VOLUME 1

REACTOR COOLANT SYSTEM

ASYMMETRIC LOADS

FINAL REPORT

Prepared by

COMBUSTION ENGINEERING, INC.

for

CALVERT CLIFFS 1 & 2

FORT CALHOUN

MILLSTONE 2

PALISADES

8007020574

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1.0 INTRODUCTION

This document presents the results of an evaluation of reactor coolant system components and supports when subjected to loads resulting from postulated pipe ruptures.

In May of 1975, it was determined by one applicant that asymmetric loading resulting from a postulated pipe rupture could have a significant effect on reactor vessel supports. Investigation of those supports and of other reactor coolant system components and supports utilizing the existing criteria for postulating pipe breaks and the existing methods of analysis led to further questions concerning the adequacy of the components and supports to withstand the effects of postulated pipe ruptures.

In January of 1978, Baltimore Gas and Electric, Consumers Power, Northeast Utilities, and the Omaha Public Power District were requested to reevaluate the reactor pressure vessel; fuel assemblies (including grid structures), control rod drives; emergency core cooling piping attached to the primary coolant pipe; primary coolant piping; reactor vessel, steam generator, and pump supports; reactor internals; and the biological shield wall for Calvert Cliffs 1 and 2, Palisades, Millstone 2, and Fort Calhoun.

In August of 1978, an evaluation plan was submitted to NRC describing the methods and codes to be employed in the evaluation.

The progress of the evaluation has been presented to NRC at various meetings since August during which interim results for various components were discussed.

In February of 1980, an interim report detailing the evaluation for the reactor pressure vessel, emergency core cooling piping, primary coolant piping, and reactor vessel, steam generator, and pump supports was submitted to NRC.

This document repeats the evaluations presented in the interim report and includes additional detail of these evaluations. In addition the evaluation of fuel assemblies (including grid structures), reactor internals, control rod drives, and the biological shield wall are presented.

2.0 NONENCLATRUE AND ABBREVIATIONS

Major Components of the Reactor Coolant System	RCS
Reactor Coolant Pump	RCP
Reactor Vessel	RV
Steam Generator	SG
Control Element Drive Mechanism	CEDM
Emergency Core Cooling System	ECCS
Core Support Barrel	CSB
Upper Guide Structure	UGS

3.0 REFERENCES AND CODES

3.1 Design Basis Pipe Breaks for the Combustion Engineering Two Loop Reactor Coolant System CEMPD-168A, Combustion Engineering Inc., June, 1977

3.2 ICES-STRUDLII, The Structural Design Language, Engineering Users Manual, First Edition, Massachusetts Institute of Technology, November, 1968

3.3 SAPIV, A Structural Analysis Program for Static and Dynamic Response of Linear Systems, University of California, Berkeley, June, 1973

3.4 Design Basis Pipe Breaks for the Combustion Engineering Two Loop Reactor Coolant System CENPD-168A, Appendix A-5, Combustion Engineering Inc., June, 1977

3.5 "Description of Loss-of-Coolant Calculational Procedures", CENPD-26, Combustion Engineering Inc., August, 1971

3.6 Standard Review Plan 6.2.1.2, "Subcompartment Analysis", February, 1975

3.7 CESSAR, "Combustion Engineering Standard Safety Analysis Report", Section 6.2.1.1-4 approved December 31, 1975

3.8 "Reactor Plant Subcompartment Analysis", CENPD-141 Revision 2, March 1978

3.9 Combustion Engineering Inc., "Method for the Analysis of Blwodown Induced Forces in a Reactor Vessel", CENPD-252-P, December, 1977 (Proprietary)

3.10 MARC-CDC, Non-Linear Finite Element Analysis Program, Control Data Crop., Minneapolis, Minn. 1976

3.11 "Structural Analysis of Fuel Assemblies for Combined Seismic and Loss of Coolant Accident Loadings", CENPD-178P, August, 1976

3.12 "Topical Report on Dynamic Analysis of Reactor Vessel Internals Under Loss of Coolant Accident Conditions with Application of Analysis to CE 800 Mwe Class Reactors", CENPD-42, 1971

3.13 "CESHOCK - A Computer Code to Solve the Dynamic Response of Lumped Mass Systems", Described and Verified in above Reference 3.12.

3.14 "SAMMSOR - A Finite Element Program to Determine the Stiffness and Mass Matrices of Shells of Revolution", Described and verified in above Reference 3.12.

