TAKEN FROM
B&W REPORT
TO NRC

"EVALUATION OF TA[illegible]
BEHAVIOR AND [illegible]
REACTOR COOLANT
SYSTEM [illegible]
IN THE 177
FUEL ASS[illegible]
PLANT

APPENDIX 5

B&W ASSESSMENT

OF

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

XC4C491

INTRODUCTION

On May 3, 1978, B&W received a communication from the Tennessee Valley Authority on the subject of Small Break Analysis in connection with the Bellefonte Nuclear Project. This latter forwarded a detailed consideration of the evolution of certain very small breaks, written by Mr. Carl Michelson, and entitled "Decay Heat Removal During a Very Small Break LOCA for a B&W 205-Fuel Assembly PWR." This has become known as the Michelson Report. Our purpose here is to document B&W's assessment of the issues raised in this report:

1. The high pressure injection water (HPI) may bypass the reactor core and exit the Reactor Coolant System (RCS) directly via the break, thus not providing for core cooling.
2. The steam generator must remove significant portions of the decay heat for certain sizes of very small breaks.
3. The pressurizer level is not a valid measure of RCS liquid inventory for certain small breaks.
4. Following depressurization of the Reactor Coolant System, the secondary side of the steam generator must be considered as a heat source and its heating effects on the Reactor Coolant System must be included within the small break evaluation.
5. Natural circulation may be interrupted by the formation of voids within the RCS. If natural circulation is terminated, a repressurization of the RCS will occur during the time that the Reactor Coolant Loops are draining prior to the establishment of steam condensation.

B&W Methodology For Selection of "Worst-Case" Small Breaks

As background for discussing the issues raised by the Michelson Report, in particular the five issues above, it would be well to describe briefly the procedure used by Babcock & Wilcox to identify critical or worst case small breaks. The spectrum of breaks evaluated is based upon the following considerations.

1. CFT Line Accident - This break, by its location, severely limits the Emergency Core Cooling Systems available for accident mitigation. Considerations of break location and single active failure dictate that core cooling must be provided by one high pressure injection train and one core flood tank, until the active low pressure injection train can be switched from its assumed injection into the broken CFT line and balanced between the two CFT lines.

2. A series of break sizes are evaluated wherein the consequences of the rupture are mitigated by various combinations of the three ECCS systems.

   A. A break is considered for which mitigation is provided by low pressure injection (LPI), Core Flood Tanks (CFT), and high pressure injection (HPI).

   B. A break is considered for which mitigation is supplied by only the CFT and the HPI systems.

   C. A break is considered for which mitigation is provided solely by the high pressure injection system.

3. Additional breaks to confirm that the above spectrum has indeed bounded the worst case are considered.

Breaks are uniformly located, with the exception of the Core Flood line break, between the high pressure injection point in the cold leg and the reactor vessel. This accomplishes two things: First, a significant portion of the high pressure injection water goes directly out the break and does not provide core cooling directly. Second, breaks at low elevations within the Reactor Coolant System drain the Reactor Coolant System of significantly more water than breaks at higher elevations. Thus, for accidents in which the high pressure injection or other ECCS systems cannot instantaneously provide core cooling and cooling must be sustained for some period of time via the initial RCS inventory, that inventory is reduced in the most rapid way possible.

XC4C493

Very small breaks, those smaller than the smallest break considered in the spectrum (0.04 ft^2) are not evaluated because they are bounded by larger breaks for the following reasons:

1. Because of the internal vent valves and the once through design of the B&W steam generator, condensation within the steam generator must occur prior to uncovering of the reactor core. For the 205 Fuel Assembly Plant considered in the Michelson Report, this occurs because the elevation of the steam generators is higher than the reactor vessel. In the 177 Fuel Assembly plant, considered in this report, this occurs because the injection location for auxiliary feedwater is near the top of the steam generator. Figure A-5-1 shows the relative elevations for the lowered loop 177 F.A. plants. The 205 FA design can be visualized from Figure A-5-1 by mentally raising the steam generators such that the bottom of the OTSG corresponds approximately to the elevation of the reactor vessel inlet and outlet nozzles.
2. If steam condensation is occurring in the primary side of the steam generator, then the RCS pressure will be at or around 1000 psig since the OTSG heat sink is controlling pressure and the OTSG is controlled at about 1000 psig.
3. The breaks evaluated in the spectrum, those with HPI mitigation only, drain the RCS loops faster and establish steam condensation earlier than do smaller breaks. At the start of the steam condensation mode, the decay heat rate for the larger break will be higher than for the smaller break. The larger break will also be losing initial RCS inventory faster than the smaller break.
4. Because it has been shown by evaluation that the HPI provides successful mitigation of a transient at a higher decay heat rate, earlier time, the HPI will provide successful mitigation of the

X040494

1. Because high pressure injection may bypass the core, B&W chooses the break location to maximize ~~that~~ possibility. All breaks are modelled at the Reactor Coolant Pump Discharge between the HPI injection location and the Reactor Vessel. Water injected in this broken cold leg is calculated to run directly out of the break without providing direct core cooling. Water injected in the intact cold legs can also bypass the core and exit the RCS via the break but it could do so only if the Reactor Vessel downcomer was full of water. Because the exit elevation from the reactor vessel downcomer is above the top of the reactor core, a full reactor vessel downcomer guarantees that a mixture of steam and water must exist throughout the core region and adequate core cooling is being provided.

2. For smaller breaks, steam generators must remove a significant portion of the decay heat for an extended period of time. The limit of this consideration becomes a no break case during which all energy removal must be via the steam generators or a leak must be created (safety valve or PORV) within the Reactor Coolant System to remove energy. Steam generators are modeled within the evaluation and heat load is computed as a function of primary and secondary system variables.

X040496

3. Pressurizer level is <u>not</u> a good indication of primary system liquid inventory. No operator action should be based on that signal alone. It is quite possible to have a smaller break causing a slow loss of RCS inventory and eventual voiding of the reactor core while maintaining a reasonable pressurizer level if high pressure injection is terminated prematurely. The only positive indication of reactor vessel liquid inventory is a subcooled indication of all RCS pressure and temperature indicators excepting those in the pressurizer. This point is considered and demonstrated within the evaluation model particularly for breaks which occur in the pressurizer itself.

