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CRUTCHFIELD,D. Operating Reactors Branch 5

SUBJECT: Forwards technical & schedular info re SEP Topic III-5.A re effects of pipe break on structures,sys & components,inside containment,per NRC 810630 request.

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October 1, 1981

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U.S. NUCLEAR REGULATORY COMMISSION

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: SEP Topic III-5.A, Effects of Pipe Break on Structures, Systems and Components Inside Containment
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Crutchfield:

A letter dated June 30, 1981 from Mr. Dennis M. Crutchfield requested that we provide a schedule within 30 days of receipt of the letter for resolution of ten open items for SEP Topic III-5.A, Effects of Pipe Break on Structures, Systems and Components, Inside Containment.

Our letter dated August 5, 1981 stated that RG&E would address items 1 through 6 in the Conclusions section of the NRC evaluation and that a schedule for resolution of items 7, 8 and 9 would be provided in October. Attachment A to this letter provides the technical and schedular information addressed in that letter.

Very truly yours,

L D White, Jr.
L. D. White, Jr.

Attachment

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ATTACHMENT A

SEP TOPIC III-5.A

HIGH ENERGY LINE BREAKS INSIDE CONTAINMENT

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

October 1981

Rochester Gas and Electric Corporation submitted a report on September 12, 1979 (reference 1) which evaluated the effects of high energy line breaks inside containment. The objective of the evaluation was to assure that the integrity of structures, systems and components relied upon for safe reactor shutdown or for mitigating the consequence of postulated pipe breaks is maintained.

The 1979 report identified the high energy lines inside the R. E. Ginna containment and the essential equipment needed to mitigate the consequences of pipe breaks. An effects-oriented approach was used for evaluating the postulated breaks in the high energy lines which assumed a pipe break anywhere along the line inside containment. As a result of this conservative approach, some breaks could not be shown to have acceptable consequences based upon the effects-oriented evaluation alone. In addition, the NRC draft evaluation of SEP Topic III-5.A (reference 2) identified areas where clarification of the methods used in our evaluation and where additional information concerning our assumptions are required for staff acceptance of our evaluation. As a result, ten items appearing in the Conclusions section of the NRC evaluation remain to be resolved to close out SEP Topic III-5.A. Each of these ten items is addressed below. The NRC evaluation item is repeated for clarity.

1. Clarify the assumptions used in the evaluation of the effects of postulated pipe breaks with respect to the jet model and the analyses of pipe motions caused by the dynamic effects of postulated pipe breaks. If the assumptions were different from those described in Sections V.A and VI.A, justify the assumptions used or demonstrate that the consequences of the possible new interactions are acceptable.

An effects-oriented procedure was used for evaluating high energy line breaks inside containment. All high energy lines (identified in reference 1) were assumed to fail and whip or form jets at any place along the line where a sustained high energy source was attached. Credit was taken for all closed valves or check valves which would restrict flow and thus limit the length of pipe subject to whip or jet formation. Each remaining line segment which could whip or form jets was evaluated individually in reference 1. Most of these line segments were shown to be contained entirely within one of the primary loop compartments or were above the operating floor of the containment. These locations are separated from all of the equipment required to mitigate the effects of the high energy line breaks by substantial concrete walls or one or more floors with steel reinforcing. The operating floor is 9 inches of concrete with a supporting steel structure. The intermediate floor is 6 inches of concrete with a steel supporting structure. The loop compartment walls are concrete walls at least 30 inches thick. Breaks in the following lines

will have no adverse effect upon mitigating equipment because of this separation:

Alternate Charging
Residual Heat Removal - Out
Residual Heat Removal - In
Reactor Coolant System
Pressurizer Safety and Relief Lines
Main Steam
Feedwater
Standby Auxiliary Feedwater

Because of the physical separation between the broken lines and the mitigating equipment, it was unnecessary to perform jet impingement calculations for these lines. No jets will reach required equipment. No jet calculational models were required to evaluate the effects upon valves, piping or cabling.

The main steam line was identified as having a potential for pipe whip impact upon the containment wall. That interaction is addressed in 2. below.

