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October 19, 1984

R. C. DeYoung, Director
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Byron Generating Station Units 1 and 2
Independent Design Inspection
NRC Inspection Report 50-454/83-32

Reference (a): August 16, 1984 letter from D. L. Farrar to J. G. Keppler.

(b): October 1, 1984 letter from T. R. Tramm to R. C. DeYoung.

Dear Mr. DeYoung:

This letter provides additional information to address NRC questions raised during the review of our response to NRC's report on their Integrated Design Inspection (IDI) and to the report of the Bechtel Independent Design Review (IDR). The bulk of the information needed to address these questions was provided in reference (b).

Attachment A to this letter contains a description of the methodology used to address pipe whip in the jet impingement study provided in reference (a).

Attachment B contains additional information on the analysis of auxiliary building flooding to address IDI Finding 2-19.

Attachment C contains a supplemental response to the NRC concerns identified as Item 2 in reference (b). This discussion places perspective on the number of IDR observations relating to FSAR discrepancies with respect to the overall IDR review effort.

Attachment D to this letter contains a supplemental response to the NRC concern identified as Item 15 in reference (b). This discussion provides further support for our conclusion that FSAR control is adequate in spite of the several IDR observations relating to FSAR accuracy.

Attachment E to this letter contains revised FSAR pages to resolve IDR Observation 8.47. These revisions make the FSAR consistent with the High Energy Line Break Report.

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Please address further questions regarding this matter to this office.

One signed original and fifteen copies of this letter and the attachments are provided for NRC review.

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Administrator

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cc: J. F. Streeter, Region III

Attachments

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ATTACHMENT A

Pipe Whip Evaluation Methodology

Pipe whip may occur when the thrust force from a broken pipe is sufficient to cause bending and formation of a plastic hinge in the pipe. In order to cause sufficient force, the break must be in a piping system capable of supplying sufficient fluid to maintain a high pressure at the break. Analyses have shown that breaks fed by pumps and breaks with high piping flow resistance between the pressure source and break will not generate adequate force to cause whip.

Large bore piping (6" diameter and greater) is restrained if located in a safety related area and shown to have a potential for whip. Whip movements are calculated as outlined in FSAR Section 3.6.2. Secondary hinges and tip deflection are considered. For evaluation of jet impingement effects the worst case position between the initial position and final position is evaluated for a given safe shutdown component.

Small bore piping (4" and smaller) has been reviewed for restraints on a break by break basis. Only those breaks with a potential for whip and the possibility of affecting safe shutdown components by pipe whip impact are restrained.

The only small bore piping breaks which could result in whip are those which are fed by a large pressure source (reactor coolant system, accumulator tanks, or the steam generators). A significant portion of the high energy line breaks in small lines can be shown not to whip because they are supplied by pumps or are flow limited due to orifices or piping flow restrictions. Systems which are used only to flow into the high energy systems are provided with check valves close to the system connection to the high energy source.

When evaluating jet impingement effects from small line breaks, pipe whip is conservatively taken into consideration for those breaks which potentially do result in whip. These are a limited number of breaks primarily in the reactor coolant letdown lines (upstream of the letdown orifices), and the steam generator blowdown lines.

The pipe is assumed to hinge at the nearest elbow sufficiently removed from the break point to develop a high moment. Because the number of cases were limited and the geometries relatively simple, no specific criteria was required. The path of the jet was assumed to terminate in the worst location with respect to the components in question consistent with the assumptions above. Because the unrestrained pipes are small (4 inches and less), the axial extent of the influence of steam and two phase jets will be 40 inches or less (per NUREG CR-2913). As a result, the conservative approach could be utilized without adversely affecting the plant design. The movement of an unrestrained pipe may be terminated by impact on a concrete structure or a larger pipe or other component. Because of the limited influence of the steam and two phase jets and the geometry associated with liquid jets, secondary hinge formation was not found to potentially increase the number of safe shutdown components affected in the few cases where the pipe movement was not terminated by structure.

The shape and extent of the jets is directly given by the applicable jet loading document. For two phase and steam jets, of primary interest inside containment, NUREG CR-2913 fully describes the jet profile with graphical representations of the jet pressure as a function of axial and radial distance from the break.

ATTACHMENT B

BYRON/BRAIDWOOD

FLOODING CLOSE-OUT PROCEDURE

The completed flooding calculation (Calculation 3C8-1281-001) was transmitted to the responsible design groups (Structural, Electrical, Control and Instrumentation, HVAC) for review. Each group reviewed the impact on their area of design. Structural incorporated the calculated flood levels into the Structural Final Load Check. Electrical walked down the areas containing safety related electrical components and identified those below the predicted flood level which could be adversely affected by flooding. In the areas of Control and Instrumentation and HVAC, affected components were identified by a review of the design documents.

As anticipated, the Structural Final Load Check confirmed that flooding would affect certain block walls which had not been designed to withstand flood loads. The potential failure of these walls has been shown to not adversely affect the safe shutdown capability. Because of the combination of increased equipment loads and flood loads, the floor in one pump room was found to be potentially overstressed. Because of the difficulty in establishing the load at which the block wall or door would fail, additional outflow area from the room is being added. No other changes have been made.

The Project Management Division has reviewed the safety related components potentially affected by flood and documented that safe shutdown capability is maintained for the postulated flooding events. No design changes were required as a result of this flood review.

ATTACHMENT C

SUPPLEMENT - A

BECHTEL RESPONSES TO NRC
FROM MEETING OF 9/14/84

Item 2 The statement, "there is no reason to expect this to be a concern elsewhere" was used frequently in close-out of observation reports. Bechtel should document the basis of why the use of this statement was appropriate for each observation report.

Bechtel Response:

The significance of the observations made with regard to insufficient control of the FSAR can be placed in proper perspective by considering the relative number of the concerns raised and the specific nature of the concerns. Only 7 Observations were made relative to the FSAR, and these concerned discrepancies of relatively minor significance. As listed in Appendices A-D of the Byron IDR Final Report, 364 different FSAR commitments were reviewed and found to be acceptably met by the Byron design which in turn were reflected in the 2120 individual evaluations noted in the Report. Many of these individual evaluations covered multiple design documents.

None of the seven observations made during the IDR with regard to FSAR control resulted in a safety significant issue. The observations represent relatively minor details. If these had gone undetected they would have had no impact on the ability of components to perform their intended safety function.

Because the FSAR is necessarily completed before all details are finalized, it is not unusual or unexpected for there to be some small differences with the final design. The FSAR describes both basic commitments and the means of implementing them. In these cases, no basic commitments were changed. The Observations related only to clarifying these commitments, or to describing equally acceptable ways of implementing the commitments. The Observations in question involved 2 cases where clarification of the FSAR was appropriate and 5 cases where the requirements were met in an equally acceptable way to that described in the FSAR.

The fact that the designs have undergone extensive reviews by the IDR and other organizations without detecting significant deficiencies in meeting commitments provides reasonable assurance that essential requirements have been met. The conditions associated with the observations noted would most likely have been detected much earlier if any of the issues had been related to a significant design parameter or function.

The small number of FSAR discrepancies noted and their conspicuous lack of significance, indicate that the Byron FSAR represents a sufficient licensing document and adequately reflects the licensing commitments and the design of the Byron plant. The IDR Observations associated with the FSAR control do not in any way constitute a safety significant issue.

ATTACHMENT D

FSAR Revisions

Changes to the Byron/Braidwood FSAR are generated both voluntarily and due to specific NRC requests. These changes are provided by Commonwealth Edison Company, Westinghouse, or Sargent & Lundy, and are transmitted either directly to the S&L Licensing Project Engineer (LPE) or indirectly to him through the Project Manager in accordance with S&L Quality Assurance Procedure GQ-3.05.

At Commonwealth Edison's request, the S&L Licensing Project Engineer compiles the FSAR changes received and generates a draft FSAR Amendment. Copies of this draft are formally transmitted to Commonwealth Edison, Westinghouse, and S&L for review. Within S&L, the draft amendment is distributed to the Project Manager and the lead Project Engineers in the Mechanical, Electrical and Structural Departments. Specialists are consulted or involved as required. Within Commonwealth Edison, the draft amendment is reviewed b the engineering staffs at the Byron and Braidwood sites and the engineering and licensing staff located in the Chicago general office. At Westinghouse, internal reviews are coordinated by the lead licensing project engineer. Comments received by the S&L LPE are resolved with the responsible engineers and the amendment is finalized. A file of input for the amendment, comments, and resolved comments is retained by the S&L LPE.

In June 1984, as a result of Byron IDI and IDR issues, the S&L Project Manager instructed the Project Team to transmit all future FSAR changes generated by S&L to the LPE via a Design Information Transmittal (DIT). The DIT is a standard form used in accordance with S&L Quality Assurance procedure GQ-3.17 for transmittal of design information between Project Team members of different divisions. Use of the DIT requires the responsible individual to give a basis for each FSAR change by providing a calculation number, drawing number, report number, or other source document reference. Also, DIT's are tracked and maintained through a formal filing system.

ATTACHMENT E

Revised FSAR Pages

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

Piping failures are postulated to occur in high and moderate energy fluid systems at locations defined using the criteria in Subsection 3.6.2. This criteria is consistent with Branch Technical Position MEB 3-1. In addition to the loss of fluid from the failed system, and the direct results of the pipe failure (i.e., pipe whip, fluid impingement, pressurization, environmental effects, water spray, flooding), a functional failure of any single active component is assumed except in those cases where the piping failure is in a dual purpose, moderate energy safety system. In these cases, the single active failure is assumed in any system other than the system which initially failed. A loss of offsite power is assumed to occur if the piping failure results in loss of offsite power or reactor trip.

The design of the plant is such that given the above, and applying the load combinations as described in Section 3.9, the function of essential systems and components will not be damaged to the extent that safe shutdown capability is lost.

3.6.1 Postulated Piping Failures in Fluid Systems Outside the Containment

The following is a summary of applicable definitions; criteria employed; potential sources and locations of piping failures; identificaiton of systems and components essential to safe plant shutdown; limits of acceptable loss of function or damage and effect on safe shutdown; habitability of critical areas following postulated piping ruptures; and the impact of the plant design on inservice surveillance and inspection.

3.6.1.1 Design Bases

3.6.1.1.1 Definitions

Throughout this section, the following definitions apply:

a. Essential Systems and Components

Systems and components required to shut down the reactor and mitigate the consequences of a postulated piping failure, without offsite power.

b. Fluid Systems

High and moderate energy fluid systems that are subject to the postulation of piping failures against which protection of essential systems and components is needed.

c. <u>High-Energy Fluid Systems</u>

Fluid systems that, during normal plant conditions, are either in operation or maintained pressurized under conditions where either or both of the following are met:

piping movement including rotational movement from static or dynamic loading. A branch connection to a main piping run is a terminal end of the branch run.

Intersections of runs of comparable size and stability are not considered terminal ends when the piping stress analysis model includes both the run and branch piping and the intersection is not rigidly constrained to the building structure.

k. Leakage Crack

A theoretical opening in the piping system, the consequences of which are evaluated on the basis of pressure and temperature differential conditions, flooding effects, and wetting of all unprotected components within the compartment.

3.6.1.1.2 Criteria

The criteria used for protection against pipe whip and the Commission's letter from Mr. Giambuso, dated December 1972, have been met for designs inside and outside the containment respectively. By virtue of the Construction Permit date for this plant, the above is the required minimum.

Subsequent criteria, including that in the Commission's letter from Mr. O'Leary, dated July 1973, and Branch Technical positions APCSB 3-1 and MEB 3-1, have been employed to the extent possible and pratical, given the stage of design/construction.

The required protection has been provided by optimization of the plant layout to minimize the number of areas affected by piping failures and to locate systems and components used for safe shutdown such that unacceptable damage would not occur. In cases where separation of systems or physical barriers provided by plant structure were not sufficient to provide protection, special protective features such as pipe whip restraints and jet impingement shields were employed.

3.6.1.1.3 Identification of Systems Important to Plant Safety

Systems important to plant safety are listed in Table 3.6-1. For a given postulated piping failure, additional systems may be required (e.g., Safety Injection is required for a LOCA). Refer to Subsection 3.6.1.3 for a more detailed discussion of systems and components important to plant safety.

The basis for defining pipe failure locations and the design approach to protect essential components are discussed in Subsection 3.6.2.

3.6.1.2 ~~~scription of Design Approval

3.6.1.2.1 Potential Sources and Locations of Piping/Environmental Effects

Potential sources of piping failures that are within or could affect Safety Category I structures are listed by system in Table 3.6-2. High-energy piping boundaries are shown in Figures 3.6-1 through 3.6-12.

Locations, orientations, and size of piping failures within high/moderate energy piping systems are postulated per the criteria given in Subsection 3.6.2.1. The dynamic effects of these postulated failures are accommodated by the methodology described in Subsections 3.6.2.2 through 3.6.2.5.

Pressure rise analyses are addressed in Subsection 3.6.1.3 Item a. There are no credible secondary missiles formed from the postulated rupture of piping.

Control room habitability is addressed in Section 6.4.

3.6.1.2.2 Impact of Plant Design for Postulated Piping Failures on Inservice Inspection

There are three areas of design necessitated for protection from piping failures which may interfere with Inservice Inspection as dictated by the ASME Boiler and Pressure Vessel Code, Section XI. They are:

a. physical separation of high/moderate energy piping in tunnels or behind barriers,

b. pipe whip restraints which may surround piping welds to be examined, and

c. impingement barriers which may interfere with weld examination or personnel/equipment access.

Design measures employed so that proper Inservice Inspection can be conducted are, respectively:

a. Tunnels containinig Section III piping have been made to allow personnel/equipment access as needed.

b. Pipe-whip restraints are of a bolted design which may be either removed from around the pipe or slid down the pipe, to allow access to any welds.

c. Impingement/separation barriers are of a removable design where interference with proper Inservice Inspection is a problem.

3.6.1.3 Safety Evaluation

In the design of this plant, due consideration was given to the effects of postulated piping breaks with respect to the limits of acceptable damage/loss of function, to assure that, even with a coincident single loss of active component, an earthquake equal to the safe shutdown earthquake, and loss of offsite power, the remaining structures, systems, and components would be adequate to safely shut down the plant. The following is a summary of the Structural, Mechanical, Instrumentation, Electrical, and HVAC items that are deemed essential and therefore designed to remain functional against (1) a high energy line rupture with resulting whip, impingement, compartment pressurization and temperature rise, wetting of compartment surfaces, and flooding, or (2) a moderate energy through-wall leakage crack with resulting wetting of compartment surfaces, and flooding, and (3) the vibratory effects of the safe shutdown earthquake.

a. Structural

All Safety Category I structures, listed in Table 3.2-1, remain functional with the exception of certain concrete block and partition walls in the auxiliary building which have not been specifically designed for loads resulting from piping failure because the failure of the wall will not cause damage to the extent that safe shutdown capability is affected. In the event walls were predicted to be loaded by postulated flooding, pressurization or jet impingement, either the walls were shown to be capable of withstanding the load or the potential effects of failure of the wall on safe shutdown components was assessed.

Pressurization and temperature rise studies for postulated breaks in all subcompartments containing normally operating high energy piping are given in Section 6.2 and Attachment A3.6 for inside and outside the containment, respectively. Flooding inside and outside containment is addressed in Attachment D3.6.

b. Mechanical

Table 3.6-3 lists all the mechanical systems which may be used for safe shutdown following any postulated pipe rupture. Note that all are seismically designed and are comprised of two full capacity, independent, redundant trains. In addition, many of the safety functions can be accomplished by two or more systems, allowing a diversity in safe shutdown procedures. For example, reactor coolant pump seal integrity is maintained if either seal injection flow (chemical and volume control system) or the thermal barrier cooling (component cooling system) is maintained. As another example, chemical shimming may be accomplished via the chemical and volume control system or the safety injection system.

It should also be noted that the essential systems are a function of the postulated initiating event. For any given event, only certain portions of an essential system may be required to achieve safe shutdown, dependent upon the postulated conditions and coincident failures.

The plant design is such that, whenever possible, all potentially essential systems are protected against loss of function resulting from any potential break. This cannot be attained when essential systems have direct communication with the postulated rupture (e.g., auxiliary feedwater connection to main feedwater or safety injection connection to reactor coolant). In these cases, the hydraulic design of the essential system is such that the "escaping" flow is not large enough to degrade the essential system flow below minimum re uirements.

Due to influences on reactivity, cooling capability, etc., break propagation is further limited as defined by Westinghouse (Reference 6) and shown in Table 3.6-4. In addition, containment leakage is always limited to an acceptable level as described in Section 3.8.

Operation of the secondary side isolation valves is critical to the safety of the plant. Therefore, the piping in the isolation valve room areas is designed well within the stress levels set for postulated breaks. In addition, the boundaries of this room, consisting of the containment and a wall at the start of the main steam tunnel, are placed as close to the isolation valves as practical, to minimize the extent of piping in the area. The piping penetrations through each are designed to withstand the loadings of piping ruptures outside this area without transferring enough strain to the isolation valves to render them inoperable. Refer to Subsection 3.8.2 for a description of their designs.

For examples of the protection afforded essential mechanical components from postulated high energy piping and the calculations that form the basis for design, refer to Subsection 3.6.2.

An assessment of the impact of flooding inside and outside containment resulting from failure of high or moderate energy line is included in Attachment D3.6. No potential flooding event affects the ability to bring the plant to a safe shutdown condition.

c. Instrumentation

Appendix B of Reference 7 lists the instrumentation required to sense critical breaks and automatically initiate protective actions to bring the plant to a safe shutdown. In some cases, instrumentation is set to initiate protective measures only when multiple reading is indicated from a number of redundant sensors (e.g., a "2 out of 4" logic). In these situations, the break may be allowed to render a sensor or sensors inoperable, with the additional sensor assumed inoperable due to a single unrelated active failure, so long as the required number of sensors necessary to signal and initiate protective measures remain.

For example in a "2 out of 4" logic, one sensor may be rendered inoperable as a consequence of the break, and the required minimum of "2 out of 4" would remain, assuming a single active failure in one sensor.

d. Electrical

Safety-related electrical components are located, to the extent possible, in areas which will not be affected by high or moderate energy line breaks. In areas such as the containment, where some electrical equipment must be located near high energy systems, redundant components are well separated to prevent failure of both trains from a common initiating event.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Described herein are the design bases for locating breaks and cracks in piping inside and outside of containment, the procedures used to define the jet thrust reaction at the break location, the jet impingement loading criteria, and the dynamic response models and results.

3.6.2.1 Criteria Used to Define Break and Crack Location and Configuration

3.6.2.1.1 Reactor Coolant Loop Piping

In any given piping system, there are a limited number of locations which are more susceptible to failure by virtue of stress or fatigue than the remainder of the system.

The discrete break locations and orientations in the reactor coolant loop are derived on the basis of stress and fatigue analysis. These postulated break locations and the methods that are used to determine them are described in Reference 1. An analysis of each individual reactor coolant loop confirms the break location defined in Reference 1. Actual seismic loads for the Byron/Braidwood site are included in the specific plant of highest stress, as calculated by equation (10) in Paragraph NB-3653, of ASME Section III which are separated by a change in direction of the pipe run are selected. If the piping run has only one change or no change of direction, only one intermediate break is postulated. A given elbow or other fitting (tee, reducer, etc.) is considered as a single break location regardless of the number or types of breaks postulated at the fitting.

b. With the exception of those portions of piping identified in Subsection 3.6.2.1.2.1.2, breaks in ASME, Section III, Class 2 and 3 piping and seismically analyzed and supported ANSI B31.1 piping are postulated at the following locations in each piping run or branch run:

1. At terminal ends of the run.

2. At each location where the stresses under the loadings resulting from normal and upset plant conditions and an OBE event as calculated by equations (9) and (10) in Paragraph NC-3652 of ASME Section III exceed $0.8\ (1.2S_h + S_a)$.

3. In the event that two intermediate locations cannot be determined by the stress limits described above, the two locations of highest stress as calculated by equations (9) and (10) in Paragraph NC-3652 of ASME, Section III which are separated by a change in direction of the pipe run shall be selected. If the piping run has only one change or no change of direction, only one intermediate break is postulated. A given elbow or other fitting (tee, reducer, etc.) shall be considered as a single break location regardless of the number or types of breaks postulated at the fitting.

4. As an alternate to (1), (2), and (3), intermediate locations are assumed at each location of potential high stress or fatigue such as pipe fittings, valves, flanges and attachments.

c. Breaks in non-seismically qualified piping are postulated at the following locations in each piping run or branch run:

e. Leakage cracks in high energy ASME Section III, Class 2 and 3 piping and seismically analyzed and supported ANSI B31.1 piping are postulated at locations where the stresses under the loadings resulting from normal and upset plant conditions and an OBE event as calculated by equations (9) and (10) in Paragraph NC-3652 of ASME, Section III exceed 0.4 $(1.2S_h + S_a)$

3.6.2.1.2.1.2 Fluid System Piping in Containment Penetration Areas

This section applies to the fluid system piping inside the isolation valve rooms, which includes the main steamlines and the feedwater lines, starting at the inside of the containment wall and extending to the first restraint outside the containment isolation valve.

3.6.2.1.2.1.2.1 Details of the Containment Penetration

Details of the containment penetrations are discussed in Subsections 3.8.1 and 3.8.2.

3.6.2.1.2.1.2.2 Break Criteria

Breaks are not postulated in the containment penetration area as defined above since the following design requirements are met:

a. The following design stress and fatigue limits are not exceeded for ASME Code, Section III, Class 2 piping and seismically qualified ANSI B31.1 piping:

   1. The maximum stress ranges as calculated by the sum of Equations (9) and (10) in Paragraph NC-3652, ASME Code, Section III, under the loadings resulting from the normal and upset plant conditions (i.e., sustained loads, occasional loads and thermal expansion) and an OBE event do not exceed 0.8 $(1.2S_h + S_a)$.

   2. The maximum stress, as calculated by Equation (9) in Paragraph NC-3652 under the loadings resulting from internal pressure, dead weight and a postulated piping failure of fluid systems piping beyond these portions of piping and excluding OBE, does not exceed $1.8S_h$. Primary loads include those which are deflection limited by whip restraints.

   3. Following a piping failure outside the first pipe whip restraint, the formation of a plastic hinge is not permitted in the piping between the containment penetration and the first pipe whip restraint. Bending and torsion limiting restraints are installed, as necessary, at

locations selected to optimize overall piping design, to prevent formation of a plastic hinge as just noted, to protect against the impairment of the leaktight integrity of the containment, to assure isolation valve operability and to meet the stress and fatigue limits in the containment penetration area.

b. Leakage cracks in the containment penetration area are postulated in accordance with Subsection 3.6.2.1.2.1.1.

c. The number of circumferential and longitudinal piping welds and branch connections are minimized as far as practical.

d. The length of these portions of piping are reduced to the minimum length practical.

3.6.2.1.2.2 Moderate-Energy Fluid System Piping Inside and Outside Containment

Through-wall leakage cracks are postulated in Seismic Category I moderate-energy ASME Section III, Class 2 and 3 and seismically analyzed and supported ANSI B31.1 piping located both inside containment except where the maximum stress range is less than 0.4 (1.2 S_h + S_a). In unanalyzed moderate-energy ASME Section III Class 2 and 3 and ANSI B31.1 piping, this exception based on stress is not taken. The cracks are postulated individually at locations that result in the maximum effects from fluid spraying and flooding, with the consequent hazards or environmental conditions developed.

3.6.2.1.2.3 Types of Breaks and Leakage Cracks in Fluid System Piping

3.6.2.1.2.3.1 Circumferential Pipe Breaks

Circumferential breaks are postulated in high-energy fluid system piping exceeding a nominal pipe size of 1 inch, at the locations specified in Subsection 3.6.2.1.2.1.

Where break locations are selected in piping without the benefit of stress calculations, breaks are postulated nonconcurrently at the piping welds to each fitting, valve or welded attachment.

3.6.2.1.2.3.2 Longitudinal Pipe Breaks

The following longitudinal breaks are postulated in high-energy fluid system piping at the locations of the circumferential breaks specified in Subsection 3.6.2.1.2.3.1.

c. $F_{steady-state}$ is determined in accordance with Figure 9-25 of Reference 4 for saturated steam and water and subcooled non-flashing water, and Figures 3.6-100, 3.6-101, and 3.6-102 for subcooled flashing water.

d. T_{imp} = Time to $F_{intermediate}$ for circumferential breaks and is determined by dividing the distance to the first elbow from the break by the sonic speed of the significant fluid wave. The sonic wave speed (C_o) is determined from Figure 9-29 of Reference 4.

e. F_{final} = The larger of F_{int} or F_{ss}.

3.6.2.2.2.1.4 Evaluation of Jet Impingement Effects

Jet impingement force calculations are required only if structures or components are located near postulated high energy line breaks and it cannot be demonstrated that failure of the structure or component will not adversely affect safe shutdown capability. The methodology used in the plant design when force calculations were found necessary is described in detail in Reference 5.

To confirm that the design approach for protection against jet impingement effects had been consistently applied throughout the design process, a thorough review of potential jet effects on safe shutdown components was completed in August 1984. A report (Reference 7) contains the results of this confirmatory review, and demonstrates that safe shutdown capability is not adversely affected by jet impingement. This effort utilized the most current information available as to the plant configuration and operating conditions. Recently, improved descriptions of steam and two-phase jet behavior were also incorporated into the review (Reference 8).

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3.6.2.2.2.2 Methods for the Dynamic Analysis of Pipe Whip

Pipe whip restraints provide clearance for thermal expansion during normal operation. If a break occurs, the restraints or anchors nearest the break are designed to prevent unlimited movement at the point of break (pipe whip). Two methods were used to analyze simplified models of the local region near the break and to calculate displacements of the pipe and restraint. These calculated displacements were then used to estimate strains in the pipe and the restraint.

An energy balance method was used to analyze carbon steel pipes since it was found possible to use a rigid-perfectly plastic moment-rotation law for pipes of this material with acceptable accuracy. The simplified models shown in Figure 3.6-15 were used to represent the local region near the break and to calculate the displacement of the pipe and the restraint when subjected to a suddenly applied constant force by the energy balance method. The restraint and structure resistances were assumed rigid-perfectly plastic. Elastic effects increase the work done by the blowdown thrust. Since these effects are neglected in the rigid-plastic energy balance model they were accounted for by increasing the gap between the pipe and the restraint by an empirical formula.

A finite difference model was used to analyze stainless steel pipes since it was found necessary to use a power law moment-curvature relationship for pipes of this material. The simplified models shown in Figure 3.6-16 were used to represent the local region near the break and to calculate the displacement in the restraint as well as the displacements and strains in the pipe.

3.6.2.2.2.2.1 Stages of Motion - Energy Balance Method

All references to points and lengths in this section can be found in Figure 3.6-15.

At the start of motion the pipe is assumed fixed at point A. Physically point A is an anchor, restraint, or elbow. In general, a hinge will form at some point B and outboard pipe segment BD will rotate as a rigid body until contact with the restraint is made at point C.

During the next stage of motion the hinge at B must move in order to satisfy the requirement that shear at a plastic hinge is zero. At the same time a hinge will form at the restraint (point C) if the plastic moment M_o is exceeded. Initially at contact, the force exerted on the pipe by the restraint is R, the restraint

In recognition of the dynamic nature of the anticipated impact loads Charpy V notch impact tests and U.T. examination of plates loaded through their thickness were specified.

3.6.2.3.1.2 Jet Deflectors

3.6.2.3.1.2.1 Nature and Location of Jet Deflectors

Jet deflectors were provided in each loop to prevent the jets emanating from the postulated longitudinal breaks at the intrados of the elbows in the hot legs at the steam generator inlets from impinging on the steam generator lower lateral supports. The deflectors, shown in Figure 3.6-22, consist of steel barrel shells tied vertically to heavy beam spanning between the steam generator column embedments and tied horizontally to embedments set in the primary shield wall.

3.6.2.3.1.2.2 Design Loads

The jet impingement load acting on the jet deflector was estimated according to the Henry-Fauske model for a subcooled homogeneous nonequilibrium flow process with the deflector treated as a simple one degree of freedom oscillator. Eccentricities of the impinging jet upon the deflector in both the radial and axial direction were postulated to reflect the uneven jet pressure distribution on the deflector bucket.

3.6.2.3.1.2.3 Design and Analysis Procedures

The jet deflectors are idealized as statically determinate pinned trusses for purposes of assessing the force in the vertical and horizontal tie members. The deflector bucket was analyzed as a circular arch.

The limiting values for member stresses were derived by increasing the AISC-69 working stress limits by 50%. The elements of the deflector are still nominally elastic under these limits. The ASCE limitation on through-plate thickness stresses were adhered to. The buckets and vertical ties are A542 steel. The horizontal ties and the embedments are A588 steel.

3.6.2.3.2 Pipe Whip Restraints Inside and Outside Containment

This subsection applies to pipe whip restraints for all piping other than the reactor main coolant piping which connects the reactor vessel, the main coolant pumps, and the steam generators.

3.6.2.3.2.1 General Description of Pipe Whip Restraints

Pipe whip restraints are provided to protect the plant against the effects of whipping during postulated pipe break. The design of pipe whip restraints is governed not only by the pipe break blowdown thrust, but also by functional requirements, deformation limitations, properties of whipping pipe and the capacity of the tests are performed on members subjected to thorough thickness tension.

3.6.2.3.2.7 Jet Impingement Shields

Jet impingement shields on the primary loops are described in Subsection 3.6.2.3.1.2. Additional jet impingement shields were not required because of the utilization of separation and redundancy to preclude jet impingement damage to safe shutdown systems and components as discussed in Subsection 3.6.1.1.

3.6.2.3.3 Criteria for Protection Against Postulated Pipe Breaks in Reactor Coolant System Piping

A lose of reactor coolant accident is assumed to occur for a branch line break down to the restraint of the second normally open automatic isolation valve (Case II in Figure 3.6-23) on outgoing lines (Note: It is assumed that motion of the unsupported line containing the isolation valves could cause failure of the operators of both valves to function) and down to and including the second check valve (Case III in Figure 3.6-23) on incoming lines normally with flow. A pipe break beyond the restraint or second check valve will not result in an uncontrolled loss of reactor coolant if either of the two valves in the line close. Accordingly, both of the automatic isolation valves are suitably protected and restrained as close to the valves as possible so that a pipe break beyond the restraint will not jeopardize the integrity and operability of the valves. Further, periodic testing capability of the valves to perform their intended function is essential. This criterion takes credit for only one of the two valves performing its intended function. For normally closed isolation or incoming check valves (Cases I and IV in Figure 3.6-23) a loss of reactor coolant accident is assumed to occur for pipe breaks on the reactor side of the valve.

Branch lines connected to the Reactor Coolant System are defined as "large" for the purpose of this criteria if they have an inside diameter greater than 4 inches up to the largest connecting line, generally the pressurizer surge line. Rupture of these lines results in a rapid blowdown from the Reactor Coolant System and protection is basically provided by the accumulators and the low head safety injection pumps (residual heat removal pumps).

Branch lines connected to the Reactor Coolant System are defined as "small" if they have an inside diameter equal to or less than 4 inches. This size is such that Emergency Core Cooling System analyses using realistic assumptions show that no clad damage is

Actual plant moments for the Byron/Braidwood Units are also given in Table 3.6-7 at the design basis break locations so that the reference fatigue analysis can be shown to be applicable for this plant. By showing actual plant moments to be no greater than those used in the reference analysis, it follows that the stress intensity ranges and usage factors for the Byron/Braidwood Units will be less than those for comparable locations in the reference mode! By this means it is shown that there are no locations other than those identified in WCAP 8082 (8172) where the stress intensity ranges and/or usage factors for the Byron/Braidwood Units might exceed the criteria of 2.4 S_m and 0.2, respectively. Thus, the applicability of WCAP 8082 (8172) to the Byron/Braidwood Units has been verified.

b. Pipe whip restraints associated with the main Reactor Coolant Loop are described in Subsections 3.6.2.3.1.1 and 5.4.14.

c. Jet deflectors associated with the main Reactor Coolant Loop are described in Subsection 3.6.2.3.1.2.

d. Design loading combinations and applicable criteria for ASME Class 1 components and supports are provided in Subsection 3.6.2.3.3.5. Pipe rupture loads include not only the jet thrust forces acting on the piping but also jet impingement loads on the primary equipment and supports.

e. The interface between Sargent & Lundy and Westinghouse concerning the design of the primary equipment supports and the interaction with the primary coolant loop is described in Subsection 3.9.3.4.4.1.

3.6.2.5.2 Postulated Breaks in Piping Other than Reactor Coolant Loop

The following material pertains to dynamic analyses completed for piping systems other than the reactor main coolant piping which connects the reactor vessel, the main coolant pumps, and the steam generators.

3.6.2.5.2.1 Implementation of Criteria for Defining Pipe Break Locations and Configurations

The locations and number of design basis breaks, including postulated rupture orientations, for the high energy piping systems are shown in Figures 3.6-25 through 3.6-99.

The above information was derived from the implementation of the criteria delineated in Subsection 3.6.2.1.

Stress levels and usage factors (usage factors for Class 1 piping only) for the postulated break locations are shown in Tables 3.6-11 and 3.6-12.

3.6.2.5.2.2 Implementation of Criteria Dealing with Special Features

Special protective devices in the form of pipe whip restraints and impingement shields are designed in accordance with Subsection 3.6.2.3.

Inservice inspection is discussed in Subsection 3.6.1.2.2.

3.6.2.5.2.3 Acceptability of Analyses Results

The postulation of break and crack locations for high and moderate energy piping systems and the analyses of the resulting jet thrust, impingement and pipe whip effects has conservatively identified areas where restraints, impingement shields, or other protective measures are needed and has yielded the conservative design of the required protective devices.

Results of jet thrust and pipe whip dynamic effects are given in Tables 3.6-13 and 3.6-14.

3.6.2.5.2.4 Design Adequacy of Systems, Components, and Component Supports

For each of the postulated breaks the equipment and systems necessary to mitigate the consequences of the break and to safely shut down the plant (i.e., all essential systems and components) have been identified (Subsection 3.6.1). The equipment and systems are protected against the consequences of each of the postulated breaks to ensure that their design-intended functions will not be impaired to unacceptable levels as a result of a pipe rupture or crack.

When it became necessary to restrict the motion of a pipe which would result from a postulated break, pipe whip restraints were added to the applicable piping systems, or structural barriers or walls were designed to prevent the whipping of the pipe.

Design adequacy of the pipe whip restraints is demonstrated in Tables 3.6-13 and 3.6-14. Data in the tables was obtained through use of the criteria delineated in Subsections 3.6.2.1 through 3.6.2.3 inclusive.

The design adequacy of structural barriers, walls, and components is discussed in Section 3.8.

3.6.3 References

1. "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop," WCAP-8082-P-A, January 1975 (Proprietary) and WCAP-8172-A (Non-Proprietary), January 1975.

2. F. M. Bordelon, "A Comprehensive Space-Time Dependent Analysis of Loss of Coolant (SATAN-IV Digital Code)," WCAP-7263, August 1971 (Proprietary) and WCAP-7750, August 1971 (Non-Proprietary).

3. "Documentation of Selected Westinghouse Structural Analysis Computer Codes," WCAP-8252, April 1974.

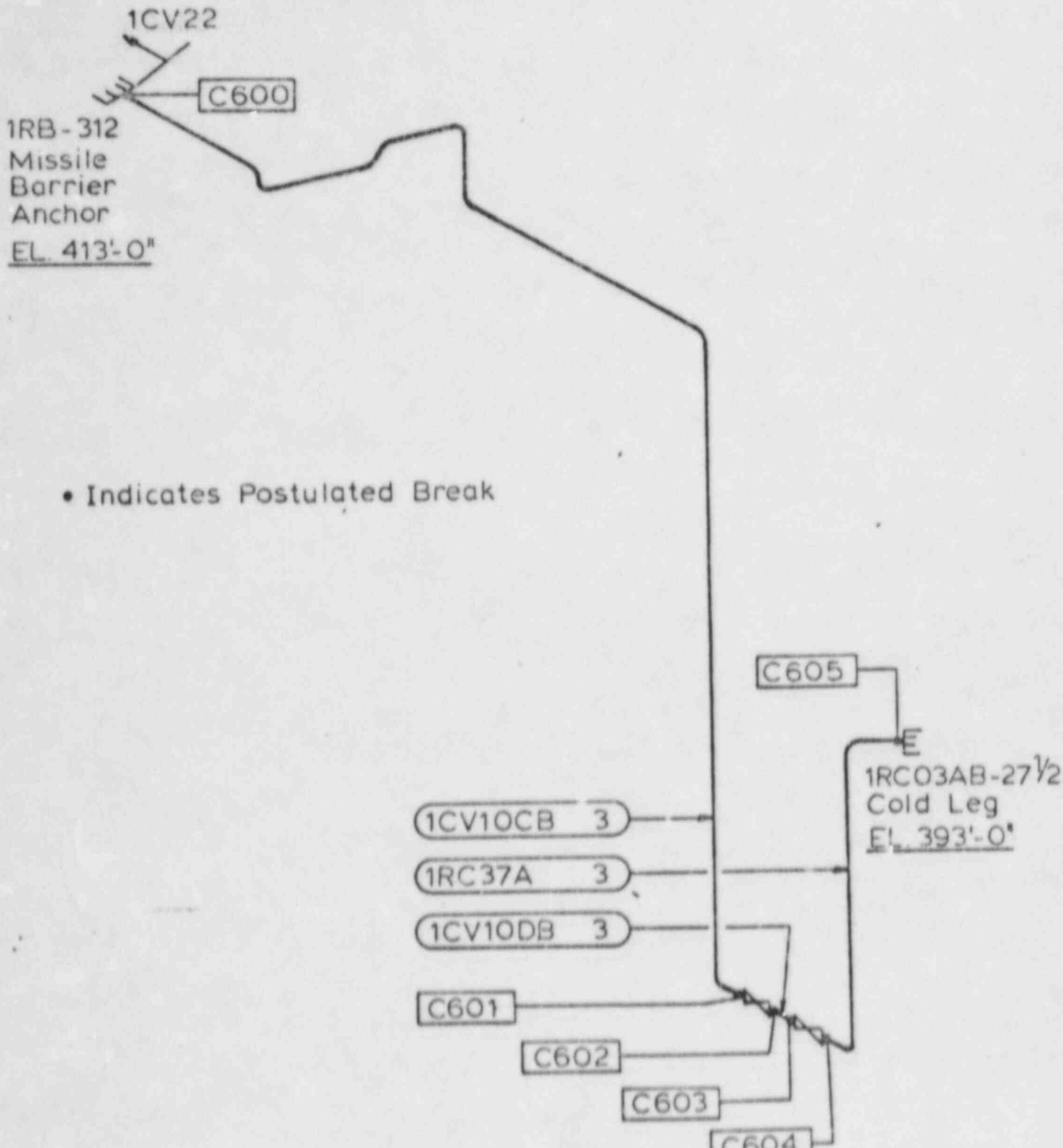
4. R. T. Lahey, Jr. and F. J. Moody, "Pipe Thrust and Jet Loads," The Thermal-Hydraulics of a Boiling Water Nuclear Reactor, Section 9.2.3, pp. 375-409, Published by American Nuclear Society. Prepared for the Division of Technical Information United States Energy Research and Development Administration, 1977.

5. Sargent & Lundy Engineering Mechanics Division Technical Procedure No. 24, "Analysis of Postulated Pipe Rupture," September 1976.

6. Westinghouse Design Criteria SS1.19, "Criteria for Protection Against Dynamic Effects Resulting from Pipe Rupture," Revision 0, March 1978.

7. "Byron 1 - Confirmation of Design Adequacy for Jet Impingement Effects," Commonwealth Edison Company, August 1984.

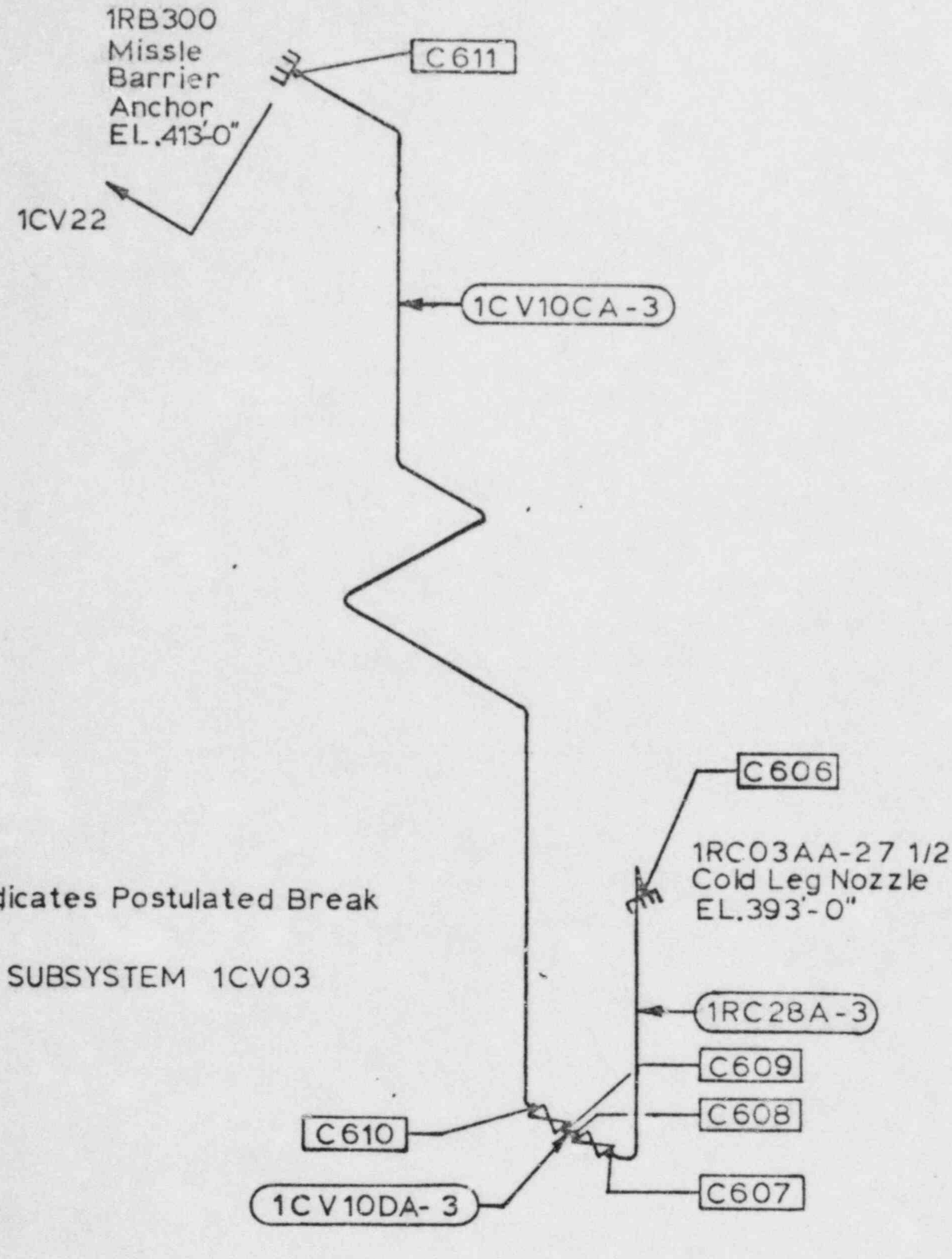
8. NUREG/CR-2913, "Two-Phase Jet Loads," January 1983.



