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Decay Heat Removal Problems Associated With

Recovery From a Very Small Break LOCA

for B&W 205-Fuel-Assembly PWR

Nuclear Systems Analysis Section

September, 1977

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1. Introduction

The Nuclear Systems Analysis Section recently undertook 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 generators must remove a significant portion of the decay heat; otherwise, reactor coolant system repressurization occurs since the break is too small to facilitate the transport of all decay heat to the environs. In this class of LOCA's, 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 pumps. An ongoing qualitative consideration of this concern

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now predicts a similar decay heat removal problem during post-LOCA recovery which needs to be resolved for a Babcock & Wilcox 205-Fuel-Assembly PWR such as the Bellefonte 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-Fuel-Assembly plant such as Bellefonte is shown in Figure 1. The plant has two once-through steam generators each with a single hot leg piping connection (38 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 44 foot long surge line (14 inch,

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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.

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 (notably ≤ $0.05 ft^2$) for which the steam generators must 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 environs. Modes 1-6

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are based on loss of off-site power, no operator action, minimum core cooling (one train) response using cold leg injection, isolation of main steam line attachments, and steam generators 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-LOCA period. These conditions are summarized in Table 3.

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The achievement of Modes 1 and 2 appears straightforward. The transition from natural circulation to core boiling during Mode 3 may be a problem because of the time delay incurred while waiting for the water level in the U-bend region of each hot leg pipe and in the steam generator tubes to drain to below the secondary side water level. In very small breaks, this could take several minutes during which time there is no significant removal of decay 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 reactor steam can communicate with an effective condensing surface.

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An additional concern associated with Mode 3 is the possibility of break 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 repressurization and subsequent two-phase fluid release through the pressurizer safety valves. However, the available full pressure makeup capability may not be commensurate with the short term boiloff rate required to remove all of the decay heat through the valves. In addition, the safety valves have not been qualified for this severe service. Another alternative might be to open the pressurizer relief valve

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to replace the isolated break 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 or supplied with on-site power or safety-grade controls.

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

Operations 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 reflooding of the horizontal hot leg pipes will disrupt normal steam flow from the core to the

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steam generator tubes and will lead to an intermittent voiding of the hot leg piping as the steam is forced through. However, phase separation 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 water elevation 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 repressurize.

In order to re-establish natural circulation decay heat removal which is Mode 6, it is necessary that Mode 5 be terminated by refilling the U-bend region of the hot leg piping with liquid and establishing the density gradients required for natural circulation. It is not clear how this can be accomplished. This could be a problem with significant accumulations of non-condensable gases in the U-bend region. If enough non-condensable gases have accumulated, natural circulation cannot be re-established (unless the gases can be removed by some means).

If Mode 5 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 decay heat removal. Makeup to the primary side may have to be terminated which is a questionable operation in view of the absence of level information. Furthermore, since the letdown system performs

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no safety function, it is in single track configuration and is operable only on off-site power. If the break was isolated earlier, it is not clear that the required letdown can be achieved. If the break has not been isolated, it may be able to provide the required letdown in time. Although valves have been provided for a direct drain to the sump to control post-LOCA boron buildup, these valves are not designed to operate at the high pressures associated with the very small break LOCA. If decay heat removal cannot be achieved, full repressurization and subsequent two phase fluid release through the pressurizer safety valves will occur. If makeup has been terminated this operation will result in a loss of liquid inventory and eventual establishment of appropriate liquid level inside the steam generator tubes for decay heat removal.

If makeup has not been terminated, the safety valves 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 severe service.

7. Available Quantitative Results

Information concerning a Babcock & Wilcox analysis for the small break LOCA for a B&W 205-Fuel-Assembly plant is included as Appendix B. This analysis is for a 0.05 square foot pump discharge break which is near the upper size limit for a very small break LOCA but at the lower calculation limit for the ECCS analysis of small breaks. It is based on the usual ECCS evaluation rules.