Pipe whip and jet loads from reactor coolant system pipe breaks have been addressed in NRC Task Action Plan A-2 (TAP A-2) and have been shown to result in acceptable consequences. Development of a leak-before-break analysis capability and supporting materials testing has been performed by Westinghouse Electric Corporation for a group of utilities including RG&E. Documentation of this work and the resulting conclusions may be found in reference 3 and reference 4.

The Residual Heat Removal-In line was identified as having a potential pipe whip impact upon a primary system component support because of the proximity of the pipe and support but the RHR line was eliminated from consideration because the plane of motion of the broken pipe would not carry it into the support. TAP A-2 work also should apply to this line and give added confidence that no adverse effects will result (see 5. below).

Breaks in three lines outside the loop compartments and in lines which are fed by the positive displacement charging pumps were shown to result in acceptable consequences based upon specific jet impingement calculations. These lines are:

RCP Seal Water In
Charging
Auxiliary Spray

For the three lines it was shown that the maximum jet force at the exit from the broken pipe would be less than 100 pounds. This was determined by multiplying the service pressure (2235 psi) by the largest opening which the charging pump could sustain at pressure with all pipe losses ignored. Applying a dynamic load factor of 2 and assuming the entire jet strikes each target still results on loads which are small and clearly acceptable.

The remaining lines (letdown, steam generator blowdown, accumulator lines, pressurizer surge and pressurizer spray), which could not be eliminated because of location or conservative calculations, are all identified as potential problem areas in reference 1 which require additional work. All of these lines are discussed in other sections of this report. Thus, except for the discussions below for specific problem areas, no detailed jet or pipe whip calculational models were required to conclude that breaks in any of the above lines will result in acceptable consequences.

2. A break in either a main steam or main feedwater line could impact the containment wall. The licensee must demonstrate that the pipe whip will not result in penetration of the containment wall. If some loss of containment function occurs an assessment must be provided of the consequences.

Because an effects-oriented evaluation of the main steam and feedwater lines could not rule out the potential for a ruptured line striking the containment wall, a mechanistic evaluation of the main steam line was performed. The analysis methods used made evaluation of the main steam line a conservative envelope for both main steam and feedwater line rupture effects upon the containment wall. The thrust force applied by the escaping fluid to the pipe was calculated by multiplying the initial pressure by the pipe cross-sectional area. The steam line force calculation thus envelopes the feedwater force calculation. The evaluation of pipe whip effect on containment wall integrity was performed for both main steam lines A and B. The piping stress analysis results from the RGE seismic upgrade program were used in the evaluation. The piping break locations were postulated at the following locations:

1 - Terminal ends of piping run.

2 - Sections where $S_{o1} + S_E J\ 0.8\ (1.2S_h + S_A)$, where the occasional loads are due to normal and upset (OBE) conditions.

3 - A minimum of two locations of maximum stress.

Table 1 provides the location of the maximum stress combinations $S_{o1} + S_E$ for each main steam line; these locations are shown in Figure 1.

TITLE			PAGE
RGE MAIN STEAM A & B PIPE WHIP			ATT. TO AM-SSA-161
PROJECT RGE	AUTHOR [signature] DATE 8/3/81	CHK'D. BY M. Y. Collins DATE 8/11/81	CHK'D. BY DATE
S.O. RGDN-1020	CALC. NO.	FILE NO. RGE 115-15A	GROUP SSA

