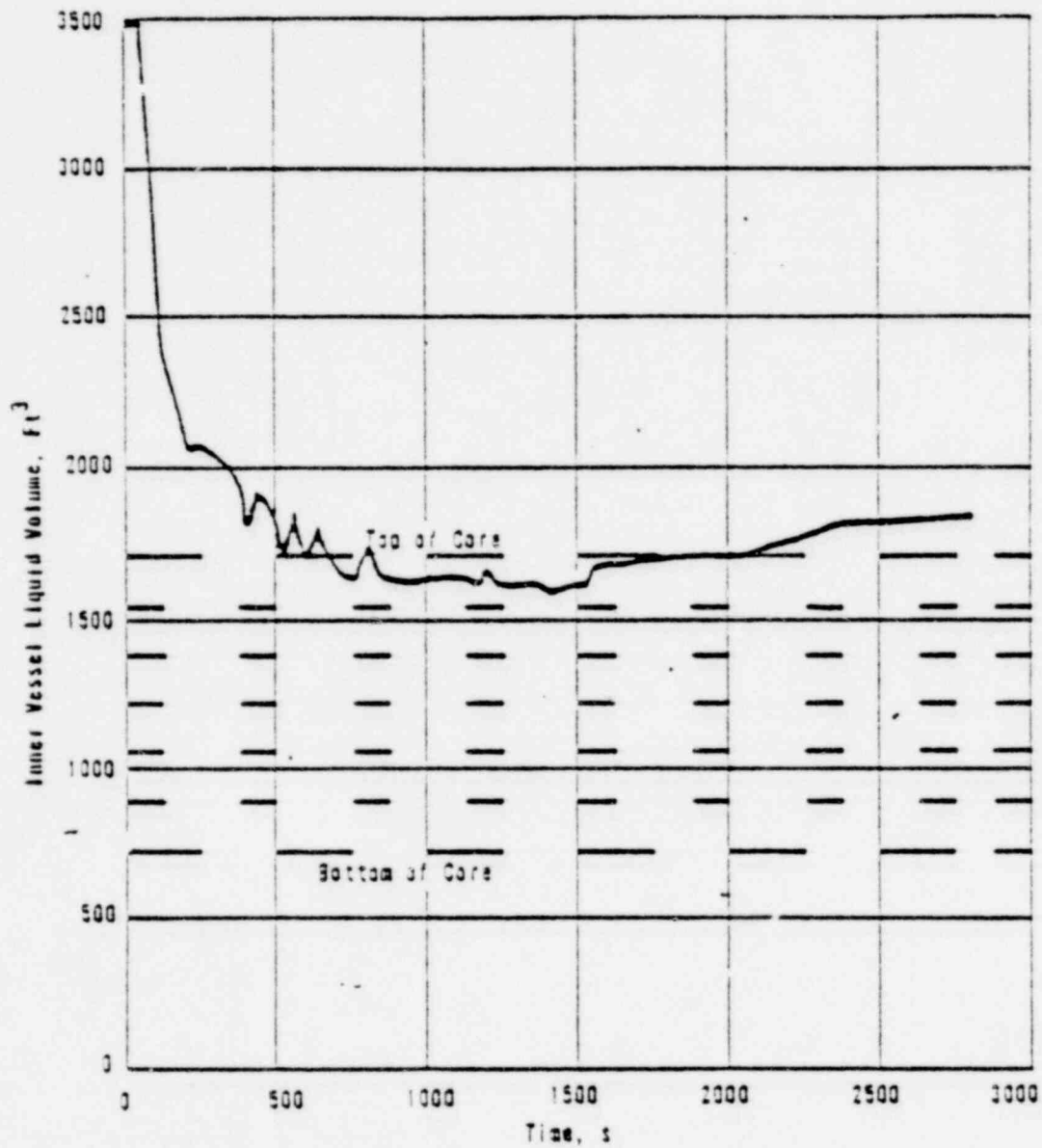


Figure 4-34. Inner Vessel Liquid Volume for 0.05-sec²
Break at Pump Discharge



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Appendix C

NRC Evaluation of BBU

Small Break LOCA Analysis



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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JAN. 08 1975

Mr. Kenneth E. Suhrke
Manager, Licensing
Babcock & Wilcox Company
Nuclear Power Generation
P. O. Box 1250
Lynchburg, Virginia 24505

Dear Mr. Suhrke:

We have completed our review of the non-proprietary B&W topical reports BAW-10062, Rev. 1, "Multinode Analysis of Small Breaks for B&W's 143-Fuel Assembly Nuclear Plants with Internals Vent Valves," and BAW-10074, Rev. 1, "Multinode Analysis of Small Breaks for B&W's 105-Fuel Assembly Nuclear Plants with Internals Vent Valves," and BAW-10075, Rev. 1, "Multinode Analysis of Small Breaks for B&W's 177-Fuel Assembly Nuclear Plants with Raised Loop Arrangement and Internals Vent Valves." A copy of our evaluation is enclosed.

From our review, we conclude that the analytical methods and models used in these three topical reports meet the requirements of 10 CFR 50 Appendix K. The results presented are acceptable for reference in licensing actions, with one exception for BAW-10075 on the 177-FA plant. This report does not contain results for the core flood tank line break, as do the other reports, but indicates that these results are furnished in the plants' safety analysis reports.

We do not intend to repeat our review of BAW-10062, BAW-10074, or BAW-10075 when these are referenced in individual licensing actions.

We request that B&W issue a revised version of these reports within three months of receipt of this letter to include the NRC acceptance letter.

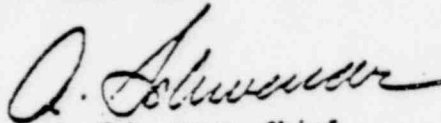
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Mr. Kenneth E. Suhre

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If our criteria or regulations change, such that our conclusions regarding BAW-10062, Rev. 1, BAW-10074, Rev. 1, or BAW-10075, Rev. 1 are invalidated, we will notify you and give you opportunity to revise and, if you desire, resubmit these topical reports for review.

Sincerely,



A. Schwender, Chief
Light Water Reactors Branch 2-3
Division of Reactor Licensing

Enclosure:
As stated

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JAN. 08 1976

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Mr. Kenneth E. Sahrke

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TOPICAL REPORTS EVALUATIONTOPICAL REPORTS SUMMARY

The topical reports evaluated in this review present results from small break spectrum (AS 0.5 ft²) analyses for B&W's 145⁽¹⁾, 177⁽²⁾, and 205⁽³⁾ FA plants with internals vent valves. The three reports were reviewed jointly because they use the same small break model to perform the small break spectrum analyses.

The small break model consists of the three computer programs discussed below used to analyze plant response to small breaks:

1. CRAFT⁽⁴⁾: A multinode representation of the plant primary system to compute blowdown hydrodynamics and water inventory in primary loop components following piping ruptures.
2. FOAM⁽⁵⁾: A single node representation of the core to compute average two-phase mixture height in the core when core uncover is computed in the heterogeneous reactor inner vessel volume of the CRAFT model.
3. THETA-8⁽⁶⁾: A multinode representation of the core hot pin fuel and clad to compute temperature transients during the loss-of-coolant accident up through core recovery based on average core coolant conditions determined in CRAFT and FOAM.

The multinode plant representations in CRAFT for small break analyses typically contain approximately half of the nodal detail used for large break studies (15 compared to 30). The reduced model has been shown⁽⁷⁾ to provide results essentially identical to those of the large break model when using identical assumptions, due to the reduced blowdown rates and internal pressure gradients computed for small breaks. Most of the nodes

detail reduction is made by lumping the two primary loops simulated in the large break model into a single loop, and by lumping the reactor inner vessel regions of core, core bypass, upper plenum, and upper head into a single volume. In addition to the nodal reduction, some large break analysis assumptions imposed for conservatism can be relaxed because of the slower blowdown of the small break. These changes are:

1. Normal system low pressure reactor scram is included.

2. All injected ECCS water is assumed to enter the downcomer and is retained in the CRAFT calculations. No conservative removal of this water at the end of blowdown is used as in the large break analyses.

3. A heterogeneous inner vessel volume is modeled for the small break in CRAFT compared to the homogeneous volumes modeled for the large break analyses.

Results obtained from the topical reports for the small breaks considered for each plant are summarized in Table 1, and show that peak clad hot spot temperatures in any of the three plant types analyzed do not approach 2200°F.

TOPICAL REPORTS EVALUATION

The small break model programs used in the analyses in the three reports reviewed here were evaluated in previous assessments for large break studies in the cases of CRAFT and THETA1-B, and the FOMU program was evaluated in the review of BAW-10064⁽⁵⁾. Applications

of these programs in the small break analyses reported in the topical's were performed in a manner which provides some conservatism in the calculations of peak clad temperatures; however, an assessment of the extent of this conservatism was not sought because the temperatures obtained were low compared to damaging temperatures.

From the results summarized in Table 1, it is noted that the analysis for the 177-FA plant with raised loops did not include the core flooding tank line break. The introduction of the report (SAW-10075)⁽²⁾ for this plant indicated that the results of the analysis for this break would be presented in the SAR for each plant application. The 0.5 ft² break for this plant was also indicated in the introduction to be reported in SAW-10053⁽³⁾. As this report presents results for a 177-FA non-vent valve plant using the large break model for the break analyses, it is considered that the 0.5 ft² break for the 177-FA plant with raised loops and internal vent valves has been adequately considered. The value shown in Table 1 for this break is considered greater than can be conservatively expected for a small break model analysis when compared to 145 and 205-FA results.

REGULATORY POSITION

Referencing the appropriate report of the three topical's evaluated in this review will be acceptable for small break spectrum analyses for SARs with the exception of SARs for 177-FA vent valve plants with raised loops which require additional documentation for the CFT line break.

The models and programs used for the small break analyses in these topical's predate issuance of Appendix K of 10 CFR 50, but are considered in conformance to the requirements of that rule.

TABLE 1

B&W PLANTS: SMALL BREAK SPECTRUM

PEAK CLAD HOT SPOT TEMPERATURE - °F

Break Size - ft ² (Pump Discharge)	145 FA	177 FA RAISED LOOP	205 FA
0.5	1196	1533 ⁽³⁾	1178
0.3	1030	1090	710 ⁽²⁾
0.1	I.C. ⁽¹⁾	700 ⁽²⁾	710 ⁽²⁾
0.05	I.C. ⁽¹⁾	-	710 ⁽²⁾
0.01	-	700 ⁽²⁾	-
(Pump Suction)			
0.3	-	824	-
0.1	I.C. ⁽¹⁾	-	710 ⁽²⁾
(CFT Line)			
0.35	760	-	-
0.44	-	-	1636

NOTES:

(1) - Initial condition - not specified

(2) - Same as initial condition

(3) - From Reference 3

CFT - Core Flooding Tank

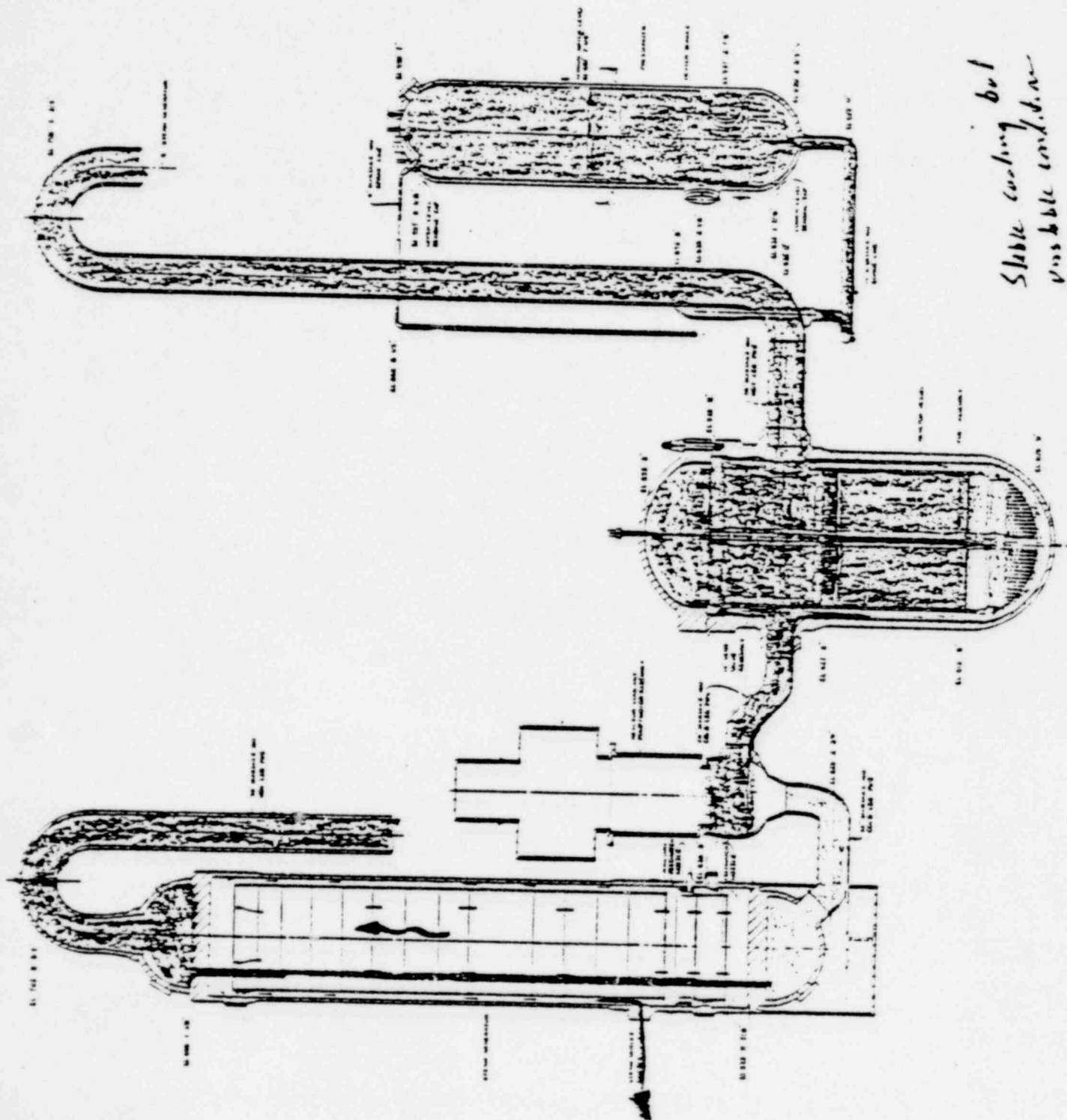
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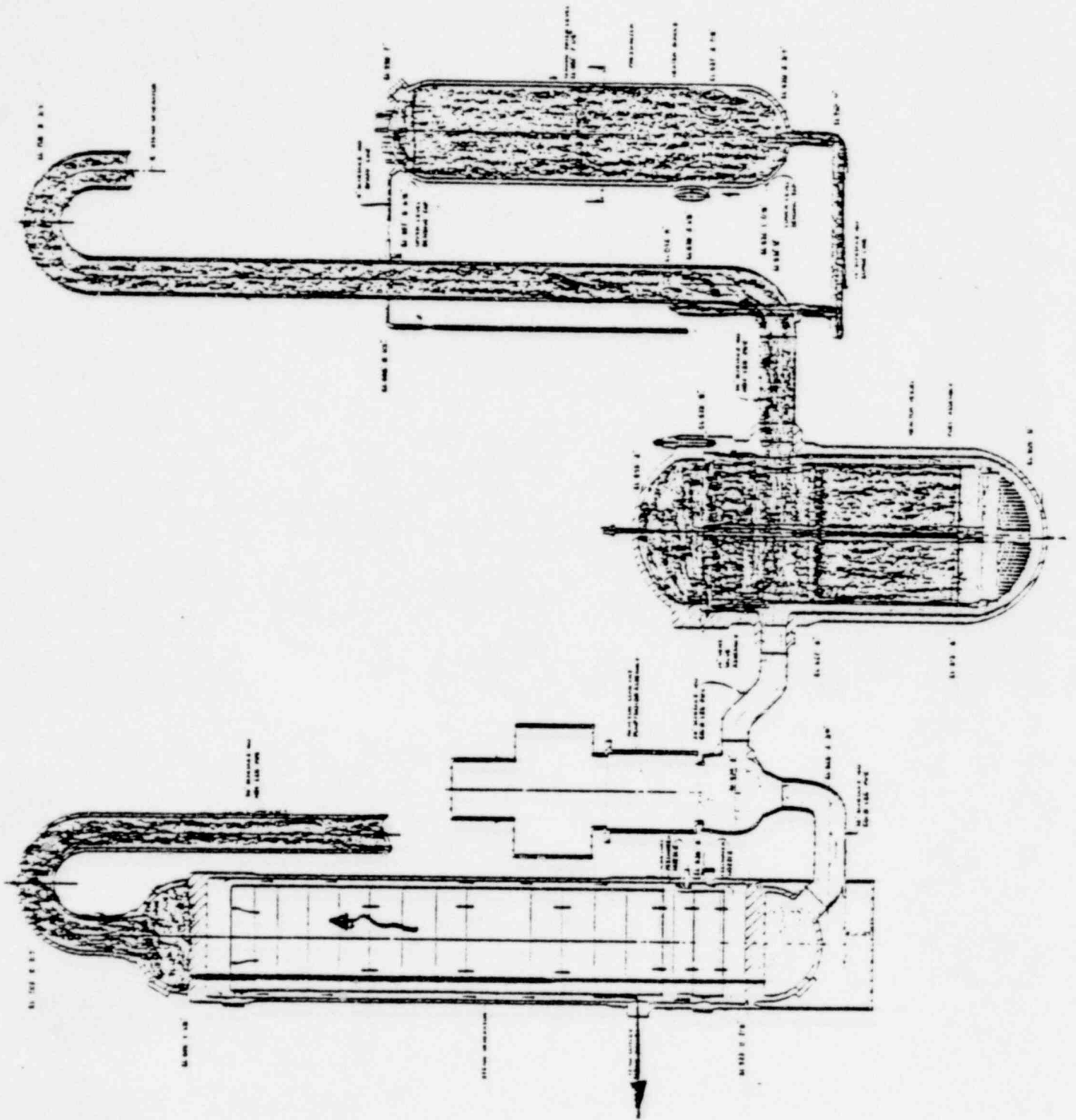
1. C. E. Parks, R. J. Allen, and L. R. Curtin, "Multinode Analysis of Small Breaks for B&W's 146-Pool-Assembly Nuclear Plants With Internals Vent Valves," B&W-10062, October 1973.
2. L. R. Curtin, J. M. Hill, and C. E. Parks, "Multinode Analysis of Small Breaks for B&W's 177-Pool-Assembly Nuclear Plants With Raised Loop Arrangement and Internals Vent Valves," B&W-10075, December 1973.
3. R. C. Jones, B. M. Dunn, and C. E. Parks, "Multinode Analysis of Small Breaks for B&W's 205-Pool-Assembly Nuclear Plant With Internals Vent Valves," B&W-10074, November 1973.
4. B. E. Bingham, W. L. Jensen, and R. A. Hedrick, "CRAFT - Description of Model for Equilibrium LOCA Analysis Program," B&W-10030, October 1971.
5. B. M. Dunn, C. D. Morgan, and L. R. Curtin, "Multinode Analysis of Core Flooding Line Break for B&W's 2566-MWt Internals Vent Valve Plants," B&W-10064, April 1973.
6. C. T. Hoeveker, and T. W. Wineinger, "THERM-B - A Computer Code for Nuclear Reactor Core Thermal Analysis," EP-1445, February 1971.
7. C. E. Parks, B. M. Dunn, and R. C. Jones, "Multinode Analysis of Small Breaks for B&W's 2566 MWt Nuclear Plants," B&W-10052, September 1972.
8. C. E. Parks, L. R. Curtin, and R. C. Stieh, "Multinode Analysis of B&W's 177-Pool-Assembly, Non-Vent Valve Plant During LOCA," B&W-10053, June 1973.

Mo. 413

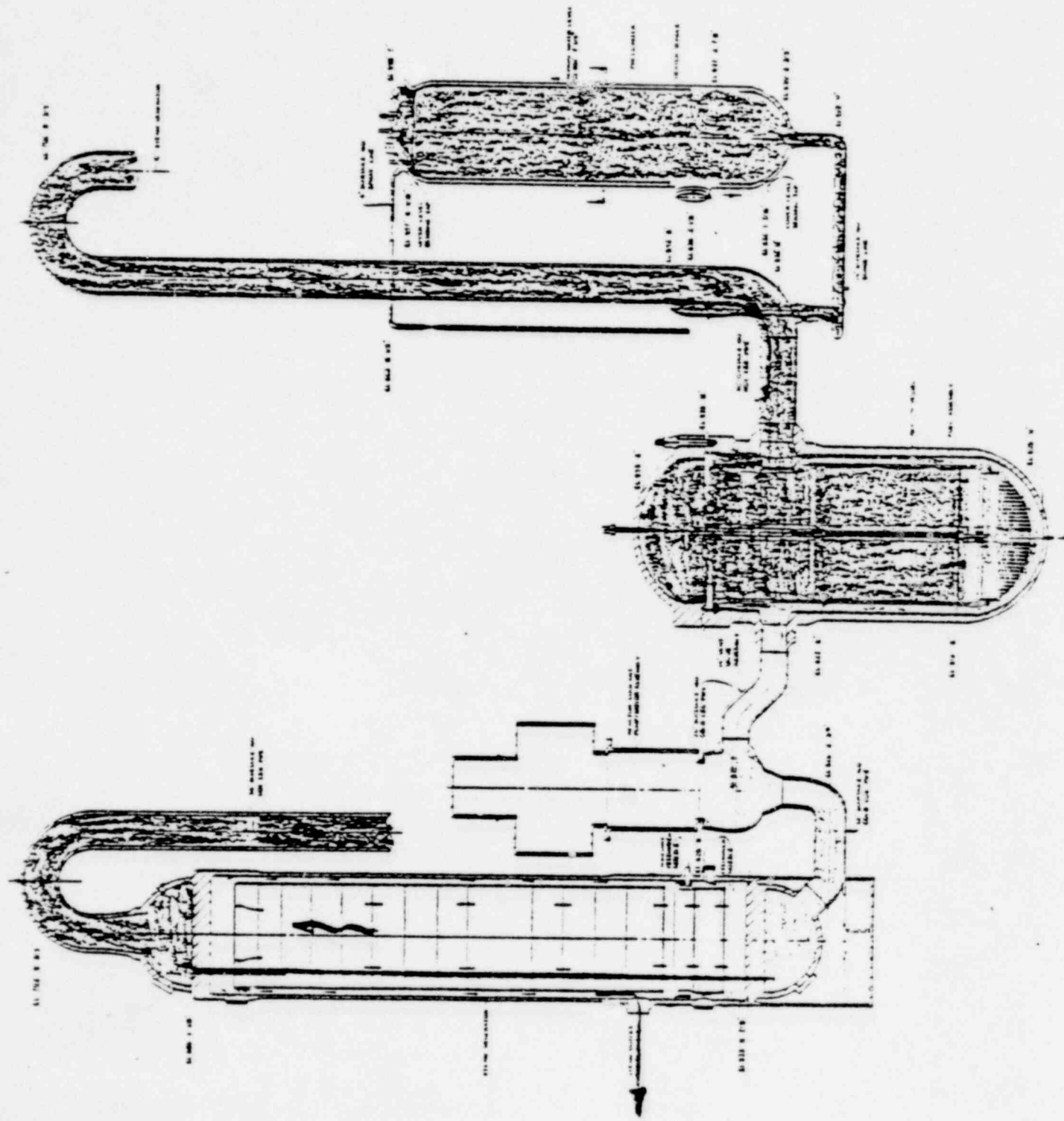
Stable cooling but
variable condensation



Ms. 29.11.11



Model 5B



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1. Introduction

During the emergency forwater meeting with Combustion Engineering on May 27-28, 1975, TVA expressed concern for a class of small break LOCA's whose depressurization rates are relatively slow (when compared to those analyzed in CESSAR as small breaks) and these might lead to inadequate makeup rates from the high pressure injection pumps. This concern was answered by Combustion Engineering in letter No. TD-CE-161 (September 18, 1975) a copy of which is included as Appendix B in this report. An ongoing qualitative consideration of the concern by TVA now predicts a dry heat removal problem during post-LOCA recovery which needs to be resolved. This problem and its associated background material are the subject of this report.

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2. Physical Arrangement and Characteristics

The physical arrangement of the reactor coolant system for a typical System 30 plant is shown in Figure 1. The plant has two steam generators each with a single hot leg piping connection (42 inch) and two cold leg piping connections (30 inch) with a reactor coolant pump provided in each cold leg. The pressure is attached to one of the hot leg pipes by an 20 foot long surge pipe (12 inch). The reactor vessel has two corresponding hot leg piping nozzles and four cold leg nozzles.

The plant elevations corresponding to various points in the reactor coolant system are given in Table 1. Other reactor coolant system characteristics of interest are given in Table 2.

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3. Predicted Post-LOCA Modes of Decay Heat Removal

The conditions predicted to be associated with each quasi-steady-state post-LOCA mode of decay heat removal are detailed in Appendix A. These conditions are for a very small break (≤ 0.1 ft²) for which the steam generator must remove a significant portion of the decay heat; otherwise, reactor coolant system depressurization occurs since the break is too small to accommodate the transport of all decay heat to the environment. Modes 1-6 are based on loss of off-site power, no operator action, minimum core cooling (no train), injection using cold leg injection, closure of main steam isolation valves, and steam generator at saturation conditions corresponding to the lowest safety valve set point.

Mode 1-7 given in Appendix A defines

the conditions which are thought to exist on the reactor coolant system to provide decay heat removal during the relatively slow changing post-LOCA period. These conditions are summarized in Table 3.

The achievement of Modes 1-3 appears straightforward although there might be some concern if the transition to core boiling is not accompanied by a relatively rapid drainage of the steam generator tubes. If an adequate condensing surface is not exposed quickly, ~~excessive steam~~ ^{excessive steam} ~~boiling~~ ^{boiling} might occur within the tubes.

