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DECAY HEAT REMOVAL DURING A VERY SMALL

BREAK LOCA FOR A B&W 205-FUEL-ASSEMBLY PWR

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SUMMARY

This report gives an account of some initial considerations of a class of very small break LOCA's (probably ≤ 0.05 ft^2) for a B&W 205-Fuel-Assembly PWR which may have an associated decay heat removal problem. The results indicate that one or more impediments to decay heat removal appear to exist which need to be better understood if proper operator response and adequate mitigation are to be assured. Of particular concern is the acceptability of intermittent natural circulation following the postulated LOCA, and system repressurization following the loss of natural circulation. Also of concern is the possibility of break isolation by operator action resulting in repressurization and slug or two-phase flow through a presurizer safety valve. These uncertainties may reflect on the adequacy of proposed emergency operating procedures and operator training for a very small break LOCA.

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1

1.0 INTRODUCTION

There appears to be a class of very small break LOCA's (probably ≤ 0 0.05 ft^2) for a B&W 205-Fuel-Assembly PWR which may have an associated decay heat removal problem. For this discussion, a very small break LOCA is one for which the steam generator must remove a significant portion of the decay heat during the initial phase of blowdown; otherwise, reactor coolant system repressurization occurs since the break is too small to facilitate the transport of all decay heat to the environs. For this class of LOCA's, depressurization rates are relatively slow (when compared to these normally analyzed as small breaks) and thus may seriously limit the makeup available from the high pressure injection pumps. An ongoing qualitative consideration of this problem now predicts the development of one or more impediments to decay heat removal during a very small break LOCA. This has become a concern that needs to be understood.

The physical arrangement of the reactor coolant system for a typical 205-Fuel-Assembly plant such as the TVA Bellefonte Nuclear Plant is shown in Figure 1. Plant elevations corresponding to various points in the reactor coolant system are indicated. Decay heat removal considerations during the post-LOCA period are based on the usual ECCS rules such as loss of offsite power, minimum core cooling (one train) response, and no short-term required operator actions.

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2.0 LOCA CHARACTERISTICS

A few elementary calculations can be helpful in developing a better understanding of the various modes of post-LOCA decay heat removal and the role which steam generators must play during a very small break LOCA. Of particular interest is the mass flow rate through a postulated break, its capability to remove decay heat, and the makeup rate to the reactor coolant system required to compensate for any lost mass. Also of interest is the time required to drain the reactor vessel top plenum and steam generator tubes during a transition to pool boiling, and the minimum time required to refill the steam generators if a level turnaround occurs.

Question the assumption

For most calculations it is assumed that a quasi-steady-state condition exists with the reactor coolant system at 1270 psia (574.4°F) and the secondary side of the steam generators relieving steam to atmosphere through a safety valve. Where applicable, it is assumed that a sufficient temperature differential exists across the steam generator tubes to transfer all decay heat not removed by the break. The decay heat is based on the ANS decay heat curve[1] using a 20 percent margin. The reactor power is 3672 Mwt. The fluid upstream of the break is assumed to be saturated water or steam. The flow areas (potential single-ended flow break areas) corresponding to a number of nominal pipe sizes of interest are listed in Table 1.

Not necessarily true

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[1] Proposed American Nuclear Society Standards - "Decay Energy Release Rates Following Shutdown of Uranium - Fueled Thermal Plants" (approved by Subcommittee ANS-5, ANS Standards Committee, October 1971).

The pipe schedules indicated are typical for the reactor coolant pressure boundary.

It should be recognized that calculations included in this section are based on general considerations of energy and mass conservation under ideal fixed conditions. Fine structure effects have been ignored to facilitate simple hand calculations. However, the results should still be useful for general guidance if the simplifying assumptions and calculational limitations are appreciated. Detailed transient calculations based on appropriate system heat transfer and fluid flow models and core thermal-hydraulic models are required to put these concepts on a firm basis.

intermittent - natural circulation is not fine structure?

2.1 Mass Flow Rate Through Break

The mass flow rate through a break assuming saturated water or steam upstream of the break is shown in Figure 2. The saturated water and steam curves are based on Moody's* Figure 3 for stagnation pressures of 1270 psia and 2500 psia with saturated liquid and vapor entrance properties. Moody discharge coefficients (C_D = actual flow/Moody calculated flow) of 0.6 and 1.0 were selected for calculating the saturated water case. The actual coefficient might be somewhere between as determined by such considerations as the break configuration and whether it is in a large or small pipe. The discharge coefficient for steam is assumed to be 1.0.

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*F. J. Moody, "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Journal of Heat Transfer, Transactions of the American Society of Mechanical Engineers, 87, No. 1, February, 1965.

The makeup rate available to the reactor coolant system to compensate for mass lost through the postulated break is shown in Figure 2 as that available from one high pressure injection (HPI) pump at the indicated system conditions. The water available for makeup is assured to be at 70° F.

2.2 Decay Heat Removed Through Break

XCS?