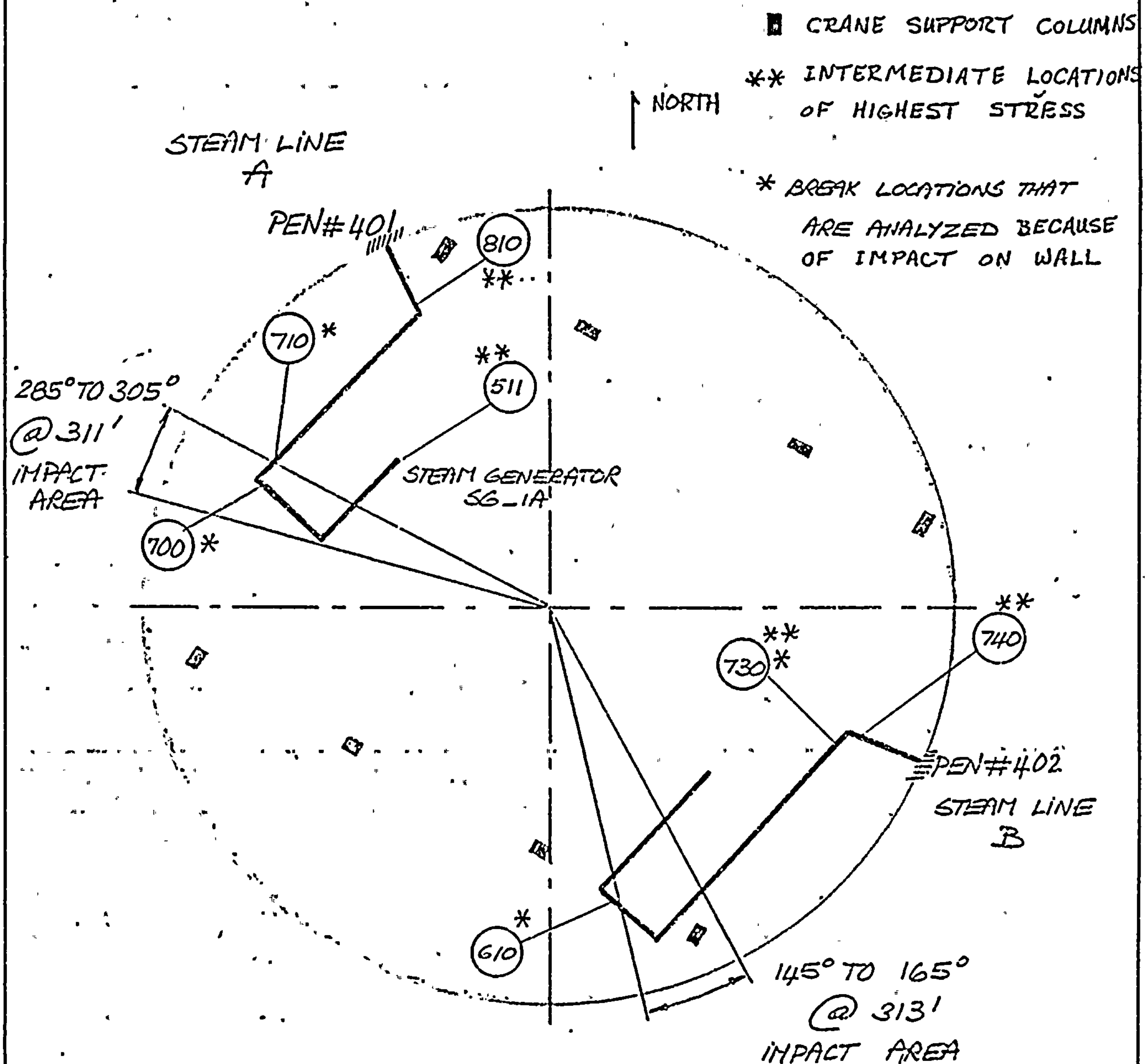

FIGURE _ 1

MAXIMUM STRESS LOCATIONS

BREAK LOCATIONS

IMPACT AREAS

REV. NO.	REV. DATE	AUTHOR DATE	CHK'D. BY DATE	CHK'D. BY DATE

WESTINGHOUSE FORM 55213D

TABLE 1

Steam Line	Node Point Number	$S_{o1} + S_E$ psi	$0.8\ (1.2S_h + S_A)$ psi
A	810	27,831	
A	511	25,361	
A	700	24,855	
A	710	24,525	29,592
B	740	27,837	
B	730	24,135	
B	610	23,843	

Since none of the combinations $S_{o1} + S_E$ exceeds the stress limit, circumferential breaks are assumed at the two intermediate points of maximum stress.

The instantaneous thrust force generated by the flashing steam water mixture was calculated according to the methods described in "Structural Analysis and Design of Nuclear Plant Facilities" J. D. Stevenson et al, ASCE, 1980.

This thrust force results in piping moments that may exceed the ultimate plastic moment at a local cross section. A plastic hinge may be formed and the kinetic moment of the thrust force may accelerate the pipe toward the containment wall.