Operation in Mode 4 appears reasonable to achieve although the reactor operators will be unaware as to what is happening to reactor vessel level.

Mode 5 is thought to be the most

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uncertain operation which will take place. The rest of Mode 5 cannot be predicted or controlled since there is no local information available to the operator. The flooding of the hot leg pipes and steam generator inlets will disrupt normal steam flow from the core to the steam generator and will lead to an intermittent riding of the hot leg piping as the steam is forced through. It appears reasonable that adequate decay heat transport might be achieved during this transition period although there should be some concern about the mechanical effects of slug flow on steam generator tubes.

In order to re-establish natural circulation decay heat removal which is Mode 6, it is necessary that Mode 5 terminate by filling the steam generator tubes with liquid and

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establishing the density gradients required for natural circulation. It is not clear how this can be accomplished while maintaining an adequate heat transport rate from the core region through the hot leg piping and inlet side tubing. There could be a problem with a persisting slug flow from the reactor vessel and significant accumulations of non-condensable gases in the steam generator U-tubes. If enough non-condensable gases have accumulated, natural circulation cannot be re-established (unless the gases can be removed by some means).

If Mode 6 cannot be established, it will be necessary to wait until the decay heat level is sufficiently low to provide prolonged core cooling by a rapid falling (subcooling) of the reactor vessel and establishment of shutdown

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Cooling System (SCS) operation helps core boiling is re-initiated. This variation of Mode 7 might be difficult to achieve unless the reactor vessel water level is sufficiently low to permit a large cold water addition to the core region. The proposed maneuver may be difficult to achieve without a reactor vessel level indication that guide the control of water addition during the waiting and refill periods.

4. Available Quantitative Results

Information concerning a Combustion Engineering analysis for the small break LOCA for a Lysium 80 plant is included as Appendix B. This analysis is for cold leg breaks which are 0.1, 0.05, and 0.005 square feet in size (about 4, 3, and 2 inch diameter, respectively). The analysis assumes loss of off-site power, the worst single

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failure of emergency equipment, and no operator action. The steam generator is assumed to remain at the saturation pressure corresponding to the lowest setpoint of the safety valves (1225 psia).

The results shown in Figure 2 of Appendix C indicate that the reactor vessel level will drop into the hot leg piping for each assumed break size within 15 to 30 minutes following the break, thus leading to the predicted occurrence of the fast sip mode of operation outlined in Appendix A. The results show that the core will remain covered at all times including Mode 4 operation.

Some correlation can be recognized between the curves in Appendix B and the modes of operation predicted in Appendix A.

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5. Area of Concern

The transition from core boiling to natural convection which is predicted in Appendix A (HDO5) to occur during reactor vessel level recovery following a very small break will require that the steam generator U-tubes become refilled. It is not clear how this can be accomplished, particularly if a significant amount of non-condensable gas has accumulated in the tubes. If the refilling is not accomplished, plant recovery from a very small break leak might be unacceptably delayed pending additional fuel cooldown.

Adding to this concern is the uncertainty associated with unknown vessel level, the adequacy of emergency operating instructions and training for this event, and the consequences of the unstable slug flow conditions which are predicted to develop as

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the reactor level recovers.

This operational difficulty appears to be generic for pressurized water reactors although it may be aggravated somewhat in the CE System 30 because of the lower head and capacity of the high pressure injection pumps. In PWR systems designed to other standards, higher head pumps with greater flow capacity at a given pressure assure that for a greater range of very small breaks the reactor vessel level will not drop below the hot leg nozzle and thereby disrupt natural circulation.

6. Additional Considerations

This qualitative consideration of recovery from a very small break LOCA did not include the problem of assuring a continuous long-term source of clean auxiliary feedwater for the steam

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generator. This difficulty might materialize if the recovery should be delayed pending additional fuel cooldown.

It should also be recognized that long-term recovery could involve recirculation of water from the primary containment sump of makeup water from the refueling water tank is no longer available or water accumulation inside of containment become limiting. The likelihood of requiring recirculation will increase if a delay for additional cooldown is experienced. This recirculation mode of operation with the reactor at high pressure is not an established design requirement and has not been evaluated for feasibility.

Additional complications appear likely if it is required to use the high pressure pumps for reactor makeup from the sump which

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operating the Shutdown Cooling System (SCS) since it involves some common piping including a common return pipe. This mode of operation has also not been a design requirement.

In addition, the high pressure pumps do not have minimum flow protection while taking suction from the sump. If the flow rate to the reactor is very low, pump damage could be a consideration.

The effect on primary containment of very small breaks was not considered. A low-low pressure pressure will appear early in the event and initiate containment isolation. A subsequent high-high containment pressure will initiate containment spray. The operators may not be able (or desire) to intercept the automatic spray. A prolonged isolation due to delay for additional cooldown will increase

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the likelihood of containment spray.

The effect of the safety injection tank starting to dump at about 610 psia was not evaluated. Unless the nitrogen pressure is reduced or the water level is lowered sufficiently, a dump will occur automatically. The tanks can be isolated manually below 415 psia but they are programmed to re-open if the pressure exceeds 500 psia.

A loss of Effetto power was assumed in preparing this report. If the very small leak should not be accompanied by an Effetto power loss, the effect of continued operation of the reactor coolant pumps would need to be evaluated. The loss of system overpressure at a high average reactor coolant temperature might lead to unacceptable pump or seal damage if the operator does not manually

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trip the pump within a few minutes. Physical damage might be induced by excessive vibration or a severe flow instability. Even if offsite power is lost, the operator must take steps within about one-half hour to re-establish seal cooling by loading the CVCS pumps on the diesel generator. If not done, an additional loss of coolant will develop through the pump seals. The consequence of this scenario was not developed.

The very small break postulated for this upset was assumed to be located at the top of a cold leg pipe. Other break locations were not considered in detail. However, the break location is not thought to be a major influence on the estimates of the predicted quench modes given in Appendix B, i.e., these estimates are on more break sizes and locations which could result in the occurrence of each predicted mode.

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Table 1Reactor Calant System Elevations

<u>Component</u>	<u>Elevation (ft)</u> *
Pressurizer:	
Vessel High Point (inside)	570
Vessel Low Point (inside)	569
Hot Leg Connection	512
Full Power Water Level	554
After Scream Water Level	544
Top of Heaters	539
Horizontal Jump Pipe (center)	519
Steam Generator:	
Highest Tube (center)	551
Lowest Tube (center)	543
Full Power Water Level	556
Tubesheet (face)	517
Reactor Vessel:	
Vessel High Point (inside)	530
Hot Leg Nozzle (high point)	512
Hot Leg Nozzle (low point)	503.5
Active Fuel (high point)	505
Active Fuel (low point)	492.5

* Approximate values.

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Table 2Reactor Coolant System Characteristics

Full Power Temperatures (°F):	Cave Entry	565
	Cave Midplane	593
	Cave Exit	621
	Pressurizer Liquid	653
	SG Steam	553
	SG Feedwater	450

Full Power Pressures (psia):	RC Pump Discharge	2319
	Cave Midplane	2288
	Cave Exit	2260
	SG Steam	1070

Safety Valve Settings (psia):	Pressurizer	2500
	Steam Generator	1285

Volumes (ft ³):	Pressurizer (steam space)	910
	Pressurizer (above heaters)	1500
	Pressurizer (total)	1800
	Reactor Vessel (above inlet)	2700*

Activation Pressures (psia):	Safety Injection Pumps	1600
	Safety Injection Tanks	610
	Reactor Scram	1750

* Approximate value.

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Table 3

Post-LOCA Modes of Decay Heat Removal²

<u>Mode</u>	<u>Condition</u>	<u>Press. Level</u>	<u>RV Level</u>	<u>System Press.</u>	<u>SG Tube Level</u>
1	Natural Circulation ^a	P-Decr.	Full	P-Decr.	Full
2	Natural Circulation ^c	S-Incr.	S-Decr. ^d	Stable	Full
3	Transition	Drain	S-Decr. ^e	Stable	Drain
4	Cave Boiling	Empty	S-Decr. ^f	Stable	Empty
5	Transition	Empty	S-Incr. ^g	S-Decr.	Refill
6	Natural Circulation ^c	Refill	S-Incr.	S-Decr.	Full
7	Forced Circulation ³	Full	Stable	S-Decr.	Full

2. Nomenclature: Press. Level is pressurizer level.
 RV Level is reactor vessel level.
 System Press. is system pressure.
 SG Tube level is steam generator tube level.
 P-Decr. is rapidly decreasing.
 S-Decr. is slowly decreasing.
 S-Incr. is slowly increasing.
 Stable is no significant changes.

a. With pressurizer pressure controlling.

c. With reactor vessel pressure controlling.

d. Reactor vessel steam bubble has formed.

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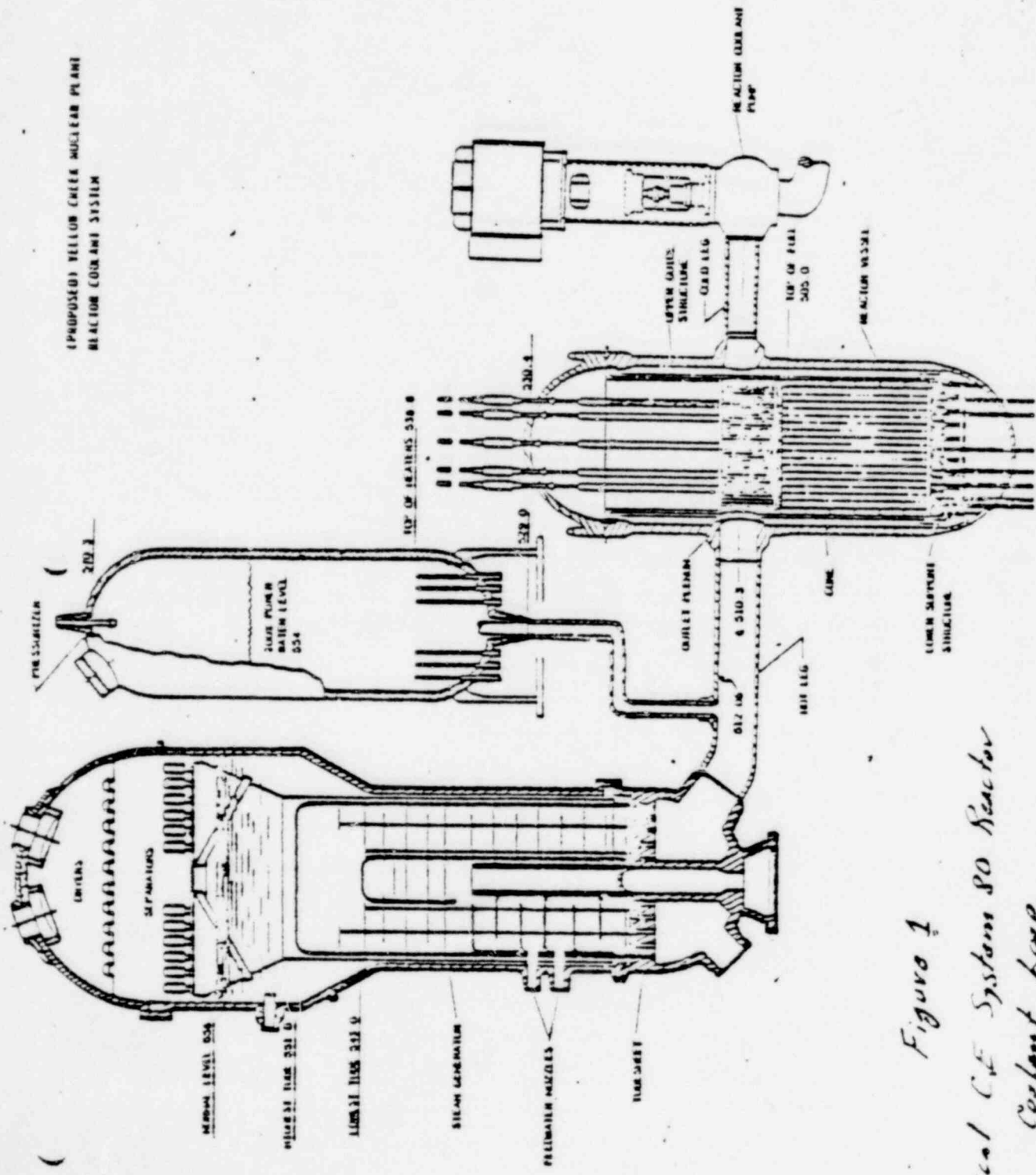
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e. Reactor vessel level has reached top of hot leg nozzles.

f. Reactor vessel level will reach a minimum and start to increase.

g. Using Shutdown Cooling System.



(PROPOSED) YELLOW CREEK NUCLEAR PLANT
REACTOR COOLANT SYSTEM

Figure 1
Typical C.E. System 80 Reactor
Coolant Loop

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Appendix A

Conditions Associated With Post-LOCA

Modes of Decay Heat Removal

For Very Small Break LOCA #1

* Numbered notes are included at end of appendix.