3.15 "DYNASOR - A Finite Element Program for the Dynamic Non-Linear Analysis of Shells of Revolution", Described and verified in above Reference 3.12

3.16 "LOAD - A Computer Code to Calculate Dynamic Axial LOCA Loads Using the Control Volume Formulation", Calculation SP80-STA-25, 5/15/78

3.17 "ASHSD - A Dynamic Stress Analysis Code of Axisymmetric Structures Under Arbitrary Loading", Described and verified in above Reference 3.12

3.18 "RUMBLE - A Computer Code to Compute Fuel Bundle Stresses Based on Deflected Shapes", Described in above Reference 3.11

3.19 "DDIFF Code Topical Report", CENPD-141, April 30, 1974

3.20 RELAP4/MOD5, ANCR-NUREG-1335, A Comprehensive For Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems

3.21 "Reactor Coolant System Asymmetric Coads Evaluation Plan", Combustion Engineering, Inc., August 4, 1978

3.22 "ANSYS Engineering Analysis System Users Manual", Swanson Analysis Systems, Inc., Houston, PA, August 1, 1978

4.0 DESCRIPTION OF EVALUATION

4.1 Summary

This document presents the results of the evaluation of reactor coolant system components and supports when subjected to loads resulting from postulated pipe ruptures for Calvert Cliffs 1 and 2, Fort Calhoun, Millstone 2 and Palisades.

The evaluation of the components and supports demonstrates that they are adequate to withstand the loads resulting from postulated pipe ruptures.

Section 4.2 of this document describes the methods of calculating break area as a function of time and presents the resultant areas and opening times for the postulated pipe ruptures. These areas and opening times form the basis for analyses described in subsequent sections.

Section 4.3 of this document describes the methods of calculating mass and energy releases from the postulated pipe breaks and resultant subcompartment pressure differences across components and walls (External Asymmetric Loads).

The resultant pressure differences are presented. These represent one of the forcing functions acting on components and walls.

Section 4.4 of this document describes the methods for calculating reactor vessel internal pressure differences and flows resulting in forces on the reactor internals and the reactor vessel (Internal Asymmetric Loads). The pressure distribution in the reactor vessel as a function of time and the drag loads on the CEA shroud are presented. These represent forcing functions on both the reactor vessel and the reactor internals.

Section 4.5 of this document describes the methods for calculating the reactions on the reactor coolant system supports, intact piping, and components in the intact loop due to the imposition of thrust forces, external asymmetric loads and internal asymmetric loads from the pipe breaks described in Section 4.2. The calculated reactions were compared to the load carrying capability of the supports and components.
For the reactor vessel supports, the calculated loads as a function of capability range from 34% for the outlet guillotine on Calvert Cliffs and Millstone 2 to 81% for the inlet guillotine on Fort Calhoun.
For the steam generator supports, a modification to the Fort Calhoun supports was performed. This modification consisted of replacing the 3 3/8" diameter rods of A36 steel with 4 1/4" diameter rods of C1018 steel. The interface connection at the rods was unchanged. With this modification, all supports for the Fort Calhoun steam generator supports are shown to have adequate margin.
The loads on the steam generator supports for Calvert Cliffs, Millstone 2, and Palisades do not exceed the load which would result in yielding of the material.

The calculated loads . . reactor coolant piping and components nozzles are below the load carrying capability as defined by the ASME Code, Section III, for Faulted Condition for elastic analysis.

Section 4.C of this document describes the methods of analysis for reactor internals and presents the results on each of the component parts of the reactor internals in terms of margin to allowable stress in accordance with the limits for Faulted Condition of the ASME code, Section III for elastic analysis. Each component part shows positive margin.

Section 4.7 of this document describes the methods of analysis for fuel and the fuel testing performed. The detailed results of the analysis are presented in Appendix A for the CE fuel for Calvert Cliffs, Millstone 2 and Fort Calhoun.
An evaluation of the fuel components shows that, except for grids in the outer row of fuel, all components, including grids in the interior bundles, can withstand the calculated loads. Specifically, the load on 2 grids in the outer fuel bundles exceeds the rated load for unirradiated grids for Calvert Cliffs and Millstone 2 by a maximum of 47 % and the load on one grid in the outer fuel bundles exceeds the rated load for unirradiated grids for Fort Calhoun by 6 %. None of the spacer grid rated loads for irradiated materials are exceeded. In addition, in order to demonstrate core coolability for the outer row of bundles for which grid capability for unirradiated material has been exceeded, a reduced channel ECCS evaluation of this bundle is being performed. This evaluation is described in Appendix E. Appendix B presents a preliminary evaluation of the Palisades fuel and describes the steps to be taken to finalize the fuel analysis.

Section 4.8 of this document describes the methods of evaluation for the control element drive mechanisms and the results of the evaluation. The results show that pressure boundary integrity of the CEDMs when subjected to vibratory motion as a result of pipe rupture is maintained.