The dynamic characteristics of the pipe required to evaluate its penetration in the containment wall for those locations where the wall is struck are:

a) the striking velocity of the pipe v_o

b) the effective pipe diameter $d = \sqrt{\frac{4A_c}{\pi}}$ where A_c is the contact area.

c) the pipe weight W, and

d) the pipe shape factor N

These variables have been evaluated for the break cases considered in Figure 1. In this evaluation, the effect of the existing pipe supports and the crane structure were neglected to maximize the impact upon the wall. This is conservative since these restraints tend to decelerate the pipe motion and, therefore, decrease the striking velocity v_o.

The results of the analysis are provided in Table 2 and the impact zones of the containment wall are shown in Figure 1.

TABLE 2

Main Steam Line	Break Location	Max. Impact Velocity ft/sec	Pipe Weight lb	Effective Pipe Diameter in
A	Pen. 401	No Impact with Containment Wall		
A	SG Nozzle	No Impact with Containment Wall		
A	810	No Impact with Containment Wall		
A	511	No Impact with Containment Wall		
A	700	195	8,578	52
A	710	No Impact with Containment Wall		
B	PEN. 402	No Impact with Containment Wall		
B	SG Nozzle	No Impact with Containment Wall		
B	740	No Impact with Containment Wall		
B	730	No Impact with Containment Wall		
B	610	29	16,463	52

It should be noted that the two intermediate locations of highest stress do not result in an impact on the containment wall and therefore no further analyses are required. Nonetheless, to provide additional confidence that postulated breaks will not yield a ruptured containment wall, the highest stressed location in each steam line which results in an impact upon the wall has been evaluated. The locations considered are #700 on MSL A and #610 on MSL B (see Figure 1).

The analyses evaluated the structural integrity of the wall considering overall wall response and evaluated the total pipe penetration depth in the wall.

Containment liner plate was not considered in the evaluation of the containment shell integrity. Characteristics for the wall were based upon prestressed concrete detail drawings for the R. E. Ginna plant. The Modified National Defense Research Committee (NDRC) formula was used for penetration depth calculations. In addition, the evaluation considered the response of the reinforced concrete wall system to resist penetration from a deformable missile. The characteristics of the missile were used to develop an applied force time history and an analysis for the overall response to the force is carried out as for an impulsive load. The analytical methods used are outlined in reference 5.

The analysis results for penetration depth (X in inches) using the NDRC formula were as follows:

For break location #700: MSL "A" X = 13.96"
For break location #610: MSL "B" X = 3.48"

The analysis for missile penetration into the wall considering overall wall response resulted in $X_m/X_c = 1.352$. This is considerably less than the allowable ductility ratio for impulse loads for flexure in structures. The rectangular impulse load considered:

Collapse load of slab = 29649K;
Plastic hinge moment = 2360 in K/in
Duration of impulse load = .00098 seconds

The conclusion of these analyses is that, even neglecting the three-eighths inch steel liner plate, structural integrity of the containment shell is assured.

3. At some locations in the "B" main steam line, a break could impact support columns for the containment crane structure and possibly cause the crane to fall. The licensee must ensure that the dynamic effects of a main steam line break will not cause the crane to fall or demonstrate that breaks need not be postulated in those locations based on a mechanistic evaluation.

Piping stresses in the "B" main steam line were determined during the RGE seismic upgrade evaluation (see 2. above). There were no locations where the stress exceeded 0.8 ($1.2\ S_h + S_A$) and thus required breaks to be postulated. The two highest stress locations between the terminal ends which are postulated to break are not located along the pipe where it passes between the crane supports (see figure 1). Breaks at the terminal ends also will not impact the crane supports. Therefore, additional analyses to demonstrate that the dynamic effects of a main steam line break will not cause the crane to fall are unnecessary. Breaks resulting in damage to the crane are not postulated to occur.

4. A break in the accumulator line between the tank skirt and the loop compartment walls could interact with the LPSI lines (resulting in a LOCA) and with the RHR outlet line. A break could also impact containment spray or safety injection lines. Breaks in this line could also interact with cables for instrumentation circuits, the LPSI valve controls and fan coolers. The licensee must demonstrate that the consequences of this scenario are acceptable, provide restraints and protection or demonstrate that breaks need not be postulated in this area based on a mechanistic evaluation.