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Mode 1²Core Heat Removal by Natural Circulation³With Pressurizer Pressure Controlling

- a. Pressurizer steam bubble pressure is above saturation pressure corresponding to reactor vessel top plenum effective liquid temperature and is augmented to pressurizer effective metal temperature.
- b. Pressurizer level and pressure are decreasing rapidly (steam bubble expanding) due to fluid loss through the break in excess of making capability.
- c. Pressurizer liquid is partially flashing to steam (thrusty cooling) in an attempt to maintain constant pressure as the steam bubble expands.

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d. Pressurized steam bubble is superheating to pressurized effective metal temperature (thereby cooling walls).

e. Without an effective heat input other than from the pressurized metal to compensate for the heat removed by flashing the pressurized liquid and heat loss to the surroundings, the pressurized pressure is decreasing rapidly as the steam bubble expands.

f. Pressurized steam bubble may expand until superheated steam is rising and condensing in the circulating liquid at pressurized convection to the hot leg piping.

Mode 2

Decay Heat Removal by Natural Circulation
With Reactor Vessel Pressure Controlling

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- a. Steam bubble forms in reactor vessel glenium when pressure pressure decreases to below the saturation pressure corresponding to top glenium liquid temperature.
- b. Reactor vessel head starts decreasing (top glenium steam bubble expanding) due to fluid loss through the break in excess of makeup capability.
- c. Reactor vessel top glenium liquid is partially flashing to steam (thru cooling) in an attempt to maintain constant pressure as the steam bubble expands.
- d. Every heat is providing an effective heat input to condensate for the heat removed by flashing the top glenium liquid, thereby assuring a liquid temperature and corresponding steam bubble pressure which follows closely

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the slowly changing effects of ice, heat reduction, heat loss to the surroundings, and heat transport through the break.

e. Pressurizing surge line is filled with semi-quiescent liquid which is subcooling to below the reactor vessel top pressure effective liquid temperature due to heat loss through the surge line in excess of heat gain by transport from the hot leg pipe.

f. Pressurizing steam bottle pressure is in equilibrium with slowly decreasing reactor vessel steam bottle pressure and is most likely contracting (pressurizing level slowly increasing) as steam from the pressurizing bottle is condensed in the pressurizing wall at a rate which is commensurate

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with heat loss to the surroundings.

Made 3²

Decay Heat Removal During Transition From
Natural Circulation³ to Core Boiling⁴ Made

a. Reactor vessel level reaches top of hot leg
nipples and starts to extend into hot leg
pipes and steam generator inlets due to
fluid loss through the break in process
of making capability.

a. Reactor vessel steam bubble enters pressure
connection and causes a heating and gravity
drain of core water in the pressure.

c. Reactor vessel steam bubble enters steam
generator inlets and causes a gravity
drain of water in the U-tubes back to
the hot and cold leg piping.

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d. Natural circulation through the reactor core and steam generator is disrupted.

e. Water in the reactor core starts to boil aggressively due to loss of natural circulation core cooling.

Mode 4²

Decay Heat Removal by Core Boiling⁴

a. Reactor vessel level continues to decrease due to fluid loss through the break in excess of makeup capability.

b. Reactor vessel steam bubble remains at saturation pressure corresponding to reactor core exit temperature and is fed directly to steam formed in the core at a rate which is commensurate with the decay heat input, steam generator condensation

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rate, and the volumetric expansion, and which follows closely the slowly changing effects of decay heat reduction, heat loss to the environs, and heat transport through the break.

C Liquid from condensation inside steam generator tubes is returned to reactor vessel by gravity countercurrent along the bottom of each partially filled hot leg pipes and by gravity flow to the pump loop seals and the cold leg pipes.

d Reactor core exit temperature and reactor vessel steam bubble pressure are decreasing in response to a decay heat reduction without a commensurate reduction in heat loss to the environs and heat transport through the break, and eventually in response to

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a reduction in steam generator pressure (and temperature) due to heat loss to the environs exceeding decay heat transferred after the safety valves are closed. If operator action is considered, the steam generator pressure may be reduced by opening atmospheric dump valves and thereby reduce the steam generator and reactor coolant system temperature).

e. Fluid loss through the break is decreasing and makeup rate is increasing due to reduction in reactor vessel pressure.

f. Reactor vessel level starts to increase as makeup capability exceeds fluid loss through the break.

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Mode 5²

Onset Heat Removal During Transition From
Cave Boiling⁴ to Natural Circulation³ Mode

- a. Reactor vessel level continues to increase to top of hot leg nozzle as makeup capability exceeds fluid loss through break.
- b. Steam generator inlet is flooded or severely choked by slug flow and entrainment thereby disrupting steam flow to the tubes.
- c. Reactor core exit temperature and corresponding reactor vessel steam bubble pressure start to increase in response to disruption of steam flow to the tubes.
- d. Reactor vessel level will decrease in response to the steam pressure increase, thereby releasing intermittent steam bubbles.

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to the steam generator tubes.

e. Reactor vessel steam bubble pressure will be communicated somewhat equally to the water filled hot and cold leg pipes due to a liquid coupling through the gap between the core barrel and the hot leg nozzles and a steam coupling through the ligament holes around the vessel flange area.

f. Reactor vessel level decrease (pressure increase) will be accompanied by a corresponding water level rise (pressure equalized) and steam pressure increase inside the steam generator tubes.

g. If sufficient water is available in the reactor vessel top plenum and if steam in the tubes can be condensed before the next cycle of steam bubbles reaches the steam generator, the tubes might become completely filled with

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liquid and the density gradients established which are required for natural circulation.

b. If sufficient quantities of non-condensable gases (principally dissolved hydrogen and oxygen released during degassing) have accumulated in the steam generator U-tubes due to prolonged sweeping of the core region and subsequent concentration during steam condensation at the top of the tubes, it will not be possible to refill the tubes and thereby establish natural circulation.

Mode 6²

Core Heat Removal After Re-establishing Natural Circulation³ With Reactor Vessel Pressure Controlling

a. Natural circulation is assumed to be re-established with a reactor vessel level

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above the hot leg noggins and with the steam generator tubes full of liquid.

b. Reactor vessel level continues to increase (top pressure steam bottle contracts) as makeup capability exceeds fluid loss through the break.

c. The rate of level increase will be commensurate with the reactor vessel steam bottle pressure which responds to changes in the heavy heat input, heat loss to the surroundings, and heat transport to the steam generator and through the break.

d. Pressurizer surge line is filled with liquid and pressurizer contains a trapped steam bottle which is in pressure equilibrium with the reactor vessel steam bottle.

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9. Pressure steam bubble is contracting as steam is condensed on pressure walls at a rate which is commensurate with heat loss to environs.

10. Pressure will eventually fall with liquid unless limited by non-condensable gases as the pressure steam bubble condenses.

Mode 7⁵

Decay Heat Removal by Forced Circulation⁶

Using Shutdown Cooling System⁷

1. Natural circulation has been previously established with reactor vessel level above hot leg nozzle and the pressure completely filled with liquid.

2. Reactor vessel level has been stabilized by adjusting make-up capability to equal

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fluid loss through the break. (In the long term, operator action may be used to re-establish a pressurized steam bubble and establish a stable pressurized level.)

C. Shutdown Cooling System (SCS) is in operation to provide forced circulation decay heat removal from the reactor core.

d. Reactor vessel top plenum is gradually filled with steam bubble at saturation pressure corresponding to top plenum liquid temperature (saturation pressure pressure control has been established).

e. Reactor vessel pressure is below SCS design operating pressure and decreasing at a rate commensurate with decay heat input, heat loss to surroundings, heat transport through break, and heat removal by the SCS.

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Notes

1. A very small break LOCA ($\leq 0.2 \text{ ft}^2 \text{ in CE}$) is one for which the steam generator must remove a significant portion of the decay heat; otherwise, reactor coolant system reflooding occurs since the break is too small to facilitate the transport of all decay heat to the surcons.

2. Assuming a very small break, reactor scram, loss of off-site power, no generator actions, minimum core cooling (no train) response using cold leg injection, closure of main steam isolation valves, and steam generator at saturation conditions corresponding to safety valve set points (as long as possible). The Safety Injection Tanks are assumed not to dump.

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3. Decay heat removal is by natural circulation through the reactor core and steam generator tubes with the core exit temperature (also assumed to be reactor vessel top plenum effective liquid temperature) established by the decay heat rate and natural convection heat transfer and transport coefficients.
4. Decay heat removal is by boiling heat transfer in the core, saturated steam heat transport from the reactor vessel to steam generator, and condensing heat transfer inside the steam generator tubes.
5. Assuming a very small break, loss of off-site power, operator actions permissible where required, closure of main steam isolation and steam dump valves, and maximum core cooling (one train) response

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using SCS for heat removal and safety injection, and charging pumps for makeup.

b. Decay heat removal is by forced circulation through the reactor core and SCS heat exchangers using SCS pumps in the shutdown cooling mode of operation with the core exit temperature (also assumed to be the reactor vessel top plenum effective liquid temperature) established by the decay heat rate and forced convection heat transfer and transport coefficients.

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Appendix B

Small Break LOCA Analysis

by Combustion Engineering

MS. A. 9.2.22

SMALL BREAK LOCA

Introduction

In a meeting held in Windsor on 5/27-5/28/75, Mr. Sabin of TVA expressed a concern for a class of small break LOCA's whose depressurization rates are relatively slow compared to those small breaks presented in CESSAR¹, Section 6.3.3.3. He worried about the possibility of inadequate HPSI flow for such breaks and also asked about the LOCA requirements for Emergency Feedwater and operator actions.

This transmittal constitutes a response to action item number 23 of Reference 2.

Summary

The results herein demonstrate that for any cold leg break less than or equal to 0.1 ft² in size, adequate liquid inventory is maintained in the reactor vessel to keep the active fuel completely covered at all times.

The results shown in CESSAR demonstrate that for those small breaks in which some degree of core uncover is predicted, the ECCS criteria³ are met and peak clad temperatures are less than 1200°F.

It has been assumed that for all breaks no operator actions are performed. At some later time following the LOCA, operator procedures for long term cooling will be implemented. This memo, however, does not address the long term cooling phase of the accident. Long term cooling performance and operator procedures will be discussed in a future submittal.

Since NRC requires the assumption of loss-of-offsite power, main feedwater flow is not available. For breaks larger than 0.1 ft^2 , those presented in CESSAR, Emergency Feedwater (EFW) was not represented and indeed was found to have little influence on the performance of such relatively large break sizes. However, for breaks equal to or smaller than 0.1 ft^2 , EFW was found absolutely necessary. For these smaller breaks, the steam generators must remove a significant part of the decay energy generated in the core. If they do not, then primary side re-pressurization occurs since the break is too small to emit all of the steam boiled off in the core.

Discussion

Calculations of breaks 0.1 ft^2 and smaller have been performed with the CELEVEL code. This code was specifically written for application to such small breaks.

The predicted two-phase level transients for several breaks 0.1 ft^2 and smaller are shown in Figure 1. The corresponding pressure transients are given in Figure 2. These results demonstrate that although the depressurization rates for such small sized break sizes are gradual, adequate HPSI pump inflow is provided to keep the core covered at all times.

The Safety Injection Tanks (SIT) become activated at approximately 900 seconds for the 0.1 ft^2 case. Neither the $.05 \text{ ft}^2$ nor the $.005 \text{ ft}^2$ cases resulted in SIT activation before the level transient is turned around. For both breaks, the level transient bottoms out at an elevation above the top of the active fuel and recovers from there via HPSI inflow alone.

For all of these breaks, steam generator heat transfer plays an important role. Since EFW is supplied to each steam generator, an adequate secondary liquid inventory is maintained to support heat removal by condensation of the steam in the U-tubes on the primary side. The secondary

inventory is assumed to be saturated at a temperature corresponding to the lower setpoint of the safety valves (1270 psia). The condensation film coefficient used was approximately 200 BTU/hr-ft²-°F. This is a conservative value which properly accounts for the influence of non-condensibles.

To clearly indicate the influence of EFW, Figure 3 has been included. Results are shown for the reactor vessel pressure and level transients for cases with and without emergency feedwater. As shown, if EFW is not available a re-pressurization occurs which in turn reduces HPSI inflow and the level transient is significantly aggravated. Neither case shown in Figure 3 should be construed to specifically represent the performance of System 80. EFW is available on all System 80 plants. The case without EFW is shown only for demonstrative purposes. For both cases shown, the HPSI capacities assumed were much less than those of System 80. Thus, the level transient are more severe.

Major Assumptions

As required by NRC³, the following conservative assumptions were applied in analyzing all small breaks (≤ 1.3 ft²):

1. Loss of offsite power assumed upon a scram signal.
2. The worst single failure of emergency equipment was assumed (failure of one emergency diesel generator to start).
3. An uncertainty margin of +20% was added to the calculated decay heat generated in the reactor.
4. Conservative setpoints and delays were assumed for scram, SIAS and equipment startup.

Assumption (1) results in a description of the accident in which the reactor coolant pumps begin coastdown, the main steam isolation valves shut, and the main feedwater pumps stop upon receipt of a scram signal. Main steam and feedwater flows are ramped to zero in 0.5 seconds.