Section 4.9 of this document describes the methods of evaluation for the emergency core cooling piping attached to the primary coolant pipes. The results shows that the calculated loads are within those allowed by the ASME code Section III, for Faulted Condition. The regions of high stress have been evaluated to determine functionability; the maximum plastic strain is less than 2% in these regions. Functionability is not impaired at these strain levels.

Section 4.10 of this document presents the evaluation of the primary shield wall. For Calvert Cliffs and Millstone, the capability of the wall to withstand the applied loads has been demonstrated. For Fort Calhoun, it has been determined that the calculated loads exceed the design loads. An evaluation of the capability of the primary shield wall for Fort Calhoun leads us to conclude that adequacy of the wall could be demonstrated by state of the art analytical methods. We believe that such an analysis is unwarranted and instead propose to show, by fracture mechanics techniques, that the postulated pipe break is incredible and that the primary shield wall is capable of withstanding the largest credible pipe break predicted by those techniques. The plan for fracture mechanics evaluation is presented in Appendix D. In addition, a feasibility study of adding pipe whipping restraints to the Fort Calhoun pump discharge piping to reduce break size has been performed and is presented in Appendix C. The analysis of the Palisades reactor cavity shield wall is currently being performed by Bechtel Power Corporation, Ann Arbor, Michigan. The time dependent reaction forces at the component supports from a Palisades plant specific analysis were sent to Consumers on June 24, 1980. The cavity wall analysis should be completed by Mid August and the report should be submitted by August 29, 1980.

4.2 PIPE BREAKS

4.2.1 DESIGN BASIS

As stated in Reference 3.21, guillotine ruptures were postulated to occur at the following locations:

a) Reactor Vessel Hot Leg Nozzle
b) Reactor Vessel Cold Leg Nozzle
c) Steam Generator Outlet Nozzle
d) Steam Generator Inlet Nozzle

4.2.2 METHOD OF ANALYSIS

For each postulated guillotine, a dynamic non-linear time history analysis was performed using methods discussed in Reference 3.1. Each analysis generated pipe end deflection time histories, from which flow area time histories and maximum flow areas were mechanistically determined.

Calvert Cliffs was selected as the model plant to be used in the generic analysis. For each postulated break, assessments were made for Millstone, Palisades and Fort Calhoun. These assessments are explained in Section 4.2.4.

4.2.2.1 Generic RV Outlet Nozzle Guillotine Analysis

A three-dimensional model of the reactor coolant system was constructed using lumped mass parameter techniques. The mathematical model represents the total Reactor Coolant System (RSC) mass and stiffness, with discontinuity at the Reactor Vessel (RV) outlet nozzle. The model is shown in Figure 4.2.1. It contains lumped mass representations of the RV, both Steam Generators (SG), all four Reactor Coolant Pumps (RCP), and piping of the 1A and 1B cold legs as well as mass detail of the ruptured hot leg. The non-linearities of the RV gapped supports are represented as in the SG #2 lower gapped support parallel to the hot leg. The lower support system of SG #1 was modelled in detail, and includes the non-linearities of the lower stop and all four vertical pads in order to calculate its movement following the postulated rupture. The resulting model consists of 71 mass dynamic degrees of freedom (d.d.o.f.) and 8 non-linear support locations.

The physical definition of the resulting model was supplied in the STRUDL computer code Reference 3.2 which generated the condensed stiffness matrix. This matrix, along with the mass definition, gapped support definition, damping, and a set of three-dimensional time history forcing functions as discussed and developed in Reference 3.1 was supplied to the DAGS computer code (Reference 3.4). DAGS generated the severed pipe end deflection time history as well as the deflection time history of the safe end of the RV outlet nozzle.

4.2.2.1 Generic RV Outlet Nozzle Guillotine Analysis (Cont'd)

Each time history was supplied to a DAGS post-processor, which generated flow area time history contributions of each end of the severed pipe, from which a conservative flow area and rise time were determined for this rupture.

4.2.2.2 Generic RV Inlet Nozzle Guillotine Analysis

The details of mathematical model analyzed (Figure 4.2.2) were essentially the same as those for the RV Outlet Nozzle Guillotine, except that the discontinuity of the piping was represented at the 1A loop RV inlet nozzle safe end, and greater mass detail was included at the 1A loop RCP discharge leg. The time history forcing function was applied at the rupture location. This model consists of 62 mass d.d.o.f.'s and 8 non-linear support locations. Analysis techniques similar to those used for the RV outlet nozzle guillotine were used for this rupture to determine the flow area and rise time.