4. The heat contained within the steam generators severely alters the course of events for those breaks large enough to depressurize below the steam generator pressure. B&W evaluation models consider the steam generators as a heat source or a heat sink depending on the relationship of primary to secondary temperatures. If the primary system is at a lower temperature than the secondary system, heat will be transferred from the secondary system back into the primary system. This heat slows the depressurization of the system, thus, controlling the flow rate from the high pressure injection for other ECCS systems.

XC4C497

5. Natural circulation will be interrupted during a very small break and a repressurization is probable for this break. This repressurization is shown in our evaluation. B&W's analyses show, however, that natural circulation extends far beyond the time of solid water natural circulation within the primary system. Following loss of natural circulation in the solid water mode, an extended period of circulation will exist in the mode we call "bubbly two-phase." During this mode of heat transfer, steam generated by the core has collected in the upper head of the reactor vessel and a steam bubble of sufficient size to expose the top of the hot leg piping has been created. Steam exits out the hot leg nozzle, mixes with the water in the hot leg piping and is carried around the system to the steam generators and condensed at that location. The bouyancy and swell effect of steam within the hot leg piping continues natural circulation for a long period of time.

The process of two phase circulation operates because the separation rate of steam from water is very low. However, at some time the void fraction in the hot legs will accumulate to the point where steam can separate from the liquid in the hot legs at a rate sufficient to interrupt circulation. During the early phases of a small break transient, the escape velocity in combination with the hot leg void fraction is low and steam cannot escape at the rate it is being pushed into the hot legs from the upper plenum. During that time, bubbly two-phase natural circulation occurs.

For breaks evaluated in accordance with the B&W small break spectrum philosophy, the pro ss of bubbly two-phase circulation does not terminate until the liquid level in the generator falls below the auxiliary feedwater nozzles on the 177 plant design. Thus, at the termination of natural circulation, the conditions

XC4C498

for steam condensation have been created and no termination of heat transfer to the generator is seen. For extremely small breaks, the termination of bubbly two-phase circulation prior to uncovering of the steam generator auxiliary feedwater nozzles is possible, because of the lower decay heat levels involved and lower rate of system depressurization. A specific case for this is included in Section 6 of this report.

B&W January 1979 Response to TVA

In B&W's January 1979 response to TVA, the following points were addressed:

1. How is decay heat removed?
2. Will a 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 valve be subjected to slug or two-phase flow?

In our response, the question on decay heat removal was broken down into two aspects:

1. Removal of decay heat from the reactor core, and
2. Removal of decay heat from the reactor coolant system.

X040499

Removal of decay heat from the core has been evaluated experimentally. After an initial cooldown from operating power, removal of decay heat from the core is accompl[illegible]ed in a boiling pot mode. Experiments have been run for [illegible] levels equivalent to those which occur in the reactor as early as five mi[illegible] following the reactor trip. These ex[illegible]ments indicate that so long as the fluid quality in the reactor core is less than approximately 70%, core clad temperatures will remain within a few degrees of saturation. Small breaks do not cause local mixtures in the reactor core in excess of 70% quality.

no core cooling problem exists so long as the reactor core is covered by fluid mixture. B&W uses this criterion (maintenance of core coverage) to assure adequate core cooling in small break analyses.

While the core can be cooled, removal of energy from the reactor coolant system is a more complicated process. There are two ways to remove the decay heat from the reactor coolant system:

1. The Break, and
2. The Steam Generators.

Both of these processes were discussed in detail in the Michelson Report. Both apply and are considered directly in the B&W evaluation models. The Michelson Report correctly asserts that for very small breaks the removal of decay heat via the steam generator must be considered. B&W agrees that it is possible to momentarily interrupt heat removal via the steam generators during the transition from bubbly two-phase natural circulation to steam condensation natural circulation. We agree that once natural circulation ceases via the formation of a steam bubble in the top of the hot legs, no heat removal from the steam generators will occur until the primary liquid inventory within the reactor coolant loops has dropped to a point below the liquid level in the steam generators (or below the auxiliary feedwater nozzles for a 177 FA plant).

The key question, however, was, could a smaller break which undergoes a loss of natural circulation become the worst case break? This can be answered by examining the nature of the repressurization once natural circulation or steam generator heat removal ceases. During this period, the pressure of the reactor coolant system is controlled on a volume balance. Steam forming within the reactor core results in an increase in pressure of the reactor coolant system, and a decrease in the specific volume of steam. Pressure stabilization occurs when the volume of fluid, of whatever state, passing out the break equals the volume of fluid being created within the reactor core or added by injection. The fact that as pressure increases the volume of steam created

within the reactor core becomes smaller (specific volume effect) and the volume of fluid passing out the break becomes larger (pressure increase leak flow) places a limit on the repressurization. Therefore, because of the reduced necessity (lower steam volume created in the core) to vent volume out of the break, system inventory loss will remain slower for a small break than it does for a larger break which did not undergo an interruption of natural circulation.

Furthermore, steam condensation occurs only after a specific volume of liquid is drained from the RCS loops. Obviously, a break size for which the RCS had to repressurize takes longer to drain this amount of liquid than a break which did not. Therefore, the smaller break must have lost liquid at a slower rate and could not be a worse case. At the time of re-establishment of natural circulation in the steam condensation mode, the same liquid inventory exists for any break size. The only difference is that the system has arrived at this point earlier and with higher decay heat levels for the larger break than for the smaller break.

B&W additionally addressed the isolation of the break after termination of natural circulation. We agreed that the scenario presented in the Michelson Report was reasonable, and system repressurization to the code safety setpoint following break isolation was probable. However, once system inventory loss to a level corresponding to condensation in the steam generator occurs, pressure would be reduced to approximately 1000 psi, core cooling would be assured, and flow out the code safety would stop. Should code safeties bec damaged because of liquid discharge, this is of no particular concern. The effective break size is rely increased and the case is now [illegible] by a slightly larger break in the break spectrum.