Assumption (2) allows credit for only one train of ECCS pumps and one train of Emergency Feedwater (EFW). Thus, only one HPSI pump and one EFW pump are available to satisfy LOCA requirements. It was additionally assumed that 25% of the HPSI inflow spills out through the break.

The flow capacity requirements for the HPSI and EFW pumps are proportional to values of decay heat. Assumption (3), therefore, results in a proportionate increase in these requirements. The HPSI pump capacity assumed is shown in Figure 4. The EFW capacity was assumed to be 375 gpm (minimum).

Assumption (4) resulted in the following setpoint and delay assumptions:

- a) Reactor scram occurs upon a low pressurizer pressure (LPP) indication of 1728 psia (minimum). Upon receipt of the LPP signal there is a 0.9 second delay before the control rods begin motion followed by a 3.5 second interval to fully insert the rods.
- b) SIAS occurs upon a pressurizer pressure indication of 1550 psia (minimum). The SIAS automatically starts the emergency diesel generators which provide power to the HPSI pumps and the EFW pumps. The total time from receipt of an SIAS to the time the HPSI pumps provide flow is 30 seconds (maximum). The total time delay from SIAS until EFW reaches the steam generators is 45 seconds (maximum).

References

1. Combustion Engineering, Inc., Standard Safety Analysis Report (CESSAR), Docket No. STN-50-470, Amendment No. 31, Section 6.3.3.3.
2. D. V. Graf to D. R. Patterson, TD-CE-117, subject: "Meeting Minutes of 5/27 and 5/28 in Windsor."
3. Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, Federal Register, Vol. 39, No. 3, Friday, January 4, 1974.

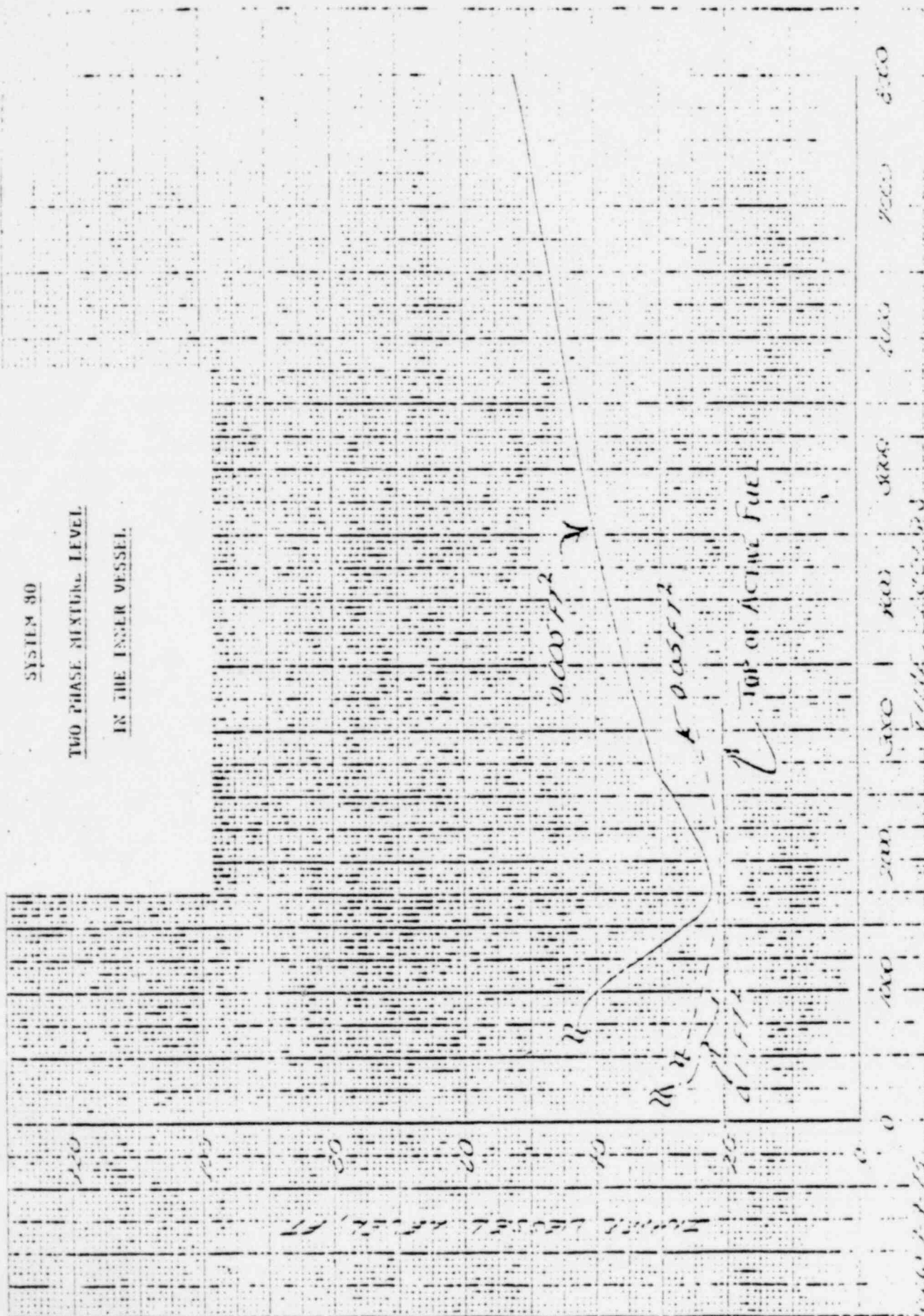


FIGURE 1

SYSTEM 80

TWO PHASE MIXTURE LEVEL

IN THE INNER VESSEL



PRIMARY SYSTEM PRESSURE



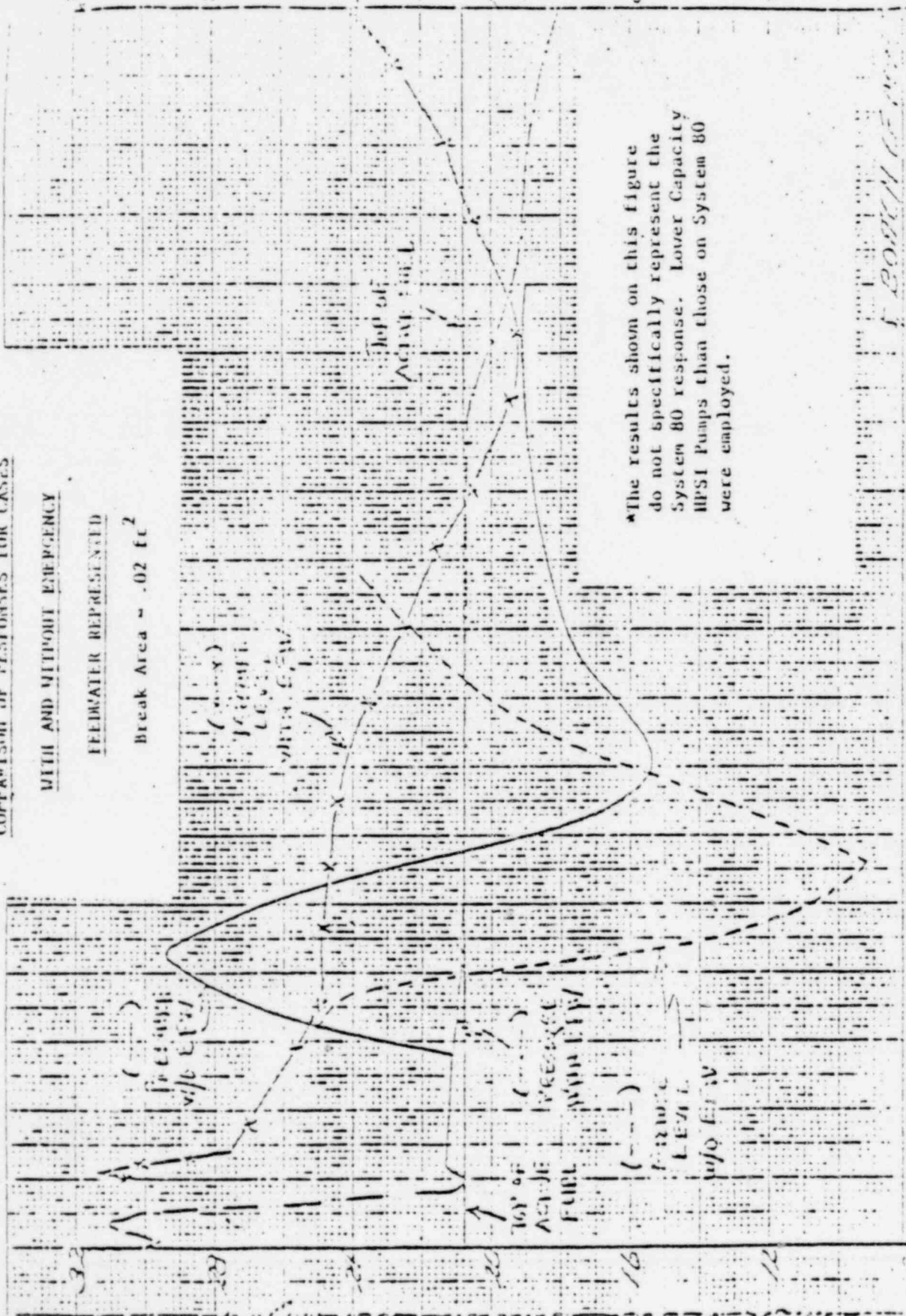
FIGURE 3

COMPARISON OF RESPONSES FOR CASES

WITH AND WITHOUT EMERGENCY

FEEDWATER REPRESENTED

Break Area = .02 ft²



The results shown on this figure do not specifically represent the System 80 response. Lower Capacity HPST Pumps than those on System 80 were employed.

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120000 100000 80000 60000 40000 20000 0