The decay heat which is removed by mass flow through a postulated break at 1270 psia (assuming saturated water or steam upstream of the break) is shown in Figure 3. The heat removed is based on the mass flow rate given in Figure 2 and a stagnation enthalpy corresponding to the designated upstream condition, i.e., saturated water or steam. A minimum value for C_D (0.6) was selected to yield a conservative (minimum) value for the heat removed. The decay heat removed is shown as a percent of total decay heat generated at the indicated time following the break.

is correct

It should be noted that saturated steam upstream of the postulated break will remove a little larger percentage of the decay heat than saturated water. This will not be the case for larger values of C_D with saturated water upstream. The larger mass flow rate for water can more than compensate for the difference in saturation enthalpy at the indicated upstream pressure.

The time after a break before all decay heat can be removed through the break (with saturated upstream conditions) is indicated in Figure 4. This is the time required for the break to be in energy equilibrium with the decay heat. The effect of

higher pressure and a range of possible C_D values for water is also shown.

2.3 Reactor Vessel Top Plenum Drain Time

The time required to drain the reactor vessel top plenum down to the top of the hot leg pipes is shown in Figure 5. The drain time is assumed to start after reactor trip. Pressurizer level is assumed to remain at the level achieved immediately after trip. The pressure is assumed to be a constant 1270 psia. In reality, the pressurizer pressure will be somewhat higher during a portion of the drain time thereby reducing the drain time indicated. Although some additional pressurizer draining may occur, it is most likely that a refilling will commence shortly after the reactor vessel steam bubble starts to form and become controlling. Therefore, the drain time given in Figure 5 is thought to be a good estimate of the maximum time that natural circulation can be sustained for a given break size after reactor trip.

I don't understand this

2.4 Steam Generator Drain Time

The time required to drain the steam generator inlet piping, plenum, and tubes down to the secondary side water level is shown in Figure 6. This drain time is calculated to start when natural circulation is assumed to be lost indefinitely, i.e., when the water level at the top of the steam generator drops to below the inside diameter of the U-bend high point and restoration of level is not expected. The drain rate is assumed to be the mass flow rate through the postulated break (Figure 2). System makeup is from one high pressure injection (HPI) pump delivering a constant

[illegible]

mass flow at the indicated system conditions upstream of the break. The secondary side water level is assumed to be increasing at a rate of 1.75 ft/min which is attributed to a 600 gpm auxiliary feedwater flow. The initial secondary side water level is assumed to be 26 ft. above the bottom tube sheet when the steam generator drain is started.

The effect of upstream pressure and initial secondary side water level on steam generator drain time is also shown. An initial secondary side level of 6 ft. is the normal auxiliary feedwater control point until an engineered safety features actuation signal (ESFAS) is initiated at 1600 psia. It is likely that the steam generator level will be in the range of 6 to 26 ft. at the time natural circulation is lost. The effect of an initial level of 48.5 ft (top of SG shroud) is included. This is an upper limit for practical consideration since the overflow enters the main steam lines. The effect of 2500 psia reactor coolant pressure (set point of pressurizer safety valves) is included for reference.

Note I don't have these figures

2.5 Steam Generator Refill Time

The steam generator refill time is the time required to refill the inlet pipe and plenum and steam generator tubes if a level turnaround should occur when the primary side water level first reaches the secondary side water level. Typical times are shown in Figure 7 for saturated water at 1270 psia upstream of the break. The initial steam generator level is that level which existed when the steam generator drain started, i.e., when natural circulation was lost. It is likely that the initial

steam generator level will be in the range of 6 to 26 ft. The curves all start at 12 minutes because this is the minimum time required to refill the inlet pipe and plenum and that portion of the steam generator tubes which is above the shroud.

3.0 MODES OF POST-LOCA DECAY HEAT REMOVAL

After the loss of offsite power and reactor coolant pump coastdown, the two basic modes of post-LOCA decay heat removal from the reactor core for very small breaks are natural circulation and pool boiling. Decay heat is removed from the reactor coolant system by single or two phase blowdown through the postulated break and by a release of steam on the secondary side of the steam generators. Any decay heat lost to the environs through thermal insulation is small by comparison and is neglected. If the break size is assumed to be sufficiently large, two or more of the following phases of operation may be experienced.

3.1 Natural Circulation

Natural circulation starts with the pressurizer still controlling system pressure at approximately 2250 psia. However, without an effective heat input (other than the pressurizer metal) to compensate for heat removed by flashing the pressurizer liquid, the pressurizer level and pressure decrease rapidly as the steam bubble expands to fill the void created by fluid lost through the break exceeding liquid mak p capability (LOCA condition). Figure 2 indicates a break area exceeding 0.005 ft^2 could create this initial condition at high pressure.

A significant system volumetric contraction and corresponding pressurizer bubble expansion occurs when reactor trip is initiated at 1945 psia. The bubble superheats to the pressurizer effective metal temperature. The bubble may expand until superheated steam is voiding and condensing in the vertical hot leg pipe of the reactor coolant system. The engineered safety features actuation signal (ESFAS) initiates at 1600 psia thereby assuring maximum availability of one high pressure injection (HPI) pump without flow control, and isolation of the letdown system.

A new steam bubble forms at the highest temperature/lowest pressure location in the reactor coolant system when the pressurizer pressure becomes less than the vapor pressure of the liquid phase at the new location. Since there is no significant heat sink in the hot leg pipes between the reactor vessel and steam generators, the highest temperature/lowest pressure location could be in the U-bend pipe at the top of each steam generator (Figure 1).