SUBSYSTEM 1CV02

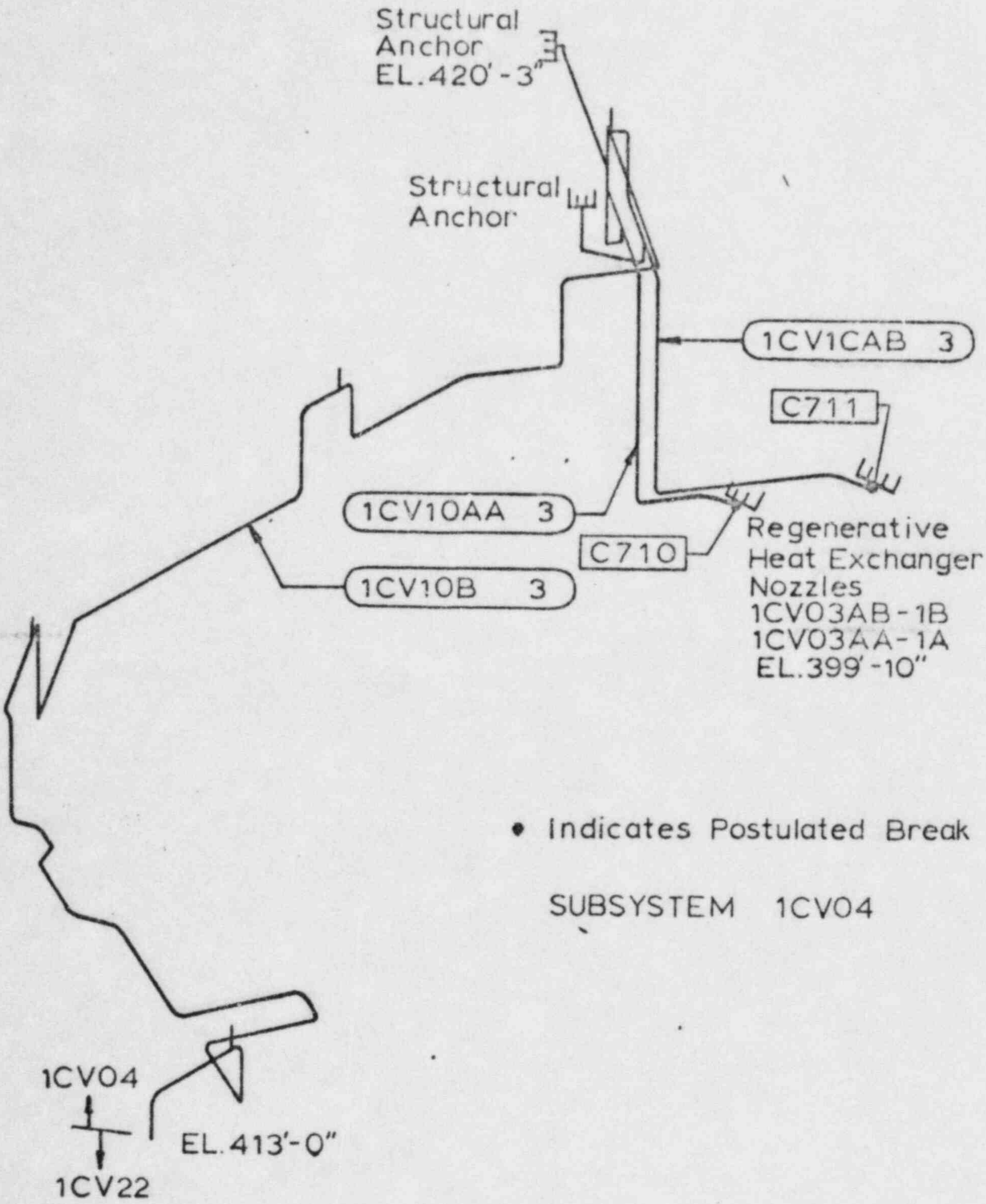
B/B

FIGURE 3C-25
CHARGING LINE
LOOP 2 SUBSYSTEM 1CV02
POSTULATED BREAK
LOCATIONS



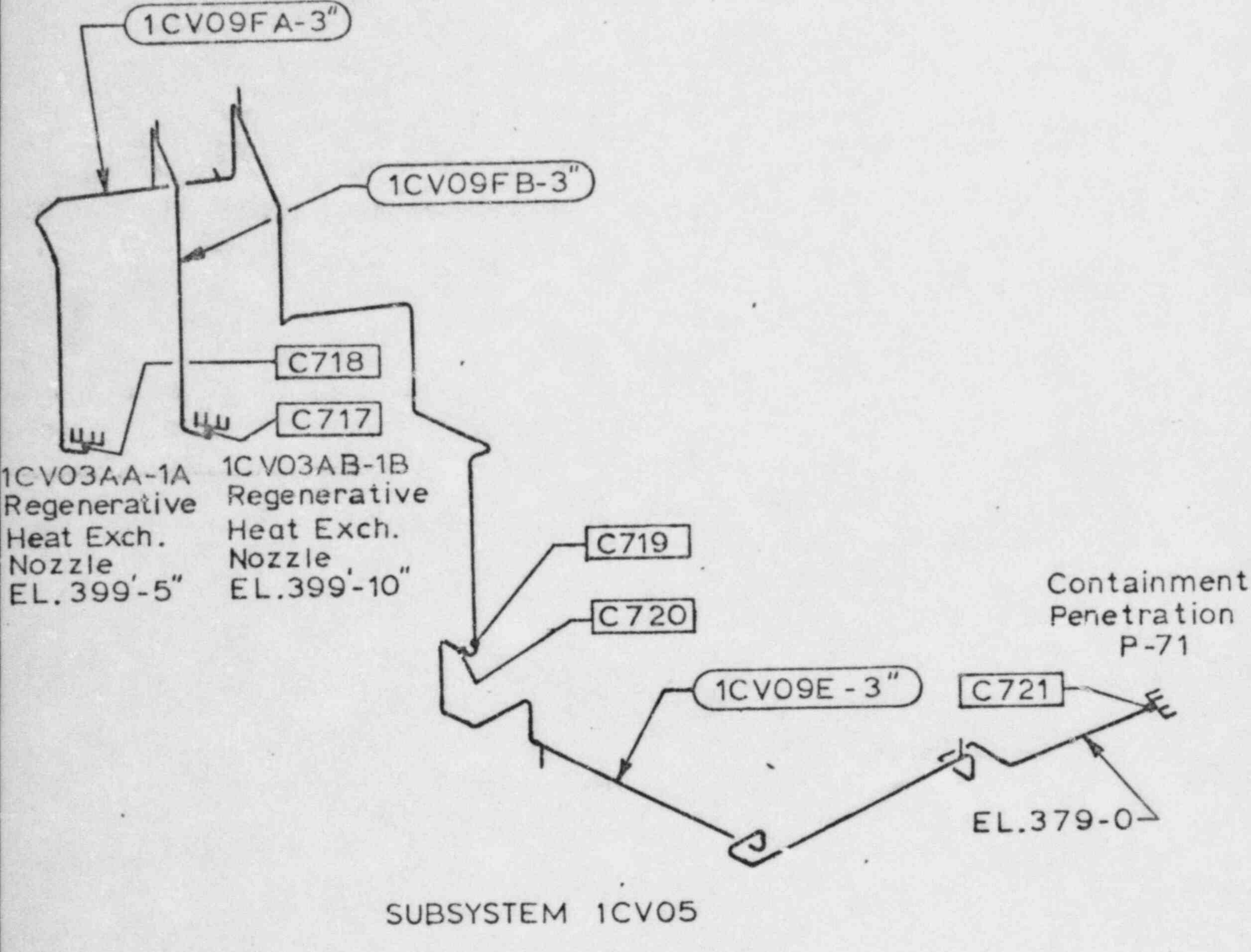
B/B

FIGURE 3.6-26
CHARGING LINE
LOOP 1 SUBSYSTEM 1CV03
POSTULATED BREAK
LOCATIONS



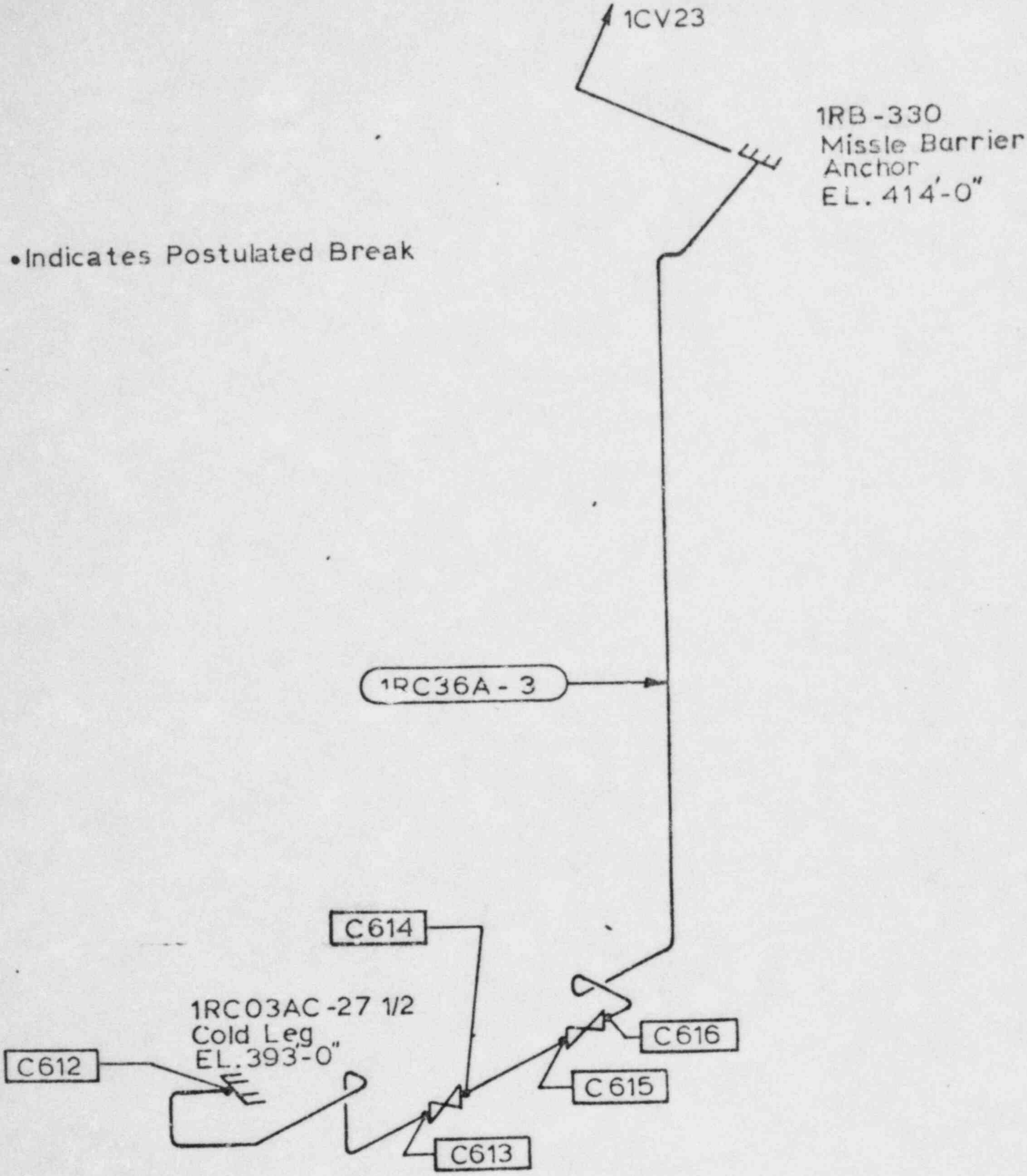
B/B

FIGURE 3.6-27
CVCS SUBSYSTEM
1CV04
POSTULATED BREAK
LOCATIONS



B/B

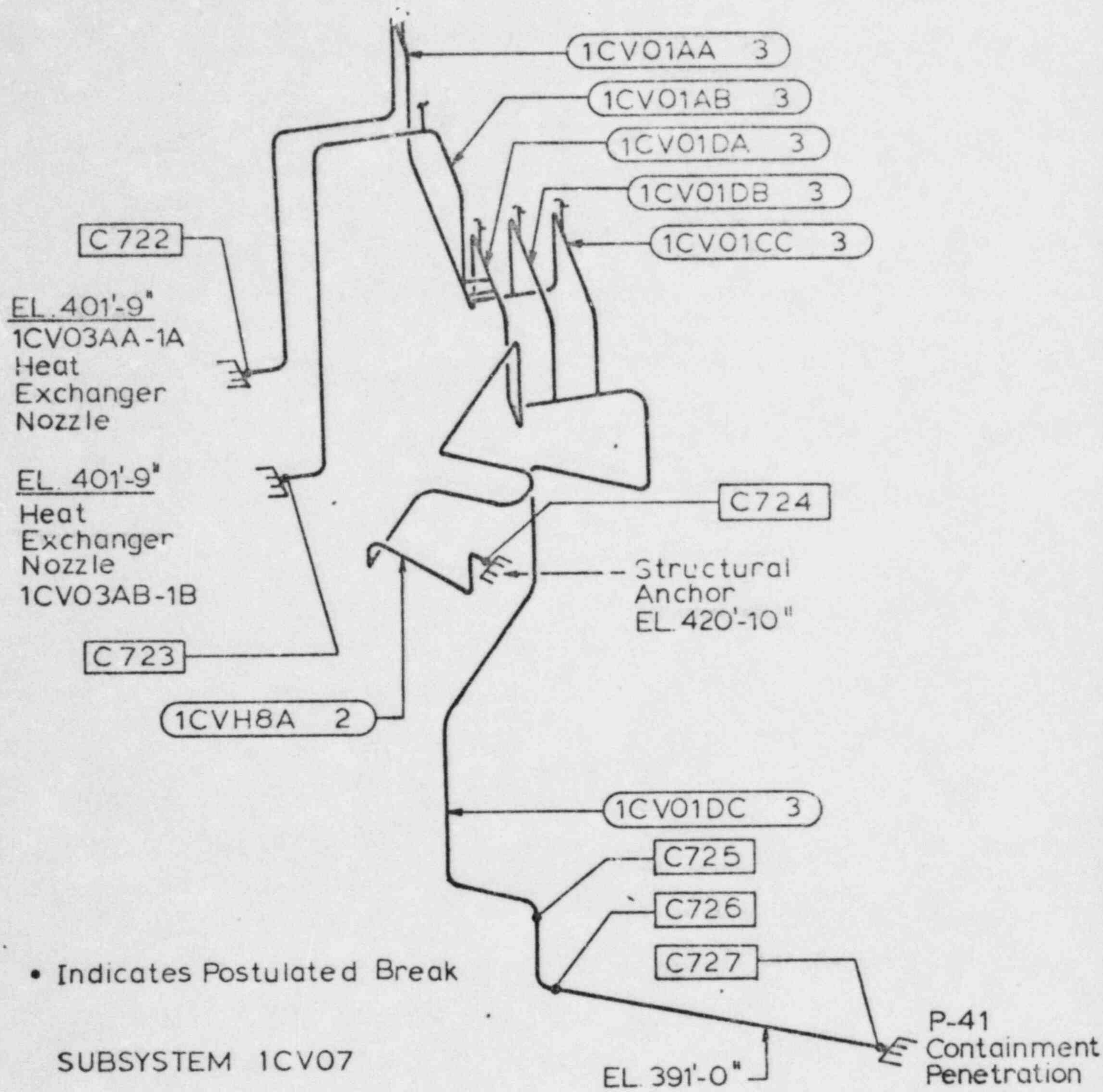
FIGURE 3.6-28
CVCS SUBSYSTEM
1CV05
POSTULATED BREAK
LOCATIONS



SUBSYSTEM 1CV06

B/B

FIGURE 3.6-29
NORMAL LETDOWN
LOOP 3 SUBSYSTEM
1CV06
POSTULATED BREAK

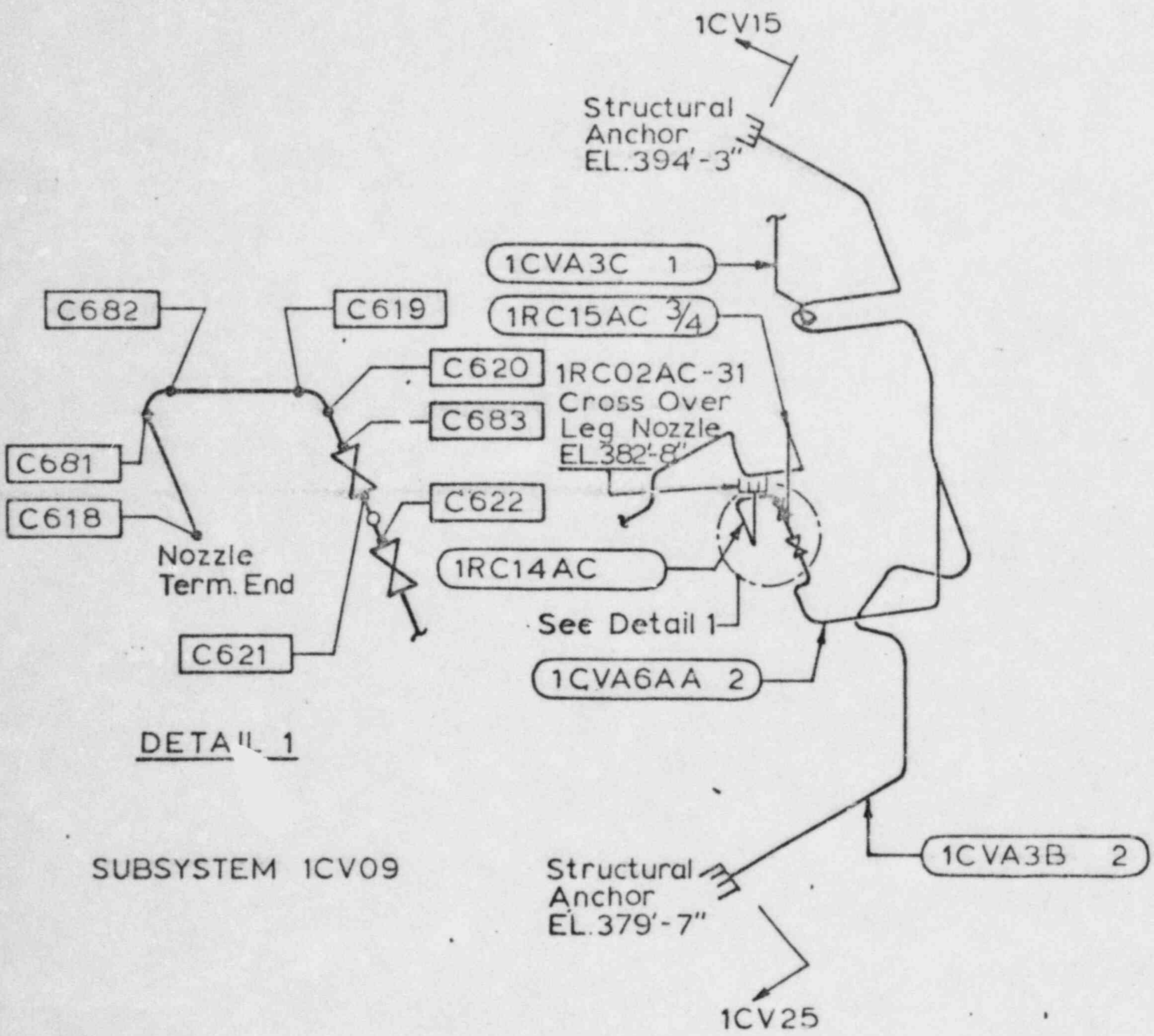


B/B

FIGURE 3.6-30

CVCS SUBSYSTEM
1CV07
POSTULATED BREAK
LOCATIONS

• Indicates Postulated Break

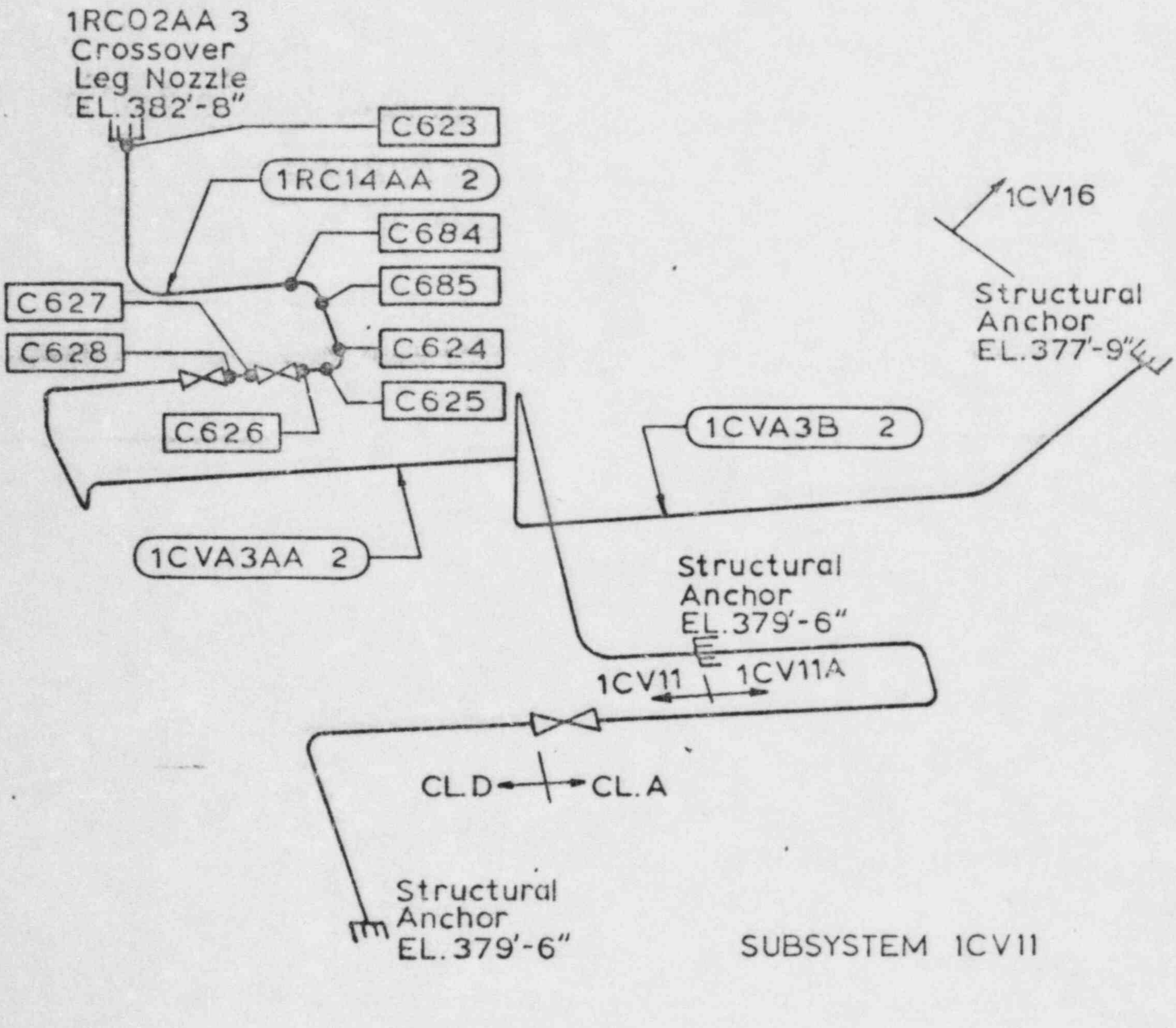


B/B

FIGURE 3.6-31

CVCS SUBSYSTEM 1CV09 POSTULATED BREAK LOCATIONS

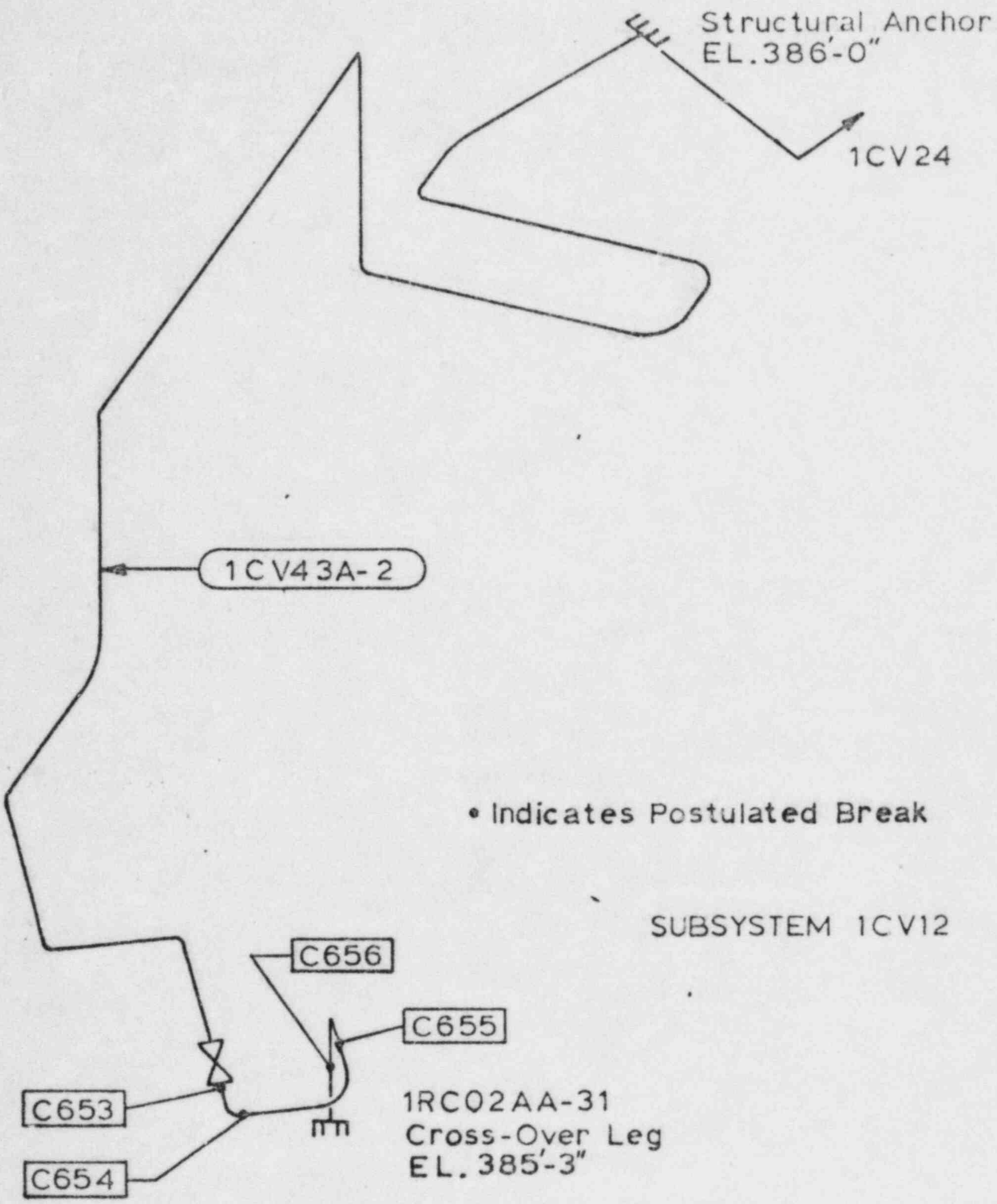
● Indicates Postulated Break



SUBSYSTEM 1CV11

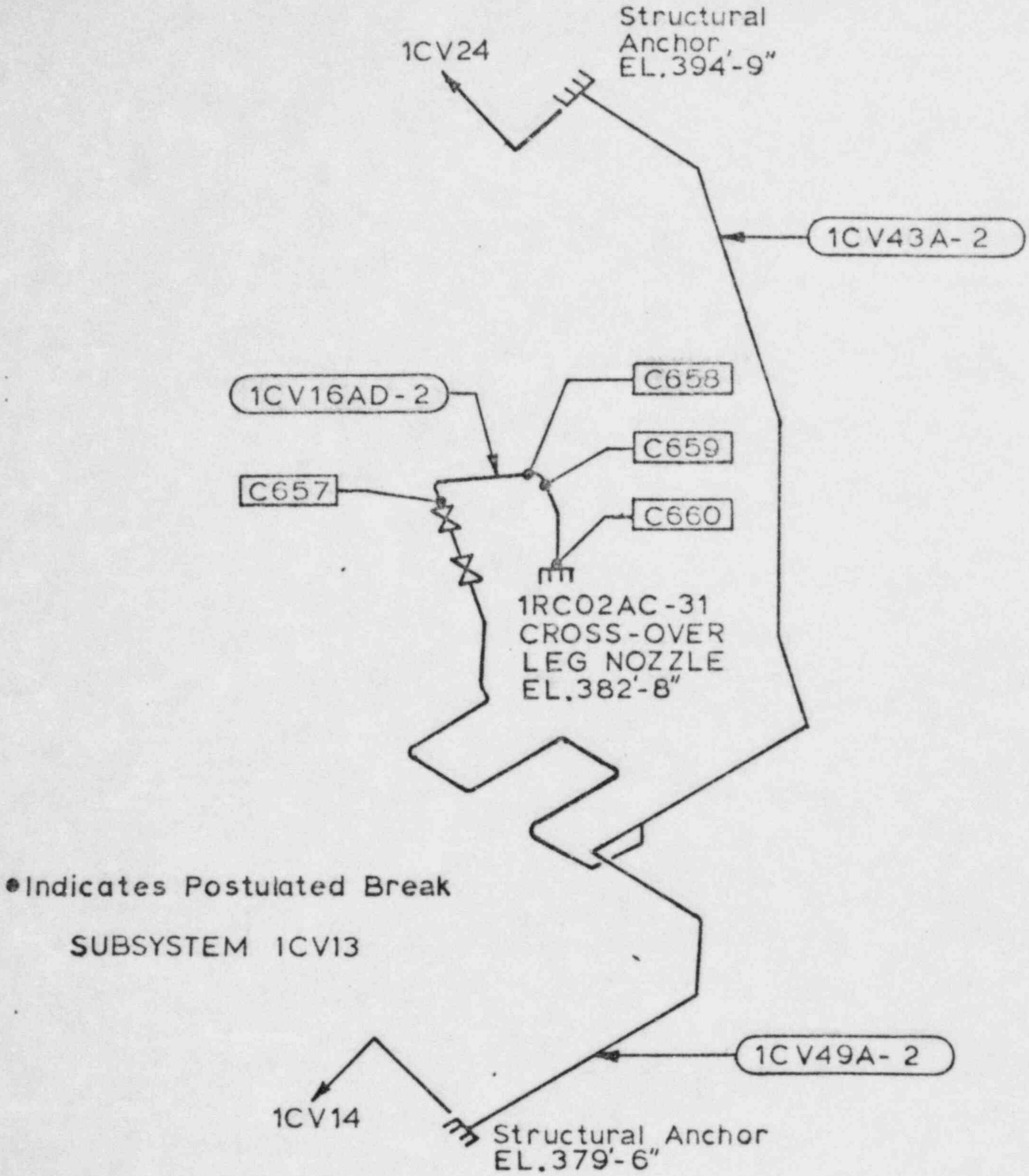
d/B

FIGURE 3.6-32
EXCESS LETDOWN
LOOP 1 SUBSYSTEM 1CV11
POSTULATED BREAK
LOCATIONS



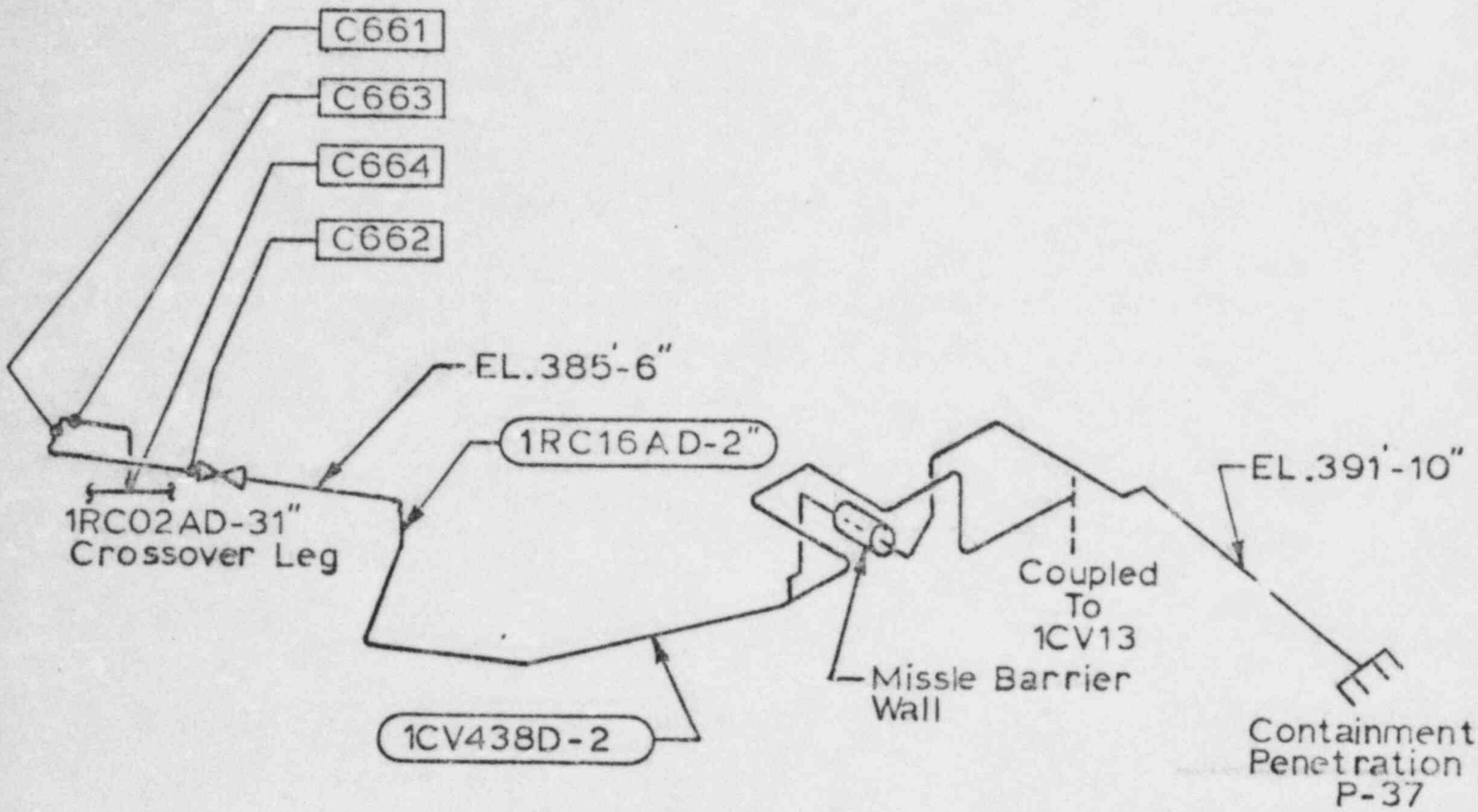
B/B

FIGURE 3.6-33
LOOP FILL LINE
LOOP 1 SUBSYSTEM
1CV12 POSTULATED
BREAK LOCATIONS



B/B

FIGURE 3.6-34
LOOP FILL LINE
LOOP 3 SUBSYSTEM
1CV13 POSTULATED
BREAK LOCATIONS



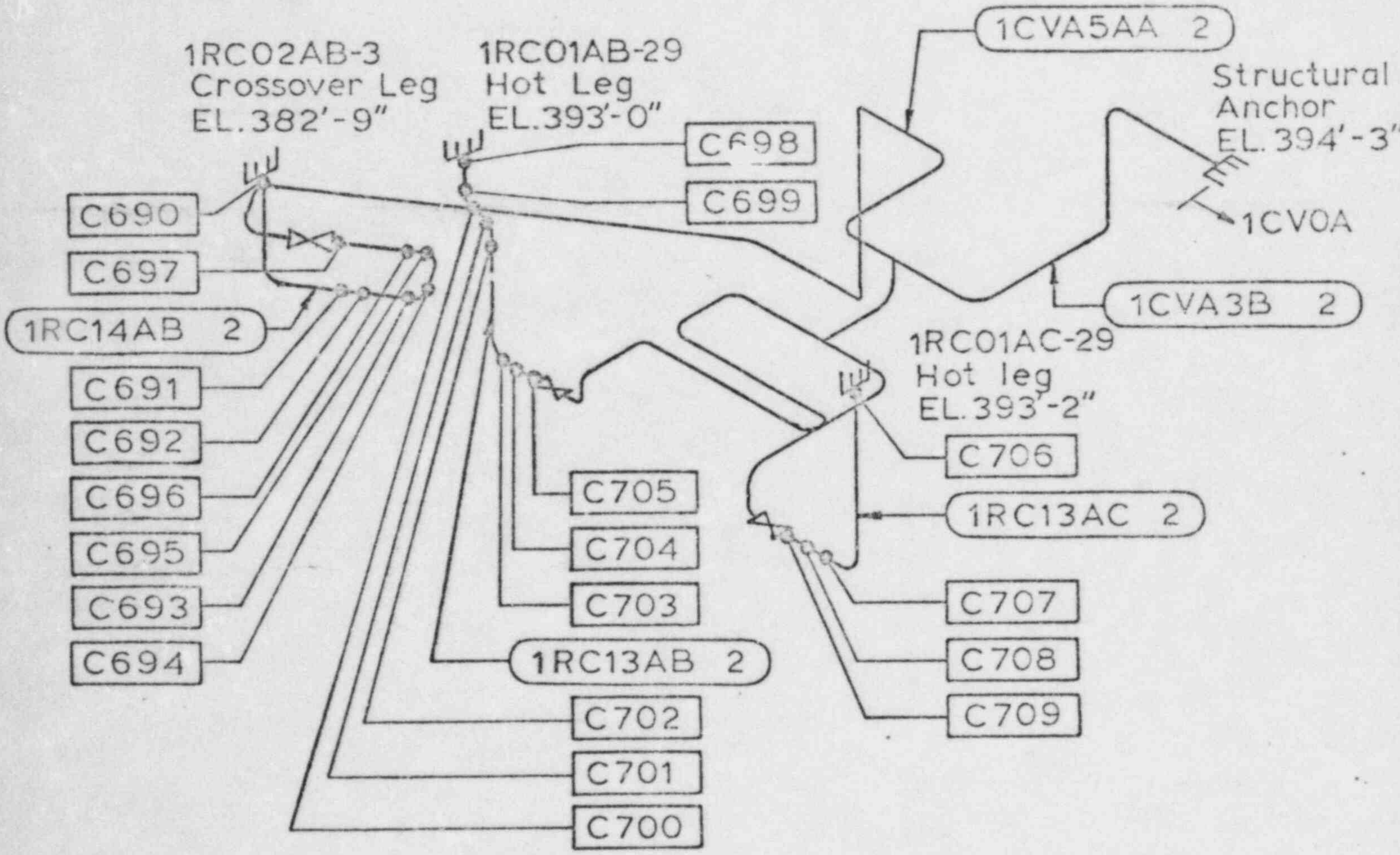
• Indicates Postulated Break

SUBSYSTEM 1CV14

B/B

FIGURE 3.6-35
LOOP FILL LINE
LOOP 4 SUBSYSTEM
1CV14 POSTULATED
BREAK LOCATIONS

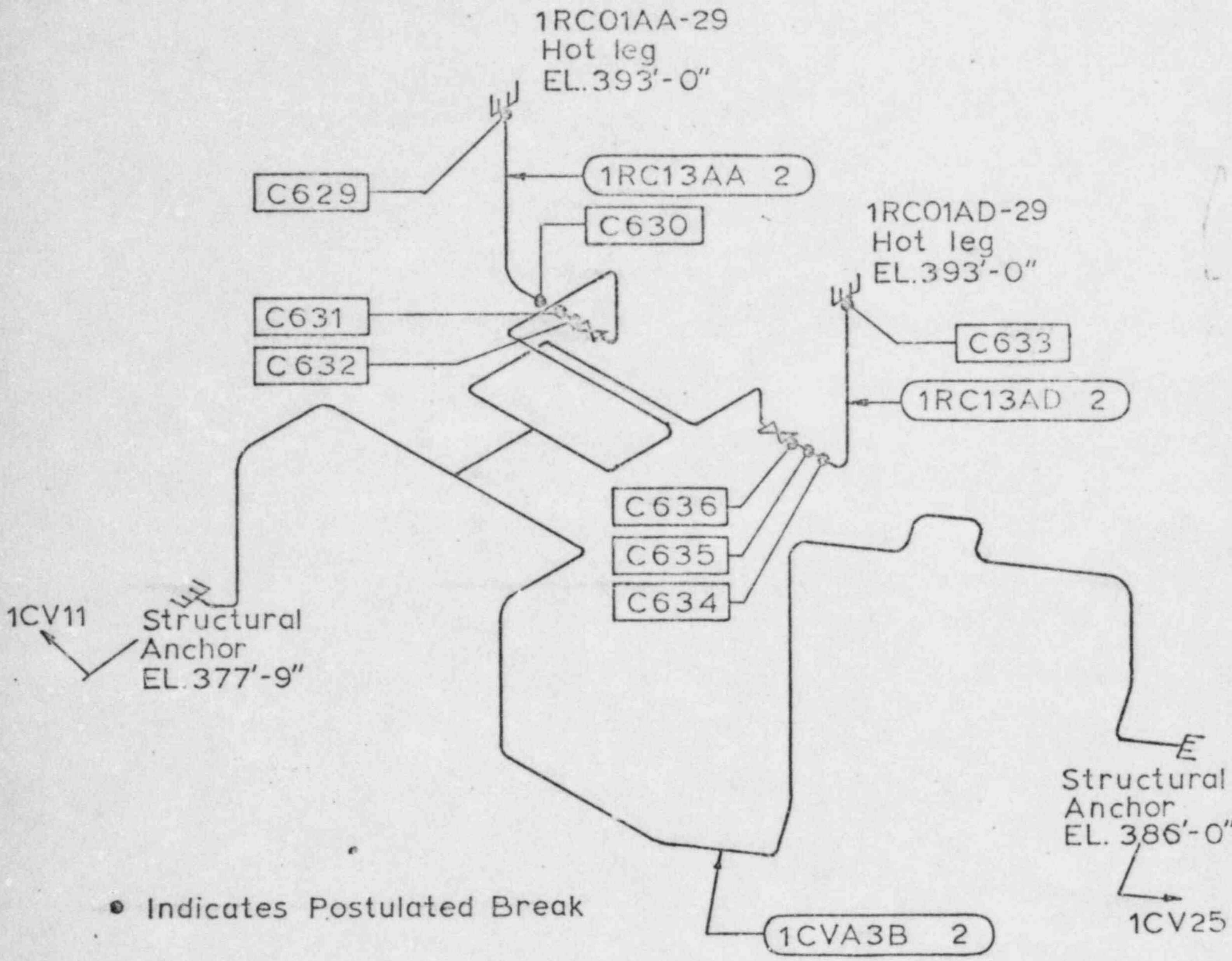
• Indicates Postulated Break



SUBSYSTEM 1CV15

B/B

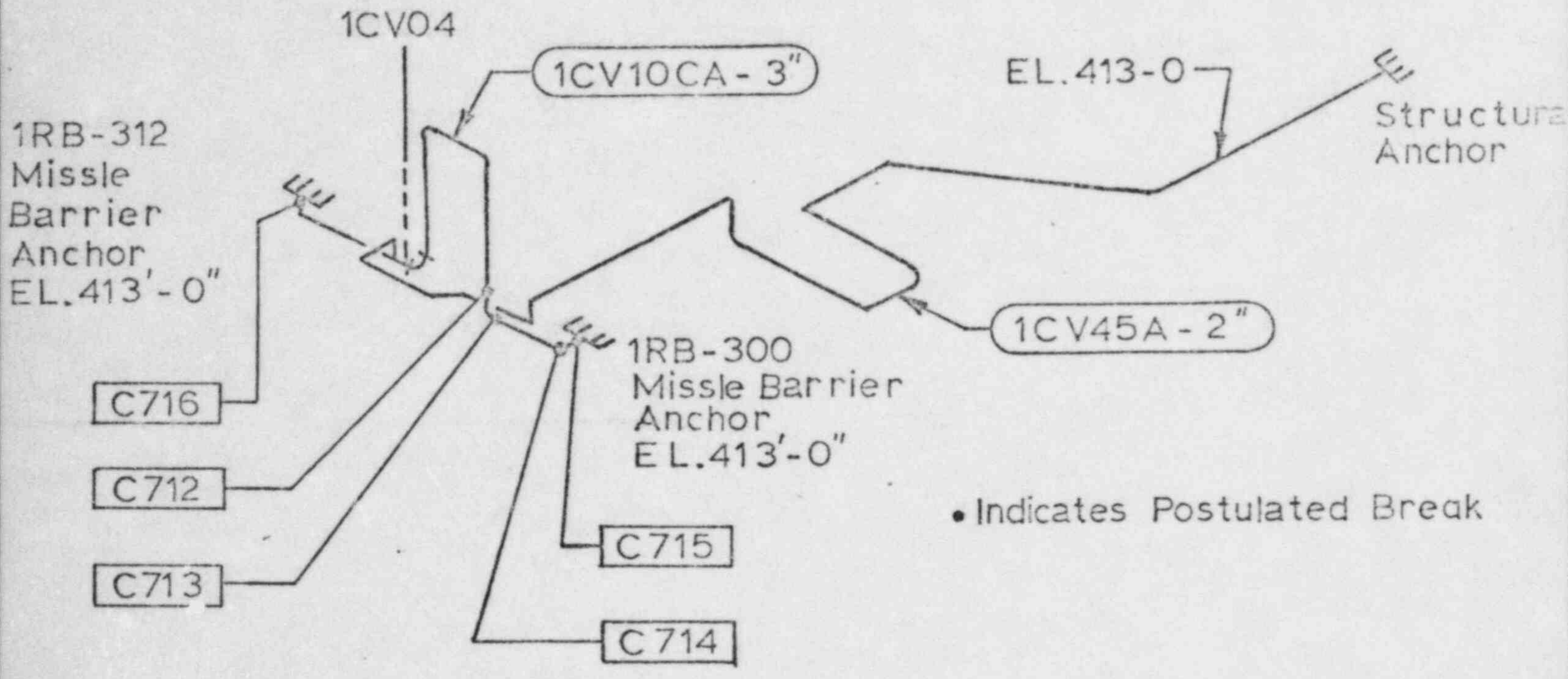
FIGURE 36-36
EXCESS LETDOWN
LOOPS 2 AND 3
SYBSYSTEM 1CV15
POSTULATED BREAK
LOCATIONS