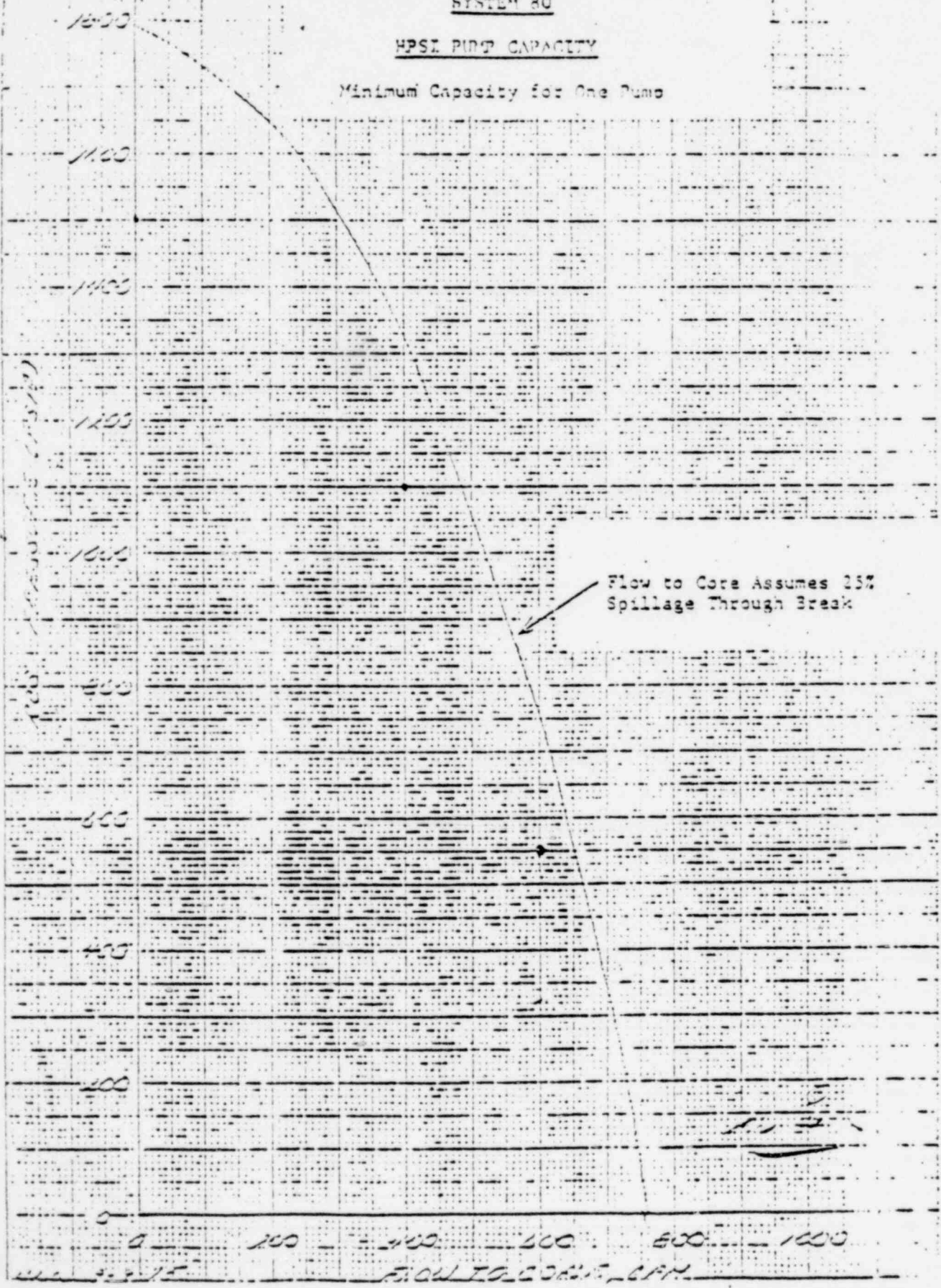
FIGURE 4

SYSTEM 80

HPSI PUMP CAPACITY

Minimum Capacity for One Pump

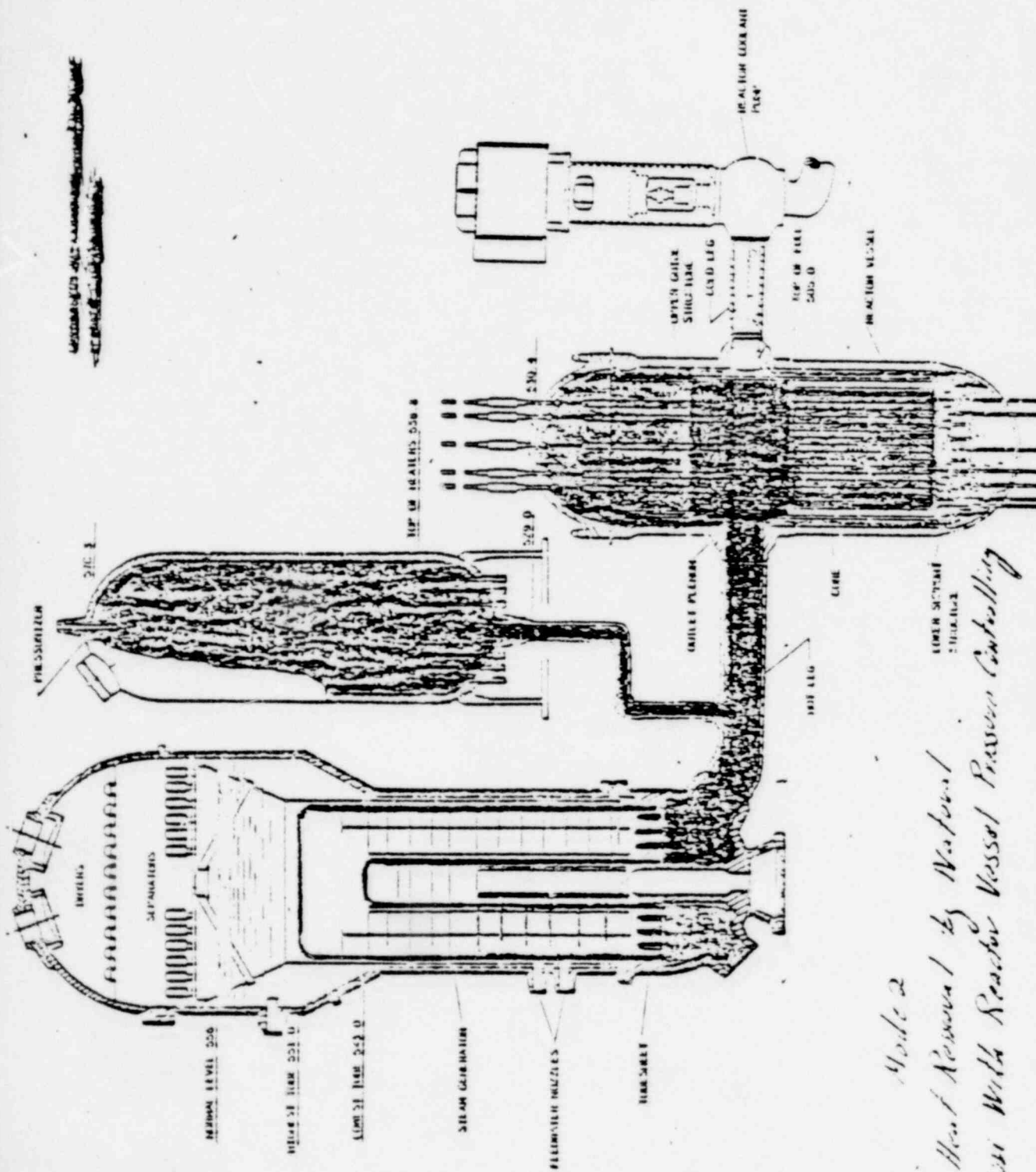
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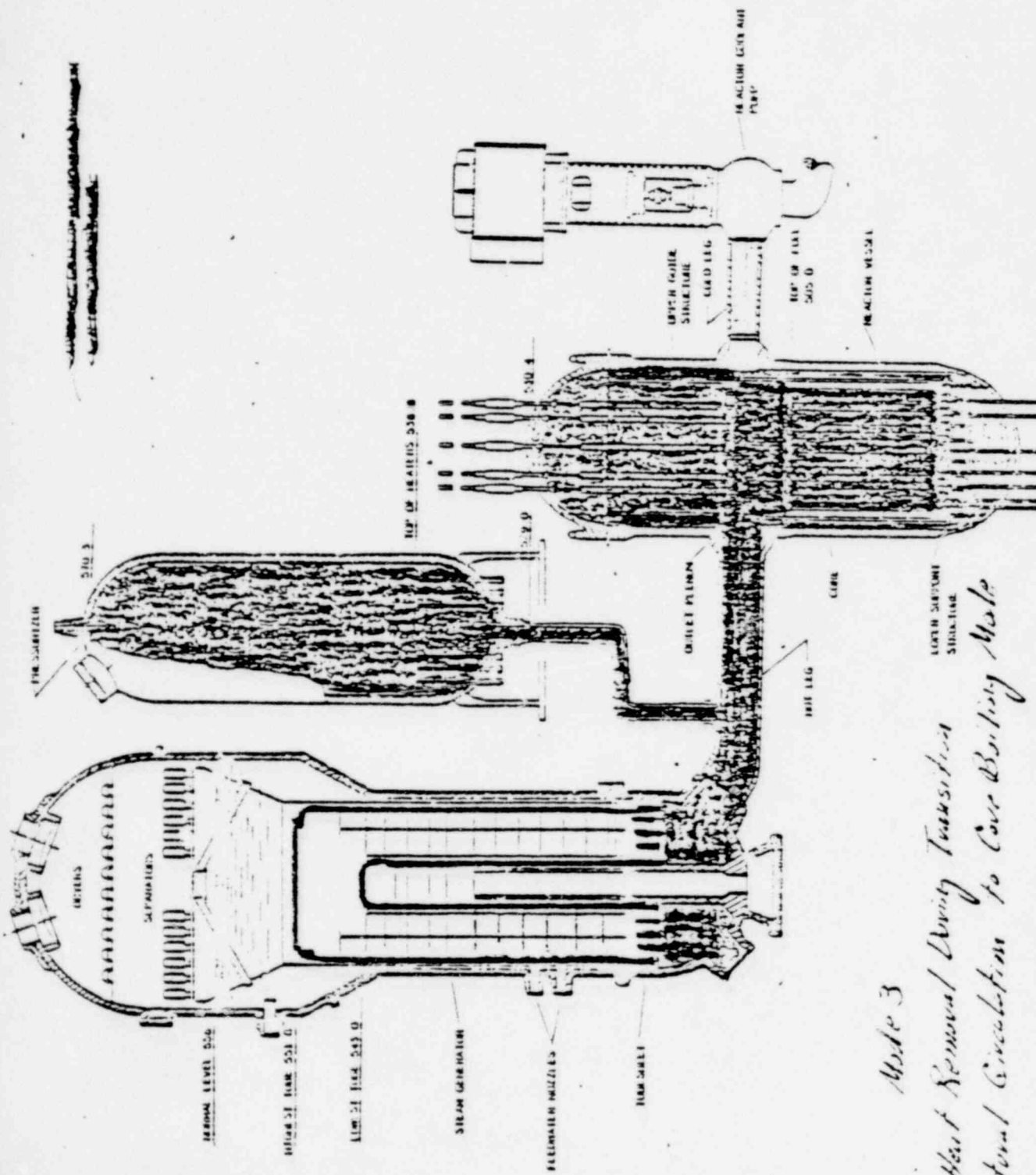
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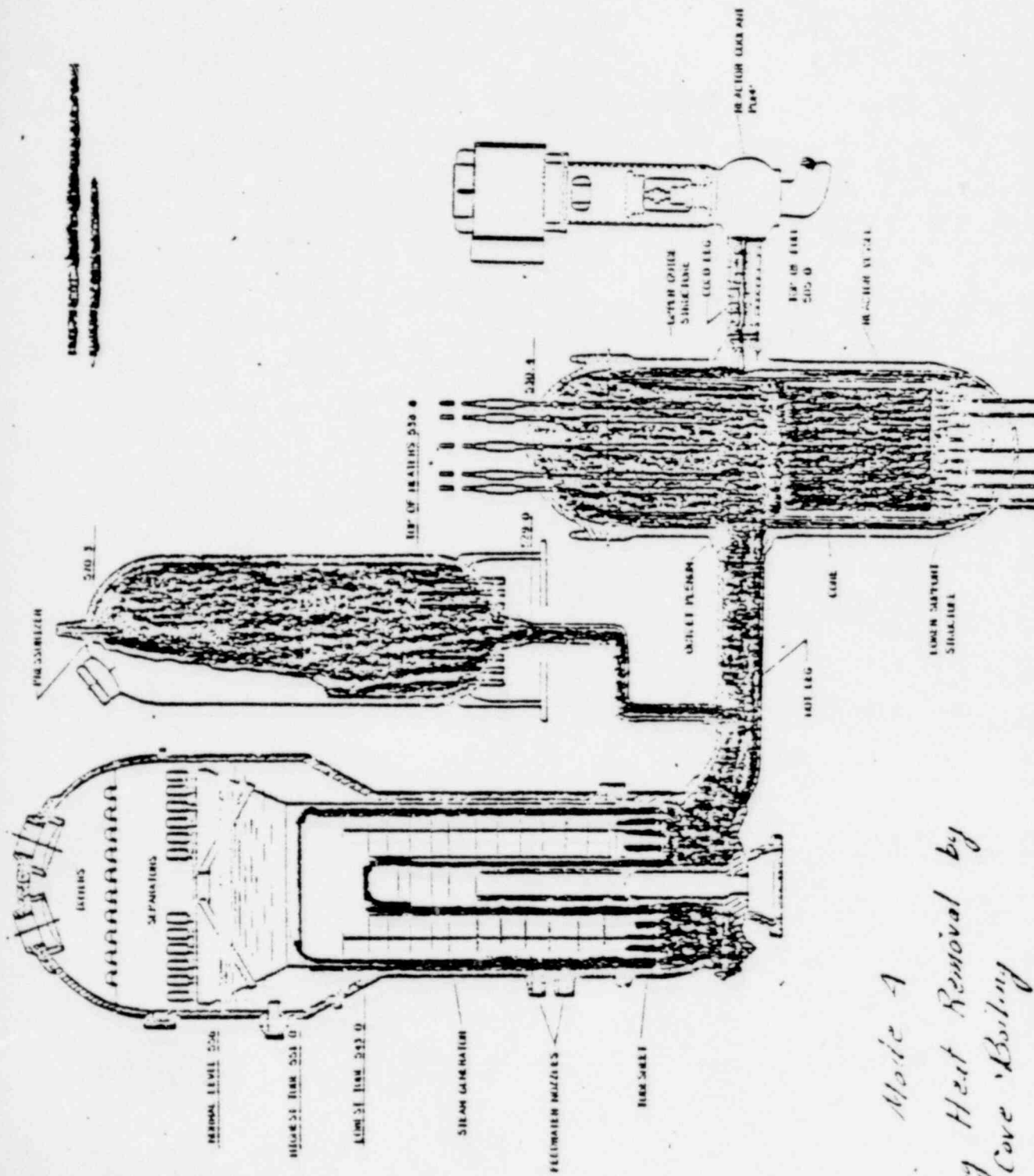
Delay that Result by Natural
Circulation With Pressure Controlling