Not only could be - but will be

If a steam bubble forms in the U-bend, it interrupts natural circulation. Water in the reactor vessel starts to boil agressively due to loss of circulating flow. The reactor core exit temperature and corresponding vapor pressure increase until the reactor vessel top plenum becomes the controlling high temperature location for steam bubble formation. This occurs when the top plenum temperature is about 1 to 2° F above the U-bend temperature.

[illegible]

The core exit temperature continues to increase due to loss of flow to the steam generators and inability of the break to remove sufficient decay heat. Figure 3 indicates that this is likely to occur in the short term for breaks up to about 0.035 ft^2. The reactor vessel level which is established in the top plenum when a steam bubble forms continues to decrease due to fluid loss through the break exceeding liquid makeup capability. The steam bubble is at a saturation pressure corresponding to the increasing core exit temperature. The steam bubble in the U-bend is compressed and condensed by the increasing reactor pressure. The U-bend is refilled with liquid and natural circulation should be restored. The restoration of natural circulation reduces core exit temperature and corresponding top plenum pressure until a new steam bubble again forms in the lower pressure U-bend region and the process repeats.

valves will probably the bubble ? your conjecture

Once formed, the reactor vessel steam bubble should be sustained and grow larger to accommodate the net blowdown of fluid from the system. The reactor vessel level which is established may experience some small additional pertubations as the steam bubbles form and condense in the U-bends, but the short-term trend should be for a decreasing level. A continuation of decay heat removal by intermittent natural circulation should be assured until the U-bends can no longer be refilled with liquid.

During the natural circulation phase following a water-side break, the pressurizer surge line and vessel slowly refill with semi-quiescent liquid. The pressurizer steam bubble is in equilibrium with a slowly decreasing reactor vessel steam bubble

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pressure and is mostly likely slowly contracting as steam from the bubble condenses on the pressurizer walls during cooldown.

For a steam-side break such as at the pressurizer top, the break vents the overpressure and flashes any saturated liquid in the pressurizer in a similar fashion to a water-side break, but the level loss rate will be lower due to a lower mass flow rate for steam through the break (Figure 2). When the pressurizer steam bubble reaches pressure equilibrium with the reactor vessel steam bubble, it starts to contract rapidly as steam continues to be removed through the break. Fluid entering the pressurizer is at system saturation conditions, but some pressurizer metal heat input remains to reduce the pressurizer level rise rate. The entire steam bubble is removed quickly through the steam-side break and it becomes a water-side break. The reactor vessel drain time should be much shorter than given in Figure 5 because this figure is based on a constant pressurizer level which is a valid assumption only if the steamside break is not at the pressurizer top.

also water will enter the PZR

3.2 Transition from Natural Circulation to Pool Boiling

Natural circulation clearly ceases if the reactor vessel level reaches the top of the hot leg pipes and reactor steam starts to break away and bubble through the pipes and accumulate at the high points (which are the U-bends). Water in the reactor vessel starts to boil agressively due to loss of circulating flow. The reactor core exit temperature and corresponding top plenum steam bubble pressure increase. The hot leg pipe pressure remains essentially equalized with the cold leg pipe pressure by reactor

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vent valve actuation. The reactor coolant pump loop seals inhibit reactor steam entry into the steam generators through the cold leg piping. The water level in each vertical hot leg pipe decreases due to fluid loss through the break exceeding liquid makeup capability.

Since natural circulation is no longer possible, the core must be cooled, in part, by pool boiling in the reactor vessel with condensation inside the steam generator tubes and pool boiling on the secondary side. The steam generators are assumed to be isolated and pressurized on the secondary side to the lowest safety valve set point. The initial secondary side liquid level is determined by plant operating conditions at the time of reactor trip. It will most likely be in the range of 6 to 26 ft above the bottom tubesheet plus any net addition from auxiliary feedwater.

The transition from natural circulation to pool boiling may be troublesome 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 below the secondary side water level. During this time, no appreciable heat is removed by the steam generators. The steam generator drain time is given in Figure 6. By definition, this drain time starts after natural circulation is lost. The drain time represents the minimum time during which system repressurization will occur if all decay heat is not being removed through the break.

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Figure 5 indicates how long it will take for the reactor vessel top plenum to drain after a break. The actual higher pressure during a portion of the drain phase will decrease the drain time. Figure 4 shows how long after a break before all decay heat can be removed through the break. For the range of smaller breaks, reactor vessel drain time for a given break size is somewhat shorter than the energy equilibrium time. This means that natural circulation ceases before the break can remove all decay heat. System repressurization will occur as required to remove the excess decay heat while waiting for the steam generators to drain following the loss of natural circulation. Increasing the pressure increases flow through the break and thereby decreases energy equilibrium time for a given break size. It should be noted from Figures 4 and 5 that repressurization to 2500 psia appears unlikely since all other curves are to the right of the 2500 psia curve for C_D = 1.0 which is the bounding condition.

Although the results indicate that full repressurization is unlikely, it should be understood that the effects of partial repressurization have not been evaluated in terms of minimum core level and peak clad temperature effects. The calculations which were performed merely confirm that repressurization may occur in order to remove decay heat when natural circulation is lost. Increasing system pressure increases flow through the break and decreases makeup available from the high pressure injection pump. This will probably result in a lower ultimate core level and a higher peak clad temperature if the core is uncovered. An ECCS type analysis based on an appropriate model for the very small break LOCA is required to determine the numbers.