The staff evaluation of the 10 inch A accumulator line break (no potential problems have been identified for the B accumulator) identifies potential problems resulting from interactions of the piping with the following equipment:

a) LPSI line
b) RHR outlet line
c) Containment spray line
d) Safety injection line
e) Instrumentation circuits
f) LPSI valve control circuits
g) Fan coolers

Breaks in the A accumulator line between the accumulator tank skirt and the loop compartment walls will not, by themselves, result in a loss of primary coolant. Check valve 867 located inside the B loop compartment will prevent loss of primary coolant. Only accumulator fluid will be lost as a result of the break. Interaction with other equipment is acceptable provided the interaction does not cause loss of primary inventory or interfere with maintaining the plant in a safe shutdown condition. Therefore, equipment required only for mitigation of LOCA's or large secondary system breaks need not remain functional for the accumulator break and items c, d, f, and g may be eliminated from consideration. The remaining items, a, b and e require evaluation.

The A accumulator line stresses have been determined in the seismic upgrade program. Stresses in the line are low and generally are only 10 to 25 percent of allowable. Two intermediate break locations have been examined based upon the highest stress locations. One of these locations is inside the loop B compartment where no interactions with surrounding equipment will result in adverse effects.

The second intermediate location is at node 420 near valve MOV 841 and is shown in Figure 2. The stress at this location is less than 4000 psi. The allowable stress is greater than 27,000 psi. The stress at node 420 is so low that a break at this location should not be postulated.

If a break is postulated at node 420, piping downstream of the node will not whip because the line is filled with cold water with no sustained high energy source available. Check valve 867A prevents blowdown of the RCS. The line between the elbow at node 420 and the tank skirt is a straight run of piping near the floor and separated from the LPSI line by at least 15 feet, a distance greater than the length of pipe between the tank skirt and node 420. Portions of the RHR line are closer to the accumulator line but are also out of reach of the pipe segment between the tank skirt and node 420. Valve 700, located in the A loop compartment, provides isolation from the primary system. If a rupture is postulated at node 420, the pipe may deflect downward or even strike the floor but the pipe deflection will not be toward the RHR line or the LPSI line (see drawing E-303-603 in reference 1).

In addition, work performed for TAP A-2 and presented in references 3 and 4 is applicable to this line. The material in this line is similar to that tested, stresses in the line are low and leakage from the line will be easily detected. In addition