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The results shown in Figure 4-34 of

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Appendix B indicate that the reactor vessel level will drop to the top of the core within 10 minutes following the break, thus leading to the predicted eventual occurrence of the first six modes of operation outlined in Appendix A. The results show that the core will remain virtually covered at all times. The NRC evaluation of the B&W small break LOCA analysis is included as Appendix C.

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5. Areas of concern

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

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condensing surface for decay heat removal. It would be an additional concern if the operator should isolate the break and thereby disrupt a timely achievement of the required primary side drain.

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, decay heat removal will cease. It is not clear that natural circulation can be re-established by continuing the refilling 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

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a condensing surface. This will require reducing or terminating the makeup (a questionable idea) and waiting for the water to drain through the break. Again, it would be an additional concern if the operator should isolate 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 this 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 B&W 205-Fuel-Assembly plant because of the once-through steam generator configuration

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6. Additional Considerations

A loss of offsite power was assumed in preparing this report. If the very small break should not be accompanied by an offsite 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 trip the pumps within a few minutes. Physical damage might be induced by excessive vibration or a severe flow instability. In addition, it is not clear what effect the flow instability will have on the maintenance of an adequate core flow for decay heat removal. The large heat input of the pumps might cause steam binding in the pump and a resulting

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flow interruption. These reactor coolant pump concerns were not investigated.

The very small break postulated for this report 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 existence of the predicted operating modes given in Appendix A, i.e., there exists one or more break sizes and locations which will result in the occurrence of each predicted mode.

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Table 1

Reactor Coolant System Elevations

Component		Elevation (ft)*
Pressurizer:	Vessel High Point (inside)	669
	Vessel Low Point (inside)	630
	Hot Leg Connection ℄	643
	Full Power Water Level	652
	After Scram Water Level	645/634**
	Top of Heaters	641
	Horizontal Surge Pipe ℄	625
Steam Generator:	Top Tubesheet (inside face)	684
	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

* Approximate values.

** Value depends upon transient selected.

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Table 2

Reactor Coolant System Characteristics

Full Power Temperatures (°F): (3600 Mwt)	Core Entry	572
	Core Midplane	601
	Core Exit	630
	Pressurizer Liquid	650
	SG Steam	603
	SG Feedwater	473
Full Power Pressures (psia):	RC Pump Discharge	2268
	Core Midplane	2235
	Core Exit	2205
	SG Steam	1075
Safety Valve Settings (psia):	Pressurizer	2500
	Steam Generator	1250/1280
Volumes (ft^3):	Pressurizer (steam space)	1050
	Pressurizer (above heaters)	1200
	Pressurizer (total)	2250
	Reactor Vessel (above hot legs)	1550*
Actuation Pressures (psia):	Safety Injection Pumps	1600
	Core Flooding Tanks	600
	Reactor Scram	1945

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* Approximate value

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Table 3

Post-LOCA Modes of Decay Heat Removal[a]

Mode	Condition	Press. Level	RV Level	System Press.	SG Tube Level
1	Natural Circulation[b]	Decr. R	Full	Decr. R	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[f]	Decr. S[g]	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. R 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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d. Reactor vessel steam bubble has formed.

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

f. While vertical hot leg pipes are empty.

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

h. While vertical hot leg pipes are filling.

i. stable after reactor vessel level has reached top of hot leg nozzles.