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During the transition to pool boiling following a water side break, the pressurizer surge line loop seal inhibits steam entry into the pressurizer. The pressurizer slowly fills as the remaining steam bubble in the pressurizer is compressed to system pressure and condenses. The water level in the vertical hot leg pipe eventually drops below the surge line connection to the pipe (Figure 1). Any further increase in pressurizer level comes from water remaining in the loop seal. The loop seal is soon purged and steam from the vertical hot leg pipe passes to the pressurizer void space as required to compensate for condensation of the steam bubble. The level should stabilize.

See notes

For a steam-side break such as at the pressurizer top, the pressurizer may refill before the reactor vessel top plenum is drained and natural circulation ceases. It should remain filled unless the vertical hot leg pipe drains to below the surge line connection. In this event, water flow through the surge line to the break changes to steam flow and water in the pressurizer and surge line is heated to saturation temperature and purged from the system. Steam bubbling through the water to reach the break may create hydraulic instabilities.

3.3 Pool Boiling

Note! [illegible] how exactly [illegible] 2.5 psi [illegible]

In order to condense steam inside the steam generator tubes, it is necessary to drop the primary side water level inside the tubes to somewhat below the existing secondary side water level. Since hot and cold leg pressures are virtually equalized by reactor vent valve actuation (minimum ΔP of 0.15 psi to open), the primary side water level is the same as in the vertical hot

leg pipes (except for a density difference correction). A portion of the steam which is generated within the reactor core will seek its way into the vertical hot legs and undergo bubble disengagement at the water-steam interface. This steam is then free to condense inside that portion of the steam generator tubes which is above the primary side water level and below the secondary side water level. Condensate inside the steam generator tubes is returned to the reactor vessel by gravity flow through the pump loop seals and cold leg pipes.

The reactor vessel level (two-phase) fluctuates above an elevation corresponding to the top of the horizontal hot leg pipes until the level in the vertical hot leg pipes drops to the horizontal hot leg elevation. At that time the reactor vessel level extends into the hot legs. It then decreases until any fluid lost through the break no longer exceeds the liquid makeup capability. For certain small break LOCA's, the reactor vessel level turnaround may not be reached until the upper portion of the core has been uncovered for a prolonged time. For certain smaller breaks, turnaround might be reached during the natural circulation phase or transition from natural circuation to pool boiling.

Not True

The core level depicted doesn't [illegible]

3.4 Transition from Pool Boiling to Natural Circulation

Following reactor vessel level turnaround, the level starts to increase and eventually returns to the top of the horizontal hot leg pipes. A water level then appears in the vertical hot leg pipes. A corresponding level (except for a density difference correction) appears in the steam generator tubes due to pressure

equilization by the reactor vent valves. Any further net addition of liquid goes into filling the vertical hot leg pipes and steam generator tubes. A portion of the steam which is generated by decay heat within the reactor core seeks its way into the vertical hot legs, disengages at the water-steam interface, and condenses inside the cooled region of the unfilled portion of the steam generator tubes.

The reactor vessel level fluctuates above an elevation corresponding to the top of the horizontal hot legs. The level should slowly increase as the large steam bubble in the upper plenum is condensed under the influence of the hydrostatic head produced by the rising water level inside the vertical hot leg pipes.

Decay heat removal is accomplished, in part, by condensation inside the steam generator tubes if the primary side water level is sufficiently below the secondary side water level and the accumulation of any noncondensible gases in the steam generator tubes does not inhibit adequate condensation. Decay heat removal by condensation ceases when the water level inside the steam generator tubes becomes greater than the secondary side water level. If the break cannot remove all decay heat, water in the reactor vessel starts to boil more agressively due to loss of one of the required heat sinks. The reactor core exit temperature and corresponding vapor pressure increase until the reactor vessel top plenum steam bubble becomes the controlling high pressure. Makeup water continues to fill the vertical hot leg pipes and steam generator tubes. The steam bubble which is trapped inside the U-bend pipe above each steam generator is

This would already have occurred

slowly compressed and condensed by the higher reactor vessel pressure. However, the U-bend may contain noncondensible gases which have accumulated at this system high point. If sufficient noncondensible gases are present, it will be impossible to refill the U-bend with fluid and establish natural circulation.

If natural circulation is re-established, the reactor vessel level is above the horizontal hot leg pipes and increasing, and the reactor coolant piping and steam generators are full of water. The reactor vessel level continues to increase until operator action is taken to trim back on the makeup rate. If natural circulation is not re-established and the break cannot remove sufficient decay heat, the reactor coolant system pressure increases until adequate heat removal can be achieved through the break or the pressurizer safety valves open.

At some point, operator action may be invoked to open the pressurizer electromatic relief valve to assure continuation of the more stable mode of pool boiling for decay heat removal or provide sufficient depressurization to go on shutdown cooling. However, this valve has not been qualified (Class IE power, etc.) to perform an essential mitigating function and can be inactivated by a postulated single failure.