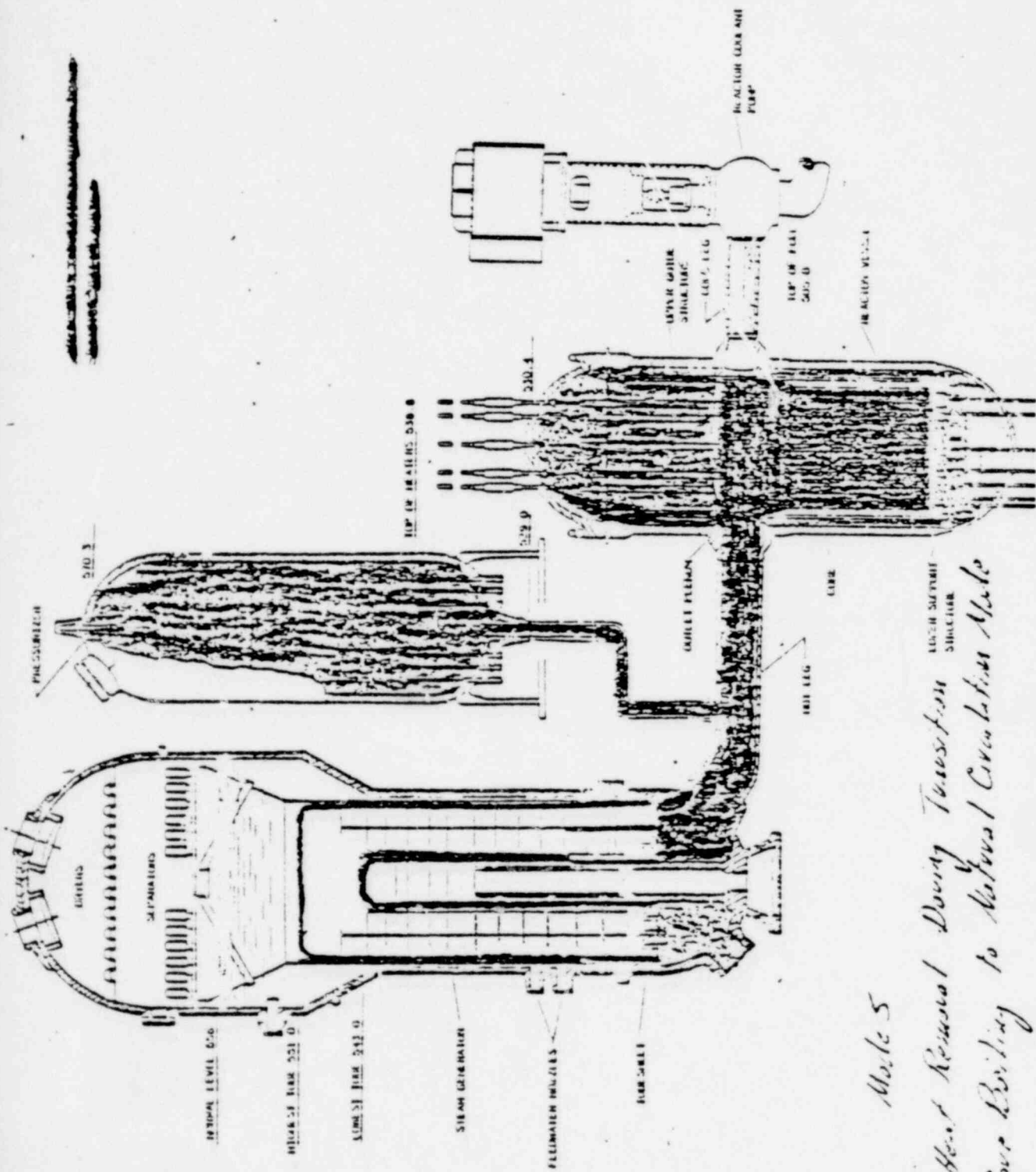
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Decay Heart Removal During Transition
From Natural Circulation to Core Binding Mode



Mode A
 Decay Heat Removal by
 Core Boiling



Model 5

Very Hot Remotely Driven Transition
From Core Boiling to Natural Convection Mode

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JAN 10 1979

NOTE TO: RSB Members

FROM: T. M. Novak

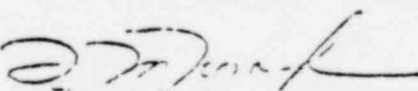
SUBJECT: LOOP SEALS IN PRESSURIZER SURGE LINE

Loop seals in the pressurizer surge line are used in some plant designs (noted in B&W). Under ordinary circumstances, these configurations are inconsequential because the saturation temperature in the pressurized (650°F) is the highest temperature in the primary system. However, under upset conditions (such as prolonged relief valve opening) and accidents where significant voids are formed in the primary system, it may be possible to end up with a two-phase mixture in the pressurizer that is not at the highest temperature in the primary system. Under these circumstances, additional loss of primary system inventory or shrinkage in the primary system may not be indicated by pressurizer level. This situation has already occurred at Davis Besse 1 when a relief valve stuck open.

The loop seal results in a manometer effect as shown in Fig. 1. If there is saturated steam at 1200 psi in the hot leg pipe and the level in the pressurizer is 60 feet, the pressurizer pressure, P_z , would only have to be about 1100 psi, which corresponds to saturation temperature about 50°F below that in the hot leg. Thus, the pressurizer temperature does not have to be significantly lower than the temperature in the primary system.

Although the safety analyses do not require termination of the makeup system, operators would control makeup flow based on the pressurizer level as part of their normal procedures. As a result, under certain conditions where the pressurizer could behave as a manometer, the operator could erroneously shut off makeup flow when significant void occurs elsewhere in the system or loss of inventory is continuing.

It is recommended that the bases for the design requirement be studied carefully for all CP reviews with the object of determining if the loop seal can be eliminated. For OL reviews, procedures should be reviewed to ensure adequate information before the operator terminates makeup flow.


Thomas M. Novak, Chief
Reactor Systems Branch

Enclosure:
Figure 1

cc: D. Ross

Contact: Sandy Israel, NRR
49-27591

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Enclosure 7

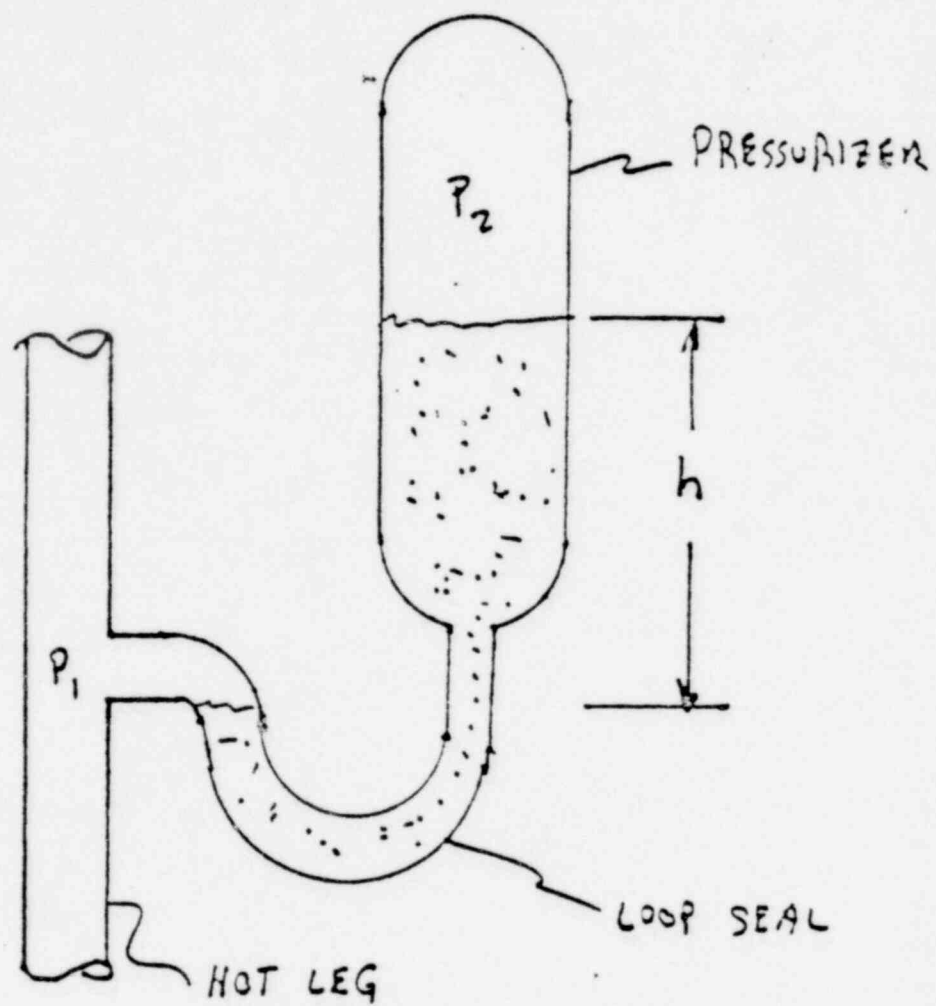


Fig. 1 SCHEMATIC OF PRESSURIZER



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 26, 1979

H. Denton
E. Case
D. Eisenhut
D. Ross
R. Tedesco
S. Hanauer
Z. Rosztoczy

Attached is a document concerning B&W transient and accident analysis methods I received this morning. This is the subject of today's meeting with B&W.

Roger J. Mattson, Director
Division of Systems Safety

cc PDR

Enclosure 8

DUPLICATE

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BABCOCK AND WILCOX COMPANY

April 26, 1979

TO: Roger J. Mattson, Director, Division of Systems Safety, NRR
FROM: D.H. Roy, Manager, Engineering, Babcock and Wilcox Company
SUBJECT: SUBMITTAL OF DETAILED ANALYSES OF B&W OPERATING PLANT
RESPONSE TO TRANSIENTS

- REFERENCES: 1. Letter Taylor to R. Mattson, dated April 25, 1979.
2. D.R. Patterson to J. McFarland, April 26, 1978,
"Emergency Core Cooling System Small Break LOCA
Analysis N4M-2-14 (AR).
3. J. McFarland to D.R. Patterson, dated January 23, 1979,
Letter D3132.
4. D.F. Ross, Jr., to E.G. Case, April 23, "Summary of Mtg.
with B&W Regarding Natural Circulation Considerations".
5. Logic Flow Charts

B&W will submit to the Staff detail results of the analyses described in the reference 1 (attached). The results of these analyses are currently (as of 0800 hrs 4/26/79) under review by a team of NRC and B&W engineers.

These analyses show that the B&W transient analysis codes are capable of predicting the response of B&W operating plants to anticipated transients compounded by postulated equipment malfunctions (e.g., delayed emergency feedwater supply, failure of pilot operated and/or code safety valves to seat following actuation), and operator actions which may interrupt safety system operation (e.g., securing or throttling of ECCS flow).

These analyses also show that the small break which results from operation with the pilot operated relief in the open position does not result in core damage as long as ECCS flow is maintained as designed including effects of single failure.

In addition, these analyses show that delay in emergency feedwater and ECCS initiation following a worst case small break in the pressure boundary of B&W plants operating at up to 2772 Mwt will not result in unacceptable consequences if the delay does not exceed 20 minutes from time of break initiation.

The concerns raised by C. Michelson in Reference 2 (attached) were addressed by B&W in Reference 3 (attached). The results of B&W's evaluations of these concerns were discussed with the NRC staff on April 17, 1979 (Reference 4, attached). B&W concludes that the concerns raised by C. Michelson do not

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invalidate the results of B&W's small break analyses performed in accordance with 10CFR50.46 and Appendix K requirements. B&W's detailed evaluation of the Michelson report and basis for these conclusions will be submitted to the staff on or before May 7, 1979.

On April 24, 1979 B&W presented to the NRC Staff event trees showing the expected paths of system behavior following loss of main feedwater initiating events and assuming various failures in subsequent operation of plant safety systems and equipment (see Attachment 4). The analyses discussed in paragraphs 1, 2, and 3 above bound the expected system performance shown in these event trees.

B&W commits to perform additional analyses as follows:

1. B&W operating plant response to a total loss of steam generator heat sink (loss of main and emergency feedwater flow).
2. Sensitivity study of response of B&W operating plants to emergency feedwater flow rate following loss of main feedwater (including overcooling event) with and without compounded equipment failure.
3. Analyses of the effect on system response of anticipatory trip on loss of main feedwater flow in comparison with trip on high RC pressure with and without emergency feedwater delay, and main reactor coolant pump availability.

These analyses will be carried out to the time stable long-term core cooling is established. The natural circulation cooling mode can and will be included if calculated to occur during any portion of the course of the transient.

A schedule for submittal of these additional analyses will develop in a meeting with the NRC Staff on April 26, 1979.

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 245

Telephone: (804) 384-5111

April 25, 1979

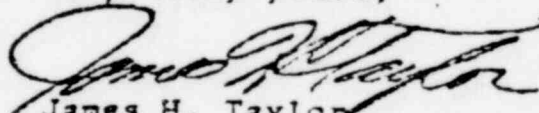
Dr. Roger J. Mattson, Director
Division of Systems Safety
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Mattson:

In the NRC Staff meeting with B&W and its 177-Fuel Assembly plant owners on Tuesday, April 24, 1979, B&W agreed to describe the scope and status of certain additional analytical work to support the continued safe operation of these plants. The attachment documents this information.

We intend to provide this information to you on or before Friday, May 4, 1979, and suggest that a detailed technical presentation of the information be made to the appropriate Staff engineers at that time with follow-up meetings as deemed necessary.

Very truly yours,


James H. Taylor
Manager, Licensing

JHT:dsf

Attach.

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DUPLICATE DOCUMENT

Entire document previously
entered into system under:

ANO 7905020348

No. of pages: 9

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24503

Telephone: (804) 384-5111

January 23, 1979

Letter No. D-3132
File Ref: M4M-2/12B14
Ref Ltr: K-5020/4-27-78

Tennessee Valley Authority
400 Commerce Avenue
Knoxville, TN 37902

Attention: Mr. D. R. Patterson
Chief Mechanical Engineer

Bellefonte Nuclear Plant Units 1 & 2
Contract No. 71C62-54114-2
B&W Reference: NSS-15 & -16
Subject: Small Break LOCA Analysis

Gentlemen:

The attached report is in response to your reference letter. Please let us know if further discussion is required.