XC4C501

In TVA's February 1979 response to B&W, the statement on volume relief out the break increasing with increasing pressure was questioned. Additionally, TVA stated "We were also wondering if you considered the system to be in 'Thermo-Balance' as well as 'Volume Balance'. Overriding in the TVA response

was the concern that although volume balance techniques for explanation of small break transients were reasonable, the final decision must rest upon a thermodynamic evaluation for the reactor coolant system.

The characterization of volume balance in determining the pressure in the reactor coolant system is a concept used by B&W to understand the reaction of the reactor coolant system to an interruption of natural circulation and other occurrences during small breaks. The actual evaluation of a transient has to include a total thermodynamic balance on the system. The B&W evaluation model is based on such a total thermodynamic evaluation and includes conservation of energy, and conservation of mass as well as spacial location of energy and mass. B&W agrees that ... "concepts such as volume balance and thermobalance should be recognized as oversimplifications which may apply only to a few special cases. The rigorous solution must be based on mass and energy conservation principles applied to the entire system including all inputs, outputs and phase changes within the system." Computer code evaluation of small breaks is necessary for complete understanding. It is possible, however, utilizing volume and thermobalance concepts, to bound the conditions of small breaks such that a small number of specific computer evaluations can be performed which provide assurance that all other small breaks will not lead to worse consequences within the reactor core. Such bounding calculations are performed by B&W, and the justification for them has been explained earlier in this report.

X040502

Assessment of Other Details in the Michelson Report

In the section above, we provided an introduction to the Michelson Report and a background for its evaluation. This covered the major issues raised by the report and our response to those issues, including documentation that the issues had been previously considered by Babcock & Wilcox. This Section will discuss the details contained in the report which have not already been adequately addressed.

Points raised in the report which are paraphrased or quoted are underlined. Within the cover letter four points are made. These are:

1. A steam bubble is likely to form in the high point U-bend at the top of each steam generator after system over pressure is lost. That is, after the primary system becomes basically saturated. This would interrupt natural circulation and decay heat removal. B&W agrees that such a steam bubble can form, however, we feel that the steam bubble will form initially in the upper head of the reactor vessel and later in the upper location of the U-bend.
2. The transition from natural circulation to condensation heat transfer in the steam generators could be troublesome because of the time delay while waiting for the steam generator tubes to drain down far enough to establish a condensation surface. Unless the break can already remove all decay heat, system repressuri-ation will occur. B&W agrees. Transition from natural circulation to condensation may involve system repressurization for very small breaks. We do not believe that this is particularly troublesome nor of concern in providing acceptable core cooling during the process, since the core remains covered during this period.

XC4C503

3. <u>There is a worry indicated that even after the steam condensation is established within the reactor coolant system steam generators. A repressurization will occur when the RCS starts to refill. This would be caused by system filling over the condensation surface. If steam generator heat removal would have been needed, it could not be re-established until complete refill of the system occurred. Oscillatory behavior may result. Also, the presence of non-condensibles may preclude final filling of the RCS.</u>

In fact, the refilling will be very gradual. RCS level will change extremely slowly in relationship to the decay heat drop-off. At first, the steam generator condensing surface will evolve to a location or size which is necessary to remove that portion of the decay heat which cannot be removed via the break. As decay heat drops, two processes will occur simultaneously. The need for steam condensation in the generator will be reduced and the system will depressurize slightly to reduce ΔT for condensation. Also, injection flow will increase slightly, thereby, both of these aspects will occur only as necessary to balance the reducing break quality and increasing break [illegible]. Eventually, steam condensation in the generator will be at a very low, almost zero, rate and the break will start removing all of the energy. At such time as the break, flowing at whatever quality is required, can remove all of the energy, the system will attempt to [illegible], and water will be stored within the loops. Steam will be passed through the mixture surface within the reactor vessel, pass through the upper head and flow to the break. For a break very high in the RCS, steam will bubble through a mixture in the hot leg regions and exit the break.

XC40504

This concern addresses very long term issues. The system progress to the point of steam condensation and long term cooling has been established. The question of eventual refilling of the primary system is raised.

The system should be viewed as a steady state once through boiler with only a very slowly decreasing power. System pressure is controlled by the need to vent a specific amount of energy (steam). An approximate balance between mass injected and mass flow out the break exists. As lower and lower power levels evolve the reduction in energy venting needed to maintain a steady state condition allows a slow depressurization. At the same time, break quality will be reduced in order that the approximate mass balance exists between injection and break flow. In mixture discharge, break correlations show that although mass discharge increases with decreasing quality, volume discharge decreases, therefore, the depressurization will be very slow. Once the system reaches saturated liquid discharge both mass discharge and volume discharge increase with increased subcooling, negative quality, and the depressurization rate will increase. System repressurization will not occur because in all cases the pressure requi[red] to vent a specific mass of saturated mixture is higher than that requir[ed] to vent the same mass flow in a subcooled state.

For smaller breaks, equalization of system pressure with HPI flow may still occur at an elevated RCS pressure. Once such a condition

XC4C5C5

exists, the reactor operator will eventually receive sub-cooled indications
on his hot leg and cold leg RTD's. At that time, he can throttle the
high pressure injection system back to reduce pressure to a level where
the LPI or the decay heat removal system can take over therole or pro-
viding core cooling. Such possibilities are outlined in the guidance
for operator management of small breaks.

The concern over non-condensibles is not valid unless considerable
non-condensibles have been generated by metal water reactions during the
transient. The volume of non-condensibles disolved in the RCS at operation
has been evaluated and shown to be insufficient to prevent natural
circulation. No small break in the B&W NSS will result in cladding
temperature in excess of the initial value at steady state operation
(about 700F). Therefore, no metal water reaction will occur and no non-
condensibles will be generated.

The statement is made that for smaller breaks reactor vessel drain
time for a given break size is somewhat shorter than an energy equilibrium
time. This means that natural circulation ceases before the break can
remove all decay heat. Increasin the pressure increases the flow to the
break and thereby decreases the energy equilibrium time f iven break s
Energy equilibrium relative to decay heat is an insufficient condition by
which to predict stabilization of system variables during a small break.
Equilibrium of the entire RCS must be considered for that prediction. Ple
refer to the earlier discussion of times to re-establish steam condensatio
cooling.