During the transition from pool boiling to natural circulation, the vertical hot leg pipe starts to refill and cover the surge line connection to the pressurizer. The pressurizer completely fills (except for noncondensible gases) as the trapped pressurizer steam bubble condenses or is vented through a steam-space break.

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3.5 Shutdown Cooling

In the long term, reactor coolant system pressure will be reduced to below 350 psia by blowdown effects of the postulated break. If the horizontal hot leg pipes are full of water at that time, it should be possible to remove all decay heat through operation of the Decay Heat Removal (DHR) system. There appears to be adequate available net positive suction head (NPSH) at each DHR pump suction to assure acceptable operation at saturation conditions if the flow rate is kept low. Any fluid still being lost through a break can be made up by one of the DHR pump loops taking suction from the ~~refueling~~ borated water storage tank or by a high pressure injection pump loop if a postulated single failure involves one of the DHR loops.

actually if break so small

4.0 WORST CASE LOCA CONSIDERATIONS

After identifying certain characteristics of a very small break LOCA and the various modes of post-LOCA decay heat removal, some thought was given to identification of the probable worst case for safety analysis purposes. A number of considerations were investigated briefly to evaluate their likely influence on the worst case selection. The more important of these are detailed below.

4.1 Discharge Coefficient and Break Location

Figure 5 indicates a Moody coefficient of 1.0 instead of 0.6 for water-side breaks shortens significantly the reactor vessel top plenum drain time for a given break size, but Figure 4 shows a compensating reduction in energy equilibrium time also occurs due

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Bullets(?)

to increased mass flow. The net effect is that only for water-side breaks less than 0.02 ft^2 with C_D = 1.0 will reactor drain time for a given break size be somewhat shorter than energy equilibrium time. For this case, natural circulation ceases before the break can remove all decay heat and some system repressurization occurs. For a water-side break with C_D = 0.6 the comparable situation develops for breaks which are less than 0.035 ft^2. In both cases, the difference between drain time and energy equilibrium time is about the same. From the viewpoint of reactor vessel drain time and energy equilibrium time for a given break area, the conservative choice over the range examined appears to be C_D =0.6.

Discharge Coeff. stuff is bull(?)!

Figure 6 indicates that a Moody coefficient of 1.0 instead of 0.6 for water-side breaks decreases the steam generator drain time for a given break size. This assures an earlier uncovery of a condensing surface inside the steam generator tubes; therefore, the conservative choice over the range examined again appears to be C_D =0.6.

In applying Figures 3 and 4 to a specific break, it should be determined that the fluid lost through the break remains representative of the fluid at the core exit. An arrangement for adequate mass transport (water, steam, or two-phase) from the core exit to the break location must be assumed if the decay heat is to be removed effectively by the break. For certain water-side break locations, the high pressure injection (HPI) pump flow may bypass the core and any decay heat generated within the core may not effectively communicate with the submerged break or steam

generator tubes. There may be no significant decay heat removal while this condition persists. Bull!

4.2 Decay Heat Removal

Decay heat can be removed from the reactor coolant system by blowdown through the postulated break and by a release of steam on the steam generator secondary side. The break is effective for heat removal at all times unless it becomes isolated. The steam generators are effective only during natural circulation or after the primary side water has drained to below the secondary side water level and condensation has become effective. Natural circulation is lost after the reactor vessel top plenum has drained (Figure 5). The steam generators become effective again after the tubes are drained sufficiently (Figure 6). The steam generators are no longer required after all decay heat can be removed through the break (Figure 4).

Correct, this is why there is a 205 prob

Not true!

The various modes of post-LOCA decay heat removal discussed in section 3.0 occur unless level turnaraound develops before the pool boiling stage. Most water-side breaks which can be classified as a LOCA lose natural circulation and reach the pool boiling stage before level turnaround. Many of these breaks reach energy equilibrium through the break with perhaps some prolonged repressurization before the steam generator can drain sufficiently to become a condenser following the loss of natural circulation. As a result, completion of the reactor vessel top plenum drainage through the break (which culminates in a loss of natural circulation) appears to mark the end of any essential usefulness of the steam generators for very small break LOCA

Not true!

mitigation. Note that the natural circulation phase appears essential.

4.3 Level Turnaround and Energy Equilibrium

Level turnaround occurs when makeup available from the high pressure injection (HPI) pump exceeds mass flow rate through the break at the existing system pressure. Figure 2 shows that at 2500 psia the flow from one HPI pump exceeds the mass flow rate through a 0.004 ft^2 water-side break (C_D = 1.0) or a 0.008 ft^2 steam-side break. Level turnaround should be immediate for these break sizes. {If the break is larger but less than 0.01 ft^2 for a water-side break or 0.035 ft^2 for a steam-side break, level turnaround occurs before the system pressure reaches 1270 psia.}

NO at High

SG Depressurization can be accomplished by lowering level →

Breaks exceeding this range lead to a prolonged loss of fluid through the break with pool boiling at 1270 psia until all decay heat can be removed through the break (energy equilibrium time) and thereby further depressurize the system. Steam generators cannot depressurize the primary system to below the set point of the secondary side safety valve unless operator action is invoked to open atmospheric dump valves. The delaying effects of reverse heat transfer from the steam generators must be accounted for when depressurizing through a break at below 1270 psia. Figure 4 shows the energy equilibrium time for a 0.01 ft^2 water-side break with C_D = 1.0 or a 0.02 ft^2 water-side break with C_D = 0.6 is over 30 minutes. A 0.035 ft^2 steam-side break reaches equilibrium in about 5 minutes. Natural circulation must be assured while awaiting energy equilibrium if repressurization is to be avoided.