4.2.2.3 Generic SG Outlet Nozzle Guillotine Analysis

The severance of the pipe from the 1A loop SG outlet nozzle was modelled, and greater mass detail was included at the 1A loop RCP suction leg (See Figure 4.2.3). This model consists of 75 mass d.d.o.f.'s and 8 non-linear support locations. Using techniques outlined in Section 4.2.2.1 the flow area and rise time were determined for this rupture.

4.2.2.4 Generic SG Inlet Nozzle Guillotine Analysis

The discontinuity of the hot leg pipe at the SG #1 inlet nozzle safe end was modelled, and greater mass detail was included at the hot leg. Effects of plasticity in the ruptured hot leg were included in the model at the RV outlet nozzle safe end and hot leg pipe interface (See Figure 4.2.4), and these effects were included in the rigorous time history analysis. This model contains 75 mass d.d.o.f.'s and 7 gapped support locations. As in the RV hot leg nozzle guillotine analysis, flow area contributions of both the component nozzle and the severed pipe end were generated on a time history basis, and the flow area and rise time were determined.

4.2.3 RESULTS OF THE GENERIC ANALYSIS

The results of the generic flow area analysis are summarized in Table 4.2-1. It can be noted from this summary that the SG outlet nozzle guillotine and the RV inlet nozzle guillotine result in full double-ended circumferential ruptures. The hot leg nozzle guillotine ruptures develop less than full area breaks because of the inherent strength of the hot leg piping and the stiffness of the supports of the major components.

4.2.3 RESULTS OF THE GENERIC ANALYSIS (Cont'd)

All four cases show rise times equal to or greater than 20 milliseconds. Pipe motion contributes almost all the motion, and except for the RV outlet guillotine case, the components do not move appreciably as compared to the pipe. For the RV outlet guillotine, the pipe motion relative to the components is small. SG motion is the major contributor and total break area development is small.

4.2.4 PLANT SPECIFIC FLOW AREA EVALUATIONS

Table 4.2-2 shows results of the specific operating plants for flow areas and rise times.

4.2.4.1 Calvert Cliffs 1 & 2

The Calvert Cliffs plant was used as the basis for the generic flow area analysis models; therefore, the generic results are directly applicable to Calvert Cliffs.

4.2.4.2 Millstone 2

The Millstone 2 plant is identical to Calvert Cliffs in RCS piping, component size and layout, therefore the generic results are directly applicable.

4.2.4.3 Palisades

An evaluation of the flow areas and respective rise times of the Palisades RCS was made by comparison of the Palisades plant RCS geometry and size to that of the generic plant. It was found that for parameters which control the flow areas and rise times; ie., pipe size and weight, pipe layout and length, system pressure, and SG and RV support schemes, the Palisades plant was virtually identical to the generic plant. Therefore, the results for the generic plant results are directly applicable.

4.2.4.4 Fort Calhoun

An evaluation of the flow areas and respective rise times for the Fort Calhoun RCS was made by comparison of the Fort Calhoun plant RCS parameters to those of the generic plant. The results of these comparisons follow.

4.2.4.4.1 RV Inlet Nozzle Guillotine and SG Outlet Nozzle Guillotine

The similarity in layout and length between the Fort Calhoun cold leg loop and the generic loop resulted in full double ended break flow areas for each of these ruptures. Because Fort Calhoun's cold leg pipes have 24" inside diameter as compared to the 30" inside diameter for the generic plant, a full area break for the Fort

4.2.4.4.1 RV Inlet Nozzle Guillotine and SG Outlet Nozzle Guillotine (Cont'd)

Calhoun plant is 905 In^2.

In order to evaluate the rise times for these two ruptures, the natural frequency of the resulting cantilevered cold leg for Fort Calhoun was compared to the generic cold leg to determine how quickly the pipe can respond (i.e., move away from the component to create flow area) following the pipe tension release. That is, the rise time is a direct function of the pipe's fundamental frequency. Since it was determined that both the Fort Calhoun and the generic thrust forces were each large enough to cause full area breaks, the amplitude of the thrust would only be a minor factor in rise time evaluation, and Fort Calhoun's smaller thrust would only have a slowing effect on the rise time.