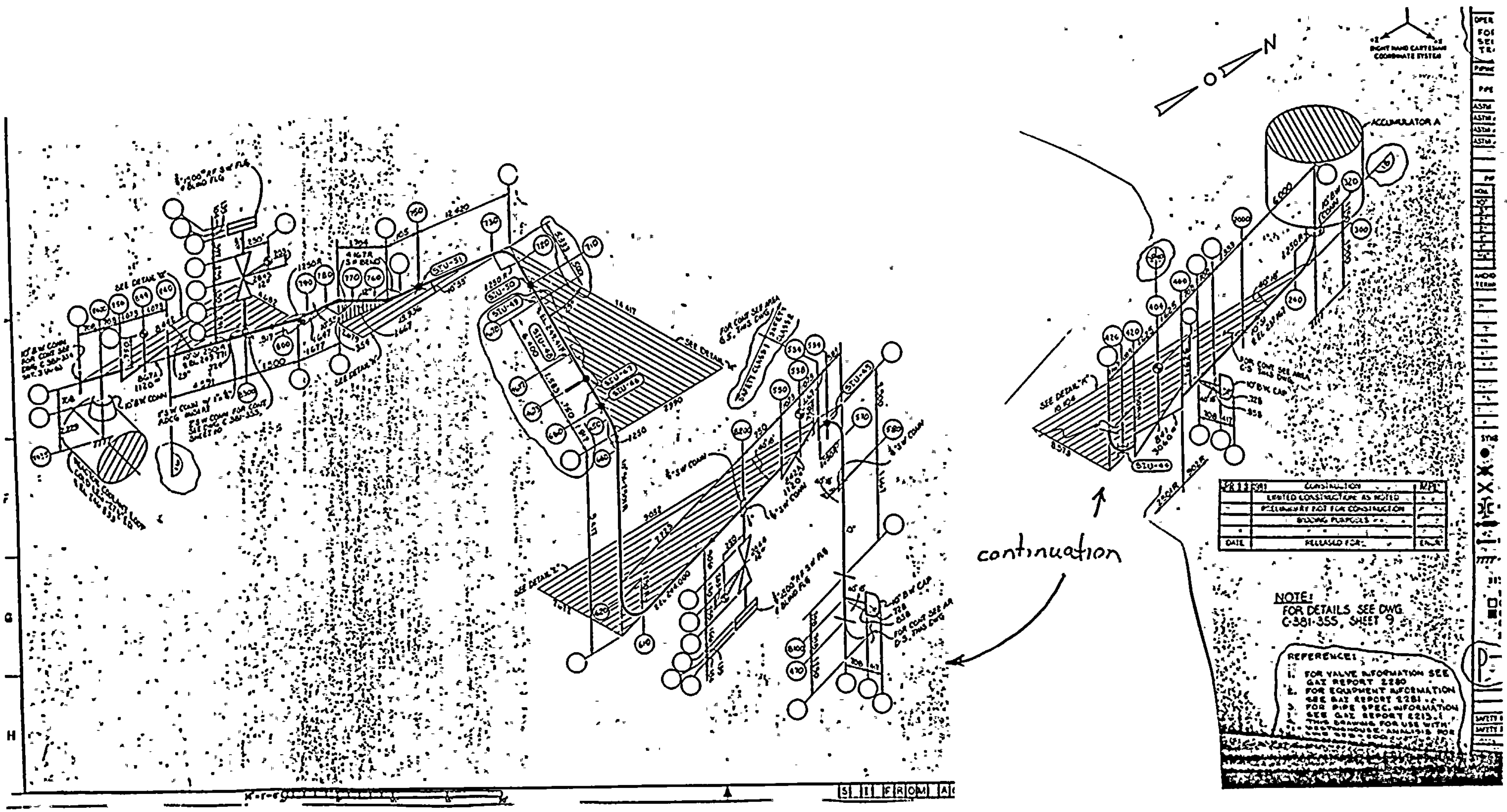
N
ACCUMULATOR A
continuation
NOTE:
FOR DETAILS SEE DWG
C-381-355, SHEET 9
REFERENCE:

Figure 2

to containment sump monitoring to detect leakage, level indication on the accumulator will give early warning of trouble. Level in the tank is governed by the Technical Specifications and is closely controlled.

Therefore, rupture of the 10 inch accumulator line should not be postulated because of the applicability of "leak before break" established by TAP A-2 work and because stresses in the line are very low. Further, even if rupture of the line at the two highest stressed intermediate locations is postulated, the break will not adversely affect required equipment.

The effect of a break in the 2 inch accumulator level taps on nearby instrument circuits does warrant further investigation which will be completed at a later date. The effect of any jets from cracks in the 10 inch accumulator line will be small. The jet force is obtained by multiplying the accumulator pressure (approximately 800 psi) times the crack opening area. The conduits and trays can easily withstand an impact force of 200 pounds or, assuming a dynamic load factor of 2, a jet force of 100 pounds. Thus, a crack opening of 100 pounds/800 psi or .125 square inch is acceptable. Cracks with easily detectable leakage in operating plants and with similar pipe wall thicknesses and system pressure have been observed. Opening areas calculated for these cracks (see Table I-5 of reference 3) are at least an order of magnitude smaller than the opening which is acceptable for jet impingement. Therefore, jets from a postulated crack in the 10 inch accumulator line require no further evaluation.

5. Breaks in the 10" pressurizer surge line could result in a large LOCA. Pipe whip from breaks in this line could impact some of the following: A LPSI valve, one SI-train, a containment spray line and the sump. Damage to one LPSI valve, and a single (independent) failure of the other LPSI valve would result in a loss of the low pressure ECCS flow which is needed to mitigate a 10" line break. Also, damage to the sump could affect long-term post-LOCA core cooling. Therefore, the licensee must provide shields and/or restraints to protect essential equipment from the dynamic effects (pipe whip and jet impingement) or demonstrate that breaks need not be postulated in these areas based on mechanistic evaluation.