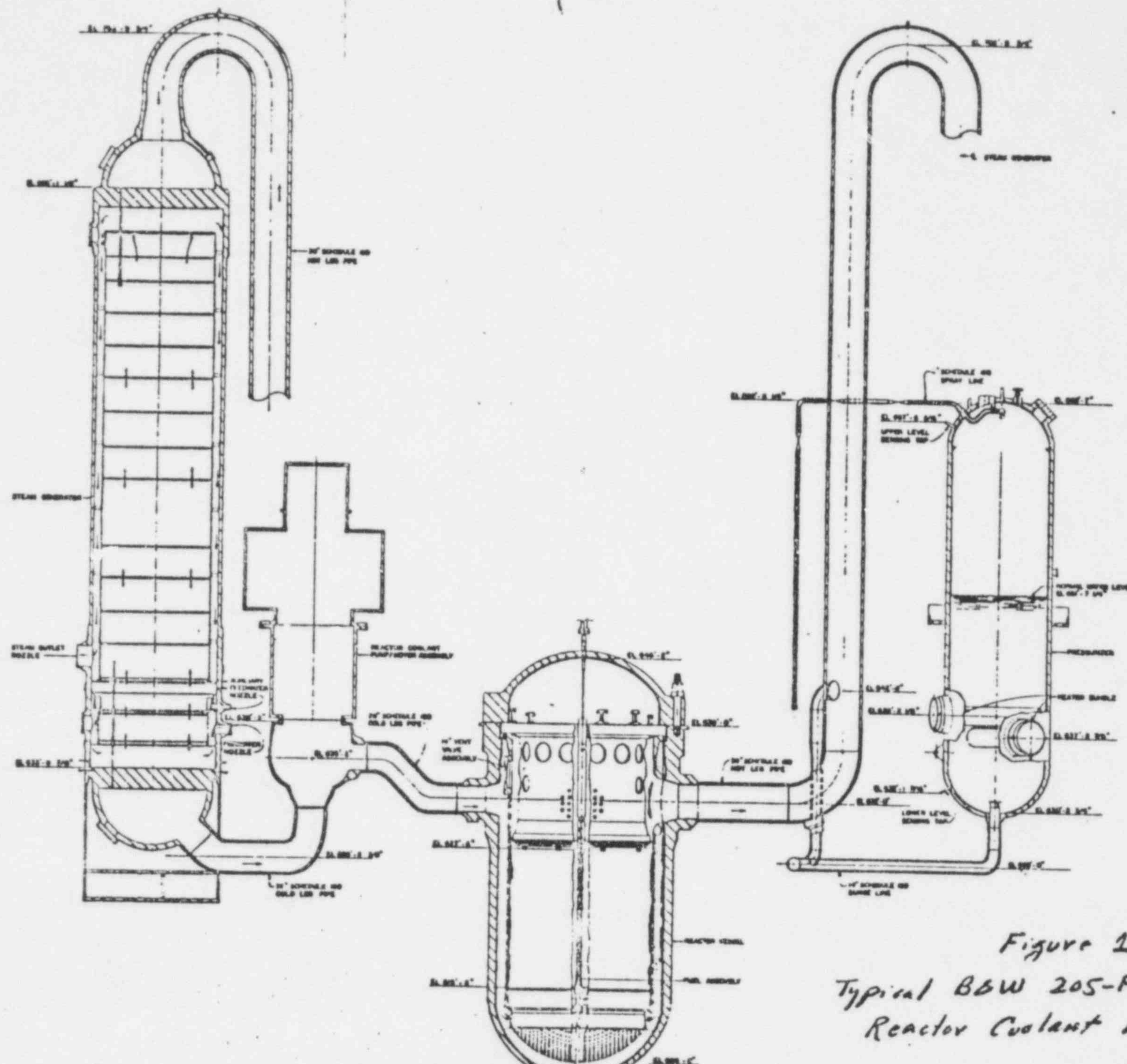

Figure 1
Typical B&W 205-Fuel-Assembly
Reactor Coolant Loop

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Appendix A

Conditions Associated With Post-LOCA Modes of Decay Heat Removal for Very Small Break LOCA*,1

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* Numbered notes are included at end of appendix.

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Mode 1[2]

Decay Heat Removal by Natural Circulation[3]
With 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.

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c. Pressurizer liquid is partially flashing to steam (thereby cooling) in an attempt to maintain constant pressure as the steam bubble expands.

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d. Pressurizer steam bubble is superheating to pressurizer effective metal temperature (thereby cooling walls).

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 environs, the pressurizer pressure is decreasing rapidly as the steam bubble expands.