The fundamental frequency for the Fort Calhoun pipe was compared to the fundamental frequency for the generic plant pipe by comparing parameters for each cold leg. The fundamental frequency (f) is proportional to mass (M), pipe length (L), and pipe bending moment of inertia (I):

By computation:

$$M_{Ft.\ Calhoun} = 59\%\ M_{Generic};\quad M' = .59$$

$$I_{Ft.\ Calhoun} = 43\%\ I_{Generic};\quad I' = .43$$

$$L_{Ft.\ Calhoun} = 90\%\ L_{Generic};\quad L' = .90$$

$$f_{Ft.\ Calhoun} = \sqrt{I'/(L')^3 M'} \times f_{Generic}$$

$$f_{Ft.\ Calhoun} = \sqrt{(.43)/(.90)^3(.59)} \times f_{Generic}$$

$$f_{Ft.\ Calhoun} = .9999\ f_{Generic}$$

Rise times for the two Fort Calhoun cold leg nozzle guillotines postulated were, therefore, evaluated to be virtually the same as those for the generic plant. They are presented in Table 4.2-2.

4.2.4.4.2 RV Outlet Nozzle Guillotine and SG Inlet Nozzle Guillotine

Fort Calhoun's steam generator and reactor vessel support and accident restraint systems are entirely different than the generic plant's systmes, and the strength and stiffness of the hot leg piping is lower than the generic plant. For these reasons, a plant specific analysis for Fort Calhoun was performed.