SUBSYSTEM 1CV16

B/B

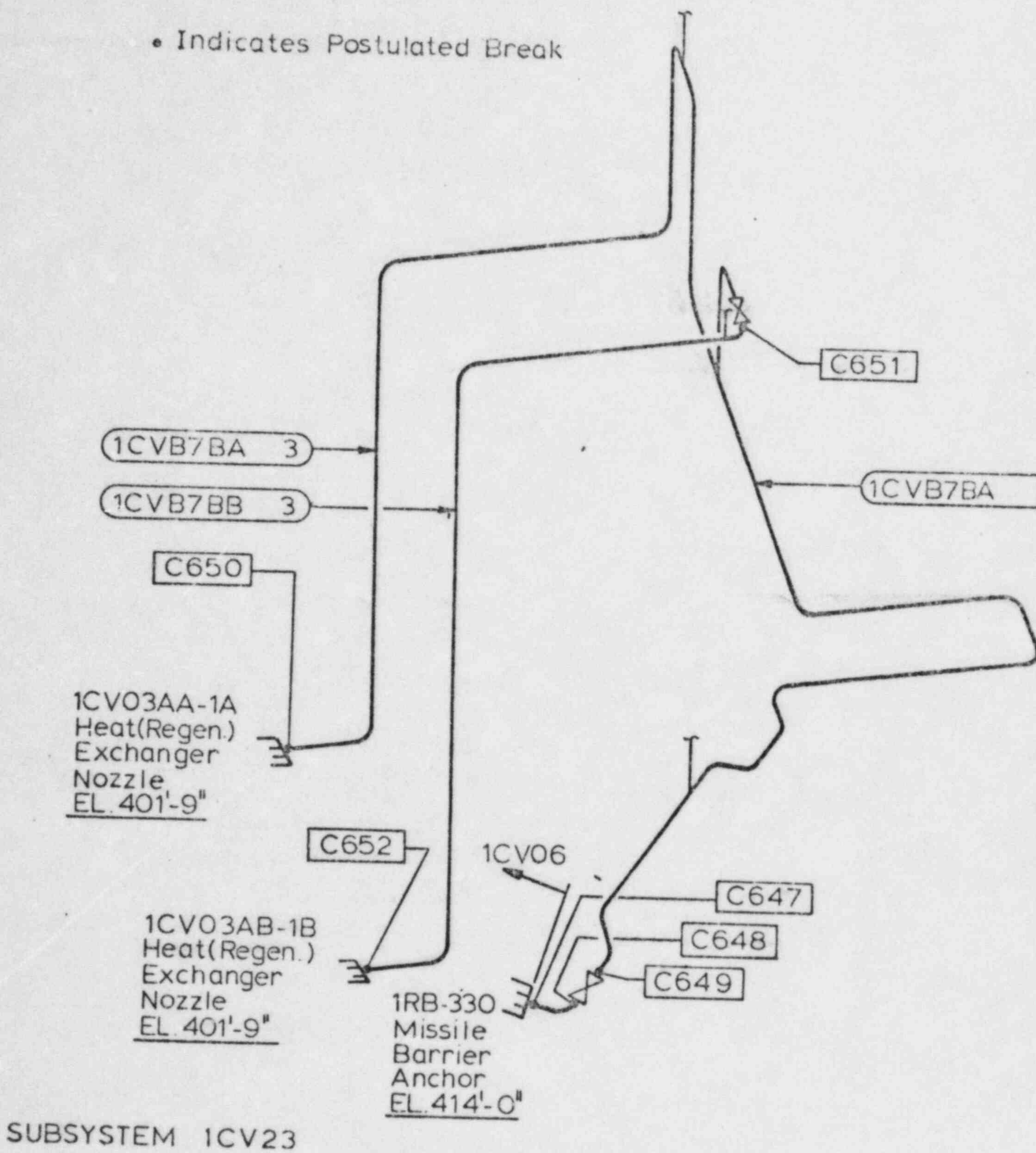
FIGURE 3.6-37
EXCESS LETDOWN
LOOPS 1 AND 4
SUBSYSTEM 1CV16
POSTULATED BREAK
LOCATIONS



SUBSYSTEM 1CV22

B/B

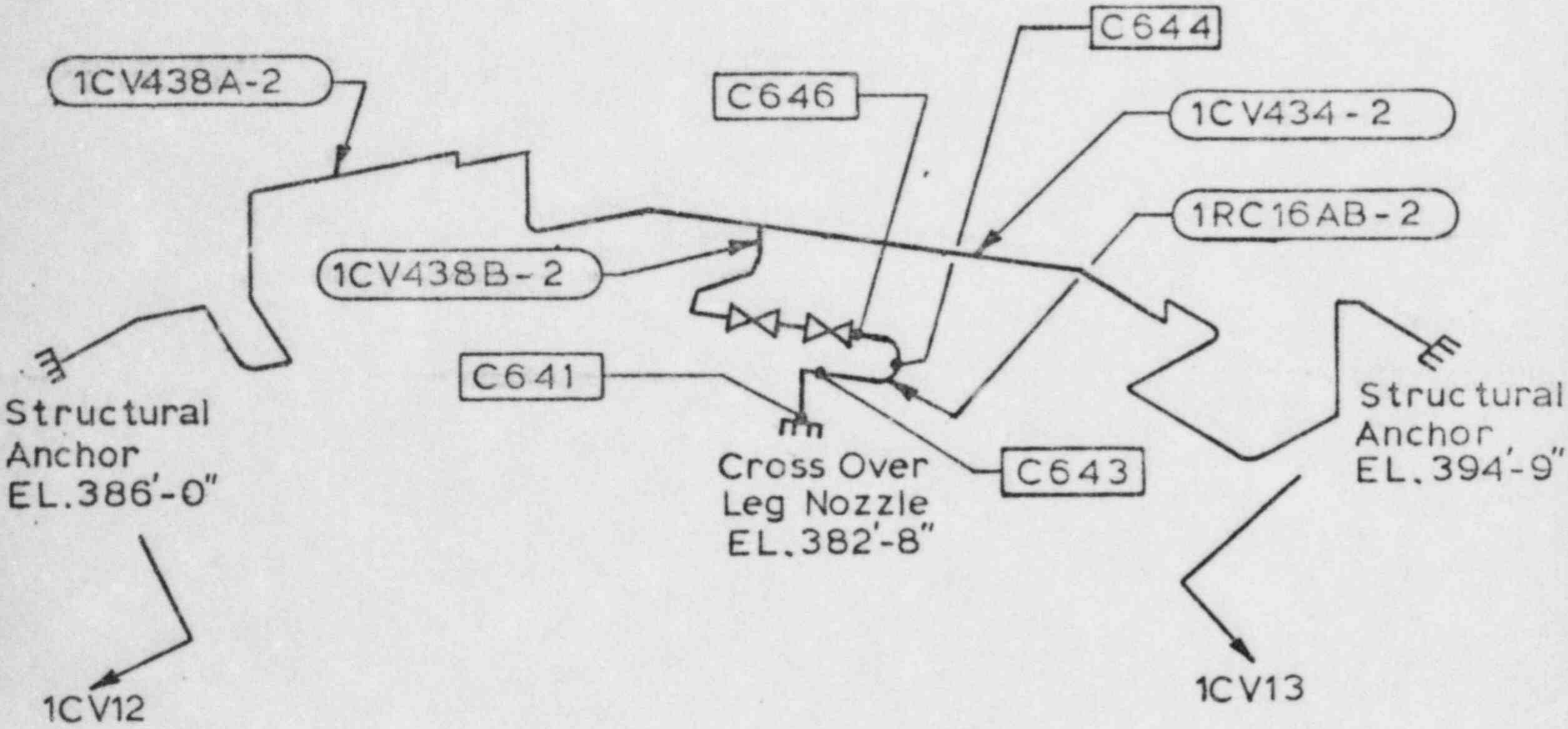
FIGURE 3.6-38
CVCS SUBSYSTEM
1CV22 POSTULATED
BREAK LOCATIONS



B/B

FIGURE 3.6-39

CVCS SUBSYSTEM
1CV23 POSTULATED
BREAK LOCATIONS

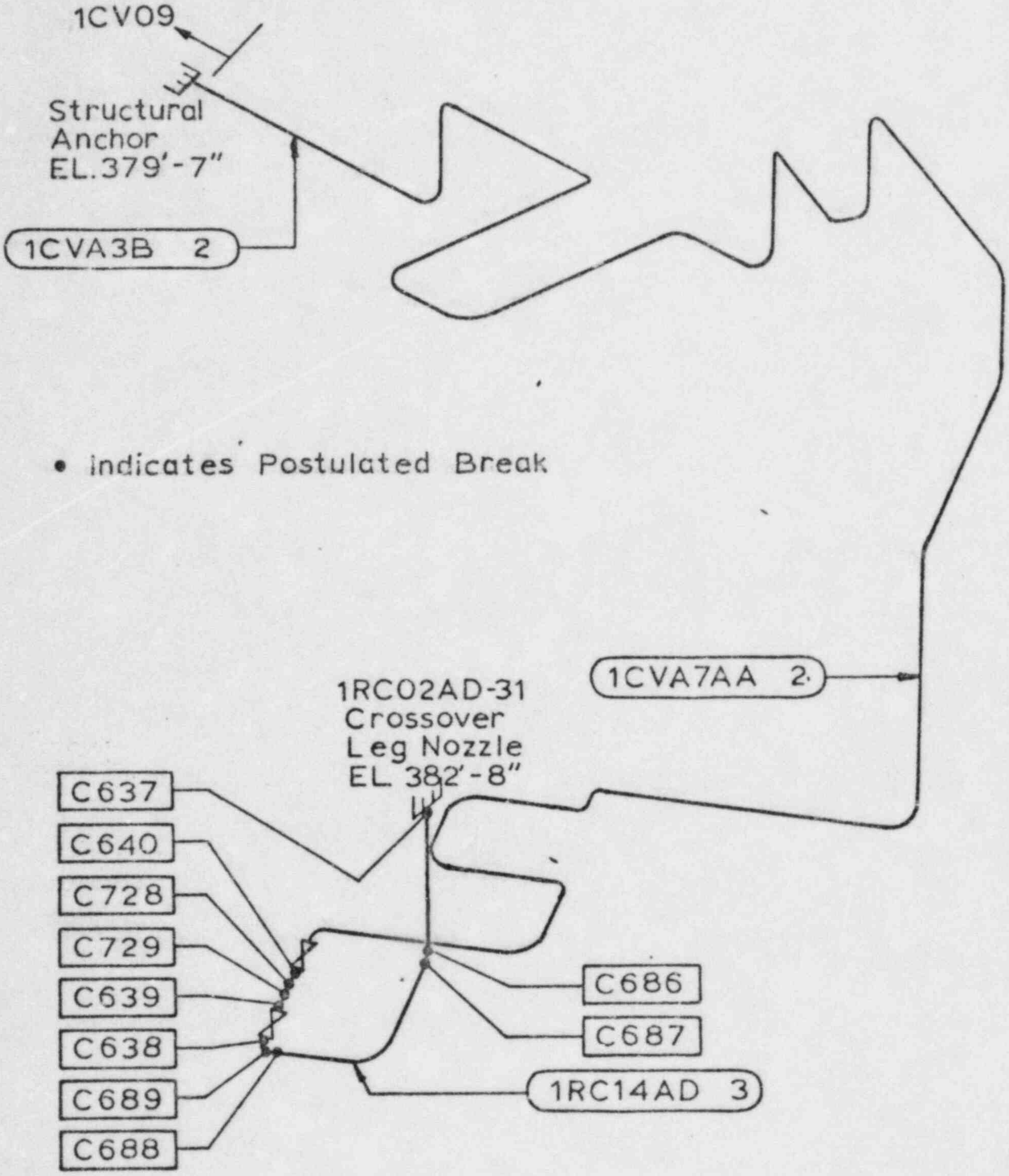


• Indicates Postulated Break

SUBSYSTEM 1CV24

B/B

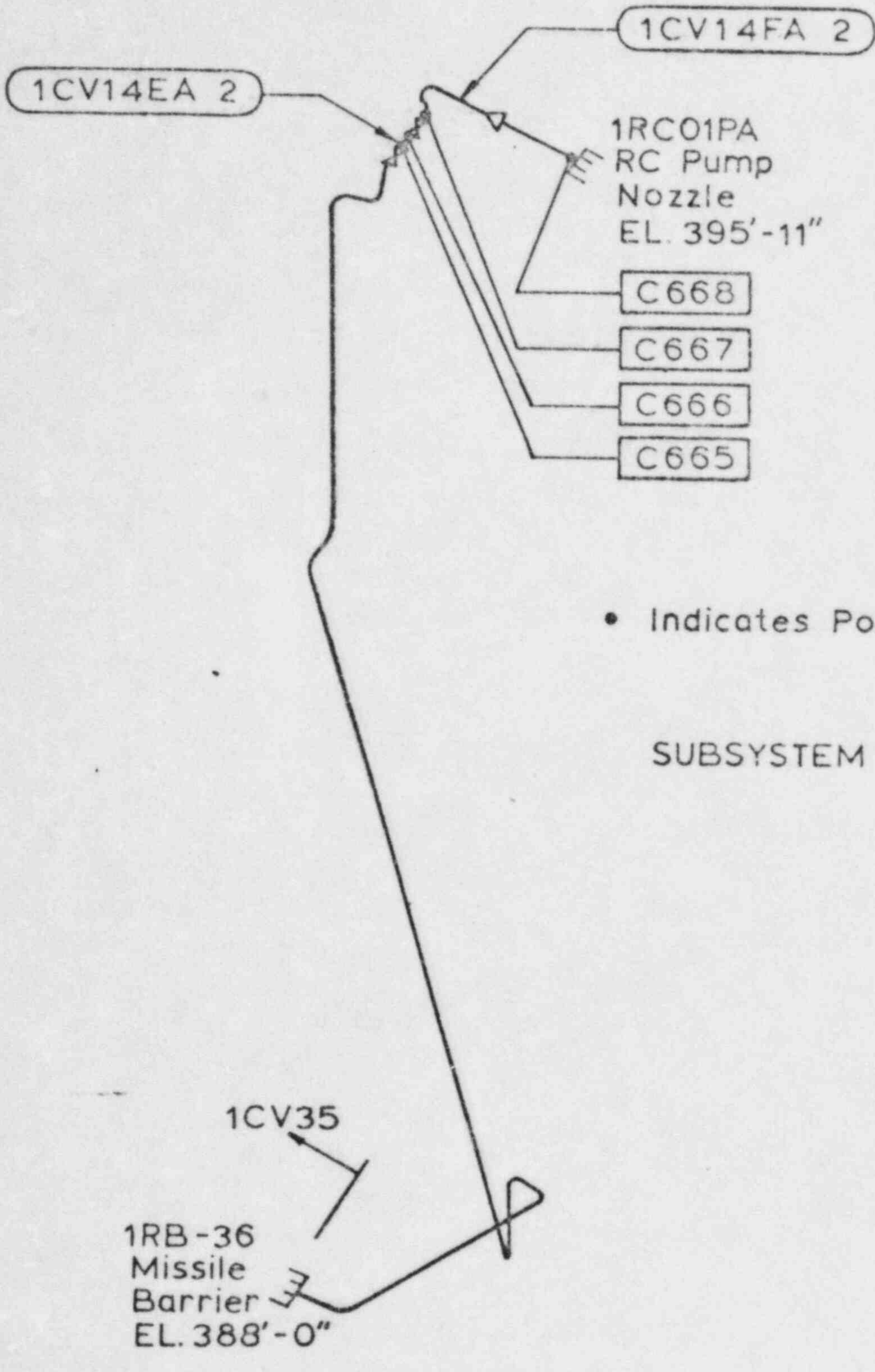
FIGURE 3.6-40
LOOP FILL LINE
LOOP 2 SUBSYSTEM
1CV24 POSTULATED
BREAK LOCATIONS



SUBSYSTEM 1CV25

B/B

FIGURE 3.6-41
EXCESS LETDOWN
LOOP 4 SUBSYSTEM
1CV25 POSTULATE
BREAK LOCATIONS



• Indicates Postulated Break

SUBSYSTEM 1CV34

B/B

FIGURE 3.6-42
RCP SEAL WATER
INJECTION LOOP 1
SUBSYSTEM 1CV34
POSTULATED BREAK LOCATIONS

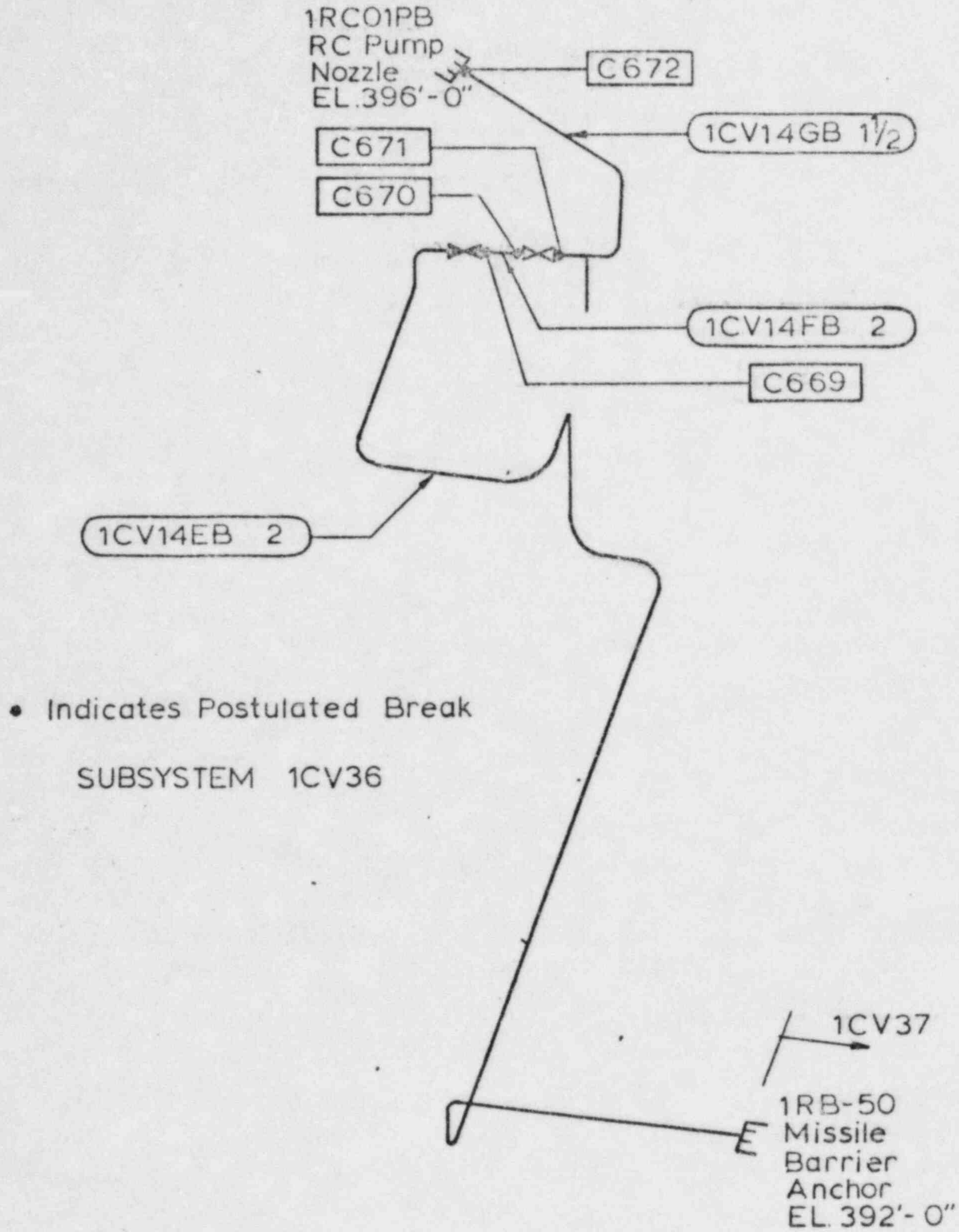
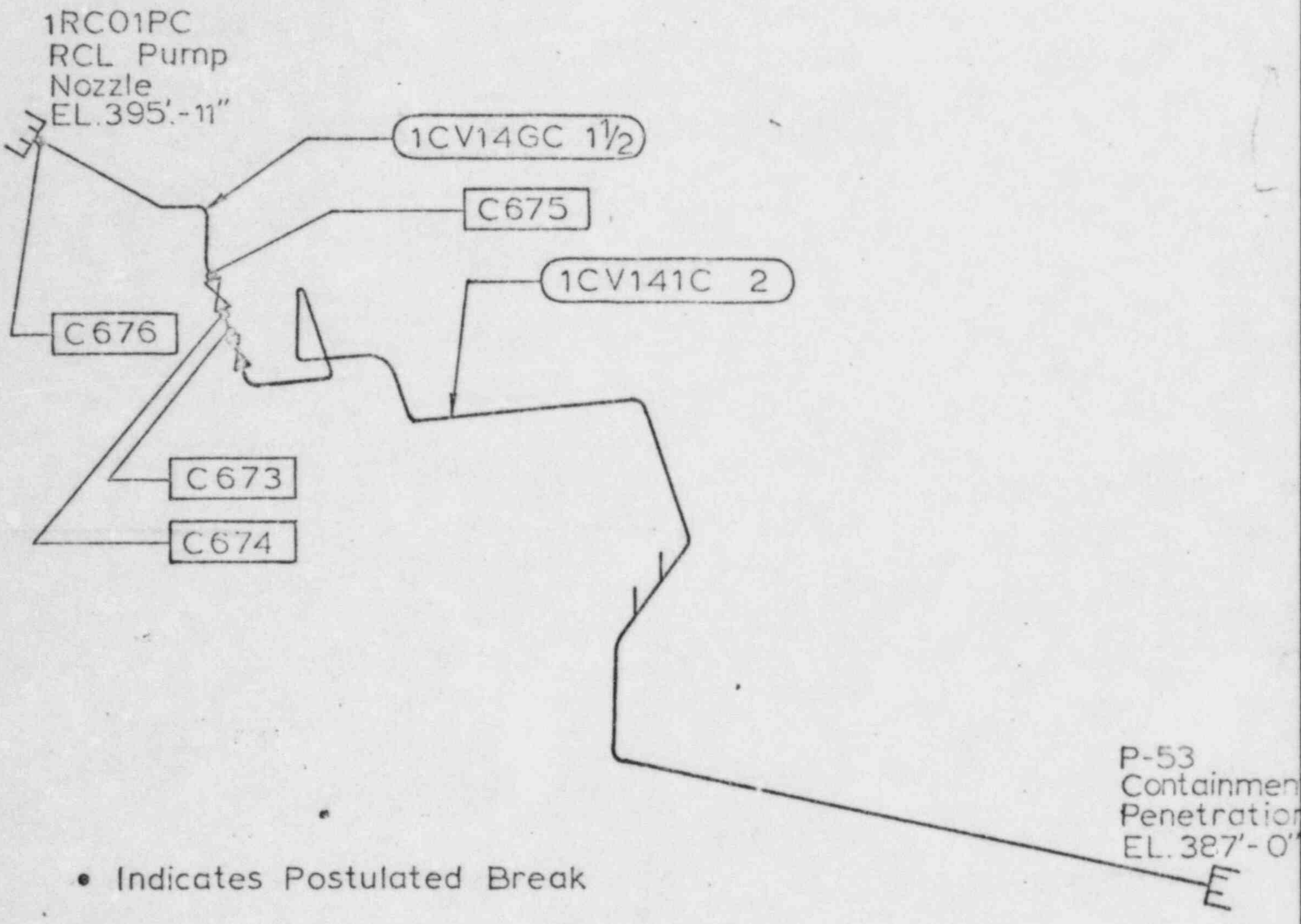


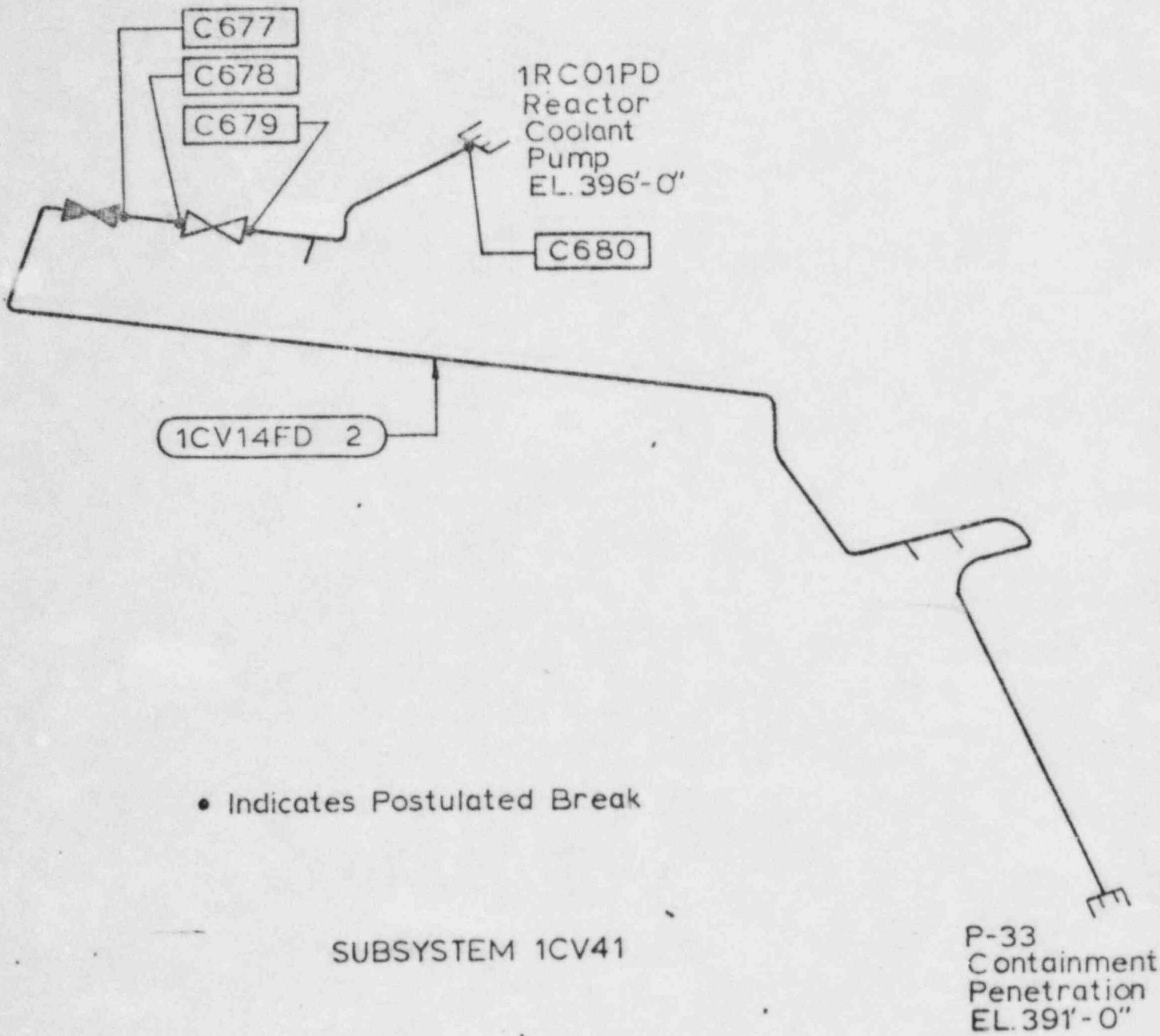
FIGURE 3.6-43
RCP SEAL WATER
INJECTION LOOP 2
SUBSYSTEM 1CV36
POSTULATED BREAK
LOCATIONS



SUBSYSTEM 1CV40

B/B

FIGURE 3.6-44
RCP SEAL WATER
INJECTION LOOP 3
SUBSYSTEM 1CV40
POSTULATED BREAK

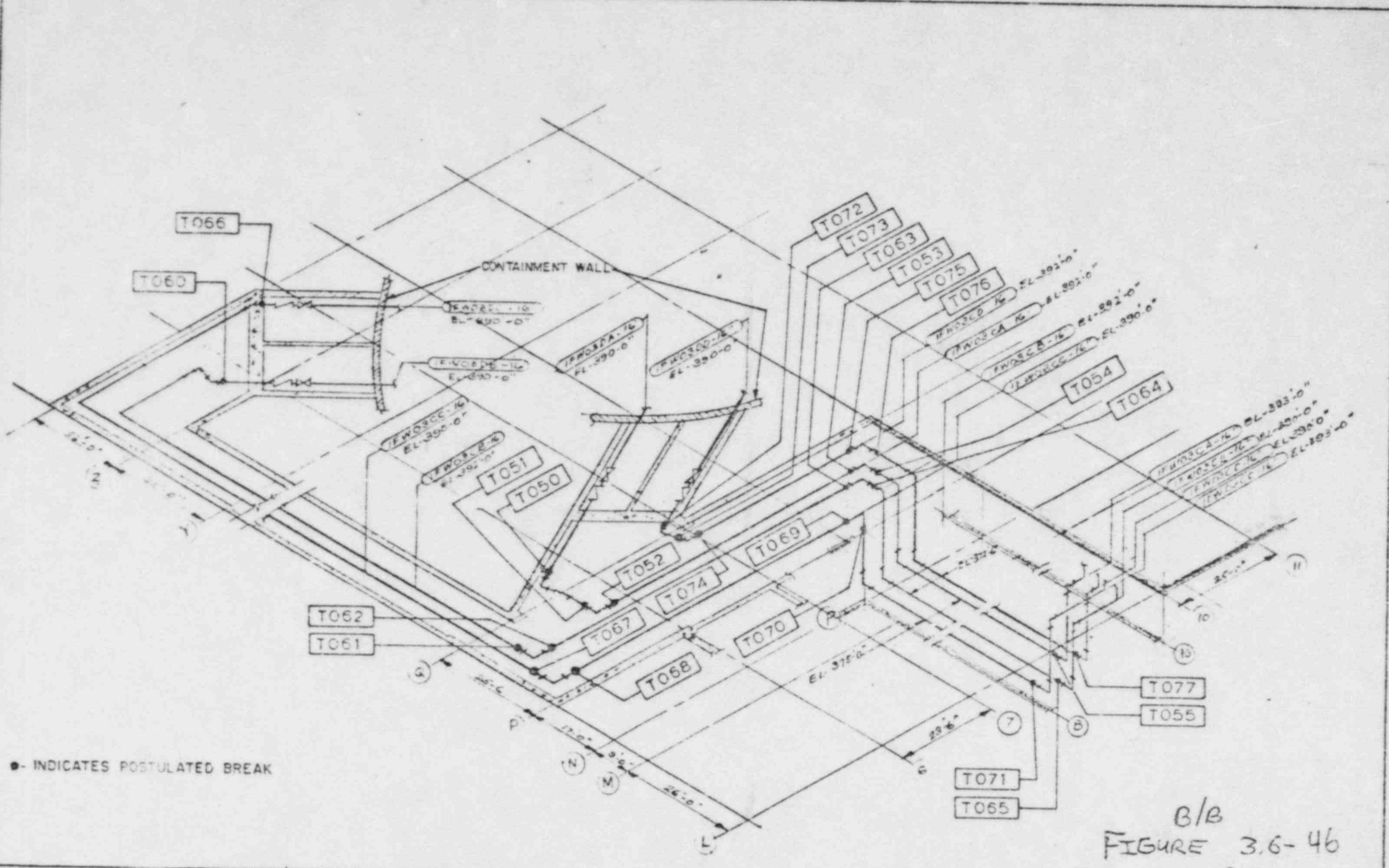


SUBSYSTEM 1CV41

B/B

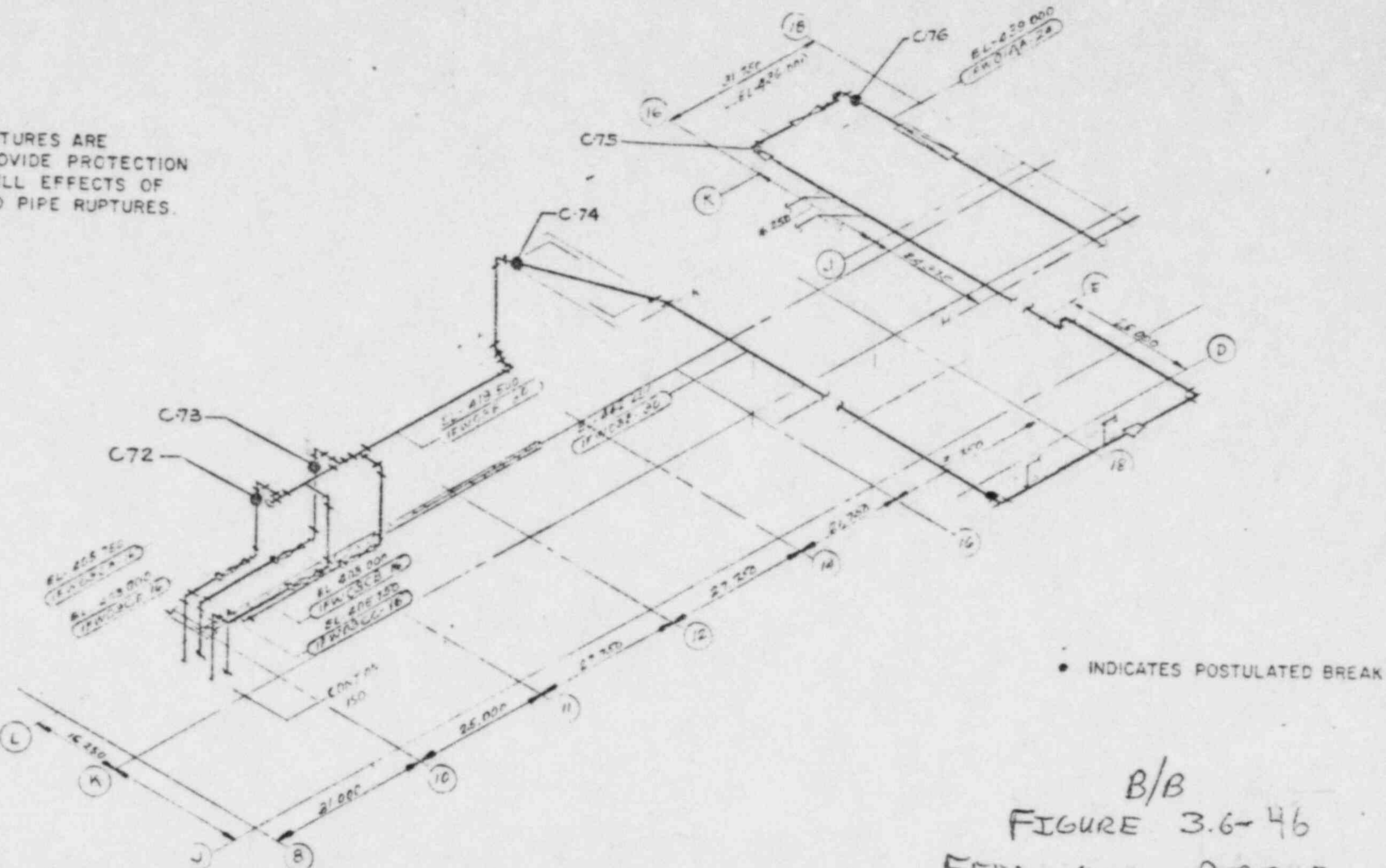
FIGURE 3.6-45

RCP SEAL WATER
INJECTION LOOP 4
SUBSYSTEM 1CV41
POSTULATED BREAK
LOCATIONS

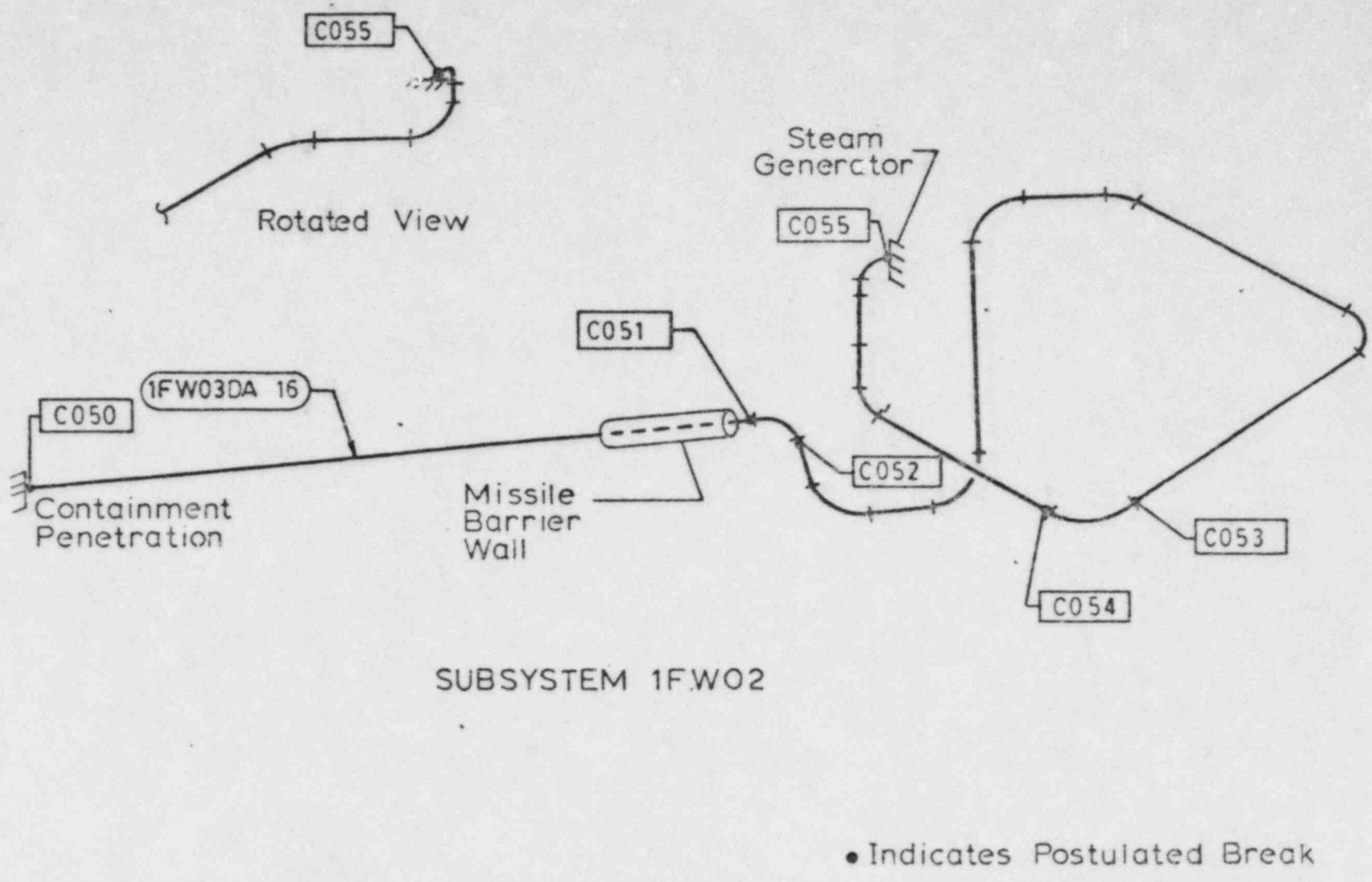


B/B
FIGURE 3.6-46
FEEDWATER PIPING SYSTEMS IN MAIN STEAM PIPE TUNNEL POSTULATED BREAK LOCATIONS (Sheet 1 of 2)

NOTE:
ADJACENT STRUCTURES ARE DESIGNED TO PROVIDE PROTECTION AGAINST THE FULL EFFECTS OF THE POSTULATED PIPE RUPTURES.



B/B
FIGURE 3.6-46
FEEDWATER PIPING SYSTEMS IN MAIN STEAM PIPE TUNNEL POSTULATED BREAK LOCATIONS
(SHEET 2 OF 2)