XC40506

Mr. Michelson relates the possibility that increasing pressure follo
cessation of natural circulation will probably result in a lower ultimate
core level and a higher peak cladding temperature if the core is uncovere

Since repressurization will be terminated before any significant possibility that the core will become uncovered, and since for breaks which require repressurization, the liquid level decrease to the point of condensation will occur at a later time than for larger breaks which do not require repressurization, the smaller break which achieves this condition at a lower core decay heat level will not lead to an uncovered core unless the larger break did. Since the larger breaks all meet the criteria of 10CFR50.46 so will the smaller ones.

In the final paragraph of Section 3.3, a statement is made to the effect that for certain small break LOCA's, Reactor Vessel turn-around may not be reached until the upper portion of the core has been uncovered for a prolonged period of time. Reactor Vessel turn-around, because of the action of the vent valves and the once through nature of the steam generator and condensation surface, will occur at some time. It will occur earlier for larger breaks and at higher decay heat levels. The only mechanism for loss of fluid from the Reactor Vessel itself, is a boiling process converting whatever water is in the Reactor Vessel to steam. At lower decay heat levels, this process takes a longer period of time. If at higher decay heat levels, it has been mitigated by the action of the high pressure injection system, then the high pressure injection system can mitigate this process at lower decay heat leve[illegible]

XC40507

In Section 3.5 Shutdown Cooling, a statement is made that any fluid still being lost through the break can be made up by one of the decay heat removal pumps taking suction from the borated water storage tank or by a high pressure injection pump loop if a postulated single failure involves one of the decay heat removal loops. The statement is true but only for a period of time. Eventually, for any of these breaks, the loss of fluid through the break will have to be made up via a pumped action from the reactor building sump. Sufficient water will have accumulated, unless the break can be isolated, within the sump to match the approximate volume of the borated water storage tank. At that time, recirculation to provide reactor coolant system makeup must be taken from the sump.

In Section 4.0 Worst Case LOCA considerations and 4.1 Discharge Coefficient and break location, paragraph three, statements are made that it should be determined that the fluid lost through the break remains representative of the fluid at the core exit. Also, statement is made that for certain water side break locations, the high pressure injection 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.

XC40508

In Section 4.2 Decay Heat Removal, the statement is made, "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 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 the small break LOCA mitigation." The drainage of the reactor vessel top plenum does not mark the end of effective natural circulation. As explained above, the steam generators are effective, although interruptable, during the entire transient evolution of a small break LOCA.

In Section 4.4, in the second paragraph, there is a discussion of the .05 ft^2 break documented in BAW-10074. A final conclusion is that level turn around must await a lower pressure and commensurate increase in pump flow because the flow discharge through the .05 ft^2 break is considerably greater than the capacity of one high pressure injection pump. For these class breaks the system depressurizes until such time as steam flow is allowed to exit the break. When steam flow exits the break, the system will rapidly depressurize to a point where energy equilibrium is achieved. In this case and in these conditions only, "energy equilibrium" is a proper tool for evaluating stabilization of the small break. With this rapid decrease in pressure, the high pressure injection flow will match the break flow and the system will achieve an equilibrium condition.

XC40509

In the following paragraph, statements are made to the effe[illegible] that the .05 ft^2 break is near the lower sized limit for the ECCS evaluation model a[illegible] near the upper limit for a very small break LOCA analysis. [illegible] ECCS evalua[illegible] tion model does not appear to take into consideration the possibility of intermittent natural circulation or the effects of steam generator drain [illegible]

Characterization of the relationship of the core outlet fluid condition to break fluid condition must involve a detailed consideration of phase separation, slip flow and steam water mixing along with proper consideration of the RCS geometry, i.e., vent valve actions. It is not possible to arrive at proper conclusions for the exact thermodynamic condition of a small break without detailed examination of these phenomena. The point that decay heat removal may not be provided by a certain portion of the high pressure injection system is valid and occurs for those breaks between the coolant pump discharge and the reactor vessel and a certain portion of the high pressure injection can exit prior to performing decay heat removal. For this reason, the Babcock & Wilcox design is controlled to achieve a specific split in high pressure injection flow between all injection points in the Reactor Coolant System. This limits the amount of water that can exit the break directly without providing core cooling to an acceptable fraction. The design provides that no more than thirty percent of the high pressure injection flow can be passed through any single line. The comment continues that there may be no significant decay heat removal while in this condition. Decay heat removal is not contingent upon high pressure injection in and of itself. The removal process is complicated, it may be via transfer to high pressure injection water, it may be occurring via transfer to initial system water, the final relief from the RCS system can be through the steam generator or through the break. In any case, an acceptable mode of removing decay heat is always provided.

X040510

during the transition from natural circulation to pool boiling. The ECCS evaluation model and the choice of .05 ft^2 break is based on bounding calculations and considerable work has been placed in proof that these breaks do provide bounding analyses as explained above. Also, it is an important part, of the ECCS evaluation model, that the drain time to establish steam condensation be included. Our evaluation includes modeling for this drain time on a sufficiently mechanistic basis to be, adequate for the evaluation of small break LOCA's.

The .05 ft^2 break relies for mitigation on successfully passing through the interruption period and reinitiation of steam condensation within the steam generators. However, since the transition for this larger break does not entail a repressurization of the primary system, the progress through that transition is not obvious from system pressure.

XC40511

In Section 4.5 Break isolation and pump shut off effect in the last part of the second paragraph, a statement is made that depressurization to 2500 psi appears likely with a commensurate reduction in makeup flow and eventual opening of the code safety valves as required to remove decay heat if the break has been isolated. Following that, in the next paragraph, the impact of passing water through the safety valves may create hydraulic instabilities and other service conditions for which the valves have not been qualified. The only impact of the failure of the safeties, if it occurs, will be to make the break slightly bigger. This larger break will behave as those already evaluated in the break spectrum. Thus, even if damage to the safety valves were to occur, core cooling would not be interrupted.