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

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Mode 2[2]

Decay Heat Removal by Natural Circulation[3] 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.

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c. Reactor vessel top plenum liquid is partially flashing to steam (thereby cooling) 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 plenum liquid, thereby assuring a liquid temperature and corresponding steam bubble pressure which follows closely the slowly changing effects of decay heat reduction, heat loss to the environs, and heat transport through the truck.

e. Pressurizer surge line is filled with semi-quiescent liquid which is subcooling to below the reactor vessel top plenum effective liquid temperature due to heat loss through the surge line in excess of heat gain by 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 likely contracting (pressurizer level slowly increasing) as steam from the pressurizer bubble is condensed on the pressurizer wall at a rate which is commensurate with heat loss to the environs.

Mode 3[2]

Decay Heat Removal During Transition From Natural Circulation[3] to Core Boiling[4] 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. As 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 generators.

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

d. Water in the reactor core starts to boil agressively 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 sink for decay heat).

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

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h. Pressurizer level increases with a commensurate water level decrease in U-bend region of the hot leg piping as remaining steam bubble in pressurizer is compressed to reactor vessel steam bubble pressure (colder loop seal water maintains a lower average water temperature in pressurizer).

i. Reactor vessel hot leg and cold leg pipes equalize to nearly the same pressure by vent valve actuation (minimum 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 generator through cold leg piping.

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k. Water level in the U-bend region of each hot leg pipe decreases due to fluid loss through break in excess 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 4[2]

Decay Heat Removal by Core Boiling[4]

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

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b. Significant heat removal occurs by condensation inside the steam generator tubes when the water level in tubes drops below 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, bubbles into the hot leg piping, and breaks away at the water-steam interface in the vertical hot legs at a rate which is commensurate with the steam condensation rate in the steam generators and local heat loss to the environs.

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 environs, and heat transport through the break.

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e. Pressurizer level continues to increase as remaining steam bubble in pressurizer cools and is compressed to reactor vessel steam bubble pressure while the hot water drawn from the hot leg pipe into the pressurizer loop seal piping is allowed to cool.

f. Pressurizer level ceases to increase when the water level in the vertical hot leg drops to below the pressurizer loop seal nozzle attachment to the hot leg pipe (subsequent damage may not occur or 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 nozzles until water level in the vertical hot legs drops 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 exceeds fluid loss through break.

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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 environs and heat transport through the break, and eventually in response to 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 temperatures).

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

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

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m. Pressurizer level increases as pressurizer continues to cool.

Mode 5[2]

Decay Heat Removal During Transition From Core Boiling[4] to Natural Circulation[3] Mode

a. Reactor vessel level continues to increase to top of reactor vessel hot leg nozzles and a water level appears in the vertical hot leg pipes as makeup capability 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 valve action (also due to gap between reactor upper internals and hot leg nozzles and a steam space coupling through the bypass holes [illegible] the [illegible] flung away.

c. [illegible] 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 cooler regions of the unfilled portions 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. Decay heat removal is accomplished by condensation inside the steam generator tubes if 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 does not inhibit an adequate condensation rate.

f. Pressurizer level increases as any remaining steam bubble in the pressurizer cools and water is drawn from the hot leg pipe into the pressurizer loop seal piping where it can cool.

g. Water level continues to increase inside steam generator tubes and vertical hot leg pipes as makeup capability exceeds fluid loss through break.

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 temperature and corresponding reactor vessel steam bubble pressure start to increase in response to disruption of steam condensation.

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j. Makeup water continues to fill the vertical hot leg piping and steam generator tubes and add to the pressure increase.

k. The U-bend region of the hot leg piping contains steam non-condensible gases at saturation conditions corresponding to about reactor vessel steam bubble conditions.

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

m. If the pressurizer 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

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pressure in commensurate with the decay heat removal requirements. (Note that since the pressurizer piping is below the vertical hot leg water level, the safety valve will pass water and not steam. Thus, the decay 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 pumps is very low for this overpressure condition).

m. If the decay 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 vertical hot leg level drops below the pressurizer piping and the pressurizer water inventory is expelled.