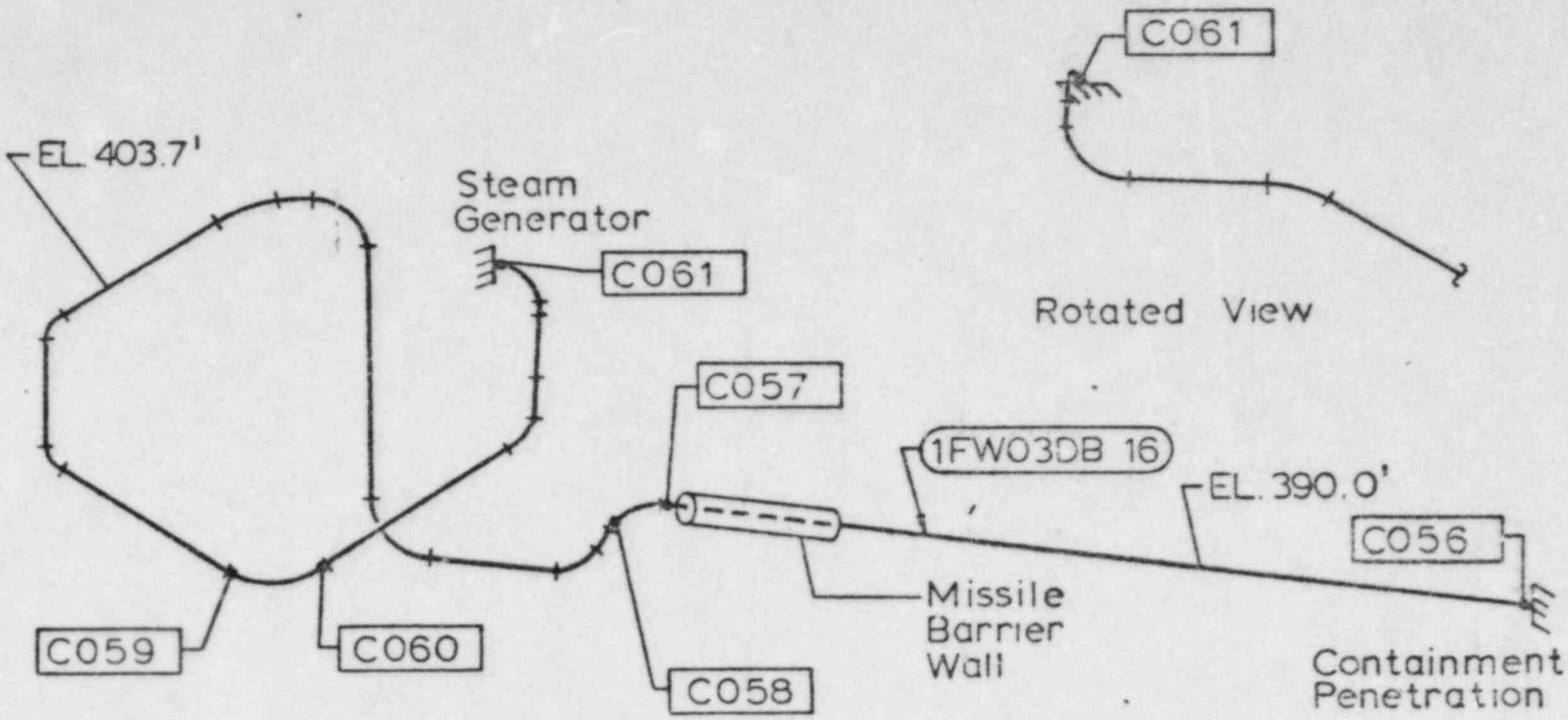


B/B

FIGURE 3.6-47

FEEDWATER LOOP 1

POSTULATED BREAK LOCATIONS



SUBSYSTEM 1FW03

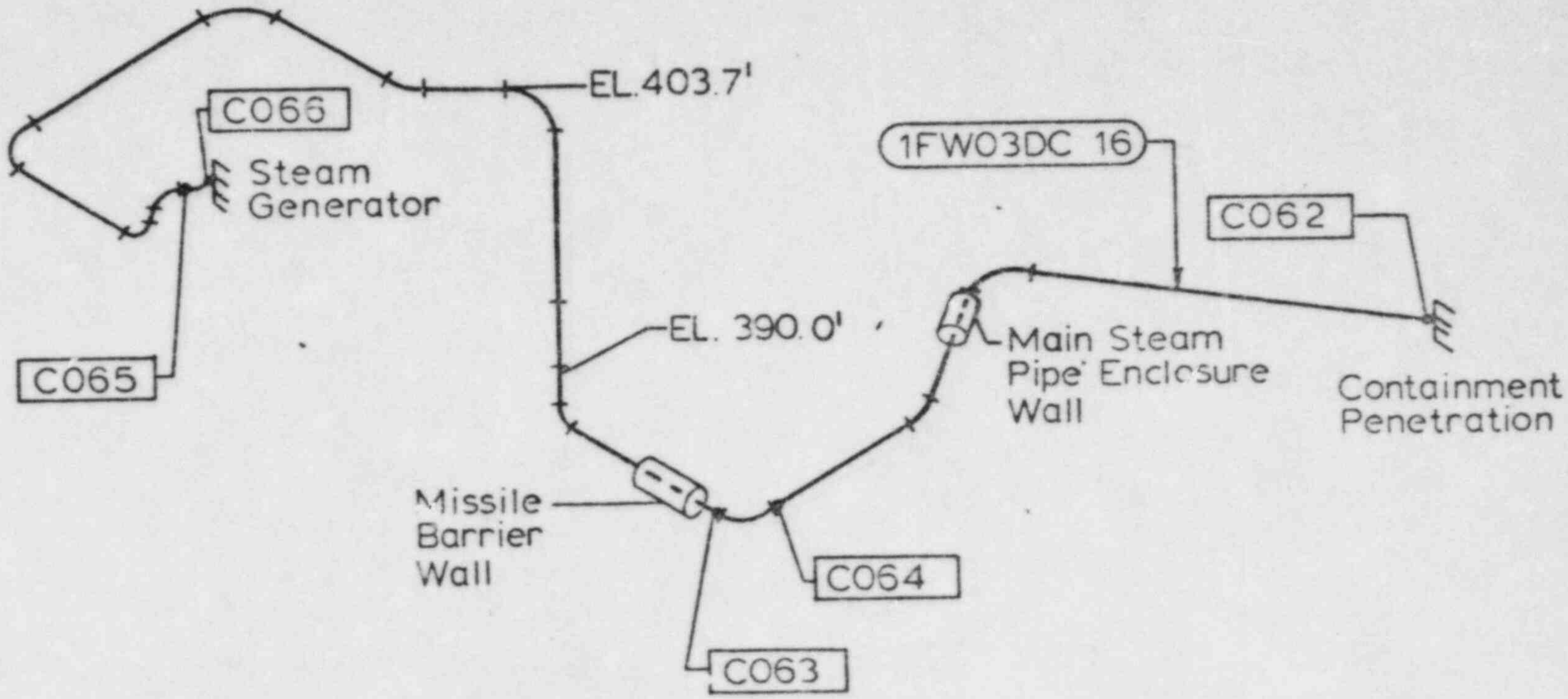
• Indicates Postulated Break

B/B

FIGURE 3.6-48

FEEDWATER LOOP 2

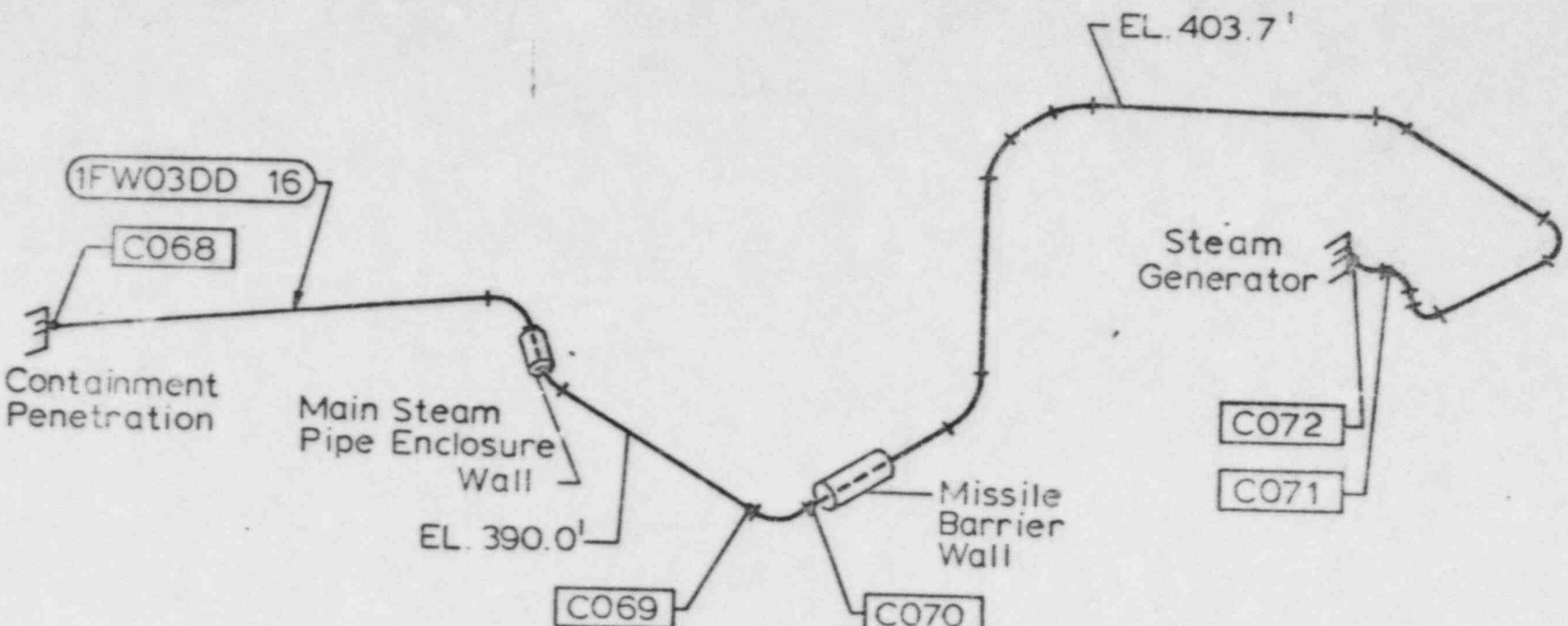
POSTULATED BREAK LOCATIONS



SUBSYSTEM 1FW04

• Indicates Postulated Break

B/B
FIGURE 3.6 - 49
FEED WATER LOOP 3
POSTULATED BREAK LOCATIONS



SUBSYSTEM 1FW05

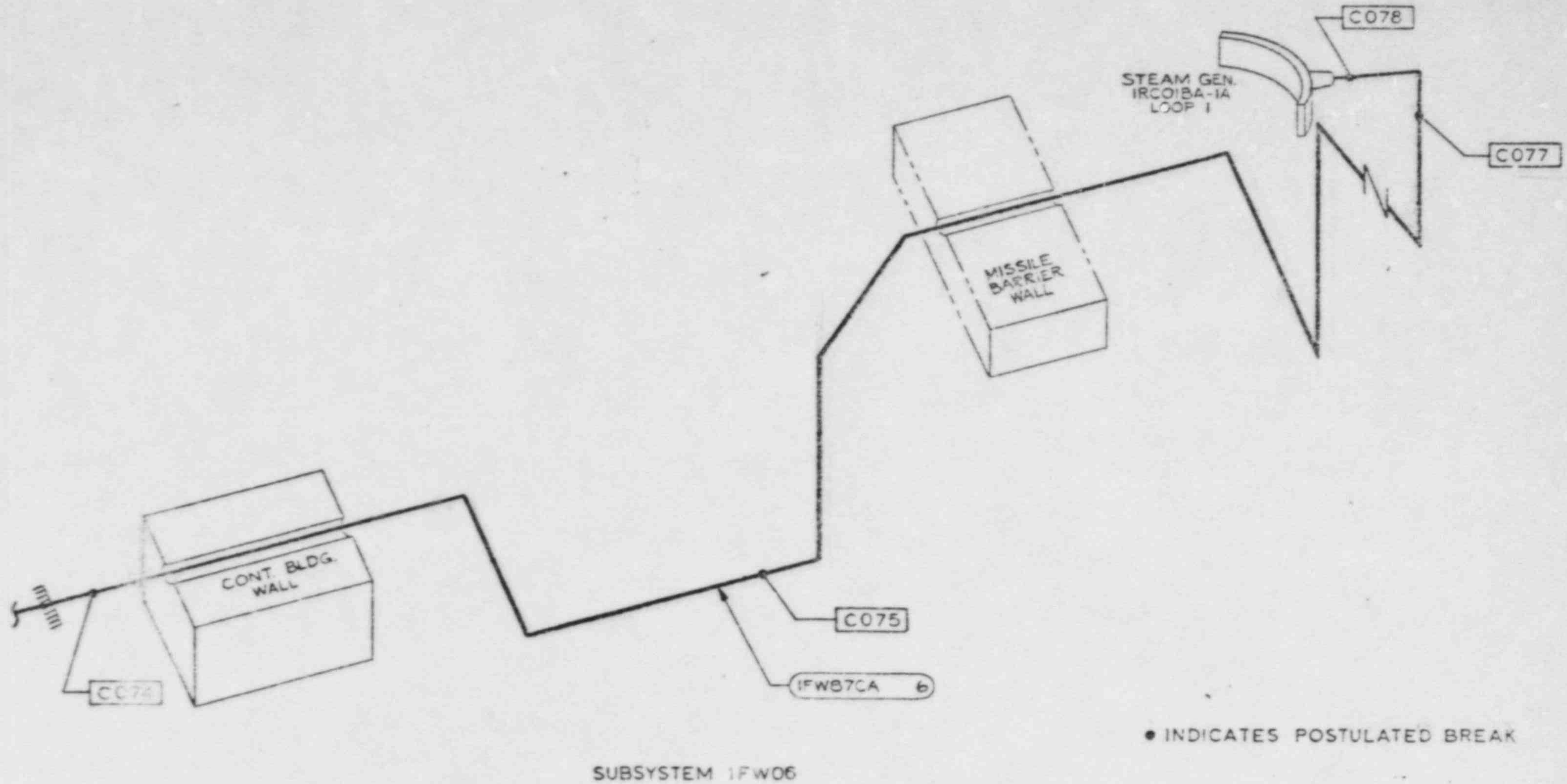
• Indicates Postulated Break

B/B

FIGURE 3.6-50

FEEDWATER LOOP 4

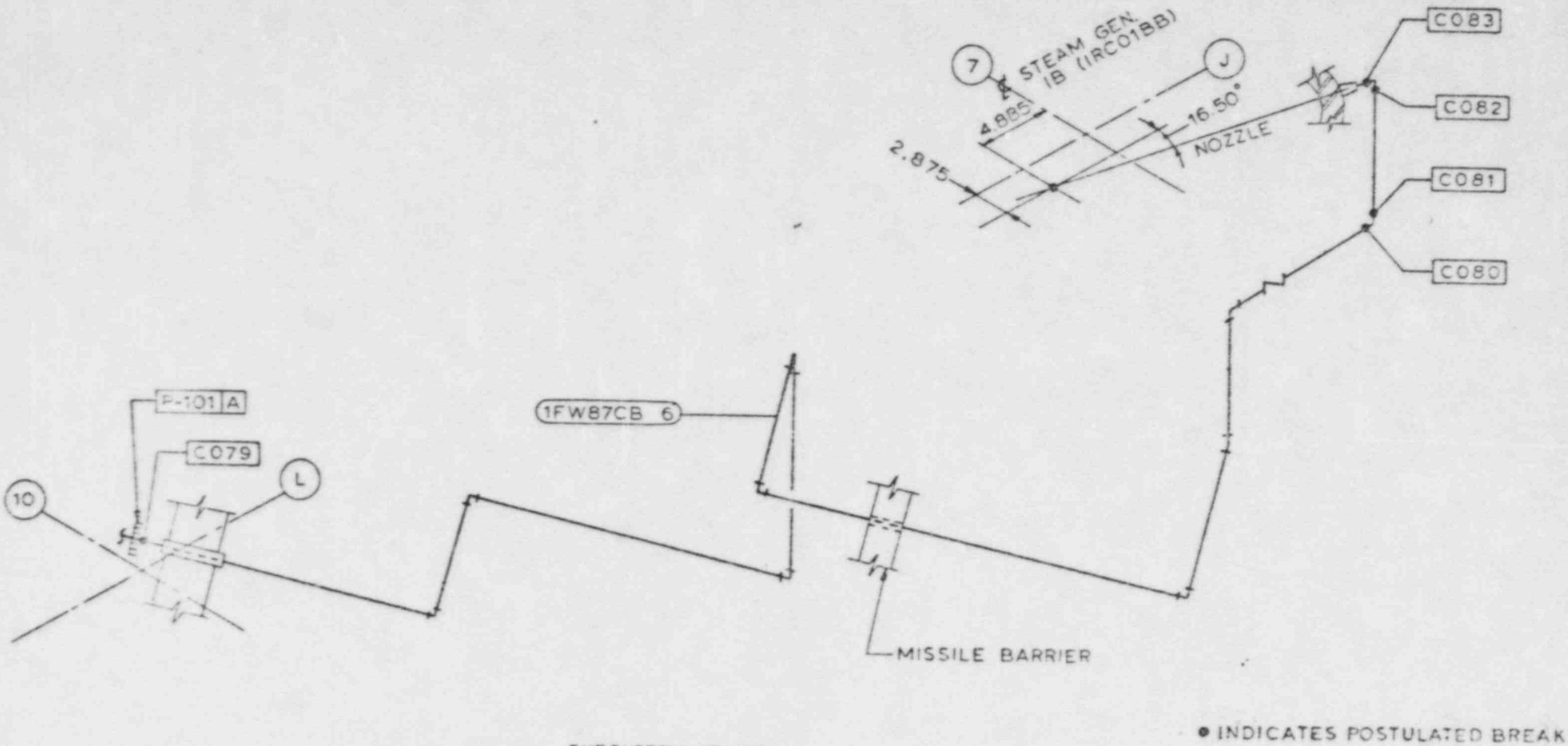
POSTULATED BREAK LOCATIONS



B/B

FIGURE 3.6-51

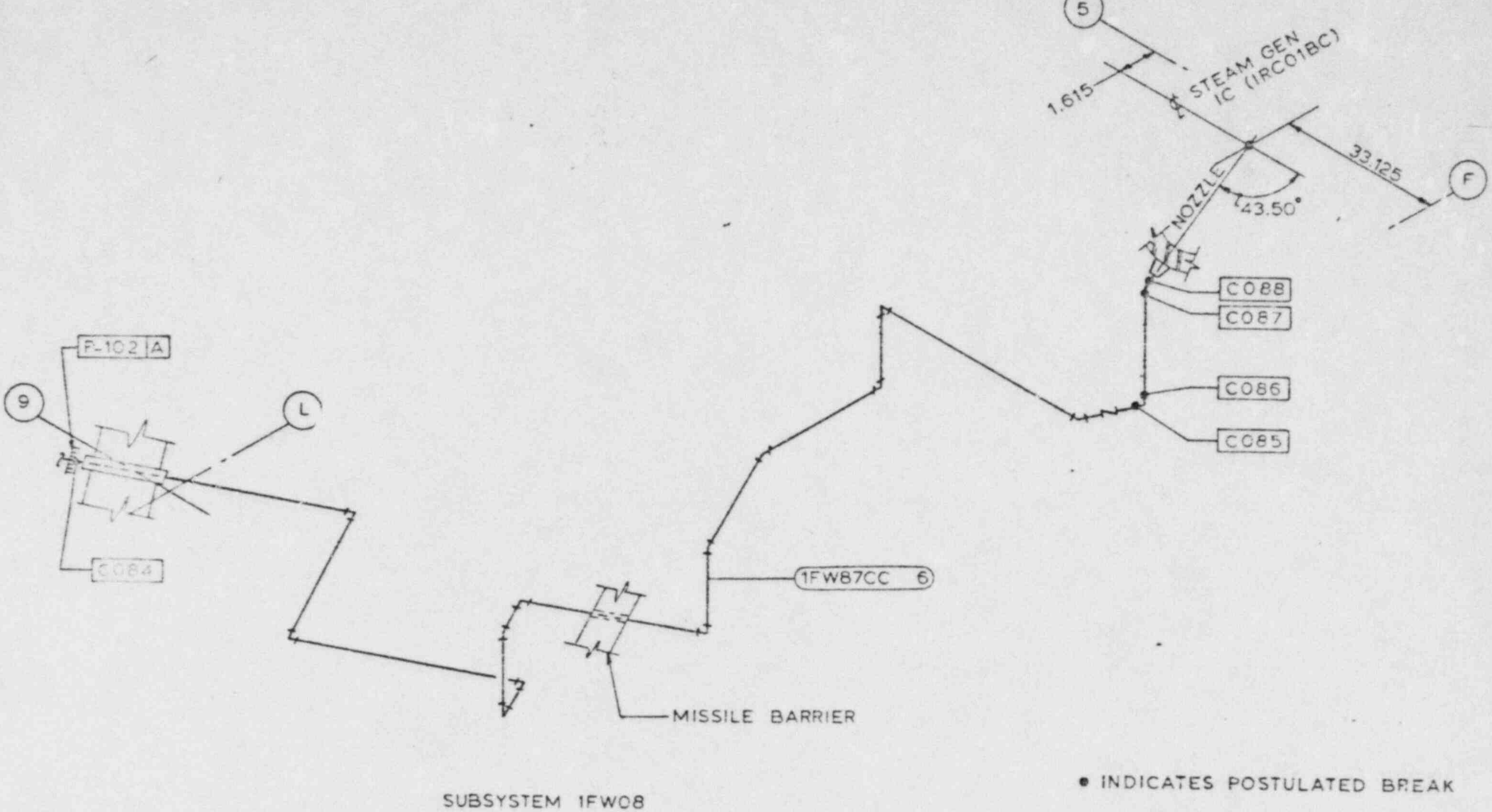
FEEDWATER - AUXILIARY FEEDWATER
LOOP 1 POSTULATED
BREAK LOCATIONS



B/B

FIGURE 3.6-52

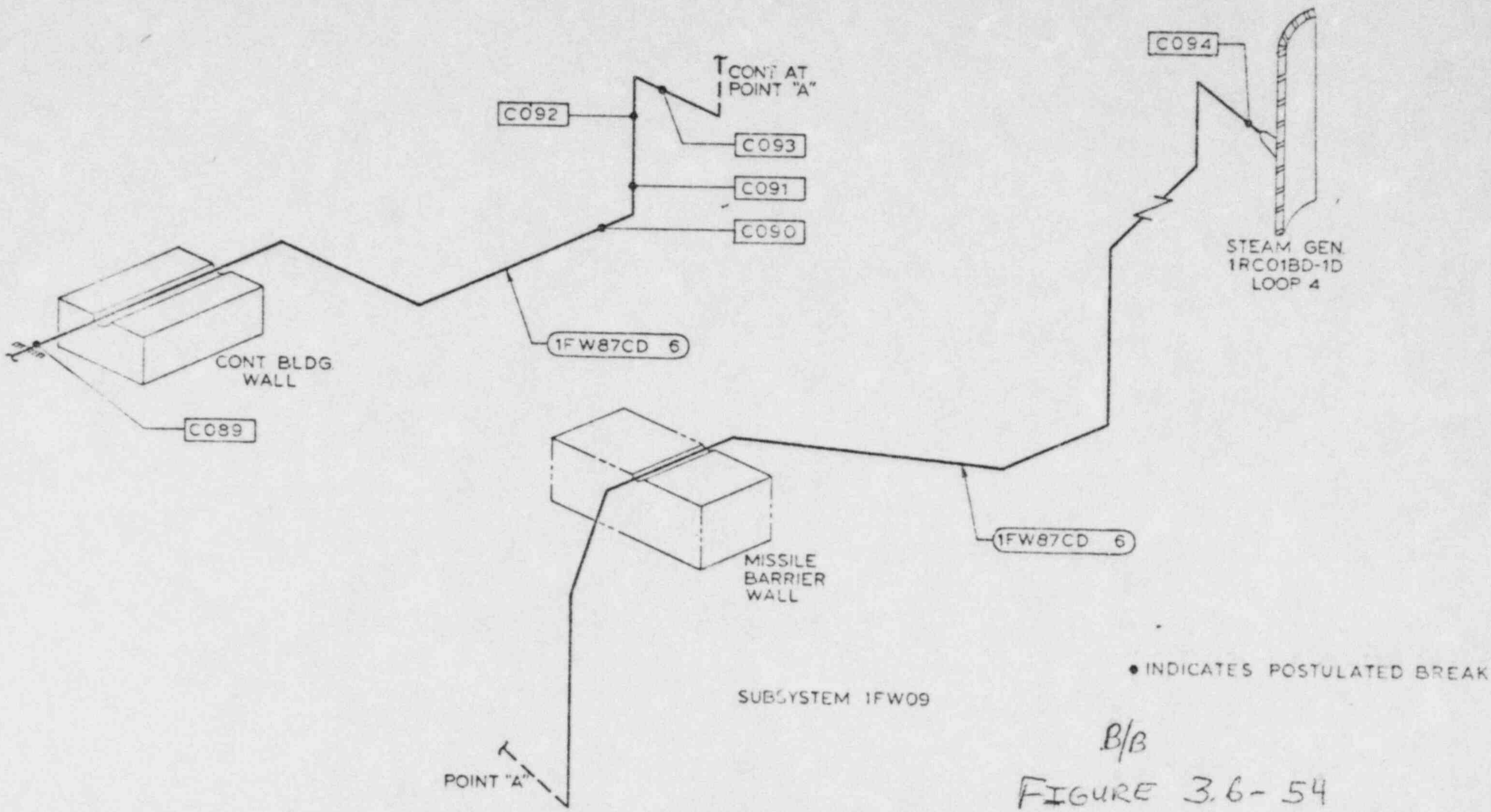
FEEDWATER - AUXILIARY FEEDWATER LOOP 2 & POSTULATED BREAK LOCATIONS



B/B

FIGURE 3.6-53

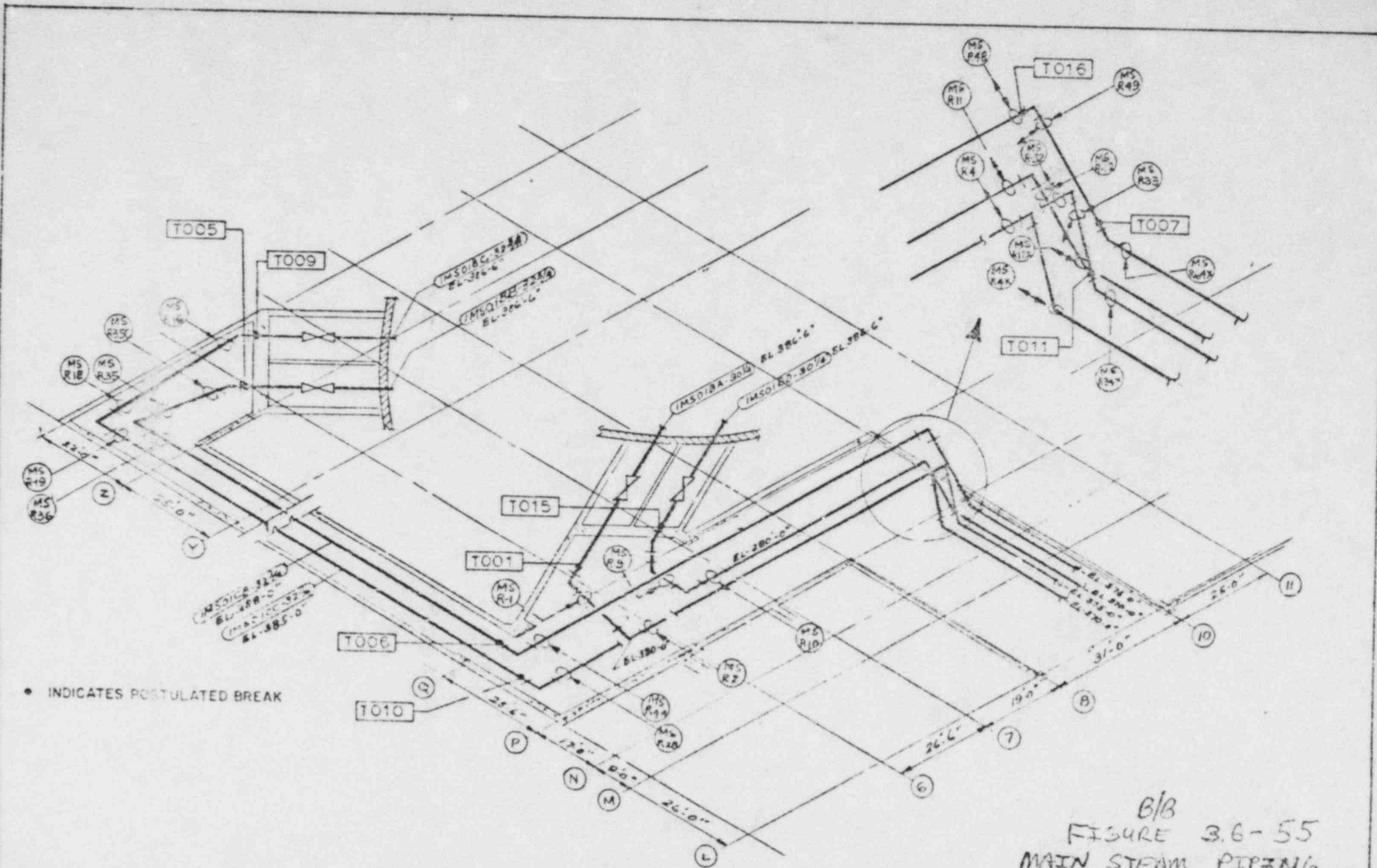
FEEDWATER - AUXILIARY FEEDWATER LOOP 3 POSTULATED BREAK LOCATIONS



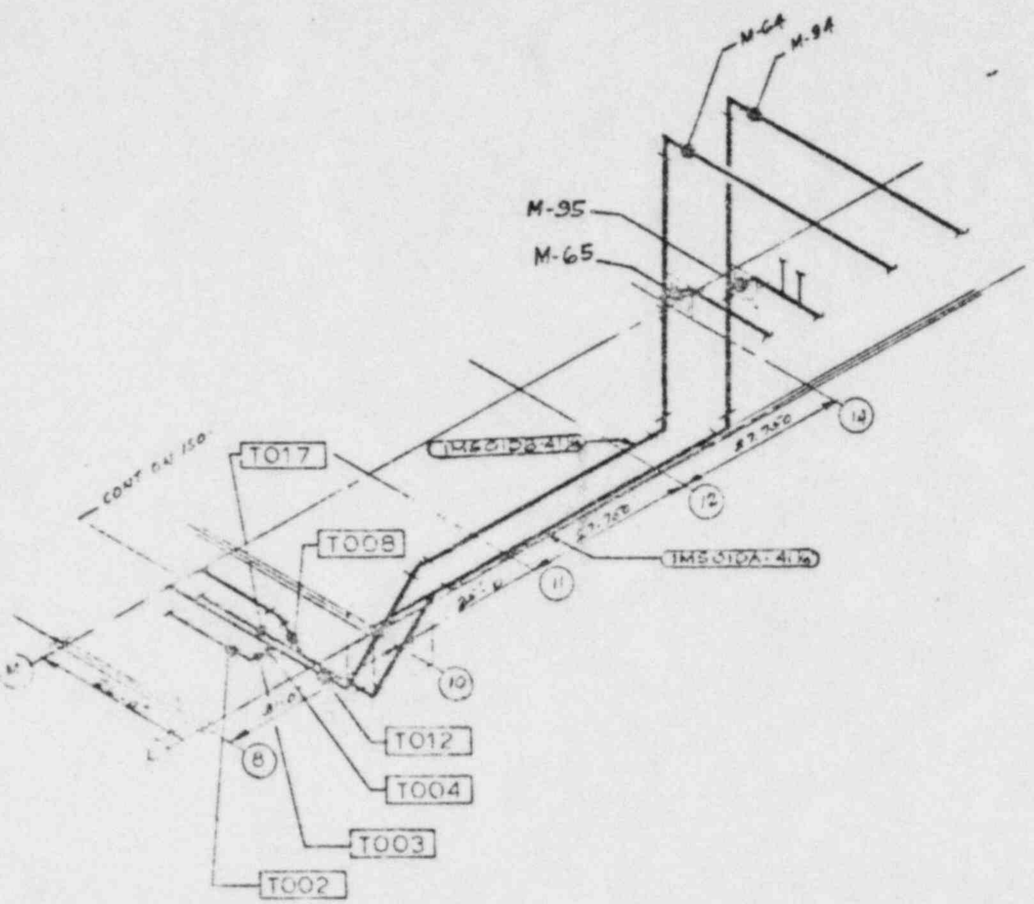
B/B

FIGURE 3.6-54

FEEDWATER - AUXILIARY FEEDWATER
LOOP 4 POSTULATED
BREAK LOCATIONS



B/B
FIGURE 3.6-55
MAIN STEAM PIPING
SYSTEMS IN MAIN
STEAM TUNNEL
POSTULATED BREAK
LOCATIONS
(Sheet 1 of 2)



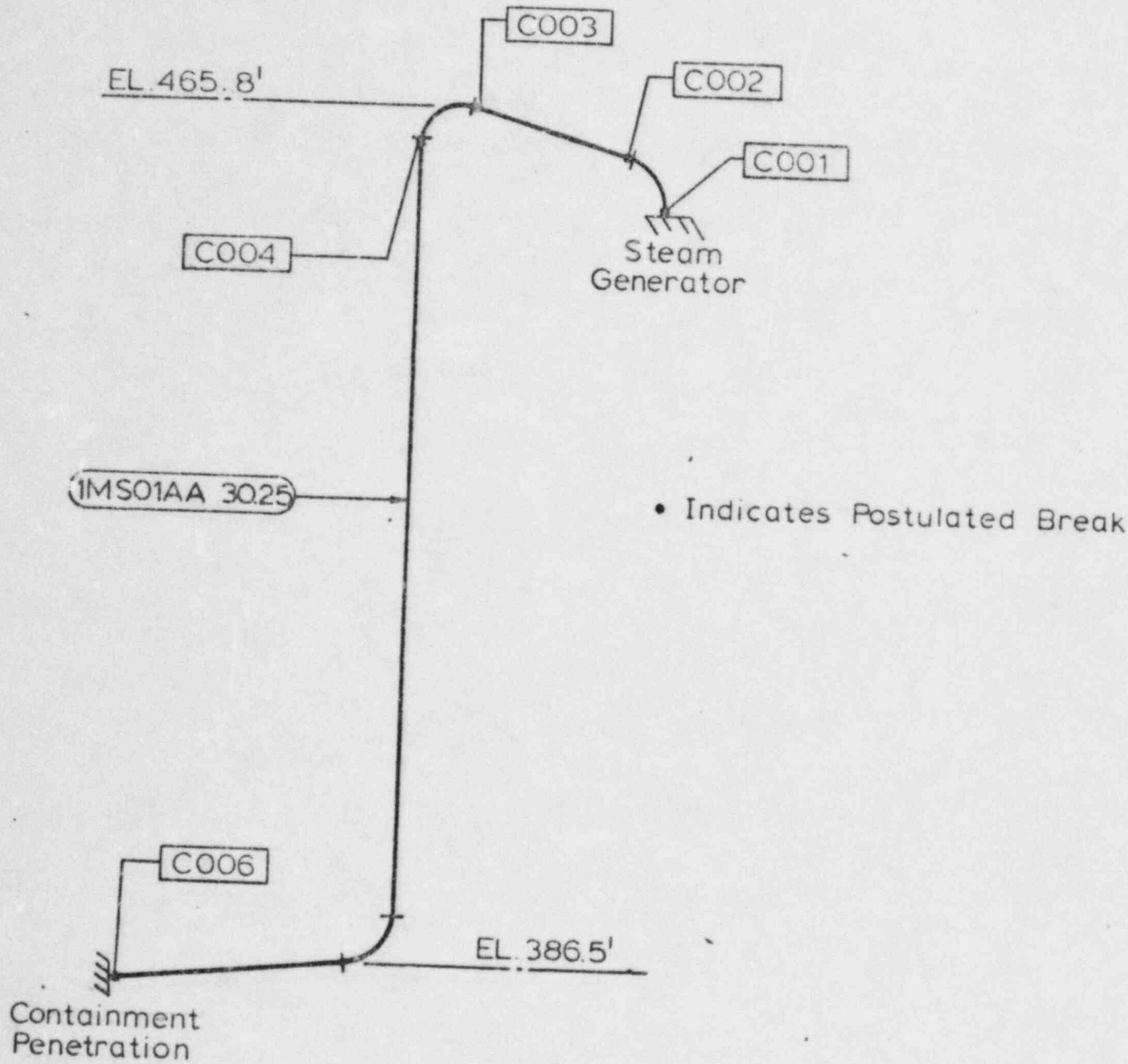
NOTE:

1. FEEDWATER LINES ARE PROTECTED AGAINST THE FULL EFFECTS OF POSTULATED MAIN STEAM PIPE RUPTURES.
2. WHERE THE PIPING IS UNRESTRAINED, ADJACENT STRUCTURES ARE DESIGNED TO PROVIDE PROTECTION AGAINST THE FULL EFFECTS OF THE POSTULATED PIPE RUPTURES.

● INDICATES POSTULATED BREAK

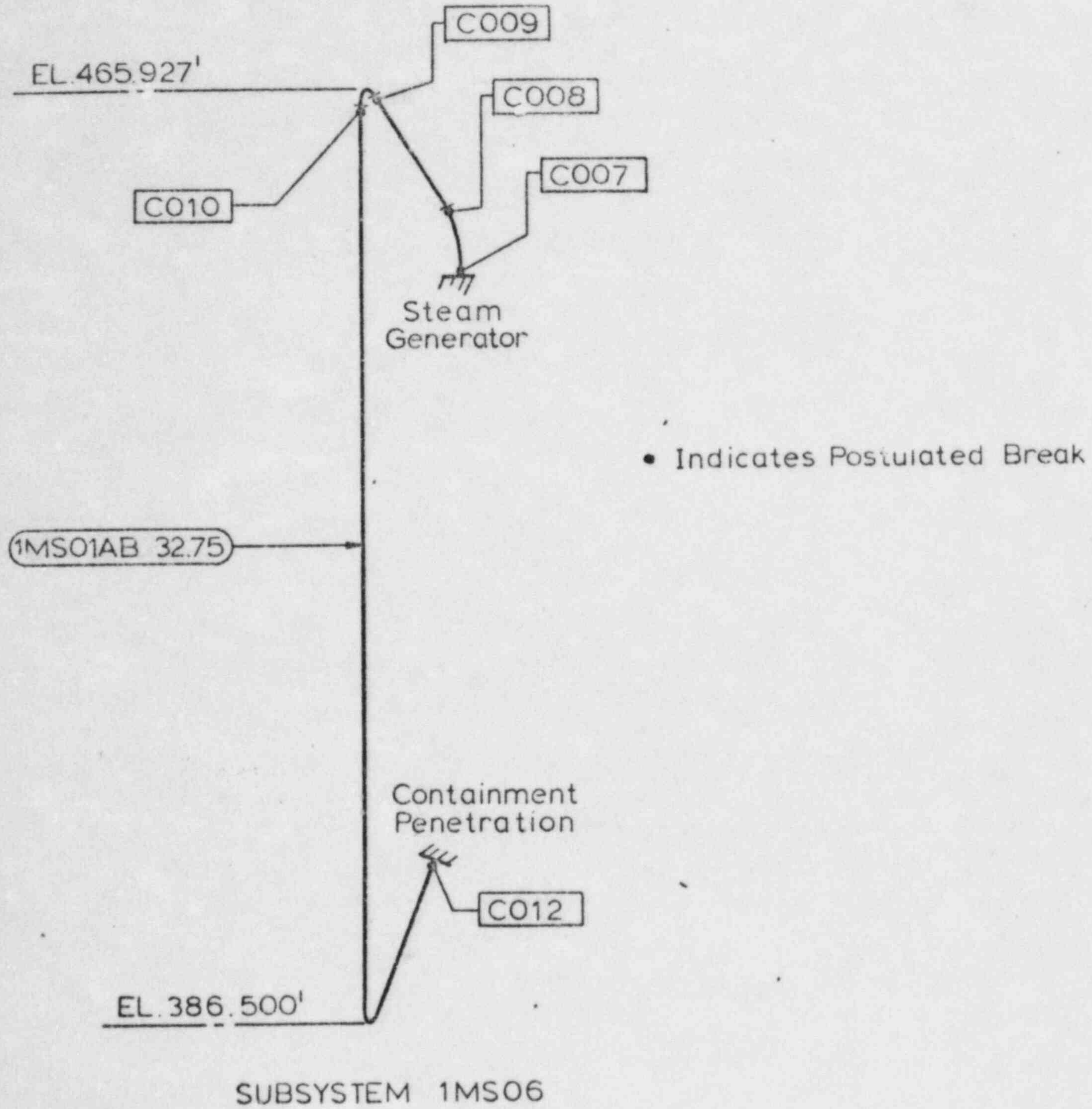
B/B

FIGURE 3.6-55
MAIN STEAM PIPING SYSTEMS IN MAIN STEAM TUNNEL POSTULATED BREAK LOCATIONS
(Sheet 2 of 2)



SUBSYSTEM 1MS05

B/B
FIGURE 3.6-56
MAIN STEAM LOOP 1
POSTULATED BREAK
LOCATIONS

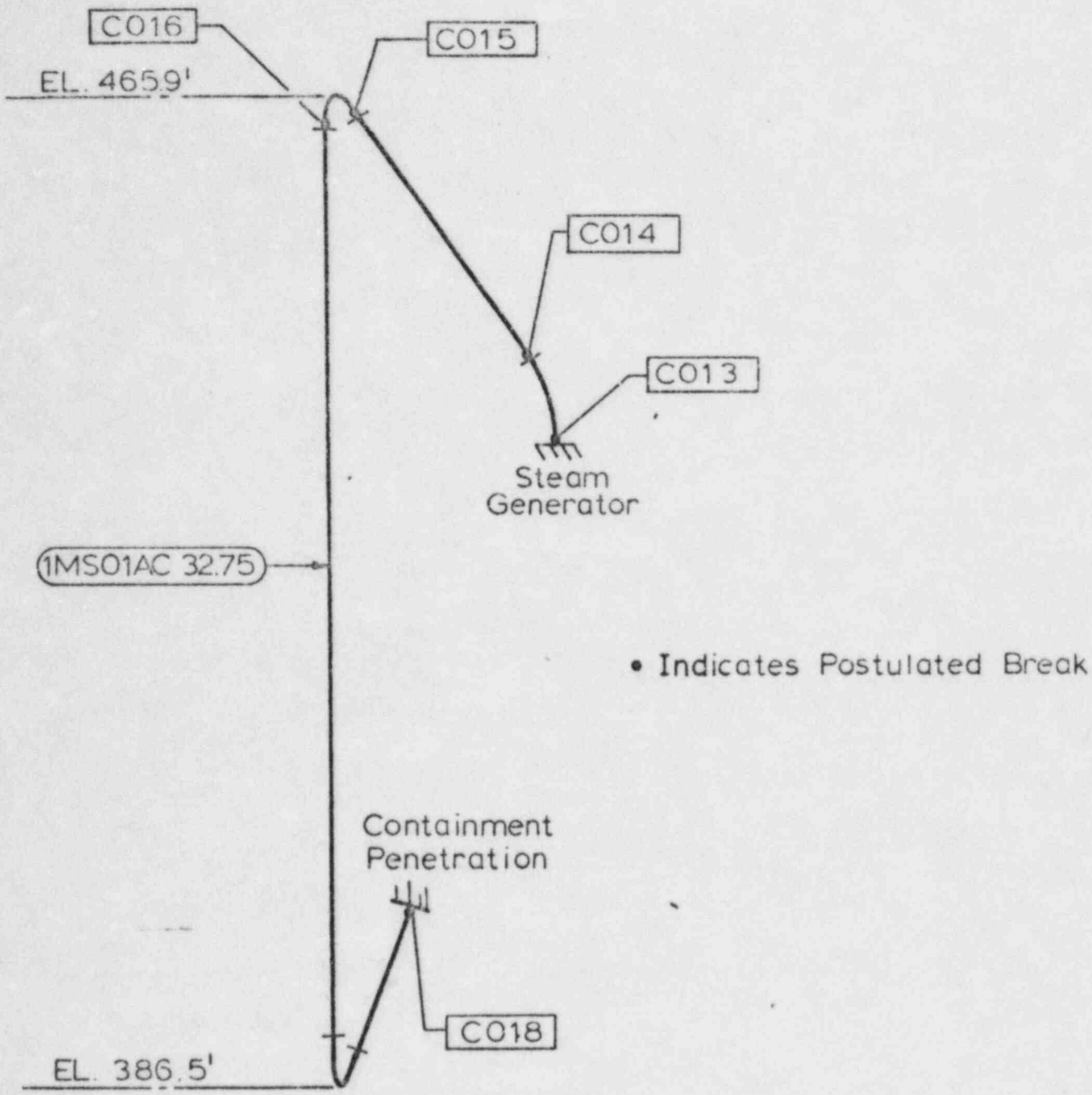


B/B

FIGURE 3.6-57

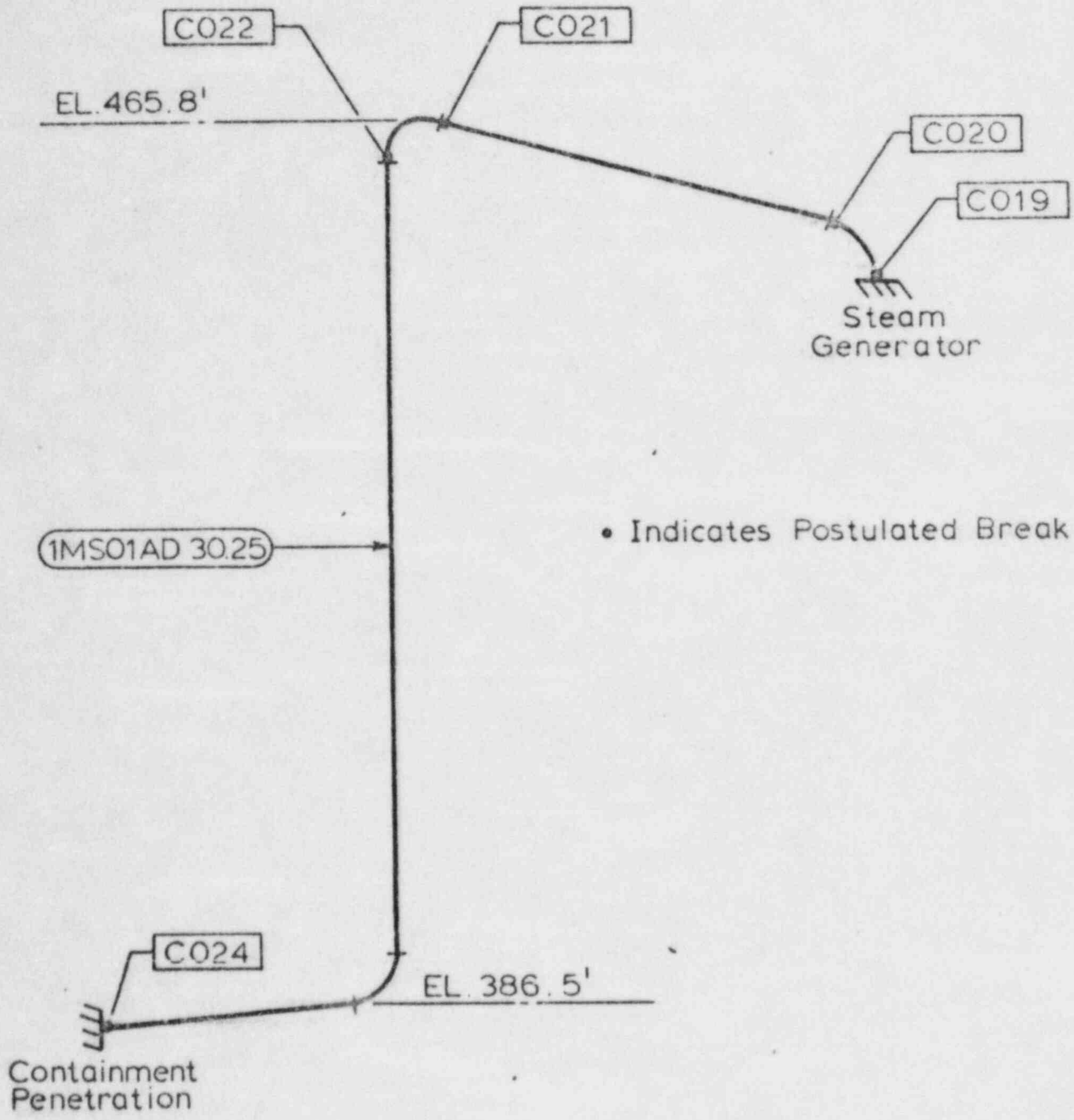
MAIN STEAM LOOP 2

POSTULATED BREAK LOCATIONS



SUBSYSTEM 1MS07

B/B
FIGURE 3.6-58
MAIN STEAM LOOP 3
POSTULATED BREAK LOCATIONS

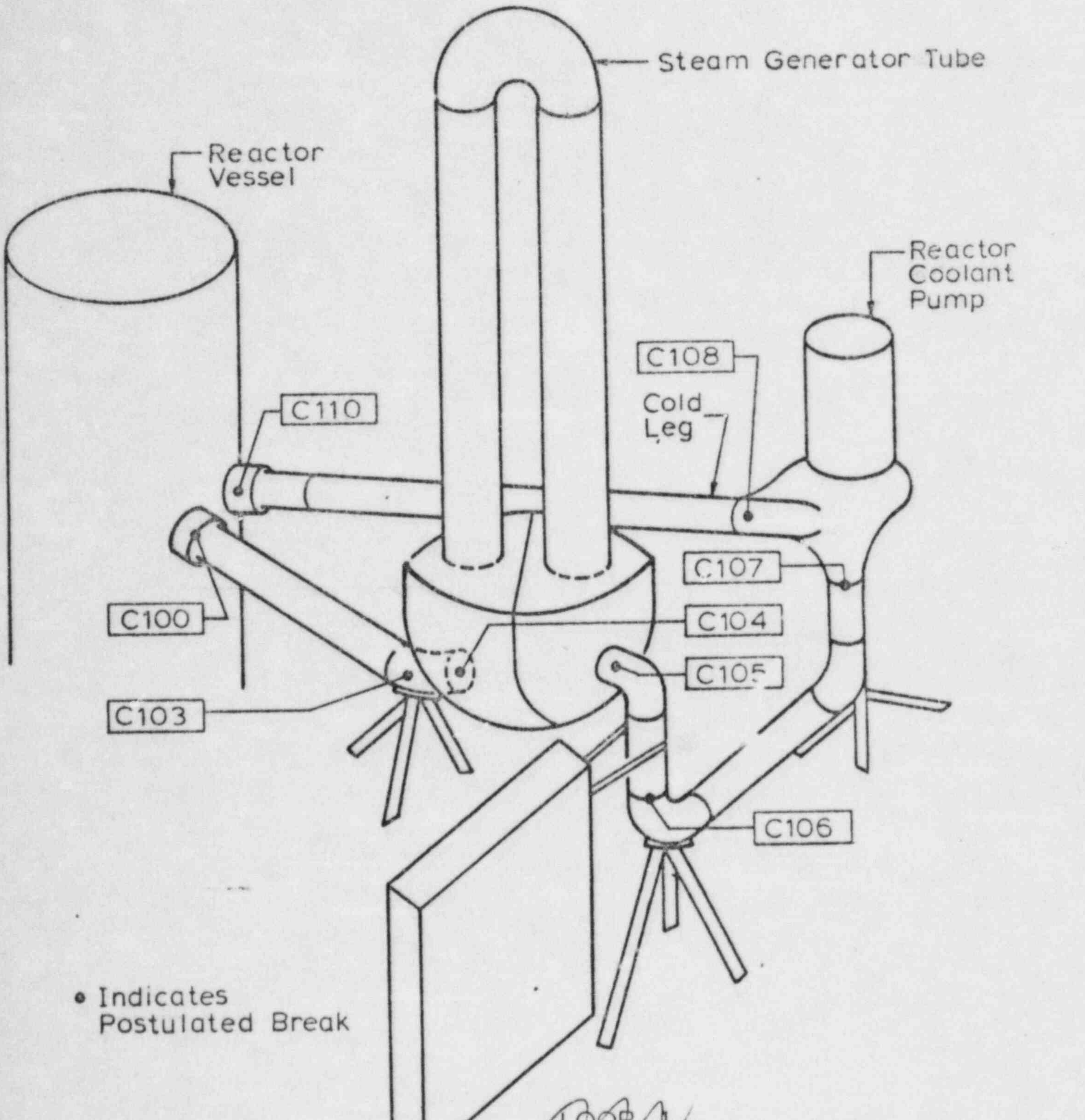


SUBSYSTEM 1MS08

B/B

FIGURE 3.6-54

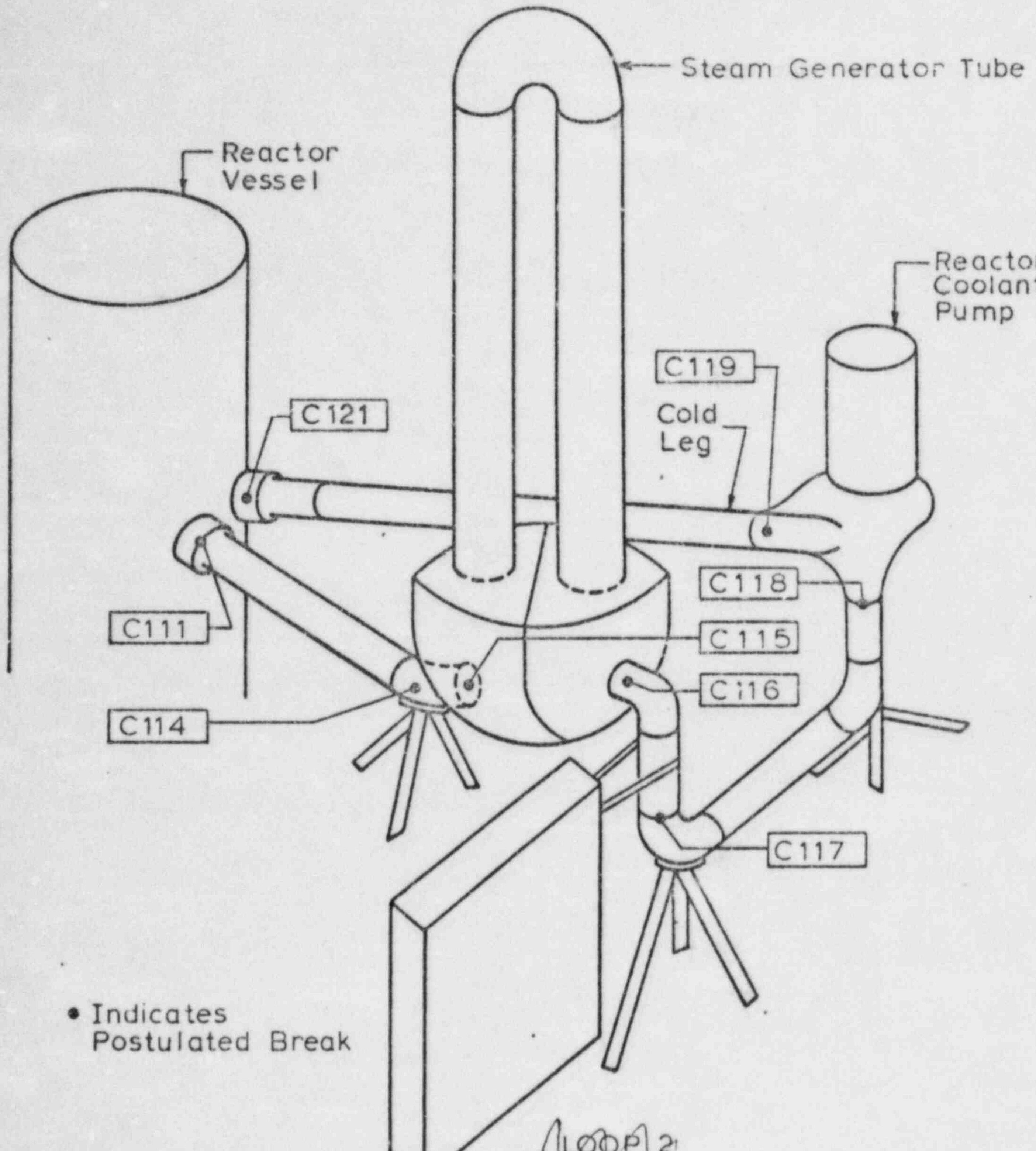
MAIN STEAM LOOP 4
POSTULATED BREAK
LOCATIONS