In the last paragraph of this section, is a statement that the full pressurization indication may convince the operator to trip the high pressure pump and watch for a subsequent loss of level. If this happens and the break has been isolated, the steam generator tube liquid level starts decreasing due to release of fluid through the safety valve until an adequate condensating surface is established. No further loss of level is likely and the safety valve should remain closed. The stable boiling mode will prevail and the pressurizer should remain full of liquid with a controlling steam bubble in the reactor vessel.

XC40512

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.

B&W Comment: The adequacy of this unstable mode of decay heat removal is verified by consideration of the processes which must evolve during a small break LOCA. The intermittent nature of natural circulation is not an unstable mode. It is an interruption until another mode of natural circulation, steam condensation, can start. The process is allowed for within the evaluation models through the use of bounding evaluations and larger break sizes which pose more severe consequences, or the potential for more severe consequences, to the reactor coolant system and the reactor core.

2. The transition from natural circulation to pool boiling/condensing invol[illegible] a time delay incurred while waiting for water inside the steam generator 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 ef[illegible] and acceptability of this repressurization needs to be determined.

XC40513

HPI termination is acceptable only when subcooled conditions have been established in the Reactor Coolant System.

In section 4.6 Pressurizer level indication discussions are about the fact that the pressurizer level is not a correct indicator of water level within the RCS. During the evaluation of the small break, pressurizer level can be stable or increasing while Reactor Coolant System is draining. These statements are true and have been discussed above.

In the very last part of Section 4.6 in the second paragraph, the conclusion is reiterated that the pressurizer surge line will prevent or can prevent low pressurizer level indication while the Reactor Coolant System is losing inventory. We again support this conclusion and feel that it is very important that the instructions issued in Bulletin 79-05A relative to termination of high pressure injection be followed in all pressurized water reactors.

Within the first paragraph of Section 5 conclusions, a statement is made that reported NSS vendor models do not appear to accommodate the very small break LOCA situations. We believe that the B&W model does consider the necessary phenomena to accommodate very small break LOCA. Because of the bounding nature of the larger breaks, extensive computer evaluation of very small break LOCA's has not been performed.

The major points of the Michelson report's conclusion are repeated here verbatim. Following each are B&W's comments to follow each of these conclusions.

XC40514

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 the adequacy of information for corrected operator actions.

B&W Comment: Babcock & Wilcox agrees. Only when subcooled conditions exist within the primary system, is it acceptable to take action to terminate or throttle the high pressure injection system.

5. The possibility of 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 analytical consideration and possible test qualification of the valves. The possibilities of break isolation are very small within the Reactor Coolant System. Three breaks can conceivably be isolated.:

XC40515

a. The break of the normal letdown line

b. The break of the PORV or power operated relief valve on the top of the pressurizer.

c. A very unlikely break between the control valve and block valve on the pressure spray line.

B&W Comment: This effect has been considered in the B&W analysis, and the breaks for the ECCS analysis have been chosen to bound the consequences of all small break LOCA's on the reactor core.

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. As
explained earlier, fluid conditions at the core exit bear
little relationship to fluid conditions at the break during
earlier portions of small breaks. Only when a true equilibrium
has evolved within the primary system will fluid
conditions at the core exit or at the exit from the
mixture above the core be representative of those at
the break. Finally, if the break is in the location
where HPI water can directly short circuit to the break,
fluid conditions at the core exit (or the exit of the
mixture above the core) will never be representative
of those at the break. Such considerations are included
in the Babcock & Wilcox evaluation models and are the
main reasons for locating the break between the Reactor
Coolant Pump discharge and the Reactor Vessel, thus allowing
a certain portion of the high pressure injection to be removed
from the Reactor Coolant System without providing direct core

XC4C516

Isolation of these breaks can conceivably cause repressurization of the Reactor Coolant System to the point where code safeties will relieve the energy from the core. There are no unacceptable reactor safety consequences of code safety valve relief even considering the possibility of code safety valve failure in the open position. These accidents, if they occur, can be handled by the ECCS systems.

6. There may be a potential for serious process disruption for unacceptable functional or pressure boundary damage to components and steam generator tubes due to the hydraulic instabilities which are likely developed during a very small break LOCA. The bubbling of saturated steam through subcooled liquid and the injection of cool makeup water into a steam filled cold leg pipe are inherently unstable processes of particular concern that need further consideration. The possibilities of system instabilities associated with high pressure injection flow or [illegible] interruptions and oscillation of decay heat removal via natural circulation are significantly smaller than the design conditions for the Reactor coolant system. The design condition of the steam generators, reactor vessel, reactor vessel internals, RCS piping and components are set by large break LOCAS in which hydraulic forces of orders of magnitude larger than those possible by these mechanisms are considered, therefore, though there is a small possibility that system instabilities can occur during small break LOCA, the resultant low loads are well within the design capacity of the NSS.