It is assumed that natural circulation can be maintained until the reactor vessel top plenum is drained. Figure 5 shows reactor vessel drain time is about equal to energy equilibrium time for 0.01 ft^2 water-side break with C_D = 1.0. Therefore, when natural circulation is lost the break should be able to remove all decay heat with no further need for the steam generator as a heat sink.

For water-side breaks in the range of 0.01 - 0.02 ft^2 with C_D = 1.0 the reactor vessel top plenum drains and natural circulation is lost up to 5 minutes before energy equilibrium is achieved. Figure 6 shows the steam generator drain time to be from 10 to over 30 minutes. Therefore, energy equilibrium is established before the steam generator becomes an effective heat sink. Some system repressurization will occur during the 5 minute delay. For breaks larger than 0.02 ft^2 with C_D = 1.0, energy equilibrium is reached before the reactor vessel top plenum drain is completed.

A similar situation exists for water-side breaks in the range of 0.02 - 0.035 ft^2 with C_D = 0.6. The delay times are about the same. For breaks larger than 0.035 ft^2 with C_D = 0.6, energy equilibrium is reached before the reactor vessel top plenum drain is completed.

For all steam side breaks up to 0.05 ft^2, the energy equilibrium time given in Figure 4 is always much less than the reactor vessel top plenum drain time given in Figure 5. In every case, the break should be able to remove all decay heat well before natural circulation is lost.

Many postulated breaks change from a water-side break to a steam-side break (or the converse) sometime during the accident scenario. For the case of a water-side break changing to a steam-side break, Figure 5 indicates the reactor drain time increases considerably if the change occurs while the top plenum is still draining. Figure 4 shows the decreased mass flow rate through the break when steam starts to flow does not have a detectable effect on the energy equilibrium time if the previous water flow is for $C_D = 0.6$. If the water flow is for $C_D = 1.0$, the energy equilibrium time after steam starts to flow increases noticably; however, a comparison of Figures 4 and 5 indicates energy equilibrium time during the steam flow phase is always very short when compared to reactor vessel drain time so no repressurization effect is anticipated.

If the postulated break changes from a steam-side break to a water-side break during the accident scenario, Figure 5 shows reactor vessel drain time decreases markedly. The break characteristics become those associated with a water-side break and some system repressurization may occur. This is the type of accident sequence which develops during a break at the pressurizer top. It initially vents the steam overpressure and eventually passes water.

4.4 NSSS Vendor Calculations

A 0.05-ft² break at the pump discharge is the smallest break analyzed and reported by B & W using their NRC approved ECCS evaluation model. Their results indicate that one HPI pump alone is sufficient to handle a break of this size. Although the

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reactor vessel liquid volume is shown to drop to the core top within about 10 minutes following the break, the results confirm that the core remains virtually covered with two-phase fluid at all times and the fuel cladding temperature never exceeds its prebreak value.

Not true!

Figure 2 indicates the water flow from a 0.05 ft² break at the pump discharge is considerably greater than the capacity of one HPI pump at 1270 psia. Figure 4 shows that energy equilibrium through the break is reached within two minutes and Figure 5 shows the reactor vessel top plenum drains in about the same time. Therefore, the steam generator does not have to function as a heat sink beyond this point. Depressurization below 1270 psia becomes possible after energy equilibrium is reached, but the rate will be slow because the steam generator heat must be removed by reverse heat transfer. During this time, the reactor coolant system continues to drain through the break. Level turnaround must await a lower pressure and commensurate increase in pump flow.

Just NOT SO.

It should be recognized that a 0.05 ft² break is near the lower size limit for the ECCS evaluation model and near the upper limit for a very small break LOCA analysis. The ECCS evaluation model does not appear to take into consideration the possibility of intermittent natural circulation or the effects of steam generator drain time during the transition from natural circulation to pool boiling. These effects are important for the case of a very small break LOCA and will be experienced before the reactor vessel level reaches the core top. They may not be considered important for an ECCS type analysis of a 0.05 ft²

break because the decay heat rate is in equilibrium with the heat lost through the break in less than 5 minutes following the break (Figure 3), thereafter eliminating the steam generator as a required heat sink. For smaller breaks, the loss of this heat sink and its ultimate effect on fuel cladding temperature needs to be considered.

4.5 Break Isolation and Pump Shutoff Effects

It may be a natural tendency for the plant operator to isolate a very small break if it can be located and valved out. This may even be a requirement of the emergency operating procedure or plan. In some cases, such as for a letdown line, the isolation may be automatic. Break isolation may be partial or complete as determined by location of the isolating device, number and size of flow paths to the break, and possible variation of effective break area by opening valves such as the pressurizer vent valve.