B/B

FIGURE 3.6-60

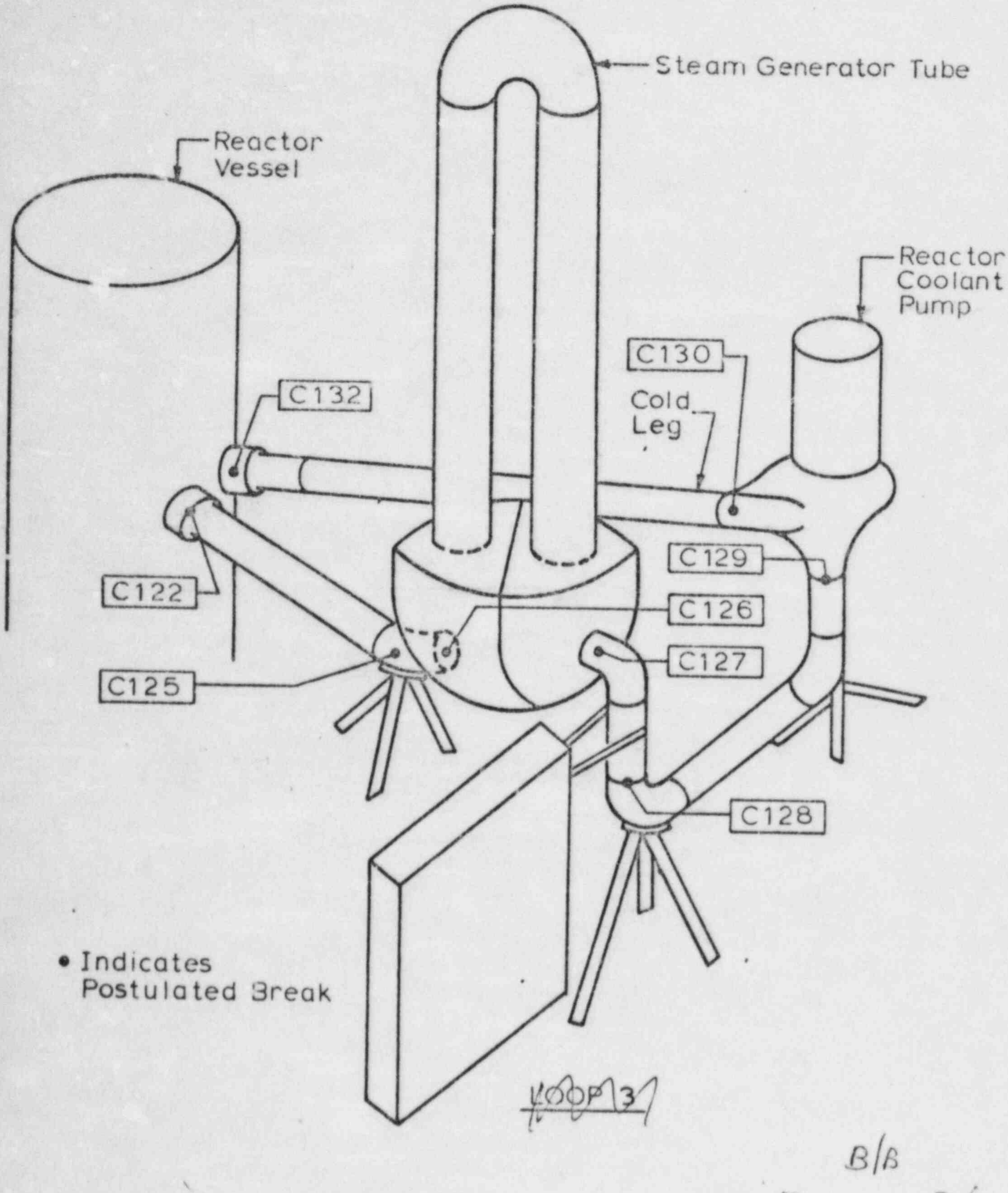
REACTOR COOLANT LOOP 1 POSTULATED BREAK LOCATIONS



B/B

FIGURE 3.6-61

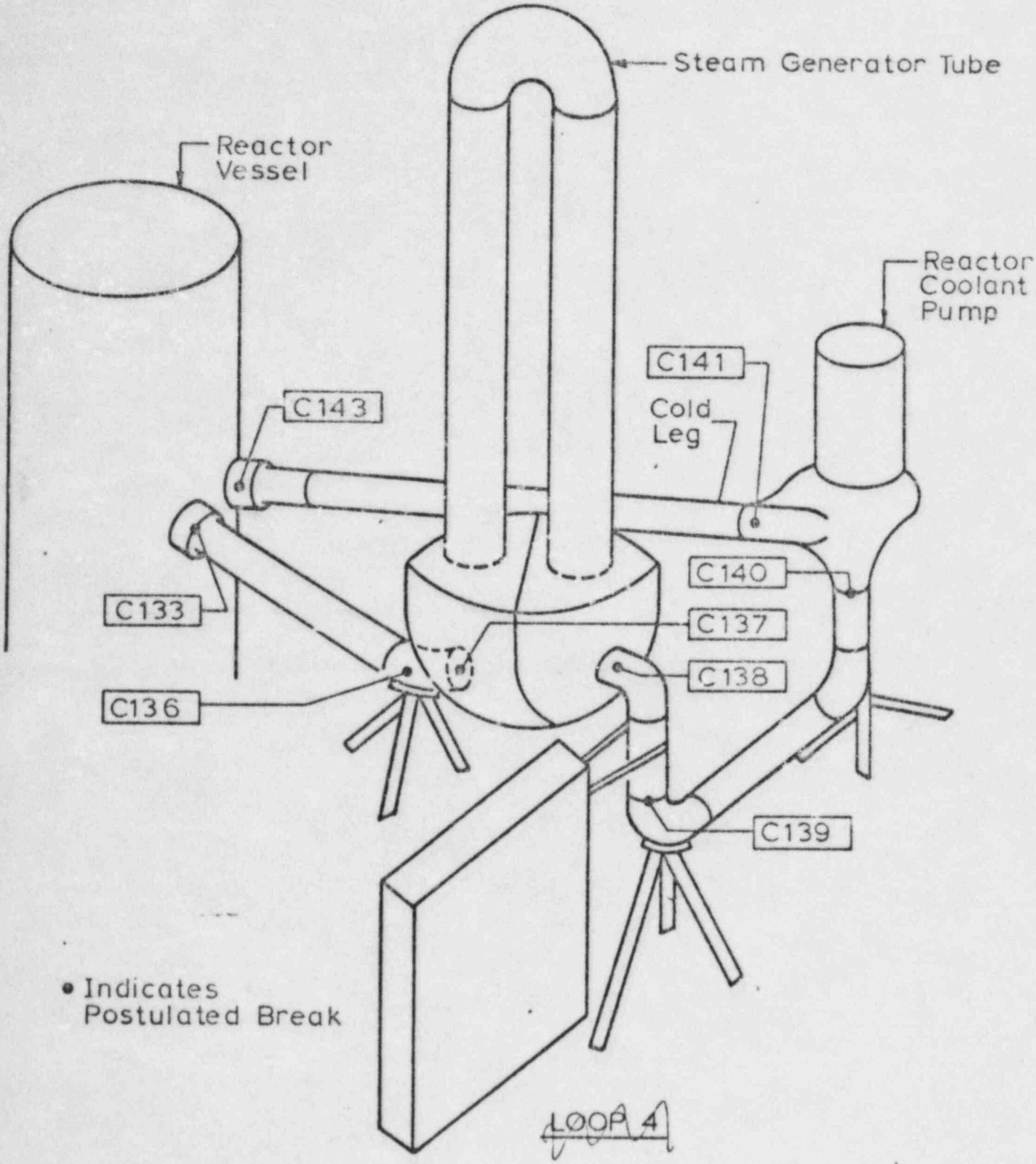
REACTOR COOLANT LOOP 2 POSTULATED BREAK LOCATIONS



B/B

FIGURE 3.6-62

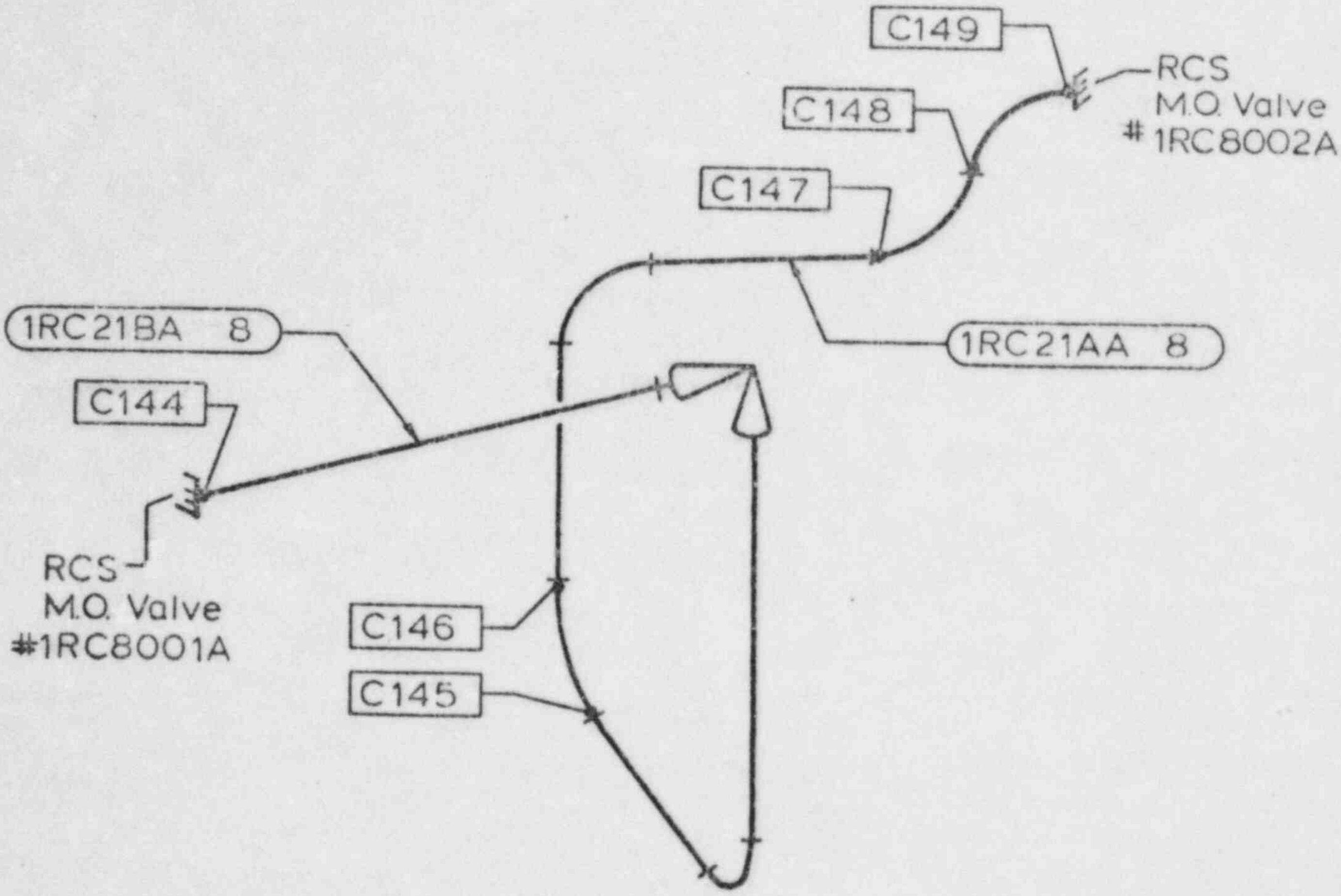
REACTOR COOLANT LOOP 3 POSTULATED BREAK LOCATIONS



B/B

FIGURE 3.6-63

REACTOR COOLANT
LOOP 4
POSTULATED BREAK LOCATIONS



SUBSYSTEM 1RC01

• Indicates Postulated Break.

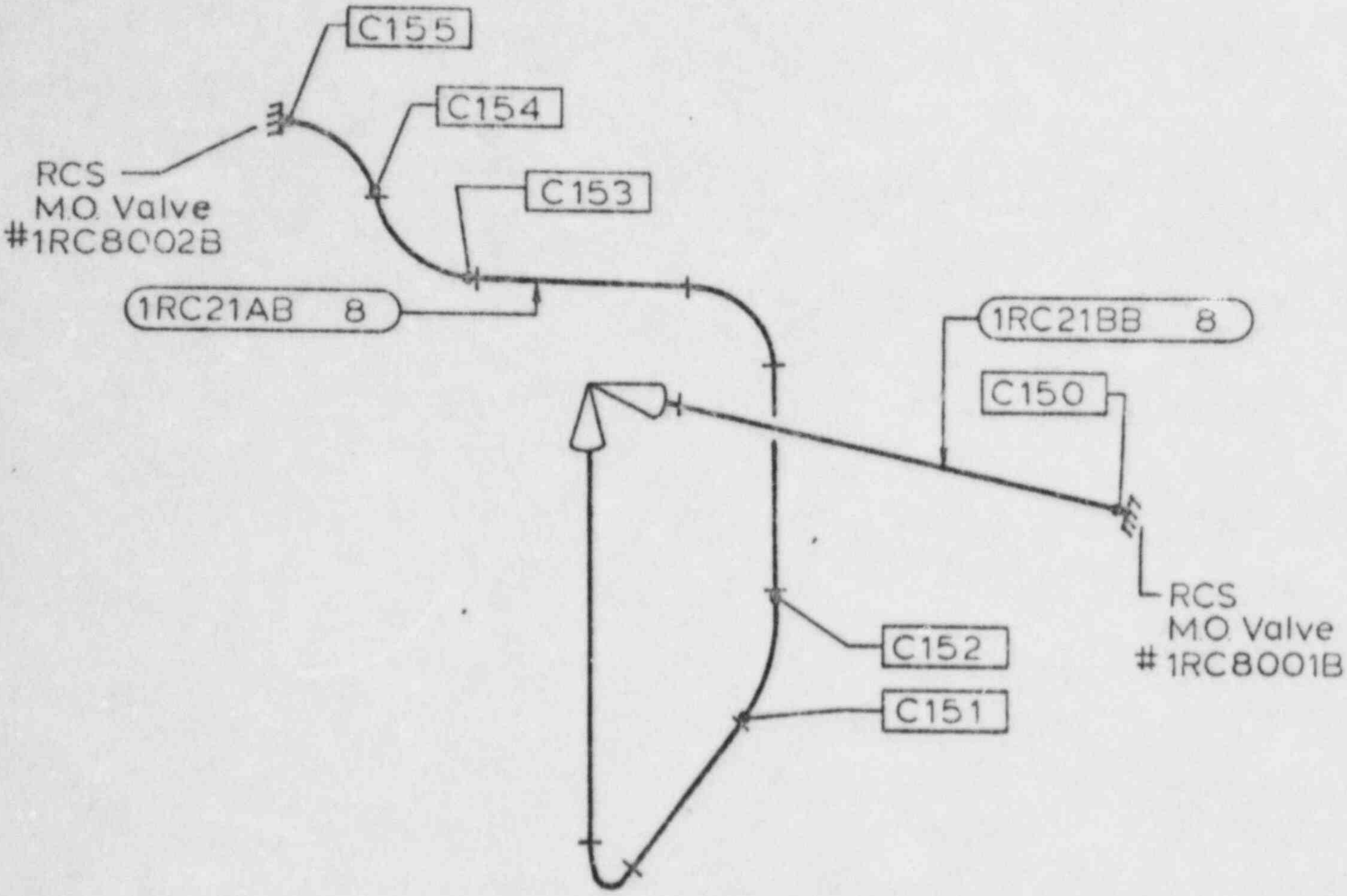
B/B

FIGURE 3.6-64

REACTOR COOLANT BYPASS

LOOP 1

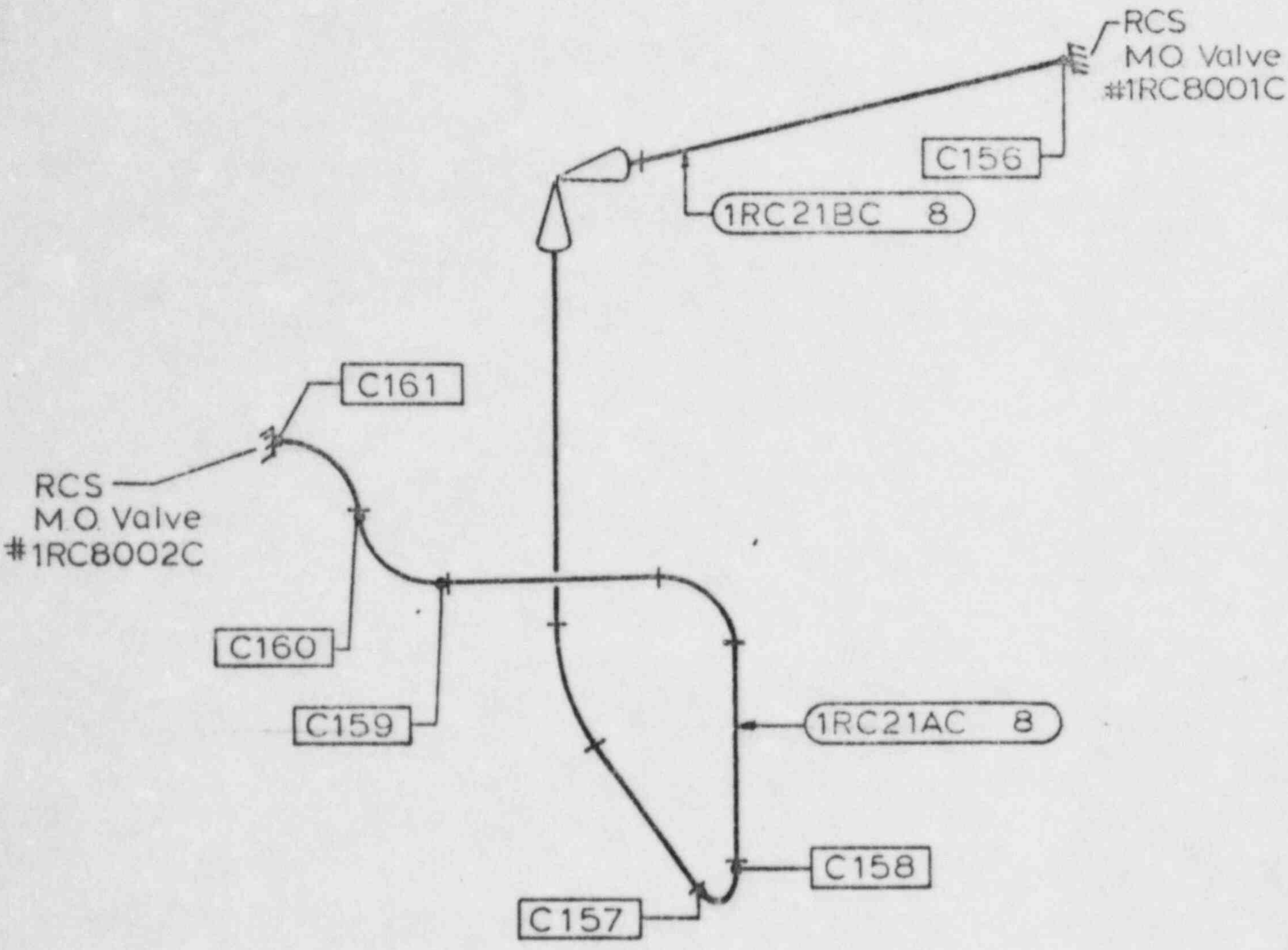
POSTULATED BREAK LOCATIONS



SUBSYSTEM 1RC02

• Indicates Postulated Break

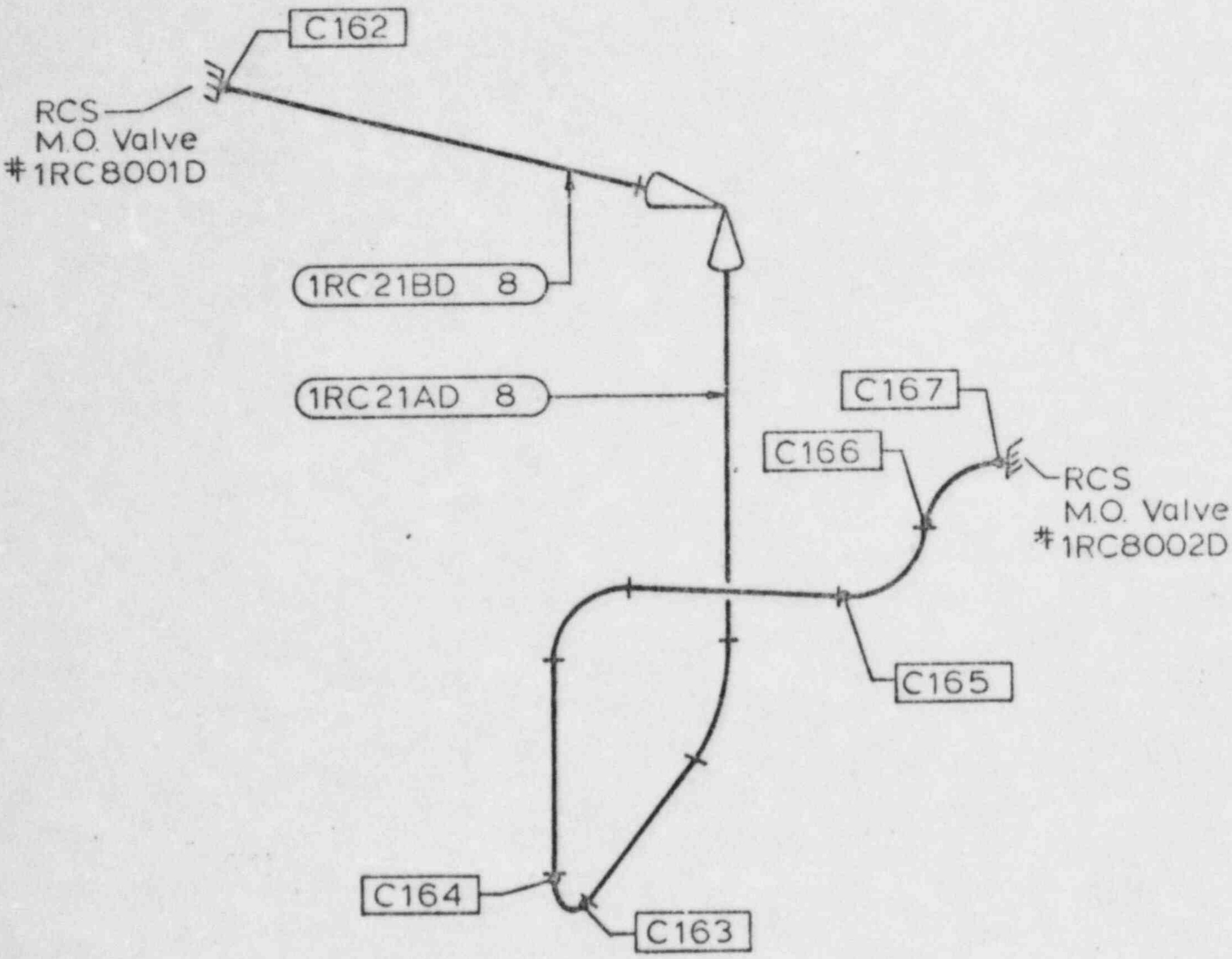
B/B
FIGURE 3.6-65
REACTOR COOLANT BYPASS
LOOP 2
POSTULATED BREAK LOCATIONS



SUBSYSTEM 1RC03

• Indicate Postulated Break

B/B
FIGURE 3.6-66
REACTOR COOLANT BYPASS
LOOP 3
POSTULATED BREAK LOCATIONS



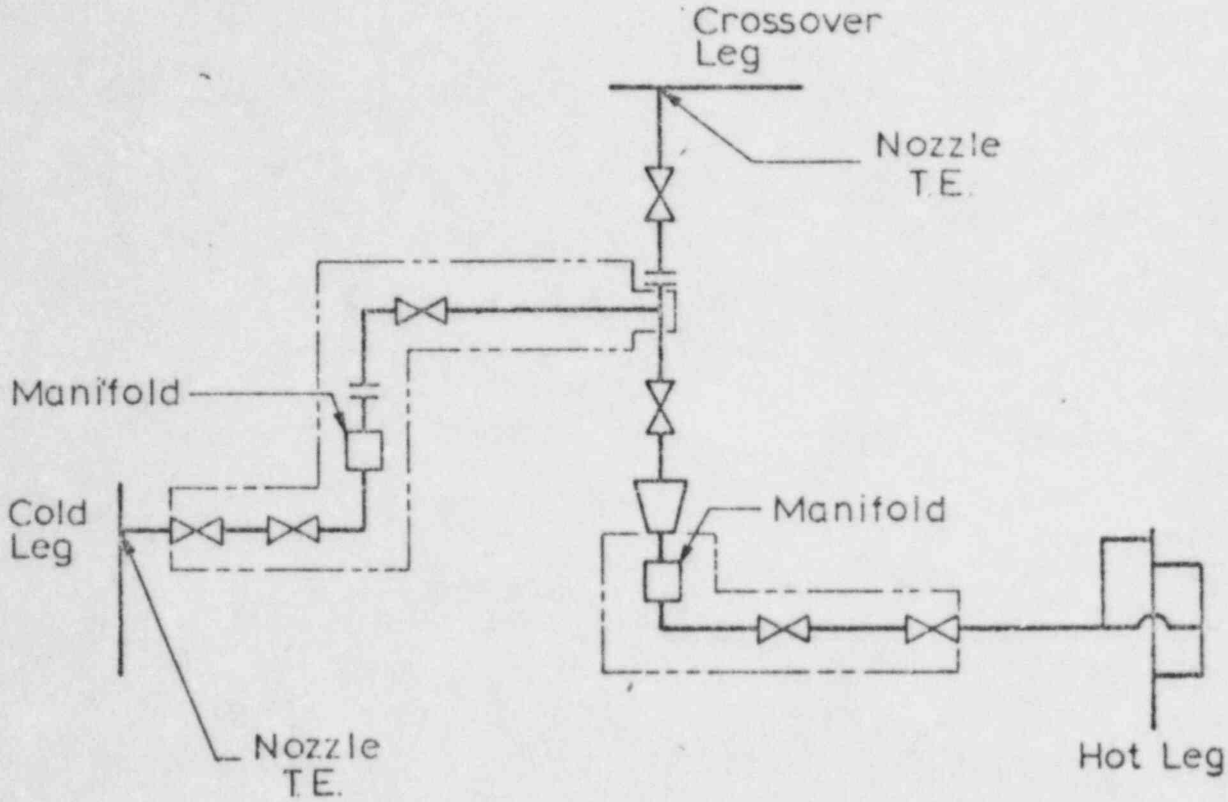
SUBSYSTEM 1RC04

• Indicates Postulated Break

B/B

FIGURE 3.6-67

REACTOR COOLANT BYPASS
LOOP 4
POSTULATED BREAK LOCATIONS



SUBSYSTEM 1RC10

NOTE:

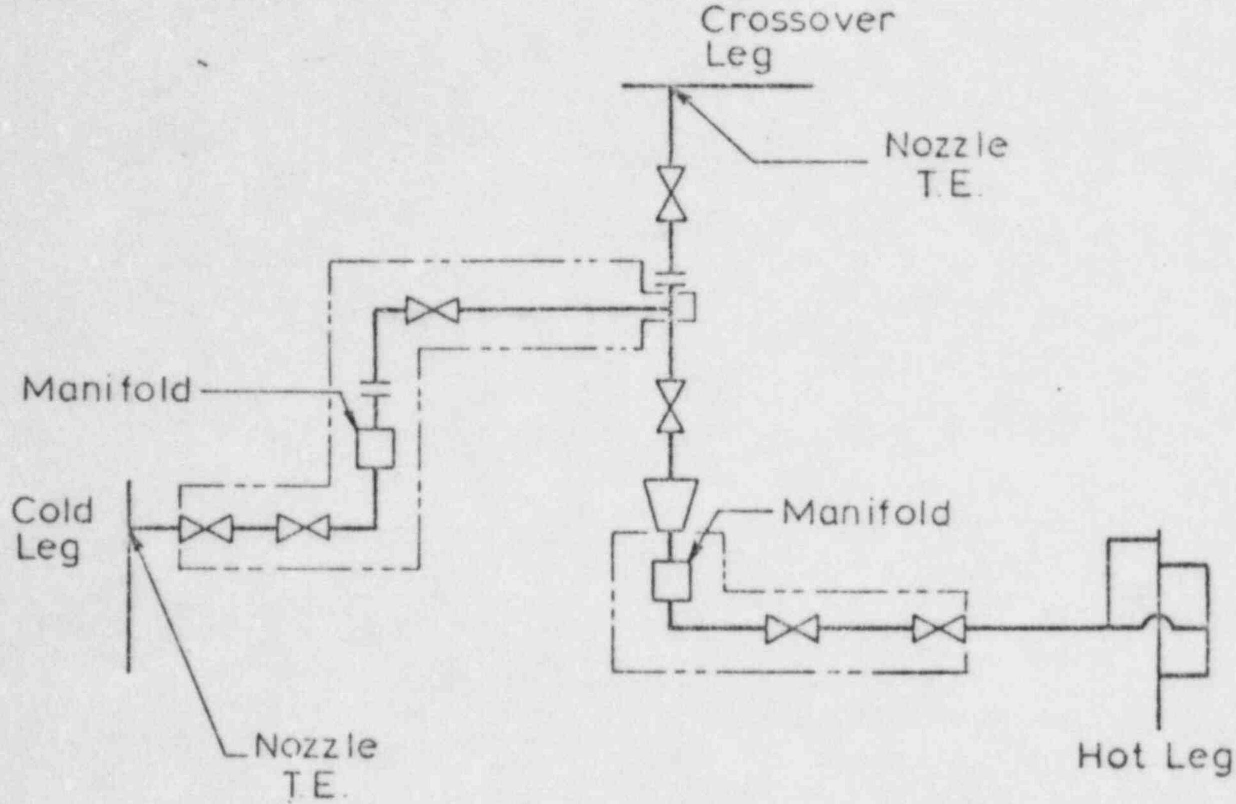
THIS IS A SCHEMATIC REPRESENTATION (NOT A TRUE ISOMETRIC SKETCH). BREAKS ARE POSTULATED AT ALL SOCKET WELDS WITHIN THE ENCLOSED AREAS.

B/B

FIGURE 3.6-68

REACTOR COOLANT SUBSYSTEM 1RC10

POSTULATED BREAK LOCATIONS



SUBSYSTEM 1RC11

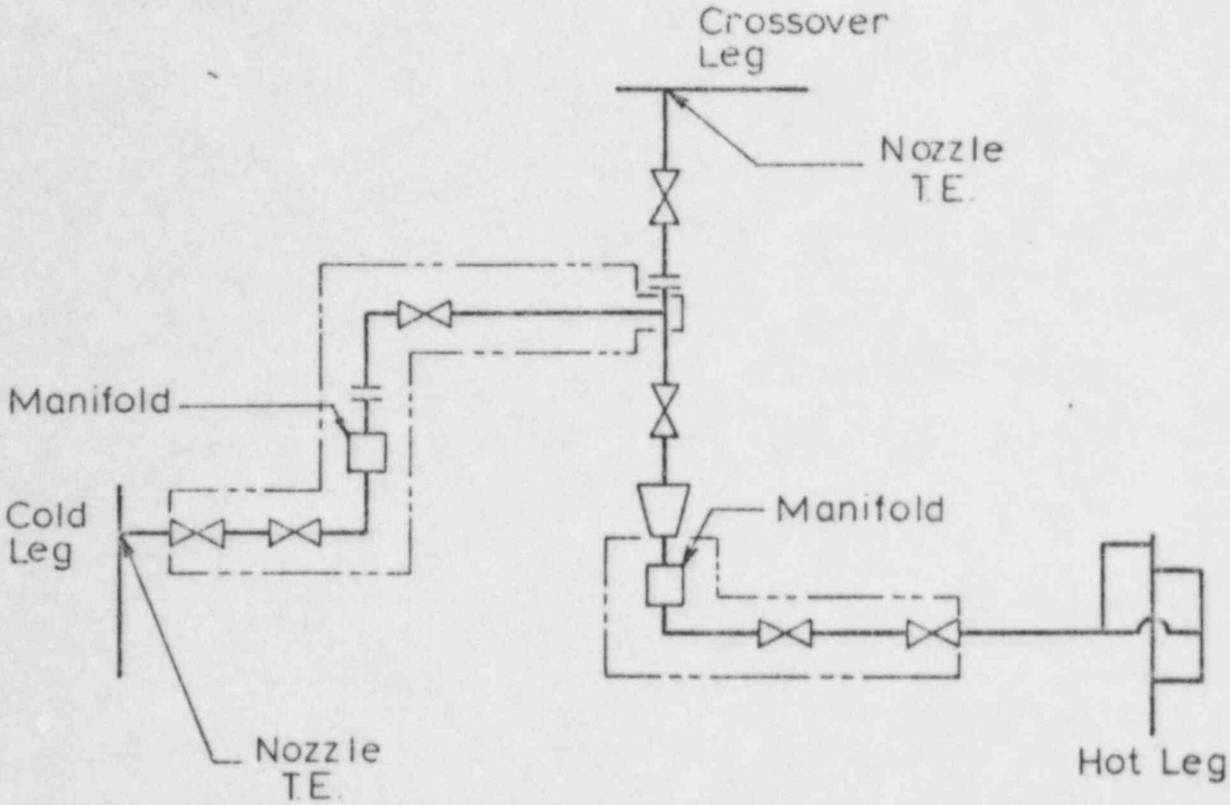
NOTE:

THIS IS A SCHEMATIC REPRESENTATION (NOT A TRUE ISOMETRIC SKETCH). BREAKS ARE POSTULATED AT ALL SOCKET WELDS WITHIN THE ENCLOSED AREAS.

B/B

FIGURE 3.6-69

REACTOR COOLANT SUBSYSTEM 1RC11 POSTULATED BREAK LOCATIONS



SUBSYSTEM 1RC12

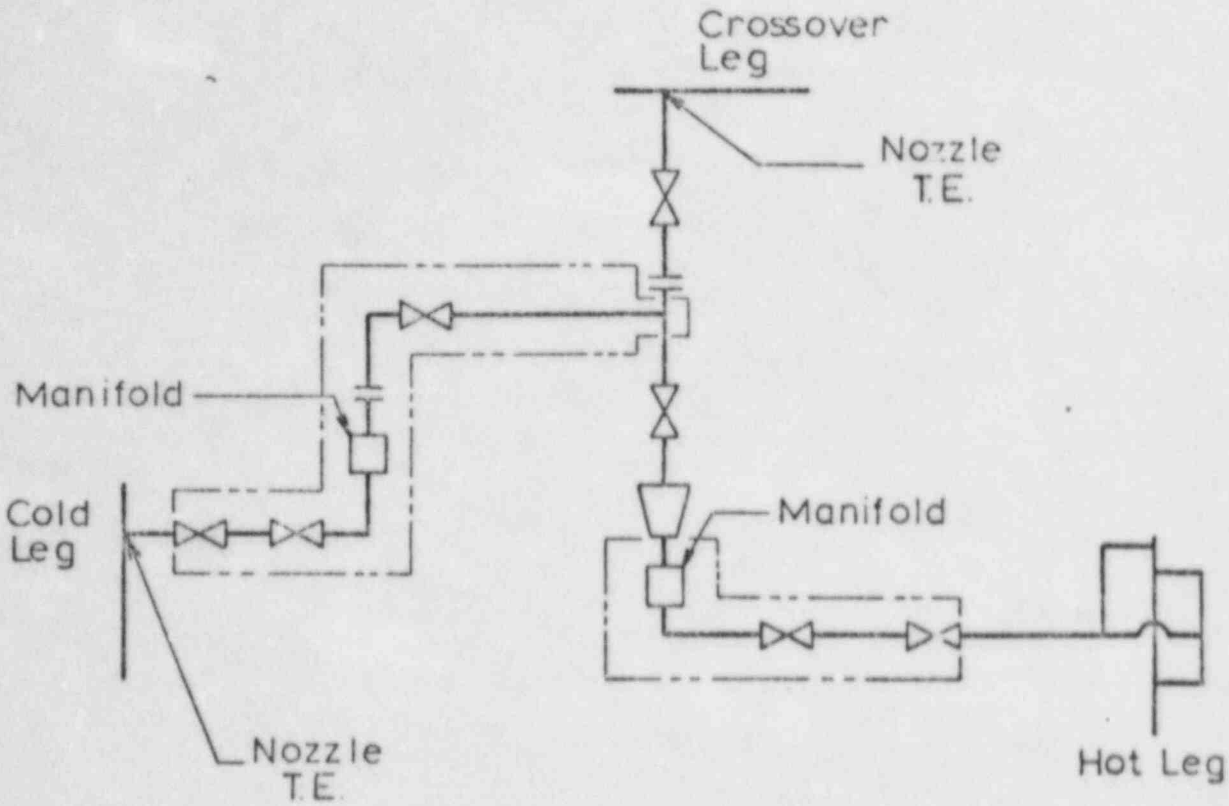
NOTE:

THIS IS A SCHEMATIC REPRESENTATION (NOT A TRUE ISOMETRIC SKETCH). BREAKS ARE POSTULATED AT ALL SOCKET WELDS WITHIN THE ENCLOSED AREAS

B/B

FIGURE 3.6-70

REACTOR COOLANT
SUBSYSTEM 1RC12
POSTULATED BREAK
LOCATIONS



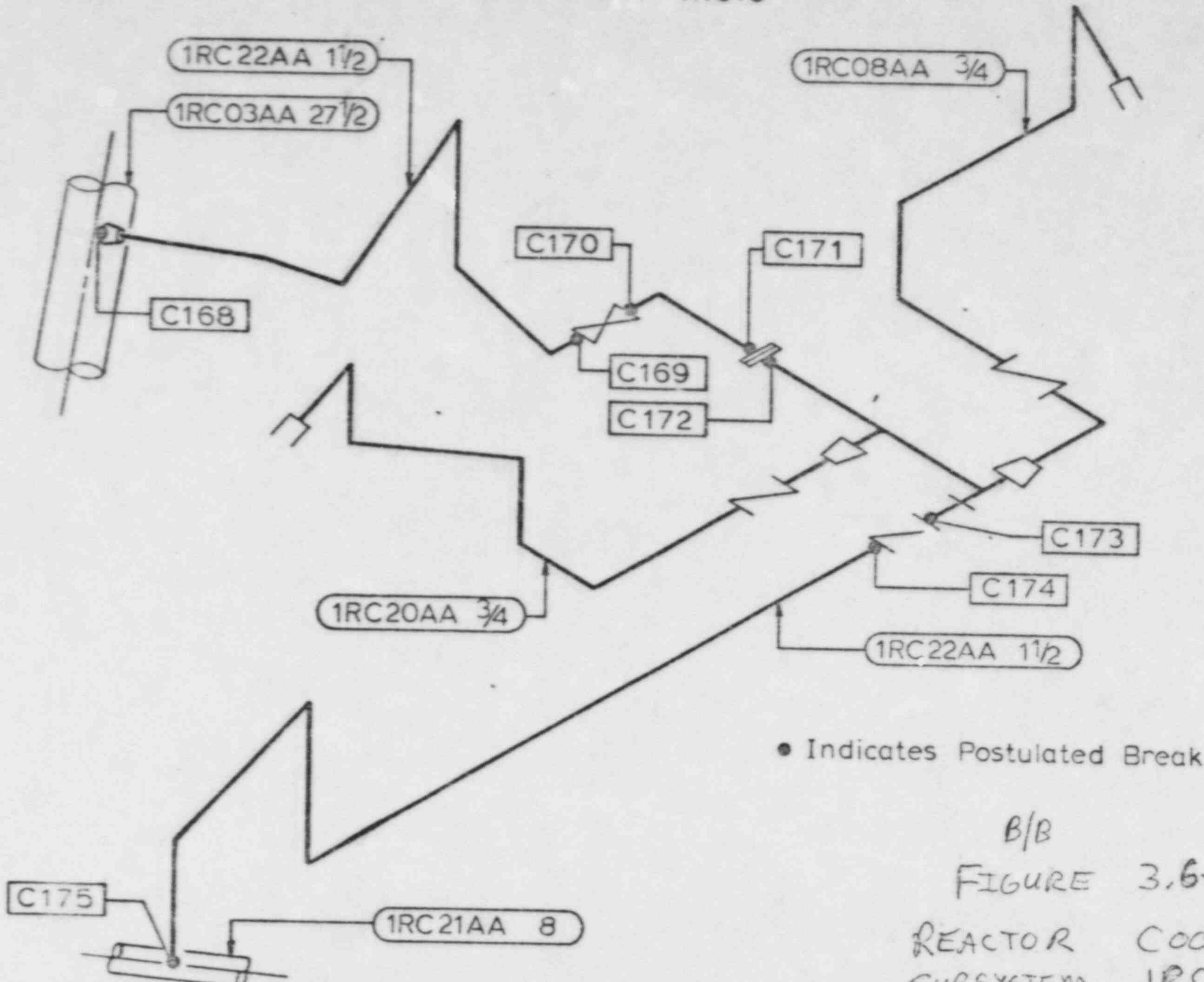
SUBSYSTEM 1RC13

NOTE :
THIS IS A SCHEMATIC REPRESENTATION (NOT A TRUE ISOMETRIC SKETCH). BREAKS ARE POSTULATED AT ALL SOCKET WELDS WITHIN THE ENCLOSED AREAS.