XC4C517

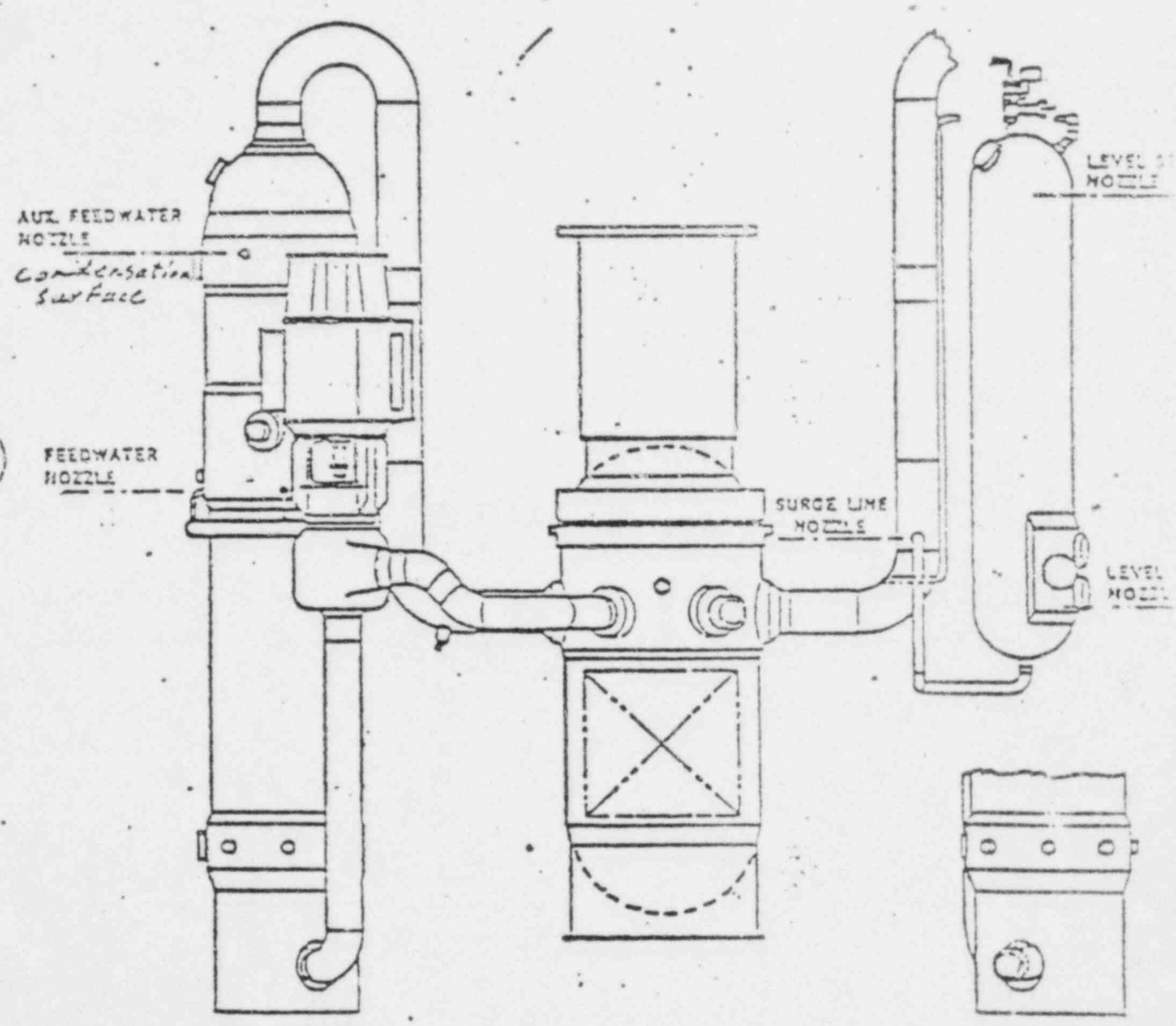
CONCLUSION

As discussed in the foregoing assessment and confirmed by analyses presented elsewhere in this report, B&W believes that the technical points raised in the Michelson report are allowed for in the design of the B&W NSS and in the ECCS evaluation model used to predict plant performance. The report independently confirms certain considerations which are a part of the B&W ECCS evaluation model.

The issues raised in the report do not alter our conclusion that the B&W ECCS evaluation model conservatively predicts the safe response of the B&W NSS to worst case Loss of Coolant Accidents, including small breaks, and that the model is an adequate basis for characterizing system behavior.

FIGURE A-5-1

NUCLEAR STEAM SYSTEM



XC4C519

-00:00:03

Some of the trip features and Condensate pumps A should not have tripped either, but did due to a wiring error

QUESTION #2 - partially addressed by the last # = -00:00:05 remarks. This is where question #2 should be addressed.

00:00:09 Remarks

Setpoint specified above? The high values had been reset to lowest after the release more early. This could change the numbers.

00:00:13 (1400:50) APPICY

In remarks, state that Cond. makeup pumps B were running before the accident. Also, state that lithium failure

2/25/82

8307080730 791108
PDR ADOCK 05000289
P HOL

was "after" 13 seconds - this should be moved to after Rx trip, Rephrase condition and list as probably caused (not recorded as there is no source)

00:00:20

00:00:15 find the real reason for the water hammer possibly condition - the existing statement came from the HEATED report on Loss of Feedwater

00:01:13 Indicated level was "screwed" up whenever the bypass valves were passing water - make this a remark and where applicable (indicated level)

00:02:00 Remark

see question 7

Condensate pumps 1 & 3 must have been tripped manually (no alarm, no log) earlier.

00:02:02 ESF equipment all went to its proper position as indicated by the board went white — good human engineering

see question 8

00:02:04 Add

see question 9

At >600# you don't get 250 gpm flow you get something less but you get enough

00:03:26 Say the RCOT was
liquid full.

84
6

after
00:04:38
Plant status summary
...system depressurization was due
to loss of inventory caused by
ERV failure and reduced HPI

00:05:54
Remark
Drain tank pressure increased
indicates relief from
pressurizer

14
00:09:13 question 12
probably a pressure
decay from 14 seconds earlier
(also computer scan interval)

00:11:5
(0411) add the remark from I&E
... 20" per minute indication
density decrease or of

1311

00:11:00 When did drain start temperature [illegible]

through 212°? At this point RB [illegible]

pressure would have started to rise

if the ~~OS~~ RCDT relief

passed enough water to see

a pressure rise. At [illegible]

RB pressure would not start to

rise until 00:14:50.

Strip chart of RCDT

See 00:03:26 ← was there

another leak (earlier)

into the RB? from 3/6 B?

00:15 100 seal flow alarm over 20 min

(04 15) ~~[illegible]~~

question 14

00:15:43 (8) check computer [illegible]

question 15 [illegible]

16:07 sequence may be in error

16:12 due to computer [illegible]

00:16:12 When did the alarm clear?