The 10 inch pressurizer surge line connects the B hot leg to the bottom of the pressurizer. The line is run along the loop B compartment wall and an exterior vertical wall of the refueling canal before turning upward to connect to the bottom of the pressurizer (see drawing E-303-603 in reference 1). Rupture of the line may require operation of the nearby LPSI, SI and containment spray to mitigate the LOCA. These lines, although nearby, are mostly routed on the underside of the refueling canal which is above the basement floor. The surge line and mitigating

equipment pipes are on walls which are normal to each other at an exterior corner over most of the pipe run. Most jets, although not all jets, from the surge line will not impinge upon the mitigating equipment lines.

However, work performed to demonstrate "leak-before-break" for TAP A-2 is also applicable to this line. The material is the same as the rest of the reactor coolant system and leakage from the surge line will be detected by the same systems that detect leakage from other RCS pipes. References 3 and 4 document analyses and test results which confirm that substantial leak rates will result from piping flaws prior to reaching crack instability even in the presence of earthquakes. Appropriate conservatism was used in all of the calculations. Loads used in the analyses to determine crack stability included thermal, pressure, deadweight and safe shutdown earthquakes. Loads used to calculate crack opening areas and leak rates included only internal pressure. The calculated leak rate of 10 gpm includes only the escaping liquid and does not account for escaping steam. The actual leakage from the postulated 7.5 inch through-wall flaw used in the analysis is probably closer to 50 gpm. Leakage will be detected by sump level, condensate from the fan coolers, humidity monitors, radioactive gas detectors and radioactive particulate detectors. Leakage rates of 10 gpm or more will also be easily detected by monitoring pressurizer level and charging pump speed.

The crack opening areas calculated in reference 3 corresponding to leak rates which are easily detectable ($\sim$ 10 gpm) are .010 square inches or less. The total jet force from an opening this size in the RCS, which operates at approximately 2235 psig, is less than 25 pounds. This force can be easily withstood by the piping systems required to mitigate small LOCAs.

It should be noted that no mechanism has been identified to produce a large flaw in a PWR coolant pipe. The leak rates which have been calculated consider only the liquid portion of the escaping fluid. Thus, leaks which are easily detectable will probably result from flaws which are smaller than those identified. Larger leak rates from smaller flaw openings will mean that the jet forces will be smaller prior to plant shutdown for repair.

Therefore, with detection of gross leakage assured and stability of the piping system assured even under large flaw conditions, rupture of the surge line should not be postulated. Shields and restraints to protect against full diameter breaks or additional analyses of smaller surge line flaws are unnecessary.

6. A break in the "A" loop pressurizer spray line could affect reach rods for the sump valves 851A and B. The licensee must ensure that adequate

protection is provided for the reach rods so that a spray line break does not restrict sump flow below the required value.

A mechanistic evaluation of the pressurizer spray line from the A loop, which passes near the reach rods for MOV.851A and 851B, has been performed. Stresses in the line were determined during the seismic upgrade program analysis. The analysis shows that breaks need not be postulated near the sump valve reach rods.

The pressurizer spray line from the A loop passes through three distinct areas within the containment: the A loop compartment, an area outside the two loop compartments, and the pressurizer compartment (see drawing D-304-602 in reference 1). On the basis of the stress analyis, four breaks are postulated to occur. These breaks are at each terminal end, one in the A loop compartment and one in the pressurizer compartment, and at the two highest stressed intermediate locations. One intermediate location is in the A loop compartment and the other is in the pressurizer compartment. Therefore, because no breaks are postulated to occur near the sump valve reach rods, no protection for these rods or additional restraints on the spray line are required.

7. The letdown line outside the "B" compartment is near the pressurizer pressure cables. The licensee must provide protection for the cables from the dyanmic effects of a letdown line break or demonstrate that breaks in this line will not prevent accident mitigation and safe shutdown.