RV OUTLET NOZZLE GUILLOTINE BREAK

AREA ANALYSIS

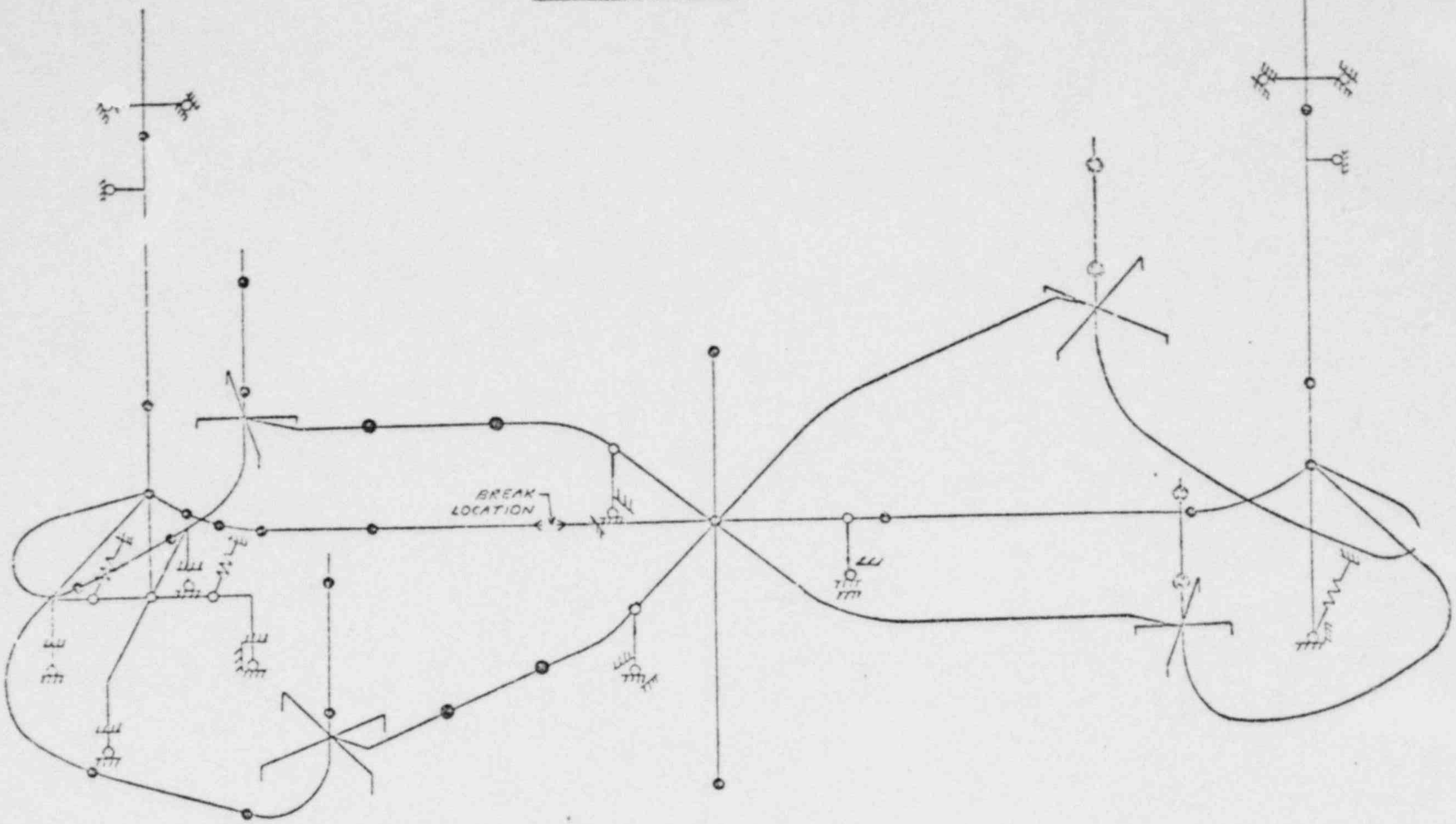

FIGURE 4.2.1

RV INLET NOZZLE GUILLOTINE BREAK

AREA ANALYSIS

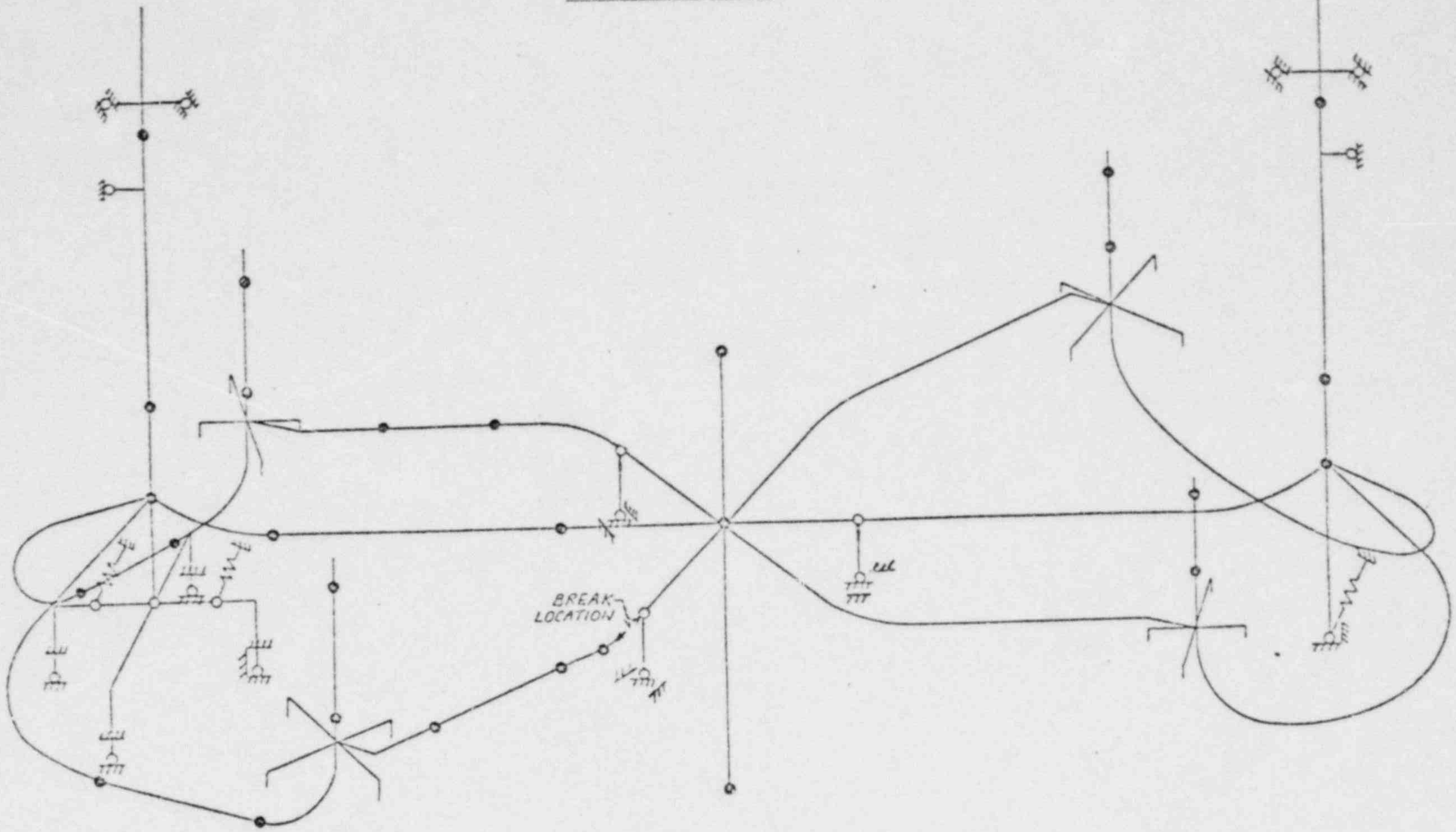

FIGURE 4.2.2

SG OUTLET NOZZLE GUILLOTINE BREAK

AREA ANALYSIS

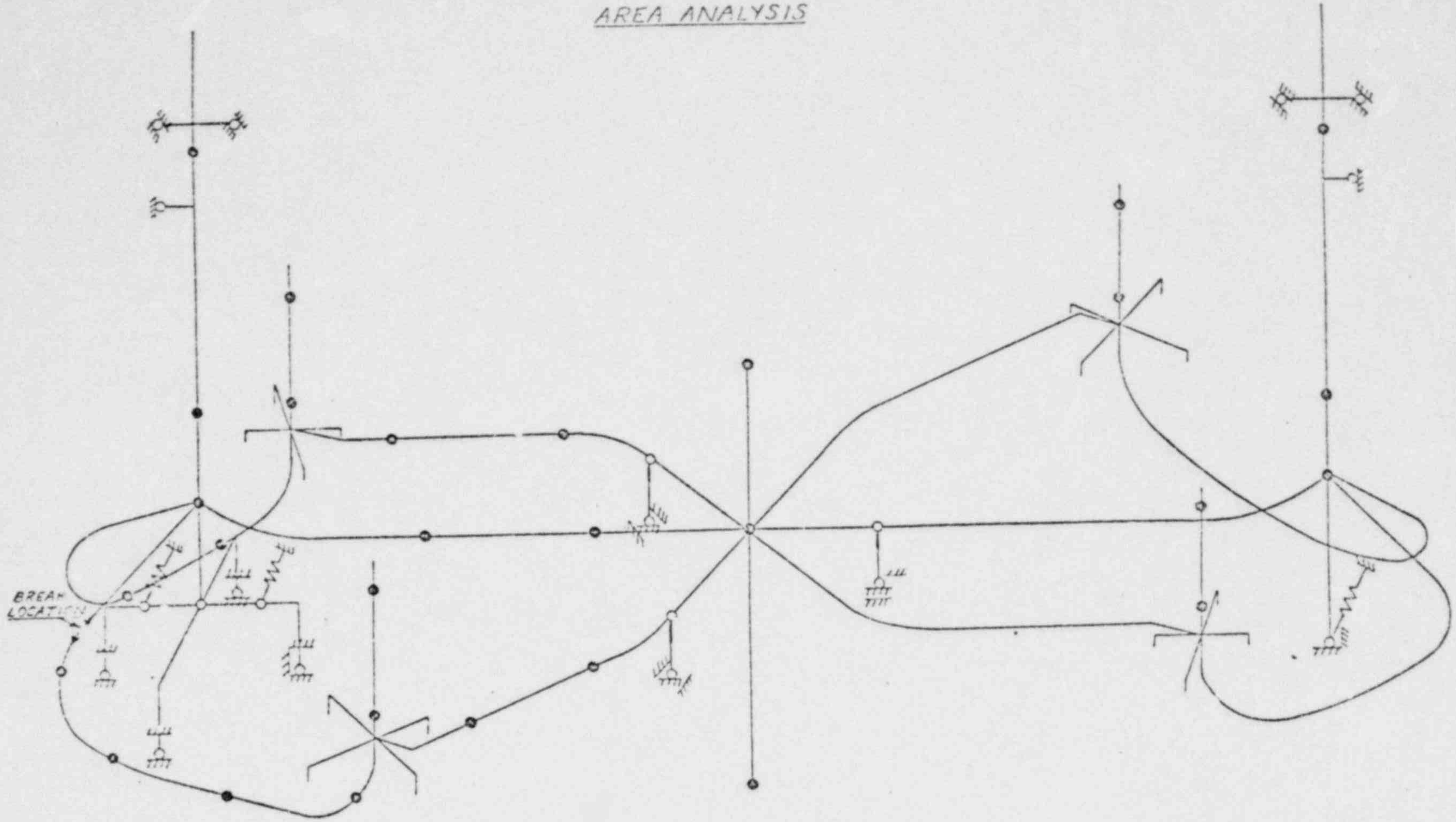

FIGURE 4.2.3

SG INLET NOZZLE GUILLOTINE BREAK

AREA ANALYSIS

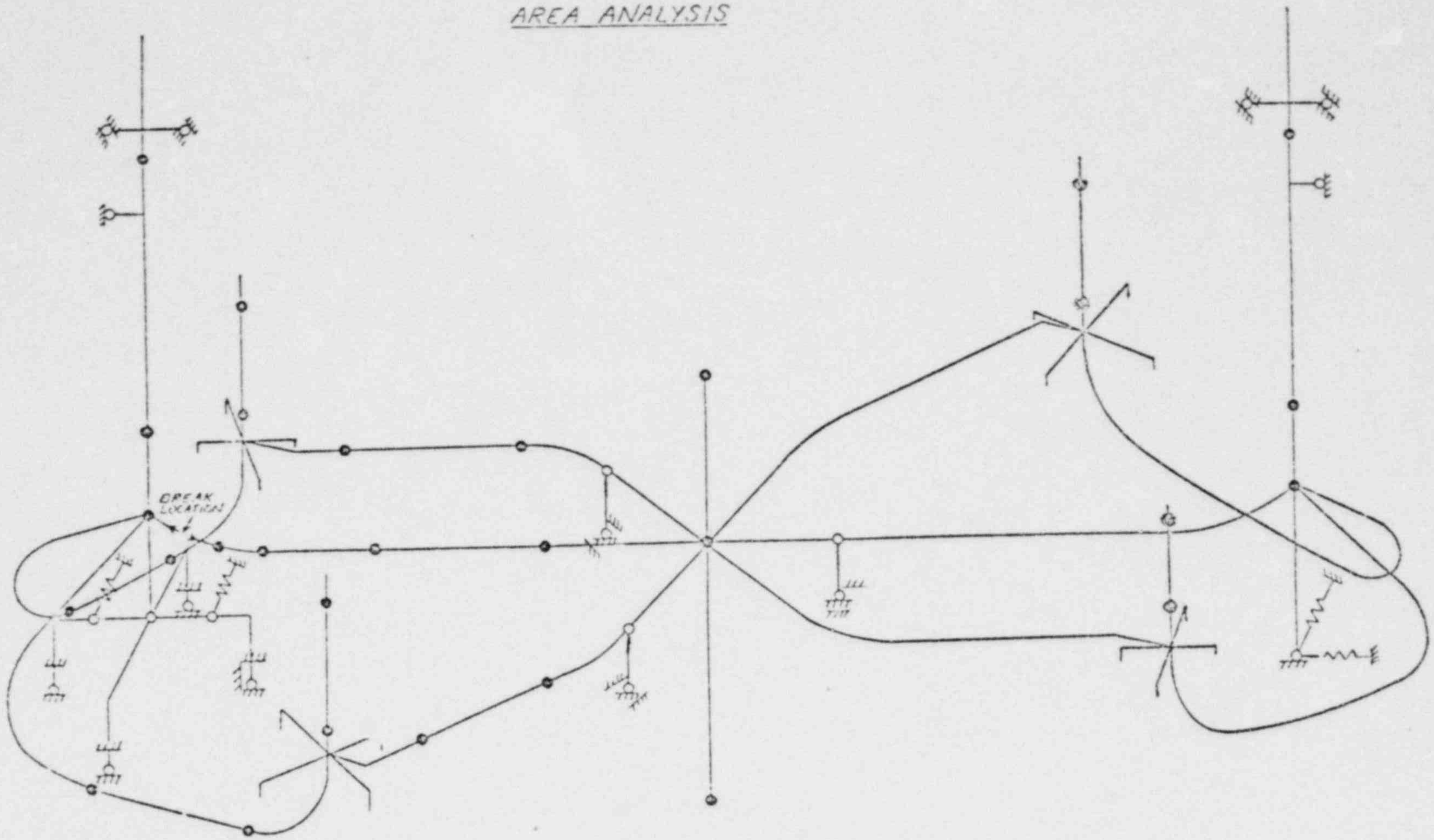