V 13 RCP 1A, 1B, 2A 2B
MUP 1A

00:38:11 there is conflicting testimony on whether the operators knew the computer [illegible]

(70)

question
22 steps in, & look at computer scan time to determine earliest the valve could have been opened

(71)

01:32:04 due the condensate booster pumps feeding through FWV-16B, the feedwater nozzle [illegible] valve

(72)

01:20:31 [illegible] the time, the make a [illegible] remark about the relief valve temperature. 402

(73)

403
403
404

Minutes

C. The bypass valve automatically closed when core water pumps were shut off, also, this shifts controller

(24) output to the atmospheric dumps. Confirm that the atmospheric dumps are open, causing the pressure drop.

01:30 (25) 15 minute dump, gives a [illegible]

Possible

01:33 (26) Pressure rise are due to possibly rapid drops are due to cooling (alternative)

01:40:00 (27) B cold leg temps decrease are probably due to back flow from the cold leg

Decrease refers to makeup pumps or makeup pumps, not reactor coolant makeup pumps

01:41 (28) Due to phase separation and liquid settling the [illegible]

TMI

01:42 Counts increasing again due to boiloff.

29

17:15:00 in Remarks, at the 2 hot 5,000's? Disregarding the temperature was not known at the granular.

30

02:19:00 confirm the decision

31 OB to check out

prior to 02:49:23

Unisolated the B steam generator for 14 seconds, causing the condensate pumps explanted

32

02:49:23

02:55:00 Check the ...

33 OB after ... or this

Declared at 02:50

03:05:-

The radiation readings conflict with 02:56:19

and with 02:49:23

also add: the generator remained isolated from the ...

03:03:39 (35) What is a quantity of decay? How did it increase?

03:12:53 (36) Remark

The pumps had run for ~ 18 minutes without providing any flow

prior to 03:46:23 (37) — The ERV had to be closed about here? When? Why?

add the generation noted.

03:59:23 (38) Were all RB ... coolers being supplied ... prior to this time? ...

04:00 → 05:30 — Start out and temperature were recorded when.

C 1:39 [illegible]

04:17:22: What are RCS pressure? [illegible]

04:21:53 pressure; Confirm no makeup pumps running. If not, why? report same.

04:19:05 (41) Was m/u pumps 1B in pull-to-lock? [illegible] #n G3

04:49:23

(42) Remark: This is because of broken vacuum and admitting air to the monitor. [illegible]

(43) 0500 Will confirm [illegible] ? confirm S/G pressure

05:15 (44) Confirm the surge line temperature of time, why is it so cool? [illegible]

05:20 Add: An operator was stationed on the roof near the atmos. relief valve as possible and did not measure any unusual readings. [illegible] OB

C this was [illegible]
be inappropriate and [illegible]
the [illegible]

02:01 (page 32) Confirm that the
3 mpph is the
minimum [illegible]
reads [illegible] if so, state

see gel [illegible]
then see 13:01 pg 76

07:04 Should there any auxiliary [illegible]
fuel handling building ventilation?

6:14:33
6:39:33
7:04:[illegible]
7:44:33

07:15 200 mR/HR at what distance?

07:38:54 Did the ~~[illegible]~~ or [illegible]
5:57 [illegible]
vent valve RCV-137?

Prior to 08:17 when were the [illegible] valve
closed? see
question 47
see entry at 07:15:17

08:31:CG Were there [illegible]
for Floyd [illegible]
closed [illegible]
or all [illegible]?

09:49:23 Confirm that the operator operated the ERV block valve and not the ERV. (see the GPU sequence) [illegible]

09:49:43 Check use of the word "Detonation"; [illegible]

09:15:00 & Did the B core flood tank ever inject? [illegible]
also 10:35:00

13:14 Plant status summary, 2nd line
Confirm that the loop A hot leg steam bubble had collapsed.
This conflicts with 522°F at 13:01 and 650 psig at 13:05
Is 522 superheat at 650#?

13:25 WHY DID it close if the block valve was open? Valve was closed at 13:00:00 (13:00:37)
Had it been shut? [illegible] 13:3.

14:30 What does this remark mean?

What was pressurizer temperature ?

day 2, Item 7; quantify how many gallons and how much activity

Day 4 PSS #3 Confirm the only 1st was sampled. Why did OPE originally describe RCS → [illegible] levels ?

Day 5 ME #8 quantify; what does '[illegible]' mean ?

Day 10 ME #3 what was sampled ?

Day 14 ME #4 what was boron ? concentration ? This comment applies to all RCS samples

Day 25 [illegible] sample M-09 - Confirm [illegible]. Could this be 238 °F ?

Pg 28

ME # 3 Provide additional details concerning the outer liner.

~~Comment 3~~

(From Tape [illegible])

Comment 3.

[illegible] the [illegible] [illegible] [illegible] the [illegible] [illegible] [illegible]:

Time	MPX	Temp	
[illegible]	[illegible]	[illegible]	
[illegible]	[illegible]	[illegible]	* [illegible]
[illegible]	[illegible]	[illegible]	
05:81:35	404	[illegible]	

[illegible] the [illegible] [illegible] [illegible] [illegible] points appear [illegible] [illegible] [illegible] [illegible] operator [illegible] [illegible] [illegible] [illegible] to be 402, 403, 404 ([illegible] [illegible] [illegible] the MPX [illegible] [illegible]). [illegible] that 403 [illegible] [illegible] [illegible] [illegible] [illegible] [illegible] to a [illegible] reading:

Actual MPX [illegible]	Possible reading error MPX [illegible]
403	402
404	403
404	404

Deft. Exh. For ID BHW 551
Pltf. Exh. in EV
Catherine Cook CC
Doyle Reporting, Inc. 2/25/82

(From [illegible])

[illegible]

[illegible]

Time	MFX	Temp
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible] [illegible]
[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]

[illegible] 402, 403, 404 [illegible] that 403 [illegible] to a [illegible] reading:

Actual MFX #	Possible Reading Error MFX #
403	402
404	403
404	404

Deft. Exh. For ID ________
Plf. Exh. in EV ________
Catherine Cook
Doyle Reporting, Inc.

```
04:44:11 DATA [illegible] RC PRESSURIZER WTR LVL 1 (DP)      105.8
04:44:21 DATA [illegible] RC PRESSURIZER WTR LVL 2 (DP)      110.0
04:44:31 DATA 0358 RC PRESSURIZER WTR LVL 3 (DP)             101.0
04:52:41 DATA 0153 COND H HOT PRESS (IN HG)                  28.[illegible]
04:53:02 DATA 0117 EMER FD PMP1 DISCH PRESS                  4.
05:12:48 ANNUNCIATOR GROUP ALARM REVIEW
REACTOR COOLANT PUMPS & MOTORS
CONT 2956 RC PUMP 2A SEAL LEAK TK LVL
CONT 2967 RCP 1A OIL LIFT PMP DSCHG PRESS
CONT 2968 RCP 2A OIL LIFT PMP DSCHG PRESS
CONT 2969 RCP 2B OIL LIFT PMP DSCHG PRESS
CONT 2970 RCP 1B OIL LIFT PMP DSCHG PRESS
CONT 2971 RCP 1A FULL SPEED
CONT 2972 RCP 2A FULL SPEED
CONT 2974 RCP 1B FULL SPEED
CONT 2975 RCP 1A BACKSTOP OIL FLOW
CONT 2976 RCP 2A BACKSTOP OIL FLOW
CONT 2977 RCP 2B BACKSTOP OIL FLOW
CONT 2978 RCP 1B BACKSTOP OIL FLOW

05:21:00 DATA 0401 RC PRESSURIZER SURGE LINE TEMP            513.0
05:21:08 DATA 0402 RC PRESS REL VLV RV2 OUT TEMP             283.0
05:21:1
      6

=            DA
            'A     0403 RC PRESS REL VLV RV1A OUT TEMP
05:21:26 DATA 0404 RC PRESS REL VLV RV1B OUT TEMP            211.3
05:21:35 DATA 0404 RC PRESS REL VLV RV1B OUT TEMP            2[illegible].5
05:21:43 DATA 0405 RC PRESSURIZER SPRAY LINE TEMP            2[illegible].6
05:31:05 DATA 0038 CIUS PUMPS OUTLET HDR PRESS               [illegible]6.6
GROUP 4                                                      194.5
05:31:33
03/23/73

SEQUENCE OF EVENTS REVIEW

05:14:05:100 3212 RC PUMP 2B OFF
05:14:05:183 3213 RC PUMP 2A OFF                             OFF
05:14:05:376 3213 RC PUMP 2A OFF                             OFF
05:14:11:031 3214 RC PUMP 1B OFF                             OFF
05:14:20:276 3176 HP GREEN CI PMP/RIPS TRIP                  OFF
05:14:20:302 3178 HP BLUE CI PMP/RIPS TRIP                   TRIP
05:14:20:308 3175 HP RED CI PMP/RIPS TRIP                    TRIP
05:14:20:355 3177 HP YELLOW CI PMP/RIPS TRIP                 TRIP
05:14:23:[illegible] 3213 RC PUMP 2A OFF                     TRIP
                                                             OFF
```

High Degrees

552

Dept. [illegible]
[illegible]
[illegible]
Doyle [illegible]

```
05:44:11  DATA  0335  RC PRESSURIZER WTR LVL 1 (DP)
04:44:21  DATA  0[illegible]7  RC PRESSURIZER WTR LVL 2 (DP)           105.3
04:44:31  DATA  0303  RC PRESSURIZER WTR LVL 3 (DP)           110.0
04:52:41  DATA  0153  COLD H HOT PRESS (IN HG)                101.0
04:53:02  DATA  0117  EMER FD PMP1 DISCH PRESS                23.[illegible]
05:12:48  ANNUNCIATOR GROUP ALARM REVIEW                      4.
REACTOR COOLANT PUMPS & MOTORS
CONT  2956  RC PUMP 2A SEAL LEAK TK LVL
CONT  2967  RCP 1A OIL LIFT PMP DSCHG PRESS
CONT  2968  RCP 2A OIL LIFT PMP DSCHG PRESS
CONT  2969  RCP 2B OIL LIFT PMP DSCHG PRESS
CONT  2970  RCP 1B OIL LIFT PMP DSCHG PRESS
CONT  2971  RCP 1A FULL SPEED
CONT  2972  RCP 2A FULL SPEED
CONT  2974  RCP 1B FULL SPEED
CONT  2975  RCP 1A BACKSTOP OIL FLOW
CONT  2976  RCP 2A BACKSTOP OIL FLOW
CONT  2977  RCP 2B BACKSTOP OIL FLOW
CONT  2978  RCP 1B BACKSTOP OIL FLOW

05:21:00  DATA  0401  RC PRESSURIZER SURGE LINE TEMP          513.0
05:21:08  DATA  0402  RC PRESS REL VLV RV2 OUT TEMP           283.0
05:21:[illegible]
6
DA
A  0403  RC PRESS REL VLV RV1A OUT TEMP
05:21:26  DATA  0404  RC PRESS REL VLV RV1A OUT TEMP          [illegible]
05:21:35  DATA  0404  RC PRESS REL VLV RV1B OUT TEMP          [illegible]
05:21:43  DATA  0405  RC PRESSURIZER SPRAY LINE TEMP          [illegible]
05:31:05  DATA  0038  CRDS PUMPS OUTLET HDR PRESS             [illegible]
GROUP 4
05:31:33
03/28/79
SEQUENCE OF EVENTS REVIEW
05:14:05:100  3212  RC PUMP 2B OFF
05:14:05:135  3213  RC PUMP 2A OFF                            OFF
05:14:05:375  3213  RC PUMP 2A OFF                            OFF
05:14:19:091  3214  RC PUMP 1B OFF                            OFF
05:14:20:275  3190  RP GREEN CH RPR/RPS TRIP                  OFF
05:14:20:[illegible]  5198  RP BLUE CH RPR/RPS TRIP          TRIP
05:14:20:[illegible]  3195  RP RED CH RPR/RPS TRIP           TRIP
05:14:20:[illegible]  3197  RP YELLOW CH RPR/RPS TRIP        TRIP
05:14:[illegible]  3213  RC PUMP 2A OFF                      TRIP
                                                             OFF
```