Complete isolation reduces the break area to zero. The remaining water inventory is determined by the original break area and the time after break before isolation is achieved. For instance, if the original break area is 0.05 ft² and $C_D = 1.0$, energy equilibrium and loss of natural circulation occur in less than 2 minutes (Figures 4 and 5). At this time, the break can remove all decay heat, but the reactor coolant system continues to drain (Figure 2). Depressurization below 1270 psia starts but is impeded by reverse heat transfer from the steam generators. At 8 minutes after the break, the steam generator tubes are drained down to the secondary side water level. If at this time the break should be isolated, there is no effective heat sink (water

cooled condensing surface not established in steam generator). The system starts to repressurize and refill. However, it takes at least 25 minutes at 1270 psia to refill the steam generators and re-establish natural circulation (Figure 7 for C_D =1.0 and 26 ft initial level). Repressurization to 2500 psia appears likely with a commensurate reduction in makeup flow and eventual opening of the pressurizer safety valves as required to remove the decay heat.

The system now behaves as discussed in section 4.3 (steam-side break changes to a water-side break) as any remaining pressurizer bubble is vented through the safety valves and the pressurizer is filled with water from the vertical hot leg pipe. The impact and passing of water through the safety valves may create hydraulic instabilities and other service conditions for which the valves have not been qualified.

A rapid filling of the pressurizer free space with liquid. produces a corresponding level drawdown in the steam generator tubes which then exposes a condensing surface. When sufficient surface is exposed, the safety valves no longer need to open and the steam generator tubes start to refill. After a time, the condensing surface is flooded again and the safety valves reopen to remove the decay heat. The alternating removal of decay heat through safety valves and by condensing inside steam generator tubes continues at 2500 psia. The reactor core should remain covered. During this time the reactor vessel steam bubble controls system pressure and supports the vertical hot leg water column and keeps the pressurizer full of saturated liquid.

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Smaller area breaks which are isolated should behave in a similar fashion except it may take a longer time to lose natural circulation. Larger area breaks can also produce similar circumstances if they are isolated.

It might if the break is not

A full pressurizer may convince the operator to trip the HPI pump and watch for a subsequent loss of level. If this happens and the break has been isolated, the steam generator tube level starts to decrease due to release of fluid through the safety valves until an adequate condensing surface is established. No further level loss is likely and the safety valves should remain closed. A stable boiling mode will prevail. The pressurizer should remain full of fluid with a controlling steam bubble in the reactor vessel.

Not true

4.6 Pressurizer Level Indication

True

The modes of decay heat removal discussed in section 3.0 point out that pressurizer level is not a correct indicator of water level over the reactor core. During the natural circulation phase, water can be draining from the reactor vessel top plenum while pressurizer level is slowly increasing. If the break is at the top of the pressurizer steam space, a rapid pressurizer refilling can occur. During the transition to pool boiling and while in pool boiling, the level should stabilize even though the core may be uncovered. Therefore, pressurizer level is not considered a reliable guide as to core cooling conditions. No other primary side level indication is provided. There is a full range level indicator on the secondary side of each steam generator.

A similar problem with pressurizer level indication is found in section 4.5 relative to HPI pump trip. A full pressurizer may convince the operator to trip the HPI pump and watch for a subsequent loss of level. Although this response appears desirable, a full pressurizer may not always be a good indication of high water level in the reactor coolant system. For instance, the steam bubble which is trapped in the pressurizer may be vented by actuation of the pressurizer vent valve due to high pressure developed in the reactor vessel top plenum or by operator action. The vent valve will subsequently close but the pressurizer may be filled solid with a subcooled liquid. The loop seal configuration of the pressurizer surge line allows the pressurizer to remain filled as the reactor coolant system water level drops until system pressure is below saturation pressure of the pressurizer liquid inventory. This may take a long time if system pressure is set by a requirement to remove some of the decay heat through the steam generator at 1270 psia. Thus a full pressurizer is not considered a reliable indication for prescribing certain operator actions such as HPI pump trip.

5.0 CONCLUSIONS

The results of this investigation verify the pressence of a class of very small break LOCA's (probably $\leq$ 0.05 ft^2) for a B & W 205-Fuel-Assembly PWR which may experience one or more impediments to decay heat removal which need to be better understood if proper operator response and adequate mitigation are to be assured. The following situations have been identified as special items of concern which require confirmation using

detailed transient calculations based on appropraite system and core thermal-hydraulic models. The reported NSSS vendor models do not appear to accommodate these very small break LOCA situations. In each case, fuel peak clad temperature is the parameter of particular interest for comparison with the ECCS acceptance criteria, but stability of the fluid process and adequacy of instrumentation and components should also be considered.

1. Intermittent natural circulation is identified as a possible mode of initial decay heat removal following a very small break LOCA (section 3.1). The adequacy of this unstable mode for decay heat removal needs to be verified.

2. The transition from natural circulation to pool boiling/condensing involves a time delay incurred while waiting for water inside the steam generator tubes to drain below the secondary side water-level (section 3.2). During this time, system repressurization will occur if all decay heat is not being removed through the break. The effect and acceptability of this repressurization needs to be determined.

3. The decay heat fraction which is removed through the break for a given mass flow rate will be less than predicted unless the fluid enthalpy upstream of the break is representative of the core exit enthalpy (section 4.1). The sensitivity to upstream enthalpy, particularly with regard to system repressurization, needs to be evaluated for those break locations wherein some core bypass may be possible.