B/B

FIGURE 3.6-71
REACTOR COOLANT
SUBSYSTEM 1RC13
POSTULATED BREAK
LOCATIONS

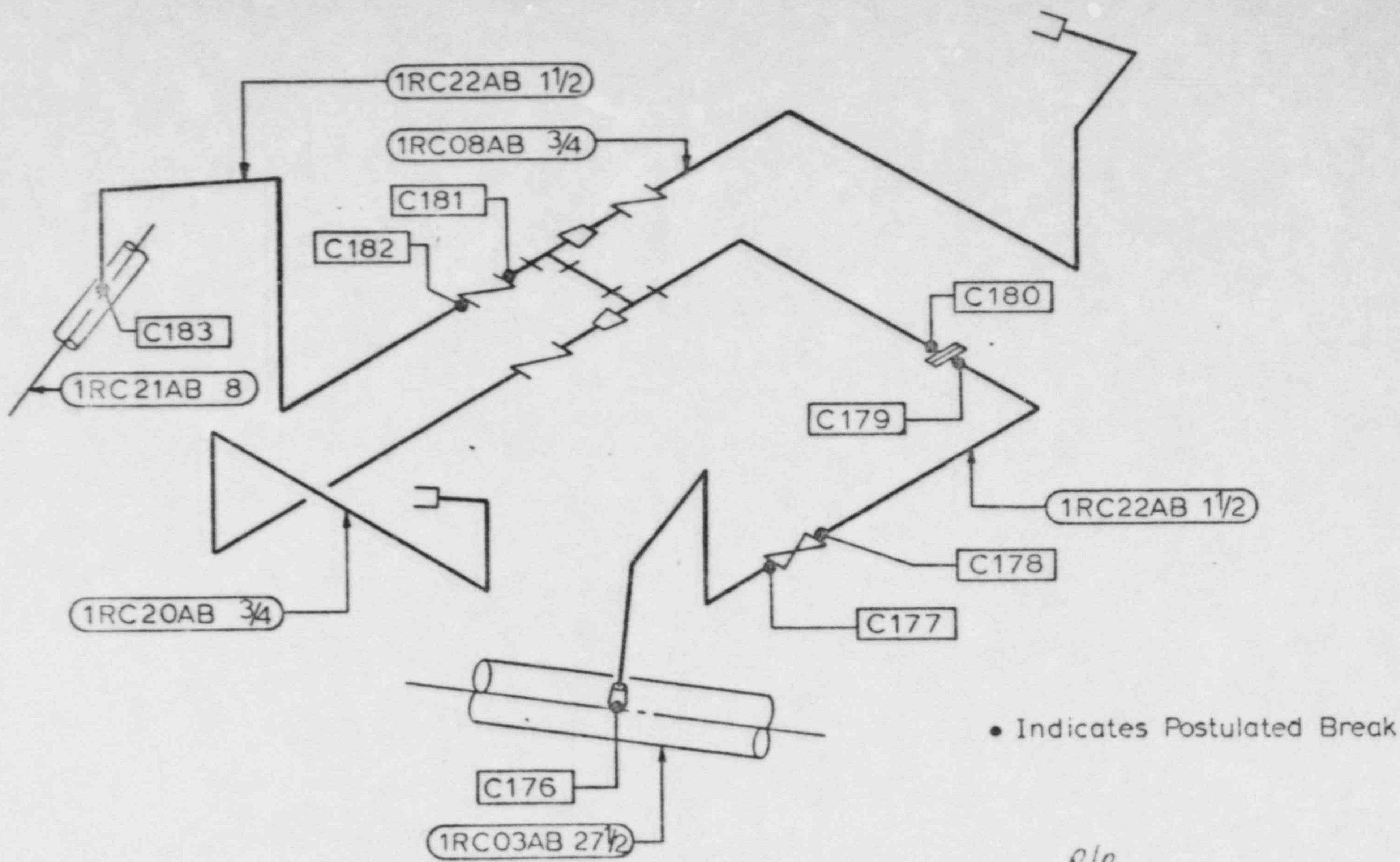
SUBSYSTEM 1RC16



B/B

FIGURE 3.6-72

REACTOR COOLANT SUBSYSTEM 1RC16 POSTULATED BREAK LOCATIONS



SUBSYSTEM 1RC17

B/B

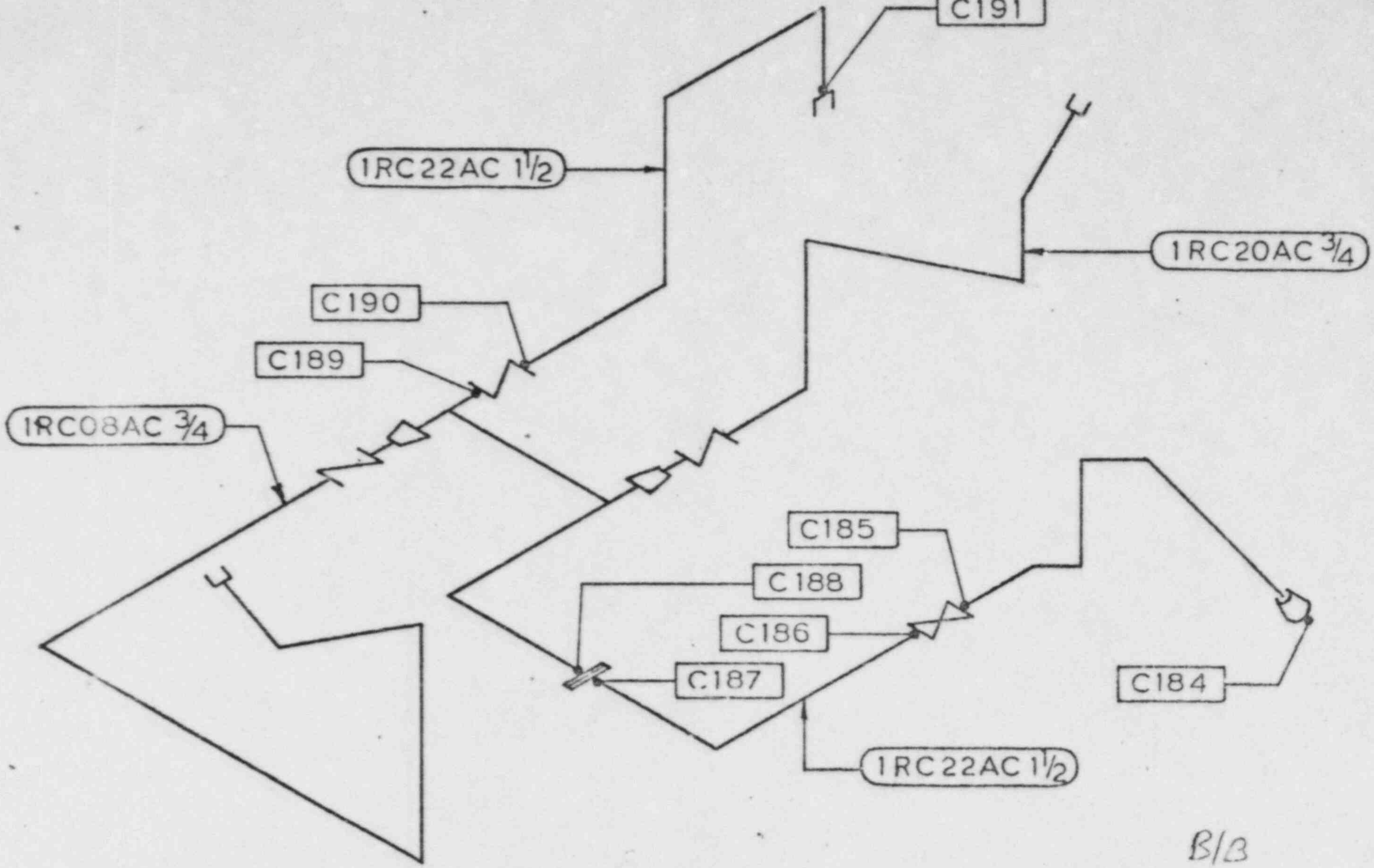
FIGURE 3.6-73

REACTOR COOLANT

SUBSYSTEM 1RC17

POSTULATED BREAK

LOCATIONS

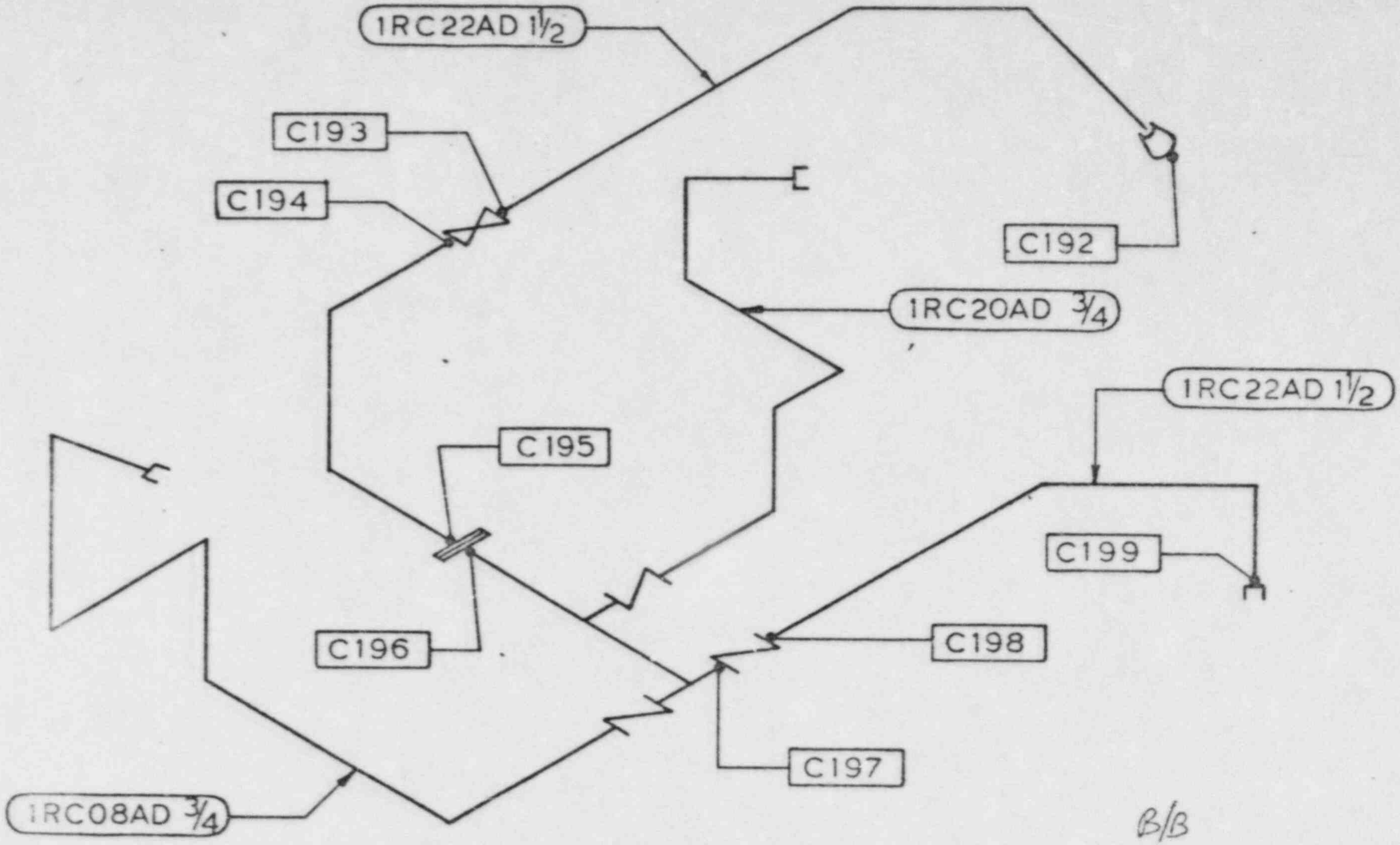


• Indicates Postulated Break

SUBSYSTEM 1RC18

B/B

FIGURE 3.6-74
REACTOR COOLANT
SUBSYSTEM 1RC18
POSTULATED BREAK
LOCATIONS



• Indicates Postulated Break

SUBSYSTEM 1RC19

B/B

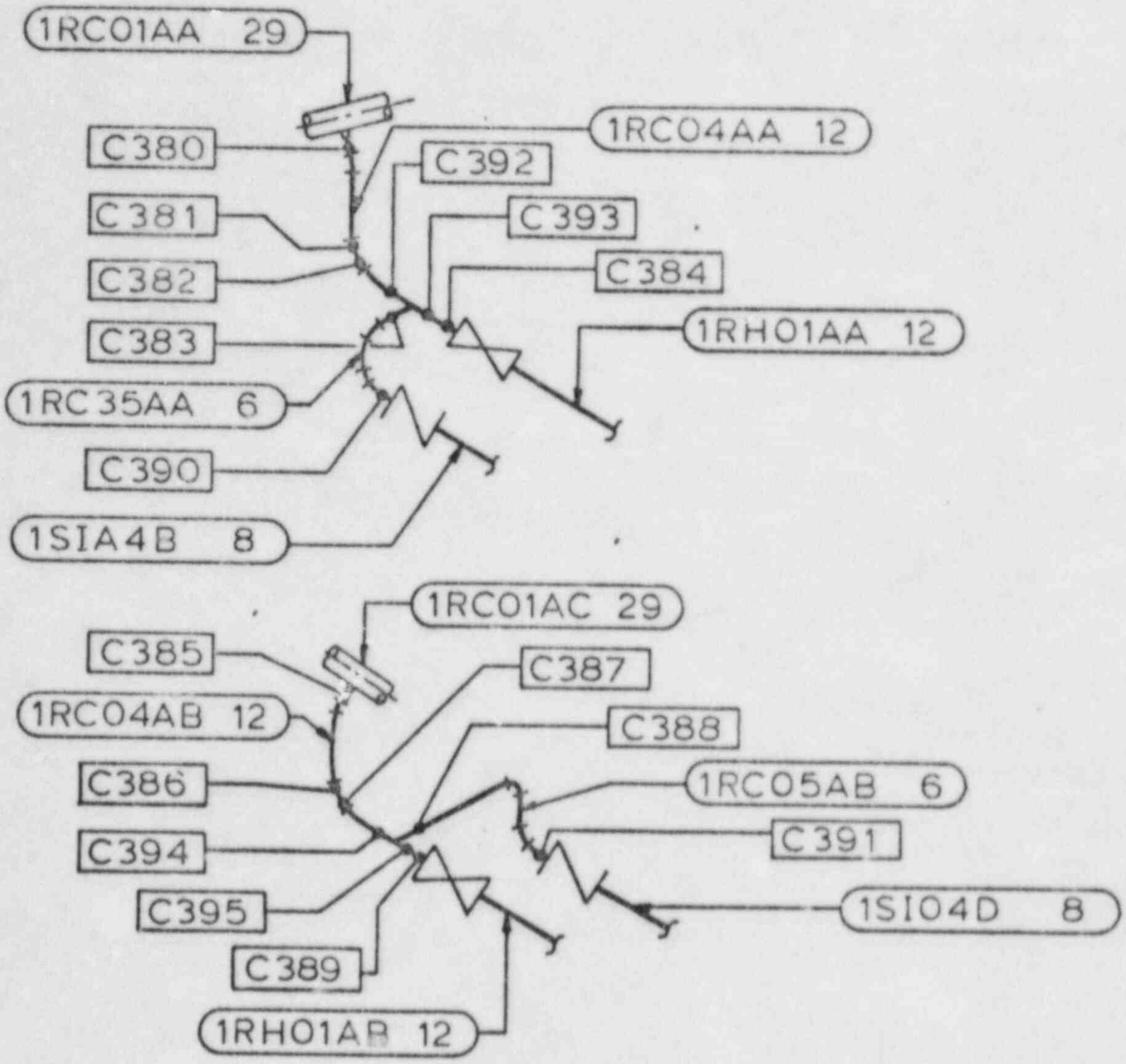
FIGURE 3.6-75

REACTOR COOLANT

SUBSYSTEM 1RC19

POSTULATED BREAK

LOCATIONS



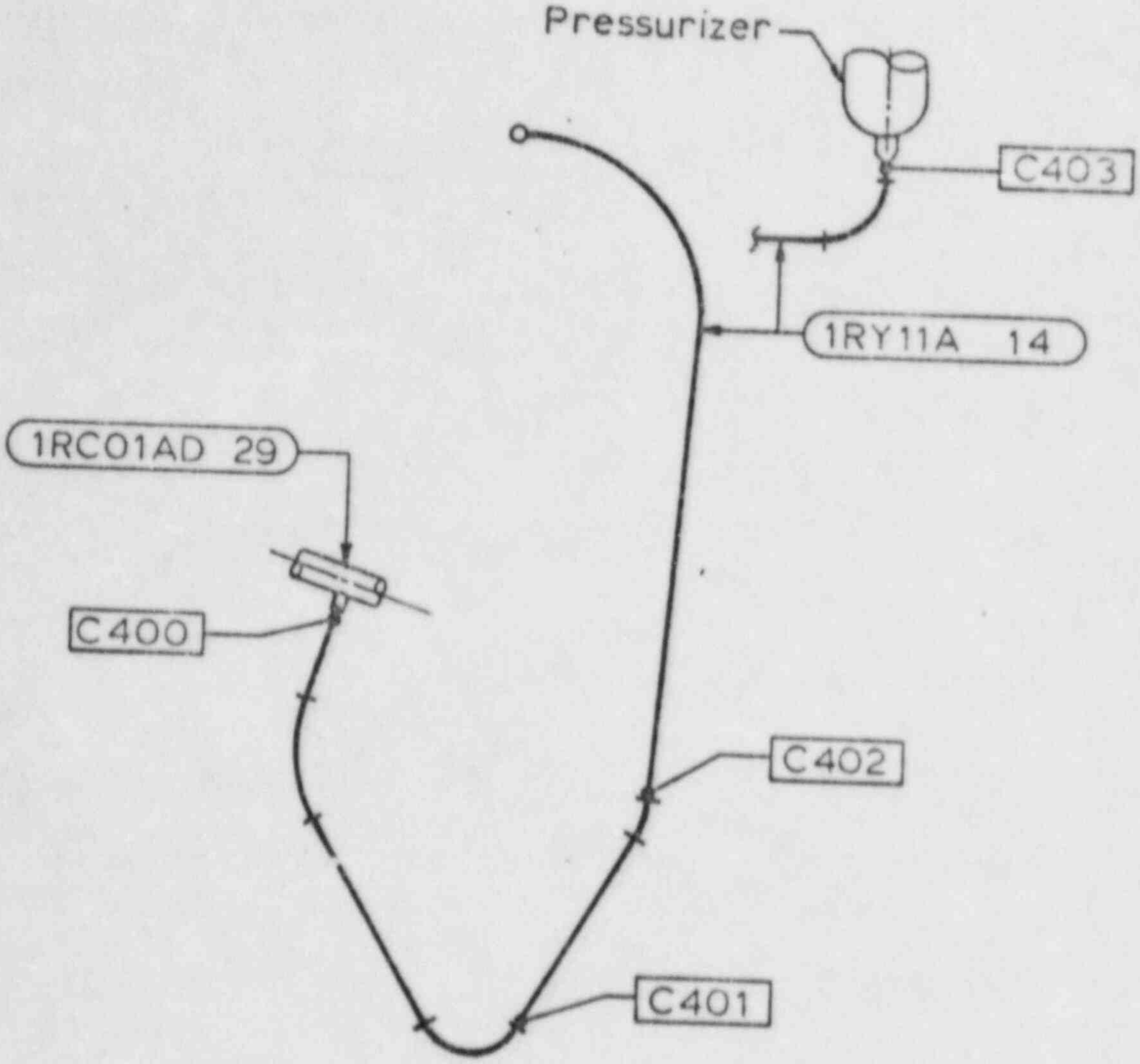
SUBSYSTEM 1RH02

• Indicates Postulated Break.

B/B

FIGURE 3.6-76

RESIDUAL HEAT REMOVAL LOOPS 1 AND 3 POSTULATED BREAK LOCATIONS



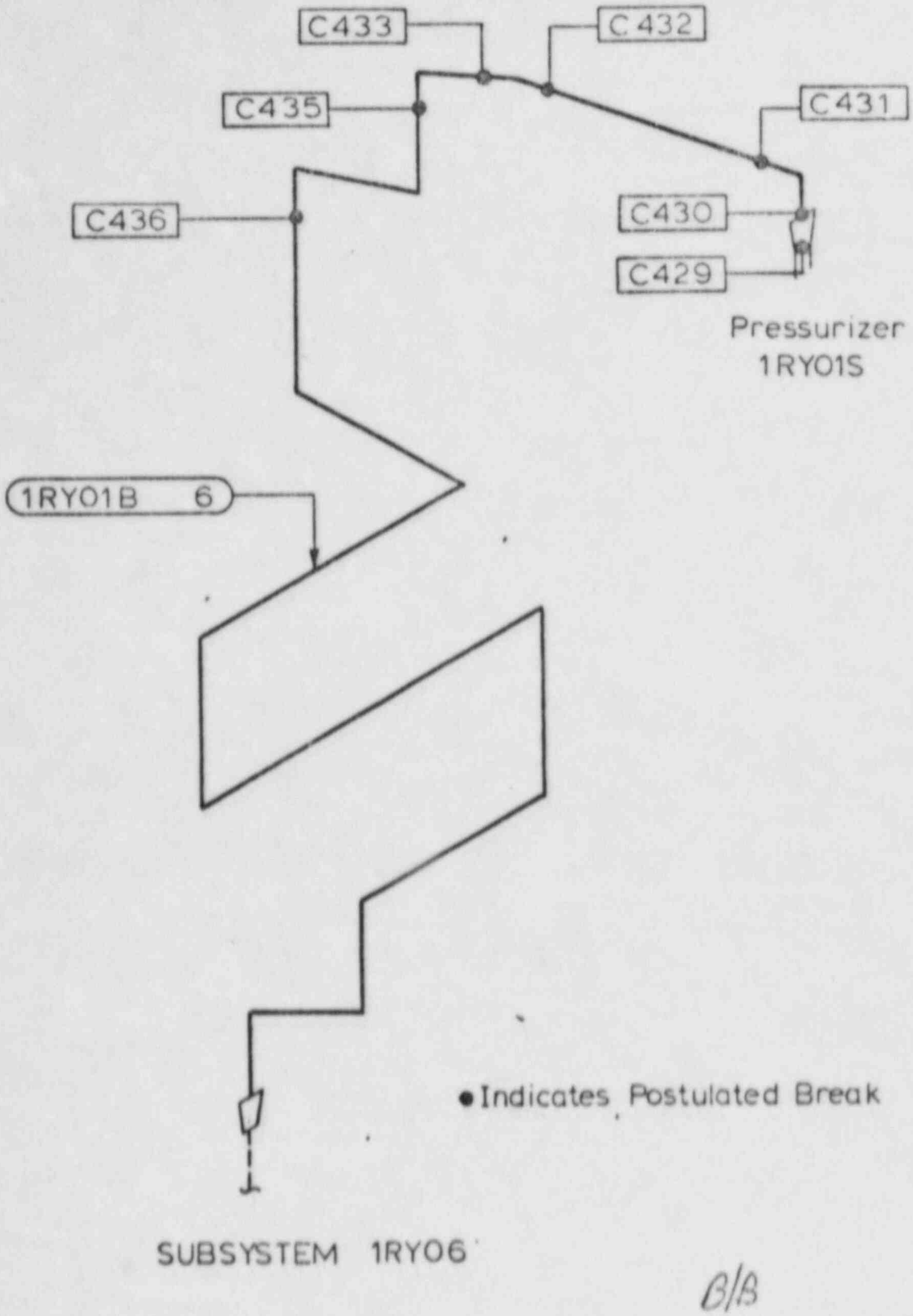
SUBSYSTEM 1RY05

• Indicates Postulated Break

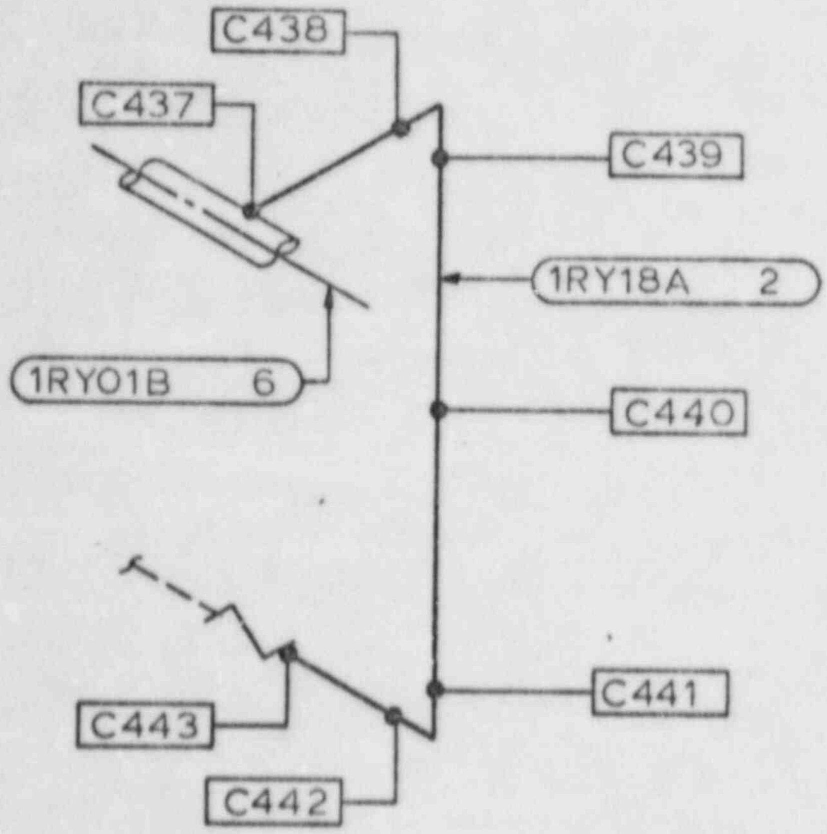
B/B

FIGURE 3.6-77

PRESSURIZER SURGE LINE POSTULATED BREAK LOCATIONS



B/B
FIGURE 3.6-78
PRESSURIZER SPRAY LINE
POSTULATED BREAK LOCATIONS
(Sheet 1 of 4)

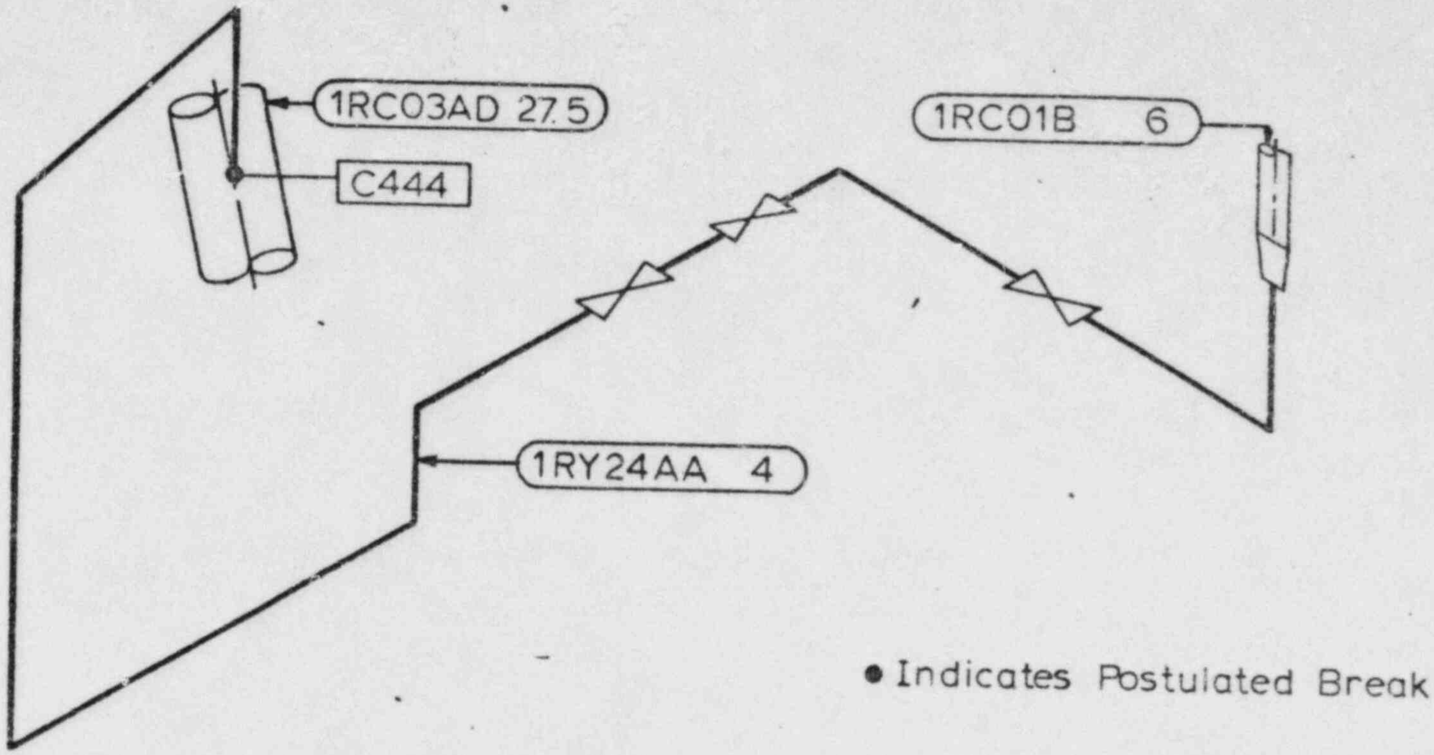


SUBSYSTEM 1RY06 (Cont'd.)

• Indicates Postulated Break

B/B

FIGURE 3.6-78
PRESSURIZER SPRAY
LINE
POSTULATED BREAK
LOCATIONS
(Sheet 2 of 4)



SUBSYSTEM 1RY06 (Cont'd.)

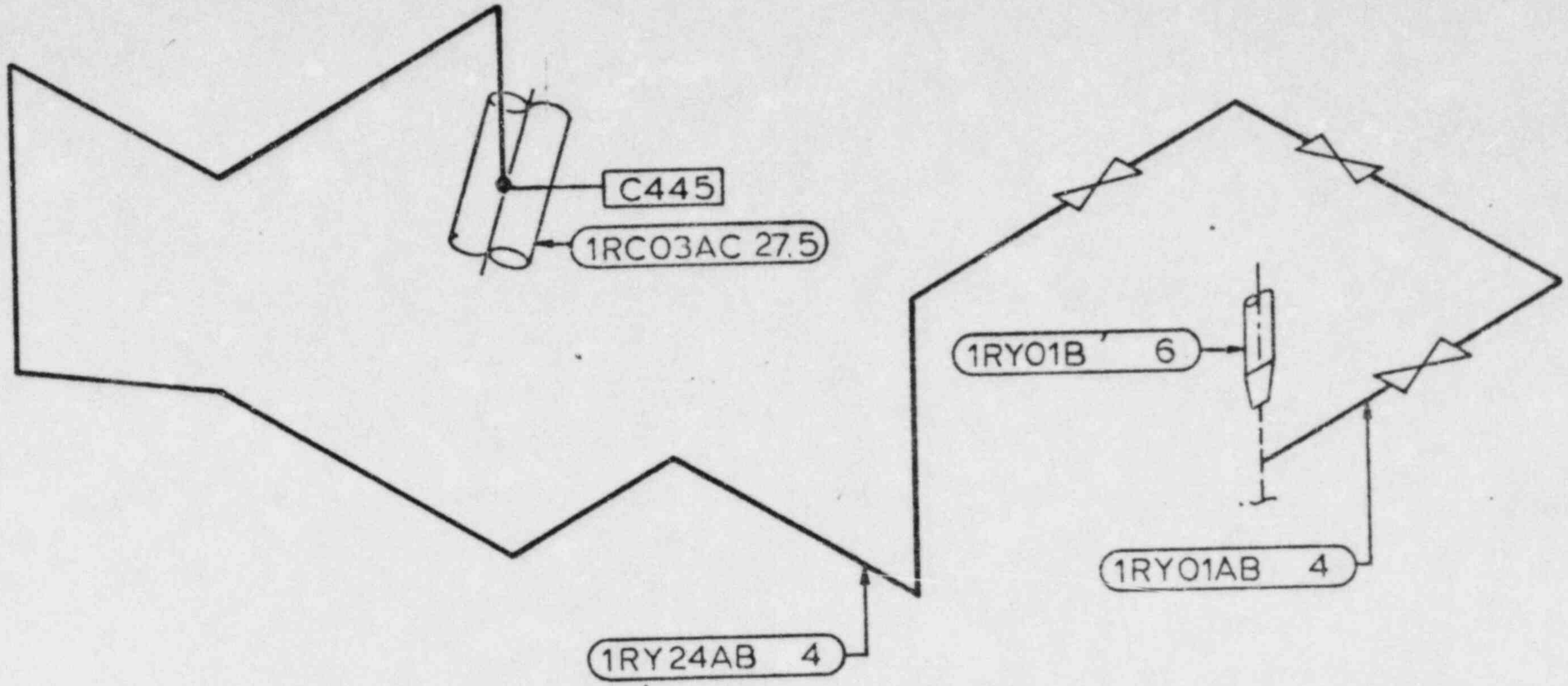
B/B

FIGURE 3.6-78

PRESSURIZER SPRAY LINE

POSTULATED BREAK LOCATIONS

(SHEET 3 OF 4)



SUBSYSTEM 1RY06 (Cont'd.)

• Indicates Postulated Break

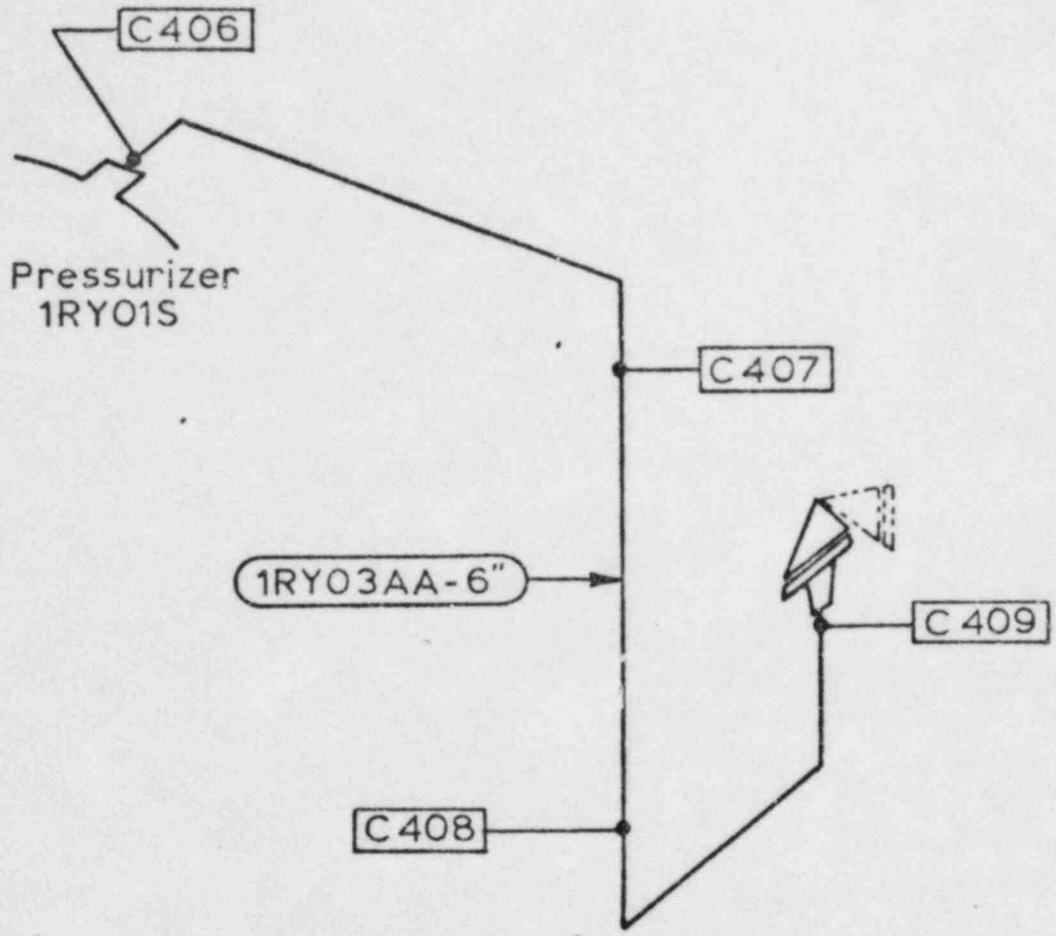
B/B

FIGURE 3.6-78

PRESSURIZER SPRAY LINE

POSTULATED BREAK LOCATIONS

(SHEET 4 OF 4)



SUBSYSTEM 1RY09

• Indicates Postulated Break

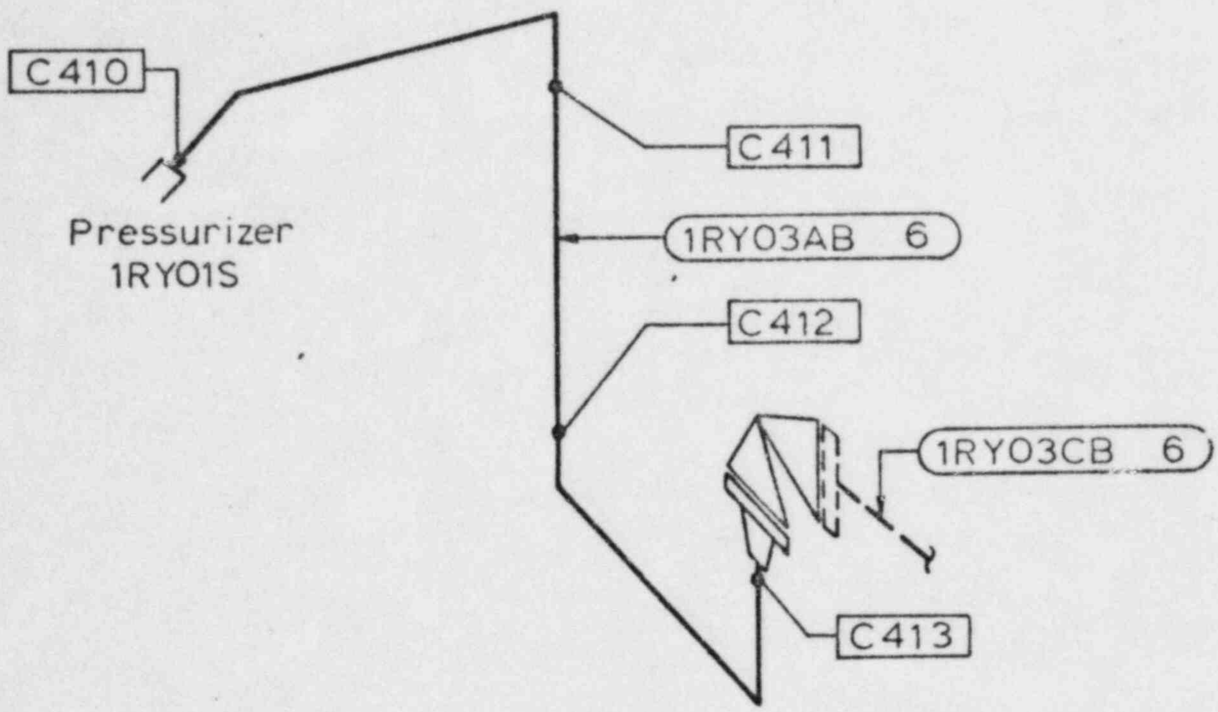
B/B

FIGURE 3.6-79

PRESSURIZER SAFETY/RELIEF VALVE LINES

POSTULATED BREAK LOCATIONS

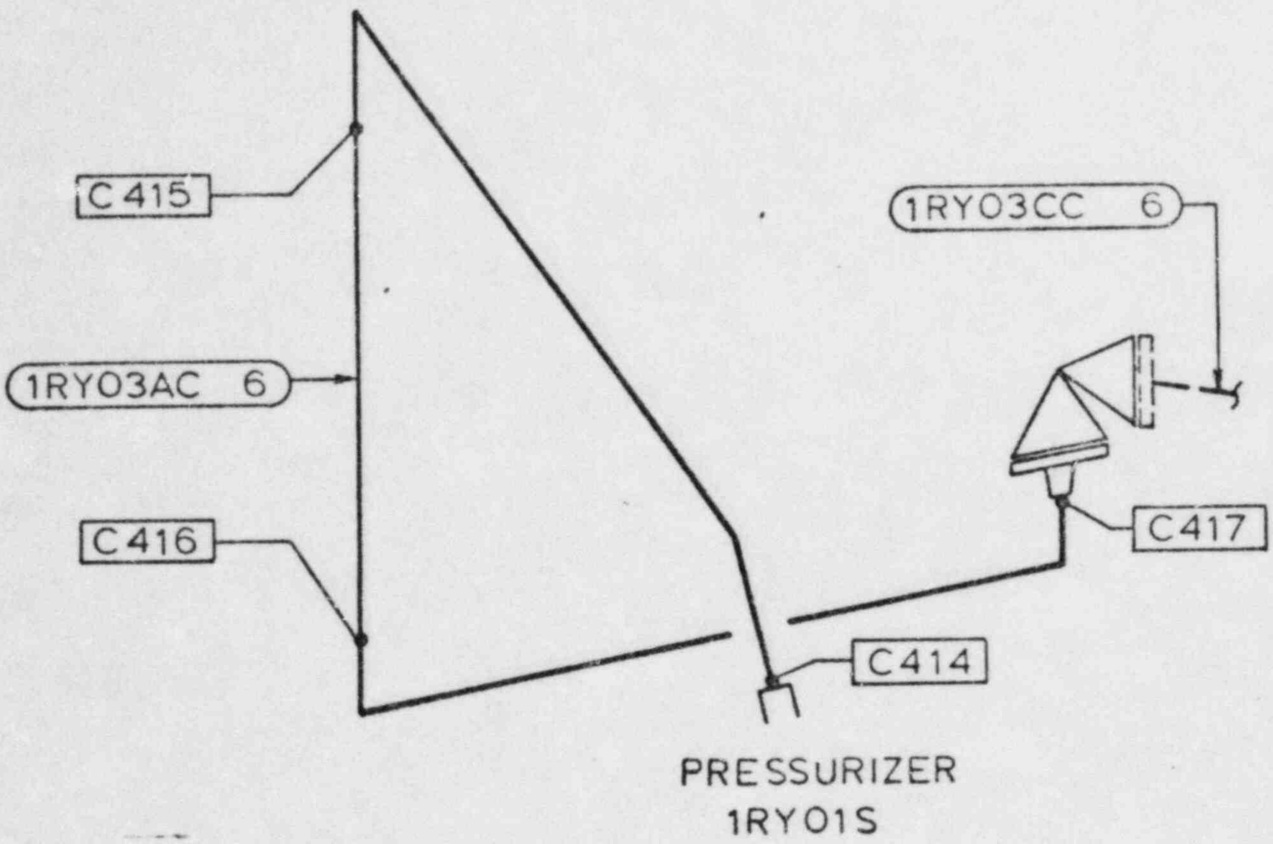
(SHEET 1 OF 4)



• Indicates Postulated Break

SUBSYSTEM 1RY09 (CONT'D)