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

Mode 6[2]

Decay Heat Removal After Re-establishing Natural Circulation[3] 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 plenum steam bubble contracts) as makeup capability exceeds fluid loss through the break.

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[illegible] d increase will be commensurate with the reactor vessel steam bubble pressure which responds to changes in the decay heat input, heat loss to the environs, and heat transport to the steam generators and through the break.

d. Pressurizer is filled with liquid but may contain a trapped steam and non-condensable gas bubble which is in pressure equilibrium with the reactor vessel steam bubble.

e. Pressurizer steam bubble contracts as steam is condensed on pressurizer walls at a rate which is commensurate with heat loss to environs.

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Mode 7

Decay Heat Removal by Forced Circulation[6]

Using Decay Heat Removal System

a. Natural circulation has been previously established with reactor vessel level above hot leg nozzles and the pressurizer essentially filled with liquid.

b. Reactor vessel level has been stabilized by adjusting makeup 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 establish a stable pressurizer level.)

c. Decay Heat Removal (DHR) System is in position to provide forced circulation decay heat removal from the reactor core.

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

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

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Notes

1. A very small break LOCA (probably < 0.1 ft² for B&W) is one for which the steam generators must remove a significant portion of the decay heat; otherwise, reactor coolant system repressurization occurs since the break is too small to facilitate the transport of all decay heat to the environs.

2. Assuming a very small break, reactor scram, loss of off-site power, no operator actions, minimum core cooling (one train) response, isolation of main steam line attachments, and steam generators at saturation conditions corresponding to safety valve set points (as long as possible).

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3. Decay heat removal is by natural circulation through the reactor core and steam generators

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 heat transport coefficients.

4. Decay heat removal is by boiling heat transfer in the core, saturated steam heat transport from the reactor vessel to steam generators, 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 atmospheric dump valves, and minimum core cooling (one train) response using DHR for heat removal and high pressure injection for makeup.

Decay heat removal is by forced circulation through the reactor core and DHR heat exchangers using DHR 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 Babcock & Wilcox

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BAW-10074A, Rev 1
Topical Report
March 1976

MULTINODE ANALYSIS OF SMALL BREAKS FOR B&W's 205-FUEL-ASSEMBLY NUCLEAR PLANTS WITH INTERNALS VENT VALVES

— Revision 1 —

by

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Lynchburg, Virginia

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 THETA1-B 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

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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-ft^2 leak area.) The CRAFT,[1] FOAM,[2] and THETA1-B[3] computer codes were used for the analysis.

A spectrum of ruptures is considered: 0.5-, 0.3-, 0.1-, and 0.05-ft^2 leak areas. Also, the report includes the double-ended rupture of the core flooding line (0.44-ft^2 leak area). These ruptures provide an appropriate spectrum for evaluation of the effects of small leaks.

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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 1636F. 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.[8] Therefore, the design of the emergency core cooling system is adequate to control small loss-of-coolant accidents. 1

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

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5 seconds of the accident. At 280 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 exiting the system. The core flooding tank actuation pressure is reached at 775 seconds. At 1300 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 1390 seconds after the break occurs and causes an increase in the system pressure. Long-term cooling is established at 2550 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-ft^2 Break at Pump Discharge

The 0.05-ft^2 break at pump discharge is the smallest break analyzed. This break responds basically in the same manner as the 0.1-ft^2 break at the pump discharge as shown in Figures 4-31 through 4-34. The reactor trip occurs 8 seconds after the break, the auxiliary feedwater pump shuts off at 1520 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 HPI 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.