Very truly yours,

James McFarland
Senior Project Manager

Robert E. Lightle
By
Robert E. Lightle
Associate Project Manager

REL:dc
Attachment

cc: W. Brent Wade
J. L. Atchison
E. L. Logan

Response to TVA Letter K-5020, Emergency Core Cooling System -
Small Break LOCA Analysis W4M-2-14(AR), April 27, 1978

Via TVA Letter K-5020, TVA transmitted to B&W a report entitled, "Decay Heat Removal During a Very Small LOCA for a B&W 205-Fuel-Assembly PWR," by C. Michelson, dated January, 1978. This report presents a simplified, hand calculation review of the small break transient and potential consequences for very small breaks not explicitly examined within the small break topical for the 205 FA plant, BAW-10074A, Rev. 1. Within this paper, the following concerns were expressed for the very small breaks:

1. How is decay heat removed?
2. Will system repressurization occur? If so, could a smaller case be a worst break?
3. If the operator isolates the break, will system repressurization occur? If so, will the pressure relief valves be subjected to slug or two-phase flow?

Responses to these concerns are developed in the subsequent paragraphs.

Before discussing these concerns, a general overview of the small break transient in a B&W 205 plant needs to be briefly discussed. Small LOCAs can be viewed as a slow transient during which the RCS can be described as a sealed manometer. Because of the internals vent valves, extensive steam bubble will form within the reactor vessel while any significant liquid inventory remains in the loop. Many experiments have been run which show that so long as a fluid, with qualities less than 70% or so, covers the core, no adverse core temperature excursion will occur at decay heat power levels. Thus, any problems with small breaks will only occur after the RCS loops have depleted their inventory.

Decay heat removal from the core region is no problem as stated above. However, decay heat removal from the system as a whole needs to be examined further. There are two ways of removing decay heat from the system; via the break and/or via the steam generator. Both of these items are discussed in detail in the TVA letter. For the very small LOCAs of interest in this discussion, it was shown that the break alone is not capable of removing all the decay heat and heat removal via the steam generator is necessary. While the TVA-predicted break size that this occurs at was not checked quantitatively, the actual break size that it occurs at is inconsequential. Such a break size does exist where the steam generators are necessary.

The role of the steam generator as a heat removal source is basically as described in the letter. Initially, natural circulation will be maintained in the system and the necessary heat removal is easily accomplished. Once a steam bubble of sufficient size necessary to fill the U-bend at the top of the hot legs is formed, natural circulation will cease. The intermittent natural circulation discussed in the letter will not occur due to the slow nature of the small break transient. Once natural circulation ceases, the system will repressurize somewhat until the SG primary side liquid level drops

below the SG secondary side level and condensation heat transfer is established. During this period between the natural circulation and condensation heat removal modes, the letter expresses concerns that the liquid inventory within the system will be depleted at a rate in excess of the rates for the breaks analyzed by BSW because of the partial repressurization of the system. TVA is concerned that this ultimately will result in more core uncover than that shown in the small break topical report BAW-10074A. This is not the case.

During the natural circulation phase, it is obvious that the smaller the break, the slower the loss of system inventory and the longer the period of natural circulation. After natural circulation ceases, system pressure will be controlled by a "volume balance." That is, the system pressure will balance at a point where the volume of fluid discharged through the break equals the volume of steam being created in the core. Since the cold leg fluid enthalpy remains unchanged during a small break transient, the volume relief out the break increases with increasing system pressure and break size. The volume of steam being generated in the core decreases with increasing pressure. As the break decreases in size, the RC system will repressurize to a higher value; thus the volume relief out the break necessary to match the volume of steam being created decreases. Therefore, the system inventory will be lost at a slower rate as break size decreases. Once the SG becomes available for condensation heat removal, the primary system pressure will depressurize to approximately the SG secondary side pressure. Since the secondary side of the SG will respond in a similar manner to that of the 0.05 ft² break analyzed in the topical, the primary side pressure response, following the advent of condensation heat removal, will be similar to that of the 0.05 ft² break. Thus, for the smaller breaks, the system inventory will always be greater than that for the 0.05 ft² break and the core will always remain covered and will not undergo a temperature excursion.

In the paper concerns are raised relative to isolation of the break after natural circulation is lost. The scenario presented in the letter is reasonable. Should the break be isolated at that time, system repressurization to the pressurizer safety valve setpoint is probable. Two-phase or liquid flow through the safety valves will also probably occur. Once the system depletes sufficient inventory to establish condensation heat transfer across the SG, the system will depressurize and no further loss of inventory will occur. The core will remain covered for this scenario and no temperature excursion occurs. Should the pressurizer safety valves become damaged because of the two-phase flow out the valves, the response of the system would then be similar to that presented in the FSAR for the pressurizer safety valve stuck open accident and no core uncover occurs. (attached)

what if
the
valve
stuck
open
←

As far as the appropriateness of the operator using pressurizer level indication to trip the RPI pumps, BSW agrees that the level indication is not a reliable indication of the state of the RCS. However, use of the pressurizer level indication, along with system temperature and pressure measurements to ensure that the system is still in a substantially subcooled state, will provide sufficient guidance for operator action.

In summary, while the TVA paper raises valid concerns and gives a detailed examination of the small break transient, the small break topical report provides sufficient analyses to ensure the ability of the BSW 205 plant ECCS system to control small break in the RCS.

3. The RCS and main steam system pressure shall not exceed 110% of the design pressure.
4. Resultant doses shall not exceed 10 CFR Part 100 limits.

15.6.1.3.2 Methods of Analysis

The analysis of the effect of the inadvertent opening of the pressurizer safety valves was performed using the small break model documented in B&W topical report BAW-10074A, Revision 1, "Multinode Analysis of Small Breaks for B&W's 205-Fuel-Assembly Nuclear Plants With Internals Vent Valves." The small break model used in that topical was demonstrated to comply with the requirements of Appendix K to 10 CFR Part 50 in Appendix A of BAW-10104, "B&W's ECCS Evaluation Model." Flow through the open safety valve was calculated by the Moody critical flow correlation. The leak area is limited by the orifice area of the safety valve, which is 0.03 ft². In order to obtain rated flow through the valve at the valve rated pressure, it was necessary to use a discharge coefficient of 0.75. Other assumptions used for the analysis are as described in BAW-10074A, Revision 1.

15.6.1.3.3 Results of Analysis

Following the inadvertent opening of a pressurizer safety or relief valve, the design pressure of the RCS is never exceeded. The RCS depressurizes to approximately the secondary side pressure of the steam generator. The secondary side pressure is maintained at approximately 1250 psia by the safety valves on the steam generator. Since the design pressure of the main steam system is 1250 psia, the system is maintained at below 110% of its design pressure.

Figure 15.6.1-1 presents the inner vessel liquid volume (lower plenum, core, and upper plenum) for this accident. Since the core remains covered by liquid throughout the transient, pool nucleate boiling will be maintained and the cladding temperatures will remain within a few degrees of the water temperature. No metal-water oxidation is incurred, and no abnormal core geometries or fuel rod damage is caused by the low cladding temperatures.

By 3200 seconds, the injection rate from the one HPIS exceeds the combined leak rate and core steaming rate and will provide long-term cooling capability. Therefore, compliance with the acceptance criteria of 10 CFR Section 50.46 is ensured.

The discharge coefficient of 0.75 was chosen to match valve performance with the Moody correlation. If 0.75 is assumed, the rated flow through the valve at the valve's rated pressure is calculated by the Moody correlation. Although this matches the valve performance, it should be noted that the discharge

coefficient assumed for this accident is not important. The system's response following the opening of a pressurizer safety valve is similar to that calculated for small breaks. B&W topical report BAW-10074A, Revision 1, discusses the consequences of small breaks and demonstrates that the core remains covered throughout the transient for breaks with leak areas less than 0.1 ft². Since the leak area for an open safety valve is only 0.03 ft², the core will remain covered throughout the transient, and no fuel rod damage will occur regardless of the discharge coefficient assumed.

15.6.1.4 Barrier Performance

The inadvertent opening of a pressurizer safety valve does not result in fuel damage or excessive pressure in the Reactor Coolant System or main steam system since pool nucleate boiling maintains the cladding temperature within a few degrees of the water temperature and peak pressures never exceed the Code design limits.

15.6.1.5 Radiological Consequences

Since no fuel clad damage occurs, there will be no increase of radioactivity in the reactor coolant or in the steam. Because the doses are a function of the amount of steam released, the potential radiological consequences are bounded by the consequences of the loss of onsite and offsite ac power to the station transient (Section 15.2.6).

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

15.6.2.1 Identification of Causes

A break in fluid-bearing lines that penetrate the containment could result in the release of radioactivity to the environment. There are no instrument lines connected to the RCS that penetrate the containment. However, other piping lines from the RCS to the Makeup and Purification System and the Decay Heat Removal System do penetrate the containment. Leakage through fluid penetrations not serving accident consequence limiting systems is minimized by a double barrier design so that no single credible failure or malfunction of an active component will result in loss of isolation or intolerable leakage. The installed double barriers take the form of closed piping, both inside and outside the containment, and various types of isolation valves.

The most severe pipe rupture relative to radioactivity release during normal plant operations occurs in the Makeup and Purification System. This would be a rupture of the letdown line just outside the containment but upstream of the letdown control valves. A rupture at this point would result in a loss of



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MEMORANDUM FOR: E. G. Case, Deputy Director, Office of Nuclear Reactor Regulation

FROM: D. F. Ross, Jr., Deputy Director, Division of Project Management, NRR

SUBJECT: SUMMARY OF MEETING WITH B&W REGARDING NATURAL CIRCULATION CONSIDERATIONS

On April 18, 1979 the NRC staff met with representatives of Babcock & Wilcox (B&W) Corporation in Bethesda, Maryland to discuss several considerations related to natural circulation in B&W reactors. A representative of Duke Power Company was also in attendance. A list of attendees is attached (Enclosure 1).

The meeting opened with a presentation of the proposed agenda. The following four (4) general areas were to be discussed:

1. Concerns raised by the ACRS in a recent letter.
2. A recent B&W precaution regarding subcooling, RCP operation and OTSG level while attempting to establish natural circulation in the RCS.
3. A report written by C. Michelson, a consultant to the ACRS, regarding potential difficulties in the removal of core decay heat for certain small break LOCAs in the 205 class B&W reactors.
4. Staff concerns related to the ICS and how it effects natural circulation in certain scenarios (loss of all RCPs, loss of off-site power).

1. ACRS Concerns

A. Greater Understanding of Natural Circulation

B&W explained the basic principles of natural circulation on the B&W system. There are three (3) basic criteria which must be met for natural circulation to be established in the RCS.

1. There must be an elevation head between the thermal centers of the system.
2. The RCS loops must be wat temperature, T_H , must be temperature for the press

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Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505

Telephone: (804) 384-5111

May 7, 1979

Dr. Roger J. Mattson, Director
Division of Systems Safety
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Mattson:

The attached document is in response to B&W's commitment to provide information related to plant behavior during certain transients with special emphasis on various small break scenarios. This completes the submittal of small break information required by the NRC staff prior to plant restart.

The information contained in this document allows the following conclusions to be reached:

1. Analytical codes are available that accurately predict plant response to various transients and small break scenarios. These codes have been benchmarked against the TMI-2 data.
2. Based upon the ability to accurately predict plant response to transients and small breaks, operating procedure guidelines have been developed. It has been concluded that adequate time and information are available to the operator to assure continued safe operation.
3. Natural circulation in both the lowered loop and raised loop configuration plants is ample to assure core cooling following a loss of power to the reactor coolant pumps. This has been demonstrated both by planned tests and several unplanned transitions from forced to natural circulation.
4. The incorporation of either the early reactor trip on loss feedwater or the modified setpoints for the pilot operated relief valve and high pressure trip is predicted to eliminate lifting the PORV during anticipated transients.

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Babcock & Wilcox

Dr. Roger J. Mattson

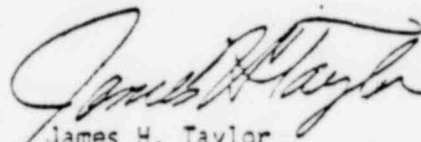
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May 7, 1979

5. Investigations into the worst break size and location have led to the conclusion that the reactor core is safely cooled for all conditions.

Discussions with the NRC staff during the past two weeks have covered much of the information in this document. I request, however, that you please bring any questions regarding this information to my attention immediately so that operation at Oconee may continue and the restart of other plants will not be delayed.

Very truly yours,



James H. Taylor
Manager, Licensing

JHT:nw

Attachment

cc: RB Borsum (B&W)

APPENDIX 5

B. ASSESSMENT

OF

"DECAY HEAT REMOVAL DURING A VERY SMALL
BREAK LOCA FOR A B&W 205 FUEL ASSEMBLY
PWR", JANUARY, 1978, C. MICHELSON

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