4. The pressurizer level indication is not a correct indication of water level relative to the reactor core (section 4.6). The safety significance of this shortcoming needs to be evaluated with regard to adequacy of information for corrective operator actions.

5. The possibility of very small break isolation by operator action and the subsequent loss of both the steam generators and break as heat sinks is of special concern (section 4.5). The rapid repressurization and eventual exposure of the pressurizer safety valves to slug or two-phase flow needs further analytical consideration and possible test qualification of the valves.

Why?

6. There may be a potential for serious process disruption or unacceptable functional or pressure boundary damage to components and steam generator tubes due to the hydraulic instabilities which are likely to develop during a very small break LOCA. The bubbling of saturated steam thorugh subcooled liquid and the injection of cold makeup water into a steam filled cold leg pipe are inherently unstable processes of particular concern that need further consideration.

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TABLE 1

FLOW AREA CORRESPONDING

TO NOMINAL PIPE SIZE

Nominal Pipe Size (in.)	Schedule No.	Flow Area (ft^2)*
1	80	.0050
1 1/2	160	.0098
2	160	.0156
2 1/2	160	.0246
3	160	.0375
4	160	.0644

*Potential single-ended circumferential break area.

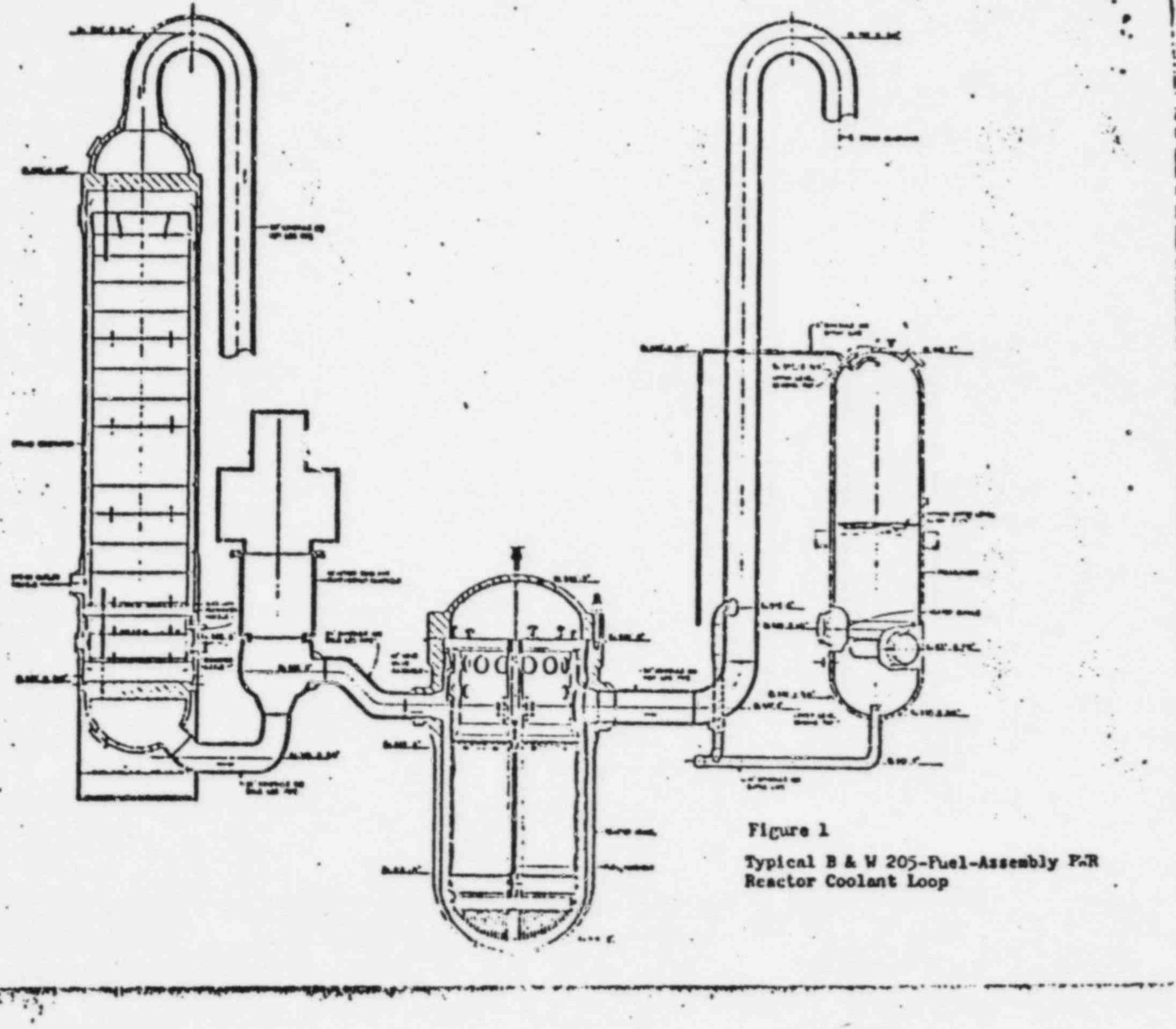

Figure 1

Typical B & W 205-Fuel-Assembly PWR Reactor Coolant Loop