FIGURE 4.2.4

TABLE 4.2.4-1

GENERIC PLANT

PIPE BREAK AREAS AND BREAK OPENING TIMES

POSTULATED RUPTURE	BREAK FLOW AREA (IN^2)	RISE TIME (MILLISECONDS)
RV INLET GUILLOTINE	1414 = 2.0A	23.
RV OUTLET GUILLOTINE	135 = .19A	20.
SG INLET GUILLOTINE	1000 = 1.4A	24.
SG OUTLET GUILLOTINE	1414 = 2.0A	20.

TABLE 4.2.4-2

PLANT SPECIFIC

PIPE BREAK AREAS AND BREAK OPENING TIMES FOR MASS AND ENERGY RELEASES

POSTULATED RUPTURE	CALVERT CLIFFS*	MILLSTONE*	PALISADES	FORT CALHOUN
RV INLET (IN^2) TIME (MSEC)	1414 23.	1414 23.	1414 23.	905 (2.0A) 23.
RV OUTLET (IN^2) TIME (MSEC)	135 20.	135 20.	135 20.	IN PROGRESS
SG INLET (IN^2) TIME (MSEC)	1000 24.	1000 24.	1000 24.	IN PROGRESS
SG OUTLET (IN^2) TIME (MSEC)	1414 20.	1414 20.	1414 20.	905 (2.0A) 20.

*GENERIC PLANT