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Figure 4-31. Core Pressure for 0.C5-ft^2 Break at Pump Discharge

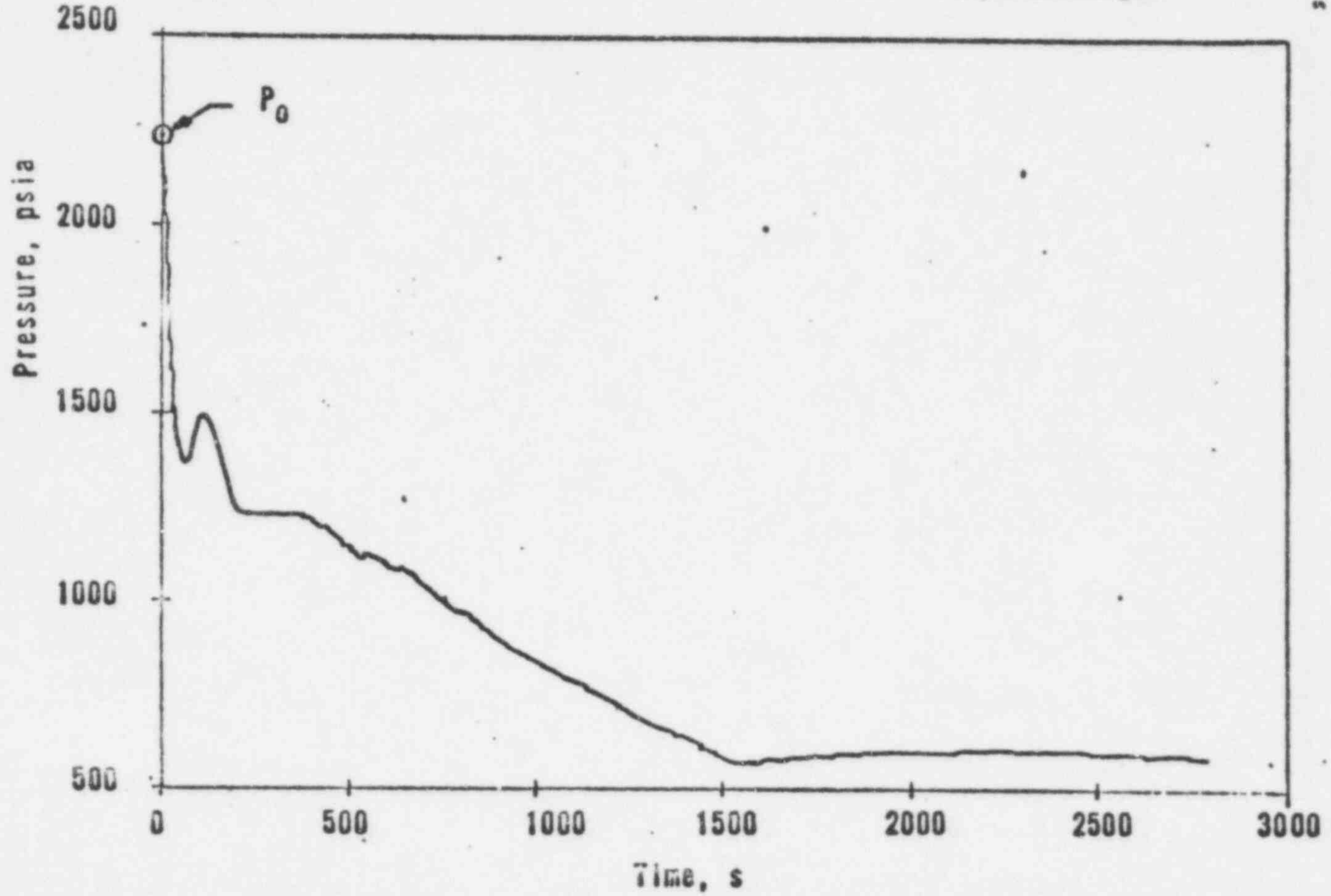
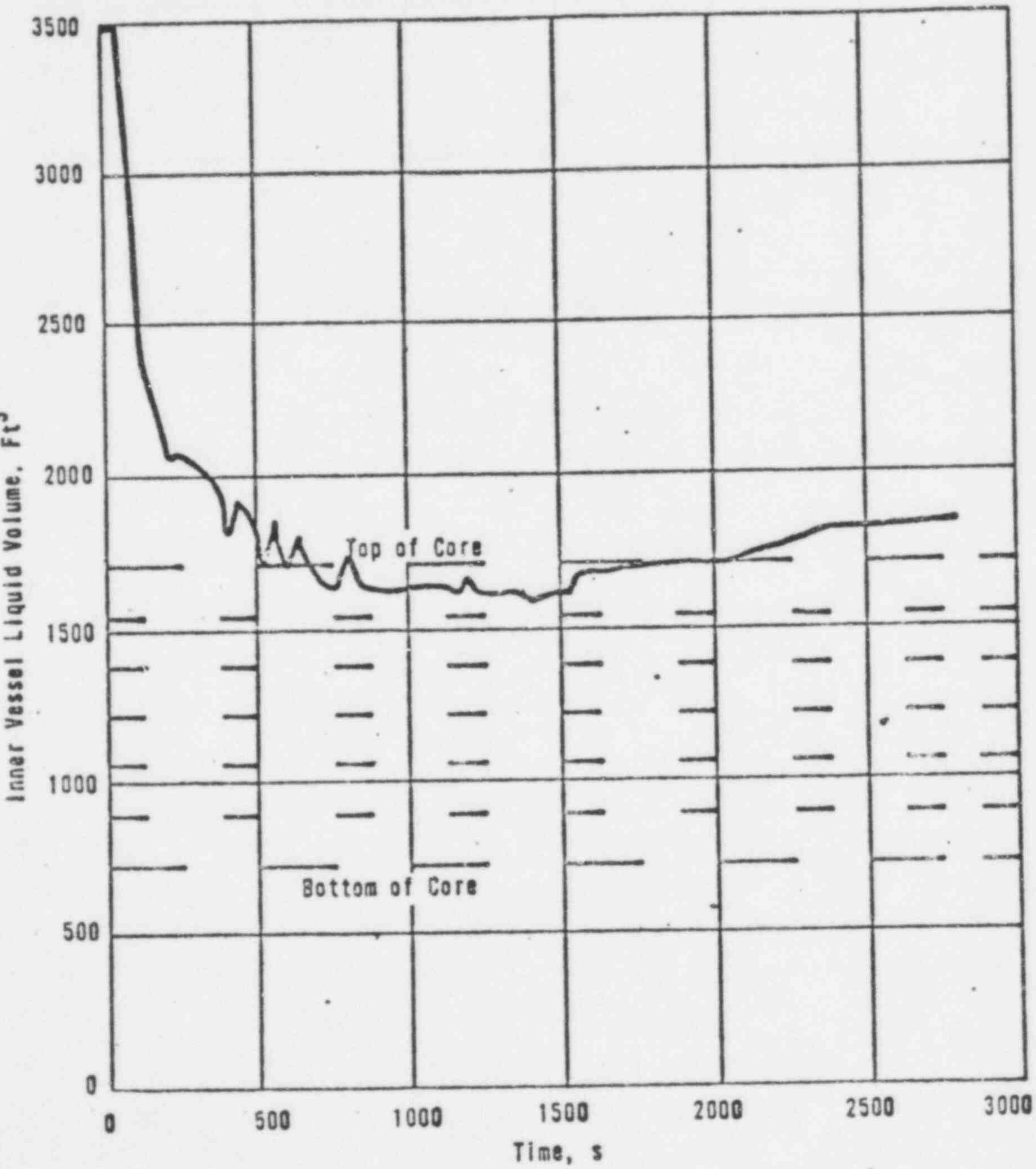


Figure 4-34. Inner Vessel Liquid Volume for 0.05-ft^2 Break at Pump Discharge



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COMPUTED DATE

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Appendix C

NRC Evaluation of B&W Small Break LOCA Analysis