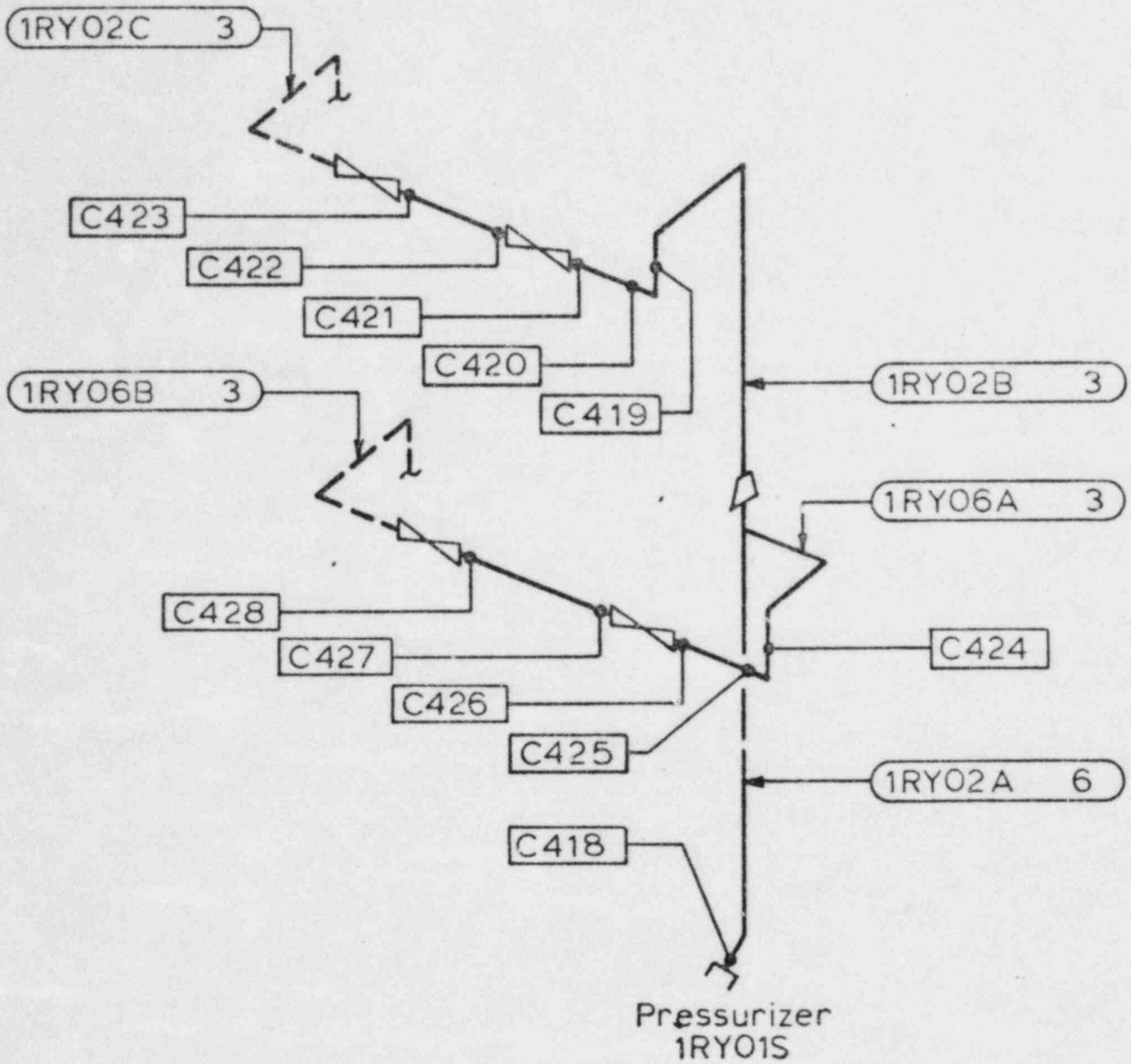
B/B
FIGURE 3.6-79
PRESSURIZER
SAFETY/ RELIEF VALVE
LINES
POSTULATED BREAK
LOCATIONS
(SHEET 2 OF 4)



SUBSYSTEM 1RY09 (Cont'd)

• Indicates Postulated Break

B/B
FIGURE 3.6-79
PRESSURIZER
SAFETY/RELIEF VALVE
LINES
POSTULATED BREAK
LOCATIONS
(SHEET 3 OF 4)

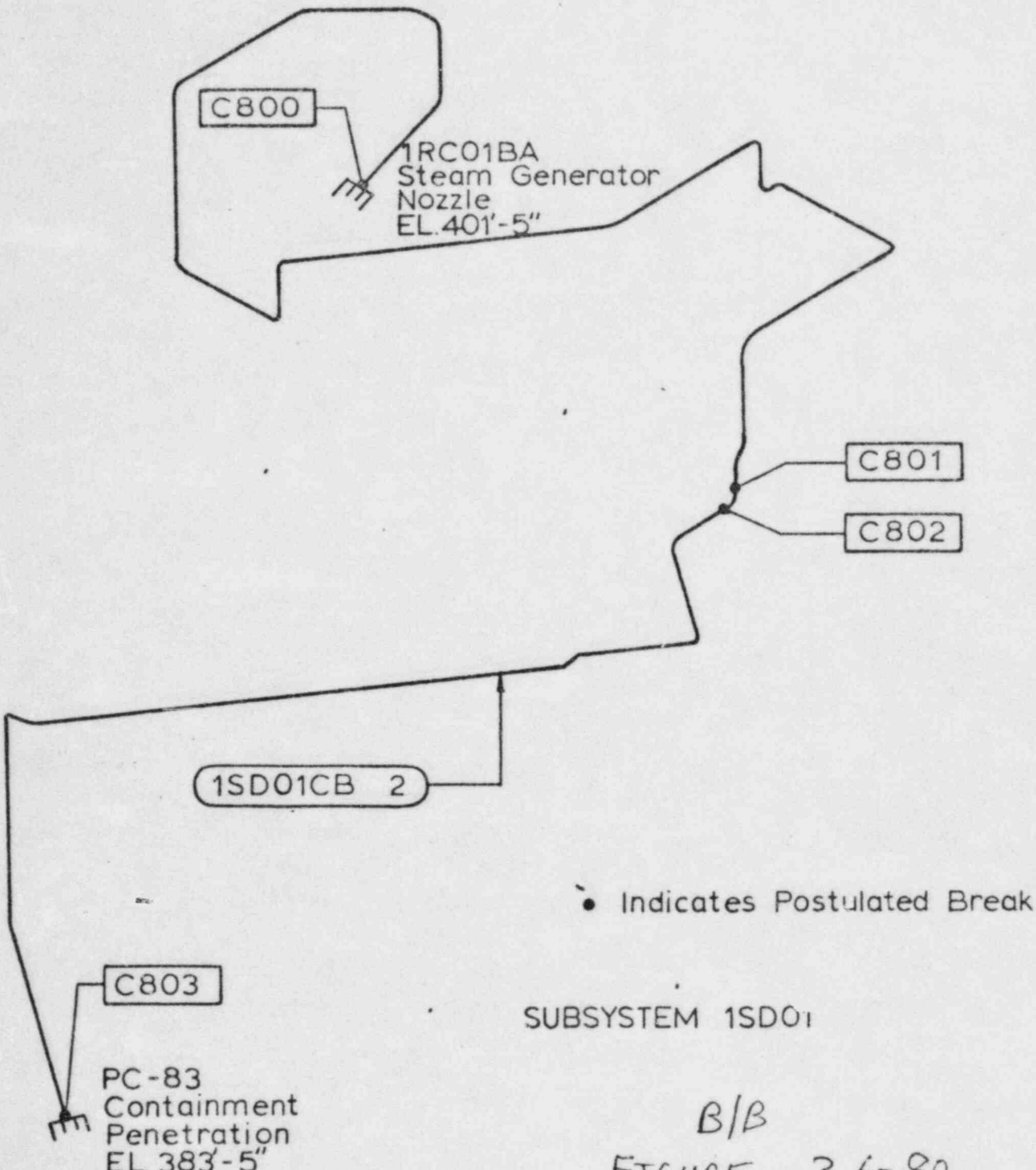


• Indicates Postulated Break

SUBSYSTEM 1RY09 (Cont'd)

B/B

FIGURE 3.6-79
PRESSURIZER
SAFETY / RELIEF VALVE
LINES
POSTULATED BREAK
LOCATIONS
(SHEET 4 OF 4)



B/B

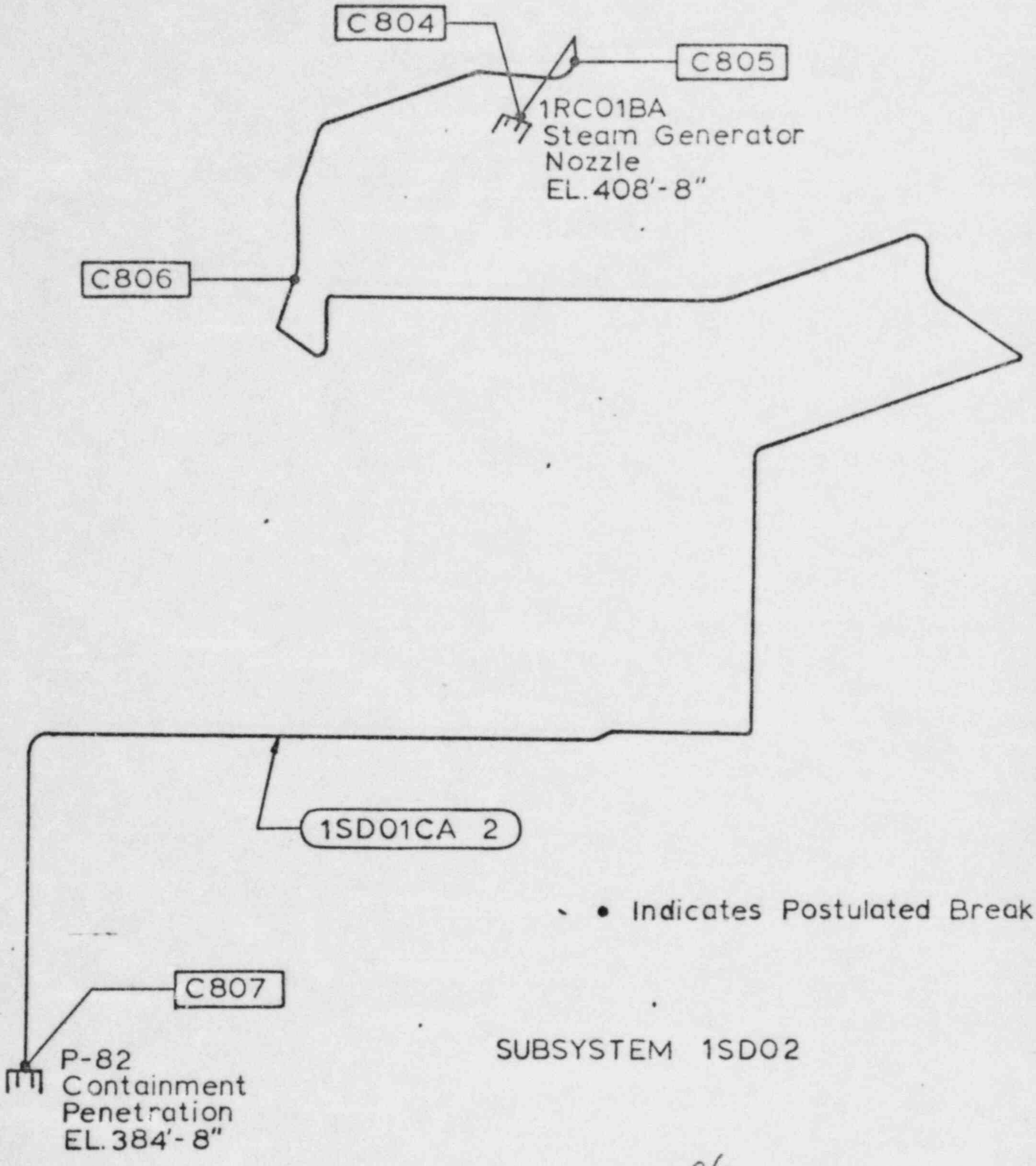
FIGURE 3.6-80

STEAM GENERATOR

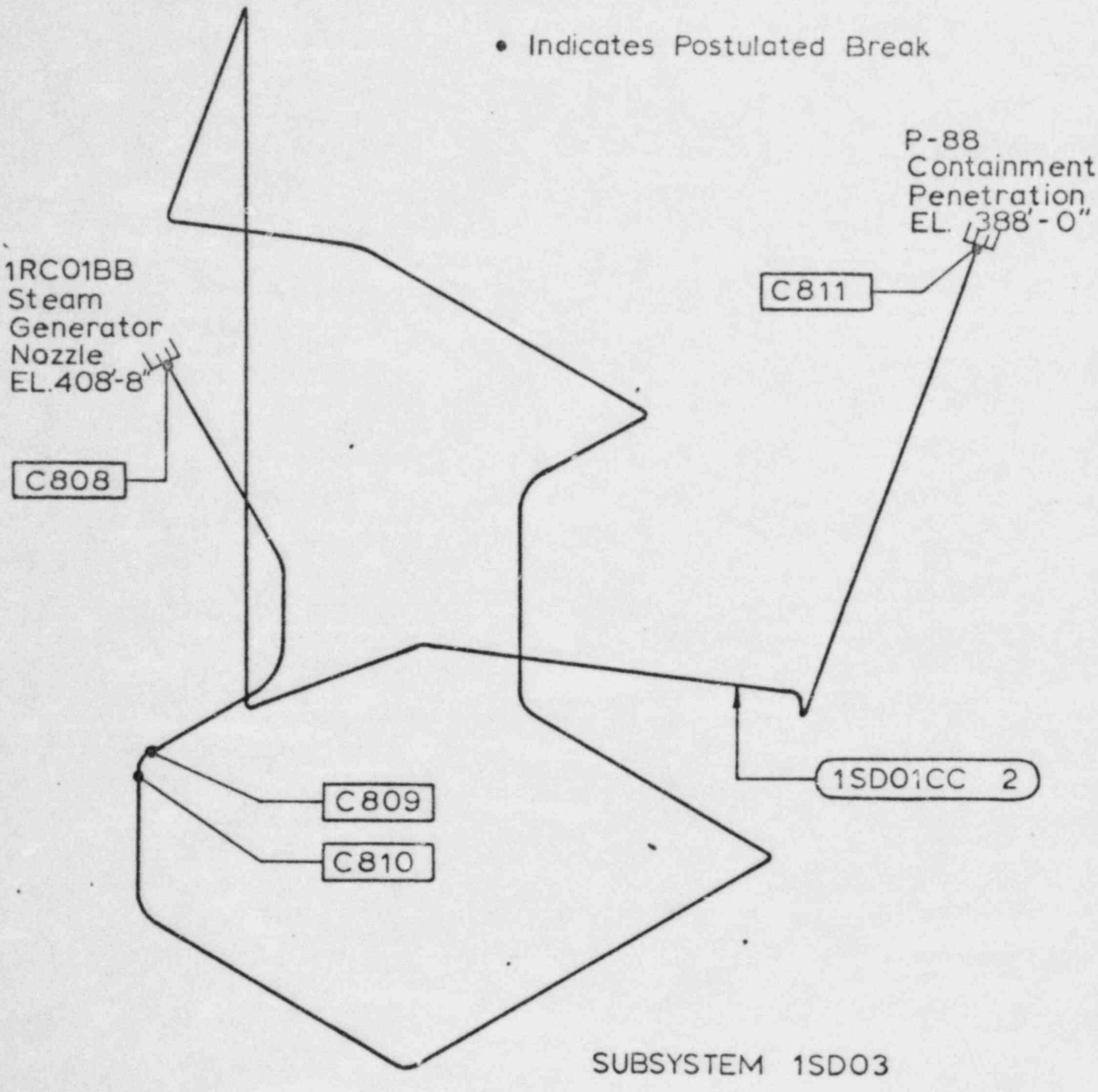
BLOWDOWN LOOP 2

SUBSYSTEM 1SD01

POSTULATED BREAK LOCATIONS



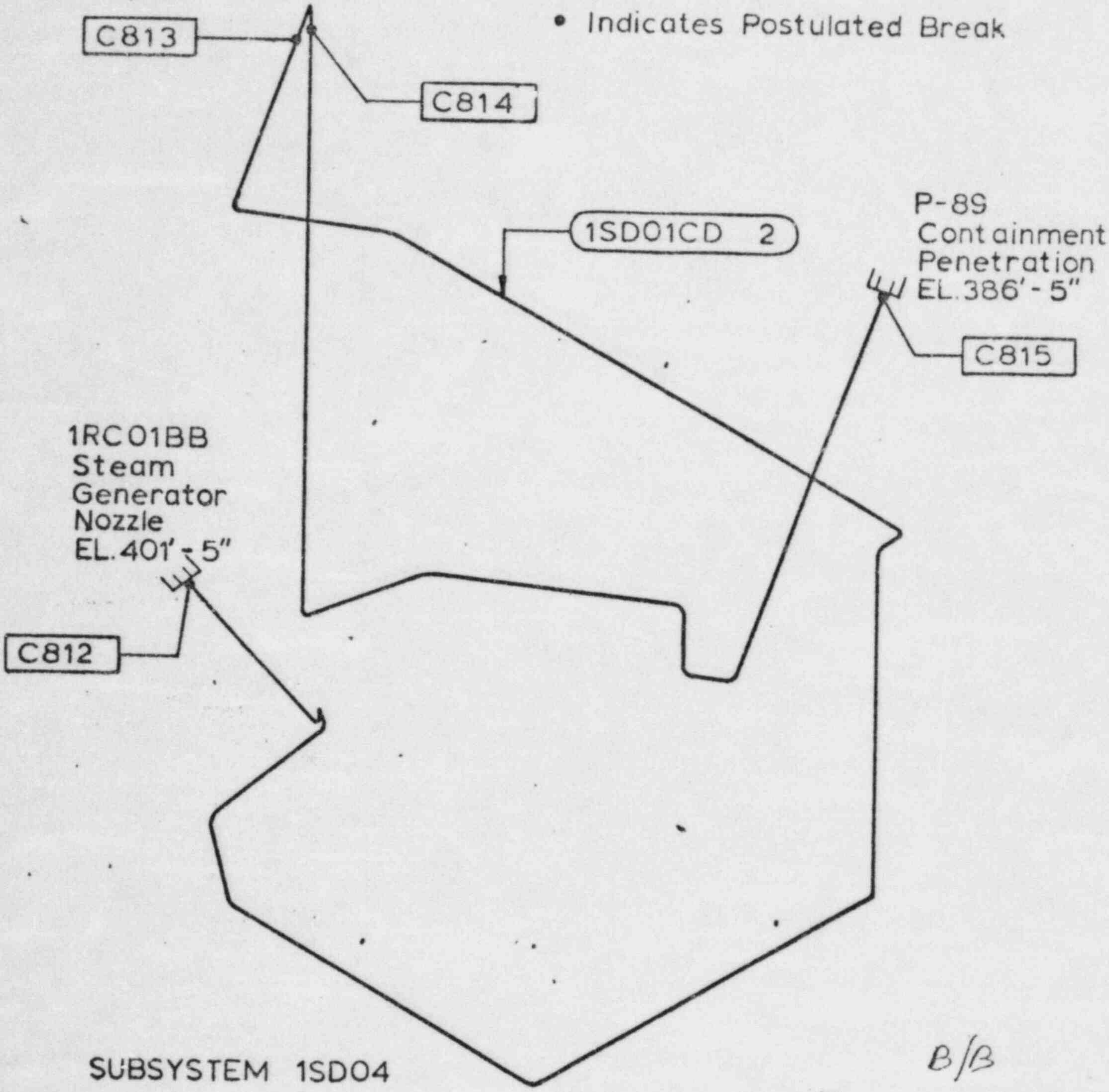
B/B
FIGURE 3.6-81
STEAM GENERATOR
BLOWDOWN LOOP 1
SUBSYSTEM 1SD02
POSTULATED BREAK
LOCATIONS