A letdown line break would result in a small cold leg LOCA. Damage to pressurizer instrumentation could prevent safety injection from being initiated from a low pressurizer pressure signal (although it is most likely that a failure would be in the safe, low, direction, actually resulting in SI). However, other instrumentation such as the high containment pressure signal (4 psig) should provide automatic safety injection initiation. If pressure does not increase quickly enough to rapidly initiate SI, manual SI could be actuated by the operator. Substantial information will be available to the operator via charging flow instrumentation, sump level, and containment radiation signals. Also, cables for at least one pressurizer pressure and one pressurizer level instrument would be unaffected by the letdown line break.

Generic Westinghouse PWR analyses (WCAP-9600) have shown that SI is not required for an hour or more following a small LOCA. Because of the amount of available instrumentation, operator action can occur within 10 minutes with no unacceptable consequences.

RG&E is still evaluating the effect of the letdown line pipe break and jet impingement on cables and cable trays. Until this work is completed, however, no immediate concerns are apparent.

8. The steam generator blowdown lines are on the same elevation as the fan coolers. Although the lines are restrained by the surrounding service water piping so that pipe whip is not of concern, protection of the fan cooler from a jet has not been established. In addition, cables for some of the steam generator level transmitters, pressurizer instrumentation and fan coolers are near the blowdown line. The licensee must either provide protection for essential equipment from the dynamic effects or show that the break effects will not prevent safe shutdown and mitigation of the break.

The steam generator blowdown line is comparable to a 2 inch feedwater line. Based on previous containment analyses, performed for large steam line breaks, it is not anticipated that containment integrity will be challenged, even with loss of all containment fan coolers. As in 7. above, damage to certain cables could prevent automatic initiation of safety injection although adequate instrumentation is available for manual operator action.

Because of the small break size, SI actuation by operator action from the control room within 10 minutes will provide all necessary safety functions for accident mitigation. The operator will have containment pressure, radiation, and sump level instrumentation available to provide information to initiate SI. If necessary, the containment spray system would be available to maintain containment conditions.

RG&E is still evaluating the effect of the letdown line pipe break and jet impingement on cables and cable trays. Until this work is completed, however, no immediate concerns are apparent.

9. Our acceptance of the consequences of breaks within loop compartments is predicated on the assumption that the dynamic effects of pipe breaks are contained by the compartment walls. Therefore, the licensee should provide appropriate references to supporting analyses for compartment wall integrity.

The compartment walls are 30 to 48 inch thick concrete walls. Analyses of subcompartments were performed during Phase B and Phase C of the Asymmetric Loads work. Documentation that the walls will withstand the effects of a full RCS guillotine break is provided in references 6 and 7 submitted for TAP A-2.

An analysis of a broken main steam line impact upon the containment wall, ignoring the containment liner, was performed in 2. above. The broken 30 inch steam line penetrated the concrete less than 14 inches. It is expected that broken 10 inch or smaller pipes inside the loop compartments will have a minor effect upon the walls.

10. As discussed in Section VI.B, conclusions on the adequacy of the review of pipe breaks in the primary RCS loop are deferred pending issuance of the USI A-2 position.

No further response required. Information to establish a "leak-before-break" criterion for stainless steel piping is given in references 3 and 4.

<u>References</u>

1. Rochester Gas and Electric letter from L. D. White, Jr. to Mr. Dennis L. Ziemann, Chief, Operating Reactors Branch #2, USNRC dated September 12, 1979.

2. USNRC letter from Mr. Dennis M. Crutchfield, Chief, Operating Reactors Branch No. 5 to John E. Maier, RGE dated June 30, 1981.

3. WCAP 9558 Rev. 2, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack," S. S. Palusamy and A. J. Hartman, Westinghouse Electric Corporation, May 1982. (SIC)

4. WCAP 9787, "Tensile and Toughness Properties of Primary Piping Weld Metal For Use In Mechanistic Fracture Evaluations," S. S. Palusamy, Westinghouse Electric Corporation, May 1981.

5. "Structural Analysis and Design of Nuclear Plant Facilities," ASCE, 1980, Sections 4.7 and 6.4.1.

6. WCAP 9628, "Westinghouse Owners Group Asymmetric LOCA Roads Evaluation Phase B," Westinghouse Electric Corporation, November 1979.

7. WCAP 9748, "Westinghouse Owners Group Asymmetric LOCA Roads Evaluation Phase C," Westinghouse Electric Corporation, June 1980.