B/B

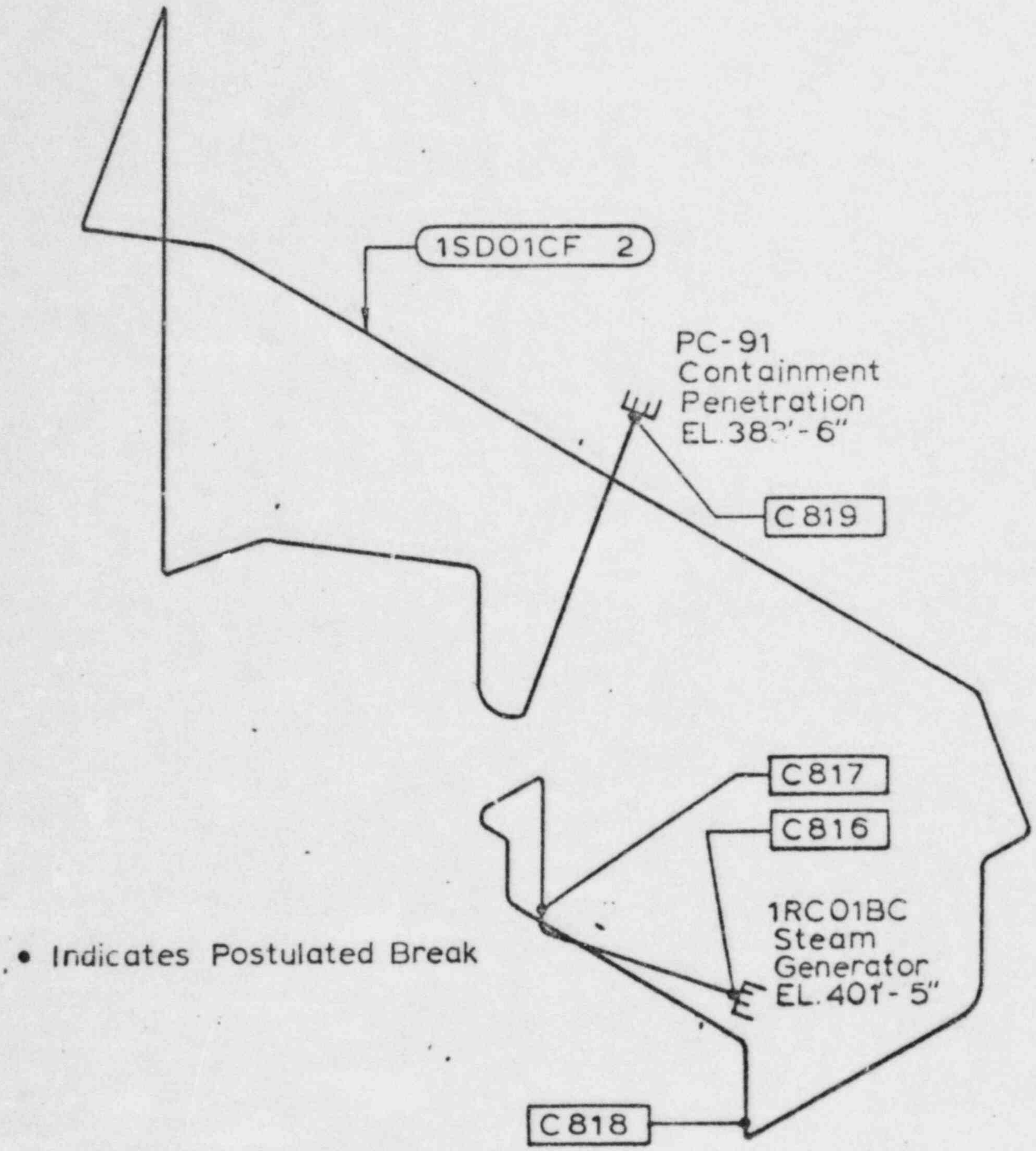
FIGURE 3.6-82

STEAM GENERATOR BLOWDOWN LOOP 2 SUBSYSTEM 1SD03 POSTULATED BREAK LOCATIONS



B/B

FIGURE 3.6-83
STEAM GENERATOR
BLOWDOWN LOOP 2
SUBSYSTEM 1SD04
POSTULATED BREAK
LOCATIONS

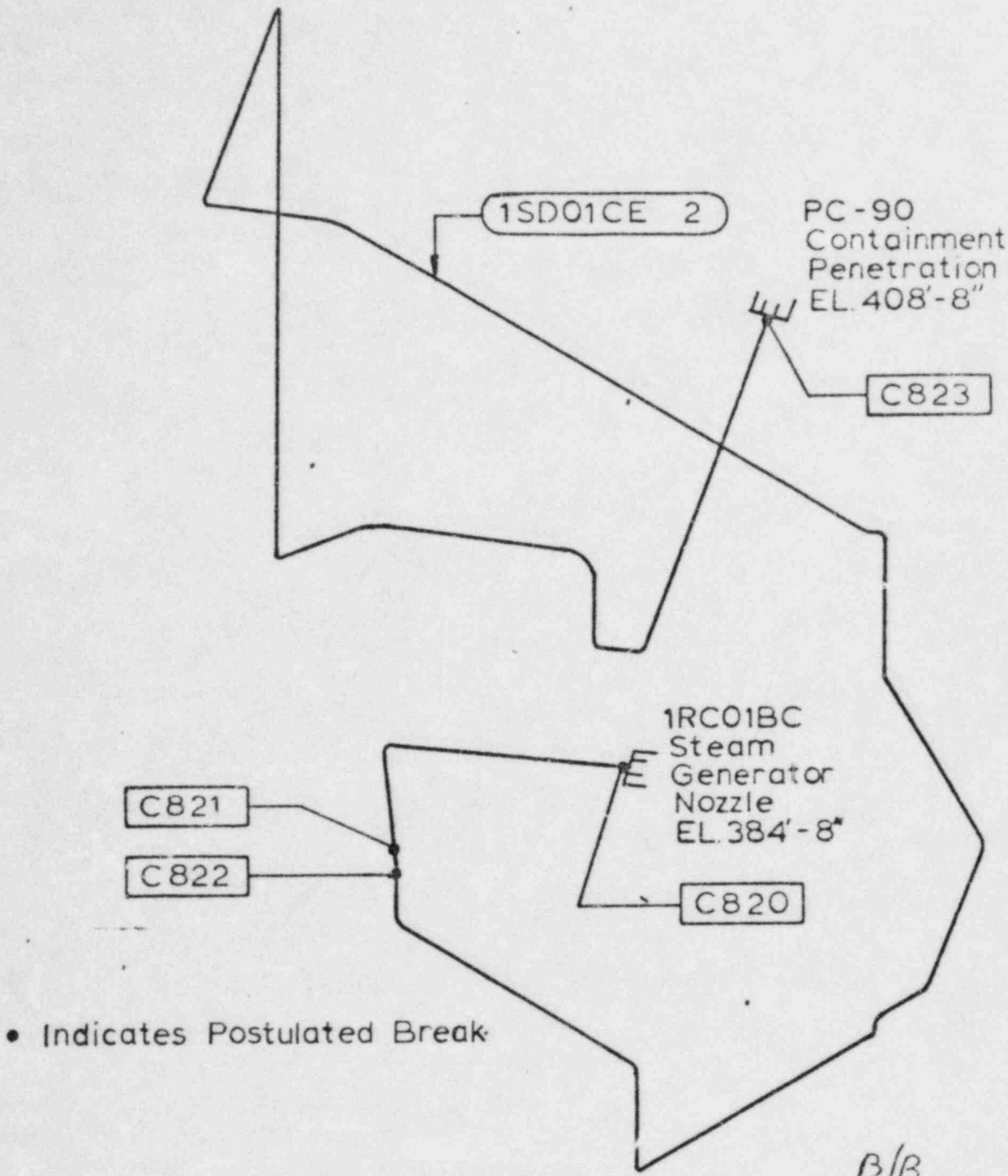


SUBSYSTEM 1SD05

B/B

FIGURE 3.6-84

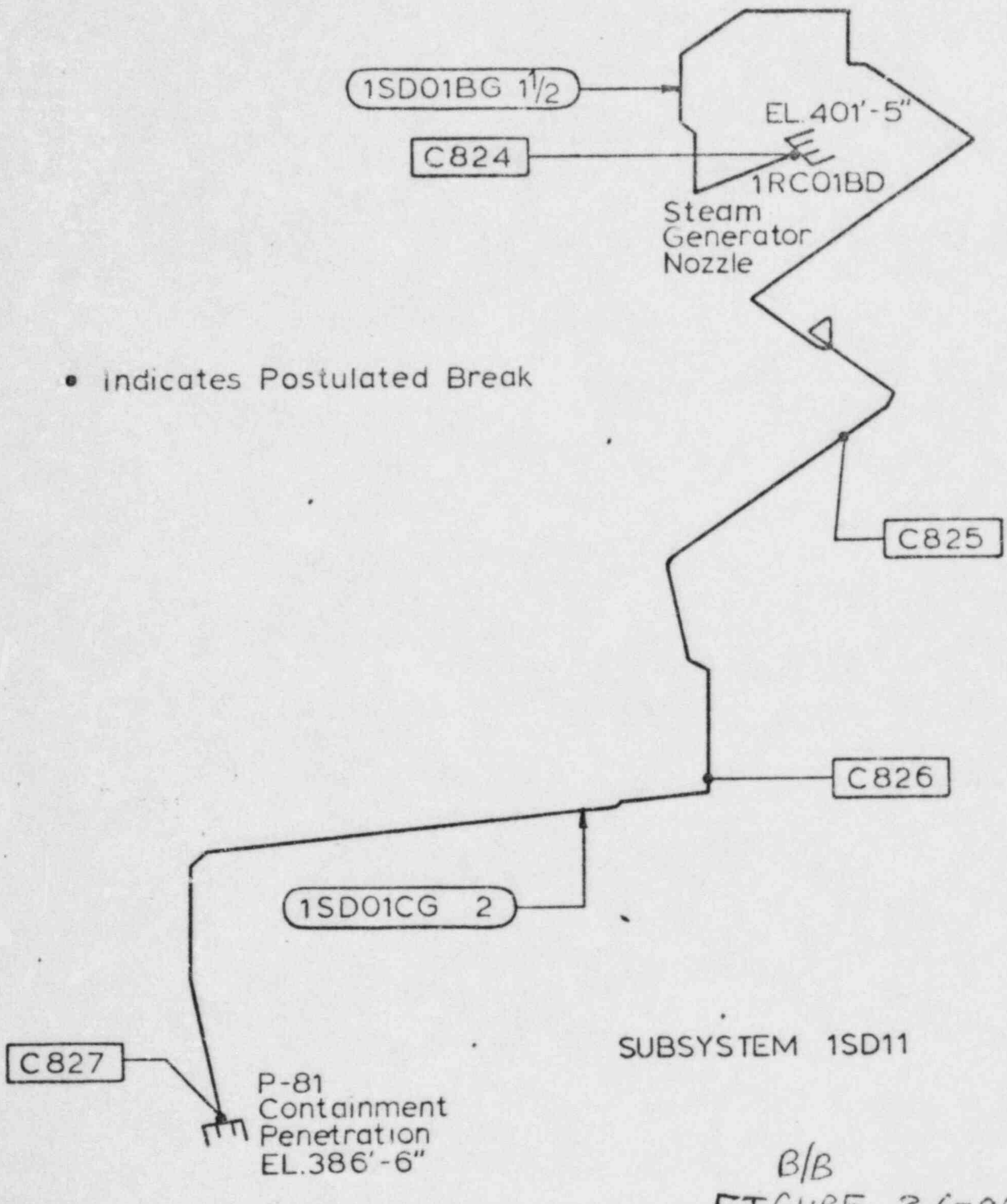
STEAM GENERATOR
BLOWDOWN LOOP 3
SUBSYSTEM 1SD05
POSTULATED BREAK
LOCATIONS



SUBSYSTEM 1SD06

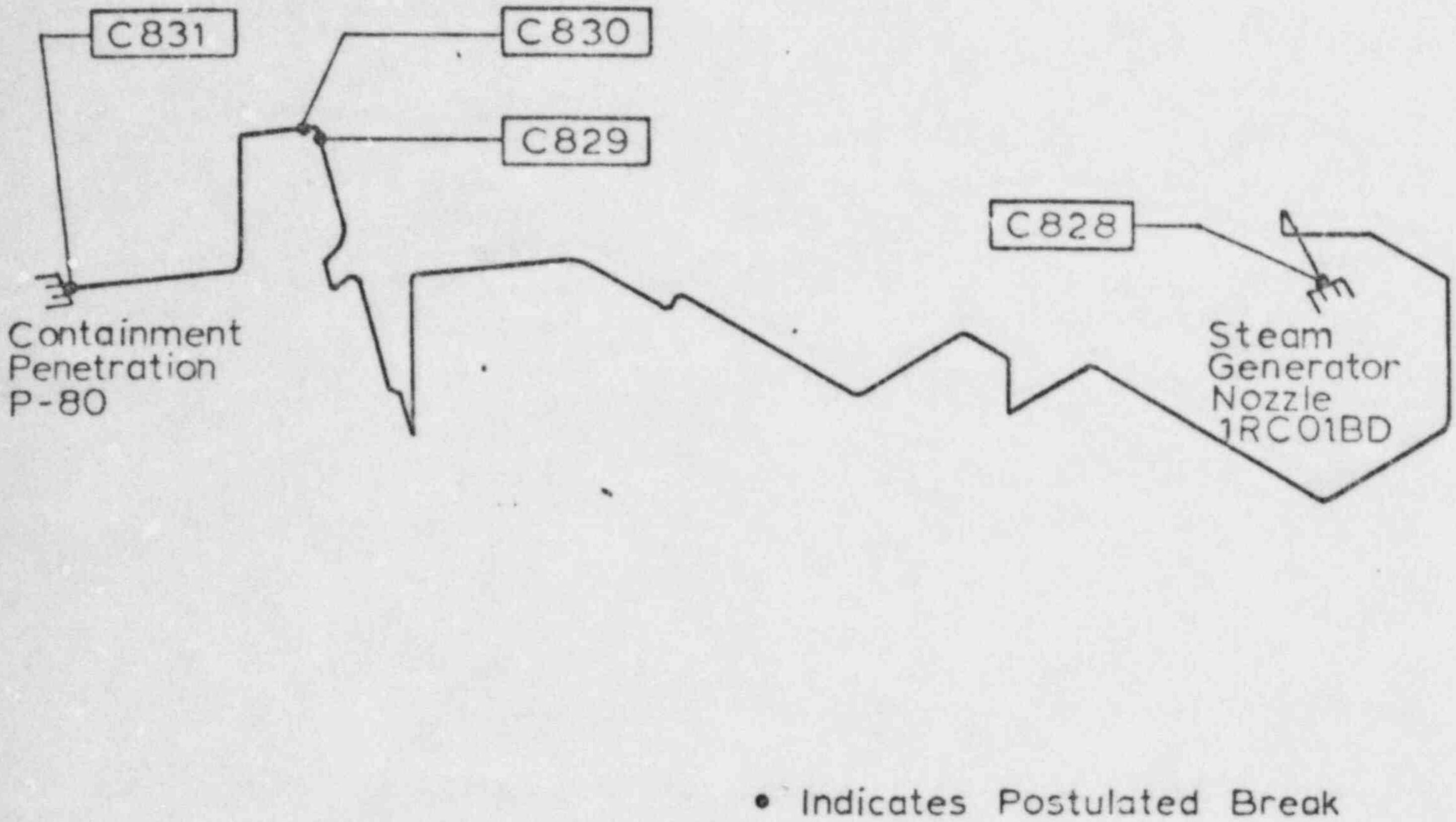
B/B

FIGURE 3.6-85
STEAM GENERATOR
BLOWDOWN LOOP 3
SUBSYSTEM 1SD06
POSTULATED BREAK
LOCATIONS



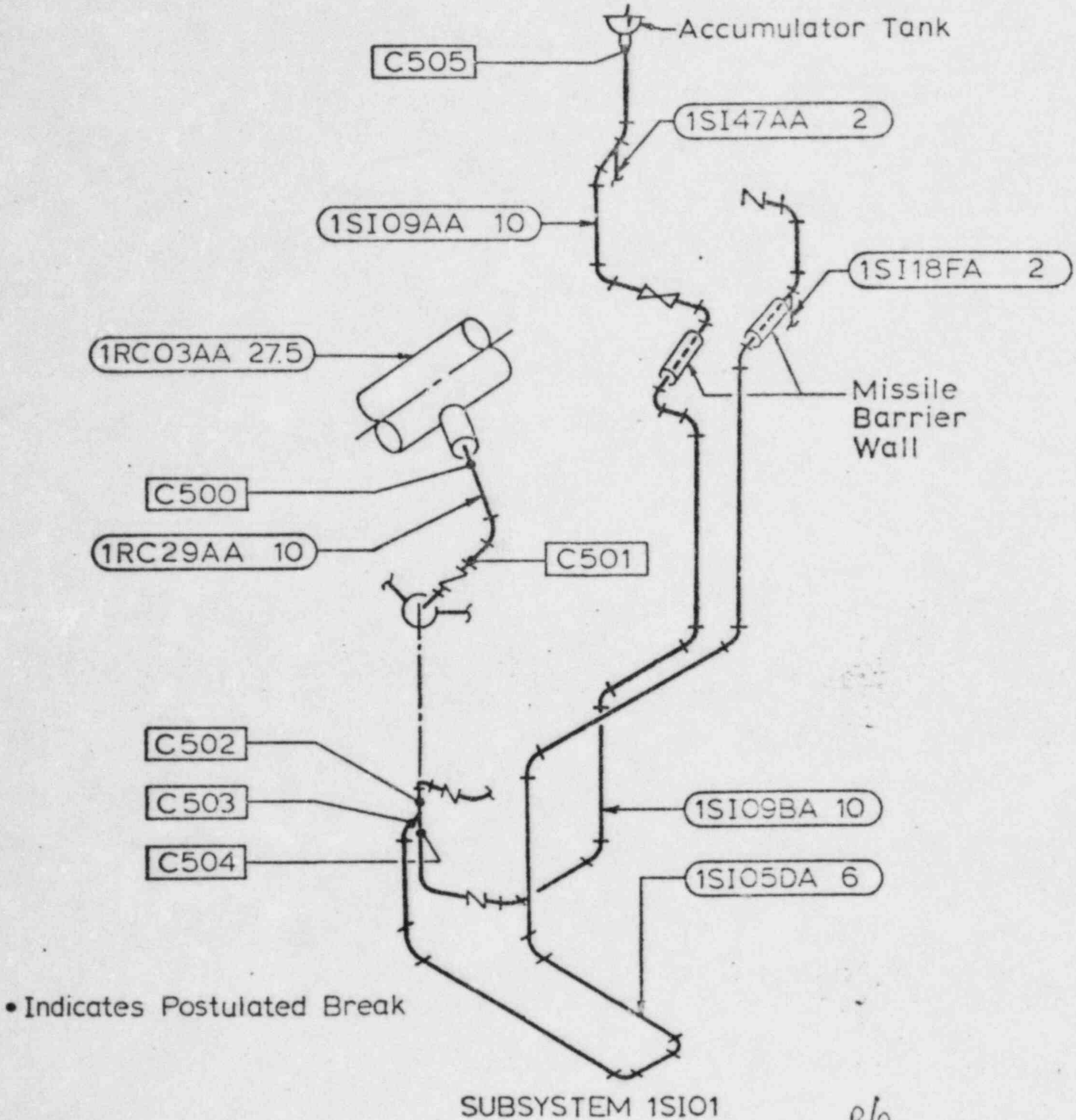
SUBSYSTEM 1SD11

B/B
FIGURE 3.6-86
STEAM GENERATOR
BLOWDOWN LOOP 4
SUBSYSTEM 1SD11
POSTULATED BREAK
LOCATIONS



SUBSYSTEM 1SD12

B/B
FIGURE 3.6-87
STEAM GENERATOR
BLOWDOWN LOOP 4
SUBSYSTEM 1SD12
POSTULATED BREAK
LOCATIONS

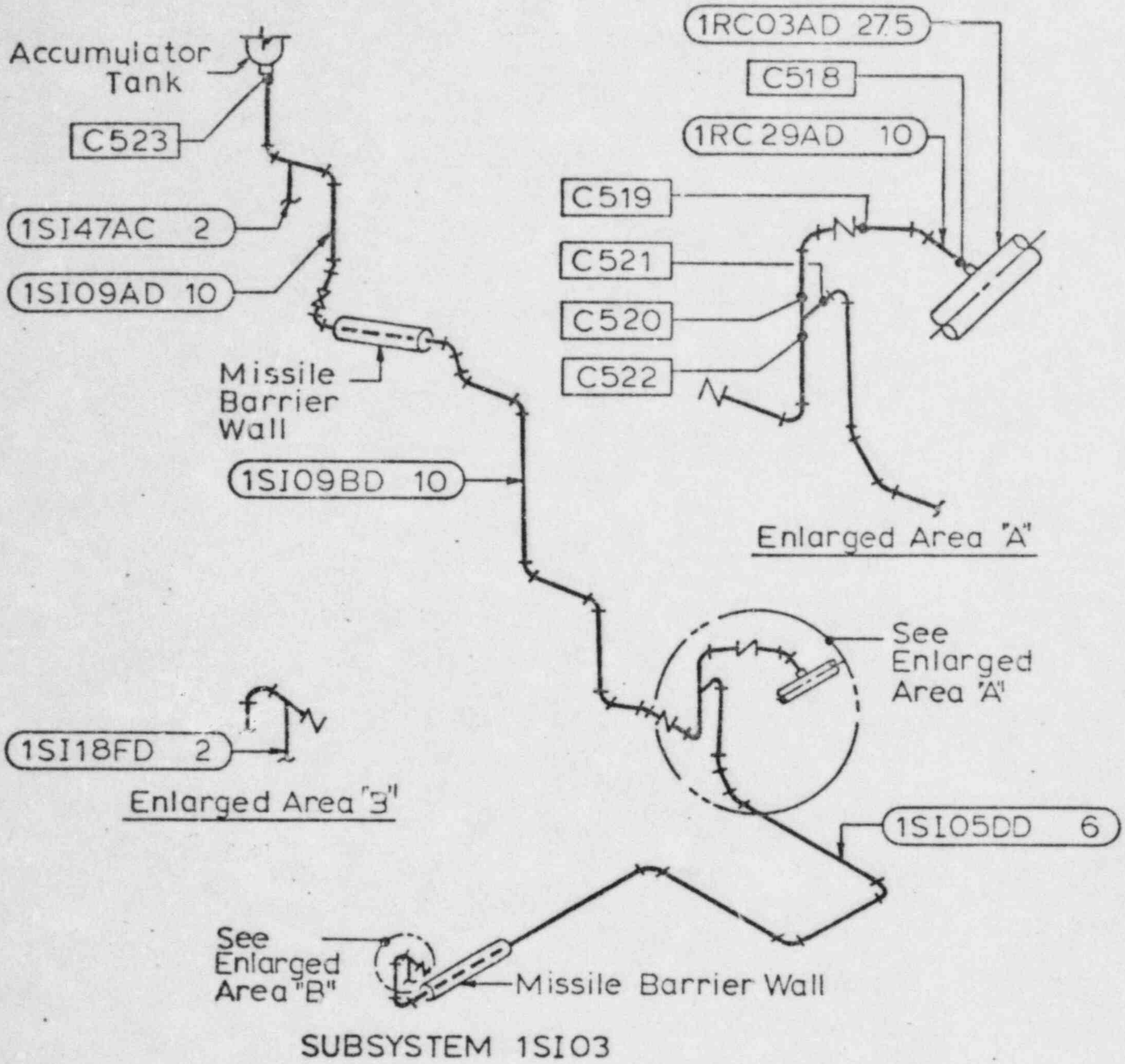


B/B

FIGURE 3.6-88

SAFETY INJECTION
LOOP 1

POSTULATED BREAK
LOCATIONS



• Indicates Postulated Break

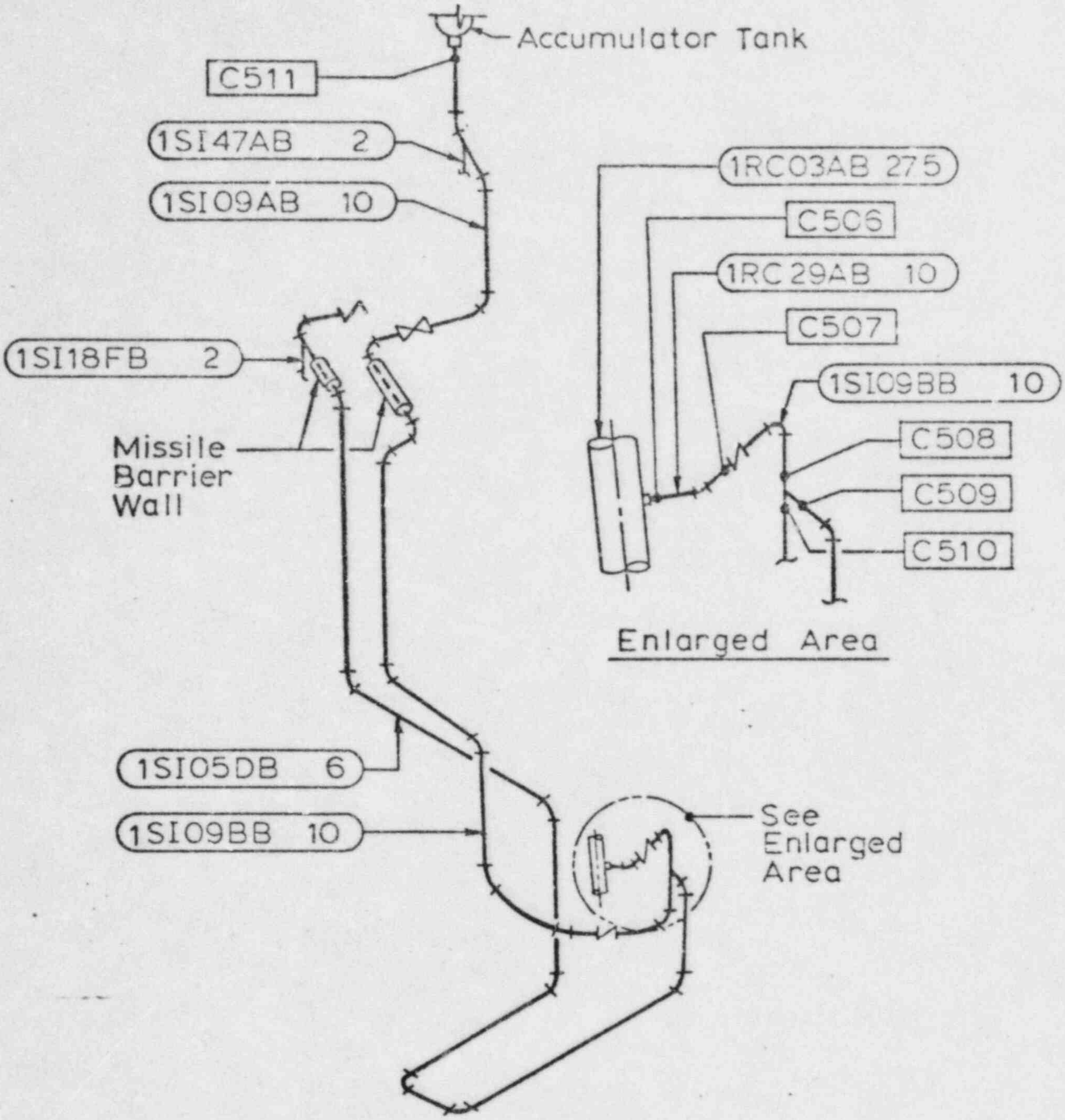
B/B

FIGURE 3.6-89

SAFETY INJECTION

LOOP 2

POSTULATED BREAK LOCATIONS



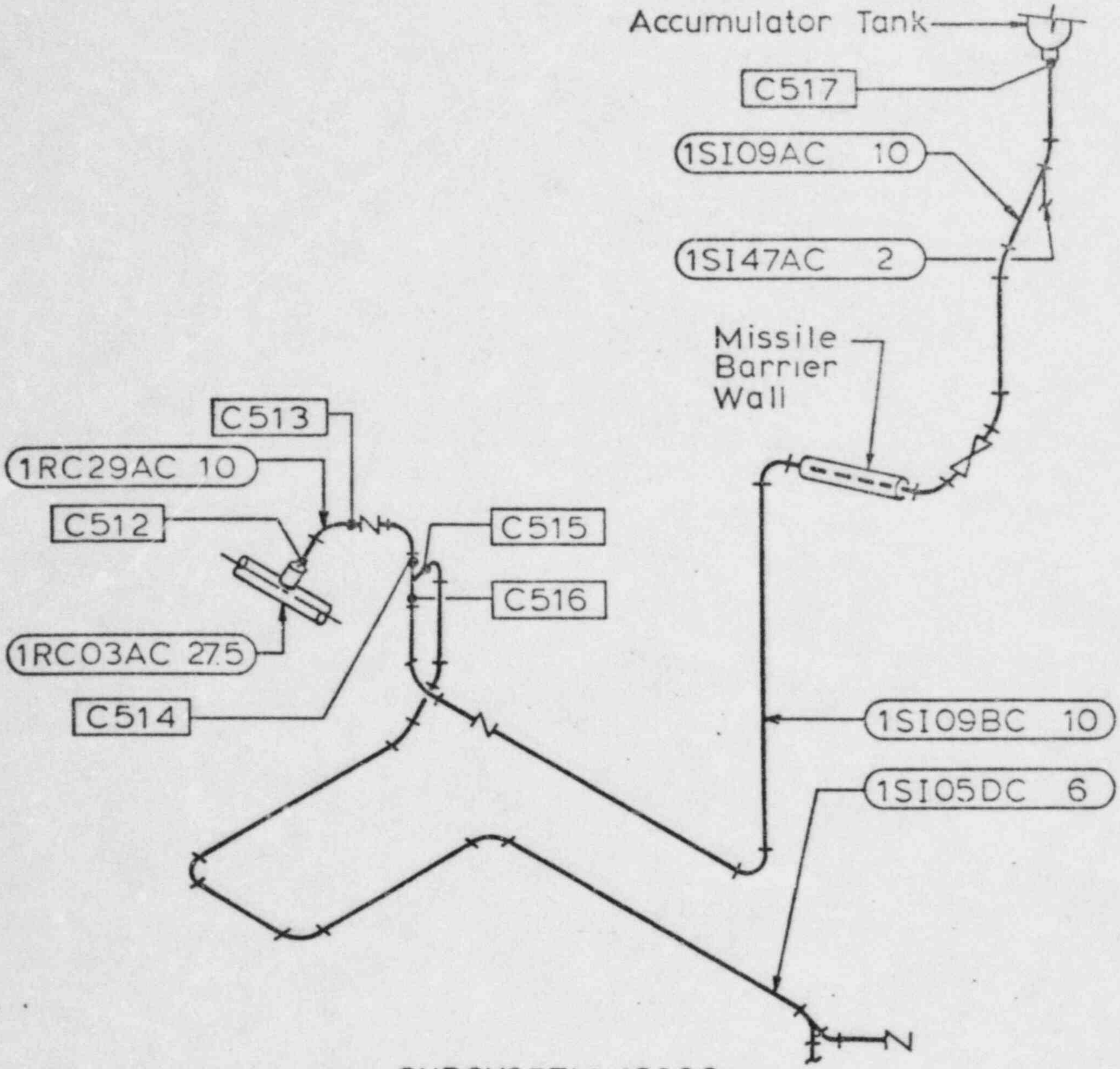
SUBSYSTEM 1SI04

• Indicates Postulated Break

B/B

FIGURE 3.6-90

SAFETY INJECTION
LOOP 3
POSTULATED BREAK
LOCATIONS



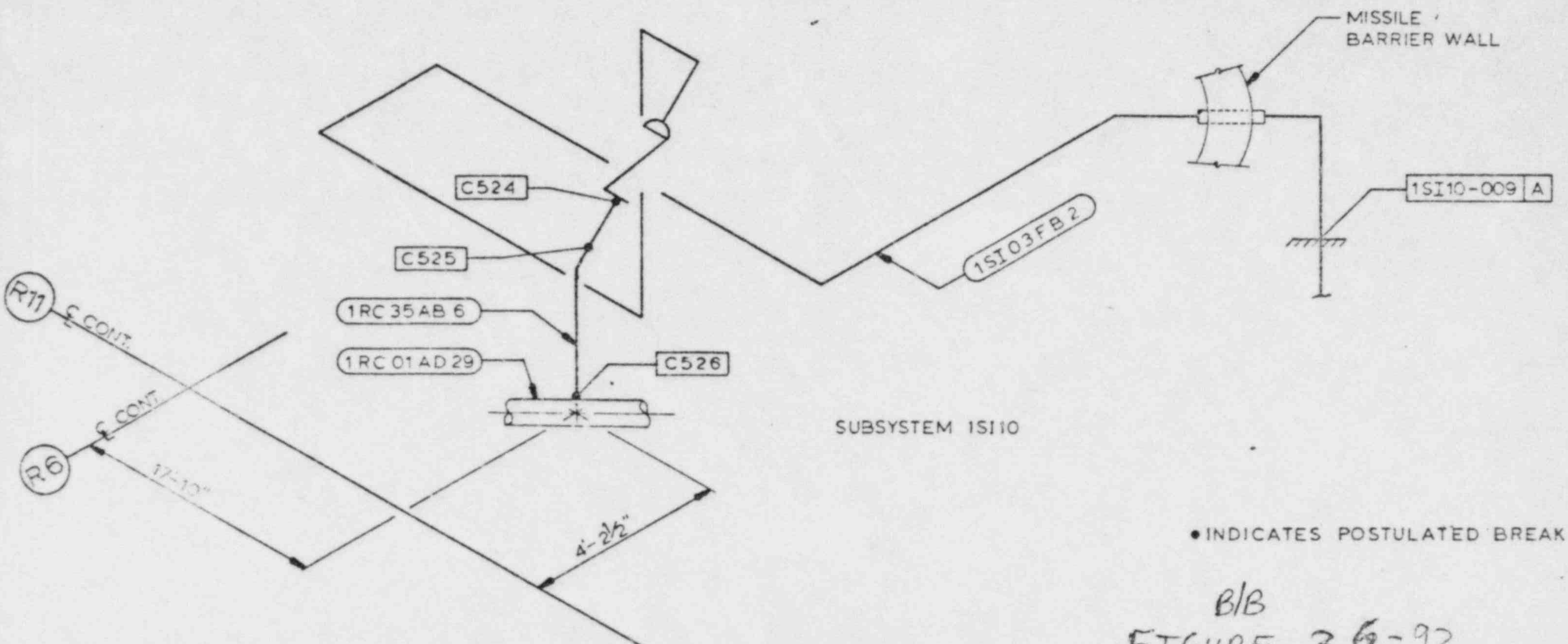
SUBSYSTEM 1SI09

• Indicates Postulated Break

B/B

FIGURE 3.6-91

SAFETY INJECTION
LOOP 4
POSTULATED BREAK
LOCATIONS

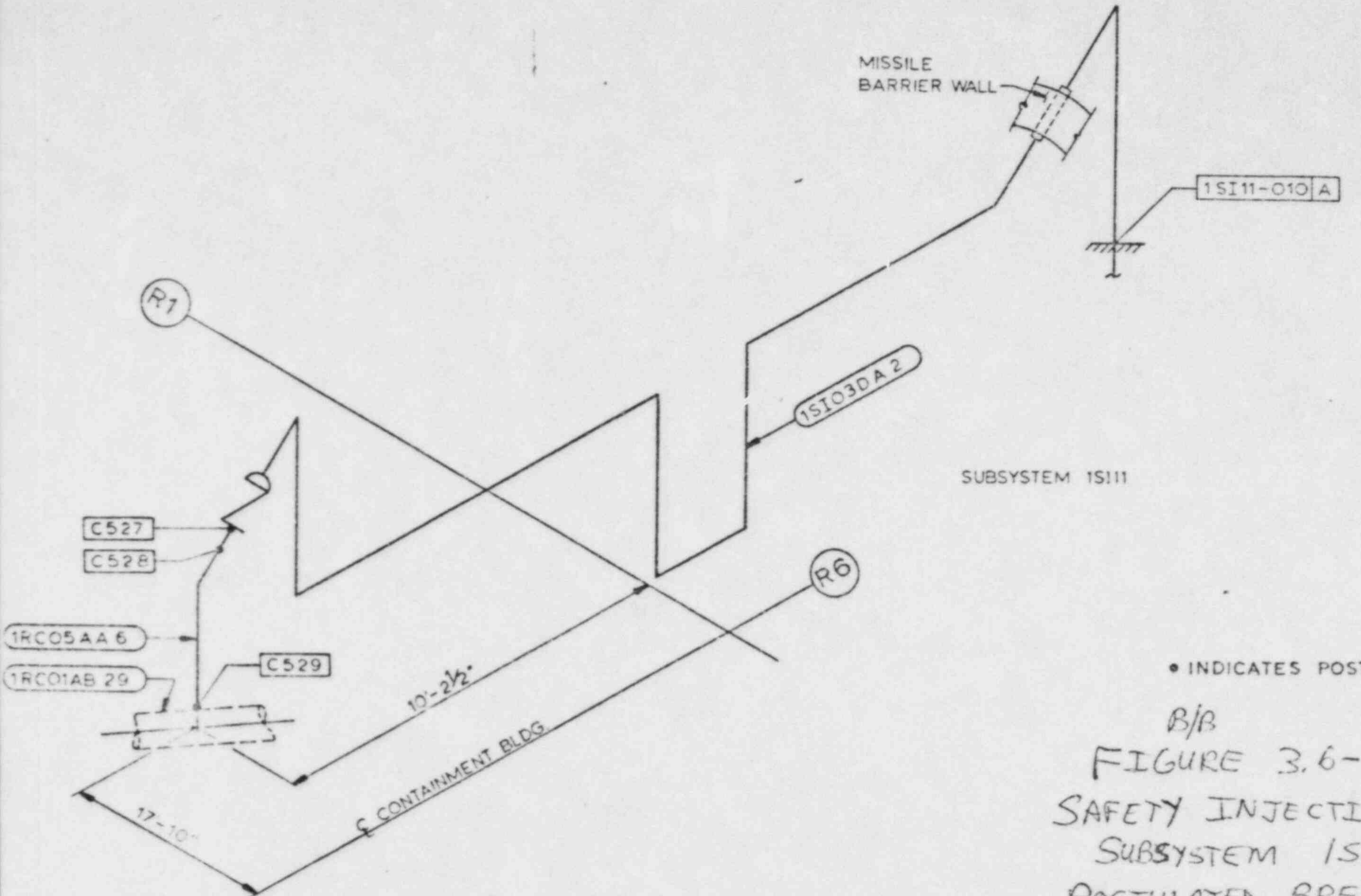


B/B

FIGURE 3.6-92

SAFETY INJECTION
SUBSYSTEM 1SI10

POSTULATED BREAK
LOCATIONS

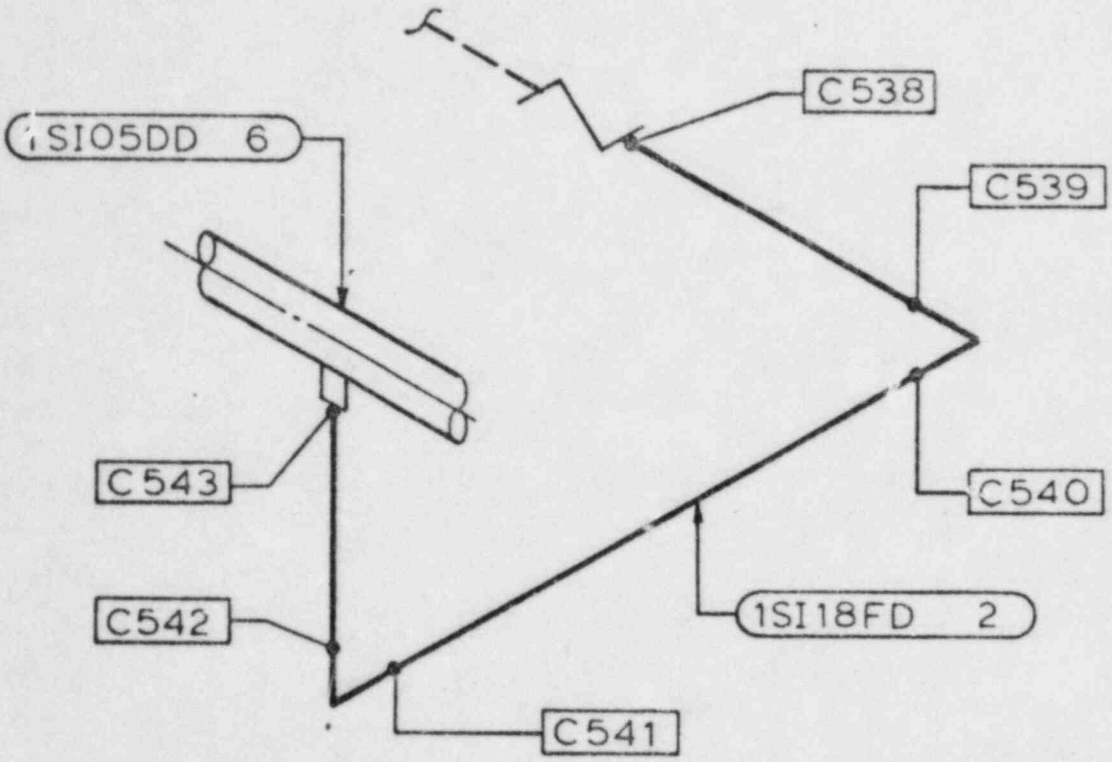


• INDICATES POSTULATED BREAK

B/B

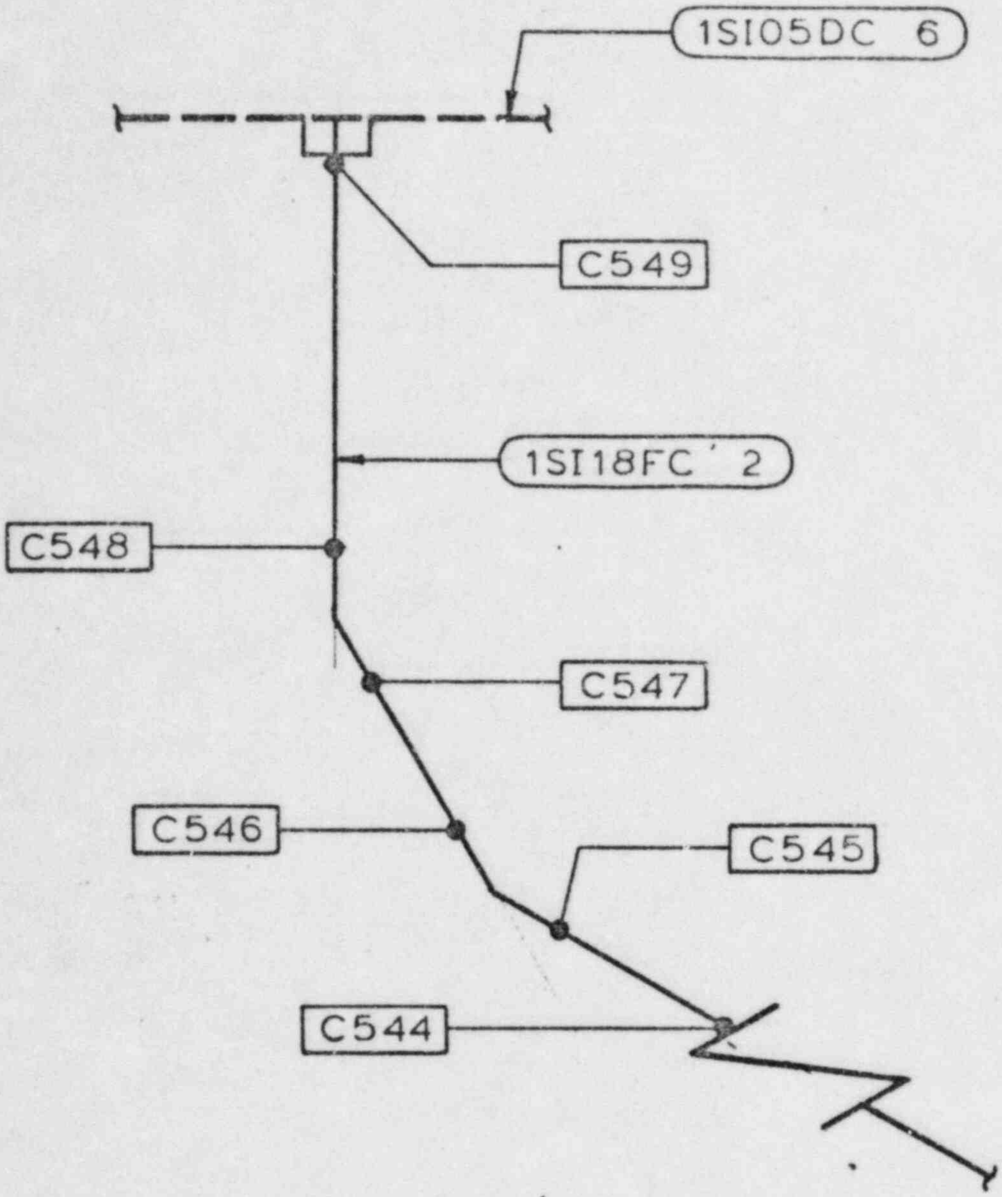
FIGURE 3.6-93
SAFETY INJECTION
SUBSYSTEM 1SI11
POSTULATED BREAK
LOCATIONS

SUBSYSTEM 1SI16



• Indicates Postulated Break

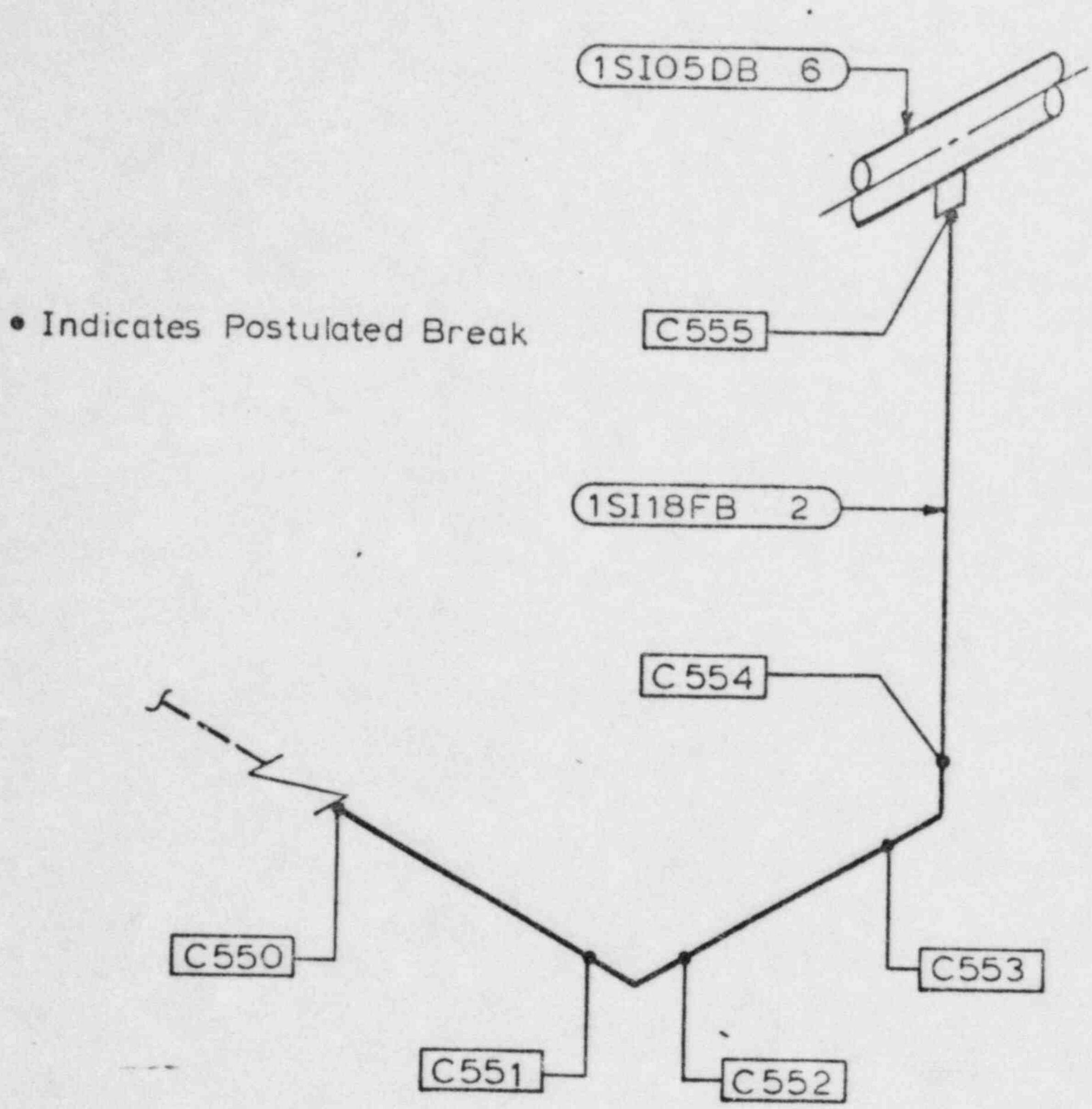
B/B
FIGURE 3.6-94
SAFETY INJECTION
SUBSYSTEM 1SI16
POSTULATED BREAK
LOCATIONS
(SHEET 1 OF 2)



SUBSYSTEM 1SI16 (Cont'd)

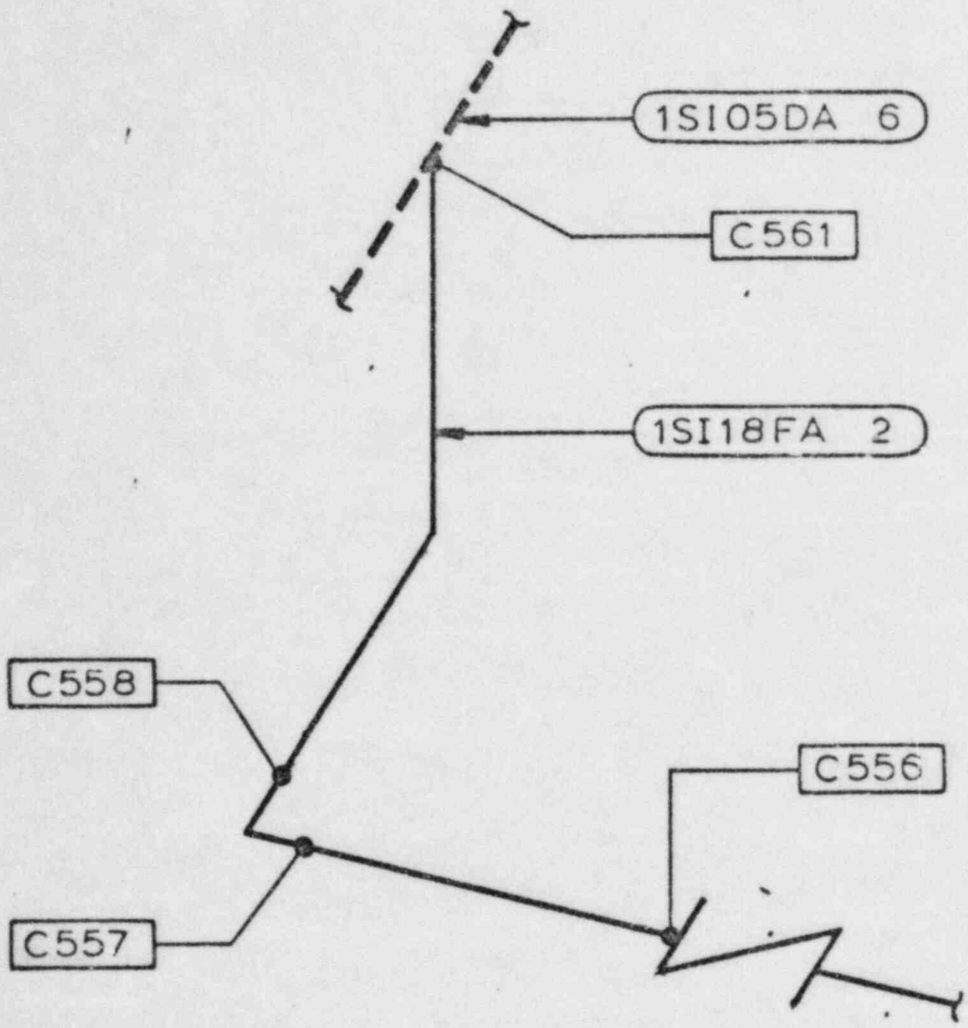
● Indicates Postulated Break

B/B
FIGURE 3.6-94
SAFETY INJECTION
SUBSYSTEM 1SI16
POSTULATED BREAK
LOCATIONS
(SHEET 2 OF 2)



SUBSYSTEM 1SI17

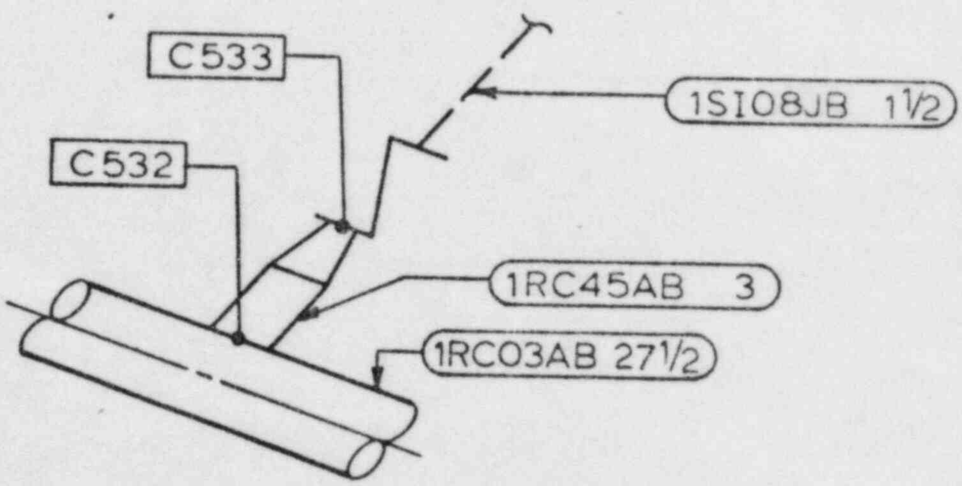
B/B
FIGURE 3.6-95
SAFETY INJECTION
SUBSYSTEM 1SI17
POSTULATED BREAK
LOCATIONS
(SHEET 1 OF 2)



SUBSYSTEM 1SI17 (Cont'd)

• Indicates Postulated Break

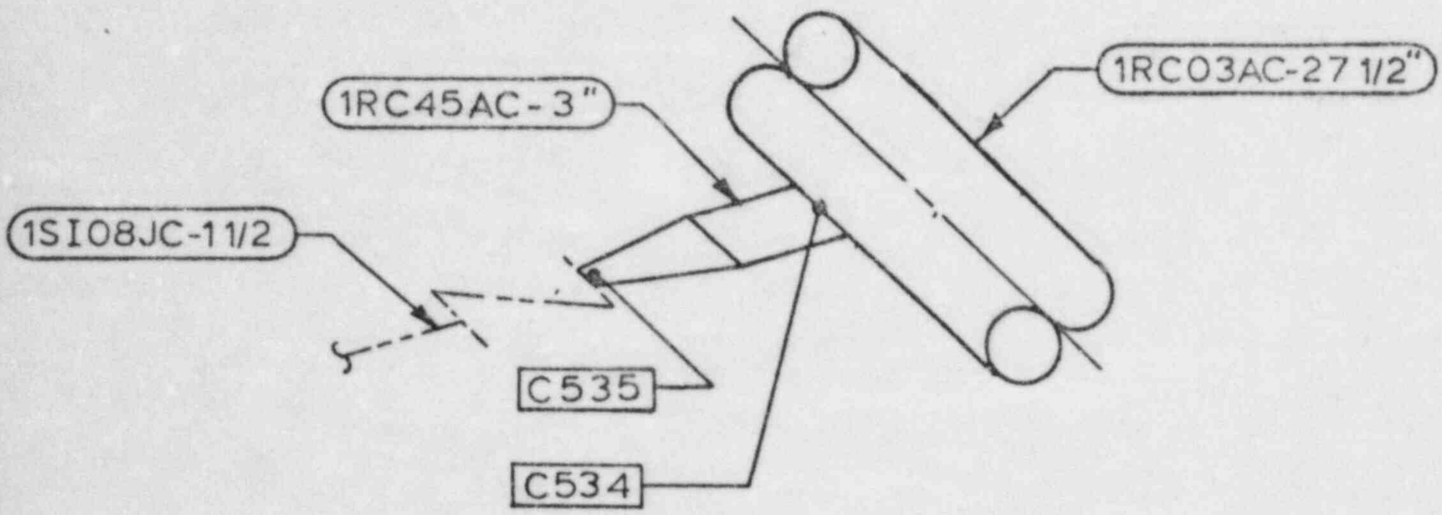
B/B
FIGURE 3.6-95
SAFETY INJECTION
SUBSYSTEM 1SI17
POSTULATED BREAK
LOCATIONS
(SHEET 2 OF 2)



SUBSYSTEM 1SI19

• Indicates Postulated Break

B/B
FIGURE 3.6-96
SAFETY INJECTION
SUBSYSTEM 1SI19
POSTULATED BREAK
LOCATIONS

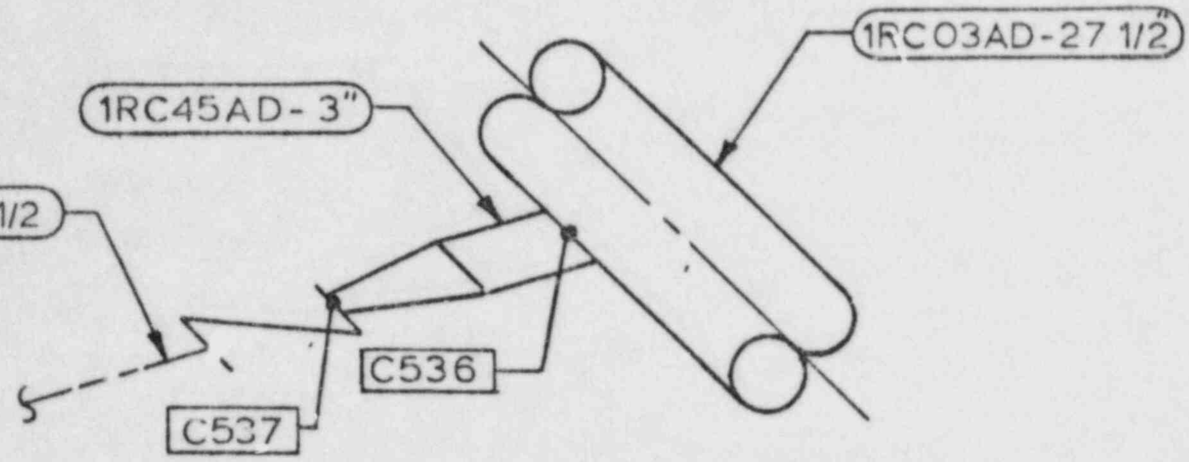


• Indicates Postulated Break

SUBSYSTEM 1SI20

B/B

FIGURE 3.6-97
SAFETY INJECTION
SUBSYSTEM 1SI20
POSTULATED BREAK
LOCATIONS

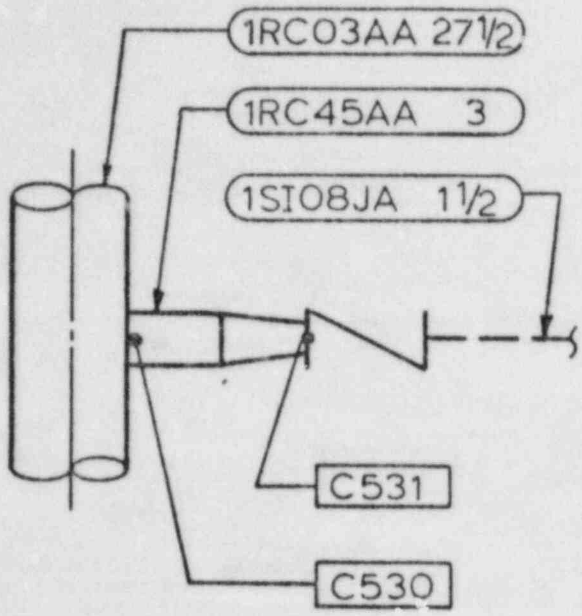


• Indicates Postulated Break

SUBSYSTEM 1SI22

B/B

FIGURE 3.6-98
SAFETY INJECTION
SUBSYSTEM 1SI22
POSTULATED BREAK
LOCATIONS

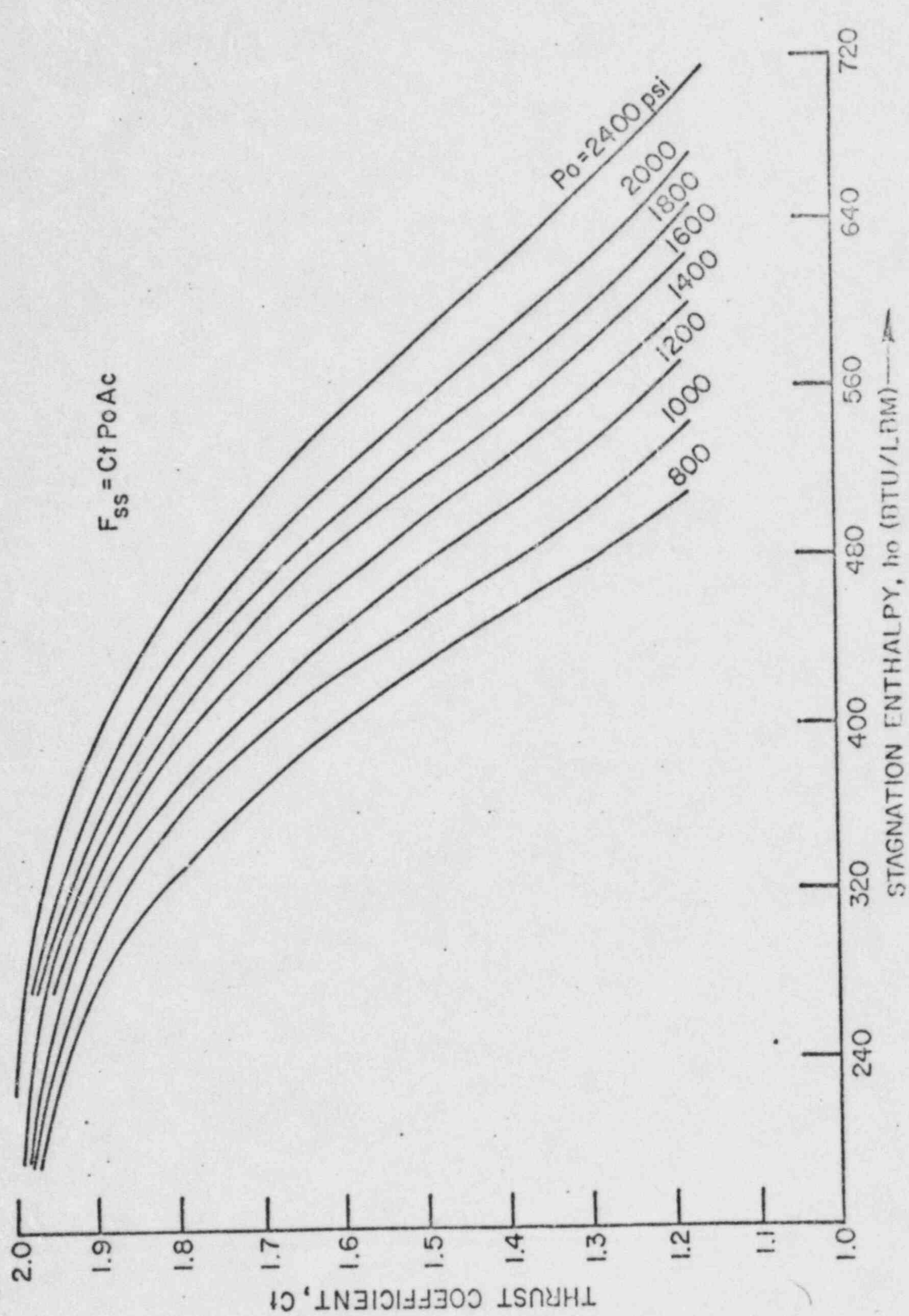


SUBSYSTEM 1SI24

• Indicates Postulated Break

B/B

FIGURE 3.6-99
SAFETY INJECTION
SUBSYSTEM 1SI24
POSTULATED BREAK
LOCATIONS

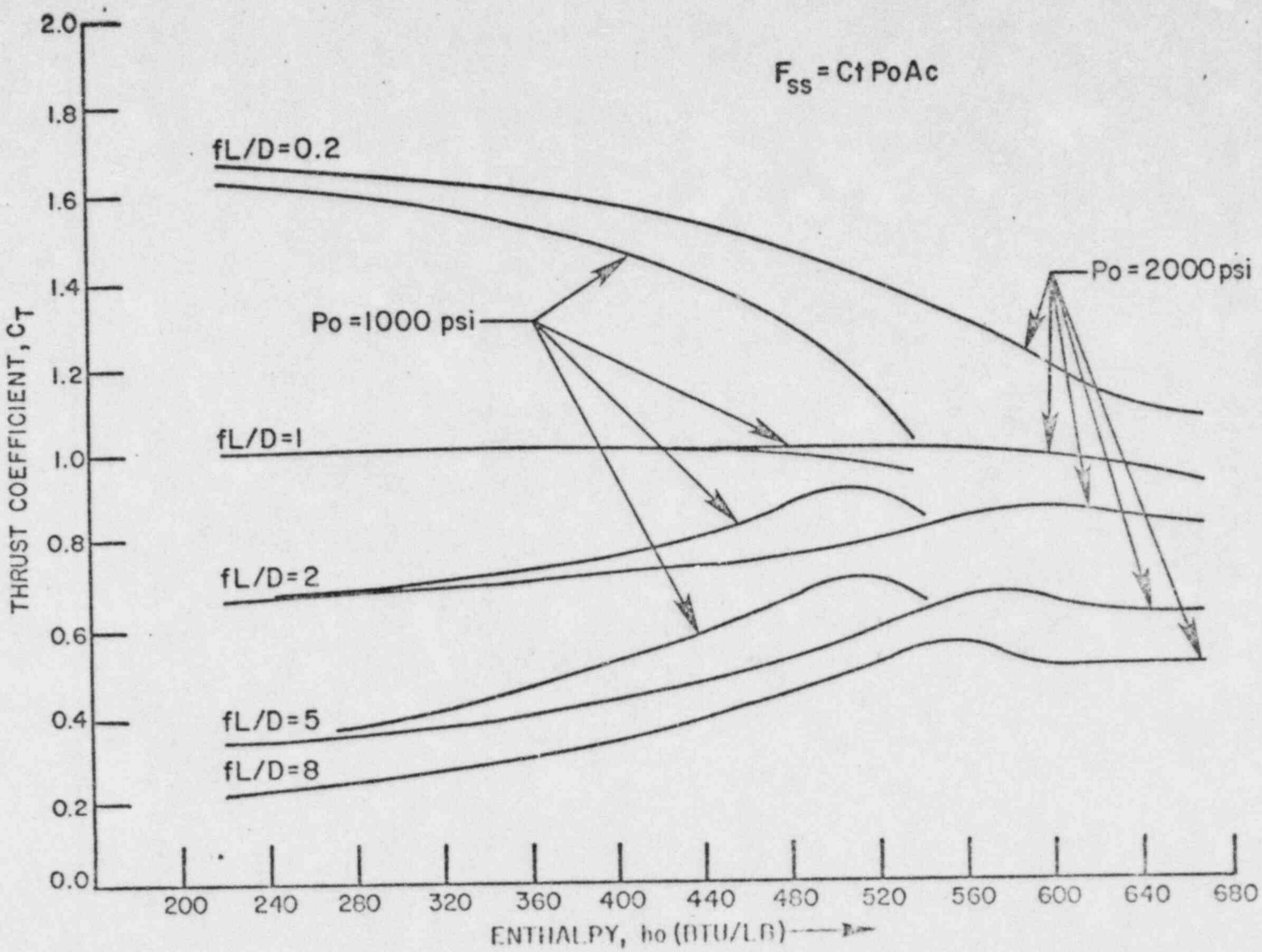


BYRON/BRAIDWOOD STATIONS

FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-100

THRUST FORCE AS A FUNCTION OF STAGNATION ENTHALPY (ho) AND PRESSURE (Po)

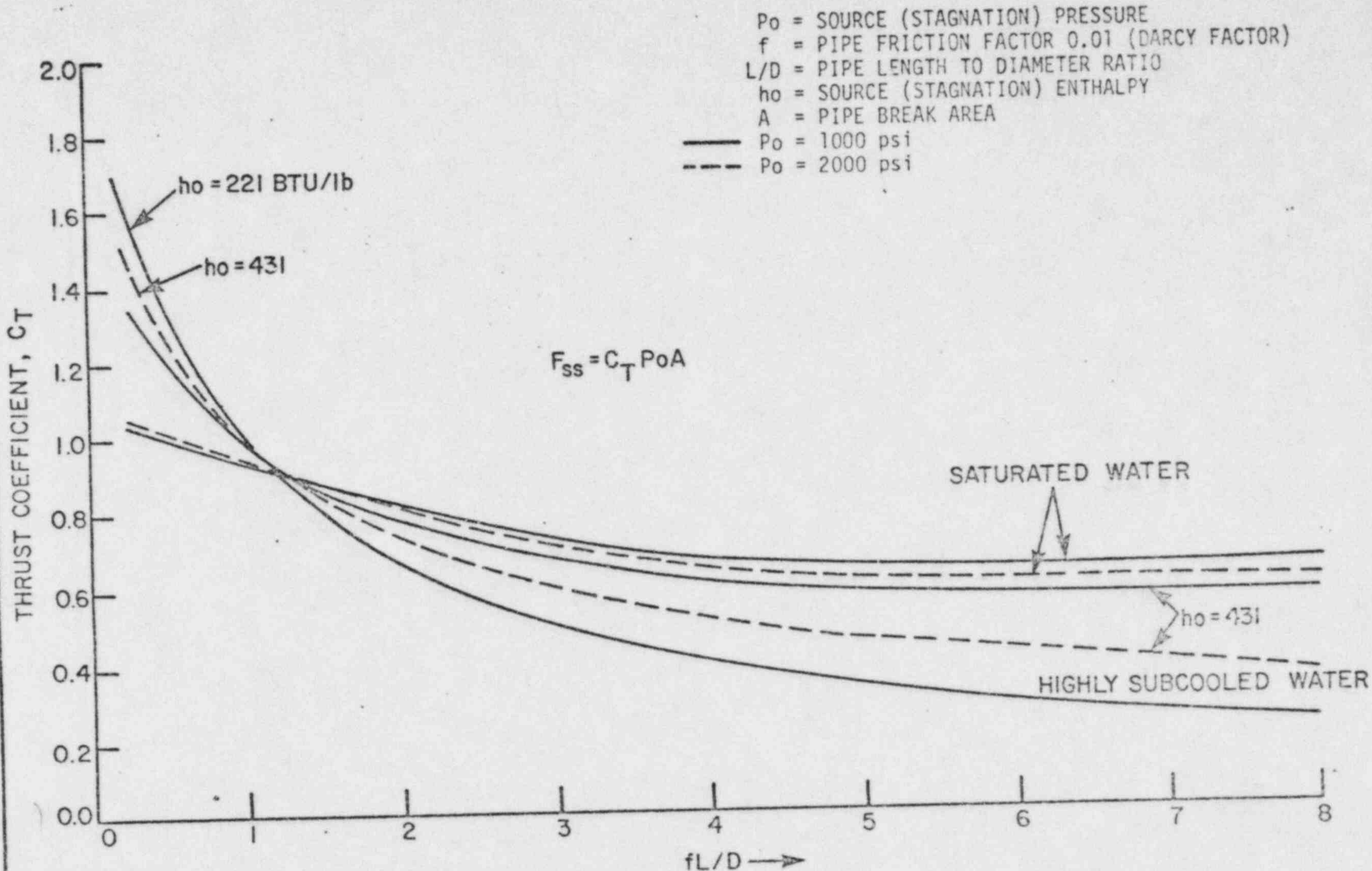


BYRON/BRAIDWOOD STATIONS

FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-101

SUBCOOLED WATER BLOWDOWN THRUST FORCE AS A FUNCTION OF STAGNATION ENTHALPY FOR VARIOUS VALUES OF PIPE FRICTION PARAMETER (fL/D)



BYRON/BRAIDWOOD STATIONS
FINAL SAFETY ANALYSIS REPORT

FIGURE 3.6-102

SUBCOOLED WATER BLOWDOWN THRUST FORCE AS A FUNCTION OF PIPE FRICTION PARAMETER (fL/D) FOR VARIOUS STAGNATION ENTHALPIES